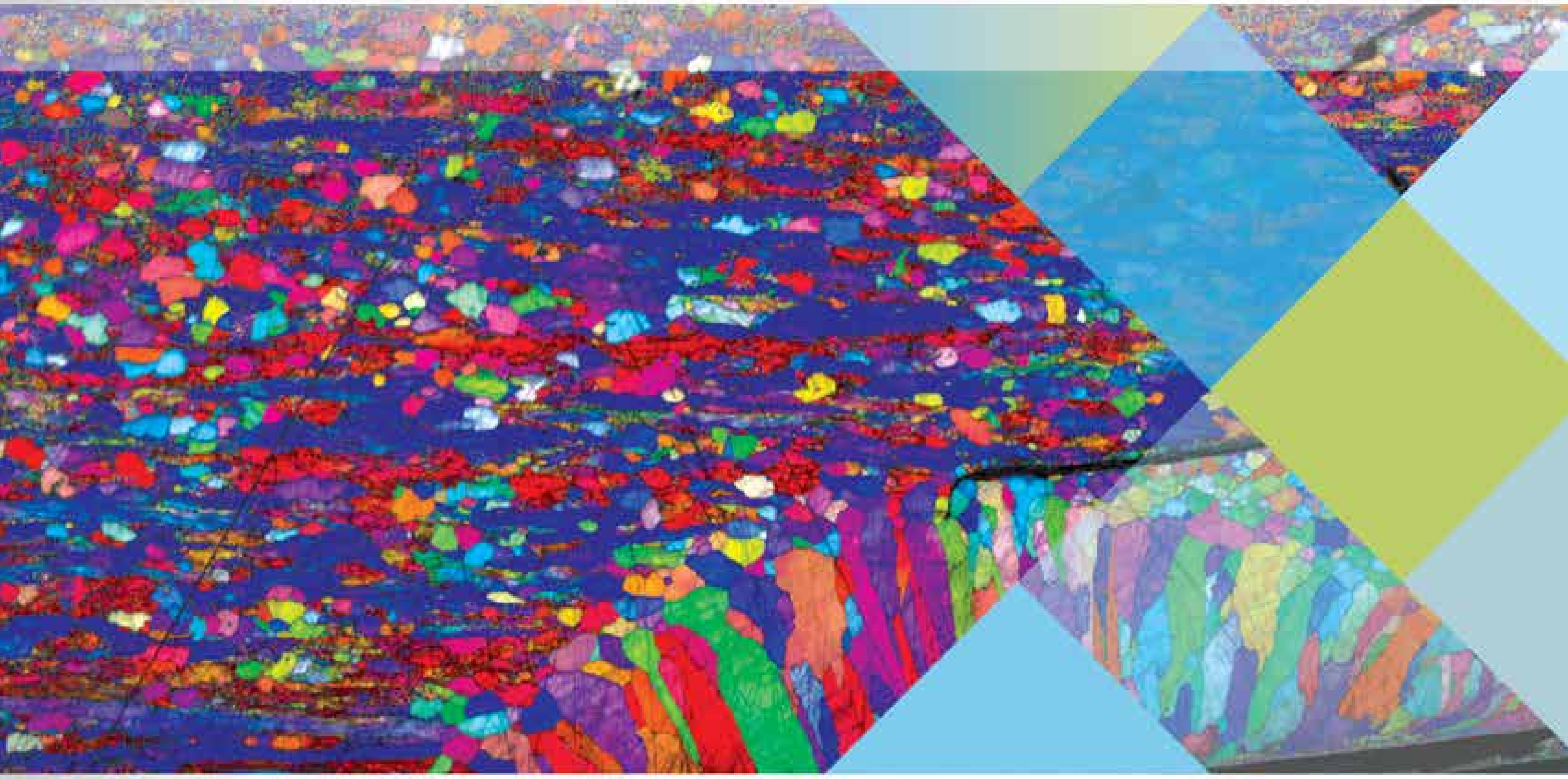
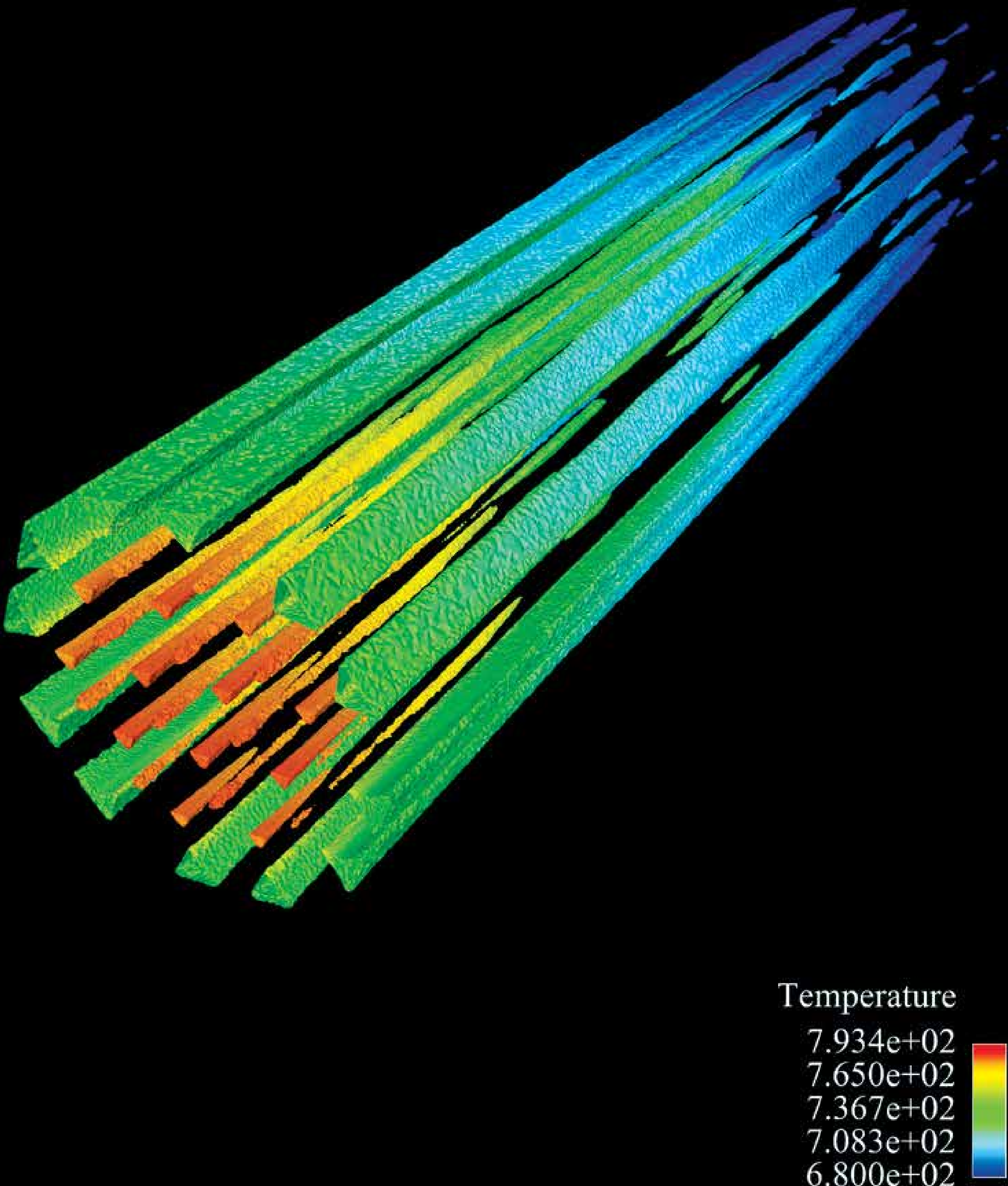
ADVANCED FUELS CAMPAIGN
2017 Accomplishments



