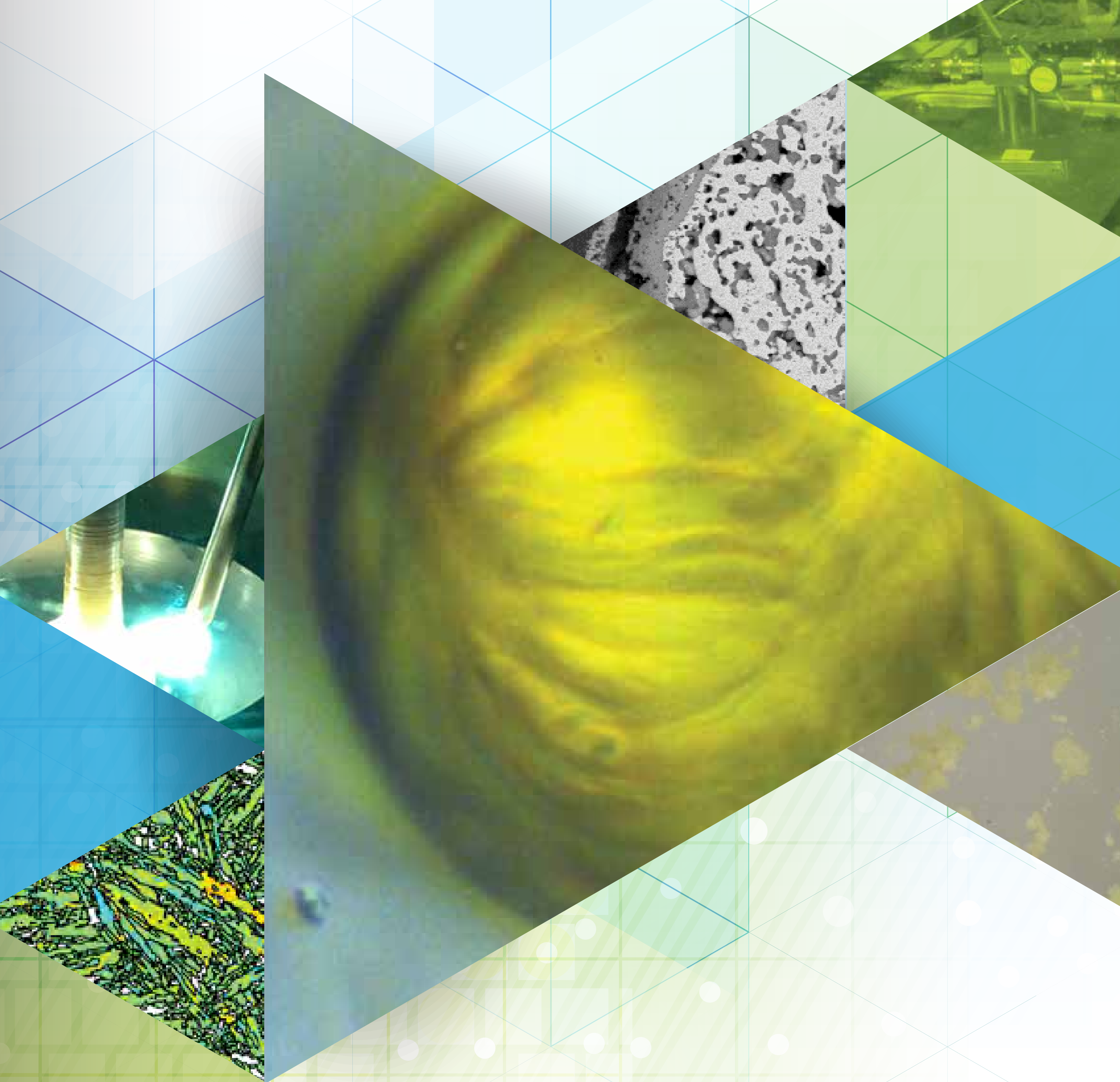
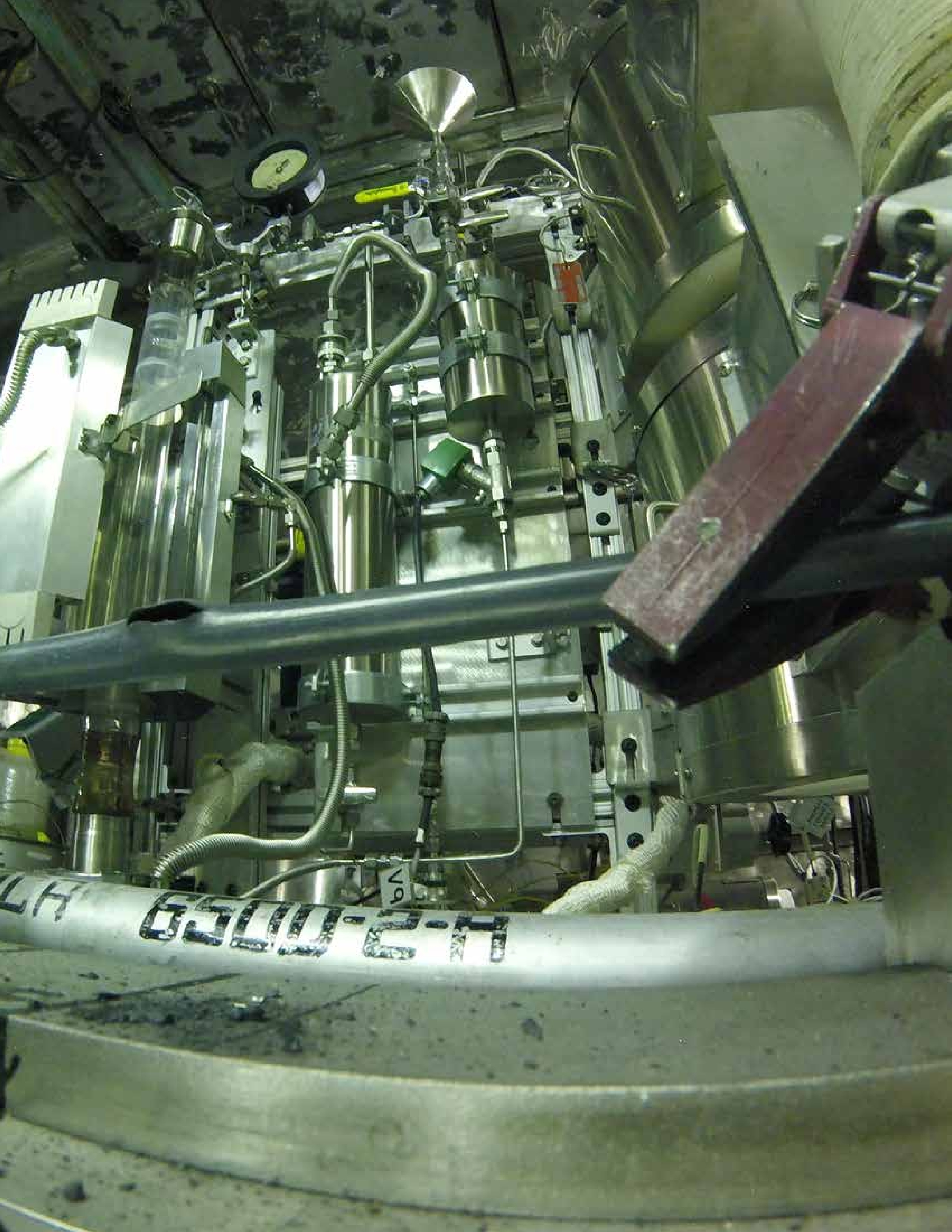


ADVANCED FUELS CAMPAIGN
2019 Accomplishments

Advanced Fuels Campaign



Nuclear Fuels Cycle & Supply Chain

Advanced Fuels Campaign
2019 Accomplishments

INL/EXT 19-56259

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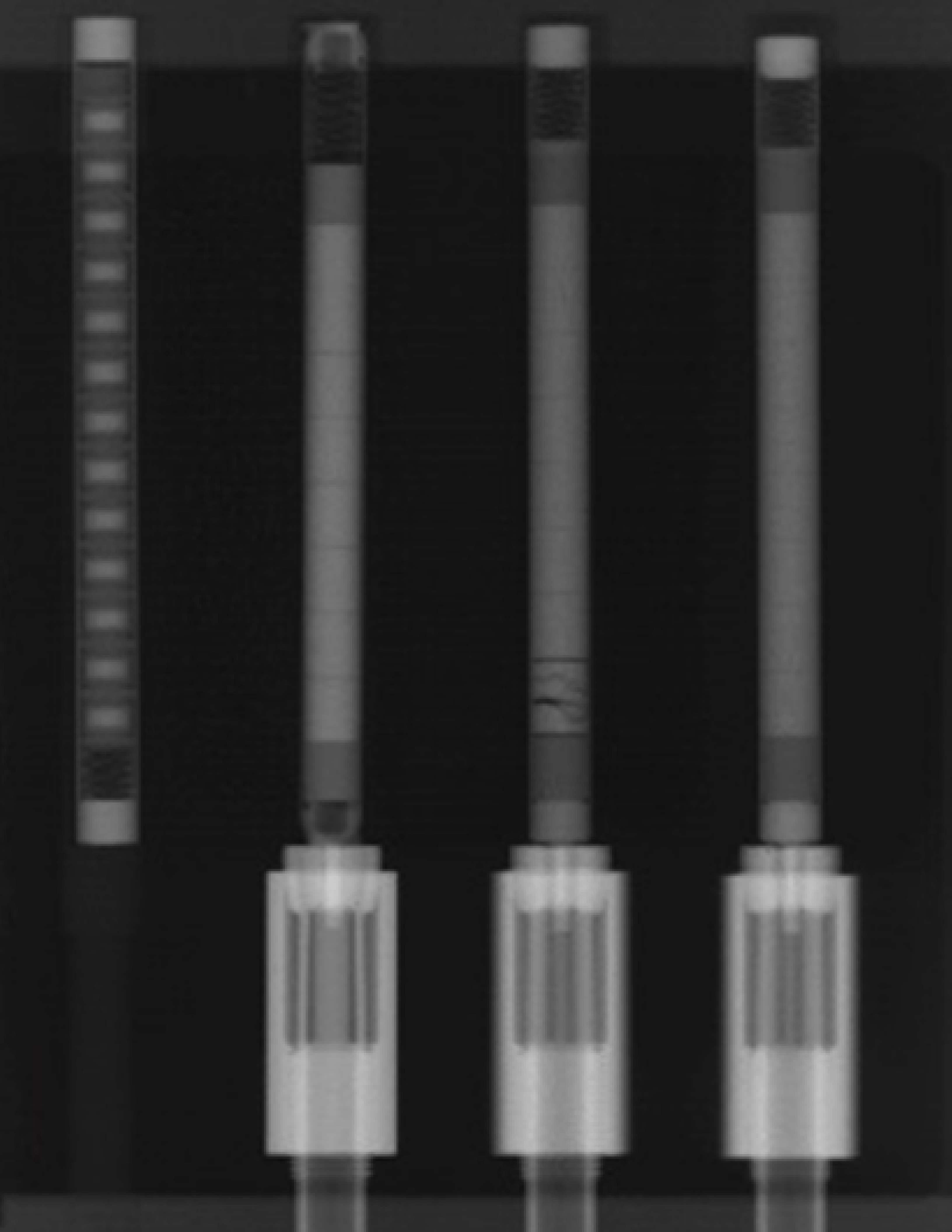

TABLE OF CONTENTS

1 AFC MANAGEMENT AND INTEGRATION

2 ADVANCED LWR FUELS

TABLE OF CONTENTS - *Continued*

3 AR FUELS

4 CAPABILITY DEVELOPMENT

5 APPENDIX

ADVANCED REACTOR FUELS SYSTEMS


TRANSIENT REACTOR
TEST FACILITY

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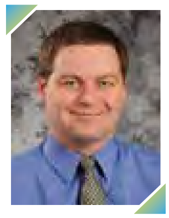

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1.2 FROM THE DIRECTOR

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The mission of the Advanced Fuels Campaign (AFC) is to perform or support research, development, and demonstration (RD&D) activities to identify and mature innovative fuels and cladding materials and associated technologies with the potential to improve the performance and safety of current and future reactors; increase the efficient utilization of nuclear energy resources; contribute to enhancing proliferation resistance of the nuclear fuel cycle; and address challenges related to waste management and ultimate disposal.

AFC pursues its mission objectives using a goal-oriented, science-based approach that seeks to establish a fundamental understanding of fuel and cladding behaviors under conditions that arise during fabrication, normal steady-state irradiation, off-normal transient scenarios, and storage/disposal. This approach includes advancing the theoretical understanding of fuel behavior, conducting fundamental and integral experiments, and supporting the mechanistic, multi-scale modeling of nuclear fuels to inform and guide fuel development projects, advance the technological readiness of promising fuel candidates, and ultimately support fuel qualification and licensing initiatives. In the area of advanced tools for the modeling and simulation of nuclear fuels, the AFC works in close partnership with the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program, participating with NEAMS in developing mechanistic fuel behavior models and providing experimental data to inform and validate its most advanced tools.

Specifically, AFC objectives in the coming five year horizon include:

1. support the industry-led development of Accident Tolerant Fuel (ATF) technologies with improved reliability and performance under normal operations and enhanced tolerance during hypothetical accident scenarios, with implementation of lead test rods/assemblies of one or more ATF concepts in commercial reactor(s) by or before 2022 and batch reloads of near-term concept(s) in the 2023-2026 timeframe;
2. lead research and development on innovative fuel and cladding technologies for applications to future advanced reactors, especially fast-spectrum reactors, including reactors that utilize both once-through and recycle scenarios;
3. continue the development and demonstration of a multi-scale, science-based approach to fuel development and testing, and contribute to the establishment of a state-of-the-art research and development (R&D) infrastructure necessary to accelerate the development of new fuel concepts; and
4. collaborate with NEAMS on the development and validation of multi-scale, multi-physics, and increasingly predictive fuel performance models and codes.

This report provides concise summaries of many of the significant AFC accomplishments made during FY-19. Of particular note are the following key accomplishments and their significance:

- The Halden Gap Assessment Final Report with recommendations to Department of Energy (DOE) Nuclear Energy (NE) on strategies and investments needed to support testing and qualification of industry-led ATF concepts without the Halden Reactor was issued in December 2018.
- Uranium silicide (U_3Si_2) fuel pellets fabricated at INL were incorporated into fuel rods as part of a Westinghouse Lead Test Assemblies (LTA) loaded into the Byron commercial power reactor, which has been operating at full power since April 2019; this represents the first DOE-sponsored commercial LTA that included a new fuel form.
- A vapor phase processing capability for advanced SiC/SiC composites was established at Oak Ridge National Laboratory (ORNL) to allow for fabrication of specimens needed for characterization, optimization, and performance testing.
- Fundamental understanding of uranium silicide fuel was extended through temperature-dependent neutron diffraction measurements, thermo-mechanical property measurements, and thermodynamic evaluations of U_3Si_2 hydride reactions, and new fabrication and characterization studies on doped UO_2 fuel concepts were initiated.
- Fabrication trials on reduced-diameter fuel pellets (UO_2 and U_3Si_2) were success, setting the stage for possible accelerated testing in ATR using the FAST approach.
- Progress was made in investigations of microstructurally engineered UN by introducing Y in an attempt to improve its corrosion and oxidation resistance
- Fracture toughness measurements using small scale cantilever beam testing was performed using unirradiated UO_2 and found to be in good agreement with data obtained using conventional testing; this methodology will be extended to irradiated fuel specimens in the future.
- Variations in minor alloying constituent levels were explored in new heats of FeCrAl cladding resulting in improved properties, improved measurements of hydrogen diffusion in FeCrAl cladding were obtained, tube production and characterization of ODS FeCrAl variants were initiated, and the FeCrAl Properties Handbook was updated and issued.
- Prototypic testing of Westinghouse and Framatome ATF concepts in Advanced Test Reactor (ATR) 2A Loop (ATF-2) continued throughout FY19, and test fuel rodlets achieved burnups ~10 GWd/MTU by year end. The ATF-2 Loop continues to be the only in-reactor option in the western world for testing of ATF concepts under prototypic Replace with pressurized water reactor (PWR) conditions.
- A major ATF-2 test train design modification for boiling water reactor (BWR)-sized rodlets was completed, and twelve new test rodlets from General Electric (GE) are to be introduced into ATF-2 during the outage preceding ATR Cycle 166B.
- Design of a new 9-pin tier for ATF-2 was completed, which could effectively increase the rodlet capacity of the ATF-2 test train by 50% (a recommendation included in the Halden Gap Assessment).
- Uranium silicide fuels in ATF-1 reached burnups of ~40 GWd/MTU, the equivalent of two full cycles in a commercial power reactor; interim examinations to date have indicated that these silicide fuels exhibit very low swelling and fission gas release, which is a significant improvement over conventional fuels under normal operations in addition to benefits they promise to provide under accident scenarios.
- Analyses performed using the BISON fuel performance code showed remarkable agreement between predicted and measured fission gas release for the standard UO_2 fuels included as experiment controls, which serves to validate the design approach of the ATF-1 test series.
- Conceptual design of ATR I-Loops as recommended in the Halden Gap Assessment was initiated, with the objective of providing prototypic BWR test conditions, ramp testing, and run to failure testing capabilities needed to qualify ATF concepts without the Halden Reactor.


30000 LBS CAPACITY
Foreign Material Exclusion

- A series of commissioning tests were completed in Transient Reactor Test Facility using the Separate Effects Test Holder (SETH) irradiation capsule, demonstrating the function of all critical systems. These were the first transient fuel tests conducted in the U.S. in ~30 years and represent a significant leap forward in experimental technology.
- The development and deployment of the Minimal Activation Retrievable Capsule Holder (MARCH) sub-module was used to conduct the first fueled Replace with Reactivity-Initiated Accident (RIA) experiments in TREAT in a water environment, and the first RIA experiments were performed on an unirradiated ATF concept (U_3Si_2).
- An Advanced Reactor Fuel Workshop was co-hosted with Gateway for Accelerated Innovation in Nuclear (GAIN) at Boise State University in March 2019. Industry advanced reactor designers briefed the campaign on their fuel concepts and research needs, which will be incorporated into campaign R&D plans for the future.
- Conceptual design of the new Fission Accelerated Steady-state Test (FAST) irradiation experiment was completed, which will use miniaturized fuel rods to accelerate burnup accumulation by as much as a factor of 10, greatly reducing the time required for reactor testing.
- FAST fabrication trials on miniature fuel rodlets were successfully completed this year, including rodlets with integral Zr, V liners; the first FAST experiments with metallic fuel rodlets are scheduled for insertion in ATR during the 168A cycle (December 2019).
- The first MiniFuel irradiation in High Flux Isotope Reactor of both coated and uncoated nitride microspheres was completed and postirradiation examination was initiated. Visual exams, gamma-spectroscopy, and fission gas release data has been obtained. This capsule-based, separate effects approach to fuel testing promises to provide fundamental data that will be valuable to advanced fuel modeling efforts. Uranium silicide (U_3Si_2) specimens were fabricated at Los Alamos National Laboratory (LANL) and shipped to Oak Ridge National Laboratory (ORNL) for a subsequent MimiFuel irradiation in HFIR next year.
- Machining capabilities for trans-uranic-bearing metallic fuels were developed and deployed in the Fuel Manufacturing Facility in order to demonstrate many of the innovative design features developed in recent years, including sodium-free U-Pu-Zr metallic fuels as an advanced driver fuel for the VTR. The first two annular metallic fuel slugs of U-20Pu-10Zr were produced.
- Extensive characterization studies were conducted on metallic fuels with additives (Pd, Sn) for lanthanide fission product immobilization, which are showing great promise toward increasing the allowable cladding temperatures in metallic fuel pins. A major update to the Metallic Fuels Handbook was issued, incorporating a considerable amount of new data generated by the campaign.
- Good progress was made on the processing and characterization of nano-structured ferritic alloys, with a view to developing a new cladding for sodium fast reactors capable of higher temperature operation, and experimental fabrication of oxide dispersion strengthened (ODS) steels by extrusion/pilgering continued.
- Postirradiation examinations (PIE) were completed on X501, an full-size U-20Pu-10Zr metallic fuel pin with Np and Am additions that was irradiated in Experimental Breeder Reactor (EBR)-II to 6% burnup; this PIE data will be used in additional assessments of similar metallic fuels irradiated in cadmium-shrouded positions in the ATR.
- The first thermal property measurements on U-10Zr metallic fuels irradiated to over 10% burnup in Fast Flux Test Facility (FFTF) were obtained in the new Thermophysical Properties Cell at the Irradiated Materials Characterization Laboratory at Idaho National Laboratory (INL).
- The AFC-4C (Alternative Metallic Fuel Alloys Experiment) and AFC-3F (Metallic Fuels Fabrication Variables Experiment) continued irradiation in cadmium-shrouded positions in ATR throughout FY19; both experiments are nearing their 10% burnup targets.
- The Integrated Recycle Test (IRT) containing metallic fuels fabricated remotely using actinide materials recovered from spent fuel continued irradiation in cadmium-shrouded positions in the ATR throughout FY19, which will provide important information on the performance of recycled metallic fuels; peak fuel burnup was nearing 3% at year end.
- Considerable enhancements to the BISON fuel performance code were made for simulations of metallic fuels, including the addition of a new mechanistic fuel swelling model, which was used to investigate the deformation behavior of Na-free, slotted metallic fuel concepts.

1.3 SHOWCASE CAPABILITIES

TREAT Flowing Sodium Loop Cartridge

Principal Investigator: Nicolas Woolstenhulme (INL)
Team Members/ Collaborator: Bryce Kelly, Greg Core,Matthew Johnson, Cole Blakely, Cody Hale, Aaron Epiney, Brandon Dalley (INL); TerraPower, LLC.

The TREAT NLC will afford the domestic and international fuel safety testing community the unique capability to test various sodium-cooled fast reactor (SFR) fuel designs at prototypic conditions of a specific SFR design.

Historically, flowing sodium loop cartridges were critical in meeting advanced reactor fuel safety testing needs at the Transient Reactor Test Facility (TREAT) at the Materials and Fuels Complex (MFC). The recent resumption of the US fuel safety testing program created the need to reestablish this capability and enable further research efforts into the behavior of sodium fast reactor fuels in prototypic environments.

Project Description:

To successfully reestablish sodium-cooled fuel safety transient testing capability for the TREAT, Idaho National Laboratory (INL) initiated a project that includes developing requirements, establishing a test loop, designing a prototype loop, fabricating a prototype loop including Annular Linear Induction Pump (ALIP) development, and demonstrating in-laboratory prototype performance via an extensive testing program. In addition to the physical work discussed above, successful execution of this project requires significant historical data mining, and nuclear safety and TREAT interface/facility requirements gathering. INL chose TerraPower, LLC as a key collaborator in the execution of the project due to the organization's resource expertise in sodium-cooled reactors. TerraPower will design, fabricate, and operate the test and prototype loops allowing informative component-specific and integrated testing. The prototype loop thermal hydraulic performance test results will be used to finalize an equipment specification for a modernized, reactor-grade TREAT sodium test loop cartridge (NLC) design. Upon design finalization, the "reactor-ready" TREAT NLC will be fabricated, inspected, and qualified to support experimental fuel safety testing. The reactor-ready NLC will provide the capability to test metallic and oxide fuels in a heated flowing sodium environment supporting both domestic and international testing needs.

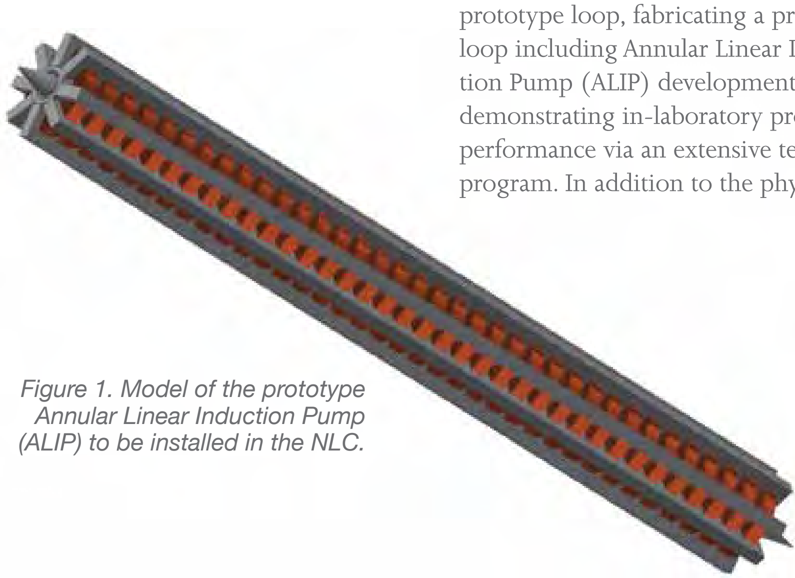


Figure 1. Model of the prototype Annular Linear Induction Pump (ALIP) to be installed in the NLC.

Accomplishments:

Fiscal year 2019 included numerous high-level goals for the project such as completing a thorough requirements gathering phase, creating a functional and operational requirements (F&OR) document for the prototype and reactor-ready NLCs, performing equipment trade studies, establishing relationships with vendors of critical components, beginning fabrication of the test loop, and completing a pre-conceptual design and initiating a conceptual design for the reactor-ready NLC.

The project team kicked off project execution in March of 2019 at TerraPower's facilities in Bellevue, WA and quickly organized a requirements gathering week to be held at INL facilities in April. Requirements gathering sessions were attended by representatives from postirradiation examination (PIE), TREAT Operations, TREAT Engineering and Nuclear Safety, Advanced Fuels Campaign (AFC) domestic and international experimenters, packaging and transportation, and others. These sessions yielded a draft F&OR for the prototype and reactor-ready NLCs and was subsequently distributed for review by all attendees. The final F&OR was issued to support downstream design activities on July 9, 2019.

Supply for the equipment and instrumentation supporting this specific testing platform all but disappeared due to the suspension of the fuels safety testing program at TREAT in the early 1990's. The project identified reestablishing the supply chain for these items as a critical short-term goal. In fiscal year (FY) 19 the project established relationships with two vendors to support ALIP and pressure transducer supply. Bryce Kelly (INL), TREAT NLC's Project Engineer, supported establishing the ALIP vendor relationship by reverse engineering and redesigning the Mk-III ALIP technology (Figure 1). Additional equipment trade studies identified a gap in electromagnetic (EM) flowmeters which can support the NLC's small form factor. In

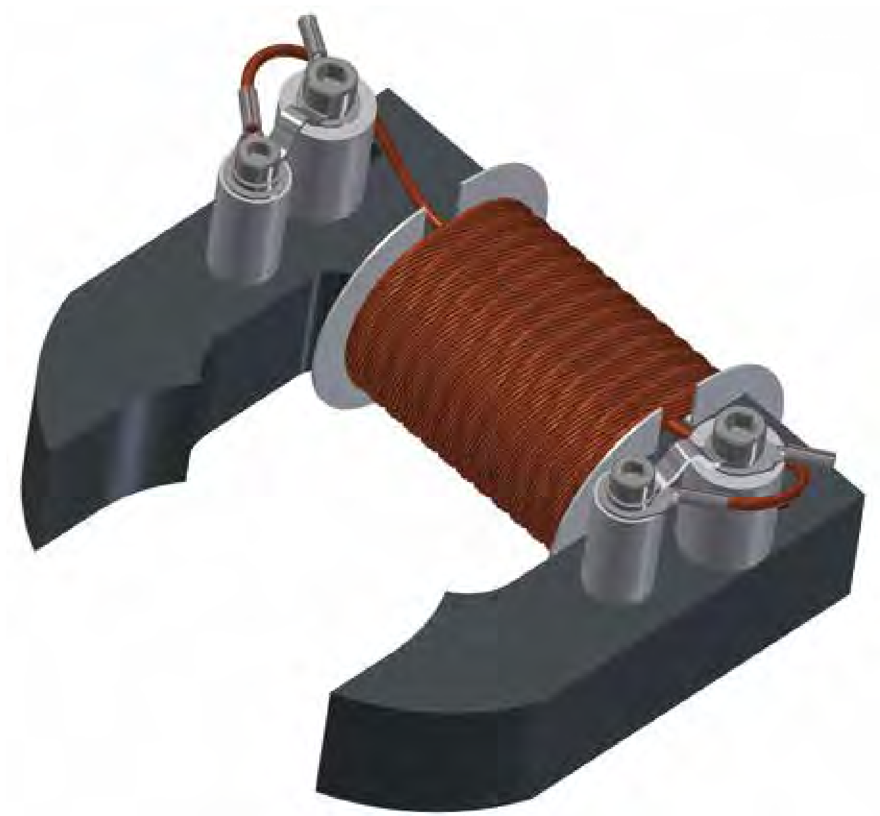
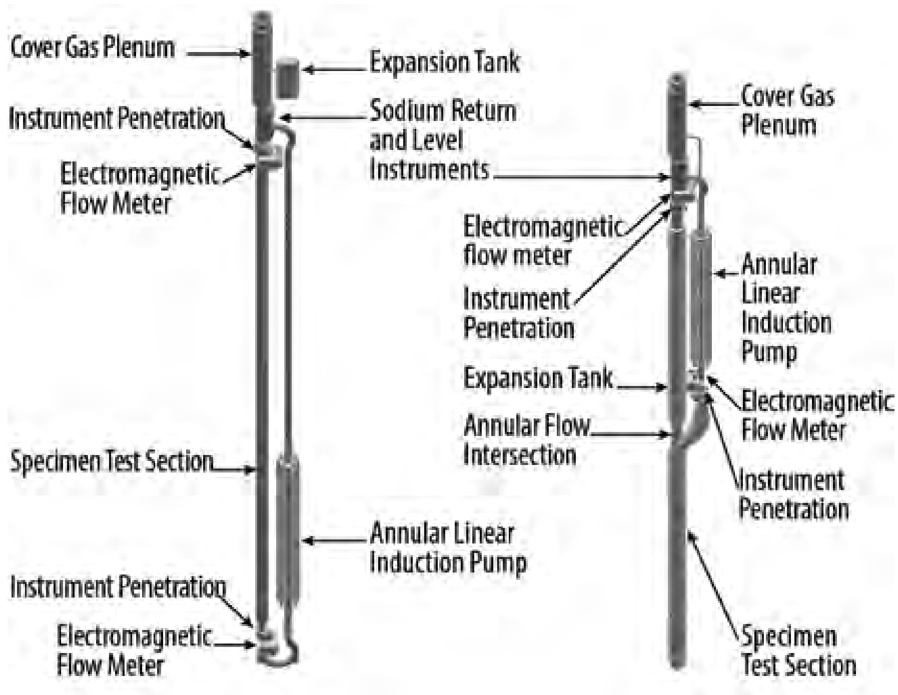


Figure 2. A schematic of the ATF-2 test train and fuel pin holders.

Figure 3. TREAT NLC Mk-IIIR (left) and Concept 4 designs (right).



response, TerraPower contracted the University of Wisconsin to investigate applying their current research in permanent magnet flowmeters to the NLC and INL embarked upon recreating the EM flowmeter technology used in the legacy loops. Figure 2. shows the EM flowmeter planned to be built at INL and installed in the test loop at TerraPower facilities.

The project completed the pre-conceptual design of the reactor-ready NLC in May of 2019 and produced two promising design options, the Mk-IIIR and the Concept 4. The Mk-IIIR entails re-fabricating the design last used in TREAT with modern materials and an updated pump design. In contrast, the Concept 4 utilizes an annular test section design allowing simpler experimental analyses and possibly reducing the cost to experimenters. Figure 3 depicts the Mk-IIIR (left) and Concept 4 designs (right). These two designs will be compared through detailed analyses and limited fabrication studies in the conceptual design phase of the project. In fact, the project initiated part of those fabrication studies in August of 2019 on a unique component, the annular flow intersection, of the Concept 4 NLC design. Structural analyses indicated that as designed, the annular flow intersection would not meet stress limitations under pressure and temperature design conditions; machining this component from one block of material would alleviate that concern by retaining extra material in the regions of maximum stress concentration. As of the end of August 2019, the component has been cut from a solid block of aluminum to actual dimensions (Figure 4). Remaining machining is expected to finish in October of 2019.

Figure 4. INL Machinist Brandon Dalley (left) and INL Project Engineer Bryce Kelly (right) with aluminum annular flow intersection component.

Development of Chemical Vapor Infiltration (CVI) for SiC/SiC Composites

Principal Investigator: Brian Jolly, ORNL
Team Members/ Collaborators: Yutai Kato, Austin Schumacher, Richard Lowden and Kevin Cooley (ORNL)

A new experimental CVD/CVI system has been designed, installed, and demonstrated allowing the fabrication of lab-scale SiC/SiC composite test specimens.

This research focuses on the design, installation, and testing of a new experimental chemical vapor deposition (CVD)/ chemical vapor infiltration (CVI) system allowing for the fabrication of ceramic matrix composites. The goal of this system is to not only produce traditional SiC/SiC composites for testing, but to also be adaptable enough to facilitate development work using a variety of precursor materials working to improve ceramic composite materials performance in nuclear applications. For this work, functionality of the equipment was demonstrated by infiltrating Nicalon silicon carbide fibers with a CVI derived matrix. Future work will include utilizing carbon, oxides and other high temperature materials as an interface layer to control bonding between the fiber and matrix material, increasing run duration/optimizing processing conditions to increase composite density, and refining graphite preform and coating chamber design.

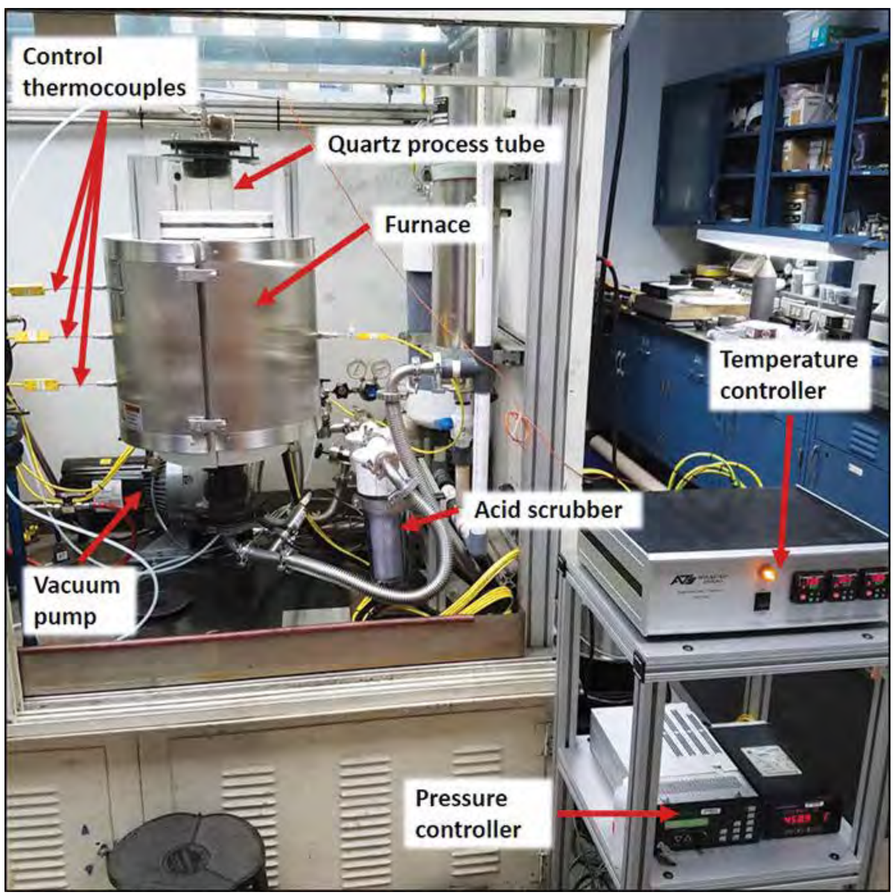


Figure 1. Overview of experimental CVD/CVI system.

Project Description:

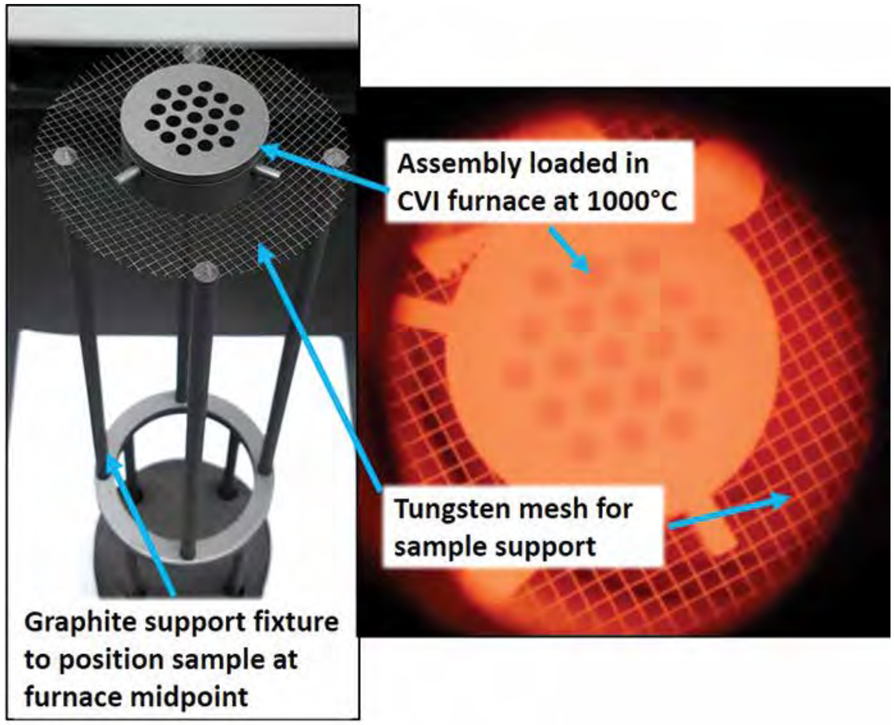
Several accident tolerant fuel (ATF) concepts have been proposed to improve fission product retention in the event of a beyond design basis accident. One concept uses a ceramic as fuel cladding and requires the ability to fabricate toughened ceramic materials. CVD and CVI are versatile and scalable technologies that enable advanced ATF cladding and core structure concepts through manufacturing of ceramic matrix composite and/or depositing overcoatings. Development of coating technologies are imminent needs for SiC composite-based claddings and core structures to ensure environmental protection in oxidizing water chemistry. Ensuring hermeticity against gaseous fission products release is an additional essential need for SiC composite-based claddings. Moreover, seal-coating technology for end-plugging is another pressing need for the ceramic claddings. In a longer term, development of improved fiber-matrix interfaces that are more oxidation-resistant in both water and steam environments than the current pyrolytic carbon technology is desired for future SiC composite-based light water reactor (LWR) claddings and core components. In addition, modification of the fiber-matrix interfaces is essential for the development of high dose radiation resistant composite for advanced reactor applications. With these future objectives, the current effort focuses on establishing a CVD/CVI capability that offers adequate configurational versatility and flexibility for precursor species and deposition conditions.

Figure 2. 25mm diameter SiC/SiC composite fabricated with new lab scale CVI system.

Accomplishments:

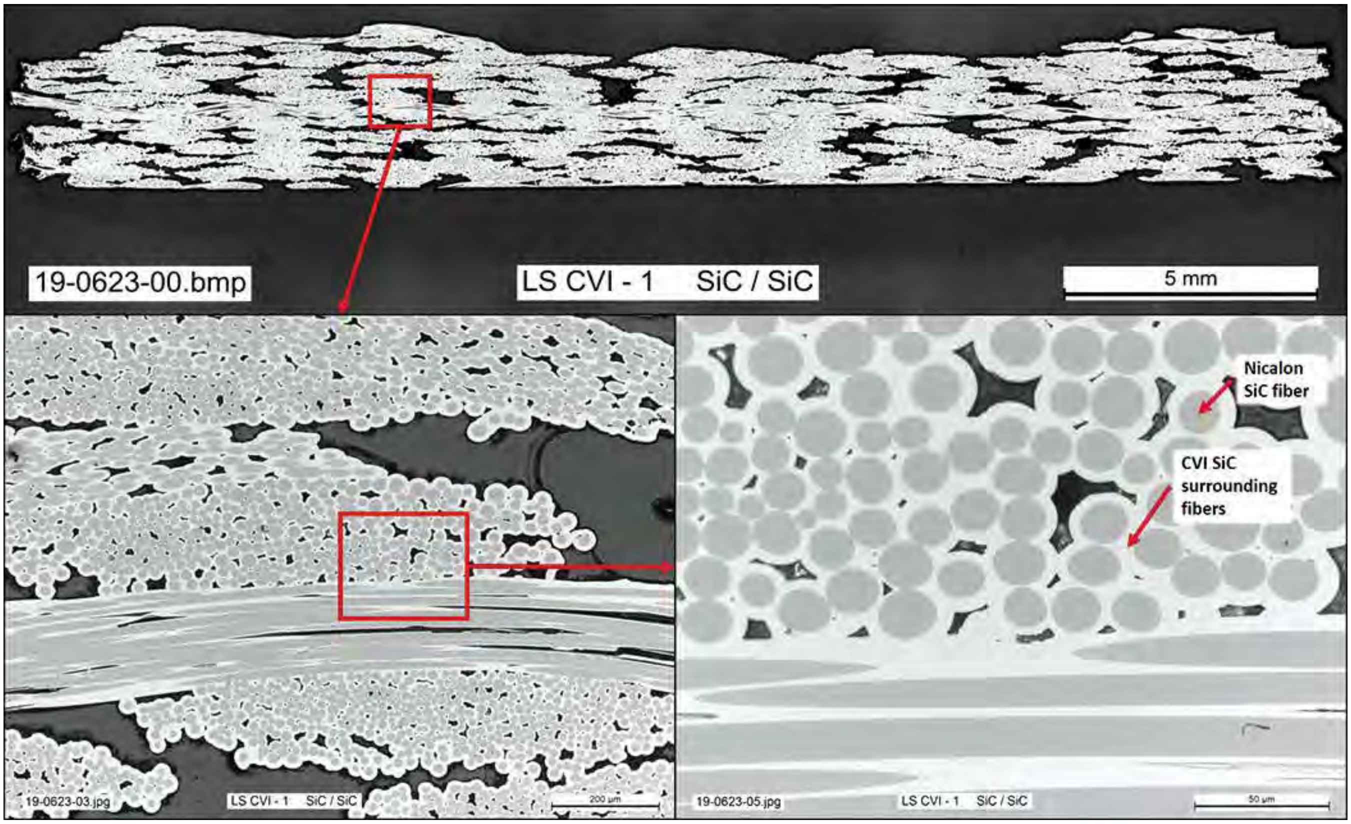
An experimental CVD/CVI system (Figure 1) has been designed, installed and demonstrated at Oak Ridge National Laboratory (ORNL) allowing

Figure 3. Graphite sample holder with SiC fiber preform sitting on graphite support fixture before and during processing at 1000°C.



the fabrication of lab-scale SiC/SiC composite test specimens. Maximum sample size for this system is ~75mm in diameter with initial tests being performed using 25.4mm in diameter disk shaped specimens consisting of multiple layers of Ceramic Grade (CG) Nicalon fabric. The experimental CVD/CVI system uses a hot-wall type configuration with a vertically oriented quartz process tube heated by a resistive 3-zone tube furnace. The quartz process tube is sealed by stainless steel flanges providing atmospheric control and serves as the coating chamber which houses the test samples. Process gasses are delivered via electronic mass flow controllers (MFCs) with a vacuum control system providing stable pressure control from ~10 Torr to atmospheric pressure. Effluents (namely HCl) from the SiC deposition process are neutralized via dry scrubber before discharge to building ventilation.

To demonstrate functionality of the CVI system, a silicon fiber composite test disk was successfully fabricated (Figure 2). Plain weave Nicalon SiC fabric from Dow Corning was used to form the SiC composite fiber preform for the first infiltration test. The SiC fabric was first cut into circular pieces 25.4mm in diameter before



being loaded into the graphite fixture. Methyltrichlorosilane was used as the SiC precursor gas with a processing temperature of 1000°C. Figure 3 shows the graphite sample holder (which houses the SiC fiber preform) sitting on the graphite support fixture before loading and during processing at temperature within the furnace. Cross sectioning and optical imaging of the SiC fabric composite Figure 4 shows successful infiltration of the CVI derived SiC matrix around the SiC fibers. Longer runs will be performed in the future to increase densification.

Figure 4. Cross section of SiC fiber composite showing successful infiltration with CVI derived SiC.

Fabrication of Small-scale Ceramic Fuels for Accelerated Fuel Qualification

Principal Investigator: Joshua T. White, LANL
Team Members/ Collaborators: John Dunwoody, Chris Grote, Najeb Abdul-Jabbar, Tarik Saleh, Ken McClellan (LANL)

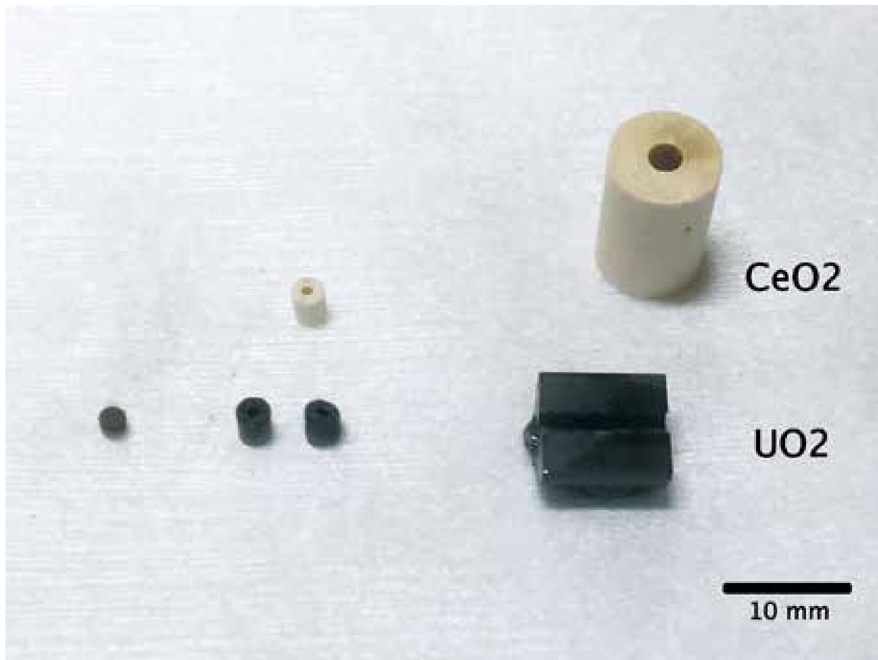


Figure 1. Demonstration of accelerated burnup test specimen geometries composed of CeO_2 and UO_2.

Interest has expanded in recent years within the Advanced Fuels Campaign (AFC) to accelerate the irradiation testing of nuclear fuels to allow a more efficient path forward to assessing the performance of fuel concepts to higher burnup. Expanding the datasets to higher burnups would improve the overall safety basis of these concepts, potentially expanding the lifetime of current generation fuels and improve the economics of the current fleet. In current irradiation testing schemes (i.e., Advanced Test Reactor (ATR)) the test articles would require on the order of 10 years to achieve ~60 MWD/tHM. Small geometry irradiation test experiments have been proposed to overcome this lengthy irradiation time frame. To this end, two irradiation approaches have been identified that would drastically shorten the irradiation schedule to on the order of a couple years although fabricability and benchmarking against historic test data will be required to validate the methodology.

Project Description:

This project aims to assess the fabricability of the two concepts outlined above. The first utilizes an integral effects testing approach at ATR where the enrichment is increased while the size is reduced (1/2, 1/3 and 1/4 scale for light water reactor (LWR) scaled pellets and 1/2 scale for fast reactor designs) to match the overall linear heat generation rate of a standard sized fuel pellet. This presents challenges not only in the ability to fabricate dense pellets with the prescribed geometries, which are anticipated to be ranging from 3 mm to 10 mm diameter in size and either solid monoliths or annular to accommodate both thermal and fast spectrum fuel designs, but also have a custom enrichment up to 30% requiring increases in authorization basis. The second approach is the MiniFuel concept, which utilizes a much smaller specimen geometry on the order of 3 mm in diameter by 0.3 mm to minimize irradiative heating in the specimen allowing for a separate effects style investigations. Both concepts present their own challenges for fabricability and will also require benchmarking against historic irradiation test data to validate the method to license a concept. However, both approaches could provide a substantial shift in the overall approach to irradiation testing of fuels in the US nuclear program. To this end, research this fiscal year (FY) has focused on fabricating ceramic nuclear fuels to the appropriate geometries described above. Initial attempts were demonstrated on a non-radiological surrogate, CeO_2, followed by fabrication studies on UO_2 and U_3Si_2.

Accomplishments:

Research at the Fuels Research Laboratory (FRL) this year has shown progress in the field of small scale test specimens for both the accelerated burnup experiments as well as for the MiniFuel irradiation test concept. Specimen preparation for the accelerated burnup experiments was able to yield >90% dense pellets for monolithic and annular geometries using UO_2, CeO_2, and U_3Si_2. Some examples of successful pellets are shown for CeO_2 and UO_2 in Figure 1 for the maximum and minimum pellet dimensions demonstrated in this

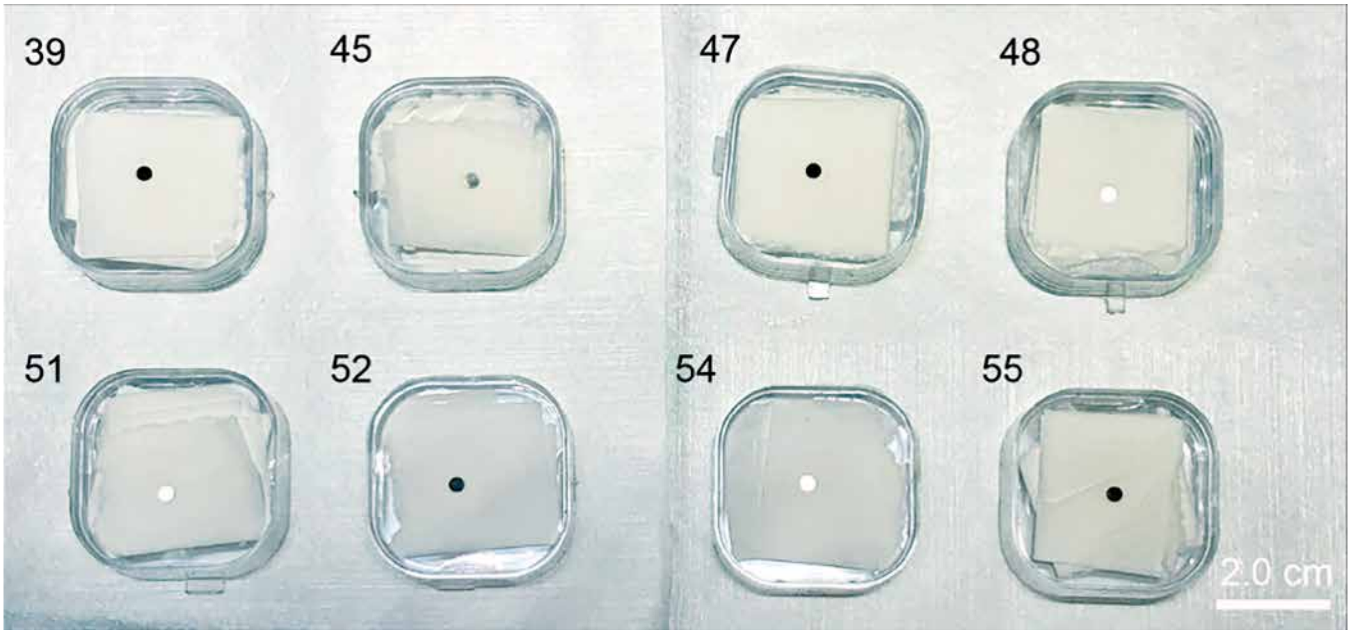


Figure 2. U_3Si_2 MiniFuel U_3Si_2 test specimens that have been prepared at LANL and shipped to ORNL for testing in HFIR.

project. Pressing defects were apparent particularly for the smallest geometry annular pellets that were 3mm in diameter with a 1 mm annular hole through the center; however, the majority of these defects were located near the punch face that could be removed from the ends to yield high density pellets with minimal defects present. Further refinements to the punch and die designs are being considered to minimize the loss observed in the high burn up pellet fabrication process, which will be considered next FY. Fabrication efforts will be useful for future irradiation experiments at ATR for both thermal and fast spectrum application of fuels.

In FY19, great strides were taken to increase the envelope of enriched fuel processing and fabrication at the Fuels Research Laboratory (FRL) at Los Alamos National Laboratory (LANL). This process was initiated in FY18 with the establishment of a standalone

Demonstration of novel small scale pellet fabrication at LANL enables separate and integral effects testing of nuclear fuels of interest for model development and assessment within the Advanced Fuels Campaign.

authorization basis and material inventory for the FRL which increased the overall allowable fissile uranium by 85%. All work authorization documents (IWDs) pertaining to highly enriched uranium (HEU) processing and the operation procedure (OP) specifically addressing accountable material processing, have been revised or created allowing custom enrichments for next FY fabrications. Technical challenges were also overcome this year fabricating depleted U_3Si_2 pellets to the MiniFuel geometry specifications, shown in Figure 2. Eight high density MiniFuel pellets >93% dense of U_3Si_2 were successfully fabricated and shipped to Oak Ridge National Laboratory (ORNL) for High Flux Isotope Reactor (HFIR) irradiations which continues upon FY18 efforts that demonstrated the ability to fabricate oxides. Based on the above investigations, it is anticipated that the FRL at LANL is well suited to accommodate future accelerated irradiation testing within the campaign for a variety of current and next generation fuel forms.

Assessment of Feedstock Synthesis Routes for High Density Fuels

Principal Investigator: Scarlett Widgeon Paisner, LANL
Team Members/ Collaborators: Joshua T. White, Nicholas R. Wozniak, Blake P. Nolen, Joseph R. Wermer (LANL)

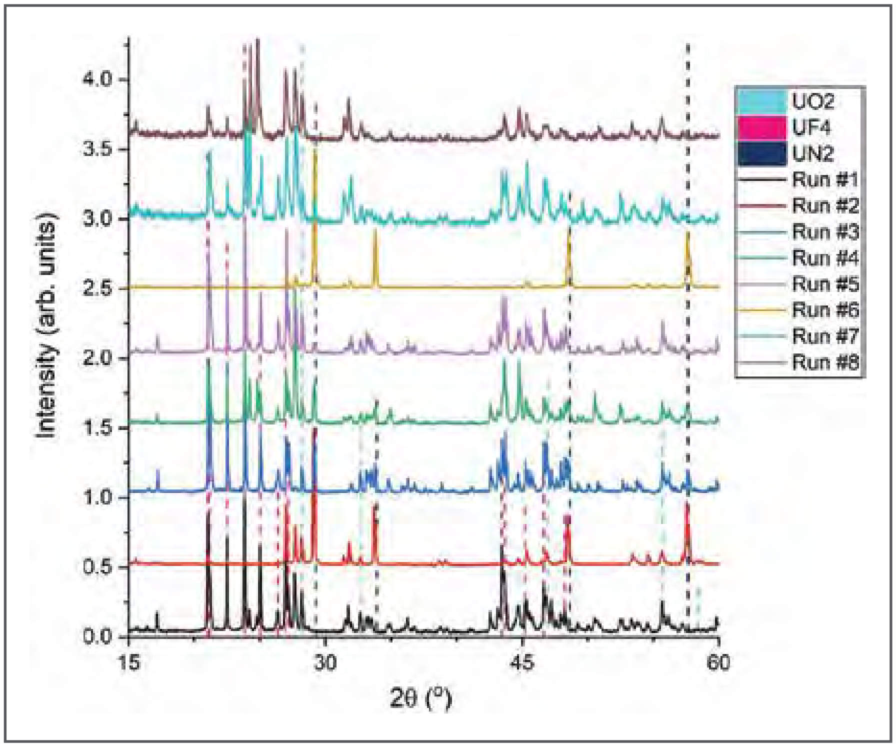


Figure 1. XRD patterns of powders after ammonolysis of UF4 powders under ammonium gas for eight runs under various conditions.

Uranium mononitride is a promising candidate for next generation, accident tolerant nuclear fuels due to its high uranium density, thermal conductivity, and melting point. The current synthesis routes for uranium nitride (UN) include (1) a carbothermic reduction of UO_2 then a subsequent nitridization (Carbothermic Reduction and Nitriding: CTRN) to produce UN or (2) hydriding uranium metal followed again by nitridization to form UN. One of the many drawbacks for these two processes is the additional processing required to make the starting materials (either UO_2 or U metal), where the starting feedstock for these materials is uranium hexafluoride (UF6). Rather than using either of the two synthesis routes described above, it would be beneficial to synthesize UN powders directly from the UF6. Here, a route known as ammonolysis is explored to convert uranium tetrafluoride (UF4) to uranium dinitride (UN2), which can be further reduced to obtain uranium mononitride.

Project Description:

The objectives of this research program is to investigate the feasibility of alternative synthesis processes for UN fuel. Here, ammonolysis of UF4 is proposed to streamline the synthesis process and will allow for a more direct pathway for UN production in comparison to other synthesis routes that have been established, such as CTRN. UF4 can be obtained from UF6 by reducing under hydrogen atmosphere, therefore eliminating the necessity to synthesize either UO_2 or U metal as the starting material can be bypassed. After the reduction process, UN2 is produced by ammonolysis, and can then be decomposed to form UN. In the past year, preliminary work on the ammonolysis step has begun and success has been achieved thus far.

Compared to the conventional CTRN process, this three-step approach expedites the synthesis of UN and also makes it more economically feasible due to reduced processing time and temperatures that are required in CTRN. Additionally, the ammonolysis approach minimizes the concentration of oxygen and carbon impurities, since these two elements are not in the starting materials and therefore is only introduced by unintentional contamination. This yields a more pure final product and enhancing the sinterability of UN powders. This research aligns well with the Department of Energy (DOE) objectives of providing accident tolerant fuels and enhancing the economics of the UN fuel production process.

Accomplishments:

During fiscal year (FY)19, focus has been placed on optimization of the experimental parameters of the ammonolysis process. In order to accomplish this, different temperatures, pressure of ammonium, and gas flow profiles were used. To carry out these experiments, UF4 powder was loaded

Results from this project present an alternative, commercially viable processing route to produce UN from common uranium feedstocks found in industry.

onto an Inconel 600 boat and loaded in an Inconel 600 reaction vessel. All sample preparation was carried out in under argon in a glovebox to avoid oxygen contamination of the UF4 powder. The reaction vessel was then transferred to a clamshell furnace, connected to the ammonium gas line, and heated to various temperatures and reactant flow conditions. After heat treatment, structural characterization was performed using X-Ray diffraction (XRD) for the first eight experiments, which are shown in Figure 1, and are plotted along with the standard XRD peaks for UF4, UO_2, and UN2. The results show that UN2 was in formed in runs #2-7, however, the reaction was not completed, as UF4 was still detected in the XRD patterns for several of the runs. In particular, runs #2 and 6 produced UN2 with relatively high yield, with another unidentified phase present in small fractions. The temperature for these two successful runs was set to 1050 and 913 °C, indicating that this process can be achieved in this narrow range of temperatures. It was also observed that these two runs were carried out under slightly above ambient pressure of the ammonium in the reactor vessel in comparison to the other six runs. It is therefore assumed

that the pressure of the system is key in determining optimal parameters for formation of UN2 powders by ammonolysis. Diffraction peaks for UO_2 are also observed in all eight runs, and this is presumed to be oxidation of the UF4 starting materials, which persist after heat treatment.

ADVANCED LWR FUELS

2.1 ACCIDENT TOLERANT FUELS

ATF Industry Advisory Committee

Committee Chair: Bill Gassmann, Exelon
Collaborators: Steven Hayes, Ed Mai, Kate Richardson (INL)

The Advanced LWR Industry Advisory Committee (IAC) 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 for commercial light water reactors. The IAC is comprised of recognized leaders from the commercial light water reactor industry. They represent the major suppliers of nuclear steam supply systems, owners/operators of U.S. nuclear power plants, fuel vendors, Electric Power Research Institute (EPRI), and Nuclear Energy Institute (NEI). Members are selected on the basis of their technical knowledge of nuclear plant and fuel performance issues as well as their decision-making authority in their respective companies. During the past year the committee provided important industry perspectives relative to the campaign's interactions with the Nuclear Regulatory Commission (NRC) on issues related to activities supporting the eventual licensing of accident tolerant fuels and the ramifications of the Halden Reactor shutdown.

The IAC meets monthly via teleconference and is currently chaired by William Gassmann of Exelon Corporation. Additional members represent Westinghouse Electric Company, Global Nuclear Fuels, AREVA, Dominion, Duke Energy, Southern Nuclear, EPRI, and NEI.

AREVA

Exelon

NEI NUCLEAR ENERGY INSTITUTE

Dominion Energy

EPRI ELECTRIC POWER RESEARCH INSTITUTE

Southern Company

DUKE ENERGY

GNF Global Nuclear Fuel
A Joint Venture of GE, Toshiba, & Hitachi

Westinghouse

Accident Tolerant Fuel (ATF) Industry Teams – Principal Investigator: Westinghouse Electric Company LLC

Principal Investigator: E. J. Lahoda, Westinhouse Electric Company LLC
Team Members/ Collaborators: Westinghouse Electric Company LLC, General Atomics (GA), Massachusetts Institute of Technology (MIT), Idaho National Laboratory (INL), Los Alamos National Laboratory (LANL), Exelon Nuclear, University of Wisconsin (UW), National Nuclear Laboratory (United Kingdom) (NNL), Army Research Laboratory (ARL)/VRC/MOOG, University of Virginia, University of South Carolina, Oak Ridge National Laboratory (ORNL), Fauske & Associates, Rensselaer Polytechnic Institute (RPI), University of Texas at San Antonio, Texas A&M University (TAMU), Air Liquide (AL), Free Form Fibers (FFF).

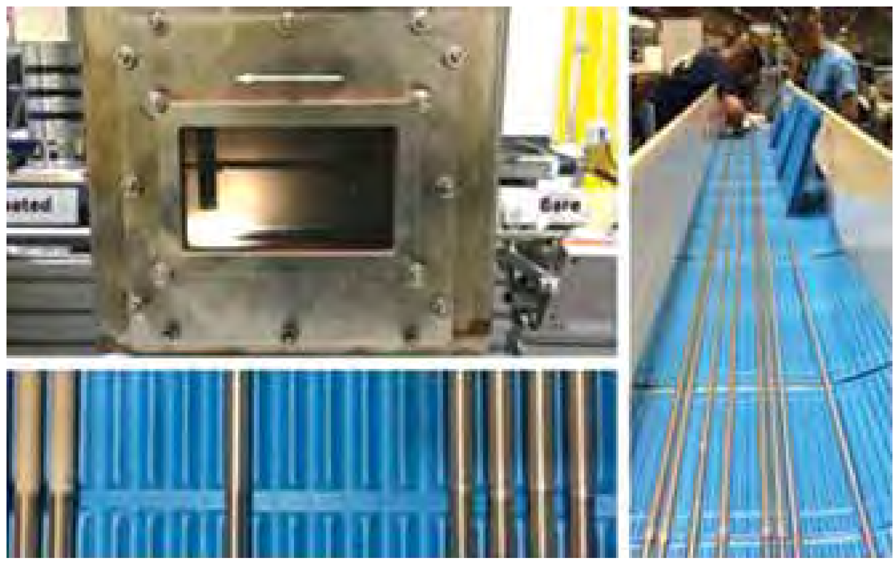


Figure 1. Rods being cold spray coated.

Westinghouse is developing two unique accident tolerant EnCore®* fuel (ATF) designs: SiGATM* silicon carbide (SiC) cladding with uranium nitride (UN) fuel, and chromium-coated zirconium alloy cladding with ADOPTTM* and UN fuels.

Project Description:

Lead test rods (LTRs) were loaded into the Byron Unit 2 reactor on April 23, 2019, which consisted of Cr coated rods with ADOPT pellets with Cr coated cladding and Zr cladding with U_3Si_2 pellets. SiC and Cr coated Zr claddings are planned to be used with ADOPT pellets in the Advanced Test Reactor (ATR) in early 2020 and with Mo pellets in BR-2 as part of the Il Trovatore program in 2020.

The immediate tasks are aimed at design and licensing with the required experimental backup to obtain U.S. Nuclear Regulatory Commission (NRC) approval for insertion of lead test assemblies (LTAs) in 2023 of Cr coated Zr with ADOPT UO_2 and >5%

Figure 2. Byron Unit 2 assemblies with EnCore fuel loaded.

enriched ADOPT pellets. Additional tasks include:

- Develop oxidation resistant UN and production technologies;
- Develop low cost methods for manufacturing SiC and Cr coated rods;
- Accelerated in-reactor and out-of-reactor testing to validate the MAAP and MELCOR codes to determine potential operational savings and to support licensing changes;
- Continue NRC licensing interactions aimed at implementation of regions in 2023 and 2027.

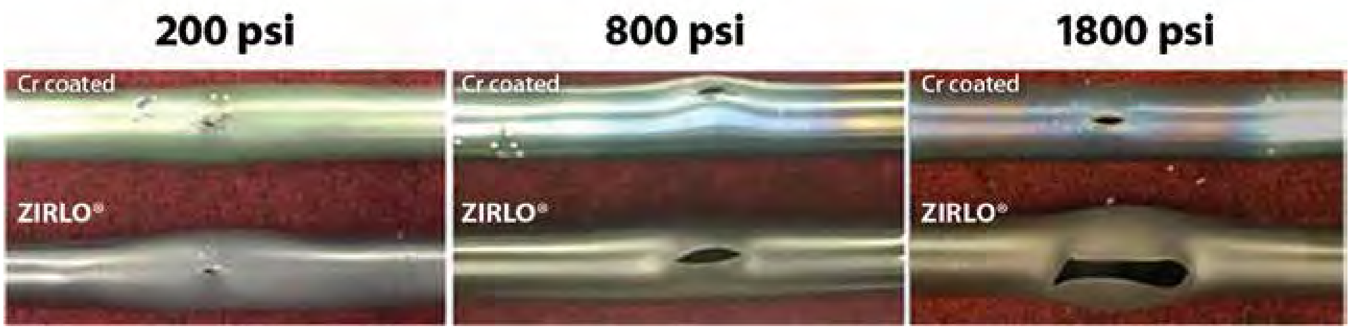


Figure 3. Burst hole-size comparisons of coated (top) and uncoated (bottom) cladding at various pressures.

Accomplishments:

The first ever loading of U_3Si_2 in zirconium cladding and ADOPT pellets in Cr coated cladding (Figure 1) was accomplished. The EnCore ATF LTRs (Figure 2) were loaded into Byron Unit 2 for plant startup on April 23, 2019.

Tests are continuing at the Massachusetts Institute of Technology reactor (MITR) on the Cr coated Zr and SiC cladding options as well as the in-rod sensor that is being developed by Westinghouse to support the ATF testing and licensing process. These tests are being run at pressurized water reactor (PWR) coolant conditions (B at ~1500 ppm, Li at 5 to 7 ppm, and H2 at 40 to 50 cm3/kg coolant). The results indicate minimal corrosion of the Cr coated Zr samples and minimal to moderate corrosion of the SiC, depending on the manufacturing conditions of the SiC and the manufacturer.

General Atomics (GA) has made significant strides in producing rodlets that consistently meet corrosion and hermeticity requirements. In addition, they have installed the production capability to manufacture 12-foot long SiC rods which
is currently undergoing shakedown testing.

The Westinghouse Advanced Modeling and Testing Technology programs have made significant progress in supporting licensing in

Westinghouse EnCore® Fuel is "game-changing" for the nuclear industry, significantly increasing safety margins in severe accident scenarios, increasing flexibility for fuel management and enabling higher burnups for longer fuel cycles which can lower operating costs.

an effort to decrease the time and cost of licensing the various ATF products. Atomic scale modeling technology, which utilizes first principles to determine physical properties of irradiated materials, is being applied to U_3Si_2 and SiC. In addition, Westinghouse remains engaged with the Department of Energy (DOE)-sponsored Nuclear Energy Advanced Modeling & Simulation (NEAMS) and Consortium for Advanced Simulation of Light Water Reactors (CASL) programs to provide additional insights into new material behaviors and performance in advanced applications.

Westinghouse in-rod sensing capability has significantly advanced with the testing of the sensors at MITR. While this capability was not employed in the current round of LTRs and test reactor activities, future applications in 2020 and 2021 are envisioned to gather real-time in-reactor test data to verify the atomic scale modeling that has been developed.

ATF Industry Teams - Framatome

Kiran Nimishakavi, Framatome
Team Members/ Collaborators: Idaho National Laboratory (INL), Oak Ridge National Laboratory (ORNL), Southern Nuclear Operating Company, Kernkraftwerk Gösgen-Däniken, Entergy Nuclear, Exelon Nuclear

Framatome's approach is to maximize levels of safety by addressing both the fuel and the cladding together through multiple, synergistic technologies for near-term evolutionary and longer-term revolutionary solution.

Framatome is continuing significant research and development efforts to develop accident tolerant fuel (ATF) technologies. Framatome's near term Enhanced Accident Tolerant Fuel (EATF) solutions are currently available to U.S. utility customers while longer term ATF concepts continue development. Both aim to provide operating margin and safety benefits through the ATF features.

Project Description:

The ultimate goal of Department of Energy (DOE's) EATF program is to develop an improved and more robust nuclear fuel design that will reduce or mitigate the consequences of reactor accidents and improve the economics reactor operations. After extensive testing, evaluation and downselection during the program's first phase, Framatome's Phase II technical approach addresses three focus areas: (i) Chromium (Cr)-coated cladding, (ii) Chromia-doped UO_2 fuel pellets, and (iii) Silicon carbide (SiC) composite cladding.

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 properties. Over the course of the EATF program, extensive processing and testing activities are being carried out in support of delivering EATF Lead Test Assemblies (LTAs), with two sets of LTAs successfully delivered in 2019, and further in support of batch implementation by the mid-2020s.

At the fuel level, chromia-doped UO_2 pellets can improve the pellet washout behavior after cladding breach and reduce fission gas release. To date, the performance of this fuel has been extensively studied in out-of-pile and in-pile test programs and modifications are being implemented to accommodate chromia-doped fuel in Framatome's fuel performance code.

For revolutionary (over-the-horizon) performance improvements, Framatome is developing a composite cladding comprising a silicon carbide fiber in a silicon carbide matrix (SiCf/SiCm). The objective is to develop a system which does not suffer from the same rapid oxidation kinetics of zirconium-based cladding while having attractive operating features such as reduced neutron absorption cross-section and higher mechanical strength at accident temperatures.

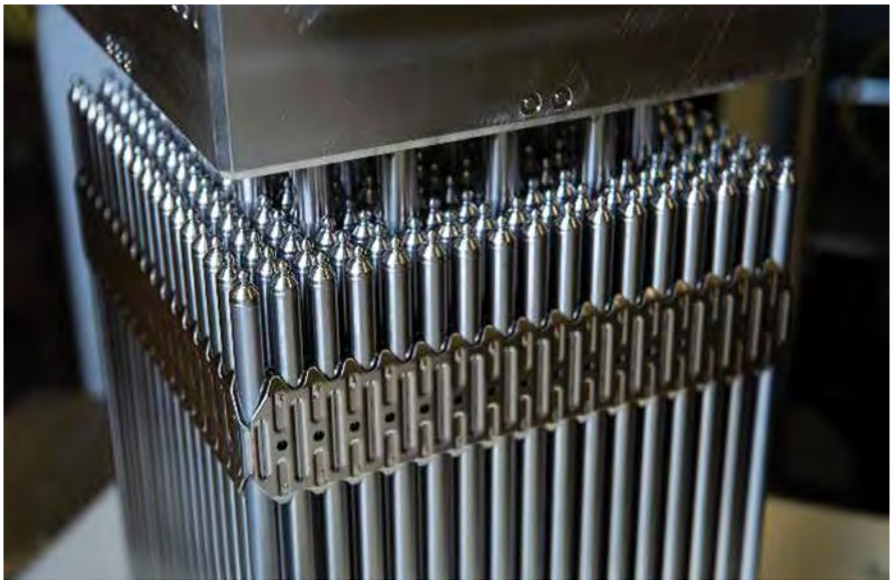


Figure 1. EATF Fuel Assembly for Vogtle Unit-2.

Accomplishments:

Framatome has continued to make substantial progress thus far in government fiscal year (GFY) 2019. The near-term solutions of chromium coated cladding and chromia-doped UO_2 fuel pellets saw significant project achievements in 2019 in both testing and delivery of lead test assemblies to US commercial reactors. Development of a more revolutionary ATF concept, composite cladding containing silicon carbide fiber, also continued with testing activities.

In January 2019, Framatome's Horn Rapids Road (HRR) facility in Richland, WA shipped EATF LTAs to Vogtle for operation in Unit -2. The LTAs feature the Framatomes GAIA W17 fuel design with chromium-coated cladding and Cr2O3-doped fuel pellets (Figure 1). Irradiation of the four LTAs began in the spring.

During third quarter GFY 2019, eight EATF LTAs were built for Entergy's ANO Unit 1 plant. The eight LTAs each have two full-length chromium-coated fuel rods containing UO_2 fuel (see Figure 2). An additional 16 full-length, inert, chromium-coated fuel rods were also fabricated. These 16 rods have stainless steel pellets in place of UO_2 fuel pellets and are scheduled

Figure 2. ANO EATF LTAs with two chromium-coated rods in corner of outer row.

to be inserted into irradiated fuel at ANO Unit 1 and operated on the fuel baffle. Manufacturing of the LTAs and the inert rods was completed in May in at HRR. The LTAs were shipped to ANO in August 2019 and are scheduled to begin irradiation in the fall of 2019. Fuel analysis supporting insertion of the Chromium-coated rods into ANO Unit 1 was also completed in the third quarter of GFY 2019.

As part of Framatome's on-going in-pile testing program, onsite visual inspections were performed on IMAGO Material Test Rods during the Gösgen outage in June 2019 after three irradiation cycles. High definition (HD) cameral visuals enabled observation of extremely fine features on the surface of the cladding and it was therefore used to evaluate the evolution of potential indications observed on the Chromium-coated cladding surface after two and three irradiation cycles.

Figure 3 shows the comparison of the same region after two and three irradiation cycles. The region is characterized by the presence of very small superficial linear indications, highlighted by the red arrows. These indications are only surface features but it was confirmed that the observations are of the same region. Figure 3 demonstrates that small indications such as the ones highlighted here do not evolve with irradiation, as they are exactly the same after two and three cycles. No delamination or deterioration of any kind was observed between two and three cycles. This shows that the Chromium-coating is stable under irradiation, even when investigated with the very high precision HD camera.

The only difference observed between two and three cycles is the change in color, shifting from gold to purple. This color variation is also observed during out-of-pile testing and is linked to the $Cr2O3$ growing at the nanometer level. When the oxide thickness remains below about 1µm, the color will evolve and change with the growing oxide thickness. This is in agreement with the out-of-pile corrosion tests showing very little variation in weight gain confirming the small amount of oxide formed for the Cr-coated cladding.

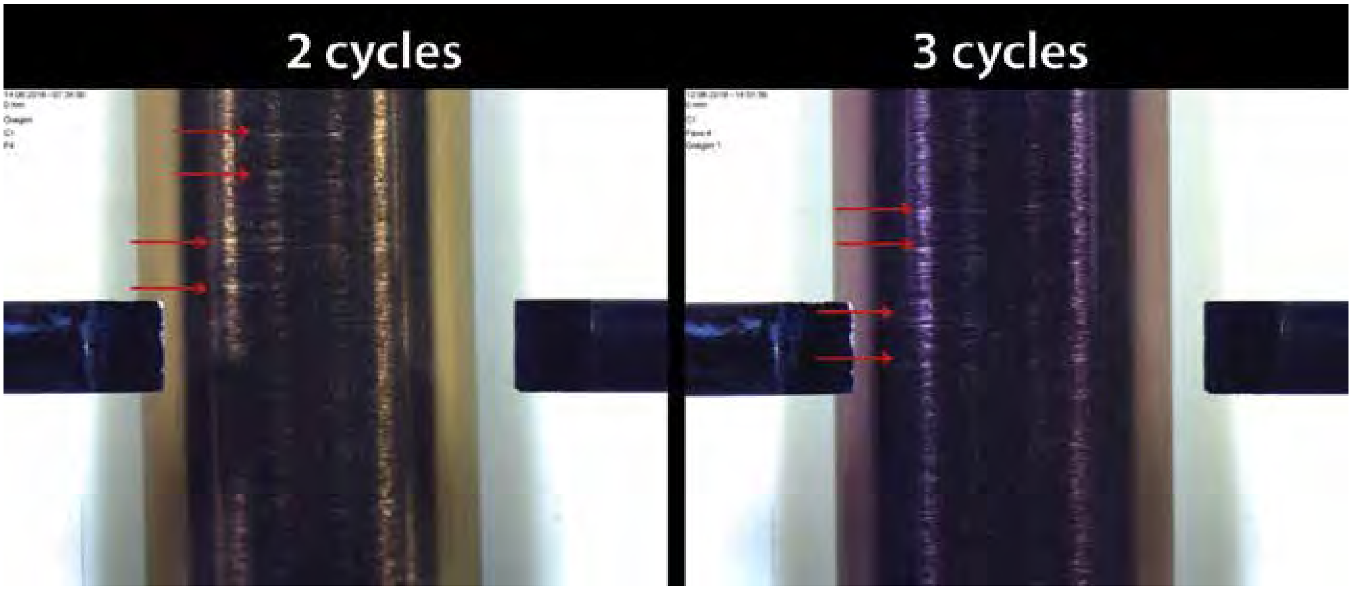


Figure 3. Comparison of a Cr-coated sample after two and three irradiation cycles showing a change in color.

During the same outage, 20 Chromium-coated lead test rods were inserted in two fuel assemblies in the Gösgen reactor as part of the GOCHROM project.

In GFY 2019, Framatome has contracted to provide two full EATF LTAs to the Calvert Cliffs Nuclear Plant #2 in early 2021. Design and qualification of this Cr-coated cladding is underway. This test program will allow for the evaluation of any interaction between a full bundle of EATF fuel rods and the rest of the fuel assembly structure and is a critical part to the overall batch licensing strategy.

Framatome is continuing to finalize a SiC-SiCf cladding design to be used for test rodlets in the ATF-2 tests. Discussions are underway with Idaho National Laboratory (INL) to identify the insertion cycles and exchange technical information necessary to prepare for this test series. The expectation is to fabricate and provide test rodlets to INL later in 2020.

Accident Tolerant Fuels (ATF) Phase II – General Electric Development of LWR Fuels with Enhanced Accident Tolerance

Principal Investigator: Raul B. Rebak, GE Research, Schenectady, NY
Team Members/ Collaborators: Russ Fawcett, Global Nuclear Fuels; Evan Dolley, GE Research; Andy Nelson, ORNL; Ed Mai, INL

GE is on track for obtaining actual commercial reactor plant irradiation data on two ATF concepts including ARMOR coated zirconium alloy and IronClad monolithic clad fueled rods.

As fuel vendors, General Electric (GE) including GE Research, GE Hitachi Nuclear, and Global Nuclear Fuels (GNF) and their partners, the reactor owners Southern Nuclear and Exelon Generation plus Oak Ridge National Laboratory (ORNL), Idaho National Laboratory (INL), and Los Alamos National Laboratory (LANL) are working in the development of accident tolerant fuels (ATF) for the current fleet of light water reactors (LWRs). Activities include basic research and testing to characterize and evaluate materials that were never used before in reactor environments, fuel rod fabrication, as well as direct assessment of fuel rods behavior by installation is operating civilian nuclear power plants. The current GE contract with the Department of Energy (DOE) Office of Nuclear Energy (ONE) DE-NE0008823 extends to February 2021.

Project Description:

The objective of the GE-led project is to develop a family of fuels that will make the current and future fleet of LWRs safer to operate. The newer family of ATF fuels will also add benefits such as: (a) Fuel cycle economics (i.e., increased burnup), (b) Increased fuel reliability, and (c) Plant operational flexibility (for example using power reactors for peak demands of electricity). GE is working in fuel concepts that are for both near term implementation and longer term development. The fuel concepts include cladding components, fuel components, and channels. For cladding, GE is developing the ARMOR coating for Zircaloy-2 tubing which will provide resistance to fretting under normal operation conditions and increased resistance to oxidation in design base accident (DBA) and beyond design basic accident (BDBA) conditions. GE is also developing the IronClad cladding concept which involves the use of a monolithic FeCrAl alloy for housing the urania fuel. Since the current zirconium alloy used for channel materials needs to be replaced as well, GE is evaluating to utilize nuclear grade silicon carbide composite materials to fabricate the channels. On the fuel side GE is exploring the modification of the current urania fuel to make them more resistant to fragmentation in the case of an accident or for extended burn up conditions.

