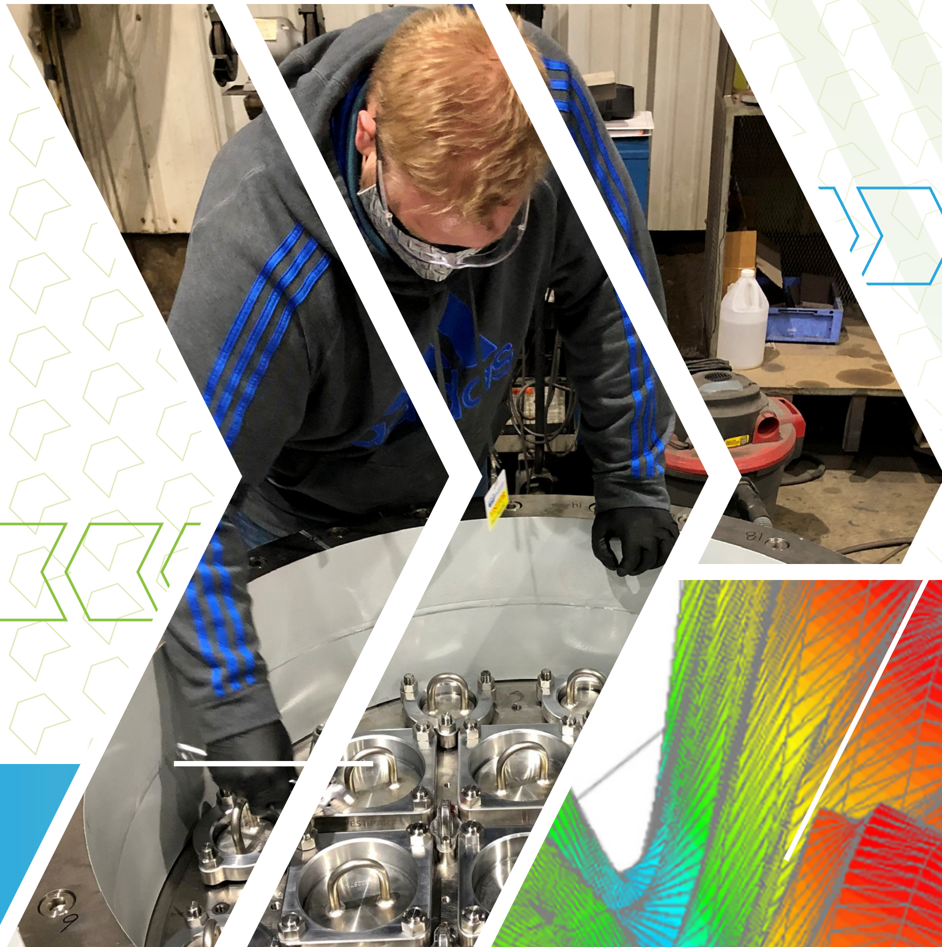
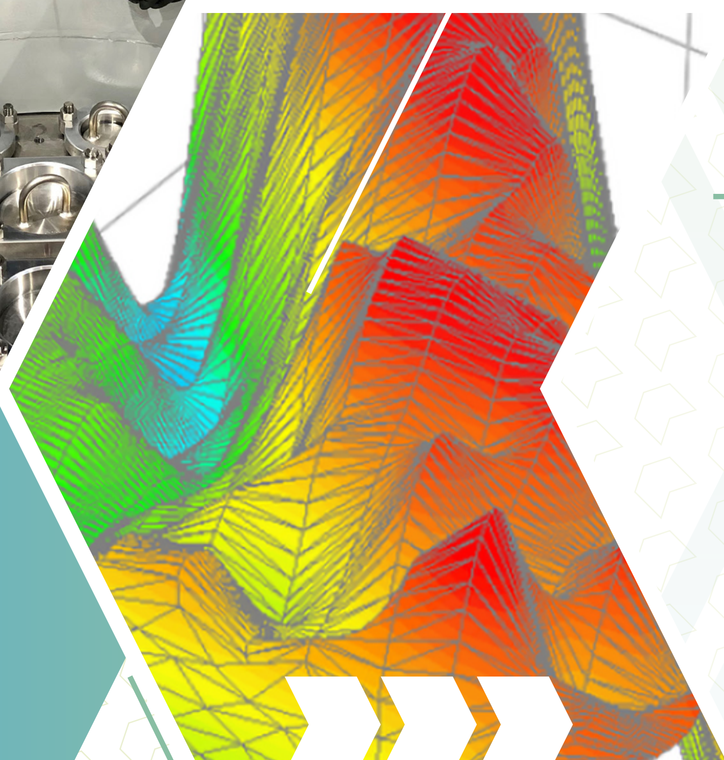
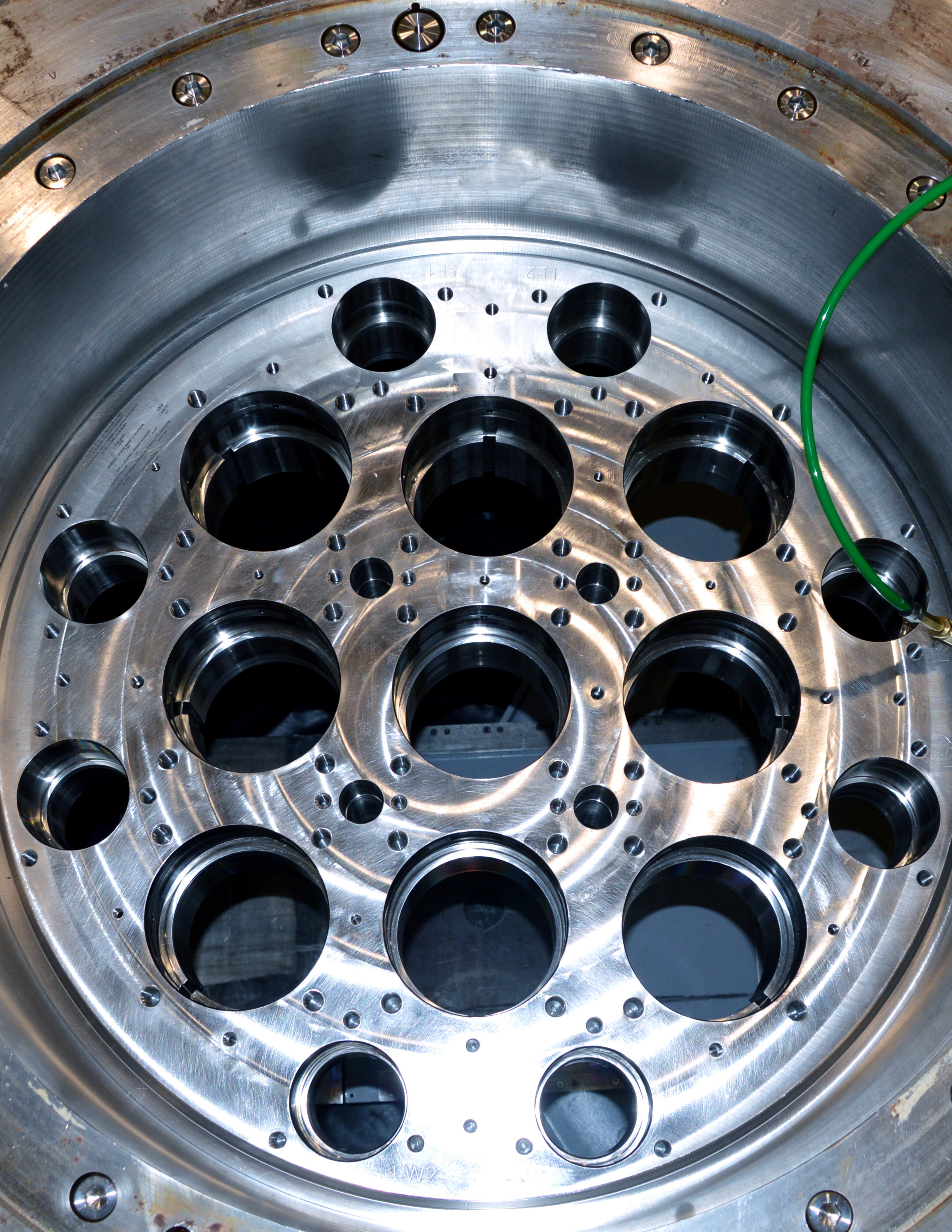




ADVANCED FUELS CAMPAIGN
2021 Accomplishments

AFC Advanced Fuels Campaign



Nuclear Fuels Cycle & Supply Chain

Advanced Fuels Campaign
2021 Accomplishments

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TABLE OF CONTENTS

1 ADVANCED FUELS CAMPAIGN INTEGRATION

2 ADVANCED LIGHT WATER REACTOR FUELS OVERVIEW

TABLE OF CONTENTS - *CONTINUED*

3 AR FUELS

4 CAPABILITY DEVELOPMENT

5 APPENDIX


60 TON
WT. 840 LBS
25 TON MAX.LOAD
TREAT Irradiation Environments
Overview
MARCH System
Experiment Modules
Future Capabilities
4

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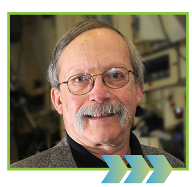

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1.2 FROM THE DIRECTOR

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It's a great pleasure to open the 2021 Advanced Fuels Campaign (AFC) Accomplishments Report. While a report of this type is, by necessity, unable to comprehensively capture all the accomplishments made by this long standing, nationally relevant research and development program, it is my hope that it reflects the quality and talent of the remarkable people that staff it and the world class scientific facilities they wield. Even in the face of a pandemic that curtailed many of our personal and collective routines this year, impactful research continued and the summaries that follow in this report provide clear evidence.

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, cladding materials, and associated technologies with the potential to improve the performance and enhance the 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.

This methodology is built on a foundation of 'analytical experiments' that merge advanced modeling and simulation with modern data-rich experimental methods to investigate the dominant physical phenomena associated with a given fuel system as necessary to drive its development to completion. AFC researchers utilize and evolve the nation's most important nuclear materials research capabilities, a comprehensive nuclear fuels testbed that spans multiple national laboratories, to serve this mission. These capabilities range from the world's best nuclear materials test reactors (ATR, HFIR, TREAT, and, hopefully in the future, VTR) to state-of-the-art materials science instruments and manufacturing technologies. AFC works in close partnership with the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program to implement advanced tools for modeling and simulation by collaborating on development of mechanistic fuel behavior models and conducting experimental studies that inform and support assessment of its most advanced tools. AFC will also work closely with the Advanced Sensors and Instruments (ASI) program to develop and deploy new sensors and instruments in experiments that will allow researchers to directly observe the evolution of materials in representative severe environments. This represents a massive leap beyond the irradiate and observe model used historically to develop and qualify nuclear fuels (colloquially referred to as 'cook and look').

As an early adopter of these methodologies and tools, AFC is applying them to a variety of fuel development and qualification initiatives that crosscut the national nuclear research enterprise. 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 to design basis and severe accident scenarios. This effort is expected to culminate with implementation of batch reloads of one or more near-term ATF concept(s) in commercial reactor(s) in the mid-2020's;

2. collaborate with industry and regulatory community to perform the R&D necessary to support extending the burnup of current commercial LWR fuels from 62 to 75 GWd/MTU by 2026;

3. lead research and development on innovative fuel and cladding technologies with applications to future advanced reactors, especially metallic fuels for fast-spectrum reactors, including reactors that utilize both once-through and recycle approaches to the fuel cycle;

4. 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 R&D infrastructure necessary to accelerate the development of new fuel concepts; and

5. collaborate with NEAMS and ASI on the execution of 'analytical experiments' that result in mechanistic fuel behavior and subsequent development and validation of multi-scale, multi-physics. Ultimately leading to increasingly predictive fuel performance models and codes.

Considering these critical missions, it is clear that the AFC program is a foundational component of the nuclear energy community's response to increasing clean energy demand.

Sincerely,

Dan Wachs


EDWARDS
HZ-PI-007
B1

ADVANCED LWR FUELS OVERVIEW

National LWR Testbed

Principal Investigator: Nick Woolstenhulme
Team Members/ Collaborator: Dave Kamerman, Nate Oldham, Colby Jensen, Clint Baker, Chris Petrie, Annabelle LeCoq, Kory Linton, Gordon Kohse, and David Carpenter

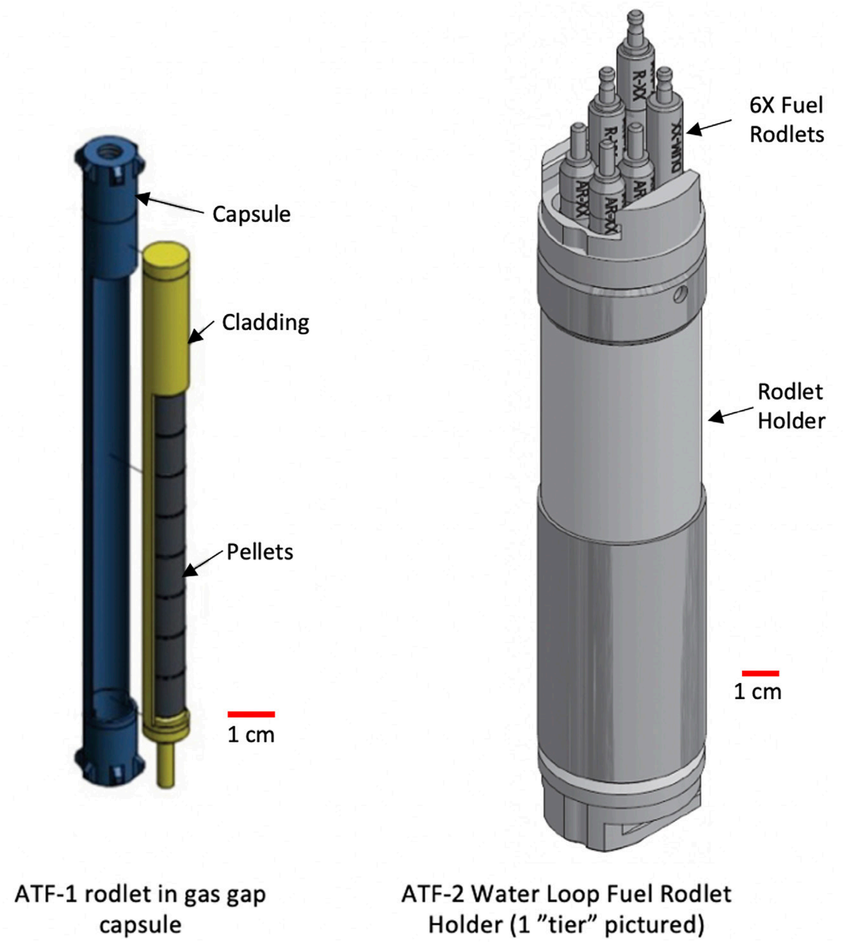


Figure 1. ATF-1 and ATF-2 rodlet holder designs

Approximately 40 years ago there were at least 9 special purpose material test reactors in the United States with relevant environments and capabilities for irradiation testing Light Water Reactor (LWR) fuels and materials. Well featured hot cell Post Irradiation Examination (PIE) capabilities also accompanied these reactors' campuses. Less than half of these reactors continue to operate today and general inactivity in LWR research areas over the preceding couple of decades caused atrophy of their LWR capabilities. During this time, LWR testing needs were met by a few international facilities which gradually consolidated over the years until only the Halden Boiling Water Reactor (HBWR) remained. The accident at Fukushima combined with economic challenges facing LWR plants drove a resurgence of interest in LWR fuel technology development for Accident Tolerant Fuels (ATF), High Burnup (HBu) license extensions, and other data needs drove revitalization of some LWR testing infrastructure using the United States' four remaining fuels and materials test reactors. The more recent and unexpected closure of HBWR obviated other capabilities that were needed and thus started a second wave of capability development is now underway to reclaim these gaps. Lastly, strategic planning has been undertaken so that the LWR testbed remains a long-lived national capability with forethought in developing its capabilities toward the next generations of water-cooled reactor fuel technology.

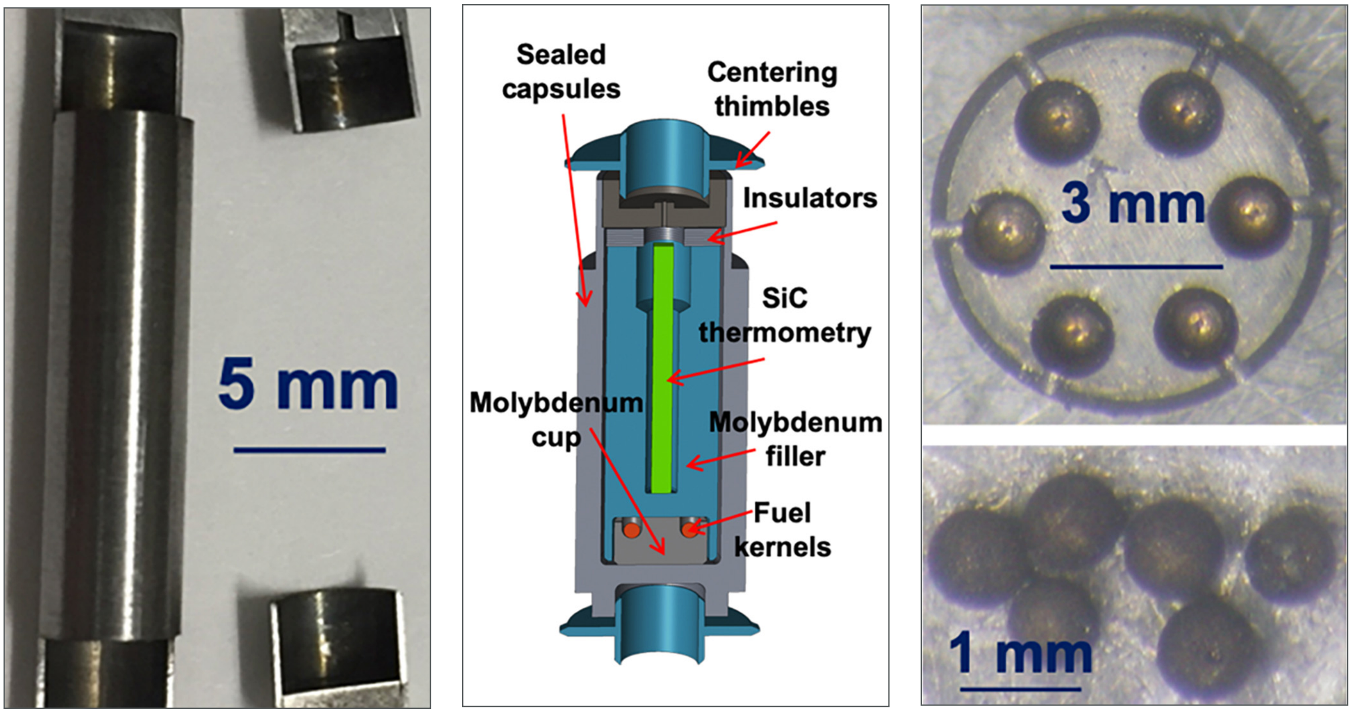


Figure 2. HFIR capsule type irradiations used to support LWR testing

Accomplishments

The initial era of revitalizing the LWR testbed focused on emerging needs for testing ATF technologies. The four remaining fuels and materials test reactors in the United States, namely the Advanced Test Reactor (ATR), Transient Reactor Test facility (TREAT), High Flux Isotope Reactor (HFIR), and Massachusetts Institute of Technology Reactor (MITR), all established new capabilities to assist in this cause.

Creation of the resulting infrastructure has been a major success whose coming of age now warrants a focused effort to brand and market it as an integrated LWR testbed so that users may better utilize its competencies. The Department of Energy's Advanced Fuels Campaign (AFC) recently established the LWR Irradiation Testing Expert Group (ITEG) with experts from each of these facilities to help facilitate, develop, and maintain the

Figure 3. Images of MITR's water loop and cladding test holder

testbed. A quick summary of the accomplishments from this first era are listed below:

- Irradiation testing of numerous ATF fuel and cladding concepts using insert gas capsule (referred to as "ATF-1") and pressurized water loop testing of several fuel rodlets (referred to as "ATF-2") in the ATR. These capabilities were commissioned for ATF program, have tested fuel technologies from all major US fuel vendors, and continue to be used today with new test planned for international partners as well. (Figure 1)
- Accelerated, separate-effects irradiation capsules for testing small specimens of ATF fuels and materials in the HFIR. These capabilities continue to be sought by the community owing to HFIR's unique high fluence capabilities. (Figure 2)
- Irradiation testing of cladding tubes to assess corrosion of ATF cladding concepts using MITR's pressurized water loop. This loop capability continues to be used and is now the subject of a Nuclear Energy University Partnership (NEUP) call for proposals to continue with further ATF cladding behavior research. (Figure 3)
- Transient testing of ATF fuel and cladding concepts in inert gas and water capsules in TREAT under conditions simulating Reactivity-Initiated Accidents (RIA). This effort not only entailed development of entirely new irradiation capabilities but was crucial connected to partner project which refurbished and restarted the TREAT reactor which

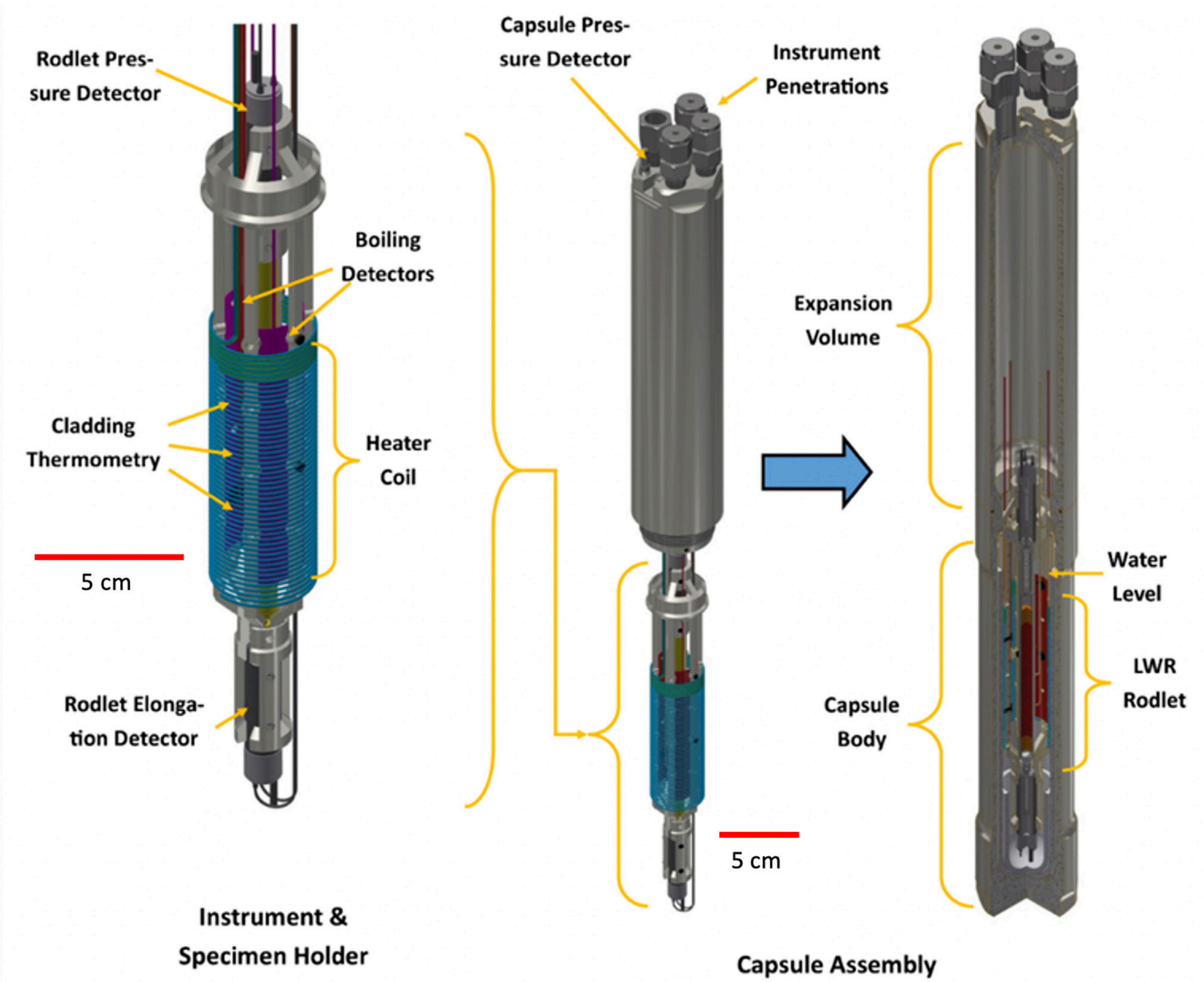


Figure 4. TREAT water capsule for RIA testing

had not operated for decades. The RIA capsule capability has enabled a current project in partnership with international partners. (Figure 4)

- Numerous enhancements in PIE equipment, hot cell performance testing, and related logistic capabilities.
- Amidst these notable developments, the unexpected closure of HBWR created further capability gaps that were needed including in-pile Loss of Coolant Accident (LOCA) testing, power ramp testing, the ability to outfit previously irradiated

With less than half of the reactors that were available in the past, the United States LWR Testbed stands poised to support the same diversity of test conditions and data needs to support development of advanced LWR fuels and materials.

rod segments with crucial instrumentation. Loss of the HBWR also caused a general reduction of test throughput for various needs. As a result, a second wave of capability development is now well underway to reclaim the HBWR gaps within domestic infrastructure including:

- A reflector-based loop at ATR ("I-Loop") to enable tests with power ramping and coolant voiding.
- Blowdown capsule for LOCA testing in TREAT.
- Hot cell-based competencies for instrumenting irradiated rod segments.

As soon as the HBWR gaps are filled, the testbed should be able to support full deployment of "evolutionary ATF" (coated-Zr, doped-UO_2) and HBu license extensions The testbed, however, will need to be maintained to support post-deployment optimization of technologies while supporting development of truly advanced fuel/cladding designs (e.g., SiC/SiC composite cladding, high-density/composite fuels) and other advancements needed to maintain the vitality of water-cooled reactors (e.g., advanced SMR fuels). Sustaining and enhancing the LWR testbed investment will require steadfastness and strategic growth to ensure that data capabilities are made available and retained for future LWR technology developments. In this regard, further opportunities have also been identified as candidates to be considered for their role in enhancing data throughput and technology developments including:

- Accelerated fuel testing capsules at ATR and expanded capabilities at HFIR
- Installation of a second cladding corrosion water test loop at MITR
- Flowing water loop in TREAT
- Expansion of ATR capacity to 2 or more I-Loops
- Thermosyphon for LWR fuel/material irradiations at high flux with representative coolant exposure in HFIR

Prioritization, strategic planning, and commissioning of the necessary capability development projects, along with their integration into the broader testbed and transition to programmatic data production, remains a prominent effort for AFC.

Figure 5. Overview of the US LWR testbed facilities

Simulating High Burnup Structures on CeO_2 and UO_2 Using Field Assisted Sintering

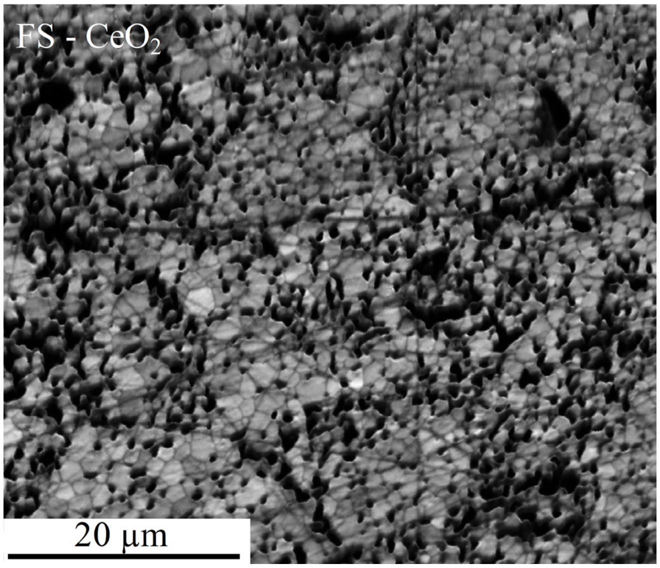
Principal Investigator: Rubens Ingraci Neto
Team Members/ Collaborator: Darrin Byler, Kenneth McClellan, and Erofili Kardoulaki

The capability to fabricate sim-HBS through FS was established at LANL and it was demonstrated by producing surrogate pellets with sim HBS.

Increasing the nuclear fuel burnups beyond 62 GWd·MTU^{-1} would reduce the fuel cycle costs and could improve the performance of nuclear power plants. However, at high burnups, nuclear fuels suffer an intense microstructure transformation that needs to be well comprehended to ensure the reliability and the safety of the operations of nuclear reactors. Most of the studies on high burnup structures (HBS) uses data collected from samples irradiated on commercial and test reactors. These studies, although highly valuable, are very expensive and also complex due to the radiotoxicity of the irradiated fuels [1, 2, 3, 4]. Besides, the time frame of these tests is incompatible with the pressing needs for understanding high burnup performance. Simulated HBS (sim-HBS), which reproduces the small grains sizes and high porosity of HBS in non-irradiated materials, can help with these efforts

Project Description:

To fabricate sim-HBS of ceramic fuels and their surrogates, two requisites need to be satisfied: (i) obtain nanopowder of the desired ceramic material, and (ii) apply a sintering method that is able to produce pellets with nanograined microstructure, resembling HBS. Field assisted sintering (FAS) has been used to fabricate sim-HBS around the world [5-11]. The current work at Los Alamos National Laboratory (LANL) aimed to produce pellets with sim HBS of CeO_2 and UO_2 through flash sintering (FS), using commercial CeO_2 nanopowder and in-house synthesized CeO_2 and UO_2 nanopowders. FS is a FAS technique that can sinter ceramics in a few seconds at low temperatures using electric fields, thus it can retain nanograined microstructures. Because the FS is faster than other FAS methods and the FS apparatus is not as complex as other FAS equipment, FS is an economic and fast alternative to produce pellets with different microstructures. At high burnups, UO_2 fuel restructures to form grain sizes varying from 0.59–0.71 µm on the surface layer (r/r0 ~1.0–0.97, ~150 µm thick), grains from 5.64–7.8 µm at the mid-radial region (r/r0 ~0.97-0.45), and smaller grains, 7.71–3.98 µm, at the fuel disc center (r/r0 ~ 0.45-0.25) [3]. The density of UO_2 also changes drastically with burnup level. The density of UO_2 fuel pellets can reduce to 78.9% of their theoretical density (TD) and their porosity can increase to 17.5 vol.% when the burnup reaches 90 GWd·MTU^{-1} [12]. Therefore, sim HBS must have similar densities and porosities. Pellets with sim HBS can be used to improve the mechanistic models that describe how thermophysical and mechanical properties change as a function of the microstructure of the material.



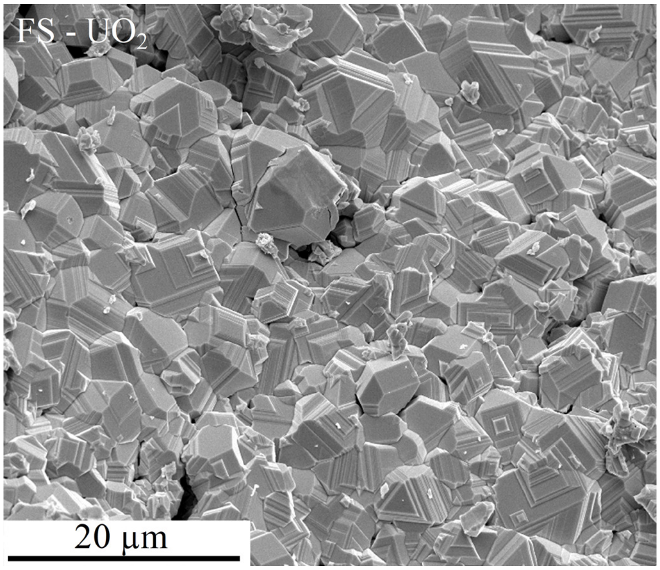
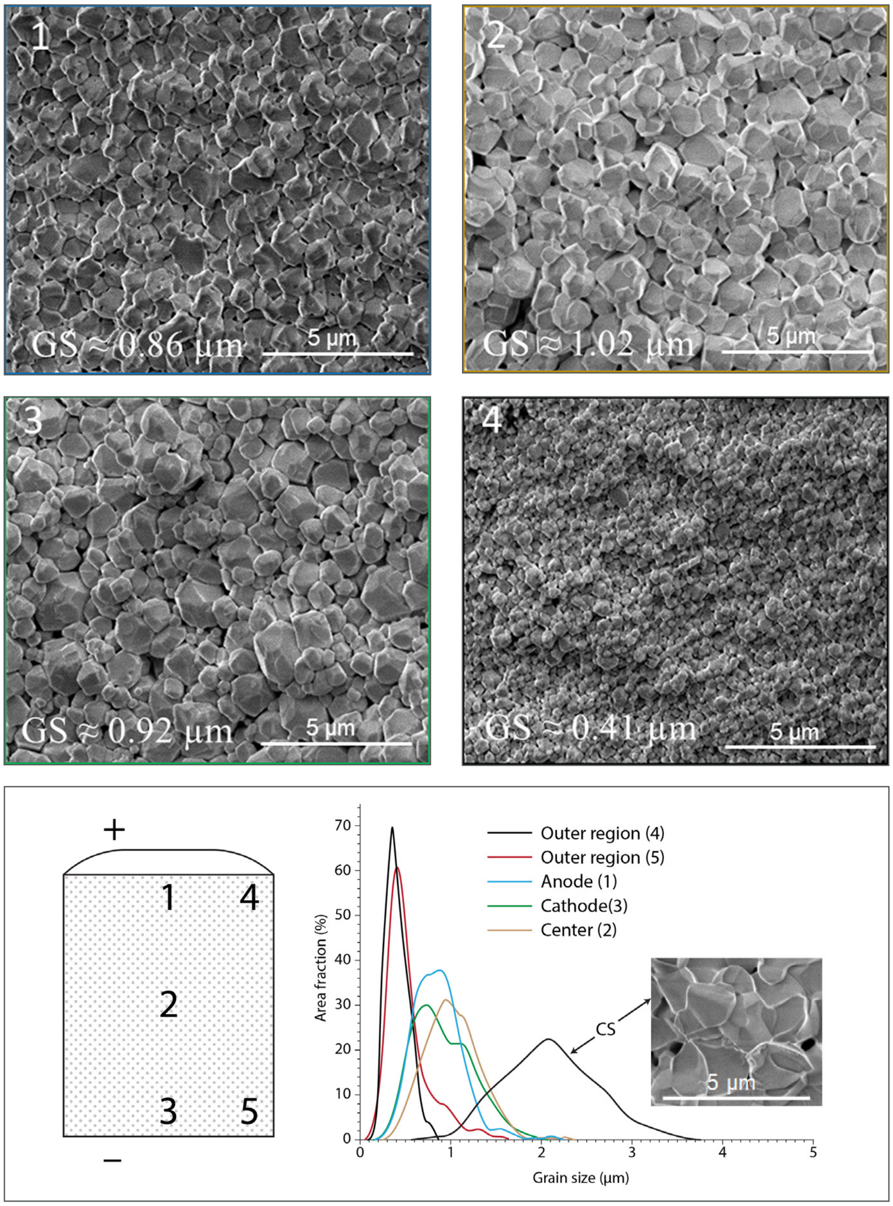


Figure 1. Microstructures of FS CeO2 (left) and UO_2 (right) pellets made with in-house synthesized powder. CeO_2 average grain size is 1.4 µm, while UO_2 microstructure shows the formation of micron-sized crystals

Accomplishments:

Nanopowders of CeO_2 and UO_2 were synthesized in-house with nanoparticles smaller than 3 nm, and then pellets of these materials were fabricated via FS. The microstructures of FS CeO_2 and UO_2 pellets are shown in Figure 1. CeO_2 pellets made by FS retained an average grain size of 1.4 µm and were densified up to 75.7% TD, while keeping an open porosity of 19%. These results are close to what is observed in actual HBS, which are characterized by submicron grains smaller than 0.7 µm, and porosities higher than 15 vol.%. To reduce the grain sizes of sim HBS even further, it is necessary to reduce the nanopowder agglomeration and aggregation. The in-house synthesized CeO_2 powder had agglomerates bigger than 50 µm and some aggregates were bigger than 5 µm. Even though FS is a very fast sintering technique capable of retaining small grain sizes, the final microstructure is dictated by the morphology of the powder precursor. Moreover, attempts to produce sim HBS of UO_2 using FS led to the formation of micron-sized UO_2 crystals inside the FS pellets, as presented in Figure 1. It is possible that the contamination by organic materials of the in-house synthesized powder combined with the electric current and oxygen deficient atmosphere during FS led to the reduction of the stoichiometry of the material. If UO_{2-x} was formed during FS, it could then melt at low temperatures (1132 °C), inducing the formation of UO_2 crystals and of UC phase, as later identified by X-ray diffraction.



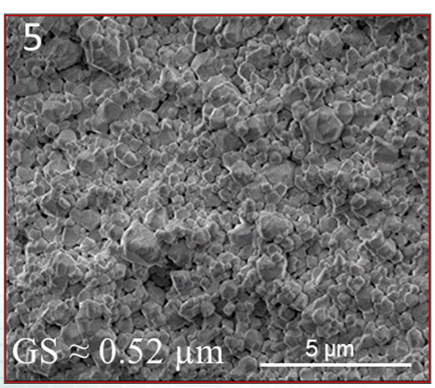


Figure 2. Grain size (GS) distribution across flash sintered CeO_2 pellet made with commercial nanopowder. Insert indicating average GS in conventional sintered (CS) sample.

In an effort to avoid powder contamination and agglomeration, as seen in the in-house CeO_2 powder, commercial CeO_2 nanopowder was used to produce sim HBS via FS. As shown in Figure 2, the pellets produced by FS retained an average grain size of 0.4 µm near the surface of the pellet, and of about 1 µm, at the center of the pellet. This heterogeneity was expected due to the temperature gradient during FS. The average open porosity of the FS pellet was 10.5 vol.% and the close porosity was 1.9 vol.%. A conventionally sintered (CS) pellet, on the other hand, had an average grain size of 2.1 µm distributed homogeneously across the pellet and an open porosity of 1 vol.% and closed porosity of 7.5 vol.%.

The capability to fabricate sim-HBS on ceramic materials was established at LANL. The efforts described here showed that FS can deliver sim HBS of ceramic materials such as CeO_2 and UO_2, although further powder optimization is necessary for the latter. Therefore, the next steps are to improve the in-house synthesis of UO_2 nanopowder and to start analyzing the thermophysical and mechanical properties of the resulting sim HBS. It is expected that this technology can help in separate effects testing efforts of HBS and that it will reduce the difficulties associated with experiments done on heavily irradiate materials.

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Integral Irradiation Tests on the ATR Pressurized Water Loop (ATF-2)

Principal Investigator: David Kamerman & Fabiola Cappia

The irradiation testing and post-irradiation examination of ATF allows for the generation of data necessary to license the new fuel designs for use in commercial operating reactors.

The Idaho National Laboratory (INL) Accident Tolerant Fuel (ATF)-2 irradiation testing and post-irradiation examination (PIE) program is focused on conducting prototypic irradiations of new light water reactor fuel designs with enhanced accident tolerance. The irradiated test pins are then shipped to INL hotcells for non-destructive and destructive testing where data needed to license the fuel designs for operation in commercial power reactors is generated.

Project Description:

The ATF-2 test train consists of 6 tiers each of which contains a 2x3 array of test pins which are nominally 15cm in length and contain 10 fuel pellets. At various times two of the tiers are combined so that longer test pins with up to 20 fuel pellets can be irradiated. Loop-2A supplies the pressurized water coolant to the ATF-2 test train via an annular in-pile tube coming in from the bottom of the Advanced Test Reactor (ATR) pressure vessel. The system is complete with a pressurizer, heat exchangers, line heaters, pumps as well as a feed and bleed secondary coolant circuit for chemistry control. The flow area in the tiers when configured with standard 17x17 pressurized water reactor (PWR) sized fuel pins is ~2.677 cm2 and ~2.509 cm2 when configured with larger boiling water reactor (BWR) sized fuel pins. This arrangement allows for disassembly of the test train in the ATR canal so that individual test pins can be removed at various time in their irradiation life and be replaced with new fresh pins. The entire design is modular allowing for large reconfigurable test matrixes.

A total of three shipments of irradiated test pins has been made from the ATR canal to the Hot Fuel Examination Facility (HFEF) hotcell for PIE. PIE of the ATF test pins has consisted of non-destructive, fission gas puncturing and destructive examinations. The non-destructive examinations included visual inspection, profilometry, neutron radiography and gamma scanning. The destructive examinations were focused on cladding performance, in particular mechanical testing.

Accomplishments:

The ATF-2 test has completed 7 cycles of irradiation in Loop 2A under prototypic PWR conditions. Forty-one test pins have been irradiated to date under the program reaching a maximum exposure of 30 MWd/kgU. The test pins have come from each of the three U.S. fuel vendors who are conducting Department of Energy (DOE) funded research and development (R&D)

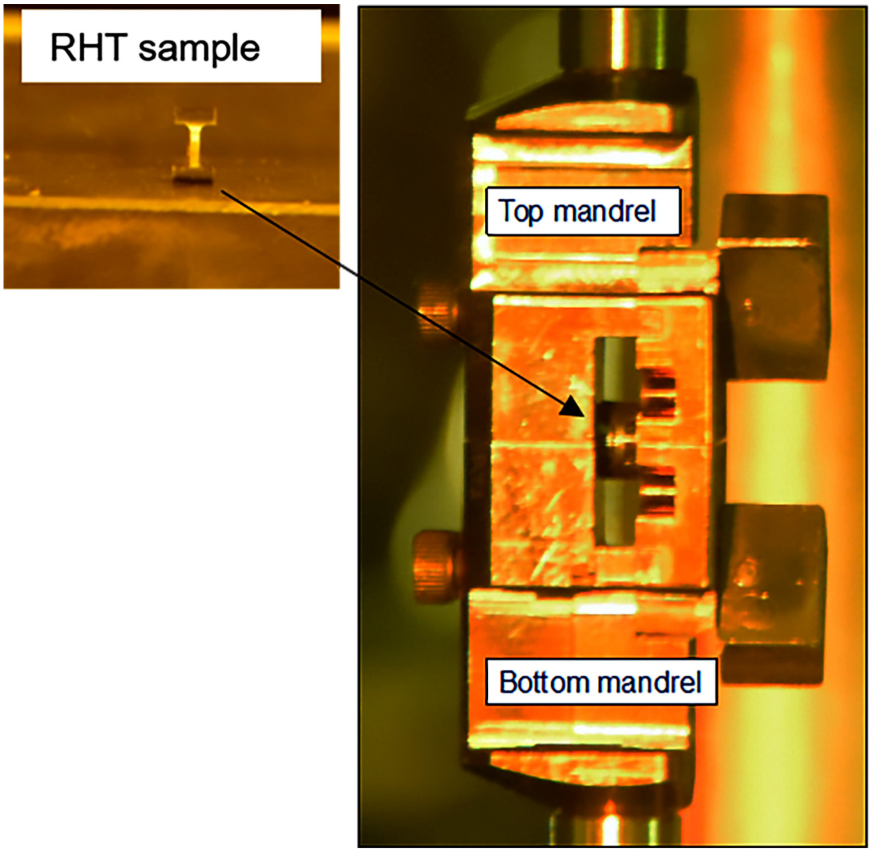


Figure 1. Ring Hoop Tension (RHT) test train installed in the remote load frame in the HFEF hot cells. The left side of the figure shows a RHT sample milled in cell before testing.

programs on ATF. Reactor power, coolant inlet and outlet temperatures, coolant pressure and coolant flow rates are monitored in-situ. Individual test pin power histories are evaluated on a cycle-by-cycle basis and published along with core power so that correlations can be made. All data is recorded and stored in an NQA-1 compliant database.

The PIE efforts were focused on the testing of ATF-2 cladding. Hydrogen content measurements were conducted on the ATF-2 baseline Zr-4 material showing negligible H pick-up at low burnup, as expected at the irradiation conditions in the Loop 2A. Mechanical testing capabilities for ring hoop tension (RHT) and axial tube tension

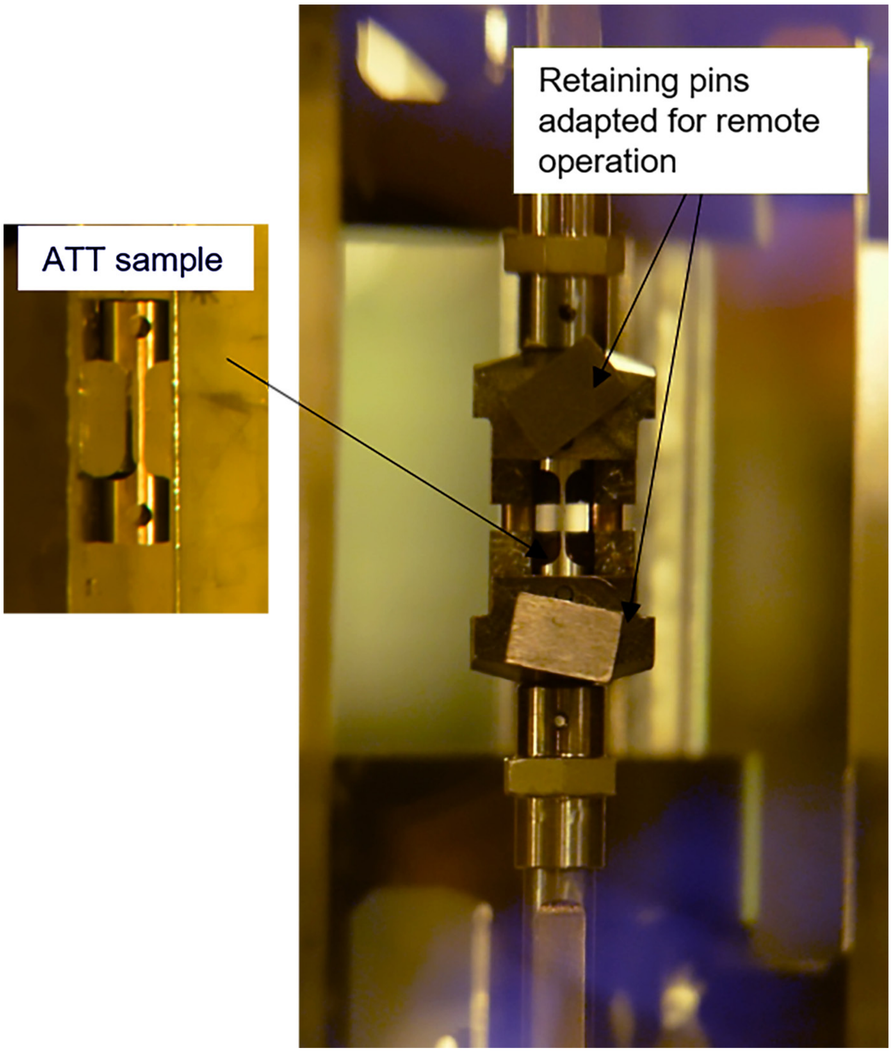
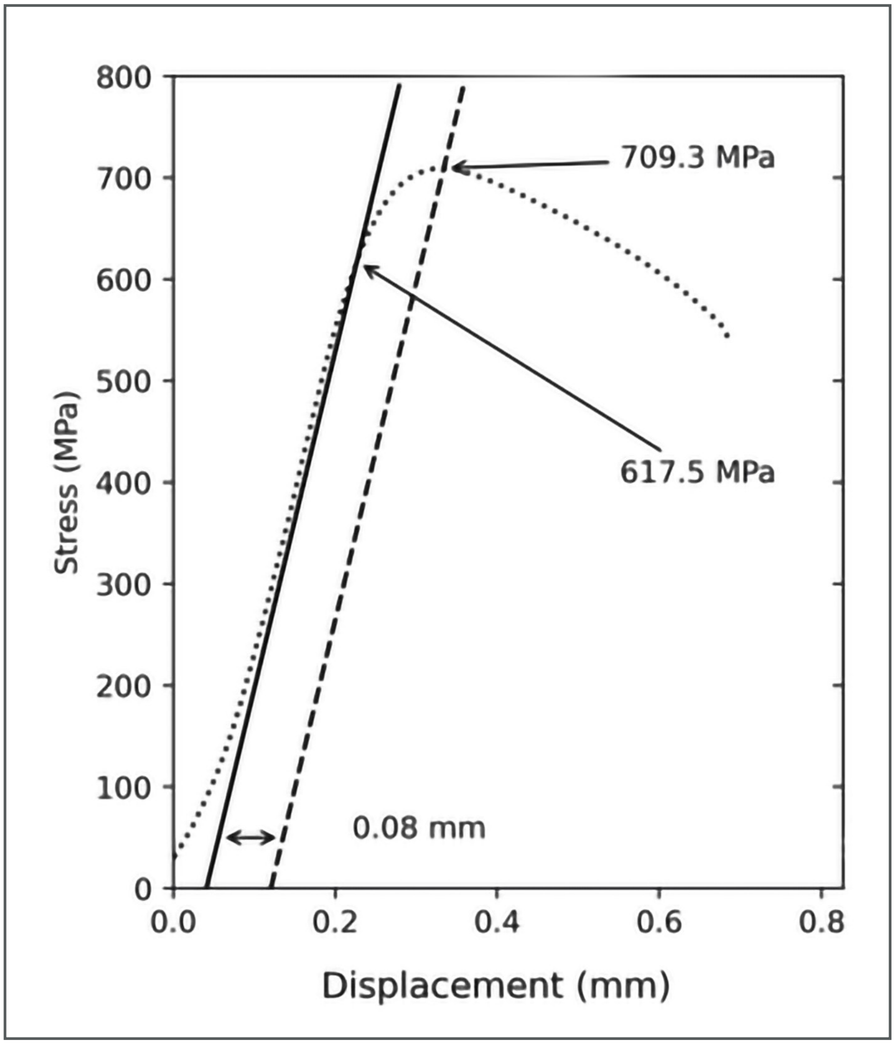


Figure 2. Axial Tension Test (ATT) test train installed in the remote load frame in the HFEF hot cells. The left side of the figure shows an ATT sample milled in cell before testing.

(ATT) testing were developed and applied for the hotcell environment of the HFEF which are shown in Figures 1 and 2, respectively. The RHT tests were coupled with finite element modeling to derive apparent yield stress and ultimate tensile stress. Results from testing 321 stainless steel and Zircaloy-4 showed consistent and repeatable performance of these test methods to those described in literature. Commissioning tests were performed on Zr-4 coated with Cr deposited by either physical vapor deposition (PVD) or cold sprayed (CS), which were provided by Dr. Martin Ševeček (Czech Technical University) and Prof. Kumar Sridharan (University of Wisconsin-Madison). The results showed that the application of either PVD or CS chrome coatings with small (micrometer scale) thicknesses do not meaningfully affect the material properties in the unirradiated state. Both ATT and RHT tests were then demonstrated using irradiated Zircaloy-4 cladding material from the ATF-2 irradiations. Figure 3 shows the results of one of the RHT tests on the irradiated Zr-4 from the baseline ATF-2 rods. Irradiated samples are milled in cell using a sample preparation method that was shown not to have an impact on the test results, giving confidence that the sample preparation and testing methods can next be used to investigate the apparent yield stress and ultimate tensile stress of irradiated coated cladding materials.



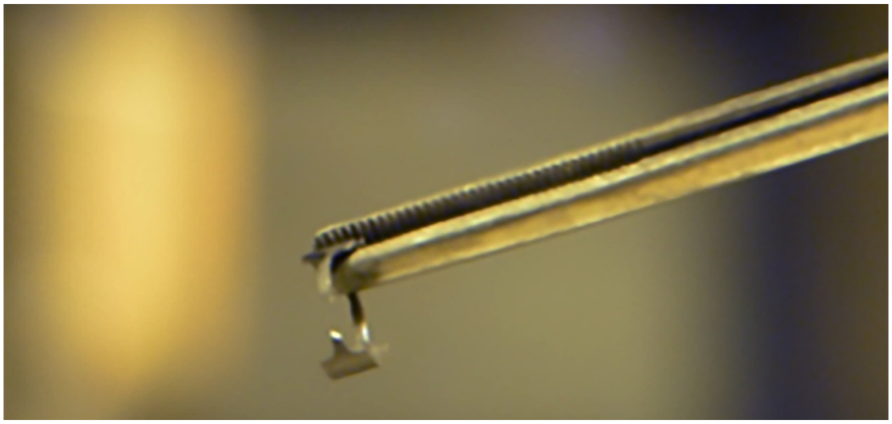


Figure 3. Stress-displacement curve of an irradiated Zr-4 sample from the ATF-2 baseline rods tested at room temperature (above left). Irradiated Zr-4 RHT sample after the tensile test, showing the broken gauge (bottom left).

Receipt of Accident Tolerant Fuel Lead Test Rods and Rapid PIE

Principal Investigator: Jason M. Harp
Team Members/ Collaborators: J. Harp, N. Capps, K. Linton, T. Jordan, Z. Burns, T. Ulrich, and A. Nelson

Figure 1. The Hatch ATF shipment arrives at the ORNL Irradiated Fuels Examination Laboratory

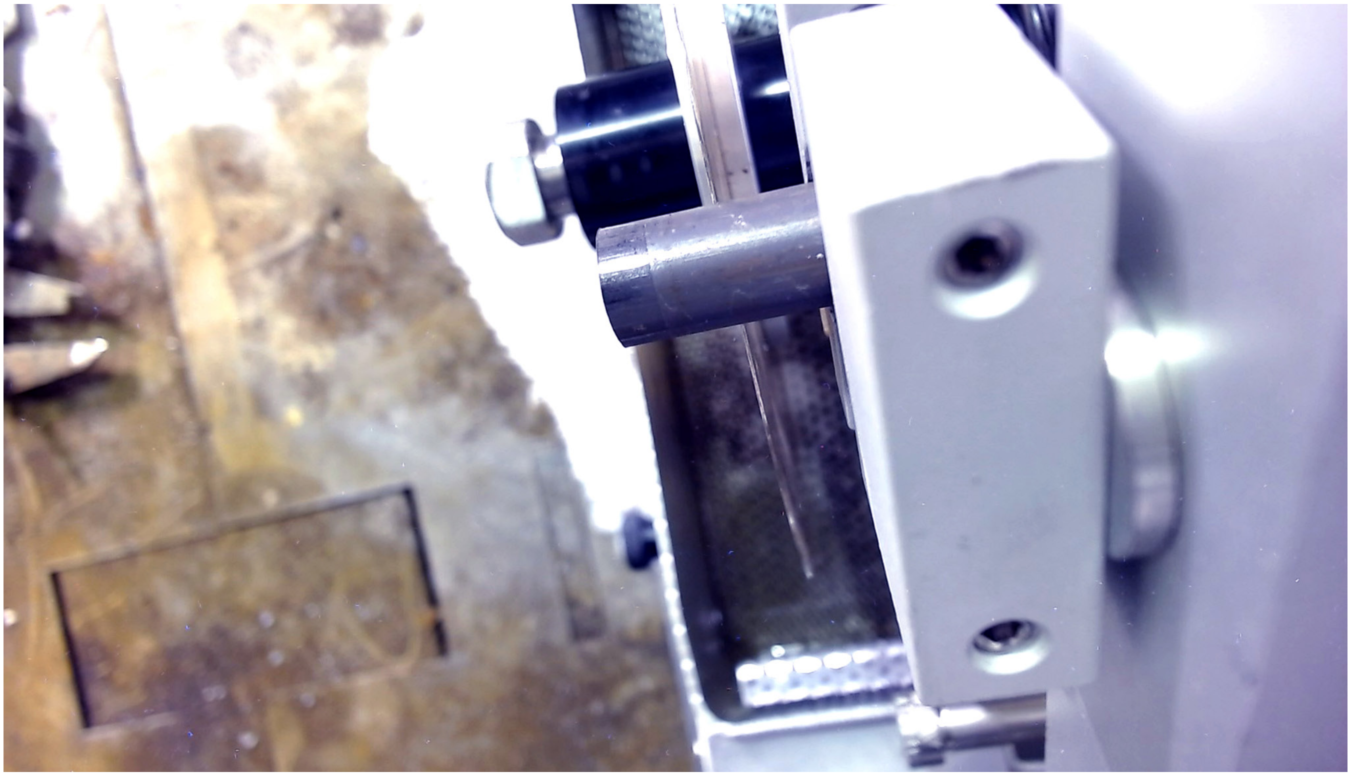
The receipt of these LTRs will enable post-irradiation examination on a variety of near-term ATF concepts and enable the generation of essential data concerning light water reactor fuel performance at elevated burnups.

Oak Ridge National Laboratory (ORNL) supported the receipt of two shipments of commercially irradiated Accident Tolerant Fuel (ATF) lead test rod (LTR) concepts from two different industrial partners. The first shipment contained General Electric (GE) ATF concepts and the second shipment contained Westinghouse ATF concepts. In addition to receiving this fuel, rapid post-irradiation examination (PIE) was initiated on the GE concepts to get a first look at a selection of these concepts.

Project Description:

The examination of commercially irradiated ATF LTRs will generate data valuable irradiation data industrial partners can use as input to future licensing packages for their ATF concepts. This fuel performance data along with data from test reactors and out-of-pile testing is used to validate fuel performance modeling codes that will be used in eventual licensing.