Advanced Fuels Campaign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Fuel Cycle Research and Development

Advanced Fuels Campaign 2017 Accomplishments

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TABLE OF CONTENTS

1 AFC MANAGEMENT AND INTEGRATION

1.1 The Advanced Fuels Campaign Team 9

1.2 From the Director (Campaign Overview) 10

1.3 International Collaborations 12

1.4 Showcase Capabilities 16

2 ADVANCED LWR FUEL SYSTEMS

2.1 Accident Tolerant Fuels 38

2.2 High-Performance LWR Fuel Development 56

2.3 Analysis 80

2.4 ATF Cladding and Coatings 84

2.5 Irradiation Testing and PIE Techniques 100

2.6 Transient Testing 112

TABLE OF CONTENTS - *Continued*

3 ADVANCED REACTOR FUELS SYSTEMS

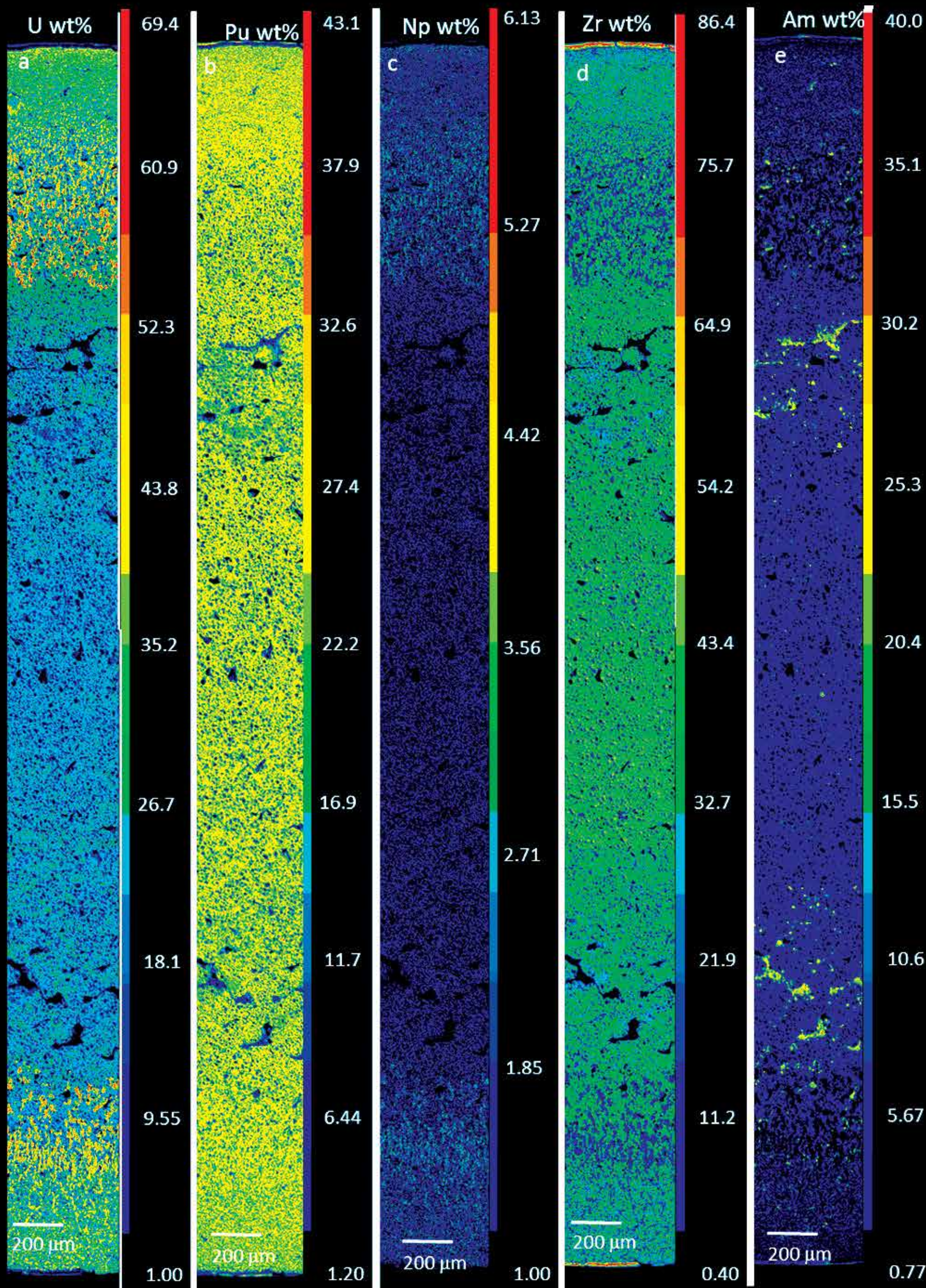
3.1 AR Fuels Development 122
3.2 AR Computational Analysis 150
3.3 AR Core Materials 160
3.4 AR Irradiation Testing & PIE Techniques 178
3.5 Capability Development 196