Accomplishments:

The two largest accomplishments for the fiscal year (FY) 2019 were the fabrication at the industrial GNF facilities in Wilmington NC of fuel rod components to be irradiated at (a) the Advanced Test Reactor (ATR) Cycle 166B at the INL, and (b) Exelon Generation's Clinton Unit 1 Cycle 20 in Illinois. For the ATR ATF-2 tests, INL received from GE twelve urania fuel rodlets. The cladding for the rodlets included standard Zircaloy-2, ARMOR coated Zircaloy-2, and C26M

Figure 1. GE ARMOR and IronClad segmented rods for the Clinton Installation.

IronClad. For the Clinton Cycle 20 insertion, GE fabricated segmented full length rods which included fueled and non-fueled segments or ARMOR coated Zircaloy-2 and IronClad C26M as well as non-fueled segments of Advanced Powder Metallurgy Tubing (APMT) and Nippon Nuclear Fuel Development (NFD) oxide dispersion strengthened (ODS) cladding. Figure 1 shows the ARMOR and IronClad rods that were installed in bundles for the Clinton Plant.

The technical goals for the current FY2019 period were, (a) to advance on rod manufacturing procedures both for ARMOR coated and monolithic IronClad cladding, and (b) to make progress on obtaining irradiation data for the ATF concepts. Both goals were highly successful since it was demonstrated that standard industrial equipment could be used for fabrication of rods, and the Clinton installation is on track to become the second commercial plant installation after the Hatch Unit 1 installation in February 2018.

University Program Integration in SiC-based ATF Cladding Area

Principal Investigator: Yutai Katoh (ORNL)
Team Members/ Collaborators: Christian Deck (General Atomics), Peng Xu (Westinghouse Electric Company)

Alignment of multi-discipline university research projects was enabled by an intense workshop of principal investigators from academia, national laboratories, and industries.

Silicon carbide (SiC) composite technology presents both tremendous benefit and great development challenges as the accident-tolerant fuel (ATF) cladding for light water reactors (LWRs). Since the challenges span from science of the environmental effects and composite behaviors to qualification and licensing of the fuels with an unconventional class of material, cooperative efforts across the academia – national laboratory – industry consortium are essential in order to strategically advance the technological readiness level of this technology. Toward the goal of integrating these multi-thread research and development (R&D) activities, coordination of the university programs was actively facilitated by creating a venue of interactions.

Project Description:

SiC composite technology for nuclear energy systems has been developed over the last decades under the leadership of national laboratories. With the recent rise of strong industry-led programs toward its applications to ATF cladding and core structures for LWRs, while the industry R&D focus on the short to mid-term efforts toward establishing manufacturing technologies, qualification, and licensing along their specific applications and approaches, national laboratory assist the industry with their unique capabilities and expertise and fill the technology gaps in high risk areas. As the foundation of the collective R&D, the academia play the important roles including strengthening the underpinning sciences and training the next generation of workforce.

Presently there are about a dozen Nuclear Energy University Program (NEUP) projects supporting the SiC ceramic and composite technologies for nuclear energy as a part of the Department of Energy (DOE's) Consolidated Innovative Nuclear Research framework, including the seven projects that newly started in Fiscal Year (FY) 2019 with the specific objective of assisting the ATF cladding development. In addition to the NEUP projects, multiple university projects are engaged through other funding mechanisms. Topics of these university projects include the multi-axial failure criteria for composite cladding tubes, development of seal/corrosion barrier coatings, understanding the radiolysis-assisted hydrothermal corrosion, radiation effects in SiC, and performance modeling for fuel systems with the SiC composite cladding.

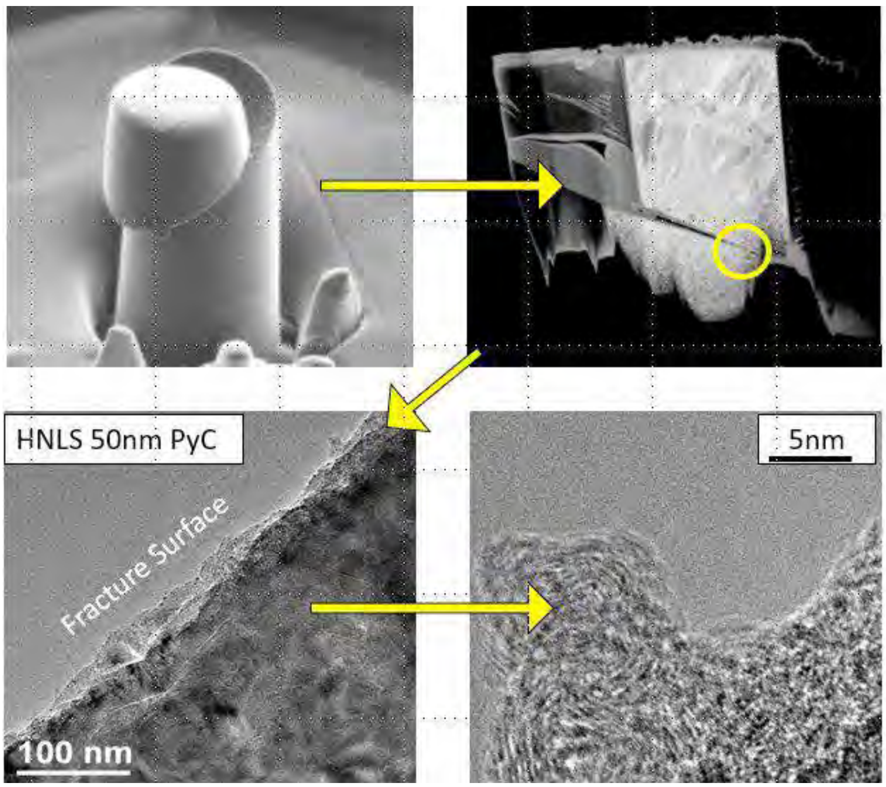


Figure 1. University of California Berkeley project led by Peter Hosemann was successful in analyzing the fiber-matrix interfacial debond/ sliding mechanisms in nuclear grade SiC composite by innovative multi-scale experimental characterizations. Experiment was conducted by graduate student Joey Kabel.

Accomplishments:

Coordination of the university programs with each other as well as with the R&D programs at national laboratories and industries is particularly important toward the accelerated development of the SiC composite cladding technology for several reasons including: 1) the physical and mechanical properties of cladding tubes are highly dependent on the architecture and method of fabrication, 2) the prototypical SiC composite cladding tube samples are long-time procurement items, 3) performance evaluations of fuel cladding require the use of highly

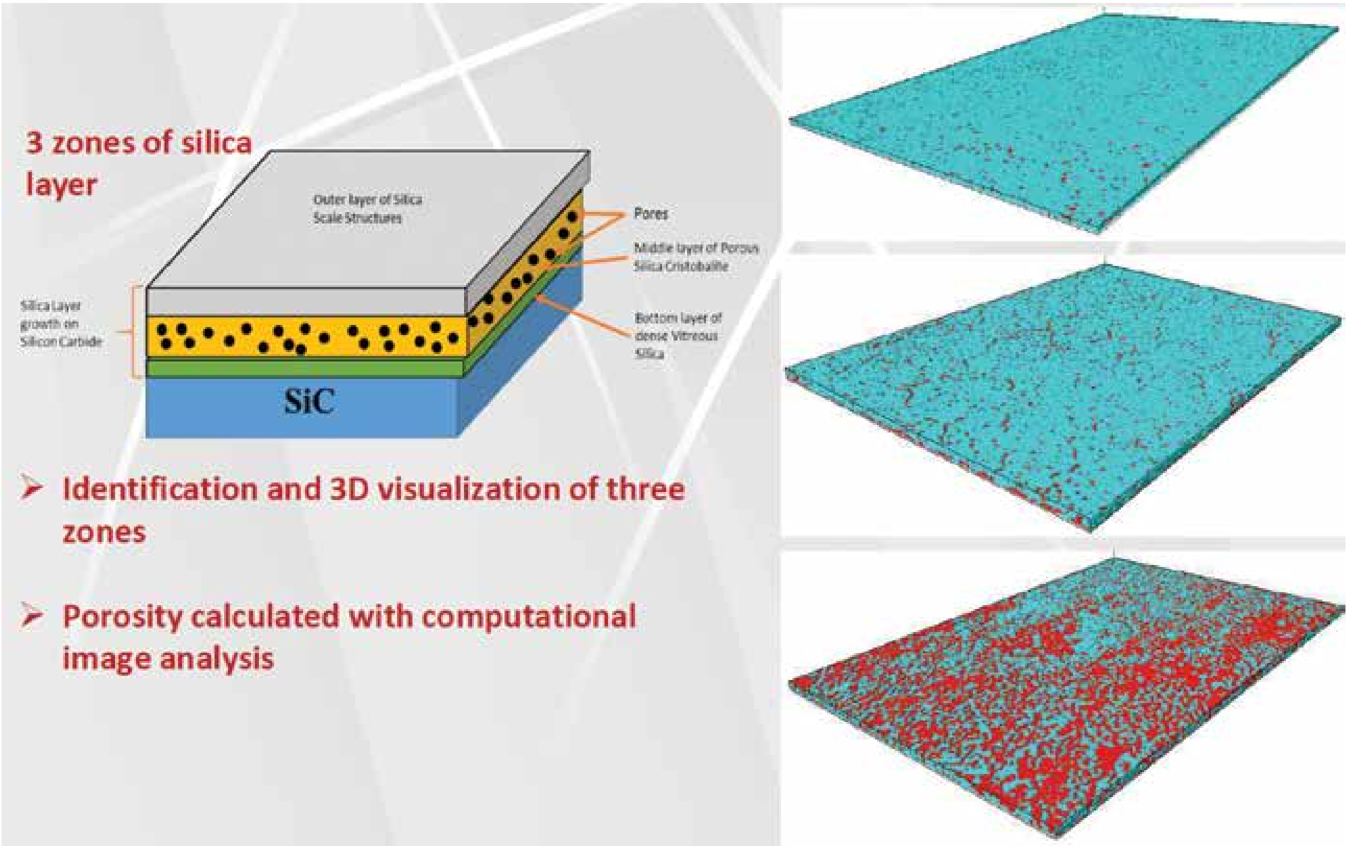


Figure 2. A multi-institution project led by Jacob Eapen of North Carolina State University identified a try-layer oxidation structure in SiC after a flowing steam exposure at 1,200°C. With the three-dimensional visualization by X-ray computed tomography (XCT), the team was successful in developing a phase field model that describes the steam oxidation behavior. XCT was performed by Prasun Majumdar of University of South Carolina.

specialized instruments and expertise, such as the high temperature steam corrosion and hydrothermal corrosion in radiation environments, which have limited availability, and 4) technology integration for this novel fuel cladding requires an extensive range of expertise that can be covered only by collecting expertise from diverse technical disciplines.

To provide the venue of interactions among the programs at universities, national laboratories, and industries, the first SiC/SiC Project Information Exchange & Coordination Meeting was co-organized by Yutai Katoh from Oak Ridge National Laboratory (ORNL), Christian Deck from General Atomics (GA), and Peng Xu from Westinghouse Electric Company (WEC). The meeting was held for two

full days from March 6 – 7, 2019, at the GA Headquarters in San Diego, California. This location was chosen as the place where the SiC composite cladding materials are designed and manufactured as the primary reason. The meeting was attended by about 30 people representing 14 programs. The overviews, planning, progress, and needs for resources and access to specialized capabilities were discussed for each program by the principal investigator (PIs) or representatives. The project alignment was discussed at the end of each session that was arranged by topical area. Overall the meeting was successful in achieving the consensus on the alignment and complementarity among the large number of projects and facilitating the PI networking for the immediate and future interactions.

Figure 3. The first SiC/SiC Project Information Exchange & Coordination Meeting was held in March 2019 at General Atomics in San Diego. Coordination of the program scopes, resource sharing, and PI networking were among the main achievements at the meeting.

2.2 HIGH-PERFORMANCE LWR FUEL DEVELOPMENT

Understanding the U_3Si_2 Crystal Structure Evolution as a Function of Temperature by Neutron Diffraction and Simulation

Principal Investigator: Sven C. Vogel, LANL

Team Members/ Collaborators: Tashiema L. Ulrich (University of South Carolina - Columbia), Joshua T. White and David A. Andersson (LANL) Elizabeth Sooby Wood (University of Texas - Austin)

The combination of high-quality neutron diffraction data and DFT calculations allowed to identify interstitial sites for Si atoms in the U_3Si_2 structures that explains the absence of secondary phases in hyper-stoichiometric U_3Si_{2+1}.

U_3Si_2 is actively researched as an accident tolerant fuel (ATF). Detailed knowledge of the crystal structure evolution, including knowledge of the anisotropic tetragonal lattice parameters, as a function of temperature and chemical composition is of paramount importance for predictions ranging from thermo-mechanical stresses to phase transformations to crystal lattice sites of fission products. Knowledge of the stability range of U_3Si_2 is of great importance to understand whether changes of the stoichiometry, either from burn-up or due to synthesis variations, will lead to undesirable phase decomposition. To investigate these topics, stoichiometric $U_3Si_{2.00}$ and hyper-stoichiometric $U_3Si_{2.01}$ were synthesized and characterized at ambient conditions and up to 1150 °C using time-of-flight neutron diffraction on the HIPPO instrument at the Los Alamos Neutron Science Center (LANSCE) at Los Alamos National Laboratory (LANL).

Project Description

Understanding the changes in behavior of U_3Si_2 in case of deviations from the line compound is imporant for the application of this system as fuel. Initial thermodynamic calculations (Middleburgh et al., 2016) indicated that the material at room temperature cannot accommodate any significant excess Si atoms, resulting in precipitation of secondary phases. Such behavior could ultimately affect fuel pellet integrity. More recent calculations revised these earlier results and predict a significant ability of U_3Si_2 to accommodate excess Si atoms without partitioning of secondary phases (Andersson et al., 2018). Ultimately, experimental data is needed to assess this issue and this research provides the data. Twelve hours neutron count times (compared to the usual 30-120 minutes) were applied to characterize stoichiometric $U_3Si_{2.00}$ and hyper-stoichiometric $U_3Si_{2.01}$ at room temperature. The ultra-high statistical quality of the diffraction data allowed researchers to assess whether secondary phases were present at a level of >0.1 wt. %. Crystallographic techniques such as difference Fourier maps applied to these data sets allow to identify sites of scattering density unaccounted for by the standard U_3Si_2 crystal structure. These sites are indicative of locations of excess Si atoms, which can subsequently be compared to DFT predictions of these sites. Adding small site occupations of Si atoms to the data analysis model allows then to identify which site fits the experimental data best. The same samples were heated to ~1150°C to assess possible differences in the crystal structure evolution as a function of temperature. Differences in the thermal expansion for different chemical compositions could lead to thermal stresses, which ultimately may lead to cracking of pellets.

Accomplishments:

The stoichiometric $U_3Si_{2.00}$ and hyper-stoichiometric $U_3Si_{2.01}$ were synthesized at LANL's Fuel Research Lab (FRL) by J. White (LANL) and E. Sooby Wood (now UT Austin) by arc-melting of weighed amounts of the constitutive elements. Because uranium metal readily oxidizes, great care was taken to control the oxygen in the atmosphere, e.g., by use of oxygen getters. The ingot was crushed into a powder, sieved, annealed, and then loaded into vanadium cans for the neutron diffraction experiments (vanadium has a typically negligible contribution to the diffraction signal). Neutron diffraction data was collected at

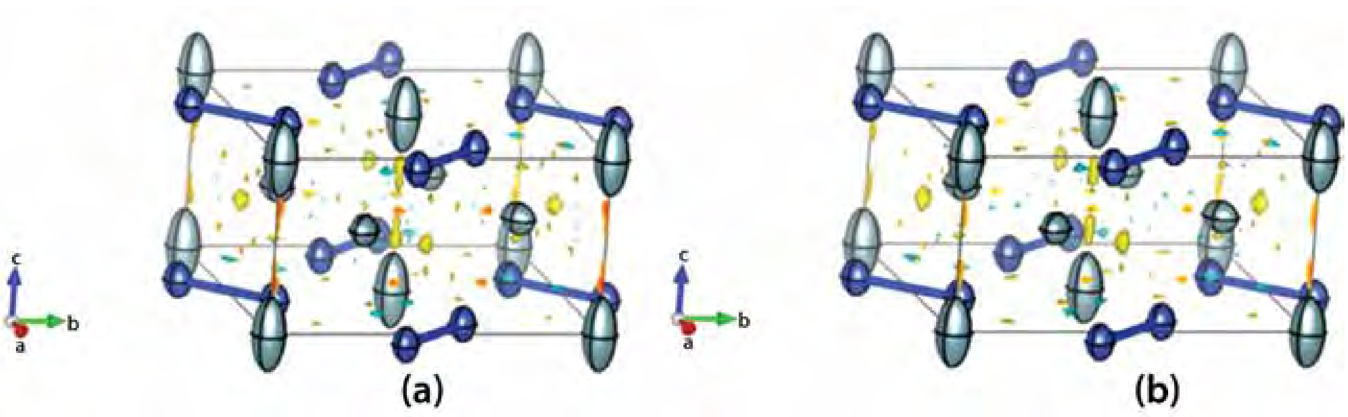


Figure 1. Visualization of the refined crystal structure of $U_3Si_{2.00}$ (a) and $U_3Si_{2.01}$ (b) samples including anisotropic atomic displacement parameters at ambient temperature. Overlaid with the crystal structure are the difference Fourier maps for ~60% of the maximum density (yellow positive difference, blue negative difference). The uranium atoms are shown in grey, with U1 on the corners of the tetragonal unit cell, while Si atoms are shown in blue. The identified 4e site for excess Si atoms is between the U1 atoms.

ambient conditions for ~12 hours per sample as well as in a vanadium furnace at temperatures up to 1150 °C (S. Vogel/LANL). The neutron diffraction data was analyzed using the Rietveld method by T. Ulrich (University of South Carolina, Columbia) and S. Vogel (LANL). The neutron diffraction data analysis provided lattice parameters of the tetragonal unit cell, atom positions of the uranium and silicon atoms, and anisotropic atomic displacement parameters as a function of temperature. Significant differences in volume expansion as well as thermal expansion along the a and c-axes were found for the two compounds. The anisotropic atomic displacement parameters showed an approximately five times larger thermal motion along the crystallographic c-axis than along the a-axis for one of the two uranium sites while the second uranium site and the silicon atom showed an approximately spherical atomic displacement at all temperatures investigated. From the unit cell and atom position parameters, the bond lengths could be computed as a function of temperature. The bond length allowed to compute strains, identifying stronger and weaker bonds within the unit cell. At no temperature were any additional phases observed. The analysis of the difference Fourier map from the high-quality ambient condition runs revealed scattering density not accounted for in the stoichiometric crystal structure. Adding Si atoms on those sites, with partial occupancy, allowed to identify the best match to the experimental data. Occupation of the 4e site (½, ½, 0.262) was identified as the most likely site of excess Si atoms in the U_3Si_2 structure. D. Andersson (LANL, in collaboration with the Nuclear Energy Advanced Modeling and Simulation (NEAMS) and the Consortium for Advanced Simulation of LWRs (CASL) programs predicted the energy of various interstitial sites using DFT calculations, which allowed researchers to confirm the experimental finding. Identification of a site where U_3Si_2 can accommodate excess Si can explain why precipitation of additional phases is not observed. Absence of this precipitation due to excess Si makes the U_3Si_2 system more robust against deviations from stoichiometry, an important finding for the application as a nuclear fuel. However, the observation that the thermal expansion changes significantly if the stoichiometry varies could lead to thermal stresses within a pellet that could ultimately lead to cracking. Further investigations are therefore warranted. The detailed experimental and simulation results were submitted to Journal of the Applied Crystallography.

Citations:

D. A. Andersson, X.-Y. Liu, B. Beeler, S. C. Middleburgh, A. Claisse, C.R. Stanek, Density functional theory calculations of self- and Xe diffusion in U_3Si_2, Journal of Nuclear Materials 515 (2019) 312-325.

S. C. Middleburg, R. W. Grimes, E. J. Lahoda, C. R. Stanek, D. A. Andersson, Non-stoichiometry in U_3Si_2, Journal of Nuclear Materials 482 (2016) 300-305.

Thermodynamic Evaluation of U_3Si_2 Hydride Reactions

Principal Investigator: Aditya P. Shivprasad, LANL
Team Members/Collaborators: Joshua T. White and Joseph R. Wermer (LANL)

This project has made great strides in understanding the degradation of U_3Si_2 during exposure to hydrogen and the methodology used here could be used to study the degradation of other high uranium-density compounds and composite systems.

U_3Si_2 is a promising, high-uranium-density fuel because of its good thermal properties and performance during irradiation and has been considered as a potential replacement for UO_2 in commercial light water reactors. Despite these advantages, limitations are observed when the coolant comes in contact with the fuel, e.g., breach of cladding, resulting in degradation of the fuel. Research focused on continuing work from fiscal year (FY)18 that studied hydrogen absorption of U_3Si_2 and relating this performance to previous steam testing. Hydrogen absorption was performed using a Sievert's apparatus at temperatures from 250 to 500 °C and pressures up to 1.5 atm of hydrogen.

Project Description

Upon exposure to steam and simulated pressurized water reactor (PWR) conditions, U_3Si_2 fuel has been shown to significantly degrade. In steam at temperatures above approximately 350 °C and atmospheric pressure, rapid oxidation and pulverization have been observed. However, no significant structural degradation was apparent during exposure at lower temperatures. Examination of the post-corrosion microstructure of U_3Si_2 exposed to steam at low temperatures showed similarities with uranium metal under similar conditions. Because uranium metal in steam is known to hydride at low temperatures and oxidize at high temperatures, the possibility of U_3Si_2 hydriding was considered and examined. As a result, this work is imperative in understanding U_3Si_2 degradation under potential LWR conditions.

Previous work in FY17 evaluated U_3Si_2 hydriding performance in hydrogenated water (5 ppm H2 at 300 °C) and in 6% H2/Ar gas showed pulverization that was correlated with the presence of a secondary phase hypothesized to be hydride lamellae. In FY18, hydrogen absorption experiments of U_3Si_2 began by examining the hydriding performance of sintered pellets and button fragments at 350 °C. This work showed a stark difference in hydrogen absorption behavior due to surface conditioning effects. In FY19, hydrogen absorption experiments were continued. Studies were focused on hydrogen absorption and desorption measurements at various temperatures ranging between 250 and 500 °C. Hydrided samples were subsequently examined using X-ray diffraction (XRD) to characterize the evolution of the hydride phase as a function of hydrogen-to-uranium (H/U) ratio and temperature.

Sievert's gas absorption measurements were performed on button fragments of U_3Si_2 with known masses at the Sigma Division of Los Alamos National Laboratory (LANL). Measurements were performed at temperatures from 250 to 500 °C in 50 °C intervals. Hydrogen absorption was performed by titration of known volumes of hydrogen into the gas manifold and monitoring of hydrogen pressure changes to calculate the moles of hydrogen absorbed during the respective aliquot. Hydrogen absorption experiments were terminated when samples no longer appreciably absorbed hydrogen, as measured by hydrogen pressure changes in the gas manifold.

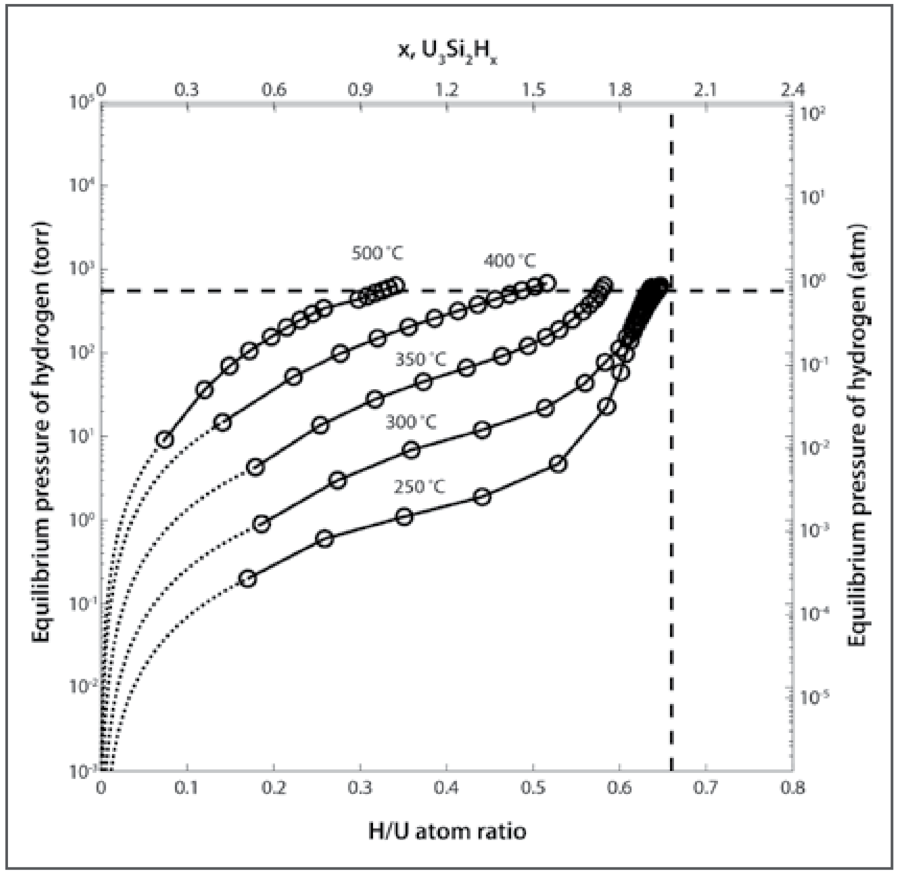


Figure 1. PCT isotherm for hydrogen absorption of U_3Si_2. Hydrogen absorption behavior shows that the hydride phase miscibility gap is at a temperature between 350 and 400 °C. Dashed lines are shown at atmospheric pressure and at a H/U ratio of 2/3 ($U_3Si_2H_2$).

Accomplishments

The goals of the hydrogen absorption measurements and subsequent phase analysis were to determine the final stoichiometry of hydrided U_3Si_2, the thermodynamics of hydride formation, and to characterize the hydride phase evolution as a function of H/U ratio. Upon hydriding, it was found that U_3Si_2 pulverized to fine particle size. Samples were subjected to hydrogen absorption and desorption experiments so that subsequent absorption measurements at different temperatures would proceed much more quickly. It was found that the first complete hydriding was accomplished in approximately one week, while subsequent absorption experiments of desorbed material were completed in under two days. Equilibrium hydrogen pressure was plotted as a function of H/U ratio and were used to develop pressure-composition-temperature (PCT) isotherms for the U_3Si_2-H system. The results of this analysis are shown in Figure 1. The XRD analysis

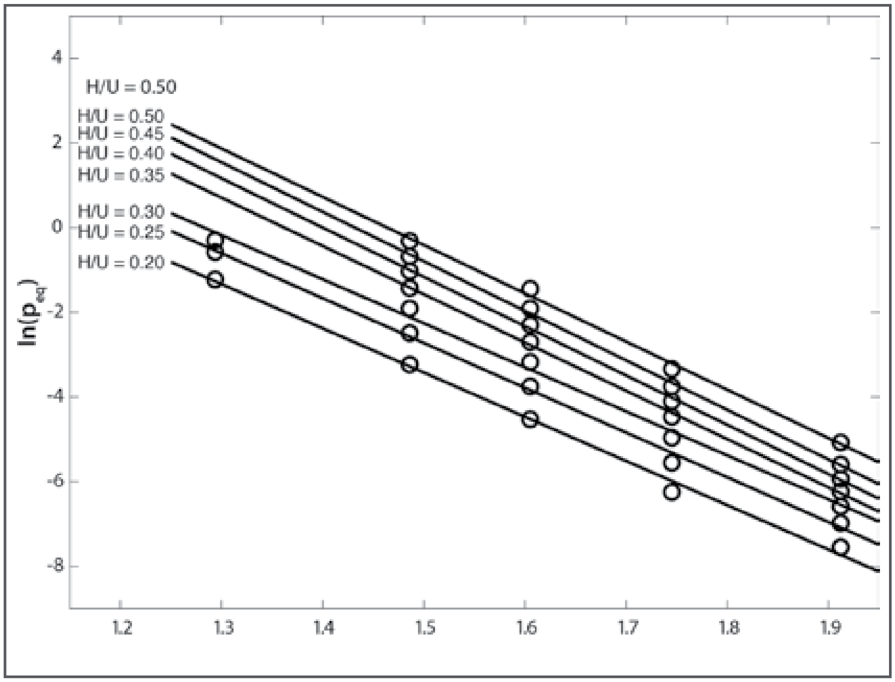


Figure 2. Van't Hoff analysis of U_3Si_2 hydriding behavior for various H/U ratios of U_3Si_2Hx with H/U ranging between 0.20 and 0.50. The figure shows similar slopes for all curves, indicating similar enthalpies of formation for the different compositions. The offsets are due to increasing formation entropy with increasing hydrogen content.

of U_3Si_2 hydrided as a function of temperature helped to highlight the formation of the hydride phase and confirm the PCT isotherms.

From this data, a miscibility gap critical temperature for hydride formation was calculated to exist between 350 and 400 °C. Below the critical temperature, hydride formation proceeds and the hydride is thermodynamically stable. Above this temperature, hydrogen is only absorbed to form a solid-solution such that U_3Si_2 and U_3Si_2Hx are the same phase. Van't Hoff analysis of the PCT isotherms was used to calculate the enthalpies and entropies of formation for several compositions (H/U ranging from 0.2 to 0.5) of the hydride phase, which are significant in understanding and modeling the absorption of hydrogen by U_3Si_2 fuel. Figure 2 displays Van't Hoff analysis for U_3Si_2 hydriding. It was also found that U_3Si_2 subjected to hydride-dehydride cycles at or near the miscibility gap critical temperature disproportionated into uranium metal and silicon. This was confirmed using XRD and is a significant finding in understanding U_3Si_2 fuel degradation by exposure to hydrogen.

Progress Towards Waterproofing Uranium Nitride

Principal Investigator: Aditya P. Shivprasad, LANL
Team Members/Collaborators: Joshua T. White and Amber C. Telles (LANL)

Uranium mononitride (UN) is a promising, high-uranium-density fuel because of its good thermal properties and performance during irradiation and has been considered as a potential replacement for uranium(IV) oxide (UO_2) in commercial light water reactors. Despite these advantages, limitations are observed when the coolant comes in contact with the fuel, e.g., breach of cladding, resulting in degradation of the fuel. Research focused on continuing work from fiscal year (FY)18 that examined the feasibility of UN-UO_2 composites for UN waterproofing purposes. Steam oxidation testing of candidate waterproofing additive materials was performed to evaluate the corrosion resistance of these materials. Composite pellets were also pressed and heated to examine chemical compatibility with UN.

Project Description:

Previous work in FY18 evaluated UN and UN-UO_2 composites for resistance to oxidation in steam. During isothermal oxidation tests, mass gain and pulverization occurred over the course of minutes, though the addition of UO_2 significantly delayed the onset of oxidation. Similarly, during temperature ramps in steam, the addition of UO_2 appeared to delay the onset of oxidation, though pulverization occurred for pellets containing more than 10 volume percent UN. This work showed that waterproofing effects are noticeable with compositing with oxidation-resistant materials.

In FY19, compositing efforts focused on evaluating UN cermets with metals that exhibit resistance to steam corrosion, such as chromium (Cr), silicon (Si), and yttrium (Y). UN-SiC composites were also examined for feasibility given the interest within the Accident Tolerant Fuel (ATF) campaign. Materials tested for steam oxidation were assessed for corrosion resistance based on the percentage of the material oxidized (a reaction coordinate) in steam. Chemical compatibility between UN and additive materials was also assessed using X-ray diffraction after thermal treatment.

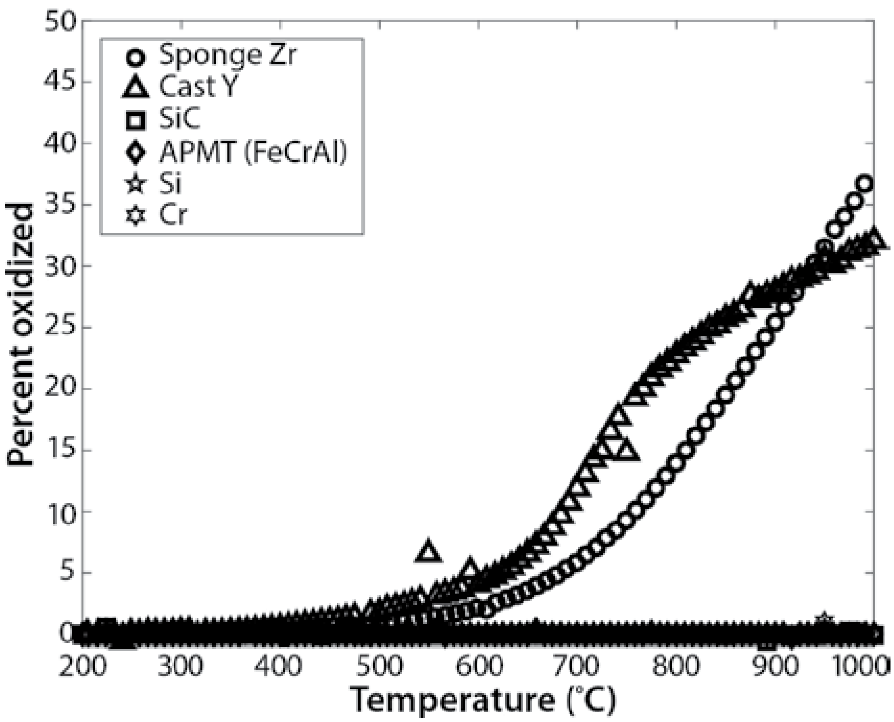


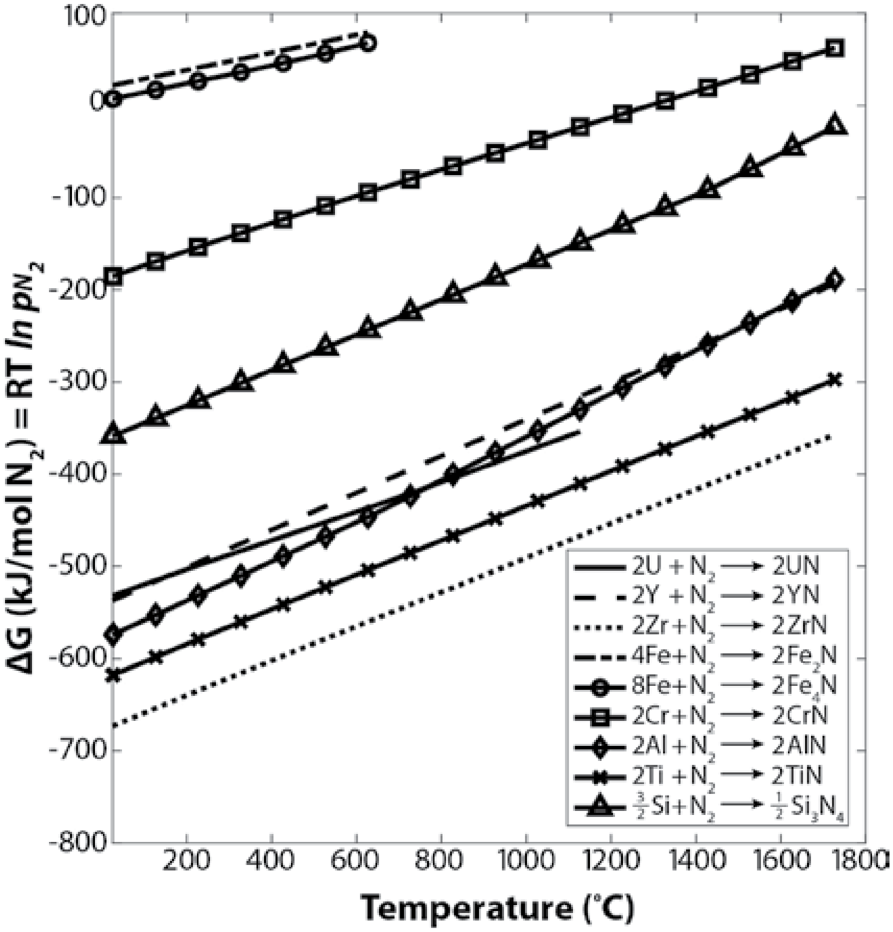
Figure 1. Oxidation fraction, in percentage, as a function of temperature for various candidate additives for UN waterproofing applications in steam up to 1000 °C. It was found that SiC, APMT, Si, and Cr exhibited the best resistance to steam corrosion.

Accomplishments:

The goal of the steam oxidation experiments was to determine the oxidation resistance of the candidate additive materials for compositing with UN, while the goal of the chemical compatibility testing was to evaluate chemical interactions between UN and the candidate additive materials.

Steam oxidation testing was performed on samples with known masses in temperatures ranging between 200 and 1000 °C using a steam furnace coupled with thermogravimetric analysis at the Fuels Research Laboratory (FRL) at Los Alamos National Laboratory (LANL). Samples were heated in argon gas maintained at oxygen levels below

Figure 2. Ellingham diagram for nitride formation of uranium and several candidate waterproofing concepts. Results showed that Zr, Al, Ti, and Y were prone to reaction with nitrogen to form nitrides at the cost of UN, while Cr, Si, and Fe were likely to remain metallic.



0.1 ppm; once the desired temperature was reached, the steam/argon mixture was flowed through the system and sample mass change as a function of temperature and time was recorded. Results of the steam oxidation testing are shown in Figure 1. Results showed that, aside from sponge Zr, all the materials exhibited adherent oxide layers. Although significant mass gain was measured for Y metal, its oxide was adherent. It was observed that Cr, Si, APMT, and SiC exhibited the best resistance to corrosion (little-to-no mass gain). Based on these results, chemical compatibility testing was performed between UN and Y, Cr, Si, and SiC.

This project has made significant advances in understanding the difficulties that arise from nuclear fuel waterproofing efforts and has resulted in selection criteria that will be useful in evaluating future waterproofing concepts.

Composite pellets of UN with Y, Cr, Si, and SiC were fabricated by mixing powders of the aforementioned materials. Yttrium dihydride (YH2) powder was used in lieu of Y metal powder due to the brittle nature of the hydride, allowing mechanical milling with the UN prior to dehydriding at an elevated temperature. Composites with Y, Cr, and Si were heated to temperatures above the melting points of the respective metals (1526, 1857, and 1410 °C) with the aim of evaluating chemical interactions and promote liquid-phase sintering. Composites with SiC were heated under temperature profiles used for two-step, pressure-less sintering of SiC.

Results showed that yttrium absorbed nitrogen from the uranium, resulting in the formation of yttrium mononitride (YN) and U metal. An Ellingham diagram for nitride formation was developed using ThermoCalc 2019a to determine which waterproofing candidates might react with nitrogen. This diagram is shown in Figure 2 which indicates that Zr, Al, Ti, and Y metals will absorb nitrogen from UN, while Si, Cr, and Fe should remain metallic.

Conversely, Si and SiC appeared to react with the material. In the case of silicon, the resultant pellets comprised of primarily uranium silicides. It is not yet clear what reactions occurred with silicon carbide, but the system is being studied further due to interest in understanding the interactions between these two compounds. The composites with chromium showed little-to-no remaining chromium metal due to the high vapor pressure of chromium at the temperatures required for liquid-phase sintering. As a result of these various outcomes, the several selection criteria have been proposed to assess future waterproofing concepts.

Demonstration of Microfluidics for Synthesis of Sol-gel Feedstocks

Principal Investigator: JW McMurray, ORNL
Team Members/ Collaborators: A McAlister, RD Hunt, KM Cooley, AT Nelson (ORNL)

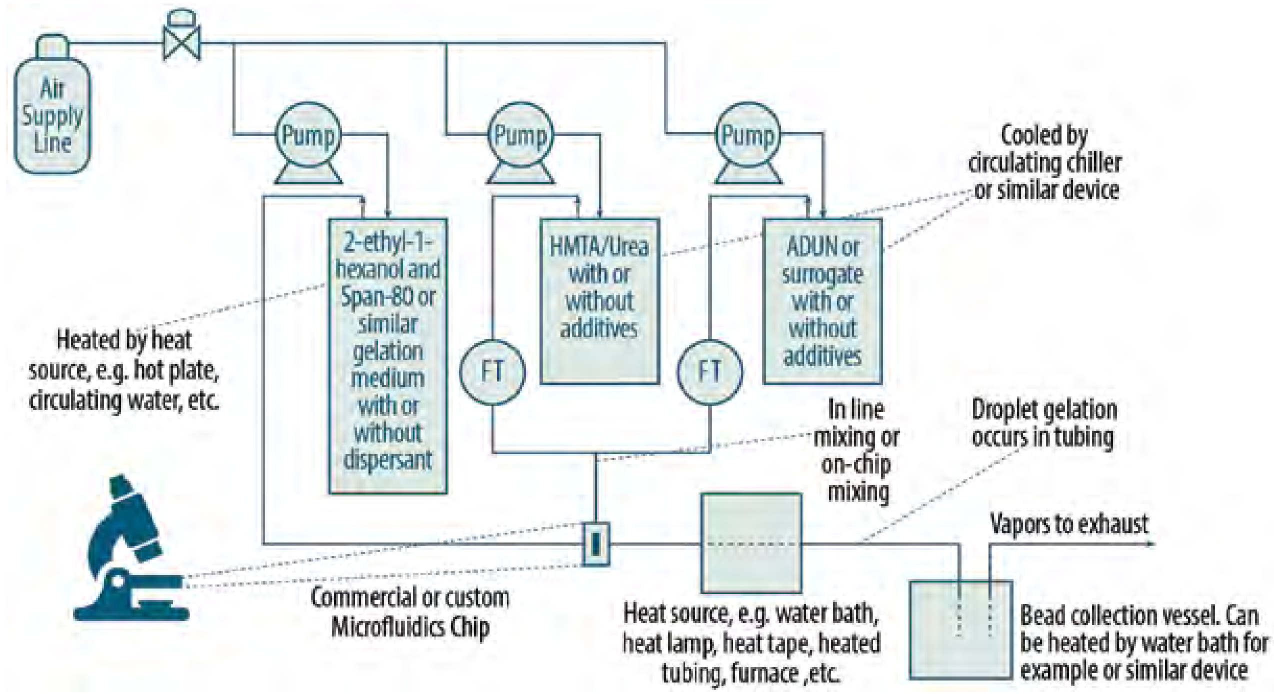


Figure 1. Flow sheet for the internal gelation system with microfluidics.

Sol-gel production has long been used for synthesis of uranium compounds for use in particle fuels (e.g., tri-structural isotropic (TRISO)) as well as transuranic-containing fuel forms where radiological concerns limit dry processing. One accepted drawback of traditional sol-gel production is that fuel kernel sizes are generally limited to a minimum of several hundred micrometers, and the yield of a conventional sol-gel process will be a distribution of sizes. Sieving is then required to narrow the particle size used for subsequent processing thereby reducing process yield. This work explores the opportunities for microfluidics to pair with the wet chemistry processes used for sol-gel fabrication and overcome these limitations.

Project Description:

The objective of this research is to develop a processing pathway that goes beyond conventional sol-gel equipment to produce particle fuel with greater ranges in size from submicron to 1mm in diameter, higher yields with tighter tolerances, and greater flexibility in process variables or additives using an automated microfluidics design. This technology enables post processing for advanced, accident tolerant fuel (ATF)

Successful development of this processing technology is critical for development of advanced particle fuel designs and enables additive manufacturing for complex fuel forms that have the potential to be safer, more reliable, and more economical for both current and future reactors.

research and development as well as the use of additive manufacturing techniques for complex fuel forms that have the potential to be safer, more reliable, and more economical for both current and future reactors.

Accomplishments:

This research demonstrates the production of uniform gel spheres using an experimental microfluidics apparatus which can produce sintered diameters well below 200 µm with a relatively small coefficient of variance. Processing designs and parameters, including feedstock chemistry, fluid flow rates, and washing procedures were explored and tuned. A schematic of the overall process flow is shown in Figure 1.

Additionally this research used Zr as a surrogate for U to produce sol-gel feedstock for ZrN and ZrC conversions. Additions of carbon to the internal gelation system with microfluidics can be problematic due to the low flow rates on the order of 100 µL/min and long run times on the order of 4 h. However, an aqueous solution of dispersed carbon, Cabot's TPX-101, was successfully used with solutions of zirconyl nitrate, urea, and hexamethylenetetramine to make zirconia microspheres with carbon. The molar ratio of carbon to zirconium was 3. A simple carbothermal reduction process using a maximum temperature of 2073 K and ultrahigh purity argon produced uniformed zirconium carbide kernels with an average diameter of 35 µm. A scanning electron microscopy (SEM) image of the output of this process, without sieving or use of a roller micrometer to control for sphericity, is shown in Figure 2.

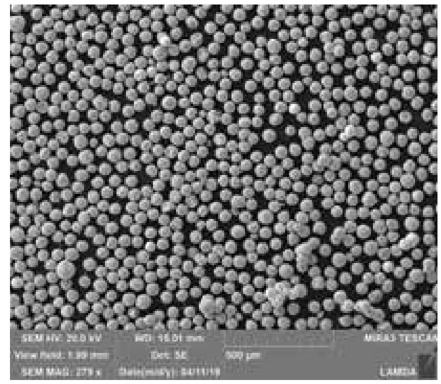


Figure 2. SEM image of 50µm ZrC spheres produced using the microfluidics process developed from this research. The phase purity was determined from X-ray diffraction (XRD).

Successful demonstration of this technology for both oxide and carbide/nitride precursors provides confidence that extension to a radiological capability is justified. Fiscal year (FY) 20 work will demonstrate this technology for uranium materials. This technology provides an opportunity for advanced particle fuels, use of sol gel kernels to generate advanced composite fuel forms, enabling of new manufacturing technologies, and other options for light water reactor (LWR) as well as advanced reactor fuels.

Thermophysical and Mechanical Property Assessment of UB_2 and UB_4; Constituent Phases for Advanced LWR Fuels

Principal Investigator: Erofili Kardoulaki, LANL
Team Members/ Collaborators: Josh White, Darrin Byler, David Frazer, Adi Shivprasad and Tarik Saleh (LANL), Tiankai Yao, Bowen Gong, Jie Lian (Rensselaer Polytechnic Institute)

This highlight marks the completion of the baseline thermophysical and mechanical property assessment for uranium boride phases, candidate phases for LWR fuels.

Uranium diboride (UB2) and uranium tetraboride (UB4) are candidate constituents for multi-phase accident tolerant fuel (ATF) due to their anticipated high thermal conductivity. These fuels have high uranium density that contributes to fission, and by tailoring the ratio of 10B/11B, can also act as an integrated burnable poison. Understanding the thermophysical and mechanical properties of uranium borides, for which only limited data are available in the literature, is of importance to determine their accident tolerance.

Project description

This highlight outlines the completion of the thermophysical and mechanical property assessment of UB2 and UB4 synthesized via arc melting and sintered to high densities via spark plasma sintering (SPS). The high density SPS samples were used to measure the thermal diffusivity via laser flash analysis (LFA) and thermal expansion via dilatometry of UB2 and UB4 and, in conjunction with specific heat literature data, their thermal conductivities were calculated from 298 to 1773 K. Finally, resonance ultrasound spectroscopy (RUS) and nanoindentation (NI) were performed to investigate the mechanical properties of the uranium borides, thus giving a complete picture on their properties and sufficient data to assess them as candidate fuel phase constituents for light water reactors (LWRs).

Accomplishments

Four uranium boride samples were analyzed using RUS: two UB_2 and two UB_4. RUS measurements were made at constant amplitude using a National Instruments NI, PXIe 1075 function generator. Frequency generation and spectra acquisition were managed by the Resonance Inspection Techniques and Analysis (RITA©) software package developed at Los Alamos National Laboratory (LANL)[1]. The elastic constants were calculated from a spectrum using the Rayleigh-Ritz method from the resonance frequencies [2–4]. Assuming each sample was comprised of a polycrystalline, isotropic material, the independent moduli are C11 and C44 and macroscopic quantities, such as bulk (K), shear (G), and Young's (E) moduli, as well as Poisson's ratio, were calculated as was done in previous work [3–6]. From the RUS results, the Debye temperatures of the uranium borides were calculated using methods detailed in the literature [7]. Nanoindentation was performed with a Hysitron TM Triboindenter TM (Minneapolis, USA) in load-controlled mode with an indent load of 15 mN. The indents were performed with a diamond Berkovich tip that was calibrated with fused silica prior to indentation on the UB_2 and UB_4 specimens. The nanoindentation curves were analyzed using the Oliver and Pharr method [8] and measuring the reduced modulus and hardness of the specimens. The elastic modulus of the

Table 1. Polycrystalline moduli of uranium borides (average of two samples) calculated by correcting for porosity from RUS and NI.

Material	G (GPa)	K (GPa)	E (GPa)	H (GPa)	ν	θ_D (K)
UB2 (RUS)	145.52 ± 6.48	127.70 ± 11.18	316.31 ± 26.96		0.1	490.4 ± 29.4
UB2 (NI)	-	-	320 ± 18	22.0 ± 2.9	-	-
UB4 (RUS)	191.60 ± 6.93	194.21 ± 1.95	432.50 ± 12.85		0.15	686.4 ± 27.2
UB4 (NI)	-	-	484 ± 29	28.4 ± 1.4	-	-

tested specimens was calculated from the reduced modulus using the equation in [8]. The values for the shear, bulk, and Young's moduli, from both RUS and NI, as well as Poisson's ratio and Debye temperatures, from RUS, averaged for the two measured samples are summarized in Table 1. From the comparison of Young's moduli in Table 1, it is observed that there is generally good agreement between RUS and NI measurements from this work.

The thermal conductivities of UB_2 and UB_4 were calculated, on cooling and heating, and the results are shown in Figure 1 along with literature data for UO_2 [9]. The conductivity for UB_2 is significantly higher than that of UB_4 throughout the measured temperature range. However, both materials have thermal conductivities significantly higher than that of UO_2. On heating, the thermal conductivity of UB_2 decreases with a minimum at 874 K, above which the conductivity starts to increase. As was expected, based on the thermal diffusivity results for UB_2, the thermal conductivity on cooling is higher throughout the temperature range and only returns to values obtained for heating at the 370-470 K range. The thermal conductivity of UB4 increases linearly with temperature throughout the measured temperature range and the values obtained for heating and cooling coincide. The thermal conductivity of UB_2 can be expressed in the following form:

$$\lambda = \frac{1}{(A + B * T)} + C * T$$

where the first term, commonly used to describe the thermal conductivity of UO_2 [10], represents the phonon contributions in terms of A, a temperature independent scattering term and B, a temperature dependent phonon-phonon scattering term. The second term represents the electron contributions to the thermal conductivity. By fitting Equation 1 to the UB_2 heating data, the following values were obtained: A=7.230×10-3 m·K·W-1, B=9.000×10-5 m·W-1 and C=1.327×10-2 W·m-1·K-2, providing an R2=0.9923. Based on

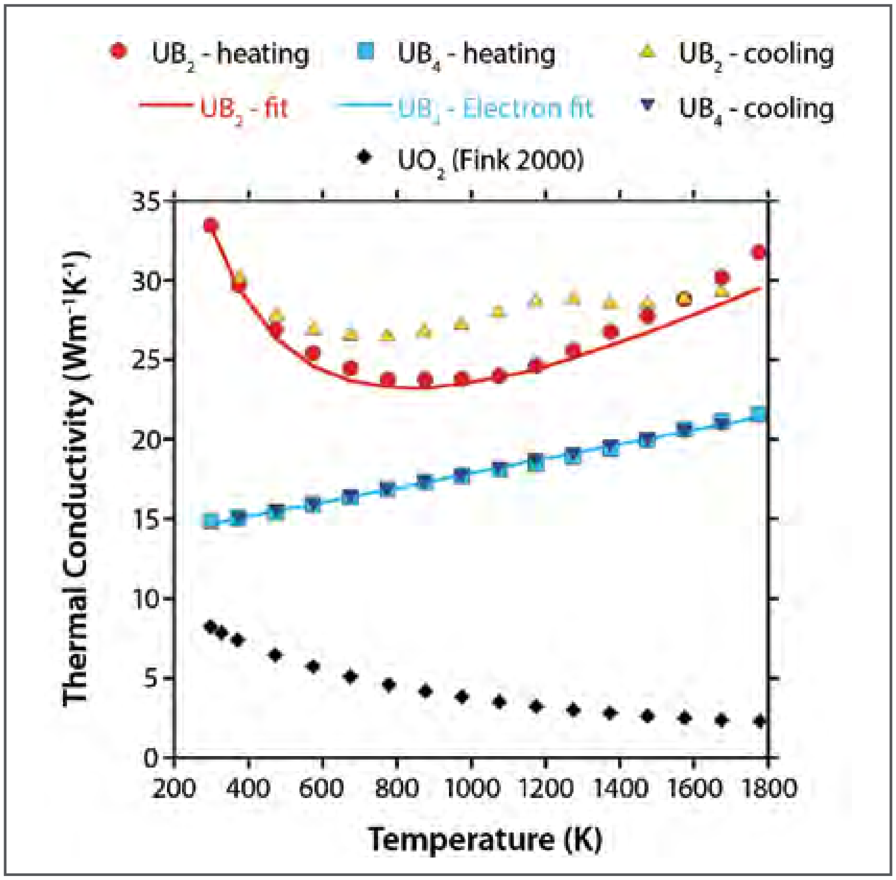


Figure 1. Thermal conductivity of UB_2 and UB_4 on heating and cooling, calculated up to 1773 K from experimental thermal diffusivity and thermal expansion data, and literature values (UB_2:[15,16] , UB_4:[16]) for specific heat. Fits are also shown for UB_2 and UB_4 as well as reference data for UO_2 [9].

these numbers, the conductivity fit to the experimental data is shown in Figure 1. Studies in the literature show that UB_2 is an electronic conductor [11] with no band gap [12] and therefore a strong electronic contribution to the thermal conductivity is anticipated. For the thermal conductivity of UB_4, a linear Wiedemann-Franz temperature dependence is observed suggesting that electronic contributions are dominant. Phonon contributions appear to be negligible given the purely linear relationship observed. This is in accordance with literature studies that report UB_4 to be an electronic conductor and a semimetal [13,14]. Therefore, a linear equation was used to describe the UB4 thermal conductivity:

$$\lambda = C * T + D$$

where C and D are constants describing the electronic contribution to the thermal conductivity. Fitting the UB_4 data to Equation 2 the following terms were obtained: C=4.573×10-3 W·m-1·K-2 and D=1.333×101 W·m-1·K-1, providing an R2=0.9960. The fit based on Equation 2 is also shown in Figure 1. The thermal conductivities of both UB_2 and UB_4 are expected to be impacted by the presence of the identified impurities, shown in Figure 1, however, this effect would be minimal given their low volume fraction.

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Nanomechanical Properties of High Uranium Density Fuels

Principal Investigator: David Frazer, LANL
Team Members/ Collaborators: Joshua T. White and Tarik A. Saleh (LANL)

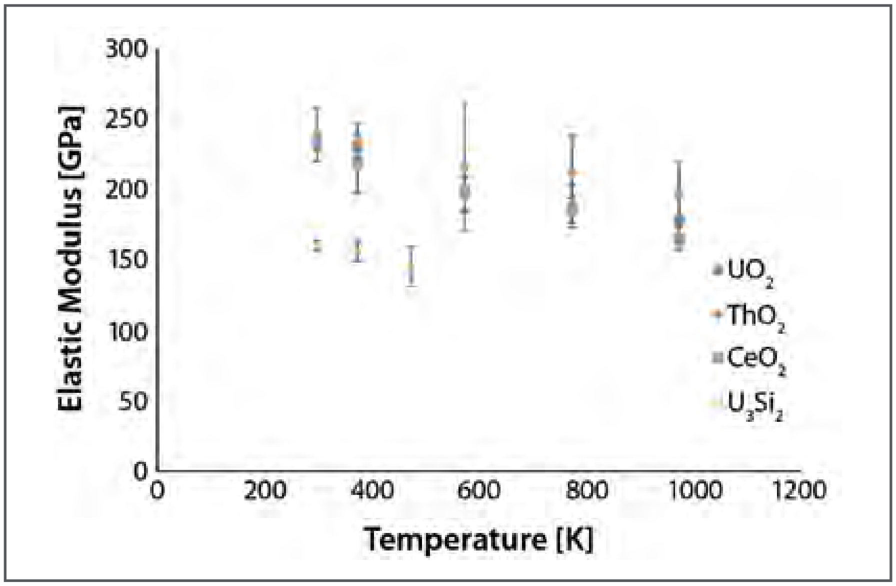


Figure 1. The elastic modulus versus temperature of the four materials tested during these experiments.

The drive for longer nuclear fuel cycles lead to the observation of pellet clad interactions (PCI) which instigated investigations in understanding this phenomenon. It was observed that the mechanical properties of the fuel play an important role in PCI. Small scale mechanical testing (SSMT) techniques can be used to measure the mechanical properties of limited volumes of material. The ability to measure mechanical properties with limited volumes of material could be beneficial to testing irradiated fuel as it reduces the source term and could allow for material to be tested outside of hot cells. In addition, SSMT has the ability to measure mechanical properties as a function of temperature, which is largely lacking in the literature for fresh fuel as well as irradiated for the majority of nuclear fuels considered for current and next generation nuclear reactors.

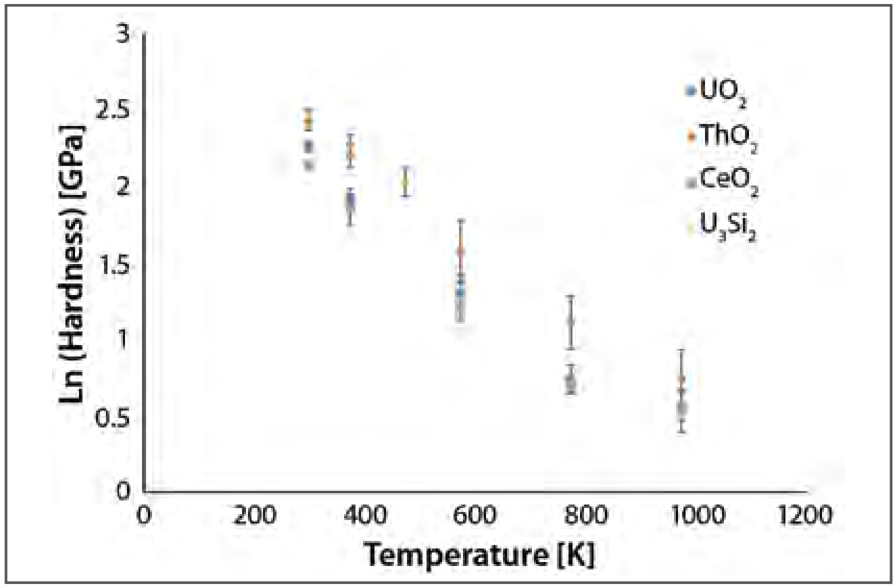
Project Description:

This research is aiming to measure the mechanical properties of high density fuel in support of the accident tolerant fuel campaign. While there has been research into the thermophysical properties of these accident tolerant fuel forms, there has been little research into the mechanical properties which are needed for the modeling of the pellet clad mechanical interaction. In this work the development of SSMT for uranium-based compounds over temperature is being evaluated and developed. SSMT can enable the rapid assessment of research size batches of different materials. The development of these SSMT techniques on fresh fuel would facilitate their use on irradiated fuel in the future. A benefit of using these techniques on irradiated fuel would be the small volume of material needed which would greatly reduce the source temperature and safely allow testing of the irradiated fuel outside of hot cells. In this work nanoindentation (NI) was used to the measure the elastic modulus and hardness of high density accident tolerant fuels at elevated temperature.

Accomplishments:

A challenge with elevated temperature testing of some uranium based compounds is oxidation. In order for the elevated temperature NI of the accident tolerant high density fuels to be successful the oxygen in the NI chamber needed to be controlled or displaced. This was achieved by modifying the nanoindenter with additional gas feedthroughs to displace the oxygen in the chamber with a cover gas such as argon or argon+6% hydrogen in order to inhibit the oxidation of the samples. Utilization of an inert gas in the chamber limits the max load of the transducer to 3-4 mN limiting the depth of the indents to approximately 100 nm in these harder ceramic materials. While this is a sufficient depth, to measure the mechanical properties it does limit the volume of material sampled to the near surface region. Development and evaluation of the high temperature NI setup used a graded approach to assess the feasibility of the technique.

Figure 2. A plot of the nanoindentation hardness versus temperature of the 4 materials tested.



First, CeO_2 was measured up to 700 °C, as it is not radioactive and not air-sensitive followed by ThO_2 as it was radioactive and not air-sensitive. While the ThO_2 is not air-sensitive it was tested in an argon atmosphere to evaluate the environmental control of the nanoindenter chamber and introduce the radiological hazard. After, the UO_2 was measured up to 700 °C as it was radioactive and ai-sensitive. The UO_2 was measured in an argon+6% hydrogen cover gas environment in order to inhibit the oxidation of the sample. Due to the UO_2 being tested in a reducing environment of argon+6% hydrogen it is believe, that the UO_2 is stoichiometric. Lastly, U_3Si_2 was measured up to 200 °C in an argon environment. U_3Si_2 was only measured to 200 °C as the specimen oxidized at elevated temperatures evidenced by the increase in hardness from the surface oxide formation. The U_3Si_2 was measured in an argon environment instead of an argon+6% hydrogen to prevent hydride formation in the U_3Si_2. Additionally, methods of inhibiting the oxidation of the U_3Si_2 sample are being explored in order to increase the testing temperature. The measured elastic modulus and hardness versus temperature plots for the materials tested can be seen in Figures 1 and 2. The values of the elastic modulus measured with high temperature NI agree well with the literature values measured through a variety of different techniques. In addition, the NI hardness values match well with literature values when there was data available for comparison. The future work for the nanoindenter includes the measuring of Uranium Nitride (UN) at elevated temperature.

First ever high temperature nanoindentation was successfully performed on high density accident tolerant fuels.

Fully Ceramic Microencapsulated Fuel Pellet Production for Irradiation in the High Flux Isotope Reactor

Principal Investigator: Rachel L. Seibert, ORNL
Team Members/ Collaborators: Jim O. Kiggans Jr., Kurt A. Terrani, and Joseph R. Burns (ORNL)

The fully ceramic microencapsulated fuel development project has conducted pre-irradiation characterization and has supplied fuel samples for irradiation testing in the high flux isotope reactor in FY20 to further the development of next generation nuclear fuel under the AFC campaign.

Fully ceramic microencapsulated (FCM) fuel is an accident-tolerant concept under consideration for use in light water and advanced nuclear reactors. FCM fuel utilizes the tristructural-isotropic (TRISO) fuel design, embedded within a silicon carbide (SiC) matrix. This design is inherently safe with the ability to achieve higher burnup, improved thermal performance, and improved reliability tailored to the fuel kernel and targeted reactor type. The work here involved the fabrication of FCM fuel pellets for irradiation studies in the high flux isotope (HFIR) reactor at Oak Ridge National Laboratory (ORNL). These pellets utilize the miniature fuel concept, which is intended to more rapidly examine and qualify different fuel forms due to small sample size and rapid burnup accumulation. Uranium nitride and two-phase uranium carbide/uranium oxide kernels were considered and will be evaluated after irradiation for performance and integrity of FCM fuels under various reactor conditions.

Project Description:

FCM pellets were fabricated to fit a modified version of the Miniature Fuel irradiation capsules designed by Petrie et al. at ORNL. They maintain an approximate diameter of 5.8 mm and 2.8 mm height. Uranium nitride and two-phase uranium carbide/uranium oxide TRISO coated fuel kernels were overcoated with a low-viscosity silicon carbide mixture using a custom-built rotating chamber rotated at 80 rpm and heated to 80°C. The coating was intended to aid in fuel particle compaction and densification during sintering. These coated particles were loaded in the middle of silicon carbide nanopowder with yttria-alumina sintering aides into a 5.8 mm graphite die. They were hot pressed at 1,800°C for 30 minutes with a pressure of approximately 9.5 MPa to densify. Neutronics calculations were performed at positions closest to the HFIR core's midplane to determine maximum neutron and gamma heating rates for the fuel. This information is used to predict fuel burnup rate and for future thermal analysis. They are carried out using Monte Carlo N-Particle 5 (MCNP5) static transport calculations and COUPLE and ORIGIN modules of the SCALE code for depletion and decay. The calculations for fuel depletion assumed fixed conditions for flux, spectrum, and reaction cross-sections. No degradation of coated particles was observed during sample compaction, and an average compact density of 97% was obtained.

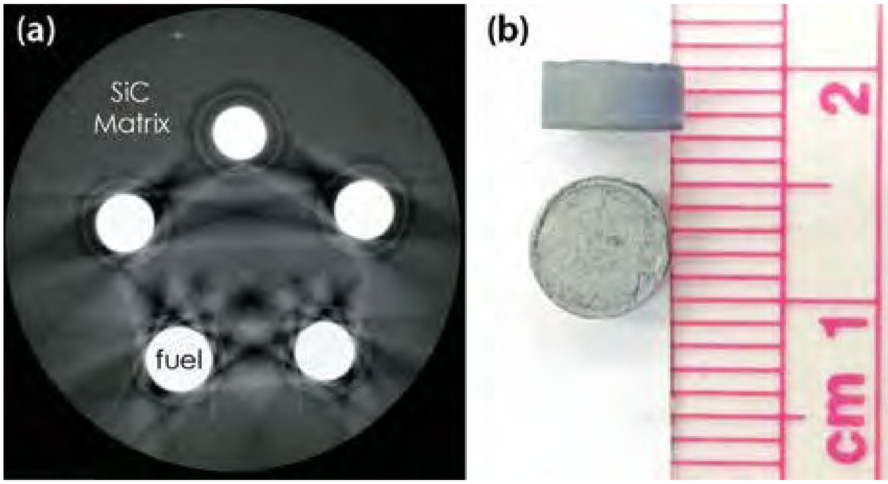


Figure 1. (a) X-ray Tomography of the cross section and (b) a photograph of the FCM fabricated fuel pellet with five centrally located fuel kernels.

Accomplishments:
This project successfully designed, fabricated, and prepared fully ceramic microencapsulated fuel for insertion into the Miniature Fuel experiment in HFIR at ORNL at the start of fiscal year (FY) 20. This irradiation will complement fuels studied in the first MiniFuel irradiation from FY18. The objective of this irradiation is to study the integrity and performance of the FCM fuel architecture for light water reactor conditions. The ORNL team successfully designed an optimized silicon carbide overcoat to aid in compaction of samples. Additionally, they designed a working procedure for pellet pressing and compaction using a hot-press, with no degradation of the coated particles. This hot-pressing method was successful despite the unconventionally small sample size used to fit the fuel capsule design. Calculations were led by Dr. Joseph R. Burns to predict the burnup, heating rates, and thermal analysis. The ability of Advanced Fuel Campaign (AFC) researchers to fabricate miniature fuel pellets in such a manner is integral to more rapidly investigating and qualifying proposed fuel forms. Postirradiation examination (PIE) will proceed in FY20 and FY21, which will validate the performance predictions and design integrity at low burnup.

Fabrication of High-density Irradiation Test Specimens for HFIR MiniFuel Testing

Principal Investigator: Najeb Abdul-Jabbar, LANL
Team Members/ Collaborators: Joshua White and Chris Grote (LANL)

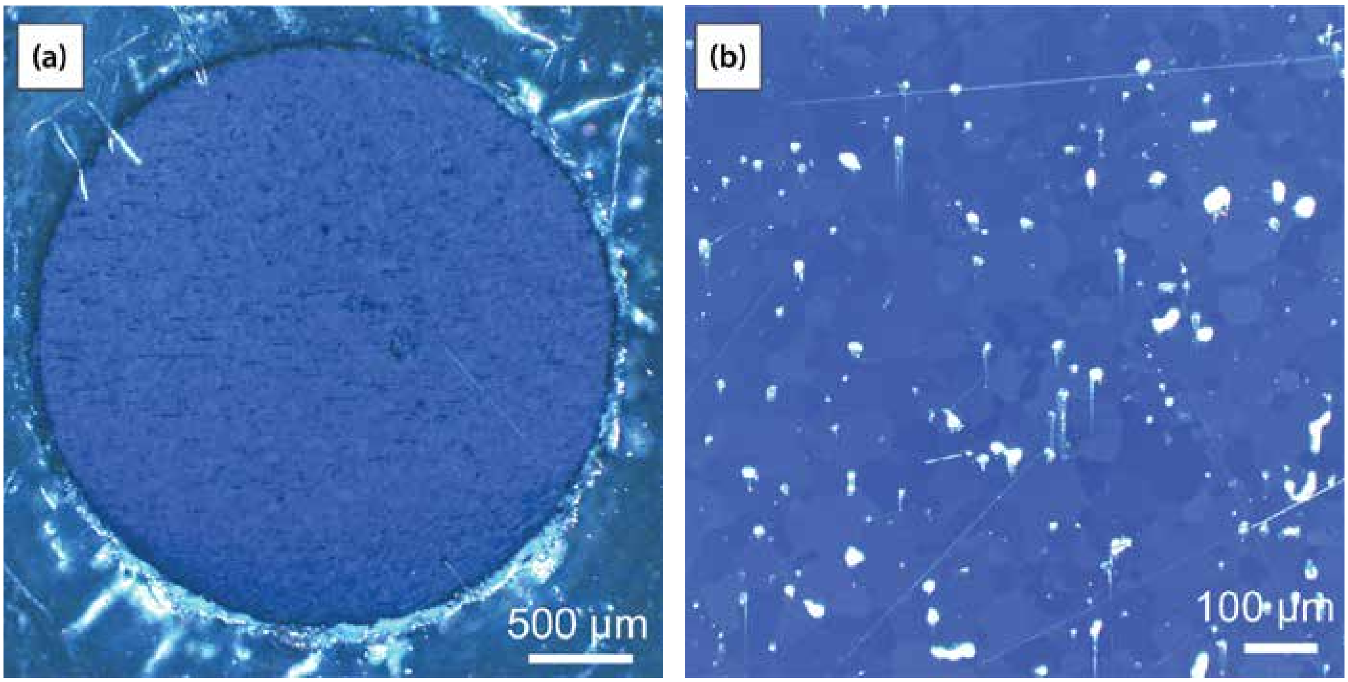


Figure 1. (a) Polarized light microscopy image of a polished U3S2 pellet that reveals overall grain structure. (b) Higher magnification image, which shows grain sizes ≥20 µm.

Characterization of conventional irradiated nuclear fuel pellets can be prohibitively expensive and other options must be explored for a higher throughput approach. An experimental framework has been developed at Oak Ridge National Laboratory (ORNL) to conduct irradiation testing on miniature fuel (MiniFuel) candidate materials. The setup requires fuel pellet samples with ~3 mm diameter and thicknesses ≤300 µm. Such size restrictions serve to greatly reduce sample radioactivity after irradiation and mitigate the need for costly handling procedures. Additionally, miniature geometries can aid in decoupling interrelated materials phenomena that are observed in full-scale fuels

Fabrication of high-density fuels at the miniature scale has been demonstrated, allowing rapid irradiation testing and characterization of emerging accident tolerant nuclear fuel materials using a separate effects testing approach.

testing. Efforts at Los Alamos National Laboratory (LANL) have focused on the synthesis and fabrication (via powder metallography routes) of high-density uranium silicide (U_3Si_2) test specimens to MiniFuel specifications.

Project Descriptions:

The primary objective of this project are fabrication and preliminary thermophysical property assessments of high-density U_3Si_2 MiniFuel test specimens. U_3Si_2 has been proposed as a candidate material as it offers enhanced thermal conductivity (this reduces the propensity for thermal stress failures) and a uranium density that is higher than conventional uranium dioxide (UO_2) fuels, which can enhance the economic viability of nuclear power plant operation. Nevertheless, full elucidation of postirradiation structure-property relationships and the thermophysical characteristics of U_3Si_2 at peak operating temperatures is necessary for potential implementation in the current nuclear energy infrastructure. This can be achieved by studying novel fuel forms (such as U_3Si_2) at the MiniFuel scale (~3 mm diameter, less than 300 µm thickness). The reduced radioactivity and bulk specimen size associated with the MiniFuel concept aids in addressing key challenges associated with rapid postirradiation testing and accelerated measurements of material properties relevant to fuel performance.

Fabrication of MiniFuel U_3Si_2 specimens was carried out at the Fuels Research Laboratory (FRL) at LANL using standard powder metallography techniques, with the aim of achieving sample density >90% theoretical

Table 1. MiniFuel U_3Si_2 specimens delivered to ORNL.

Specimen ID	Mass [g]	Thickness [µm]	Diameter [mm]	ρ^* geometric [g/cm^3]	%TD*$_{geometric}$
35-P-19-039	0.0212	290	2.842	11.524	94.5
35-P-19-045	0.0179	250	2.823	11.439	93.8
35-P-19-047	0.0200	278	2.836	11.412	93.5
35-P-19-048	0.0190	266	2.811	11.510	94.3
35-P-19-051	0.0170	242	2.813	11.330	92.9
35-P-19-052	0.0178	244	2.821	11.652	95.5
35-P-19-054	0.0205	283	2.845	11.373	93.2
35-P-19-055	0.0189	271	2.801	11.329	92.9

density (~12.2 g/cm3). Powder x-ray diffraction and inert gas fusion elemental analysis were respectively used to determine phase purity and oxygen-hydrogen-nitrogen concentrations. Microstructures were examined using optical and electron microscopy to gauge overall sample porosity, defects, and grain size. Preliminary thermophysical property characterization involved laser flash analysis (LFA) to measure thermal diffusivity and differential scanning calorimetry (DSC) to measure specific heat capacity.

Accomplishments:

Eight MiniFuel high-density U_3Si_2 specimens were fabricated and delivered to ORNL for irradiation tests at the High Flux Isotope Reactor (HFIR). Specimen densities (measured using geometric methods) ranged from 92.9% to 95.5% of the theoretical U_3Si_2 density (~12.2 g/cm3). Detailed sample specifications are listed in Table 1. Microstructure examination (see Figure 1) showed grain sizes ≥20 µm (measurements based on the ASTM-E112 standard). Oxygen contents in the MiniFuel specimens were ~284 ppm, with negligible amounts of hydrogen and nitrogen present.

Preliminary thermophysical characterization of MiniFuel samples has highlighted the challenges associated with measuring specific heat capacity and thermal diffusivities at the miniature scale. Heat capacity measurements on MiniFuel U_3Si_2, copper, 304 and Inconel 600 samples show deviations from reference data and is attributed to low sample masses that result in a reduced DSC signal to noise ratio. This challenge becomes more acute for heavier actinide materials, where specific heat capacities are intrinsically low. Nevertheless, there are feasible engineering solutions that will be explored to improve the reliability of DSC measurements on MiniFuel specimens. Inherent difficulties are also encountered in thermal diffusivity measurements that stem from the instrumental limit of a ~3 ms laser pulse in the LFA system at LANL. However, a new LFA apparatus is being acquired that will be capable of pulses faster than the diffusion of heat in a thin MiniFuel specimen.

2.3 LWR CORE MATERIALS

Data Collection to Support NRC Licensure of Coated Cladding for Industry

Principal Investigator: Yong Yan, ORNL
Team Members/ Collaborators: Ben Garrison, Mike Howell, and Andy Nelson (ORNL)

High temperature oxidation and post-quench ductility testing of industry ATF coated cladding concepts allows for quantification of improvements in mechanical properties following design-basis accident.

Nuclear utilities are now confronted with two pressing issues: achieving acceptable power generation economics and maintaining very high safety standards. The damage to the Fukushima Daiichi nuclear facilities in Japan caused by the loss of coolant accidents (LOCAs) has underscored the importance of factoring severe accidents into nuclear reactor safety research and analyses. Numerous industry teams are developing a waterside hard coating material for use on boiling water reactor (BWR) and pressurized water reactor (PWR) cladding that resists fretting wear and is more tolerant of the LOCA conditions. Quantification of the benefits imparted by the applied coatings with respect to high temperature oxidation is a critical activity. As part of the General Electric (GE) Accident Tolerant Fuel (ATF) program, Oak Ridge National Laboratory (ORNL) is collaborating with Global Nuclear Fuels (GNF) to evaluate the performance of the thin ARMOR coating. This work provides the technical basis for Nuclear Regulatory Commission (NRC) licensing.

Project Description:

The objective of this project is to develop data that establishes ARMOR's high temperature oxidation (HTO) and post quench ductility (PQD) behavior relative to the uncoated condition. This data will be used as design input for establishing NRC LOCA licensing basis for ARMOR coated fuel rods. The objective is met by comparison of weight gain and offset strain of coated and uncoated samples and will help to demonstrate that ARMOR "does no harm" while also highlighting its oxidation benefits in LOCA. The HTO and PQD data will also be used as input to the future Licensing Topical Report (LTR) that will be submitted to the NRC to demonstrate that the LOCA performance of ARMOR is equivalent to or better than uncoated Zircaloy-2.

The HTO testing with coated and uncoated coupons will be conducted to determine the weight gain vs. time trends at 800 to 1400°C for equivalent times up to 7000 seconds. The PQD study is performed with coated and uncoated cladding samples, which includes (1) steam oxidation at 1000 and 1200°C and then followed by water quench at 800C; (2) ring compression testing (RCT) with the oxidized samples at elevated temperature. Oxidation times for PQD testing is guided by Cathcart- Pawel (CP) formulation for 2-sided Equivalent Cladding Reacted (ECR) of 13%, 17%, and 23%. For coated coupons, coating is applied to all surfaces including edges. For the cladding PQD tests, the inner surface will not be coated and will be exposed to steam during testing. The ductility of steam oxidized cladding samples will be determined by RCT. Thermogravimetric Analysis (TGA) scoping tests are conducted to inform and select the test times that will establish the HTO weight gain vs. time kinetics tests.

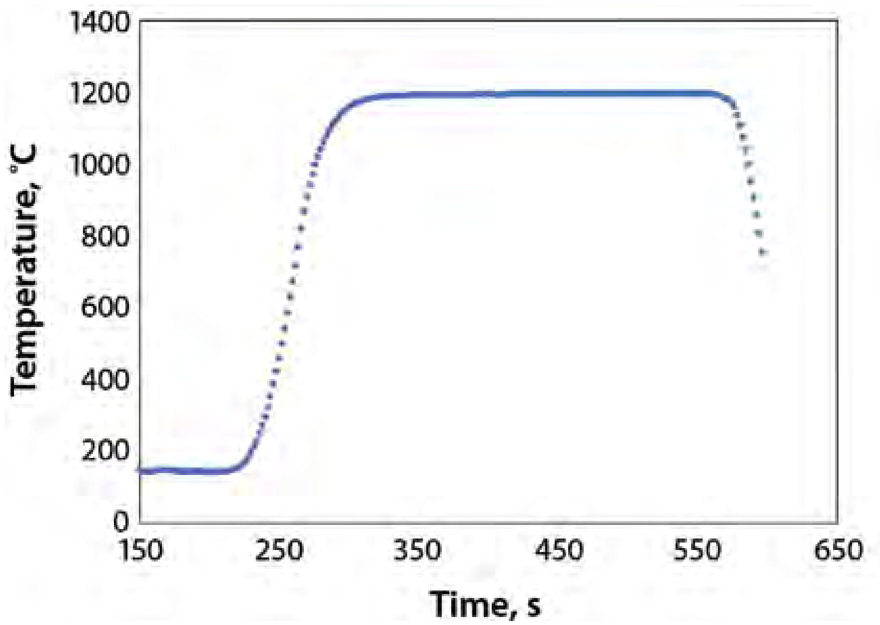


Figure 1. Temperature history of the 1200ºC oxidation test with coated Zircaloy-2 cladding sample for CP-ECR = 17%.

Accomplishments:

The main activities in FY19 included: (1) PQD testing with coated and uncoated cladding samples at 1000 and 1200°C, and (2) HTO study with coated and uncoated coupons at 800, 1000, and 1200°C.