The first shipment arrived at the ORNL Irradiated Fuels Examination Laboratory on November 5, 2020, from the Edwin G. Hatch Nuclear Power Plant Unit 1 (Hatch-1). These ATF LTR segments were irradiated in collaboration between the US Department of Energy, General Electric Research, Global Nuclear Fuels, and ORNL. The GE-2000 cask from this shipment can be seen in Figure 1. The shipment contained 38 fully sealed segments including 32 UO_2 fuel segments clad with Zircaloy-2 and coated with a GE proprietary coating known as ARMOR and 6 un-fueled cladding segments of an iron-chromium-aluminum (FeCrAl) alloy known as IronClad. Once the segments arrived at the facility, they were unloaded from the shipping cask and

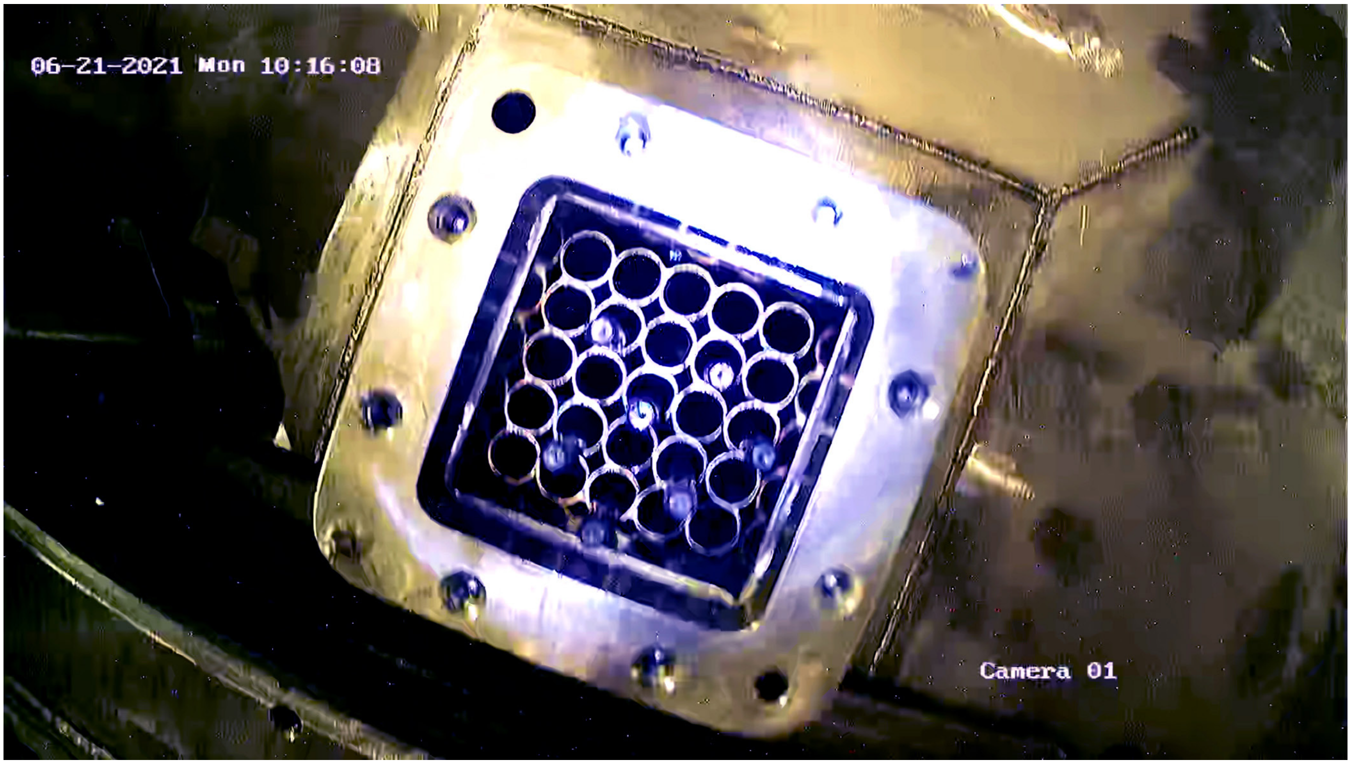


catalogued by checking the segment ID printed on the end-cap of each segment. Rapid examination of the ARMOR segments was performed to provide GE with preliminary data regarding the performance these segments. Sectioning of an ARMOR segment can be seen in Figure 2.

The second shipment of fiscal year (FY)21 arrived at ORNL June 15, 2021, containing ATF and high burnup LTR's that will be examined on behalf of Westinghouse. The unloading of the NAC LWT cask that contained the 3 ATF LTR's and the 4 high burnup LTR's is shown

Figure 2. Segmenting ARMOR

Figure 3. The cask containing Westinghouse ATF and high burnup LTRs is unloaded at ORNL



in Figure 3, and the unloading of the fuel into the hot-cell is shown in Figure 4. The Westinghouse ATF concepts in this shipment contained Cr-coated zirconium alloy cladding fueled with both standard UO_2 and Westinghouse's doped UO_2 known as ADOPT. The high burnup LTR's will be used to collect data relevant to extending the burnup of light water reactor fuel beyond the current regulatory limit of 62 MWd/kg U. Some of the planned exams include non-destructive evaluation of the fuel, simulated Loss of Coolant Accident (LOCA) burst testing, transient fission gas release evaluation, and microstructural examination of the high burnup UO_2 to better understand fuel fragmentation behavior.

Figure 4. The 3 Westinghouse ATF LTR's and the 4 High Burnup LTR's ready for unloading into the ORNL Irradiated Fuel Examination Laboratory hot-cell

Integrated Experimental and Modeling Approach to Assessment of Enhanced Grain Size of UO_2 Fuel Pellets

Principal Investigator: Joshua T. White
Team Members/Collaborators: Arjen van Veelen, Brandon S. Battas, Tashiema L Ulrich, Michael Tonks, Michael W.D. Cooper, and Joshua T. White

Methods shown here provide a framework to integrate modeling and experimentation, accelerating the validation and qualification of model development.

Uranium dioxide (UO_2) fuel pellets power more than 90 percent of civilian nuclear reactors operating today. With the increasing demands for carbon neutral energy, the continuous improvement of the efficiency and safety nuclear reactors is a top priority. The addition of oxide dopants to UO_2 fuel provides significant improvement to the micro-structural properties, specifically a two to fivefold increase of the grain size1-5. This is important because it has been demonstrated that grain size affects fission gas release from fuel pellets, as well as creep and hardness of the fuel pellets. In order to develop and license enhanced UO_2 as Accident Tolerant Fuel (ATF), understanding the mechanistic impacts of different dopants on fuel performance must be demonstrated. This study aims to develop sintering kinetic models for undoped UO_2 with grain sizes comparable to that of doped UO_2 with the aim to use this as reference material for studying doped-UO_2 fuel performance.

Project Description:

This study set out to understand the mechanistic impacts of different sintering conditions on the activation energies that control the microstructural parameters, such as grain size and density. Our results show that activation energies decrease with increasing stoichiometry at $0.0003 < x < 0.2$ (Figure 1). We also measured the grain size as a function of oxidizing atmosphere (pO_2) and concluded that largest and smallest grain sizes were obtained under CO_2 and Ar atmospheres, respectively. We combined this with meso- (MARMOT) and atomic scale modeling to predict vacancy concentrations with variations in O/U ratio. The model sheds light on the mechanisms and the energetics (bulk and grain boundary) that govern sintering, which highlight the strong thermodynamic driving force for vacancies to diffuse to the grain boundaries and then diffuse along the grain boundaries to eliminate internal porosity. The experimental and modeling work correspond well for near-stoichiometric compositions, but more work is necessary to account for defect clustering that will impact the applicability of the model for high O/U ratios. This work underpins the fundamental mechanisms that will further elucidate the understanding of irradiation-enhanced densification to better predict material responses to irradiation. Research in this area support the continued interest within the advanced fuel campaign (AFC) to advance understanding of near-term ATF fuel through integrated modeling and experimental efforts clarifying the role of fission gas release and irradiation induced densification in large grained UO_2.

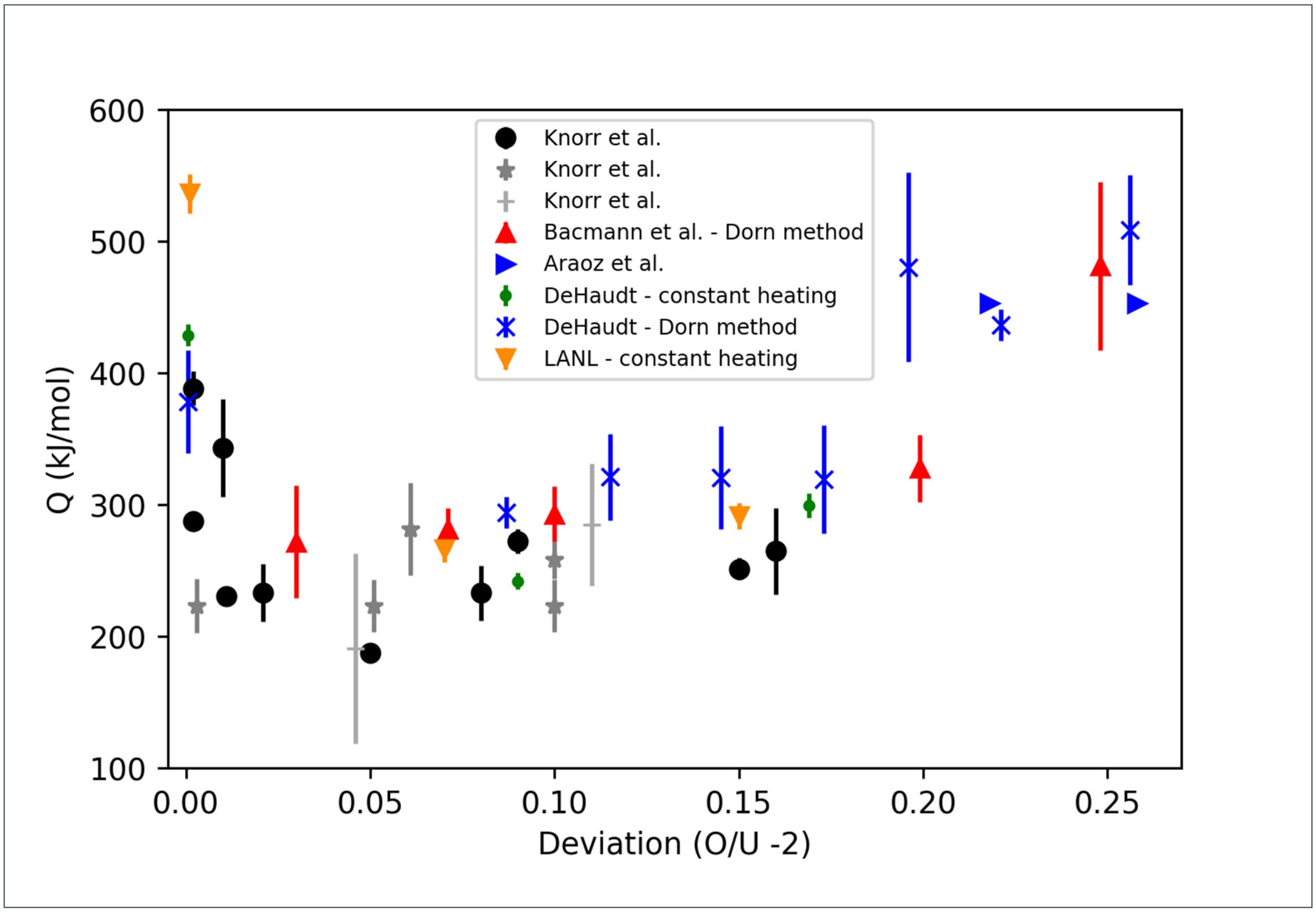


Figure 1. Change in sintering activation energy with varying stoichiometry. Published data is adopted from deHaudt et al. (2001).

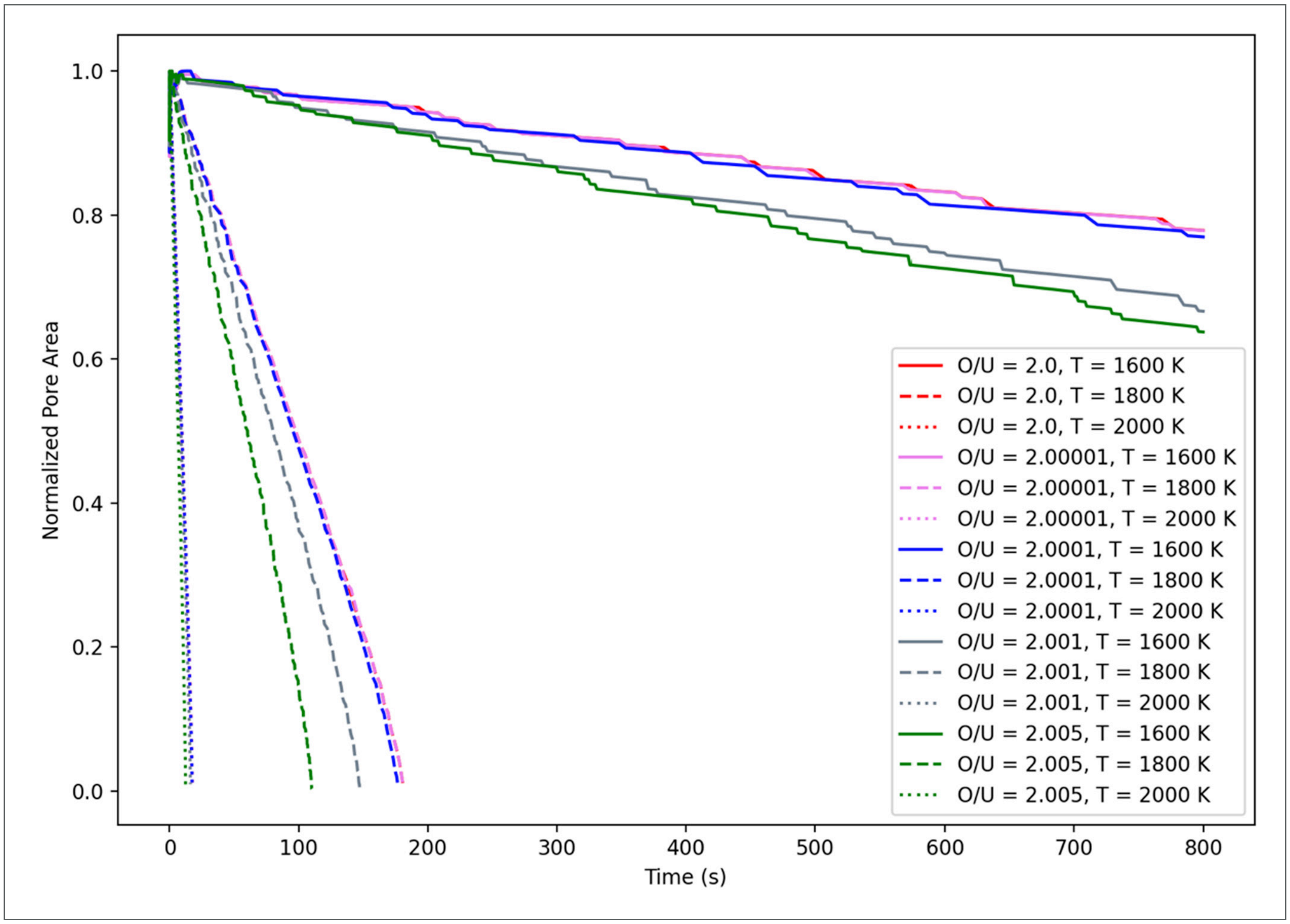


Figure 2. The reduction of internal pore area from the MARMOT simulations of various fixed temperatures and stoichiometries

Accomplishments:

In this project we provide datasets such as grain size, activation energies, porosity and correlate this with modeling efforts. Samples were fabricated by Joshua White, Tashiema Ulrich, and Arjen van Veelen at the Los Alamos Fuels Research Laboratory. The analyses of the grain size, porosity was done by Tashiema Ulrich under the supervision of Joshua White at Los Alamos National Laboratory. The data suggest that larger grain sizes are achieved under more oxidizing atmospheres, with largest grain sizes achieved under CO_2 atmospheres. However, more data is necessary to obtain grain size in low density materials in order to predict grain size kinetics. Activation energies were calculated by Arjen van Veelen, using a custom-made script written in Python. The experimental data provided the basis for validating the combined atomic scale and meso-scale modeling efforts on densification of a simple 2D 4-particle geometry. These models were developed by Michael Cooper, Brandon Battas, and Prof. Michael Tonks from the University of Florida. The modeling is in qualitative agreement with experimental data capturing the increase in densification rate for increased O/U. Application to high stoichiometries (e.g., O/U=2.1) is not possible within the point defect assumptions of the model and extension of the model would require the inclusion of defect clustering in future work. The most significant finding is that slight variation from perfectly stoichiometric to marginally hyperstoichiometric UO_2 translates to a significant increase in the densification rate, which may account for the large variation in activation energies observed in experimental data for nominally stoichiometric UO_2 (Figure 2). The modeling also showed that this behavior is due to the increase in grain boundary vacancy caused by the slight increase in O/U. Although the simulations are in good qualitative agreement with experimental trends, more data is needed in terms of densification/grain growth versus stoichiometry and phase field simulations containing more complex particle geometries that are better representative of the microstructure of real pellets. This will have to be combined with experimental activation energies obtained at lower density samples.

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Over Power Transient Testing of LWR Fuels

Principal Investigator: David Kamerman
Team Members/ Collaborators: Leigh Ann Astle, Charlie Folsom, Colby Jensen, Austin Fleming, Connor Woolum and Wes Smith

This test program will help inform the development of safety margins for high burnup light water reactor fuel in reactivity-initiated accidents.

The HERA project (High Burnup Experiments in Reactivity initiated Accidents) is a part of the Nuclear Energy Agency's (NEA's) Framework for Irradiation Experiments (FIDES) initiative. The project aims to conduct integral Reactivity-Initiated Accident (RIA) testing of both pre-hydrided and actual high burnup test specimens.

Project Description:

HERA is a joint experimental program (JEEP) operating within the NEA framework for irradiation experiments (FIDES). HERA is dedicated to the understanding of light water reactor (LWR) fuel performance at high burnup under RIA. In-pile RIA experiments have been performed on high burnup fuels (above 60 GWD/MTU) in the CABRI reactor in France, and the NSRR reactor in Japan. However, the majority of these experiments have taken place with heavily corroded Zircaloy claddings in test reactors with pulse widths that are more narrow (5ms – 30ms full-width-half-max (FWHM)) than what would be likely in a commercial LWR (30ms – 80ms FWHM). Heavy waterside corrosion and narrow pulse widths are both known to increase the vulnerability of LWR fuel to pellet cladding mechanical interaction (PCMI). The HERA proposal is designed to (1) quantify the impact of pulse width on fuel performance, offering new insight into the applicability of existing data, (2) generate new data on high burnup fuel under pulse conditions prototypic of LWRs (3) quantify the additional margin provided by modern cladding alloys to PCMI failure limits and (4) offer improved data for modelers using specially designed tests that eliminate key uncertainties in high-burnup fuel tests.

Accomplishments:

This year Idaho National Laboratory (INL) developed a new static water capsule design to be used in the HERA experiments. The design features several new instruments including a fuel centerline thermocouple and 2 water pressure sensors to help detect the moment of any cladding failures. The design also features an improved cladding thermocouple holder to improve the reliability and aid in the ease of assembly for the cladding thermocouples. Weld development for the integral junction cladding thermocouples also yielded very fruitful outcomes with a reliable process being developed to better attach the cladding thermocouples in a consistent and repeatable manner.

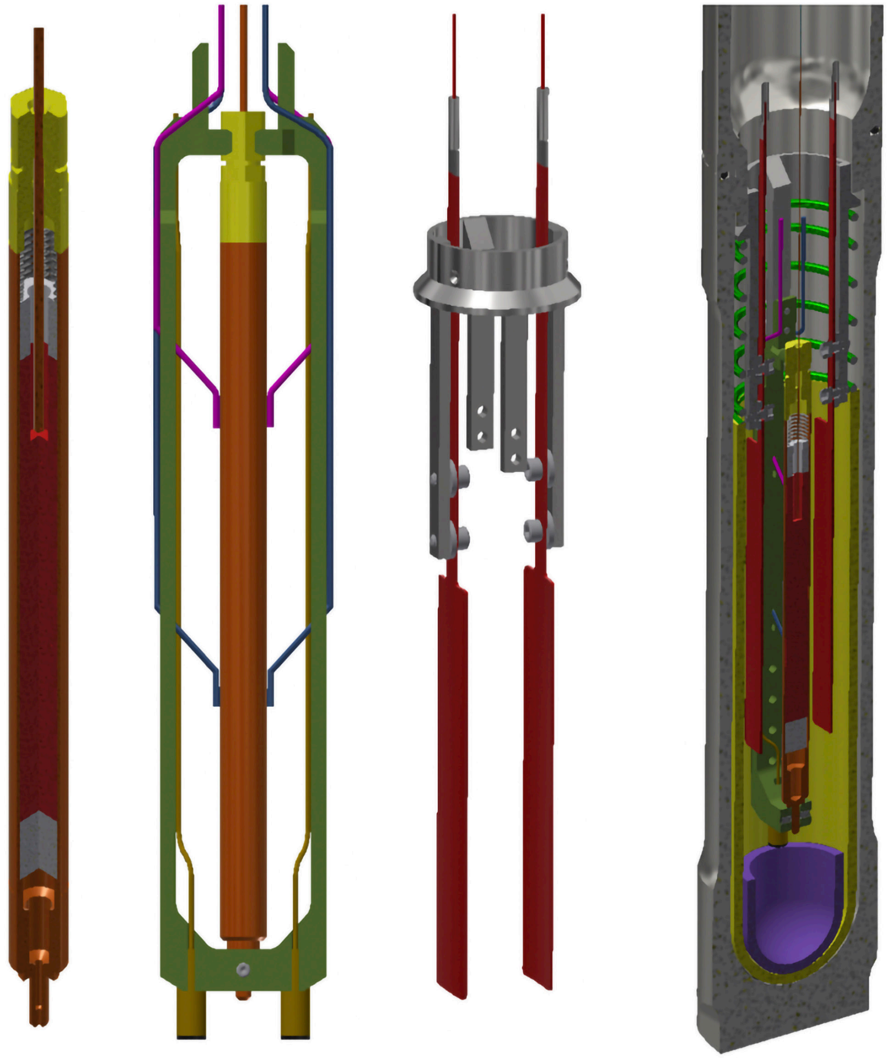

Figure 1. HERA static water capsule test train design

Figure 2. Cladding thermocouple weld

Additionally, a reliable method for pre hydriding Zircaloy-4 cladding with hydride rims of ~100 microns was developed using a sealed partial pressure gaseous diffusion method. Destructive examinations of the hydrided cladding revealed consistent hydrogen concentrations and rim thicknesses across the center 10 cm of the hydrided cladding tubes.

Completion of the capsule design, methodology for creating hydrided cladding, and methodology for welding integral junction cladding thermocouples positions the HERA project well to be successful next year when the initial irradiations will take place.

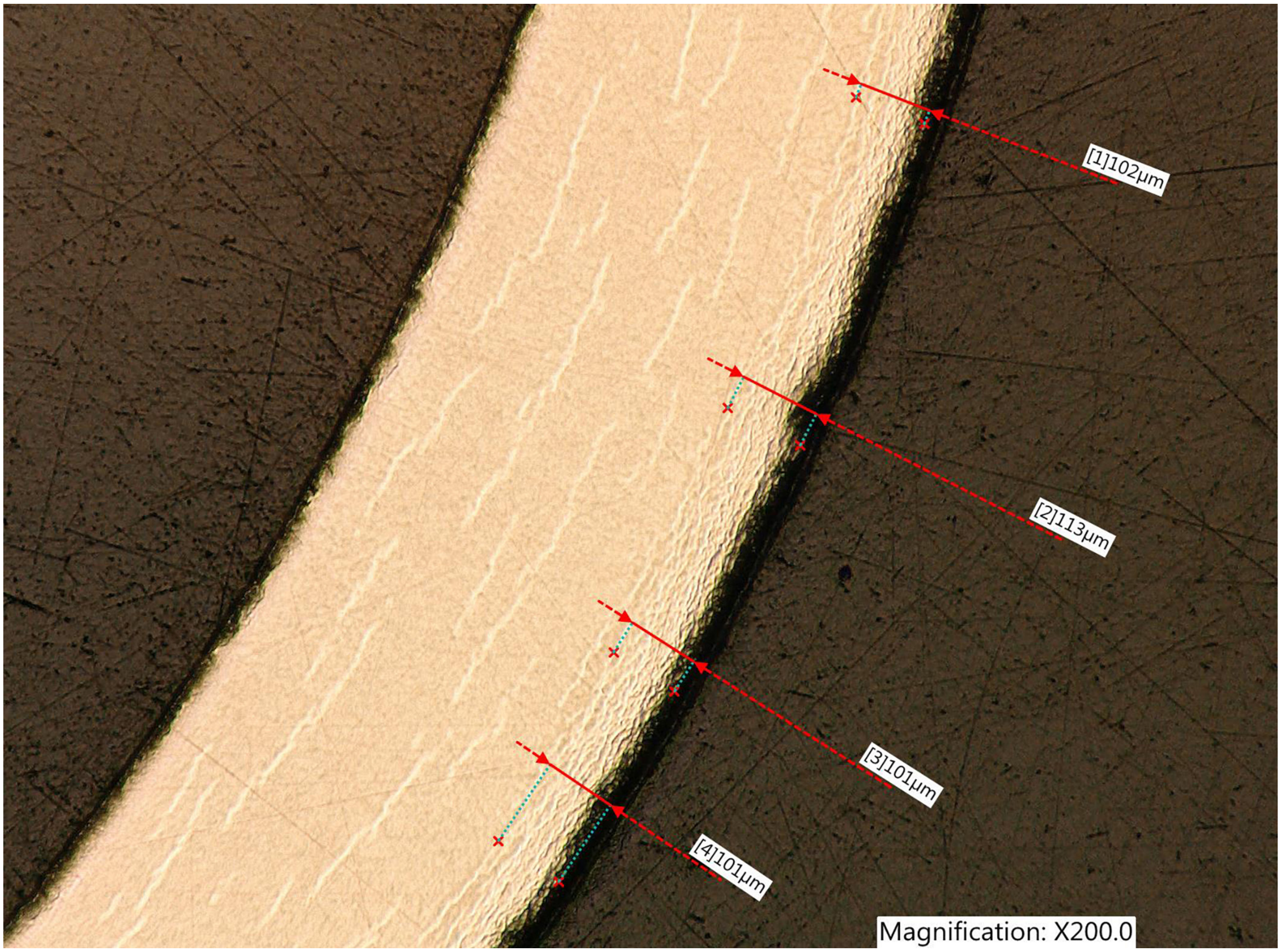

Figure 3. Hydrided cladding with 100 µm rim

First Characterization of Irradiated Fuel with Pulsed Neutrons at LANSCE

Principal Investigator: Sven C. Vogel (LANL)
Team Members/ Collaborator: Thilo Balke (LANL & Purdue University), Charles A. Bouman (Purdue University), Luca Capriotti (INL), Jason M. Harp (ORNL), Alex M. Long (LANL), Danielle C. Schaper (LANL), Anton S. Tremsin (UC Berkeley), and Brendt E. Wohlberg (LANL)

Neutrons offer bulk, non-destructive characterization of irradiated materials for which other bulk methods, e.g., X-ray diffraction or tomography, are not suitable due to the immense gamma background emitted from the samples. In particular, pulsed neutrons provide information from the ability to resolve the neutron energy using their time-of-flight. This enables the potential to utilize neutron absorption resonance spectroscopy to characterize the spatial distribution of isotopes, so-called energy-resolved neutron imaging or neutron resonance imaging. As was demonstrated previously [1-5], neutron resonance spectroscopy allows to characterize the distribution of fission and neutron capture products non-destructively. Besides neutron resonance-based radiography and tomography, neutron diffraction provides information on the microstructure (texture, identification of crystallographic phases and measuring their phase fractions). Both of these techniques may ultimately be applied to the entire bulk volume of an irradiation capsule to guide destructive post-irradiation examination to identify regions of interest.

Project Description:

Advanced post-irradiation examination methods utilizing the unique capabilities offered by pulsed neutrons are developed at Los Alamos National Lab's (LANL) Los Alamos Neutron Science Center (LANSCE). Characterization of the entire irradiated sample volume of an irradiated fuel with pulsed neutron techniques will ultimately identify normal and abnormal regions within the entire irradiated volume, add otherwise unavailable data and guide destructive evaluation. These pulsed neutron techniques therefore maximize insight from expensive irradiation campaigns. As a step towards this goal, an Advanced Test Reactor (ATR) irradiated U-10Zr-1Pd sample was characterized with an Nuclear Science User Facilities (NSUF) funded Rapid Turnaround Experiment (RTE) in the 2020 and 2021 LANSCE run cycles. U-10Zr metallic fuels are researched as host materials for potential transmutation fuels and the addition of palladium strives to bind lanthanides, thus hindering fuel-cladding chemical interactions (FCCI). This is the first pulsed neutron characterization of an irradiated fuel at LANSCE. Part of the project is also the development of a cask designed to enable the techniques demonstrated here on a small disk sample to entire irradiated fuel capsules. The development of this cask is described elsewhere in this report. Besides post-irradiation examination, the project also applies pulsed neutron techniques to Advanced Fuel Campaign (AFC) funded fuel development at LANL.

Accomplishments:

The technical goal of this project is to apply pulsed neutron characterization to an irradiated fuel sample manageable by remote handling (i.e., not requiring a cask), both to gain insight on the particular sample as well as a step towards characterizing larger amounts of irradiated fuel.

For this goal, a 6mm diameter, 1.8mm thick disk was cut from the AFC-3A-R5A irradiation sample [6]. The post-irradiation isotope composition of the material is expected to be 39.9 wt%. 238-U, 46.5 wt.% 235-U, 0.3 wt.% 239-Pu, 10.2 wt% Zr, 2.6 wt.% Pd. A custom sample holder was designed for pulsed neutron experiments at LANL and the sample was loaded into that container. Figure 1 shows the sample container, an aluminum rod with a cavity for the disk-shaped irradiated fuel sample. The aluminum rod was enclosed in a

First characterization of irradiated fuel with pulsed neutrons at the Los Alamos Neutron Science Center is an important step towards pulsed neutron characterization of entire irradiation capsules.

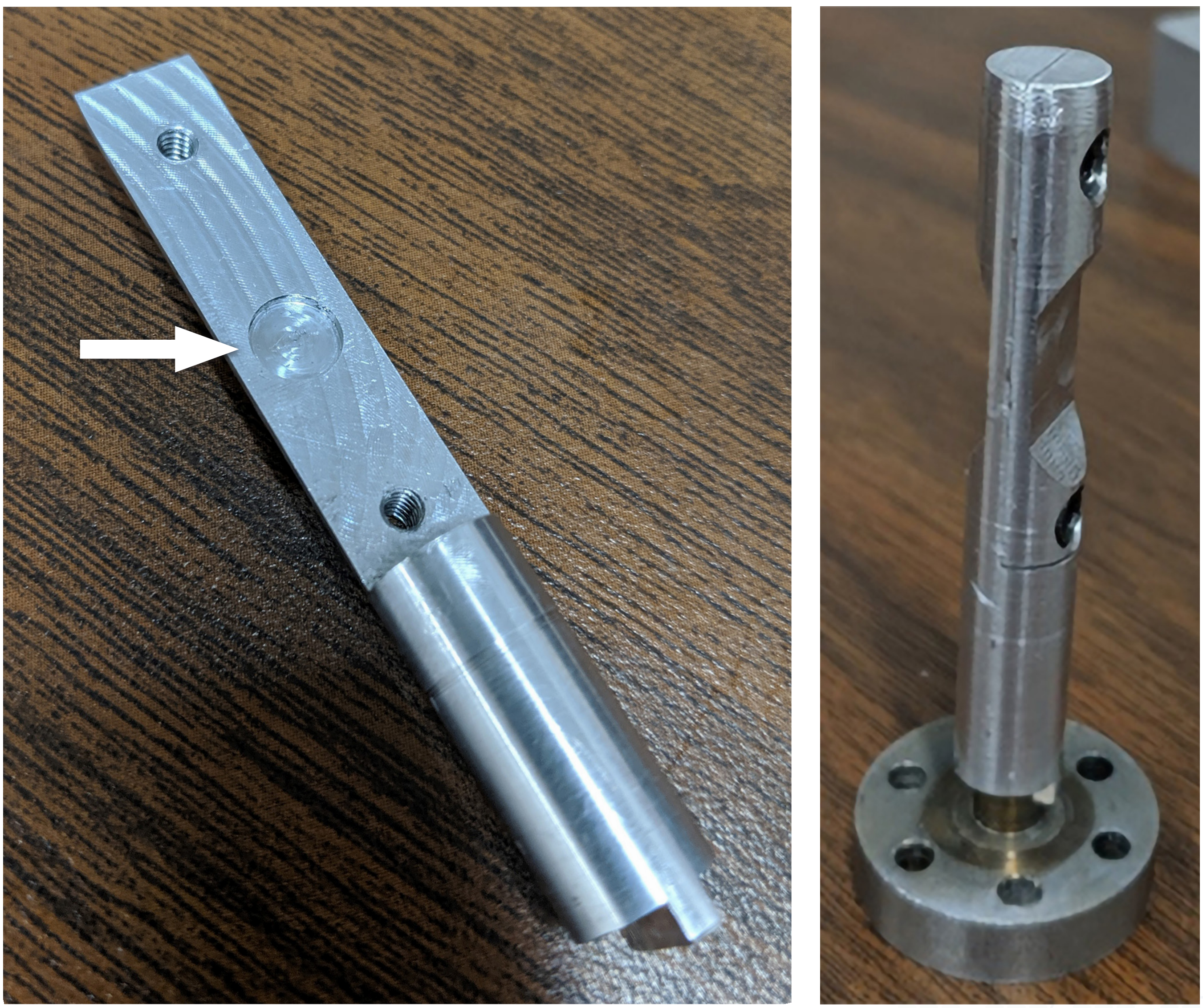

Figure 1. Opened aluminum sample holder with the cavity for the disk of irradiated U-10Zr-1Pd in the thinned section marked by an arrow (left). Sample holder closed and attached to the lid of a vanadium can that provides secondary containment of the sample (right).

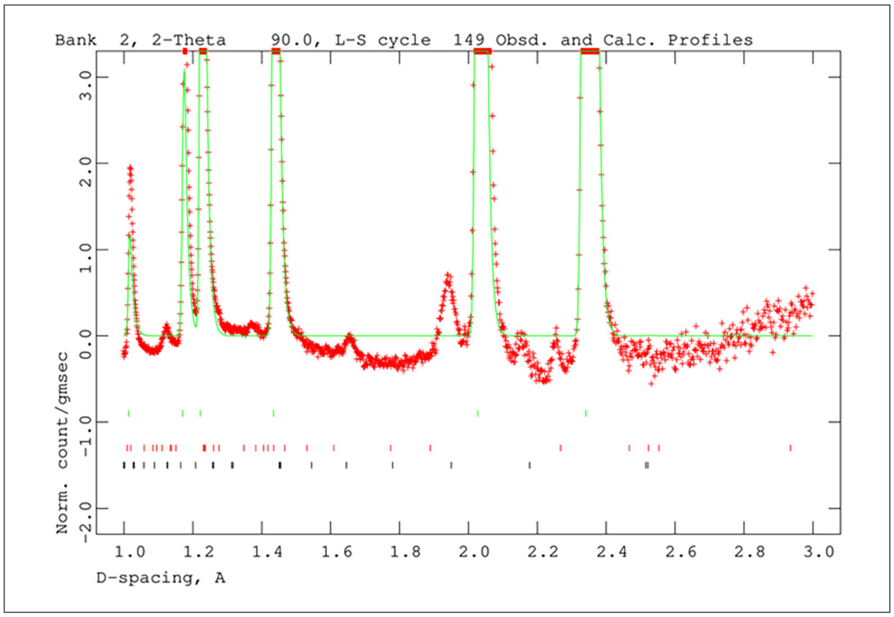

Figure 2. Diffraction data collected on the HIPPO instrument showing the strong diffraction peaks from the aluminum sample holder with the Rietveld fit (green curve, black tick marks refer to Al peaks). The weaker peaks not fit in this preliminary analysis originate from the U-Zr phases in the irradiated material and can be indexed with a cubic crystal structure. Notably absent are diffraction peaks from orthorhombic α-uranium that are observed in fresh U-10Zr fuel e.g., between 2.5 and 2.7 Å (positions indicated by red tick-marks). The thermal neutron absorption of the 46.5 wt% enriched fuel made neutron diffraction challenging.

standard neutron scattering vanadium can as a secondary containment.

The sample and sample container were shipped to LANL in a shielding container. The sample emits at a dose rate of ~3R/hr on contact which decreases to <20 mR/hr at 2 meters distance and is therefore manageable with remote handling, without requiring a cask. The disk-shaped material is larger than samples prepared for analysis using electron or X-ray methods and is therefore an intermediate step towards characterization of bulk samples at LANSCE. However, since it covers the full diameter of the irradiated fuel slug, some insight on redistribution of elements, spatially resolved information on microstructure, e.g., phase composition and texture, will be possible using the pulsed neutron-based methods developed for fuel characterization at LANSCE.

Supported by AFC, a custom sample holder was designed and fabricated. The sample was encapsulated in the sample container in the hot cell at INL (Luca Capriotti/INL) and shipped to LANL where the material was characterized at the High-Pressure/Preferred Orientation (HIPPO, Sven Vogel/LANL) neutron time-of-flight diffractometer and the energy-resolved neutron imaging (ERNI, Alex Long/LANL) beamline at LANSCE in the fall of 2020 and summer 2021. Data analysis and interpretation are still on-going. Improved experimental setup and novel imaging data analysis methods for the energy-resolved neutron imaging, the latter supported by a LANL Laboratory Directed Research and Development (LDRD), are for the first time applied to this sample (Thilo Balke, Alex Long, Sven Vogel, Brendt Wohlberg/all LANL, Anton Tremsin/UC Berkely, Charles Bouman/Purdue University). The novel imaging analysis methods allow to better interpret the inherently low counting statistics data collected during the energy-resolved neutron imaging measurement.

First preliminary results on the powder diffraction data analysis, showing the crystallographic phases present in the irradiated material, are shown in Figure 2. Non-spatially resolved neutron resonance data, providing a higher sensitivity to isotopes present only in smaller concentrations, was also collected and is presented in Figure 3. Note that the results shown here are preliminary results and the analysis and interpretation (Luca Capriotti/

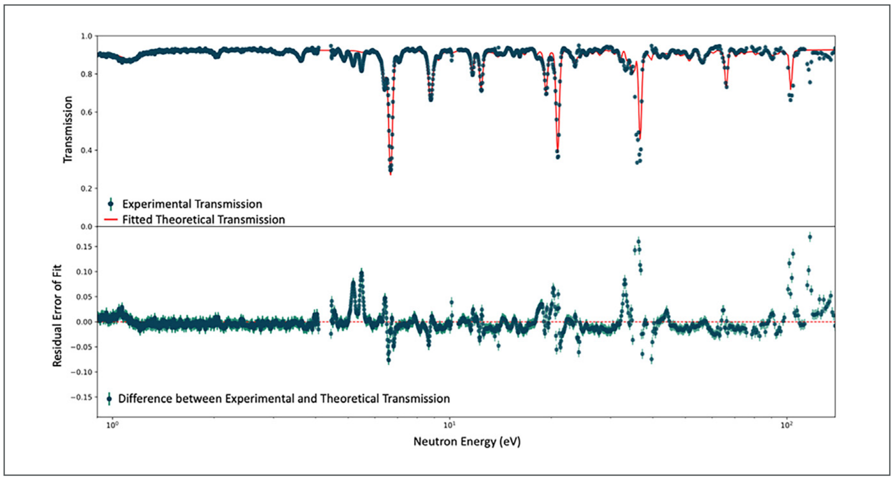

Figure 3. Preliminary analysis of the neutron transmission. The dips in the transmission spectrum are due to neutron absorption resonances, indicating the presence of specific isotopes. The fit (red curve) allows to quantify the isotope density. As is evident in the difference curve between measured data and fit in the lower part of the figure, several resonances are not yet identified in the preliminary results shown here.

INL, Jason Harp/ORNL) is on-going. Improvements of the sample holder for future characterization of such disk-ship samples are also discussed. Data analysis for diffraction and resonances as well as data interpretation and writing of a manuscript describing this effort are funded by AFC. Development of advanced energy-resolved imaging data analysis methods demonstrated here are funded by a LANL LDRD and NSUF funding for receiving, handling and data collection on the HIPPO and FP5/ERNI beamlines at LANSCE is gratefully acknowledged.

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Progress towards Pulsed Neutron Characterization of Entire Irradiation Capsules with SHERMAN, the Sample Handling Environment for Radioactive Materials Analyzed with Neutrons

Principal Investigator: Sven C. Vogel (LANL)
Team Members/ Collaborator: Eric J. Larson (LANL), James R. Angell (INL), D. Travis Carver (LANL), Aaron Craft (INL), Christopher Fairbanks (LANL), Brian Gross (INL), Alex M. Long (LANL), Jason M. Harp (ORNL), Peter Hosemann (UC Berkeley), and Vedant K. Mehta (LANL)

SHERMAN, the Sample Handling Environment for Radioactive Materials Analyzed with Neutrons, will enable multi-modal pulsed neutron characterization of entire irradiation capsules to guide destructive examination as well as providing data hitherto unavailable.

A bottleneck of nuclear energy research and development (R&D) is the lack of bulk probes to identify non-destructively the most scientifically valuable volume of less than 1 mm^3 volume to be prepared destructively for characterization out of a several cm^3 total volume of irradiated material contained in an irradiation capsule. Pulsed neutron-based techniques enable energy-resolved neutron tomography (i.e., in the epithermal regime where the absorption of isotopes such as ^{235}U and ^{239}Pu is sufficiently low to provide sufficient neutron transmission), neutron absorption resonance spectroscopy (including the potential to reconstruct isotope densities in 3D), and neutron diffraction. These techniques were developed and applied to fresh fuels at Los Alamos Neutron Science Center (LANSCE) in the past years and add parameters and characterization opportunities not available by other methods for post irradiation examination of entire capsules.

A cask named SHERMAN (Sample Handling Environment for Radioactive Materials Analyzed with Neutrons, Figure 1), allowing to apply these techniques to irradiated fuels, was designed at Los Alamos National Laboratory (LANL), University of California Berkeley (UC Berkeley) and Idaho National Laboratory (INL). The cask design has several constraints that must be fulfilled in order to allow loading of the irradiated fuel at INL, transportation to LANL, and pulsed neutron characterization at the LANSCE at LANL. The cask is designed to characterize irradiated mixed oxide (MOX) fuel samples of the AFC-2C and AFC-2D irradiation campaigns, emitting estimated dose rates of 900R/hr on contact.

Project Description:

The cask design has to fulfill criteria imposed by radiation safety, i.e., dose rates on the cask surface acceptable for handling by workers with a 900 R/hr sample inside, as well as structural integrity, i.e., simulated drop tests that must not result in direct shine paths for the sample to the outside. The cask has to fit into the cargo bay of a BEA Research Reactor (BRR) Type-B shipping cask to be able to transport the irradiated fuel on public roads from irradiation facilities at INL or Oak Ridge National Laboratory (ORNL) to LANL. Sample loading, i.e., attaching irradiation capsules to the SHERMAN sample stick, has to be possible in hot cells. Finally, pulsed neutron characterization, i.e., beam paths to the sample and the diffraction, neutron resonance spectroscopy, and radiography detectors have to be possible.

Once implemented, the combination of energy-resolved neutron imaging (ERNI), neutron resonance spectroscopy, and neutron diffraction will provide data for the entire irradiated volume that is at present not available. Utilizing the various parameters provided by this multi-modal neutron characterization, regular and irregular volumes can be identified and the regions with the highest scientific value for destructive post-irradiation examination can be identified and prepared for detailed analysis. These pulsed neutron techniques therefore maximize insight from expensive irradiation campaigns. As a step towards this goal besides the development of SHERMAN, an Advanced Test Reactor (ATR) irradiated U-10Zr-1Pd sample, which emits at a dose rate of ~3R/hr on contact and can be handled with remote handling tools, was characterized. This effort is described elsewhere in this report.

Figure 1. The SHERMAN logo

Accomplishments:

The technical goal of this project is to design a cask that allows to load, transport, handle on-site and characterize with several pulsed neutron techniques irradiated fuel samples. The design has to be authorized for operation at the irradiation and characterization facilities (James Angell/INL), e.g., INL and LANL, respectively. The cask has to fit into a transport cask for highly radioactive materials for transportation on public roads to transport the material from INL to LANL and back. Remotely operated shutters have to enable access with neutron beams and pathways for the transmitted and diffracted neutrons. Furthermore, the cask has to be aligned in the beamline as well as enable motion of the sample (irradiation capsule) e.g., for neutron tomography, scans along the axis of the capsule, or texture measurements. All of these operations have to

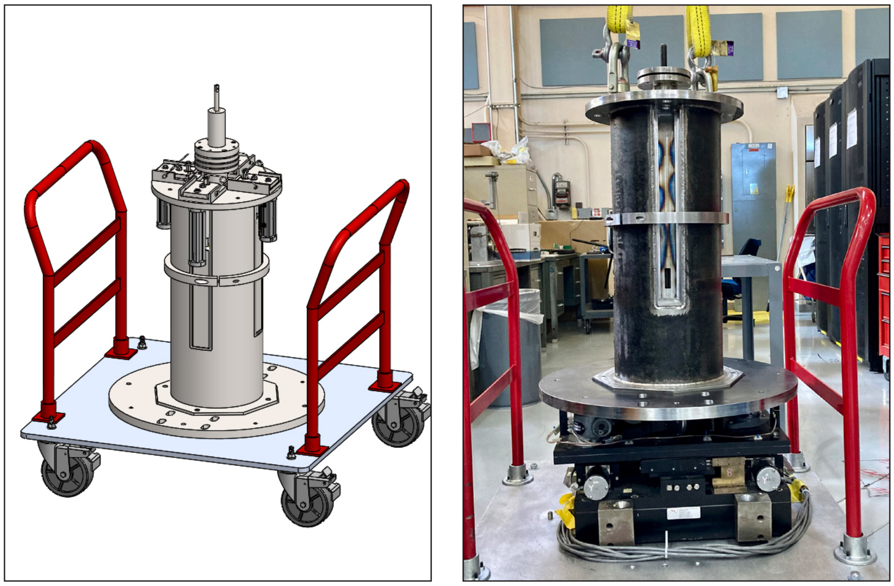

Figure 2. (Left) Drawing of the SHERMAN cask on the transportation cart for movement at LANSCE. (Right) Photograph of the prototype fabricated in FY21 on top of the translator stage for alignment in the beamline and on the cart. Both drawing and photograph are without the motion stage for sample movement in the beam for scans along the axis of an irradiation capsule or rotation for neutron CT.

meet safety criteria such as simulated drop tests (e.g., during crane operations, Vedant Mehta/LANL), shielding simulations (to design the cask such that the dose rate on contact of the cask is below 100 mrem/hr for safe handling by workers, Brian Gross/INL), and safe movement of the SHERMAN cask by crane or cart. As an important step, a time-of-flight neutron imaging detector to be used for this characterization at LANSCE was tested previously at the Neutron Radiography Reactor (nRAD) facility at INL with an irradiated fuel and was able to withstand the emitted dose rate while still collecting neutron imaging data (Aaron Craft/INL) [1,2].

After an initial design in collaboration with UC Berkeley (Peter Hosemann/UCB), the cask was designed over the past years following the aforementioned design criteria (Eric Larson, Travis Carver, Alex Long, Chris Fairbanks, Sven Vogel, all LANL) [3]. In fiscal year (FY)21, a prototype (funded by a LANL TED Pathfinder grant, PI Sven Vogel/LANL) was fabricated. Figure 2 (a) shows the design of the cask and the cart to be used for movement. Figure 2 (b) shows a photograph of the prototype on the cart with the translator table for alignment in the neutron beamline below. Both drawing and photograph show SHERMAN without the motion control stage to move the sample inside the cask. The photograph shows SHERMAN also without the beam shutters, providing a view on the beam window. The prototype is not yet filled with lead. During shielding simulations for the irradiated AFC-2D MOX several shine paths above the desired limit were identified. To mitigate those, tungsten bars and disks to be embedded in the lead inside the cask was purchased with a LANL infrastructure investment grant. The drop test simulations using the ABAQUS finite element software are on-going at the time of this writing (Vedant Mehta/LANL) and the purchased tungsten bars will also contribute to the structural stability of the cask. While design, fabrication and commissioning were affected by the COVID pandemic, in-beam tests at LANSCE are planned for the fall 2021. The LANL SHERMAN design team are shown with the SHERMAN prototype in Figure 3. This effort (development of the concept in collaboration with INL and ORNL, design, drop test and shielding simulations, commissioning etc.) is funded by AFC except for fabrication of the SHERMAN prototype and the purchase of additional tungsten shielding, for which LANL funding is gratefully acknowledged.

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Figure 3. LANL SHERMAN design team with the SHERMAN prototype (from left to right Vedant Mehta/finite element analysis, Sven Vogel/project lead & HIPPO beamline scientist, Chris Fairbanks/mechanical design, Alex Long/ERNI instrument scientist, Eric Larson/mechanical design, Andy Chaves/technician, missing is Travis Carver/motion control).

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High-Temperature Steam Oxidation of Irradiated FeCrAl in the Severe Accident Test Station

Principal Investigator: Yong Yan
Team Members/ Collaborators: Y. Yan, K. Linton, J. M. Harp, Z. Burns, T. Jordan, and B. Johnston

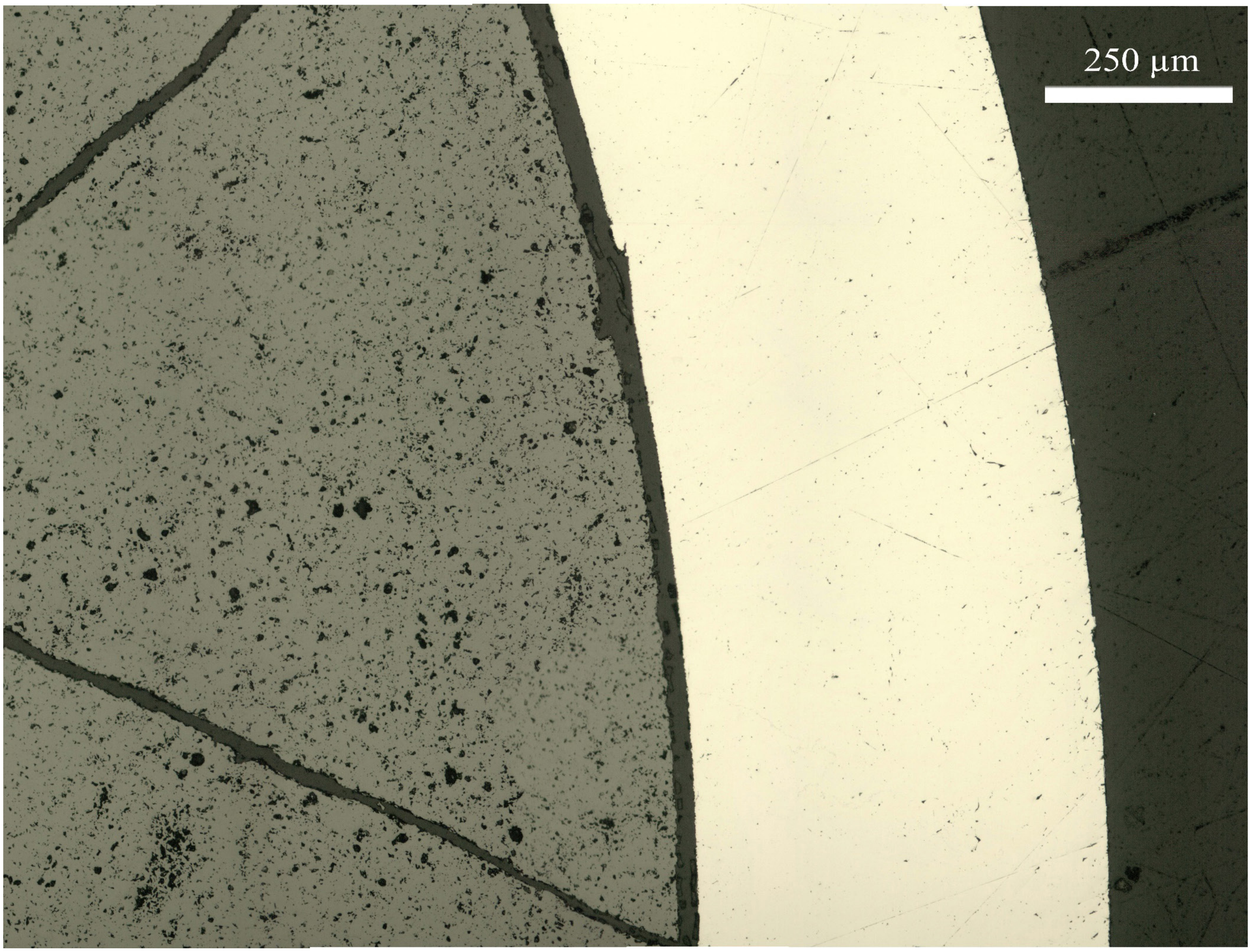

Figure 1. High-magnification image of ATF-18 cross section showing a gap between the irradiated FeCrAl cladding and UO_2 fuel

The first fueled irradiation experiments of Accident Tolerant Fuel (ATF) concepts, referred to as ATF-1, began in 2014 to evaluate fuel–cladding compatibility and provide irradiated material for future ATF testing. The irradiation included three rodlets of FeCrAl alloy C35MN (Fe-13Cr-5Al) cladding combined with UO_2 pellets fabricated at Oak Ridge National Laboratory (ORNL) targeting a range of burn-up values. One of the three rodlets, ATF-18, was irradiated to 10 GWd/MT at approxi-

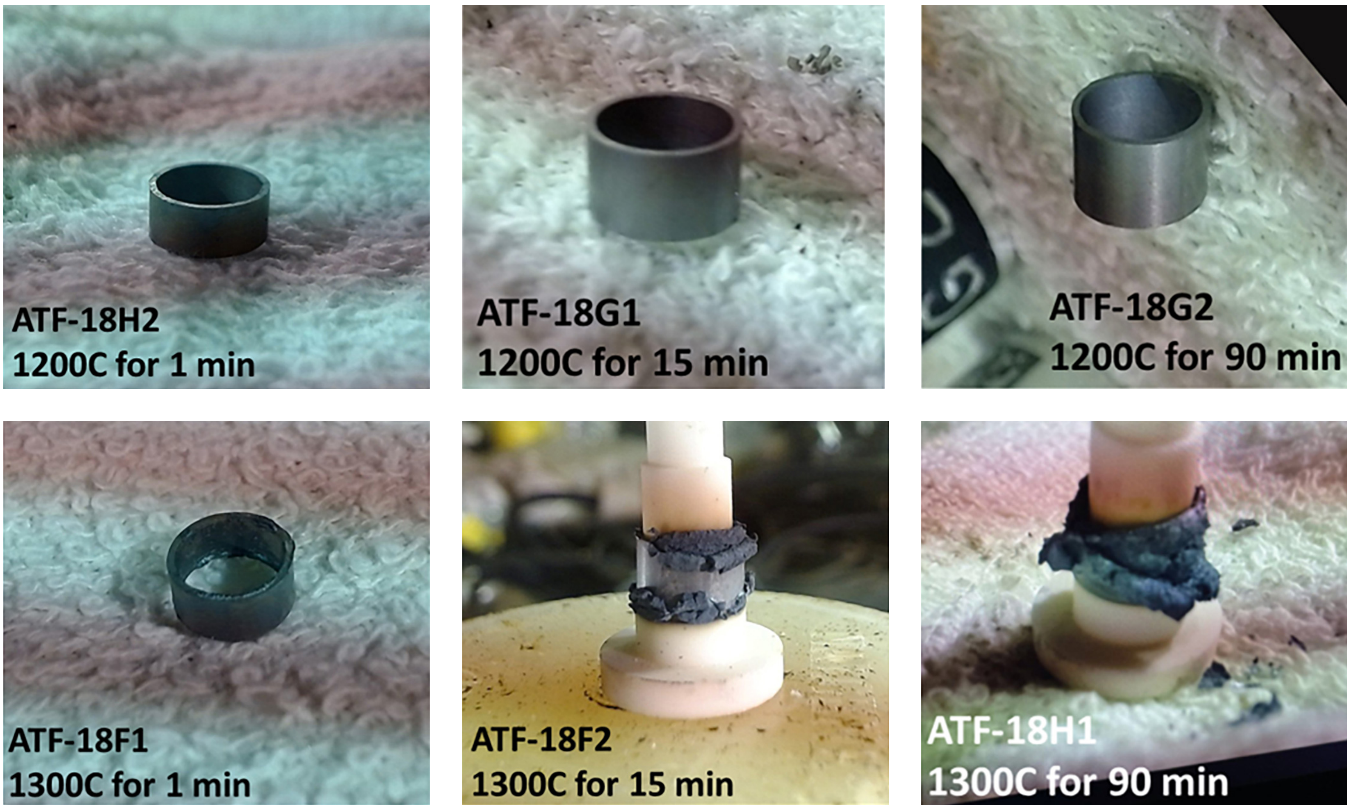

Figure 2. Images of irradiated FeCrAl after high-temperature oxidation testing at 1200ºC and 1300ºC for 1 minute, 15 minutes, and 90 minutes as indicated

mately 400°C. After irradiation, initial post-irradiation examination (PIE) was performed at Idaho National Laboratory (INL) and the rodlet was shipped to ORNL's Irradiated Fuels Examination Laboratory (IFEL) hot cell facility for additional PIE in the Severe Accident Test Station (SATS).

Project Description:

In fiscal year (FY)21, ATF-18 samples were sectioned from the irradiated rodlet and oxidation kinetics were evaluated to access the candidate cladding high-temperature oxidation performance following irradiation. Tests were conducted at 1200 and 1300°C with weight measurements taken before and after oxidation testing. Cross-sections of the cladding were metallographically mounted and optical microscopy was performed. Measurements of the oxidation layer before and after high-temperature testing were collected.

Pretest characterization was performed on the ATF-18 rodlet to determine the fuel, fuel-cladding bond, and cladding behavior. Figure 1 shows a

ORNL SATS high-temperature oxidation testing of irradiated C35MN FeCrAl showed resistance to oxidation up to 1200°C but quickly begin to oxidize at 1300°C.

high-magnification image of the gap between the fuel and cladding. A fuel-cladding bond was not formed, which is consistent with expectations for a low burnup irradiation.

All in-cell tests of irradiated ATF-18 specimens were conducted with a standard sample holder under the same test conditions for unirradiated surrogate specimens. The specimen was first ramped from room temperature to 600°C under an argon atmosphere at a rate of 20°C/min. Following this initial ramp, the argon supply was shut off, steam was supplied to the test section, and the temperature was ramped to 1200°C at 7.5°C/min. The sample was either held at 1200°C or slowly ramped to 1300°C at 1.82°C/min. The hold times and terminal temperatures for the in-cell tests are summarized in Figure 2.

The results indicate the irradiated FeCrAl C35M alloy provided good thermal stability up to 1200°C, though higher-magnification images revealed some cracks initiated from the outer surface. At 1300°C enhanced oxidation was observed on the inner surface at several areas of the ATF-18F1 (see Figure 3d), which was held at 1300°C for 1 min. This phenomenon was not observed for unirradiated ATF-18 surrogate specimens oxidized in steam at 1300°C up to 240 min.