4 APPENDIX

4.1 Publications 200
4.2 FY-17 Level 2 Milestones 208
4.3 AFC Nuclear Energy University Project (NEUP) Grants 210
4.4 Acronyms 217

ADVANCED REACTOR FUELS SYSTEMS

1.1 The Advanced Fuels Campaign Team
1.2 From the Director
2.3 International Collaborations
2.4 Showcase Capabilities



U wt%
a
69.4
60.9
52.3
43.8
35.2
26.7
18.1
9.55
1.00
200 µm
Pu wt%
b
43.1
37.9
32.6
27.4
22.2
16.9
11.7
6.44
1.20
200 µm
Np wt%
c
6.13
5.27
4.42
3.56
2.71
1.85
1.00
200 µm
Zr wt%
d
86.4
75.7
64.9
54.2
43.4
32.7
21.9
11.2
0.40
200 µm
Am wt%
e
40.0
35.1
30.2
25.3
20.4
15.5
10.6
5.67
0.77
200 µm

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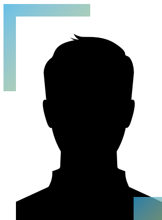

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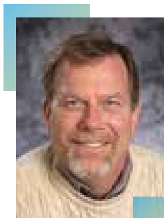

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1.2 FROM THE DIRECTOR

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The mission of the Advanced Fuels Campaign (AFC) is to perform research, development and demonstration (RD&D) activities for advanced fuel forms (including clad- ding) to boost the performance and safety of the nation's current and future reactors; enhance proliferation resistance of nuclear fuel; effectively utilize nuclear energy resources; and address the longer-term waste management challenges. This includes development of a state-of-the art research and development (R&D) infrastructure to support the use of a "goal-oriented, science- based approach."

In support of the Nuclear Technology Research and Development (NTRD) program, AFC is responsible for developing advanced fuel technologies to augment the current and future fuel cycle options.

In philosophy, AFC pursues a "goal-oriented, science- based approach" aimed at a fundamental understanding of fuel and cladding fabrication methods and performance under irradiation, enabling the pursuit of multiple fuel forms for future fuel cycle options to support qualification and licensing of technologies. This approach includes fundamental experiments, theory, and advanced modeling and simulation to ultimately support integral qualification and licensing activities. The modeling and simulation activities for fuel performance are carried out under the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program, which is closely coordinated with AFC. In this report, the word "fuel" is used generically to include fuels, targets, and their associated cladding materials.

In FY-17, many AFC goals and initiatives have achieved success and have progressed significantly..

The campaign management staff is responsible coordinating and integrating the activities performed by many organizations, DOE's national laboratories, U.S. industry partners, and universities funded through the DOE Nuclear Energy University Projects (NEUP) program. The campaign is responsible for developing and executing international collaborations on nuclear fuel research and development, primarily with France, Japan, the European Union, Republic of Korea, and China, as well as various working groups and expert group activities in the Organization for Economic Cooperation and Development Nuclear Energy Agency (OECD-NEA) and the International Atomic Energy Agency (IAEA). Our goal in the international community is to achieve a high level of coordination across the nuclear technology community and to improve the competitiveness of U.S. nuclear technology base. Three industry-led projects made significant progress in fuels and materials development. All are closely integrated in the AFC accident-tolerant fuels (ATF) program. Accomplishments made during FY-17 are highlighted in this report, which focuses on completed work and results. The key FY-17 technical area outcomes include:

KEY FY-17 Technical Outcomes	
Phase II, year one of the Industry led ATF development and qualification projects	✓
PWR condition test loop in the ATR supporting ATF technologies	✓
Nuclear Quality Assurance Level-1 (NQA-1) data archiving system	✓
Standard for nuclear technology material property handbooks and the first handbook versions for U_3Si_2, FeCrAl alloy, and SiC material systems	✓
Scanning Electron Microprobe Analysis instrument (EPMA) in a fully shielded environment allowing the microscale level chemical and structural evaluation of highly irradiated nuclear fuel samples	✓
First EPMA analysis on a full cross section sample from the FUTURIX-FTA irradiation experiment	✓
First draft comparison of thermal verses fast spectrum irradiation behavior of nuclear fuels	✓
Co-extrusion of a zirconium lined ferretic martensitic fuel pin with a U-10Zr fuel alloy	✓
Severe Accident Test Station (SATS) Loss of Coolant Accident (LOCA) test station in a shielded hot cell allowing evaluation of fully irradiated high dose nuclear fuel samples	✓
Remote casting fabrication of high dose recycle metallic nuclear fuel	✓
PIE on the first SiC samples irradiated in the HFIR high radial heat flux experiment	✓
Continued irradiation of ATF technologies in the ATR, HFIR, and MIT reactors	✓
Designed irradiation loops in the Norwegian Halden reactor	✓
PIE on the initial low dose ATF fuel technologies irradiated in the ATR and initiated PIE on the first samples to reach medium dose	✓
C26M FeCrAl tubing fabricated for GE lead test rods in the Hatch nuclear power reactor	✓
U_3Si_2 pellets fabricated for Westinghouse SiC and coated Zr cladding experiments in ATR	✓

1.3 INTERNATIONAL COLLABORATIONS

The Advanced Fuels Campaign (AFC) researchers are active in international collaborations with Korea, France, Japan, China, EURATOM, and OECD-NEA. These interactions and collaborations are managed through a combination of participation in Generation IV Global International Forum projects, International Nuclear Energy Research Initiative (INERI) projects, and participation in bilateral and trilateral government-to-government agreements. Active work has been executed in collaboration with the Korean Atomic Energy Research Institute (KAERI) and with the Halden reactor operated by the Institute for Energy Technology (IFE) laboratory of Norway.