Two-sided benchmark oxidation of standard Zircaloy 4 cladding tube samples was performed with a tube furnace to evaluate the oxidation system and test conditions by comparing the weight gain and oxide layer thickness to the CP-predicted values. The measured sample weight gain is in excellent agreement with the CP-predicted values at 1200°C, indicating the oxidation system and test conditions satisfy the NRC requirements for the PQD study. HTO tests of coated and uncoated Zircaloy-2 cladding samples were performed with the same furnace used for the benchmark testing of Zircaloy 4 described above (100% steam flowing through a quartz tube at steam flow = 5.5 mg/cm2/s). Figure 1 shows a temperature history of the 1200ºC oxidation test with a coated cladding sample for CP-ECR = 17%. The ramp rate was about 20°C/s from the 200°C to1000°C and a few degrees per second thereafter. The average cooling rate from the hold temperature to the 800°C quench temperature is ≈ 10°C/s. The oxygen pickup of coated samples was much lower than uncoated

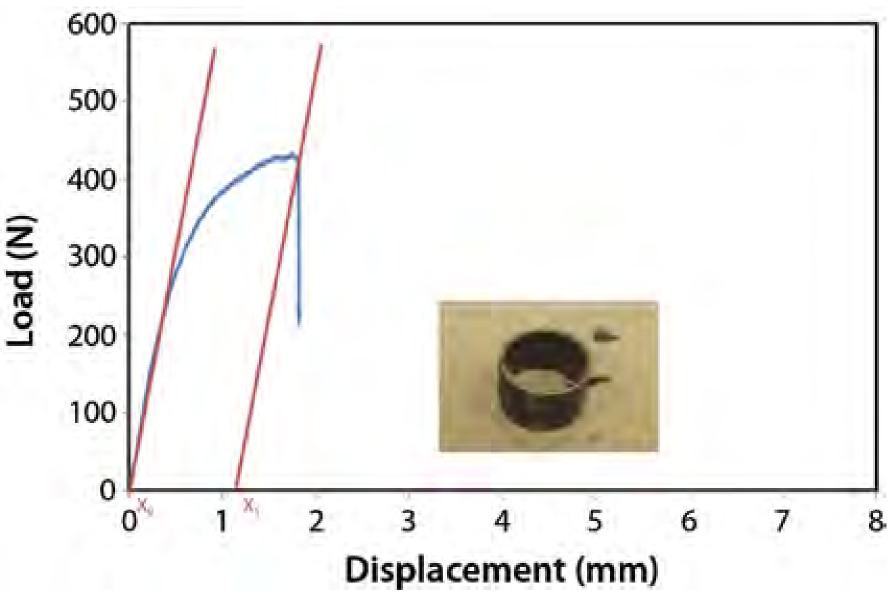


Figure 2. Load-displacement curve from the RCT of uncoated Zircaloy-2 specimen oxidized at 1200C for 17% CP-ECR.

samples at both 1000 and 1200°C. For 17% ECR tests at 1200°C, the weight gains of coated samples are 50% (or lower) of uncoated samples, indicating protection of the coated layer.

Following oxidation and quench, 7-8 mm rings were cut from the oxidation cladding samples. RCTs were performed at 135°C at a displacement rate of 0.033 mm/min. The load-displacement curves were analyzed by the offset-displacement method. The test results show that both coated samples are much more ductile than uncoated samples. The load-displacement curve from the RCT of uncoated specimens oxidized at 1200C for 17% CP-ECR is shown in Figure 2. The microstructure of the steam-oxidized coated and uncoated cladding specimens was examined. Metallographic mounts were prepared and examined using optical microscopy to image the oxide, alpha, and prior-beta layers and their interfaces.

HTO testing with coated and uncoated coupons was started for temperatures at 800, 1000, and 1200C. As the heating rate and steam flow rate of the TGA are low, only limited TGA tests were performed to select the times for the HTO study. The oxygen pickup of coated coupons is much lower than uncoated coupons.

Digitization of FeCrAl Property Handbook for NRC Collaboration

Principal Investigator: M. Dylan Richardson, ORNL
Team Members/ Collaborators: Kevin G. Field and Andrew T. Nelson (ORNL)

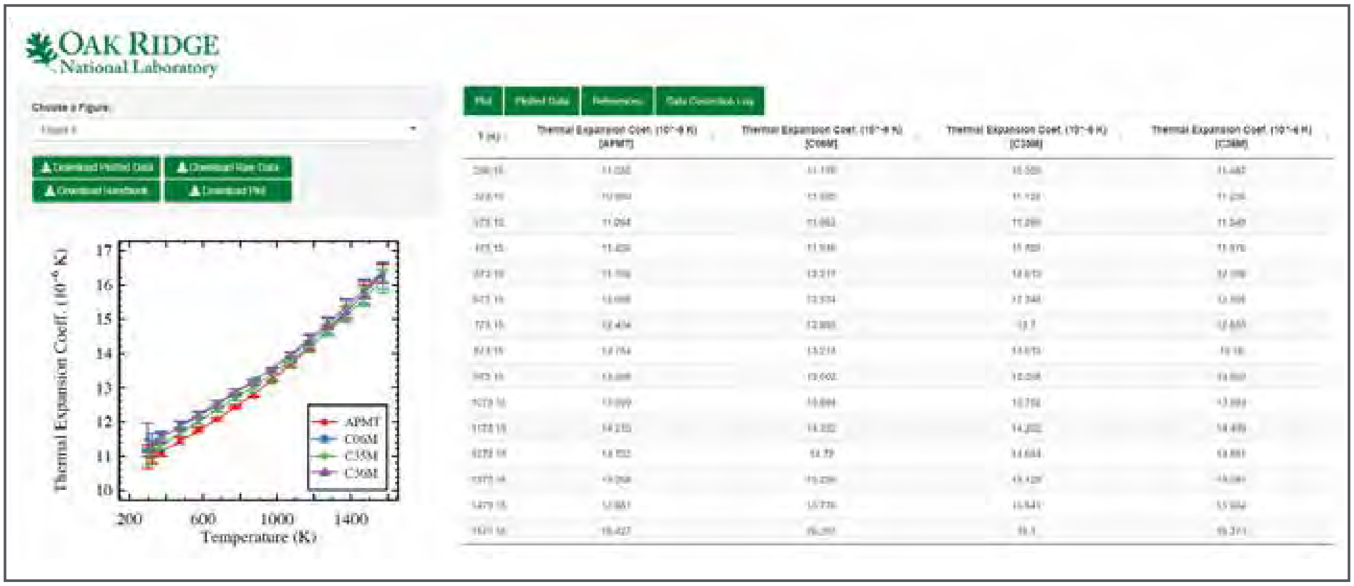


Figure 1. Web interface screenshot used to plot selected thermal expansion data from FeCrAl alloys.

In order to allow for the data found within the AFC document "Handbook on the Material Properties of FeCrAl Alloys for Nuclear Power Production Applications" to be easily accessible, digitization efforts were made with all data and plots. This was accomplished via a Microsoft Excel Document, as well as with a web app developed using a package in R called Shiny.

Project Description:

For the digitization of this handbook, there were four main objectives:

1. to allow for dissemination of the handbook figures, plotted data, and raw data,
2. to provide the figures and associated data in a consumable manner,
3. to provide the raw data associated with each figure, and
4. to provide commentary on the quality assurance (QA) for the various datasets.

With the sheer amount of data found within the FeCrAl handbook, being able to quickly navigate to the data or figures is extremely useful in preparation for codes and/or documents for further support of FeCrAl related activities.

Accomplishments:

The data and figures were digitized in two different manners. First, they were digitized via a Microsoft Excel Document. Because of the amount of data and number of figures, a table of contents was created within the document to allow for swift navigation between the raw data, plotted data, and associated figures. Additionally, a web app was developed using a package in R called Shiny. This web app has the figures, plotted data, and references separated via tabs. It allows for the plotted and raw data to be downloaded as a Comma Separated Value (CSV). It also has links to download high quality images of the plots and the current version of the handbook.

Property Improvement of ATF Wrought FeCrAl

Principal Investigator: Yukinori Yamamoto, ORNL
Team Members/ Collaborators: Bruce A. Pint and Kenneth Kane (ORNL)

Proper combinations of minor alloying additions in ATF wrought FeCrAl alloy can improve the processability of the seamless thin-wall tubes, which significantly contributes to the cost-effectiveness of the cladding production.

Development efforts for Accident Tolerant Fuel (ATF) wrought FeCrAl alloys have been pursued since fiscal year (FY) 2013. FeCrAl alloys were selected as an attractive candidate ATF cladding material because of their excellent oxidation resistance in high-temperature steam environments up to 1475°C, resulting from sufficient amounts of Cr and Al additions. After comprehensive evaluations of model Fe-Cr-Al-Y alloys (Gen. I) and engineering FeCrAl alloy series (Gen. II) with a wide range of major and minor alloy compositions, an alloy consisting of Fe-12Cr-6Al-2Mo-0.2Si-0.03Y, wt.% (identified as "C26M") was down-selected for seamless, thin-wall tube production to be used as an ATF cladding.

Project Description:

Research has been focused on achieving a balance between high-temperature performance and tube fabricability. Body-centered-cubic (BCC) Fe-based materials with Cr and Al additions typically suffer from poor ductility because of their relatively high ductile-brittle transition temperatures. Microstructural stability of the refined grain or subgrain structure at high temperatures is critical to maintaining both the processability of the alloys during tube production and the mechanical properties of the final tube products. Previous efforts successfully produced seamless, thin-wall FeCrAl alloy tubes combined with the expected fine grain structure in the final products through commercially available fabrication process (including a vacuum induction melting, hot-isostatic pressing, extrusion, gun-drilling, drawing and annealing). However, because of a relatively narrow process window to balance the processability of C26M tube production and the properties in the final product, further demands appeared to increase the cost-effectiveness by applying a large reduction in each drawing pass, wide tolerance of intermediate annealing conditions, increased material yield through less susceptibility of unexpected crack formation during drawing process, together with controlling the fine grain structure and sufficiently high oxidation resistance at elevated temperature. Minor alloying additions of yttrium and zirconium (and some other elements, not shown in the report) in C26M base alloys have been attempted to evaluate the effects on various properties related to the processability and the properties.

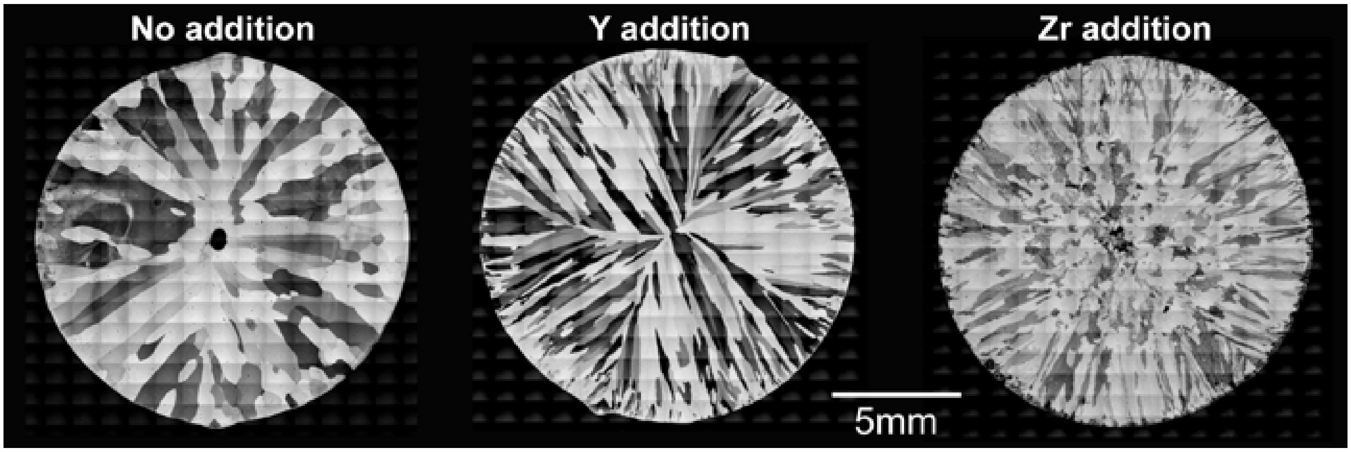


Figure 1. Cross-sectional macrostructure of as-cast Fe-12Cr-6Al-2Mo-0.2Si (C26M) base alloys.

Accomplishments:

Comparisons of lab-scale C26M base alloy heats with minor alloying additions were conducted, in terms of microstructure evolution during solidification and thermo-mechanical processing, room-temperature deformability, and high-temperature oxidation resistance in a steam-containing environment. It was found that the Y addition dominantly controlled the formation of fine columnar grains in as-cast microstructure (Figure 1), and caused inhomogeneous material deformation (and therefore microstructure evolution) during the hot-process. On the other hand, the Zr addition promoted the formation of equi-axed grains (Figure 1) and supported more homogeneous deformation during hot-process, which resulted in a significant stabilization of fine, recrystallized grain structure with ~20-25 µm of average grain size (Figure 2). By comparing with the Y addition, the effect of the Zr addition

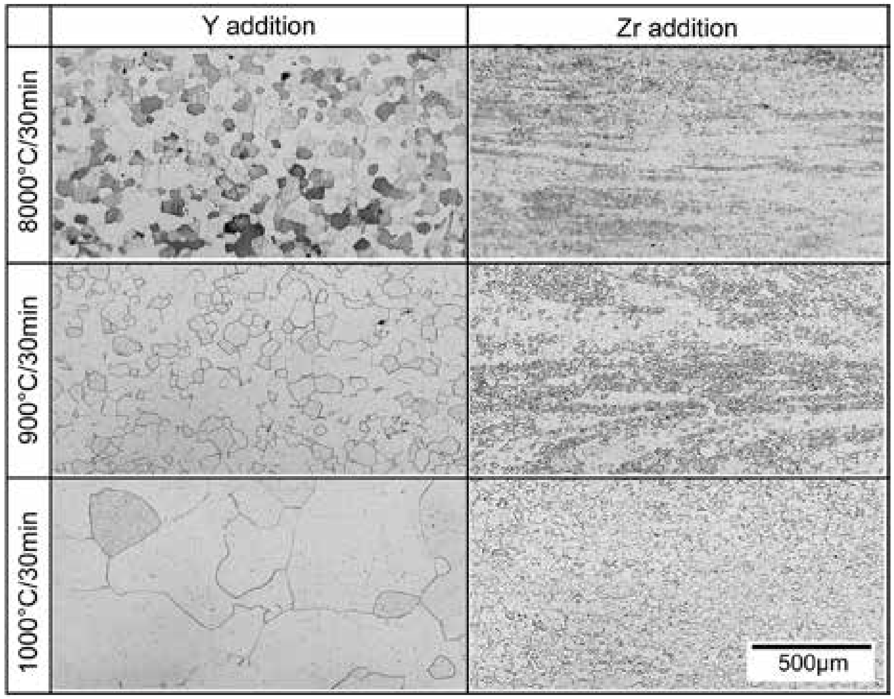


Figure 2. Optical micrographs showing cross-sectional microstructure of hot-rolled and annealed C26M base alloys with Y or Zr additions.

on improving the microstructure stability was significant, even at 1,000°C, as shown in Figure 2. The fine grain structure resulted in increasing the room-temperature ductility for up to ~60% without any trade-off of the tensile strengths (Figure 3). The wide temperature tolerance with the high grain-size stability during annealing will be a significant advantage in maintaining processability with various combinations of hot- and cold-process, including seamless, thin-wall FeCrAl tube production (e.g., large drawing reduction per pass). However, it was also found that the Zr addition promoted internal oxidation to decrease the oxidation resistance during ramp test in steam environments up to 1450°C (Figure 4). The Y added alloys (with no Zr), on the other hand, showed improved oxidation resistance with protective alumina-scale formation at the same test condition. The observed improved properties suggested that the C26M base alloys still hold a potential opportunity to optimize the balanced and further improved properties through refining the combination of minor alloying additions.

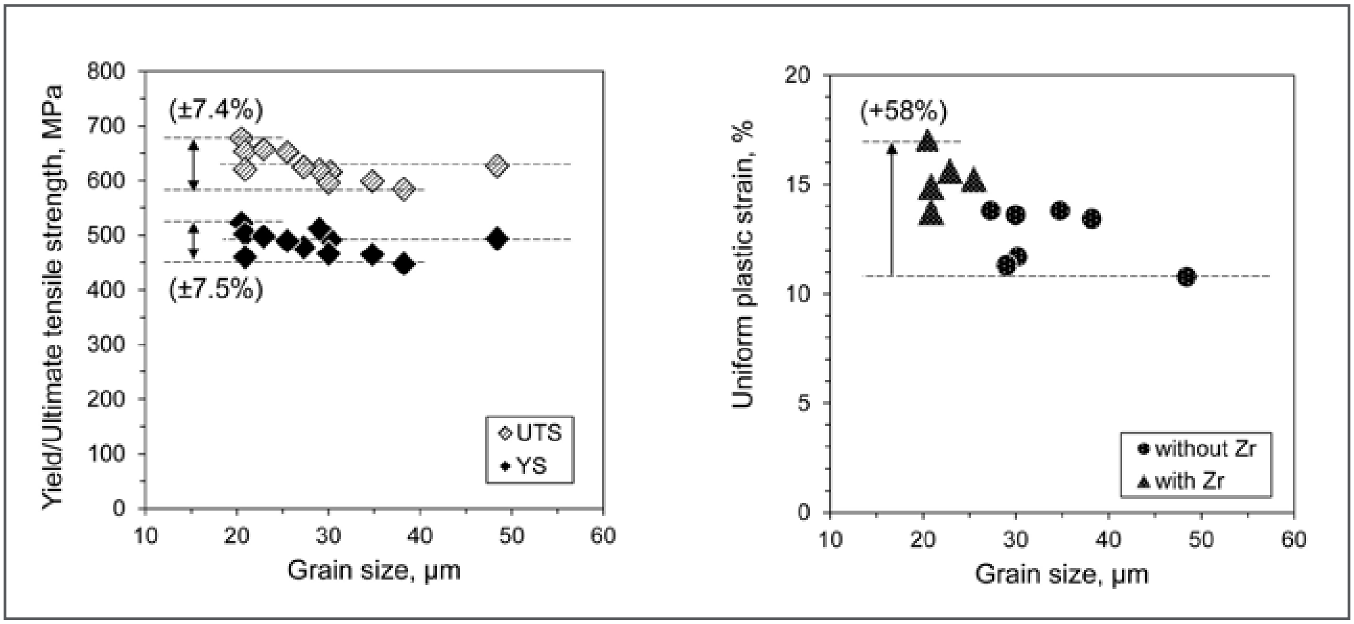


Figure 3. Room-temperature tensile properties of hot-rolled and annealed C26M base alloys with various grain size.

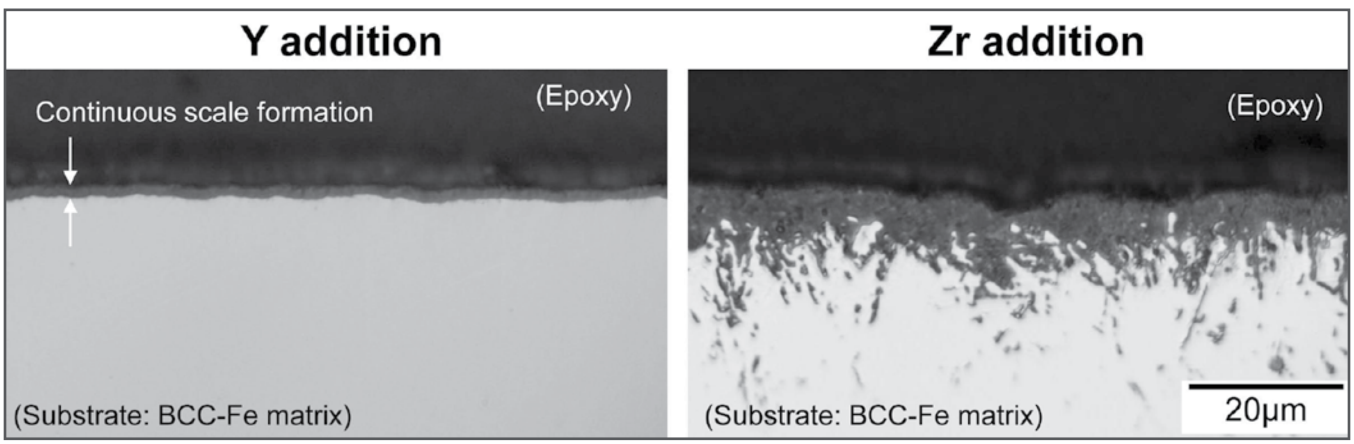


Figure 4. Surface cross-sectional views of C26M base alloys after steam ramp test up to 1450°C.

Hermeticity Evaluation of SiC/SiC Composite Tubes Relevant to LWR Cladding Application

Principal Investigator: Xunxiang Hu, ORNL
Collaborators: Takaaki Koyanagi and Yutai Katoh (ORNL)

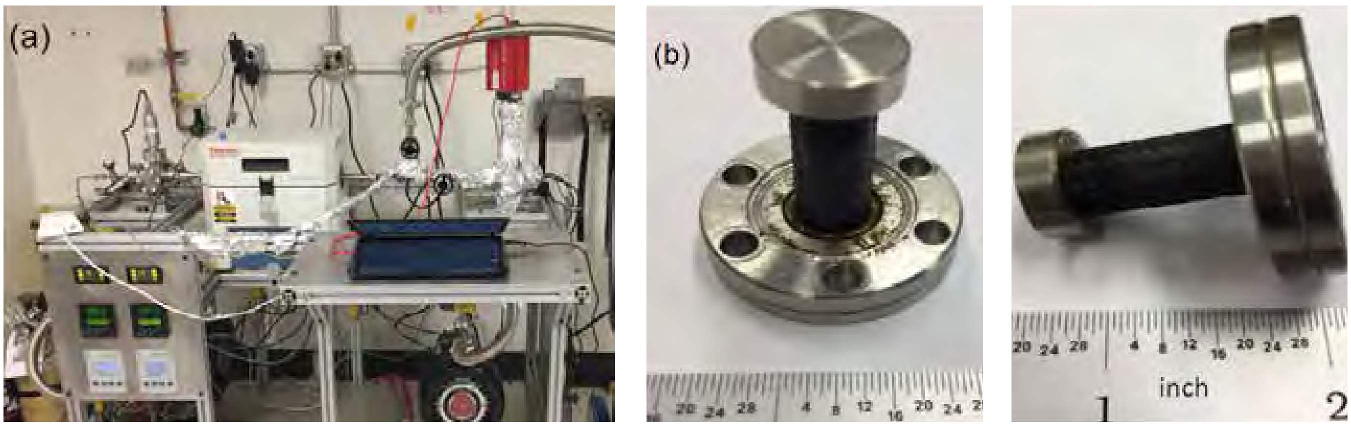


Figure 1. a) Permeation testing system at LAMDA. b) A tubular SiC/SiC composite tube mounted to a stainless steel sample fixture using epoxy.

The hermeticity of SiC/SiC composite tubes subject to LWR-relevant environments was evaluated using the high-resolution permeation testing station at ORNL.

Continuous SiC fiber–reinforced SiC matrix ceramic (SiC/SiC) composites are perceived as one of the leading candidate materials for accident-tolerant fuel cladding in light water reactors (LWRs). Potential loss of containment of the fission products due to penetrating cracking has been identified as one of the most critical technical issues for SiC-based nuclear fuel cladding. Therefore, evaluation of the hermeticity of SiC/SiC composite claddings subject to LWR-relevant environments is critically important. Gas permeation flux through SiC-based cladding is considered an important indicator for evaluating the hermeticity of the studied materials. However, gas permeation data for SiC and SiC/SiC composites in tubular configurations are very limited, especially for samples subjected to neutron irradiation.

Project Description:

The project goal is to establish an experimental capability for measuring the gas permeation flux through SiC/SiC composite tubes under unirradiated and irradiated conditions as a function of gas pressure at room temperature. The major challenges associated with testing tubular samples involve sealing the open ends of tubular samples and achieving acceptable leakage rate resolution in testing lab-scale tubular samples. This project is developing a high-resolution permeation testing system for the hermeticity evaluation of SiC/SiC composite tubes. The expected project output is measurement of the gas permeation flux in SiC/SiC composite tubes subject to LWR-relevant environments by employing the developed permeation testing system.

Accomplishments:

An ultra-high–vacuum permeation testing system (Figure 1) was designed and constructed at the Low Activation Materials Development and Analysis (LAMDA) lab of Oak Ridge National Laboratory (ORNL), which has a high detection resolution of 8.07×10^{-12} atm cc/s and 2.83×10^{-12} atm cc/s for helium and deuterium, respectively. The tubular samples were hermetically mounted to a stainless steel sample fixture using epoxy. A special irradiation vehicle was designed and fabricated to test SiC-based cladding under conditions representative of LWR to allow for a constant tube outer surface temperature in the range of 300~350°C under a representative high heat flux (~0.66 MW/m2) during one cycle of irradiation in the High Flux Isotope Reactor (HFIR) (equivalent to 2 dpa). The permeation testing system was employed to evaluate the hermeticity of the SiC/SiC tubes, provided by General Atomics (GA), French Alternative Energies and Atomic Energy Commission (CEA), and Korea Atomic Energy Research Institute (KAERI), following high-heat and low-heat neutron irradiation.

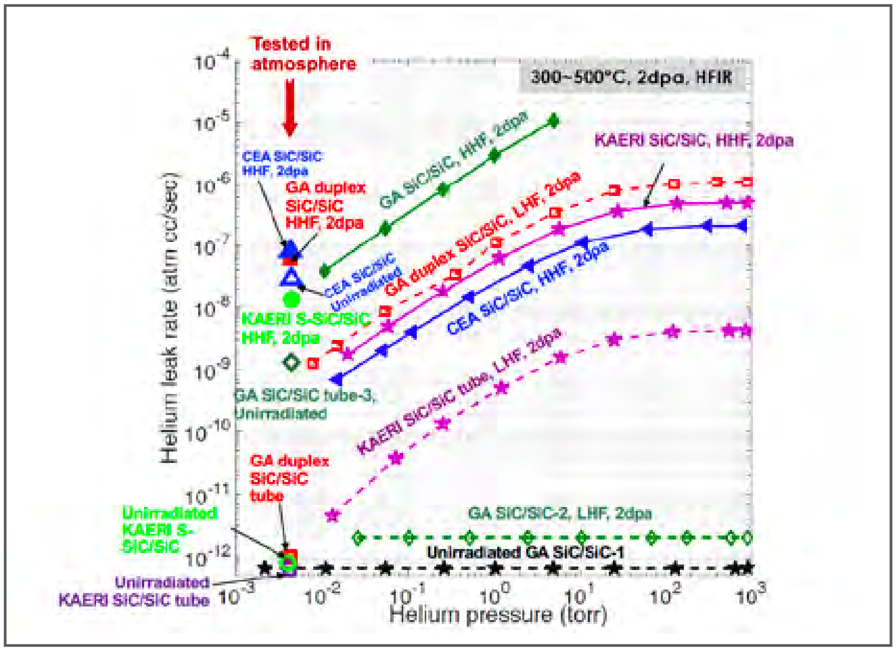


Figure 2. Helium leakage rate of SiC/SiC composite tubes following neutron irradiation (300~500°C, 2dpa, HFIR) under high (HHF) and low (LHF) heat flux as a function of helium pressure.

The results (Figure 2) show that the helium leakage rate increased with increasing helium pressure and then started to level off when the pressure rose above 50 torr. Only one neutron-irradiated GA SiC/SiC composite tube irradiated under a low heat flux remained hermetic. All the other tubes, following neutron irradiation under both high and low heat fluxes, lost their gas tightness. It is evident that neutron irradiation under HHF leads to increased helium leakage rates compared with the same sample before and after irradiation. Although a comparison of the same sample before and after neutron irradiation is not available, neutron irradiation significantly altered the helium leakage rate of the studied samples compared with the same type of SiC/SiC composite tube. Note that HHF neutron irradiation leads to a higher helium leakage rate for the studied samples compared with samples following LHF irradiation. The loss of hermeticity of the studied samples indicates the presence of penetrating cracks. Ongoing work aims at understanding the failure mechanisms by combining thermomechanical analysis and postirradiation examination (PIE).

Neutron Irradiation on First-Generation Coated SiC/SiC Composites

Team Members/ Collaborators: Takaaki Koyanagi, Stephen S. Raiman and Yutai Katoh (ORNL), Peter J. Doyle and Caen Ang (UTK), David M Carpenter and Kaicho Sun (MIT)

Figure 1. Image showing SiC fiber–reinforced SiC matrix composite tube coated with CrN. The tube length is 110 mm.

Extensive development of silicon carbide (SiC) fiber–reinforced SiC matrix composite cladding for light water reactors (LWRs) has identified benefits of the concept, including its inherent accident resistance and neutronic performance. Although SiC cladding appears to be attractive, critical feasibility issues such as hydrothermal corrosion and potential loss of fission gas retention due to cracking under normal operating conditions must be addressed. A possible solution to both issues is the application of a mitigating coating on the outer surface of the cladding. The team at Oak Ridge National Laboratory (ORNL) developed first-generation metal and ceramic thin coatings on a SiC composite tube (Figure 1). This research investigated responses of the coatings to neutron irradiation.

Project Description:

The objective of this research is to understand the effects of neutron irradiation on the coating microstructure and then to provide feedback toward optimization of coating microstructure and

The postirradiation examination of Cr, CrN, and TiN coatings on SiC identified the key degradation process and provided feedback for improvement of coating performance.

processing. The coating needs to be free from significant debonding and cracking to avoid degradation of the corrosion behavior and the hermeticity of the coating under an irradiation environment.

The test specimens included cathodic arc physical-vapor-deposited (PVD) Cr, CrN, and TiN; electro-plated Cr; and vacuum-plasma-sprayed Zr. Neutron irradiation was conducted under LWR-relevant temperature and dose conditions (at 300°C to 4.8 × 1024 n/m2 [>0.1 MeV], equivalent to nominal neutron damage of ~0.5 displacements per atom) under an inert Ar atmosphere at the Massachusetts Institute of Technology reactor (MITR). Multiscale microstructural characterizations were conducted using optical microscopy, x-ray diffraction (XRD), scanning electron microscopy (SEM), and transmission electron microscopy (TEM) at the Low Activation Materials Development and Analysis (LAMDA) Laboratory at ORNL.

The degradation process expected is debonding/delamination at the interface of the coating and substrate as a result of irradiation-induced stress; stress is built up when the coating and substrate swell differently under irradiation. This damage mechanism is a well-known challenge for dissimilar interfaces under irradiation; it also applies to SiC/coatings, as SiC swells by up to ~2 % in volume at 300°C. Therefore, the technical goal of the coating development is to achieve a high-adhesion-strength and/or ductile coating that withstands such swelling-induced stresses.

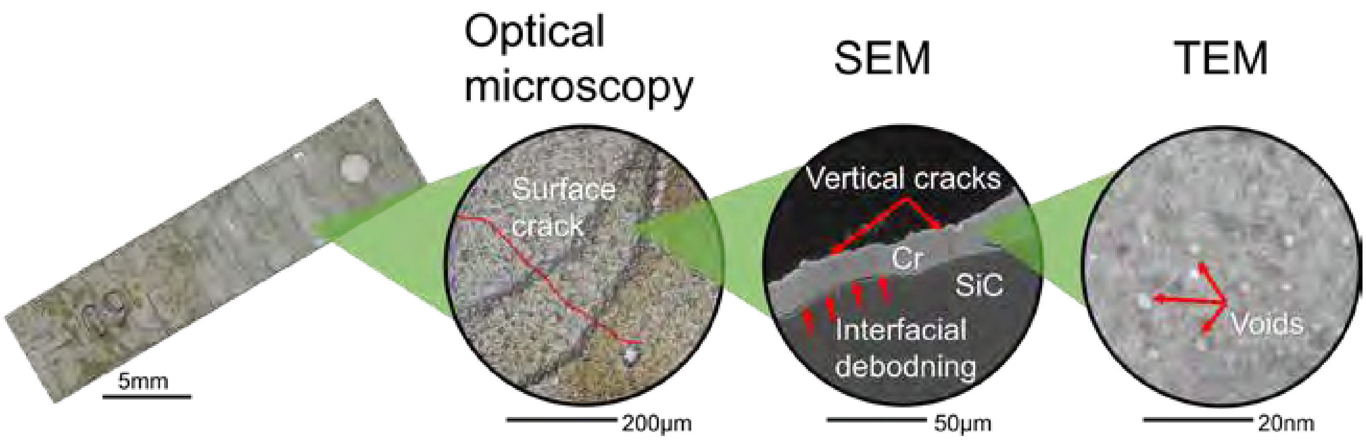


Figure 2. Multiscale microstructural characterization of neutron-irradiated Cr-coated SiC specimen using optical microscopy, SEM and TEM.

Accomplishments:

The microstructural characterizations of the neutron-irradiated coating specimens were completed in fiscal year (FY) 2019. Low-magnification optical microscopy found that PVD is the most promising processing route to produce a radiation-resistant coating. The electro-plated and plasma-sprayed coatings experienced macroscopic cracking and delamination as results of neutron irradiation. Therefore, the microstructures of the PVD coatings were investigated in detail. Multiscale analysis of the microstructure identified the radiation damage processes: the PVD Cr coating exhibited irradiation-induced surface cracking, which was associated with partial interface debonding and cracks propagating vertically to the coating/substrate interface (Figure 2). The interfacial debonding indicates (1) significant differential swelling between Cr and SiC and (2) insufficient interface adhesion. Limited void formation in the irradiated Cr (Figure 2) and XRD lattice constants suggests larger swelling of SiC than of Cr, which results in tensile stress in the Cr coating and explains the vertical cracks it experienced. A significant finding is that such radiation-induced cracking was almost absent in the PVD TiN coating, which warrants further investigation of this coating for future research. Another important finding is that different grades of CrN coating showed different responses to neutron irradiation; in a coating with an ~10 nm Ti interlayer between the SiC substrate and the CrN coating (Figure 3), the number of radiation-induced cracks was reduced. This result implies that a ductile interlayer

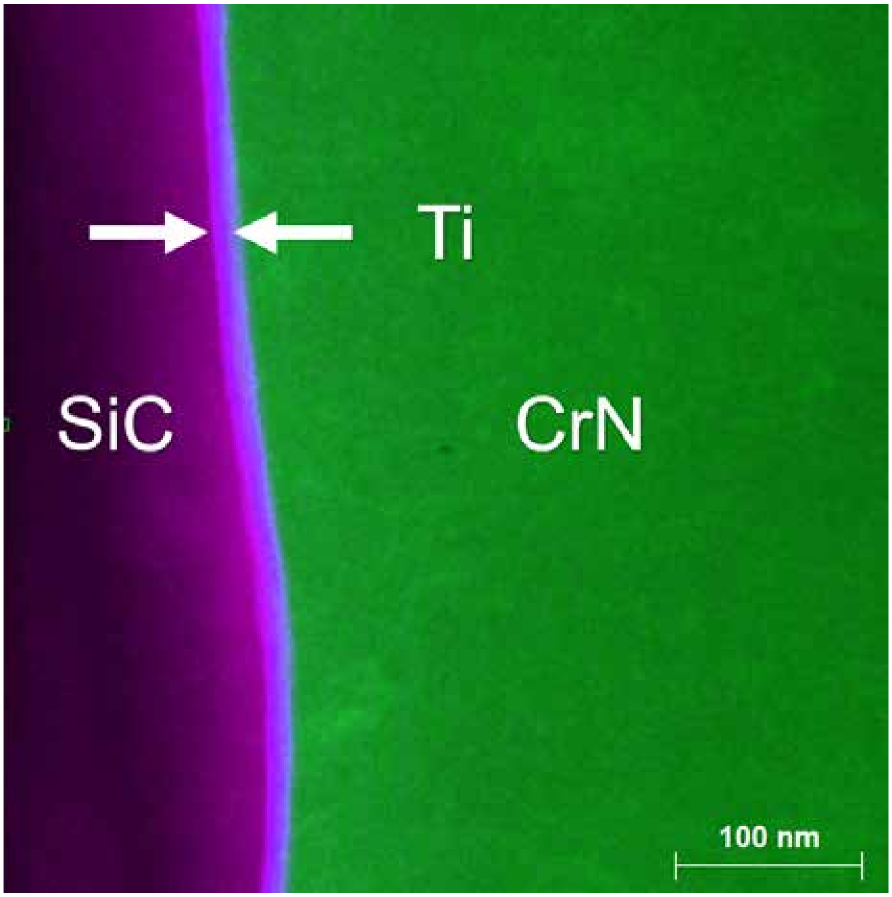


Figure 3. Layered elemental map of CrN coating on SiC with Ti nano-interlayer imaged by energy-dispersive x-ray spectroscopy.

is a solution to mitigate debonding and cracking of the coating due to irradiation-induced stresses. Based on the findings of this research, a second-generation coating for SiC composite cladding was designed to modify the coating microstructure (e.g., grain size and texture) for better ductility and to introduce an improved ductile interlayer at the coating/substrate interface.

Currently, the first-generation coatings neutron-irradiated under an LWR coolant environment are being investigated. The results of this research with neutron irradiation under an inert gas environment will be useful in understanding the synergetic effects of neutron irradiation and hydrothermal corrosion, which is a critical research topic for realizing SiC-based fuel cladding for LWRs.

ODS FeCrAl Tube Production and Characterization

Principal Investigator: Caleb P. Massey, ORNL
Team Members/ Collaborators: Sebastien N. Dryepondt, Maxim N. Gussev, Philip D. Edmondson and Kory D. Linton (ORNL)

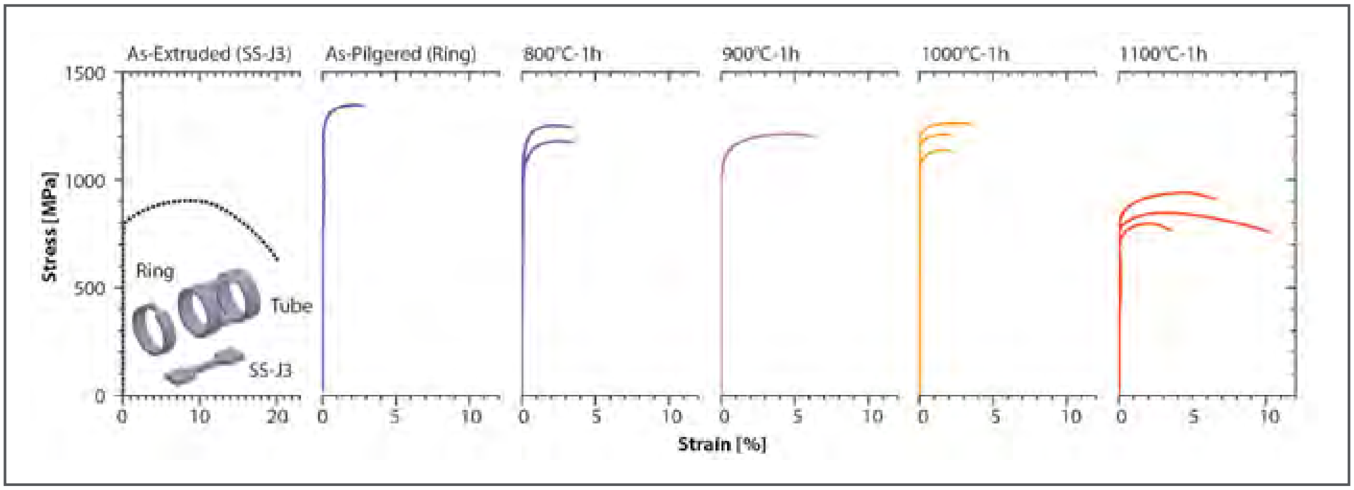


Figure 1. Stress-strain curves of ODS FeCrAl tubes as a function of processing and heat treatment.

DFeCrAl alloys have become strong candidates for accident tolerant fuel (ATF) cladding in light water reactor (LWR) designs due to their superior high temperature strength and oxidation resistance in comparison to existing Zr-based alloys. As part of an ongoing initiative to further improve upon the accident tolerance of the FeCrAl alloy system, advanced oxide dispersion strengthed (ODS) FeCrAl alloys are being studied that have the potential to simultaneously increase the strength and irradiation resistance of this ferritic alloy system. A high number density of nanoscale (~6 nm diameter) oxide precipitates serve as sinks for defects generated during radiation damage events, consequently decreasing the irradiation-induced hardening of ODS FeCrAl alloys during normal reactor operating conditions. These precipitates also serve as obstacles to dislocation motion within the microstructure, leading to increased high temperature strength in comparison to wrought FeCrAl variants.

The ultimate objectives of the current research on the ODS FeCrAl system are to (1) develop thermomechanical processing routes that provide the microstructures that provide

The modernization and optimization of thermomechanical treatmeatments used to fabricate advanced ODS FeCrAl is rapidly propelling this fuel cladding candidate material to increasing levels of technology readiness.

maximal irradiation resistance and high temperature strength without sacrificing alloy ductility, (2) to illustrate that thin-walled tubes of ODS FeCrAl alloys can be successfully manufactured using currently available commercial routes, and (3) to begin compiling material property data through separate effects tests for future comparison with currently qualified alloy systems.

Over the last half-decade, much research has aimed at designing new ODS FeCrAl alloys that have much finer dispersed particles. The lower Cr level would help minimize the irradiation-enhanced precipitation of a Cr-rich alpha-prime phase in the operating temperature regime of existing light water reactors, while the precipitates would serve to increase the strength of the alloy for possibly enhanced burst properties in comparison to Zr-alloys or even wrought FeCrAl variants. Due to the higher strength of the ODS FeCrAl alloy, one of the major goals of this project was to form as-extruded rods into thin-walled tubes through collaboration with an international commercial collaborator, which was successfully reported in 2018. However, the severe deformation associated with the fabrication of the thin-walled ODS FeCrAl tube resulted in a highly textured and brittle microstructure, which is not ideal for use as cladding in its current form. This project has thus aimed to develop processing-microstructure-mechanical property relationships for annealed thin-walled ODS FeCrAl tubes to assess the ability to regain ductility after recovery and/or recrystallization heat treatments.

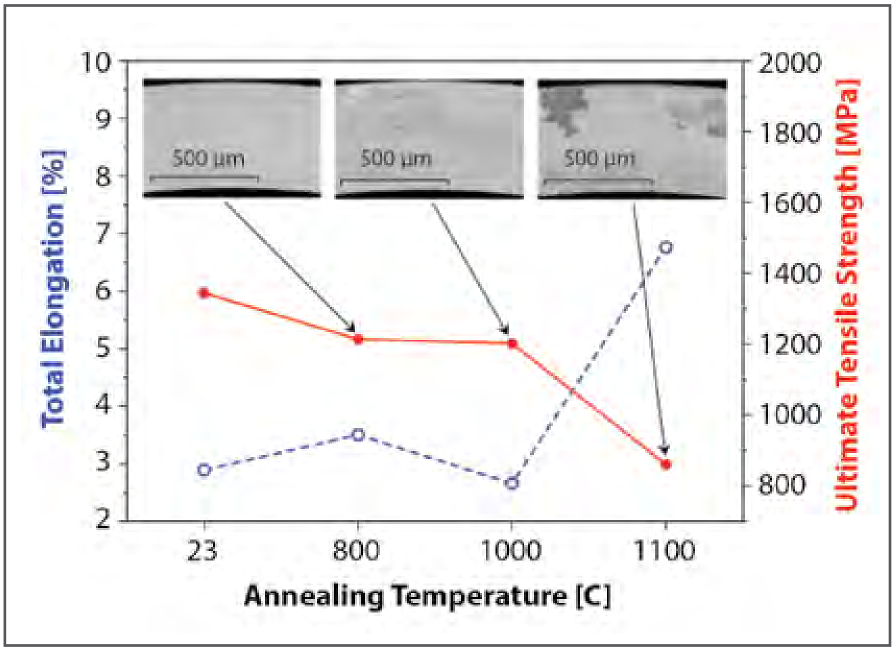


Figure 2. Effect of annealing temperature on total elongation and ultimate strength of ODS FeCrAl.

Accomplishments:

Progress has been made on understanding how the highly deformed grain structure in pilgered thin-walled ODS FeCrAl tubes adversely affects mechanical response. In the as-received condition, the thin-walled ODS FeCrAl tubes exhibited a total elongation less than 3% prior to fracture in a brittle manner. Even partial recrystallization was unable to increase the ductility of the alloy until large area fractions of recrystallized grains existed after a high temperature anneal at 1100°C for 1h. This is a significant challenge that needs to be alleviated in future large batch productions of ODS FeCrAl tubes. A secondary obstacle is that once significant amounts of recrystallization have occurred, the alloy's strength deteriorates through a variety of processes (i.e., particle coarsening, a drop in dislocation density, a drastic increase in grain size), so innovative methodologies are being implemented to retain the beneficial high temperature strength and irradiation resistance of these alloys even after recrystallization treatements.

The insights gained from the aforementioned annealing experiments are also supplemented by recent work on optimizing future larger scale productions of ODS FeCrAl rods and tubes through an in-depth study of the kinetics of particle nucleation, growth, coarsening in the same alloy powders prior to alloy consolidation. Using a combination of small-angle neutron scattering and atom probe tomography, the dispersion characteristics of the smallest nanoscale precipitates were tracked as a function of annealing temperature and time so that the effect of future annealing treatments on precipitates in these ODS FeCrAl alloys can be modeled. This is an important progression in the advancement of ODS FeCrAl since an improvement in the particle dispersion characteristics prior to tube fabrication widens the range of thermal annealing ranges that can be applied to as-pilgered thin-walled tubes without adverse affects to the dispersion of complex oxide precipitates within the microstructure. Using this coarsening model, an innovative two-step annealing procedure has been implemented for the creation of new master rods for more in-depth pre- and postirradiation mechanical properties of ODS FeCrAl Tubes.

2.4 LWR IRRADIATION TESTING & PIE TECHNIQUES

MiniFuel Irradiation of LWR Fuel Concepts

Principal Investigator: Christian Petrie, ORNL
Team Members/ Collaborators: Joseph Burns, Annabelle Le Coq, Alicia Raftery, Robert Morris, Jason Harp, Andrew Nelson and Kurt Terrani (ORNL)

Figure 1. Uranium mononitride (UN) kernels tested in the initial MiniFuel irradiation experiment (photography credit to Carlos Jones/Oak Ridge National Laboratory (ORNL), U.S. Department of Energy (DOE)).

One of the major limiting steps to qualifying a new fuel system is acquiring irradiation performance data. Historically this involved performing large, expensive, integral testing under prototypic conditions. While this approach has been successful, it is no longer economically feasible and the long timeframe (approaching 25 years) discourages the adoption of new fuel technologies. This motivates the need to accelerate fuel irradiation testing and adopt a new approach for fuel qualification that relies on the integration of accelerated, separate effects irradiation tests with modern modelling and simulation tools to rapidly acquire a more fundamental understanding of fuel performance prior to the execution of confirmatory integral tests.

Project Description:

The objectives of this work are to establish a separate effects irradiation testing capability at the High Flow Isotope Reactor (HFIR) to support accelerated fuel qualification and to demonstrate this new testing capability experimentally. The tests should enable rapid burnup accumulation under highly controlled experiment conditions to facilitate a more fundamental understanding of nuclear fuel performance. This contrasts with the historical approach where performance data were largely acquired only from complex integral tests that did not have the ability to isolate specific fuel phenomena. The experimental facility should have the ability to perform economical testing of many fuel specimens and have the flexibility to accommodate a wide range of fuel compositions, enrichments, and even geometries without requiring detailed analyses for each specific fuel variant. Successful demonstration of the new facility would include development of an adequate mechanical design, detailed neutronic and thermal analyses, fabrication and assembly of initial test experiments, irradiation in HFIR, and postirradiation examination to confirm that the test was performed as expected.

Figure 2. MiniFuel capsule components (photography credit to Carlos Jones/ORNL, DOE).

Accomplishments:

The first MiniFuel experiments tested UN kernels and tri-structural isotropic (TRISO) coated particles at temperatures (500–600°C) relevant to light water reactors (LWRs). The effects of carbon impurities, burnable absorbers, and density are being evaluated in the first irradiation target, which completed irradiation to a nominal burnup of 1% fission of initial metal atoms (FIMA) in November 2018. A second target with a nominal burnup of 6% FIMA is scheduled for completion in calendar year 2021. The 6 individual capsules from the first irradiation target were extracted from the target and punctured in the Oak Ridge National Laboratory (ORNL) Irradiated Fuels Examination Laboratory (IFEL) to measure fission gas release. The kernels were then transferred to the Irradiated-Microsphere Gamma Analyzer facility within IFEL for careful extraction of the kernels and the passive silicon carbide temperature monitors, which will

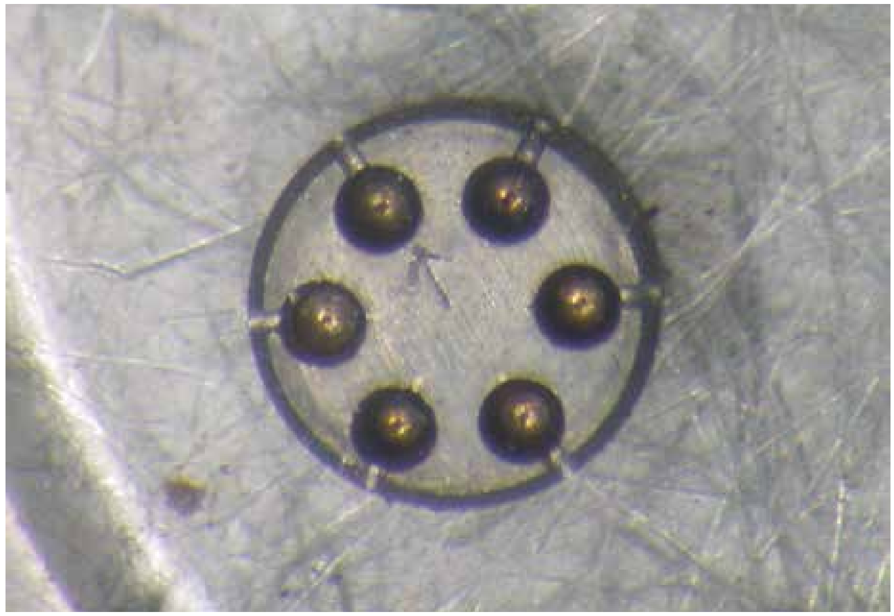


Figure 3. Hot cell extraction of one set of UN kernels contained inside a molybdenum cup.

be used to confirm the irradiation temperatures. Future non-destructive examination will include mass (scale) and volume (x-ray computed tomography) measurements which will be compared with pre-irradiation measurements to quantify fuel swelling. The kernels will then be sectioned and polished in preparation for microstructural characterization. Efforts are currently underway to prepare the next set of MiniFuel irradiations, with plans to include miniature fully ceramic microencapsulated (FCM) pellets with UN kernels and a silicon carbide matrix, as well as uranium silicide disk specimens fabricated at Los Alamos National Laboratory (LANL).

The success of the first MiniFuel experiment has also attracted attention from industry. General Atomics (GA) was awarded funding from DOE, Office of Nuclear Energy (ONE) to use MiniFuel to quantify swelling and fission gas release of uranium monocarbide (UC) fuels to support their Energy Multiplier Module (EM2), which is a gas-cooled fast reactor. GA has an aggressive schedule to collect fuel performance data within 2 years, consistent with the larger industry

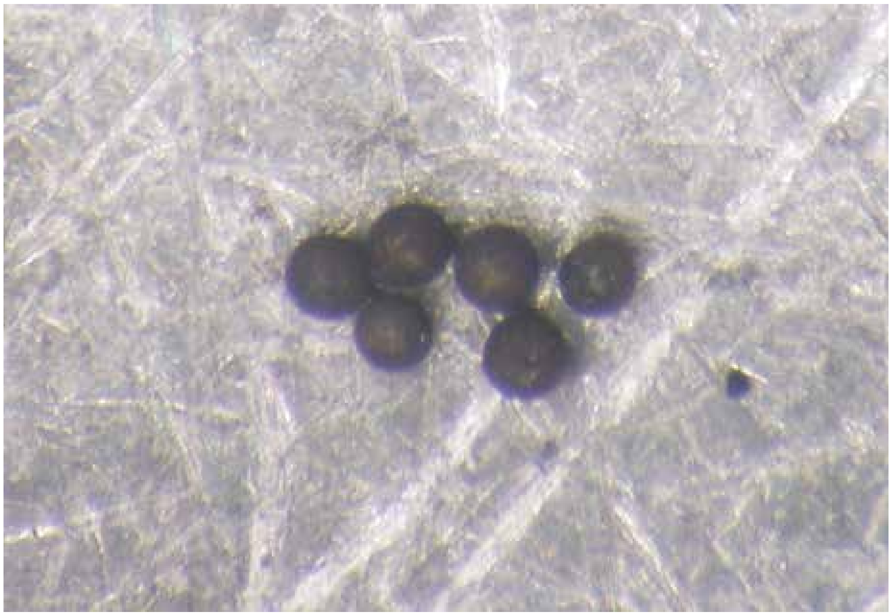


Figure 4. Closeup of one set of UN kernels after irradiation.

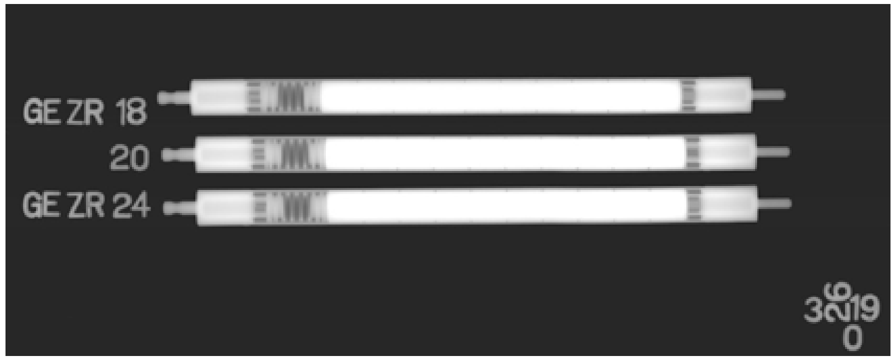
effort of accelerating the fuel qualification. The Nuclear Science User Facilities (NSUF) Program also recently selected a proposal lead by Kairos Power LLC for funding to test uranium oxycarbide (UCO) TRISO particles, developed under the Advanced Gas Reactor (AGR) program, at higher particle powers and lower temperatures to support their fluoride salt-cooled high temperature reactor. While the project will span 5 years, the fuel will reach end-of-life burnup levels in less than 100 days of irradiation in HFIR.

The first successful demonstration of an accelerated, separate effects fuel irradiation test in the High Flux Isotope Reactor gives increased confidence that the MiniFuel testing capabilities could become a critical part of an accelerated fuel qualification process for the nuclear industry.

Preparations to Add GE Rodlets to ATF-2

Principal Investigator: Connor Woolum, INL
Team Members/ Collaborators: Brian Durtschi and Gary Hoggard (INL); Global Nuclear Fuels (GNF); General Electric Global Research (GERC)

Figure1. Radiographic image of GE fuel rodlets.



Twelve ATF fuel rodlets fabricated by GNF in a collaborative effort among GNF, GE, and INL were received at INL in preparation for insertion in ATR Cycle 166B.

The Accident Tolerant Fuels (ATF) Program inserted the first test train into the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL) in 2018 in support of testing novel concepts designed to enhance the accident tolerance of light water reactor (LWR) fuels. As a follow-on to the initial ATF-2 insertion, continued collaboration with industry partners has resulted in plans to irradiate additional ATF concepts as part of the ATF-2 campaign.

Project Description:

Irradiation testing of ATF candidate concepts in prototypic environments is pivotal to understanding performance of the fuel-cladding system and also to support future fuel licensing efforts for commercial use. The ATF program has supported irradiation testing of concepts driven by industry partners since its inception and continues to do so through a variety of in-pile and out-of-pile testing.

The ATF-2 test train currently contains over two dozen fuel rodlets undergoing irradiation testing within the 2A loop of ATR. Several of these rodlets are planned to be removed from the test train after ATR Cycle 164A and replaced with General Electric (GE)/ Global Nuclear Fuels (GNF) concepts with enhanced accident tolerance. These concepts include both coated cladding concepts and novel FeCrAl cladding materials, paired with UO_2 fuel. In addition to these novel concepts, rodlets representative of a standard Boiling Water Reactor (BWR) fuel pin are included in the test matrix to serve as reference specimens.

Irradiation testing of these ATF concepts within ATR will enable a more thorough understanding of their irradiation performance and behavior. This increased understanding and performance data is critical both for licensing purposes and also to continue efforts focused on further enhancing the safety of fuel-cladding systems for current and future commercial nuclear reactors.

Figure 2. GE fuel rodlet for ATF-2B insertion.

Accomplishments:

A total of 16 fuel rodlets were fabricated by GNF, 12 of which are slated for installation in the ATF-2B test train, scheduled to be inserted in ATR cycle 168B. This fabrication and assembly effort represents the culmination of a significant amount of work by GNF, General Electric Global Research (GERC), and INL personnel. The rodlets were received at INL in March of 2019, after several months of work by all parties involved in the effort.

GE/GNF established a test matrix of rodlets to add to the ATF-2 test train, and subsequent iterations on the design ensured the rodlets would be suitable for fabrication, assembly, irradiation testing, and postirradiation examination (PIE). The fuel rodlet design was constrained by a variety of factors. The design had to be compatible with the overall ATF-2 test train design and also compatible with fabrication and assembly techniques. Potential impacts of the design on irradiation performance were considered, as was design impacts to rodlet handling and PIE. Additionally, the design and assembly of the rodlets had to allow for their inspection and qualification for insertion into ATR.

Upon receipt of the rodlets at INL, additional inspections were performed to verify rodlet integrity after shipping. A helium leak check demonstrated the rodlets hermeticity, and radiography verified that no rodlet internals had become damaged or dislodged.

Throughout the effort INL, GE, and GNF personnel collaborated closely in order to execute the assembly and qualification of the test rodlets. The collaborative efforts proved extremely successful and will result in the insertion of 12 rodlets in ATR Cycle 166B, currently scheduled for Fall 2019.

Additional samples beyond those planned for ATR insertion were assembled as part of the same campaign, intended to serve as spares in the event a rodlet was damaged during shipping. Given no rodlets were damaged in transit, these additional samples may be used for future fuel safety research in Transient Reactor Test Facility (TREAT) as part of the ATF-3 campaign, or for out-of-pile testing.

Design of Irradiation Creep Tests for Coated and Composite Cladding

Principal Investigator: Padhraic Mulligan, ORNL
Team Members/ Collaborators: Patrick Champlin, Andrew Nelson, Scarlett Clark, Christian Petrie, Alan Frederick, Kory Linton (ORNL) Dan Lutz, Sarah DeSilva, Yang-Pi Lin, Francis Bolger (GNF)

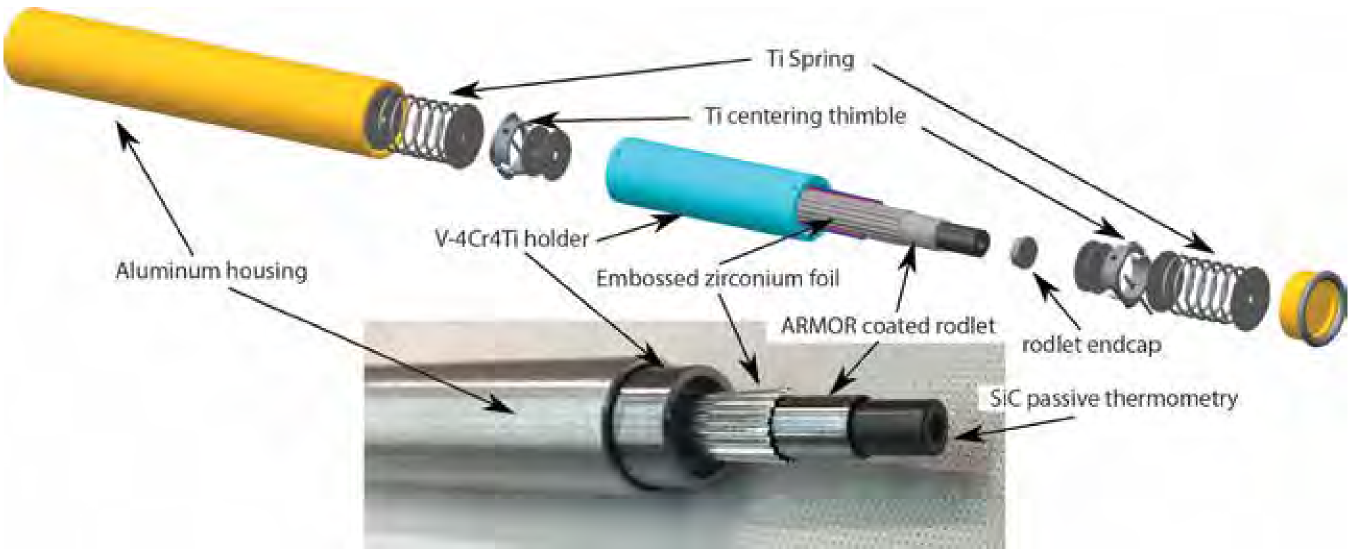


Figure1. Pressurized rodlet assembly used for irradiation in HFIR.

ARMOR coated fuel cladding has the potential to delay the onset of hydrogen generation in loss of coolant accident (LOCA) scenarios, enabling greater safety margins in existing nuclear reactors. This work seeks to experimentally determine the impact of ARMOR coating on irradiation induced creep in biaxially stressed, Zircaloy-2 rodlets in the High Flux Isotope Reactor (HFIR).

Project Description:

Irradiation induced creep is a critical mechanical phenomenon for materials in nuclear applications. While the creep rate of standard zirconium fuel cladding is well known, the impact of coatings of various thicknesses and properties on cladding creep has not been quantified. To accomplish this measurement, thin-walled zirconium rodlets were fabricated for placement inside standard HFIR irradiation capsules. Rodlets are coated with various thicknesses of Global Nuclear Fuels (GNF's) coated cladding concept (ARMOR) and internally pressurized to generate a circumferential hoop stress in the rodlet wall. An embossed, compressible zirconium foil surrounds the pressurized rodlet in the irradiation capsule, maintaining a constant heat transfer path, and therefore constant temperature, as the rodlets creep radially

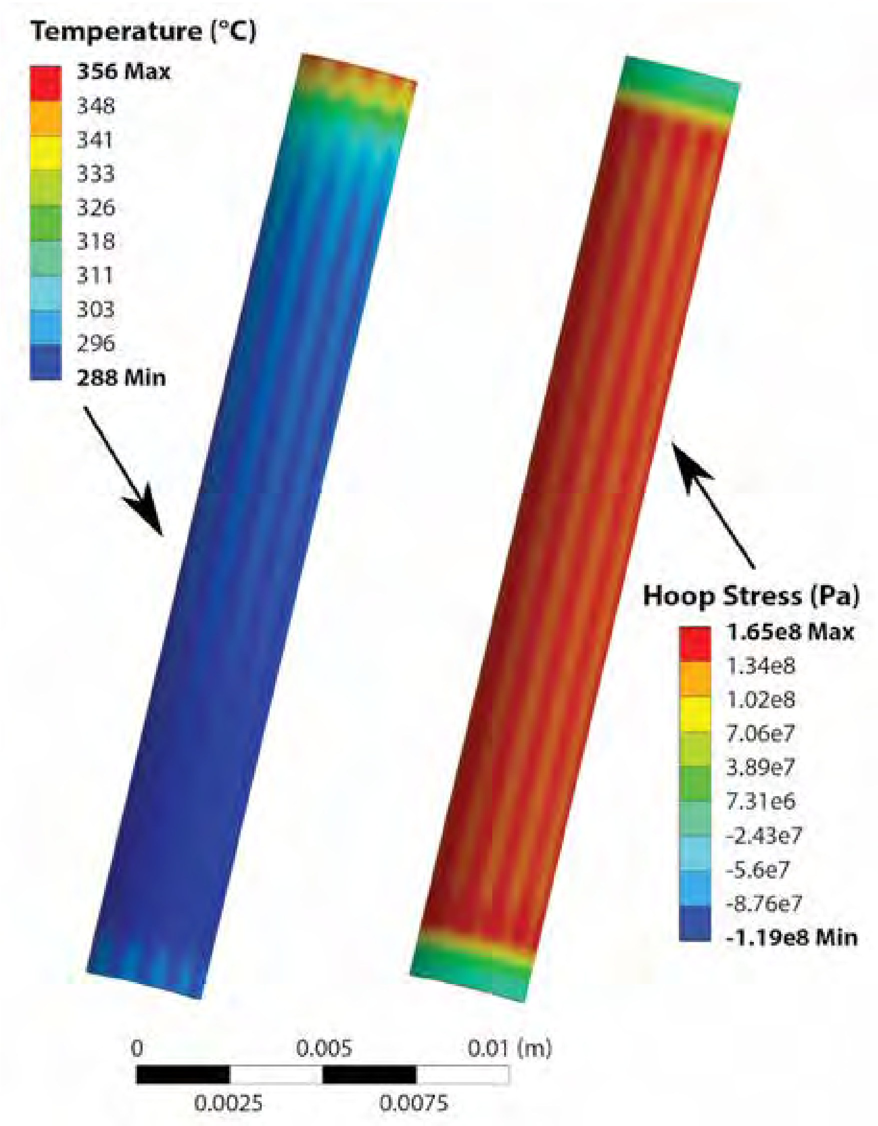


Figure 2. Temperature and hoop stress distribution in pressurized rodlet during irradiation, determined using ANSYS finite element software.

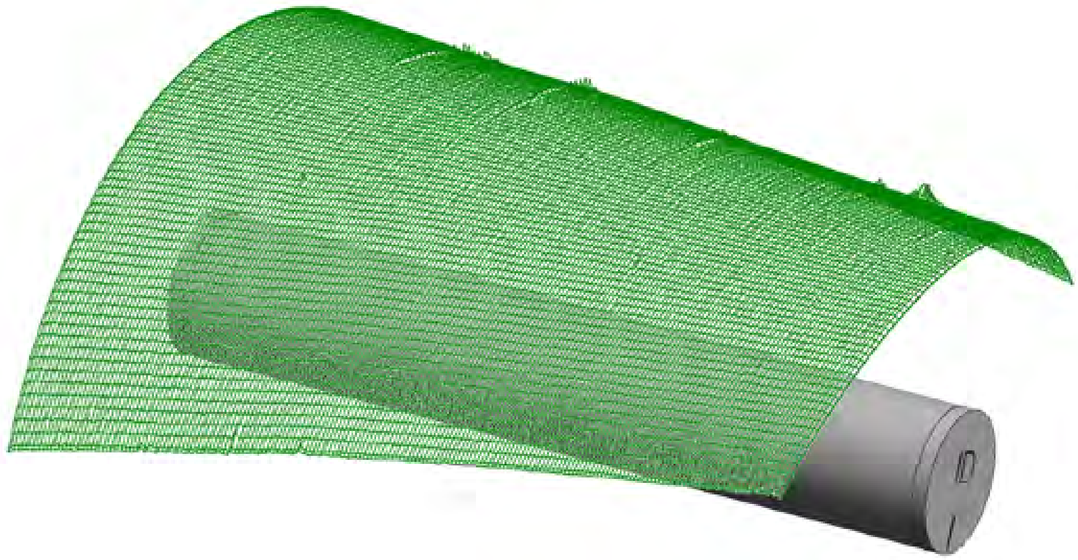


Figure 3. Representative dimensional (dimensions removed) scan of ARMOR coated rodlet to facilitate measurement of irradiation creep during postirradiation examination.

during irradiation. The diameter of each rodlet is carefully measured before and after irradiation using several techniques including non-contact laser profilometry, conventional contact micrometry, and high-resolution digital image processing. The creep deformation measured in coated rodlets will then be compared to the creep observed in uncoated rodlets to determine the impact of ARMOR coating on cladding irradiation creep. Out-of-pile test will be conducted in parallel to quantify thermal creep in coated rodlets. Work is being conducted to NQA-1 standards for inclusion in a Nuclear Regulatory Commission (NRC) license application, allowing for accelerated deployment of this safer and more reliable technology to the country's current reactor fleet.

Accomplishments:

Project design goals sought to replicate prototypic light water reactor (LWR) cladding stress and temperature

This experiment will provide the first quantitative measurement of irradiation induced creep in coated zircaloy, a critical parameter for NRC licensure.

conditions on a reduced scale, compatible with the dimensional constraints of standard HFIR irradiation capsules. Finite element analysis models using HFIR specific heat generation rates and boundary conditions were solved to determine irradiation capsule component dimensions for design and safety purposes. Safety calculations have been documented and are under review for HFIR irradiation. Nominally high and low circumferential hoop stress states in the thin-walled zirconium rodlets were calculated from these models and used to determine pre-irradiation rodlet fill pressures. Resistive forces caused by the embossed foil surrounding the rodlets were analyzed and compensated by increasing the rodlet internal pressure. Zirconium thin-walled tubes and endcaps were machined and electron beam welded at ORNL, and subsequently ARMOR coated by GNF. Several methods capable of measuring precise (<1 μm) diameters pre- and post- irradiation were identified and will be used to quantify dimensional changes in the rodlets. Microstructure, tensile strength, and texture measurements were performed at Oak Ridge National Laboratory (ORNL) on the Zircaloy-2 barstock used to fabricate rodlets. A separate out-of-pile thermal creep test assembly was designed by ORNL and is being installed to determine ARMOR coating impact on thermal creep.

Postirradiation Examination of FeCrAl Irradiated at Light Water Reactor Conditions

Principal Investigator: Fabiola Cappia (INL)
Team Members/ Collaborators: Brian Frickey, Shane Haney, Robert Cox, Francine Rice, Glen Papaioannou, David Sell, Katelyn Wheeler, John Stanek (INL), Jason Harp and Kevin Field (ORNL)

The PIE results of this irradiation campaign provide fundamental data to assess the fuel-cladding interaction between UO_2 and ATF candidate iron-based cladding.

One of the main strategies in current Research and Development (R&D) of Accident Tolerant Fuels (ATFs) aims at developing and qualifying new cladding materials that could overperform Zr-alloy claddings currently used in Light Water Reactors (LWRs). Particularly, Zircaloy suffers from rapid exothermic oxidation accompanied by hydrogen gas generation under certain accident conditions. If these phenomena could be mitigated, a large gain in safety margin could be achieved, which would imply economic benefit through increased operational flexibility. Among others, advanced iron-based alloys proved to have desirable attributes in terms of improved oxidation resistance in water-vapor containing environment, which is a fundamental property to increase coping time in accidental scenarios. Several efforts have been undertaken to develop and qualify 'nuclear grade' FeCrAl alloys, studying radiation tolerance, mechanical properties at elevated temperatures, aqueous corrosion resistance, oxidation resistance and compatibility with the fuel.

Project Description:

The FeCrAl alloys have been studied for nuclear applications since the 1960s, due to their attractive resistance to oxidation at high temperature. Comprehensive studies of FeCrAl alloys optimized for deployment as nuclear cladding has been performed, but data assessing reactivity between the FeCrAl alloys and LWR fuels during in-reactor operation remains limited. As part of the ATF-1 irradiation tests, three Oak Ridge National Laboratory (ORNL) Fuel-Cladding Chemical Interaction (FCCI) rodlets were designed and fabricated to specifically study the potential FCCI between advanced FeCrAl alloys and UO_2 under irradiation. The rodlets consist of a series of FeCrAl disks and UO_2 fuel disks whose design resembles diffusion couples. The design incorporates several layers of clad-fuel specimens enabling the test of different cladding materials and surface treatments within a single irradiation. The selected alloys comprised one alloy from Phase I of the ORNL FeCrAl alloy development strategy, namely B135Y3, two alloys from Phase II, C35M and C37M, as well as the commercial alloy, Kanthal

APMT®, used as reference. Both as-fabricated and pre-oxidized alloys have been tested, in order to evaluate the effectiveness of pre-existing internal oxide layer in reducing FCCI. The disks of FeCrAl and fuel are retained within an external ring which limits radial swelling of the UO_2 fuel as well as favors constant contact between the fuel and cladding disks, forming what is called a H-cup configuration. Each rodlet included a total of 13 H- cups (1 fuel disk and 2 cladding disks per H-cup). The rodlets were inserted for irradiation in the Advanced Test Reactor (ATR) in December 2016. The first rodlet completed irradiation in 2018 and postirradiation examination (PIE) has been conducted on this first rodlet throughout the fiscal year.

Accomplishments:

The PIE took place in the Hot Fuel Examination Facility (HFEF) at Idaho National Laboratory (INL). The PIE included rodlet visual examination, neutron radiography, gamma scanning, and fission gas release measurements. Thermal neutron radiography is shown in Figure 1. Each H-cup is visible in the radiograph and no major dimensional deformation of the disks occurred. The

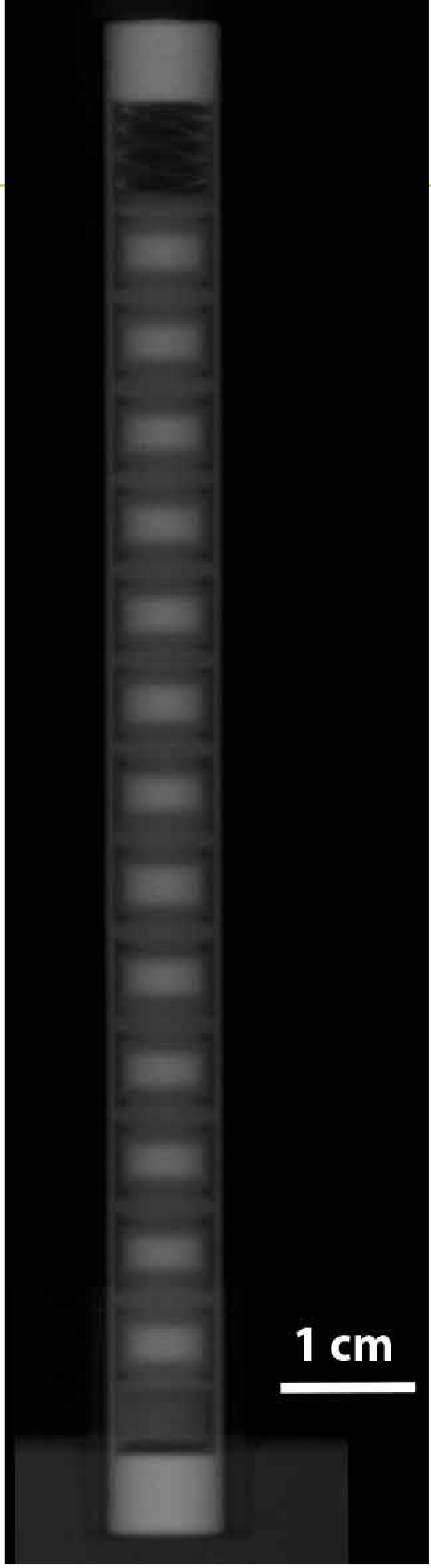


Figure 1. Thermal neutron radiography of the ORNL FCCI rodlet.

Figure 2. Retrieval of the H-cups in HFEF.

gamma scan showed that the fission product 137Cs remains confined in the fuel disks within each H-cup. Following rodlet non-destructive examinations, disassembly of the rodlet to retrieve the single H-cups was performed in the containment box of HFEF (Figure 2). Each H-cup was mounted in a metallography mount, ground and polished to reveal the FeCrAl-UO_2 interfaces. One of the metallography mounts is shown in Figure 3a. It corresponds to the cross section of one of the samples with C35M disks pre-oxidized at 1000°C for two hours. The fuel is in contact with the FeCrAl disk along most of the surface, but small gaps could be observed towards the outer part of the fuel slice. A high-magnification image of the central part of the lower UO_2-FeCrAl interface is shown in Figure 3b. The microstructure of the fuel shows extensive porosity,

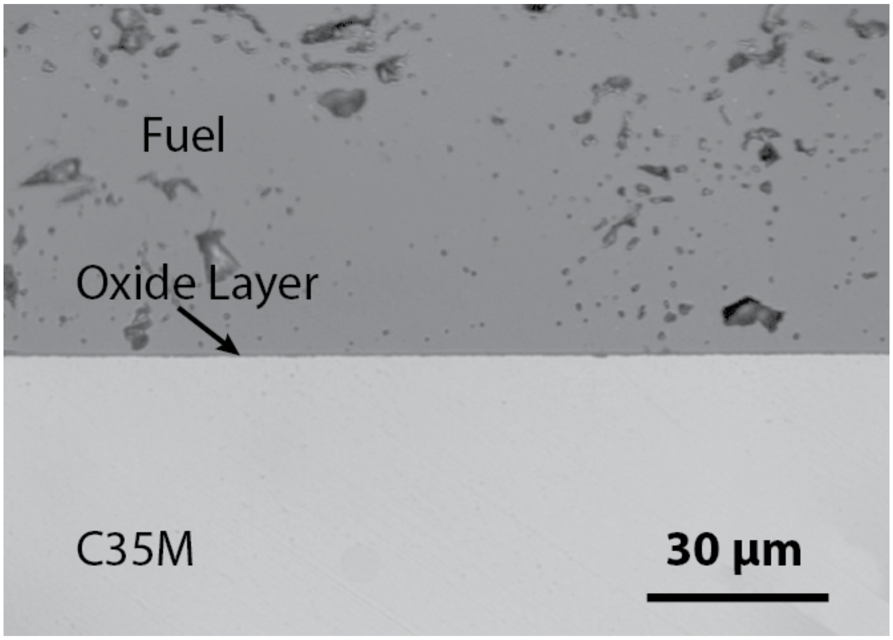


Figure 3. a) Cross-section of one of the H-cups. b) High magnification optical microscope image of the fuel-cladding interface.

consistent with the corresponding fission gas release measurements. The large surface-to-volume ratio of the fuel disks in this configuration could be the factor favoring larger fission gas release compared to standard LWR fuels irradiated to similar conditions. The pre-existing oxide layer is visible in the micrograph. The thickness is consistent with the measurements performed pre-irradiation. The morphology of the interface does not reveal an obvious chemical interaction between the fuel and the alloy at low burnup (8.7 GWd/tHM). Similar observations were made on the other capsules containing the non-oxidized alloy. The optical microscopy analyses on the other samples are in progress and will be compared to provide a comprehensive evaluation of the various interaction layers.

Accident Tolerant Fuels - Series 1 (ATF-1)

Principal Investigator: David Kamerman, INL
Team Members/ Collaborators: Stanton Byington, Bryon Curnutt, Cody Hale, Kelly Ellis, Chris Murdock and Connor Woolum (INL)

ATF-1 is testing new technologies such as improved fuel and/ or cladding concepts using a demonstrated design-vehicle in the ATR irradiation test environment.

The ultimate goal of the Accident Tolerant Fuel (ATF) program is to demonstrate improved fuel and/or cladding concepts offering the potential to replace the Zircaloy-UO_2 system currently used throughout the light water reactor (LWR) industry. To support this goal, the congressional appropriation language for fiscal year (FY) 2012 included specific language for Department of Energy-Nuclear Energy (DOE-NE) to initiate an aggressive research, development, and demonstration (RD&D) program for LWR fuels with enhanced accident tolerance. The test data collected as part of the ATF program will support demonstration of lead test rods (LTRs) or lead test assemblies (LTAs) in a commercial LWR within 10 years (i.e., by the end of FY 2022).

As a step toward this goal, an irradiation test series of drop-in capsule experiments, denoted ATF-1, was fabricated and inserted into the Idaho National Laboratory (INL) Advanced Test Reactor (ATR) beginning in FY 2015. As part of feasibility testing, ATF-1 experiment fuel cladding is not exposed to the ATR Primary Coolant System (PCS) as the fueled rodlets are encapsulated in stainless steel. Upon reaching pre-defined irradiation test objectives, the experiments are discharged from ATR and shipped to the Material and Fuels Complex (MFC) for postirradiation examination (PIE) and/or transient testing in the Transient Reactor Test Facility (TREAT).

Project Description:

The irradiation experiment assembly (Figure 1) loaded into the ATR consists of the basket, top spacer, and seven vertically-stacked capsule assemblies and/or dummies in each of the three channels (for a total of twenty-one (21) capsules/dummies per basket). The capsule used in the experiment provides the pressure boundary for the experimental materials located within. In the event of a rodlet cladding failure, the stainless steel (316L) capsule prevents fission products from entering the ATR PCS. The test rodlet is intended to represent a miniature length pressurized water reactor (PWR) fuel rod, nominally

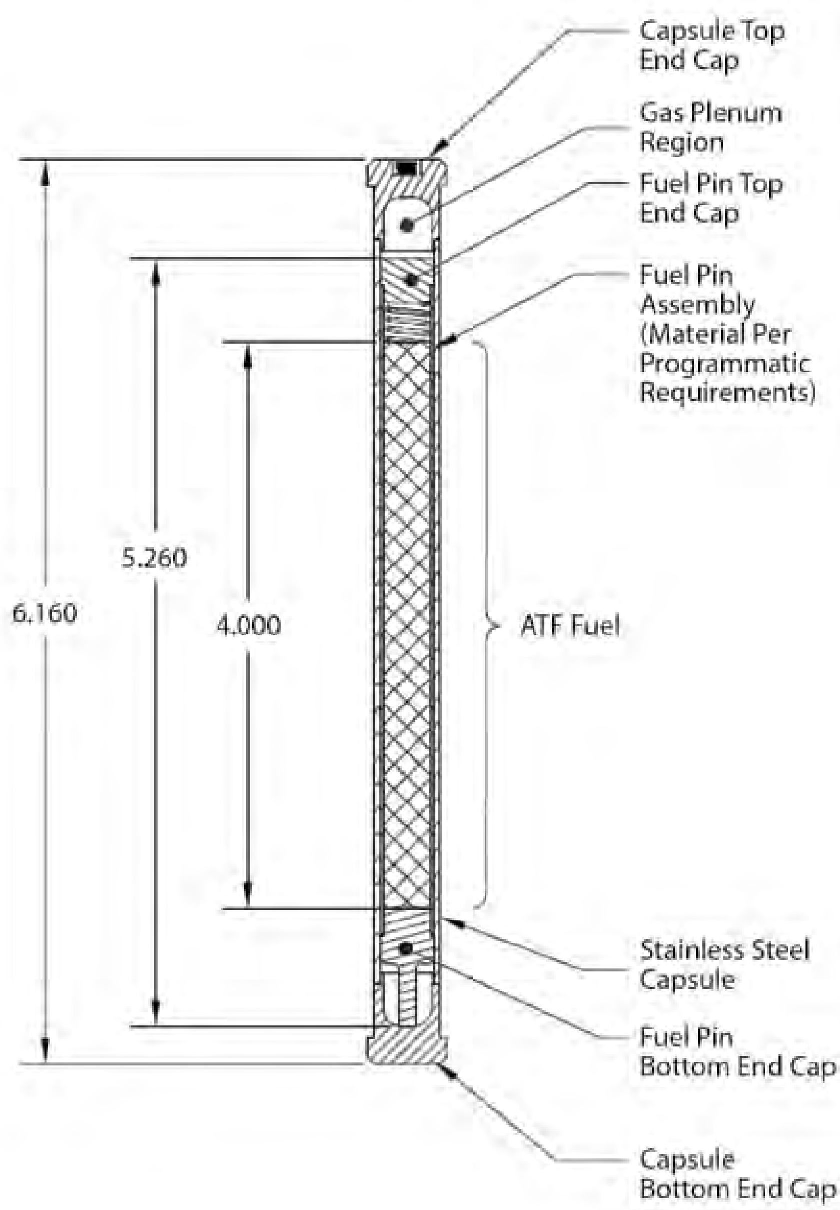


Figure 1. ATF-1 assembly.