Accomplishments:

This series of SATS testing at ORNL represents the first time that ATF-1 rods irradiated in ATR at INL were tested using a methodology first envisioned nearly a decade ago and realized through sustained Department of Energy- Nuclear Energy (DOE-NE) investment. The workflow demonstrated here has shown that Advanced Fuel Campaign (AFC) possesses the infrastructure and experience to perform integral irradiation testing of light water reactor (LWR) material concepts up to and including fuel safety testing. The results of these tests also provide evidence that irradiated FeCrAl alloys exhibit oxidation behaviors dissimilar from those observed in the unirradiated state. This is an important outcome that will motivate additional research to understand the performance of this family of alloys under design basis accident scenarios.

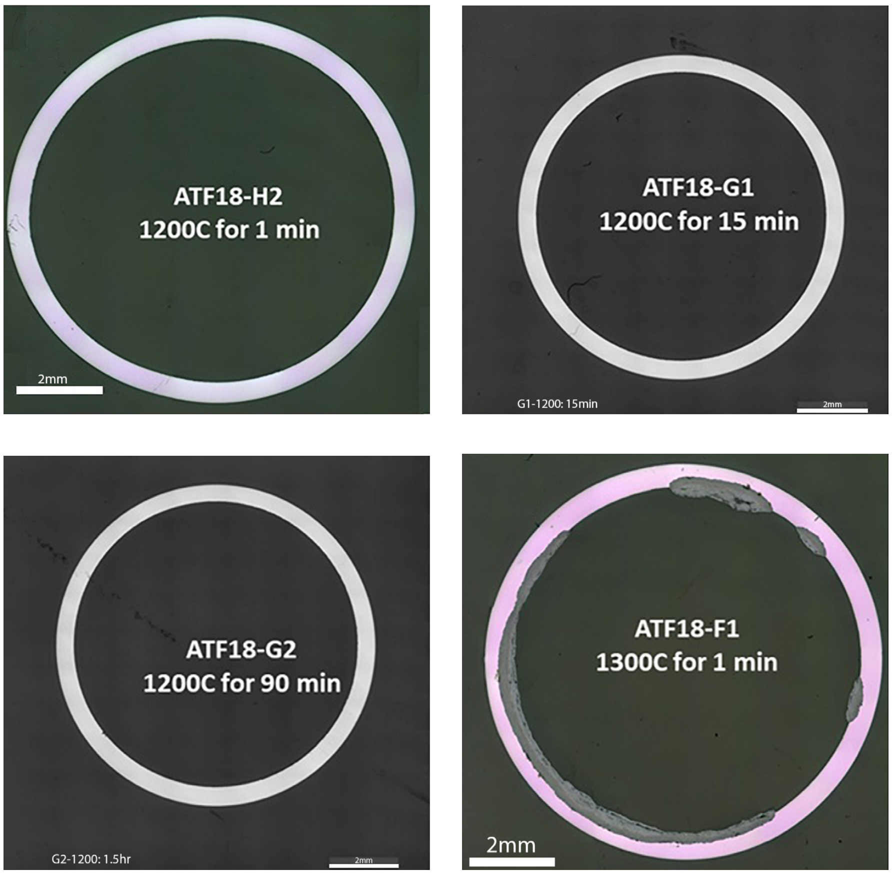

Figure 3. Low-magnification images of irradiated ATF-18 FeCrAl after high-temperature oxidation testing at the terminal temperature and hold times listed in the image text

Microstructural Characterization of High Burnup UO_2

Principal Investigator: Fabiola Cappia
Team Members/ Collaborators: Tsvetoslav Pavlov, and David Frazer

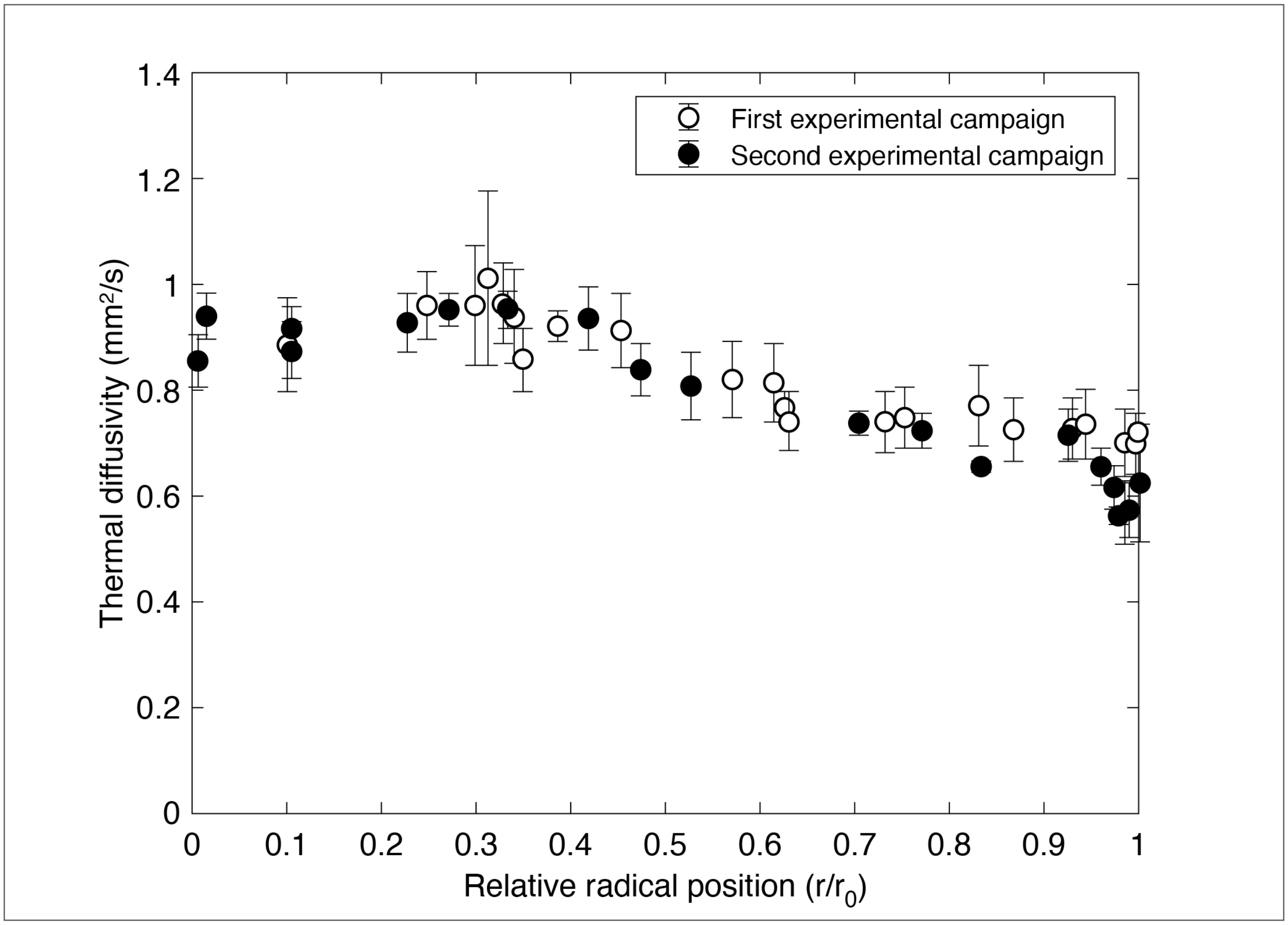

Figure 1. Porosity radial profile

The material behavior and properties at the microscopic scale are linked to the macroscopic performance that are of technological interest. Nuclear fuels undergo irradiation-induced modifications that deeply alter their structure and properties, thus affecting performance. Experimental evidence that quantifies such modifications are necessary to determine the fuel behavior and performance at high burnup. Significant progress has been made in the last two decades in the developments of analytical materials science techniques that can be applied to highly radioactive materials. The availability of new techniques has enabled investigations previously not possible which deepen the understanding of the fuel characteristics and properties at high burnup. The better knowledge of material behavior and irradiation-induced phenomena helps the prediction of its performance.

Project Description:

Safety and transient testing to evaluate performance under off-normal conditions is an essential pillar for both the development of Accident Tolerant Fuels (ATF) and the optimization of fuel operation economics beyond current discharge burnups. The knowledge of the initial microstructure of both fuel and cladding allows a more robust interpretation of the subsequent transient testing results, provides validation of the physical phenomena underlying the model predictions, and eliminates the uncertainties related to the limited knowledge of the sample status before the test, which has been shown to have significant impact on the transient testing outcomes. One example is the phenomenon of fine fragmentation that occurs in UO_2 light-water reactor (LWR) fuel during a Loss of Coolant Accident (LOCA) [1]. The amount of fine fragmentation increases with burnup, which has raised safety concerns due to the increased likelihood of dispersal of such small fuel particles after cladding burst and due to the increased transient fission gas release [2]. Therefore, efforts have been devoted to the assessment of the pulverization threshold that could determine the conditions under which fine fragmentation is predominant [3–7]. However, the lack of information regarding the initial conditions of the fuel and the connections between those conditions and the pre-transient irradiation history have hindered the development of a fully mechanistic fragmentation and pulverization criterion. In this context, the scope of the present work is to apply a wide portfolio of advanced characterization techniques to determine properties that are relevant for safety and performance of high burnup UO_2.

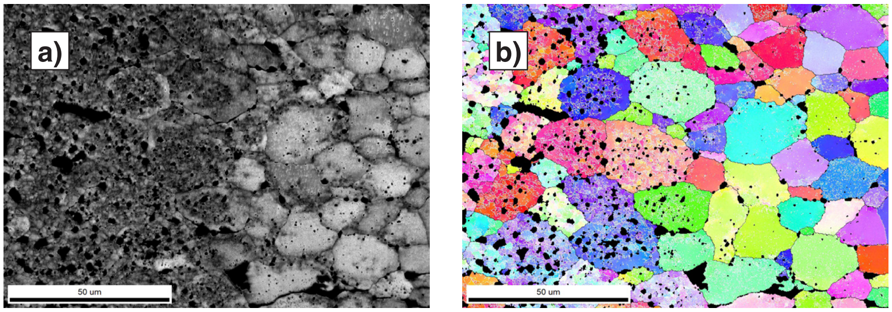

Figure 2. (a) Image quality (IQ) and (b) inverse pole figure (IPF) maps taken at radial positions r/r0 = 0.55–0.60 at the interface between the central restructured area and the non-restructured region

Accomplishments:

A fuel sample from a legacy high burnup rod available at Idaho National Laboratory (INL) has been analyzed using multiple experimental techniques. Three macro-areas have been investigated: (1) fuel microstructure including grain subdivision and porosity distribution, (2) thermal properties and (3) micro-mechanical properties. The sample had an average pellet burnup of 76 GWd/tHM and was cut from the high-power region of the mother rod. Images of the entire fuel radius were collected using a Thermo Fischer Helios Plasma Focused Ion Beam (PFIB) and processed via image analysis to obtain

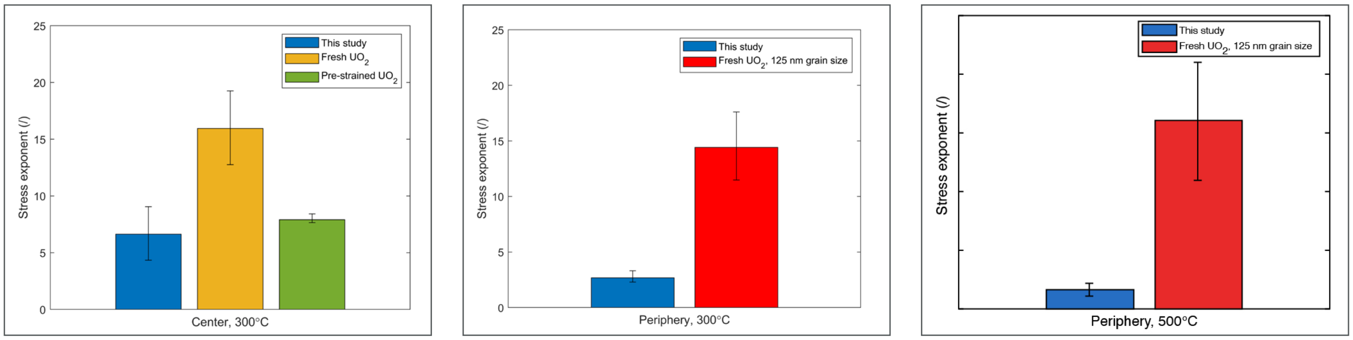

Figure 3. Comparison of the stress exponents measured on the irradiated fuel with those measured on fresh UO_2 with different microstructures. (left) The center region of the pellet at 300°C, the periphery at (middle and right, respectively) 300°C and 500°C

The current studies provided a comprehensive description of the irradiation-induced modifications occurring in high burnup UO_2 fuel and are inputs to any integral and semi-integral results interpretation, as well as source for meso-scale models to describe performance of high burnup fuels.

the radial porosity profile, showed in Figure 1. Electron Backscattered Diffraction (EBSD) orientation maps were collected using the EDAX Hikari Super EBSD detector of the PFIB and analyzed to study grain subdivision across the pellet radius. Three major zones were identified: (1) a central area till $r/r0 \approx 0.55$ characterized by grain subdivision by polygonization and porosity between 2% to 4%, (2) an intermediate region from $r/r0 \approx 0.55$ till $r/r0 \approx 0.75–0.80$ with low porosity and absence of grain subdivision and (3) the rim region with the exponential increase of porosity and grain subdivision typical of the high burnup structures (HBS) formation, both in transition and fully developed. Examples of the grain structure are shown in Figure 2. An Alemnis standard assembly equipped with the high-temperature module was used for the elevated temperature nanoindentation and nanoindentation creep experiments on the sample. The nanoindentation creep measurements were performed at 300°C and 500°C. The results of the nanoindentation creep tests are shown in Figure 3, where the stress exponents obtained for the irradiated fuel are compared to data from fresh and pre-strained fresh material. The results highlighted that the creep deformation mechanism of irradiated material is very different from the fresh material and similar to the pre-strained material. At 300°C the stress exponent for both the pre-strained sample and the irradiated UO_2 would suggest that the materials are deforming by dislocation glide in the material, while at 500°C the exponent is indicative of deformation by grain boundary sliding. The results highlight important effect of irradiation defects (i.e., dislocation) in determining the deformation behavior of irradiated material. Finally, local thermal properties have been measured using the Thermal Conductivity Microscope, a new technique developed at INL which provides both thermal diffusivity and thermal conductivity measurements with micrometric resolution. Figure 4 shows the radial profile of the thermal diffusivity as a function of the fuel radius. The data, in combination with the microstructural characterization could be used to validate microstructure-based meso-scale models of thermal properties, which are critical to correctly predict fuel temperatures

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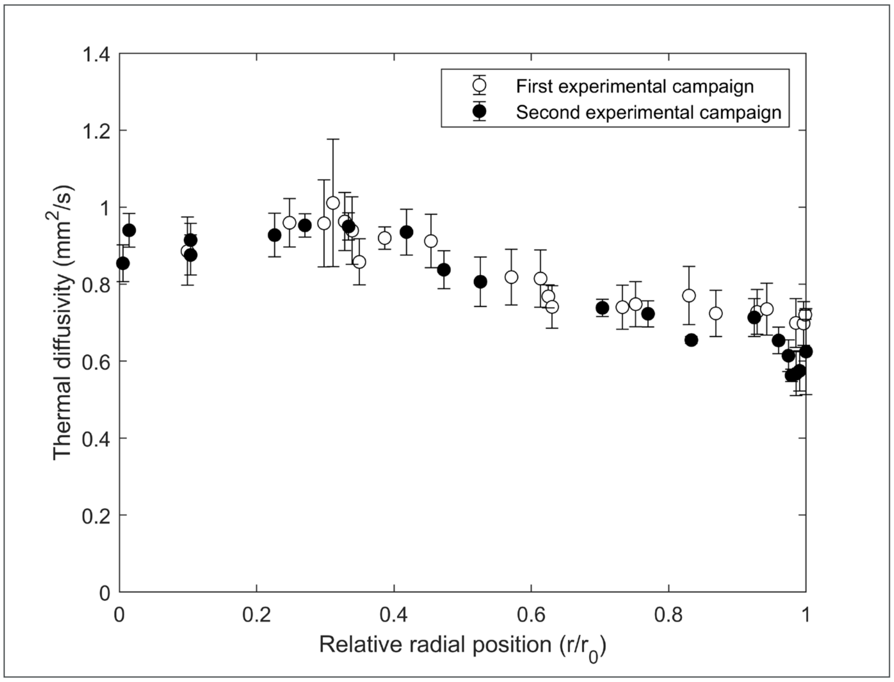


Figure 4. Radial profile of the measured thermal diffusivity of the high burnup UO_2 sample

Characterization and Testing of Cr-Coated Zircaloy

Principal Investigator: Stuart Maloy
Team Members/ Collaborators: Hyosim Kim, Nan Li, Cheng Liu, J. Kevin Baldwin, Kumar Sridharan (U. Wisconsin), Tyler Dabney (U. Wisconsin), Andy Nelson (ORNL), and Tim Graening (ORNL)

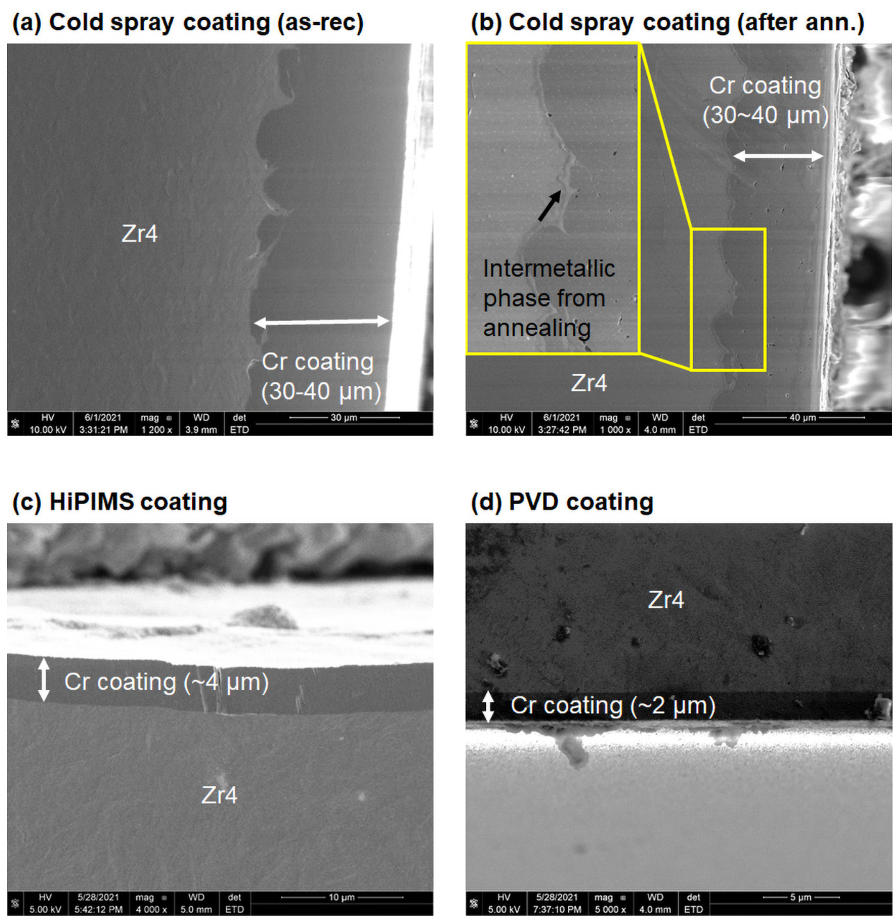


Figure 1. Scanning Electron Microscopy (SEM) images of (a) cold spray coating without annealing, (b) cold spray coating after annealing at 1000 °C for 1 hr, (c) HiPIMS coating, and (d) Physical Vapor Deposition (PVD) coating

Because of the events related to the Fukushima reactor incident, innovative new cladding materials are being investigated for light water reactors (LWRs) with improved resistance to accident conditions such as a Loss of Coolant Accident (LOCA). Materials need to be developed and tested to meet these challenging conditions.

Some engineering alloys are presently available with promising properties, but these alloys were not specifically developed for LWR applications. Thus, we are continuing development of Cr-coated zircaloy cladding material in collaboration with Oak Ridge National Laboratory (ORNL) and performing mechanical testing and ion irradiation testing on this alloy

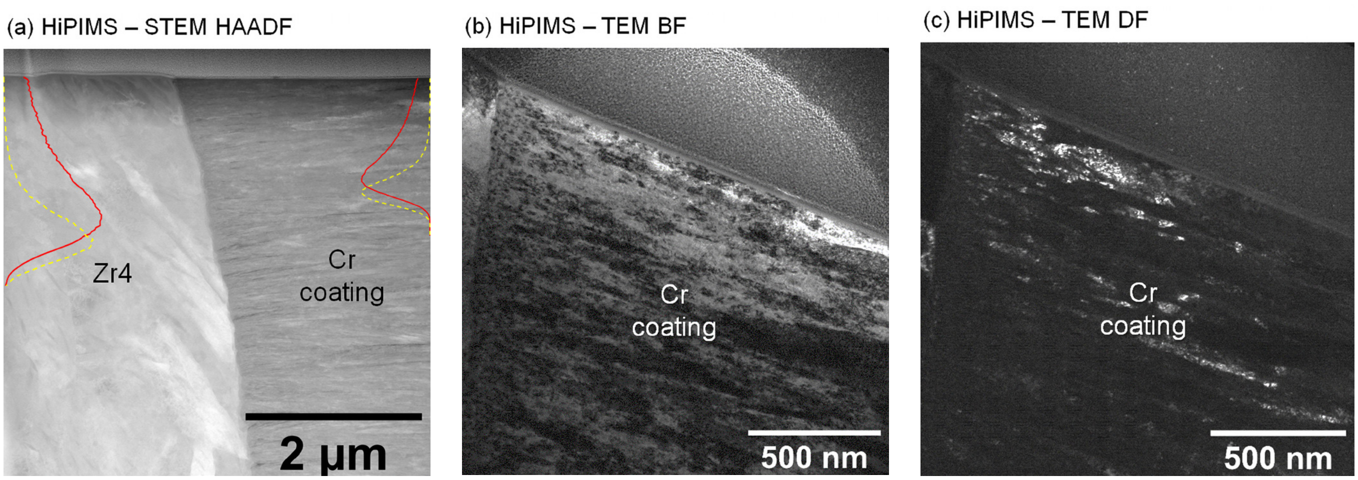


Figure 2. (a) Scanning transmission electron microscopy (STEM) high angle annular dark field (HAADF) image of HiPIMS coated Zr4 plate sample after 50 dpa irradiation with stopping and range of ions in matter (SRIM) dpa (red solid line) and Cr ion (yellow dashed line) profiles superimposed. (b) Transmission Electron Microscopy (TEM) bright field (BF) and (c) TEM DF images of the HiPIMS Cr coating after irradiation

Project Description:

The overall objectives of this research are to measure the interface strength of Chrome coatings on Zircaloy and understand the microstructural features that lead to this interface strength. In addition, ion irradiation is being used to understand how irradiation affects the interface strength.

Accomplishments:

Chrome coatings were produced by cold spray at the University of Wisconsin, by physical vapor deposition at Los Alamos National Laboratory and by High Power Impulse Magnetron Sputting (HiPIMS) at Acree Technologies Inc. and supplied by ORNL. Crossectional images of these coatings are shown in Figure 1. The thickest coating was produced by cold spray technology (40-50 microns) while the thinnest was produced using physical vapor deposition (2 microns). Ion irradiation was performed on the coatings in crossection to 50 dpa at 300°C using Cr ions showing the least amount of damage in the HiPIMS coating which has a very fine-grained microstructure (Figure 2). The fine grains provide sinks for defects produced under irradiation leading to improved radiation tolerance. An annealing treatment was performed on the cold spray coating at 1000°C and it produced an intermetallic at the interface. Upon irradiation of this annealed coating, the interme-

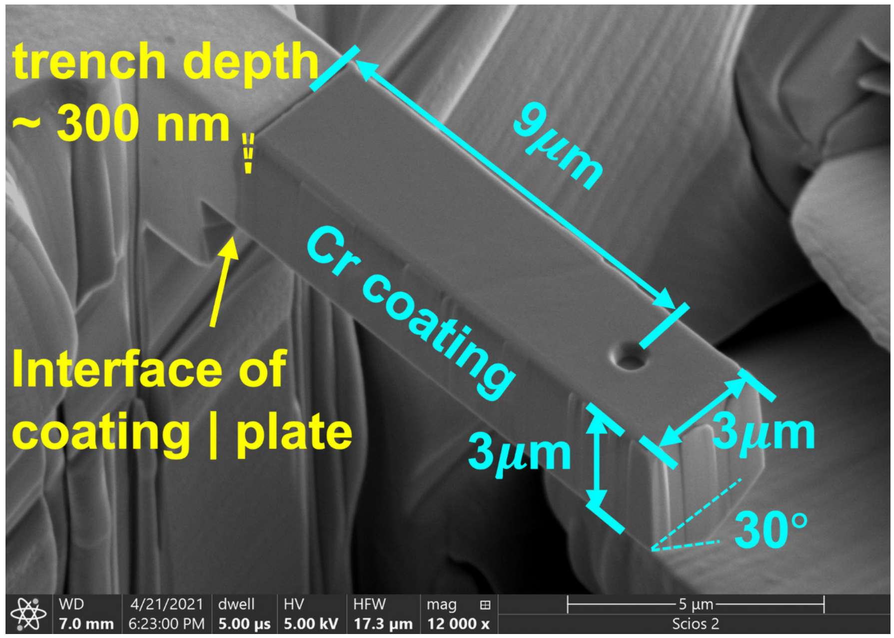


Figure 3. The SEM image of the tested notched cantilever with characterized dimensions

The results from this research show the detailed microstructural features in chrome coatings produced by vastly different manufacturing techniques and how they contribute to strength and irradiation tolerance in these coatings.

tallic became amorphous. Microscale bend testing was also performed on the coatings using a novel technique where the bend specimen is produced with focused ion beam (FIB) machining (Figure 3). Figure 4 shows an image of the bend specimen in the middle of the test showing the elastic and plastic deformation of the microbend specimen during testing. A strong university and lab collaboration was required to be successful in this research. The results show some of the first interface measurements on these coatings produced by different manufacturing techniques and initial measurements of the radiation tolerance of these coatings using ion irradiation.

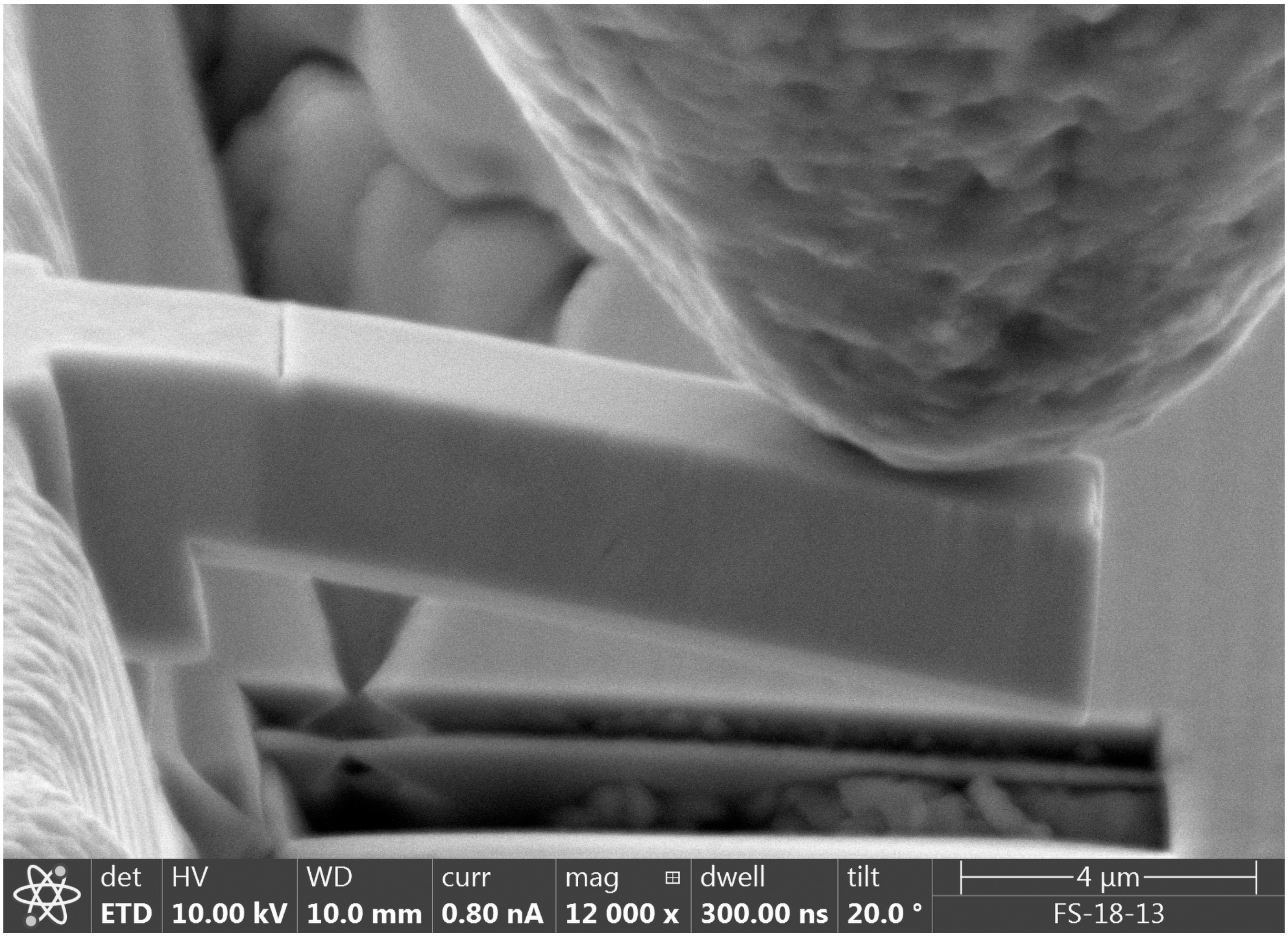


Figure 4. SEM image of the cantilever beam during the bend test

Cr-Coated Characterization of Zircaloy-4 Claddings

Principal Investigator: Tim Graening
Team members/ Collaborators:Kory Linton, Nathan Capps, Ben Garrison, Ken Kane, Peter Mouche, Ryan Sweet, and Andy Nelson

Cr-coated Zircaloy-4 tubes have been produced to facilitate an improved understanding of the relationship between coating properties, architecture, and defects and overall cladding performance.

Accident Tolerant Fuel (ATF) concepts have been developed and tested in diverse research programs presently around the world. Industry teams have developed proprietary coating compositions and geometries, but significant knowledge gaps exist in terms of the resulting properties of the coating and the cladding and how those components impact each other in reactor conditions. Furthermore, the national laboratory and university teams currently studying the industry concepts are limited in their ability to freely disseminate and discuss results obtained on commercial coatings. The Advanced Fuels Campaign (AFC) is therefore working to develop an understanding of how coatings applied to zirconium cladding alloys impact a broad range of performance factors including fundamental properties, steady state performance, and transient test conditions. These capabilities and experience will enable improved fundamental understanding of coating behaviors and strengthen AFC's ability to support Nuclear Regulatory Commission (NRC) to license coated Zr cladding concepts.

Project Description:

The objective of this research is to develop and investigate Cr-coated Zircaloy claddings to identify and overcome current knowledge gaps. Closing those gaps ultimately enables us to simulate and understand the coated cladding behavior inside a reactor under normal operation and during accident scenarios. The first step towards that goal is to provide an initial assessment of the most important coating properties and the experimental methods available for their determination. An extensive literature study was conducted to develop standardized property requirements, measurement methods, and reporting guidance for coated fuel claddings. Most critically, the manufacturing of a non-proprietary Cr coating is needed to facilitate microstructural examinations, characterization of baseline mechanical properties, and to supply materials for transient testing and eventual irradiation testing. Coated cladding manufacturing employs many different coating methods, which all result in different properties. All three ATF industry teams include coated Zr concepts in their portfolio, but the

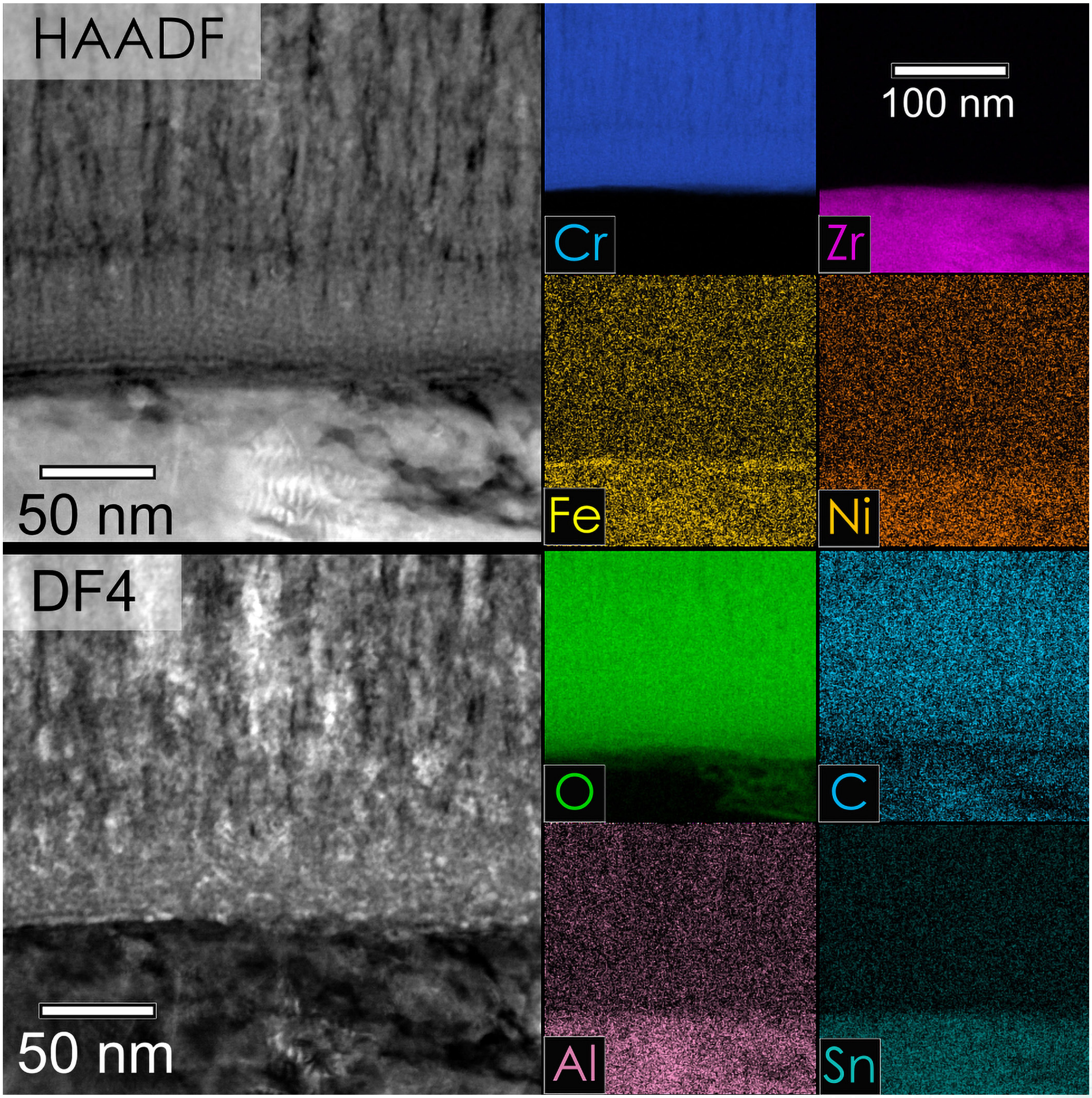


Figure 1. TEM EDS map of the interface between Cr coating and Zircaloy-4 cladding

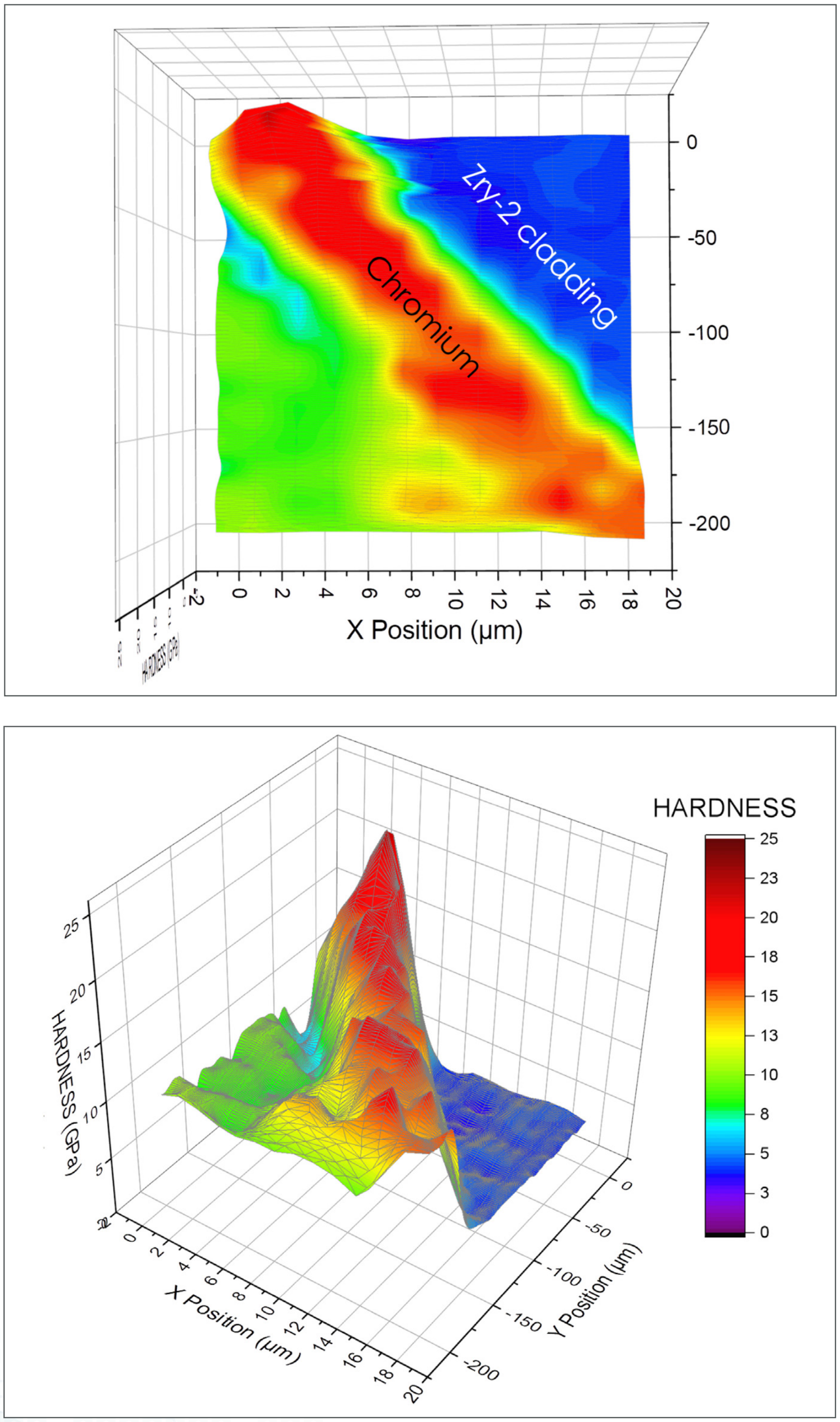


Figure 2. Nanoindentation of Cr-coating on Zry-2

details of processing remain proprietary. Test results are similarly guarded. These factors combine to limit open dialogue, discussion, and improvement of testing methods and coating performance. This reality prompted AFC to begin efforts to produce coated claddings free of these restrictions. A range of coating methods were explored in fiscal year (FY)21, characterized, and testing in the as-received condition and after both loss of coolant accident (LOCA) and high strain rate testing to establish the baseline for further investigations.

Accomplishments:

A comprehensive literature study was performed and resulted in a priority rating system for coating-cladding properties as informed by sensitivity analyses performed using BISON. Thermal expansion, residual stress and creep properties are the most important properties to investigate. Chromium was chosen as the coating given its prevalence across the industry concepts and existing literature demonstrating its performance across relevant conditions.

Selected parameter sets for High Powder Impulse Magnetron Sputtering (HiPIMS) of Cr on Zry-2 and Zry-4 tubes were tested to create a strong adhesive Cr-coating on the Zircaloy cladding with a 7- and a 5-micron thickness, respectively. Transmission electron microscopy (TEM) energy dispersive spectroscopy (EDS) methods were utilized to investigate the interface shown in Figure 1, but these investigations did not reveal a brittle Laves phase. Nanoindentation maps between coating and cladding material, displayed in Figure 2, were utilized to determine hardness and elastic modulus across the interface are necessary parameters for future simulations. Coatings were first attempted on the as-received cladding surfaces, but it was determined that the surface roughness of the pilgered tubes was too high to achieve continuously adhered coatings.

Coated tubes containing surface defects were used to test the potential impact of a cracks in the Cr shell using modified burst tests (MBTs) to simulate reactivity-initiated accidents (RIA) and burst tests in the severe accident test station at ORNL to simulate LOCA. Digital image correlation and a mirror setup was used during MBT to measure the strain on the entire circumference of one tube specimen as shown by the four projections on the same specimen in Figure 3. It was found that both the coated and uncoated material did not burst at hoop strains up to and above 8% at room temperature and at 275°C. Furthermore, at the plane strain state, the coated material did not burst at a hoop strain of 6%. For LOCA performance, it was found that the uncoated and coated material both followed the historical Cathcart-Pawel relation for burst temperature vs. hoop stress as shown in Figure 4. Cracked coatings have not shown a negative impact on the base properties of cladding during those tests.

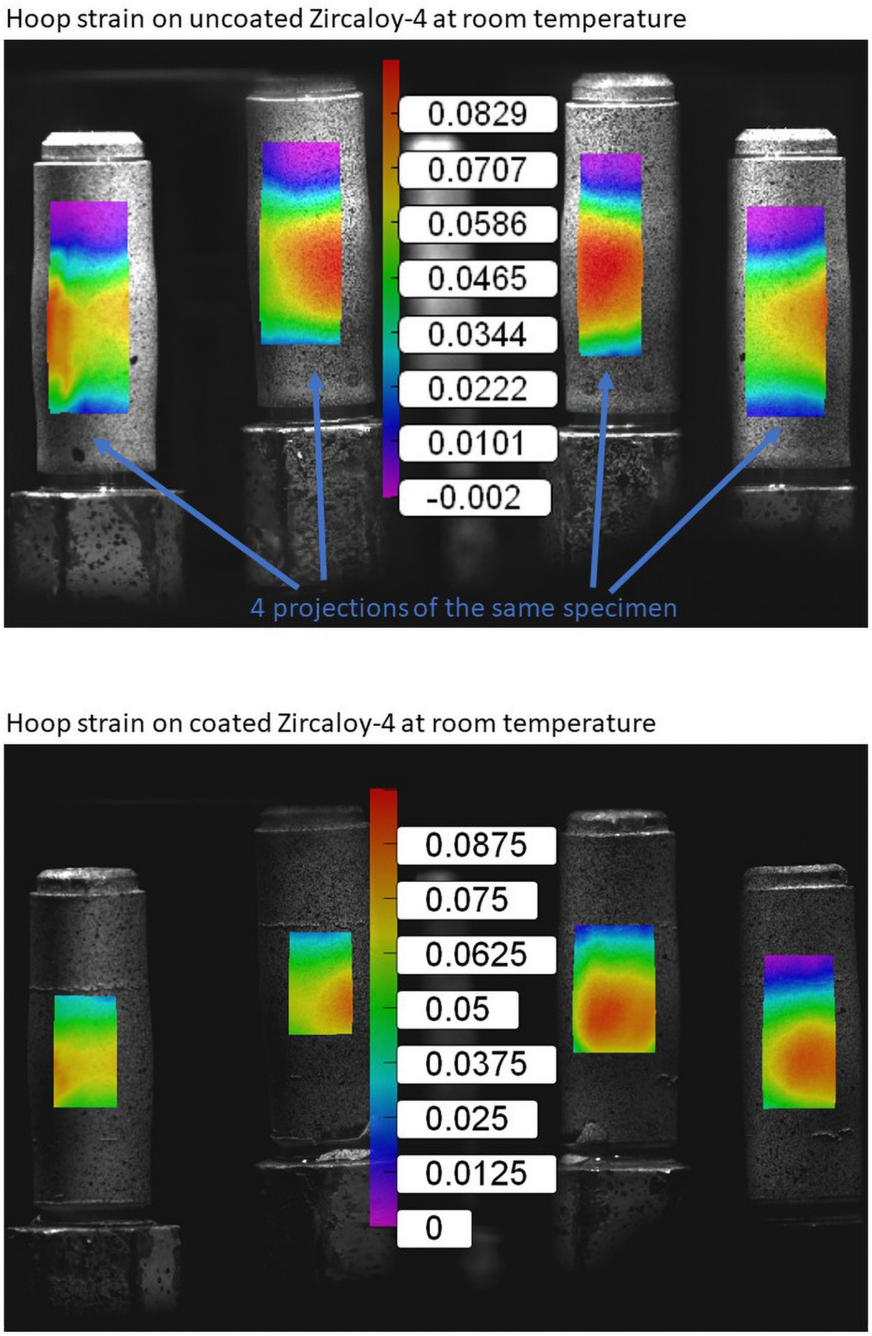


Figure 3. MBT hoop strain measurements using DIC for uncoated and coated Zircaloy-4 at room temperature

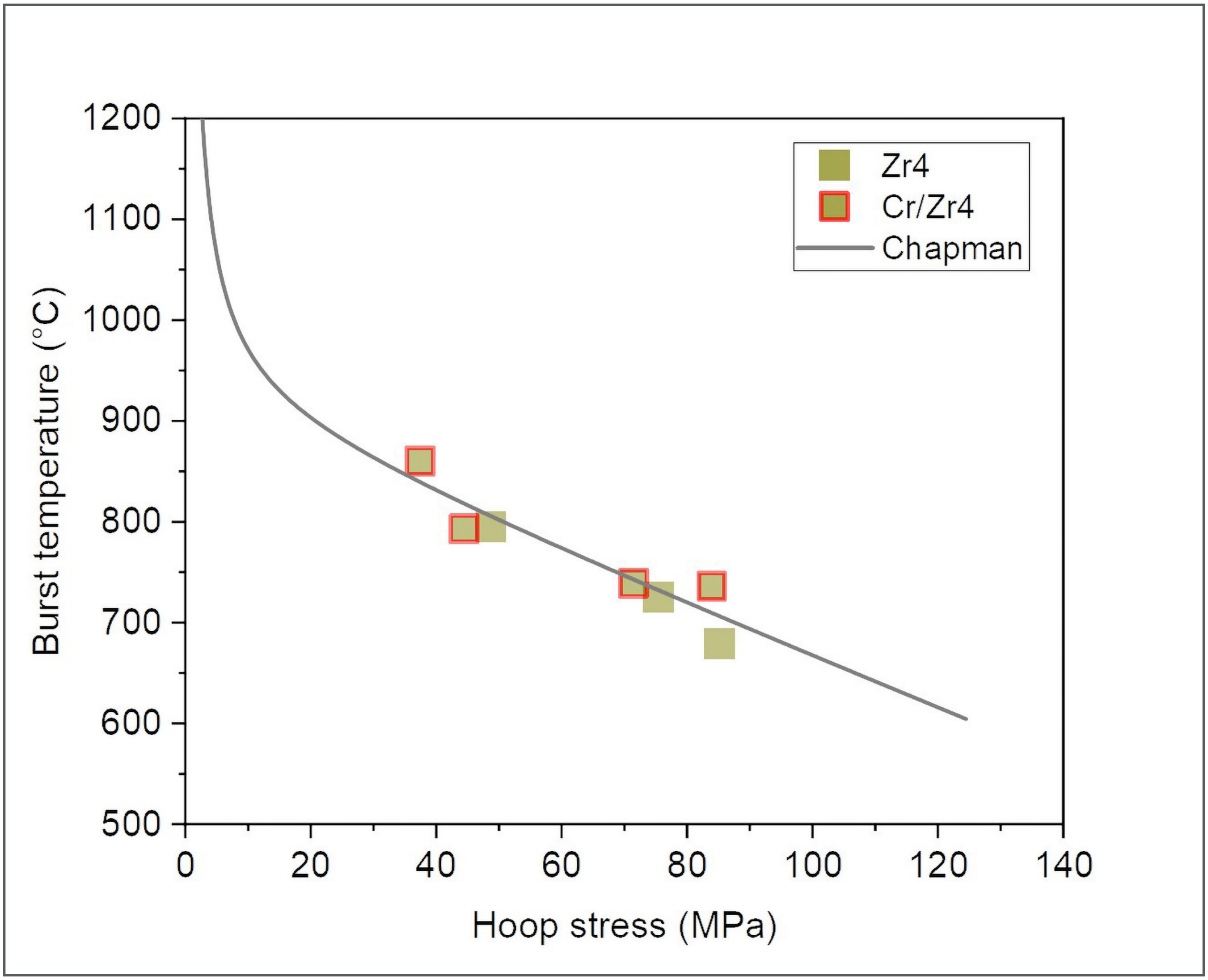


Figure 4. LOCA burst temperature vs. hoop stress

Coating development using HiPIMS was successful in the next attempt using three different parameter sets, shown in Figure 5 with different magnifications. All parameter sets exhibit no visible cracks, however one parameter set showed ablation of the coating, caused by very high compressive stresses in the coating. Cr coatings have been applied to both flat coupons, PWR, and BWR tube geometries for testing performed at both Oak Ridge National Laboratory (ORNL) as well as Los Alamos National Laboratory (LANL). Ongoing work will include broader property and performance investigations including planned irradiation testing to begin in FY22

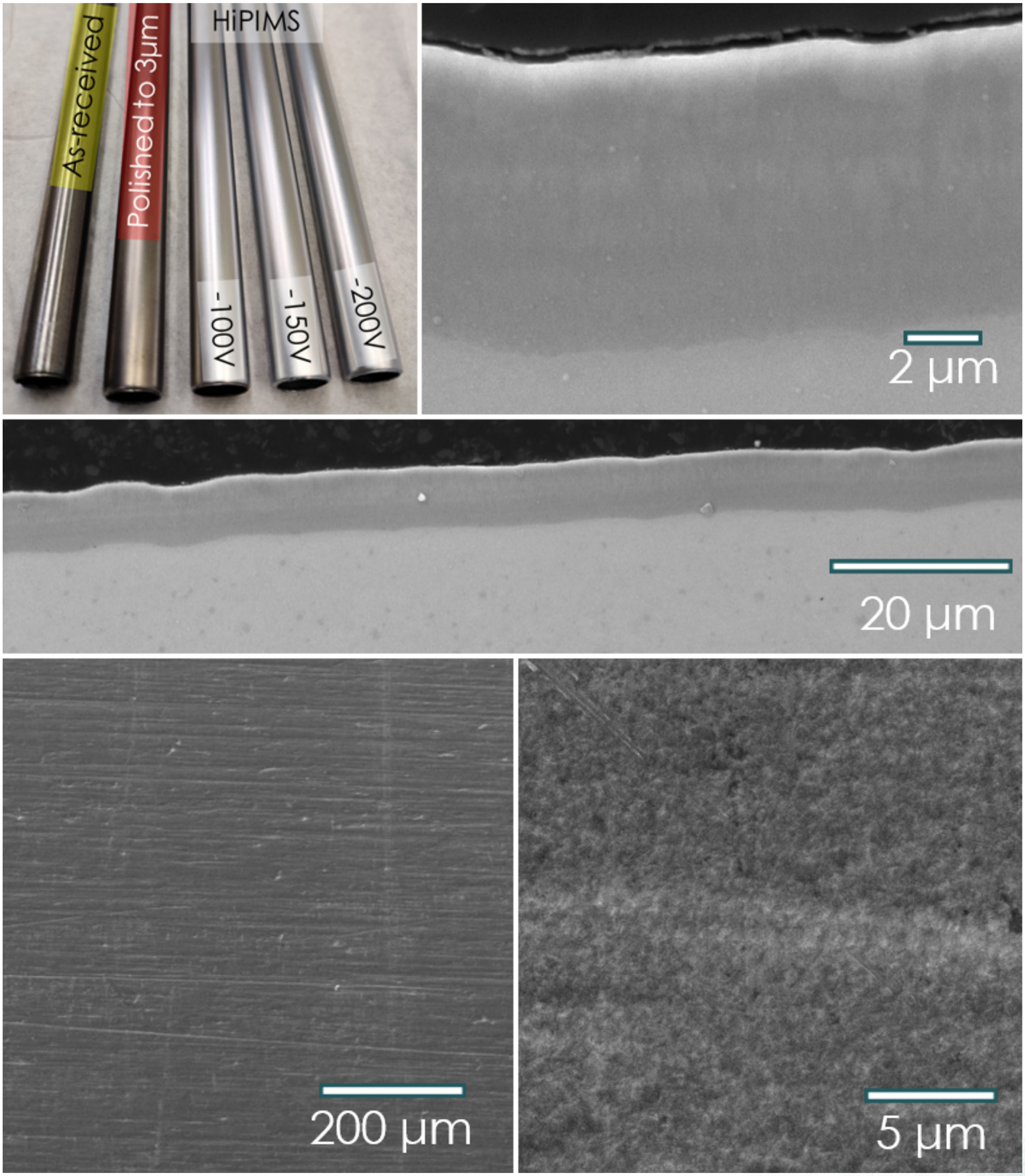


Figure 5. Cr-coated Zry-4 tubes with an adhesive coating without any cracking in radial and axial orientation

SiC/SiC Channel Box Irradiation

Principal Investigator: Christian Petrie
Team Members/ Collaborators: Wilna Geringer, Adam James, Kurt Smith, Joseph R. Burns, Annabelle Le Coq, Nicholas Russell, Christian Deck, Takaaki Koyanagi, and Yutai Katoh

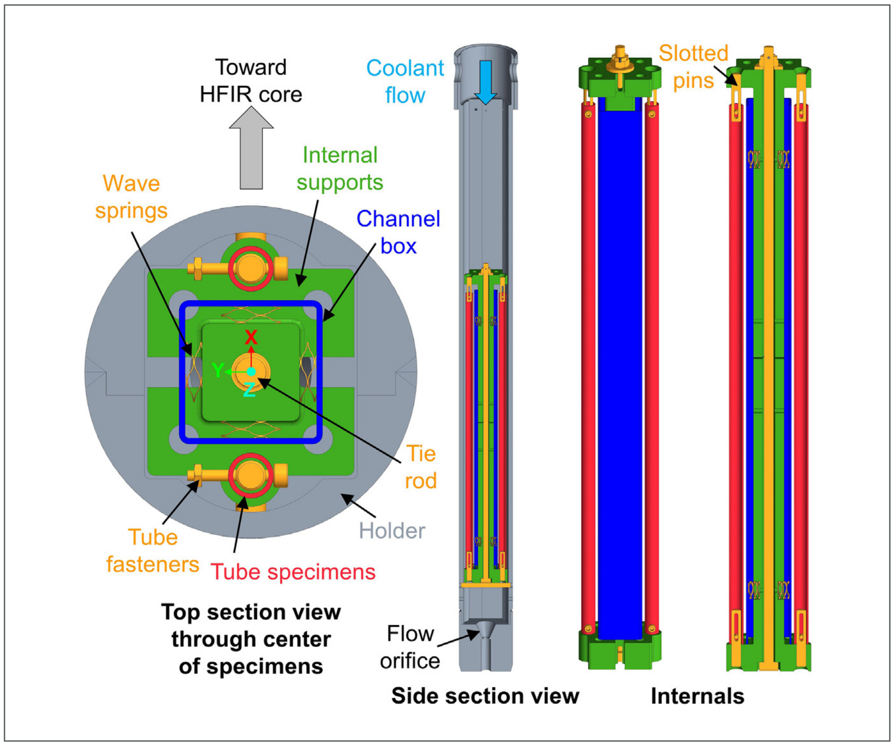


Figure 1. Design of the channel box experiment

Silicon carbide (SiC) fiber–reinforced, SiC ceramic matrix composites (SiC/SiC composites) have long been considered as a potential accident tolerant fuel (ATF) because of their high-temperature strength, dimensional stability under irradiation, minimal neutron absorption, and superior oxidation kinetics compared to zirconium (Zr) alloys in steam and air environments. In addition, SiC/SiC materials could be a nearer term replacement for current Zr alloy channel boxes in boiling water reactors (BWRs) that could mitigate the significant heating and hydrogen generation caused by exothermic steam oxidation reactions with Zr-based materials during severe accidents. However, there is concern that spatial variations in neutron flux and/or temperature could result in significant lateral bowing of a SiC/SiC channel box due to differential radiation-induced swelling. This bowing could potentially interfere with coolant channels or control blade movements. While some recent models have predicted the extent of bowing for a SiC/SiC channel box, those models are lacking in validation from experimental irradiations.