US/Japan CNWG Collaboration on Advanced Fuels

With respect to fast reactor driver and transmutation fuel development, for the last ~10 years the US has prioritized resources towards advancing the technical readiness level of metallic fuels. In order to stay current and relevant in oxide fuels, the Advanced Fuels Campaign uses international collaborations to leverage more limited research for these fuels. Accordingly, cooperative research on mixed oxide (MOX) fuel is being performed by the US and Japan under the Fuel Cycle R&D and Waste Management Sub-Working Group within the Civil Nuclear Energy Working Group (CNWG) bilateral arrangement.

Coordinated activities on the properties, performance and analysis of advanced oxide fuels are being performed by DOE and JAEA laboratories. Specifically, the effort consists of coordinated activities in the areas of basic fuel and cladding properties, post irradiation examination technique development, and gathering of additional data on irradiation performance from completed MOX and minor actinide (MA)-bearing MOX tests. These activities in turn support the overall goal of jointly developing a MA-MOX fuel performance code using the BISON framework developed at the Idaho National Laboratory. Under this effort additional fuel irradiation tests, for instance in the Joyo or TREAT reactors, will also be proposed and planned as data needs are identified during fuel performance code development.

In FY17, technical expert meetings were held in Japan and in the US at INL to advance specific tasks on basic properties of fuels and cladding, development of PIE data, and modeling and simulation of irradiated transmutation MOX fuel. Key accomplishments derived from the expert coordination meetings included providing initial training at INL on Bison model development for a JAEA researcher in early 2017.

US-France Advanced Nuclear Fuels R&D Collaboration

DOE and CEA continued to work toward definition and design of the Americium-Bearing Blanket (AmBB) experiment proposed for irradiation in the Advanced Test Reactor at INL. The AmBB concept proposes to investigate the possibility of transmuting Americium in the breeding blankets (or reflector region) of future sodium fast reactors and would put 10-15% Am into either depleted UO_2 or depleted U-Zr blanket rods. Such AmBB rods would operate in low power and low temperature regimes for extended periods of time where no performance data currently exists. On the US side, INL produced 10 g of purified Am metal that is designated for use in fabricating the U-15Am-10Zr rodlets that would be contributed to this experiment. However, at the end of FY17, the agreement between DOE and CEA for this experiment has still not been signed.

OECD-NEA Expert Group on Accident Tolerant Fuels for LWRs

The Organization for Economic Cooperation and Development / Nuclear Energy Agency (OECD/ NEA) Nuclear Science Committee approved the formation of an Expert Group on Accident Tolerant Fuel (ATF) for LWRs (EGATFL) in 2014. Chaired by Kemal Pasamehmetoglu, INL Associate Laboratory Director for Nuclear Science and Technology, the mandate for the EGATFL defines work under three task forces: (1) Systems Assessment,

(2) Cladding and Core Materials, and (3) Fuel Concepts. Scope for the Systems Assessment task force (TF1) includes definition of evaluation metrics for ATF, technology readiness level definition, definition of illustrative scenarios for ATF evaluation, and identification of fuel performance and system codes applicable to ATF evalua-


ADVANCED TEST REACTOR

tion. The Cladding and Core Materials (TF_2) and Fuel Concepts (TF_3) task forces are working to identify gaps and needs for modeling and experimental demonstration; define key properties of interest; identify the data necessary to perform concept evaluation under normal conditions and illustrative scenarios; identify available infrastructure (internationally) to support experimental needs; and make recommendations on priorities. Where possible, considering proprietary and other export restrictions (e.g., International Traffic in Arms Regulations), the Expert Group will facilitate the sharing of data and lessons learned across the international group membership.

The Systems Assessment task force is chaired by Shannon Bragg-Sitton (Idaho National Laboratory [INL], U.S.), the Cladding Task Force is chaired by Marie Moatti (Electricite de France [EdF], France), and the

Fuels Task Force is chaired by Masaki Kurata (Japan Atomic Energy Agency [JAEA], Japan). The original Expert Group mandate was established for June 2014 to June 2016. In April 2016 the Expert Group voted to extend the mandate one additional year to June 2017 in order to complete the task force deliverables; this request was subsequently approved by the Nuclear Science Committee. All three task forces are expected to publish their respective deliverable reports in Fall 2017 following the final task force meeting in November 2017.

IAEA Coordinated Research Project on Accident Tolerant Fuels for LWRs (ACTOF)

The Fuel Performance and Technology Technical Working Group (FPTTWG) within the International Atomic Energy Agency (IAEA) established a coordinated research project (CRP) on ATF for LWRs (ACTOF) in 2015 (CRP-T12030).

CRPs are typically initiated with a technical workshop, followed by a solicitation for proposals on potential projects under the CRP. Each CRP runs approximately 4 years, with a joint plan for the work established based on proposals submitted by various member institutes/organizations. Studies under that joint plan are typically managed through a series of consultants meetings and small contracts. A technical meeting on ATF for LWRs was initially held in October 2014 at ORNL to launch the ACTOF CRP. Focused on nuclear fuel performance and safety, the objective of ACTOF is to support options for the development of nuclear fuel with improved tolerance of severe accident conditions through experiments to acquire data on new fuel and cladding materials and modeling of new fuel designs using ATF materials. ACTOF is expected to provide information to IAEA Member States to support decision making on ATF choices and to provide data, analyses, and advanced techniques to understand and predict the integral performance of ATF designs under normal, transient, and severe accident conditions. The first Research Coordination Meeting (RCM) on ACTOF was held in November 2015 and was attended by 14 organizations across 11 countries, including Westinghouse and Battelle Energy Alliance (BEA, with INL as the participating laboratory) in the U.S. The Westinghouse contribution to the CRP will include information associated with the design and development of U_3Si_2 and UN-U_3Si_2 composite fuel, SiC composite cladding, Ti_2AlC coated Zr cladding, and SiC wrapped Zr cladding. The INL contribution will provide implementation of material models and properties for FeCrAl and U_3Si_2 in INL's fuel performance code BISON, validate models against experiments, perform simulations of fuel rod behavior with ATF cladding and/or fuel (under normal and accident conditions), and perform sensitivity studies on critical mate- rial properties using BISON interfaced with DOE uncertainty quantification tools. Additional participants currently include Karlsruhe Institute of Technology (KIT) in Germany,

VTT Technical Research Center in Finland, A.A. Bochvar Institue

(VNIINM) in Russia, Bhabha Atomic Research Centre (BARC) in India, and Korea Atomic Energy Research Institute (KAERI) in Korea. Proposals will still be accepted until the next RCM, tentatively scheduled for Sprint/Summer 2017.