Figure 2. ATF-1 basket cut-away.

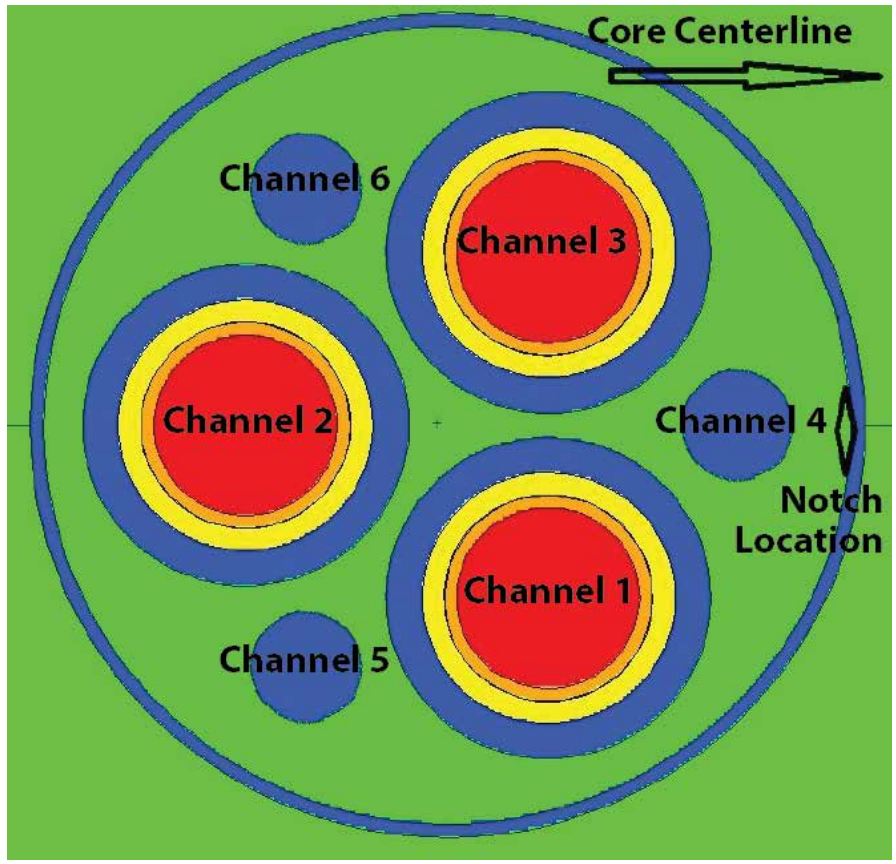


Figure 3. ATF-1 basket diagram.

prototypic in the radial dimension. For the ATF-1 irradiation test series, the test rodlet generally contains fuel in the form of pellets or slices, holdown springs, insulator pellets, and a gas plenum. The rodlets were fabricated from various cladding materials and fuel forms. A typical ATF-1 capsule and rodlet assembly is shown in Figure 2. The basket of the experiment assembly maintains the capsule assembly configuration within the ATR. The current basket design is an aluminum three-hole basket with additional flux wire monitor channels as displayed in Figures 3 and 4. An additional figure identifies the basket channel numbering scheme for capsule and flux monitor wire insertion.

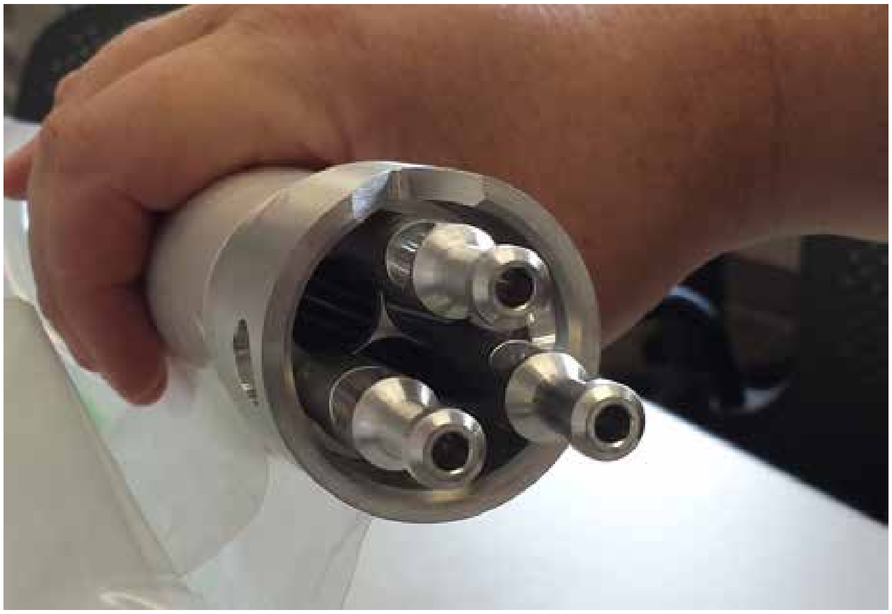


Figure 4. ATF-1 basket1.

Accomplishments:

At the conclusion of FY2019, fifteen of the thirty-one ATF-1 capsules had reached target burnup and had either completed PIE or are in process. FY2019 scope included two ATR cycles, 164B and 166A. Cycle-specific analyses were performed for these two cycles. Additional flux-wire monitors were fabricated to support the on going irradiations for these and future cycles. ATR canal reconfiguration work was also performed to support the insertion of the ATF-1 experiments and baskets into the ATR.

Multiple ATF-1 capsules reached target burnup either just prior to FY2019 or during. ATF-41 reached target burnup in the 162B cycle and shipped 4/22/19 for PIE at the Hot Fuel Examination Facility (HFEF). ATF-10 reached target burnup in the 164A cycle and shipped 4/22/19 for PIE at HFEF. ATF-30, ATF-34, and ATF-44 reached target burnup in the 164B cycle and shipped 4/22/19 for PIE at HFEF. ATF-44 represents the last of the Los Alamos National Laboratory (LANL)-1 concepts in the ATF-1 series.

Scoping for a new ATF-1 test group of capsules for Framatome was completed. FY2020 scope will include design, fabrication and assembly of these capsules for a future insertion in the ATR. Scoping for two more ATF-1 shipments in FY 2020 was also performed.

ATF-2 Progress

Principal Investigator: Ed Mai, INL
Team Members/ Collaborators: Bryon Curnutt, Brian Durtschi, Stephen Evans and Gary Hoggard, Connor Woolum (INL)

Figure 1. ATF-2 assembly team with ATF-2 test train and the Advanced Test Reactor Complex Test Train Assembly Facility.

The ATF-2 test train is the only experimental apparatus in the United States dedicated to steady-state testing of commercial light water reactor fuels.

The Nuclear Technology Research and Development (NTRD) Advanced Fuels Research, Development, and Demonstration (RD&D) program supports the Department of Energy (DOE) Office of Nuclear Energy (NE) mission and objectives through development of advanced fuel technologies within the Advanced Fuels Campaign (AFC). The accident tolerant fuel (ATF)-2 experiment results will support the Advanced Light Water Reactor (LWR) Fuels Development part of the AFC.

Irradiation of the ATF-2 experiment began June 12, 2018, and it is currently being irradiated in the Idaho National Laboratory (INL) Advanced Test Reactor (ATR). Design improvements and reconfiguration of the test train are planned for an October 2019 ATR insertion (cycle 166B outage) designated ATF-2B.

Project Description:

In 2011, following the Great East Japan Earthquake, resulting tsunami, and subsequent damage to the Fukushima Daiichi nuclear power plant complex, DOE-NE, in collaboration with the nuclear industry, shifted RD&D emphasis to accident performance of fuels. Subsequently, in fiscal year (FY)12, Congress included specific language for DOE-NE to initiate the Enhanced ATF program, and DOE-NE designated funding for Funding Opportunity Announcement (FOA) competitive awards to identify, develop, and test advanced LWR fuel concepts. Three industry-led teams were selected: Westinghouse Electric Company, LLC (WEC); General Electric Global Research (GERC); and AREVA (now known as Framatome). Thus, ATF-2 is a joint effort of the INL with WEC, GE, and Framatome. General Atomics (GA) is a partner with WEC.

ATF-2 experiment results will play a critical role in confirming certain performance aspects of new fuel/cladding concepts under pressurized water reactor (PWR) operating conditions. The ultimate goal of the ATF program is to demonstrate improved fuel/cladding concepts offering the potential to replace the Zircaloy-UO_2 system currently used throughout the LWR industry.

ATF-2 consists of specimens that are pin-type which are scaled-down in size from commercial-size fuel pins. The specimens consist of sealed cladding tubes that contain fuel pellets. The cladding tubes and fuel pellets

Figure 2. Installation of ATF-2 in the Advanced Test Reactor Loop 2A in the Center Flux Trap.

consist of various materials of interest chosen by the parties involved such as variations of zirconium cladding and coatings and doped UO_2 fuel pellets. The specimens are 0.360 – 0.374 inch diameter and six to twelve inches long. These are held vertically in Loop 2A by holders and irradiated under prototypical PWR conditions.

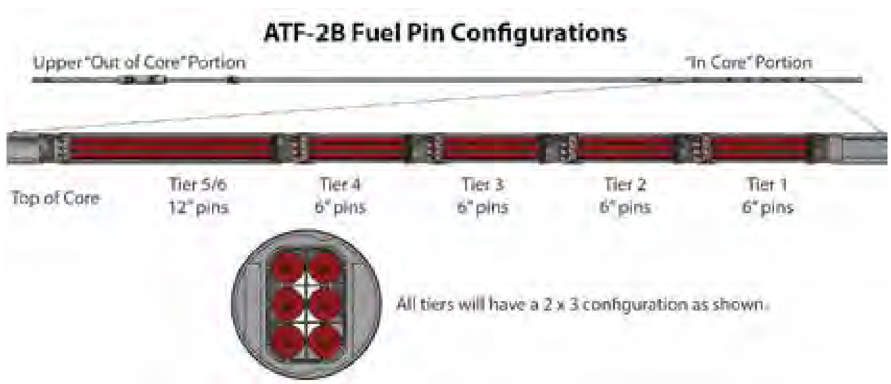


Figure 3. ATF-2B diagram with new design improvements for BWR type specimens (Tiers 3 & 4) and optimized prototypical operation.

Accomplishments:

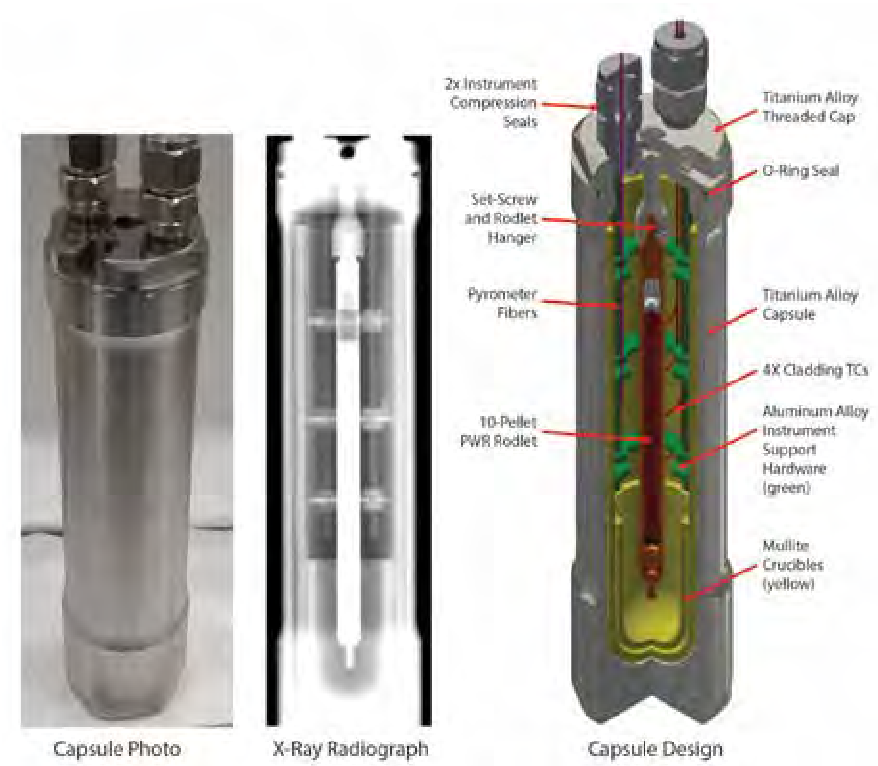
ATF-2 completed 176 days of irradiation in ATR through September 30, 20219. In addition, new design improvements were completed to accommodate larger diameter boiling water reactor (BWR)-type specimens and reconfigure the existing specimens to optimize desired linear heat generation rates and cladding temperatures for PWR prototypic conditions. This completed a level 2 milestone.

BWR-type specimens have been received from Global Nuclear Fuel (GNF) and INL is in the final stages of fabrication of new tooling and experiment test train hardware. Reconfiguration of ATF-2 to ATF-2B is on-schedule to be completed in October 2019.

Completion of the SETH-A/E Test Series to Commission TREAT Experimental Capabilities

Principal Investigator: Nicolas Woolstenhulme (INL)
Team Members/ Collaborators: Austin Fleming, Devin Imholte, Connor Woolum, Cody Race, Connie Hill, Dan Chapman, Spencer Snow, Doug Dempsey, Clint Baker, David Kamerman and Colby Jensen, Daniel Wachs (INL)

Figure 1. Photo, X-radiograph, and design rendering of the SETH capsule.



The first series of SETH capsule irradiation tests in TREAT successfully commissioned several modern experiment capabilities, familiarized engineering staff with experimental methods, and gathered crucial nuclear heating data to pave the way for future fuel safety research in TREAT.

The Transient Reactor Test facility (TREAT) at Idaho National Laboratory is a graphite-based shaped-transient reactor constructed in the late 1950's to support nuclear heated fuel safety research. After an extended operational hiatus, TREAT was refurbished and restarted in late 2017 to reclaim the domestic capability for this type of research. The first series of modern fueled experiments were performed starting in 2018 in order to commission modern irradiation capabilities and gather nuclear heating data to aid design of near future accident tolerant fuels (ATF) objectives.

Project Description:

TREAT performed transient research on multiple fuel systems concerning several reactor types including light water reactor (LWR) during its early history (1960's era). Due to the later arrival of LWR-focused transient test reactors (e.g., Power Burst Facility, now decommissioned) TREAT was incrementally upgraded and reconfigured with an increasing focus on engineering scale sodium loop tests in the era leading up to its operational suspension in 1994. Following system refurbishment and return to operation in late 2017, TREAT's configuration represented a more capable facility, but historic data and experimental capabilities lacked relevance for modern ATF/LWR objectives. A series of first-of-a-kind transient shapes were performed on a nuclear mockup experiment to demonstrate TREAT's capability and develop model calibration data for LWR category transients including reactivity-initiated accident and loss of coolant accident with great success.

An innovative approach to experiment design was also developed so that TREAT, and its supporting facilities, could return to its roots with higher throughput capsule-based experiments while leveraging modern enhancements including advanced in-situ instrumentation. This approach, termed the Minimal Activation Retrievable Capsule Holder (MARCH) irradiation vehicle system, makes use of a modular mechanical arrangement, structural materials which minimize radioactive products, and strategic categorization of reusable safety function components all to create a cost/schedule effective method for experiment design, irradiation, and logistics. A MARCH module

Figure 2. First SETH capsule installed in the irradiation vehicle suspended over a storage hole at TREAT.

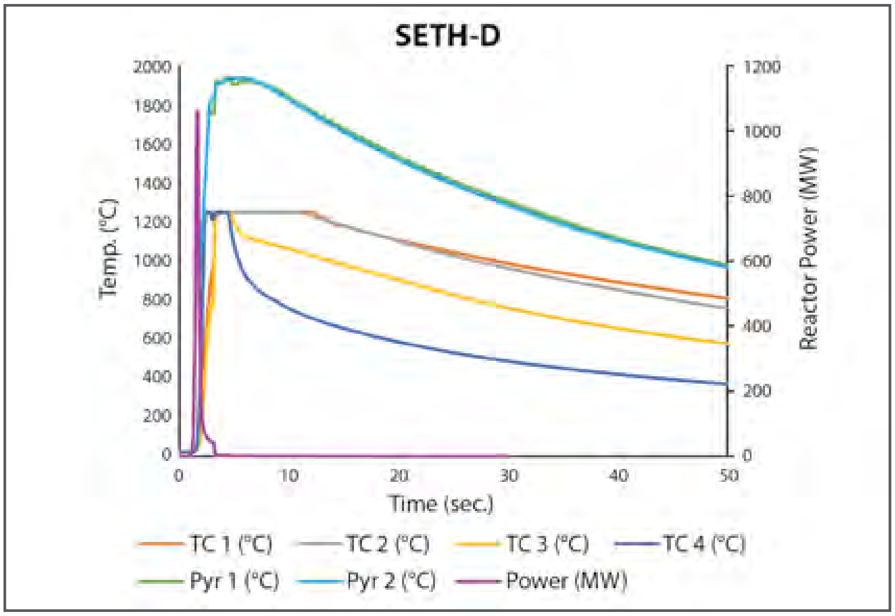


Figure 3. Reactor transient and specimen temperature data from SETH-D.

design was developed, entitled the separate effects test holder (SETH), to enable transient fuel irradiations in inert gas capsules. A series of SETH irradiations were completed in 2019, both to commission TREAT's modern experiment capabilities and to obtain nuclear heating data from LWR specimens in a modern core configuration. These outcomes were essential developments in enabling experiments on ATF technologies and water environment capsule (currently underway). This inaugural irradiation campaign paved the way for TREAT to reclaim its role in fuel safety research to help develop and license advanced nuclear fuels.

Accomplishments:

INL's nuclear science and technology directorate largely oversaw the design, predictive analysis, and fabrication of irradiation test hardware for the MARCH system and SETH capsules. The fuel fabrication division fabricated a series of small LWR rodlets each using 4.9% enriched UO_2 pellets in zirconium alloy cladding and oversaw development of capsule assembly methods. TREAT experiment engineering personnel developed and installed experiment supporting system including modernized experiment gas control system and experiment data acquisition systems. This coordinated effort gave way to the first fueled experiment capsule (SETH-A) irradiation. Thermocouples were attached to the SETH-A rodlet cladding during this low energy transient whose purposes was to obtain nuclear heating data (expressed as a ratio of specimen-to-core energy deposition, termed energy coupling factor, ECF) by calorimetry in-situ. The SETH-A rodlet was later extracted from the capsule and characterized by gamma spectroscopy as another method of correlating ECF. The second capsule (SETH-B) was irradiated multiple times to investigate the effect of increasing transient energy on ECF, to work out data acquisition parameters for first-of-a-kind use of multispectral pyrometry, and to refine parameters for operating the refurbish fuel motion monitoring system (FMMS, aka fast neutron hodoscope). SETH-C was irradiated a single time at higher energy deposition to help explore cladding temperature instrumentation response time and range. SETH-C also achieved a world first with use of additive manufacturing to produce the irradiation capsule. The use of additive manufacturing enabled reduced cost in manufacturing the mostly-hallow subsequent capsules while paving the

Figure 4. Post-transient neutron radiography of SETH-C, -D, and -E capsules using TREAT's neutron radiography capability.

way for greater instrument density in more complex geometries in future designs.

The ECFs measured by calorimetry and gamma spectroscopy in the SETH-A through -C irradiations were found to be in good agreement with each other ranging from 2.0 to 2.2. J/ gUO_2MJ. The SETH -C rodlet, which achieved a peak cladding temperature of 1378 °C, according to pyrometer measurements, was characterized in TREAT's neutron radiography stand and later extracted from its capsule in an unshielded glovebox (confirming the viability of the MARCH low activation approach to postirradiation exam for fresh specimens) where both activities confirmed that the rodlet was intact and undisturbed.

The final irradiations in the SETH series (SETH-D and -E) were performed at even higher energy to cause cladding melt. The SETH-D transient was designed to achieve partial cladding melt by using ECFs measured in previous tests to predict performance using the BISON code. This target condition was achieved, giving observational confirmation of the previously measured ECFs, as indicated by post-transient neutron radiography where only the top pellet was held by surviving cladding. SETH-E caused gross cladding melting and the transient was shaped to favor observation of fuel downward motion by the FMMS. Both SETH-D and -E far exceeded the survival and calibration range for cladding thermocouples while the pyrometer measurements successfully measured temperatures beyond the melting point of the zirconium alloy cladding, demonstrating the value of this advanced measurement method in transient testing.

In-pile Pyrometry for Transient Testing Applications

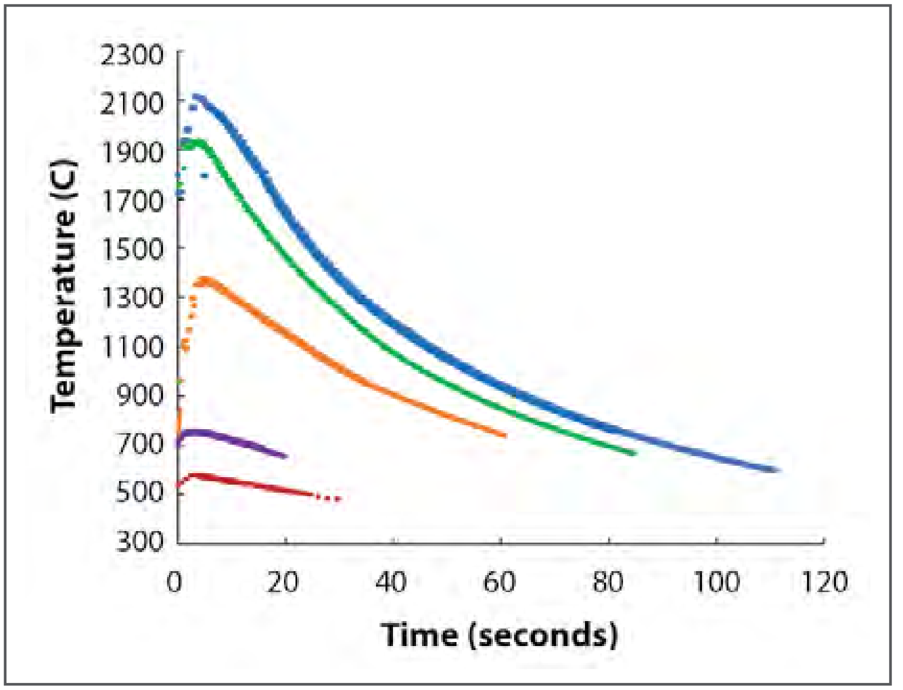
Principal Investigator: Austin Fleming, INL
Team Members/ Collaborators: Colby Jensen, INL

Providing accurate in-pile cladding surface temperature measurements is a technically challenging problem from a metrology standpoint. The conventional technique is to use thermocouples which are welded to the cladding surface. Using thermocouples for transient in-pile applications can result in several hundred degrees of error due to the fin effect, the heat capacity, and the thermal resistance between the cladding and thermocouple. Fiber-optic based in-pile pyrometry can provide high-speed, non-contact, temperature measurements. Pyrometry provides high temperature measurement capability, well beyond other convection techniques, and the non-contact nature of the measurement eliminates the heat transfer challenges that are associated with thermocouples.

Project Description:
The objective of this research is to develop a fiber-optic based, high-speed pyrometer for in-pile temperature measurement applications. One of the main applications of the sensor is for cladding surface temperature measurements during accident conditions. The data objectives for Reactivity Initiated Accidents (RIA) and Loss of Coolant Accidents (LOCA) have similar but slightly different challenges for each case. Both require high-temperature measurements in extremely harsh environments. For LOCA experiments the non-contact nature has the ability to enable balloon temperature measurement under prototypic conditions (not possible with thermocouples). RIA experiments require a high speed measurement capability to resolve the temperature history and peak cladding temperatures reached during the event. Pyrometry has the potential to satisfy all of these requirements which would provide valuable data to researchers for fuel/cladding development, testing, and qualification. While pyrometry systems are commercially available, there are unique requirements and challenges associated with in-pile applications. Some of these include: radiation-induced emission, radiation-induced attenuation, the harsh environment optical components need to survive, viewing through participating media (water & steam), oxidizing surfaces, high-pressure/high-temperature feed throughs, and unique alignment challenges. Many of these challenges are unique to the nuclear industry, but the research can leverage the experience from industries such as oil & gas to help solve some of the common challenges.

Figure 1. Photograph of the ATF-SETH capsule with two pyrometers located on the left side of the rodlet.

Figure 2. Plot of the measured temperatures from the pyrometers in five of the ATF-SETH experiment transients in TREAT.



Accomplishments:

Steady progress has been made toward the development of a high-speed, in-pile pyrometer for high-temperature measurements. This research has been conducted with a graded approach which includes laboratory-based testing, preliminary in-pile tests, and deployment in fueled experiments. Laboratory-based testing has used design of experiment approaches to evaluate pyrometry techniques to determine which are the most appropriate for in-pile applications. This testing included parameters such as misalignment, target shape, oxidation, surface roughness, and time response. An active single wavelength pyrometry technique was evaluated alongside a multi-spectral based system. Through these tests, the multi-spectral demonstrated less sensitivity to misalignment, surface roughness, and target shape. This leads to more accurate measurements under prototypic conditions. This testing was followed by preliminary sensor testing in the coolant channels of Transient Reactor Test Facility (TREAT). This testing was conducted by inserting an optical fiber into a

A high-speed, fiber-based pyrometer has been developed, tested, and deployed in the Transient Reactor Test (TREAT) facility for use in fueled experiments to provide accurate high temperature measurements.

titanium tube which was inserted into the air coolant channels between fuel elements in TREAT. From this preliminary testing, the impact of radiation induced emission and attenuation on pyrometer performance has been evaluated. Through this opportunity techniques have been developed to account for the radiation-induced emission in the measured spectrum to minimize its impact on low-temperature, high-flux conditions. For most high-temperature applications the radiation-induced emission is negligible compared to the thermal emission, which has been successfully demonstrated. Following this preliminary in-pile testing, the fiber optic-based pyrometer was deployed in the fueled TREAT experiment series referred to as Accident Tolerant Fuel - Separate Effects Test Holder (ATF-SETH). This experiment series used two pyrometers per capsule viewing the cladding surface of a fuel rodlet. These pyrometers successfully measured temperatures ranging from ~400 C to ~2100 C throughout this experiment series. These measured temperatures were used for calorimetry to obtain the power coupling factor between the TREAT driver fuel and the fuel rodlet in the experiment. The power coupling factor measured with this technique was confirmed with alternative methods. This test series represented the first successful pyrometry measurements for an in-pile fueled experiment.

Neutron Tomography of Irradiated Fuel Pins

Principal Investigator: Aaron Craft, INL
Team Members/ Collaborators: William Chuirazzi, Josh Kane, Nik Cordes, Nick Boulton and the NRAD Team (INL)

This work represents the world's first digital neutron tomography of irradiated nuclear fuel.

Neutron radiography is typically performed to assess the two-dimensional (2D) internal geometric condition of irradiated nuclear fuels. This project performed three-dimensional (3D) neutron computed tomography (nCT) using a recently developed digital camera-based neutron imaging system located at Idaho National Laboratory (INL's) Neutron Radiography Reactor (NRAD) North Radiography Station. Capsules containing accident tolerant fuel (ATF) rodlets underwent transient testing and were then brought to NRAD for nCT. A set of digital neutron radiographs were acquired of two different capsules, Separate Effects Test Holder (SETH)-D and SETH-E, which were then mathematically reconstructed into 3D tomograms that reveal internal features of the capsules that were not visible in 2D radiographs. The results of this work represent the world's first digital neutron tomography of irradiated nuclear fuels.

Project Description:

Transient testing of advanced fuels campaign (AFC) fuels can result in significant disruption of the fuel sample inside an experiment capsule. The internal geometric condition can provide knowledge about the irradiation conditions experienced and failure mechanisms, but must be assessed non-destructively to maintain the internal integrity of the disrupted sample. Irradiated samples are traditionally visualized using 2D neutron tomography with film-based methods. However, a digital camera-based neutron radiography system was recently developed in the North Radiography Station at NRAD. Neutron computed tomography is accomplished by taking a series of 2D radiographs of a sample for multiple projection angles, then mathematically reconstructing these projections into a 3D tomogram. This project performed nCT for two post-transient ATF rodlet samples inside SETH capsules. The resulting 3D rendering provides literally an additional dimension of information to assess the condition of a sample and reveals features that were not previously accessible using 2D radiography. Additionally, the 3D tomographic reconstruction lends itself more readily to extracting quantitative information about samples including density variations, dimensional inspection of features, crack and other defect detection, and quantification of relocated and melted material. This new capability to perform digital neutron tomography of nuclear fuel represents a significant advancement from current capabilities at INL that clearly demonstrates INL's leadership in nuclear fuels testing and evaluation and promotes the technical superiority of nuclear energy research and development (R&D) in the United States.

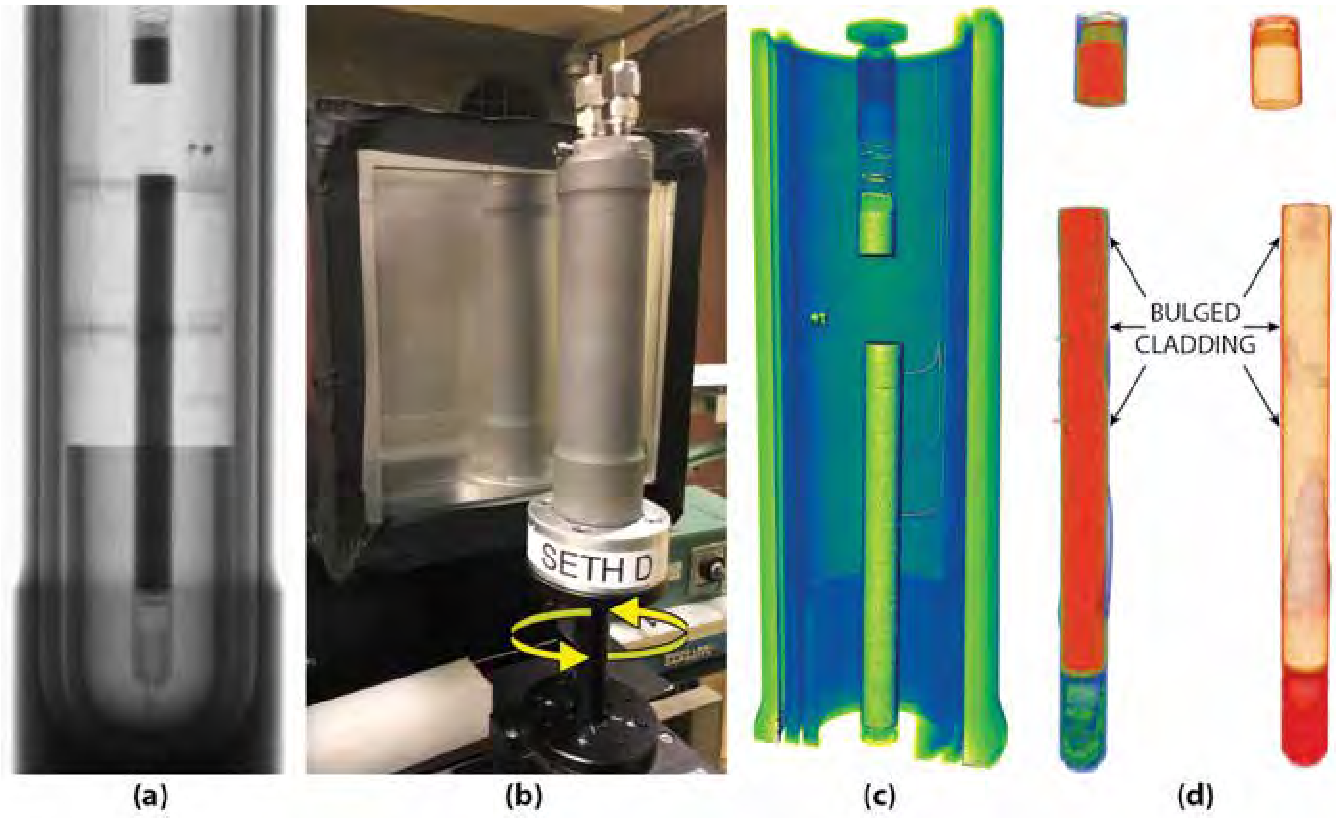


Figure 1. a) Example of neutron radiograph of SETH-D with stitched top and bottom fields of view. b) SETH-D capsule in front of the digital neutron imaging system in NRAD's North Radiography Station (NRS). c) A cut-away of the 3D rendering showing the internal configuration of the SETH-D capsule. d) 3D renderings of the rodlet showing regions of bulged cladding.

Accomplishments:
A digital camera-based neutron imaging system was developed and installed in NRAD's North Radiography Station to test novel scintillator screens for neutron imaging. An adapter to mount SETH capsules was fabricated and mounted to a sample positioning stage to accommodate AFC programmatic samples. An instrument control system was developed in collaboration with colleagues at Technical University Munich using their NICOS software that was specially formatted for a stand-alone setup apart from their instrument network. Altogether, the imaging system, sample stage, and control system makes up a neutron tomography system capable of acquiring multiple neutron radiographs at multiple projection angles using automated instrument control software.

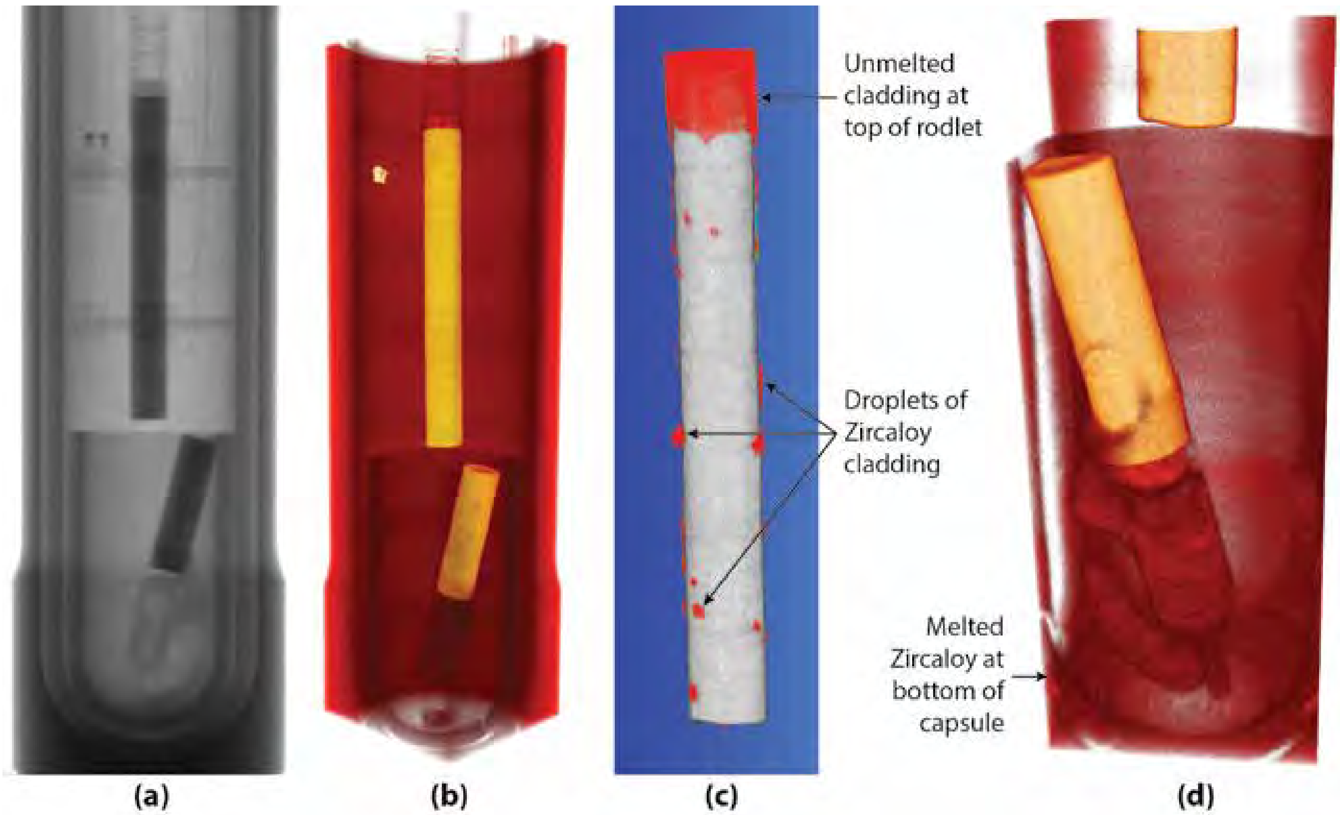


Figure 2. a) Example of neutron photograph of SETH-D with stitched top and bottom fields of view. b) A 3D rendering of SETH-E. c) Segmented view showing zircaloy droplets down the sides of the fueled region in red. d) A 3D rendering showing the lower portion of a rodlet and melted cladding at the bottom of the SETH-E capsule.

The first sample tested was the SETH-D capsule containing an ATF rodlet that previously underwent transient testing. The first nCT dataset includes 360 radiographs in 1° increments, which took 6 hours to acquire. A longer nCT scan acquired 1081 radiographs in 18 hours. Compared to the current film-based method that takes a full day to produce a single image and is limited to 14 radiographs per day, this digital imaging system produced 1081 radiographs in just 18 hours, representing an improvement in production capacity and rate of image production by a factor >100. The resulting tomogram revealed the broken rodlet and ballooning of the cladding.

The second sample tested was the SETH-E capsule containing an ATF rodlet that experienced a higher-power transient than SETH-D. The nCT scan included 1081 projections and the resulting 3D tomographic reconstruction revealed significant fuel disruption and other features that were not visible in the 2D radiographs. The Zircaloy cladding melted around the fueled region and collected in the bottom of the capsule. Droplets of molten cladding solidified along the fuel column resembling melted wax running down the side of a candle. The 3D tomogram enables potential future analysis that could quantify the volume of cladding relocated into different areas inside the capsule.

These initial results represent the world's first digital neutron tomography of irradiated unclear fuel.

The current digital neutron imaging system was a prototype system for initial capability development. The lessons learned from this system will inform the design of a next-generation neutron imaging system that will provide higher resolution and contrast, which will ultimately lead to improved feature detectability and enable extraction of quantitative information that will inform the fuel qualification process. The researchers involved in this project are working with many colleagues at universities, industry, national and international laboratories to pursue multiple activities towards establishing nCT capabilities for postirradiation examination (PIE) and defining the state of the art for visualizing nuclear materials with neutron beams. Early successes such as those reported here demonstrate the feasibility and applicability of nCT for PIE and garner the support required to accelerate further development of these capabilities.

Investigation of I-Loops for Ramp Testing

Principal Investigator: N. Woolstenhulme, INL
Team Members/ Collaborators: N. Oldham, T. Maddock, K. Horman, M. Ramirez and H. Hiruta (INL)

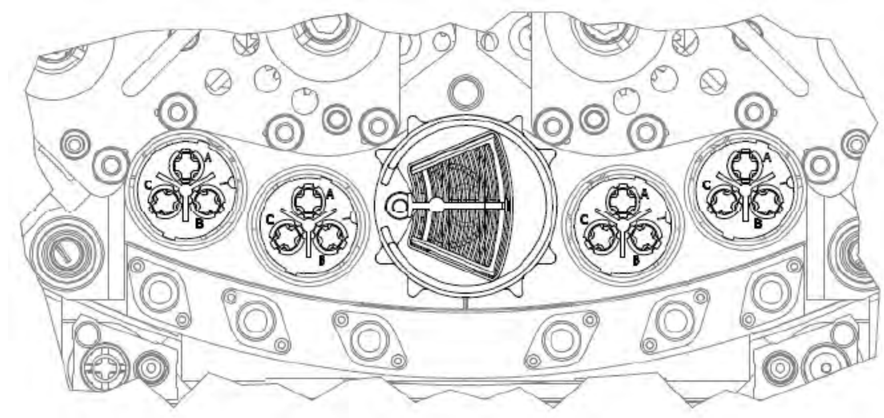


Figure 1. Booster Fuel in Large-I Position.

Preliminary design and safety evaluations have been performed and show that the I-Loop design effort in ATR is a viable strategy to address LWR fuel irradiation testing capability gaps left by the HBWR closure as well as enhance testing capabilities including ramp testing.

The Advanced Test Reactor's (ATR) existing Light Water Reactor (LWR) loop in the center flux trap provides ample flux for both prototypic and accelerated burnup accumulation in approximately 30 test rodlets, a handful of which can be comprehensively instrumented. This loop alone cannot satisfy the capabilities and demand required for LWR fuel testing. The Halden Boiling Water Reactor's (HBWR) was a main provider of irradiation testing of LWR fuel. However, the closure and decommissioning announcement of the Halden Reactor has left a major void.

Additional LWR loops (I-Loops), which are not subject to some of the availability and size constraints present in flux trap-based loops, will be needed in order to address Halden capability gaps using ATR. Anticipated testing will be performed in two separate loops conditions, Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR), running base irradiations and ramp testing with instrumented specimens.

Project Description:

The HBWR in Norway has been a primary irradiation test facility for LWR nuclear fuels and materials behavior to assess fuel performance for nuclear regulators. Due to technical issues, financial risk, and political challenges, Halden's parent organization Institute for Energy Technology (IFE) voted to permanently close the reactor. This is a major loss to the LWR industry because there are few test facilities available for advancing reactor technologies.

ATR has many irradiation positions including nine flux trap positions with in-pile tubes with customizable coolant conditions. All of these positions are highly scheduled for many years by other testing programs with one exception, the center loop position. The center loop is primarily designated for LWR fuels burnup accumulation. This loop alone cannot support the demand required for LWR because of an overall limitation in specimen capacity, inability to perform flux tailoring, and complication with coolant-voiding conditions (e.g., BWR, dry out).

There are several other positions more available within ATR's neck shim housing and inner reflector. While useful for other capsule-based and instrumented-lead type experiments, these positions don't possess desirable useable diameters to implement LWR loops. Only the Large and Medium I-positions, which reside outside of the reactivity control cylinders in the outer beryllium reflector, have adequate diameters for LWR loop installation because of high availability and a neutron flux similar to the HBWR.

Accomplishments:

Enabling an I-Loop for ramp testing as well as filling the Halden void requires a boosted reflector flux, a custom in-pile tube, and reactor vessel modifications. Designs in these three areas are rapidly progressing with no major obstacles.

Figure 2. Booster element in holder.

A standard ATR driver fuel element placed in ATR's Large-I irradiation position (Figure 1) boosts neutron flux outside the cores control cylinders. The standard element was chosen over a fuel ring because the fuel element design is already in-place and has an established supply chain. This "booster fuel" element requires a holder (Figure 2) to position the element within the reactor hardware

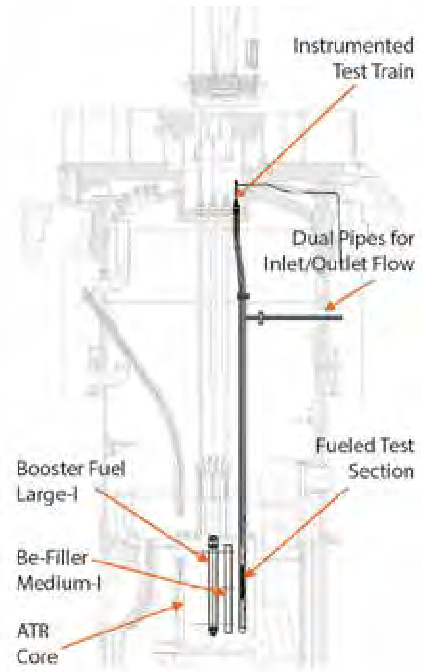


Figure 3. I-Loop reactor pressure vessel layout.

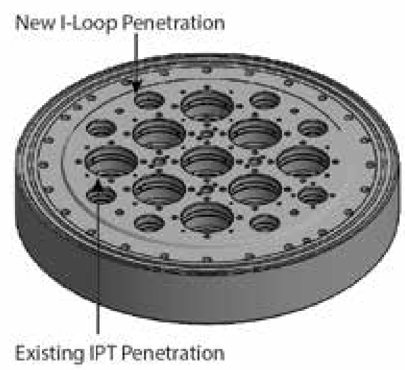


Figure 4. Modified ATR top head closure plate.

in the Large-I position. The holder is composed of aluminum structural materials with the inner filler of beryllium for neutron reflection and moderation.

Monte Carlo investigations have been performed and have demonstrated that inclusion of a standard ATR driver fuel assembly in a Large-I position can increase neutron population in the outer reflector for a 10% thermal neutron flux increase in the fuel specimen. This novel, yet straightforwardly achievable, approach to ATR core management can elevate thermal neutron flux to the same level available in the HBWR at $5 \cdot 13$ n/cm^{2}s [3] during routine ATR cycles. Average heating rate comparison of a fuel rod in HBWR and a PWR vs. in the ATR I-Loop position is shown in Table 1. Early results show that the I-Loop is capable of matching and even exceeding a HWBR power rate.

An annular loop layout using nuclear grade Zr-2.5Nb alloy (which has been used extensively in high fluence pressurized water conditions in CANDU reactors) works to accomplish this aim in the I-Loop design. Thermal neutron flux in Medium-I positions are slightly higher than the Large-I positions at $4 \cdot 10^{13}$ n/cm^{2}s during typical ~50 day ATR cycle powers (five lobes each at ~25 MW lobe power, ~125MW total core power); making medium-I positions the preferred location for I-Loops. Each I-Loop enables a 2×2 rodlet array (giving up to 16 total 30 cm rodlets across ATR 1.2 m active core length per loop). The I-Loop design is also compatible with other test train configurations such as a cross section with two or three individual rodlets in discrete flow tubes for varying thermal hydraulic conditions within a single test assembly, or a single rodlet cross section to reduce rod-to-rod self-shielding for increased nuclear heating with additional volume for

Table 1. Reactor heating rate comparison.

Reactor	PWR	HBRW	ATR I-Loop
kW/m	~20	~15	~18
kW/ft	~6	~4.5	~5.5

instrumentation.

An I-Loop will be installed so that test train extraction and instrument leads route through the top closure plug while permanent coolant plumbing will penetrate through the side of the reactor pressure vessel in existing L-flanges in a manner typical for many successful lead-out type experiments performed at ATR. The loop will be designed so that it can a variety of tests including PWR/BWR base irradiations, ramp testing, and creep testing. Ramp testing is performed by flux manipulation via a helium-3 screen or movement within the reactor's flux gradient. The slight offset (~20cm offset over ~6m length) of these in-pile tubes will require that test trains are designed with some compliance to facilitate insertion and extraction. See Figure 3 for ATR in-vessel facility layout. Test train extraction in this fashion also permits the transport of irradiation tests to ATR's sizeable spent fuel storage pool, the adjacent dry transfer cubicle hot cell, or to a variety of off-site hot-cell for several options in between-cycle and postirradiation examinations.

ATR's native facility design is heavily based around the nine flux trap loops as manifest in existing penetrations through the pressure vessel and shielding structures for high-pressure plumbing and test train extraction via overhead casks. Facility modification to include these new I-Loops, however, is a retrofit operation with some key mechanical constraints. The reactor pressure vessel's existing top closure plate will be replaced with a new plate so that eight new peripheral penetrations exist in addition to the current nine for the flux traps (see Figure 4). This closure plate is a relatively small part of the pressure vessel head (1.2m diameter) and can be removed during the upcoming core internal change-out (CIC), planned in 2021. This would create a rare opportunity for this modification with slight impact planned operations.

Figure 5. Investigation of I-Loops for ramp testing: project team photo.

CHF Studies - Merging M&S, Out-of-pile and In-pile Experiments

Principal Investigator: Colby Jensen, INL
Team Members/ Collaborators: Nicolas Woolstenhulme and Charles Folsom (INL), Richard Christensen (University of Idaho), Nicholas Brown (University of Tennessee Knoxville)

Unique capability at the TREAT facility to study in-pile critical heat flux may provide the primary building block to allow for a change in the current fuel safety criteria and enable safe operation at higher powers, which result in a substantial increase in the performance of current and future LWRs.

Cladding-to-coolant heat transfer during transient irradiation conditions remains a critical area of uncertainty in understanding nuclear reactor safety. This uncertainty impacts analytical predictions of both accident progression and fuel performance behavior. This uncertainty manifests itself in the use of conservative models that widely bound the onset and duration of transition phenomena. Improved understanding and predictive models to describe the transitions to film boiling and its duration, as well as improved material performance through accident tolerant fuel (ATF), could meaningfully increase safety and design margins in both the current commercial fleet and advanced light water reactors (LWRs) (including small modular reactors). Experiments are currently being developed to leverage the Transient Reactor Test Facility (TREAT) reactor, with its unique accessibility for instrumentation, to investigate transient and accident scenarios involving critical heat flux (CHF) in LWR systems.

Project Description:

The availability of the TREAT facility with its unique accessibility for instrumentation is ideally suited to fundamentally advance understanding of nuclear reactor safety—in particular, transient and accident scenarios involving CHF in LWR systems. This project is developing an in-pile experiment for TREAT to answer key questions regarding CHF under prototypic irradiation with specific interest in transient conditions with close integration of modeling and simulation and in- and out-of-pile experimental facilities. The modeling and out-of-pile experimental scope of this project will extend to fully prototypic LWR thermodynamic conditions while in-pile experimental testing is focused on transient and irradiation effects. The following

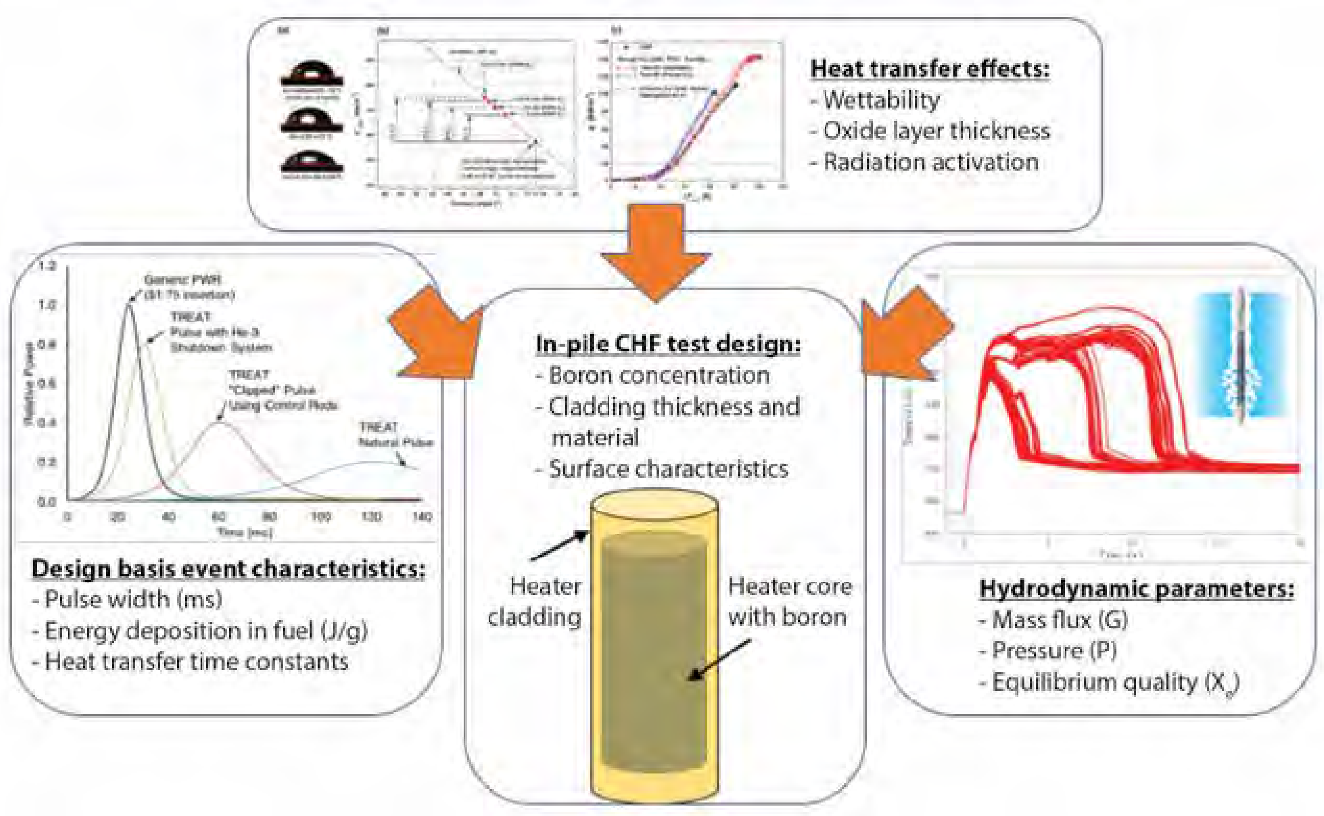


Figure 1. Schematic illustration of the relationship of the important physics inputs for boiling prediction and the in-pile boiling test design.

fundamental questions are being addressed through modeling and experiments:

1. Irradiation Effects: What is the fundamental cause of observed differences in boiling behavior of pre-irradiated cladding?
2. Transient Effects: How does rapid transient heating behavior impact CHF?

The first step in the development of the design leading to the ultimate in-pile experiments is the development of an envelope of appropriate test conditions. Figure 1 presents a schematic illustration of the relationship of the physics inputs and the in-pile boiling test design. The research scope of this project encompasses both data analytics and experimental components. Data

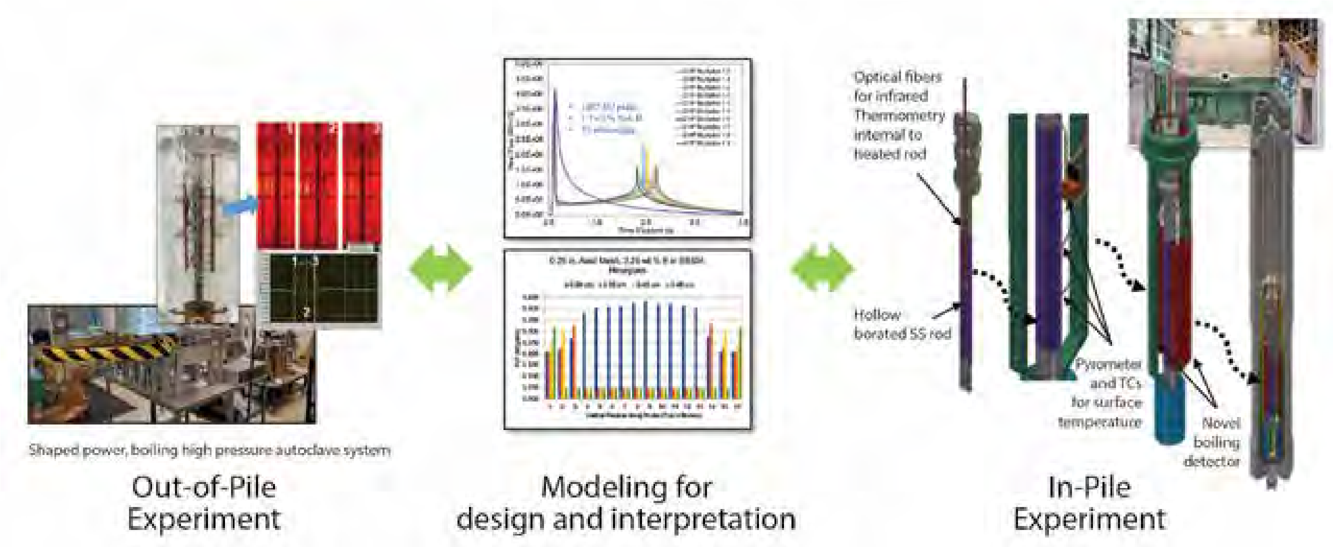


Figure 2. Relationship of out-of-pile experiments, modeling and simulation, and in-pile experiments to investigate transient in-pile boiling.

analytics consist of a combination of theoretical and modeling and simulation analyses to provide design inputs and develop the experimental test matrix. The experiment design requires analyses to support the safety basis of the experimental device in TREAT.

Key experimental capabilities developed under this project include (1) a unique-in-the-world, out-of-pile pulse power boiling system for existing high pressure water autoclave, (2) a non-fueled, nuclear heated rod design to support separate effects studies of in-pile boiling, and (3) an experiment capsule design incorporating advanced instrumentation to obtain high confidence desired results. First experiments are planned to begin before the end of 2019.

Accomplishments:

The primary technical accomplishments thus far include:

- A novel, non-fuel, nuclear-heated fuel simulator rod has been designed to perform CHF studies in TREAT. The original proposal concept has proven out through modeling work to show transient CHF is achievable in TREAT without nuclear fuel! Collaboration with the University of Tennessee-Knoxville (UTK) has been a vital part of this work to date with crucial contributions by student participants. The simulator rod design has been tuned to achieve specific experimental goals to detect onset of CHF using the planned instrumentation package resulting in an elegant but simple, borated stainless steel, tapered wall thickness heater tube design.

- A final design for the in-pile experiment has nearly been achieved awaiting final analysis. The design incorporates the fuel simulator rod with an advanced instrumentation package including surface temperature measurements via thermocouples and fiber-coupled infrared pyrometry, internal heater rod temperature measurements via fiber-coupled infrared pyrometry, and an impedance-based boiling detector for detecting voiding behavior in the capsule. Most of the instrument package is novel for this application and has already shown great promise in out-of-pile testing to obtain high value data supporting the goals of the project.
- The project has designed and is nearing construction completion of a unique capacitive-based power control system to perform transient boiling experiments in a high-pressure water autoclave under prototypic conditions (excluding flow). This effort is closely supported by the University of Idaho (UofI) through all phases of development. The design is truly unique for full pressure and temperature on prototypic LWR rod cross-sectional geometry. The system will be efficient for accomplishing the goals of this project and extending to qualification of TREAT devices and a variety of thermal-hydraulic experiments. The design has been modeled extensively and a subscale, fully functional prototype has been tested to show full function and operability. All supporting laboratory control and safety documentation has been established to support system operation.

This project has become a proving ground for water reactor fuel behavior under boiling conditions. These capabilities and the resulting data will be foundational to transient testing of ATF (design and interpretation of results) while providing further opportunity to take advantage of enhanced material performance. The project has already led to follow on studies for cladding performance studies during anticipated operational occurrences in LWRs and potential studies of other advanced fuels under CHF conditions. The first CHF experiments will begin in the TREAT facility this winter.

Separate Effects Testing in TREAT for ATF Fuels

Principal Investigator: G. Beausoleil & F. Cappia, INL
Team Members/ Collaborators: L.A. Emerson, D. Murray, D. Sell, C. Christensen, A. Winston, D. Wahlquist, M. Bachhav, and B. Miller (INL)

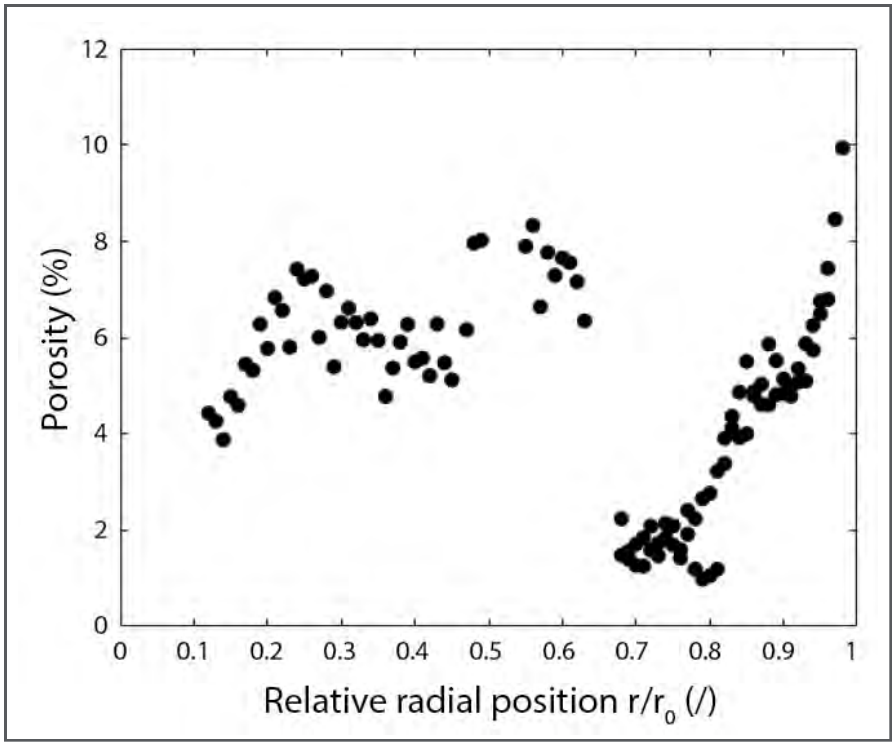


Figure 1. Measured radial porosity profile on a polished cross section. Fuel average burnup: 72 GWd/tHM.

This work was designed to investigate the causes of fuel fragmentation during accident scenarios by characterizing retained fission gas on the grain boundaries of high burnup UO_2.

This work performed examinations on high burnup UO_2 from commercial power plants with the intention of better understanding of fission gas and fragmentation behavior. Fuel fragmentation, relocation, and dispersal (FFRD) is a major factor in fuel lifetime limits imposed on UO_2 by regulators, both domestic and foreign, and is a major area of interest by industry groups (e.g., Electric Power Research Institute (EPRI)). The characterization of the fuel focused on the differences in porosity between the high-burnup (rim) zone, the dark (intermediate) zone (DZ), and center zones of the fuel as well as the fission gas content of each region. Work involved thermal treatments to analyze fission gas and high resolution microscopy techniques to investigate pore structure.

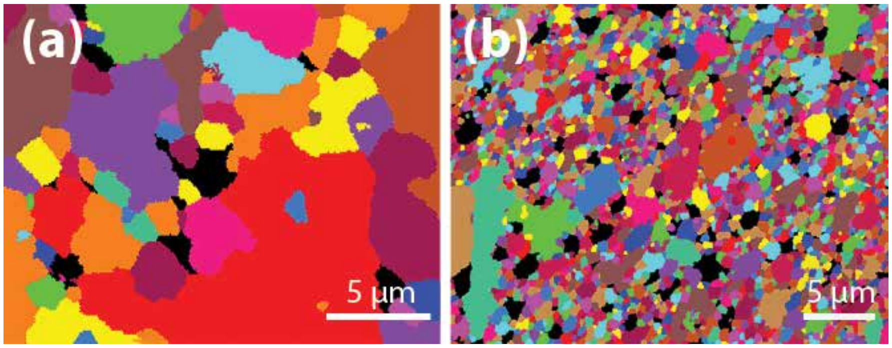


Figure 2. Selected EBSD maps at different radial locations: a) pellet intermediate radial position b) pellet rim.

Project Description:

It is known that fuel pellet fragmentation appears to be largely driven by a rapid expansion of fission gas bubbles along grain boundaries, but quantitative assessment of the local gas inventory and related microstructural factors playing a major role in fuel fragmentation is still incomplete. The project focuses on the characterization of the initial local fission gas inventory and grain structure at different radial positions in high-burnup fuel. This information is crucial to better interpret results from future integral Loss of Coolant Accident (LOCA) tests planned on the same mother rod.

Fission gas measurement experiments will be used to assess the retained fission gas at different regions within the fuel. As UO_2 is irradiated within a reactor, the material forms three concentric zones with different appearances developed along the radius. These zones are referred to as the high-burnup structure (HBS) or the rim zone, the central zone of the fuel, and the intermediate zone between the rim and center, generally referred to as the 'dark zone' which can contain several sub- regions with variable bubble concentration. It is postulated that a significant amount of gas is retained in the dark zone and at intermediate radial positions, making these areas prone to fine fragmentation. The fission gas measurement experiment will utilize focused ion beam techniques to remove samples in radial increments so that the fission gas within each zone can be compared.

The fuel will also be characterized using scanning electron microscopy (SEM) techniques including imaging and electron backscatter diffraction

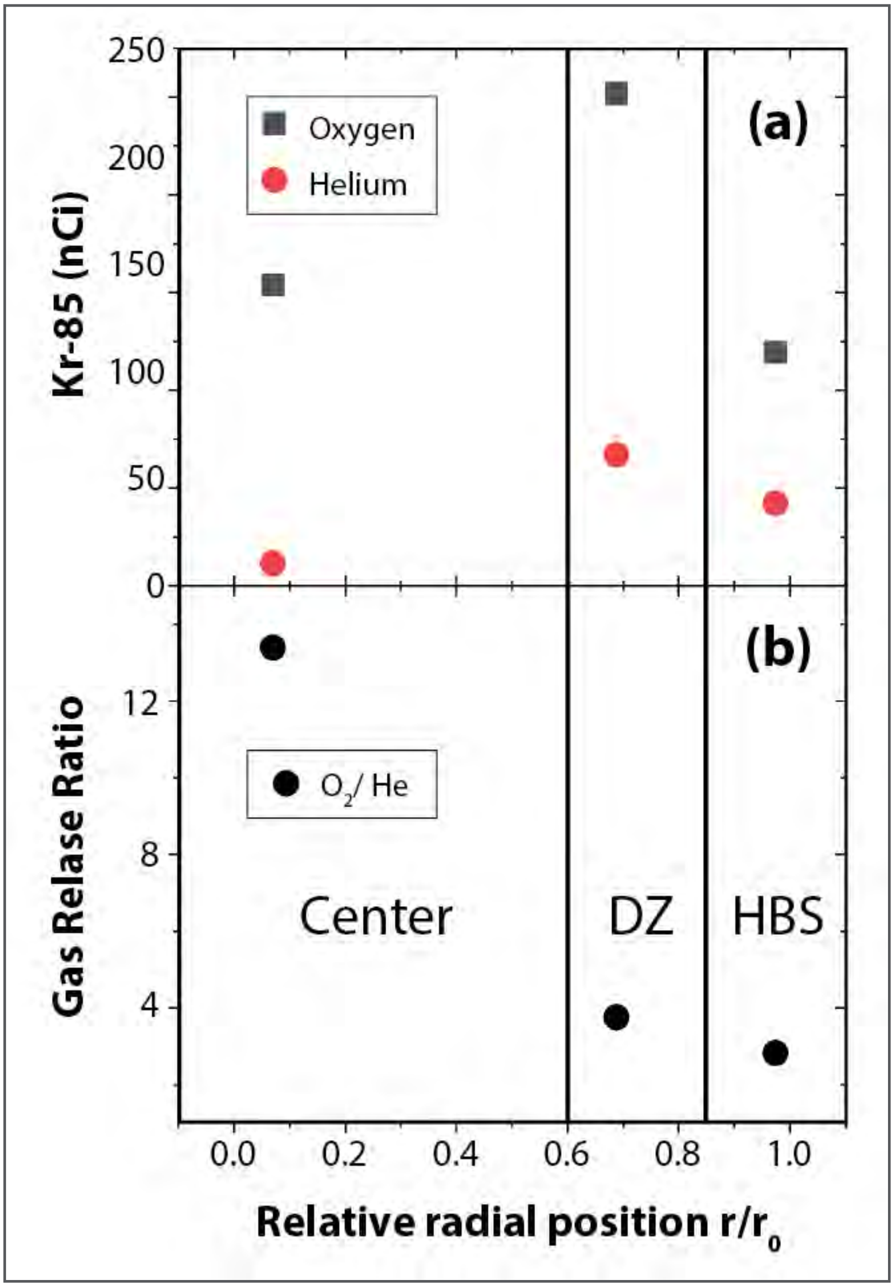


Figure 3. Kr-85 measured from oxidation study: a) shows the comparison between capsules heat treated with O_2-He mixture and pure He. b) shows the relative difference in the release quantity. The lines indicate the relative separation between the HBS, DZ, and center regions.

(EBSD) patterns as well as electron probe microanalysis (EPMA). The goal of this characterization will be to identify the fuel microstructure and porosity of each region. It is important to understand the statistics of the porosity, such as pore volume fraction and number density, so that the fission gas analysis can be meaningful. Understanding both enables researchers to analyze the pressures experienced by the fuel microstructure during an accident scenario.

Accomplishments:

In order to provide a thorough description of the as-irradiated microstructure, a combination of experimental techniques has been used to gather data from a polished cross section of LWR UO_2 fuel irradiated at 72 GWd/tHM average burnup. Detailed image analysis from SEM images acquired with the Backscattered Electron Detector has been performed along the pellet radius. At the same time EBSD patterns have been collected radially and correlated to the local gas inventory.

The plasma focused ion beam (PFIB) at the Irradiated Materials Characterization Laboratory (IMCL) at Idaho National Laboratory (INL) was successfully used to mill out and remove large (~300 μm) cubes for fission gas examination. The team also successfully designed and fabricated air tight capsules capable of retaining the fuel samples during heat treatment. The cubes were placed within the capsules and subjected to 300 °C heat treatments in both oxidizing and inert atmospheres. The oxidizing atmosphere was used to preferentially attack the grain boundaries of the samples and thus force the release of the grain boundary retained fission gas without inducing any significant transport or release of intra-granular retained fission gas. This was performed at the Hot Fuel Examination Facility (HFEF) at INL. After the heat treatments, the gas was collected and analyzed using gamma spectroscopy and gas mass spectrometry. That analysis is still ongoing.

Radial composition and burnup characterization were also attempted using atom probe tomography (APT). The shielded FIB at IMCL was used to remove APT tips and APT was performed using the LEAP system at the Center for Advanced Energy Studies (CAES). Performing APT on highly irradiated fuel is exceptionally challenging as porosity within the sample can cause the tip to fail catastrophically during mounting or measurement. A few tips were successfully measured from all three regions; however, additional investigation is required in order to corroborate present results and confirm identification of fission products. The presence of many fission products causes significant overlaps in the spectra collected, and the thermal tails in the fissile materials makes signal deconvolution challenging.

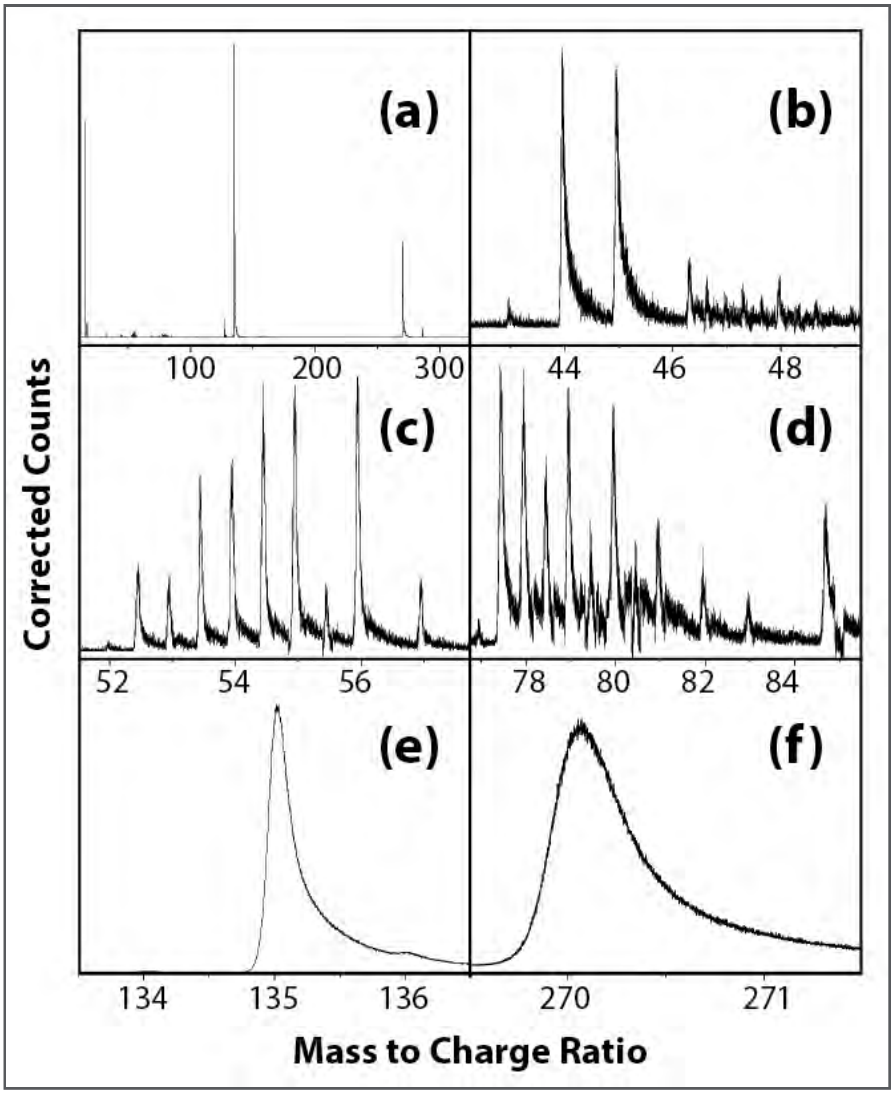


Figure 4. APT count corrected spectrum: a) shows the full mass to charge spectrum (excluding hydrogen peaks). b) through d) show fission product and U/Pu peaks. e) & f) show UO & UO_2 peaks.

Mechanical Testing on Simulated HBU Cladding to Prepare for AFC LTR Evaluation

Principal Investigator: David Kamerman, INL
Team Members/ Collaborators: W. David Swank, David Cottle, Austin Mathews, M. Nedim Cinbiz, Robert Hansen and Jason Schulthess (INL)

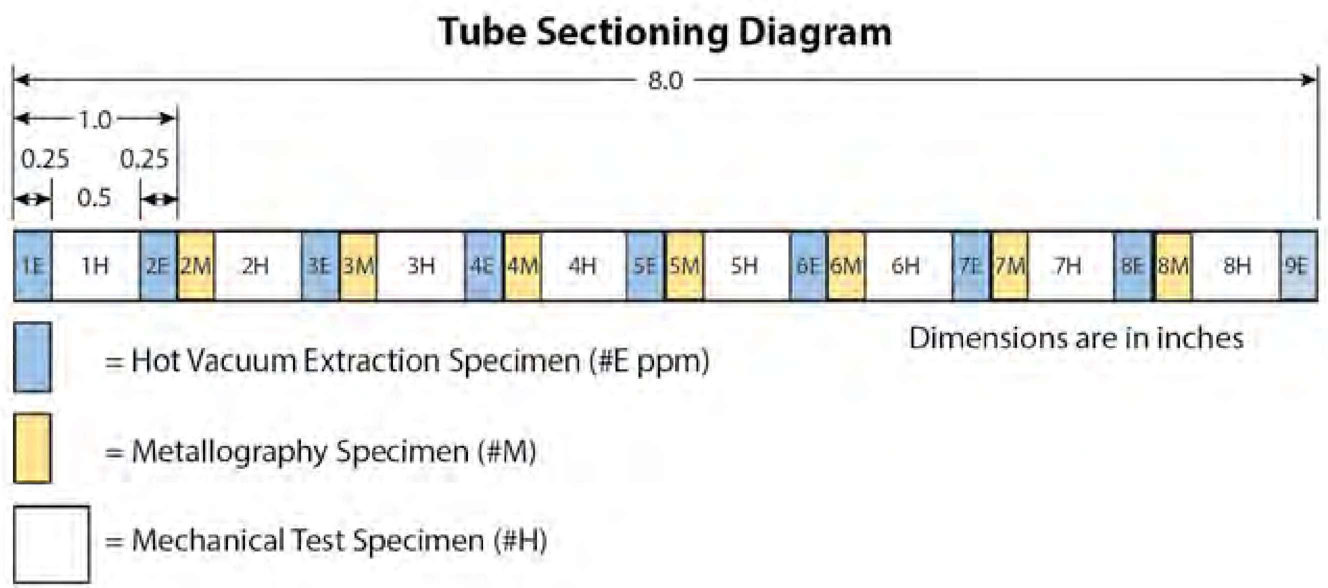


Figure 1. Cladding sectioning diagram following hydrogen charging.

The purpose of this research project is to develop simulated high burnup Zircaloy cladding that can be used to develop mechanical testing techniques for light water reactor (LWR) cladding coming from irradiations in the Advanced Test Reactor (ATR), the Transient Reactor Test Facility (TREAT) reactor, or commercial reactors as part of lead test assembly programs. Zircaloy cladding will be artificially degraded by exposure to hydrogen gas resulting in the formation of brittle zirconium hydrides which is a chief embrittlement mechanism for Zircaloy cladding. The hydrided Zircaloy will then be characterized through use of Hot Vacuum Extraction to determine the hydrogen content and optical metallography to determine the orientation and distribution of the hydrides in the Zircaloy matrix. Following material characterization, the degraded samples will be subject to tensile loads to determine the effect of the brittle hydrides on the cladding's mechanical strength and fracture toughness.

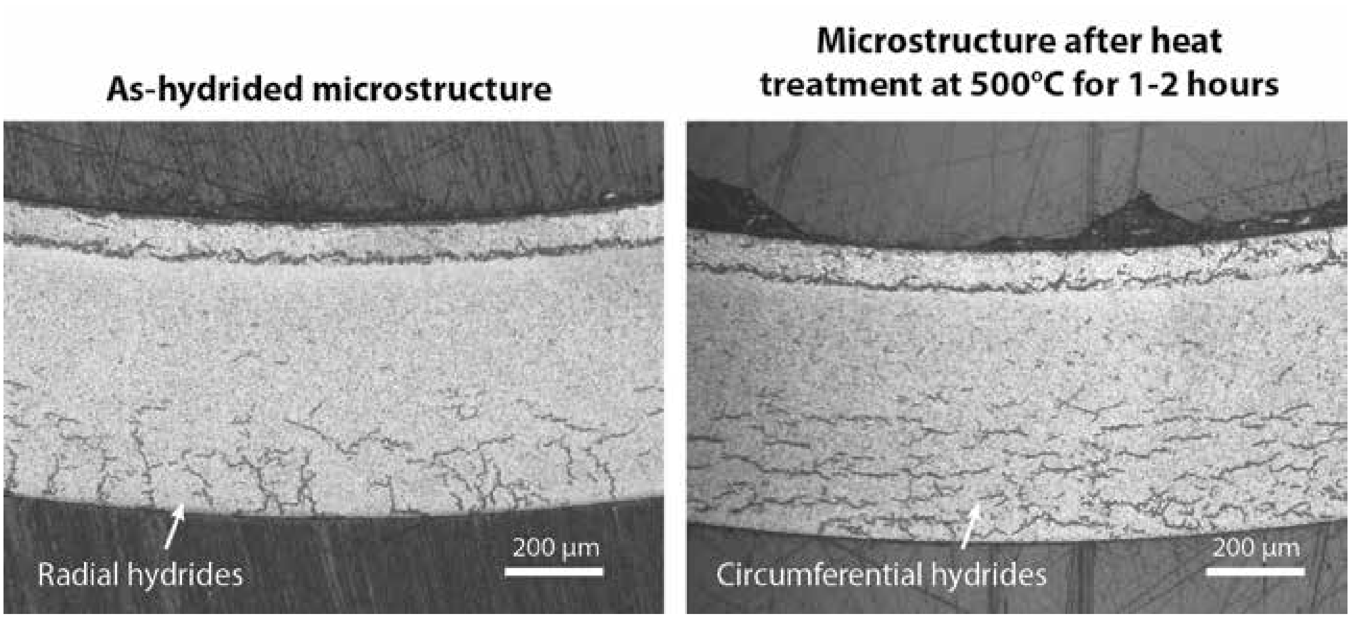


Figure 2. Optical micrographs of zirconium hydrides in recrystallized zircaloy-2 cladding.

Project Description:

The first objective of the research was to artificially age or degrade Zircaloy cladding through a hydrogen charging process. Sections of Zircaloy cladding were cut into 8-inch sections and polished with an Emory cloth to remove the native oxide from the outside of the cladding. The cladding segments were then placed into a sealed quartz tube and brought to a vacuum of 10-5 Torr. Partial pressures of pure hydrogen gas were then introduced into the quartz tube based on the volume of the tube and the desired final concentration of hydrides in the metal matrix. The samples were then heated to approximately 400°C. The partial pressure in the quartz tube was monitored while the sample was heated, and the temperature was held at 400°C until the pressure approached the 10-5 Torr value indicating that all of the hydrogen had been absorbed in the metal matrix. The sample was then removed from the furnace. The sample was sectioned according to the diagram shown in Figure 1 resulting in 9 samples which can be used for hot vacuum extraction, 7 samples which can be used for metallography mounts, and 8 samples which can be used for mechanical testing.