Project Description:

The objective of this project is to experimentally evaluate the lateral bowing of a miniature SiC/SiC channel box and SiC/SiC tubes (for fuel cladding applications) following neutron irradiation under a fast neutron flux gradient. General Atomics fabricated a 38 cm long channel box with a 30 mm × 30 mm cross-section as well as SiC/SiC composite tube specimens with typical pressurized water reactor cladding dimensions. A novel irradiation vehicle was designed (Figure 1) to accommodate the specimens and allow them to bow and swell axially without constraint during neutron irradiation in Oak Ridge National Laboratory (ORNL's) High Flux Isotope Reactor (HFIR). Within the permanent reflector of the reactor, the fast neutron flux decreases by ~35% moving from the

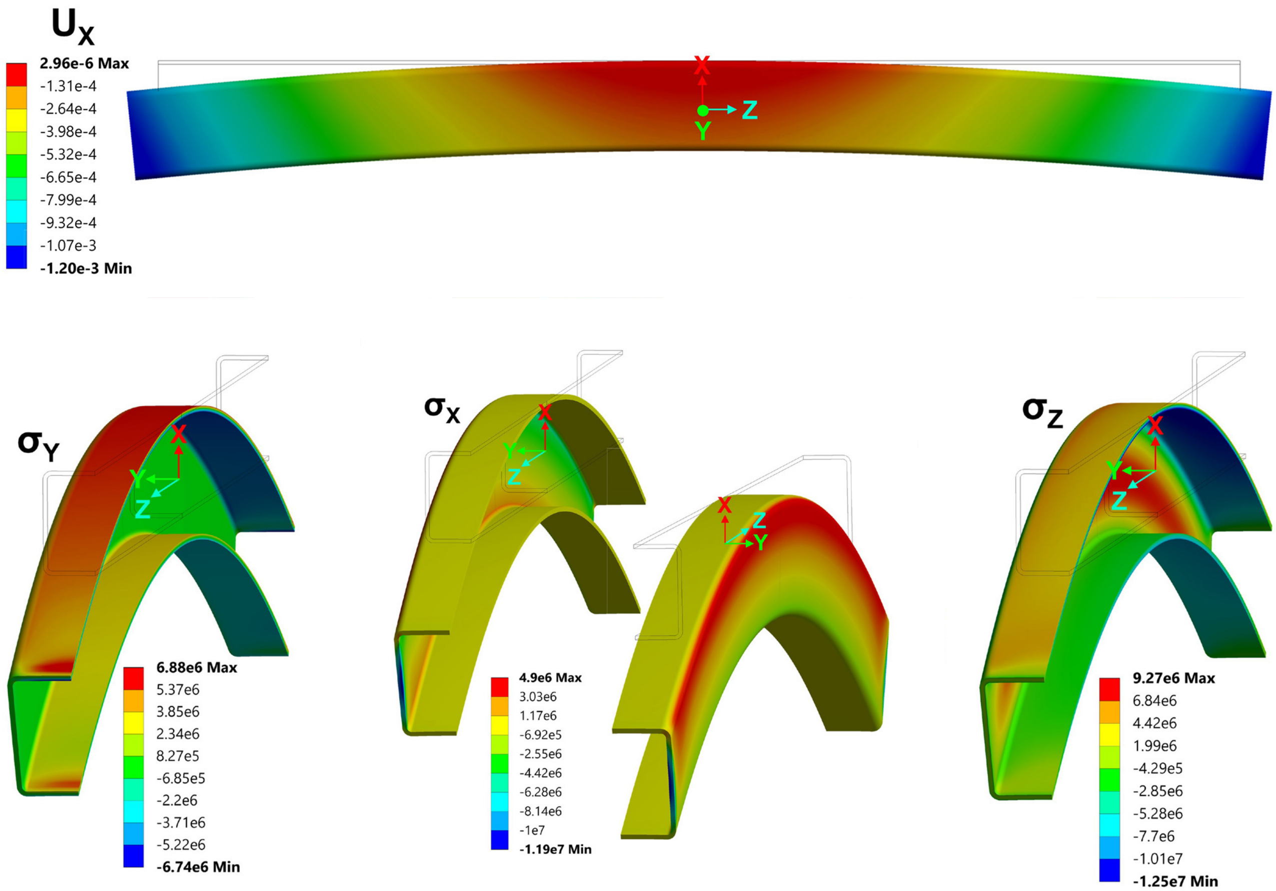


Figure 2. Simulated channel box bowing after two HFIR cycles showing (top) lateral displacement UX (in m), with visual deformations artificially scaled by a factor of 10, and (bottom left, middle, and right, respectively) normal stresses (in Pa) with visual deformations artificially scaled by a factor of 50

core-facing to rear-facing sides of the channel box. The goal of this work is to accurately characterize the geometry and surface profile of the specimens before and after irradiation to evaluate local deformations and overall lateral bowing and compare with values predicted using high-fidelity modeling and simulation. Surface profilometry was performed using a custom measurement system that utilizes an optical micrometer attached to a precision stage. In addition, a digital microscope was attached to the same stage and used to record the precise positions of fine engraving marks that were etched along all faces of each sample. This allows for determination of local strain in addition to macroscopic dimensional changes.

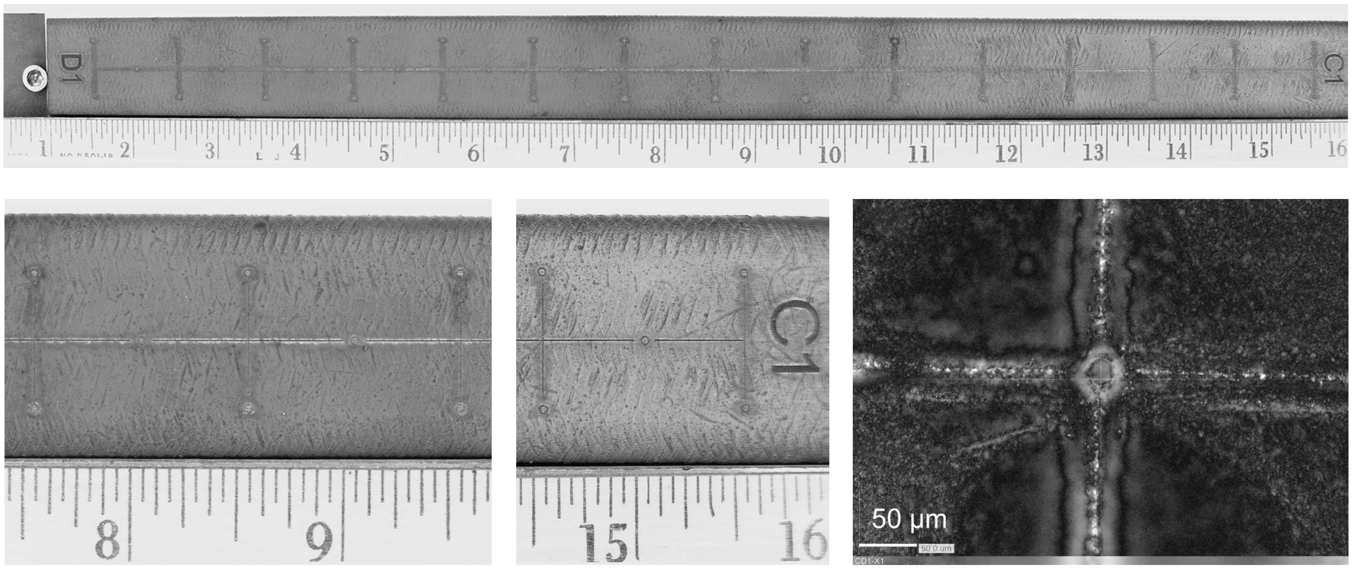
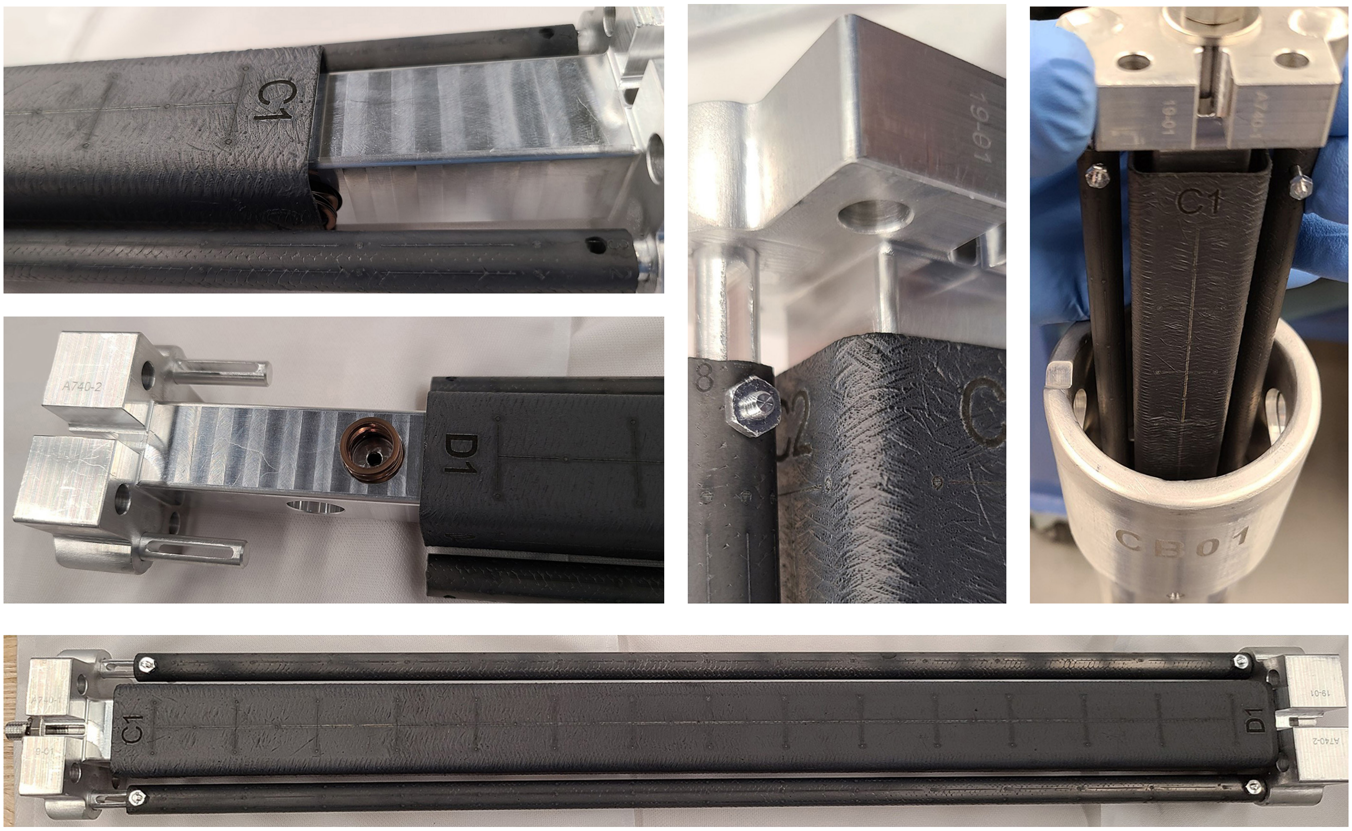


Figure 3. Photograph (top) of the engraved channel box specimen; (left and middle, respectively) closer views of the engraved markers and (right) a processed image of a single marker

This is the first neutron irradiation of a SiC/SiC composite channel box to evaluate the extent of lateral bowing under a fast neutron flux gradient to ensure that bowing of a SiC/SiC BWR channel box would not interfere with control blade movements during commercial reactor operation.

Accomplishments:

Detailed three-dimensional Monte Carlo neutronic calculations were performed to determine the spatial variations in dose rate throughout the entire sample geometry. These dose rates were used as inputs to structural finite element simulations along with swelling correlations that were previously established in the literature to predict the deformation of the specimens due to spatial variations in dose. The ~35% reduction in dose rate from the front face (facing toward the core) to the back face (facing away from the core) of the 30 mm square channel box specimen results in a calculated differential radiation-induced swelling of 0.55 vol % after two irradiation cycles. The resulting bowing is calculated to be 1.02 mm (Figure 2) at the end of the two-cycle irradiation. Calculated normal stresses are on the order of megapascals and less than 13 MPa for all evaluated cases. In addition to typical dimensional inspections (length, width, depth, and wall thickness), the channel box specimen was engraved with fine markers for mapping local strains (Figure 3) and



profiled using an optical micrometer to accurately characterize any bowing that may be present before irradiation. Those measurements will be repeated post-irradiation to quantify radiation-induced bowing. The pre-irradiation bowing was measured to be as large as 0.67 mm on one face and as small as 0.25 mm on another, indicating that the expected radiation-induced bowing (~1 mm) can be measured. The spacing between engraved markers was measured using a 3D stage and a digital microscope. Additional analysis is being performed to provide a fine adjustment to the marker spacing by determining the distance from the center of each image to the centroid of the engraved markers. It is expected that the spacing can be quantified to within less than 10 μm, which is equivalent to ±0.01% linear strain (±0.03% volumetric swelling). The experiment was successfully assembled (Figure 4) and inserted into HFIR during cycles 492 and 493 (May through July of 2021). During these cycles, the samples were irradiated for a total of 51.2 effective full power days, resulting in a maximum dose of 0.09 dpa. Efforts are underway to ship the experiment to the Irradiated Fuels Examination Laboratory (IFEL) for post-irradiation examination.

Figure 4. Photographs of the internal assembly

Assessment of the Fission Gas Release Model in BISON Applied to UO_2 MiniFuel

Principal Investigator: Amani Cheniour (Oak Ridge National Laboratory)
Team Members/ Collaborators: Giovanni Pastore (University of Tennessee, Knoxville), Jason Harp (Oak Ridge National Laboratory), Christian Petrie (Oak Ridge National Laboratory), and Nathan Capps (Oak Ridge National Laboratory)

The results of this analysis have been used to develop a UO_2 MiniFuel irradiation test matrix is to help to extend the model's operational range and to generate new data to validate the FGR model capabilities to higher burnups and transient conditions.

The manuscript provides a comprehensive assessment of the current BISON fission gas release model and its ability to model fission gas release in MiniFuel. Furthermore, the model was validated to high burnup fission gas release experiments to assess its ability to accurately predict high burnup fission gas release.

Project Description:

The initial MiniFuel fuel performance analysis was designed to inform subsequent MiniFuel irradiations campaigns. BISON was used to model and predict steady-state and transient fission gas release (FGR) for a variety of conditions. A representative BISON model was developed with the appropriate thermal hydraulic boundary conditions to accurately replicate MiniFuel irradiation conditions. The model was designed to assess the impact of temperature, sample burnup, grain radius, fission rate, and diffusion constants on FGR. Current monolithic MiniFuel experiments generally target irradiation of fuel samples at constant temperatures, ranging from 500 to 600°C, and the simulated amount of FGR outside of athermal release would not be measurable. BISON results indicate that high temperatures >900°C are required to observe appreciable FGR (1%) at burnups <62 MWd/kgU. Above 62 MWd/kgU, FGR will occur at temperatures >800°C; however, fuel sample burnup must exceed 100 MWd/kgU before appreciable FGR is observed at lower temperatures. These results indicate that MiniFuel irradiations would need to consider extremely high burnups (100 MWd/kgU) or high temperatures >800°C at 62 MWd/kgU to measure FGR under steady-state conditions. Low-temperature MiniFuel irradiations may still offer the possibility to provide microstructural information regarding fission gas (FG) diffusion, bubble nucleation, bubble growth, and FGR prior to a temperature transient. Furthermore, these microstructural features could help identify key mechanisms leading to pulverization during a temperature transient. Future work should consider developing temperature-transient testing capabilities to evaluate transient FGR as well as pulverization of MiniFuel samples following irradiation in High Flux Isotope Reactor (HFIR). The application of transient conditions in UO_2 fuels significantly affects the amount of released FG to the plenum volume in a fuel rod. The fuel performance code BISON can model transient FGR in UO_2 and doped UO_2 fuels. The BISON FGR model includes physics-based diffusion-controlled

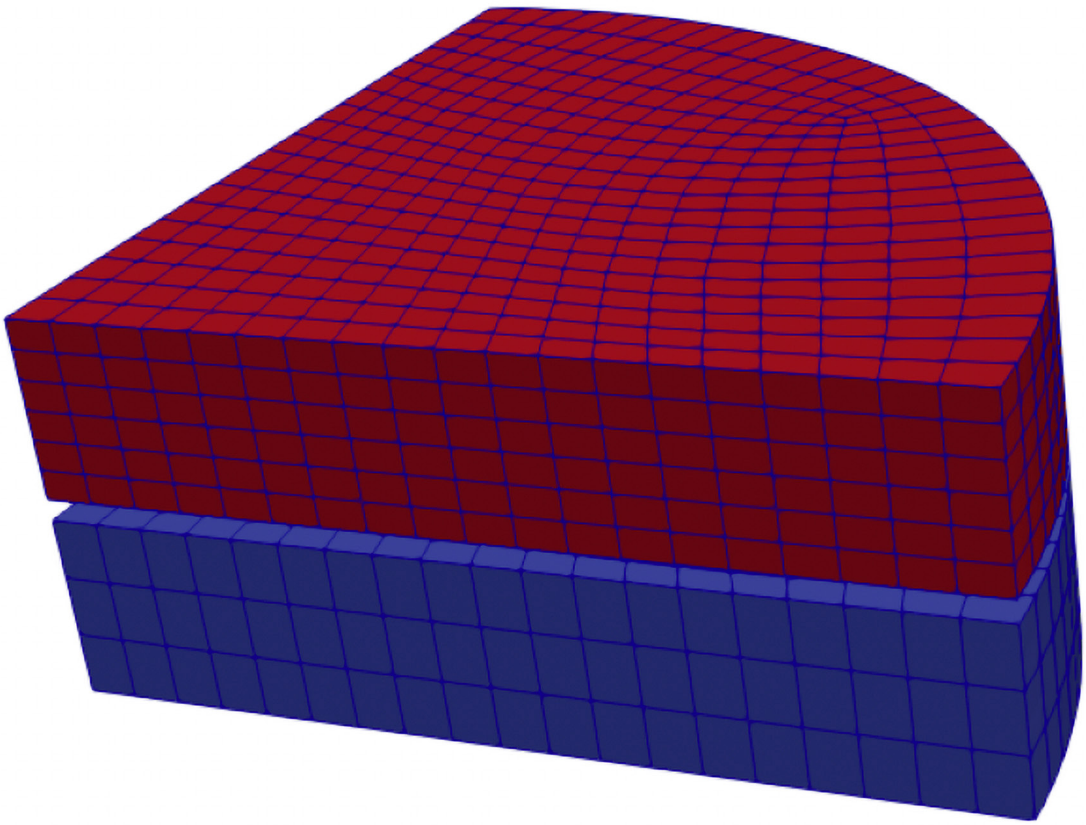
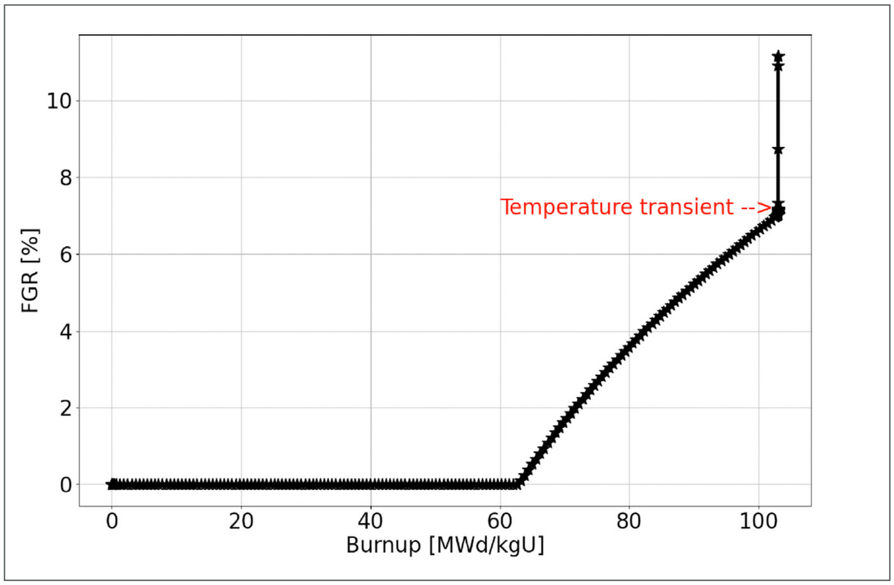


Figure 1. MiniFuel mesh and geometry showing the fuel disk in red and the refractory metal cup disk in blue (quarter disks are used assuming symmetry)



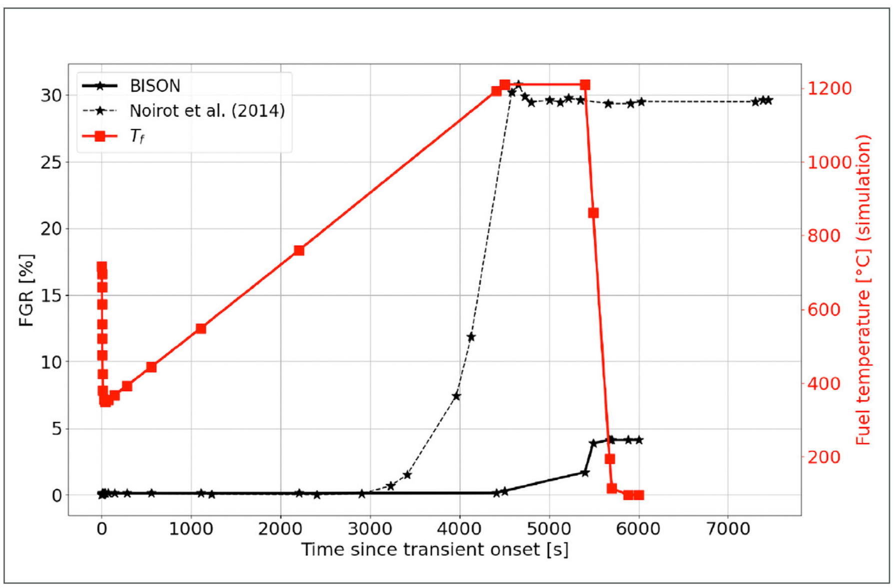


Figure 2. Evolution of FGR with burnup before and during the transient (top) and of FGR during the temperature transient compared to experimental FGR data from Noirot et al. (2014), along with the fuel temperature (bottom)

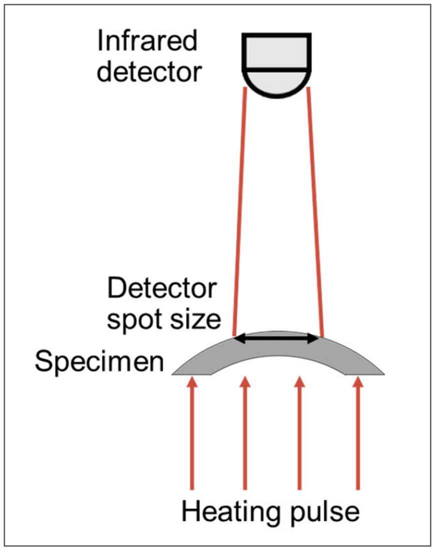
FGR calculations in addition to fully temperature-dependent microcracking-based FGR; the latter component was originally developed based on experimental results from power ramp tests. In this work, the model was exercised and evaluated under several temperature transient scenarios to fully assess its capabilities. BISON accounts well for the effects of temperature conditions and microstructural features on diffusion-controlled FGR. However, the model, which does not yet include a representation of FGR driven by fuel fragmentation during Loss of Coolant Accident (LOCA)-type transients, largely underestimates transient FGR in the considered transient cases when compared with experimental data.

Accomplishments:

There has been a recent push to accelerate fuel qualification by developing capabilities to reduce irradiation periods. This project looked to build on this push by using modeling and simulation to inform experimental development. The demonstration leveraged the Mini-Fuel irradiation capsule designed to irradiate "mini" fuel samples under isothermal temperature conditions. Steady-state irradiations involving MiniFuel effectively decouple the fuel temperature from the fission rate (i.e., power), improving our ability to generate and understand the microstructures observed in commercial reactor fuels. Furthermore, this process offers the possibility to gather in situ data as well as post irradiation or transient data such as thermal conductivity, specific heat, fission gas diffusion and release. However, accelerating fuel qualification is not solely reliant on generating large amounts of data but also on developing an informed test matrix designed to rapidly generate impactful data. This process is reliant on fuel performance codes, such as BISON, to evaluate MiniFuel irradiations using existing material models. This process pinpoints model/data gaps, identifies desired irradiation conditions, and subsequently supports model validation and development. This work describes the use of BISON to perform a number of sensitivity studies designed to understand conditions that lead to FGR according to the model predictions under steady-state isothermal irradiation conditions and temperature transient conditions. The model is applied to a UO_2 MiniFuel disk and shows an overall good qualitative agreement with experimental FGR annealing tests under different temperature conditions. It also accounts well for microstructural effects on FGR. When quantitatively compared with FGR data from previously irradiated 103 MWd/kgU UO_2 disks under thermal annealing, the model shows a less satisfactory agreement with the experimental data. Finally, a UO_2 MiniFuel test matrix is proposed to help to extend the model's operational range and validate the new FGR model capabilities to higher burnups and transient conditions and will provide valuable means to validate BISON's FGR model. Lastly, two major mechanisms behind the experimentally observed FG burst release are currently not accounted for in the model. First, the formation of high burnup structures (HBS) results in a significant drop in the average grain size, allowing for more FG at grain faces because of shorter diffusion times. Second, the overpressurization of HBS bubbles leads to important fragmentation during large and rapid temperature variations. Therefore, further model development in BISON is necessary to better capture transient FGR. Moreover, further irradiation experiments on UO_2 MiniFuel specimen at higher burnups and fuel temperature will provide valuable means to validate BISON's FGR model.

Characterization and Performance of SiC/SiC Tubes

Principal Investigator: Takaaki Koyanagi (ORNL)
Team Members/ Collaborators: Christian M. Petrie, Peter A. Mouche, Xunxiang Hu, José D. Arregui-Mena, and Yutai Katoh (ORNL)



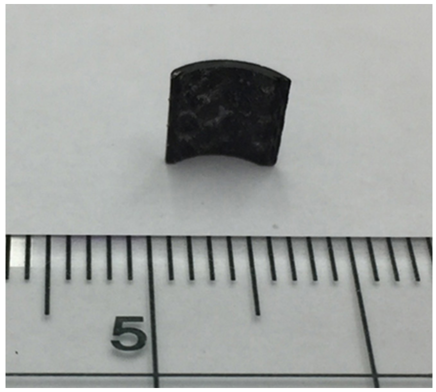


Figure 1. (Left) Schematic illustration of the flash diffusivity test; (right) the SiC/SiC composite tube after specimen preparation for thermal diffusivity measurement

SiC fiber-reinforced SiC matrix (SiC/SiC) composite technologies have exponentially advanced for accident-tolerant fuel cladding applications in light water reactors (LWRs). Recent research and development have improved the tube processing technology and advanced understanding of the thermomechanical behavior of the cladding under normal and off-normal operating conditions. A team at Oak Ridge National Laboratory (ORNL) is addressing the remaining technology challenge—the potential loss of fission gas retention due to cracking under normal operating conditions—by understanding material degradation mechanisms and developing mitigation technologies.

Project Description:

The objective of this research is to understand the effects of neutron irradiation on the hermeticity of prototypic SiC/SiC composite cladding and to develop a mitigation coating on the outer surface of the cladding. Understanding radiation effects helps predict fuel performance with SiC/SiC composite cladding systems for reliable reactor operations. The coating technology could be critical for safely operating and extending the lifetime of SiC-based cladding.

This project conducted post-irradiation examination (PIE) of several prototypic SiC/SiC composite tubes with different fiber architectures. Neutron irradiation in the High Flux Isotope Reactor (HFIR) at ORNL was conducted by using a novel radiation vehicle that generated a temperature gradient across the tube thickness to simulate a radiation environment of LWR fuel rods. Comprehensive microstructure and physical property characterizations were conducted at ORNL's Low Activation Materials Design and Analysis Laboratory. The correlation between the microstructure and hermetic properties provides important feedbacks on how to opti-

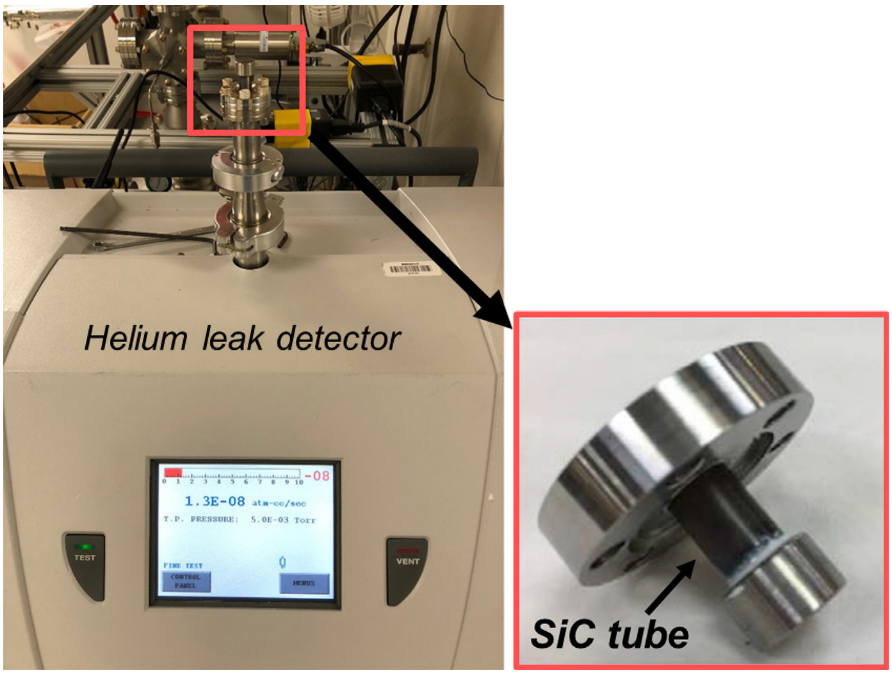


Figure 2. He leak test of the SiC/SiC composite tube

mize the cladding design (e.g., full composite cladding, duplex cladding, coated cladding).

Cr mitigation coating was deposited on SiC/SiC composite tube via the state-of-the-art hybrid physical vapor deposition of a high-power impulse magnetron sputtering (HiPIMS) followed by cathodic arc (CA) deposition. The ORNL team designed the coating process, and industrial partner Acree Technologies Incorporated performed the processing.

Accomplishments:

PIE of the irradiated SiC/SiC tube specimen was completed in fiscal year (FY)21. The results showed that irradiation under a temperature gradient representative of the LWR condition causes radiation-induced micro-cracking and consequently results in a loss of gas tightness of current grades of SiC/SiC composite cladding.

The second-generation Cr coating on the outer surface of SiC/SiC composite cladding was developed to mitigate hermeticity loss via irradiation-induced micro-cracking found by a PIE of the prototypic cladding materials.

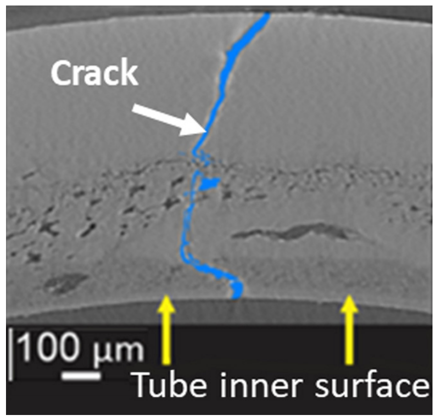


Figure 3. X-ray radiograph of the irradiated SiC/SiC composite tube showing the through-thickness crack in blue

The evaluation methods used included a laser flash thermal diffusivity test, a gas permeability measurement using a leak detector and permeation gas station, and x-ray computed tomography. A new diffusivity test method was developed specifically for the tube geometry. This study investigated the thermal diffusivity of coupons with a curvature machined from the SiC/SiC composite tubes via a modern flash diffusivity apparatus (Figure 1). The detector spot size was reduced to improve measurement accuracy. The thermal property was used to back-calculate the irradiation temperature and its distribution by using a finite element model to ensure the intended irradiation condition. He leak tests were conducted before and after irradiation (Figure 2) and revealed that irradiation with the temperature gradient caused a loss of tube hermeticity. This is because irradiation with the temperature gradient causes cracking damage attributed to strain from the gradient of irradiation-induced swelling. The x-ray computed tomography analysis proved that this occurred, and through-thickness cracking was found locally (Figure 3). In summary, PIE generated the critical experimental data that the prototypic SiC/SiC composite cladding needs to be improved to mitigate hermeticity loss.

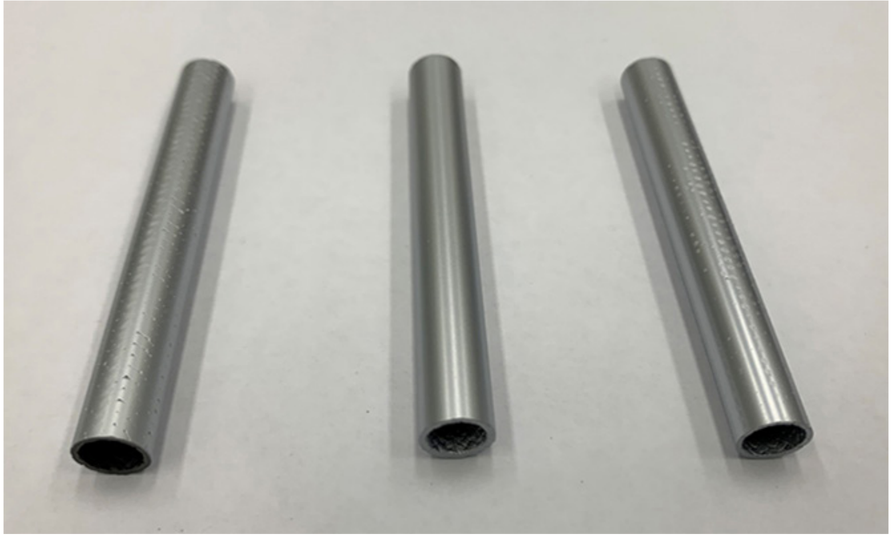


Figure 4. SiC/SiC composite tubes coated with Cr mitigation coating. The tubes are each 75 mm long.

In this research, second-generation Cr coating on the cladding outer surface was discovered to be a potential mitigation strategy for overcoming the cracking issue (Figure 4). To increase the coating adhesion and ductility of the first-generation Cr coating that the ORNL team designed, hybrid physical vapor deposition methods were employed. Using an initial HiPIMS Cr layer before depositing the CA Cr improved the adhesion of the HiPIMS Cr and reduced the cracking/spalling of the CA Cr. This combined structure is desirable for use as a coating for SiC/SiC composite fuel cladding because it maximizes adhesion while allowing for a ductile top layer that can blunt cracks that initiate in the substrate. Currently, PIE of the hybrid Cr coating irradiated in HFIR is being conducted. The experiment will provide critical data that assess the radiation resistance of the mitigation coating.

A Critical Review of High Burnup Fuel Fragmentation, Relocation, and Dispersal under Loss-of-Coolant-Accident Conditions

Principal Investigator: Nathan Capps (Oak Ridge National Laboratory)
Team Members/ Collaborators: Colby Jensen (Idaho National Laboratory), Fabiola Cappia (Idaho National Laboratory), Jason Harp (Oak Ridge National Laboratory), Kurt Terrani (Oak Ridge National Laboratory), Nicolas Woolstenhulme (Idaho National Laboratory), and Daniel Wachs (Idaho National Laboratory)

The content of this manuscript provides a clear path for connecting testing and data acquisition to commercial application by providing concrete means for filling data gaps and prioritized testing regimes.

Experimental data produced by fuel fragmentation testing of high burnup fuel is limited in the literature, and collection of new data limited by both the time required to irradiate commercial fuels to relevant burnup as well as the cost. This data is essential to facilitate ongoing efforts to extend rod average burnup to ~75 GWd/tU in the coming years. The uncertainties and caveats inherent to complex hot cell testing are present in the existing data, but not always identified. Advanced Fuels Campaign (AFC) research staff therefore performed a comprehensive review of all the publicly available data related to high burnup fuel fragmentation, relocation, and dispersal. The report summarizes the loss of coolant accident (LOCA) integral test, separate effects heating test, and detailed microstructure analysis examinations.

Project Description:

The purpose of this review is to perform a critical, holistic assessment of fuel fragmentation, relocation, and dispersal under loss-of-coolant conditions to identify data gaps in the experimental data base. Phenomena have been identified and well defined through the various testing programs, and general agreement regarding the governing parameters (temperature, burnup, heating rate, cladding deformation, etc.) has been established. However, there is a significant data gap connecting research to commercial application. Major identified data gaps consist of (1) a comprehensive understanding of pretransient fuel rod conditions (fuel temperature, fuel stress, rod internal pressures, microstructure, etc. (see Figure 1). The first figure provides a snapshot at the end of a cycle for a core containing high burnup fuel. The figure highlights that there are at minimum two different operating regimes (center of the core and core periphery) that may impact end of life fuel rod conditions. Additional data gaps are (2) definition of prototypic loss-of-coolant accident conditions (heating rate, fuel temperatures, fuel stress), and (3) identification of the differences between nuclear and electrical integral loss-of-coolant accident tests. Figure 2 is an example of fragmented fuel from the Severe Accident Test Station (SATS) facility at ORNL. SATS uses an electrically heated furnace to heat sample up from ~330oC, whereas fuel in a commercial reactor will be at power prior to the transient. Halden data suggest fuel operating at power prior to the transient may experience less fragmentation than observed in an electrically heated furnace. Finally, connecting fuel rod performance (cladding ballooning, burst opening, potential relocation, fragmentation suscep-

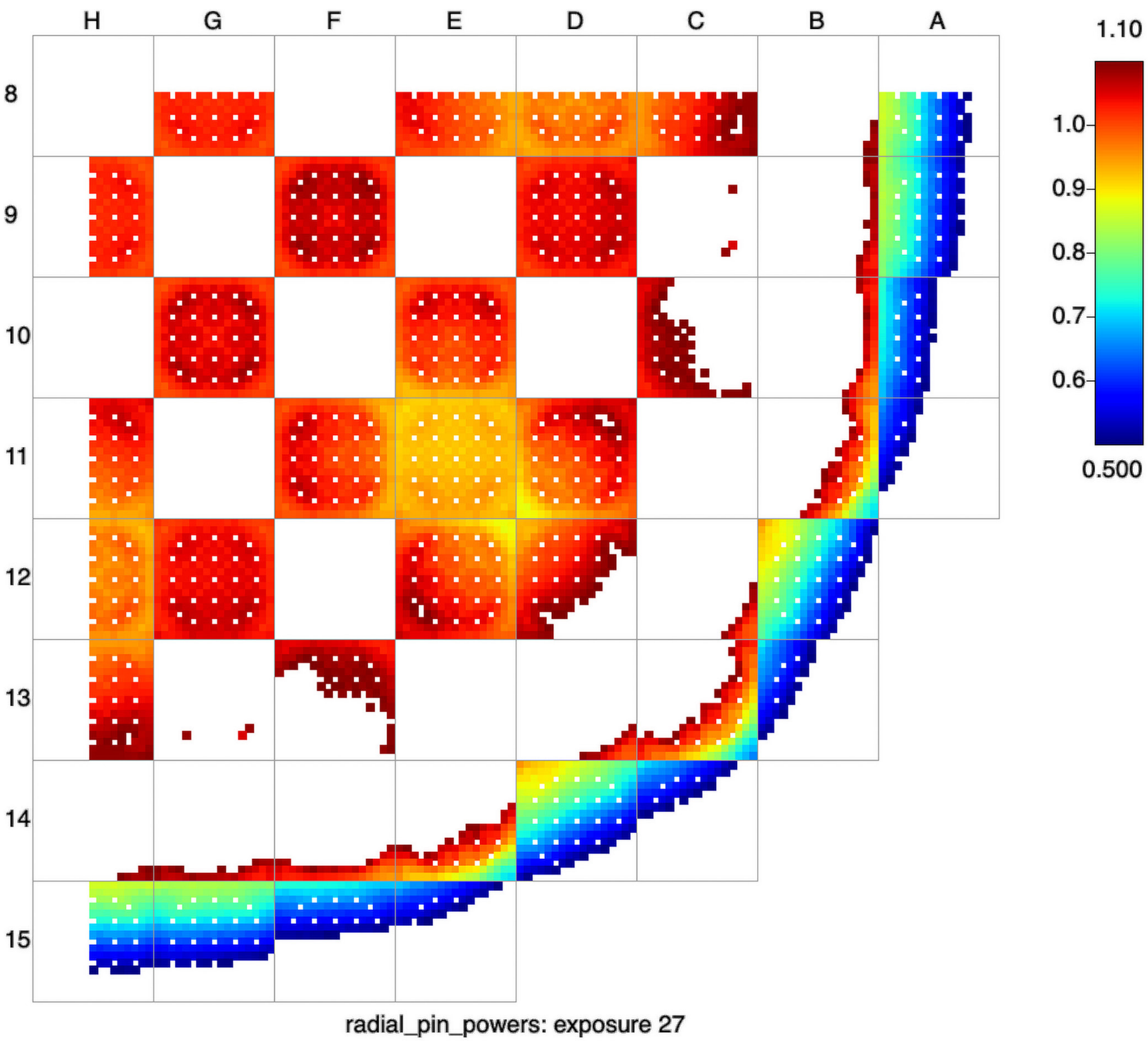


Figure 1. EOC rod average peaking for every high burnup (>62 GWd/MTU) fuel rod in the core (core average LHR = 18.8 kW/m)

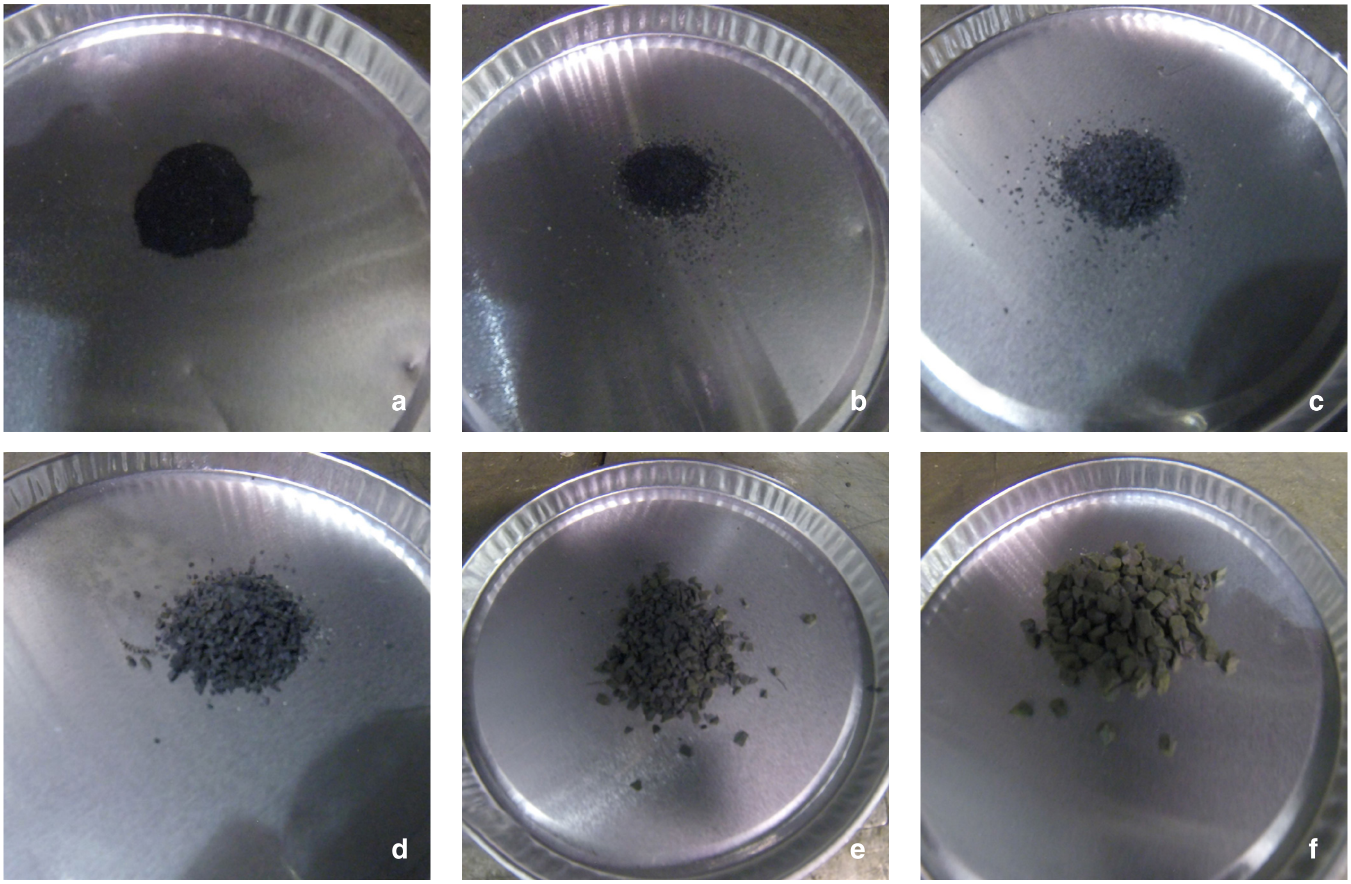


Figure 2. HBFF images [39] of NA#2 high burnup LOCA test performed at the Sever Accident Test Station: (a–d) fine fragments < 1 mm, (e) medium fragments ranging from 1–2 mm, and (f) large fragments >2 mm

tibility, etc.) to fuel fragmentation, relocation, and dispersal severity. The content of this manuscript provides a clear path for connecting testing and data acquisition to commercial application by providing concrete means for filling data gaps and prioritized testing regimes. The role of the AFC High Burnup (HBu) program is to use this manuscript identify, prioritize, and filling data gaps that will enable extending burnup beyond 62 GWd/tU.

Accomplishments:

This manuscript is supporting U.S. nuclear industry desire to extend the rod average burnup to ~75 GWd/tU by 2026. Given the aggressive schedule of this goal and limited available resources, the manuscript supports subsequent efforts indented to prioritize, integrate, and coordinate parallel efforts being conducted by all stakeholders to the extent possible in order to fill technical knowledge gaps. This complex integration effort is being led by the Collaborative Research on Advanced Fuel Technologies (CRAFT) for Light Water Reactors (LWRs) program including representatives from major interests in industry, United States Nuclear Regulatory Commission, and from AFC program representation. The purpose of CRAFT is to disseminate the information provided by the technical community, assess their relative progress, and aid in research and development (R&D) scope prioritization. This manuscript has been critical to the high burnup program within the campaign and used to develop and document the experimental and analytical activities required to generate critical path data and information to aid in regulatory review and inform Topical Reports related to the Burnup Extension mission. Additionally, the review has provided the Nuclear Regulatory Commission (NRC) with critical information to support the development of their internal Research Information Letter as well as informed the SCIP IV program directions.

2.2 ATF INDUSTRY ADVISORY COMMITTEE

Accident Tolerant Fuel Industry Advisory Committee

Committee Chair: Bill Gassmann, Exelon
Collaborators: Daniel Wachs, Ed Mai and Phyllis King

The Advanced LWR Fuel Advisory Committee was established in 2012 to advise AFC's National Technical Director on the direction, development, and execution of the campaign's activities related to accident tolerant fuels for commercial light water reactors. Last year the committee charter was revised, and this year the committee was asked to provide an industry perspective on LWR fuels issues that are broader in scope than just accident tolerant fuels. The IAC is comprised of recognized leaders from diverse sectors of 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, EPRI, and NEI. Members are invited to participate on the committee based on their technical knowledge of nuclear plant and fuel performance issues as well as their decision-making authority in their respective institutions. During the past year the committee provided important industry input relative to utility and fuel vendor perspectives on the potential benefits of extending the burnup of current fuels; continued efforts in testing and evaluation of new accident tolerant fuels, especially relative to the lead test assemblies operating in numerous commercial plants that include ATF rods; and testing infrastructure needs and gaps created by the loss of the Halden Reactor in Norway.

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

BWXT
BWX Technologies, Inc.

framatome

NEI
NUCLEAR ENERGY INSTITUTE

Dominion Energy

TerraPower

Southern Company

DUKE ENERGY

EPRI
ELECTRIC POWER RESEARCH INSTITUTE

Westinghouse

GENERAL ATOMICS

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

Exelon

ATF Industry Teams - Framatome

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

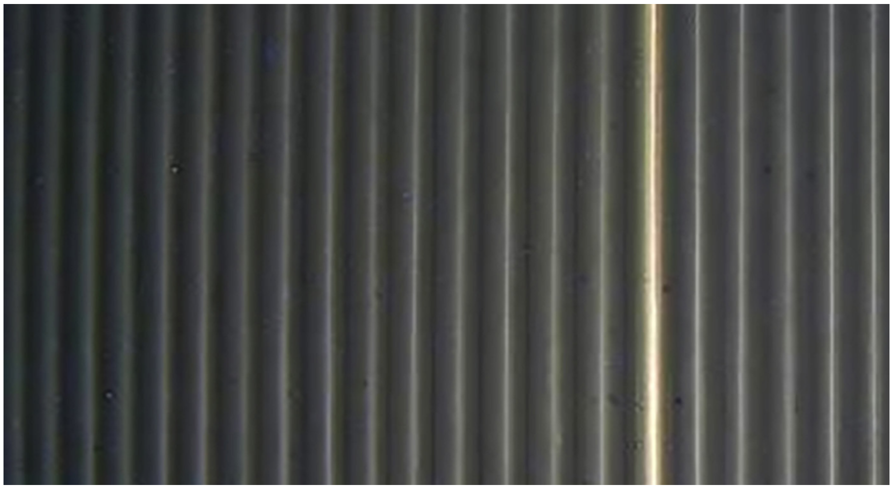


Figure 1. Cr-coated fuel rods after 1 cycle in Vogtle

Framatome's ATF solutions are designed to deliver value to all current US reactor designs in both the BWR and PWR operating conditions. Additionally, extensive irradiation testing is ongoing in European reactor designs as well.

Framatome made significant progress in development, testing, and licensing of Framatome's accident tolerant fuel (ATF) solutions. It relies on a two-phased approach to balance benefits with the anticipated timeline for full-core deployment. The first phase consists of near-term evolutionary solutions which are focused on improving safety and fuel cycle economics while being compliant with insertion of batches in mid-2020s. The second phase consists of longer-term solutions aimed to offer further performance improvements during beyond design basis accidents.

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

Project Description:

The goal of DOE's EATF program is to develop an economical and more robust nuclear fuel design that will reduce or mitigate the consequences of reactor accidents while maintaining or improving existing performance and reliability levels in daily operations. After extensive testing, evaluation, and down selection, Framatome's technical approach addresses three focus areas: (i) Coatings for pressurized water reactor (PWR) and boiling water reactor (BWR) claddings, (ii) Chromia-doped and Chrome-variant UO_2 fuel pellets, and (iii) Silicon carbide (SiC) composite materials.

A dense Cr-coating on a zirconium-based cladding substrate has the potential for improved high temperature steam oxidation resistance and high temperature creep performance, as well as improved wear behavior. Over the course of the EATF program, extensive processing and testing activities are being carried out on Cr-coated $M5_{Framatome}$ cladding in support of batch implementation by the mid-2020s.

Building on both the experience gained and scientific knowledge achieved from the PWR Cr-coated cladding development, a coating material that is suitable for BWR application has been developed. Several out-of-piles tests were performed to ensure adherence and coating quality. Currently, the lead test rods with BWR coating segments are being irradiated in the Monticello reactor.

Chromia-doped UO_2 pellets can improve pellet wash-out behavior after cladding breach and reduce fission gas release. The performance of this fuel has been extensively studied in out-of-pile and in-pile test programs. Chromia-doped fuel topical report for PWR application has been submitted to the Nuclear Regulatory Commission (NRC) in June 2021.

Framatome's EATF pellet development is also focused on improving thermal-mechanical properties, especially thermal conductivity. Variants of Cr-doped UO_2 fuel pellets have been developed and out-of-pile testing showed a significant increase in thermal conductivity compared to UO_2. As part of the proof-of-concept, investigation has been focused on understanding evolution of thermal conductivity and microstructure under irradiation.

For revolutionary performance improvements, Framatome is developing a composite cladding comprised of SiC fiber in an SiC matrix (SiCf/SiC) as well as studying applications for other critical components such as BWR channel boxes and structural tubes. The objective is to develop a fuel system which does not suffer from the same rapid oxidation kinetics of zirconium-based cladding while having attractive operating features.

Accomplishments:

Onsite visual inspections were performed on Cr-coated Lead Test Rods (LTRs) after first cycle irradiation in Vogtle Unit-2. The Cr coating was very adherent to the underlying $M5_{Framatome}$ cladding with no signs of coating delamination. Cr-coated rods showed lustrous-gold appearance suggesting that the oxide formed on the surface of the cladding was very thin (Figure 1). These results confirm the excellent performance of the Cr-coated cladding which was demonstrated in the out-of-pile tests and other in-pile programs. Similar performance was observed for both inert and UO_2 fuel loaded Cr-coated rods, which completed one cycle at Arkansas Nuclear One (ANO) in April 2021 and in GOCHROM

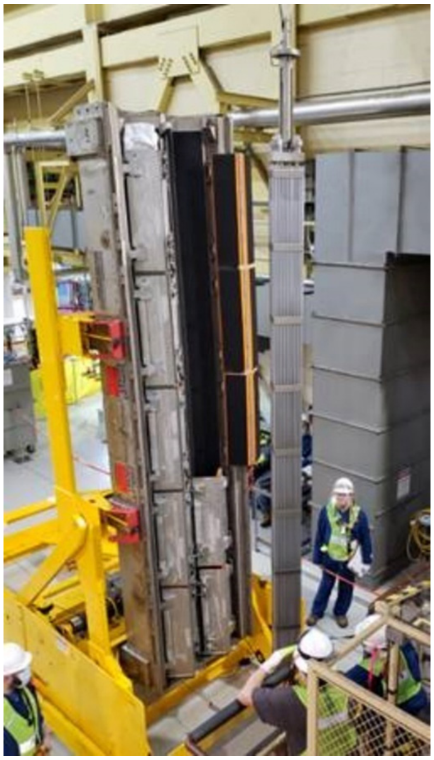


Figure 2. Cr-Cr LTA at Calvert Cliffs

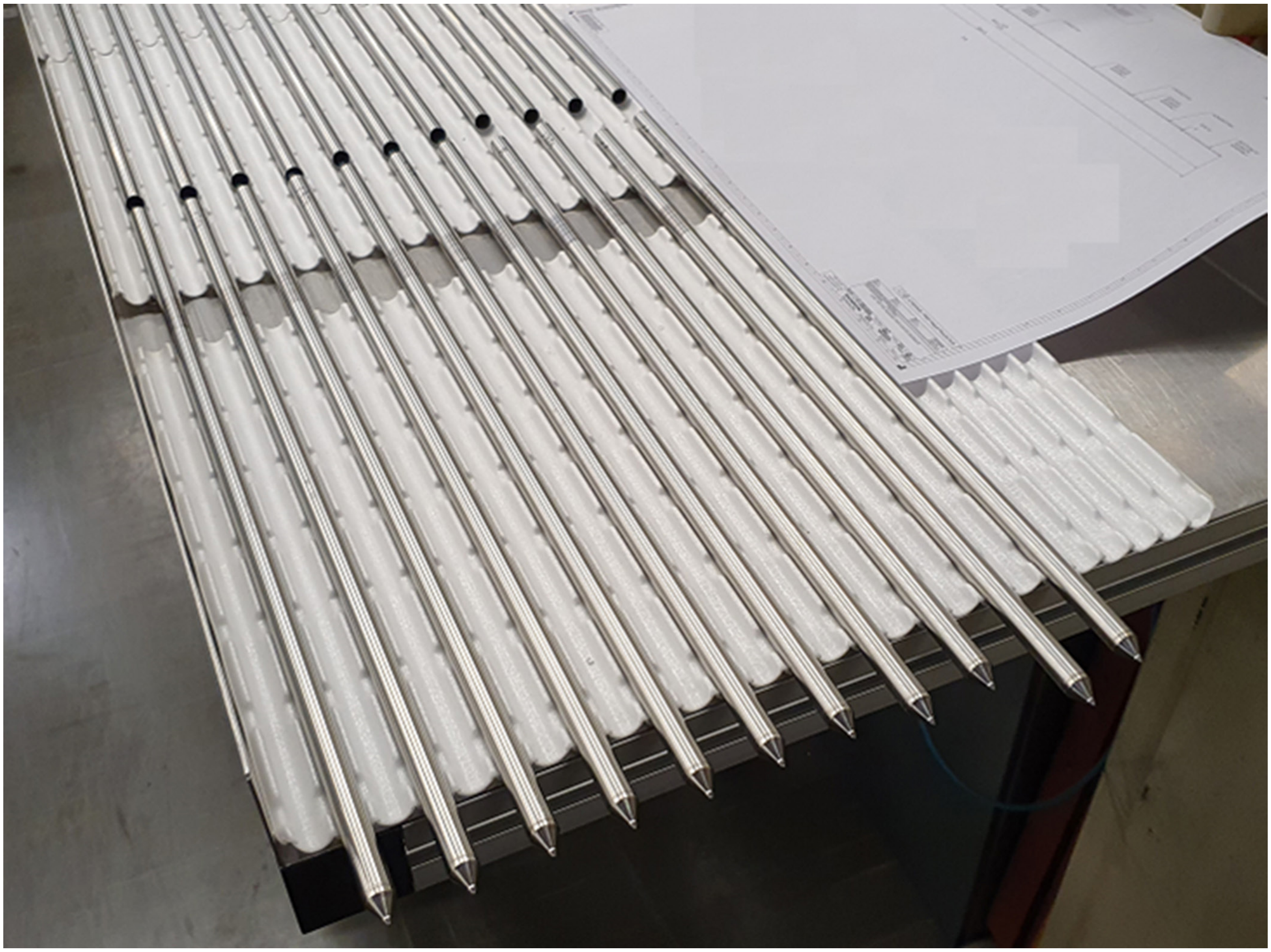


Figure 3. BWR coated segments for Monticello LTR campaign

Cr-coated fuel rods after one and two cycles of irradiation in Gösgen reactor.

One full EATF Lead Test Assembly (LTA) was fabricated and shipped to the Calvert Cliffs in January 2021 (Figure 2) The LTA consists of 176 Cr-coated clad rods with Cr_2O_3-doped UO_2 and 12 UO_2-Gd_2O_3 rods. Irradiation of full EATF assembly began in March 2021 with post-irradiation examination (PIE) scheduled after completion of the 1st cycle in 2023.

Framatome fabricated and installed two new prototype machines. These new prototypes use the same physical vapor deposition (PVD) process as the previous full-length coating machine. The operating conditions of two new prototypes are currently being validated by Framatome. In addition, pre-design phase for a pilot machine has been launched to better understand the key aspects of the pilot design.

Figure 4. SiC based rodlets for MITR irradiation

Framatome identified a protective coating for BWR fuel claddings. The coating showed a significant reduction in corrosion and oxidation behavior in autoclave and high temperature steam environment compared to uncoated cladding. Segmented rods with BWR coating were fabricated in Framatome's prototyping lab and shipped to Monticello for testing in a commercial reactor (Figure 3). Monticello LTR irradiation began in May 2021 and onsite PIE are planned after each cycle. Further out-of-pile testing of the potential BWR coating solution is being performed to generate comprehensive data for comparison with hot cell measurements.

A program was developed aiming to test Framatome's SiCf/SiC rodlets under PWR representative conditions. Framatome's SiC-based unfueled rodlets were prepared for irradiation in the Massachusetts Institute of Technology Research Reactor (MITR) in late 2021 (Figure 4). Framatome completed fabrication of SiC rodlets and they will be shipped to MITR in fall 2021. In parallel, Framatome's SiC-based unfueled rodlets are being manufactured and prepared for irradiation in INL's Advanced Test Reactor (ATR) after Core Internal Change-out (CIC) Shutdown.