1.4 SHOWCASE CAPABILITIES

The INL has a complete engineering scale extrusion line, capable of extruding bare uranium alloys on the kilogram scale.

Engineering-scale Extrusion Laboratory for Metallic Fuels

The most recent engineering scale transmutation fuel fabrication was the operation of EBR-II, in which all of the fuel was injection cast. More recently, fuel used for the AFC series of irradiation tests has also been cast. However, other methods of fabrication have been used for metallic fuel product as well, such as extrusion. Over the years much of the uranium extrusion capabilities within the DOE complex have been shut down and lost. Recently a 150 ton extrusion press as well as the needed support equipment has been installed and operated at the Idaho National Laboratory. This extrusion line will now allow engineering scale extrusions to be performed on experimental uranium based metallic fuels.

Figure 1. Installed engineering scale extrusion line.

Project Description:

The extrusion capability has been developed by leveraging funding from related programs. The extrusion line is made up of several discreet pieces of equipment; billet caster, CNC lathe, salt bath, extrusion press and run out table, draw bench, and a 4-die swager. The billet casting system is a vacuum induction melting system used to alloy and cast the desired billets for extrusion. The furnace has a maximum temperature of 1650°C and is attached to a molecular turbo vacuum pump. Billets to be cast are approximately 37 mm diameter and 150 mm in length are cast for extrusion, although depending on the mold other shapes and sizes are also possible. After casting the uranium alloy billet is machined to final length and diameter using a CNC lathe. After machining, the billet is heated to extrusion temperatures using a salt bath. The salt bath uses a lithium-potassium carbonate salt and can be heated to as high as 1000°C. After heating the billet is moved to the Butech-Bliss 150 ton extrusion press. Extruded diameters as small as approximately 6 mm have been extruded. Included in the engineering scale extrusion line is also a Fenn 10-8

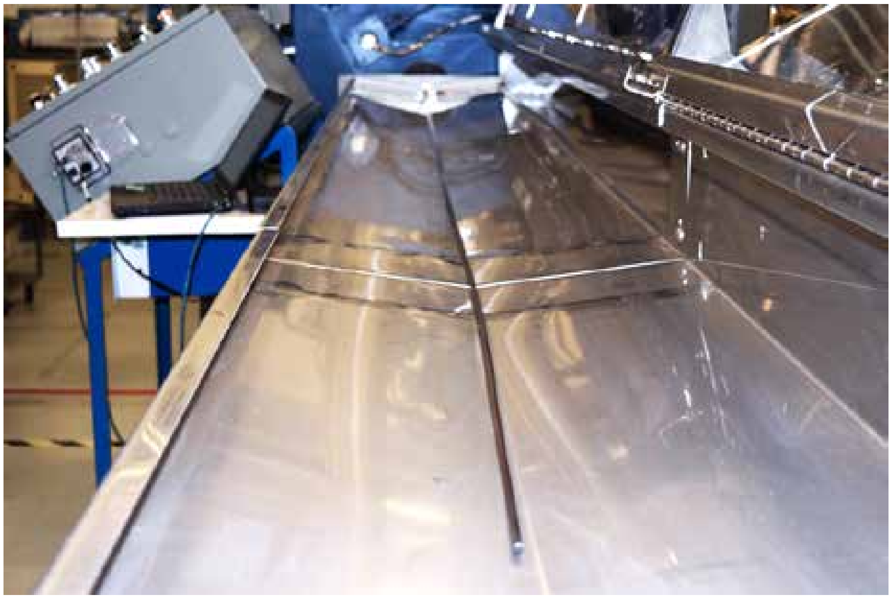


Figure 2. As-extruded rod in the extrusion press run out table.

hydraulic draw bench with a 3 meter stroke, and a maximum pulling force of 8000 lbs and a Fenn 4F-4 4-die swager. The draw bench can be used to either stretch straighten the extruded product or could also be used to further reduce the diameter of the fuel. The swager was installed in order to reduce the diameter of one end of the extruded product to allow it to be inserted through the drawing die. It could also be used for reduce the diameter of the fuel over the entire length if desired. A quench furnace has also been installed with hot zone of approximately 60 cm. The furnace is currently designed to heat under a vacuum atmosphere, but as appropriate can be backfilled with inert gas and the sample quench in a liquid quenching media, such as water. With these capabilities uranium alloys can routinely be extruded on the kilogram scale.

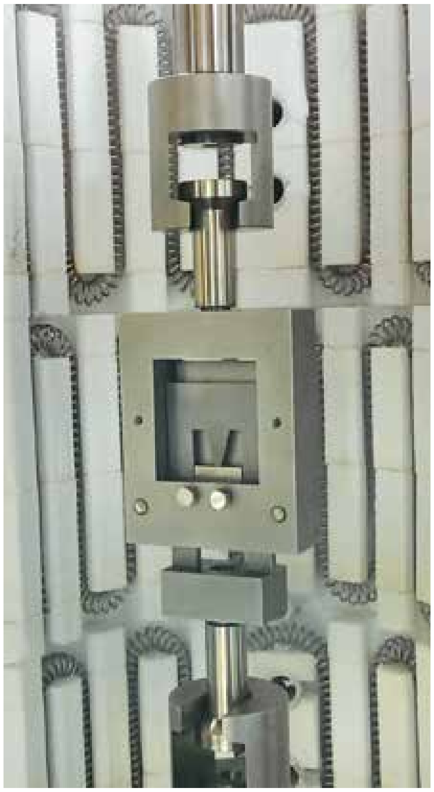
Accomplishments:

All of the equipment making up the extrusion line has been installed and is currently operating. A total of 62 uranium or uranium alloys extrusions have been performed. Figure 1 shows the extrusion press installed in the facility. The extrusion line can be used for either solid fuel or annular fuel forms. Solid extrusions have been carried using uranium, U-6Zr, and U-10Zr alloys. Using the current tooling the starting diameter of the billet is 34.9 mm, with the extruded diameter being as low as 6 mm. A typical example of an extruded rod is shown in Figure 2. Annular copper surrogates have also been extruded with an outside diameter of 8.6 mm and an inside diameter of 6.1 mm.