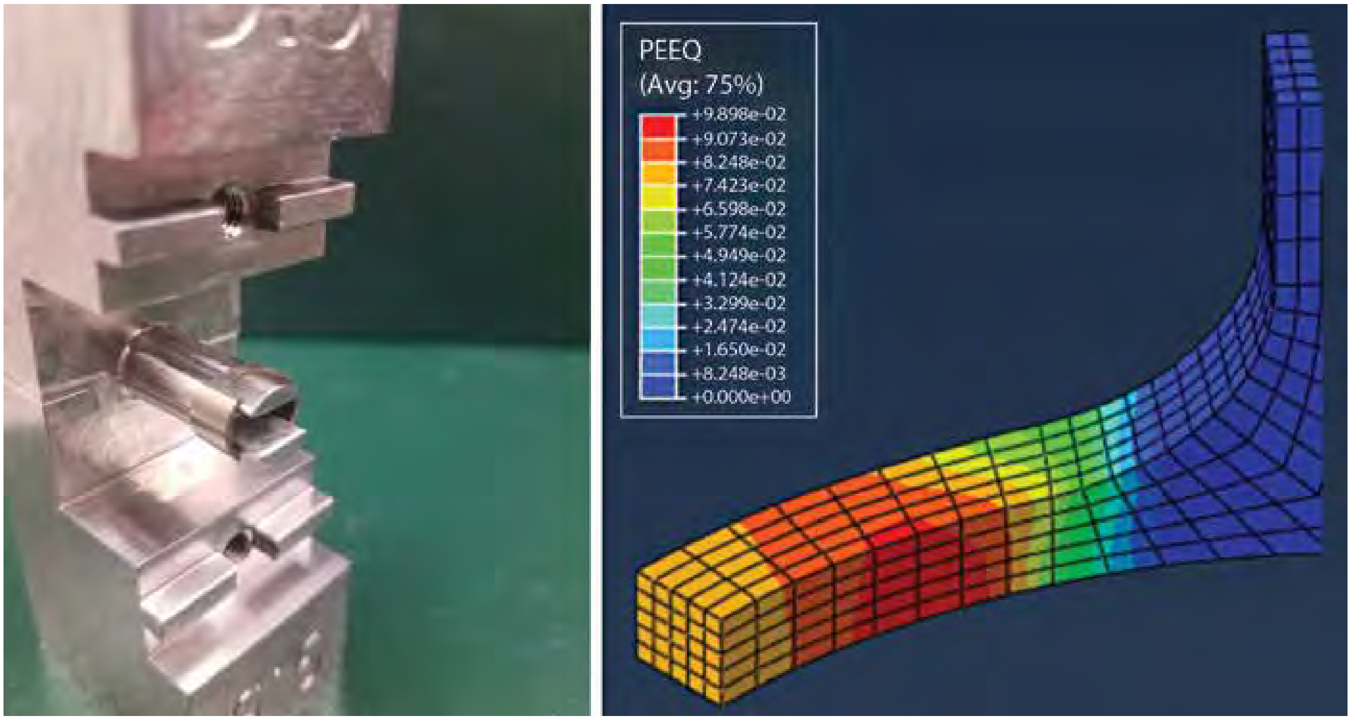


Figure 3. Ring hoop tension test, specimen and grip fixture design along with finite element predictions of strain distribution in the gauge region.

A variety of mechanical tests will be used to evaluate the strength and fracture toughness properties of the hydride cladding, among them the ring hoop tension test. The goal of the ring hoop tension test is to replicate the stress state of a uni-axial tension test with a cladding geometry. Several specimen designs, grip designs, and loading arrangements were modeled using finite element software to determine an optimum configuration. Loading grips and sample specimens were then fabricated according to the designs that performed best in the finite element simulations.

Accomplishments:

The preliminary results of the hydrogen charging were promising. Removal of the native oxide layer successfully resulted in the hydrogen being absorbed in the Zircaloy matrix. However, the distribution of the cladding's hydrogen content along the 8-inch axial length was highly non-uniform, with peak concentrations being 2-3 times that of the average. It was determined that temperature non-uniformities in the furnace were the main cause of this non-uniformity. Efforts were undertaken to flatten the temperature profile in the furnace and, centering features were added to the quartz tube to ensure that the center of the cladding always aligned with the hottest location in the furnace. This successfully reduced the variation in hydrogen contents to less than 30%. The procurement of a longer furnace is also expected to further improve the hydrogen uniformity.

Assessment of the mechanical properties of high burnup cladding is essential to establish a licensing criteria for modern and advanced nuclear fuel designs in light-water reactors.

Three different Zircaloy alloys were used in the experiments: a stress relieved annealed Zircaloy-4 cladding, a fully recrystallized Zircaloy-2 cladding, and a cold worked Zircaloy-2 cladding. For the Zircaloy-4 sample, hydrogen charging results were consistent with those seen in literature with a uniform distribution of circumferentially oriented hydrides appearing through the specimen wall thickness. The hydride microstructure of the recrystallized annealed Zircaloy-2 resulted in radial hydrides forming at the cladding outer surface. These samples were then subjected to an additional heat treatment in a box furnace at 500°C for 2 hours. Following the heat treatment, the hydrides reoriented in the circumferential direction with a somewhat higher concentration toward the cladding outer surface. The result of the hydrogen charging and subsequent heat treatment on the hydride morphology is shown in Figure 2. Results of the hydriding runs with cold-worked Zircaloy-2 are forthcoming.

Assessing the strength of cladding materials in the circumferential or hoop direction is a challenging but necessary task for cladding materials such as Zircaloy which exhibit directional anisotropy. The challenge is to maintain a uni-axial stress state in the hoop direction through the gauge section of a ring tension sample over a wide range of plastic strains. Traditional loading with hemispherical inserts and gauge region parallel to the applied load results in strong bending moments through the thickness of the cladding. One remedy for this is the use of a central "dog-bone" shaped insert to counter the bending moment. However, this can result in non-uniform plastic strains at the edges of the gauge region. Another remedy is to load the sample perpendicular to the applied load on top of the hemispherical inserts. However, this arrangement can lead to non-uniform loads in the gauge region if the friction is too high or the gauge region too long. Finite element simulations were conducted to design a test that minimized these effects. Results showed that a specimen with a gauge width of 1mm and length of 4mm when loaded in the perpendicular arrangement showed promising results when the friction coefficient between the sample and loading grip was kept below 0.05. Figure 3 shows this test arrangement next to finite element contours of the plastic strain distribution in the sample. The results of forthcoming mechanical tests will be compared to these finite element simulations to validate their results.

SATS In-Cell and Out-of-Cell Fuel Safety Testing

Principal Investigator: Yong Yan, ORNL
Team Members/ Collaborators: Ken Kane, Zach Burns and Kory D. Linton (ORNL); Ken Yueh (EPRI)

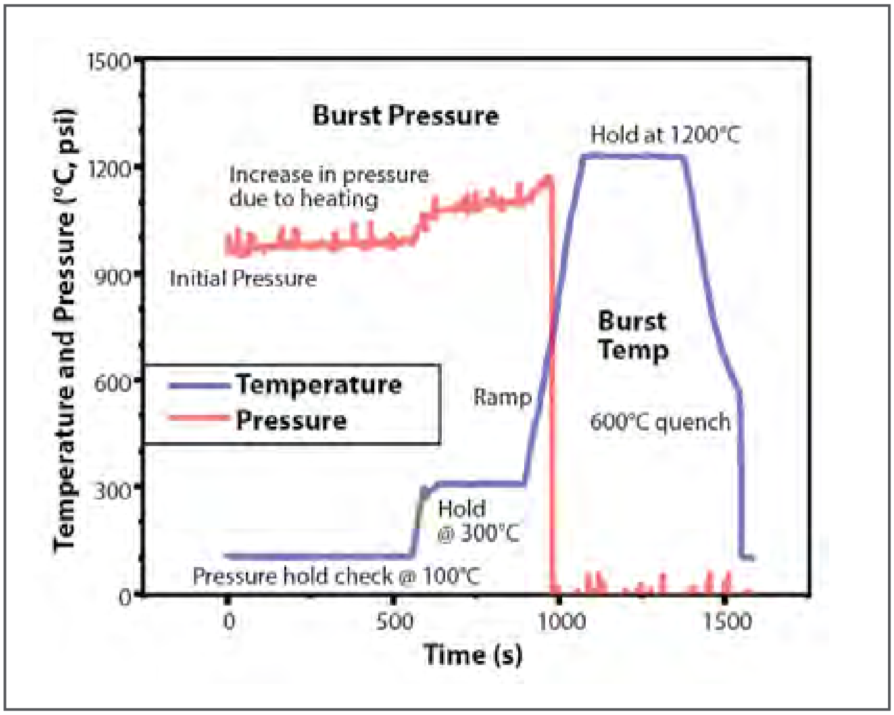


Figure 1. Typical temperature and pressure profile during a LOCA test sequence. The temperature is increased and held to ensure proper sealing of cladding tube assembly. Burst is indicated by sudden and rapid loss of pressure, and burst temperature is taken at this occurrence.

The high-temperature steam oxidation behavior of zirconium alloy cladding under design basis Loss of Coolant Accident (LOCA) scenarios has been studied across a broad set of conditions at several facilities around the world. The Severe Accident Test Station (SATS) at Oak Ridge National Laboratory (ORNL) was developed and demonstrated in order to domestically re-establish this testing capability for advanced nuclear fuel cladding concepts with enhanced accident tolerance, known as accident-tolerant fuel (ATF). SATS is an in-cell and out-of-cell capability used to reproduce conditions typical of design basis accident (DBA) and DBA scenarios including high-temperature, steam, internal pressure, and water quench.

The two key highlights from fiscal year (FY) 19 include: 1) industry supported out-of-cell, unfueled testing of General Electric (GE)-ARMOR cladding and 2) Electric Power Research Institute (EPRI) proposed in-cell burst and fuel fragmentation testing of commercial high-burnup fuel (Nuclear Science User Facilities (NSUF) award). The GE-ARMOR testing was performed in support of ATF vendor data needs and provides important results on the improved embrittlement performance of coated claddings. The EPRI NSUF in-cell testing demonstrates the readiness of ORNL rodlet re-fabrication techniques and SATS testing capabilities for upcoming vendor testing needs while providing valuable data on fuel fragmentation behavior of high burnup fuels under accident scenarios.

Both FY19 fuel safety related testing activities provide critical data supporting validation of nuclear fuel performance codes, such as BISON, and improve models designed to predict fuel system behavior during accident scenarios.

Project Description:

Out-of-cell SATS testing was performed to evaluate the impact of ARMOR coating on burst behavior of Zircaloy-4 cladding under LOCA scenario conditions. A typical out-of-cell LOCA testing procedure began with cladding test train assembly using zirconia rods to simulate fuel rods on the interior of

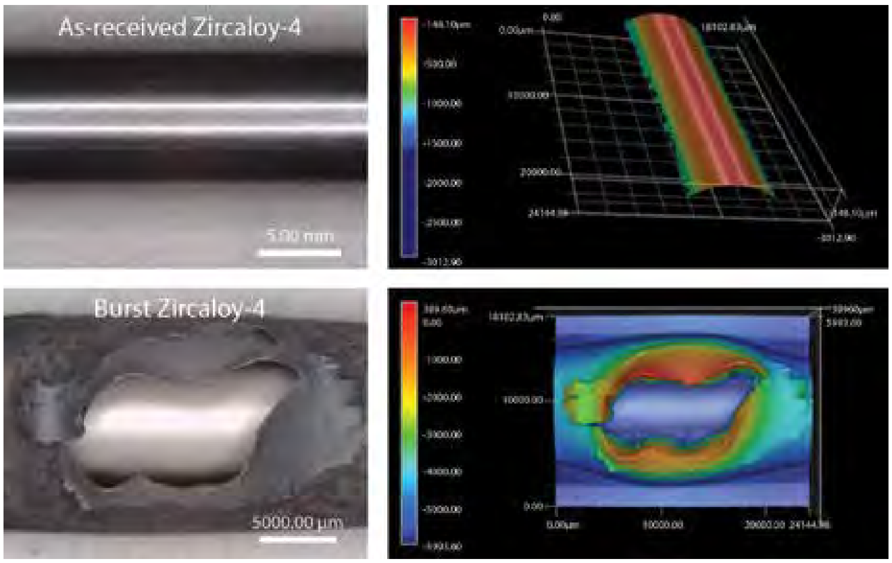


Figure 2. Images of as-received and burst Zircaloy-4 tubes with associated structured light scanning images of burst area. Burst area is measured from the re-constructed object from the structured light scan.

cladding. Four Type-S thermocouples were secured onto the cladding via Pt-Rh wire and were used to control and monitor the temperature during testing while a transducer was used to monitor internal cladding pressure.

Testing was conducted at several initial internal cladding pressures to capture and quantify burst behavior over a range of temperatures, with replicate tests being performed at all pressures for coated and uncoated tubes. The full out-of-cell LOCA burst testing sequence included internal pressurization at room temperature, 5°C/s temperature ramp to 100°C followed by a hold, and then another 5°C/s temperature ramp to 300°C, also followed by a hold. Then, the temperature was ramped at 5°C/s to 1200°C, significantly increasing cladding internal pressure resulting in cladding burst. The temperature and pressure profile for LOCA test sequence is shown in Figure 1.

In-cell SATS testing was used to generate burst and fuel fragmentation data from high burnup fuels. This data will enable the industry and regulatory agencies to best address fuel fragmentation in standard LWR fuel designs irradiated to high burnup. This approach proposes reconditioning of reactor-irradiated fuel at specific power levels and simulated LOCA testing in hot-cell furnaces and Transient Reactor Test Facility (TREAT) testing at Idaho National Laboratory (INL).

A high burnup pressurized water reactor H. B. Robinson (HBR) segment was refabricated into a LOCA test train (HBR#1). Test train fabrication included

Use of the Severe Accident Test Station both in and out of cell allows for evaluation of cladding performance under conditions relevant to design basis accident scenarios and comparison of ATF concepts to current LWR cladding materials.

Figure 3. The Severe Accident Test Station pictured during HBR#1 testing. The test station includes the integral LOCA testing module (foreground furnace) and the BDBA high-temperature furnace test apparatus (behind).

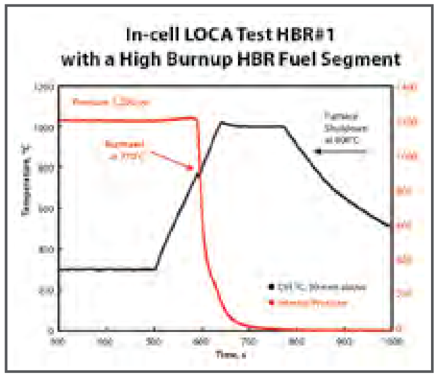


Figure 4. Temperature and pressure histories for the in-cell LOCA burst & fragmentation test HBR#1.

fuel rod sectioning, metallographic mounting and polishing, fuel leaching, outer oxide layer removal, limited fuel removal for end-plugs, and end-plug welding. After assembly of the test train, two Type-S thermocouples were strapped to the outer surface of the cladding, approximately 2 inches above the sample centerline. One of them was used to control the sample temperature. The fragmentation test was conducted in steam at 1,000°C, but without water quench. The full LOCA fragmentation test sequence included heating the fuel segment in flowing steam to 300°C and internally pressurizing the fuel segment to 1,200 psi, increasing the temperature to 1,000°C at 5°C/s, holding in steam for 120s at 1,000°C, cooling at 3°C/s to 800°C, and furnace cooling from 800°C to room temperature. After fragmentation testing, strain measurement of the burst region and sieve analysis of fuel fragmentation were performed.

Accomplishments:

The main activities in FY19 included: (1) industry supported out-of-cell testing of ATF concept GE-Global Nuclear Fuels (GNF) ARMOR cladding, (2) in-cell refabrication and SATS burst testing performed with an HB Robinson HBU cladding segment, (3) post-SATS test cladding and fuel characterization

Aside from evaluating changes in burst pressure and temperature due to ARMOR coating, out-of-cell burst behavior was quantified in a non-destructive manner via structured light scanning. The aim was to quantify effectiveness of ARMOR coating to physically impart better tolerance in LOCA scenario by reducing burst surface area. Images of as-received and burst uncoated Zircaloy-4 cladding with associated structured light scans are shown in Figure 2.

For in-cell fuel fragmentation testing, Figure 3 shows the temperature and pressure histories for the in-cell LOCA fragmentation test HBR#1. After the HBR fuel specimen was assembled to the test train, two Type-S thermocouples were strapped to the outer surface of the cladding approximately 2 inches above the sample centerline. One of these was used to control the furnace power to give a hold temperature of 1,000°C at that location. The fuel fragmentation test was conducted in steam at 1,000°C, but without water quench.

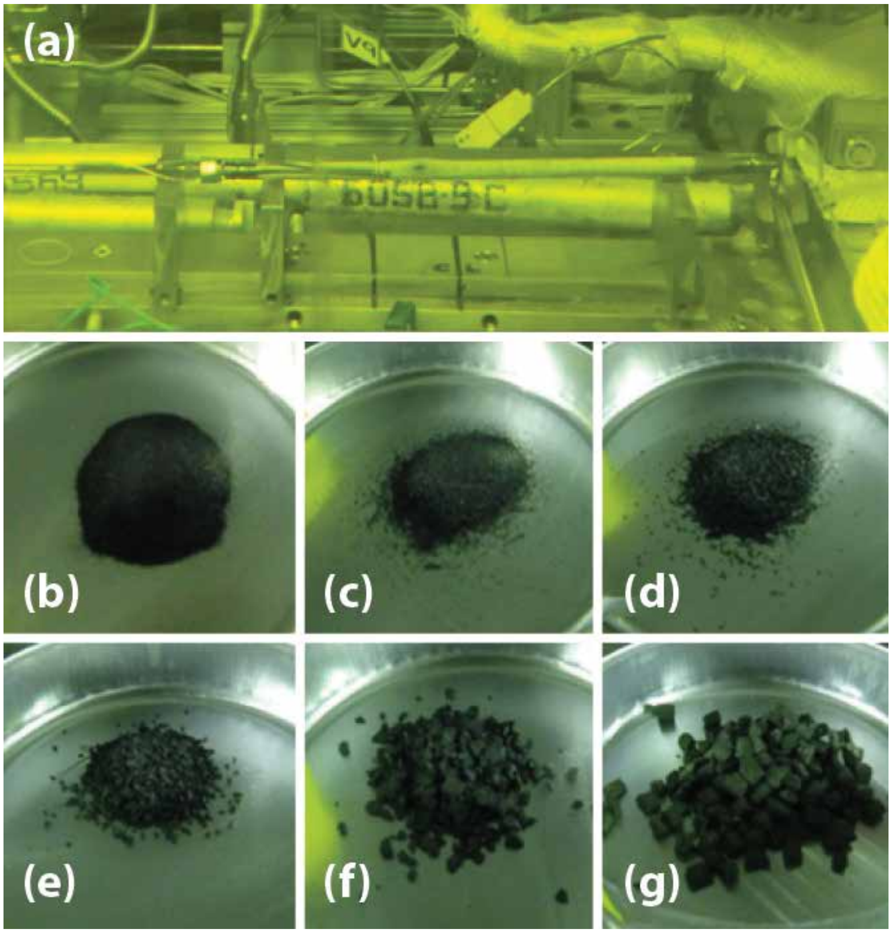


Figure 5. (a) Posttest appearance of LOCA Burst & fragmentation test sample HBR#, (b) fragmented fuel particles collected : (a) <0.125 mm; (b) 0.125–0.250 mm; (c) 0.250–0.500 mm; (d) 0.500–1.00 mm; (e) 1.00–2.00 mm; and (f) >2.00 mm.

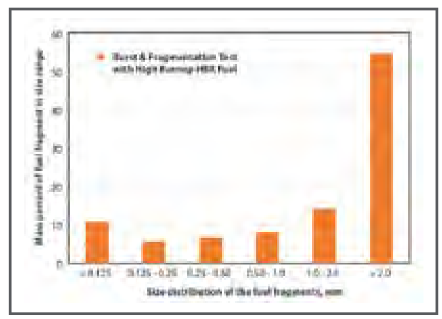


Figure 6. Size distribution of fuel fragments collected after ORNL Burst & LOCA fragmentation and shake testing.

Figure 4 shows the post-test LOCA burst & fragmentation test HBR#1 sample. In FY 18, ORNL performed a LOCA test with a North Anna high burnup specimen in a full LOCA sequence at 1200°C, and the HBR specimen was tested up to 1,000°C without water quench. Although the cladding geometry and cladding material were different between these tests, their burst temperatures were remarkably close, around 770°C. However, the HBR sample had a relatively small balloon and burst opening, as shown in Figure 3. Sieve analysis and strain measurement were performed. The total fuel collected from HBR#1 sample was 59.8 g. A few fine particles can be seen in Figure 5. The fragment size distribution of the fuel collected after LOCA fragmentation and shake testing of the HBR#1 sample was quantified, and the result is shown in Figure 6.

Length-Dependence of Severe Accident Test Station Integral Testing

Principal Investigator: B. Garrison, ORNL
Team Members/ Collaborators: M. Howell, M. Gussev and K. Linton (ORNL); M. N. Cinbiz (INL)

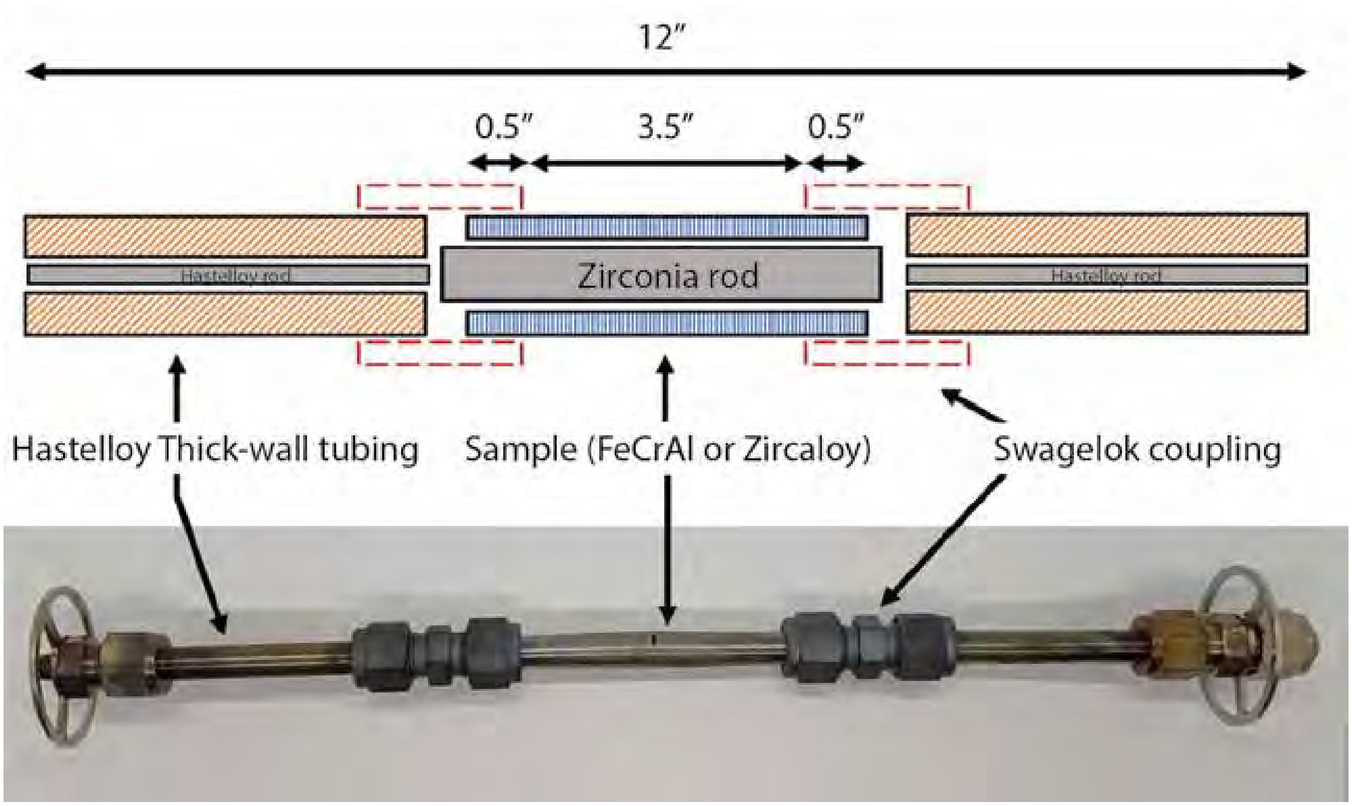


Figure 1. Setup of a 4-inch nuclear grade FeCrAl specimen post-LOCA test with thick-walled Hastelloy extensions to preserve temperature distribution and internal gas volume of the standard 12-inch specimen.

The performance of the Accident Tolerant Fuel (ATF) candidates must be evaluated under simulated loss-of-coolant-accident (LOCA) for the development of the licensing path for these materials. The cladding integrity is a primary focus of fuel safety testing and remains a significant research priority for the Advanced Fuels Campaign (AFC) on behalf of ATF vendor teams. The integral fuel safety testing ranges from separate-effects tests, including tube testing, simulated-LOCA testing, to the integral tests at the Idaho National Laboratory's (INL) Transient Reactor Test Facility (TREAT) reactor. As ATF concepts mature and undergo irradiations in commercial and test reactors, the availability of in-cell simulated-LOCA testing at the Oak Ridge National Laboratory (ORNL)'s Severe Accident Test Station (SATS) provides a direct comparison of simulated-LOCA test conditions previously performed on light-

The length dependence of integral LOCA test FeCrAl segments was studied between 4-inch and 12-inch found no difference in burst conditions resulting from the shortest length cladding segment, increasing the number of samples available for integral fuel safety testing from a lead test rod or irradiation specimen.

water reactor (LWR) cladding. SATS testing can expose a single 12-inch cladding segment to the temperature, pressure, oxidation and water-quench conditions anticipated during a LOCA or the high temperature module can be deployed to subject segments to beyond design basis (DBA) accident relevant conditions.

With the limited irradiated ATF material available for SATS testing, a standard SATS integral LOCA test of a 12-inch segment becomes a significant limiting factor on the amount of in-cell testing that can be performed due to the overall length. Additionally, High Flux Isotope Reactor (HFIR) irradiations of a single 12-inch test train remain a less-cost effective, practical barrier to utilizing target rod irradiations for in-cell burst testing. By researching the length-dependence of SATS integral LOCA testing, future irradiation and postirradiation examination campaigns may benefit from the increased test matrix made available with shorter test segments. For shorter segments to be utilized in SATS, the pressure fittings at the ends of the segments must maintain the internal pressure during the temperature ramp and the cladding must demonstrate similar balloon and burst behavior as a 12-inch segment.

Project Description:

As-drawn nuclear-grade unirradiated FeCrAl cladding tubes were tested at 12, 8, 6, 5, and 4-inch lengths at 1200 psi initial internal pressure. Four inches was found to be sufficient to induce burst during LOCA tests, so more tests with 4-inch length tubes were subsequently run at 550 psi and 850 psi initial internal pressures. In order to preserve a similar internal gas volume and temperature distribution with shorter-length specimens, thick-walled Hastelloy extensions were added to the bottom and top of the FeCrAl specimen so the total length was that of a 12-inch specimen. A schematic and image of

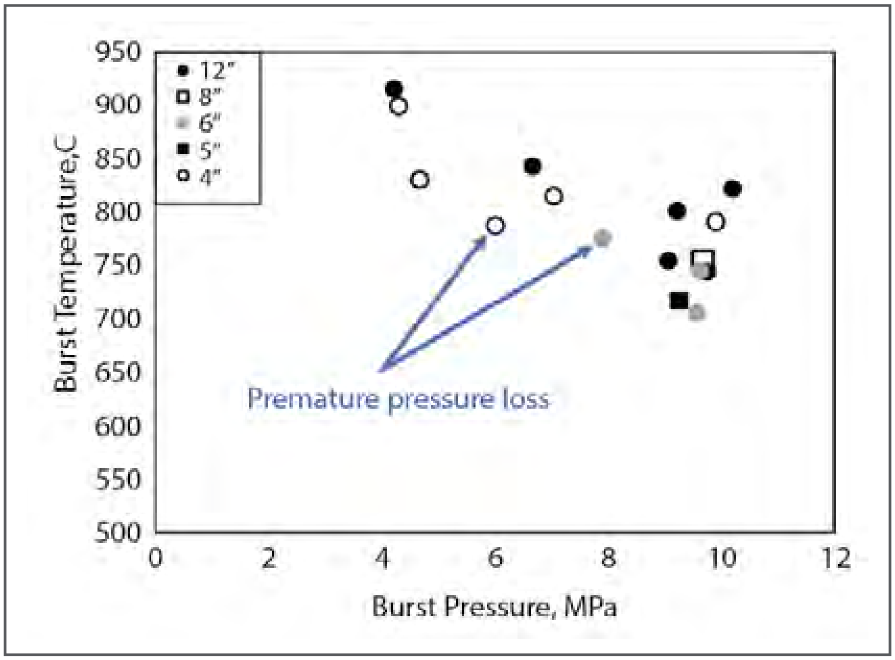
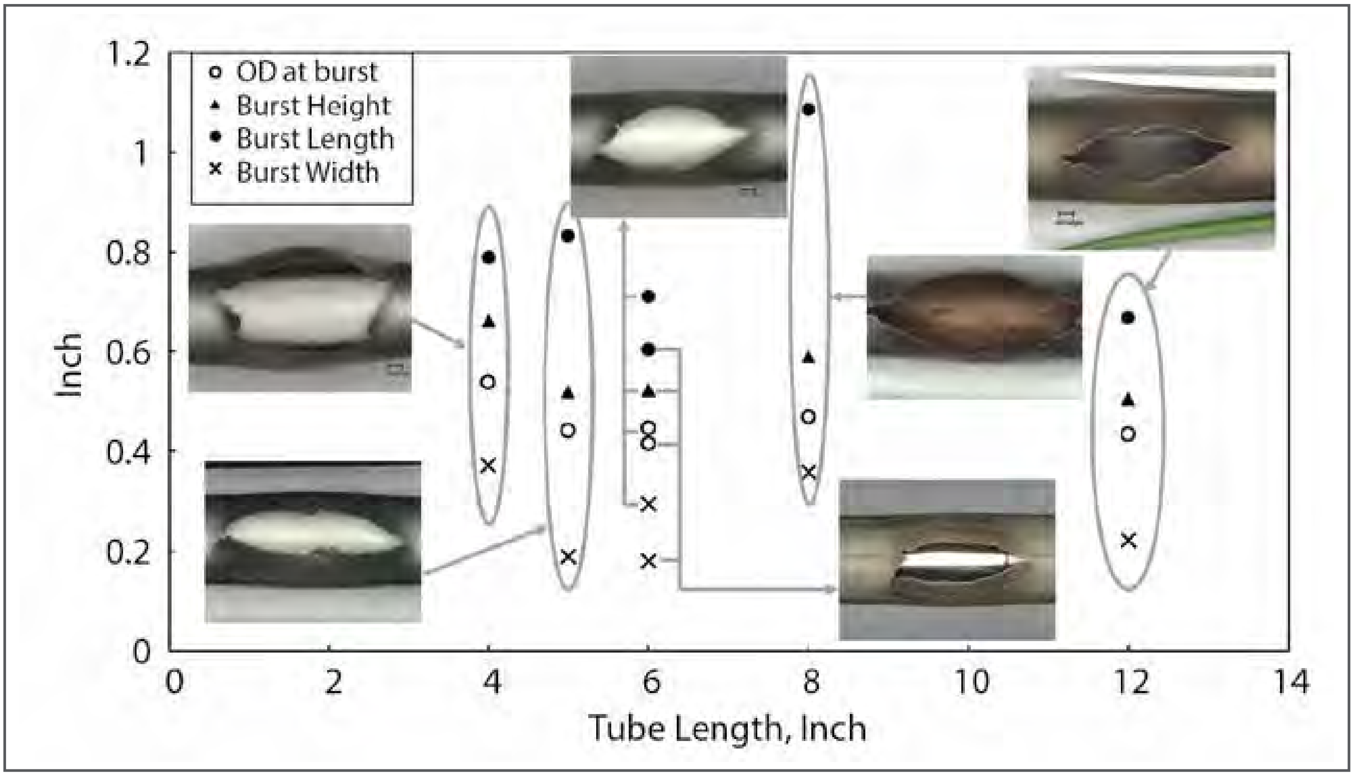


Figure 2. Results from design-based loss of coolant accident tests performed with varied-length nuclear-grade FeCrAl. No significant difference could be determined between 4-inch and 12-inch specimens during SATS LOCA tests, so the 4-inch specimens were found to produce valid results.

the short-sample setup with Hastelloy extensions is shown in Figure 1. Thermocouples were attached to the front and back of the specimens 2" above and below centerline for 12, 8, and 6-inch specimen lengths, 1.5" above and below centerline for 5-inch specimen length, and 1" above and below centerline for 4-inch specimen length. The thermocouples were attached using Platinum/30% rhodium wire to secure them to the tube surface. The bottom, front thermocouple was used for the control temperature, and the remaining 3 thermocouples were used to measure temperature. Burst location was tracked after the test, and the thermocouple closest to the burst was used to determine burst temperature. The tubing was found to have a variable wall thickness of ± 3%, so specimens were epoxy-mounted to measure specimen wall thickness post-test for hoop stress calculations. The burst opening length, width, height, and tube Outer Diameter (OD) at the burst location were measured with a micrometer to characterize the burst for all specimens. One out of five 4-inch and one out of three 6-inch specimens lost pressure prematurely but recovered and still burst after ballooning took effect. More data would determine if this result can be considered a low-pressure test on account of pressure loss or if the test should be rejected. Standard specimens approximately have a 20% failure rate, so if that specimen is considered a failure, then the failure rate of 4-inch specimens appears to be consistent with the standard 12-inch specimen.



Accomplishments:

This research was performed to study the length dependence of nuclear-grade FeCrAl cladding required to run a LOCA relevant experiment on the SATS. To accomplish this, experiments with varied lengths and pressures were performed using unirradiated FeCrAl thin tube cladding. The SATS system simulates a design basis LOCA by increasing the temperature of an internally pressurized tube in a steam environment in a postulated DBA. This increasing temperature consequently increases the internal pressure and hoop stress on the tube specimen until it bursts. Tube performance was quantified by the burst temperature, burst pressure, hoop stresses and burst characteristics of the tube specimen. Burst temperature and pressure results are shown in Figure 2. Burst measurements and images were performed with a micrometer and Keyence microscope respectively. Burst measurements and images for variable-length tubes tested at the same initial pressure are shown in Figure 3. Burst results from the various lengths of tube specimens were compared and demonstrated that specimens as short as 4" in length were within the scatter of standard 12-inch specimens for C26M FeCrAl material, so it was determined that a 4-inch specimen would produce valid results.

Figure 3. Results of burst measurements with varied-length nuclear grade FeCrAl specimens tested at the initial pressure. No significant differences could be determined for specimen lengths as short as 4-inch long.

2.6 LWR COMPUTATIONAL ANALYSIS

Accident Tolerant Control Rods

Principal Investigator: Michael Todosow, BNL
Team Members/ Collaborators: Arantzazu Cuadra, BNL

The combination of accident tolerant fuel/ cladding and advanced control rods with improved thermo-chemical/ mechanical characteristics enhances the performance and safety of LWRs in normal and accident conditions beyond those achievable by ATF alone.

Advanced "accident tolerant" control rods that have improved thermo-chemical/mechanical characteristics more than current silver-indium-cadmium (AIC) or boron-carbide (B4C) control rods, yet retain/enhance the poisoning effects/ requirements while maintaining structural integrity/functionality during normal operation and accident conditions are a natural complement to accident tolerant fuel (ATF) concepts. The combination of accident tolerant fuel/cladding and advanced control rods enhances the performance and safety of light water reactors (LWRs) in normal and accident conditions beyond those achievable by ATF alone.

Project Description:

Requirements and desirable characteristics of advanced/accident tolerant control rods include:

Options must:

- Maintain or exceed absorption capabilities (shutdown, ejected rod worth, etc.) for desired target lifetime
- Maintain structural integrity and functionality under operating and accident scenarios (ability to be inserted/withdrawn, handle loads/ shocks, such as due to SCRAM)
- Be chemically and mechanically "robust" in reactor temperature and radiation environment
- High miscibility with fuel materials to avoid possibility of re-criticality

Desirable characteristics include:

- High melting temperature: Absorber, cladding/sheath
- Minimal adverse chemical interactions: Between absorber and cladding/sheath, coolant, etc.
- Minimize adverse mechanical interactions, e.g., Swelling, wear/fretting, ballooning, bowing
- Resource availability, manufacturability, cost

The neutronic performance of several options for advance control rods that satisfied the above criteria was evaluated. Impacts considered control rod worth, as well as fuel and moderator temperature coefficients, and soluble boron worth.

Accomplishments:

The neutronic performance of options for more "robust" control rods (higher temperature, reduced hydrogen generation, minimize adverse chemical and mechanical interactions) to complement fuels with enhanced accident tolerance was evaluated via scoping calculations for a detailed TRITON model of a Westinghouse 17x17 fuel assembly.

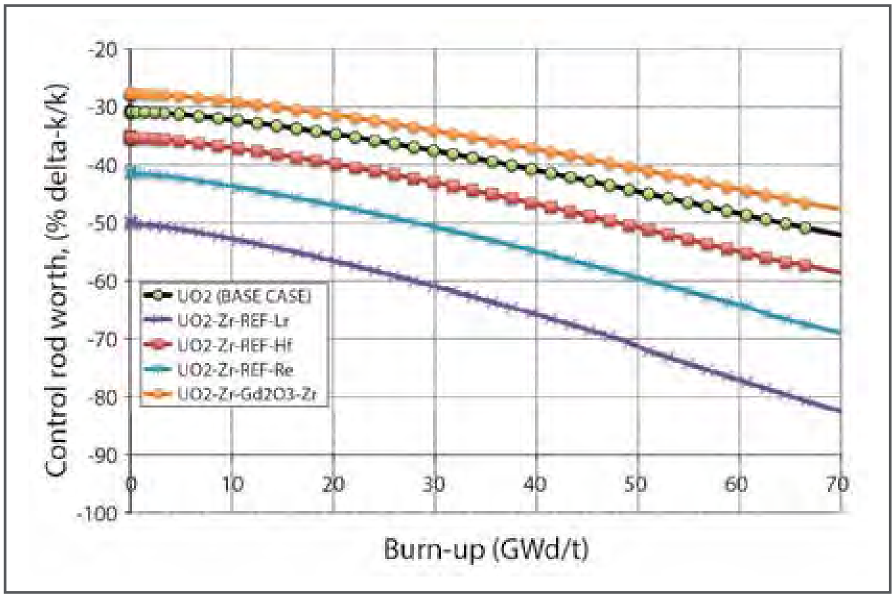


Figure 1. Control rod worths as a function of fuel burnup.

The standard Silver-Indium-Cadmium (AIC) control rod (24-rodlets) was replaced by selected "improved" control rods utilizing materials with higher temperature capabilities and comparable or better absorption properties. The reference AIC rod plus its steel cladding was replaced with solid rods of hafnium (Hf), iridium (Ir) and rhenium (Re). In addition, a gadolinium-oxide (Gd2O3) absorber with a Zircaloy cladding (same dimensions as the AIC rod) was evaluated. The control rod worths at beginning-of-life (BOL) from TRITON were compared to results from MCNP with good agreement. Control rod worths as a function of fuel burnup are shown in Figure 1 based on "branch calculation" of rod-out/rod-in, i.e., the control rod material remained at its initial composition. These results suggest that the number of control rods needed to satisfy control/shutdown requirements can be reduced, but may have a negative impact on ejected and stuck-rod worths/impacts. A possible approach for retrofit would be to reduce the number of "fingers" in the standard control rod design from 24 to a number that can match the worth of AIC rods.

Re-fabrication of Irradiated Fuel Rods and LOCA Burst Testing

Principal Investigator: L.-Y. Cheng, BNL
Team Members/ Collaborators: M. Todosow and A. Cuadra (BNL)

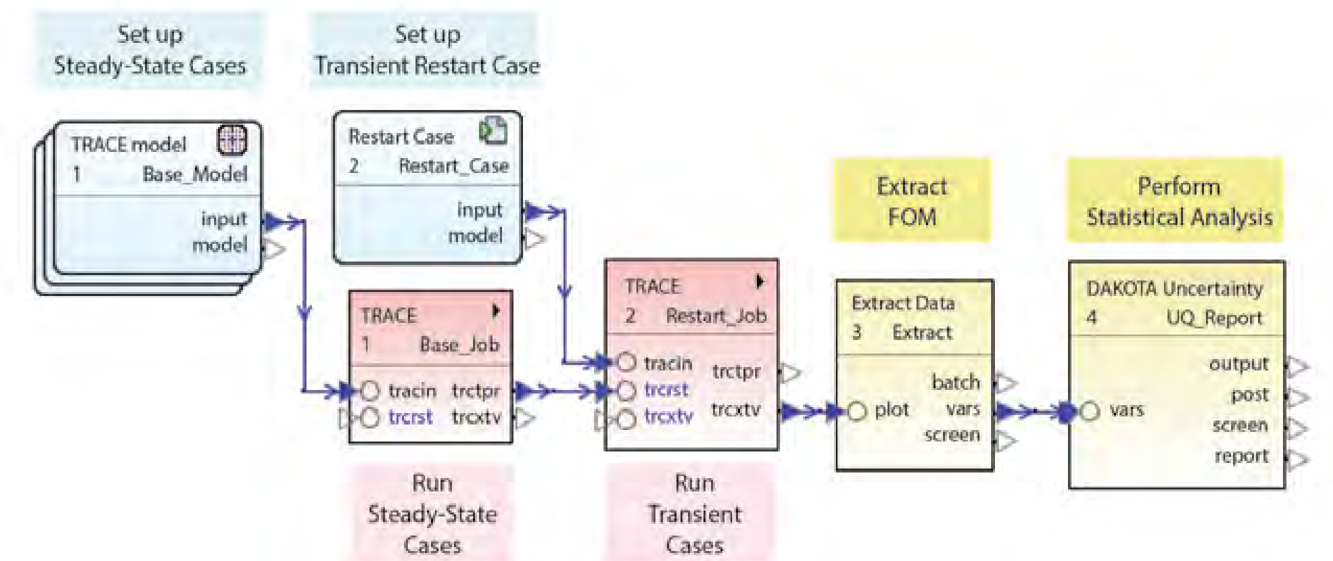


Figure 1. SNAP uncertainty job stream steps.

One aspect of evaluating the performance of accident tolerant fuel (ATF) under accident conditions is an uncertainty and sensitivity analysis to identify and quantify all potentially important parameters that impact the code predicted figure-of-merit (FOM), such as the peak cladding temperature (PCT). A framework to conduct uncertainty quantification (UQ) and sensitivity analysis (SA) has been established by using the TRACE-DAKOTA interface in SNAP (a graphical user interface to perform TRACE runs) and applying it to a previously developed 3-loop pressurized water reactor (PWR) plant model for the assessment of ATF fuel concepts. Using input models generated by random sampling of sensitivity parameters, the TRACE accident analyses produced results for calculating the FOM (or the response function). The UQ step computed the FOM value that conformed to some specified tolerance interval/limit. The SA step calculated correlation coefficients that indicated the relation between input (individual sensitivity parameter) and output (the FOM).

Project Description:

The analytical effort is to consider uncertainty in the assessment and comparison of different fuel and cladding options under accident conditions. There are two parts to the evaluation of fuel performance, UQ and SA. The UQ process, by statistical sampling of results of the accident analyses, evaluates the value of the FOM (e.g., the PCT) such that it satisfies

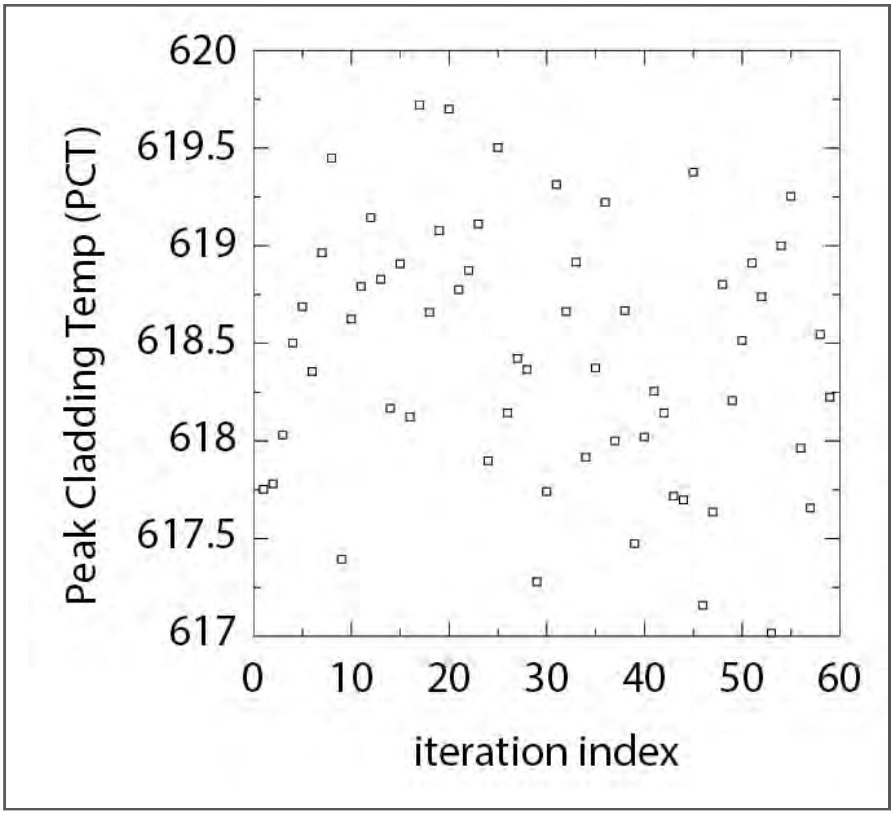
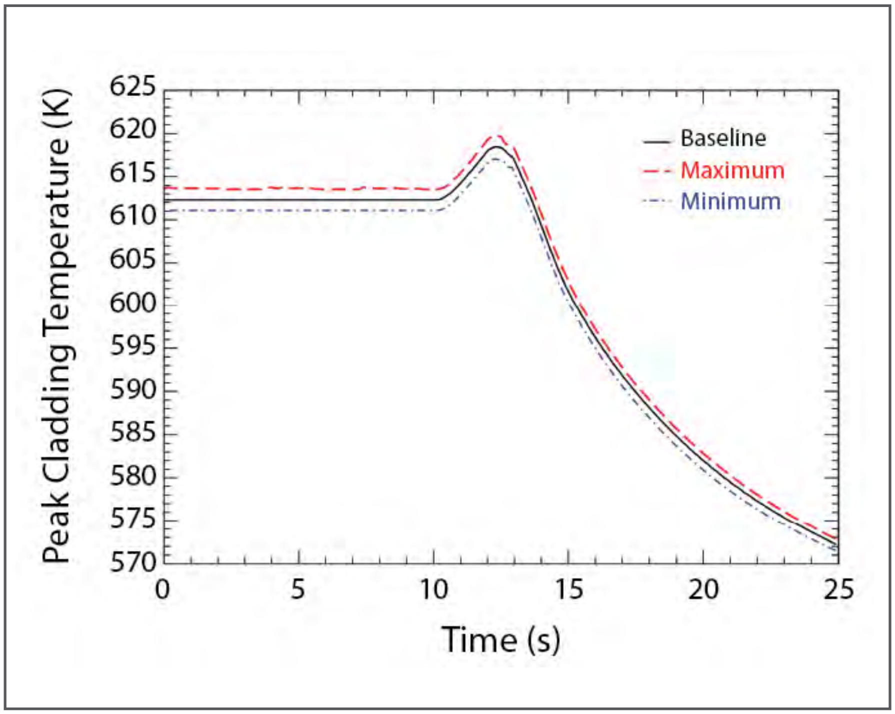


Figure 2. Peak cladding temperature (PCT) vs. iteration index.

some specified tolerance limits given in terms of probability and confidence level. Uncertainties in some of the ATF design parameters are inevitable while the ATF options are under development. Inherent in the UQ process is the ability to account for uncertainties and it facilitates a systematic comparison of the performance of different ATF options under accident conditions. The SA process evaluates the relative impact of different design parameters on the FOM. The quantification of the SA is by way of determining the correlation coefficient (normalized version of the covariance between two random variables) between some sensitivity input to the accident analysis (e.g., the thermal conductivity of fuel) and some output (e.g., the PCT) of the analysis. The statistical data required by the SA is assembled through multiple accident analyses with each analysis accounting for the random variability of some select sensitivity parameters. The correlation coefficients provide indication by magnitude, the strength of the relation between input and output magnitude the strength of the relation between input and output.

Figure 3. Comparison of PCTs.



The assessment of uncertainty for accident analysis informs ATF designers in a systematic and quantitative fashion how each fuel parameter will impact the figure-of-merit used in comparing performance of different design options and can focus property measurements to improve steady-state and accident performances.

Thus, the SA informs the designers where to focus their efforts to refine design parameters that have the biggest impact on the outcome of an accident. Applying UQ and SA to accident analysis helps stakeholders, the Advanced Fuels Campaign (AFC) and industry, formulate areas of research that are most beneficial to the safety performance of different ATF concepts.

Accomplishments:

A loss of offsite power (LOOP) accident for a PWR with standard fuel (UO_2 pellet and Zr cladding) was used to demonstrate the UQ/SA exercise. The analysis considered six (6) model sensitivity parameters that are related to fuel performance in an accident. They are core power, fuel gap conductance, fuel thermal conductivity, clad thermal conductivity, clad specific heat and fuel specific heat. The FOM chosen for this analysis was the PCT. All sensitivity parameters were assumed to have normal distribution with lower and upper bounds and a defined standard deviation. DAKOTA first sampled (Monte-Carlo or Latin Hypercube) the six prescribed sensitivity parameters and generated sets of variates for the sensitivity parameters. The sets of variates were then used by SNAP to generate input for the parametric cases. After completing the steady-state run for each sampled case a TRACE restart run was conducted for the transient calculation. The final step was to pass on the FOM (in this analysis the PCT) for DAKOTA to perform the UQ/SA analysis. The SNAP UQ job stream that automated the interface between DAKOTA and TRACE is shown in Figure 1.

Based on the first-order Wilks formula, a minimum of 59 parametric cases is required to quantify the 95/95 one-sided tolerance limit for the

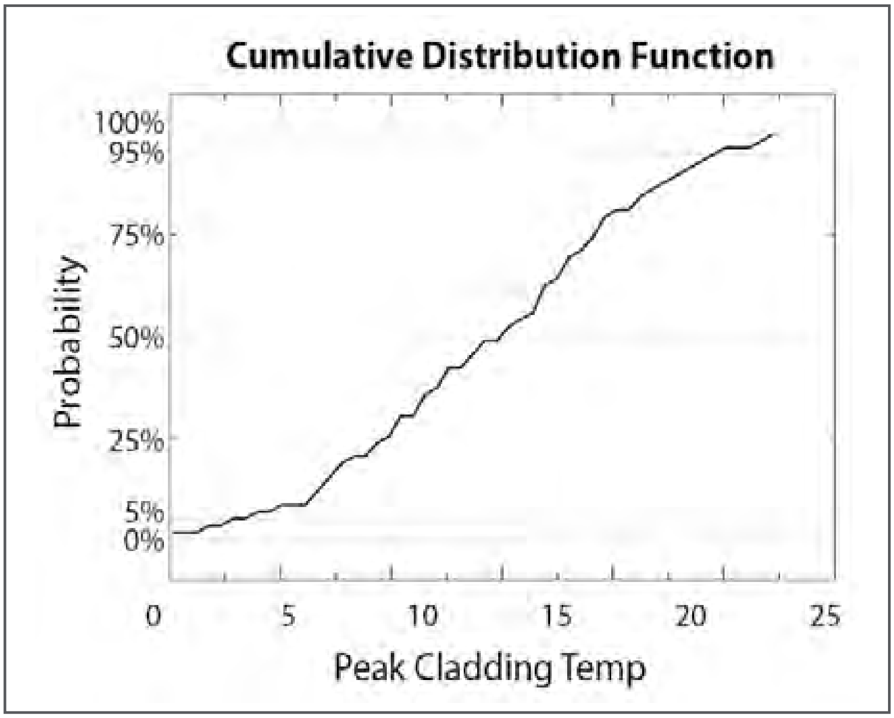


Figure 4. Cumulative distribution function (CDF) for the PCT.

PCT. That is, it is 95% confident that the maximum PCT from the 59 calculations bounds at least 95% of the PCT population distribution. Figure 2 shows the calculated PCT versus the iteration index (1 through 59) using the Latin Hypercube sampling method. Figure 3 compares the PCT from three cases, the maximum PCT, the minimum PCT and the baseline case (no uncertainty). The results indicate that the chosen sensitivity parameters with their probability distributions do not have significant impact on the PCT. This is not unexpected because of the mild transient as a result of the LOOP (a primary pump trip followed by reactor trip). Figure 4 shows the CDF for the PCT. The 95/95 PCT limit corresponds to the maximum PCT calculated for the 59 parametric cases and is 619.72 K. A second UQ calculation using the Monte-Carlo sampling method resulted in similar results with a slightly different PCT limit of 619.65 K.

For the SA, DAKOTA calculated the response correlations for the FOM. The correlation of interest is the partial correlation coefficient. It reflects the correlation between two variables (any of the sensitivity parameters and the PCT) after adjusting for (i.e., removing) the effects of other variables. The partial correlation coefficients indicate the PCT has a strong positive relation to core power, a strong negative relation to the fuel thermal conductivity, a mildly negative relation to both the gap conductance and the clad thermal conductivity, and a much smaller relation to the specific heat of the fuel and the cladding. The SA results suggest that for the assumed uncertainties the PCT for a LOOP is influenced mainly by fuel parameters that raise the initial fuel temperature. The partial rank correlations suggest the same conclusion.

Effect of Diamond Doping on the Fission Gas Release In UO_2 Fuel Irradiated to 7.2 GWd/tHM.

Principal Investigator: Pavel Medvedev

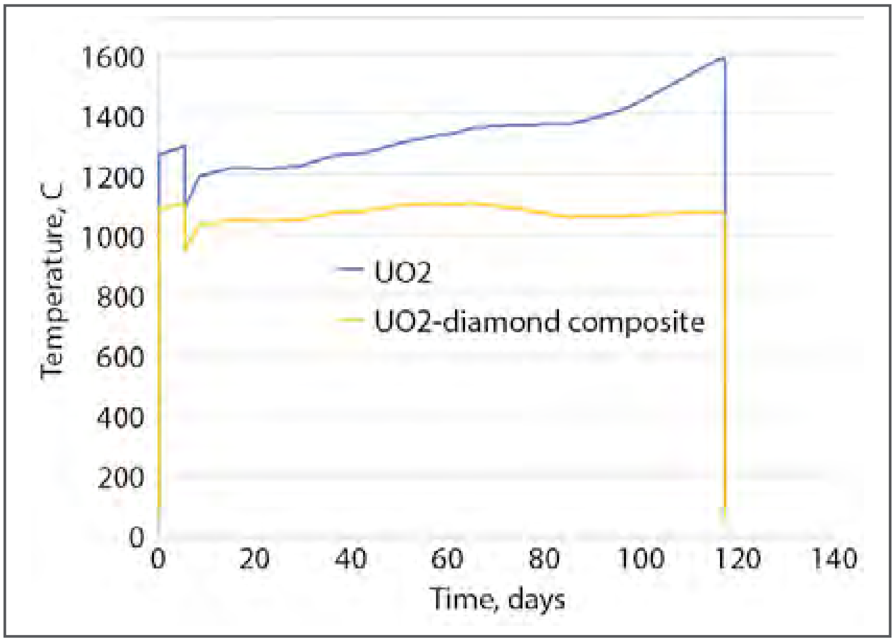


Figure 1. Comparison of the calculated peak fuel temperature for the standard UO_2 fuel and UO_2-5 vol % diamond composite.

An array of innovative fuels and materials combinations has been irradiated in the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL) with an objective to develop an Accident Tolerant Fuel (ATF) in response to the Fukushima accident. ATF-1 is an initial test series whose irradiation commenced in 2015. Detailed description of the ATF-1 test matrix, experiment design, irradiation conditions and the current status of irradiation and postirradiation examination (PIE) has been provided (Core et al., 2017).

UO_2 pellets doped with 5 vol % diamond fabricated by spark plasma sintering were included in the ATF-1 irradiation experiment in an effort to assess the ability of diamond, a material with the highest thermal conductivity of any known material, to improve the performance of the UO_2 fuel. The concept of using diamond for this purpose, along with the fabrication method and characterization is well documented (Chen et al., 2015; Morrell, 2015).

Results of the PIE of selected ATF-1 rodlets have been reported recently (Cappia, 2019). It was found that UO_2 pellets doped with 5 vol % diamond exhibited a fission gas release of 1.08%. In order to determine whether the diamond doping had an impact on the fuel performance, fuel performance modeling was carried out to calculate fission gas release in undoped UO_2 under identical irradiation conditions. Results of these calculations are the main subject of the present paper.

BISON, a modern finite-element based nuclear fuel performance code, was utilized for the analysis presented here. BISON has been under development at the INL since 2009. The BISON light water reactor (LWR) validation database includes 73 integral fuel rod assessment cases evidencing that simulation results compare quite well with LWR experimental measurements. Good agreement between predicted and measured low burnup fission gas release is specifically called out (Assessment of BISON, 2018), making BISON particularly suitable for the present work.

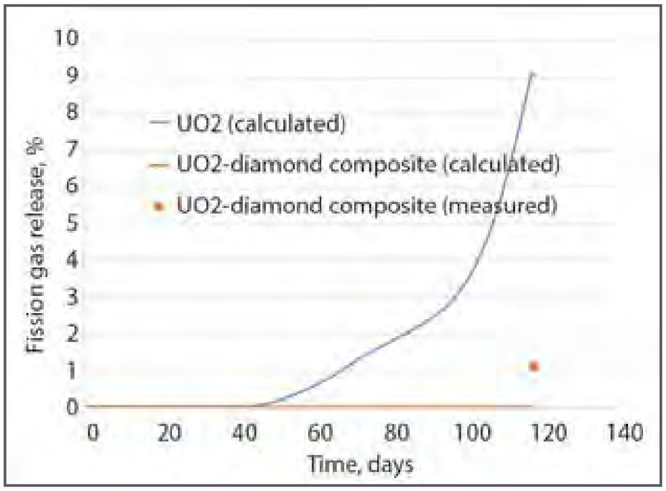


Figure 2. Comparison of the calculated fission gas release for the standard UO_2 fuel and for the UO_2-5 vol % diamond composite.

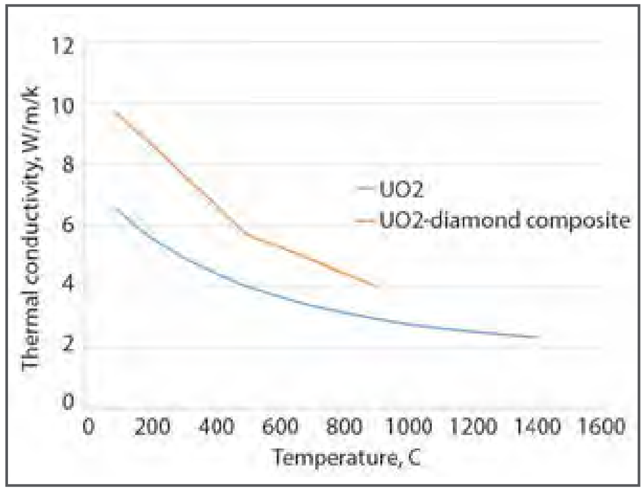


Figure 3. Comparison of the thermal conductivity for the standard UO_2 fuel and for the UO_2-5 vol % diamond composite.

Test Description

A UO_2-5 vol % diamond composite rodlet containing eight fuel pellets and two depleted uranium oxide insulator pellets was irradiated as part of the ATF-1 experiment in a small I position of the ATR operated by INL. This was a non-instrumented drop-in test that utilized the double encapsulation concept to achieve prototypic cladding temperatures. In this design, the test rodlet is housed in a sealed helium-filled stainless steel capsule. As the ATR inlet coolant temperature is only 52°C and the desired cladding temperature is 360°C, a significant thermal resistance between the rodlet and the coolant is required to achieve the desired cladding temperature. Double encapsulation introduces a helium gap between the test rodlet and the capsule, thus providing the necessary thermal resistance.

Results

Peak fuel temperature

Comparison of the peak fuel temperature for the standard UO_2 fuel and UO_2-5 vol % diamond composite is shown in Figure 1. In the beginning of life the peak temperature of the standard UO_2 fuel is 180°C higher. At the end of life, the peak temperature of the standard UO_2 fuel is 511°C higher. Standard UO_2 fuel exhibits higher temperature due to its lower thermal conductivity and due to the increase of the thermal resistance of the fuel-cladding gap which occurs after the onset of the fission gas release.

Fission gas release

Comparison of the fission gas release for the standard UO_2 fuel and for the UO_2-5 vol % diamond composite is shown in Figure 2. These results

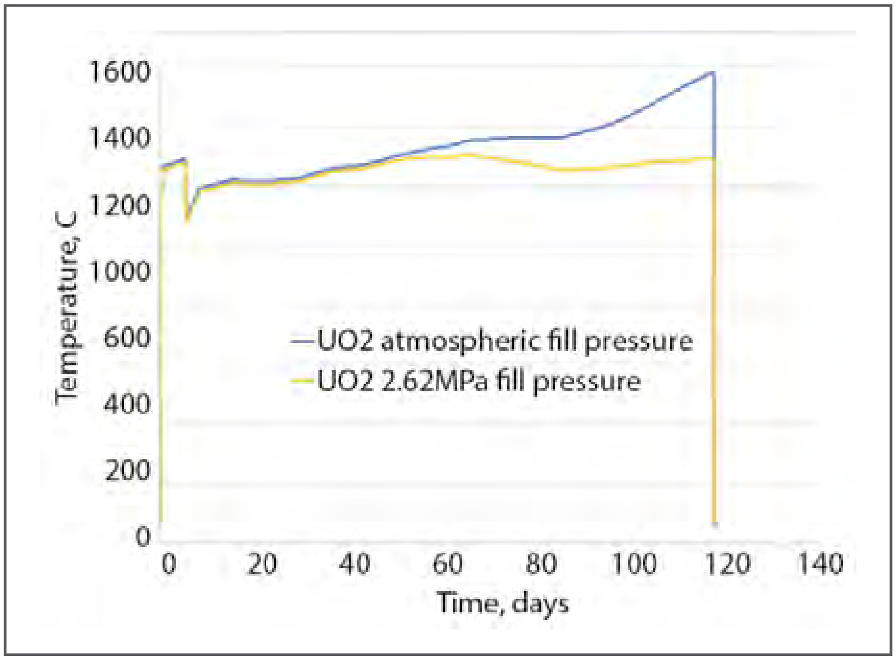


Figure 4. Effect of helium fill pressure on the peak fuel temperature.

demonstrate the impact of diamond doping on the fission gas release in in UO_2 fuel irradiated to 7.2 GWd/tHM. Evidently, the use of diamond doping resulted in marked reduction of the fission gas release. Therefore, it is concluded that diamond doped UO_2 shows promise as an ATF based on the results of this initial screening test.

Thermal conductivity

Comparison of the beginning of life thermal conductivity for the standard UO_2 fuel and for the UO_2-5 vol % diamond composite is shown in Figure 3. Comparison of the thermal conductivity for the standard UO_2 fuel and for the UO_2-5 vol % diamond composite.

Effect of helium fill pressure on the fuel temperature and fission gas release

As mentioned above, ATF-1 rodlets were filled with helium gas at the atmospheric pressure of 0.084 MPa. To assess the effect of helium fill pressure on the fuel temperature and fission gas release, a fuel performance calculation assuming prototypic LWR helium fill pressure of 2.62 MPa was executed and compared with that of the reduced fill pressure rodlets. As evident from Figures 4 and 5, rodlets with reduced fill pressure exhibit higher fuel temperature and fission gas release. When the fission gas release begins, rodlets with reduced fill pressure experience greater degradation of the gap conductance because the same amount of released

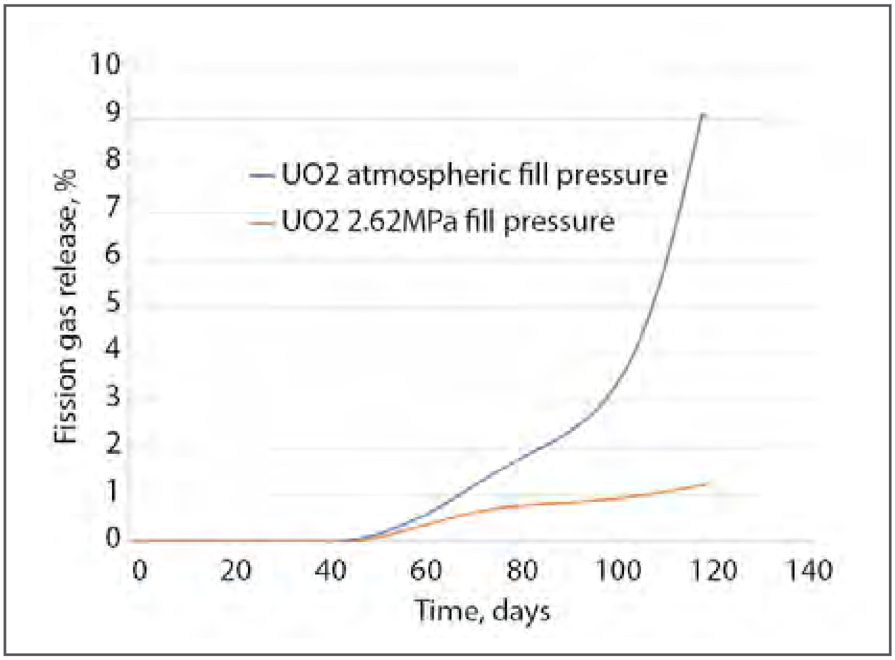


Figure 5. Effect of rodlet fill pressure on fission gas release.

fission gas yields greater resultant fission gas concentration in the plenum gas. These findings suggest that after the onset of the fission gas release, ATF-1 test conditions are more aggressive than typical LWR fuel operating conditions.

Conclusions

UO_2 pellets doped with 5 vol % diamond particles were irradiated in the ATR at linear heat generation (LHGR) of up to 310 W/cm to the burnup of 7.2 GWd/tHM. Fission gas release, measured during PIE, was determined to be 1.08%. Fission gas release modeling performed using BISON fuel performance code showed that undoped UO_2 fuel irradiated under the identical conditions would have had fission gas release of 9.09%. Noting that diamond has the highest thermal conductivity of any known material, these results suggest that doping with a good thermal conductor is an effective means to reduce fission gas release in UO_2 fuels.

BISON calculations show that ATF-1 experiment conditions are more aggressive than typical LWR fuel operating conditions due to ATF-1 reduced fill pressure. This finding is important for interpreting ATF-1 test results and drawing comparisons with the baseline LWR fuel performance. Recognizing that the ultimate goal of ATF-1 test series is screening of the early accident tolerant fuel concepts, the more aggressive conditions of the ATF-1 are fully appropriate.

Development of BISON Models for Coated Cladding

Principal Investigator: Ryan Sweet, ORNL
Team Members/ Collaborators: Brian Wirth and Andy Nelson (ORNL)

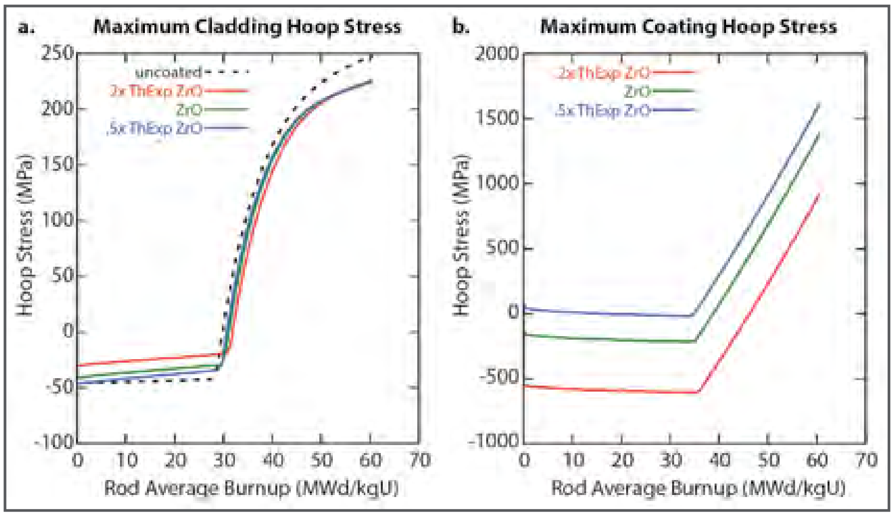


Figure 1. These simulations demonstrate a minimal effect of zirconium oxide coatings on the cladding hoop stress state a) however the coating hoop stress b) varies a sizeable amount based on the property mismatch with the cladding.

Developed coating analysis capabilities have garnered interest from several fuel vendors for assessing their near-term accident tolerant fuel concepts.

The motivation for developing alternative cladding materials for light water reactor (LWR) fuel systems is to improve reactor safety during high-temperature transient conditions, where zirconium cladding rapidly oxidizes. As the cladding reacts with the coolant, it both releases energy and produces hydrogen gas. Coatings applied to the exterior surface of zirconium cladding have been identified as a possible solution to help limit cladding oxidation at elevated temperatures. These coating materials must remain adherent to the cladding substrate to realize the oxidation protection and are expected to modify the established behavior of the zirconium alloy cladding during reactor operation. This work focuses on the development of constitutive models for proposed coating materials and an assessment of the integral thermal-mechanical performance of fuel rods with coated cladding.

Project Description:

The goal of this effort is twofold; to identify the constitutive behavior characteristics that a coating material must possess to remain beneficial, and to develop accurate coating analysis capabilities, which allow us to collaborate with industry partners on near-term accident tolerant fuel (ATF) solutions.

For this project, the BISON fuel performance code is used due to its multi-dimensional and multi-physics capabilities, as well as its extensive library of LWR performance models. Constitutive properties for the proposed coating materials are implemented and simulations are performed to evaluate the behavior of both the coating and cladding under LWR conditions. Key cladding performance indicators, such as creep down and stress development after gap closure, are investigated to ensure that the coating is not negatively impacting the cladding behavior. Correspondingly, the coating behavior itself is examined to identify if the coating is in danger of cracking, delamination, or spallation. Additionally, parametric analyses of core thermomechanical properties allow us to determine where additional investigation is needed, and can help inform experimental conditions.

As material property testing commences at Oak Ridge National Laboratory (ORNL), refinements to the constitutive models will be made and additional simulations will be performed of integrated experiments to benchmark these developed capabilities and elucidate supplemental data.

Accomplishments:

For this analysis, simulations were performed using a 15μm zirconium dioxide coating to demonstrate the coating capabilities of BISON. While these results are not intended to provide quantifiable results for a particular case, they do display general trends of a material with these properties, i.e., increased fuel rod temperatures due to poor coating thermal conductivity and increased coating stresses due to the absence of a creep or plasticity model. As well, an initial parametric evaluation has been performed to demonstrate the effect of varying the thermal expansion coefficient (by 2× and ½×) on the coating and cladding stress state. Successful use of BISON to model the effects of zirconium dioxide on cladding properties and fuel performance also represents an important first step in modeling of other coating materials.

This analysis is performed using prototypical Boiling Water Reactor (BWR) operating conditions and fuel geometry. Figure 1a shows the maximum cladding hoop stress resulting from the presence of a 15μm coating, with a constant thermal expansion coefficient during the modeled operation, up to 60 MWd/kgU. As the thermal expansion coefficient is increased, the cladding stress state is more tensile and the cladding creep down behavior is slowed, resulting in delayed gap closure. The maximum hoop stress in the coating based on the thermal expansion coefficient is shown in Figure 1b. Here, there is a significant change in the predicted hoop stress with variation of the thermal expansion in the coating. Leading up to gap closure, the hoop stress decreases slightly, becoming more compressive. After gap closure occurs, the hoop stresses become rapidly become very large. Although no creep or plasticity models are used in this analysis of the impact of the coating, large tensile stresses typically indicate mechanical failure.

Figure 2 shows the radial distribution of hoop stress across the coating and cladding at 60 MWd/kgU. There is an extremely large jump in stresses moving radially into the coating due to the thermal expansion mismatch and the lack of a plasticity model to relieve stresses. The cladding, however, shows only a slight difference based on the coating properties.

This highlights the need to identify coating properties that are compatible to those of zircaloy for favorable performance, and these results demonstrate that a framework is in place to model the coating and fuel rod performance as more representative material properties are established. In an effort to provide high-fidelity simulation results and leverage the versatility of BISON as a finite element tool, current efforts will also include modeling ongoing thermal-mechanical property testing. This consists of developing test geometry and implementing relevant conditions to simulate the property test and assess the accuracy of the current material models.

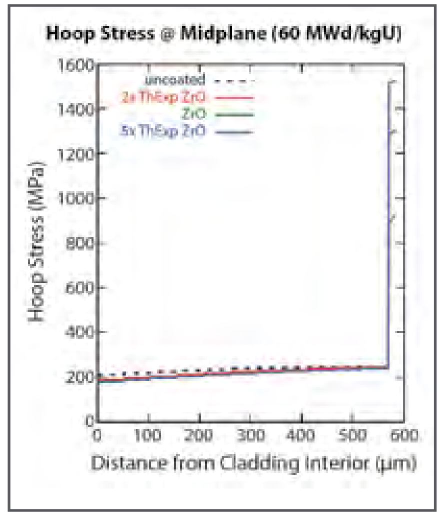


Figure 2. The hoop stress plotted as a function of cladding radius. A drastic increase in hoop stress is shown between the cladding to the coating due to thermal expansion mismatch between the materials.

Use of Advanced Core Neutronic Simulation Tools to Improve Mechanistic Models of SiC/SiC Performance

Principal Investigator: Nicholas Brown (University of Tennessee, Knoxville)
Team Members/ Collaborators: Jacob Gorton (University of Tennessee, Knoxville; (ORNL); Benjamin Collins and Andrew Nelson (ORNL)

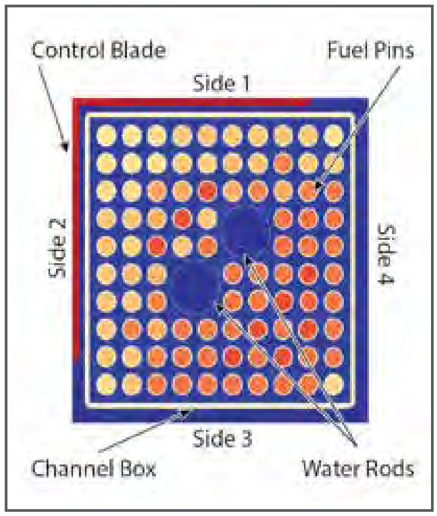


Figure 1. Cross-section of 10x10 BWR fuel assembly model.

Using VERA-CS to obtain temperature and fast neutron flux distributions in the channel box will provide accurate and high-fidelity boundary conditions that are useful in determining the feasibility of SiC/SiC as a channel box material in BWRs.

Channel boxes surrounding each fuel assembly in a boiling water reactor (BWR) make up a significant portion of the zirconium in the reactor core. In line with the goals of the Accident Tolerant Fuel (ATF) program, silicon carbide (SiC) fiber-reinforced, SiC matrix composite, or SiC/SiC, is being considered as a potential replacement for zirconium as a channel box material. SiC/SiC has been shown to be significantly more oxidation-resistant than zirconium, so using SiC/SiC as a channel box material may reduce the hydrogen production in a BWR in the event of a severe accident. However, SiC/SiC is known to undergo irradiation swelling, and the nonuniform fast neutron flux distribution in a BWR may lead to significant deformation of the channel box. Research efforts have aimed at evaluating the performance of SiC/SiC as a channel box in BWR fuel assemblies using advanced modeling tools with coupled thermal-hydraulics-to-neutronics capabilities.

Project Description:

The modeling tools used for this analysis are the coupled versions of MPACT and CTF within the Virtual Environment for Reactor Applications – Core Simulator (VERA-CS) developed by the Consortium for Advanced Simulation of LWRs (CASL). MPACT is a deterministic reactor physics code, and CTF is a thermal hydraulics subchannel code. These computational tools have been coupled together by CASL to create a feedback loop in which the temperatures and densities calculated by CTF are used to update the cross sections used by MPACT, which then calculates the radial and axial power profile in the model. The power profile is fed back into CTF, and the process iterates until convergence criteria are met. Several coupled codes have been developed for BWR application, including TRACE/PARCS and PARCS/PATHS, but one of the additional benefits of using VERA-CS is the level of fidelity provided by the output. MPACT and CTF provide results on a pin and subchannel resolved

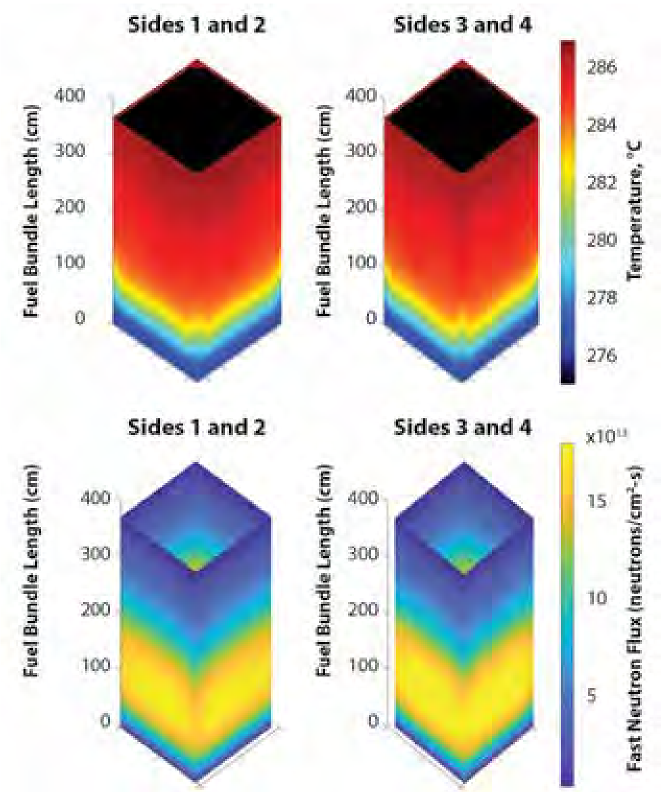


Figure 2. Temperature and fast neutron flux distributions when the control blade is fully withdrawn.

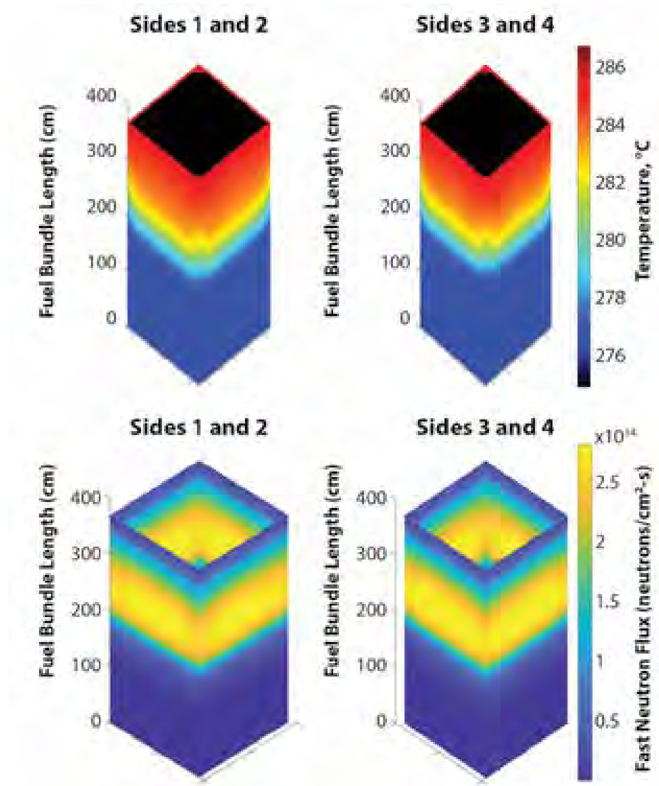


Figure 3. Temperature and fast neutron flux distributions when the control blade is halfway inserted.

scale, while codes like PARCS/PATHS provide results on a fuel assembly scale. It is this level of fidelity and the coupled feedback loop that led to VERA-CS being chosen as the tool for this study. Models utilizing SiC/SiC channel boxes were developed in VERA-CS of a single 10x10 BWR fuel assembly and of a square mini-core consisting of sixteen fuel assemblies.