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

Principal Investigator: Raul B. Rebak, GE Research, Schenectady, NY 12309, USA
Team Members/ Collaborators: Russ Fawcett, Global Nuclear Fuels; Evan Dolley & Andrew K. Hoffman, GE Research; Andy Nelson & Jason Harp ORNL, David Kamerman, INL, and Kenneth McClellan, LANL

GE is the only fuel vendor currently developing a monolithic thin walled tube of FeCrAl as cladding for the fuel.

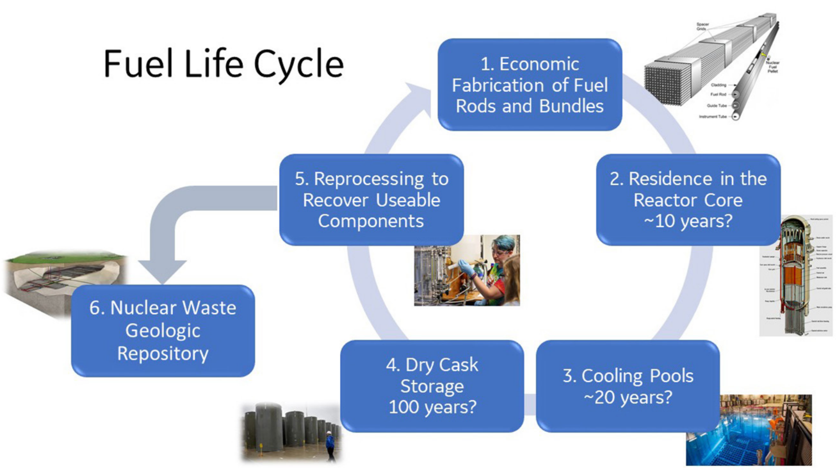


Figure 1. ATF fuel cycle involves the economical fabrication of the fuel, residence in the reactor (+unlikely accident scenarios), cooling pool storage, dry cask storage, reprocessing, and final geologic repository disposal

General Electric (GE), including GE Research, GE Hitachi Nuclear Energy, and Global Nuclear Fuel (GNF), has a contract with the US Department of Energy Office of Nuclear Energy to develop Accident Tolerant Fuels (ATF) for the current fleet of Light Water Reactors (LWR). GE is working closely with the reactor-owners' utilities Southern Nuclear and Exelon Generation plus Oak Ridge National Laboratory (ORNL), Idaho National Laboratory (INL), and Los Alamos National Laboratory (LANL) in this fuel development. Activities include basic research and testing to characterize and evaluate advanced concepts (never used before in reactor environments) for fuel rod fabrication, as well as their direct assessment in operating commercial nuclear power stations. Areas of development include (1) ARMOR and other coatings for Zircaloy, (2) IronClad cladding, (3) Higher 235U enriched fuel, (4) Extended fuel burn up, and (5) Modified fuel forms. This report covers accomplishments under contracts DE-NE0008823 and DE-NE0009047.

Project Description:

The objective of the GE led project is to develop a family of fuels that will make the current fleet of LWRs safer to operate. The average age of the LWRs in the USA and in the western world is increasing because no new reactors are being connected to the grid. There is a consensus that the operation life of the current reactors fleet needs to be increased. The introduction of the ATF fuels into the operating reactors will facilitate this life extension, increasing safety of operation, and reducing operation costs. That is, the newer family of ATF fuels will add benefits such as (a) Fuel cycle economics (i.e., increased burnup), (b) Increased fuel reliability, and (c) provide plant operational flexibility. GE is working in ATF fuel concepts that are for near term implementation and for longer term development (advanced concepts). The fuel developments include cladding components, fuel components, and channels for boiling water reactor (BWR) applications. All of the new

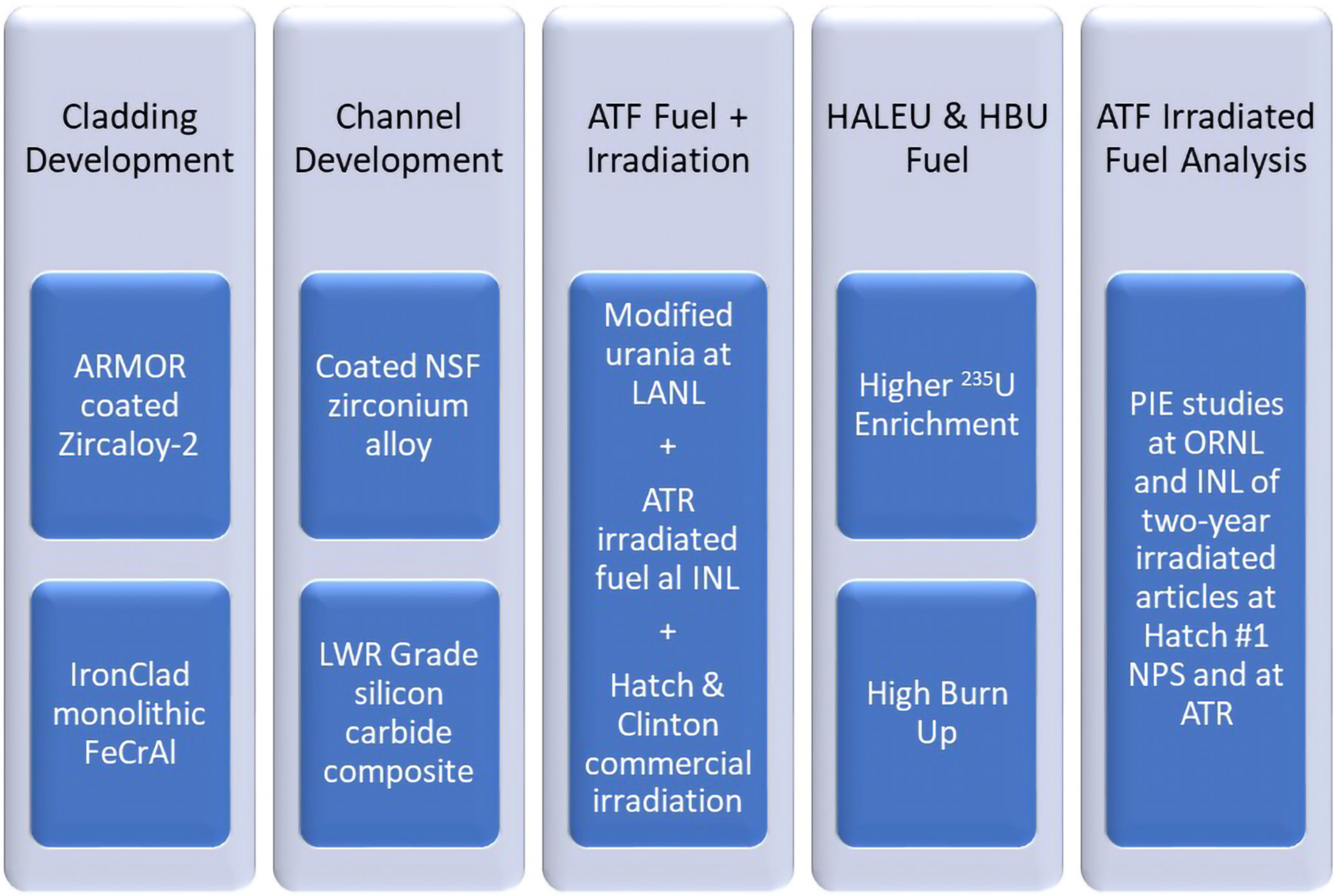


Figure 2. ATF concepts being investigated by GE

ATF concepts need to be characterized for performance in the entire fuel cycle (Figure 1). For claddings, GE is developing the ARMOR family of coatings for Zircaloy-2 tubing which will provide outstanding resistance to fretting under normal operation conditions and increased resistance to oxidation in Design Basis Accident (DBA) and Beyond Design Basis 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 (Figure 2). 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. Since the channel does not require hermeticity but requires compatibility with hot water and resistance to attack by steam, the development of silicon carbide (SiC) composites for channels is a logical first step before SiC composites can be implemented for fuel cladding. On the fuel side GE is exploring the modification of the current urania fuel to make it more resistant to

Figure 3. The GE 2000 cask containing GNF articles removed from Hatch NPS #1 delivered to ORNL hot cells

fragmentation in the case of an accident or for extended burn up conditions. GE is also concurrently working on higher 235U fuel enrichment and higher fuel burn-up concepts (Figure 2).

Accomplishments:

In spite of the Covid pandemic disruption, significant accomplishments can be claimed for the Fiscal year 2021.

One of the major accomplishments was the post irradiation examination (PIE) in the hot cells at ORNL of the first version of ARMOR articles removed from the Hatch NPS number 1 (Figure 3). These ARMOR articles were exposed to typical BWR environments for one cycle of 24 months from February 2018 to February 2020. Similarly, in June 2021, irradiated ARMOR and IronClad articles were removed from the Advanced Test Reactor(ATR) after receiving a dose of approximately 20 GWd/MTU. Poolside optical inspection did not show any type of obvious degradation after exposure in a simulated pressurized water reactor (PWR) environment (Figure 4). The articles will be moved to the INL hot cells in September 2021 for PIE studies.

GE is continuing with the characterization of two main compositions of IronClad (C26M and Advanced Powder Metallurgy Tubing (APMT) which have bounding compositions of chromium (12% and 21%, respectively). Research studies include the evaluation of (1) resistance of these alloys to 300-800°C creep, (2) resistance to fretting corrosion, (3) effect of composition on the thermal formation of alpha prime, (4) dissolution behavior in hydrogenated and oxygenated environments, (5) effect of presence of zinc on their passivation behavior, and (6) reprocessing capabilities of FeCrAl compared to Zircaloy cladding. All the results on IronClad are highly promising compared with the current cladding of Zircaloy-2. For example, Ironclad materials have higher resistance to fretting than Zircaloy since under the same tested conditions in 288°C water containing 2 ppm oxygen, the wear

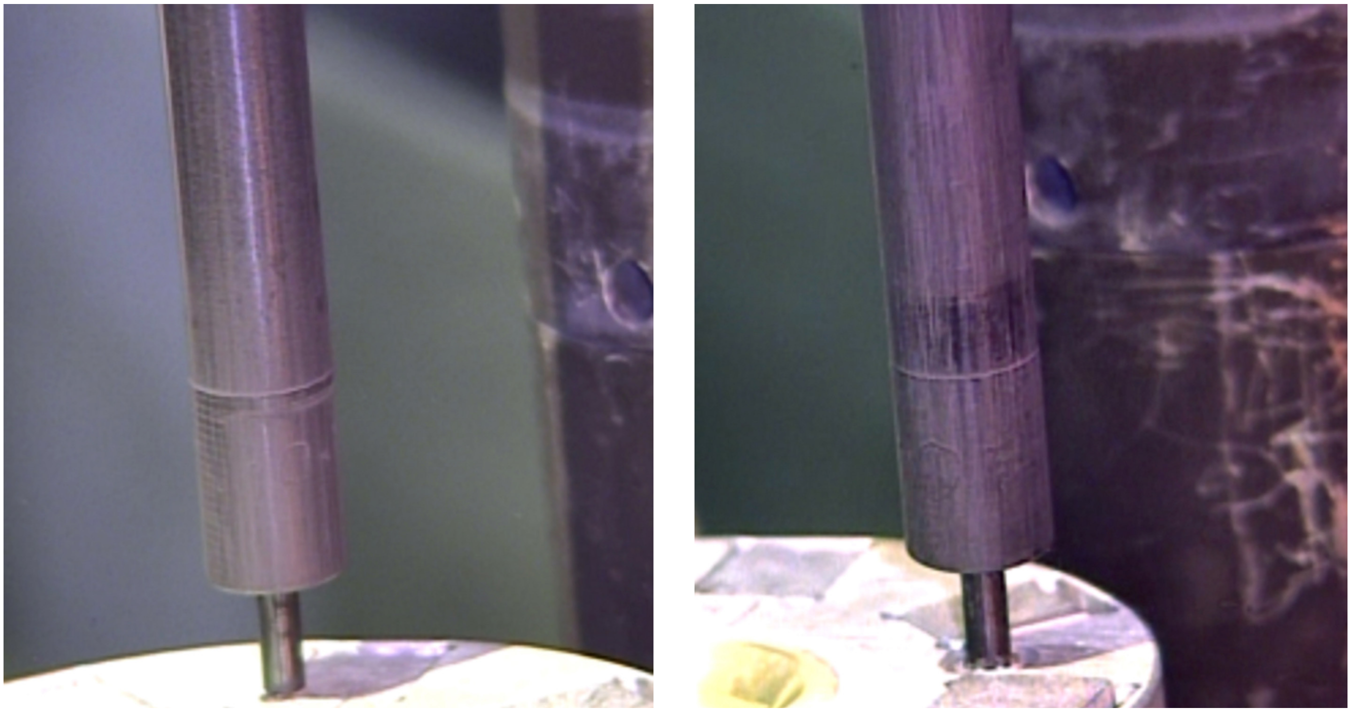


Figure 4. GE rodlets removed from ATR after ~ 20 GWd/MTU irradiation

depth of Zircaloy2 tubing was more than 600 μm while for C26M the wear was ~200 μm deep and for APMT was ~130 μm deep. Thermal aging studies of APMT tubing showed that even after aging for 1000h at 450°C (when alpha prime formation was identified), the tube had a 315°C elongation ductility of ~10% compared to a ~15% ductility for the non-aged tube. PIE studies of the rodlets removed from the ATR will demonstrate if neutron irradiation contributes to any loss of ductility of APMT. Immersion tests for one year at 288°C showed that the presence of 5 ppb of dissolved Zn^{2+} considerably decreased the dissolution rate of IronClad tube coupons, especially for the lower Cr C26M. Immersion and electrochemical corrosion tests conducted in sulfuric, hydrochloric, and nitric acid at 30-90°C showed that the dissolution rates of the IronClad tubing was several orders of magnitude faster than the dissolution rate of the current Zircaloy-2 tubes, which will facilitate the reprocessing development of IronClad. GNF's manufacturing plant is moving forward to ready the facility for LEU+ operations based on Criticality Safety Analysis (CSA) and Integrated Safety Analysis (ISA) work. GNF has been working on a risk informed licensing approach for High Burnup (HBU) application, which has been already discussed with the Nuclear Regulatory Commission. GNF is also working with the Nuclear Energy Institute regarding federal funding and the implementation of ATF concepts in the current light water reactor fleet.

Accident Tolerant Fuel (ATF) and High Burnup - Higher Enrichment (HBHE) Fuel Industry Teams – Principal Investigator: Westinghouse Electric Company LLC

Principal Investigator: E. J. Lahoda
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), University of Virginia, University of South Carolina, Oak Ridge National Laboratory (ORNL), Rensselaer Polytechnic Institute (RPI), University of Tennessee (UT), University of Texas at San Antonio, Air Liquide (AL), and Royal Institute of Technology (Sweden) (KTH)

Westinghouse EnCore® Fuel is "game-changing" for the nuclear industry, significantly increasing safety margins in severe accident scenarios, increasing flexibility for fuel management and, coupled with HBHE for longer fuel cycles which can lower operating costs, EnCore Fuel becomes the clear utility choice for increased safety and cost efficiency.

Westinghouse is working to commercialize unique accident tolerant EnCore®* fuel designs with the capability of using higher ^{235}U enriched ADOPTTM* fuel (Cr_2O_3+Al_2O_3 doped UO_2) or $U^{15}N$ fuel to achieve burnups of around 75 MWd/kgU; Cr coated Zr-Alloy or SiGA®* silicon carbide (SiC) cladding with higher ^{235}U enriched fuel or $U^{15}N$ fuel uranium nitride (UN) fuel.

Project Description:

Lead test rods (LTRs) of Cr coated cladding were retrieved and sent to ORNL in October 2020 (Figure 1), and Cr coated Zr claddings with ADOPT and UO_2 pellets from the Advanced Test Reactor (ATR) in March 2021 (Figure 2). Plans to load SiC clad tubes with Mo pellets into the ATR in 2022 and to perform leaker rod tests in the Westinghouse autoclaves in Churchill, PA are progressing. Cr coated Zr cladding and SiC cladding will be loaded into the BR-2 reactor as part of the European Union funded Il Travatore program at the end of 2021.

The immediate tasks are focused on supporting design and licensing activities to develop the required experimental backup to obtain U.S. Nuclear Regulatory Commission (NRC) approval for insertion of region quantities of Cr coated Zr with ADOPT pellets and with >5% enriched ^{235}U pellets to increase achievable burnups to 75 MWd/kgU, allowing the potential of more economical 24-month cycles in pressurized water reactors. Additional tasks include:

- Develop oxidation resistant UN and production technologies;
- Develop low-cost methods for manufacturing SiC and Cr coated rods;
- High temperature out-of-reactor bundle testing of Cr coated cladding to validate the MAAP and MELCOR codes to determine potential operational savings and to support licensing changes will begin in the 4th quarter of 2021 and in 2022 at the Karlsruhe Institute of Technology (KIT).

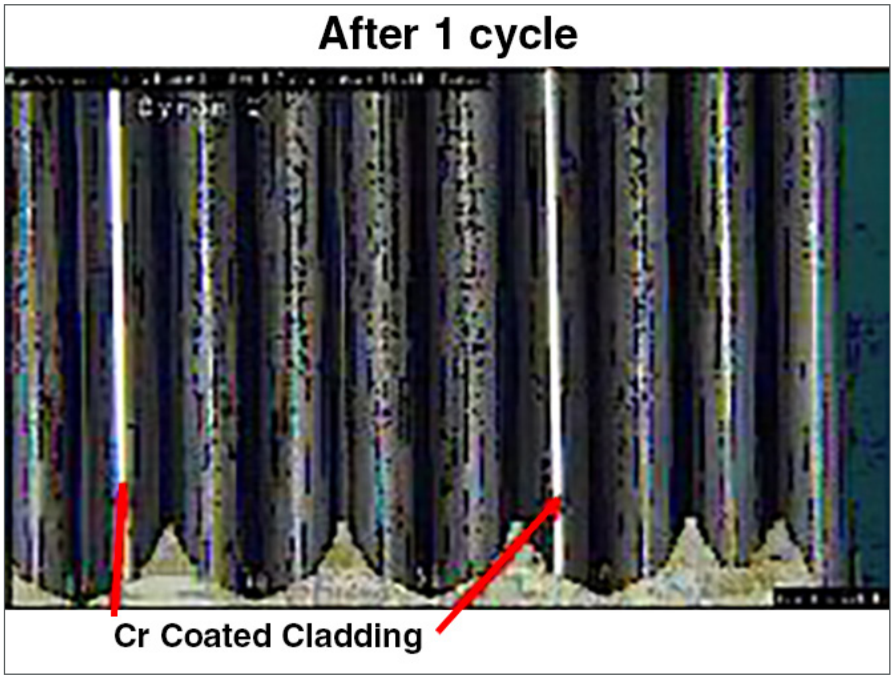


Figure 1. Cr coated cladding after first cycle

- Mechanical, autoclave and in-reactor testing are continuing for the Westinghouse Cr coated cladding product to support an upcoming topical report.
- Development of integral fuel rod temperature, pressure and fuel swelling sensors that can be remotely monitored in conjunction with atomic scale modeling to reduce the time for licensing.

Accomplishments:

A topical report on extending burnup up to 68 MWd/kgU was submitted to the NRC in December 2020. Review meetings were held with the NRC and requests for additional information on the previously submitted ADOPT fuel topical were supplied. Irradiated ATF lead test rods (Cr coated rods and ADOPT pellets) were shipped to ORNL to begin post irradiation examination activities and mechanical and performance testing (Westinghouse's

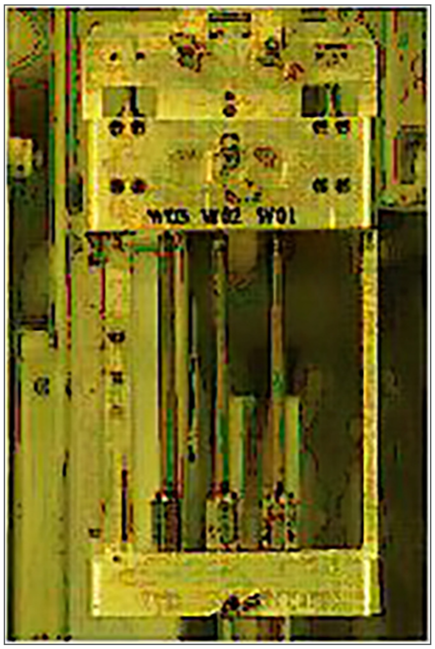


Figure 2. Rods irradiated under PWR conditions to ~9 MWd/kgU in the ATR

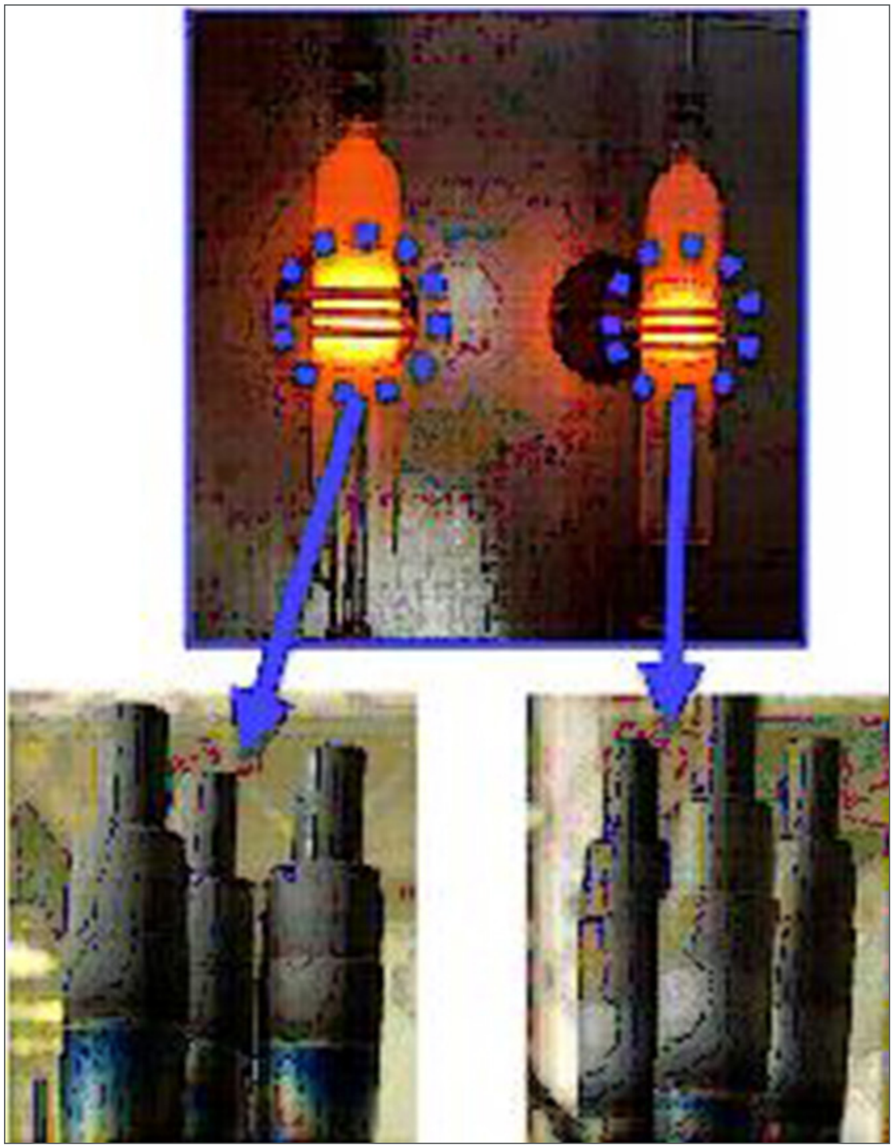


Figure 3. SiGA cladding end plug yield increased to 95%

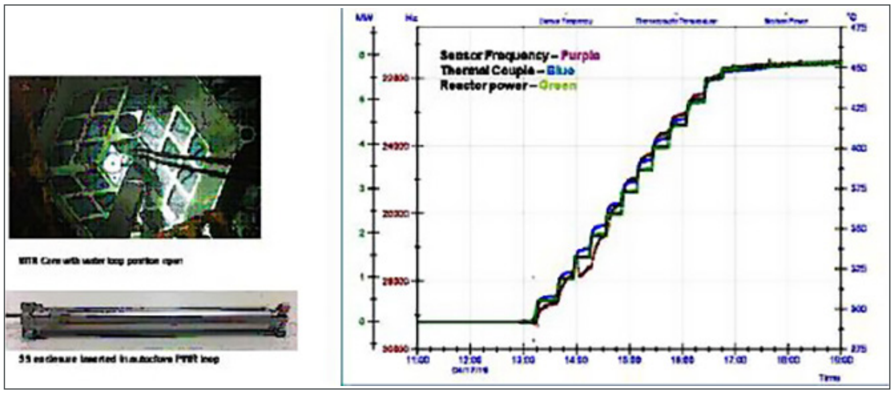


Figure 4. Sensor frequency tracks internal thermocouple and reactor power

EnCore Fuel Takes the Next Step towards Regulatory Approvals (westinghousenuclear.com). Plans are on-going to ship more rods to INL following agreement with the state of Idaho in 2022 for Transient Reactor Test Facility (TREAT) and ramp testing.

Synthesis of the dense UN pellets with controlled microstructure by spark plasma sintering (SPS), and the establishment of the physical density, pellet consolidation – materials processing conditions including sintering temperature, duration and pressure was completed at Rennselaer Polytechnic Institute and the effects of pellet density on oxidation on-set temperature and rate were evaluated at the University of Texas at San Antonio. Mini-fuel UN tests in the BR-2 reactor in Belgium are being planned for late 2021 or early 2022. Los Alamos National Laboratory is generating physical property data for UN and ADOPT pellets to support licensing as well as UN powder and pellets for testing.

AL completed evaluation of methods for ^{15}N enrichment and concluded that it appears feasible to provide ^{15}N at economically acceptable costs. GA is continuing development of their SiGA clad fuel. End plug joining (Figure 3) and the scaleup to 14-foot rods are being addressed. GA is also pursuing rodlet insertions at the ATR with Mo pellets with liquid metal bonding after the Core Internals Change-out (CIC) is completed at ATR in early 2022.

Integral fuel rod temperature sensors were successfully tested at Massachusetts Institute of Technology Reactor (MITR) and found to accurately track the in-rod temperatures at reactor conditions (Figure 4).

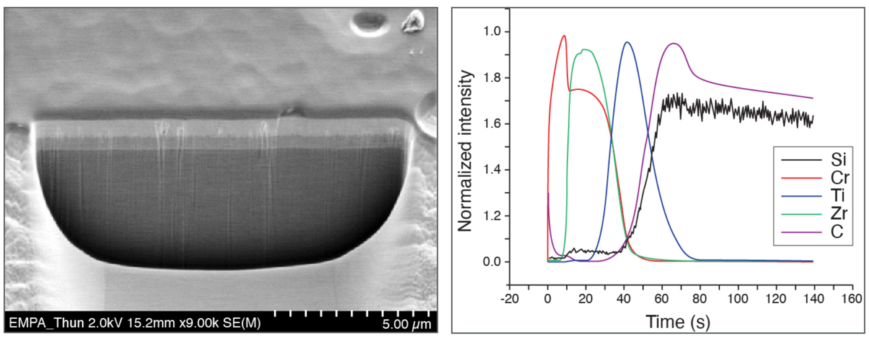
University-Led Investigations of Behavior of SiC Composite Cladding as Accident Tolerant Fuels (ATF) for Light Water Reactors (LWRs)

Principal Investigators: Peng Xu & Yutai Kato
Team Members/ Collaborators: Peter Hosemann; The Regents of the University of California, Berkeley; Christian Deck, General Atomics; Julie Tucker, Oregon State University; Djamel Kaoumi, North Carolina State University; Yongqiang Wang, Los Alamos National Laboratory

Seven university-led projects including 9 universities, 3 national laboratory, and 2 industry collaborators address key technical challenges and overcome research and development huddles for the SiC technology for ATF applications.

SiC composites are considered as a promising ATF cladding material for LWRs. The excellent neutronic properties, high strength and high temperature stability and oxidation resistance make it a prime candidate for ATF. However, because this is revolutionary change to the existing zirconium alloy-based cladding, the behavior of SiC cladding under normal operation conditions and accident conditions need to be understand, and several technical challenges should be addressed, such as the corrosion resistance and retention of hermeticity. In 2018, the Nuclear Energy University Program (NEUP) awarded 7 proposals to study the key technical challenges SiC cladding is facing at pressurized water reactor (PWR) conditions as a prime ATF candidate. The projects were well coordinated, and the research outcomes were augmented by a close participation and engagement from national laboratories and nuclear industry. All projects will finish at the end of this year, after providing key technical data and insights that support SiC development moving forward.

To mitigate the corrosion of SiC in PWR water, the team acquired SiC samples with a Physical Vapor Deposition (PVD) coating that consists of Zr, Ti, and Cr from the industry partner General Atomics, and from the Swiss Federal Laboratory (Electron Probe Micro Analysis (EPMA), Thun CH). Initial characterization of the PVD coatings have been made, including Transmission Electron Microscopy (TEM), diffraction, Chemi-STEM, nanoindentation, and corrosion tests. Figure 1 shows the coating cross section for an EPMA sample. In addition to the PVD coating, the team also partnered with Ionbond and produced four Chemical Vapor Deposition (CVD) coatings with the selected elements of TiC/TiCN, TiC/TiCN (optimized parameters) ZrN, and TiN. All CVD coatings show visible defects at the surface such as cracks, pits, or delamination. The microstructure and composition of the Ionbond CVD coatings have been further investigated using Scanning Electron Microscopy (SEM), Energy Dispersive Spectroscopy (EDS), TEM, SIMS, and ChemiSTEM giving



	Ti	Zr	Cr	Total
nm	418	522	367	1307
nm/min	4.6	5.7	4.0	

Figure 1. (Top left) FIB cross section of the uniformly deposited Ti-Cr-Zr coating. (Top right) GDOES intensity plot for showing defined peaks with limited intermixing. (Bottom) PVD depostion rates for each element.

a comprehensive view over what has been produced. These analyses primarily found that oxygen was within the coating in non-negligible amounts, heat treatment caused a detrimental effect to adhesion by increasing oxygen and depleting base element concentrations. The CVD coating was further characterized using nanoidentation, giving a view into how these coating will do in high temperature environments and giving baseline values for the hardness and elastic response of these coating systems. Comprehensive corrosion testing has been performed on the CVD coatings. None of the coatings were protective enough in order to form a passivation layer which shielded it from the simulated boiling water reactor (BWR) conditions. Further optimization of the CVD coating and the final selection of the PVD or CVD coating are underway.

Dynamic System Scaling (DSS) for Enhanced Interpretation of Irradiation Tests

Principal Investigator: Alexander Duenas
Team Members/ Collaborators: Dan Wachs, Guillaume Mignot, Jose Reyes, Qiao Wu, and Wade Marcum

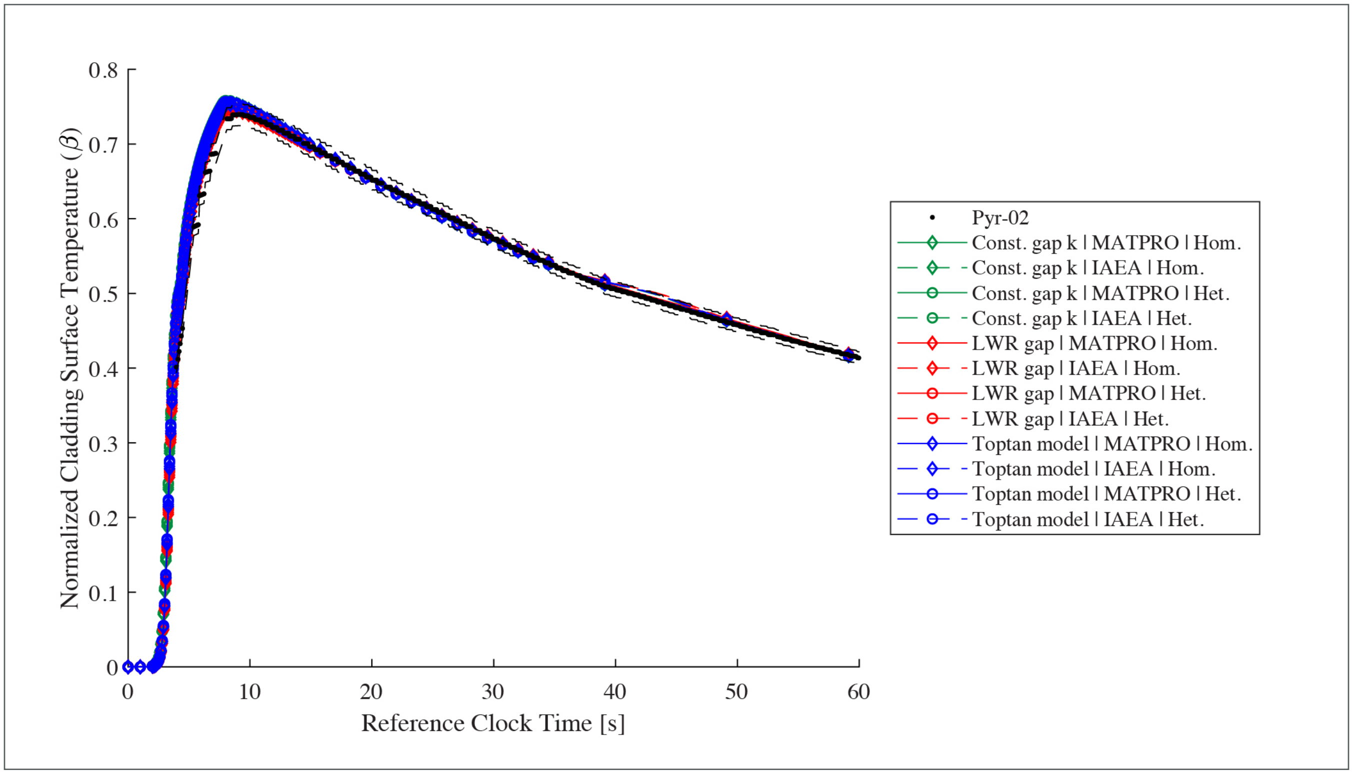
DSS can provide objective metrics to identify and quantify distortions between scaled SET and IET data and full-scale LWR conditions.

Efforts for accelerating fuel qualification are leveraging new irradiation experiments with reduced geometric length scales and reactor conditions to generate test data on shorter timelines and provide data necessary for fuel performance code validation studies. Irradiated fuel performance test data produced from test reactors often are at reduced or altered geometric length scales and are not typically capable of replicating exact initial, boundary, or test conditions in light water reactors (LWR). Distortions introduced by these varying power, geometrics and temporal scaling factors need to be identified and quantified to determine how representative test data is of the full-scale nominal conditions in a commercial reactor. The Dynamical System Scaling (DSS) methodology can quantify scaling distortions over transients and determine similitude between scaled fuel rodlet performance and full-length fuel rod performance.

Project Description:

DSS is being used to develop a framework to quantify transient scaling distortions in reduced-scale fuel rodlets under transient testing. This scaling theory's utility is being able to identify regions of local scaling distortion and regions of similitude throughout a transient. DSS can be used to support fuel performance code validation studies, experiment design, and evaluating similitude between datasets from different test reactors and irradiation conditions. Emerging Nuclear Regulatory Commission (NRC) guidance on fuel qualification acknowledges the need to demonstrate that experiment data is representative of full-scale reactor conditions by quantifying the impact from geometric and temporal distortions. This approach will provide researchers with the tools to design irradiation experiments at altered geometric and temporal scales that produce test data scalable to LWR conditions.

Recent test data from the Separate Effects Test Holder (SETH) and Critical Heat Flux Static Environment Rodlet Transient Test Apparatus (CHF-SERTTA) experiments at TREAT will be used to analyze separate effects experiments that replicate distinct fuel and cladding thermal and mechanical responses from Reactivity Initiated Accident (RIA) power pulses. The objective will be to identify regions of scaling agreement which can allow data from multiple separate effects test to capture separate phases of transients. The potential outcome from this is that the number of integral effects



tests could be reduced if the overall behavior can be produced from several well scaled separate effects experiments. The data synthesis component of DSS permits detailed comparisons between simulated predicted data and accompanying experiment data, which will complement current advanced instrumentation and irradiation testing This will provide objective metrics to measure the similitude between a fueled experiment design and its corresponding full-scale conditions.

Accomplishments:

Initial data synthesis efforts with DSS were applied to the SETH-C experiment. An accompanying BISON model of the experiment was parametrically varied to identify regions

Figure 1. Temperature comparison

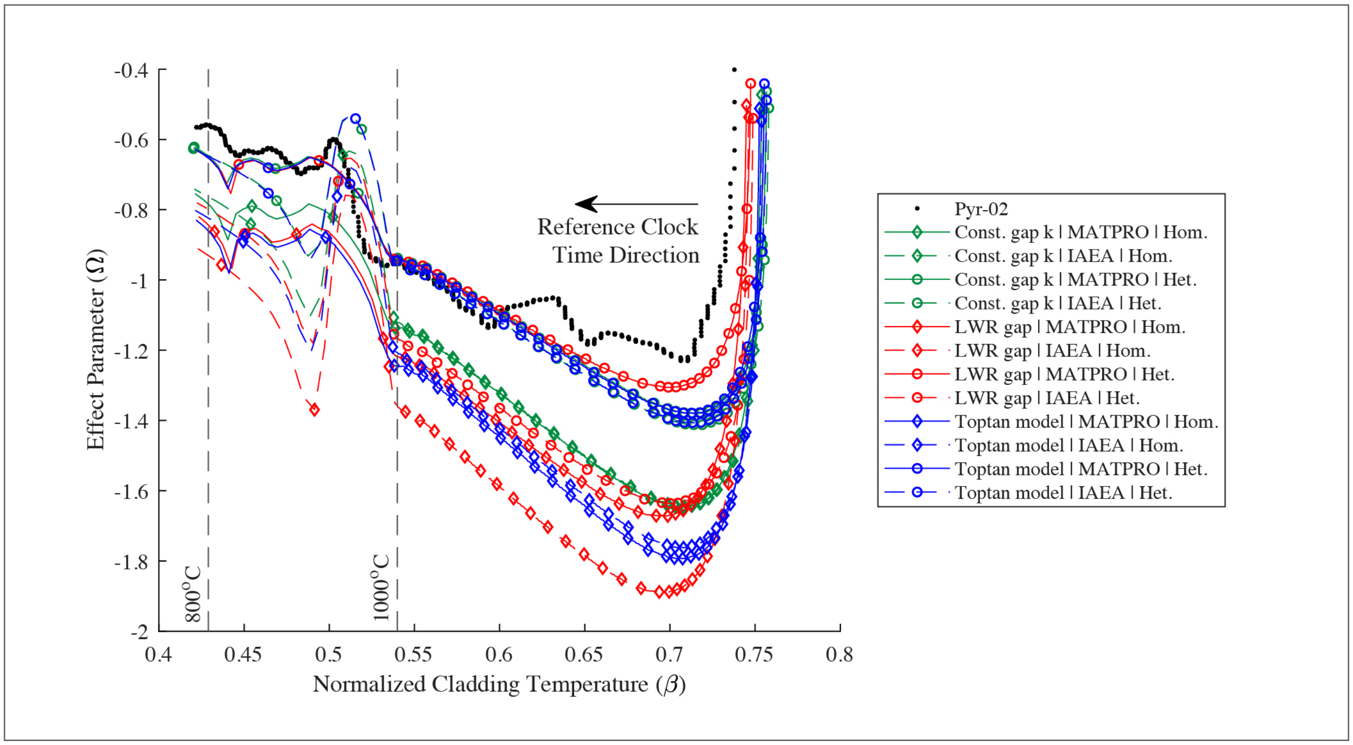


Figure 2. Ramp down phase scaled process curve comparison

of modeling agreement, and sensitivities to gap conductance models, radial power deposition profile, and cladding material property databases. Three gap conductance models in BISON were considered, along with two radial profile power depositions, and the two available Zr-4 material property databases were considered resulting in twelve model combinations to analyze. The measured power pulse was applied to the BISON simulation and cladding temperature results were compared to the measured pyrometer data. Figure 1 shows qualitative comparison between the predicted temperature traces and measured data showed good agreement, however a DSS analysis revealed several features that were not easily observed from a traditional comparison.

Analyzed temperature data after peak cladding temperature produced scaled process curves shown in Figure 2. For ideal similitude to be achieved, the curves produced from the various BISON simulations would directly overlay on top of the measured data curve. Separation from the measured scaled process curve indicates an area local distortion. It was seen that the area of largest modeling discrepancy corresponded to temperatures undergoing the α-β phase transition region in Zr-4 cladding. This feature would not have been easily distinguishable from the temperature comparison in Figure 1. This allowed the MATPRO and International Atomic Energy Agency (IAEA) material property databases to be objectively compared with the SETH-C test data.

The total separation computed between the predicted and measured process curves to provide an objective metric to rank the similitude of different modeling options. It showed that a heterogeneous radial power profile corresponded to less separation consistently regardless of gap conductance model or cladding material property database used. An advantage of this ranking was that it showed that simplified cases produced similar separation to the more complex models which can be used by researchers to justify modeling decisions. These findings were subsequently published in Nuclear Science and Engineering

Radiation Resistant High Entropy Alloys for Fast Reactor Cladding Applications

Principal Investigator: Adrien Couet (University of Wisconsin-Madison)
Team Members/ Collaborators: Calvin Parkin, Michael Moorehead, Mohamed Elbakhshwan, Dan Thoma, Kumar Sridharan (University of Wisconsin-Madison), Lingfeng He, Mukesh Bachhav (Idaho National Laboratory), Meimei Li, Weiying Chen (Argonne National Laboratory), David Armstrong (Oxford University, UK), Alan Savan, and Alfred Ludwig (Ruhr University of Bochum, Germany)

Figure 1. MaDCoR Laboratory group photo including members of the HEA team

The advent of high entropy alloys (HEAs) presents new opportunities and challenges for alloy design. The project evaluated the phase stability, mechanical properties, radiation effects, and corrosion behavior of various HEAs. To effectively probe the expansive compositional spaces of HEA families, laser metal deposition 3D printed arrays of samples and combinatorial magnetron-sputtered thin films were combined with CALPHAD phase predictions and high-throughput experimental techniques to generate large data sets conducive to machine learning applications. More targeted mechanical and irradiation experiments and advanced microscopy were conducted with selected promising compositions.

Project Description:

The purpose of this research was to holistically evaluate the design benefit of multiple principal elements in solid-solution to nuclear structural materials. Previous modeling and experiments suggest the compositional complexity of such a base matrix may postpone the onset of rapid swelling, so the project set out to experimentally determine the mechanisms that influence void swelling in such alloys. The project focused on the effect of a compositionally complex base matrix, without the

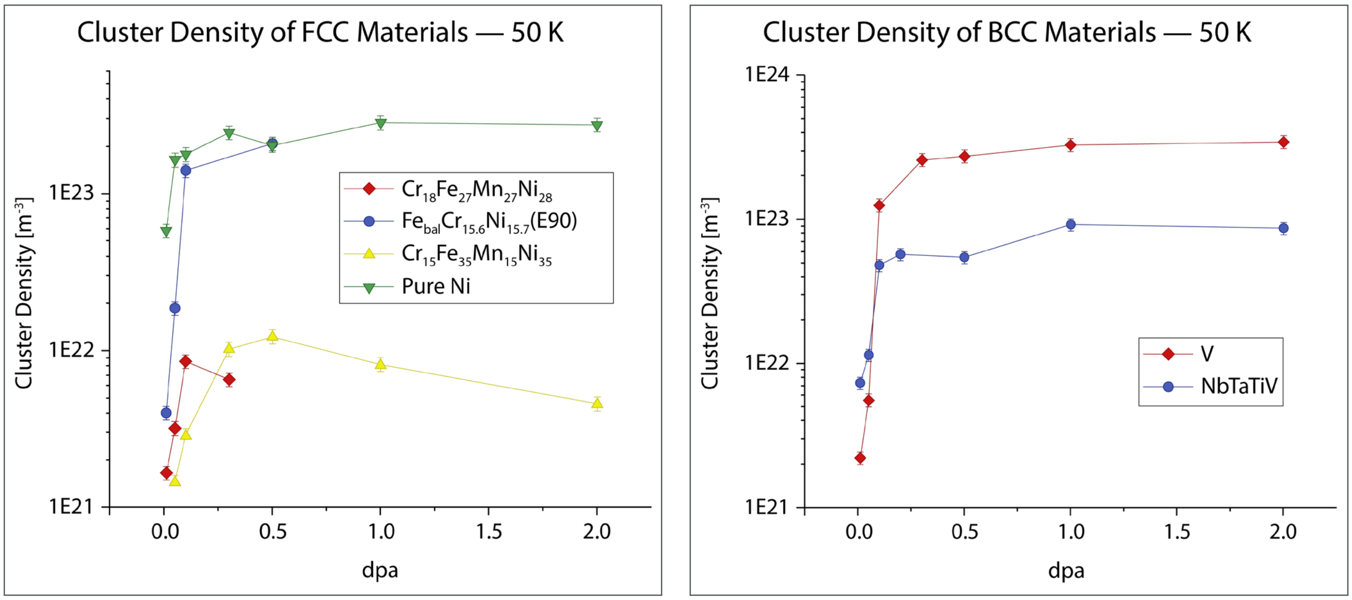


Figure 2. Defect cluster densities of FCC (left) and BCC HEAs (right) compared to less compositionally complex reference materials

added complexity of multiple phases. As such, single-phase alloy candidates were selected for further study. Experimental studies of radiation damage mechanisms support the Department of Energy (DOE's) objectives by enhancing the fundamental understanding of extended defect formation and growth which guides alloy design efforts and further modeling efforts. In addition to a damaging radiation field, nuclear structural materials must perform at the intersection of corrosive coolants and mechanical stresses, all at high temperature. The project set out to benchmark the mechanical properties and thermal stability of selected alloys against other well-understood structural materials, as well as the corrosion behavior in current and advanced reactor coolants. HEAs present the unique challenge of an extensive compositional space, with properties that may vary drastically even over relatively small atom percents. High-throughput sample fabrication, experimentation, and CALPHAD phase predictions were developed to address this problem, and support DOE efforts to expediate nuclear materials discovery, and to some extent qualification, to meet the increasing demands of future reactor designs.

Accomplishments:

Several irradiation damage experiments and the respective post-irradiation examination were conducted by Calvin Parkin and supplemented by other teammates:

- A paper concerning high temperature and cryo in-situ irradiations at the IVEM-Tandem experiment at Argonne National Laboratory was published in Acta Materialia. In the cryogenic irradiations, diffusion of point

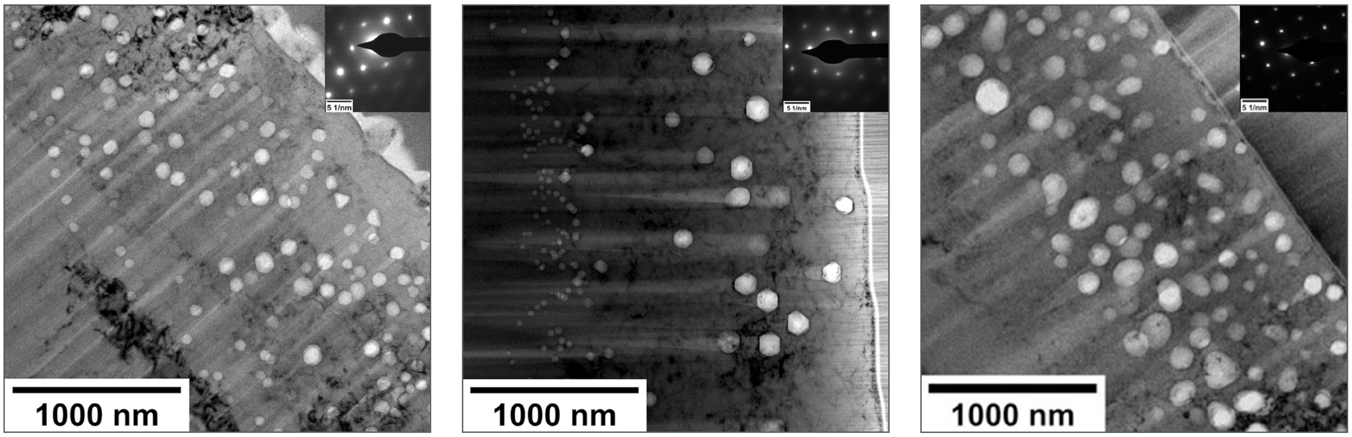


Figure 3. TEM micrographs of voids in $Cr_{18}Fe_{27}Mn_{27}Ni_{28}$, $Cr_{15}Fe_{35}Mn_{15}Ni_{35}$, and 316H stainless steel irradiated to 150 dpa at 500 C using 4.0 MeV Ni heavy ions

A compositionally complex base matrix may delay the nucleation stage and extend the incubation time of irradiation-induced extended defects such as voids and dislocation loops and may be used as a design parameter to tune alloys for radiation damage resistance in addition to the classical sink strength density approach.

defects was suppressed, and it was revealed by quantification of defect clusters that in body centered cubic (BCC) and face centered cubic (FCC) HEAs, the primary defect production is reduced compared to less compositionally complex materials (see Figure 2). High-temperature irradiations revealed a difference in the size of faulted dislocation loops in Cr18Fe27Mn27Ni28 and Cr15Fe35Mn15Ni35 alloys after 2 dpa, which was attributed to the effect of composition on vacancy mobility, stacking fault energy, and nucleation of interstitial loops.

- Bulk heavy ion irradiations of Cr18Fe27Mn27Ni28, Cr15Fe35Mn15Ni35, and 316H stainless steel were performed to 150 dpa at 500 C and 600 C and voids were characterized using Transmission Electron Microscopy (TEM). The swelling measurements from these experiments are used to construct temperature-swelling and local dpa-swelling curves to compare void swelling across materials (HEAs vs 316HSS) Early characterization indicates that while 316HSS experienced relatively uniform void swelling across the irradiation depth, void populations in the two HEAs experienced lower void density overall and more variation in void size, especially with depth. See Figure 3.
- The first high-throughput irradiation of an as-deposited magnetron-sputtered combinatorial CrFeMnNi thin film to 5 dpa at room temperature was performed at the UW-Madison Ion Beam Laboratory (IBL) using 4 MeV Ni++ heavy ions. The development of the automated 2D irradiation stage at the UW IBL was led by Michael Moorehead and Dr. Benoit Queylat. Likely because the as-deposited thin film has nano-grains, no voids formed, and energy dispersive spectroscopy (EDS) revealed no significant elemental redistribution. Most points on the thin film remained stable on the substrate during irradiation, setting up the next step, which is to irradiate an annealed wafer with a reduced sink density (i.e., larger grain size) to better probe the role of the compositionally complex matrix on irradiation resistance.
- HEA, which began homogeneous and single-phase, were characterized for irradiation-induced chemical redistribution using Super-X EDS near voids and along depth profile as part

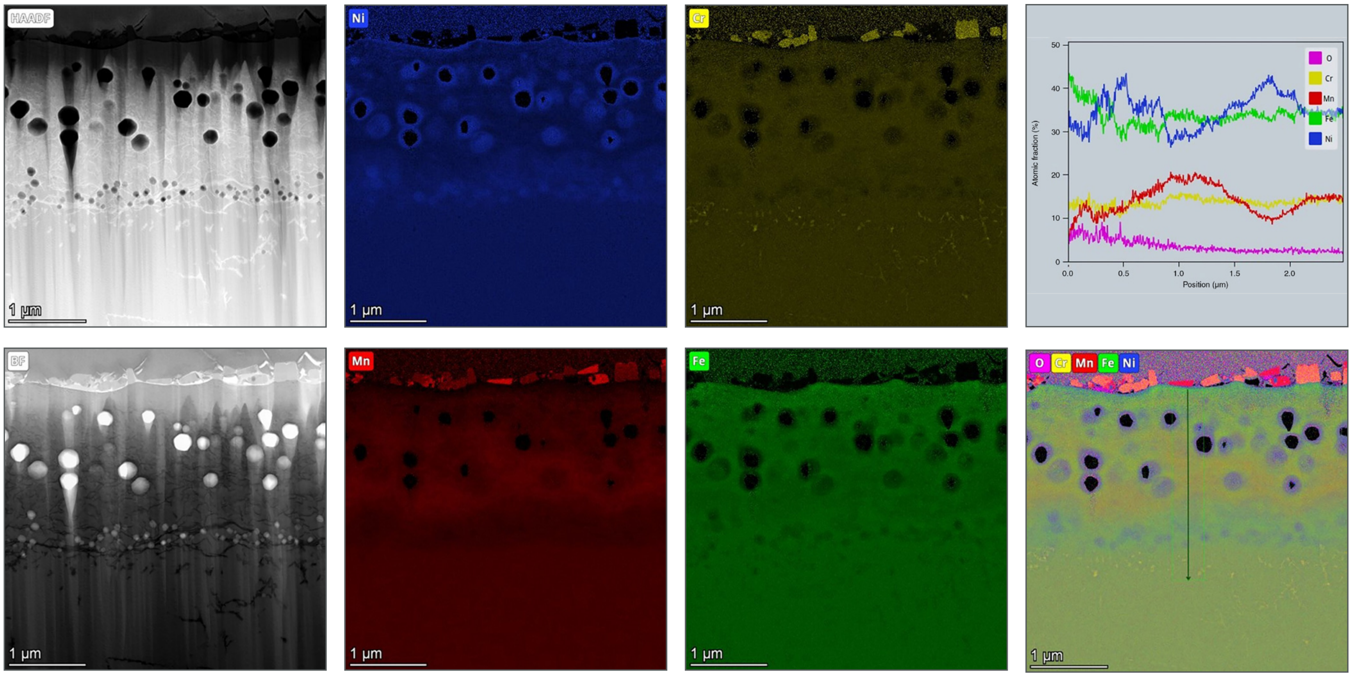


Figure 4. Super-X EDS chemical maps and depth profile line scan of $Cr_{15}Fe_{35}Mn_{15}Ni_{35}$ irradiated to 100 dpa at 500 C using 5.0 MeV Ni heavy ions

of a Nuclear Science User Facility (NSUF) Rapid Turnaround Experiment (RTE) #20-4095, which was awarded March 2020. These efforts were supervised by Dr. Lingfeng He at Idaho National Laboratory. Clear Ni enrichment near the surface of voids is observed in all cases. Mn shows signs of diffusing away from voids, with a more diffuse interface, while Fe and Cr are simply depleted due to enriched Ni. Away from voids, Mn and Ni show depth-dependent enrichment and depletion (see Figure 4), which will provide information about vacancy diffusion.

- The team has also begun steps toward experimental measurement of stacking fault energy by measurement of partial dislocation spacing by creating a focused ion beam lamella from a Cr18Fe27Mn27Ni28 specimen previously fractured in tension and sending off material to Dr. Boris Maiorov at Los Alamos National Laboratory for resonant ultrasound spectroscopy measurements of shear modulus and Poisson's ratio. A new sample of Cr18Fe27Mn27Ni28 was prepared to obtain faulted loop statistics needed for a publication.
- Calvin Parkin was awarded RTE #21-4233 to investigate void nucleation and growth in HEAs under Kr++ ion irradiation and He+ co-implantation at the IVEM-Tandem facility at Argonne National Laboratory. The work will be supervised by Dr. Meimei Li and Dr. Weiying Chen. Since previous IVEM experiments did not result in void formation (see Couet_2), this new RTE will provide valuable insight on the effect of compositional complexity on He-diffusion and the void nucleation in a stabilizing environment.

Development of Advance High-Cr Ferritic/Martensitic Steel for Reactor Applications

Principal Investigator: K.D. Clarke
Team Members/ Collaborators: C.J. Rietema, M.A. Walker, S.K. Ullrich, M.M. Hassan, M.R. Chancey, C.B. Finfrock, T.R. Jacobs, G.R. Bourne, D.V. Marshall, B.P. Eftink, O. Anderoglu, Y.Q. Wang, T.A. Saleh, S.A. Maloy, and A.J. Clarke

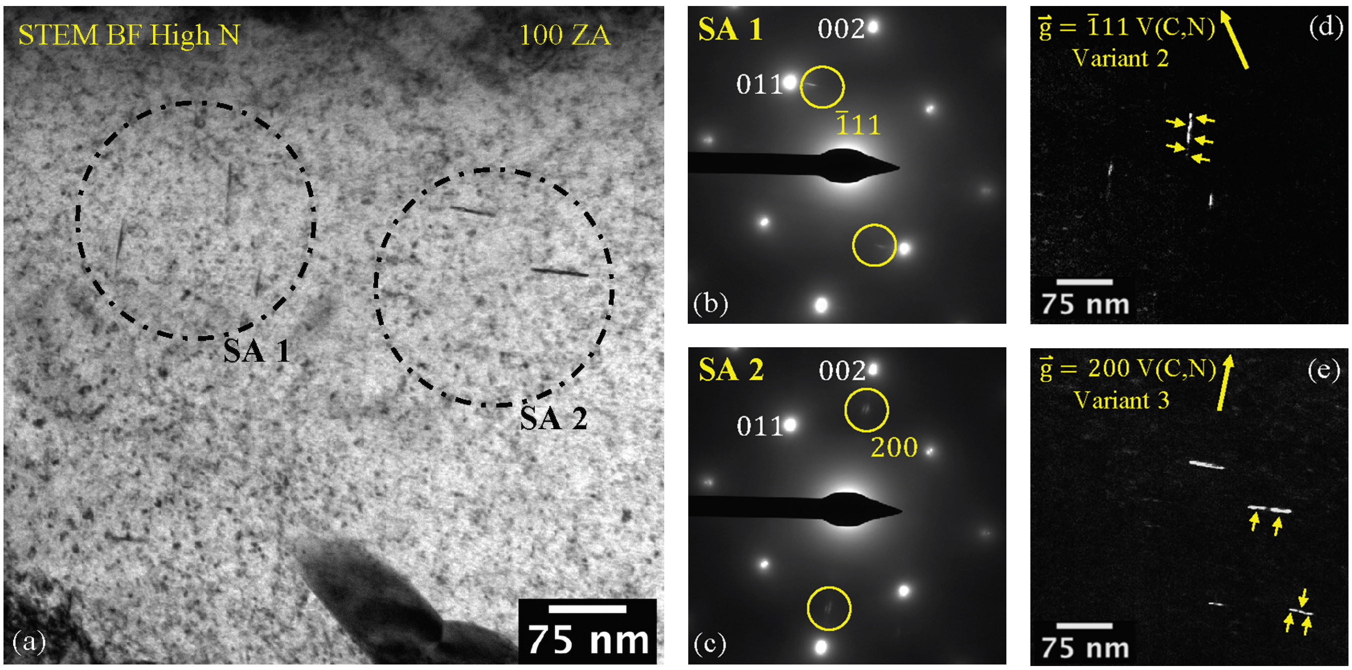
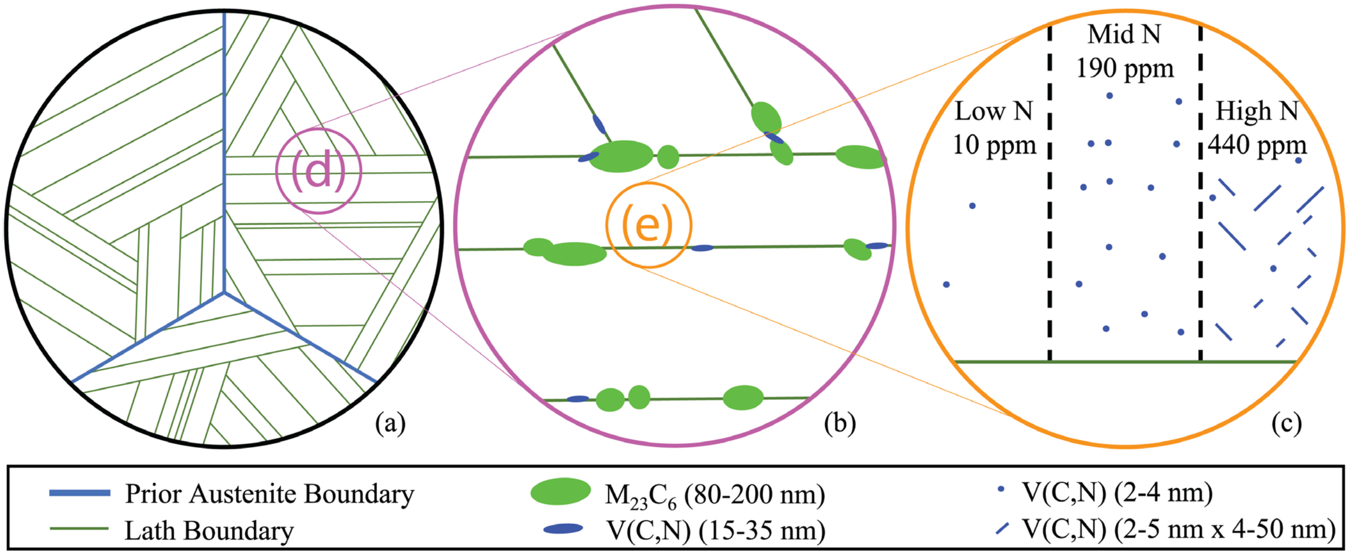


Figure 1. A schematic of the general microstructure of 12Cr-1MoWV steel at three different length scales after conventional heat treatment. (a) shows the prior austenite grain structure with laths of tempered martensite having developed during heat treatment. (b) shows a magnified region within (a) revealing the presence of M23C6 and V(C,N) decorating the lath boundaries. (c) shows a magnified region within (b), illustrating the different morphologies and volume fractions of intralath V(C,N) found across the low, mid, and high N alloys.