Radiochemical Processing Laboratory - Pacific Northwest National Laboratory

Figure 1. Fracture toughness testing for irradiated materials is performed in MEC-2 hotcell at RPL (right) after radioactive specimens are received and prepared for testing in a sister hotcell MEC-1. This facility has been established for simplified and streamlined mechanical testing capabilities, including fracture, fatigue and tensile tests.

Fracture toughness testing capability is among the many post-irradiation testing and evaluation capabilities at the Radiochemical Processing Laboratory (RPL; Figure 1) at PNNL, which is a Hazard Category II Non-Reactor Nuclear Facility and is located on the Hanford site in the south central region of Washington State. The fracture toughness testing capability, which also needs fatigue cracking to introduce a sharp precrack to the test specimen, has been established on an Instron® 8801 servohydraulic testing system. This testing system is an omni-purpose mechanical testing machine and its frame is installed in the MEC-2 hotcell and equipped with a three-zone Mellen furnace operated in inert gas environment.



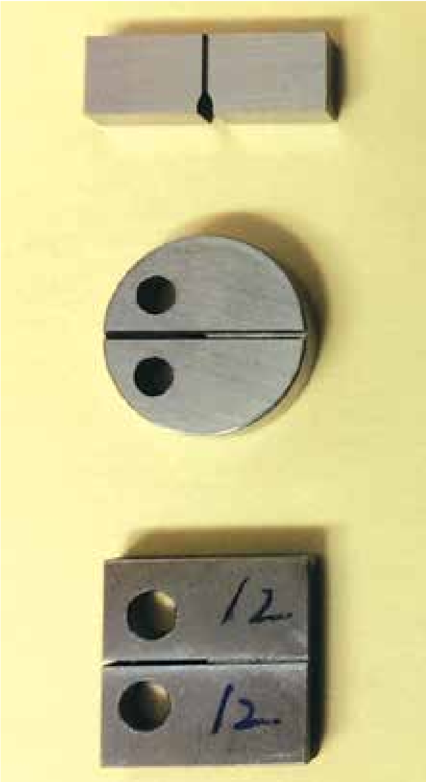
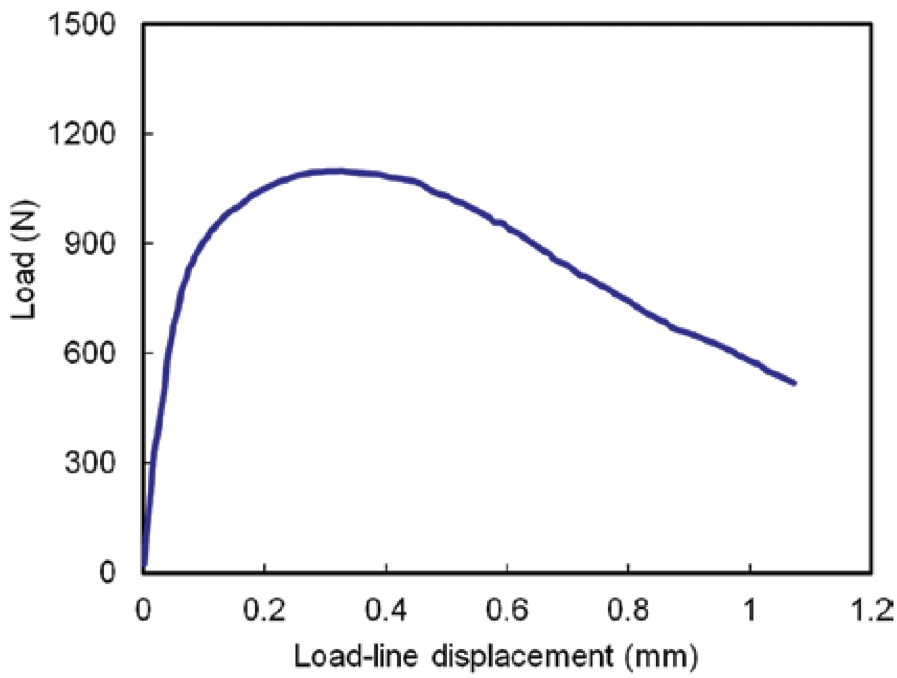


Figure 2. A cradle-type fracture testing grip loaded on the load train within three-zone temperature control furnace (left). Three miniature fracture specimens that have been used for irradiated materials testing in AFC campaign (right): the rectangular bend bar (15 mm L) and square compact tension (12.5 x 12.5 mm) specimens were developed in AFC campaign, while the 12.5 mm diameter disk compact tension specimen is commonly used in the nuclear materials community.

The load train of the mechanical testing system can hold various cradle-type grips that uniquely developed for highly efficient mechanical tests, including uniaxial tensile, high and low cycle fatigue, fracture resistance (J-R), and bending tests in controlled environments (Figure 2). Among these capabilities, the procedure of static J-R fracture testing and fracture toughness calculation demands the most complicated steps and significant background knowledge. As part of the Advanced Fuel Cycle (AFC) campaign, a streamlined procedure from specimen loading to fracture toughness determination has been established for testing and evaluation of irradiated materials in shielded facilities. In pursue of fast and efficient fracture toughness testing, the precracking and stable fracture testing steps were highly simplified: no external gage attachment is needed and a unique crack growth monitoring method is used. Further, the procedure to calculate fracture resistance (J-R) curve from test data was modified accordingly. In the simplified fracture testing, therefore, only two components of data, i.e., a load versus load-line displacement curve and a fracture surface photograph, are required to be obtained for calculation of fracture resistance (J-R) curve and fracture toughness parameters (K_{JQ} and J_Q).