Accomplishments:

The coupled MPACT/CTF models in VERA-CS were used to calculate fast neutron flux and temperature distributions in the single fuel assembly and mini-core models under typical BWR operating conditions. Several characteristics of BWRs, such as the generation of vapor in the core, part-length rods, and axial regions

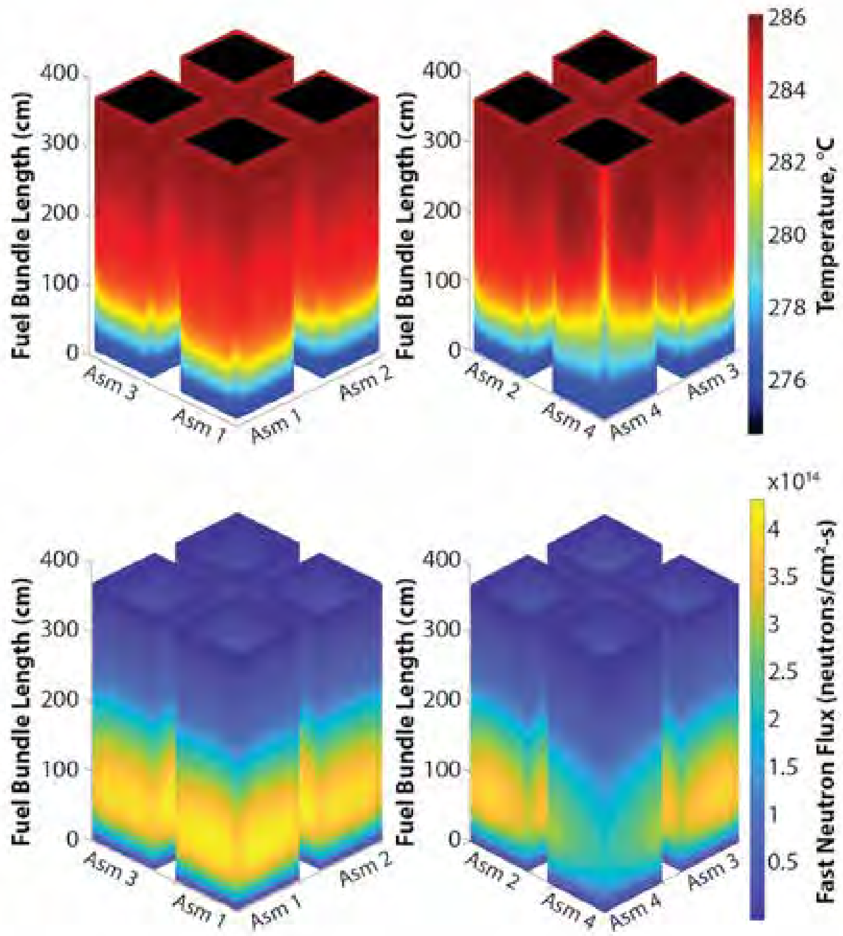


Figure 4. Temperature and fast neutron flux distributions in a control cell with the control blade fully inserted.

containing different fuel loading patterns and gadolinia content, lead to nonuniform fast neutron flux in the core. Multiple axial regions with different fuel loading patterns are able to be modeled in VERA-CS, and the coupled codes predict void fraction distributions representative of real BWRs. Part-length rod modeling is still under development by CASL, but a workaround was used in which extremely low fuel density and enrichment (both on the order of 10-6) were used in the upper regions of the model where part-length rods would end. This workaround allows for MPACT and CTF to accurately predict power and temperature distributions, although the coolant flow patterns predicted in CTF are affected. Other features of BWRs are included in the models, such as cruciform control blades that are inserted from the bottom of the core and large water rods. A 2-D cross-section of a single fuel assembly model is shown in Figure 1 with key characteristics highlighted, including a quarter of a control blade. The different colored fuel pins indicate different levels of fuel enrichment and gadolinia content. Each fuel assembly in the mini-core model is identical to the one shown in Figure 1, but the control blade is located in the center of the 4x4 array of assemblies.

Fast neutron flux and temperature distributions were obtained in the channel box for varying degrees of control blade insertion. Examples of the distributions are shown in Figures 2 and 3, which show the temperature and fast flux distributions in the single assembly model channel box with the control blade fully withdrawn and halfway inserted, respectively. Additionally, the temperature and fast neutron flux is shown in Figure 4 for a control cell (four fuel assemblies) from the mini-core model. Obtaining these distributions under varying operating conditions is important for informing deformation studies that will determine the feasibility of using SiC/SiC as a BWR channel box material since too much deformation could lead to interference with control blades or stress-induced failures.

ADVANCED REACTOR FUELS

Demonstrate Machining Capabilities for Pu Bearing Fuels

Principal Investigators: Leah Squires and Randy Fielding, INL
Team Members/ Collaborators: James Newman, Ryan Johnson, Steven Monk and Blair Grover (INL)

This project demonstrates the ability to create numerous fuel geometries for irradiation testing via machining as cast fuel.

One key aspect to make disposal and storage of used metallic nuclear fuel easier is elimination of the sodium which is typically used to create a thermal path out of the fuel and allow for fuel swelling. Due to its reactive nature, sodium in used fuel contributes a unique set of challenges when it comes to disposal. Since the main purposes of the sodium are thermal transfer and to allow the fuel to swell, alternate methods by which to do these things are being explored. One such method is to allow for space internal to the fuel pin for the swelling to occur. In order to do this, holes or spaces in the fuel may be created either during the casting process or afterwards. This study looked at methods to machine plutonium bearing alloys in preparation for feasibility testing of metallic fuel forms suitable for irradiation without the sodium bond.

Project Description:

In order to eliminate the sodium in used fuel, this work investigated alternative methods for creating a thermal path and allowing fuel to swell. In particular, changes in the fuel geometry to accommodate swelling while allowing a mechanical thermal bond show promise for achieving this goal. Changing the fuel geometry will require that the fuel is either cast in the novel geometry, forged through a deformation process, or machined after casting. Machining, although not optimal for full scale operations, is a useful tool for fabricating novel geometries which would require a substantial amount of process and equipment development. Due to the high temperature phase change of plutonium, machining of the material presents a challenge. The high temperature phase of the material is brittle and difficult to machine and heat from the machining process is thought to bring about the phase change. The goal of this research was to find ways to machine Pu bearing material that will allow for irradiation testing of novel fuel geometries within the constraints of current gloveboxes and facilities. Some geometries require drilling while others require turning on a lathe. Previous work on U-Zr fuels has shown it is also necessary to provide a smooth outer surface on the fuel for consistent thermal conductivity in mechanically bonded fuels. The key objective of this project was to demonstrate these capabilities using as cast plutonium bearing alloys.

Accomplishments:

The main goal of this work was to demonstrate the feasibility of machining plutonium bearing fuels. Due to the brittle nature of plutonium fuel alloys they can be challenging to machine. This is further complicated by constraints of current facility

gloveboxes and equipment. A small modified hobby lathe was placed in an inert atmosphere glovebox for the process. This lathe is shown in Figure 1. The first geometry attempted was annular. This fuel geometry is of interest because the center void allows the fuel to swell inwards while maintaining a mechanical bond to release heat without straining the fuel cladding, thereby eliminating the need for sodium. This geometry has shown good behavior through simulations and testing in U-10Zr alloys but has not been tested in plutonium bearing fuels. In order to reduce the potential

Figure 1. Modified hobby lathe with fuel slug in the chuck to be machined.

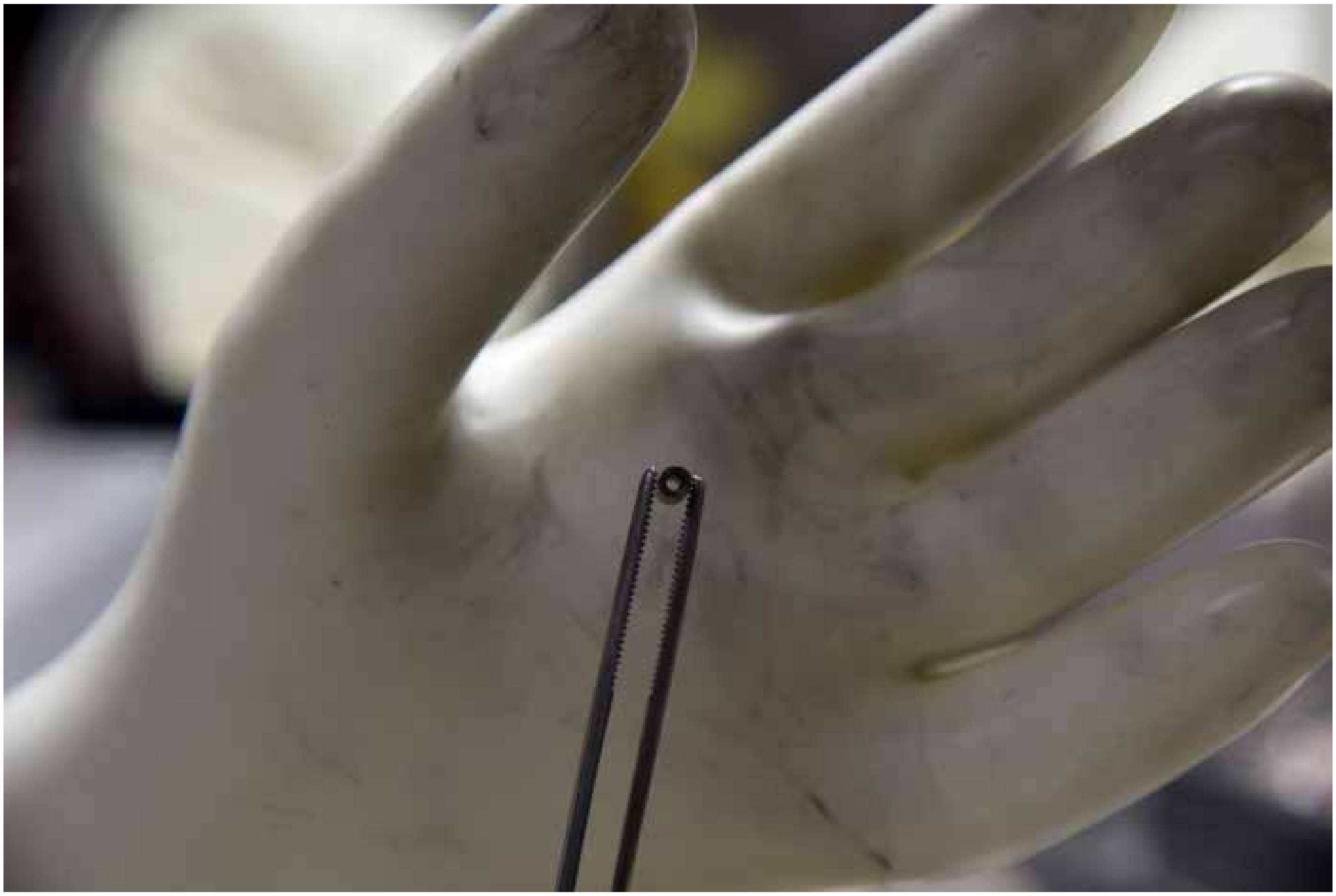


Figure 2. Machined annular fuel pin.

of overheating the fuel and cutting tool during the machining process, machine oil was used to cool the machined surface. First the fuel slug was held in place and turned down using the lathe attachment in order to reduce its diameter and also to smooth the surface. The fuel was then held in place while a drill bit was used to hollow out the center and create an annular geometry. Machine oil was applied liberally to the drill bit each time it was removed from the center of the fuel rod. This was successfully performed on a 0.75" long, 0.192" outer diameter slug of U-20Pu-10Zr fuel using a 0.116" drill bit to create an annular fuel pin with little under 65% smear density. Figure 2 shows the machined annular slug. It was found during this process that the machine oil was not necessary to keep the materials from overheating. The second geometry machined was a "clover leaf". This geometry requires that slots be machined down the length of the fuel pin to allow space for swelling and heat transfer without

cladding strain and thus eliminates the need for sodium. It may also be feasible to directly cast this geometry using standard casting processes. First the fuel slug was turned down to smooth the surface. Then a mill attachment was placed on the hobby lathe and the slots were made along the length of the fuel. First three rounded slots were made then another sample was made using different types of drill bits to make one rounded slot, one square slot and one triangular slot along the length of the fuel pin to show the machining capabilities. The results are shown in Figure 3. Again it was found that the machine oil was unnecessary. This work developed a useful method for creating test fuels of differing geometries that can be incorporated into a drop in irradiation test. It will be useful moving forward to have a variety of options for changing the fuel slug geometry.

Figure 3. Slotted fuel pin with one rounded slot, one square slot and one triangular slot.

Evaluation of Metallic Fuel Additives in U Fuel (U-Sn and U-Pd)

Principal Investigator: Michael T. Benson, INL
Team Members/ Collaborators: Yi Xie and James King (INL)

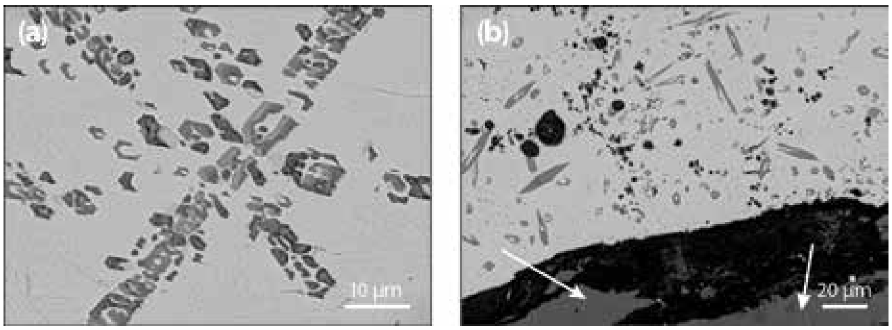


Figure 1. SEM backscatter images of U fuels with Sn additive. a.) U-Sn, b.) U-Sn-Ln.

Fuel-cladding chemical interaction due to fission product lanthanides can be prevented using additives to react with the lanthanides, thus extending the lifetime and increasing the potential burn-up of a metallic fuel.

Fuel-cladding chemical interaction (FCCI) occurs when the nuclear fuel or fission products react with the cladding material. A major cause of FCCI in U-based fuels during irradiation is fission product lanthanides (Ln), which migrate to the fuel periphery, coming in contact with the cladding. The result of this interaction is degradation of the cladding, and will eventually lead to rupture of the fuel assembly. Tin and palladium are being investigated as minor component additives to control FCCI in metallic fuels specifically due to lanthanides. The role of the additive is to prevent FCCI by forming very stable intermetallic compounds with the lanthanides, thus preventing interaction with the cladding. Previous studies investigated additives in a U-10Zr fuel. These studies have been extended to include U fuel, without Zr present.

Project Description:

The technical objectives of this research are to investigate additives to metallic fuels for improved FCCI performance. Previous investigations using tin and palladium as fuel additives have shown promising results for controlling FCCI. The current work is a continuation of that work, using U as the fuel, without Zr. The objective of this work is to characterize the microstructure of the metallic fuels with these additives, and to evaluate performance in out-of-pile diffusion couple tests.

An additive that effectively controls FCCI will help the Department of Energy (DOE) meet its objectives of a safe, reliable, and economic reactor by significantly improving fuel performance. By preventing FCCI due to the fission product lanthanides, cladding ruptures will be prevented, improving fuel safety and reliability, and higher fuel burn-up will be possible, thus improving reactor economics by decreasing the amount of fuel required, and decreasing the amount of nuclear waste generated.

Accomplishments:

The technical goals of this research are to explore the effectiveness of adding minor additives to a metallic fuel to bind fission product lanthanides, with the end goal of preventing or decreasing FCCI. To this end, both tin and palladium were investigated during fiscal year (FY) 19.

This report covers the evaluation of using fuel additives (Pd and Sn) in U fuel, without Zr. To date, the as-cast and annealed microstructures of USn, USnLn, UPd, and UPdLn have been evaluated (where Ln = 53Nd-25Ce-16Pr-6La, and added to simulate fission product lanthanides). Diffusion couples between Fe and USn, UPd, and UPdLn are complete, and the SEM analysis is finished. The diffusion couple between USnLn and Fe is in progress.

The microstructures of U-Sn and U-Sn-Ln are shown in Figure 1. The microstructure is relatively simple in U-Sn, as shown in Figure 1a, with an intermetallic compound having the composition U2Sn (based on Energy Dispersive X-Ray Spectroscopy (EDS) analysis), and U with a small amount (~2 at. %) dissolved Sn. The intermetallic compound is not known, and requires further investigation. Upon addition of lanthanides, a large precipitate of Sn-Ln was observed, as shown in the bottom of Figure 1b. The dark region is SnLn, while the lighter grey region (shown with red arrows) contains less Sn, but has not been identified yet. Aside from the large precipitate, the microstructure contains U2Sn and smaller precipitates of SnLn.

The microstructures of U-Pd and U-Pd-Ln are shown in Figure 2, along with EDS maps of U and Pd for the U-Pd alloy. Based on the EDS maps and quantitative EDS analysis, UPd3 deposited along the uranium grain boundaries. The only other structures present in the microstructure are square precipitates of UO_2. Upon addition of lanthanides, a large precipitate formed. The dark region on the left side of the image shown in Figure 2b is the 1:1 PdLn intermetallic compound.

The large additive-lanthanide precipitates present in both U-Sn-Ln and U-Pd-Ln are due to the immiscibility of these phases in the U matrix.

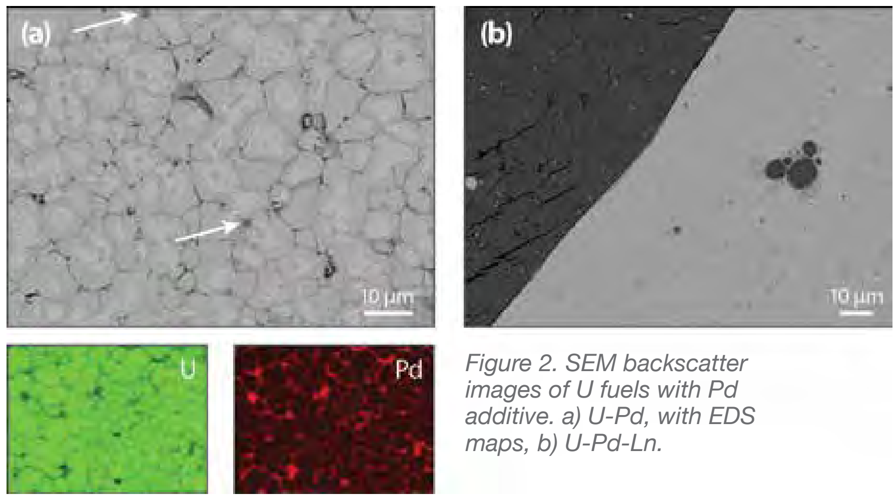


Figure 2. SEM backscatter images of U fuels with Pd additive. a) U-Pd, with EDS maps, b) U-Pd-Ln.

In the corresponding Zr alloys, i.e. U-10Zr-4Sn-4Ln and U-10Zr-3.86Pd-4.3Ln, the additive-lanthanide compounds are dispersed throughout the alloy. Although these compounds are still not miscible in the U-Zr matrix, there may be some miscibility in the molten phase due favorable interactions between the additives and Zr. There is segregation during cooling, but the result is dispersion of the additive-lanthanide compounds throughout the matrix.

Diffusion couples have been run between the alloys and Fe (the couple between U-Sn-Ln had to be repeated and is underway). Analysis of the data is in progress. A manuscript is in progress for both the annealed structures and the diffusion couples.

FAST Irradiation Test Assembly Development, Fuel Fabrication and Trials

Principal Investigators: Randall Fielding and Blair Grover (INL)
Team Members/ Collaborators: Korbin Tritthart, Steve Steffler and Darrell Jonak (INL); Elliott Marsden (INL Intern)

The fuel fabrication package has shown the feasibility of fabricating the FAST rodlets, enabling the number of rodlets in an integral fuel irradiation test to increase dramatically.

The Fission Accelerated Steady State Test (FAST) was designed to greatly increase the burnup rate of fuel, which can lead to an accelerated qualification path. In addition to increased burnup rate, FAST is also designed to greatly increase the number of samples that can be irradiated in a test position. Both of these attributes have been accomplished by reducing the sample size to be irradiated. Decreasing the size of the fuel samples and increasing the total number of samples to be inserted in the reactor increases fabrication difficulties and schedule risks. Work this year has focused on fuel sample casting and rodlet assembly, including loading, welding, and sodium settling and bonding.

Project Description:

Work in this area showed the feasibility of fabrication of the FAST rodlet despite the increased difficulty. Feasibility was shown by developing methods for casting, rodlet loading, closure welding, and finally sodium bonding. Reduced diameter cladding was also fabricated. Mock-up test hardware is shown in Figure 1. In addition to simply developing a process, because the tests call for up to 50 rodlets the process must be consistent and time efficient as this is a very large increase in number of samples. The test design calls for the sodium bonded fuel slugs to be 0.084 in. (2.13 mm) diameter and 0.75 in. (19.05 mm) in length. A fuel casting method is needed to consistently produce these fuels. Many of the test specimens will be sodium bonded. Again, as with casting due to the number of samples and the small size, a consistent manner of loading and eventually bonding the rodlets is needed that can be used for tens of samples in an efficient manner. The cladding walls are designed to be 0.009 in. (0.23 mm) thick and will be welded to a solid endplug. Welding of a thin walled tube to a solid plug is difficult and must be demonstrated before test fabrication can begin. As these steps are demonstrated the feasibility of fabricating the FAST rodlets will also be demonstrated and optimized to allow the FAST test to be fabricated and inserted on schedule. The FAST tests will greatly increase the amount of high burnup data on metallic fuel alloys and because of the increase in the number of samples the tests will be used as a scoping test for advanced reactor fuels.

Figure 1. Mockup hardware for the FAST irradiation test to be used in the ATR outboard A and Small I positions.

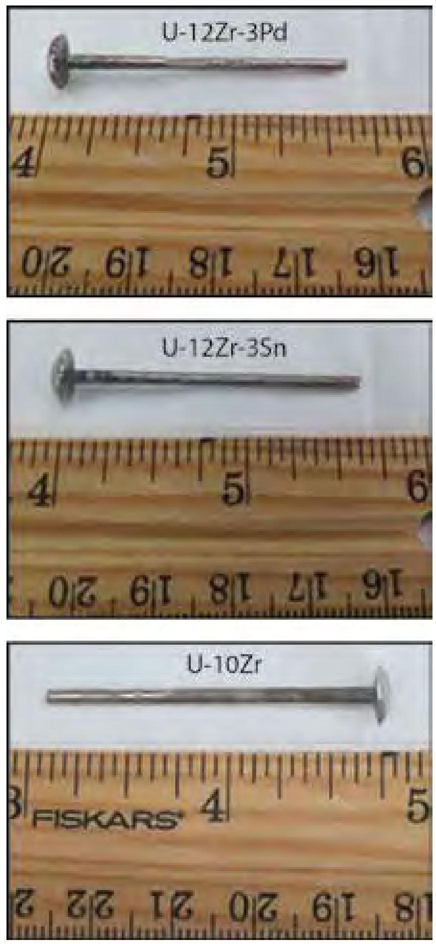


Figure 2. Prototypic FAST fuel slugs casting using the pressure differential assisted gravity casting technique.

Accomplishments:

The first step in showing feasibility of fabricating the FAST test was to cast surrogate fuel slugs in a manner that could consistently produce acceptable slugs. Nearly all of the previous AFC metallic fuel tests were cast using an arc casting technique. In arc casting the fuel is heated by means of an electric arc to a temperature above the melting point, after which the fuel flows into the mold. Although, this method has been used almost extensively for the AFC tests it can be non-consistent during casting and is seen as a large schedule risk. This risk is greatly compounded by the small size of the FAST samples. To mitigate this risk, modifications were made to the arc melting apparatus which will force the molten fuel into the mold

Figure 3. Welding and loading fixture to be used for FAST rodlet fabrication, shown before glovebox installation.

using a pressure differential. This system has been tested with several alloys producing several slugs, as shown in Figure 2.

Following casting, a pulsed gas tungsten arc welding method was developed along with the loading hardware, as shown in Figure 3. Using this method of welding, the thin walled cladding tubes are joined to the solid endplug by melting the entire outer surface of the endplug. The welds zones are small enough that standard weld radiography will likely not be able to be used to detect weld flaws. However, two rodlets from a batch of ten were randomly selected and interrogated using micro-x-ray computed tomography, which showed the welds to be solid with no discernable discontinuities in the weld area. After welding, the surrogate fuel slugs were settled into the sodium and bonded. During fabrication of the previous AFC tests, settling and bonding were performed in batches of 5-6 rodlets, and rodlets were manually lifted and dropped in the furnace many times over the bonding cycle.

Because of the number of FAST tests, manually impacting each rodlet will be a long process and the small size of the rodlets and fuel slug may also decrease the success rate seen during bonding. To determine a more efficient bonding path two separate methods were investigated 1) lateral vibration of the loaded furnace assembly and 2) vertical vibration of the individual rodlets while in the furnace. Lateral vibration was accomplished by setting the furnace, loaded with the rodlets, on a vibration table and intermittently vibrating throughout the hour long bonding process. Vertical vibration of the individual rodlets was done through vibrating each rodlet with a modified vibro-peen tool. After testing both methods much better results were seen by vertically impacting the rodlet intermittently throughout the bonding process. Figure 4 shows surrogate FAST rodlets after bonding. With these results it can be confidently concluded that the FAST rodlets can be fabricated in an efficient manner allowing the total number of irradiation samples to be irradiated to increase substantially.

Figure 4. Completed FAST mockup rodlets.

Assessment of Viability of Scaled Annular Pellet Fabrication Technologies

Principal Investigator: Chris Grote, LANL

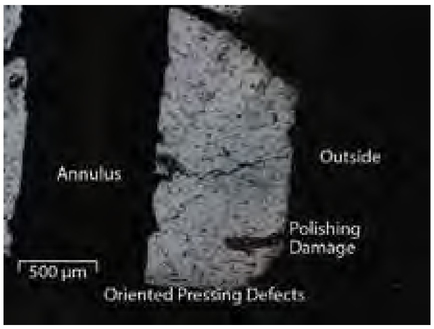


Figure 1. Polished microstructure of 3mm radius annular pellet.

High-density pellets can successfully be manufactured with geometries of interest to accelerated fuel testing and qualification.

Fabrication of annular pellets and/or scaled traditional pellets has potential to support accelerated burn-up testing concepts. To this end, a variety of geometries have been proposed that support the accelerated fuel qualification program within the Department of Energy Nuclear Energy Advanced Fuels Campaign. (DOE-NE AFC) This includes the traditional solid monolithic cylindrical geometries with diameters down to 2 mm, as well as annular pellet designs with sizes ranging from 10 to 3 mm with annular holes that scale with pellet diameter. Work this year has centered on the development of pellets using these unique geometries and is demonstrated for CeO_2, UO_2, and U_3Si_2. Polished microstructures are shown detailing the defects present in specific geometries/designs as well as methods to mitigate these features in future fabrications.

Project Description:

Irradiation testing to high burnup fuel (to ~60G Wd/MTU) requires significant time in current test reactors limiting the ability to effectively model how the fuel will behave when it approaches long duration energy generation. While reaching this point conventionally is achievable in test reactors, it requires a decade to achieve using current irradiation testing schemes. Accelerated burnup would be approaching the threshold of "high burnup" more quickly, on the order of 1-2 years. To achieve this faster rate, the fuel is likely going to be subject to thermal effects commiserate with the increased heat by more fission. In a normal fuel pellet, this increased temperature would operate less efficiently because the thermal diffusivity of the pellet only allows it to transfer heat so quickly, thus causing a high centerline temperature and noticeable stress along the diameter of the pellet. Solutions could include smaller pellet radii or annular designs, which would have a greater surface area giving off heat relative to their volume, or "shaped" fuel designs modifying the thermal behavior.

The concept that drives the annular pellet design is reduction of the thermal gradient caused from the high "centerline" temperature at the center of a normal pellet to its surface. By removing the center of the pellet, the new centerline will instead be closer to an edge (whether that be the inner or outer edge), allowing cooling to wick away more heat than the traditional design. This increase in heat removal would allow the pellet to be subjected to higher energy output, and thus accelerate the burn-up process.

Accomplishments:

In an effort to develop methodology, Ceria was used as a non-radiological material surrogate for both testing the punch and die sets outside of a glovebox testing the viability of fabricating annular pellets on a well-known system. For this reason, only the largest and smallest annular designs were tested.

10 mm w/ 3mm	5mm w/ 1mm	3 mm w/ 1mm	10 mm	5mm	3mm	2mm
CeO_2	91.54	90.57	91.173	96.00	96.37	CeO_2
UO_2	93.83	88.49	88.8	93.82	85.47	UO_2

Table 1. Densities of annular and small radii pellets fabricated at Los Alamos National Laboratory (LANL) Fuel Research Lab.

The geometric densities for the sintered pellets were 90% dense, and showed a 14-15% diametric shrinkage on the outer diameter after sintering. The large pellets have ~10% shrinkage on the inner diameter, with the smaller pellets showing up to 20% shrinkage. Since nuclear material is of interest, these preliminary studies show a level of success that indicate the recipe will work in a glovebox with materials of interest.

Uranium dioxide is the industry standard for fuel and will be a necessary benchmarking fuel for the accelerated burnup testing. Historic knowledge on solid pellets meant prioritizing the annular pellets, hence the 10 mm and 3 mm pellet checks for UO_2. For these tests, depleted Uranium was used because of the reduced hazards associated with it, but will behave the same for when higher enrichments are used later for the actual irradiation testing. The annular pellets ranged from 87.5-93.9% dense which is well within successful parameters for an exploratory study without specific process improvements. Additionally, comparison of the 3mm annular pellets to ~3mm monolithic pellets gives strong credence to the idea that annular pellets can actually achieve higher densities, perhaps due to the relief from the annulus surface. Compared to pellets previously made at the Fuel Research Lab (FRL), the annulus increased the density of pellets by an average of 4%.

For the end goal of helping with qualification, the microstructure effects of the annulus are also important. New pressing defects are expected due to new interaction surfaces, particularly the surface of the annulus with the die. Not only does the increased contact area/volume ratio increase, but the relaxation and sintering processes might change as well, since they have an extra surface to precipitate new defect behavior. A 3 mm w/1mm annulus pellet was examined and found to contain some noticeable pressing defects. The annulus of the pellet left a smooth bore, which is the geometry we would ideally want for using pellets. On the other hand, there are clear defects presented as pores that run perpendicular to the direction of pressing. These are likely caused from either a lack of pressure that caused incomplete compressing of the powder due to a low pressure during initial testing.

Mechanical Testing of High Dose Irradiated Materials for Fast Reactor Applications

Principal Investigator: Benjamin P. Eftink (LANL)
Team Members/ Collaborators: Matthew E. Quintana, Tobias J. Romero, Paul L. Caccamise, Tarik A. Saleh and Stuart A. Maloy (LANL); Cheng Xu (TerraPower, LLC); Gary Was (University of Michigan)

Figure 1. Image of hot cell at Los Alamos National Laboratory (LANL) in CMR Wing 9 where mechanical testing occurred.

Ferritic/martensitic (FM) steels and nanostructured ferritic alloys (NFA) are being developed for next generation high dose reactor applications. Applications such as transmutation of long lived isotopes in used fuels require cladding materials that can withstand high irradiation doses, potentially in the hundreds of displacements per atom (dpa), at intermediate to elevated temperatures and in contact with a liquid metal coolant. Changes in the mechanical properties occur after such harsh conditions. Mechanical testing as part of this project probes changes in mechanical properties due to neutron irradiation of fast reactor relevant alloys.

Project Description:

The main technical objectives of the project include: i) determining the changes to mechanical properties of advanced ferritic steels (such as oxide dispersion strengthened (ODS) steels 14YWT and MA957) to high dose irradiations ii) determining the changes to mechanical properties of traditional ferritic/martensitic steels (such as HT-9 or T-91) to high dose irradiations iii) understanding the effect of irradiation temperature on irradiation

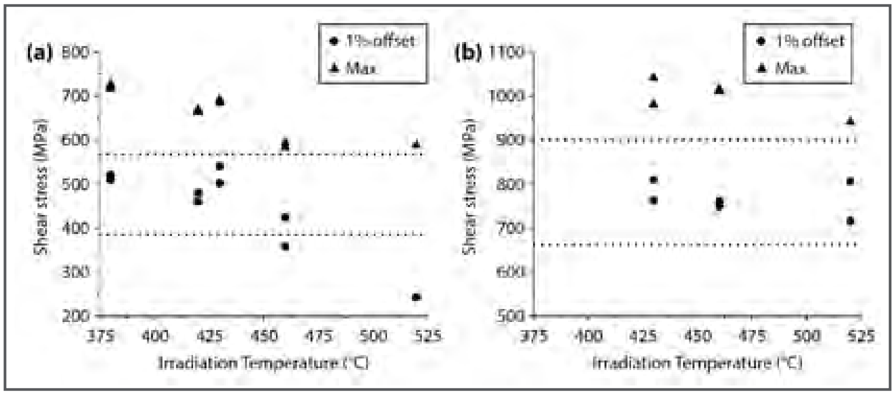


Figure 2. Comparison of 1 % offset effective shear stress and maximum effective shear stress at doses between 15 and 20 dpa for a) HT-9 and b) 14YWT as a function of irradiation temperature. Dotted lines mark the average max and 1 % offset shear strain for the unirradiated control samples tested at room temperature.

induced hardening and iv) evaluating mechanical testing techniques for small material volumes.

All of these technical objectives directly support the development of safe, reliable and economic next generation reactors, particularly sodium cooled fast reactors. The ferritic steels investigated are relevant fuel cladding and in-core structural materials, based on the prior performance of traditional HT-9 in the experimental breeder (EBR)-II fast reactor and the promise of better irradiation tolerance of advanced ferritic alloys including ODS 14YWT.

Cladding materials for Generation IV fast reactors, such as the sodium fast reactor (SFR), will be subjected to irradiation damage levels in the hundreds of dpa and temperatures of 400-600 °C, which is significantly higher than the requirements for current generation light water reactors (LWRs). Doses expected in SFR at lower temperatures (e.g., 350-400 °C) are lower but strong hardening is observed at those temperatures. Thus, data is needed to characterize mechanical properties after low temperature irradiation also. Understanding the effect of both irradiation dose and irradiation temperature on the evolution of mechanical properties of structural materials is critical to ensure the safe and reliable operation of next generation nuclear reactors.

Evolution of mechanical behaviors of structural materials at high neutron irradiation doses is complex, and understanding it is critical to the safe implementation of advanced reactor designs.

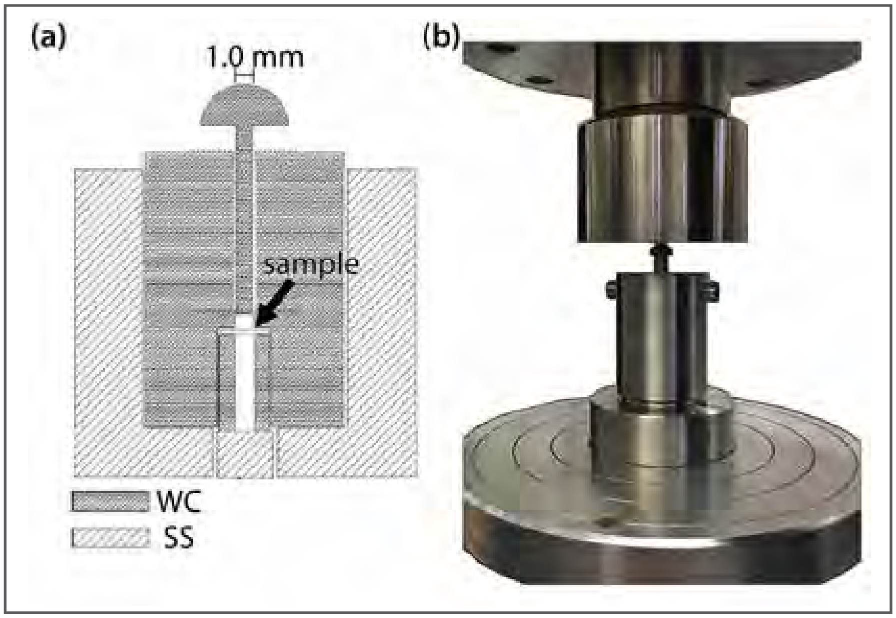


Figure 3. a) Schematic of Shear Punch Test fixture for 3 mm x 0.25 mm Transmission Electron Microscopy (TEM) samples using a 1 mm punch. WC and SS are tungsten carbide and stainless steel, respectively. b) Image of shear punch fixture.

Accomplishments:

The main technical objectives of the project include: i) determining the changes to mechanical properties of advanced ferritic steels (such as oxide dispersion strengthened (ODS) steels 14YWT and MA957) to high dose irradiations ii) determining the changes to mechanical properties of traditional ferritic/martensitic (F/M) steels (such as HT-9 or T-91) to high dose irradiations iii) understanding the effect of irradiation temperature on irradiation induced hardening and iv) evaluating mechanical testing techniques for small material volumes.

Two main thrusts have contributed to progress toward the technical objectives listed above. First, traditional F/M steel HT-9 and advanced ODS 14YWT that were irradiated at the BOR-60 reactor (an SFR, at the RIAR laboratory in Dimitrovgrad, Russia) were mechanically tested using shear punch. The samples were irradiated to doses ranging from 15 to 35 dpa and temperatures between 380 and 520 °C. These irradiation conditions are relevant to next generation sodium cooled fast reactors. Shear punch testing was conducted at the Wing 9 hot cells at the CMR building at LANL, Figure 1 shows the mechanical testing setup inside the hot cell. Response was material dependent, with the traditional HT-9 exhibiting greater irradiation hardening than the advanced 14YWT at the irradiation temperatures below 520 °C. The even dispersion of oxides in the advanced ODS 14YWT is believed to reduce the irradiation hardening in that alloy by acting as recombination sites for irradiation induced vacancies and interstitials. Additionally, the advanced ODS 14YWT alloy was impacted less by irradiation temperature, Figure 2 shows the impact of irradiation temperature on irradiation hardening. These results contributed to technical objectives i)-iii). It was also shown that shear punch testing

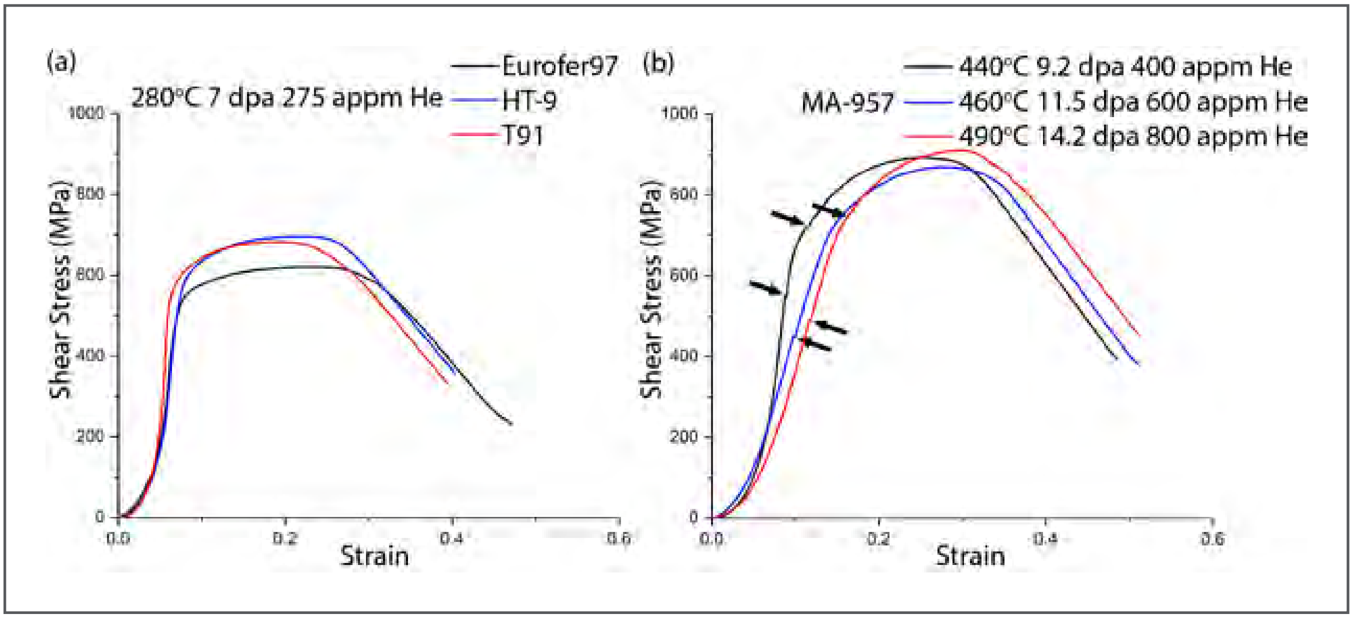


Figure 4. Shear punch stress vs strain. a) Curves for Eurofer97, HT-9 and T91 irradiated at 280 °C to a dose of 7 dpa and He accumulation of 275 appm. b) Curves for MA-957, irradiation temperatures, doses and He accumulation presented on Figure. Note the serrations in the curves for MA-957 which correspond to fracture events on the sample surfaces, marked by arrows.

is an effective mechanical testing technique for neutron irradiated steels, supporting technical objective iv). Figure 3 shows the setup of the shear punch testing.

Written accomplishments include the completion of a milestone "Report and perform shear punch testing of BOR60 irradiated HT-9 and 14YWT", as well as the submission of a manuscript "Shear punch testing of neutron irradiated HT-9 and 14YWT" to the journal JOM.

For the second main thrust, shear punch mechanical testing was performed to compare neutron irradiated reduced activation steel Eurofer97, and traditional F/M steels HT-9 and T91. Additionally, the response of advanced ODS MA-957 as a function of dose and He production was investigated. What was specifically accomplished was determining the reduced activation steel and the traditional F/M steels exhibited similar stress-strain responses at irradiations of 7 dpa at 280 °C, and the advanced MA-957 irradiated at temperatures between 440 and 490 °C to 9 and 14 dpa with corresponding He accumulations of 400 and 800 appm exhibited similar stress-strain responses, observed in the stress-strain curves in Figure 4. Fracture of the shear punch surfaces of the MA-957 reveals that despite high irradiation temperatures, which reduces irradiation hardening, He accumulation and sample texture may contribute to unacceptable fracture even in advanced ferritic alloys. This second thrust also directly contributes to technical objectives i)-iv)

The second main thrust resulted in the written accomplishment of the milestone report "Report and perform fast reactor materials testing from STIP V irradiation".

High Fidelity Simulation of High Dose Neutron Irradiation

Principal Investigator: Gary S. Was, University of Michigan
Team Members/ Collaborators: Steven Zinkle, Brian Wirth (University of Tennessee); Arthur Motta (Pennsylvania State University); Todd Allen, Fei Gao, Emmanuelle Marquis, Zhijie Jiao (University of Michigan); Stuart Maloy (Los Alamos National Laboratory)

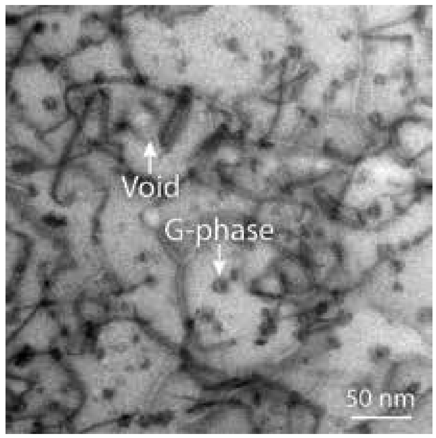


Figure1. Annular bright field image of HT9 irradiated with dual ions (Fe and He) at 460 °C to 16.6 dpa. Electron beam is parallel to <311> zone axis. (He Li)

The SNAP program is developing the capability to simulate reactor irradiated microstructures and mechanical properties to high dose, paving the way for predicting microstructure and properties to high dose to support the development of advanced reactor materials.

Ion irradiation coupled with the absence of residual radioactivity are the attributes that make this route to advancing our understanding of radiation effects so attractive. The lack of activation means that samples can be handled as if they were unirradiated, eliminating the need for the extremely high investment in time and cost connected with the use of hot cells and dedicated characterization instrumentation. In total, high damage rates mean that 20 years in reactor can be achieved in 2 days in an accelerator meaning that ion irradiation is 10-1000x less costly and 10-100x quicker than test reactor irradiation.

Project Description:

The objective of this proposal is to demonstrate the capability to predict the evolution of microstructure and properties of structural materials in-reactor and at high doses, using ion irradiation as a surrogate for reactor irradiations. "Properties" includes both physical properties (irradiated microstructure) and the mechanical properties of the material. Demonstration of the capability to predict properties has two components. One is ion irradiation of a set of alloys to yield an irradiated microstructure and corresponding mechanical behavior that are substantially the same as results from neutron exposure in the appropriate reactor environment. Second is the capability to predict the irradiated microstructure and corresponding mechanical behavior on the basis of improved models, validated against both ion and reactor irradiations and verified against ion irradiations. Taken together, achievement of these objectives will yield an enhanced capability for simulating the behavior of materials in reactor irradiations.

The program consists of four major elements, or thrusts: 1) establishment of the capability to conduct dual- and triple- ion irradiations that capture the key elements of the BOR-60 reactor neutron spectrum and development of both ion and reactor irradiation programs, 2) a description (both experimental and computational) of the evolution of the irradiated microstructure over a wide dose range relevant to fast and thermal

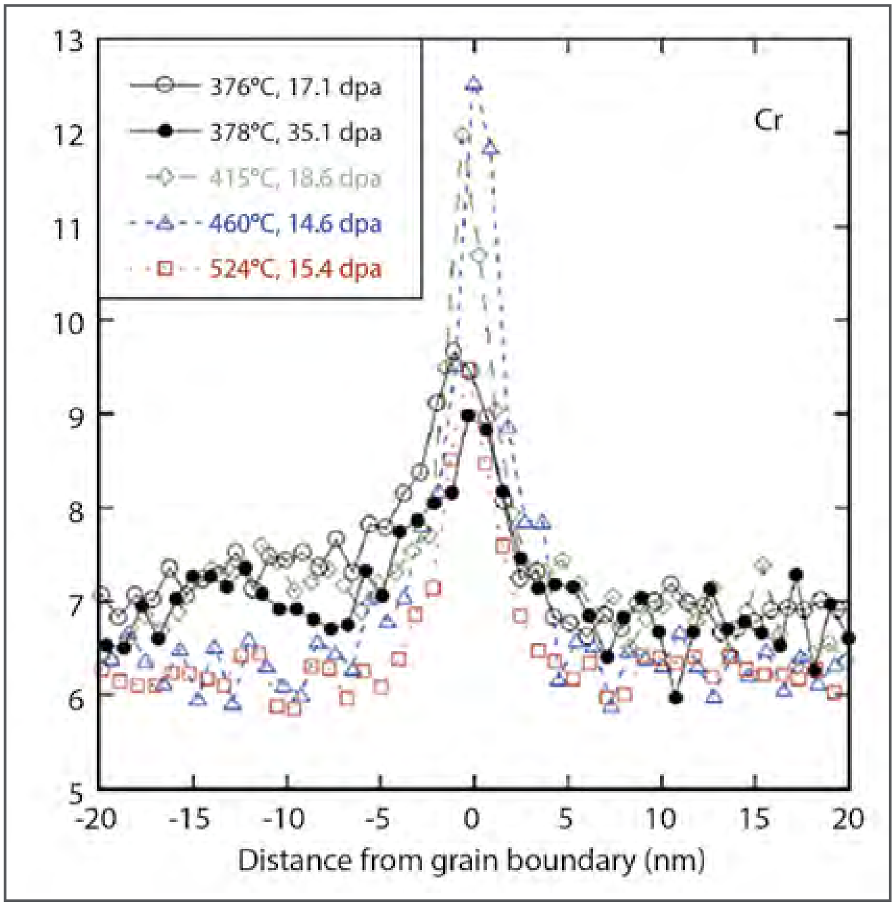


Figure 2. Radiation-induced segregation of chromium at a grain boundary in T91 irradiated in the BOR-60 reactor under different irradiation conditions. (Zhijie Jiao)

reactors, 3) establishment of the microstructure – property relationship for irradiated materials, and 4) engaging the worldwide radiation effects community through creation of workshops and working groups to address ion irradiation techniques and irradiated sample preparation and analysis. Cross-cutting activities through these major thrusts consist of A) ion/neutron irradiation of the same alloys/heats, B) incorporation of both damage and transmutation effects, and C) meshing of experiment and modeling efforts across all aspects of the program.

If successful, this program will establish the capability to predict the microstructure and mechanical property changes in reactor compo-

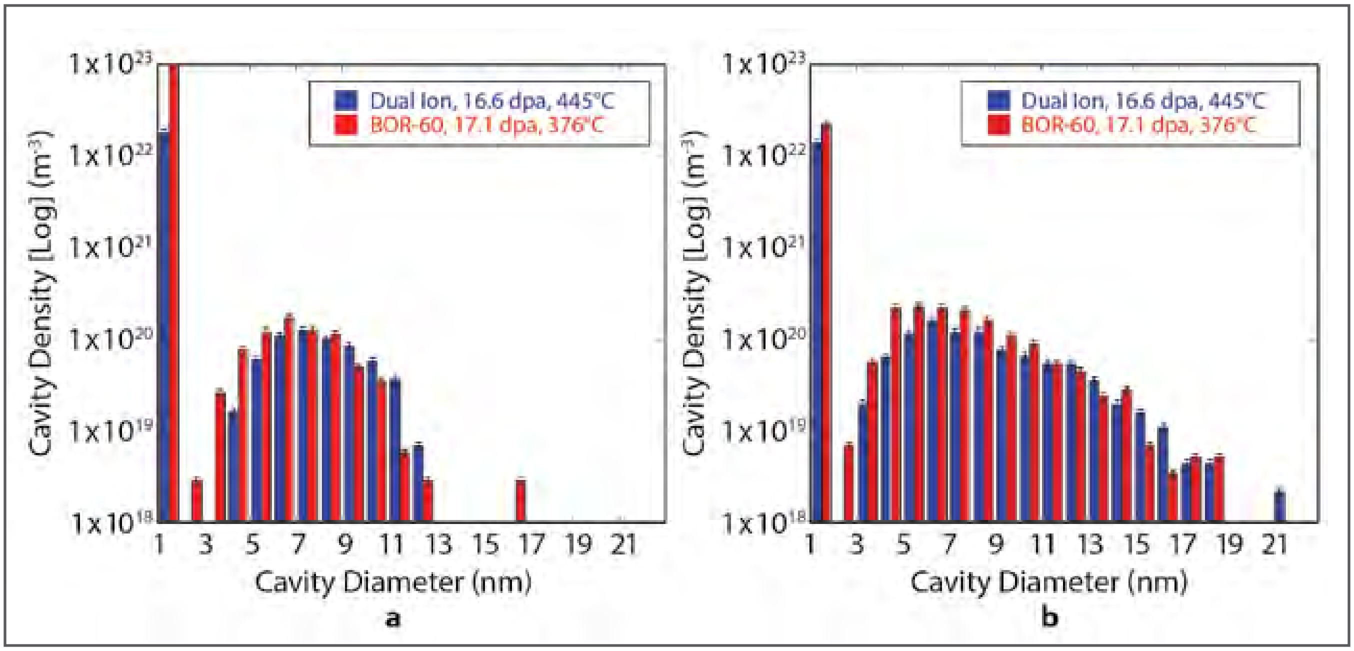


Figure3. Comparison of the cavity size distribution at dpa in dual ion irradiated T91 with 4 appm He/dpa and BOR-60 irradiated T91. (Stephen Taller)

nents, providing an important tool for utilities in predicting component degradation, and providing the basis for development of new alloys that improve the safety, reliability and economics of reactors in the current fleet and also advanced reactors.

Accomplishments:

The microstructure of alloy T9 irradiated in the BOR-60 reactor and with dual ion (DI) beams at 360°C was characterized to establish the optimum ion irradiation conditions for ion irradiation to produce microstructures that match those from reactor irradiation at nominally 360°C. The actual reactor irradiation temperature was 378°C. Dislocation loops, cavities, radiation-induced precipitates and radiation-induced segregation observed in BOR60 irradiated T91 were also observed in DI-irradiated samples. A temperature shift of 70°C (to 445°C) for DI irradiation was found to produce comparable dislocation loop and cavity microstructures with BOR60 irradiation. The bimodal distribution of cavities was also accurately captured by DI irradiation. Segregation of Cr to grain boundaries and formation of Ni-Si rich clusters was insignificant in the DI irradiated T91 compared to the BOR60 irradiated T91. The enrichment of Cr at the grain boundary in DI irradiated T91 is much lower than that observed in BOR60 irradiated

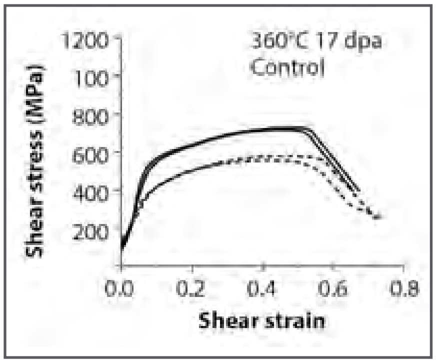


Figure 4. Shear stress vs. shear strain for 14YWT irradiated in the BOR-60 reactor to 17b dpa at 380°C. (Ben Eftink)

samples. The size of Ni/Si-rich clusters following ion irradiation was comparable to that from the BOR-60 irradiation and the density was found to be 10-100x lower in DI irradiated T91 compared to BOR60 irradiated T91.

A cluster dynamics model was developed to predict microstructural evolution in high-energy particle irradiated ferritic martensitic (F/M) steels, in which we have recently implemented a model for the biased absorption of self-interstitial defects at small, nanometer-sized cavities that can provide an intrinsic nucleation barrier for the onset of void swelling. The model also includes the influence of helium content on modifying the intrinsic cavity bias. Incorporation of cavity bias was effective in creating a nucleation period for void swelling that is commonly observed in experiment.

The indentation size effect that occurs during nanoindentation (NI) testing has been examined on annealed pure Fe and Fe-Cr binary alloys under constant strain rate conditions. In parallel, the effect of matrix strength on the indentation size effect has been examined by performing NI tests on binary Fe-18Cr specimens that were thermally aged at 475C for 100 to 900 h in order to produce a wide range of matrix hardening associated with Cr-rich alpha prime precipitates of varying precipitate size and density. These fundamental NI "single-effects" data are essential to reduce the quantitative uncertainty when analyzing the hardening contributions from dislocation loops, cavities and precipitates in neutron and ion irradiated 9-12Cr steels. To complement NI, shear punch tests were conducted on neutron irradiated advanced reactor relevant materials from the BOR-60 reactor. These tests provide a comparison between conventional HT-9 and advanced 14YWT at high doses up to 35 dpa and temperatures between 380 and 520 °C which are relevant conditions for advanced fast reactors. The results brought insights particularly to the impact of irradiation temperature, which is less pronounced in the advanced 14YWT alloy.

Advanced Fast Reactor Cladding Development

Principal Investigator: David T. Hoelzer (ORNL)
Team Members/ Collaborators: Caleb P. Massey and Rachel L. Seibert (ORNL)

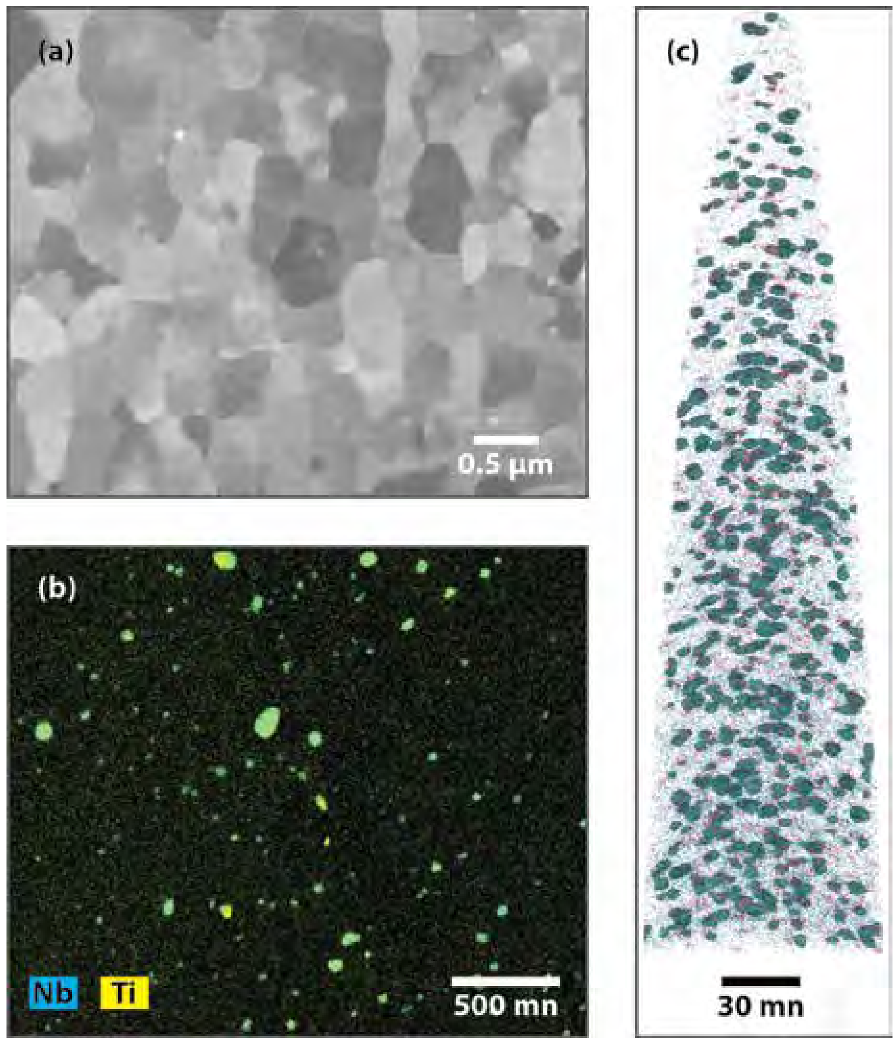


Figure 1. Microstructural features of Oak Ridge Fast Reactor Advanced Fuel Cladding (OFRAC) a) Back-scattered Electron (BSE) Scanning Electron Microscopy (SEM) micrograph of the ultra-fine grains, b) Scanning Transmission Electron Microscopy (STEM) Energy Dispersive X-Ray Spectroscopy (EDS) map of Nb and Ti overlaid to show the Nb(C,N) particles and c) LEAP element map of the Ti-Y-O nano-size oxide particles.

The Nuclear Technology Research and Development (NTRD) program is investigating advanced nuclear fuel cladding that will enable the development of new fuel for irradiating to very high burnup (e.g., >40%) in a fast neutron flux. The cladding must withstand mechanical property degradation due to swelling and creep deformation upon exposure to very high neutron doses of >200 displacements per atom (dpa) at

The development of advanced oxide dispersion strengthened (ODS) ferritic alloys such as 14YWT and OFRAC offer many advantages over conventional steels for fuel cladding in sodium cooled fast reactors such as high temperature creep strength and improved tolerance to high dose neutron irradiation environments.

high temperatures while maintaining structural integrity in contact with the fuel and coolant. The cladding material must meet this challenging environment while possessing the ability to be fabricated into thin wall tubing and joined by welding.

Project Description:

The leading candidate for high dose environments of sodium-cooled fast reactors is 12Cr-1MoVW (HT-9) ferritic/martensitic steel (F/M steel) due to its resistance to irradiation induced swelling and creep up to 550°C. For higher thermal efficiency of fast reactors, advanced materials are required that can perform at higher irradiation temperatures. Advanced radiation tolerant materials containing a dispersion of nano-size oxide particles are being developed to meet the higher irradiation temperatures and enable higher fuel burnup. Although the high strength properties and radiation tolerance of the oxide dispersion alloys are attractive, these alloys also present greater challenges in fabricating useful parts such as thin wall tubes. Thus, in this project, research is focused on improving the high temperature strength of HT-9 and on development and fabrication of oxide dispersion strengthened alloys, such as 14YWT and a new alloy referred to as OFRAC into thin wall tubes.

Accomplishments:

Building upon extensive research and development of the nanostructured ferritic alloy (NFA) 14YWT starting in 2001 at ORNL, a new NFA referred to as OFRAC was developed for advanced fuel cladding in sodium fast reactors (SFR) starting in 2016. The composition of OFRAC is Fe-12Cr-1Mo-0.3Ti-0.3Nb-0.3Y2O3 (wt.%). The purpose for selecting this composition was to promote partial phase transformation of the microstructure to form grains of the γ-fcc (face centered cubic) austenite phase at high temperatures between ~910°C and ~1125°C from the matrix consisting of α-bcc (body centered cubic) ferrite phase during

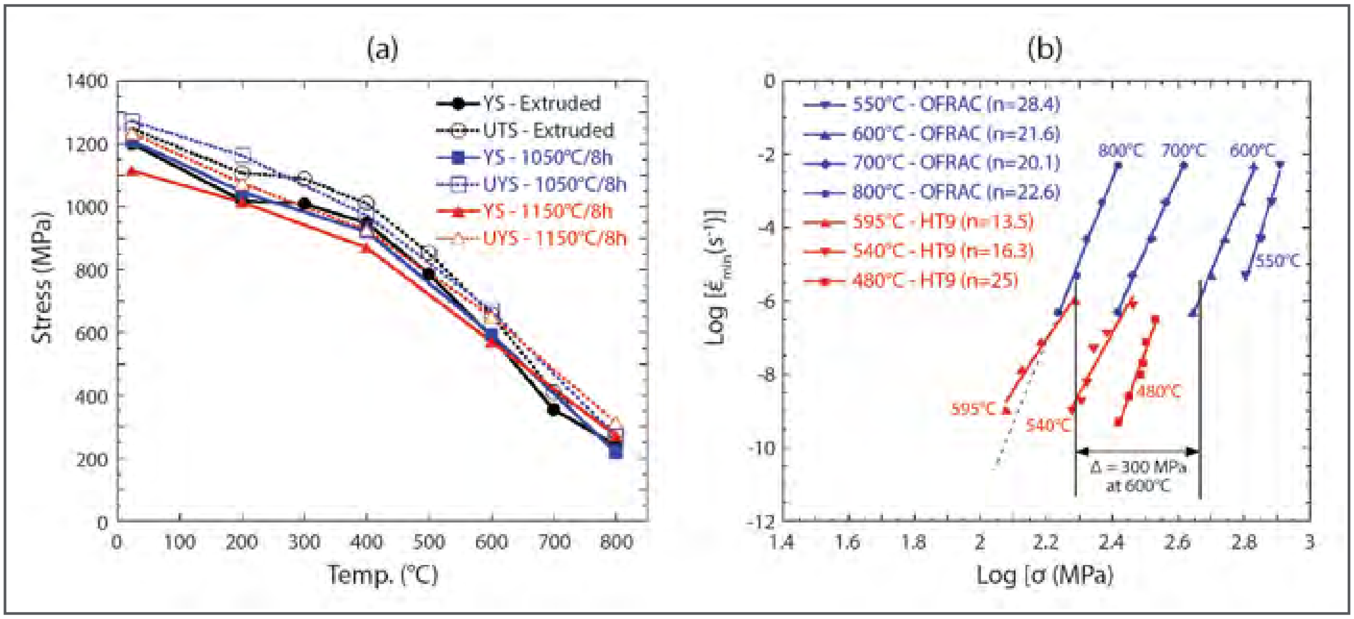


Figure 2. Mechanical properties of OFRAC. (a) comparing the yield and ultimate tensile stresses of OFRAC from 25ºC to 800ºC in as-extruded condition and after annealing for 8 hours at 1050ºC and 1150ºC and (b) comparing the creep properties of HT-9 with OFRAC, which was obtained from strain rate jump tests.

thermo-mechanical processing and to sequester interstitial C and N introduded as contamination during ball milling by the addition of Nb to form Nb(C,N) particles (FCC: Fm¯("3")m space group). Titanium is added to form nano-size Ti-Y-O oxide particles that are characteristic of 14YWT. Since the mechanical alloying conditions used for producing 14YWT were the same for producing OFRAC, the microstructure of OFRAC consistes of ultra-fine grains containing a high concentration of nano-size Ti-Y-O oxide particles and a dispersion of coarser Nb(C,N) particles (Figure 1).

The tensile and creep properties were evaluated to investigate the performance of OFRAC at high temperatures (Figure 2). A highlight of the tensile tests conducted over the temperature range of 25°C to 800°C was that annealing samples of OFRAC for 8 h at 1050°C and 1150°C, which is ~0.70 and 0.77 of the melting point, had little effect on the strength and ductility properties (Figure 2a), indicating the microstructure is remarkably stable at high temperatures. The creep properties of OFRAC were obtained from strain rate jump tests at temperatures ranging from 550°C to 800°C and were compared to data for HT-9 (Figure 2b), which is a 12Cr tempered martensitic steel that was developed for fuel cladding and other components in fast reactors. The comparison shows that creep performance of OFRAC is superior to HT-9 since the test temperature required to induce a given creep rate at the same applied stress was ~300 MPa higher at 600°C.

A significant achievement was fabrication of a thin wall tube of OFRAC by pilger-rolling at room temperature in collaboration with Nippon Nuclear Fuel Development Co., Ltd (NFD), Japan (Figure 3). No cracks formed in the OFRAC tube after 84% reduction in wall thickness and 790% increase in length. The final OFRAC tube was 1,780 mm long, 8.51 mm outside diameter and 0.51 mm wall thickness, indicating cold-pilger is a viable method for producing tubing.

Figure 3. Optical images of the OFRAC thin wall tube and cross-section of the tube (inset)

Advanced Fast Reactor Cladding Development

Principal Investigator: Stuart Maloy, LANL
Team Members/Collaborators: Ben Eftink, Jonathan Gigax, Hyosim Kim and Yongqiang Wang (LANL) S Byun (PNNL) Lin Shao and Frank Garner (TAMU)

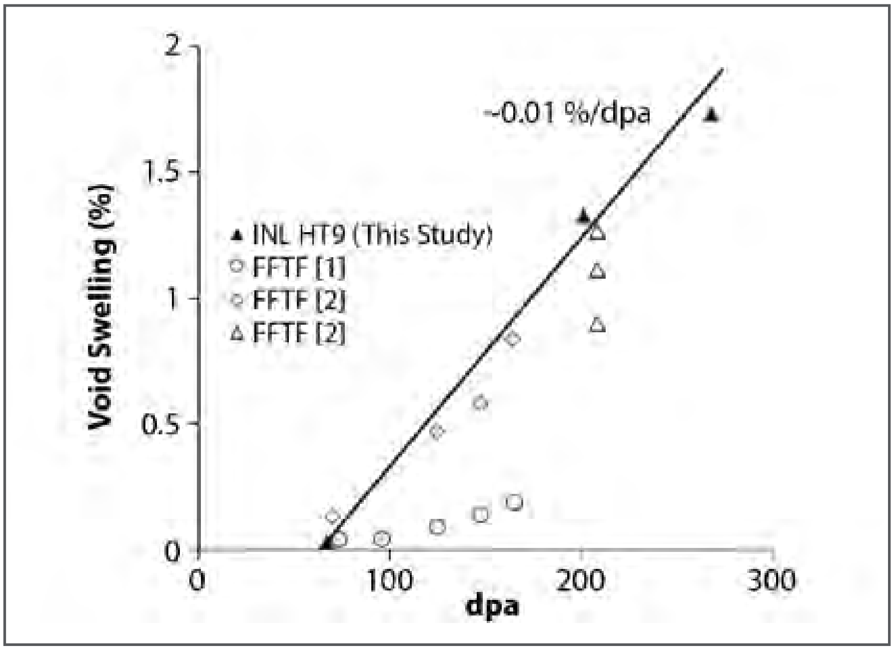


Figure 1. Comparison of heavy-ion induced void swelling in INL HT9 (this study) and void swelling from irradiation in the Fast Flux Test Facility (FFTF).

The Nuclear Technology Research and Development (NTRD) program is investigating advanced nuclear fuel cladding that will enable the development of new fuel for irradiating to very high burnup (e.g., >40%) in a fast neutron flux. The cladding must withstand mechanical property degradation due to swelling and creep deformation upon exposure to very high neutron doses of >200 dpa at high temperatures while maintaining structural integrity in contact with the fuel and coolant. The cladding material must meet this challenging environment while possessing the ability to be fabricated into thin wall tubing and joined by welding.

Project Description:

The leading candidate for high dose environments of sodium-cooled fast reactors is 12Cr-1MoVW (HT-9) ferritic/martensitic steel (F/M steel) due to its resistance to irradiation induced swelling and creep up to 550°C. For higher thermal efficiency of fast reactors, advanced materials are required that can perform at higher irradiation temperatures. Advanced radiation tolerant materials containing a dispersion of nano-size oxide particles are being developed to meet the higher irradiation temperatures and enable higher fuel burnup. Although the high strength properties and radiation tolerance of the oxide dispersion alloys are attractive, these alloys also present greater challenges in fabricating useful parts such as thin wall tubes. Thus, in this project, research is focused on improving the high temperature strength of HT-9 developing and testing improved radiation tolerant heats of HT-9.

Accomplishments:

Significant accomplishments have been achieved related to testing the radiation tolerance of new heats of HT-9 and performing

Excellent radiation tolerance was observed in new heats of HT-9 through high dose ion irradiations to doses greater than 200 dpa.

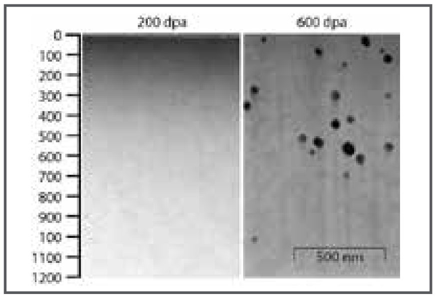


Figure 2. Void swelling in delta ferrite grains after irradiation to 200 and 600 peak dpa.

thermomechanical treatments on new heats of HT-9. Ion irradiations were performed using Fe ions in the ion irradiation facilities at Texas A & M to doses of 800 dpa at 450C. Data shows that the Idaho National Laboratory (INL) heat of HT-9 shows swelling less than 2% at doses up to 200 dpa (see Figure 1). In addition, microstructural analyses after irradiation show that more swelling was observed in delta ferrite areas compared to tempered martensitic regions (see Figure 2). In addition, improved heat treatments were developed for HT-9. By normalizing and then tempering at different temperatures, materials showing excellent toughness were developed which exceeded the toughness measured on older heats of HT-9 (see Figure 3).

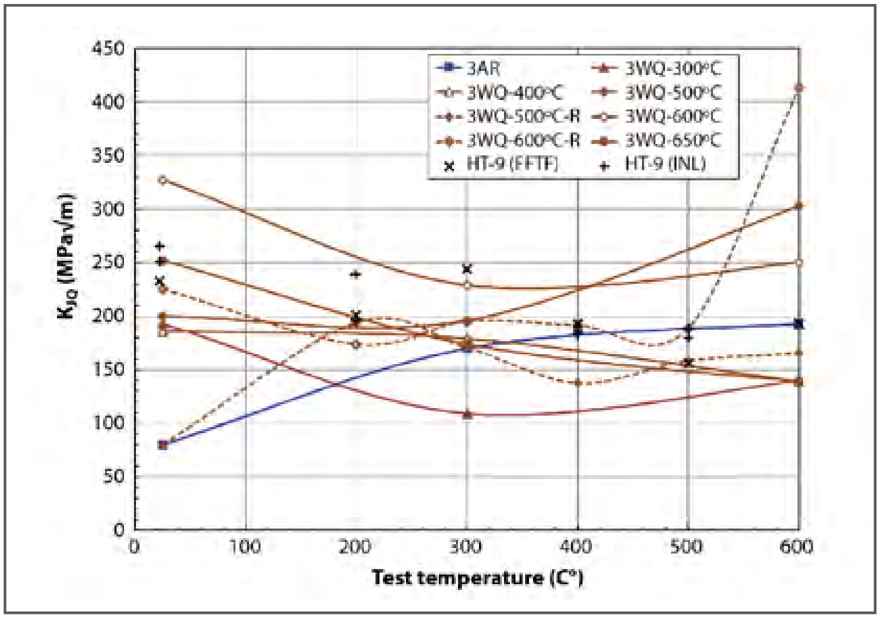


Figure 3. Fracture toughness of 12 Cr steel (HT-9 steel) heat-4 (N-added) after various thermomechanical treatments, whose final conditions include (a) as-rolled (AR), water-quenched (WQ), and single-tempered (1 h), and (b) double-tempered conditions. Note that R indicates repeated testing and S the second tempering was for 15 minutes instead of typical 30 minutes.

Advanced Reactor Tube Processing Development

Principal Investigators: Marie McCoy and Curt Lavender (PNNL)
Team Members/Collaborators: Ron Omberg, Mark Smith and Mike Dahl (PNNL); Dr. Stu Malloy (LANL)

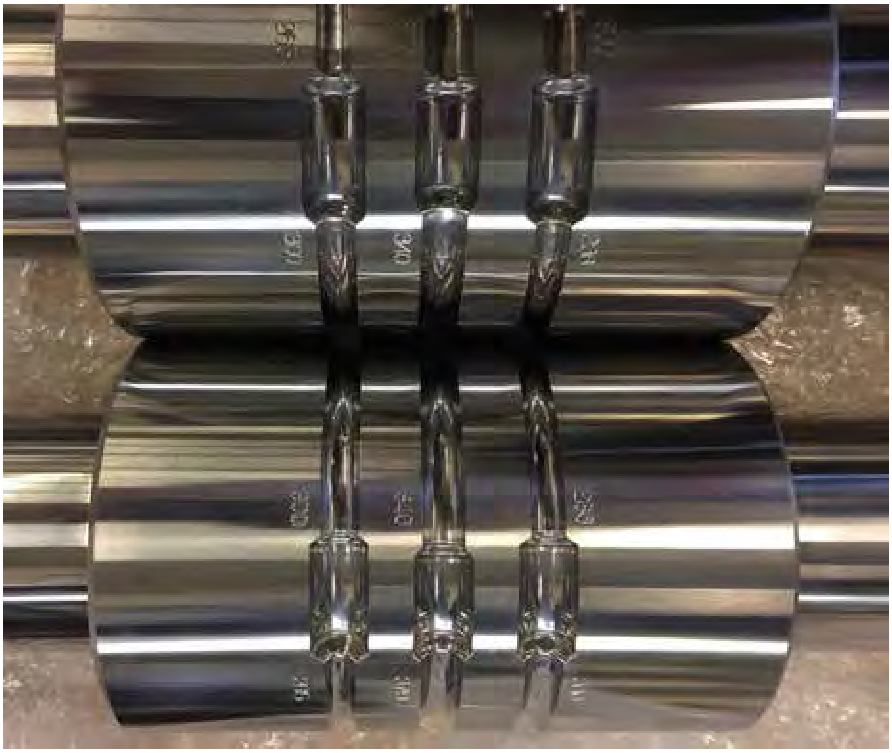


Figure 1. As machined roller dies used for pilgering in roller mill.

The main goal of this activity is to perform the research and development as needed to maximize nuclear fuel utilization while minimizing high level waste. In both the modified open and closed fuel cycle options, these goals are accomplished through irradiating nuclear fuel and cladding to a very high burnup in a fast neutron flux.

Projection Description:

One of the most promising alloys to provide the radiation tolerance required at high burnups is 14YWT. This alloy has a fine dispersion of oxide particles that have the advantage of serving as sinks for defects and thereby achieving very high burnups, but for which these finely dispersed oxide particles also make the processing of this alloy into engineering forms very difficult. The objective of this project is to perform the research and development (R&D) required to processes this alloy into engineering forms by extruding it into billets and then forming these billets into engineering tubes using the pilgering process.

Accomplishments:

A rolling mill at the Pacific Northwest National Laboratory (PNNL) was modified so that it could accomplish the pilgering process and thereby form thick-walled billets into thin-walled tubes. This required modifying the rolling mill by designing, fabricating, and installing roller dies and mandrels. The roller dies are shown in Figure 1. Subsequent to this, it was necessary to establish the basic pilgering parameters such

Figure 2. 14YWT starting thick-wall tube and thin-wall tube after pilgering.

as feed rates, rotation rates, and reduction schedules per pass, as applicable to the successful forming of 14YWT thick-walled tubes into thin-walled tubes. These parameters were developed by pilgering successfully in sequence stainless steel, and oxide dispersion alloy MA956, and then the objective alloy 14YWT. Two thin-wall tubes of 14YWT were successfully pilgered this fiscal year and these are shown in Figures 2 and 3. Characterization of the finished tubes was performed in a joint collaborative effort between PNNL and Dr. Stu Malloy at the LANL.

Achieving the objectives noted above, that is developing alloys that can attain very high burnups in a fast neutron flux and thereby maximize nuclear fuel utilization and minimize nuclear waste, requires an alloy that can perform its function in a radiation environment at doses greater than 400 displacements per atom (dpa) which promote low temperature embrittlement, radiation induced segregation, high temperature helium embrittlement, swelling and accelerated creep, and chemical interaction with the fuel. 14YWT is such an alloy and the ability to form it into engineering shapes is being developed.

Figure 3.Thin-wall 14YWT tube after pilgering – length 8.13 Inches.

Fracture Toughness of 9Cr and 12Cr Steels After New thermomechanical Treatments for Fast Reactor Cladding Application

Principal Investigator: Thak Sang Byun, PNNL
Team Members/ Collaborators: David A. Collins, Timothy G. Lach, Jung Pyung Choi, and Emily L. Barkley (PNNL)

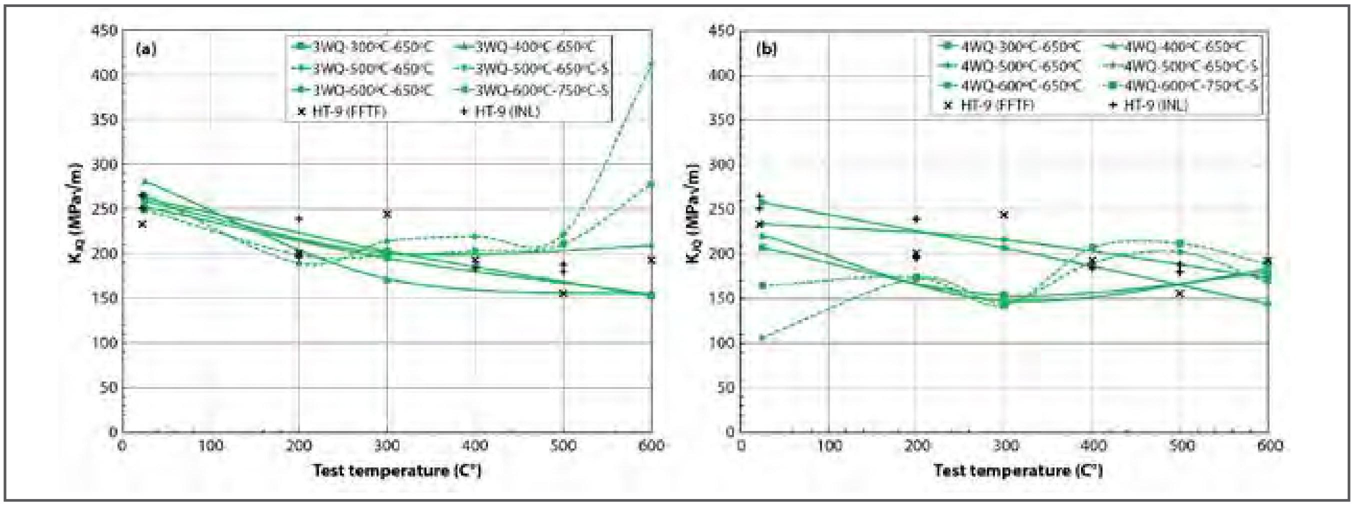
High-Cr steels with fine ferritic/martensitic (FM) structures are candidate materials for fast reactor core structures because of their high creep strength and excellent resistance to high temperature radiation damage such as void swelling. After the traditional thermomechanical treatment consisting of normalizing, water quenching, and tempering, however, the high-Cr steels show limited mechanical properties, e.g., significant decrease of strength above ~450°C. Since the high-Cr steels are tempered at high temperatures (typically around 750°C), significant coarsening can occur during the final heat treatment. It is believed that the high temperature tempering can maximize ductility but reduce high temperature strength and fracture toughness. It is also likely that such a fully-tempered structure will experience a lowered radiation resistance when compared to the steels with finer microstructures. This research was planned to explore new processing routes that can produce ultrafine microstructures and thus can improve the high temperature mechanical properties of the high-Cr steels.