The fuel cladding in sodium-cooled fast reactor designs call for irradiation resistant materials able to withstand a wide range of temperatures and pressures, all within a corrosive environment. The high Cr (12 wt%) ferritic/martensitic (FM) steel HT9 is a candidate material, but irradiation at low temperatures (<300 °C) can sharply reduce ductility. Thus, improving ductility of the fuel cladding at lower temperatures is critical to ensure safe and efficient reactor operation. Alloys designed to minimize irradiation hardening after low-temperature irradiation, while maintaining excellent high-temperature irradiated properties, will improve reactor efficiency and help generation IV reactor technology progress. The work performed here to optimize HT9 microstructures to enhance ductility during low-temperature irradiation by fundamental understanding of nitrogen (N) alloying effects is a collaborative effort between Colorado School of Mines (CSM), Los Alamos National Laboratory (LANL) and the University of New Mexico (UNM).



Project Description:

The recent identification of N as a potential element of interest for controlling the irradiation resistance of FM steel has led to the detailed investigation of how the irradiated microstructure and hardness change with increased N content. Initial characterization of HT9 alloys containing low (10 ppm), mid (190 ppm), and high (440 ppm) N led to the first-time observation of ultrafine (<10 nm) vanadium carbonitride precipitates (shown in Figure 1) that change morphology and volume fraction with increasing N content (shown in Figure 2). It is hypothesized that if the ultrafine precipitates are stable under irradiation, they could act as sink/recombination sites for self-interstitial atoms (SIAs) and transmutation He. Initial characterization efforts also led to the development of a new high-throughput method for comparing the relative amount of vanadium carbonitride and interstitial N in FM steels utilizing time-of-flight secondary ion mass spectrometry (ToF-SIMS). Not only will the method be useful for future irradiation studies interested in changes in precipitates volume fraction, but the method is also applicable to other steels research, e.g., nitriding for wear applications or rapid heat treatment optimizations. The three alloys were then irradiated to a dose of 1 dpa with 1.5 MeV protons at 300°C. Analysis of the steels post-irradiation

Figure 2. Ultrafine intralath vanadium carbonitrides V(C,N) in the High N alloy. (a) shows two variants of strongly diffracting V(C,N) in a bright field (BF) TEM image taken slightly off of the [100] Body Centered Cubic (BCC)-Fe zone axis. Selected areas (SAs) 1 and 2 correspond to the selected area diffraction patterns (SADPs) in (b) and (c) respectively, which have the relevant diffraction spots circled for the two variants of V(C,N) visible in (a). (d) and (e) show that the ultrafine precipitates in the irradiated alloy are still organized in clusters of smaller particles (indicated with arrows) forming along the habit plane. (d) and (e) are Centered Dark Field-Transmission Electron Microscopy (CDF-TEM) images taken with the diffraction spots circled in (b) and (c).

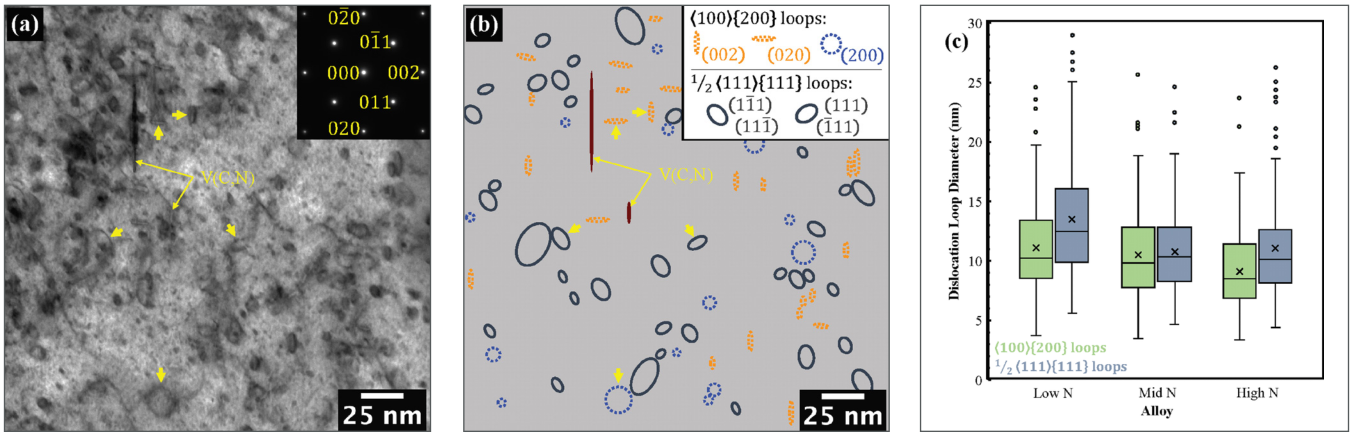


Figure 3. (a) Color inverted annular dark field (ADF) STEM image of the High N alloy after irradiation with 1.5 MeV protons to 1 dpa at 300˚C with many visible loops. Imaging down the [100] zone axis (selected area diffraction pattern (SADP) in the upper-right corner of (a)) satisfies the g·b criteria for the imaging of nearly every variant of a<100> and a/2<111> dislocations. The yellow arrows point to an example of each of the possible common dislocation loop variants. (b) A schematic of the indexed dislocation loops from (a). Because the loops are assumed to be circular and tend to form on distinct planes ({100} and {111}), the projected shape of the dislocation loops based on the viewing direction of the crystal can be used to identify each dislocation loop; the major axis of each ellipse corresponds to the diameter of the dislocation loop in plane. (c) The distribution of a<100> and a/2<111> dislocation loop diameters. The mean diameter is represented by the 'x', and outliers are shown as points above box and whisker plots.

Elevated N content in the ferritic/martensitic steel HT9 has been found to decrease irradiation hardening after 1 dpa of 1.5 MeV protons at 300 ˚C, and likely reduce irradiation swelling at higher irradiation doses, both of which improve performance in next-generation reactor applications.

showed that the high N alloy exhibits reduced irradiation hardening relative to the low N alloy. Additionally, based on the defect structure of the alloys, in particular the dislocation loop type, size, and number density there appears to be a decrease in SIA cluster mobility with increasing N content. An example of the dislocation loop structure is shown in Figure 3 along with chart summarizing differences in loop size across the alloys. Overall, this work indicates that increasing the overall N content in ferritic/martensitic alloys may be beneficial to irradiated properties.

Accomplishments:

To understand the effect of N on the irradiation hardening behavior of F/M steel three primary technical goals were carried out: the production and initial characterization of the experimental alloys, the proton irradiation of the alloys, and the characterization of the irradiated experimental alloys. First the experimental alloys were designed and produced, and the initial microstructure and properties were characterized. The low and high N alloys were designed at LANL and produced by Sophisticated Alloys while the mid N alloy was designed at CSM and produced by Cleveland Cliffs. The initial mechanical characterization for the three alloys included subsized tensile testing and microhardness testing and was performed at the UNM. The microstructural characterization was done at CSM and included traditional metallography, texture analysis using electron backscatter diffraction, dislocation density analysis via x-ray diffractometry, and precipitate analysis via transmission electron microscopy (TEM) and ToF-SIMS. This portion of the research generated two of the publications listed below. The next technical goal was the irradiation of each of the alloys. The irradiation was performed successfully using a 3 MeV tandem ion beam line in the Ion Beam Materials Laboratory at LANL. The last technical goal was the comparison of irradiation hardening amongst the alloys in addition to a detailed characterization of each alloy. To evaluate irradiation hardening, microhardness and nanoindentation were done at CSM. The characterization work was done primarily via scanning transmission electron microscopy also at CSM. Figure 4 shows graduate student Connor Rietema loading an irradiated sample into an electron microscope to prepare a specimen for analysis in the TEM. This portion of the research has been written up as a manuscript that is currently under review for publication. All three technical goals are outlined in detail with the finding explained in the PhD thesis listed below.

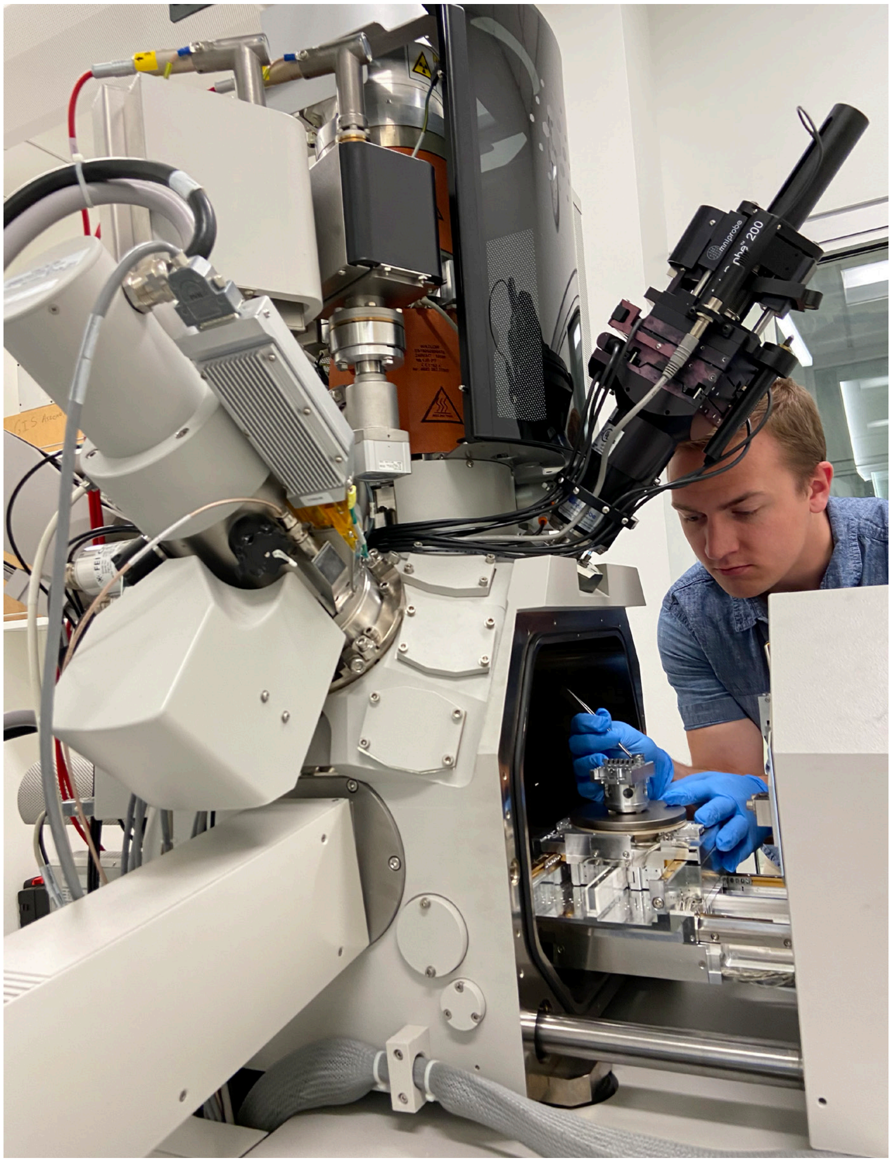


Figure 4. A proton irradiated sample being loaded into a dual-beam focused ion beam/scanning electron microscope by graduate student Connor Rietema to produce a specimen for analysis in the transmission electron microscope

Nanostructured Composite Alloys for Extreme Environments

Principal Investigator: Osman Anderoglu
Team Members/ Collaborators: N. Mara, M. Radhakrishnan, M. Knezevic, J. Gigax, and N. Li

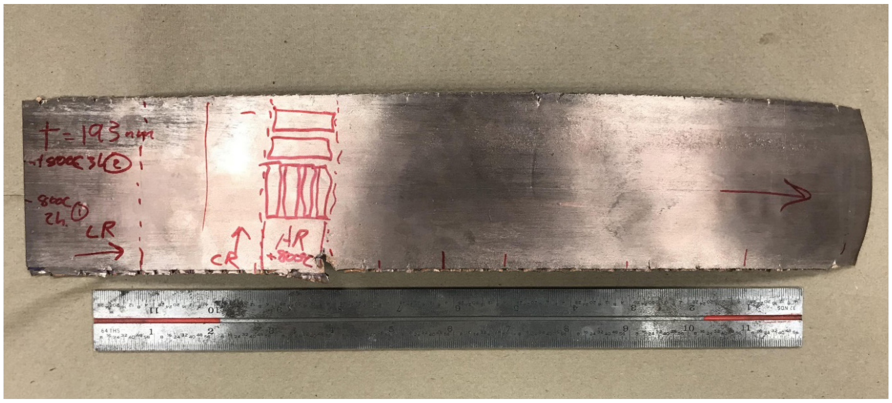


Figure 1. A photograph of the as-rolled h = 200nm ARB Cu/Nb. A ruler measured in inches is included for scale

For the first time, accumulative roll bonding processed bulk nanolayered composites showed remarkable void swelling resistance at very high irradiation doses (200-450 dpa, 400-600C).

One of the aims of the Fuel Cycle R&D (FCRD) program is to develop cladding materials for very high burn up. This can be accomplished with materials that can withstand very high irradiation doses. Conventional ferritic/martensitic steels are one of the candidate alloys show significant embrittlement at low temperatures (<400°C) and swelling at medium temperatures (~450°C) depending on the dose. A non-conventional promising alternative is nanolayered composite alloys. Thanks to a very high density of stable interfaces which provides recombination sites for irradiation induced point defects these alloys show remarkable radiation resistance. Furthermore, they can be designed for envisioned medium (coolant, temperature, etc.) so that they can perform in the extreme environments.

Project Description:

The objective of this proposal is to develop extreme performance bulk nanocomposite alloys that can withstand very high irradiation doses at elevated temperatures for demanding nuclear environments such as fast reactor cladding applications. Nanolayered (20-600 nm individual layer thickness) Cu/Nb and Zr/Nb composites were produced using innovative accumulative roll bonding (ARB) technique. Mechanical property and thermal stability studies were performed up to 600°C followed by microstructural characterization. The selected alloys were irradiated up to 200 dpa using 5 MeV self-ions at 400-600°C. High dose irradiations were performed for the first time on these alloys as part of the project. Remarkable stability under irradiation was shown. Findings show an innovative approach for production of advanced alloys for demanding applications. The study on model bilayers indicates possibility of producing cladding materials both for safer operation of current operating reactors as well as new alloys for other demanding environments of future advanced reactors.

Accomplishments:

ARB processing of Cu/Nb and Zr/Nb nanocomposites down to 20nm individual layer thickness including cross-rolling to change the interface structure at the same layer thickness was accomplished (N. Mara, J. Cheng, U of Minnesota; Danial Savage, Marko Knezevic, U of New Hamshire) while a new ARB processing capability was also established at UNM (O. Anderoglu, M. Radhakrishnan). After processing, microstructure characterization including scanning electron microscopy , detailed texture analysis using neutron diffraction and transmission electron microscopy (TEM) were

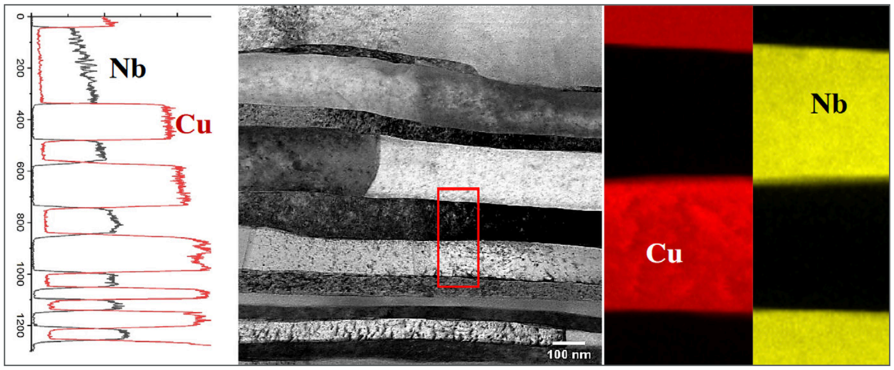


Figure 2. TEM image Cu/Nb composite after 5MeVCu irradiation at 400°C to 100 dpa. Layer structured is well preserved with no indication of mixing.

completed investigating the layer size, morphology, and texture (J. Cheng, M. Radhakrishnan, N. Mara, O. Anderoglu). To establish structure-property relations, mechanical testing as a function of layer thickness including at high temperatures (M. Radhakrishnan, O. Anderoglu) as well as thermal conductivity measurements were completed (M. Khafizov, Ohio State U.). Mechanical testing was also completed perpendicular to layer thickness (J. Gigax, LANL) showing similar results as parallel to layer thickness measurements. High dose irradiations were completed at 400C (200 dpa) and 600C (450 dpa) (O. Anderoglu, UNM; Y. Wang, LANL; MIBL, U of Michigan). Detailed investigations including TEM and Atom Probe Tomography (APT) were performed.

Cu/Nb: 5MeV Cu irradiations at 400C up to 200 dpa (~200nm below the surface) did not show any voids but some clusters presumably due to ballistic collisions were found on both Cu and Nb layers. However, at 600C and 450 dpa, although no cavities were obtained, increase in layer size were observed. Since this was observed on unirradiated layers as well, it was thought to be thermally driven.

In Zr/Nb, the irradiation caused chemical mixing of layers in all the layer thicknesses studied and the mixed layer thickness extended to multiple bi-layer periods. The observed mixing is attributed to the liquid phase interdiffusion within thermal spikes, and a stable mixed layer was favored by the liquid-phase miscibility of Zr and Nb.

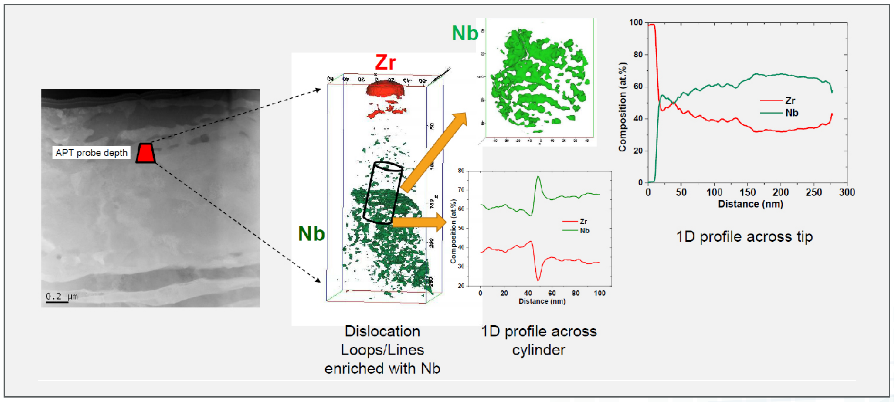


Figure 3. Dark field TEM and APT of Zr irradiated Zr/Nb showing mixing of the layers

ATF Cladding Tube Mechanical Evaluation

Principal Investigator: Ben Eftink
Team Members/ Collaborators: Mathew Hayne, Peter Beck, James Valdez, Cheng Liu, Thomas Nizolek, Tarik Saleh, Stuart Maloy, and Ben Eftink

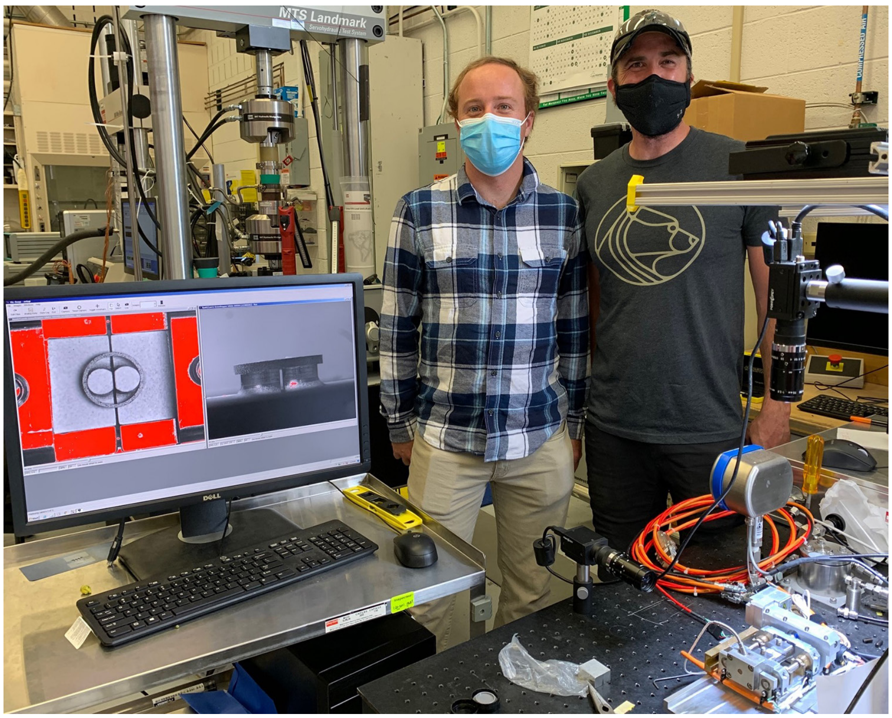


Figure 1. Graduate student Peter Beck (left) and postdoc Mathew Hayne (right) performing ringpull testing with DIC on Cr coated Zircaloy. Testing is conducted on a miniature and portable load frame, bottom right, mounted to an optical table. Computer monitor shows the view of the DIC cameras.

Accident tolerant fuel claddings of FeCrAl and Cr coated Zircaloy alloys provide safety improvements for light water reactor (LWR) fuel claddings. These materials are being studied for early implementation in LWRs. Understanding how they respond to mechanical stress, particularly in the hoop direction of the tube is necessary before wide implementation. For Cr coated Zircaloy tubes, specifically the integrity of the Cr coating during mechanical loading is not known. Evaluating cladding tubes mechanically in the hoop directions is challenging and testing methods require special considerations. Testing the tensile properties in the axial (length of the tube) direction is straightforward, however, does not always translate to the hoop direction properties due to microstructural anisotropy from tube extrusion processing. Since the hoop direction is the direction that sees the greatest stress in real world situations, it is necessary to test the hoop direction mechanical properties directly.

Project Description:

This project had two main technical objectives: 1) evaluate the hoop direction mechanical properties of accident tolerant Cr coated Zircaloy and FeCrAl alloys and 2) further the understanding of the ringpull mechanical testing technique. The project team included an early career staff member, postdoc, and a graduate student visiting as part of a Nuclear Energy University Project (NEUP; see Figure 1).

Implementing accident tolerant cladding concepts in the next few years requires confidence in the claddings' mechanical performance. This includes understanding the tubes response in the hoop direction and for the Cr coated Zircaloy material, the integrity of the coating during mechanical loading. By mechanical testing of these cladding tubes, we are understanding better how they will hold up in service conditions including high mechanical strain (due to internal gas pressure) accident scenarios. This directly

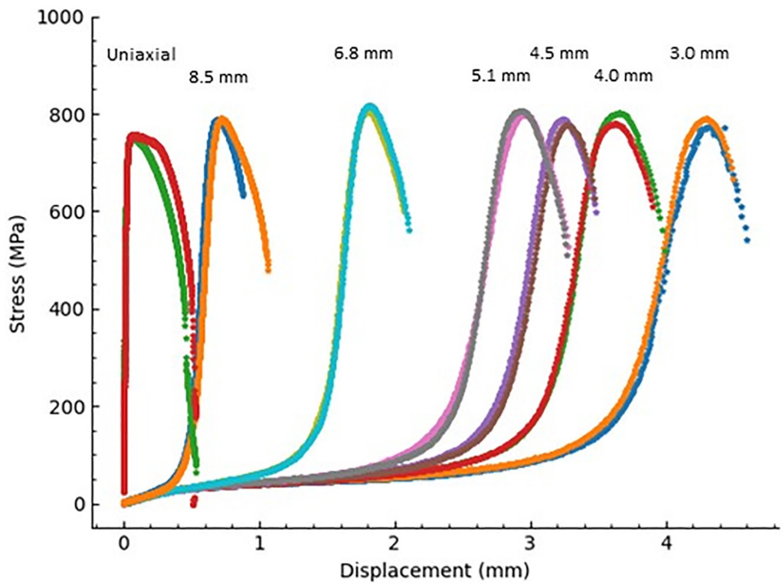


Figure 2. Engineering stress vs. displacement curves of C26M welded material measured though no-gauge ring pull tests using a mandrel diameter of 3.0, 4.0, 4.5, 5.1, 6.8, or 8.5 mm as well as uniaxial tension tests

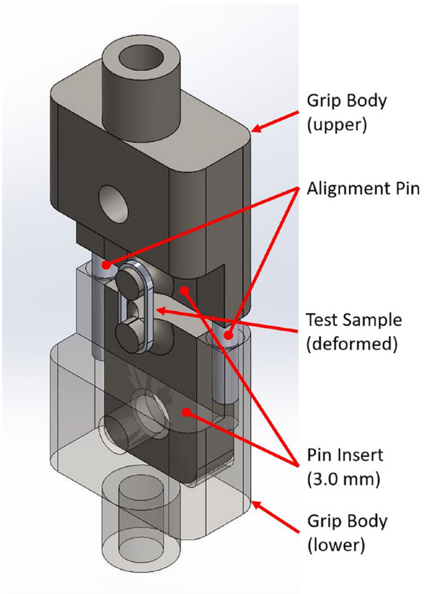
supports the Department of Energy (DOE) objectives of safe, reliable and thereby economical operation of the nation's current reactor fleet.

The second component of the project investigated the interpretation of ringpull test data. Specifically, we used digital image correlation (DIC) to measure local strains to interpret the load/displacement data from the tests. This interpretation was done with a systematic study on a range of ringpull fixture mandrel sizes chosen based on our analytical strain calculations. This component of the project is important because different institutions use different ringpull test fixtures that can have different mandrel diameters. In addition, cladding tubes come in a range of diameters and thicknesses. Figure 2 shows how the stress strain curves change with different mandrel diameters. When interpreting past results and planning future ringpull tests, understanding the impact of the ratio of mandrel and tube diameters is critical to meet DOE objectives.

Accomplishments:

The project goals were achieved by ringpull testing both Zircaloy and FeCrAl cladding tubes using our newly developed fixture, Figure 3, on our hotcell friendly and miniature load frame, Figure 1. During the ringpull testing we were able to extract the yield and ultimate tensile strength of the

Mechanical evaluation of cladding tubes is not straight forward, this work package developed and tested new ringpull testing protocols that have shown to extract mechanical properties from accident tolerant fuel claddings.



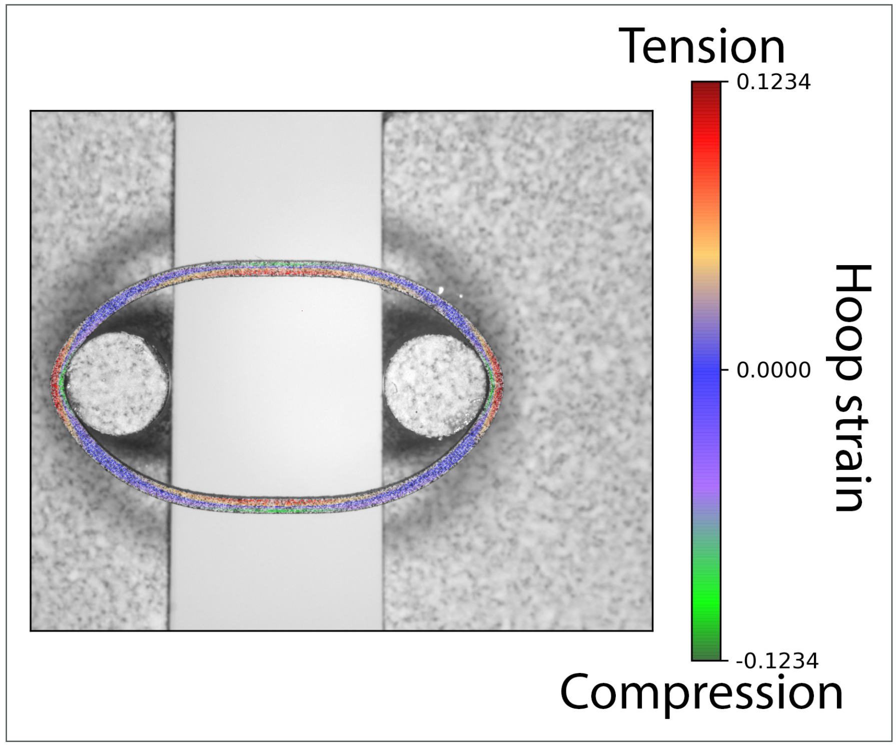


Figure 3. (Above) Solidworks 3D model showing the assembled grip body with 3.0 mm diameter mandrel inserts used for the gaugeless ring pull testing

Figure 4. (Right) DIC visualized hoop (tangential) strain using custom developed interpretation software

cladding tubes in the hoop direction. In the case of the FeCrAl cladding tubes, a comparison was made to tensile tests from samples of the same tube and the strengths between the testing techniques were similar.

Additionally, substantial progress was made towards the ringpull technique testing and analysis. Based on analytical strain calculations and verification by experimental testing on a newly designed test fixture, Figure 3, the relationship between tube diameter and mandrel diameter is better understood. When the tube/mandrel diameter ratio is too large, then there is a tendency to fail the sample where the tube is in contact with the mandrel and the potential of brittle samples to prematurely break. With a tube/mandrel diameter ration too small, there are issues with sample gauge dimensions being far from and incomparable to tensile tests. We also conducted ringpull and tensile tests on annealed samples from the same FeCrAl tube to isolate the impact of strength and ductility on the ringpull test results. That comparison is still being analyzed.

Lastly, we developed post processing DIC strain analysis tools that are planned to be released open-source. This toolset is capable of separating out different strain components including the hoop strain, identifying tensile and compressive locations during the ring deformation, calculating stress and strain from the load/displacement data, and extracting quantitative strength and ductility values. The analysis tools also incorporate figure building capabilities for visual communication of the data. Figure 4, for example, shows the hoop (tangential) strain results during a test, there are both tensile and compressive strains in the ring

ADVANCED REACTOR FUELS

FAST/Accelerated Testing for Advanced Fuel Designs

Principal Investigator: Geoffrey Beausoleil
Team Members/ Collaborators: Christopher Murdock, Nate Oldham, Luca Capriotti, Randall Fielding, and Bryon Curnutt

The FAST tests performed in FY21 by the Advanced Fuels Campaign (AFC) demonstrated the viability of reduced scale fuel testing for accelerated fuel R&D with burnup rates that were up to six times faster than conventional testing methods.

The Fission Accelerated Steady-state Test (FAST) approach to irradiating fuel completed its first irradiation cycle within the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL). The FAST method to irradiating was shown to accelerate irradiation rates by up to a factor of six and is based upon the ability to increase the power density of a test pin by scaling the fuel pin down while maintaining an equivalent linear heating rate. The tests were initially designed to investigate alloy fuels for sodium fast reactors (SFRs). The tests performed in the first cycle include controls fuel pins, sodium free annular fuel pins, fuel with liners, and fuel with alloy additives.

Project Description:

The objectives of the FAST experiment are, at a basic level, to improve the timeline of advanced fuel research and development (R&D) efforts and to support accelerated qualification methodologies. The project lends itself to support high throughput, rapid prototyping of fuel designs to be able to provide qualitative comparisons to the performance of standard fuel designs. The ability to produce high burnup fuel pins of both a control and novel fuel design within the same capsule and irradiation conditions is critical to the development of next generation reactors. Novel fuel designs can then be paired against controls in post-irradiation furnace tests or transient testing in the Transient Reactor Test Facility (TREAT). This allows rapid qualitative comparisons of performance between novel designs against established performance metrics. The solution of simply accelerating irradiation, however, requires consideration of many other phenomena that FAST can provide answers for. Phenomena such as fission product diffusion and alloy re-distribution are heavily influenced by temperature gradients, time at temperature, and concentration gradients. By scaling the geometry to a reduced size and reducing the total time under irradiation, the balance of time and distance for diffusion centric phenomena is tested. This has important implications to developing high burnup alloy fuel as the roles of alloy redistribution and fission product concentrations at the cladding are both important fuel performance metrics. This adds additional value to the use of FAST as it provides critical data to the development and implementation of physics-based models by virtue of the departure from fully prototypic irradiation and thus an improved, broader basis for validating performance codes against.

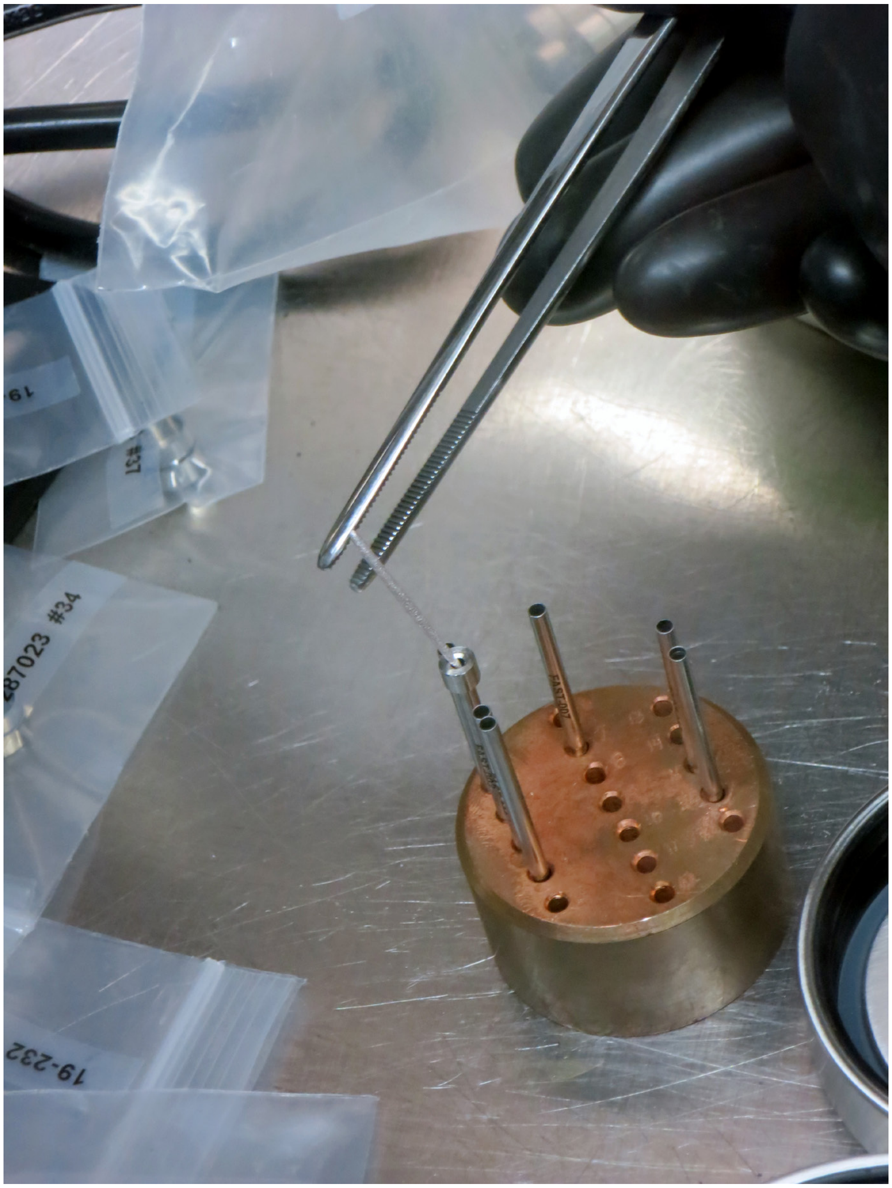


Figure 1. Technicians at the Materials & Fuels Complex (MFC) load sodium into the rodlets. All fuel fabrication and assembly work at MFC occurs within a glovebox to ensure oxidation risks are limited. After loading, the sodium is melted so that it forms a uniform bond between the fuel and cladding. The sodium is used to maintain adequate heat transfer between fuel and the cladding during early irradiation stages until the fuel is able to swell and make physical contact with the cladding.

Figure 2. Assembled FAST rodlets. Each rodlet contains a 0.75" long fuel pin that is scaled at ½ of the radius of standard Experimental Breeder Reactor (EBR-II) or AFC fuel pins. Each capsule can hold up to two of these rodlets, notionally a standard U-10Zr control rodlet and one of the novel fuel designs to be tested.

Accomplishments:

a. Murdock, Oldham, Fielding, and Curnutt completed the final experiment fabrication, assembly, as-built analysis, and insertion of the first phase of FAST experiments into the ATR. These experiment scope included a series of U-10Zr (10 weight %) 75% smear density alloy fuel pins in the following designs: conventional sodium bonded solid pin controls; sodium free annular pins; sodium bonded solid pins with Zr liners inside of the cladding; and sodium bonded solid pins with an additional 3-4% alloy additives of Pd, Sn, and Sb. The first irradiation of a FAST experiment was a critical success for the accelerated fuel qualification efforts being supported by multiple laboratories and industry partners.

b. Murdock, Capriotti, and Curnutt completed the planning, analysis, and execution of the first shipment of irradiated FAST pins into ATR. As-run power histories suggest burnups of FAST pins ranging from 2 %FIMA (fissions of initial metal atoms) to nearly 7 %FIMA. The range of burnup is a product of different parameters including linear heat generation rate and configuration of the experiment. These pins will provide a first look into the actual viability of FAST testing and the impacts of scaled irradiation effects. The remaining experiments will continue irradiation in ATR and be characterized accordingly.

c. Beausoleil and Capriotti led a phenomena identification and ranking table (PIRT) effort between INL and Oak Ridge National Laboratory for U-Pu-Zr fuel in SFRs and the Versatile Test Reactor (VTR). The PIRT results are important for being able to plan future experiments and to guide post-irradiation examination (PIE) efforts from FAST. While the first phase of FAST does not include the U-Pu-Zr ternary fuels, ternary fuels are a high priority consideration for the second phase of FAST testing and many of the phenomena associated with the ternary fuels are highly relevant to the U-Zr binary fuels currently being tested. This is documented in an article published in Nuclear Technology (Beausoleil et al., 2021).

Figure 3. X-ray radiography of the assembled FAST capsules to be inserted into the small I position within ATR. As shown, some of the rodlets became slightly dislodged from the holding clip and are tilted or off centered. Thermal analysis was performed on these eccentricities and validated that there was no risk to the experiment objectives. This is due to the combination of the large sodium bond to balance out the temperature and the relatively far distance between the cladding and the gas gap.

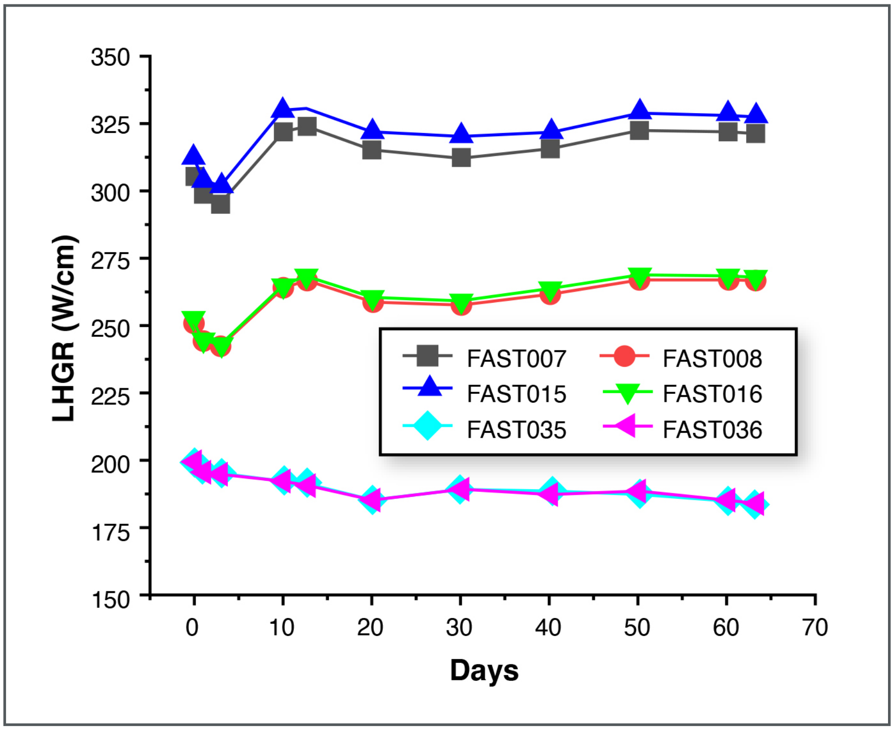


Figure 4. Linear heat generation rate (LHGR) of select FAST rodlets from the as-run neutronics analysis. LHGRs remained relatively constant throughout the irradiation cycle which is indicative of a relatively constant irradiation temperature (neglecting degradation in thermal transport properties as burnup increases). LHGRs were matched within 10% of the target LHGR.

Transient Testing of SFR Fuel Designs

Principal Investigator: Colby Jensen

Team Members/ Collaborators: Robert Armstrong, Cole Blakely, Devin Imholte, Randall Fielding, Nicolas Woolstenhulme, Dan Chapman, Andrew Chipman, John Bess, Austin Fleming, Nathan Gardner, Sterling Morrill, Trevor Smuin, Doug Dempsey, Jason Schulthess, Fabiola Cappia, Luca Capriotti, and Daniel Wachs

The ARES joint project with INL and JAEA has completed detailed final design for three advanced reactor fuel experiments and is ready to irradiate the first capsule to enable improved performance of advanced reactors.

Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) is a joint project between the U.S. Idaho National Laboratory (INL) and the Japanese Atomic Energy Agency (JAEA) to investigate the transient fuel performance of irradiated advanced metallic and mixed oxide (MOX) fuel designs from Experimental Breeder Reactor (EBR-II) experiment programs. Transient fuel performance of fast reactor fuels has been well-established internationally. The continued optimization of fuel designs and associated operational limits to improve performance and economics demands the continued experimental evaluation of these behaviors. These goals require the development of improved fuel performance behavioral models implemented in advanced modeling and simulation tools, which in turn require modern experiment complements with new data streams. The ARES project relies on the Transient Reactor Test (TREAT) facility and the development of associated in-pile testing infrastructure to provide sodium fast reactor (SFR) boundary conditions to accomplish its goals.

Project Description:

The ARES project is investigating the transient performance of advanced designs of irradiated metallic and MOX fuels as a collaboration between INL and JAEA to support improved performance of advanced reactors. The objectives are to investigate fuel failure modes in high burnup metallic and MOX fuels while developing and validating transient fuel performance models and establishing testing infrastructure at the TREAT facility. The ARES project is comprised of five primary components in this first phase. Figure 1 presents the ARES project timeline.

1. Design of an in-pile SFR testing capsule and associated hot cell infrastructure to remotely load irradiated fuel specimens using the best modeling and simulation capabilities.

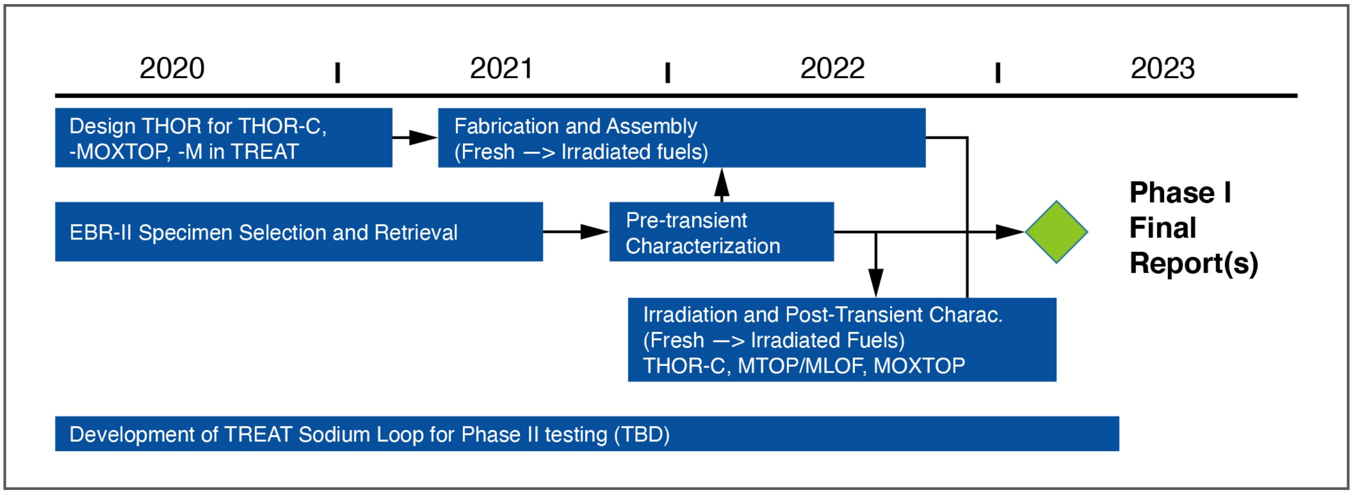
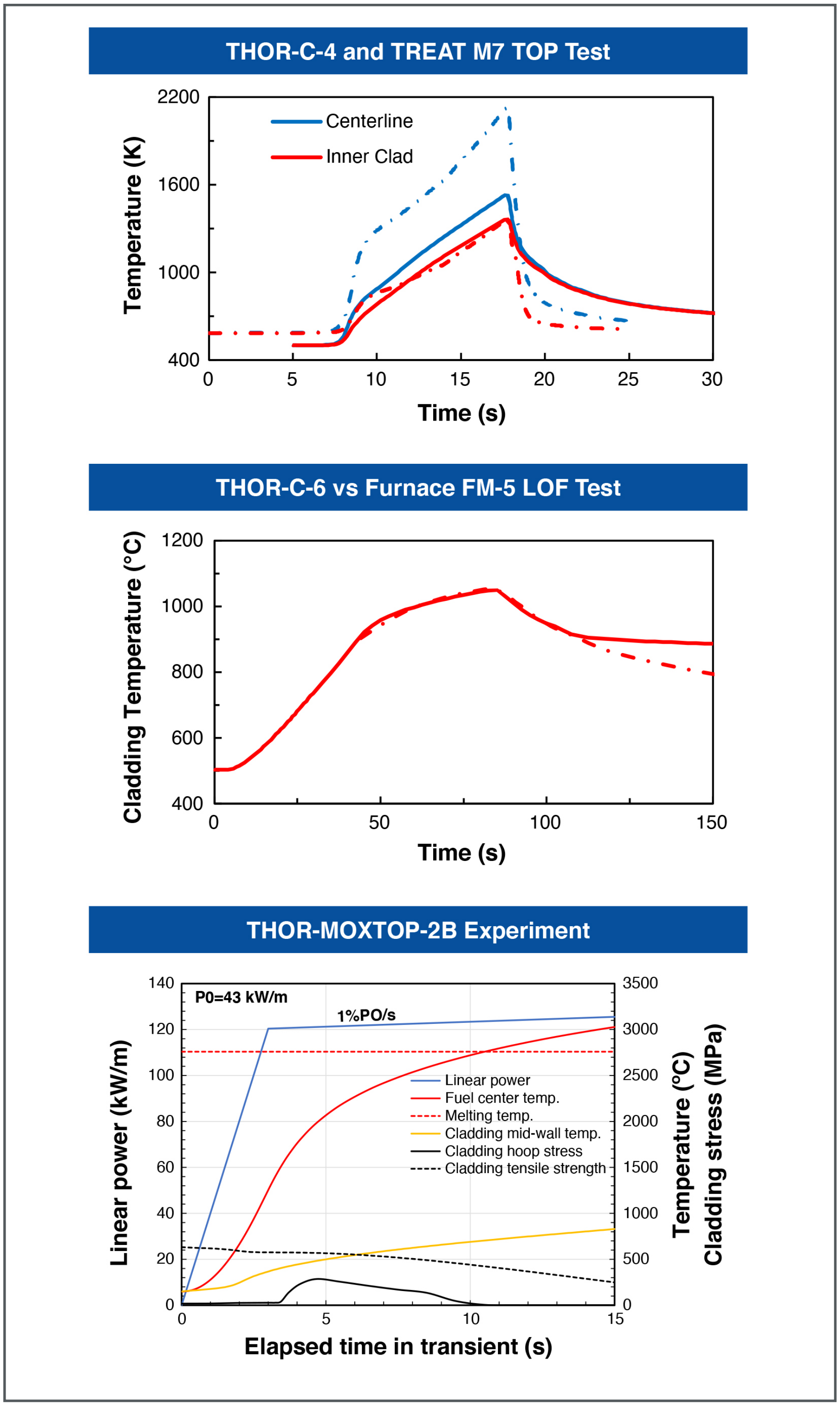


Figure 1. Overview of current ARES project schedule

2. Fresh fuel commissioning tests to qualify the TREAT testing platform for providing required nuclear heating and thermomechanical boundary conditions and measuring the associated fuel response.
3. Four integral experiments will be performed in TREAT including two irradiated metallic fuel specimens (called M-TOP/M-LOF series) and two MOX fuel specimens (called MOXTOP series).
4. Pre- and Post-Transient Examinations will be performed on the EBR-II-irradiated metallic and MOX fuel test specimens.
5. Data synthesis and detailed experimental evaluation will be performed using BISON and JAEA fuel performance codes.

The outcomes of these experiments will directly serve needs of the U.S. and Japanese SFR research and commercial communities. The project is establishing crucial infrastructure at TREAT and associated hot cells to address existing data gaps and extend the existing database while paving the way to future experiments. The Temperature Heat Sink Overpower Response (THOR) test device is already planned to support five different test series and four different customers with additional consideration by industry users.

Figure 2. Select examples of predicted results for final ARES experiment transients, each titled accordingly



Accomplishments:

After its first full year of execution (halfway mark), the ARES project has:

- Completed final design of
  - the THOR Commissioning (THOR-C experiment) series to commission a new test platform in TREAT,
  - the Mixed Oxide Transient OverPower (MOXTOP) experiment series in close collaboration with JAEA to design and select test parameters,
  - and (very near completion) for the THOR Metallic (THOR-M) experiments.
- Fabricated, assembled, and loaded the first THOR-C capsule, ready for irradiation in TREAT planned within days of fiscal year end
- Developed methods for and completed neutronic characterization using modern tools of the MOX test specimens irradiated in EBR-II
- and has been transferred from the storage facility to INL hot cells where preliminary visual exams indicate they are in good condition.

Each of these accomplishments represents a significant accomplishment by themselves but also towards completion of the ARES project goals. The final design of multiple irradiation experiments required significant effort from a large multidisciplined design team. As an important example, the THOR experiments have required new understanding of TREAT capability to tailor reactor power histories to represent relevant fuel conditions. Over several months, test objectives were translated into achievable design targets working closely with JAEA collaborators. Both JAEA and INL were required to simulate these conditions using their own modeling tools to validate design objectives will be met and concur on the final targets. A similar process was followed for metallic fuel experiments at INL, working to balance the constraints of achieving design targeted conditions but also to develop and ensure compliance with the corresponding safety case for shaped transients. Figure 2 presents several examples of final predicted experiment temperatures. The latter exercise proved challenging yet very fruitful towards interpreting TREAT Safety Analysis Report (SAR) language that is a first for TREAT since restart for complex transients and energy-demanding experiments. In all, eleven separate experiments were designed between the three ARES project experiments.

The complete assembly of the first THOR capsule (and using sodium) represents a first of a kind for TREAT experiments and facilities at the Materials & Fuels Complex (MFC). The scale and logistics of the test device required careful cooperation in new supporting facilities and has aided in developing understanding to address future challenges with assembly in the Hot Fuel Examination Facility (HFEF) facility for irradiated specimens. More details of the THOR capsule design including schematics and pictures are found in another article in this report.

Finally, working with the irreplaceable stockpile of legacy EBR-II irradiated materials has required significant effort to recover test specimens, pertinent data and records, and develop an approach to obtain detailed neutronic characterization of the as-irradiated state of the fuel pins. Working with Idaho State University and their developed EBR-II benchmark model, experiment irradiation conditions were generated and passed to a depletion code to create best-estimate predictions for material isotopic inventories. These results were crucial to calculating energy coupling within the TREAT reactor to ensure energy targets are achieved and will be verified via experimental methods in the next year on the MOX pins. Figure 3 shows an example of calculated axial distribution of U235 in a MOX pin, a photo showing the recovered MOX pin, and a historical neutron radiograph of the MOX pins post-irradiation in EBR-II.

With many successes, significant challenges were also encountered such as developing new understanding of the TREAT safety case for complex and energetic power shapes in TREAT, which required more analysis iterations than were anticipated. The logistics of the capsule assembly required intense experimental validation of solutions to maintain sodium within the capsule where required. Transferring legacy irradiated fast reactor fuels between facilities required significant logistical coordination and firsts in facilities not accustomed to handling MOX. Significant hurdles remain to complete the ARES work requiring unique logistics such as transferring irradiated materials between facilities while ensuring facility material inventory limits remain within limits, accomplishing extensive characterization of specimens before and after the TREAT experiments, and development of THOR handling capabilities in HFEF to load irradiated test specimens.

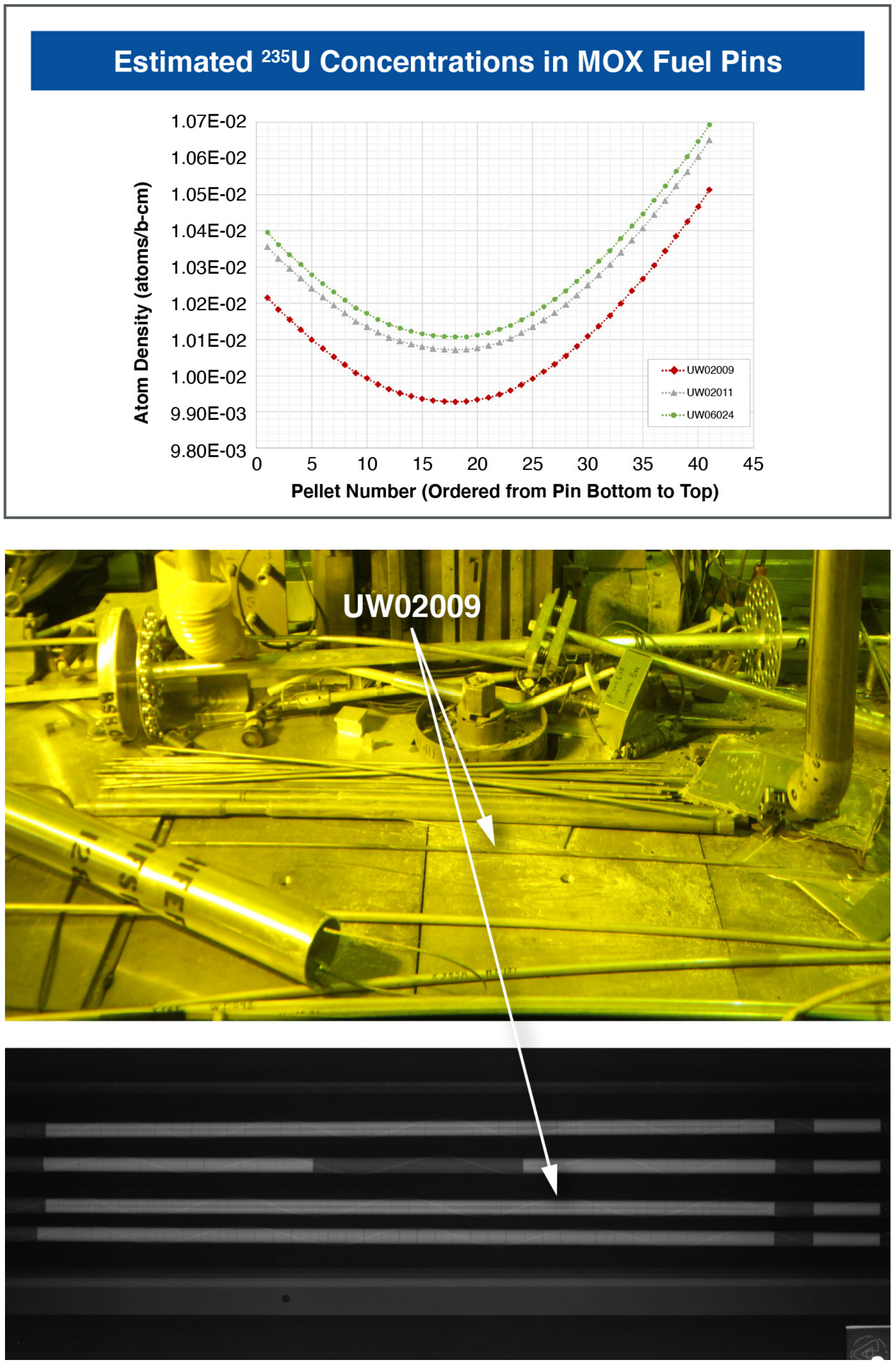


Figure 3. THOR-MOXTOP experiment specimens: (Top) predicted axial U235 concentration in pins, (Middle) Recovered neutron radiograph showing pin UW02009, (Bottom) photo of test pin UW02009 in hot cell during retrieval from storage facility.