This fracture toughness testing and evaluation technology (Figure 3) was proven to be particularly effective for processing irradiated miniature specimens, which have been widely used in post-irradiation evaluation of nuclear materials. In particular, the new procedure not using attached displacement gage enabled us to test high dose specimens at high temperatures, where obtaining correct crack growth data with a traditional testing method is almost impossible because of high friction at specimen-jig contacts and difficulties in attaching displacement gage to miniature specimens. Because of its excellent efficiency, the new fracture testing and evaluation procedure is currently used for nonirradiated standard sized specimens as well as for irradiated miniature specimens.



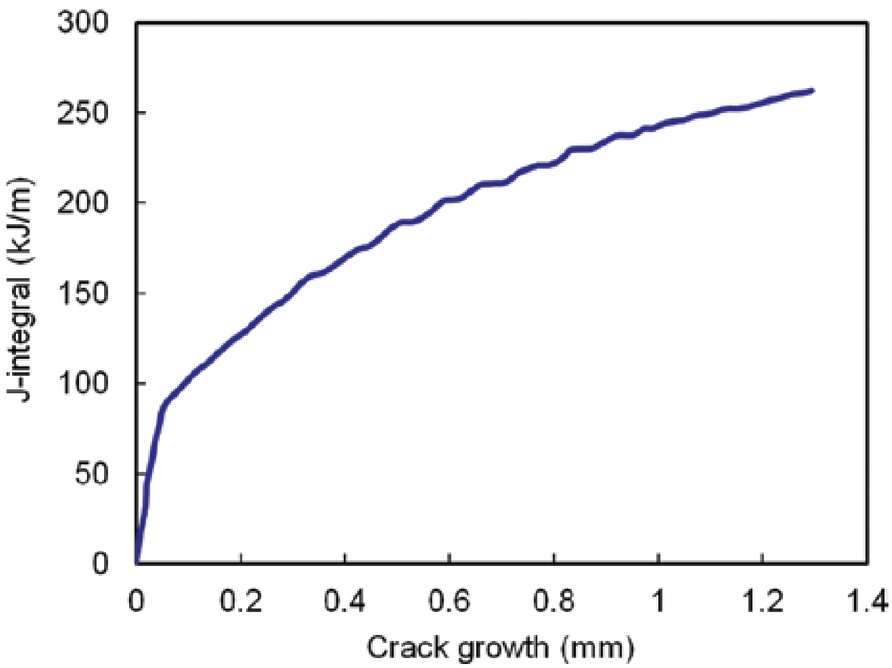


Figure 3. Testing and evaluation procedures to obtain fracture toughness were simplified for hotcell facilities. The load versus load-line displacement record (upper left) and fracture surface photograph (lower left) are the only test data needed to calculate the final output or fracture resistance (J-R) curve (upper right).

In-Cell Installation and Demonstration of the Severe Accident Test Station

Lab	Heating Method	Temperature Control	Temperature Measurement	Out of Cell	In Cell	Fuel	Target Maximum Temperature (°C)
ORNL SATS	Radiant + (resistance)	Computer	Thermocouples	**Yes**	**Yes**	**Yes**	1200 **(1700)**
ANL	Radiant	Computer	Thermocouples	Yes	Yes	**No**	1200
CEA-Saclay	Resistance	Manual	Pre-set	Yes	**No**	**No**	1200
Halden	Reactor + heaters	Pre-set	Thermocouples	**No**	**No**	Yes	1200
JAEA	Radiant	Computer	Thermocouples	Yes	Yes	**No**	1200
Studsvik	Radiant	Computer	Thermocouples	**No**	Yes	Yes	1200

Table 1. Summary of domestic and international accident testing capabilities.

A Severe Accident Test Station (SATS) capable of examining the oxidation kinetics and accident response of irradiated fuel and cladding materials for design basis accident (DBA) and beyond design basis accident (BDBA) scenarios has been successfully installed and demonstrated in the Irradiated Fuels Examination Laboratory, a hot cell facility at Oak Ridge National Laboratory. Two test station modules provide various temperature profiles, steam, and the thermal shock conditions necessary for integral loss-of-coolant accident (LOCA) testing, defueled oxidation quench testing, and high-temperature BDBA testing. Installation of the SATS system restores the domestic capability to examine postulated and extended LOCA conditions on spent fuel and cladding and provides a platform for evaluating advanced fuel and accident-tolerant fuel cladding concepts.

Project Description:

The high-temperature steam oxidation behavior of zirconium alloy cladding under design basis loss-of-coolant accidents (LOCA) and over a broader set of conditions has been studied at several facilities around the world. Currently only a limited number of facilities domestically and internationally can examine the oxidation behavior and post-quench ductility of cladding materials (Table 1). DBA evaluation of Zr-based alloys using the LOCA Integral Test system was originally developed at Argonne National Laboratory (ANL) and provided significant contribution to the US Nuclear Regulatory Commission test criteria for LOCAs. After the ANL unit was decommissioned, the domestic capability was lost. In the aftermath of the Tōhoku earthquake and tsunami and the resulting Fukushima Daiichi nuclear accident, research goals have been directed towards developing novel materials with enhanced accident tolerance in light water reactor designs. These materials require evaluation in comparison to current UO_2/Zr-based alloy fuel systems to understand the DBA response and enhanced safety margins of BDBA scenarios.