Project Description:

This research aimed to improve the mechanical properties of high-Cr steels by developing new thermomechanical processes for application to advanced fast reactors. A reactor core structure with an improved high temperature capability will be essential for the development of a high-performance reactor with higher thermal efficiency. In this project a simultaneous improvement of fracture toughness and strength is specifically targeted in the design of thermomechanical processing for various 9Cr and 12Cr steels. Another principle that has guided the process development task is the common knowledge that containing more defect recombination sites or higher defect sink strength is required for higher radiation resistance. Three 9Cr steels and two 12Cr steels with and without nitrogen addition were selected for this research. A series of new thermomechanical process treatments (TMTs) were designed and applied to those five steels to find a new TMT route that can result in the increase of both strength and fracture toughness. It was demonstrated for the HT-9 steels that some tailored TMTs increased or maintained the fracture toughness of the high-Cr steels while some of those treatments could significantly increase the strength of the steels to the higher levels comparable to those of nanostructured ferritic alloys. The fracture toughness data of both 9Cr and 12Cr steels after various thermomechanical treatments were analyzed and compared with those of their reference materials. A two-step tempering route is selected for application to the high-Cr steels for reactor core structures.

Accomplishments:

This research aimed to find a new processing route that can produce an optimized microstructure of each high-Cr steel with improved mechanical properties including excellent high temperature fracture toughness. It was considered that (i) a highly controlled treatment, such as a rapid cooling, which can effectively refine the quenched lath structure, and (ii) the following tempering process, should be feasible for the practical processing of thin core components such as fuel cladding and fuel duct. In order to determine the best processing route for the high-Cr steels, the main task of the year was focused to thoroughly evaluate their fracture toughness behavior after a variety of thermomechanical treatments. For the testing campaign, twelve TMTs were applied to two 12Cr alloys and a select of five TMT routes to three 9Cr alloys. Fracture toughness tests were carried out over a wide temperature range of 22–600ºC, and the results were compared with reference data for the 9Cr and 12Cr steels. Summarized here are the key observations and conclusions from the comparison of fracture toughness data among the alloys and with the reference materials, which aimed to answer to the question whether any of the new TMTs can improve the fracture resistance of the high-Cr steels. First, the two final TMTs with two-step temping, 500°C/1h + 650°C/15mi and 600°C/1h + 750°C/15min, consistently delivered high fracture toughness with the 12Cr (HT-9) steels



over the whole test temperature range, as presented in Figure 1. It is a notable result that the HT-9 heat-3 without nitrogen addition after two-step tempering treatments are able to maintain their fracture toughness over 200 MPa√m in the high-temperature range of >200°C. When compared to those of Fast Flux Test Facility (FFTF) and Idaho National Laboratory (INL) HT-9 steels, the fracture toughness of the HT-9 steel heat-3 actually improved with those TMT routes while that of the HT-9 steel heat-4 did not show any improvement. The single-step tempering at 600°C also resulted in high fracture toughness for both HT-9 steels. Considering the benefit of a higher strength that can be obtained with a lower degree of tempering, it is recommended that the HT-9 steels are tempered at 600°C with or without a short additional tempering at 750°C. Second, similar fracture resistance behavior was observed in the three 9Cr steels. Both of the two-step temperings after water quenching, 500°C/1h + 650°C/15min and 600°C/1h + 750°C/15min, yielded high fracture toughness with the three 9Cr steel compositions. Third, addition of nitrogen or other modification in composition did not improve the fracture resistance of the high-Cr steels. That is, the addition of nitrogen (~0.044%) to the original HT-9 composition is actually detrimental to the fracture resistance of the alloy. Similarly, the 9Cr steel-3 contains relatively smaller amount of molybdenum (0.5% instead of ~1.5% for the other two 9Cr alloys) and additional ~0.21% vanadium and highest amount of nitrogen (0.045%), but this compositional modification, from tradition 9Cr-1Mo steels, actually reduced fracture toughness of the alloy, in particular, at low temperatures.

Figure 1. Fracture toughness of 12 Cr steel (HT-9 steel) heat-3 and heat-4 after various thermomechanical treatments consisting of water-quenching (WQ) and two-step tempering. Note that S indicates the second tempering was for 15 minutes instead of typical 30 minutes.

It is possible to improve the high-temperature mechanical properties of high-Cr steels by simply changing the final thermomechanical schedule, without increasing any manufacturing cost.

3.3 IRRADIATION TESTING AND POSTIRRADIATION EXAMINATIONS

Fission Accelerated Steady-state Testing (FAST)

Principal Investigator: Geoffrey Beausoleil, INL
Team Members/ Collaborators: Nate Oldham, Bryon Curnutt, Kyle Gagnon, Chris Murdock, and Randy Fielding (INL)

The AFC-FAST experiment is providing the campaign with high burnup fast reactor fuel in a fraction of the time previously available with 50 fueled specimens being irradiated up to 5 %FIMA in a single ATR cycle.

In an effort to accelerate the irradiation time for advanced reactor fuels, a revised capsule design was analyzed and developed at Idaho National Laboratory (INL) for the Advanced Fuels Campaign (AFC). This design incorporates a highly enriched, reduced scale fuel pin that is doubly encapsulated by two steel capsules. This design alloys accelerated irradiations and reduced sensitivity to fabrication variances and eccentricities. The capsule designs utilize existing experiment designs from the AFC capsules in the Advanced Test Reactor (ATR) outer A position (FAST-OA) and the ATF-1 capsules in the small I positions (FAST-SI).

Project Description:

Previous advanced reactor fuels, namely fast reactor fuels, have been successfully irradiated within the ATR. However, these tests have been problematic in that they require adjustments to the experiment design to make the power profile more prototypic (introduction of a cadmium shroud), a tight-tolerance helium bond between the cladding and the capsule wall (a high-heat flux zone for increasing the cladding temperature to prototypic temperatures), high sensitivity to fabrication variances and eccentricities (lower limit of fabrication tolerances still have a significant effect on the outcome of the experiments), and ultimately very long irradiation times in order to achieve high burnup targets. These challenges have been addressed by the FAST experiment design.

The FAST experiments will contain metallic fuel for sodium fast reactors (SFR) and will be inserted into the ATR during the 168B cycle (tentatively March 2020). The experiments will include up to fifty different fuel specimens that are designed to provide data for fuel performance model validation and irradiated material for postirradiation (PIE) testing (e.g., furnace testing for creep). The test matrix encompasses a large array of fuel concepts including:

- Control specimens for comparison to historical irradiations
- Fuel additives and liners for mitigating fuel-cladding chemical interactions (FCCI) at high burnup
- Variable smear densities for studying unrestrained fuel swelling
- Sodium free annular fuel for once-through fuel systems
- Varying fuel alloy specimens (U, U-Pu, and U-Pu-Zr) with liners to test increased heavy metal loading
- Very low burnup for evaluating early microstructure changes and comparing accelerated irradiation effects

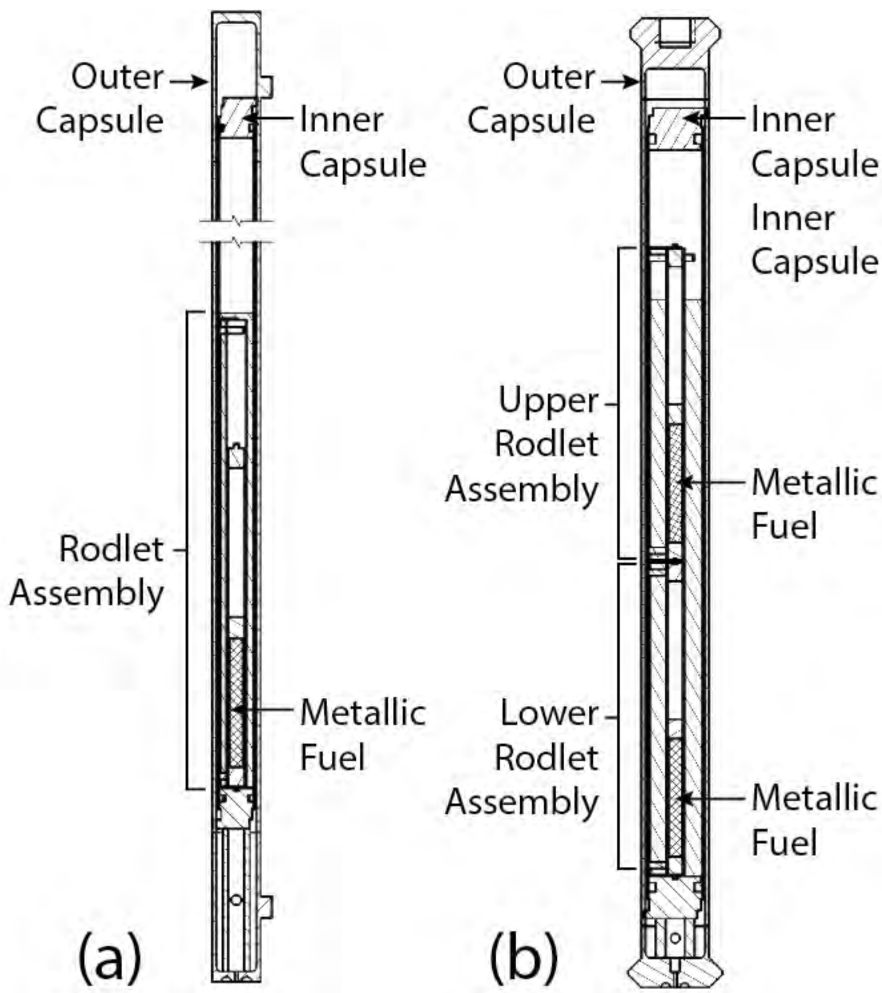


Figure 1. Schematic of the FAST capsule design. a) The FAST-OA capsule with the single rodlet per capsule design. b) The FAST-SI capsule with the double rodlet per capsule design.

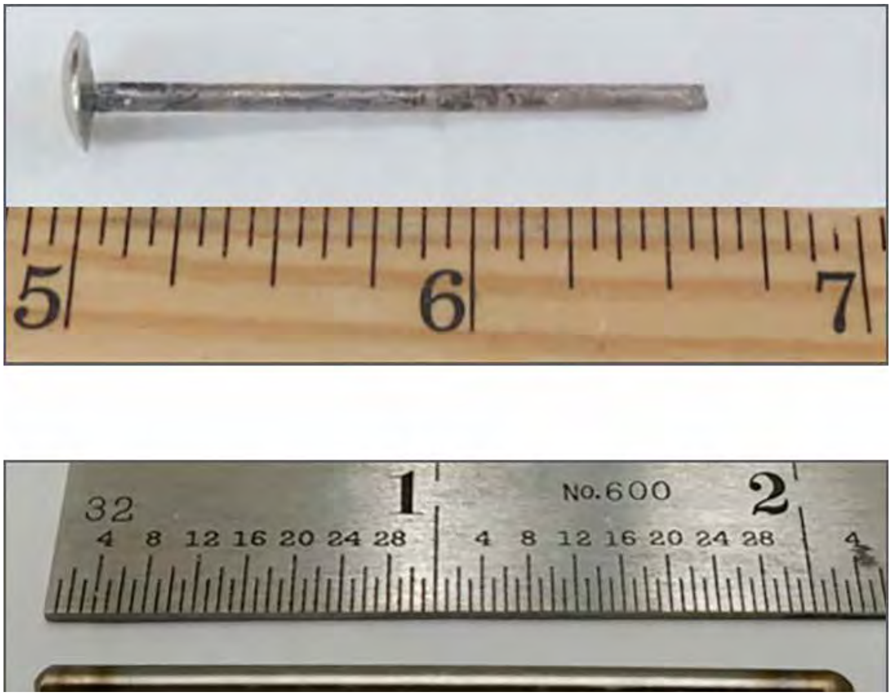


Figure 2. Top image shows an as cast mock-up pin of the fuel. Fuel pins will be cast to this dimension and then machined to the desired dimension/geometry.

to high burnup targets

This project will provide INL an accelerated irradiation platform that can be easily adopted to multiple fuel forms and reactor designs. The benefits will enable the laboratory and Department of Energy (DOE) to meet industry needs for quickly investigating advanced fuel concepts and achieving the objectives of deploying safe, reliable, and economic operation of next generation reactors.

Accomplishments:

Previous work (INL/EXT-18-45933) provided scoping analyses of the FAST concept. Those results demonstrated the feasibility of reducing the size of the fuel pin and cladding while maintaining near-prototypic conditions. From the scoping analysis the team proceeded into this year with the goals of performing preliminary and final design analyses, technical design, and mockup fabrication of the FAST concept. The thermal analysis was performed (G. Beausoleil, INL) using finite element software to investigate the fuel designs of interest, the impact of fabrication tolerances, the consequence of eccentricities and component offsets, and fill gas variations (He/Ar mixtures). The thermal analysis provided important insight into the thermal gradients within the fuel specimens, which can significantly affect fuel performance phenomena.

Mechanical analysis was also performed (K. Gagnon, INL) using finite element analysis software. While the capsules being used in FAST have been previously analyzed for their use in ATF-1 and AFC experiments, there was a slight adjustment incorporated into the design so that pressurized gas can be used in a high-heat flux gap between capsules. This modification required mechanical analysis in order to ensure that the capsules still met the requirements of the ATR Safety Analysis Report.

Neutronic analysis was performed (B. Curnutt, INL) using MCNP-ORIGEN software on generic experiment concepts to determine power profiles for the fuel and non-fissile heat generation rates for components within the capsule. These values were used to provide enrichments to the fuel fabrication team and to verify that the heat loads of the capsule are within the safety limits of the original capsule designs.

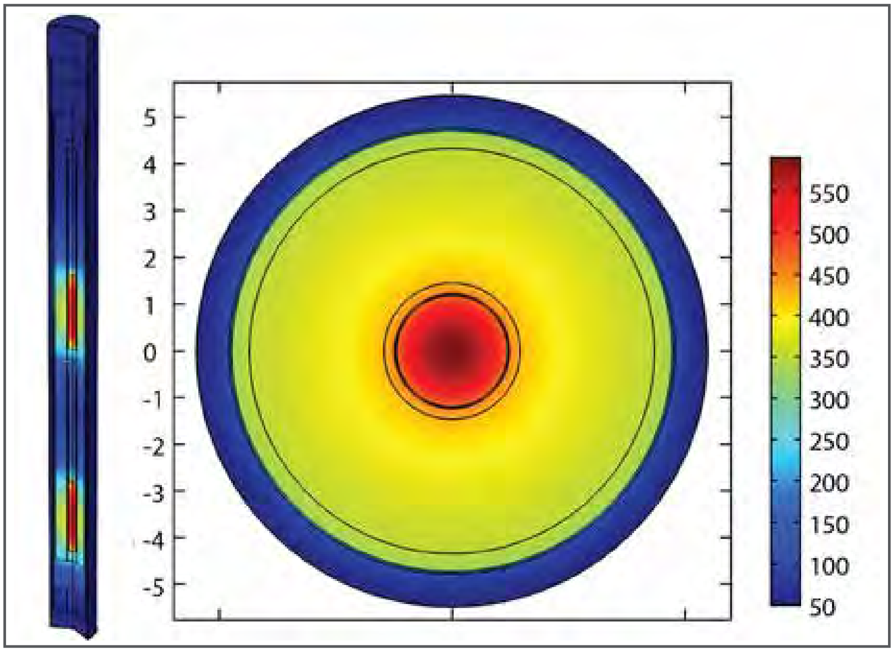


Figure 3. Thermal analysis of the FAST capsule in the small I position.

Preliminary design drawings were issued with only a minor variation to the original capsules (N. Oldham, INL). The variation was the incorporation of a weep hole in the capsule to facilitate pressurized gases in the gas gap. The pressurization simplified the thermal modelling efforts and, as shown by mechanical analyses (K. Gagnon, INL), had no impact on the structural integrity of the capsules.

Sodium interactions were evaluated for the experiments in light of a proposed accident scenario from the ATR experiment safety group due to the presence of large quantities of bond sodium and a possible failure of the capsule within ATR coolant water. It was shown that the capsules, upon failure, would be able to sustain the pressures generated by a water-sodium reaction with impeded egress of the produced hydrogen.

It was also identified that, due to the reduced size of the fuel specimens, the capsules would be able to accommodate two rodlets stacked axially within the capsule. This enabled a doubling of total number of experiments able to be performed with no adverse effect on the thermal and neutronic analysis results. In total, 50 different fuel specimens will be irradiated in the first insertion. This has significant implications to the scope and breadth of the project.

Progress and Accomplishments of AFC Experiments in ATR

Principal Investigator: Chris Murdock, Nicolas Woolstenhulme and Colby Jensen, INL
Team Members/ Collaborators: Randy Fielding, Chris Murdock, Dan Chapman and Emily Swain (INL)

The near-term focus for these irradiation tests is to develop the data necessary to select the most promising design(s), enable irradiation of lead test assemblies in the Versatile Test Reactor, and pave the way for industrial use of advanced metal fuels with improved benefits for utilization economy and disposition pathways.

The Advanced Fuels Campaign (AFC) series of irradiation tests have long been underway at the Advanced Test Reactor (ATR). These irradiations have addressed research and development (R&D) needs for many of the key benefit areas of advanced fuels in fast spectrum reactors including actinide transmutation, fuel utilization through ultra-high burnup fuels, and disposition-facilitating direct disposal once-through fuel systems. The AFC tests simulate sodium fast reactor (SFR) test conditions in a thermal spectrum test reactor by placing fuel rodlets in capsules with engineered gas gaps to elevate specimen temperature while surrounding the capsules with cadmium thermal neutron filters to achieve representative fast reactor neutronic conditions. Typically, irradiation has been performed in ATR's outboard A positions where proximity to driver fuel favors fast neutron populations. This approach has been compared to historic tests performed in genuine sodium fast reactors with fairly good results.

Project Description:

The AFC irradiations have been divided into 4 major campaigns to date (AFC-1 through -4) with several incremental sub-campaigns denoted by a letter (e.g., AFC-3A, -3B). Depending on desired burnup, some rodlets can take several years to complete irradiation. As a result, these experiment sub-campaigns typically represent a new batch of rodlets produced and shuffled into the irradiation positions duration ATR's refueling outages and occasionally causing main campaigns to overlap slightly in the irradiation schedule. These new specimen batches have often corresponded with new fabrication developments, sharpened research focus areas resulting from previous test postirradiation exams (PIE), and other refinements based on input from academic, industrial, and international partners. The AFC-1 and -2 irradiations are now complete and successfully addressed

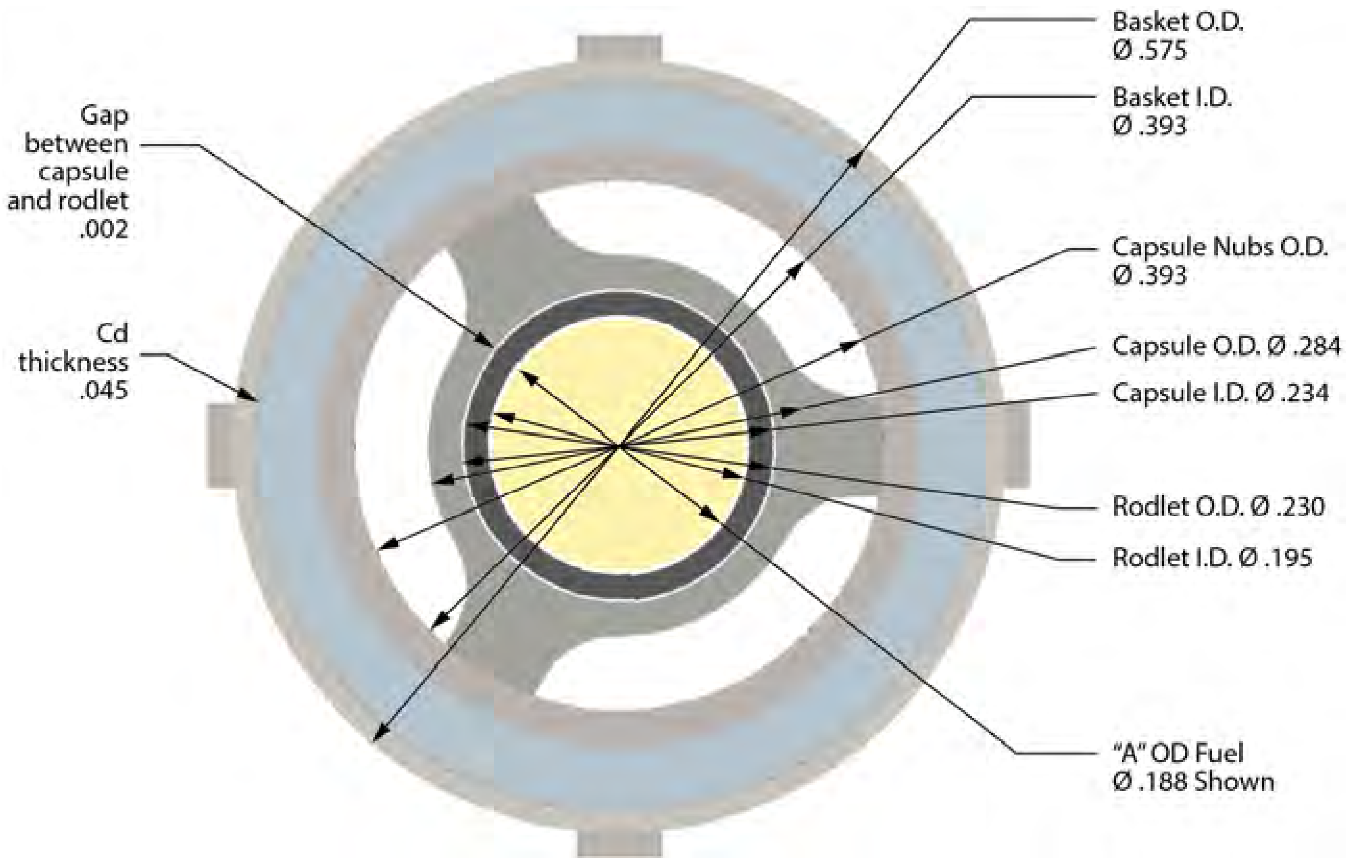


Figure 1. Typical cross section of AFC-series irradiation tests.

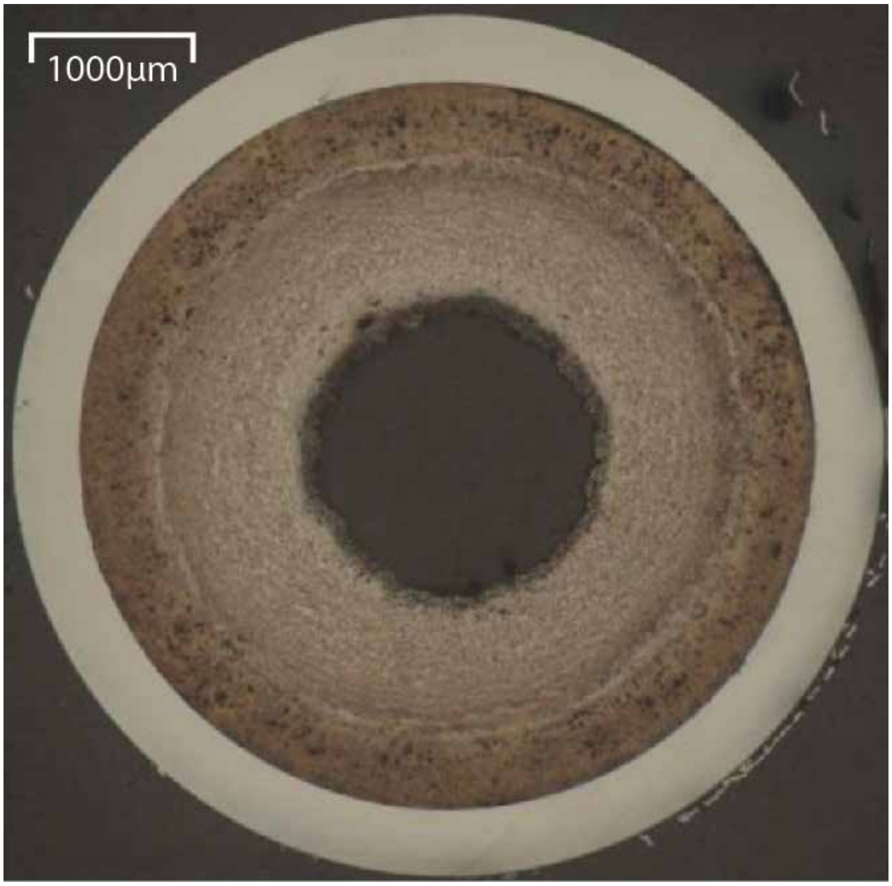


Figure 2. PIE image of low burnup annular fuel specimens from AFC-3D.

research needs for several fuel compositions including both metallic and ceramic fuels types. One of the major accomplishments of the AFC-1 and -2 campaign showed the relative insensitivity of fuel performance when adding actinides to metal fuels for the purpose of transmuting them and greatly reducing radiotoxicity of spent fuels.

The AFC-3 and -4 campaigns have focused on advanced metallic fuel concepts focusing largely on two objectives including fuel design modifications able to 1) perform well without fuel-to-cladding bond sodium (thus enabling direct disposal fuel cycles) and 2) mitigate the life-limiting phenomena associated with fuel cladding chemical interaction (FCCI). The former of these objectives has focused development on the use of annular fuel geometries in close contact with the helium-filled cladding where the latter of these objectives has focused developments both on the use of FCCI-mitigating cladding internal liners and the use of fuel alloy additives which tend to react and form stable intermetallic with lanthanides. Most of the AFC-3 specimens, and some of the AFC-4 specimens, have already competed their planned irradiation. Presently the AFC-3F, -4B, -4C, and -4D sub-campaigns are undergoing irradiation. Another irradiation campaign is also underway in concert with the Korean Atomic Energy Research Institute (KAERI) via the joint fuel cycle study cooperative research and development agreement. This campaign is the first phase of the Integral Recycling Test (AFC-IRT1) irradiation experiment purposed to exhibit fuel performance of metal fuels which were produced in part from recycled spent oxide fuels. All the IRT specimens are non-annular slugs, sodium bonded to the cladding, and contain a significant fraction of transuranic elements. Some of the IRT specimens also have FCCI-mitigating cladding liners.

Accomplishments:

During fiscal year (FY) 2019 the AFC irradiation team oversaw the continued irradiation of AFC-3F, -4B, -4C, -4D, and -IRT1. Due to some extended maintenance outages and two "PALM" cycles (where the AFC experiments are extracted from the core due to very high power conditions), two complete ATR standard cycles were complete accumulating ~120 effective full power days on AFC tests in FY19. Supporting continued irradiation of these experiments requires vigilance in tracking as-run irradiation conditions and projecting future operations both for safety compliance reasons, and to inform programmatic decisions such as reconfiguration of capsules within the irradiation assembly to leverage ATR's axial flux gradient in achieving desired fuel temperature conditions. In one case these efforts resulted in a simple capsule reconfiguration strategy that will be employed in coming cycles to ensure that the AFC-IRT1 test stays within target temperature ranges while accommodating emergent plans for lobe power increases to support other fuels programs.

The AFC irradiation team also collaborated with the advanced fuel performance modeling and fuel fabrication teams in generating a preliminary test matrix for the upcoming AFC-4F and -G tests. The matrix is not yet finalized, but the team achieved adequate maturity to help guide near term developments in fuel fabrication capabilities. The AFC-4F and -G tests will address employ U-Zr and U-Pu-Zr alloys with promising FCCI-mitigating liners, strategic inclusion of rare earth elements to help accelerate FCCI effects thus supporting a hastened timeline for final liner type selection, and the use of annular extruded and slotted fuel designs to help enable direct disposal sodium-free designs. These efforts are synergistically planned to leverage, support, and augment the concurrent FAST-1 irradiation test. Co-planning of both AFC and FAST tests will enable comparisons to validate the veracity of and identify artifacts resulting from the novel burnup acceleration strategy planned for FAST-1. Similarly, the complimentary nature of these tests will enable AFC test series to focus more specifically on geometry and scale phenomena at representative radial dimensions.

Experiments for In-Situ Microstructural Evolution

Principal Investigators: Nicolas Woolstenhulme and Colby Jensen (INL)
Team Members/ Collaborators: C. Baker, R. Schley, C. Adkins, L. Sudderth, C. Hill, D. Chapman, S. Snow and H. Guymon (INL)

This project has successfully integrated the objectives of two DOE programs to develop tools necessary to perform temperature controlled irradiations in TREAT with advanced instrumentation capable of characterizing microstructural changes for fundamental irradiation behavior of metal fuels.

The Transient Reactor Test facility (TREAT) at Idaho National Laboratory (INL) was constructed in the late 1950's, provided thousands of transient irradiations before being placed in standby in 1994, and was restarted in 2017 in order to resume its crucial role in nuclear-heated safety research. Advances in modern computational capabilities and a resurgence of interest in novel reactor technology have created an opportunity for emphasizing modernized science-based and separate effects test capabilities at TREAT. A high temperature heater module, an instrumented multi-specimen capsule train, and in-pile application of laser resonant ultrasound spectroscopy were developed for TREAT's modular irradiation vehicle and demonstrated in fiscal year (FY)19 to enable continuation of this type of research on metal fuels.

Project Description:

TREAT is typically thought of as a reactor best used for simulating postulated accidents with short duration extreme power maneuvers. However, TREAT's ease of access for in-situ instrumentation and collocation with fuel fabrication and postirradiation examination (PIE) facilities also make it uniquely capable for some niche fundamental irradiation science applications not typically associated with fuel safety research. TREAT is not well-suited to burnup or neutron damage accumulation, but examples of these types of applications can make use of oscillating power histories for phase-based properties measurements and in-situ characterization of fundamental irradiation behavioral changes under active fission. An innovative design approach to enable this type of testing leverages the relatively low radioisotope accumulation during brief irradiations by arranging small fresh fuel specimens in low activation hardware "modules" so that they can be easily extracted and shipped for examination within weeks. This concept, termed the Minimal Activation Retrievable Capsule Holder (MARCH) irradiation vehicle system, was developed and deployed culminating in completion of TREAT's first fueled irradiation series in early 2018, the first fueled experiments since its shutdown in 1994. These first tests were employed for pulse-type transient heating of light water reactor rodlets as foundational work for accidental tolerant fuel (ATF) development but focused on engineering scale performance rather than enabling in-situ microstructure-based irradiation science. Further work was performed in 2019 under he Advanced Fuel Campaign (AFC's) advanced reactor metal fuels effort as well as Department of Energy's (DOE) newly formed in-pile instrumentation (I2) program to address this capability gap.

Accomplishments:
Engineers from INL's irradiation experiment design and supporting groups developed three major systems in 2018 with pertinence to microstructural-focused experimentation in TREAT. All of these systems make use of the MARCH system for its modular mechanical layout. The first development was a heater module which can be installed in the MARCH pressure pipe to enable electrical heat up to 700°C prior to irradiation, or to create longer duration temperature-controlled histories during irradiation. The heater module mounts on an upper ring which enables smaller test modules to pass through. One heater module was constructed and used successfully in the irradiation discussed below. A second heater module is under fabrication with optimized thermocouple placement to enable full temperature operations.

The second major development was supported by the I2 program for an in-pile demonstration of a novel sensor applying resonant ultrasonic spectroscopy – laser (RUSL) to infer changes in material microstructure. This method uses laser ultrasonics for measuring changes in the elastic stiffness tensor caused by grain restructuring (recrystallization). In this arrangement a fiber delivers an amplitude modulated laser to the base of the cantilever beam which excites

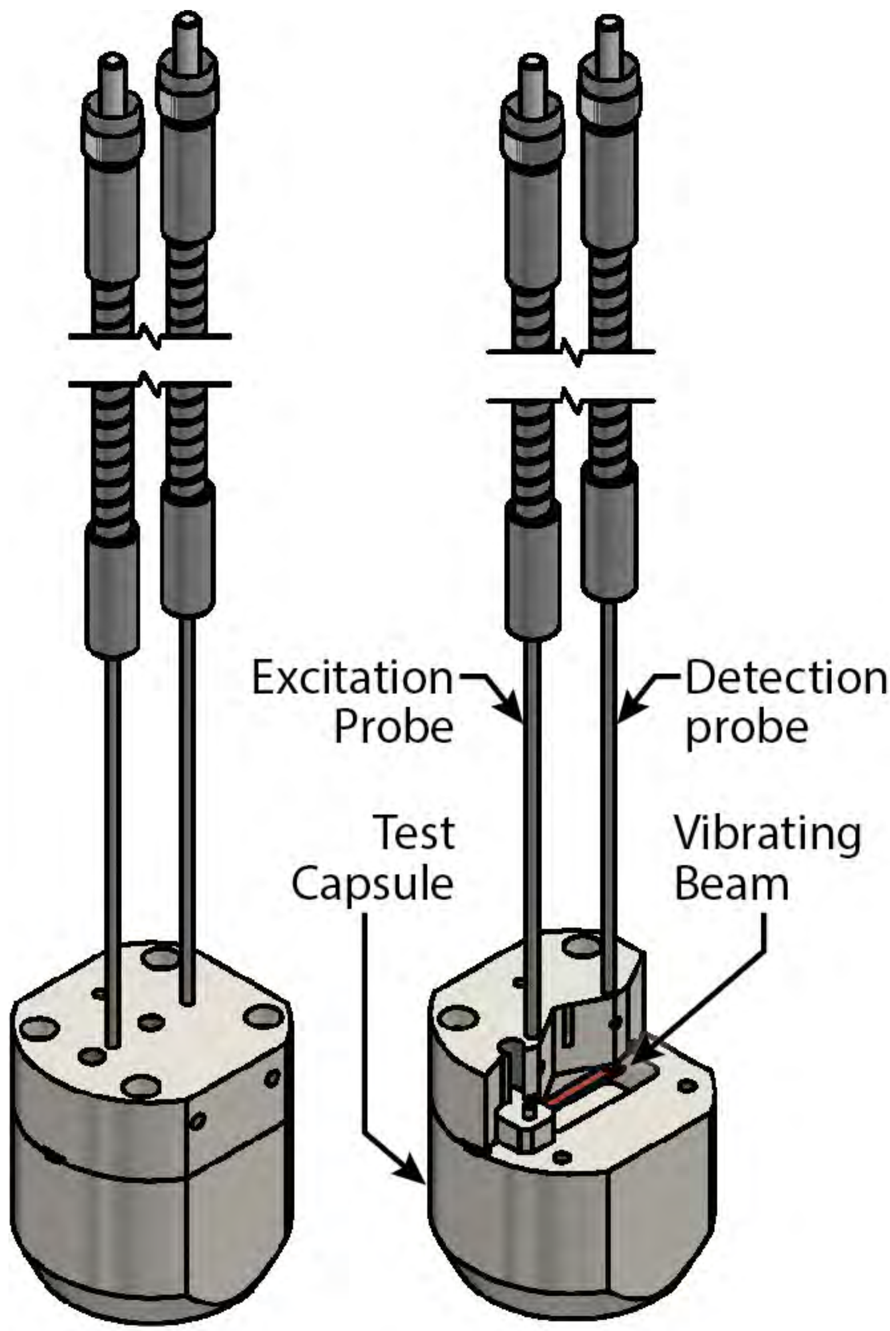


Figure 1. Design rendering of the RUSL irradiation device.

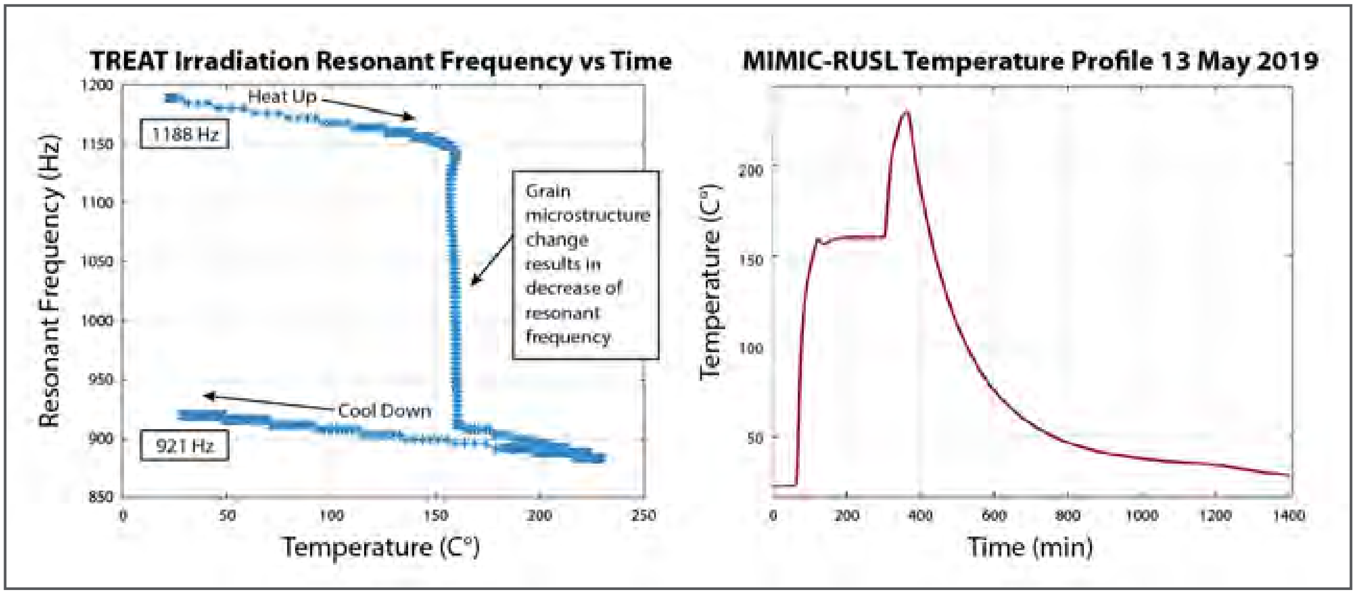


Figure 2. Data from the RUSL copper beam irradiation performed in TREAT.

the first flexural mode of the cantilever. A second fiber, near the tip of the beam, measures oscillations in light reflected from its surface (due to beam defocusing/deflection) to determine displacement so that changes can be monitored by measuring the resonant frequency. Such an approach is ideal in TREAT where total radiation dose is typically low enough to avoid darkening of optical fibers. The first RUSL irradiation was performed in TREAT in mid-2018 using a small copper specimen so that in-reactor data could be compared to out-of-pile tests which had previously been performed in a lab furnace (where the recrystallization behavior is caused by temperature alone). The specimen's fast neutron damage rate was predicted to be inadequate to cause different behavior in-reactor, making this test ideal for demonstrating whether the method was viable in the presence of neutron/gamma bombardment. Preliminary data analysis shows good similarity between the in-pile and out-of-pile data.

Lastly, the Characterization-scale Instrumented Neutron Dose Irradiation (CINDI) module design was completed, underwent final design review, and is now under final fabrication to prepare it for irradiation. CINDI consists of a few Swagelok ® capsules, each with thermocouple temperature monitoring and the ability to house several 5mm diameter disk-like specimens, arranged in a vertical hanging structure for elevated temperature irradiation within the

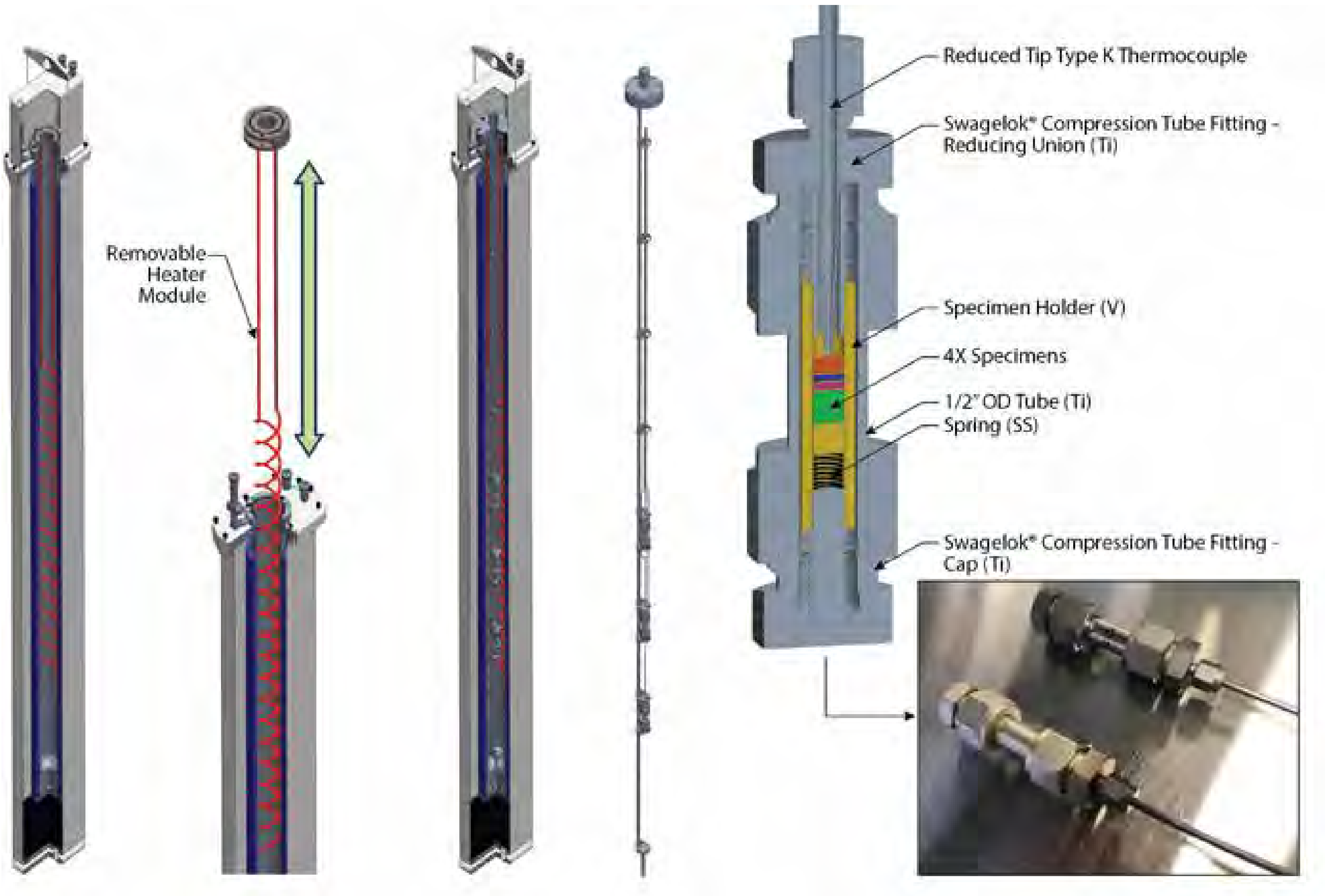


Figure 3. Renderings and capsule photo of the CINDI design.

heater module. CINDI is designed to be compatible with MARCH in the double containment mode (required for plutonium bearing tests) as well as being compatible with INL facilities able to assemble plutonium bearing samples into its capsules. The specimens for the first CINDI irradiations were completed (including U-Zr, U-Pu-Zr, and Pu-Zr metal fuel alloys). The CINDI capability essentially transforms MARCH into a tube furnace surrounded by a reactor in order to support low-dose irradiations (up to ~1E15 fissions/gram) in a well-controlled and monitored temperature environment for postirradiation (PIE) characterization and development of lower-length-scale fundamental fission damage models. The combination of the three capabilities described herein is now being explored as a means of measuring irradiation-affected phase stability of metal fuel alloys in future tests.

Disc Irradiation for Separate Effects Testing with Control of Temperature (DISECT)

Principal Investigator: W.J. Williams and D.M. Wachs (INL)
Team Members/ Collaborators: C. Hale, M. Sprenger, T. Maddock, F. Di Lemma and L. Sudderth (INL)

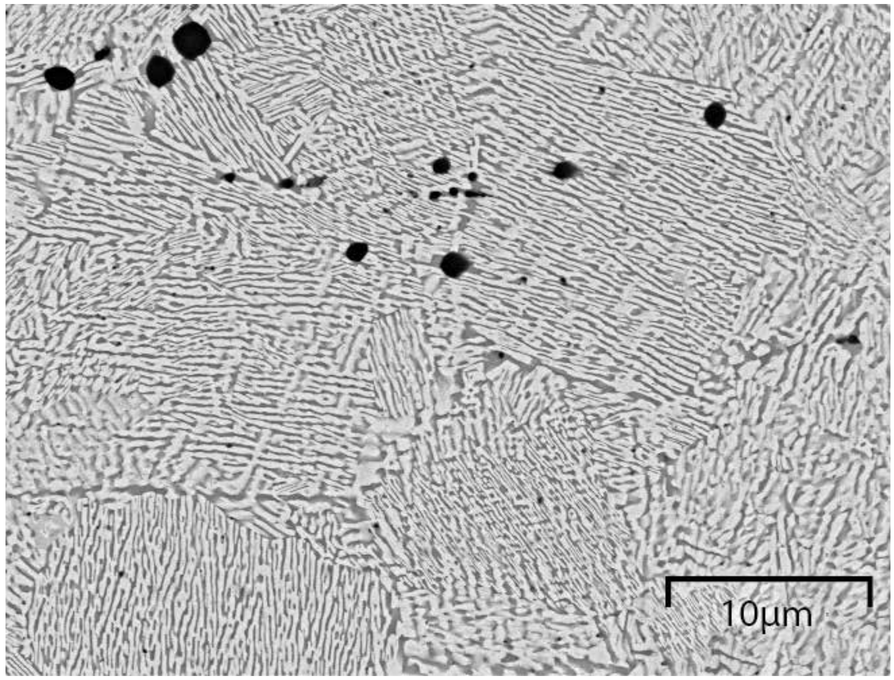


Figure 1. Moderate magnification backscatter SEM images of the low enriched U-10wt.%Zr following a two hour anneal at 900°C and a cooling rate of 60°C/hr.

The Disc Irradiation for Separate Effects Testing with Control of Temperature (DISECT) experiment is an Advanced Fuels Campaign (AFC) guided separate effects study centered upon the characterization of various U-Xwt.%Zr alloys, (X=6, 10, 20, and 30). The experiment utilizes a novel instrumented test vehicle designed by the United States (US) Department of Energy (DOE) Office of Nuclear Energy(NE) Nuclear Science User Facilities (NSUF) in conjunction with the Belgian Nuclear Research Centre (Studiecentrum voor Kernenergie (SCK•CEN)) and Idaho National Laboratory (INL). The test vehicle allows for the decoupling of fission rate, fission density, temperature, and composition, something not achievable in integral fuel tests. Fresh U-Zr foils have been characterized and are awaiting irradiation in 2020 in the BR2 reactor. Characterization includes Scanning Electron Microscopy (SEM), Transmission Electron Microscopy (TEM), chemistry, and neutron diffraction. The separate effect testing is intended to provide a more comprehensive understanding of in-pile behavior to facilitate the transition from empirical to mechanistic modeling and simulation.

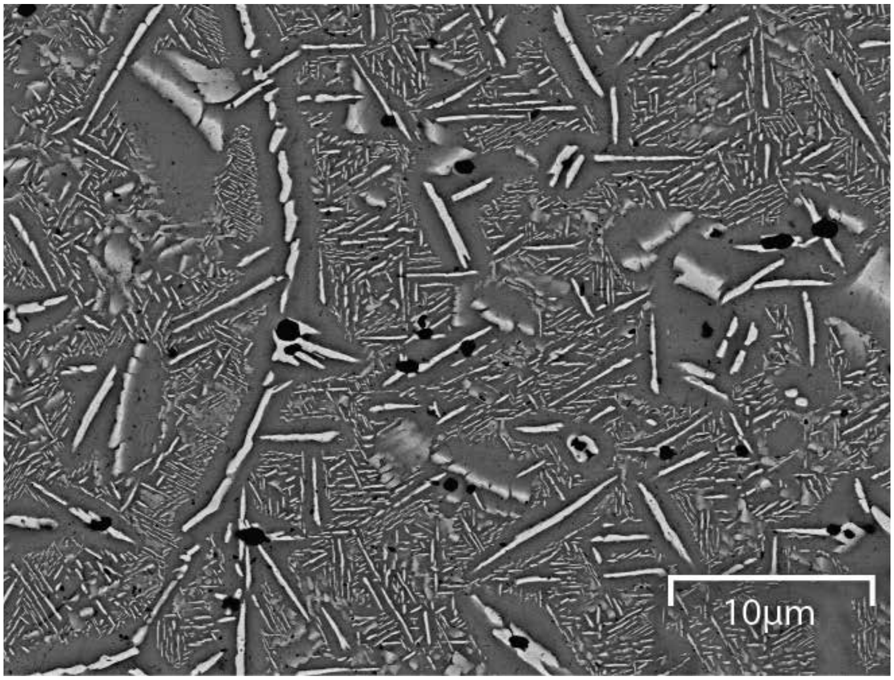


Figure 2. Moderate magnification backscatter SEM images of the low enriched U-30wt.%Zr following a two hour anneal at 900°C and a cooling rate of 60°C/hr.

Project Description:
The technical objectives of the experiment began with the guidance of test vehicle design and in-pile instrumentation as well as the development of a unique fuel fabrication method that allows for the decoupling of historically convoluted reactor conditions such as power and temperature. Beyond experiment design, phenomena identification and ranking tables (PIRT) were created for the U-Zr system. The PIRT analysis was meant to identify reactor conditions and material properties that directly influence the performance-limiting phenomena present in the U-Zr system, specifically fuel swelling and constituent redistribution. The results of this study are being used to guide pre and postirradiation characterization of the alloys in order to provide a fundamental understanding of the microstructural, and subsequent performance, evolution of the alloy. This work includes routine techniques such as SEM as well as advanced characterization such as neutron diffraction and TEM coupled with in-situ heating. These advanced techniques allow for the investigation

DISECT is a venue for separate effects testing on U-Zr alloys intended to advance the fundamental understanding of the microstructural evolution, both in and out of pile, in order to aid in the transition from empirical to mechanistic modeling required to advance metallic fuels to moderate and high burnups.

of bulk and atomistic understanding of the phase transformation kinetics taking place during expected operating conditions. Coupling these with postirradiation examination (PIE) will allow for the study of irradiation's influence on phase transformations and microstructure. Understanding the precise nature of microstructural evolution taking place under irradiation will allow for the tailoring of reactor conditions or fuel design needed to extend the fuel lifetime. Additionally, being able to perform more accurate predictive modeling will allow for a more precise identification of safety margins.

Accomplishments:

Preliminary thermal and neutronics analysis were completed on the test vehicle at INL and SCK•CEN, respectively. These analyses indicate that the fuel and vehicle design are capable of irradiating samples under targeted fast reactor conditions while deconvoluting thermal, compositional, and neutronic effects. These analyses were critical to the completion of phases 1/3 and 2/3 of the CEE (Committee for the Examination of Experiments), a safety review for reactor insertion, with the third scheduled to take place in early fiscal year (FY) 20. Following the completion of safety reviews, fabrication continued, and preirradiation characterization began.

The SEM, taking place at the Center for Advanced Energy Studies (CAES) and INL's Fuels and Applied Science Building (FASB), was performed on U-10wt.%Zr and U-30wt%Zr samples throughout the novel fabrication process, pictured in Figures 1 & 2, respectively. This work illustrates microstructural recovery of the U-10wt.%Zr alloy after rolling and annealing to that expected following typical heat treatments. Additionally, the SEM analysis was used to characterize the microstructure of both the 10 and 30wt% alloys under two unique heat treatments, describing the thermal history's influence on microstructure. Purdue University was awarded two Rapid Turnaround Experiments, RTEs, namely 'In-Situ Phase Analysis of Phase Transitions in U-(6, 10, 20, 30) wt%Zr Fresh Fuels using Neutron Diffraction' and 'Phase Evolution of Uranium-Zirconium Alloys under In-Situ TEM Heating.'

The neutron diffraction work, completed at Los Alamos Neutron Science Center's (LANSCE) Lujan Neutron Scattering Center pictured in Figure 3, made precise measurements on crystal structure, phase fractions, and texture throughout fabrication. Further examination was also performed with in-situ heating and controlled cooling with the intent of furthering the understanding

Figure 3. Researcher Walter Williams preparing for neutron diffraction at LANL's lujan neutron scattering center's high-pressure-preferred orientation beamline.

of both phase transformations and phase transformation kinetics in the system. The samples included various alloys ranging from U-6wt%Zr to U-30wt%Zr, corresponding to compositions observed in-pile following constituent redistribution. As phase transformations are a key factor in the occurrence of constituent redistribution, the understanding of them is subsequently critical in the accurate prediction of temperature-to-melt, a key safety factor in all nuclear fuels. Increasing this understanding will allow for the re-evaluation of reactor safety margins and operating conditions. TEM analysis with in-situ heating is ongoing and intends to couple the atomistic neutron diffraction data to the nanostructure present in the fuel. The highly controlled out-of-pile test conditions can then be contrasted with the postirradiation characterization to deconvolute thermal and neutronic effects on the microstructure. While the pre-irradiation characterization data has been collected, the characterization itself is ongoing.

SEM Comparison Between Three Transmutation Fuel Experiments: X501, FUTURIX FTA, AFC-1H

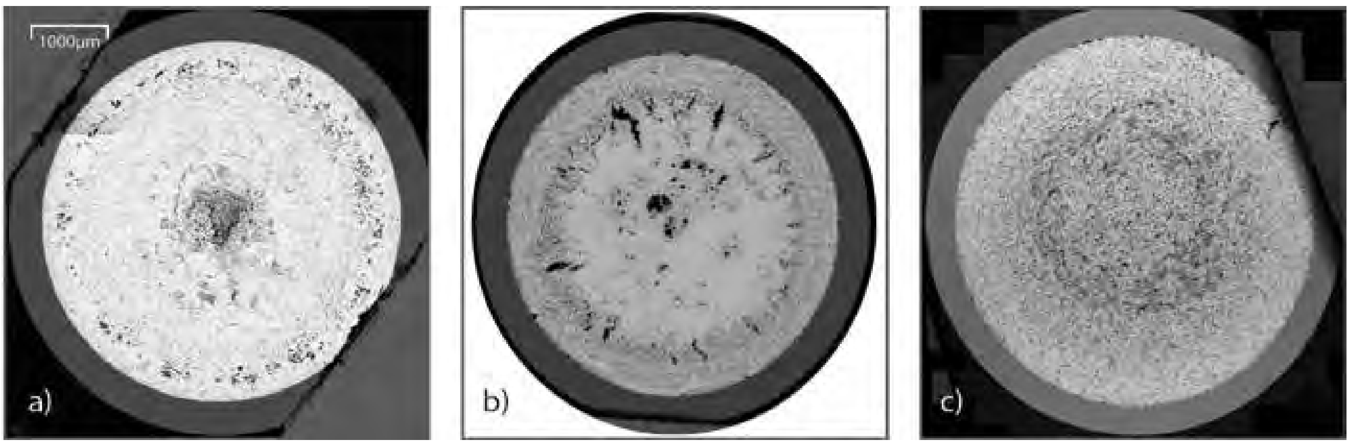
Principal Investigator: Luca Capriotti, INL
Team Members/ Collaborators: Jason Harp, INL

Scanning electron microscopy on three transmutation metallic fuel experiments have shown that the microstructure evolution under irradiation is not affected by minor actinides addition.

A long term objective of the Advanced Fuels Campaign (AFC) is the investigation into enabling technologies that allow for the destruction of long-lived minor actinides generated in irradiated nuclear fuel through transmutation in fast reactors. In an effort to better understand the fuel performance implications of adding minor actinides to a fuel system, several AFC irradiation experiments have examined the performance of metallic, nitride, and ceramic fuels in approximated fast reactor neutron spectra at the Idaho National Laboratory Advanced Test Reactor (INL ATR). In addition and complementary to ATR experiments, systematic characterizations of experiments irradiated in true fast reactors are performed in order to assess ATR in testing fast reactor fuel. For this purpose, postirradiation examination (PIE) and advance microscopy are performed on selected fuel pins from the FUTURIX-FTA (Phenix sodium fast reactor) and the Experimental Breeder Reactor (EBR-II) X501 experiments.

Project Description:

The PIE and microscopy exams such as scanning electron microscopy (SEM) have been completed on three irradiated transmutation metallic fuel experiments in order to characterize and compare the different behavior in pile and microstructures evolution under irradiation. For the SEM analysis, samples were taken from AFC-1H experiment in ATR and FUTURIX-FTA experiment in the Phenix sodium fast reactor (France). These are sibling experiments of 35U-29Pu-4Am-2Np-30Zr



composition to validate ATR testing of fast reactor fuel. The third sample analyzed was taken from a unique transmutation experiment in EBR-II, X501 (U-20.2Pu-9.1Zr-1.2Am-1.3Np composition).

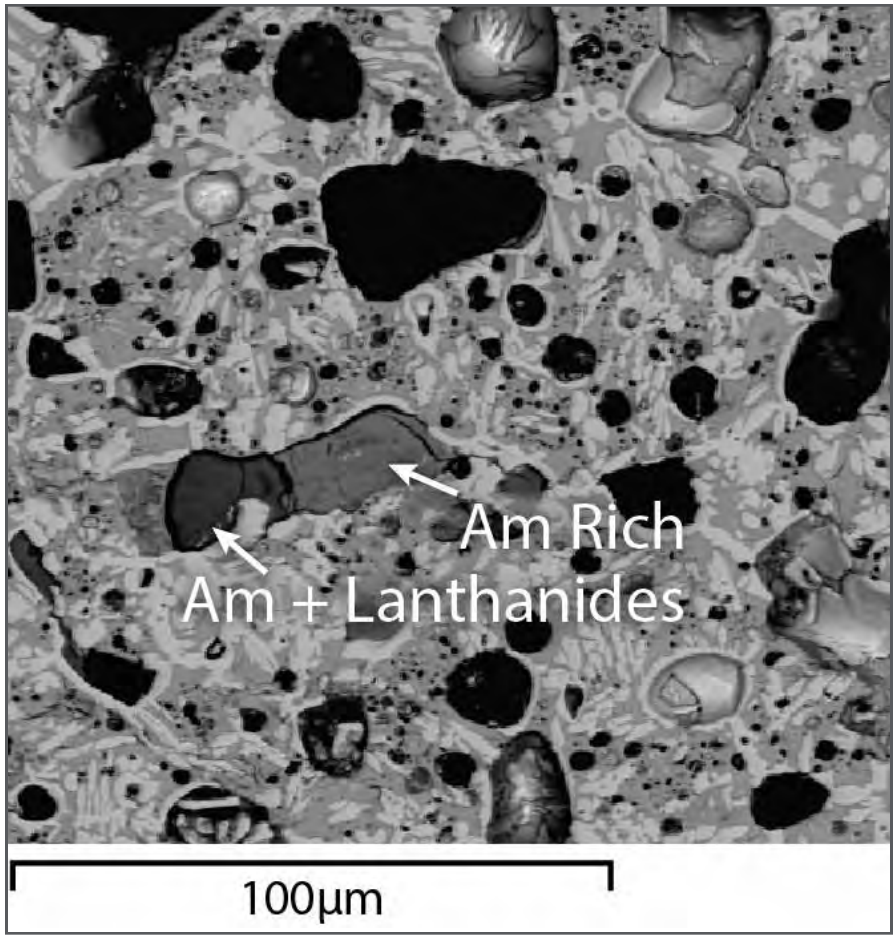
This project confirms that irradiation in ATR creates the correct radial power profile therefore, the correct radial temperature profile to reproduce thermal-driven phenomena.

Accomplishments:

SEM examinations have been performed on three irradiated transmutation metallic fuel samples taken respectively from X501, AFC-1H and FUTURIX-FTA experiments. The microstructure of X501 sample (Figure 1a) presents the three main regions while the AFC-1H (Figure 1b) and FUTURIX-FTA (Figure 1c) show a predominant central region that expand for more than half of the cross sections. Energy dispersion X-ray (EDX) line scan measurements revealed that the Zr element redistribution largely occurs in the same manner between AFC-1H and FUTURIX-FTA samples and it is less readily apparent compared to the X501 sample and with historically

Figure 1. Back-scattered electron (BSE) montage of the 3 SEM samples cross-section. In the upper left and lower right area for a) and c) there is some copper tape used to ground the sample electrically. In this figure, brighter areas indicate areas of higher electron density associated with higher atomic number elements.

Figure 2. BSE image of the central portion of X501 fuel sample where an Am – lanthanide precipitate is highlighted.



U-20Pu-10Zr metallic fuel. The Zr concentration is higher in the center regions of the three fuel samples compare to the as fabricated compositions, inferring the existence of a cubic-like structure (gamma phase). Elsewhere in the fuels there are several different phases with different amounts of Zr, U and Pu. To notice is a phase in the intermediate region of X501 where the Zr content drops to 2.5 wt.%; this phase is not well understood and to assess its impact on the fuel solidus temperature, further research is needed.

Minor actinides act similarly in all of the three samples. Np behaves very similarly to U and is present in the fuel matrix while Am is behaving like a lanthanides and is segregated in precipitates / secondary phases also with noble metals (Figure 2).

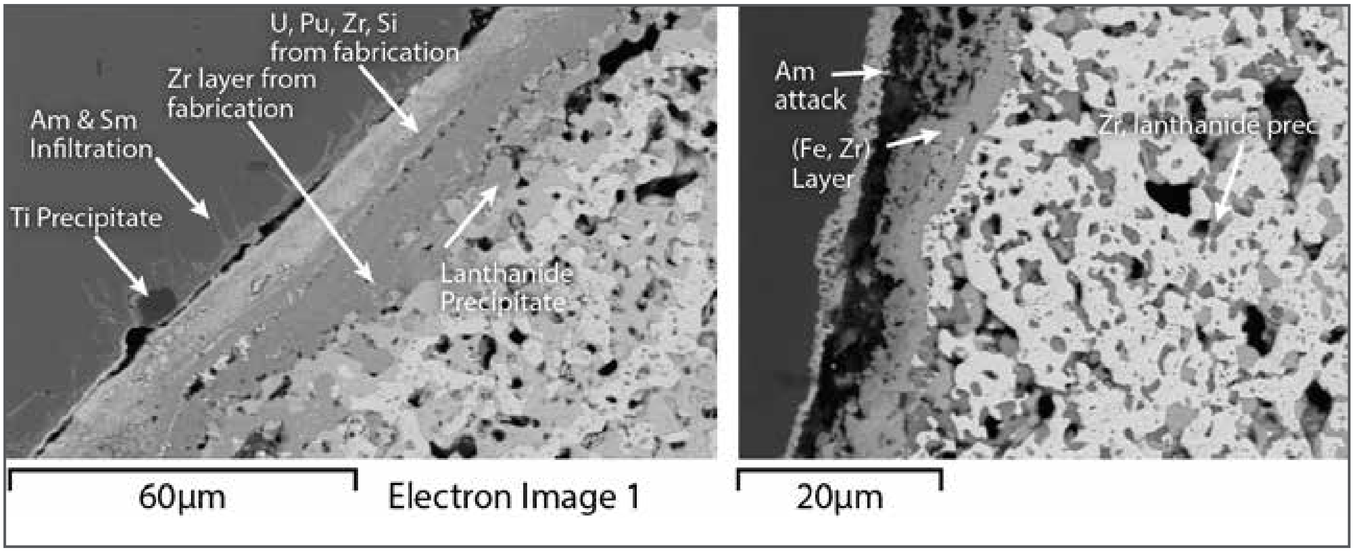


Figure 3. a) High magnification BSE detail of the FUTURIX-FTA (left) and b) X501 (right) microstructure near the cladding.

A higher magnification image, in Figure 3a, shows details near acladding microstructure observed in FUTURIX-FTA sample. The cladding is AIM1 (austenitic steel), which contains ~ 0.5 wt.% Ti and it appears the Ti has precipitated in the cladding during irradiation. There are also precipitates of lanthanides (La, Ce, Nd, and others) in the fuel side near the Zr layer, which was formed during fabrication. However there does not appear to be any attack of the cladding from the major fission product lanthanides (La, Ce, Nd). The most significant feature observed in Figure 3a is an integranular infiltration of Am and Sm into the cladding. This infiltration is very different from AFC-1H and X501 in which there is a planar front of Am as shown in Figure 3b. This difference is likely largely driven by the difference in cladding composition where the AFC-1H and X501 experiments were clad with a ferritic-martensitic (F/M) steel, but more research into the phenomenon is needed.

The overall microstructure evolution seems not affected by minor actinides addition compared to expected behavior of conventional U-19Pu-10Zr ternary metal fuels. A different fuel cladding chemical interaction (FCCI) phenomena is clear between the use of AIM1 (austenitic steel) and HT9 F/M cladding.

Preliminary Postirradiation Examination on a U-Pu-Zr-Ga Pin Tested in EBR-II

Principal Investigator: Luca Capriotti, INL
Team Members/ Collaborators: Jason Harp, INL

Characterization of the behavior in pile of the EBR-II U-Pu-Zr-Ga metallic fuel pin experiment is of paramount importance to assess the feasibility of the usage of weapons-grade Pu as fast reactor metallic fuel without removing Ga.

A long-term objective of the Department of Energy's (DOE's) Advanced Fuels Campaign (AFC) is to develop and demonstrate the technologies needed to transmute long-lived transuranic actinide isotopes contained in spent nuclear fuel into shorter-lived fission products, which is important to minimize high level waste. As part of this development, candidate fuel compositions and forms are irradiated in a cadmium-shrouded positions, to simulate a fast spectrum reactor, at the Idaho National Laboratory (INL's) Advanced Test Reactor (ATR), and are subsequently examined at the Hot Fuel Examination Facility (HFEF). Complementary postirradiation examination (PIE) is performed on several metallic fuel pins irradiated in the experimental breeder reactor's (EBR-II) sodium fast reactor (SFR). This is performed to validate the cadmium-shrouded testing in the ATR as prototypical of a true fast reactor. This accomplishment focuses on the PIE performed on a selected EBR-II fuel pin from a unique experiment X521. This experiment contained U-19.8Pu-10Zr-0.2Ga metallic fuel pins to assess the feasibility of the usage of weapons-grade Pu as fast reactor metallic fuel without removing Ga.

Project Description:

The X521 experiment was meant to characterize the behavior in pile of Ga addition to a well characterized metallic fuel system and included five experimental fuel pins of alloy U-19.8Pu-10Zr-0.2Ga. They were irradiated in EBR-II in mid-1994 and by the time EBR-II was shut down (in the fall of 1994), the fuel elements all achieved a peak burnup of 1.5 at%. However, due to funding limitations, no postirradiation work has been conducted on these fuel elements so far.

A PIE campaign on X521-G594 (one of the five fuel pins) has been performed at HFEF of the Material and Fuel Complex (MFC) at INL. The PIE campaign consisted of non-destructive examination such as visual inspection, neutron radiography, dimensional exams, gamma spectroscopy and destructive exams such as fission gas release collection and optical microscopy. These examinations will assess the feasibility and safety of the usage of weapons-grade Pu as fast reactor metallic fuel without removing Ga through a timely and costly reprocessing.

Accomplishments:

Preliminary PIE on X521-G594, a unique metallic fuel pin of U-19.8Pu-10Zr-0.2Ga composition, has been performed at INL. The fuel pin was irradiated in EBR-II at a peak burnup of 1.5 at%.

Neutron radiography of the examined pin revealed typical metallic fuel behavior. The fuel appears to have swelled radially to fill the cladding and a small amount of fuel is dissolved into the sodium at the top of the pin plenum. Furthermore, the presence of some lift off the pin is evident and from the neutron radiography (shown in Figure 1) is possible to see some neutron attenuation in the central region. This could indicate either the

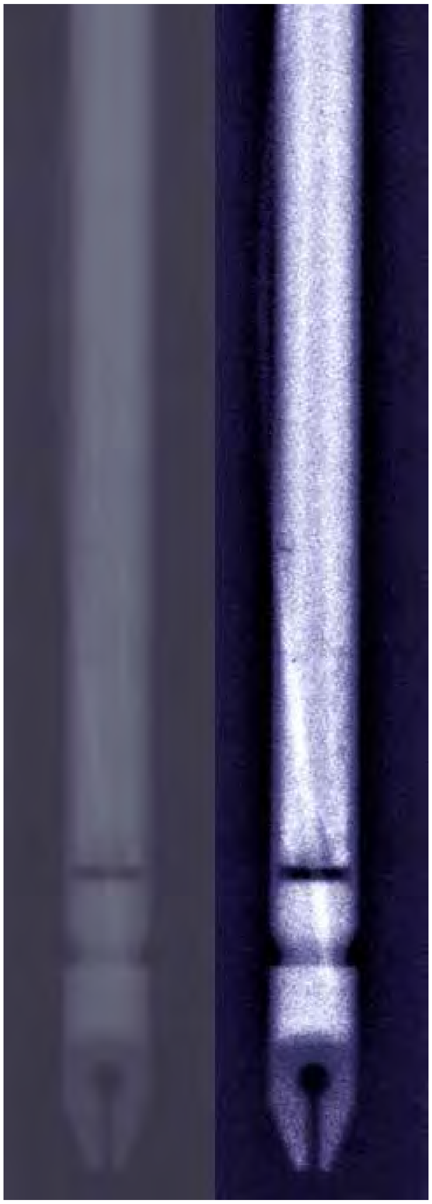


Figure 1. Neutron radiography image (left) and the same image with enhanced contrast showing the particular of the bottom of the fuel pin, lift off and the neutron attenuation in the central region.

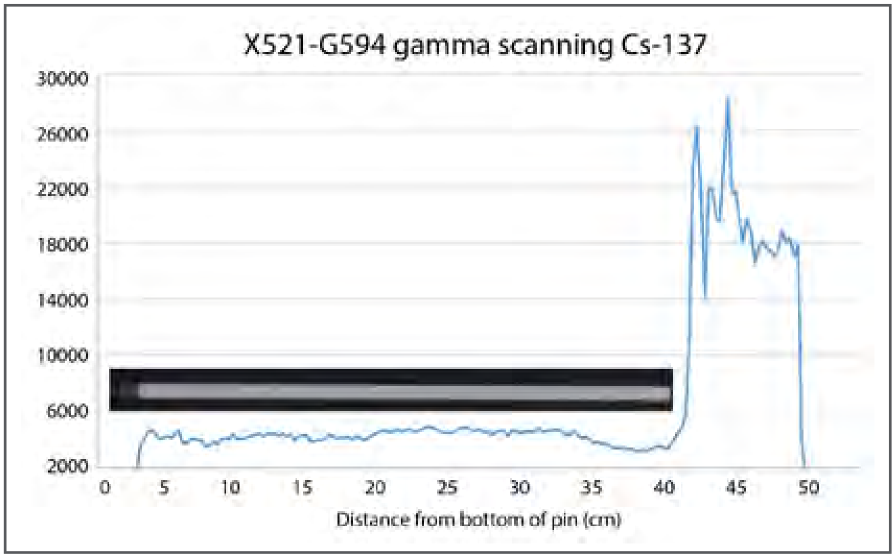


Figure 2. Gamma scanning profile of Cs-137 along the fuel pin. Neutron radiography of the pin is also shown as reference.

beginning of change in bulk density in the central region of the pin or a change in composition.

The gamma spectrometry technique reveals the behavior of gamma emitter isotopes within the fuel stack. Typical isotopes important to understand the performance of a metallic fuel system are Ru-106, Cs-137, Co-60 and Eu-154. In this specific case all isotopes have decayed, due to the long time pass between irradiation and exam, apart from Cs-137 which is often dissolved in the Na bond between the fuel and the cladding. The Cs-137 migrates with the Na above the fuel column producing a Cs activity spike above the fuel as observed in Figure 2.

Preliminary destructive examinations as optical microscopy are underway. Figure 3 shows a metallography image taken from the center of the fuel stack. The fuel has swelled to contact the cladding for the major extent (it is possible to notice some part of circumferential gap not closed) and re-structuring of the microstructure has already begun at this low burn-up. Rounded porosity in the center of the fuel are visible and from them the existence of a cubic structure can be inferred. From the central region towards the periphery various degrees of porosity can be noticed. Close to the cladding large and swirled porosity are under

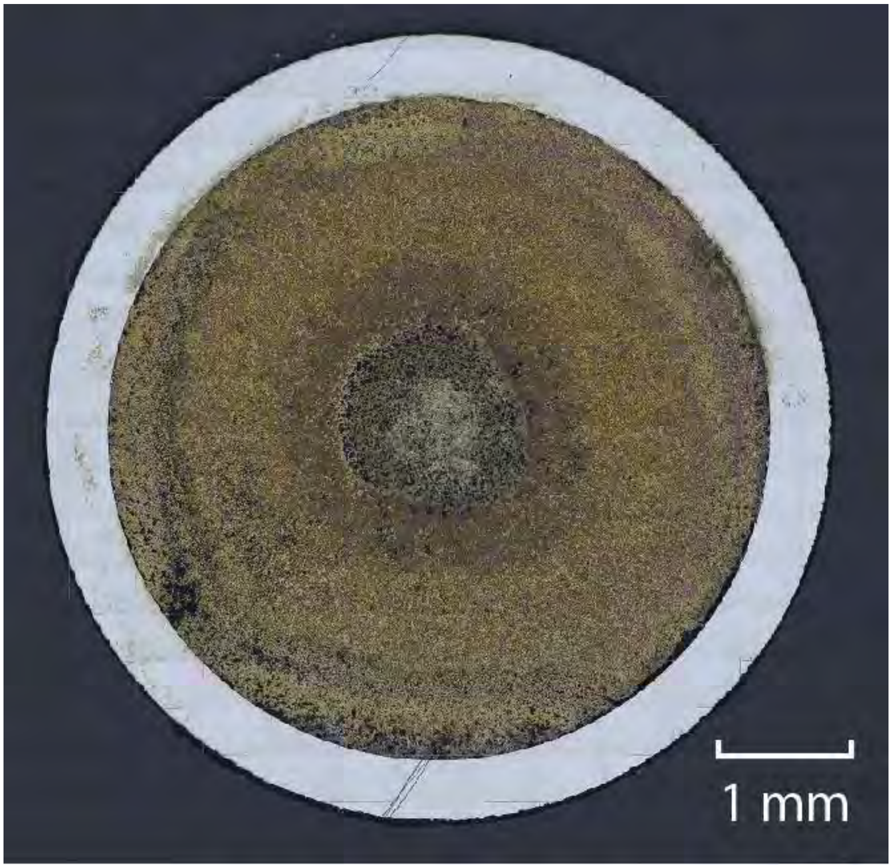


Figure 3. Metallography image taken from the center of the fuel stack, showing the different microstructure features along the fuel radius.

development in many areas; this can be linked to a predominant presence of the orthorhombic U phase at these radial positions, as observed in typical U-Pu-Zr ternary alloy.

Preliminary considerations on the data collected are that the behavior of the fuel pin at this low burnup looks not drastically affected by the addition of this small amount of Ga compared to the expected behavior of well-known U-19Pu-10Zr ternary metal fuels. However, a more complete microstructure study, employing also electron microscopy techniques, is needed to characterize the behavior of Ga within the fuel matrix. This is planned for next year.

3.4 FUEL SAFETY TESTING

Characterization of The Metallic Foam for Source Term Analyses

Principal Investigator: F. G. Di Lemma, INL
Team Members/ Collaborators: K. Wright, A. Winston, L. Capriotti, C. Jensen and D. Wachs (INL)

Figure 1. The EPMA instrument scientist (Ms. Karen E. Wright) performing a sample loading operation on an irradiated sample.

Under the Department of Energy (DOE) Advanced Fuels Campaign (AFC), fuel safety research for metallic fast reactor fuels has recently become a major focus. Knowledge of fission products transport behavior in nuclear fuel is crucial to achieving a mature technology that can be used by reactor designers. This research applies advanced characterization techniques to evaluate the behavior (e.g., chemical composition, fission product migration and microstructural evolution) of the metal 'foam' structure found in certain metallic fuel pins. The foam structure forms at the top of the fuel column where volatile fission products are believed to accumulate, also where fuel failures have been found to occur under failure-limits testing of Experimental Breeder Reactor (EBR)-II pin designs in Transient Reactor Test Facility (TREAT). Detailed understanding of this structure is still lacking but is needed to support transient fuel performance and source term determination analyses, mandatory for licensing of new nuclear fuel forms.

Project Description:

For the first time, the foam structure formed in certain metallic fuel pins is analyzed in detail using advanced characterization techniques, now available in the Irradiated Materials Characterization Laboratory (IMCL) at Idaho National Laboratory (INL). A U-Pu-Zr metallic fuel pin, irradiated to 11.09% peak burnup in EBR-II and subjected to an overpower event in the OPT-1 experiment (1992), was selected to assess the evolution of microstructure and the chemical form of fission products after a transient. This irradiated rod underwent a moderate overpower ramp of 32% over a 5-minute period and represents a first step to reviving detailed studies of transient irradiation performance of metallic fuels since the early 1990's. These exams will aid detailed understanding of transport and potential release mechanisms following unexpected

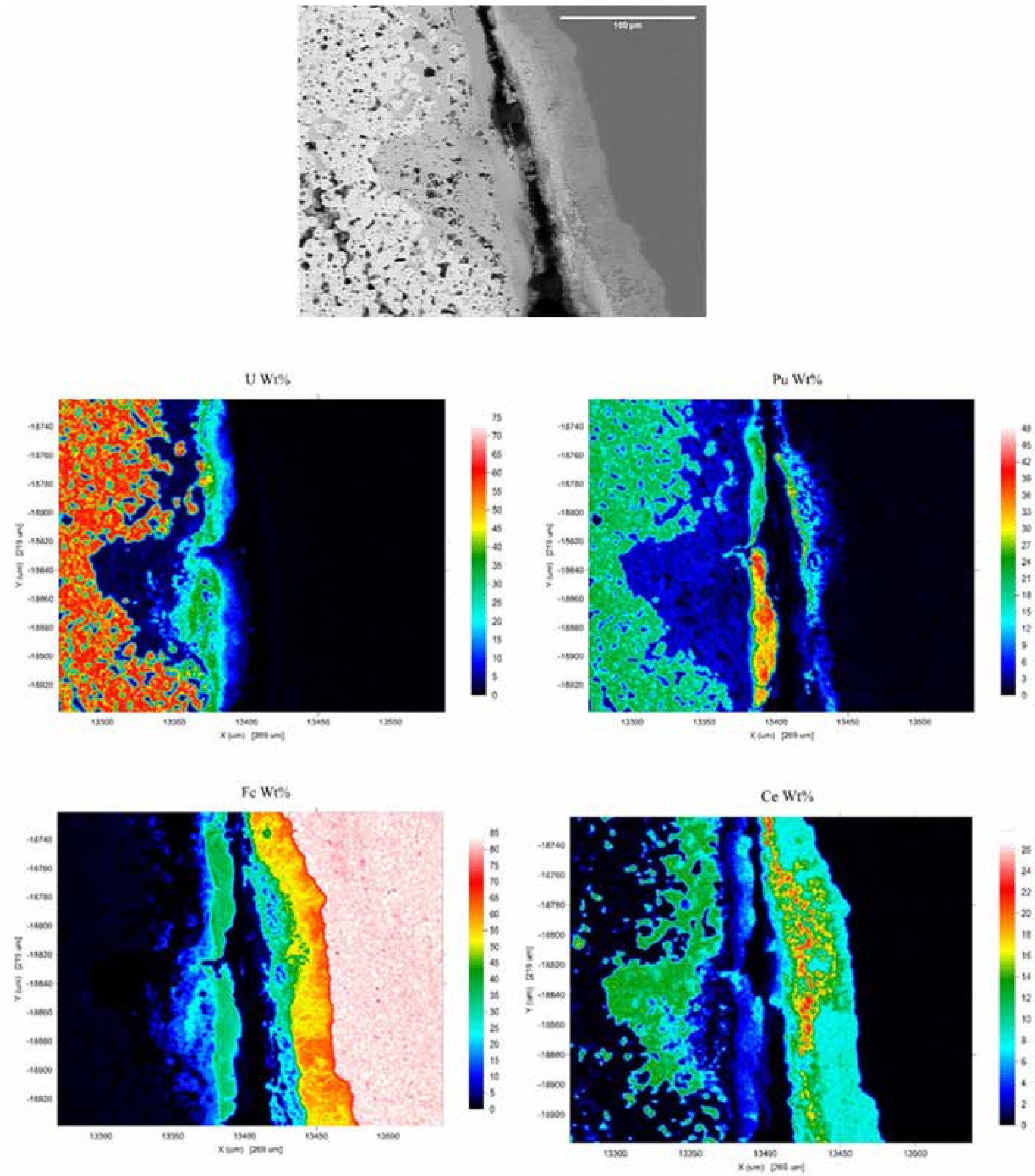


Figure 2. On the left-hand side, optical image of the top of the fuel, on the right-hand side, elemental wavelength dispersive X-ray (WDX) maps obtained by Electron Probe Micro Analysis (EPMA), showing the three regions and Pu and RE precipitates.