Status and Future for Integral Scale Fast Spectrum Fuel Pin Irradiations

Principal Investigator: Nick Woolstenhulme
Team Members/ Collaborators: Bryon Curnutt, Joseph Nielsen, Kevan Weaver, Colby Jensen, and Austen Fradeneck

The ARCTIC experiment capability can bridge research gaps, attract new experimenters, and cultivate a rich area for development of fast spectrum fuels and core materials.

Development of advanced fuels and materials for fast spectrum nuclear reactors is an immense challenge with tremendous potential in maximizing the value of advanced nuclear plants. Historical developments and modern materials show promise for overcoming these challenges, but the lack of fast neutron test capabilities in the western world presently hamper meaningful progress. For years the Advanced Fuels Campaign (AFC) has led the charge by using an existing thermal-spectrum reactor, the Advanced Test Reactor (ATR), to irradiate fuel technologies intended for use in fast spectrum reactors. The so-called "AFC-series" irradiations used cadmium lined irradiation baskets to filter thermal neutrons and achieve a semi-representative neutron spectrum to help flatten fuel slug radial power profiles to represent fast spectrum environments. This method was successful but acknowledged for its inadequacy in representing fast neutron damage in cladding, supporting full scale specimens, and generally in achieving the irradiation timescales needed to support accelerated innovation cycles. A new concept for overcoming these challenges was generated under the AFC program in early 2020, but budget reductions caused cessation of new AFC irradiation designs. This concept was later proposed, awarded, and investigated under the Idaho National Laboratory (INL) Laboratory Directed Research and Development (LDRD) program. Hence this scope is technically an accomplishment funded outside of AFC in 2021 but is summarized here because the concept was originally generated under AFC funding, was performed by personnel who are largely committed to the campaign and has potential future relevance to the program.

Project Description:

This project used modern modeling to investigate a new method for fast neutron irradiations in ATR. This new method built upon past ideas, where fast flux is increased by surrounding the specimens with fissionable "booster fuel," but diverges from historical

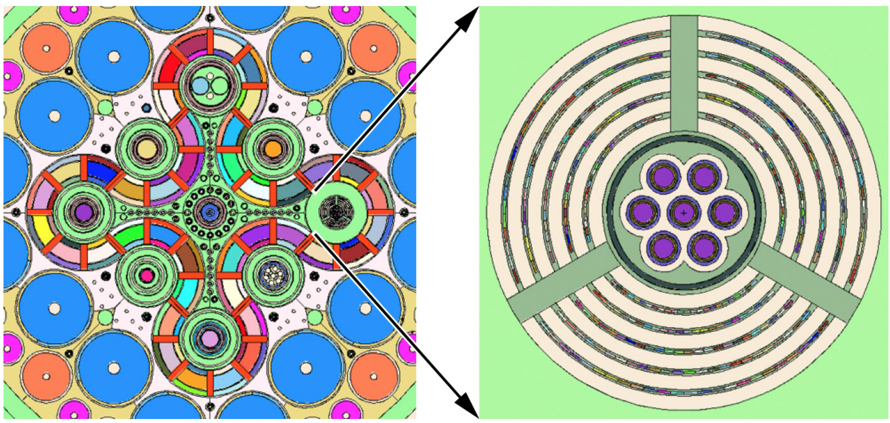


Figure 1. MCNP rendition of the ATR core and ARCTIC concept in the NE flux trap

approaches by using an "off-the-shelf" fuel assembly as the booster fuel. This concept has been termed the Advanced fast Reactor Concept in Thermal-spectrum Irradiation Capability (ARCTIC). These studies used modern modeling to investigate the hypothesis that ARCTIC would result in a worthwhile fast flux increase whose deployment in ATR would be timely in maturing fast spectrum research while the community awaits availability of a true fast spectrum test reactor.

Accomplishments:

This project included brainstorming, categorization, and prioritization of several concepts for specimen types and flux/spectra modification schemes. These options were then investigated by predictive models. A detailed Monte Carlo N-Participle (MCNP) model was constructed (Figure 1) for the favored concept where an existing fuel assembly design in production for the Belgium Reactor -2 (BR2) was placed in an ATR flux trap to boost fast flux to a specimen holder with various thermal neutron filtering materials (cadmium and europium options were investigated). A typical sodium fast reactor type seven-pin hexagonal arrangement of U-10Zr metallic fuel was used as the specimen bundle.

These predictions showed that the total power output and surface heat flux from the BR2 booster assembly would be well within both ATR thermal hydraulics safety limits and the qualified performance envelope for the booster fuel. Most importantly, predictions of various design candidates showed that the fuel pins could achieve fission rates typical of a sodium fast reactor in neutron flux spectra with fast (> 100 keV) to thermal (< 0.625 eV) ratios ranging from 50 – 150. This would be a marked improvement compared to the same ratio of ~20 seen in the current state-of-the-art method where only thermal flux filtering is used. This finding is important and shows that ARCTIC will increase cladding atom displacement damage, reduce thermal neutron transmutation effects in cladding, and flatten radial power peaking in the fuel. All these effects work together to yield an irradiation method far more representative of true fast reactors. The europium oxide and cadmium filter options were both shown to effective at hardening the spectrum, with europium performing slightly better, as shown in Figure 2.

Various thermal finite element calculations were also performed near the completion of this project to determine the best method for managing fuel pin temperatures. Several concepts were investigated and two emerged as viable candidates in achieving desired conditions. The first method used a seven-hole molybdenum insert within a capsule with thin liquid sodium thermal bonds between cladding and capsule surface gaps. This method minimized moderator and yielded the highest fast-to-thermal neutron flux ratios (~100-150) but showed modest temperature gradients transversally across the fuel pins which could affect fuel constituent redistribution in and non-prototypic way. This concept was hence determined to be the most favorable for tests designed to investigate cladding performance in environments with the

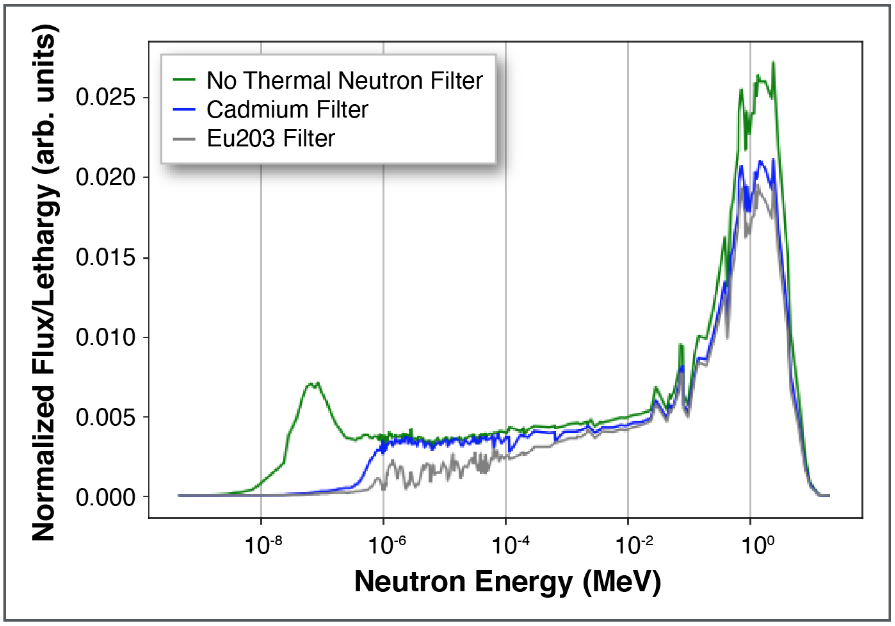


Figure 2. Neutron flux spectra comparison within BR2 with and without thermal neutron filtering

highest achievable fast neutron flux. The second concept modeled each of the seven pins independently encapsulated and utilized cladding to capsule gas gaps to increase critical fuel temperatures relative to ATR coolant temperature. This approach appeared to be well suited for achieving the desired temperatures and gradients for irradiations whose data objectives are primarily focused on fuel material performance but include more moderator within the BR2 element resulting in relatively lower fast-to-thermal ratios (~50-100). Continued refinement of detailed mechanical design and thermal hydraulic conditions was noted as primary area for maturation of the ARCTIC concept in any future development efforts.

Metallic Alloy Fuels for Transmutation

Principal Investigator: Randall Fielding & Luca Capriotti
Team Members/ Collaborators: Brian Newbold, and Hot Fuel Examination Facility and Fuel Manufacturing Facility Operators

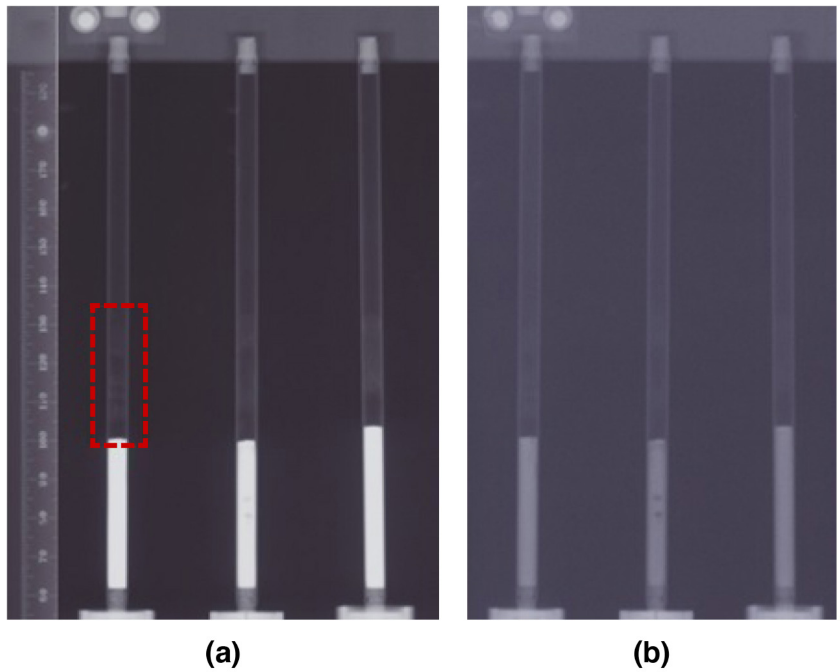


Figure 1. Neutron radiography of IRT1 rodlets, (a) thermal neutron radiography (Dy foil) and (b) epithermal (In foil). Each panel shows R1, R2, and R5 from left to right. Highlighted in the red square is the sodium plug on top of IRT1-R1.

In collaboration with the Joint Fuel Cycle Studies project post-irradiation examination (PIE) was performed on three rodlets from the irradiation test (IRT)-1 fuel irradiation test. The follow-on irradiation test, IRT-2, was fabricated using U-TRU material recovered through recycling of used light water reactor fuel.

Project Description:

This work was one of the main goals of the Joint Fuel Cycle Studies (JFCS) project; to fabricate, irradiate and perform PIE on metallic, transuranic (TRU) bearing fuels, using feedstock recycled from used oxide fuels. The IRT-1 test was fabricated using U-TRU material recovered from used mixed oxide (MOX) fuels. An addition of various lanthanide elements was added to the fuel to simulate fission products that may be carried over during recycling high burnup fuel. Three test rodlets underwent PIE showing acceptable irradiation behavior. Following irradiation of the IRT-1 test, a follow-on test, IRT-2 was fabricated. The IRT-2 test used U-TRU feed material recovered from used light water reactor fuel. The fuel alloys did not include the addition of surrogate fission products, as was done for the IRT-1 test, to determine the irradiation behavior of recycled used

The IRT tests have fabricated, irradiated, and performed post irradiation examination on fuel sources from recycled spent oxide fuel.

oxide fuels with the as recycled amount of carry over contaminates. This work supports reduction of potential waste created by current and future light water reactors making the nuclear fuel cycle safer and more economically competitive. The successful irradiation of the IRT-1 test has shown initial feasibility of a recycled oxide fuel cycle. This work can be continued by irradiation of the IRT-2 test and by continuing irradiation of remaining IRT-1 samples not used for destructive examination.

Accomplishments:

Luca Capriotti performed the PIE on IRT1-C1, C2, and C5. Non-destructive PIE results consist of visual exams, neutron radiography, dimensional inspection, and gamma-ray spectrometry. Destructive PIE results consist of fission gas analysis, optical microscopy of the fuel, and electron microscopy of a fuel and cladding section.

Visual examination showed no obvious defects in the rodlets, although, some discoloration was noticed in the fuel region. Neutron radiography images were taken of the capsules and rodlets, shown in Figure 1. Overall, the rodlets appear to have behaved well, in line with historical expectations. The axial growth for all the three IRT1 rodlets was higher compared to the literature for U-20Pu-10Zr EBR-II pins at this burnup; the reason for this is not known. Gamma ray spectrometry of the rodlets showed behavior of gamma emitting fission products which followed trends historically observed for a solid, sodium bonded fuel.

The fission gas release results for IRT-R2 of ~70±10%, is in good agreement with historical norms for 75% smeared density U Pu Zr (or U-10Zr) fuel behavior beyond ~3 atom % burnup. However, for IRT-R1 and -R5 the fission gas release was lower, around 32-36%. This may be related to a slightly lower calculated burnup of the two rodlets, the possibility that the porosity is still not fully interconnected, and the smear density of these rodlets was around 79%. Fuel metallography for IRT-R1 and -R5 did not show a microstructure developed entirely as it would be expected for metallic fuel at the same burnup and irradiation temperature, seen in Figure 2. Metallography of IRT-R2 presented a microstructure where at least 3 regions could be seen,

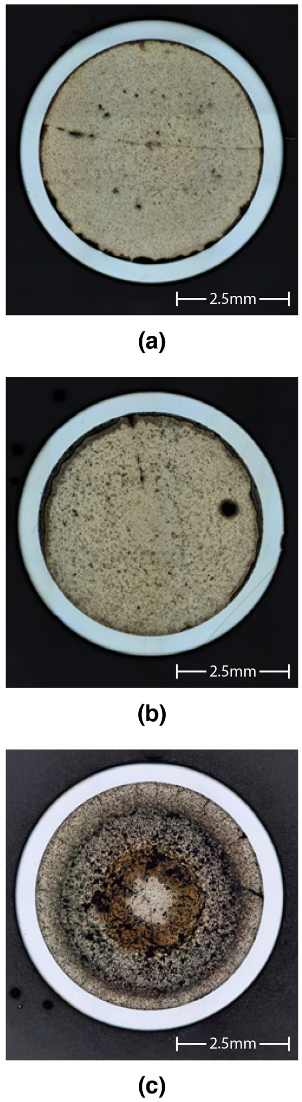


Figure 2. Optical microscopy of the fuel a) IRT1-R1 b) IRT1-R2, and c) IRT-R3

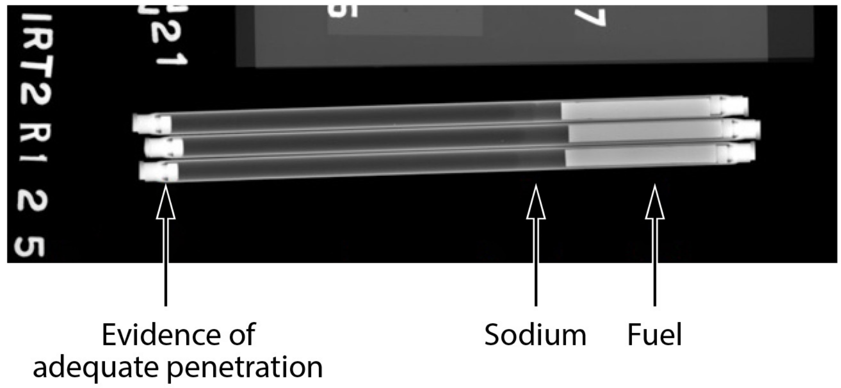


Figure 3. Overall radiography of the IRT-2 irradiation test

as seen in Figure 2. Scanning electron microscopy on the IRT1-R2 cladding sample did not show a wastage layer (layer affected by intrusion of fuel components or fission products / rare earth elements (RE) or corrosion. Some Cr enriched grain boundaries and Nb precipitates were observed. Microscopy of the fuel was performed on the center, intermediate and periphery regions of the fuel radius. Results showed several pore formations and elemental re-distribution. Based on the non-destructive examination results, the IRT-1 capsule/rodlets-1, 2, 5 behaved similar to Experimental Breeder Reactor (EBR)-II experiments. Although destructive examinations did reveal some differences were seen which will require further analysis but may have been due to the low burnup or sample location.

Randall Fielding and Brian Newbold oversaw and managed assembly of the IRT-2 test. The IRT-2 fuel irradiation test is made up of 3 rodlets to replace the three IRT-1 rodlets removed for PIE. Unlike IRT-1, no surrogate fission products were added to IRT-2, rather, the recycled U-TRU was used as recovered. The design called for a Cr/CrN coated FC92, however, due to fabrication issues this cladding was replaced by another KAERI provided cladding with a Cr coating. The two remaining rodlets used HT9 cladding, one included a 12 μm Zr foil placed in between the cladding and fuel. The IRT-2 fuel slugs were cast using the arc casting method in the AFCI glovebox located in the Fuel Manufacturing Facility (FMF). Following casting the rodlets were loaded with sodium and fuel and were welded closed. Following welding, the fuel slugs were settled and bonded, and final rodlet inspections performed. The rodlets are awaiting a confirmation of irradiation power and cycle before re-encapsulate and to set the final fill gas mixture to provide the target PICT. Figure 3 shows a radiograph of the completed rodlets.

CAPABILITY DEVELOPMENT

Expansion of LWR Loop Testing Infrastructure

Principal Investigator: Nate Oldham
Team Members/ Collaborators: Kendell Horman, Lex Strain, Kelly Ellis, Thomas Maddock, Justin Johnson, Bubba Ricker, and Josh Tonks

I-Loop 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

Expand access and capability to the Advanced Test Reactor (ATR) Medium-I position with a flowing water loop, allowing testing of fuels and materials in prototypic light water reactor (LWR) conditions.

Project Description:

The closure of the Halden Boiling Water Reactor (HBWR) and growing demand for extending LWR fuel performance limits, has created urgency to find a testing solution to support near-term fuel testing needs. This need comes at a time when industry, via Electric Power Research Institute (EPRI), and the Department of Energy (DOE), via the Accident Tolerant Fuels (ATF) Program, are pushing for performance and capability increases for new fuels. The I-Loop project aims to fill the gap created by HBWR's closure and provide a similar capability at ATR, by installing a flowing water loop in one Medium-I position and providing access to the other Medium-I positions for future experiments. In order to provide better access to the I-positions of ATR, a new Top Head Closure Plate (Mark II) has been installed, expanding the capabilities of the ATR for continued fuel and materials testing. To support the flowing water loop, the refurbishment of the 1A cubicle will be required including new pumps, piping, heat exchangers, and other equipment. In order to provide the neutronics to approximate prototypic LWR heating rates, nuclear grade Zr-2.5Nb will be used for the experiment tubes. The improvements will enable advanced fuel qualification to continue at the Idaho National Laboratory (INL) and support the industry and DOE desire for these capabilities in the United States, to support the continued safe use of existing LWR reactors and provide for future fuel testing for next generation reactors.

Accomplishments:

ATR's original Top Head Closure Plate (Figure 1) was successfully replaced with the Mark II (Figure 2) design in July 2021 during the reactors Core Internal Changeout (CIC) outage. This is a critical accomplishment because a CIC is the only time to perform such a large effort and because the next CIC outage is estimated in the year 2038. The replacement of the Top Head Closure Plate provided the opportunity for the replacement of the o-ring seals which have been in-service since the first startup in 1967. The Mark II design possesses several new features that allow for testing and maintenance of the reactor to ensure ATR operates safely and reliable well into the future. The main design feature of the Mark II is the additional eight I-penetrations on the periphery of the nine in-pile

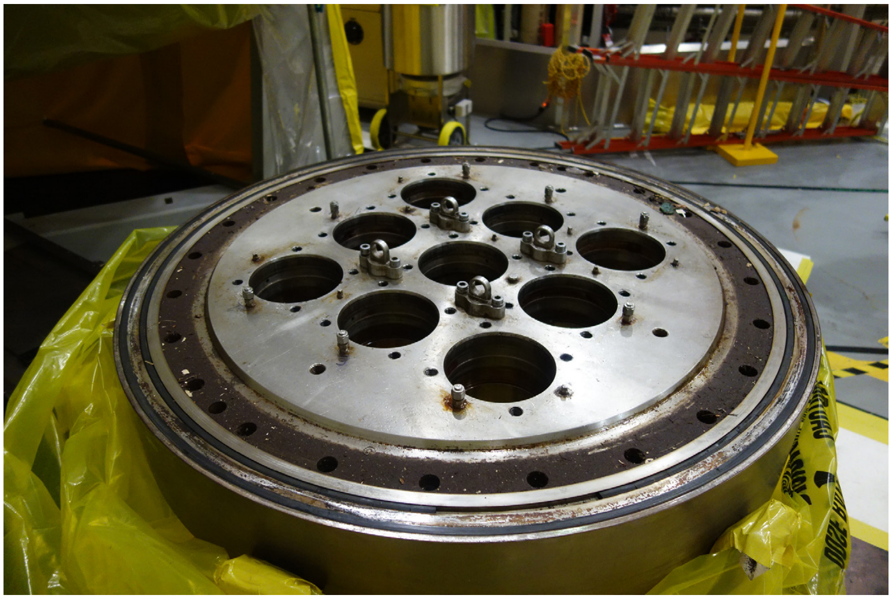


Figure 1. Original ATR top head closure plate

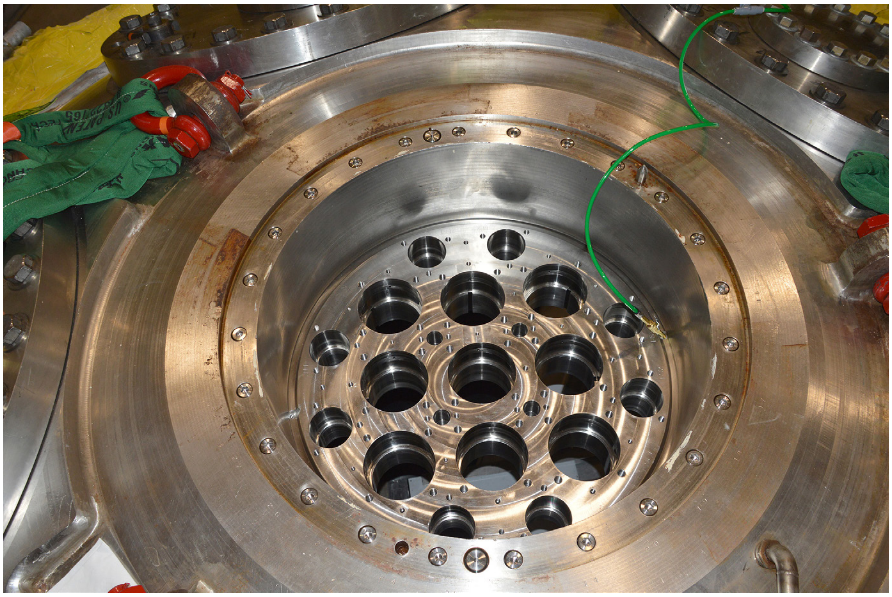


Figure 2. Top head closure plate – mark II installed on ATR vessel (top view)

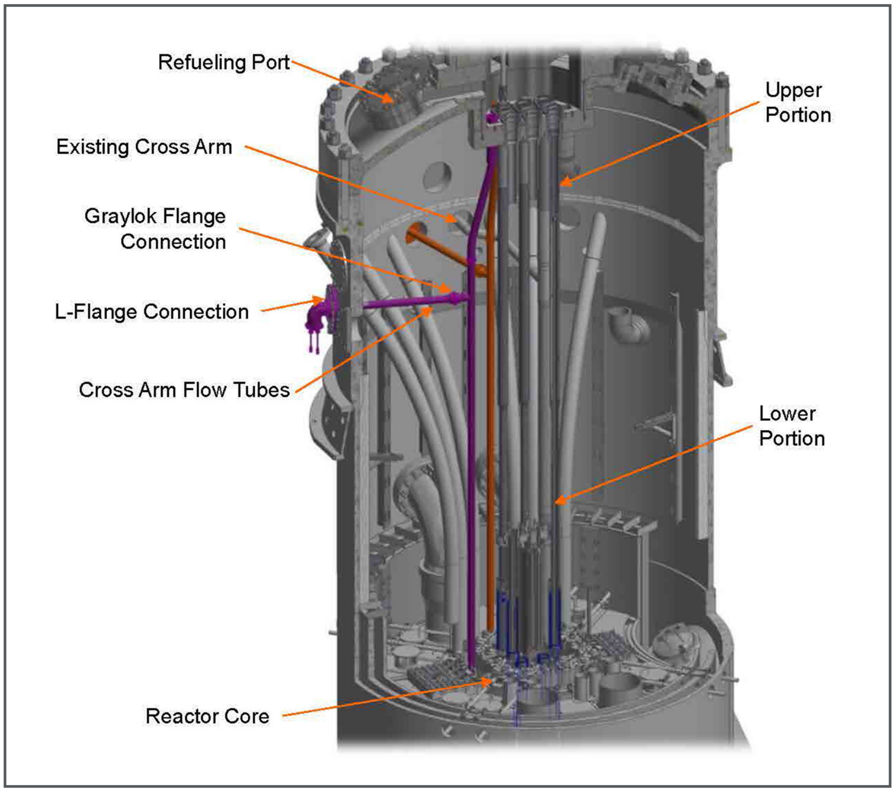


Figure 3. I-Loop tubes in the ATR

Figure 4. Mark II installation team on ATR main floor

tube penetrations. The I-penetrations allow for all types of experiment vehicles to be inserted through the reactor top head and open the possibility for facilities such as I-Loop to be located in the reactor's reflector Medium-I positions.

The I-Loop project also completed a conceptual design of the in-vessel hardware known as an I-Loop Tube (Figure 3). The I-Loop Tube serves as the experiment facility that houses the experiment test rig and specimens. The I-Loop Tube design established a that concept is viable the following critical capabilities:

- LWR fuel testing in both pressurized water reactor (PWR) and boiling water reactor (BWR) environments
- Flux similar to Halden's central core flux (~5.2 kW/ft)
- Loop coolant pressure of 15.2 MPa (2,200 psi) and 320°C (608°F)
- Insertion/removal of test trains using existing casks
- I-Loop Tube installation during an extended maintenance outage
- Test train of a 2x2 array of LWR fuel rods
- The test needs subjected to power manipulations that are decoupled from the ATR core

The I-Loop project is in a great position to continue with design refinement and prototype activities in the next fiscal year. The following years will consist of equipment procurement, installation, and commissioning tests.

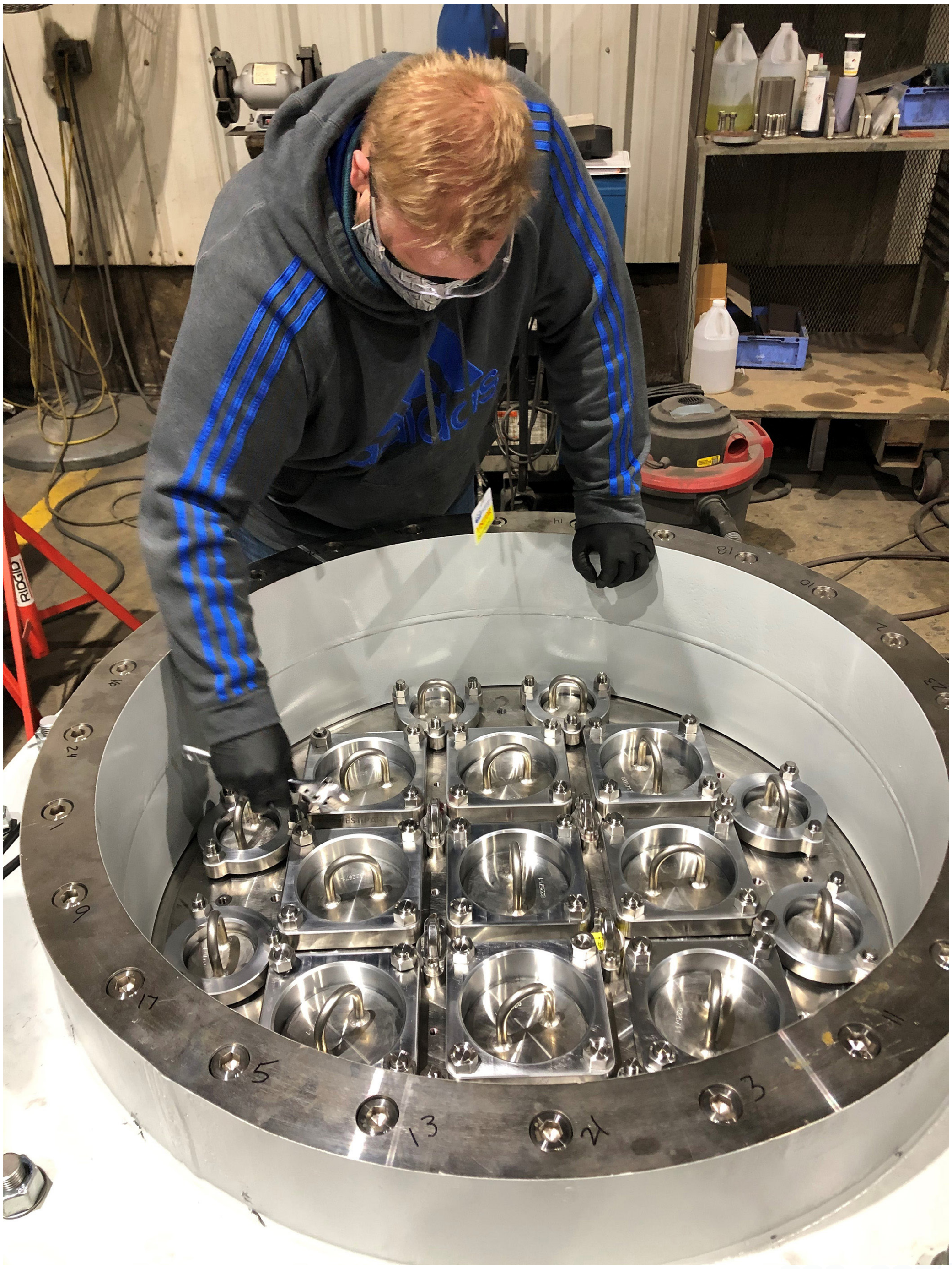


Figure 5. Mark II hydro testing assembly (Nate Oldham)

Fuel Rod Refabrication Capability

Principal Investigators: Jason Schulthess
Team Members/ Collaborators: Spencer Parker, Evans Chambers, Jordan Argyle, Kim Davies, Justin Yarrington, Clayton Turner, Cad Christensen, James Chandler, John Stanek, Mark Cole, and Mike Bybee

New refabrication and experiment assembly capability to support follow on irradiation and out of pile testing of previously irradiated fuel.

Irradiated fuel rod refabrication is a crucial enabling capability that bridges between base reactor irradiation and subsequent experimentation and/or re-irradiation. It is key to performing meaningful research and development (R&D) on fuels with any level of burnup, especially at the Transient Reactor Test (TREAT) Facility and opens the door to R&D for materials tested in commercial nuclear power plants. Refabrication allows access to fuel at any point in its lifetime, allowing opportunity to apply instrumentation and perform experiments to measure performance under a variety of specified conditions. Secondly, the ability to repackage previously irradiated fuel into experiment vehicles is an enabling capability that ensures experiments are instrumented and contain desired boundary conditions for experiment objectives. Measurements on high burnup fuels are impractical, if not impossible, without refabrication and experiment preparation capability.

Project Description:

A fuel refabrication system and process has been designed, constructed, installed, and demonstrated in the Hot Fuel Examination Facility (HFEF) hot cell to enable basic fuel rod refabrication. The basic refabrication capability includes segmenting full length fuel rods, defueling the ends to make space for new end caps, attaching and welding in place new end caps, and pressurizing and seal welding the new rods, followed by leak check inspection, which are then available for follow on irradiation or out of pile testing.

The basic refabrication welding systems have been designed to provide maximum flexibility and reliability to perform the three necessary welds. Two circumferential welds to attach the end caps, and one seal weld to ensure hermeticity of the rodlet. All three welds are performed using commercially established micro-Tungsten Inert Gas (TIG) welding methodology. The circumferential welds are performed using a custom designed welding lathe, which holds the welding torch in a single position and rotates the rodlet, which provides the most consistent welding on these materials. The lathe system further incorporates an automated variable voltage controller, which supports consistent welds if the rods are found to be non-circular (e.g., elliptical). The seal welding system also utilizes micro-TIG methodology and encapsulates the rodlet in a pressurized chamber capable of pressurizing the rodlet from 500 psi to 2250 psi to allow for experiments to be conducted with variable rodlet internal pressures.

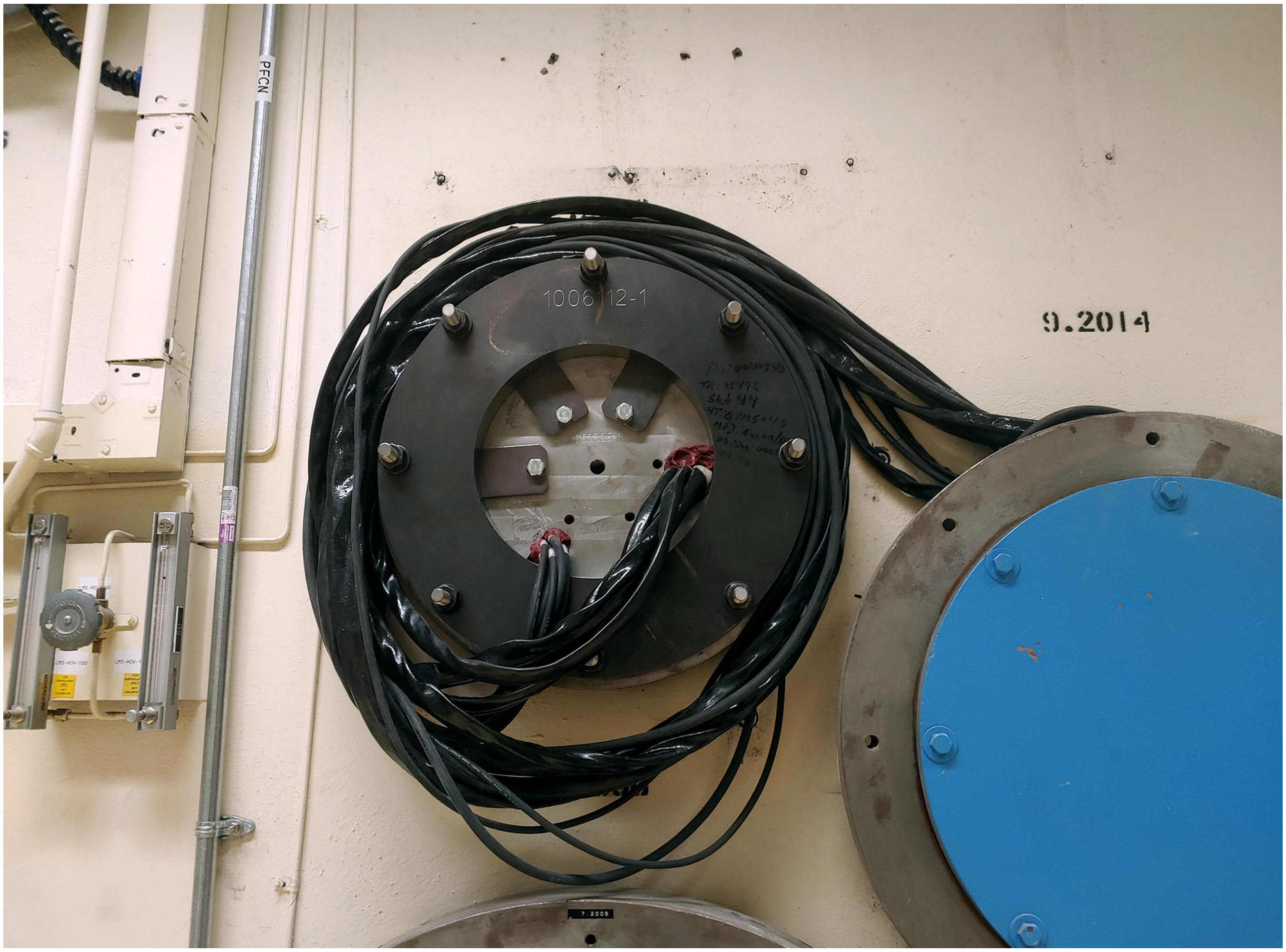


Figure 1. New HFEF feedthrough installed to support refabrication/welding equipment

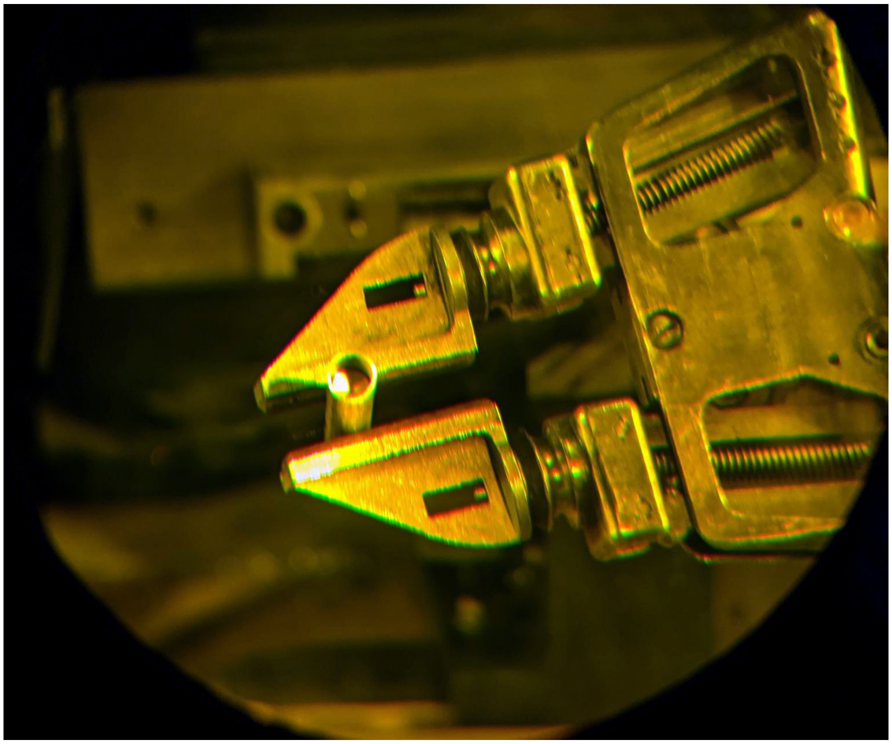


Figure 2. Surrogate fuel rodlet with ends successfully mechanically defueled using newly installed in-cell equipment

Additionally, the capability to assemble irradiation experiments with previously irradiation fuel was developed and demonstrated with the successful assembly and shipment of the ATF-R experiment in the Minimal Activation Retrievable Capsule Holder–Static Environment Rodlet Transient Test Apparatus (MARCH-SERTTA) experiment vehicle.

Accomplishments:

During FY2021, final designs of equipment and methodologies were completed enabling fabrication to begin and complete. Following fabrication all equipment was thoroughly tested out of cell and applicable independent facility reviews completed. Equipment installations into the hot-cell were completed and in-cell checkout/demonstration performed.

For base facility infrastructure, a new control system was fabricated and installed, including installing new feedthrough cables in the hotcell wall to support experiment assembly and instrumentation checkout. An entirely new feedthrough was fabricated and installed to support the power and control requirements for the welding systems. The HFEF-15 cask was returned to service following the completion of maintenance, upgrades, and readiness reviews. New electrical power and gas supply were routed to support the experiment assembly, welding systems, and the upgrades to the in-cell mill.

Defueling development was twofold and included both mechanical and wet/chemical defueling processes. The existing HFEF in-cell 3-axis mill was successfully modified to incorporate a 4th-axis rotation stage and vacuum collection system. Following successful installation, mechanical defueling and preparation of cladding tube ends for welding was demonstrated on the in-cell system, including removal of the cladding oxide layer.

Wet/chemical defueling utilizes an acid bath to dissolve the fuel and remove it from the cladding. This process uses 6M HNO_3 heated to near boiling (~150° C). Once at temperature, the fuel sample is introduced to the system, and the acid etches the fuel out of the cladding. Once dissolution is completed, the acid bath will be neutralized with MgO to eliminate the hazardous nitrates and then solidified using a WIPP disposal criteria approved sorbent. A demonstration of this procedure was performed previously at the Analytical Laboratory using a 5 mm length tubular segment of an irradiated fuel rod from the H.B. Robinson plant. For fiscal year (FY)21, efforts focused on completing scale up activities and facility impact and safety analysis to support performing chemical defueling on larger samples in HFEF. Equipment has been procured and is pending installation and demonstration.

Two micro-TIG welding systems have been designed, fabricated, and installed to perform the circumferential welds and seal welds necessary to attach new endcaps and set the desired rod internal pressure prior to follow on experiments. These welding systems utilize the new hot cell penetration feed through that was installed to supply the utilities and control system for the welding operations.

Assembly of the first experiment with previously irradiated fuel was enabled by the restoration of the HFEF-15 cask to service and significant new handling equipment and control systems, in addition to operational and facility readiness. Demonstration of this new capability was completed when the ATF-R experiment was assembled into the MARCH-SERTTA experiment vehicle and successfully transferred to the TREAT facility from HFEF. The transient irradiation of this experiment was subsequently performed on July 29th 2021, marking the first experiment with previously irradiated fuel of the modern TREAT era.

Figure 3. ATF-R experiment assembly being filled with water during assembly process at HFEF

Figure 4. Successful weld demonstrated on oxidized cladding during out of cell testing of circumferential welder

In-Pile LOCA Testing at TREAT

Principal Investigator: Robert Armstrong
Team Members/ Collaborators: Colby Jensen, Charles Folsom, Nicolas Woolstenhulme, Devin Imholte, and Changhu Xing

The TREAT TWIST device is designed to be the future LWR fuel safety testing device with world-leading, unique capability to investigate high burnup fuel performance during LOCA (FFRD behaviors) supporting industry goals to achieve burnup extension.

In recent years, the U.S. nuclear energy industry has set goals for licensing fuel burnup beyond current limits of 62 GWd/t [1], targeting up to ~75 GWd/t. Industry has identified Loss of Coolant Accident (LOCA) performance of high burnup fuel, in particular fuel fragmentation relocation and dispersal (FFRD), as an issue requiring additional research and development (R&D) support. Additionally, the closure of the Halden research reactor has left no possibilities for in-pile LOCA testing. Working closely with industry and national laboratory researchers, the fuel life history and corresponding microstructure as well as the specific LOCA conditions of relevance have been a major focus for guiding design. In particular, the fuel conditions related to potential rod failures occur shortly after the blowdown event with a cladding heat up rate driven largely by relatively high stored energy in the fuel, a more dynamic condition than has ever been evaluated via integral experiments previously. Therefore, the Advanced Fuels Campaign (AFC) is developing in-pile LOCA testing capabilities in the Transient Reactor Test Facility (TREAT) to support the burnup extension goals and investigate high burnup fuel behaviors while establishing unique in-pile LOCA test capabilities.

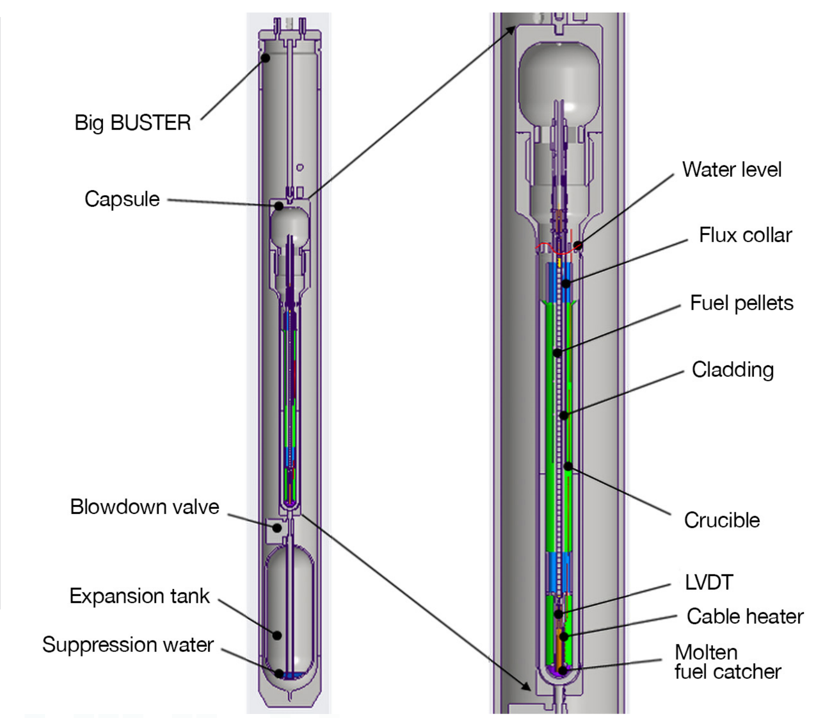


Figure 1. TWIST design rendering

Project Description

To support in-pile LOCA testing in TREAT, an experiment vehicle known as the Transient Water Irradiation System in TREAT (TWIST) is being developed. TWIST will be used as a consolidated water reactor testing system in the TREAT facility. The TWIST design is based on previous LOCA device concepts from recent years with the goal of better representing fuel conditions starting from normal operations fuel through the LOCA event. TWIST consists of a pressurized static water

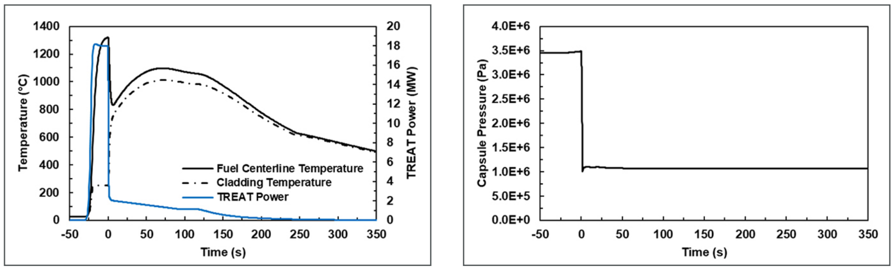


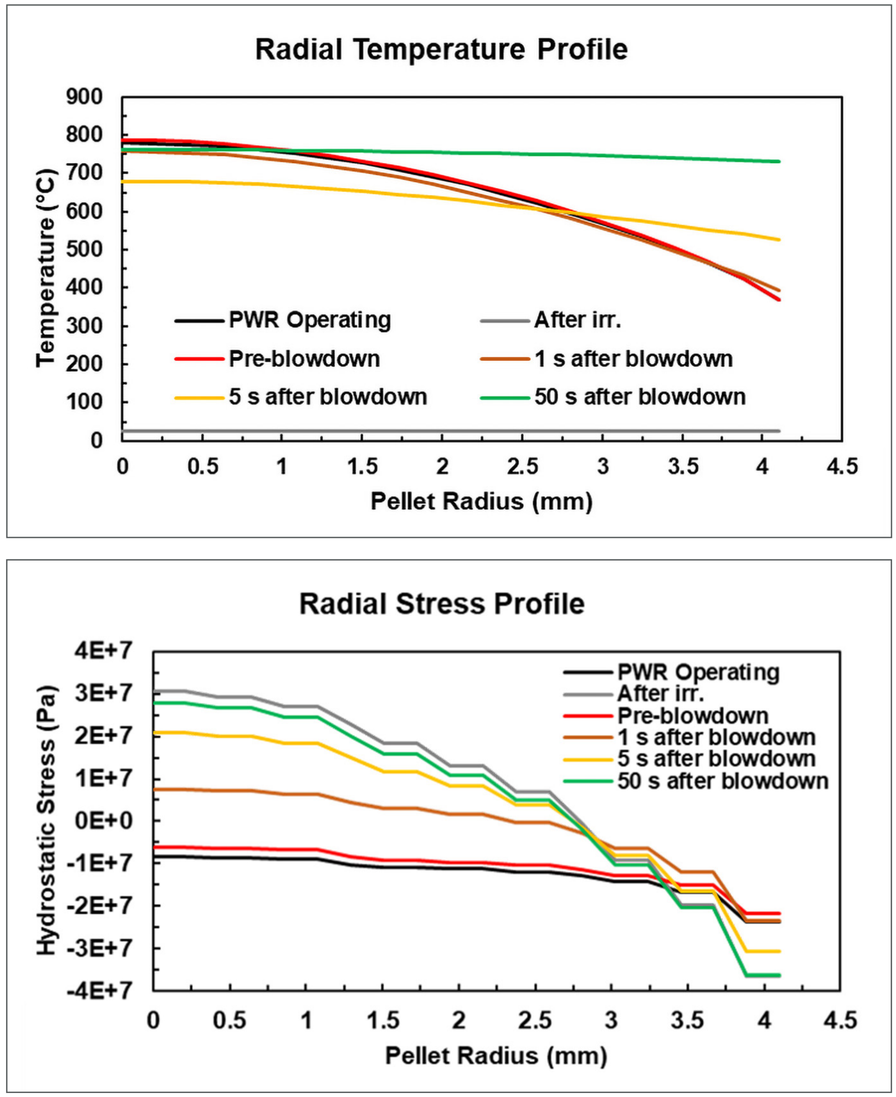
Figure 2. RELAP5-3D predictions of the fuel and cladding temperature, TREAT core power, and capsule pressure during TWIST LOCA transient. Blowdown at t = 0 seconds

capsule that can house up to a 50-pellet fuel rodlet connected to a lower pressure expansion tank via a remote triggered valve. Opening this valve triggers a rapid depressurization and draining of the water surrounding the fuel simulating the blowdown portion of a LOCA. A typical TWIST LOCA experiment consists of two segments. In the first, the fuel rod is brought up to quasi-steady-state conditions meant to mimic the operating conditions of a rod in a commercial light water reactor (LWR). This is done by ramping up TREAT power over a period of seconds, then holding the power constant for ~30 seconds resulting in a radial temperature profile throughout the fuel and linear heat rate that is consistent with commercial LWRs. Achieving a prototypic radial temperature profile is important as it allows for returning test fuel to its operating thermomechanical state, though the ~seconds timescale will not allow for full reconditioning of the fuel microstructure. The radial temperature profile also establishes representative stored energy conditions in the fuel. After the desired conditions are reached, the second segment is initiated by opening the valve to blowdown the water in the specimen-containing capsule into the expansion tank. At the same time the valve is opened, TREAT power is reduced to simulate decay heat in the fuel rod. Together, these two segments allow for TWIST to simulate a prototypic LWR LOCA in which cladding temperature rapidly increases up to ~100°C/s due to the redistribution of stored energy in the fuel, followed by further clad heat up from the simulated decay heat. The conditions have been shown to match well with predictions for limiting rods in LWRs.

Accomplishments

Leveraging modeling and simulation tools, the design of the TWIST experiment vehicle is nearing completion. RELAP5-3D modeling of TWIST has shown that it is capable of meeting the thermal-hydraulic objectives of simulating a LOCA from steady-state operating conditions. The ability to

Figure 3. Evolution of radial temperature and stress profiles through the TWIST LOCA transient. Predictions are generated using the fuel performance code BISON.



represent a fully prototypic thermal evolution in the fuel will be unique in the world and will be crucial to investigate questions related to FFRD behavior of high burnup fuels. The fuel performance code BISON has been used to identify fuel rod design parameters such as rod plenum volume and initial pre-pressurization ensuring proper cladding balloon targets to investigate the FFRD phenomena are met. BISON has also been used to gain a better understanding of fuel stress behavior during a LOCA. Modeling results have indicated that fuel stresses change rapidly at the initiation of the LOCA due to the redistribution of the stored energy in the fuel where the center portion of the pellet decreases in temperature and the outer periphery increases. It is believed that this may have an impact on FFRD behavior as both the fuel stress state and temperature are important when describing an overpressurization of fission gas-filled pores in the fuel–a leading hypothesis of the driving force of fuel fragmentation [2]. In general, temperature ramp rate effects in the cladding and the fuel have been shown to have strong impacts on both cladding and fuel behaviors including fission gas release [3], never evaluated during an integral LOCA experiment and requiring fission heating of the specimen.

The design of TWIST has also focused strongly on its instrumentation package which is needed to provide data to the multiscale advanced modeling efforts within DOE R&D programs. Several first-of-a-kind in-situ instrumentation approaches have been developed and are being integrated into TWIST including: optical fiber based noncontact temperature measurements in the cladding balloon region, an electro-impedance sensor to measure cladding balloon extent, and the TREAT Fuel Motion Monitoring System to provide in-situ monitoring of fuel relocation.

Along with the design of the in-pile TWIST experiment vehicle, an out-of-pile equivalent experiment facility has been designed and is under construction. This facility will support initial and ongoing in-pile TWIST experiments by providing detailed characterization of the thermal-hydraulic performance and the ability to develop and qualify instrumentation, both of which would not be easily achievable in a nuclear environment.

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Development of the Temperature Heat sink Overpower Response Capsule for Use at Transient Reactor Test Facility

Principal Investigators: D. Devin Imholte
Team Members/ Collaborators: Colby Jensen, Nicolas Woolstenhulme, Austin Fleming, Todd Birch, Randall Fielding, Scott Wilde, Cole Blakely, and Robert Armstrong

The THOR device is an important new test platform for TREAT supporting many customers already, allowing for passive heat rejection from test specimens to simulate important advanced reactor fuel conditions.

The Temperature Heat sink Overpower Response (THOR) capsule is a Transient Reactor Test (TREAT) Facility irradiation experiment vehicle for simulating transient overpower events typical of Sodium Fast Reactor (SFR) fuels. The heat sink within THOR, combined with TREAT's shaped transients, allows for unique transient temperature control and energy densities characteristic of overpower events. In addition, THOR's advanced instrumentation portfolio enables flexibility for measuring fuel temperature and timing of fuel failure. The THOR capsule has been modified to support a variety or fresh and pre-irradiated fuel specimen types including U-10Zr, U-19Pu-10Zr, UO_2, Mixed Oxide (MOX) and others.

Project Description:

Given the broad range of fuel specimen types and test objectives, the THOR capsule has been designed to have an advanced instrumentation portfolio that is customizable for measuring different phenomena of interest. At the same time, the THOR capsule has been standardized across these experiments to take advantage of similar hardware, while maintaining robust design to withstand hot cell assembly and transient events. The heat sink has been specifically designed to include an integral 1000 W cable heater for achieving high SFR operating and accident temperatures of interest, along with thermocouples and distributed temperature sensing fibers for measuring fuel and cladding temperature response thru the heat sink. Figure 1 shows images of a fully instrumented heat sink, along with photographs of the recently assembled THOR-C-1 heat sink. With the capsule being a welded Inconel-625 pipe enclosure, it is designed for the highest temperature conditions of interest in SFR accidents.

The THOR capsule also has a configuration that is optimal for hot cell assembly. Whereas the fresh-fuel capsules have a straight hanger rod and requires electrical connections to be made between the capsule and

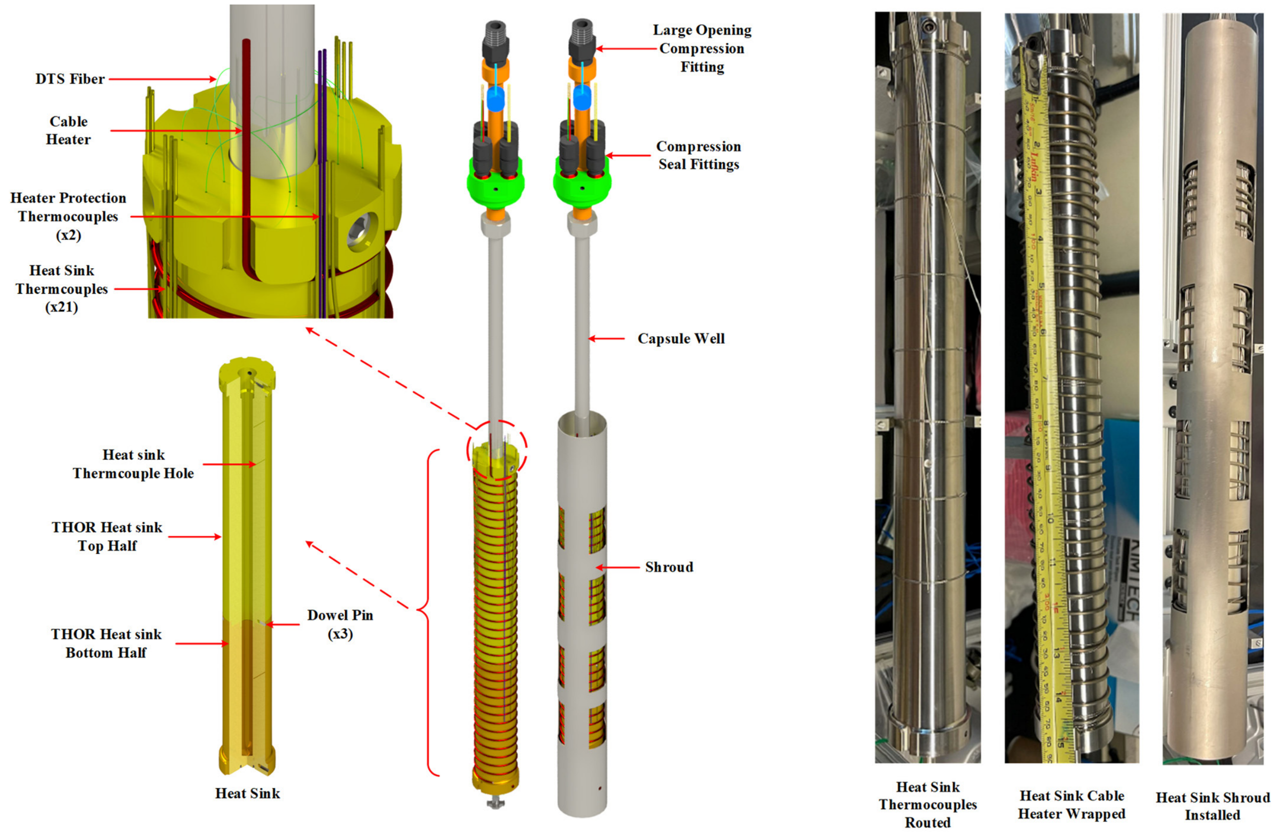

Figure 1. THOR heat sink (left) and THOR-C-1 assembled heat sink (right)

the flange, the pre-irradiated fuel capsules have a hinged pipe that pivots to allow for fuel and sodium loading with the hot cell. This avoids making capsule-to-flange connections in the hot cell, and places more focus on performing the sodium loading, fuel loading and sodium bonding. Figure 2 provides an overview of various THOR

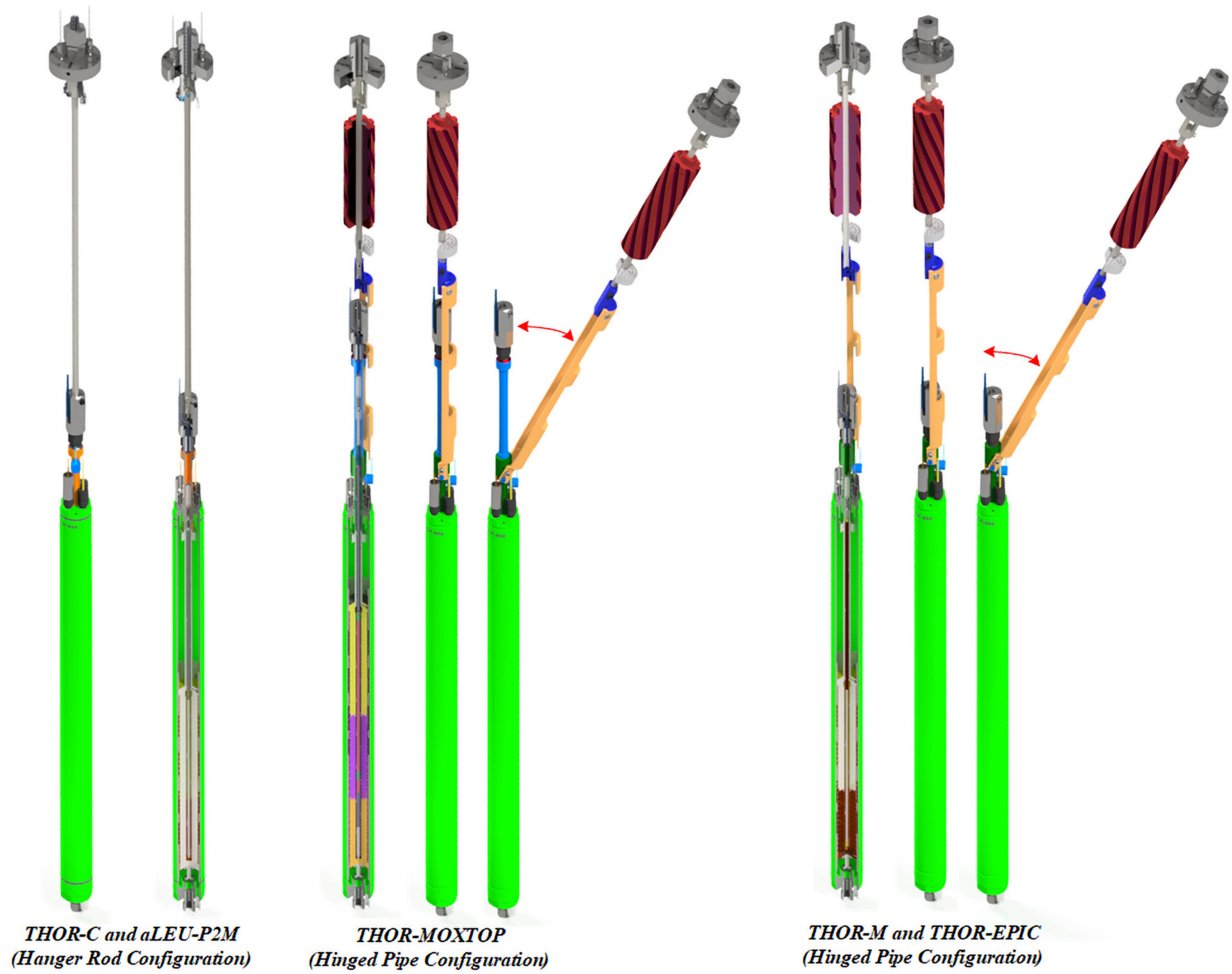

Figure 2. THOR capsule configurations – fresh (THOR-C, aLEU-P2M) and pre-irradiated fuel (MOXTOP, THOR-M, EPIC)

capsule adaptations, tuned for each specific experiment objectives.

There are several experiment campaigns that the THOR capsule supports. The first experiment campaign will be the THOR Commissioning (THOR-C). The primary THOR C experiment objective is to commission a sodium heat sink testing capability in TREAT. This will be accomplished by executing several transients of unirradiated (i.e., "fresh") U-10Zr to demonstrate the ability of the capsule to measure and detect transient overpower phenomena typical of SFR fuels. THOR-C will have three separate configurations for measuring cladding rupture, annular fuel temperature and in-situ fuel elongation.

The THOR Advanced Low-Enriched Uranium Power-to-Melt (aLEU-P2M) experiment (1) assesses the effect of thermally conductive inserts in fresh UO_2-based fuel by measuring

Figure 3. Assembly of THOR-C-1 capsule in the MFC pyrochemistry glovebox

Figure 4. Assembled THOR-C-1 capsule

their net effect on radial thermal conductivity using transient nuclear heating and (2) determines their power-to-melt threshold in transient overpower ramps. This experiment is funded by the aLEU program.

The remaining THOR-based experiments are all concerning pre-irradiated fuel. The THOR Mixed Oxide Transient OverPower (MOXTOP) experiment campaign is a collaborative effort with the Japanese Atomic Energy Agency (JAEA) to establish the fuel cladding mechanical interaction threshold for pre-irradiated oxide fuels for the JAEA JOYO reactor irradiation experiments, and for qualification of the same oxide fuels for future commercial SFRs.

The THOR Metallic (THOR-M) is essentially a continuation of the THOR-C experiment campaign, with an emphasis on simulating transient overpower and loss-of-flow events for pre-irradiated U-19Pu-10Zr fuels. Its primary objective is to test pre-irradiated metallic U-Zr and U-Pu-Zr fuel alloys at power/thermal conditions extending to fuel failure using the THOR capsule. Ultimately, these tests will continue incomplete historical metal fuels testing and inform development of fuel performance models and fuel safety criteria for industry.

Finally, the THOR EBR-II Pre-Irradiated Conductivity (EPIC) experiment aims to determine thermal conductivity of pre-irradiated U-Pu-Zr fuels that are each irradiated to various burnup levels. This experiment is funded by the Nuclear Science User Facility (NSUF) program.

Accomplishments:

The THOR-C, aLEU-P2M, MOXTOP, THOR-M and EPIC experiments are all at different stages of the design process. Even so, all of them have notable achievements. The first THOR-C experiment, THOR-C-1, has progressed thru fabrication, testing and assembly and will be irradiated in TREAT. Included in this stage has been establishing a sodium bonding procedure and molten sodium encapsulation method that will inform all subsequent THOR-based experiments. Figure 3 shows technicians performing the final torquing of the THOR-C-1 capsule compression fitting. Figure 4 shows the totally fueled and sealed THOR-C-1 capsule.

Fabrication is underway for the iron heat sink components for the aLEU-P2M and MOXTOP experiments. These heat sinks have precise machined features that are pushing the limits of electrical discharge machining (EDM) technology. These features enable high-resolution temperature measurement, including 21x Type K thermocouples and a distributed temperature sensing (DTS) fiber for in-situ cladding and fuel temperature measurement at different axial and radial locations. The aLEU-P2M and MOXTOP experiments have all completed final design review, while the THOR-M and EPIC experiments have completed preliminary design review and are awaiting their final design review. One of the most significant areas of progress across these THOR experiment campaigns has been establishing a standard method for simulating shaped transients in TREAT, specifically with regards to the safety implications. This method will inform other TREAT experiments in the future

Advanced Fuels Campaign High Flux Isotope Reactor Irradiation Testing Highlights

Principal Investigator: Annabelle Le Coq
Team Members/ Collaborators: Christian Petrie, Kory Linton, Jason Harp, Jacob Gorton, Zane Wallen, Patrick Champlin, David Bryant, and Padhraic Mulligan

Irradiation testing in ORNL's High Flux Isotope Reactor (HFIR) supports advancements in the development of various accident-tolerant fuel (ATF) material concepts.

Materials and fuels irradiation testing at Oak Ridge National Laboratory (ORNL) leverages the High Flux Isotope Reactor (HFIR) capabilities to study various Accident-Tolerant Fuel (ATF) material concepts. HFIR irradiation experiment capsules (also referred to as "rabbits") and full-length target designs allow small-scale specimens to be irradiated at a particular damage dose and temperature. Experiments can be inserted in HFIR's flux trap, removable beryllium (RB) reflector, or permanent beryllium reflector's vertical experiment facilities (VXFs), depending on the irradiation goals and specimen geometry. Irradiation testing at ORNL is a major part in the development and qualification of novel material and fuel concepts to improve the accident-tolerance of existing and future reactor designs. Current HFIR irradiation testing of ATF concepts include MiniFuel targets, cladding tube rabbits and targets, and creep tube rabbit experiments.

Project Description

MiniFuel experiments

MiniFuel experiments, designed at ORNL allow separate effects accelerated irradiation testing of miniature fuel specimens [1]. The experiment design accommodates various fuel geometries such as kernels (∅ 425 to 800 μm), tri-structural isotropic (TRISO) particles, or disk specimens, and various fuel forms (for example, UN, UC, UO_2, U_3Si_2, UCO). Post-irradiation examination (PIE) of the first MiniFuel target accommodating UN kernels and TRISO particles was completed and includes irradiation temperature analysis, fission gas release measurements, and fuel integrity analysis [2]. This first target achieved the designed target temperature ranging from 450°C to 550°C and accumulated a burnup of 10 MWd/kg U. Two other Advanced Fuels Campaign (AFC) targets containing UN and UCO compacts, and TRISO particles have completed irradiation and are intended for similar fuel performance evaluation. To enhance the MiniFuel capability, the existing VXF position experiment design was modified for RB positions, which are more readily available and closer to the reactor core. The RB MiniFuel design is illustrated in Figure 1 [3]. Experiments inserted in such positions can achieve heat generation rates and burnup accumulation rates more than twice the VXF positions, enabling higher irradiation temperatures and reduced irradiation time to reach similar burnups. In addition, the RB

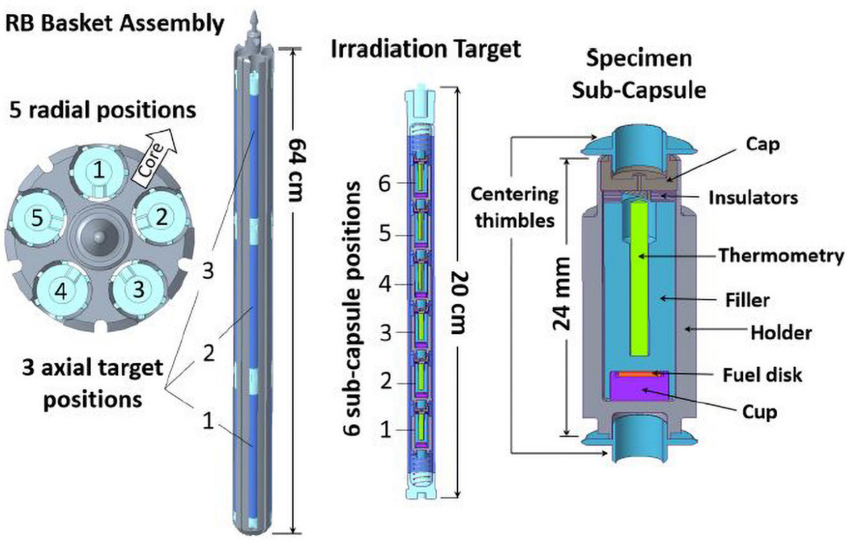

Figure 1. MiniFuel RB irradiation vehicle accommodating 15 MiniFuel targets

MiniFuel design provides additional testing positions, fifteen targets being able to be inserted in an aluminum basket in one RB position versus the 9 targets accommodated by a VXF basket. This design would allow UO_2 fuel specimens to reach temperatures beyond 1,000°C, enabling observation of fission gas release for moderate burnups in the range of 40–50 MWd/kg-U.

Cladding tube specimens irradiation testing

Coated-zirconium cladding tube specimens were inserted in rabbit capsules for HFIR irradiation testing at temperatures relevant for pressurized-water reactors (PWRs) (300-400°C) and damage of 2 or 8 displacements per atom (dpa) [4]. The specimens are short PWR cladding segments about 24 mm long and 9.5 mm diameter. In addition, PWR geometry plane-strain and tensile tube specimens about 12 mm long were also inserted in HFIR for irradiation. The plane-strain specimen geometry is a standard tube specimen featuring two notches at each end of the specimens for hoop strain in plain strain state measurements. The tensile tube specimen geometry presents two rings connected by two gauge regions for uniaxial tension testing. A rabbit capsule can accommodate two 24-mm long tube specimens and four 12-mm long plane-strain or tensile specimens.

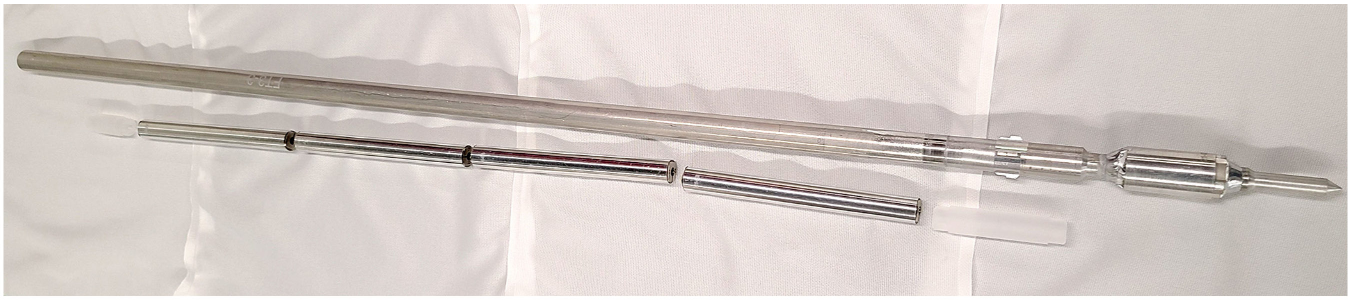

Figure 2. Parts layout for capsule assembly of tensile cladding tube specimens

Figure 2 shows one capsule parts layout for irradiation of tensile tube specimens. Post-irradiation the specimens will be examined for coating integrity evaluation and mechanically tested to evaluate the neutron irradiation effects on specimens with representative PWR cladding geometry. Longer cladding specimens (about 102 mm long) were also inserted into HFIR irradiation targets (see parts layout in Figure 3). Four longer cladding specimens can be inserted in a single target. PIE on such specimens can provide information on thermal and mechanical fatigue and burst behavior.

Creep tube irradiation testing

To quantify irradiation creep on ATF cladding concepts and the effect of coatings on standard cladding creep, pressurized FeCrAl and coated Zr tubes specimens were inserted in HFIR. The specimens were internally pressurized to generate a hoop stress up to 180 MPa and were irradiated to 2 or 8 dpa at temperatures representative of PWR cladding temperatures. Au automated profilometer measurement system was developed to allow diameter measurements of the pressurized tubes in the hot cells. This low-cost and custom measurement technique was effective at capturing tube deformation on the micron scale and determining the irradiation creep rate in the tube specimens. It was observed that the coating does not impact irradiation creep in Zr tubes. Future PIE may include passive thermometry analysis via dilatometry and depressurized tube diameters measurements to evaluation the change in Young's modulus after irradiation.

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[1.] C.M. Petrie, J.R. Burns, A.M. Raftery, A.T. Nelson, K.A. Terrani, "Separate Effects Irradiation Testing of Miniature Fuel Specimens," Journal of Nuclear Materials, 526 (2019).

[2.] J.M. Harp, R.N. Morris, C.M. Petrie, J.R. Burns, K.A. Terrani, "Postirradiation Examination from Separate Effects Irradiation Testing of Uranium Nitride Kernels and Coated Particles," Journal of Nuclear Materials, 544 (2021).

[3.] J.P. Gorton, Z.G. Wallen, C.M. Petrie, "Modifications to MiniFuel Vehicle to Enable Higher Temperature UO2 Irradiation Capabilities," ORNL/SPR-2021/2096, Oak Ridge National Laboratory, Oak Ridge, TN, August 2021.

Figure 3. Parts layout for target assembly of long cladding tube specimens

[4.] P.A. Champlin, C.M. Petrie, A.G. Le Coq, K.R. Smith, K D. Linton, "Thermal Analysis and Irradiation Growth of Coated Zirconium Alloy Cladding Specimens in HFIR," ORNL/TM-2020/1567, Oak Ridge National Laboratory, Oak Ridge, TN, August 2020.

[5.] P. Mulligan, et al. "Post-irradiation Characterization of ARMOR Coated Zircaloy Pressurized Creep Tubes", ORNL/SPR-2021/2157, Oak Ridge National Laboratory, Oak Ridge, TN, 2021.


NOT

APPENDIX

5.1 PUBLICATIONS

Author	Title	Publication
O. Anderoglu	Layer Dissolution in Accumulative Roll Bonded Bulk Zr/Nb Multilayers Under Heavy-Ion Irradiation	JNM, 2021 accepted
Thilo Balke, Alexander M. Long, Sven C. Vogel, Brendt Wohlberg and Charles A. Bouman	Hyperspectral Neutron CT with Material Decomposition	In 2021 IEEE International Conference on Image Processing (ICIP), pp. 3482-3486. IEEE, 2021
G.L. Beausoleil et al.	Integrating Advanced Modeling and Accelerated Testing for a Modernized Fuel Qualification Paradigm	Nuclear Technology, 2021: p. 1-20
J. Bess et al.	Utility of EBR-II Benchmark Model to Enable MOX Fuel Pin Characterization	ANS Transactions 2021, 124(1) 238-241
Nathan Capps, Colby Jensen, Fabiola Cappia, Jason Harp, Kurt Terrani, Nicolas Woolstenhulme and Daniel Wachs	A Critical Review of High Burnup Fuel Fragmentation, Relocation, and Dispersal under Loss-Of-Coolant Accident Conditions	Journal of Nuclear Materials, Volume 546, 2021, 152750, ISSN 0022-3115, https://doi.org/10.1016/j.jnucmat.2020.152750
N. Chaari, J. Bischoff, K. Buchanan, C. Delafoy , P. Barberis, J. Augereau and K. Nimishakavi	The Behavior of Cr-coated Zirconium Alloy Cladding Tubes at High Temperatures	STP1622 on 19th International Symposium on Zirconium in the Nuclear Industry
P.A. Champlin, C.M. Petrie, A.G. Le Coq, K.R. Smith and K D. Linton	Thermal Analysis and Irradiation Growth of Coated Zirconium Alloy Cladding Specimens in HFIR	ORNL/TM-2020/1567, Oak Ridge National Laboratory, Oak Ridge, TN, August 2020
Amani Cheniour, Giovanni Pastore, Jason Harp, Christian Petrie and Nathan Capps	Assessment of the Fission Gas Release Model in BISON Applied to UO_2 MiniFuel	To be submitted to Journal of Nuclear Materials, under review
Bryon Curnutt, Nicolas Woolstenhulme, Joseph Nielsen, Kevan Weaver, Colby Jensen and Austen Fradeneck	A Neutronics Investigation Simulating Fast Reactor Environments in the Thermal Spectrum Advanced Test Reactor	Manuscript submitted to Nuclear Technology Aug 2021
A. Duenas, D. Wachs, G. Mignot, J. N. Reyes, Q. Wu, and W. Marcum	Dynamical System Scaling Application to Zircaloy Cladding Thermal Response During Reactivity-Initiated Accident Experiment	Nuclear Science and Engineering, (2021), DOI: 10.1080/00295639.2021.1955591
Bowen Gong, Lu Cai, Penghui Lei, Kathryn E Metzger, Edward J. Lahoda, Frank A. Boylan, Kun Yang, Jake Fay, Jason Harp and Jie Lian	Cr Doped U_3Si_2 Composite Fuels under Steam Corrosion	Corrosion Science 177 (2020)

Author	Title	Publication
Bowen Gong, Tiankai Yao, Penghui Lei, Lu Cai, Kathryn E Metzger, Edward J. Lahoda, Frank A. Boylan, Afiqa Mohamad, Jason Harp, Andy Nelson and Jie Lian	U_3Si_2 and UO_2 Composites Densified by Spark Plasma Sintering for Accident-Tolerant Fuels	Journal of Nuclear Materials 534 (2020)
Adrian Gonzales, Jennifer K. Watkins, Adrian R. Wagner, Brian J. Jaques and Elizabeth S. Sooby	Challenges and Opportunities to Alloyed and Composite Fuel Architectures to Mitigate High Uranium Density Fuel Oxidation: Uranium Silicide	Journal of Nuclear Materials Volume 553, September 2021
J.P. Gorton, Z.G. Wallen and C.M. Petrie	Modifications to MiniFuel Vehicle to Enable Higher Temperature UO_2 Irradiation Capabilities	ORNL/SPR-2021/2096, Oak Ridge National Laboratory, Oak Ridge, TN, August 2021.
A. Gouws, D. Hagen, A. Chen, E. Kardoulaki, J.J. Beaman and D. Kovar	Onset of Selective Laser Flash Sintering of AlN	Int J Appl Ceram Technol. 00 (2021) 1– 11. https://doi.org/10.1111/ijac.13840.
T. Graening	AFC Burst Activities with Coated Zircaloy-4 Under Accident Conditions	M2FT-21OR020204071-ORNL/SPR-2021/2117, July 2021
T. Graening	Development of Standardized Property Requirements, Measurement Methods, and Reporting Guidance for Coatings	M4FT-21OR020202013, ORNL/SPR-2021/6, January 2021
T. Graening	Microstructure Investigation and Mechanical Properties of Coated Zircaloy Cladding	M2FT-21OR020202011, ORNL/SPR-2021/2038, June 2021
M. Grosse, M. Steinbrueck, J. Stuckert, J. Yanga, M. Sevecek and L. Czerniak	High-temperature Behaviour of Chromium Coated Zirconium- based Fuel Cladding Materials	TopFuel2021, October 24, 2021
J.M. Harp, R.N. Morris, C.M. Petrie, J.R. Burns and K.A. Terrani	Postirradiation Examination from Separate Effects Irradiation Testing of Uranium Nitride Kernels and Coated Particles	Journal of Nuclear Materials, 544 (2021)
C. Jensen et al.	Final Design of the MOXTOP Experiment in TREAT	INL/LTD-21-62032 Rev. 0, March 2021
E. Kardoulaki, D.M. Frazer, J.T. White, U. Carvajal, A.T. Nelson, D.D. Byler, T.A. Saleh, B. Gong, T. Yao, J. Lian and K.J. McClellan	Fabrication and Thermophysical Properties of UO_2-UB_2 and UO_2-UB_4 Composites Sintered Via Spark Plasma Sintering	Journal of Nuclear Materials, 544 (2021) 1-12. https://doi.org/10.1016/j.jnucmat.2020.152690.

Author	Title	Publication
E. Kardoulaki, D.D. Byler, J. Bárta and K.J. McClellan	Tri-arc Growth and Characterization of U_3Si_2 and U_3Si_5 Single Crystals	J. Cryst. Growth. 558 (2021) 1-9. https://doi.org/10.1016/j.jcrysgro.2021.126025.
Zeses Karoutas, Kathryn Metzger, Luke Olson, Luke Hallman, Ed Lahoda, Michael Sivack, John Lyons, Luke Czerniak, Frank Boylan, Zachary McDaniel, Robert Terry, Antoine Claisse, Denise Adorno Lopes and Jonathan Wright	Westinghouse Encore® Accident Tolerant Fuel and High Energy Program	TopFuel2021, October 24, 2021
Takaaki Koyanagi, Hsin Wang, Christian M. Petrie, Christian P. Deck, Weon-Ju Kim, Daejong Kim, Cédric Sauder, James Braun and Yutai Katoh	Thermal Diffusivity and Thermal Conductivity of SiC Composite Tubes: The Effects of Microstructure and Irradiation	Journal of Nuclear Materials 557 (2021): 153217.
Donghwi Lee, Barret Elward, Paul Brooks, Rajnikant Umretiya, Jessika Rojas, Matteo Bucci, Raul B. Rebak and Mark Anderson	Enhanced Flow Boiling Heat Transfer on Chromium Coated Zircaloy-4 Using Cold Spray Technique for Accident Tolerant Fuel (ATF) Materials	Applied Thermal Engineering, 185 (2021) 116347, https://doi.org/10.1016/j.applthermaleng.2020.116347
Denise Adorno Lopes, Antoine Claisse, Magnus Limbäck Kathryn Metzger Ed Lahoda, Michael Sivack Frank Boylan, Zachary McDaniel and Robert Terry	Atomistic Modeling at Westinghouse: From Design to Performance of Fuel Compounds	TopFuel2021, October 24, 2021
M. Moorehead, P. Nelaturu, M. Elbakhshwan, C. Parkin, C. Zhang, K. Sridharan, D. Thoma and A. Couet	High-Throughput Ion Irradiation of Additively Manufactured Compositionally Complex Alloys	Jounal of Nuclear Materials, 152782, Jan. 2021
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Author	Title	Publication
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R.R. Ingraci Neto, K.J. McClellan, D.D. Byler and E. Kardoulaki	Controlled Current-rate AC Flash Sintering of Uranium Dioxide	Journal of Nuclear Materials, 547 (2021) 1-13. https://doi.org/10.1016/j.jnucmat.2021.152780.
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R. K. Ratnayake, D. F. Hussey, L. Olson, G. Wang, W. Byers and R. Becker	ATF Coating Response in KOH Adjusted PWR Water	TopFuel2021, October 24, 2021
C.J. Rietema	The Role of Nitrogen in the Structure and Properties of Proton Irradiated 12Cr1MoWV Ferritic/Martensitic Steel for Advanced Nuclear Reactor Cores	Ph.D. Thesis, Colorado School of Mines, 2021
C.J. Rietema, M.M. Hassan, O. Anderoglu, B.P. Eftink, T.A. Saleh, S.A. Maloy, A.J. Clarke and K.D. Clarke	Ultrafine Intralath Precipitation of V(C,N) in 12Cr-1MoWV (wt.%) Ferritic/Martensitic Steel	Scr. Mater. 197 (2021). https://doi.org/https://doi.org/10.1016/j.scriptamat.2021.113787
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D.C. Roache, C.H. Bumgardner, T.M. Harrell, M.C. Price, A. Jarama, F.M. Heim, J. Walters, B. Maier and X. Li	Cr-Coated Cladding Demonstration Program for DOEL-4	2021 46th Annual Reunion of Spanish Nuclear Society, Granada Spain, October 6-8 2021.

Author	Title	Publication
D.C. Roache, C.H. Bumgardner, T.M. Harrell, M.C. Price, A. Jarama, F.M. Heim, J. Walters, B. Maier and X. Li	Design and Safety Evaluation of Cr-Coated Lead Test Rods for Doel Nuclear Power Plant Unit 4	TopFuel2021, Santander Spain.
D.C. Roache, C.H. Bumgardner, T.M. Harrell, M.C. Price, A. Jarama, F.M. Heim, J. Walters, B. Maier and X. Li	Experimental Evaluation of Chromium Cold-Spray Coated Cladding	TopFuel2021, Santander Spain
D.C. Roache, C.H. Bumgardner, T.M. Harrell, M.C. Price, A. Jarama, F.M. Heim, J. Walters, B. Maier and X. Li	Poolside Inspection of EnCore® Fuel Lead Test Rods at Exelon Byron Unit 2	TopFuel2021, October 24, 2021
D.C. Roache, C.H. Bumgardner, T.M. Harrell, M.C. Price, A. Jarama, F.M. Heim, J. Walters, B. Maier and X. Li	Poolside Inspection of EnCore® Fuel Lead Test Rods at Exelon Byron Unit 2	2021 ENYGF (European Nuclear Young Generation Forum), Tarragona Spain, September 27-30 2021
D.C. Roache, C.H. Bumgardner, T.M. Harrell, M.C. Price, A. Jarama, F.M. Heim, J. Walters, B. Maier and X. Li	Unveiling Damage Mechanisms of Chromium-Coated Zirconium-Based Fuel Claddings at LWR Operating Temperature by In-Situ Digital Image Correlation	Submitted draft, Materials Science & Engineering A, paper submitted July 20, 2021
D.C. Roache, A. Jarama, C.H. Bumgardner, F.M. Heim, M.C. Price, T.M. Harrell, J. Walters, B. Maier and X. Li	Damage Mechanisms of Chromium Coated Zirconium Alloy Fuel Claddings by In Situ Digital Image Correlation and Acoustic Emissions	Presented at Materials Science & Technology, Virtual, November 4, 2020
D.C. Roache, A. Jarama, C.H. Bumgardner, F.M. Heim, J. Walters, J. Romero, B. Maier and X. Li	Unveiling Damage Mechanisms of Chromium-coated Zirconium-based Fuel Claddings by Coupling Digital Image Correlation and Acoustic Emission	Mater. Sci. Eng., A, 774 (2020)
Michael A. Shockling, Aaron M. Everhard, Ho Q. Lam and Uriel Bachrach	Assessment of Loss-Of-Coolant Accident Fuel Dispersal for High Burnup Core Designs	TopFuel2021, October 24, 2021

Author	Title	Publication
Elizabeth S. Sooby, Brian A. Brigham, Geronimo Robles, Zachary Acosta and Joshua T. White	Steam Oxidation Performance of Uranium Mononitride during Thermal Ramp and Isothermal Conditions	June 2021 ANS meeting
Martin Steinbrueck, Mirco Grosse, Ulrike Stegmaier, James Braun and Christophe Lorrette	High-temperature Oxidation of Silicon Carbide Composites for Nuclear Applications	TopFuel2021, October 24, 2021
Sven C. Vogel, Thilo Balke, Charles A. Bouman, Luca Capriotti, Jason M. Harp, Alex M. Long, Danielle C. Schaper, Anton S. Tremsin and Brendt E. Wohlberg	First Examination of Irradiated Fuel with Pulsed Neutrons at LANSCE (Preliminary Results)	Milestone report for the Nuclear Technology Research and Development Program, Advanced Fuels Campaign #M3FT-20LA020201023, LA-UR-20-27504 (2020)
Sven C. Vogel, Thilo Balke, Charles A. Bouman, Luca Capriotti, Jason M. Harp, Alex M. Long, Danielle C. Schaper, Anton S. Tremsin and Brendt E. Wohlberg	Pulsed Neutron Characterization of Irradiated U-10Zr-1Pd	Summary published in the transactions for the 2021 American Nuclear Society Meeting, June 14-16, 2021
H. Wang, B. Gould, M. Moorehead, M. Haddad, A. Couet and S. Wolf	In Situ X-ray and Thermal Imaging of Refractory High Entropy Alloying during Laser Directed Deposition	Journal of Materials processing, Jan. 2022, Vol 299, Pp 117363
Walter Williams, Maria Okuniewski, Sven Vogel and Jianzhong Zhang	An In-situ Neutron Diffraction Study of Crystallographic Evolution and Thermal Expansion Coefficients in U-22.5at.%Zr During Annealing	JOM Vol. 72 2020 2042–2050
N. E. Woolstenhulme, G. L. Beausoleil, C. Jensen, C. Petrie and D. Wachs	Irradiation Testing Methods for Fast Spectrum Reactor Fuels and Materials in DOE's Thermal Spectrum Test Reactors	ANS Transactions 2021, 124 (1), 168-171
N. Woolstenhulme, C. Jensen, C. Folsom, R. Armstrong, J. Yoo and D. Wachs	Thermal-Hydraulic and Engineering Evaluations of New LOCA Testing Methods in TREAT	Nuclear Technology, Vol. 207, Iss. 5, pp. 637-652, 2021

Author	Title	Publication
N. Woolstenhulme, C. Jensen, D. Kamerman, N. Oldham, D. Wachs, C. Petrie, K. Linton, D. Carpenter and G. Kohse	Development of the LWR Fuels and Materials Irradiation Platform in US Test Reactors	Proceedings of Top Fuel 2021 Conference
Yi, Xie, Sven C. Vogel, Jason M. Harp, Michael T. Benson and Luca Capriotti	Microstructure Evolution of U–Zr System in A Thermal Cycling Neutron Diffraction Experiment: Extruded U–10Zr (wt.%)	Journal of Nuclear Materials 544 (2021): 152665
Y. Yan, K.D. Linton, J.M. Harp, Z. Burns, T. Jordan and B. Johnston	High-Temperature Steam Oxidation of Irradiated FeCrAl in the Severe Accident Test Station	ORNL/SPR-2021/2035, Oak Ridge National Laboratory, Oak Ridge, TN, May 2021
Kun Yang, Erofili Kardoulaki, Dong Zhao, Bowen Gong, Andre Broussard, Kathryn Metzger, Joshua T. White, Michael R. Sivack, Kenneth J. Mcclellan, Edward J. Lahoda and Jie Lian	Uranium Nitride (UN) Pellets with Controllable Microstructure and Phases – Fabrication by Spark Plasma Sintering and their Thermal-mechanical and Oxidation Properties	Under revision for Journal of Nuclear Materials
Kun Yang, Erofili Kardoulaki, Dong Zhao, Bowen Gong, Andre Broussard, Kathryn Metzger, Joshua T. White, Michael R. Sivack, Kenneth J. Mcclellan, Edward J. Lahoda and Jie Lian	Uranium Nitride (UN)+Cr Composite Nuclear Fuels with Enhanced Mechanical Performance and Oxidation Resistance	TopFuel2021, October 24, 2021
Liang Yin, T.B. Jurewicz, M. Larsen, M. Drobnjak, C.C. Graff, D. R. Lutz and R.B. Rebak	Uniform Corrosion of FeCrAl Cladding Tubing for Accident Tolerant Fuels in Light Water Reactors	Journal of Nuclear Materials, 554 (2021) 153090, https://doi.org/10.1016/j.jnucmat.2021.153090
Guanjie Yuan, Paul F. Kreutzer, Peng Xu, Luke Olson, Edward J. Lahoda and Dong Liu	In Situ High Temperature X-ray Tomography of SiC/SiC Composites under C Ring Compression Test	TopFuel2021, October 24, 2021

5.2 FY-21 LEVEL 2 MILESTONES

Work Package Title	Site	Work Package Manager	Level 2 Milestone
AFC Coordination and Integration	INL	Mai, Edward	Complete Draft 2020 Accomplishments Report
TREAT Refabrication Capability (Capital)	INL	Cole, Mark	Demonstrate Ability to Perform Re-fabrication of Fuel Rods in Mockup
Fabrication and Characterization of Coated Cladding	ORNL	Nelson, Andy	Issue Report Documenting Characterization of Coated Zry-4 Cladding for Round Robin Testing
ATF Irradiation and Accelerated Qualification	ORNL	Petrie, Christian	HFIR SiC Bowing Test Ready to Insert
ATR I-Loop Installation (capital - TEC)	INL	Strain, Lex	Receive Top Head Closure Plate from Vendor
Westinghouse ATF FOA	PNNL	Zbib, Ali	PNNL - W/Byron Spent Fuel Rods Feasibility Assessment
HERA	INL	Emerson, Leigh Ann	Conduct Final Design Review for Simulated HBU Test for HERA
Transient Testing	ORNL	Linton, Kory	Issue Report on Impact of Cr Coatings on Transient Response
Demonstration TREAT Tests on Previously Irradiated Fuel	INL	Emerson, Leigh Ann	Complete First TREAT Test with Previously Irradiated Fuel
Advanced Ceramic Fuel Development	LANL	White, Josh	Report on the Integrated Experimental and Modeling Approach to Enhanced Grain Size UO_2
ATF PIE - Coordination /Support	INL	Cappia, Fabiola	Complete Mechanical Testing Demonstrations
Perform Evaluation of FAST Application to U-10Zr/HT9 System	INL	Murdock, Chris	Receipt of Irradiated FAST-1 Capsules at HFEF
Integral Irradiation Testing of ATF in ATR	INL	Hoggard, Gary	Issue ATF-2B Final Irradiation Report
LOCA Commissioning Capsule Tests	INL	Dempsey, Doug	Complete Assembly of Out-of-Pile LOCA Testbed Device, the TWIST Prototype, to Support LOCA Testing in TREAT
Interface Mechanical Property Testing for Coated Zircaloy	LANL	Maloy, Stuart	Issue Report on Bulge Testing and Microscale Testing on Coated Zircaloy
ARES Joint Work with JAEA to Study Off-Normal Behavior of Fast Reactor Fuels	INL	Smuin, Trevor	Perform First Modern Transient Irradiation Experiment on Metallic Fuel
TREAT Reinstrumentation Capability	INL	Cole, Mark	Issue a Report on the Progress of Researching, Designing, and Ultimately Developing Fuel Rodlets Instrumented with a Fuel Centerline Thermocouple
ATR I-Loop Installation (Capital - TEC)	INL	Strain, Lex	Ready to Install and Test THCP

TOP HAT
1700 LB
DWG 732550
TREAT
LOOP CASK
MAX LOADED
DWG 732533
TAYLOR

5.3 AFC NUCLEAR ENERGY UNIVERSITY PROJECTS (NEUP) GRANTS

Active Projects Awarded in 2016

Nuclear Energy University Cooperative Agreements

Lead University	Title	Principal Investigator
The Ohio State University	Alloying Agents to Stabilize Lanthanides Against Fuel Cladding Chemical Interaction: Tellurium and Antimony Studies	Christopher Taylor
Rensselaer Polytechnic Institute	Oxidation and Corrosion-resistant Uranium Silicide Fuels	Jie Lian

Active Projects Awarded in 2017

Nuclear Energy University Cooperative Agreements

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 Nevada, Reno	Development and Experimental Benchmark of Computational Models to Predict Cladding Temperature and Vapor Removal from UNF Canisters during Drying Operations	Miles Greiner
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 Cooperative Agreements

Lead University	Title	Principal Investigator
Virginia Polytechnic Institute and State University	C-SiOC-SiC Coated Particle Fuels for Advanced Nuclear Reactors	Kathy Lu
University of California, Berkeley	Understanding of Degradation of SiC/SiC Materials in Nuclear Systems and Development of Mitigation Strategies	Peter Hosemann
University of California, Berkeley	Bridging the Length Scales on Mechanical Property Evaluation	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	Peng Wang
Purdue University	Microstructure-Based Benchmarking for Nano/Microscale Tension and Ductility Testing of Irradiated Steels	Janelle Wharry
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
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 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
University of South Carolina	Development of Multi-Axial Failure Criteria for Nuclear Grade SiCf-SiCm Composites	Xinyu Huang

Active Projects Awarded in 2019

Nuclear Energy University Cooperative Agreements

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	Wolfgang Windl
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 Post Irradiation 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 2020

Nuclear Energy University Cooperative Agreements

Lead University	Title	Principal Investigator
University of Wisconsin-Madison	Investigation of Degradation Mechanisms of Cr-coated Zirconium Alloy Cladding in Reactivity Initiated Accidents (RIA)	Hwasung Yeom
University of Nevada, Las Vegas	Single- and Polycrystalline Diamond Electrodes for Spectroelectrochemical Characterization of Various Molten Salts	Cory Rusinek
Boston University	Connecting Advanced High-Temperature X-ray and Raman Spectroscopy Structure/Dynamics Insights to High-Throughput Property Measurements	Uday Pal
University of Wisconsin-Madison	Maintaining and Building Upon the Halden Legacy of In-situ Diagnostics	Michael Corradini
University of California, Berkeley	Femtosecond Laser Ablation Machining & Examination - Center for Active Materials Processing (FLAME-CAMP)	Peter Hosemann
Rensselaer Polytechnic Institute	Chemical Interaction and Compatibility of Uranium Nitride with Liquid Pb and Alumina-forming Austenitic Alloys	Jie Lian
Georgia Institute of Technology	Linear and Nonlinear Guided Ultrasonic Waves to Characterize Cladding of Accident Tolerant Fuel (ATF)	Laurence Jacobs
Pennsylvania State University	High Throughput Computational Platform for Predictive Modeling of Thermochemical and Thermophysical Properties of Fluoride Molten Salts	Shunli Shang
Rensselaer Polytechnic Institute	First-Principles Free Energies by Hybrid Thermodynamic Integration for Phase Equilibria and Fission Product Solubility in Molten Salts	Ravishankar Sundararaman

Active Projects Awarded in 2021

Nuclear Energy University Cooperative Agreements

Lead University	Title	Principal Investigator
University of Tennessee at Knoxville	Safety Implications of High Burnup Fuel for a 2-Year PWR Fuel Cycle	Nicholas Brown
University of Tennessee at Knoxville	Fuel-to-Coolant Thermomechanical Behaviors Under Transient Conditions	Nicholas Brown
University of Florida	High-Fidelity Modeling of Fuel-to-Coolant Thermomechanical Transport Behaviors Under Transient Conditions	Justin Watson
University of Wisconsin-Madison	Post-DNB Thermo-mechanical Behavior of Near-term ATF Designs in Simulated Transient Conditions	Hwasung Yeom
University of Tennessee at Knoxville	Modeling High-Burnup LWR Fuel Behavior Under Normal Operating and Transient Conditions	Giovanni Pastore
Massachusetts Institute of Technology	Experimental Investigation and Development of Models and Correlations for Cladding-to-Coolant Heat Transfer Phenomena in Transient Conditions in Support of TREAT and the LWR Fleet.	Matteo Bucci
Oregon State University	Characterizing Fuel Response and Quantifying Coolable Geometry of High-Burnup Fuel	Wade Marcum
University of Pittsburgh	Fragmentation and Thermal Energy Transport of Cr-doped Fuels under Transient Conditions	Heng Ban
Texas A&M University	Multiscale Modeling and Experiments for Investigating High Burnup LWR Fuel Rod Behavior Under Normal and Transient Conditions	Karim Ahmed
Pennsylvania State University	Estimation of Low Temperature Cladding Failures During an RIA Transient	Arthur Motta

5.4 ACRONYMS

ADF .... Annular Dark Field
AEC .... Atomic Energy Commission
AFC .... Advanced Fuels Campaign
AFQ .... Accelerated Fuel Qualification
AGR .... Advanced Gas Reactor
AL .... Air Liquide
aLEU .... Advanced Low-Enriched Uranium
ALIP .... Annular Linear Induction Pumps
AMET .... Automated Welding Systems
ANO .... Arkansas Nuclear One
ANS .... American Nuclear Society
APMT .... Advanced Powder Metallurgy Tubing
APT .... Atom Probe Tomography
AR .... As-Rolled
ARCTIC .... Advanced Reactor Concepts in Thermal-spectrum Capability
ARES .... Advanced Reactor Experiments for Sodium Fast Reactor Fuels
ATF .... Accident Tolerant Fuel
ATR .... Advanced Test Reactor
ATT .... Axial Tube Tension
BCC .... Body Centered Cubic
BDBA .... Beyond Design Basis Accident
BF .... Bright Field
BNL .... Brookhaven National Laboratory
BOC .... Beginning of Cycle
BOL .... Beginning of Life
BR2 .... Belgium Reactor - 2
BRR .... BEA Research Reactor

BSE .... Backscatter Electron

BWR .... Boiling Water Reactor

CA .... Cathodic Arc

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

CRAFT .... Collaborative Research on Advanced Fuel Technologies

CS .... Cold Sprayed

CS .... Conventionally Sintered

CSA .... Criticality Safety Analysis

CTRN .... Carbothermic Reduction and Nitriding

CVD .... Chemical Vapor Deposition

CVI .... Chemical Vapor Infiltration

DBA .... Design Basis Accident

DCMS .... DC Magnetron Sputtering

DFT .... Density Functional Theory

DI .... Dual Ion

DIC .... Digital Image Correlation

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

DOE .... Department of Energy

DOT .... Department of Transportation dpa.. Displacements per Atom

DSC.. Differential Scanning Calorimetry

DSS .... Dynamical System Scaling

DTS.. Distributed Temperature Sensing

DZ.. Dark Zone

EATF .... Enhanced Accident Tolerant Fuel

EBR.. Experimental Breeder Reactor

EBSD.. Electron Backscatter Diffraction

ECF .... Energy Coupling Factor

ECR.. Equivalent Clad Reacted

EDM.. Electrical Discharge Machining

EDS .... Energy Dispersive Spectroscopy

EDX .... Energy Dispersion X-Ray

EM .... Electromagnetic

EM2 .... Energy Multiplier Module

EOC .... End of Cycle

EPIC.. EBR-II Pre-Irradiated Conductivity

EPMA.. Electron Probe Micro Analysis

EPRI.. Electric Power Research Institute

ERNI.. Energy-Resolved Neutron Imaging

F&OR.. Functional and Operational Requirements

FAS .... Field Assisted Sintering

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

FGR .... Fission Gas Release

FIDES .... Framework for Irradiation Experiments

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

FS .... Flash Sintering

FWHM .... Full-Width-Half-Max

FY .... Fiscal Year

GA .... General Atomics

GE .... General Electric

GERC .... General Electric Global Research

GET .... Gamma Emission Tomography

GFY .... Government Fiscal Year

GNF .... Global Nuclear Fuels

GS .... Grain Size

GTRF .... Grid-to-Rod-Fretting

HAADF .... High Angle Annular Dark Field

HALEU .... High Assay Low-Enriched Uranium

HBFF .... High Burnup Fuel Fragmentation

HBHE .... High Burnup - Higher Enrichment

HBR .... HB Robinson

HBS .... High Burnup Structures

HBU...High Burnup

HBWR ....Halden Boiling Water Reactor

HD ....High Definition

HEA ....High Entropy Alloy

HERA ....High Burnup Experiments in Reactivity-initiated Accidents

HEU ....Highly Enriched Uranium

HFEF...Hot Fuel Examination Facility

HFIR...High Flux Isotope Reactor

HiPIMS..Hybrid High Power Impulse Magnetron Sputtering

HIPPO ....High Pressure/Preferred Orientation

HRR...Horn Rapids Road

HTO ....High Temperature Oxidation

H/U...Hydrogen-To-Uranium

IAEA ....International Atomic Energy Agency

ICWUPS...In Cell Weld Under Pressure System

IFE ....Institute for Energy Technology

IFEL ....Irradiated Fuels Examination Laboratory

IMCL ....Irradiated Materials Characterization Laboratory

INL ....Idaho National Laboratory

IPF ....Inverse Pole Figure

IQ ....Image Quality

IR...Infrared

IRT...Irradiation Test

ISA...Integrated Safety Analysis

ITEG ....Irradiation Testing Expert Group

JAEA ....Japanese Atomic Energy Agency

JEEP ....Joint Experimental Program

JFCS ....Joint Fuel Cycle Studies

IWD...Work Authorization Documents

KAERI .... Korean Atomic Energy Research Institute

KIT .... Karlsruhe Institute of Technology

KTH .... Royal Institute of Technology

LAMDA .... Low Activation Materials Development & Analysis

LANL .... Los Alamos National Laboratory

LANSCE .... Los Alamos Neutron Science Center

LDRD .... Laboratory Directed Research and Development

LFA .... Laser Flash Analysis

LHGR .... Linear Heat Generation Rate

LOCA .... Loss of Coolant Accident

LOOP .... Loss of Offsite Power

LRM .... Linear Reactivity Model

LTA .... Lead Test Assembly

LTR .... Lead Test Rod

LTR .... Licensing Topical Report

LVDT .... Linear Variable Differential Transformer

LWR .... Light Water Reactor

MARCH .... Minimal Activation Retrievable Capsule Holder

MARCH-SERTTA .... Minimal Activation Retrievable Capsule Holder–Static Environment Rodlet Transient Test Apparatus

MBTS .... Modified Burst Tests

MCNP .... Monte Carlo N-Particle

MFC .... Materials and Fuels Complex

MFCs .... Mass Flow Controllers

MIT .... Massachusetts Institute of Technology

MITR .... Massachusetts Institute of Technology Reactor

MMU .... Manchester Metropolitan University

MOX .... Mixed Oxide

MOXTOP .... Mixed Oxide Transient OverPower nCT....Neutron Computed Tomography

NE....Nuclear Energy

NEA....Nuclear Energy Agency

NEAMS....Nuclear Energy Advanced Modeling and Simulation

NEUP....Nuclear Energy University Project

NFA....Nanostructured Ferritic Alloys

NFD....Nippon Nuclear Fuel Development

NHL....North Holmes Laboratory

NI....Nanoindentation

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

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

OSU....Oregon State University

PBF....Power Burst Facility

PCI....Pellet Clad Interactions

PCMI....Pellet-clad Mechanical Interaction

PCS....Primary Coolant System

PCT....Peak Clad Temperature

PCT....Pressure Composition Temperature

PFIB....Plasma Focused Ion Beam

PI....Principal Investigator

PIE	Post-irradiation Examination
PIRT	Phenomena Identification and Ranking Tables
PNNL	Pacific Northwest National Laboratory
PQD	Post Quench Ductility
PVD	Physical Vapor Deposition
PWR	Pressurized Water Reactor
QA	Quality Assurance
R&D	Research & Development
RAC	Radial, Axial and Capsule
RB	Removeable Beryllium
RCT	Ring Compression Testing
RD&D	Research, Development, and Demonstration
RE	Rare Earth Elements
REWS	Rodlet Endcap Welding System
RHT	Ring Hoop Tension
RIA	Reactivity-Initiated Accident
RITA	Resonance Inspection Techniques and Analysis
RPI	Rensselaer Polytechnic Institute
RT	Room Temperature
RTE	Rapid Turnaround Experiment
RUS	Resonant Ultrasound Spectroscopy
SA	Sensitivity Analysis
SADP	Selected Area Diffraction Pattern
SAR	Safety Analysis Report
SATS	Severe Accident Test Station
SCK•KEN	Studie Centrum Voor Kemenergie: Belgian Nuclear Research Centre
SEM	Scanning Electron Microscopy
SERTTA	Static Environment Rodlet Transient Test Apparatus
SETH	Separate Effects Test Holder

SFR .... Sodium Fast Reactor

SHERMAN .... Sample Handling Environment for Radioactive Materials Analyzed with Neutrons

SiC .... Silicon Carbide

SIMS .... Secondary Ion Mass Spectrometry

SPERT .... Special Power Excursion Reactor Test

SPS .... Spark Plasma Sintering

SRIM .... Stopping and Range of Ions in Matter

SS .... Stainless Steel

SSMT .... Small Scale Mechanical Testing

STEM .... Scanning Transmission Electron Microscopy

TAMU .... Texas A&M University

TD .... Theoretical Density

TE .... Total Elongation

TEM .... Transmission Electron Microscopy

TESB .... TREAT Experiment Support Building

THOR .... Temperature Heat Sink Overpower Response

THOR- aLEU-P2M .... THOR Advanced Low-Enriched Uranium Power-to-Melt

THOR-C .... THOR Commissioning

THOR-EPIC .... EBR-II Pre-Irradiated Conductivity

THOR-M .... THOR Metallic

TGA .... Thermogravimetric Analysis

TIG .... Tungsten Inert Gas

TM .... Temperature Monitor

TMS .... The Mineral, Metal and Materials Society

TMT .... Thermomechanical Treatments

ToF-SIMS .... Time-of-Flight Secondary Ion Mass Spectrometry

TREAT .... Transient Reactor Test Facility

TRISO .... Tri-structural Isotropic

TRU .... Transuranic

TWERL .... TREAT Water Environment Recirculating Loop

TWIST .... Transient Water Irradiation System in TREAT

UC .... Uranium Carbide

UE .... Uniform Elongation

UN .... Uranium Nitride

UofI .... University of Idaho

UQ .... Uncertainty Quantification

U.S. .... United States

USC .... University of South Carolina

UT .... University of Tennessee

UTA .... University of Texas-Austin

UTK .... University of Tennessee-Knoxville

UTS .... Ultimate Tensile Strength

UW .... University of Wisconsin

VCPS .... Visco-plastic Self Consistent

VERA-CS .... Virtual Environment for Reactor Applications – Core Simulator

VTR .... Versatile Test Reactor

VXF .... Vertical Experiment Facilities

Wd .... Watt-days

Wd/MTU .... Watt-days per Metric ton Uranium

WEC .... Westinghouse Electric Company

WQ .... Water Quenching

XCT .... X-ray Computed Tomography

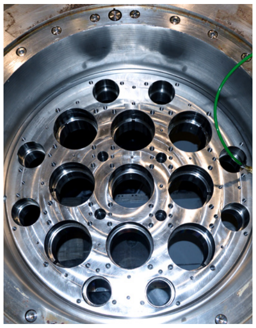
XRD .... X-Ray Diffraction

YH2 .... Yttrium Dihydride

YN .... Yttrium Mononitride

YS .... Yield Strength

5.5 DIVIDER PHOTO CAPTIONS



Page 2

Advanced Test Reactor (ATR) vessel's newly installed top head closure plate with additional experiment penetrations

Page 4

ORNL hot-cells receiving commercially irradiated lead test rods in support of the DOE-NE and Westinghouse collaborative research on Accident Tolerant Fuel and LWR burnup extension.

Page 8

Receipt of HFEF-15 Cask loaded with ATF-R experiment from HFEF into the cask stand at TREAT.

Page 14

Assembly of the THOR-C1 experiment in the Fuels and Applied Sciences Building at INL.

Page 158

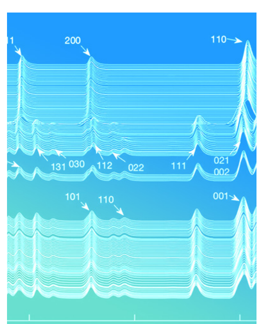
ORNL staff unloading a NAC-LWT cask filled with commercially irradiated lead test rods at the Irradiated Fuels Examination Laboratory (IFEL) hot-cell facility.

Page 164

An inside view of the Advanced Test Reactor (ATR) looking up at the vessel top head/experiment penetrations.

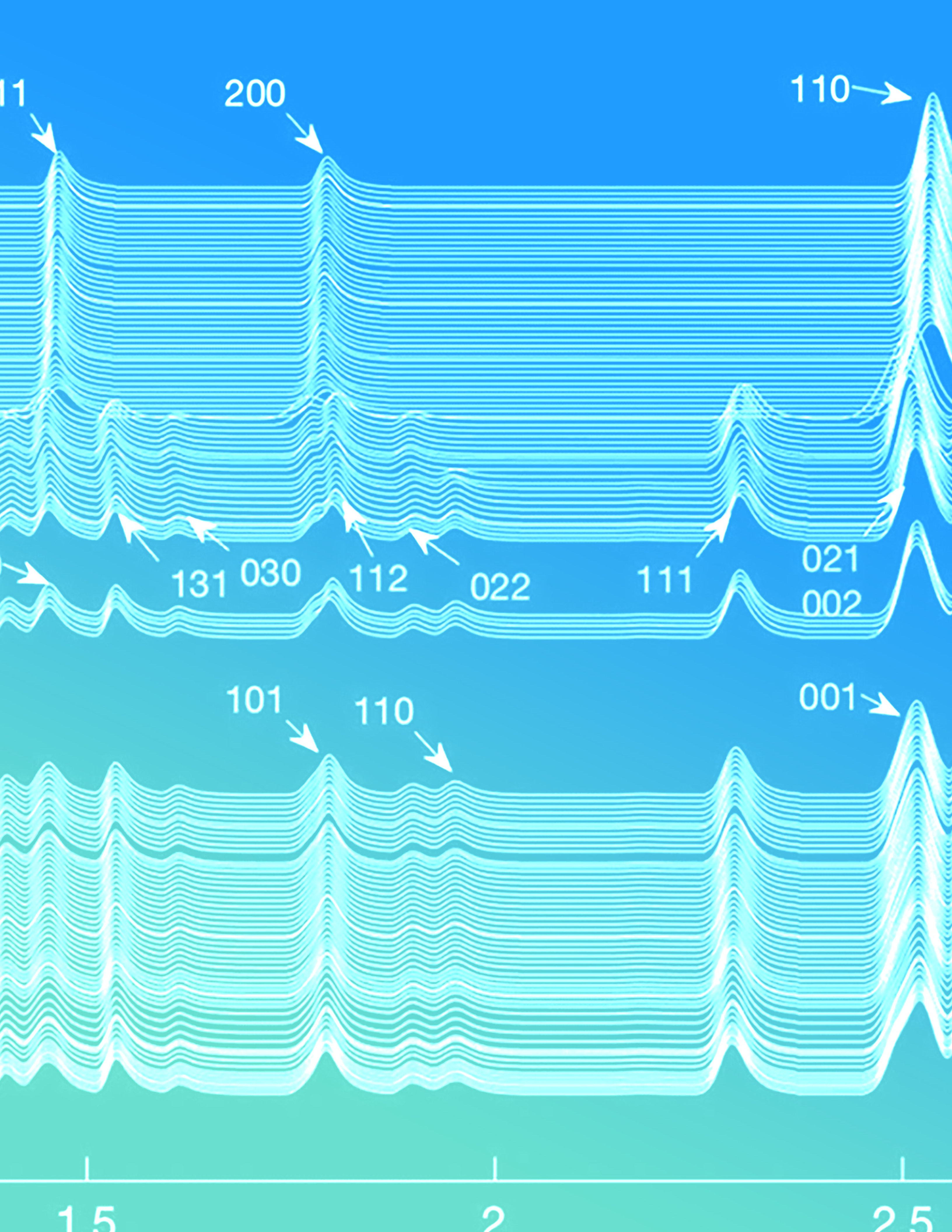
Page 169

Shipment of the HFEF-15 Cask loaded with ATF-R experiment from HFEF to TREAT.



Page 187

Waterfall plot of normalized Rietveld refinements of U-22.5 at %Zr cooled at 1°C/min with an average Chi squared value of 0.90. The hkl planes for various peaks are indicated for the αU, αUZr2, and αUZr phases



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