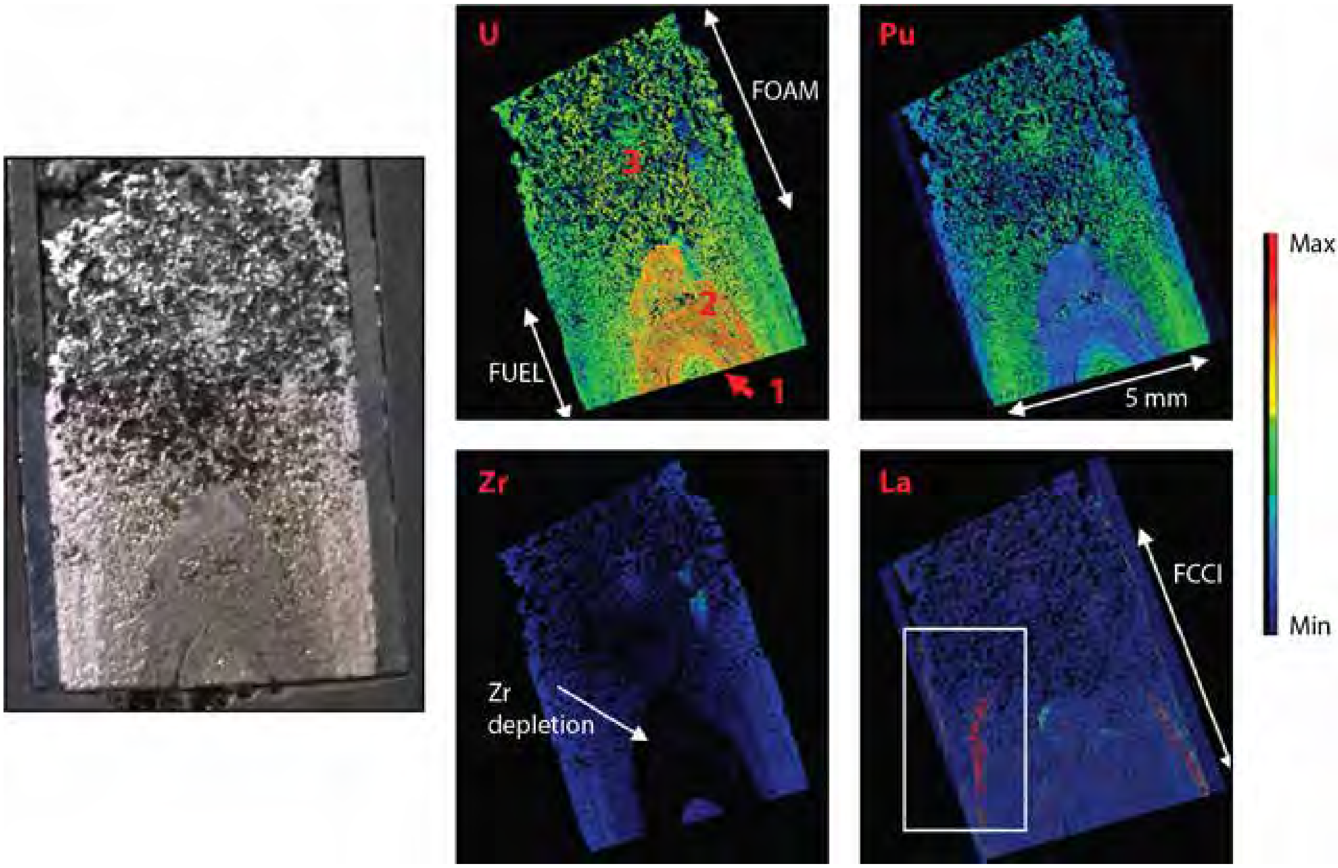


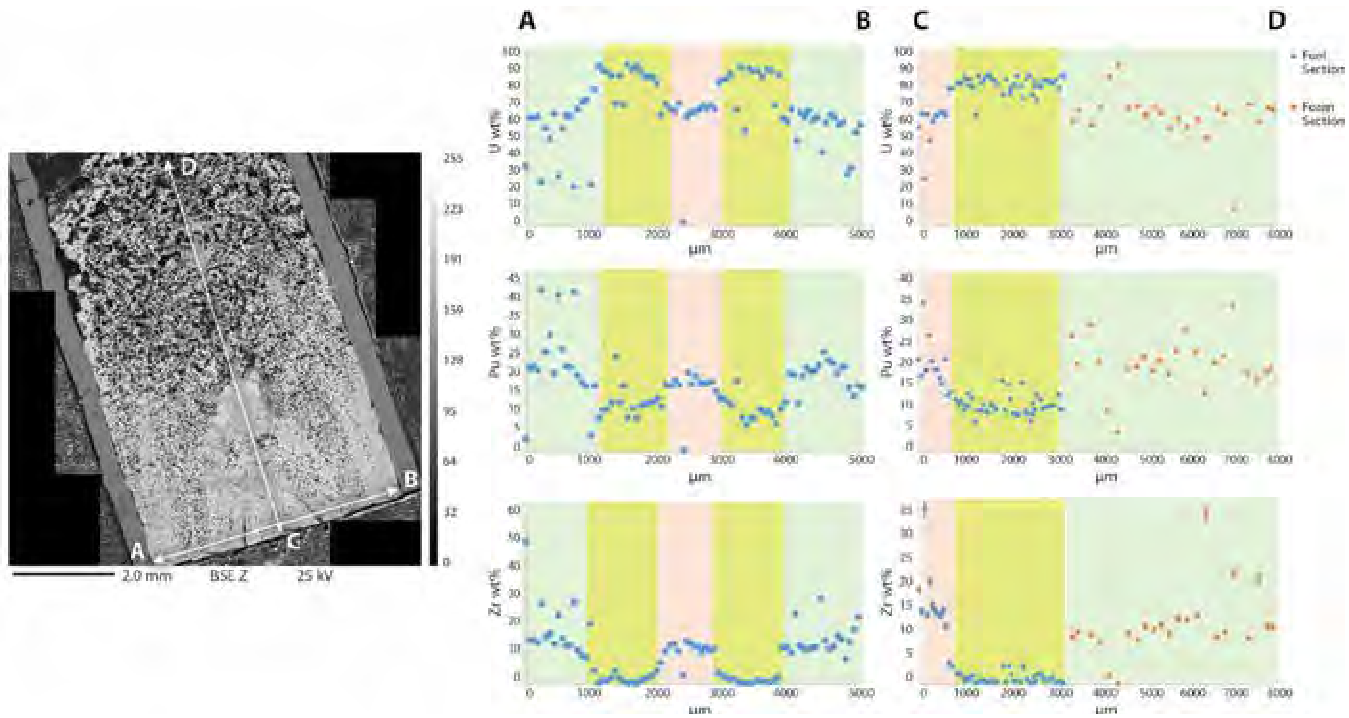
Figure 3. EPMA points analyses of the main elements' redistribution (U, Pu, Zr) across the 2 main axes (radial A-B and axial C-D) as shown in the back-scattered electron (BSE) image on the left.

events, and support determination of a possible source term. Samples of the "foam" structure region at the top of the fuel pin were prepared and have been analyzed using EPMA to reveal a detailed compositional and microstructural understanding of this region.

Accomplishments:

The metallic fuel specimen with metal "foam" was retrieved from the Hot Fuels Examination Facility (HFEF) at INL. A section from the top of the fuel rod was chosen to assess the foam microstructure and integrity after a moderate transient. Optical microscopy was conducted on the specimen in HFEF for preliminary examination supporting following steps. Subsequently, the sample underwent final preparations in IMCL for measurements by the EPMA. The polishing procedure used a nonaqueous solution in an attempt to retain water-soluble fission products. Samples were then analyzed by EPMA in IMCL using the shielded Cameca SX100 R (Figure 1), capable of handling irradiated fuel. Both quantitative point analyses and WDX maps were acquired to investigate the microstructural evolution of the foam structure, fission product behavior, and redistribution of primary base-alloy compositional elements.

The analysis results revealed three typical alloy redistribution regions (Figures 2-3): 1) the central region with slightly enriched in Zr-Pu; 2) the annular neighboring area showing Zr



depletion; 3) and the outmost region with a U-Pu-Zr composition similar to the as fabricated alloy. The redistribution of these main elements is strongly influenced by the temperature gradient. Thus, the spatial composition data coupled to known phase diagram and modeling, will provide insight into possible phases formed and the temperature reached in the various regions. Moreover, the region with low Zr concentration is of particular interest for fuel performance due to the presence of phases with lower solidus temperature.

Fuel Cladding Chemical Interaction (FCCI, Figure 4) was also investigated in this specimen, as this phenomenon is considered a limiting factor to fuel performance and influences the performance limits of the fuel during transient conditions. FCCI was observed in a limited area at the top of the fuel, where lanthanides and Pu migration extend approximately 50 µm into the cladding and Fe and Ni was found up to approximately 100 µm into the fuel. Finally, only a minor presence of volatile fission products (e.g., Cs and Xe) was found in the foam region, inconsistent with previous indications from gamma spectroscopy on the pin. The nonaqueous sample preparation method is believed to not be successful in retaining the formed chemical compound, as Na was concomitantly wash out during sample preparation.

By these analyses we can conclude that the samples behaved in line with steady state irradiation of metallic fuel, preliminary indicating that the moderate transient ramps did not influence the fuel performance. These analyses confirmed safety of this fuel form even under unplanned events.

Figure 4. Elemental WDX maps obtained by EPMA on the cladding-fuel interface, showing FCCI.

Irradiation Testing of Slotted Metallic Fuel

Principal Investigator: Pavel Medvedev, INL

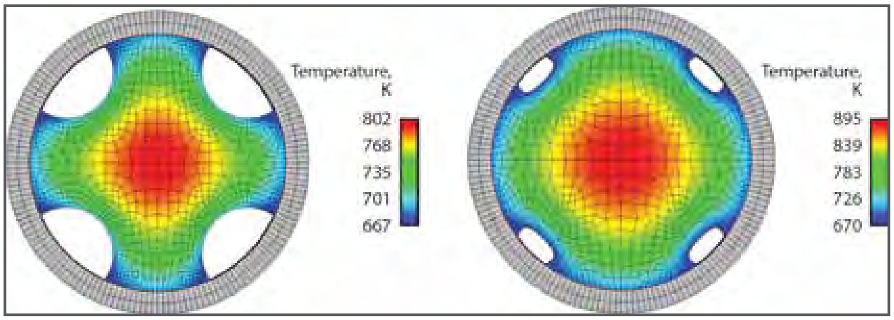


Figure 1. Four-slotted fuel before and after irradiation.

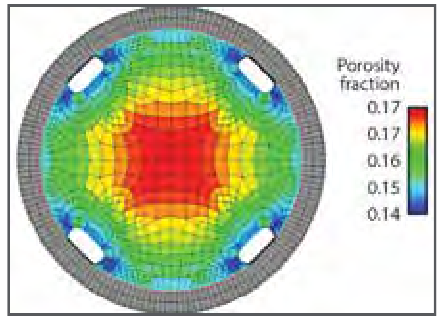


Figure 2. Porosity distribution in four-slotted slotted fuel after irradiation.

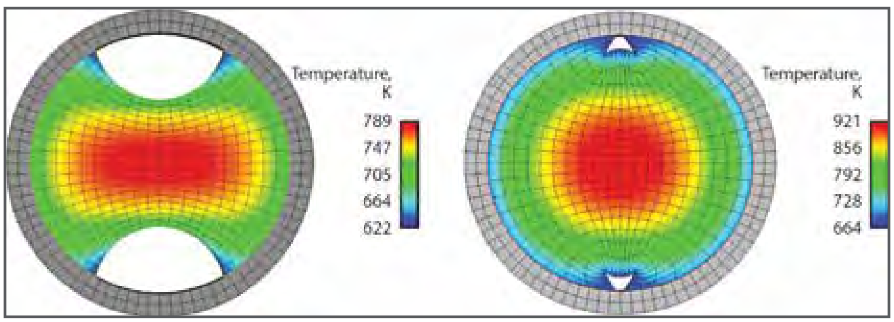


Figure 3. Two-slotted fuel before and after irradiation.

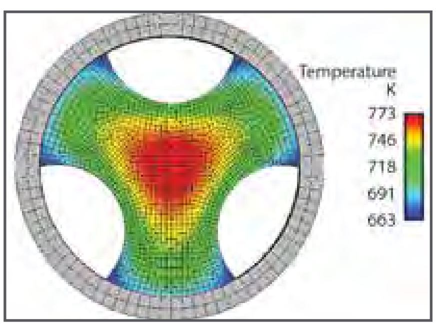


Figure 4. Nominal cladding thickness.

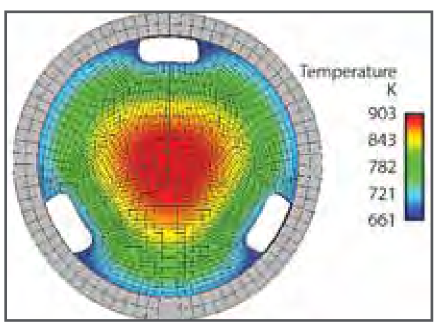


Figure 5. Reduced (50%) cladding thickness.

Irradiation testing of slotted metallic fuel has been proposed by the Advanced Fuel Campaign (AFC) in order to develop high burnup fuel designs (INL/EXT-18-45933). Pre-fabricated fuel slots are designed to accommodate fuel swelling.

Project Description:

The key challenge is understanding the evolution of the fuel slug geometry during irradiation. Closure of prefabricated slots without significant cladding strain is a desired behavior. Excessive cladding strain due to fuel swelling occurs if slots fail to close, signifying undesired behavior.

Slotted Fuel Deformation

- BISON simulation of U-10Zr sodium bonded slotted fuel irradiation predicted favorable behavior under prototypic conditions
- Swelling and creep of the fuel cause prefabricated slots to close without significant cladding deformation

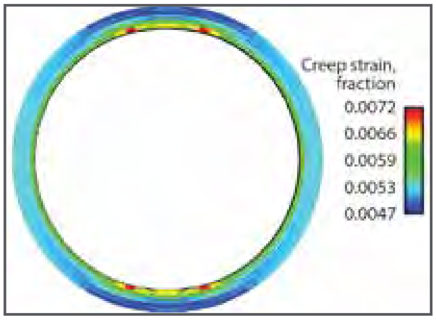


Figure 6. Nominal cladding thickness.

- Distribution of fission gas porosity in the fuel confirms that fuel deformation is driven by fission gas induced swelling

Cladding Behavior

- Cladding ovalization depends on the cladding wall thickness
- (Cladding deformation (10 x magnified) when using two-slotted fuel

Cladding provides driving force for slot closure

- Simulation of unconstrained two-slotted fuel swelling shows non-closure of the slots
- Constraint of the fuel by the cladding defines the nature of fuel deformation

Fuel Creep is the mechanism responsible for closure of the slots

- BISON simulation performed assuming reduced fuel creep show non-closure of the slots and extreme cladding deformation
- Creep-resistant fuel compositions are no suitable for slotted fuel design

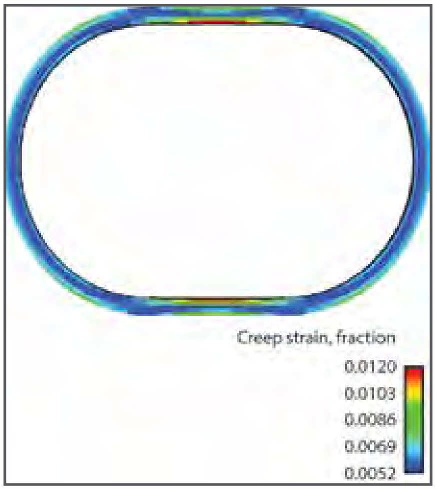


Figure 7. Reduced (50%) cladding thickness.

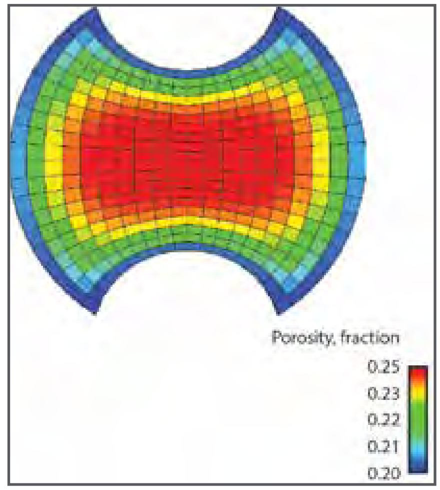


Figure 8. EOL shape of unconstrained two-slotted fuel (no cladding).

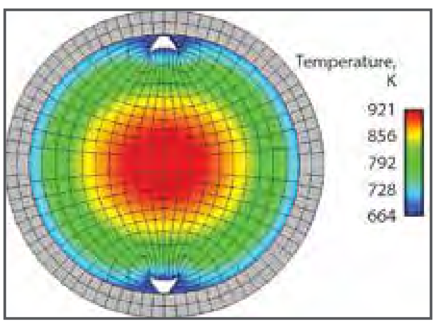


Figure 9. EOL shape of constrained two-slotted fuel (with cladding).

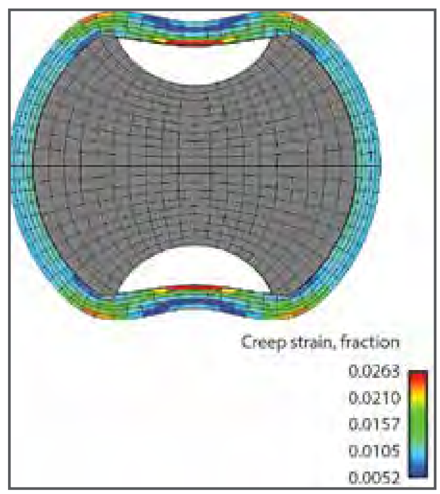


Figure 10. Reduced fuel creep.

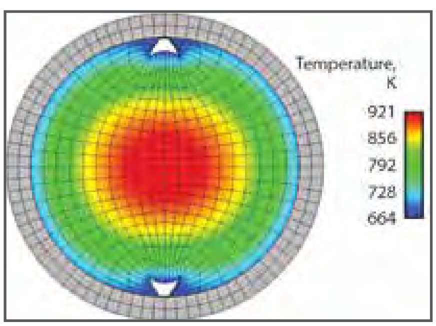


Figure 11. Nominal fuel creep.

Validation of BISON Swelling Model for Metallic Fuel

Principal Investigator: Christopher Matthews, INL
Team Members/ Collaborators: Cetin Unal, INL

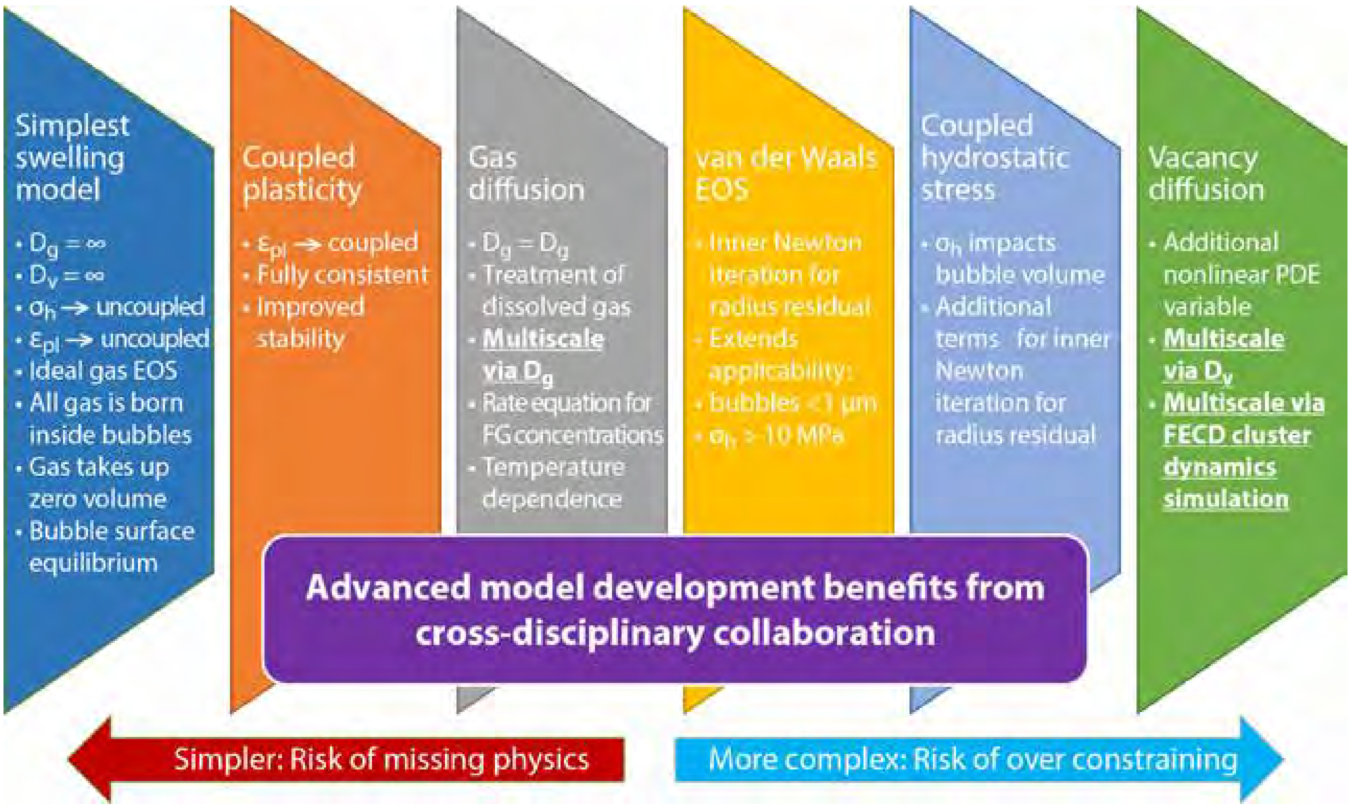
Collaboration between modelers from different backgrounds and across DOE programs has led to strides in multi-scale modeling capability and enhanced robustness of metallic fuel BISON simulations.

The BISON fuel performance code is well underway to being extended to model zirconium based metallic fuel. The challenges presented by the advanced U-Zr and U-Pu-Zr fuel types have resulted in a coupled thermo-mechanical-diffusion simulation required to describe historical observations, as well as predict future performance. Advances in fundamental models such as swelling, fission gas release, and zirconium diffusion will help provide the necessary foundation for advanced models such as cladding creep failures and fuel-cladding chemical interaction (FCCI). Benchmark comparisons to past postirradiation examinations (PIE)are utilized to assess model performance, with advanced calibration techniques playing a role when data is sparse or unavailable.

Project Description:

The objective of this research is to provide a code that can predict behavior of zirconium based metallic fuel for use in future advanced Gen IV commercial or test reactors such as the Versatile Test Reactor (VTR) currently under development. The favorable thermal conductivity and proven safety record of metallic fuel experienced during operation of experimental breeder reactor (EBR)-II lends U-Zr and U-Pu-Zr fuel naturally to the requirements of enhanced safety in next generation reactors. With FCCI as the primary limiting factor of metallic fuels, enhancements in understanding and modeling of the key phenomena involved in FCCI can help bolster the economic viability of metallic fuel through extending the fuel lifetime. Simultaneously, the coupled nature of nuclear fuel requires the implementation of many fundamental models in order to provide a baseline capability for advanced concepts and off-normal behavior.

Due to the sparse availability of modern data, historical EBR-II data is used as a primary source of information for model formulation and comparison. Previous codes have been limited to heavily empirical models due to limited resources, understanding, and data. By leveraging many of the capabilities within BISON, advanced models that capture complex behavior can be implemented in a viable way. In addition, on going work within the Advanced Fuels Campaign (AFC) has helped bolster the limited data set with new irradiations and higher fidelity data collection. Modern computational tools such as Bayesian calibration have been implemented to help bridge over limited data. In general, metallic fuel modeling in BISON aims to create a tool that is predictive in order to provide confidence for core designers in the safe, reliable, and efficient use of metallic nuclear fuel.



Accomplishments:

Leveraging several mechanical modeling experts within Los Alamos National Laboratory (LANL), and collaborating closely with members from the Department of Energy (DOE) Nuclear Energy Advanced Modeling and Simulation (NEAMS) program at Idaho National Laboratory (INL) and LANL, we were able to resolve previously unsolvable physics relating to swelling and fission gas release. This was achieved by both implementing physics that is more representative of the system, as well as improving the underlying thermo-mechanical framework in MOOSE.

In general, the primary metric of interest for core designers utilizing metallic fuel is the amount of creep swelling in the cladding due to the risk of flow channel blockage. Following previous assessments of BISON for metallic fuel from

Figure 1. Evolution of the coupled swelling, fission gas release, porosity, and creep models in BISON. The interplay between model complexity and ability to mechanistically predict behavior is complex, with strides towards pushing the envelope this year achieved through interactions with LANL and INL researchers in the AFC and NEAMS.

previous years, the current models utilized in BISON have some success in matching EBR-II data, but fail to capture cladding strain when varying rod parameters such as plenum height and pellet radius. Consequently, tuning parameters for the currently implemented empirical models that impact cladding strain (i.e., internal plenum pressure, fuel swelling, and cladding creep) will have limited applicability to off-normal fuel pin designs, high burnup fuel, and accident scenarios. In a move away from empirical formulations and mechanistically capture the behavior of the fuel, a new coupled fuel swelling, fission gas release, porosity, and creep model has been implemented in BISON. The model was employed following collaboration with a LANL team lead by Laurent Capolungo, and will allow BISON to capture the behavior of fuel outside the EBR-II operational experience envelope such as slotted or annular fuel [1,2]. In addition, fission gas release of the fuel will be calculated using physically based models rather than the currently implemented curve-fit from EBR-II data. The model will also benefit from continuing collaboration with NEAMS via lower-length scale information such as fission gas diffusivity and fundamental material parameters, leading to a truly multi-scale solution to fuel behavior.

In addition to the new coupled plastic strain model, a new multi-scale model for cladding creep has been implemented. Again, working with NEAMS, primarily through Laurent Capolungo and his team, a lower-length scale informed model has been implemented in BISON via a reduced-order model that rapidly calculates creep strain using an extensive set of previously calculated runs from the LANL code Visco-Plastic Self-Consistent (VCPS). Leveraging our BISON modeling experience with Capolungo's constituent modeling expertise allowed rapid implementation of creep models for HT-9 and SS316 cladding.

Finally, in order to be applicable to the greater scientific community, BISON needs to not only calculate values that match historical data, but it must be robust and easy to use for non-MOOSE experts. Part of this FY was spent ensuring the models that are in place that are used for metallic fuel modeling are appropriately implemented, verified, and documented. In addition, extensive framework development by Alex Lindsay and Daniel Schwen from INL lead to the implementation of Automatic Differentiation and Automatic Scaling, increasing the robustness of the MOOSE-based simulations when using complex models. We have converted all metallic fuel BISON models to the new system, allowing complex coupled behavior such as the fuel strain model discussed above to be feasible in production runs.

Citations:

[1.] Matthews, C., Bieberdorf, N., Capolungo, L., Andersson, D., Combined Visco-Plasticity and Swelling in Metallic Nuclear Fuel, LA-UR-19-25483, (2019).

[2.] Prakash, Naveen, Matthews, Christopher, Versino, Daniele, and Unal, Cetin. A general constitutive framework for thecombined creep, plasticity and swelling behavior of nuclear fuels in an implicit hypoelastic formulation. 2019. doi:10.2172/1493517.

CAPABILITY DEVELOPMENT

4.1 FINAL HALDEN RECOMMENDATIONS

Final Halden Recommendations

Principal Investigator: Colby Jensen, INL
Team Members/ Collaborators: Daniel Wachs, Nicolas Woolstenhulme, Nate Oldham, Steve Hayes and Kate Richardson (INL)

For decades, the Halden Boiling Water Reactor (HBWR) in Norway has been a key resource for assessing nuclear fuels and materials behavior to address performance issues and answer regulatory questions. Halden contributions to modern global Light Water Reactor (LWR) technologies have been expansive and crucial to an industry with decreasing financial resources and fewer available test facilities. This loss of HBWR represents a great challenge and opportunity for swift response by the Research and Development (R&D) community to fill the resulting capability gaps. A study was carried out with the primary objectives of identifying the core fuels and materials experimental capabilities available at the HBWR, assessing potential capability gaps specifically related to the Department of Energy (DOE) Accident Tolerant Fuels (ATF) program, and providing recommendations for a path forward for DOE ATF. The primary capabilities required for ATF are highly aligned with broader LWR testing needs.

Project Description:

The recent closure of the HBWR and growing demand for enhancing LWR fuel performance for safety and economic reasons has created an urgency to find a testing solution to support near-term R&D needs. Through the DOE ATF program, nuclear fuel vendors and industry have goals to support licensing and capitalizing on enhanced performance and safety of advanced fuels. First full batch reloads of some ATF fuel designs are planned to begin at commercial nuclear power plants in 2023 with goals for taking advantage of enhanced performance and safety by 2026. At the same time, the U.S. nuclear industry led by the Electric Power Research Institute (EPRI) is pushing a strategy towards burnup and enrichment extension targeting similar timeframes for full implementation. These major goals and other advanced LWR technology R&D require advanced testing capabilities like those found at the HBWR to enable efficient and successful resolutions. Therefore, the Advanced Fuels Campaign (AFC) was tasked to perform a detailed assessment of the potential implications of the HBWR closure and provide recommendations for paths forward. A report documenting an international assessment of irradiation testing gaps and capabilities related to the closure of the HBWR was finalized in December 2018. Since then, significant progress continues to be made in developing the given recommendations.

Accomplishments:

An international workshop was held in Idaho with attendance from more than twenty major nuclear organizations, including nuclear industry, regulators, and R&D facilities, both domestically and internationally, to provide consensus on: key testing capabilities used at the Halden reactor closure, a survey of capabilities at remaining available reactor facilities, and identification of capability gaps and potential resources that could fill those gaps (Figure 1).

- A report was drafted to document all agreed-upon findings and provide preliminary recommendations to fill capability gaps. The preliminary report was reviewed by major stakeholders with feedback provided to contribute to the generation of final recommendations and the results presented at multiple industry-R&D technical meetings.
- The preliminary recommendations were investigated in more detail to provide evaluations and final recommendations. These final recommendations included input from needs surveys requested from ATF fuel vendors and preliminary engineering evaluations. The final draft of the report was again reviewed by major

Figure 1. Group photo from international workshop held in Idaho to jointly assess testing capability gaps and international facility capabilities.

stakeholders and presented at several technical meetings. The final report was released in December 2019.

In summary, the unfortunate loss of the HWBR has further emphasized the important reawakening of DOE facilities under the ATF program to better utilize major existing infrastructure to meet the needs of the modern LWR industry, in addition to the advanced reactor community. Significant irradiation testing capabilities are available and being used at major U.S. testing facilities including Advanced Test reactor (ATR), High Flux Isotope Reactor (HFIR), the Transient Reactor Test Facility (TREAT), and Massachusetts Institute of Technology Reactor (MITR). Revitalized LWR testing capabilities at BR-2 and construction at the Jules Horowitz Reactor (JHR) are promising European complements to expanded DOE capabilities, while Russia maintains a full suite of LWR R&D capabilities. The ATR and TREAT Facility (and Material Fuels

Collaboration between modelers from different backgrounds and across DOE programs has led to strides in multi-scale modeling capability and enhanced robustness of metallic fuel BISON simulations.

Complex (MFC)) have the necessary key capabilities to absorb the breadth of the Halden mission gaps (though probably not the full testing capacity desired by the international community) and are conveniently situated within 25 km of each other. Still, additional scientific infrastructure is necessary to meet the needs of the ATF program and continue sustaining the current LWR fleet.

The primary testing gaps immediate partnership between DOE laboratory and Halden technical staff will ensure efficient development and maturation of these capabilities. The integrated technical program coordination for broader LWR fuels and materials irradiation testing (beyond ATF) provided by the Halden Reactor Project is a potential program gap that could be addressed through DOE and industry cooperation.

Primary recommendations made to fill existing capability gaps include: working with Halden staff to transfer technology and knowledge while (1) accelerating development of the loss of coolant accident (LOCA) device for TREAT, (2) adding new pressurized water loops into I-positions in ATR, (3) continuing to establish refabrication and reinstrumentation facilities at MFC, and (4) qualifying identified baseline instrumentation for use in DOE test reactors (see Figure 2). Overall, the assessment and recommendations have received very positive response and support by the U.S. nuclear community, with strong recognized importance to sustaining the current reactor fleet and future advanced LWR technology R&D.

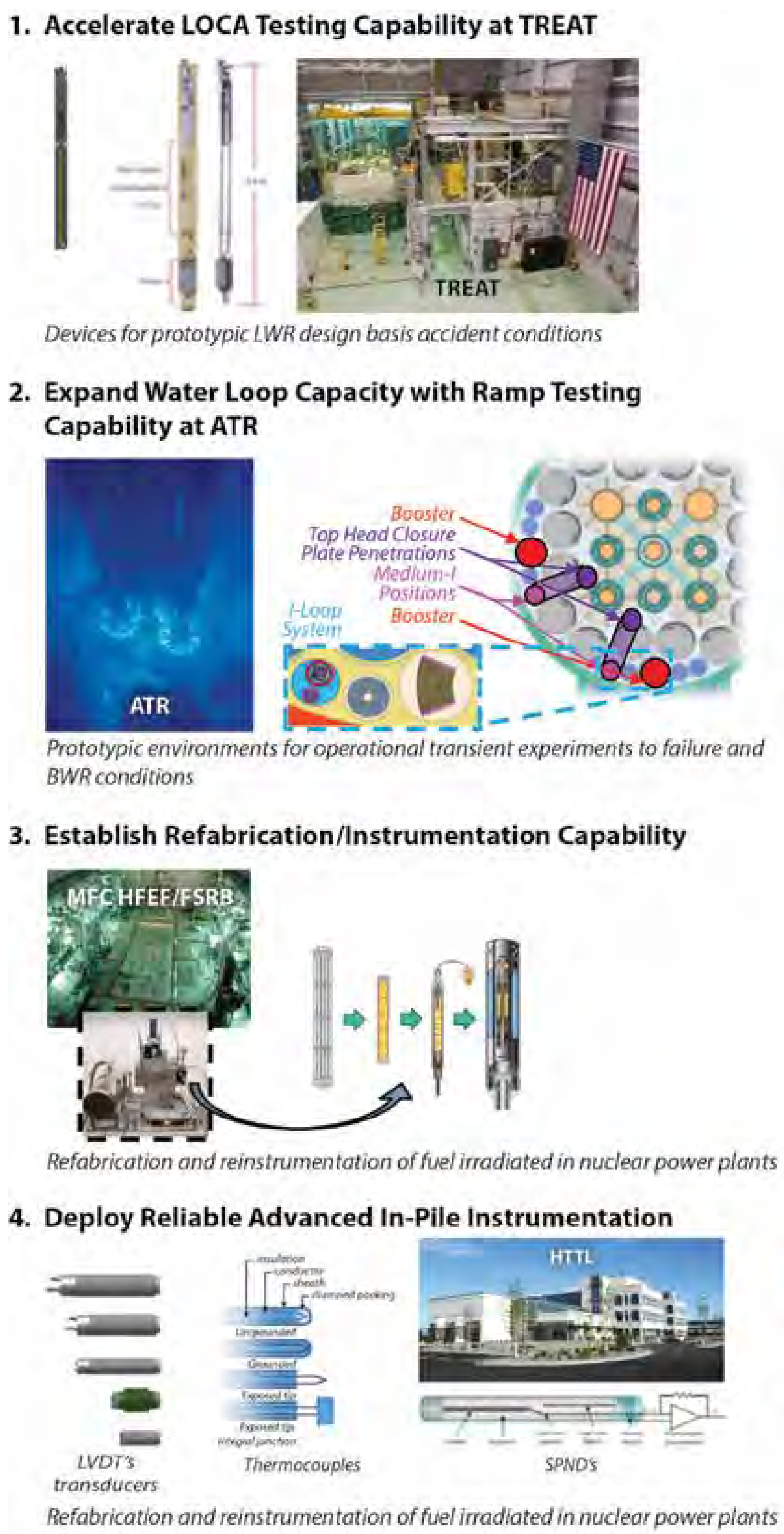


Figure 2. Summary of primary recommendations to fill irradiation testing gaps with the loss of the Halden research reactor.

APPENDIX

4.1 PUBLICATIONS

Author (s)	Title	Publication
N.M. Abdul-Jabbar, C.J. Grote and J.T. White	Assessment of thermophysical property characterization of MiniFuel scale geometries	LA-UR-19-24287 (2019), Los Alamos National Laboratory
N.M. Abdul-Jabbar and J.T. White	Processing and characterization of U_3Si_2 at the MiniFuel scale	LA-UR-19-26376 (2019), Los Alamos National Laboratory
C. Ang, D.Carpenter, K.Terrani and Y.Katoh	Preliminary Characterization and Projections of PVD Coatings On SiC Cladding for Light Water Reactors	In Proceedings of the 42nd International Conference on Advanced Ceramics and Composites, Ceramic Engineering and Science Proceedings (No. 3, p. 119). John Wiley & Sons. (2018, November)
C. Ang, C.Kemery and Y. Katoh	Electroplating chromium on CVD SiC and SiCf-SiC advanced cladding via PyC compatibility coating	Journal of Nuclear Materials 503 (2018): 245-249
E. Aydogan, C.J. Rietema, U. Carvajal-Nunez, S.C. Vogel, M. Li and S.A. Maloy	Effect of High-Density Nanoparticles on Recrystallization and Texture Evolution in Ferritic Alloys	Crystals 9(3) (2019)
E. Aydogan, J.S. Weaver, U. Carvajal-Nunez, M.M. Schneider, J.G. Gigax, D.L. Krumwiede, P. Hosemann, T.A. Saleh, N.A. Mara, D.T. Hoelzer, B. Hilton and S.A. Maloy	Response of 14YWT alloys under neutron irradiation: A complementary study on microstructure and mechanical properties	Acta Materialia 167 (2019) 181-196
G. Beausoleil, F. Cappia, L.A. Emerson, D. Murray, D. Sell, C. Christensen, A. Winston, D. Wahlquist, M. Bachhav and B. Miller	Separate Effects Testing in TREAT for ATF Fuels	Portions of this work were presented in a poster session at The Mineral, Metals, and Materials Society (TMS) 2019 annual meeting in San Antonio TX.
G.L. Beausoleil, G.L. Povirk and B.J. Curnutt	A Revised Capsule Design for the Accelerated Testing of Advanced Reactor Fuels.	Accepted to Nuclear Technology June 2019
M.T. Benson, Y. Xie, L. He, K.R. Tolman, J.A. King, J.M. Harp, R.D. Mariani, B.J. Hernandez, D.J. Murray and B.D. Miller	Microstructural characterization of annealed U-20Pu-10Zr-3.86Pd and U-20Pu-10Zr-3.86Pd-4.3Ln	Journal of Nuclear Materials 518 (2019), 287-297
M.T. Benson, Y. Xie, J.A. King, K.R. Tolman, R.D. Mariani, I. Charit, J. Zhang, M.P. Short, S. Choudhury, R. Khanal and N. Jerred	Characterization of U-10Zr-2Sn-2Sb and U-10Zr-2Sn-2Sb-4Ln to assess Sn+Sb as a mixed additive system to bind lanthanides	Journal of Nuclear Materials 510 (2018), 210-218

Author (s)	Title	Publication
J. D. Bess, N. E. Woolstenhulme, C. B. Jensen, J. R. Parry and C. M. Hill	Nuclear Characterization of a General-Purpose Instrumentation and Materials Testing Location in TREAT	Annals of Nuclear Energy, 124 (2018) 270-294
J. Bischoff et al.	AREVA NP's enhanced accident-tolerant fuel developments: Focus on Cr-coated M5 cladding	Nuclear Engineering and Technology 50(2) (2018), 223-228, ISSN 1738-5733
J.R. Burns, C.M. Petrie and D. Chandler	Burnup Calculation Methodology for a Small-Scale Fuel Irradiation Experiment in the High Flux Isotope Reactor (HFIR)	Transactions of the American Nuclear Society Vol. 120 (2019), p. 841-844
B.J. Curnutt and G.L. Beausoleil	The Fission Accelerated Steady State Test (FAST) – A Revised Capsule Design for the Accelerated Testing of Advanced Reactor Fuels	ANS proceedings June 2019
L. Czerniak, E. Lahoda, M. Sivack, J. Lyons, W. Byers, G.Wang, R. Oelrich, P. Xu and R. Lu	Development of Silicon Carbide as a Nuclear Fuel Cladding	Submitted to TopFuel2019, Seattle, WA, September, 2019
T. Dabney, G. Johnson, B. Maier, H. Yeom and K. Sridharan	Development of Cold Spray FeCrAl Coatings for Accident Tolerant Fuel	Proceedings of the 2019 American Nuclear Society Annual Conference, pp. 387-390, Minneapolis, MN, 2019
T. Dabney, G. Johnson, H. Yeom, B. Maier, J. Walters and K. Sridharan	Experimental Evaluation of Cold Spray FeCrAl Alloys and Coated Zirconium-alloy as Potential Accident Tolerant Fuel Cladding	Submitted for publication to Nuclear Materials and Energy, June 2019
F. Di Lemma et al.	Metallic Fast Reactor Separate Effect Studies for fuel safety	American Nuclear Society Winter Meeting 17-21 November 2019
F. Di Lemma et al.	Metallic Fast Reactor Separate Effect Studies for fuel safety	MiNES (Material in Nuclear Energy Systems) Baltimore 6-10 October 2019
B. Eftink, M. E. Quintana, T. J. Romero, C. Xu, T. A. Saleh and S. A. Maloy	Shear Punch Testing of Neutron Irradiated HT-9 and 14YWT	Submitted, Journal of Metals
F. Franceschini, V. Kucukboyaci, D. Stucker, E.J. Lahoda and Z. Karouta	Modeling of Westinghouse Advanced Fuels EnCore and ADOPT with the CASL Tools	Submitted to TopFuel2019, Seattle, WA, September, 2019

Author (s)	Title	Publication
D. Frazer, J.T. White and T.A. Saleh	Nanomechanical Properties of high uranium density fuels	NTRD-M3FT-19LA020201026, LA-UR-19-27315
B. Garrison,M. Howell, M. N. Cinbiz, M. Gussev and K. Linton	Length-Dependence of Severe Accident Test Station Integral Testing	Results from this work will be presented as a paper and oral presentation at the 2019 ANS Top Fuel conference in September 2019
J.G. Gigax, H. Kim, E. Aydogan, L.M. Price, X. Wang, S.A. Maloy, F.A. Garner and L. Shao	Impact of composition modification induced by ion beam Coulomb-drag effects on the nanoindentation hardness of HT9	Nuclear Instruments & Methods in Physics Research Section B-Beam Interactions with Materials and Atoms 444 (2019) 68-73
C. Grote	Assessment of Viability of Scaled Annular Pellet Fabrication Technologies	Assessment of Viability of Scaled Annular Pellet Fabrication Technologies
F.M. Heim, B.P. Croom, C.H. Bumgardner and X. Li	Scalable measurements of tow architecture variability in braided ceramic composite tubes	Journal of the American Ceramics Society 101 (2018), 4297-4307
F.M. Heim, B.P. Croom, C.H. Bumgardner and X. Li	Scalable measurements of tow architecture variability in braided ceramic composite tubes	Presentation delivered at MS&T18 Conference, 10/15/2018
Z. Jiao, S. Taller, K. Field, G. Yeli, M. P. Moody and G. S. Was	Microstructure Evolution of T91 Irradiated in the BOR-60 Fast Reactor	Journal of Nuclear Materials. 504 (2018) 122-134
B. Jolly, Y. Kato, R. Lowden, A. Nelson, A. Schumacher and K. Cooley	Development of Vapor Processing Capability for Advanced SiC/SiC Composites	Milestone Report: ORNL/SPR-2019/1125 (2019)
Y. P. Lin, R. M. Fawcett, S. S. DeSilva, D. R. Lutz, M. O. Yilmaz, P. Davis, R. A. Rand, P. E. Cantonwine, R. B. Rebak, R. Dunavant and N. Satterlee	Path Towards Industrialization of Enhanced Accident Tolerant Fuel	Paper A0141, TopFuel 2018, 30 September – 04 October 2018, Prague, European Nuclear Society https://www.euronuclear.org/.../topfuel/topfuel2018/.../TopFuel2018-A0141-fullpaper.pdf
R.Y. Lu, J. L. Walters and J. Qu	Assessment of Wear Coefficients of Accident Tolerance Fuel Claddings with Coated Materials	Submitted to TopFuel2019, Seattle, WA, September, 2019
J. L. Lyons, J. Walters, J. Romero, A. J. Mueller, J. Partezana, W. A. Byers, G. Wang, A. Parsi, H. Shah and R. Oelrich	Westinghouse Chromium-Coated Zirconium Alloy Cladding Development and Testing	Submitted to TopFuel2019, Seattle, WA, September, 2019

Author (s)	Title	Publication
B.R. Maier, H. Yeom, G. Johnson, T. Dabney, J. Hu, P. Baldo, M. Li and K. Sridharan	In Situ TEM Investigation of Irradiation-induced Defect Formation in Cold Spray Cr Coatings for Accident Tolerant Fuel Applications	Journal of Nuclear Materials, 512, 2018, 320
B.R. Maier, H. Yeom, G.O. Johnson, T. Dabney, J. Walters, J. Romero, H. Shah, P.Xu and K. Sridharan	Development of Cold Spray Coatings for Accident-Tolerant Fuel Cladding in Light Water Reactors	Journal of Minerals, Metals, and Materials Society (JOM), 70(2), (2018), 198
B. Maier, H. Yeom, G. Johnson, T. Dabney, J. Walters, P. Xu, J. Romero, H. Shah and K. Sridharan	Development of cold spray chromium coatings for improved accident tolerant zirconium-alloy cladding	Journal of Nuclear Materials, Volume 519, 2019, Pages 247-254, https://doi.org/10.1016/j.jnucmat.2019.03.039.
C.P. Massey, S.N. Dryepondt, P.D. Edmondson, M.G. Frith, K.C. Littrell, A. Kini, B. Gault, K.A. Terrani and S.J. Zinkle	Multiscale investigations of nano-precipitate nucleation, growth, and coarsening in annealed low-Cr oxide dispersion strengthened FeCrAl powder	Acta Materialia 166 (2019) 1-17
C.P. Massey, D.T. Hoelzer, R.L. Seibert, P.D. Edmondson, A. Kini, B. Gault, K.A. Terrani and S.J. Zinkle	Microstructural Evaluation of a Fe-12Cr Nanostructured Ferritic Alloy Designed for Impurity Sequestration	Journal of Nuclear Materials 522 (2019) 111-122
C. Matthews, N. Bieberdorf, L. Capol-ungo and D. Andersson	Combined Visco-Plasticity and Swelling in Metallic Nuclear Fuel	LA-UR-19-25483, (2019)
R. Oelrich, Z. Karoutas, P. Xu, J. Romero, H. Shah, J. Walters, E. Lahoda, M. Sivack, J. Lyons, L. Czerniak, F. Boylan, R. Āali, A. Bowman, M. Limbäck, A. Claisse and J.Wright	Overview Of Westinghouse Lead EnCore Accident Tolerant Fuel Program	Submitted to TopFuel2019, Seattle, WA, September, 2019
C.M. Petrie, J. Burns, R. Morris and K.A. Terrani	Miniature Fuel Irradiations in the High Flux Isotope Reactor	Proceedings of the 40th Enlarged Halden Programme Group Meeting, Lillehammer, Norway (2017)
C.M. Petrie, J. Burns, A. Raftery, A.T. Nelson and K.A. Terrani	Separate Effects Irradiation Testing of Miniature Fuel Specimens	Journal of Nuclear Materials, under review
C.M. Petrie, T. Koyanagi, R.H. Howard, K.G. Field, J.R. Burns and K.A. Terrani	Accelerated Irradiation Testing of Miniature Nuclear Fuel and Cladding Specimens	Proceedings of Top Fuel 2018 (2018), p. A0159

Author (s)	Title	Publication
N. Prakash, C. Matthews, D. Versino and C. Unal	A general constitutive framework for thecombined creep, plasticity and swelling behavior of nuclear fuels in an implicit hypoelastic formulation	doi:10.2172/1493517
R. B. Rebak, R. J. Blair and V. K. Gupta	Corrosion Evaluation of Iron-Chromium-Aluminum Alloys in Used Fuel Cooling Pools	Paper 12944, Corrosion/2019, NACE International, Houston, TX.
R. B. Rebak, V. K. Gupta, M. Drobnjak, Dd. J. Keck and E. J. Dolley	Overcoming Sensitization in Welds Using FeCrAl Alloys, Paper A0052	TopFuel 2018, 30 September – 04 October 2018, Prague, European Nuclear Society, https://www.euronuclear.org/events/topfuel/.../TopFuel2018-A0052-fullpaper.pdf
R. B. Rebak, V. K. Gupta and M. Larsen	Oxidation Characteristics of Two FeCrAl Alloys in Air and Steam from 800°C to 1300°C	JOM (2018) 70: 1484. https://doi.org/10.1007/s11837-018-2979-9
R. B. Rebak, S. Huang, M. Schuster, S. J. Buresh and E. J. Dolley	Fabrication and Mechanical Aspects of Using FeCrAl for Light Water Reactor Fuel Cladding	PVP2019-93128, PVP ASME Conference, San Antonio, TX, July 2019
R. B. Rebak, T. B. Jurewicz and E. J. Dolley	Assessing the Electrochemical Behavior of Ferritic FeCrAl in High Temperature Water	Paper A0053, TopFuel 2018, 30 September – 04 October 2018, Prague, European Nuclear Society, https://www.euronuclear.org/events/topfuel/.../TopFuel2018-A0053-fullpaper.pdf
R. B. Rebak, T. B. Jurewicz, and Y.-J. Kim	Electrochemical Behavior of Accident Tolerant Fuel Cladding Materials under Simulated Light Water Reactor Conditions	ASTM STP 1609, Advances in Electrochemical Techniques for Corrosion Monitoring, pp. 231-243 (2019)
M.D. Richardson, G.W.Helmreich, A.M.Raftery and A.T. Nelson	Resolution Capabilities for Measurement of Fuel Swelling using Tomography	ORNL/SPR-2019/1071, Oak Ridge National Laboratory, Oak Ridge, TN (2019)
R. S. Schley, D. H. Hurley, Z. Hua and S. J. Reese	In-Pile Instrument to Measure Changes in Grain Microstructure	Proceedings of the ANS NPIC & HMIT conference, Feb 9-14, 2019, Orlando FL

Author (s)	Title	Publication
M. Schuster, E. J. Dolley, T. B. Jurewicz and R. B. Rebak	Environmental Degradation Resistance of ATF FeCrAl Cladding Tube Specimens During the Fuel Cycle	19th International Conference on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactor, American Nuclear Society, Boston, MA, August 2019
R.L. Seibert, J.R. Burns, J.O. Kiggans and K.A. Terrani	Fabrication of Fully Ceramic Microencapsulated Compacts for Miniature Fuel Specimen Irradiation	To be published in Transactions of the American Nuclear Society Volume 21 (2019)
R.L. Seibert, J.O. Kiggans and K.A. Terrani	Fabrication of fully ceramic micro-encapsulated fuel pellets for HFIR irradiation	ORNL/SPR-2019/1133, Oak Ridge National Laboratory, April 2019
R.L. Seibert, K.A. Terrani, J.O. Kiggans, J.W. McMurray, B.C. Jolly, C.M. Petrie and A.T. Nelson	Fabrication and irradiation test plan for fully ceramic microencapsulated fuels	ORNL/TM-2019/1088, Oak Ridge National Laboratory, January 2019
E.S. Wood, C. Moczygemba, G. Robles, S. Nesloney, C. Grote, L. Cai, P. Xu and E. Lahoda	Fabrication and Steam Oxidation Testing of Alloyed Uranium Silicide Fuels	Submitted to TopFuel2019, Seattle, WA, September, 2019
S. Taller, Z. Jiao, K. Field and G. S. Was	Emulation of Fast Reactor Irradiated T91 using Dual Ion Beam Irradiation	Journal of Nuclear Materials, in press
T.L. Ulrich, S.C. Vogel, J.T. White, D.A. Andersson, E.S.Wood and T. M. Besmann	Temperature-Dependent Crystal Structure of U_3Si_2 by High Temperature Neutron Diffraction	Submitted to Acta Materialia
S.C. Vogel, T.L.Wilson and J.T. White	Crystal Structure Evolution of U-Si Nuclear Fuel Phases as a Function of Temperature	No. LA-UR-18-28584. Los Alamos National Lab.(LANL), Los Alamos, NM (United States), 2018
S.C. Vogel, T. L. Wilson, E. S. Wood, J.T. White and T. M. Besmann	Temperature-Dependent Crystal Structure of U_3Si_2 by High-Temperature Neutron Diffraction	Accepted for publication in Proceedings of Global/Top Fuel 2019, September 22-26, 2019, Seattle, WA
W.J. Williams, M.A., Okuniewski, L. Sudderth, D. Wachs and S. Van Den Berghe	Assessment of Swelling and Constituent Redistribution in Uranium- Zirconium Fuel Using Phenomena Identification and Ranking Tables (PIRT)	Ann. Nucl. Energy 136, 2019
W.J. Williams, M.A. Okuniewski, D. M. Wachs and S. Van Den Berghe	Fabrication and characterization of U-Zr foils for the DISECT project	Transactions of the American Nuclear Society. pp. 348–351, 2019

Author (s)	Title	Publication
T. L. Wilson, T.M. Besmann, S.C. Vogel and J.T. White	Crystal Structure Characterization of Uranium- Silicides Accident Tolerant Fuel by High Temperature Neutron Diffraction	Accepted for publication in Advances in X-ray Analysis, 63, (2019) Proceedings of the 68th Denver X-ray Conference, Lombard, Illinois, U.S.A. 8/5 – 8/9 2019
N. Woolstenhulme, C. Baker, J. Bess, D. Chapman, D. Dempsey, C. Hill, C. Jensen and S. Snow	New Capabilities for In-Pile Separate Effects Tests in TREAT	Transactions of the American Nuclear Society Summer Meeting 2018, Philadelphia, PA
N. Woolstenhulme, C. Baker, C. Jensen, D. Chapman, D. Imholte, N. Oldham, C. Hill and Spencer Snow	Development of Irradiation Test Devices for Transient Testing	Nuclear Technology, February 28, 2019
N. Woolstenhulme, J.Bess, P. Calderoni, B. Heidrich, D. Hurley, C.Jensen, R. Schley and K. Tsai	Overview of I2 Irradiation Deployment Activities in TREAT	Proceedings of the ANS annual meeting, Jun 9-13, 2019, Minne-apolis, MN
N. Woolstenhulme, A Fleming, T. Holshuh, Colby Jensen, D. Kamerman and D. Wachs	Core-to-specimen Energy Coupling Results of the First Modern Fueled Experiments in TREAT	Manuscript submitted Apr 2019, under review for publication in Annals of Nuclear Energy
N.R. Wozniak, J.T. White, B.P. Nolen and J.R. Wermer	Assessment of Feedstock Synthesis Routes for High Density Fuels	FT-19LA02020102, 22-February-2019
Y. Xie, M.T. Benson, L. He, J.A. King,R.D. Mariani and D.J. Murray	Diffusion behaviors between metallic fuel alloys with Pd addition and Fe	Journal of Nuclear Materials, 525 (2019), 111-124
H. Yeom, T. Dabney, G. Johnson, B. Maier and K. Sridharan	Oxidation of Cold Spray Cr Coatings in High Temperature Steam Environments	Submitted to Proceedings 2019 American Nuclear Society Annual Conference, pp. 383-386, Minne-apolis, MN, 2019
H. Yeom, B.R. Maier, G.O. Johnson, T. Dabney, J. Walters and K. Sridharan	Development of Cold Spray Process for Oxidation-Resistant FeCrAl and Mo Diffusion Barrier Coatings on Opti-mized ZIRLOTM	Journal of Nuclear Materials 507 (2018), 306
C. Zheng, J.-H. Ke, S.A. Maloy and D. Kaoumi	Correlation of in-situ transmission electron microscopy and microchemistry analysis of radiation-induced precipitation and segregation in ion irradiated advanced ferritic/ martensitic steels	Scripta Materialia 162 (2019) 460-464

4.2 FY-17 LEVEL 2 MILESTONES

Work Package Title	Site	Work Package Manager	Level 2 Milestone
Baseline Integral Tests in TREAT - INL	INL	Dempsey, Doug	Issue a summary technical report for conceptual design of MARCH-SERTTA
Campaign Management - INL	INL	Mai, Edward	Final Report to DOE on Halden Gap Analysis for ATF
Baseline Integral Tests in TREAT - INL	INL	Dempsey, Doug	Complete ATF-SETH testing in TREAT
ATF SiC Cladding and Core Component Development - ORNL	ORNL	Katoh, Yutai	Systematic Technology Evaluation Plan (STEP) for SiC as BWR channel box
Baseline Integral Tests in TREAT - INL	INL	Dempsey, Doug	Update ATF-3-1 RIA Test Plan
Advanced Ceramic Fuel Performance and Qualification - LANL	LANL	White, Josh	U_3Si_2 pellets fabricated with controlled microstructures ready for HFIR MiniFuel irradiation
AR Integral Transient Fuel Performance Testing - INL	INL	Emerson, Leigh Ann	Develop Test Plan for International Transient Testing Project for Fast Reactor Fuels
Campaign Management - INL	INL	Mai, Edward	Co-host (w/GAIN) Advanced Reactors Fuels Industry Workshop
Irradiation Testing Capability - INL	INL	Guymon, Helen	Conduct Super-SERTTA Preliminary (60%) Design Review
Fuel Fabrication - INL	INL	Fielding, Randy	Demonstration of fabrication of reduced diameter metallic fuel rodlets for FAST
Advanced Ceramic Fuel Development - LANL	LANL	White, Josh	Report on waterproofing UN studies
Evolution of Fuel Structure, Chemistry, & Properties following Irradiation - ORNL	ORNL	Nelson, Andy	Report on structure, chemistry, property assessment capabilities for PIE of MiniFuel geometries
Baseline Integral Tests in TREAT - INL	INL	Dempsey, Doug	Update ATF-3-2 LOCA Test Plan
Fuel Characterizations - INL	INL	Giglio, Jeff	Update of Metallic Fuels Handbook
Baseline Integral Tests in TREAT - INL	INL	Dempsey, Doug	Ready to perform first RIA in TREAT on unirradiated ATF (ATF-3-1)
Small Scale Mechanical and Ion Irradiation Testing of FeCrAl alloys - LANL	LANL	Maloy, Stuart	Issue report on advanced microscale testing on FeCrAl tubing and correlate measurements to macroscale testing results
Advanced FR Cladding Development - LANL	LANL	Maloy, Stuart	Report on and Extrude ODS tubes to dimensions needed for fueled ATR irradiations

Work Package Title	Site	Work Package Manager	Level 2 Milestone
Materials Modeling in Support of ATF Fuel Development - ORNL	ORNL	Wirth, Brian	Summary Report on Advanced Fuel Mechanics Modeling of SiC-clad Fuels
ATF-2 loop testing and redesign - INL	INL	Hoggard, Gary	Design approved for modifications to the ATF-2 test train for BWR rodlets and increased pin capacity
ATF PIE - INL	INL	Harp, Jason	Update Report on ATF-1 PIE
Advanced Fabrication Development - INL	INL	Fielding, Randy	Demonstrate machining capability for plutonium-bearing fuel alloys
PIE and Analyses - INL	INL	Harp, Jason	Issue PIE Report on legacy EBR-II and FFTF metallic fuel experiments
Fuel Modeling Support - LANL	LANL	Unal, Cetin	Issue Assessment Report of metallic fuel swelling model vs. data
CX PIE Infrastructure - INL	INL	Martinson, Steven	Demonstrate Property Measurement on Irradiated Fuel in IMCL
Irradiation of ATF Concepts in HFIR - ORNL	ORNL	Petrie, Christian	Monolithic ATF MiniFuel Sample Capsules Ready for HFIR Insertion

4.3 AFC NUCLEAR ENERGY UNIVERSITY PROJECTS (NEUP) GRANTS

Active Projects Awarded in 2015

Nuclear Energy University Project Grants

Lead University	Title	Principal Investigator
Massachusetts Institute of Technology	Multilayer Composite Fuel Cladding for LWR Performance Enhancement and Severe Accident Tolerance	Michael Short

Active Projects Awarded in 2016

Nuclear Energy University Project Grants

Lead University	Title	Principal Investigator
University South Carolina	Phase Equilibria and Thermochemistry of Advanced Fuels: Modeling Burnup Behavior	Theodore Besmann
North Carolina State University	Microstructure Experiments-Enabled MARMOT Simulations of SiC/SiC-based Accident Tolerant Nuclear Fuel System	Jacob Eapen
Purdue University	Microstructure, Thermal, and Mechanical Properties Relationships in U and UZr Alloys	Maria Okuniewski
Pennsylvania State University	A Coupled Experimental and Simulation Approach to Investigate the Impact of Grain Growth, Amorphization, and Grain Subdivision in Accident Tolerant U_3Si_2 Light Water Reactor Fuel	William Walters
University of Idaho	A Science Based Approach for Selecting Dopants in FCCI-Resistant Metallic Fuel Systems	Indrajit Charit
The Ohio State University	Alloying Agents to Stabilize Lanthanides Against Fuel Cladding Chemical Interaction: Tellurium and Antimony Studies	Christopher Taylor

Active Projects Awarded in 2017

Nuclear Energy University Project Grants

Lead University	Title	Principal Investigator
University of Wisconsin-Madison	Extreme Performance High Entropy Alloys (HEAs) Cladding for Fast Reactor Applications	Adrien Couet
University of Wisconsin-Madison	Critical Heat Flux Studies for Innovative Accident-Tolerant Fuel Cladding Surfaces	Michael Corradini
Colorado School of Mines	Development of Advanced High-Cr Ferritic/ Martensitic Steels	Kester Clarke
Massachusetts Institute of Technology	Determination of Critical Heat Flux and Leidenfrost Temperature on Candidate Accident Tolerant Fuel Materials	Matteo Bucci
University of New Mexico	An Experimental and Analytical Investigation into Critical Heat Flux (CHF) Implications for Accident Tolerant Fuel (ATF) Concepts	Anil Prinja
Missouri University of Science and Technology	Gamma-ray Computed and Emission Tomography for Pool-Side Fuel Characterization	Joseph Graham
Virginia Commonwealth University	Evaluation of Accident Tolerant Fuels Surface Characteristics in Critical Heat Flux Performance	Jessika Rojas
University of New Mexico	Nanostructured Composite Alloys for Extreme Environments	Osman Anderoglu

Active Projects Awarded in 2018

Nuclear Energy University Project Grants

Lead University	Title	Principal Investigator
University of California, Berkeley	Understanding of degradation of SiC/SiC materials in nuclear systems and development of mitigation strategies	Peter Hosemann
University of Minnesota, Twin Cities	Probabilistic Failure Criterion of SiC/SiC Composites Under Multi-Axial Loading	Jialiang Le
University of Wisconsin-Madison	Advanced Coating and Surface Modification Technologies for SiC-SiC Composite for Hydrothermal Corrosion Protection in LWR	Kumar Sridharan
University of Michigan	Mechanistic Understanding of Radiolytically Assisted Hydrothermal Corrosion of SiC in LWR Coolant Environments	Gary Was
University of Florida	Multiaxial Failure Envelopes and Uncertainty Quantification of Nuclear-Grade SiCf/SiC Woven Ceramic Matrix Tubular Composites	Ghatu Subhash
University of Notre Dame	Radiolytic Dissolution Rate of Silicon Carbide	David Bartels
University of South Carolina	Development of Multi-Axial Failure Criteria for Nuclear Grade SiCf-SiCm Composites	Xinyu Huang
University of California, Berkeley	Bridging the length scales on mechanical property evaluation	Peter Hosemann
Purdue University	Microstructure-Based Benchmarking for Nano/Microscale Tension and Ductility Testing of Irradiated Steels	Janelle Wharry
University of Utah	Benchmarking Microscale Ductility Measurements	Owen Kingstedt
University of Nebraska, Lincoln	Bridging microscale to macroscale mechanical property measurements and predication of performance limitation for FeCrAl alloys under extreme reactor applications	Jian Wang
Virginia Polytechnic Institute and State University	C-SiOC-SiC Coated Particle Fuels for Advanced Nuclear Reactors	Kathy Lu
University of Tennessee at Knoxville	A novel and flexible approach for converting LWR UNF fuel into forms that can be used to fuel a variety of Gen-IV reactors	Craig Barnes

Active Projects Awarded in 2019

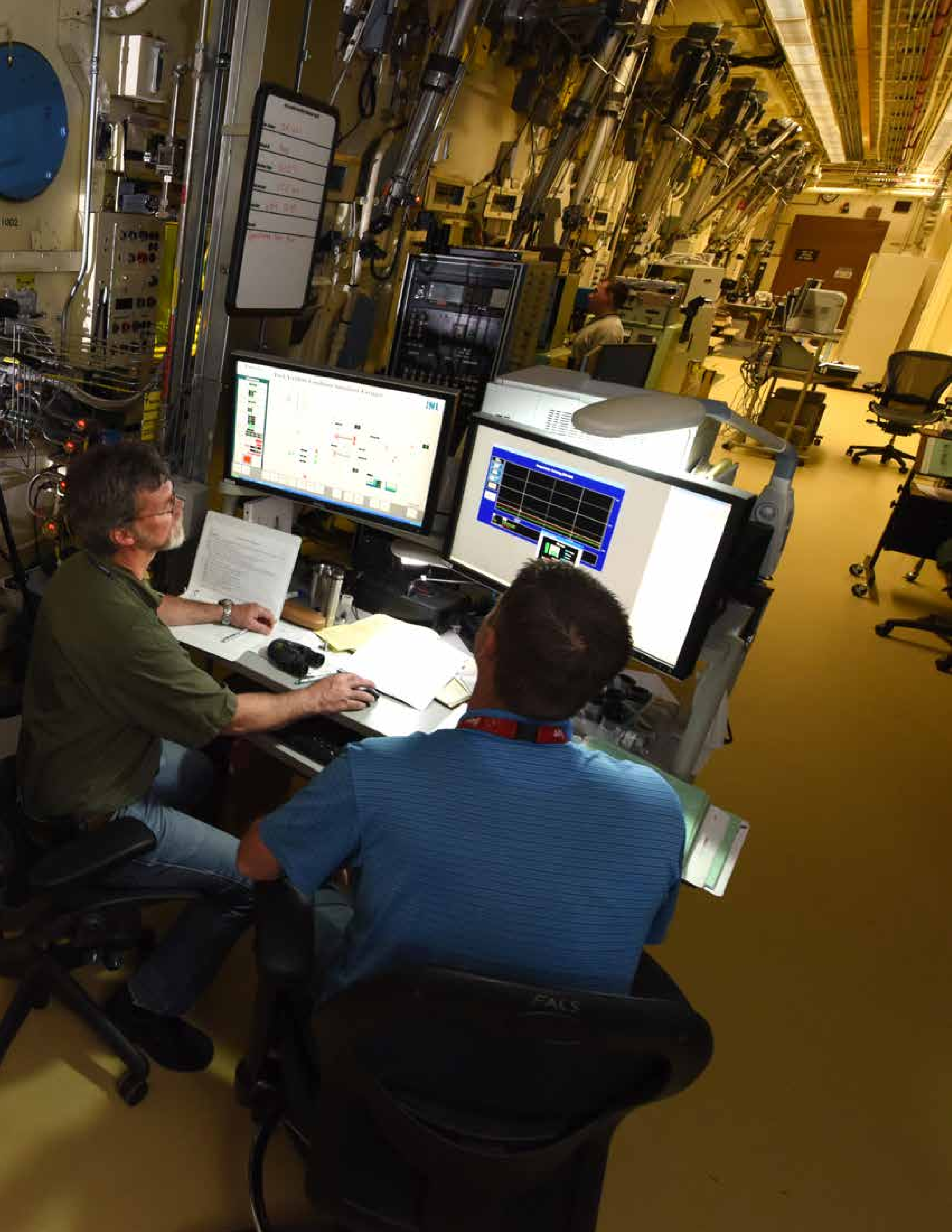
Nuclear Energy University Project Grants

Lead University	Title	Principal Investigator
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	Ji-Cheng Zhao
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 South Carolina	Remote laser based nondestructive evaluation for postirradiation examination of ATF cladding	Lingyu Yu
University of Tennessee at Knoxville	Radiation-Induced Swelling in Advanced Nuclear Fuel	Maik Lang
University of Minnesota, Twin Cities	High throughput assessment of creep behavior of advanced nuclear reactor structural alloys by nano/ microindentation	Nathan Mara

Active Projects Awarded in 2021

Nuclear Energy University Project Grants

Lead University	Title	Principle Investigator
University of California, Berkeley	Bridging the length scales on mechanical property evaluation	Peter Hosemann
Purdue University	Microstructure-Based Benchmarking for Nano/ Microscale Tension and Ductility Testing of Irradiated Steels	Janelle Wharry
University of Utah	Benchmarking Microscale Ductility Measurements	Owen Kingstedt
University of Nebraska, Lincoln	Bridging microscale to macroscale mechanical property measurements and predication of performance limitation for FeCrAl alloys under extreme reactor applications	Jian Wang
Virginia Polytechnic Institute and State University	C-SiOC-SiC Coated Particle Fuels for Advanced Nuclear Reactors	Kathy Lu
University of Tennessee at Knoxville	A novel and flexible approach for converting LWR UNF fuel into forms that can be used to fuel a variety of Gen-IV reactors	Craig Barnes



4.5 ACRONYMS

AFC .... Advanced Fuels Campaign

AGR .... Advanced Gas Reactor

AL .... Air Liquide

ALIP .... Annular Linear Induction Pumps

ANS .... American Nuclear Society

APMT .... Advanced Powder Metallurgy Tubing

APT .... Atom Probe Tomography

AR .... As-rolled

ARL .... Army Research Laboratory

ATF .... Accident Tolerant Fuel

ATR .... Advanced Test Reactor

BCC .... Body Centered Cubic

BDBA .... Beyond Design Basis Accident

BOL .... Beginning of Life

BSE .... Back-scattered Electron

BWR .... Boiling Water Reactor

CAES .... Center for Advanced Energy Studies

CASL .... Consortium for Advanced Simulation of LWRs

CDF .... Cumulative Distribution Function

CEA .... Commissariat à l' Énergie Atomique

CEE .... Committee for Examination of Experiments

CG .... Ceramic Grade

CHF .... Critical Heat flux

CIC .... Core Internal Change-out

CINDI .... Characterization-scale Instrumented Neutron Dose Irradiation

CP .... Cathcart-Pawel

CSV .... Comma Separated Value

CTRN .... Carbothermic Reduction and Nitriding

CVD .... Chemical Vapor Deposition

CVI .... Chemical Vapor Infiltration

DBA .... Design Basis Accident

DFT .... Density Functional Theory

DI .... Dual Ion

DISECT .... Disc Irradiation for Separate Effects Testing with Control of Temperature

DOE .... Department of Energy

dpa .... Displacements per Atom

DSC .... Differential Scanning Calorimetry

DZ .... Dark Zone

EATF .... Enhanced Accident Tolerant Fuel

EBR .... Experimental Breeder Reactor

EBSD .... Electron Backscatter Diffraction

ECF .... Energy Coupling Factor

ECR .... Equivalent Cladding Reacted

EDS .... Energy Dispersive X-Ray Spectroscopy

EDX .... Energy Dispersion X-Ray

EM .... Electromagnetic

EM2 .... Energy Multiplier Module

EPMA .... Electron Probe Micro Analysis

EPRI .... Electric Power Research Institute

F& OR .... Functional and Operational Requirements

FASB .... Fuels and Applied Science Building

FAST .... Fission Accelerated Steady-state Testing

FAST-OA .... Fission Accelerated Steady-state Testing – Outer A position

FAST-SI .... Fission Accelerated Steady-state Testing – Small I position

FCC .... Face Centered Cubic

FCCI .... Fuel Cladding Chemical Interaction

FCM .... Fully Ceramic Microencapsulated

FCRD .... Fuel Cycle Research and Development

FFF .... Free Form Fibers

FFRD .... Fuel Fragmentation, Relocation, and Dispersal

FFTF .... Fast Flux Test Facility

FIMA .... Fission of Initial Metal Atoms

FM .... Ferritic-Martensitic

FMMS .... Fuel Motion Monitoring System

FOA .... Funding Opportunity Announcement

FOM .... Figure of Merit

FRL .... Fuels Research Laboratory

FY .... Fiscal Year

GA .... General Atomics

GAIN .... Gateway for Accelerated Innovation in Nuclear

GE .... General Electric

GERC .... General Electric Global Research

GFY .... Government Fiscal Year

GNF .... Global Nuclear Fuels

HBR .... HB Robinson

HBS .... High Burnup Structure

HBWR .... Halden Boiling Water Reactor

HD .... High Definition

HEA .... High Entropy Alloys

HEU .... Highly Enriched Uranium

HFEF .... Hot Fuel Examination Facility

HFIR .... High Flux Isotope Reactor

HHF .... High Heat Flux

HRR .... Horn Rapids Road
HTO .... High Temperature Oxidation
H/U .... Hydrogen-To-Uranium
I2 .... Inpile Instrumentation
IAC .... Industry Advisory Committee
IFE .... Institute for Energy Technology
IFEL .... Irradiated Fuels Examination Laboratory
IMCL .... Irradiated Materials Characterization Laboratory
INL .... Idaho National Laboratory
IRT .... Integral Recycling Test
IWD .... Work Authorization Documents
KAERI .... Korean Atomic Energy Research Institute
LAMDA .... Low Activation Materials Development & Analysis
LANL .... Los Alamos National Laboratory
LANSCE .... Los Alamos Neutron Science Center
LFA .... Low Heat Flux
LHF .... Laser Flash Analysis
LHGR .... Linear Heat Generation
LOCA .... Loss of Coolant Accident
LOOP .... Loss of Offsite Power
LTA .... Lead Test Assemblies
LTR .... Lead Test Rod
LTR .... Licensing Topical Report
LWR .... Light Water Reactor
MARCH .... Minimal Activation Retrievable Capsule Holder
MCNP5 .... Monte Carlo N-Particle 5
MFC .... Materials and Fuels Complex
MFCs .... Mass Flow Controllers

MIT .... Massachusetts Institute of Technology
MITR .... Massachusetts Institute of Technology Reactor
nCT .... Neutron Computed Tomography
NE .... Nuclear Energy
NEA .... Nuclear Energy Agency
NEAMS .... Nuclear Energy Advanced Modeling and Simulation
NEI .... Nuclear Energy Institute
NEUP .... Nuclear Energy University Project
NFA .... Nanostructured Ferritic Alloys
NFD .... Nippon Nuclear Fuel Development
NI .... Nanoindentation
NLC .... Sodium Test Loop Cartridge
NNL .... National Nuclear Laboratory
NRAD .... Neutron Radiography Reactor
NRC .... Nuclear Regulatory Commission
NSUF .... Nuclear Science User Facilities
NTRD .... Nuclear Technology Research & Development
OA .... Outboard A
OD .... Outer Diameter
ODS .... Oxide Dispersion Strengthened
OFRAC .... Oak Ridge Fast Reactor Advanced Fuel Cladding
ONE .... Office of Nuclear Energy
OP .... Operation Procedure
ORNL .... Oak Ridge National Laboratory
PCI .... Pellet Clad Interactions
PCS .... Primary Coolant System
PCT .... Peak Clad Temperature
PCT .... Pressure Composition Temperature

PFIB .... Plasma Focused Ion Beam

PI .... Principal Investigator

PIE .... Postirradiation Examination

PIRT .... Phenomena Identification and Ranking Tables

PNNL .... Pacific Northwest National Laboratory

PQD .... Post-Quench Ductility

PVD .... Physical Vapor Deposited

PWR .... Pressurized Water Reactor

QA .... Quality Assurance

R&D .... Research & Development

RCT .... Ring Compression Testing

RD&D .... Research, Development, and Demonstration

RIA .... Reactivity-Initiated Accident

RITA .... Resonance Inspection Techniques and Analysis

RPI .... Rensselaer Polytechnic Institute

RTE .... Rapid Turnaround Experiment

RUS .... Resonant Ultrasound Spectroscopy

RUSL .... Resonant Ultrasonic Spectroscopy-Laser

SA .... Sensitivity Analysis

SATS .... Severe Accident Test Station

SCK·KEN .... Studie Centrum Voor Kemenergie: Belgian Nuclear Research Centre

SEM .... Scanning Electron Microscopy

SETH .... Separate Effects Test Holder

SFR .... Sodium Fast Reactor

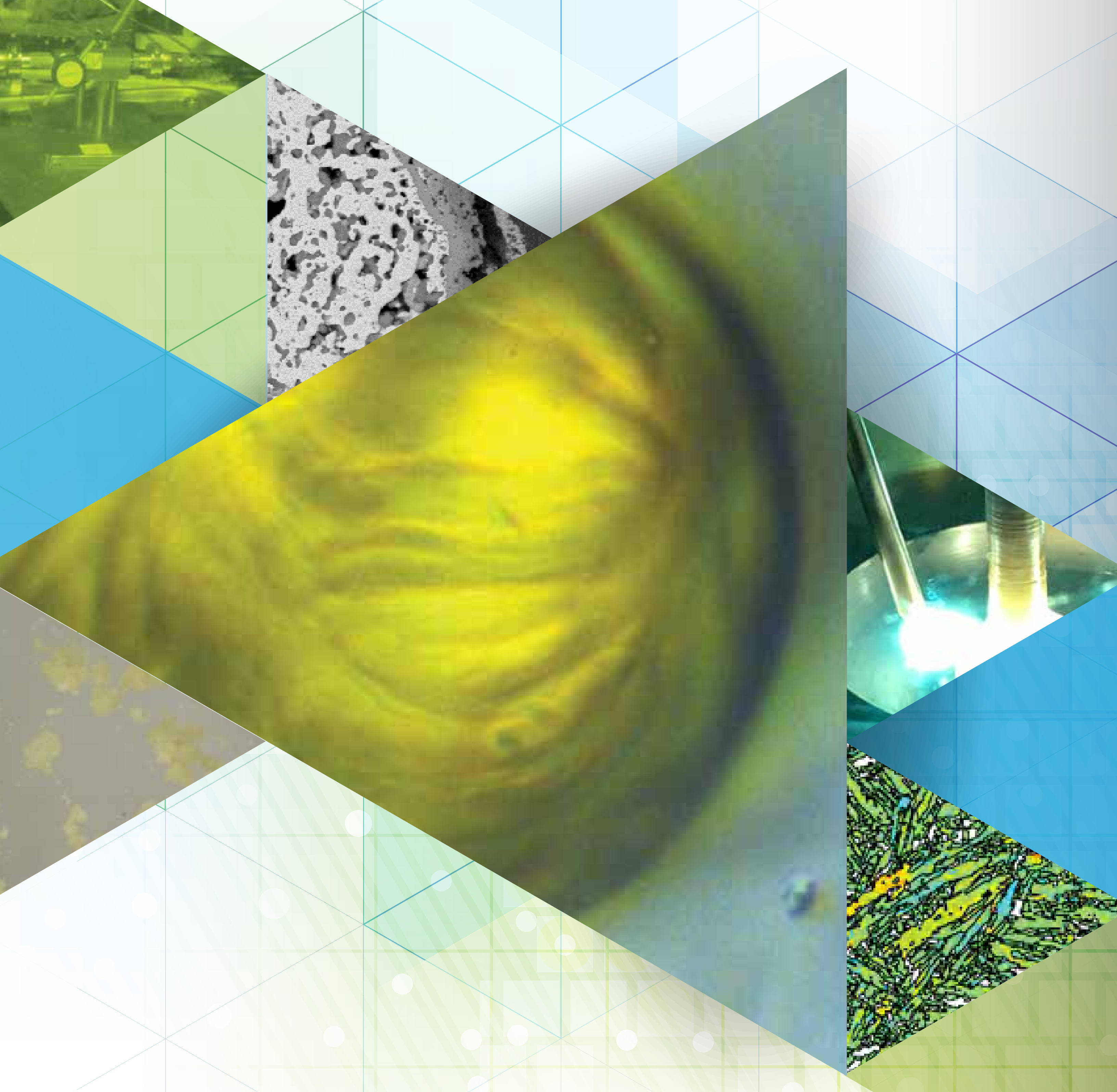
SiC .... Silicon Carbide

SPS .... Spark Plasma Sintering

SS .... Stainless Steel

SSMT .... Small Scale Mechanical Testing

STEM	Scanning Transmission Electron Microscopy
TAMU	Texas A&M University
TEM	Transmission Electron Microscopy
TGA	Thermogravimetric Analysis
TMS	The Mineral, Metal and Materials Society
TMT	Thermomechanical Treatments
TREAT	Transient Reactor Test Facility
TRISO	Tri-structural Isotropic
UN	Uranium Nitride
UofI	University of Idaho
UQ	Uncertainty Quantification
U.S.	United States
USC	University of South Carolina
UTA	University of Texas-Austin
UTK	University of Tennessee-Knoxville
VCPS	Visco-plastic Self Consistent
VERA-CS	Virtual Environment for Reactor Applications – Core Simulator
VTR	Versatile Test Reactor
Wd	Watt-days
Wd/MTU	Watt-days per Metric ton Uranium
WEC	Westinghouse Electric Company
WQ	Water Quenching
XCT	X-ray Computed Tomography
XRD	X-Ray Diffraction
YH2	Yttrium Dihydride
YN	Yttrium Mononitride



AFC Advanced Fuels Campaign

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