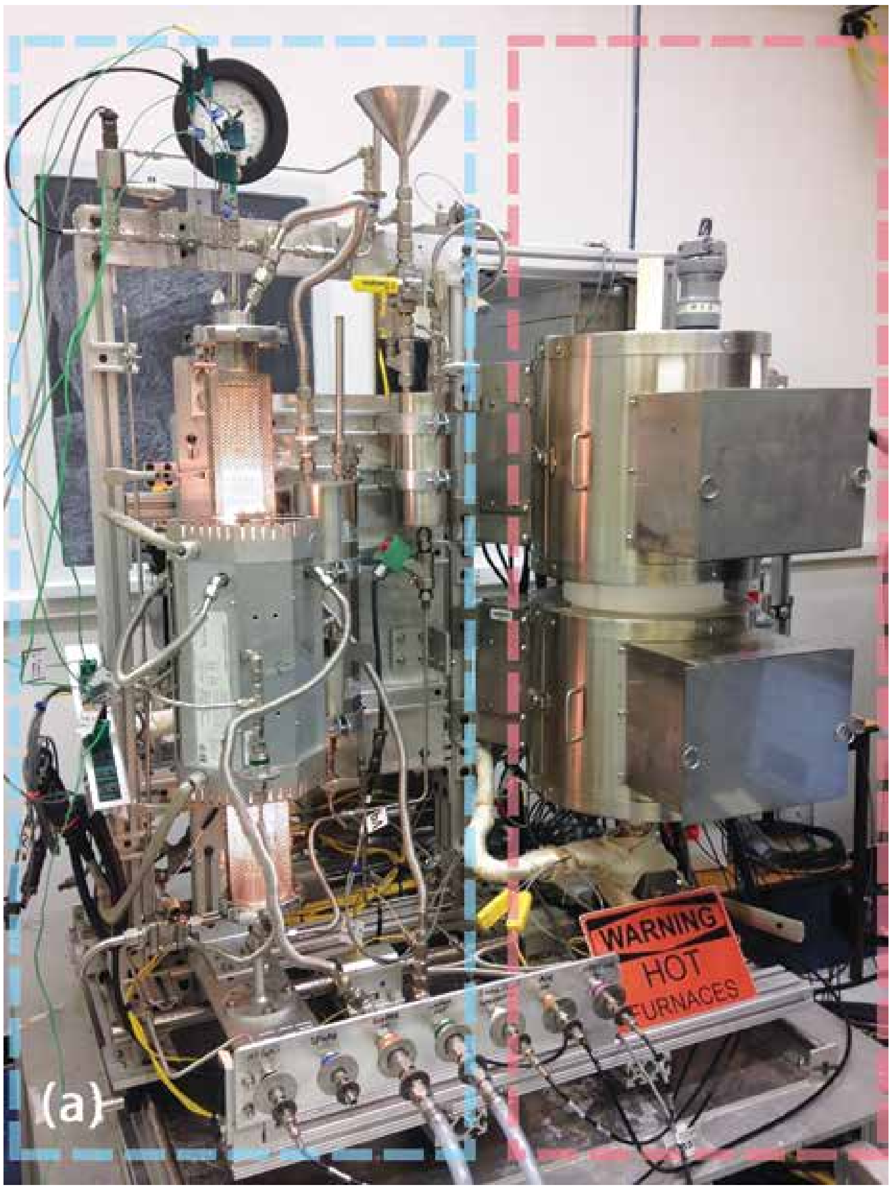


Figure 1. (a) The Severe Accident Test Station is a single unit with two modules, one for design basis accident integral LOCA testing (outlined in blue) and one for beyond design basis accident high-temperature testing (outline in red). (b) LOCA integral test apparatus.

Accomplishments:

Figure 1 provides an overview of the test station prior to in-cell insertion; it consists of two modules on a single test stand. The integral LOCA test apparatus is outlined in blue, while the high-temperature test furnace is outlined in red. The SATS system was initially deployed out-of-cell to examine nonirradiated materials. The same modules were then replicated and modified for in-cell operation (i.e., in ORNL's fuel hot cell in Building 3525) to examine irradiated fuels and materials. Throughout the design of the SATS system, operational constraints within the hot cell (e.g., spatial and other constraints from the use of manipulators to operate the unit, and limits on the volume and types of gases released from the system) have been thoroughly captured to enable an efficient transition from the out-of-cell module to the in-cell capability. Table 2 summarizes the range of test parameters for each module and test type.

Before moving the SATS system into the hot cell, many thermal verification LOCA tests were performed to aid in optimizing the apparatus and the test train for remote operation in the hot cell. Tests were conducted with specimens made from 17×17 pressurized water reactor Zircaloy-4 (300 mm long, 9.50 mm outer diameter, 0.570 mm wall thickness) filled with dense zirconia pellets leaving a cold radial gap of ~0.1 mm. Figure 2 shows test samples from four tests at 1200°C, with the only difference being the internal pressure applied (Results: Zr4-19 no internal pressure and no burst; Zr4-23 pressurized to 4.14 Mpa, burst occurred at 866°C, max strain 27%; Test Zr4-21 pressurized to 6.21 Mpa, burst occurred at 826°C, max strain 37%; and Test Zr4-22 pressurized to 8.27 Mpa, burst at 790°C, max strain 47%). Verification test results show good agreement with prior ANL LOCA test results and provided confidence to move forward with in-cell installation. Figure 3 shows the SATS system after being installed in-cell.

A full sequence in-cell LOCA demonstration test was conducted with the SATS system on August 7, 2017. A LOCA integral test with 300-mm-long samples consists of the following sequential steps: stabilize temperature, internal pressure, and steam flow at 300°C; ramp the temperature (~5°C/s) through ballooning and burst to ~1200°C; hold at 1200°C for 1–5 min; slow cool (~3°C/s) to 700–800°C; and finally, water quench. Figure 4 shows the temperature and pressure profile for integral LOCA demonstration test Zr4-26, performed on the in cell SATS.

	DBA Module		BDBA Module
	LOCA Integral Test	Oxidation-Quench Test	High-Temperature Test Station
Sample Spec	Fueled Rod	Defueled Rod	Rod or Coupon with 3mm hole
Sample Segment, mm	~200–300	~25–50	~25–50
Pressure, MPa	~8, max 20	0.1	0.1
Max Temp, °C	1200	1200	**1700**
Heating Rate, °C/s	5	5; max 20	.25; max .33
Steam Flow Rate, $mg/cm^2 \cdot s$	~5.7	~5.7	3.0–7.0
Gas Environment	Steam or Argon	Steam or Argon	Steam or Argon
Quench, °C	@ 20–800	@ 20–800	None
Quench Condition	Rising Water Around Sample	Rising Water Around Sample	None
Quench Flow Rate, mm/s	≥15	≥15	None
Test Time, min	≥30	≥30	Multiple Days

Table 2. Summary of test parameters for each module and test type.

Figure 2. Four post-LOCA samples tested at 1200YC with internal pressurization of 0.0, 4.14, 6.21, and 8.27 MPa.

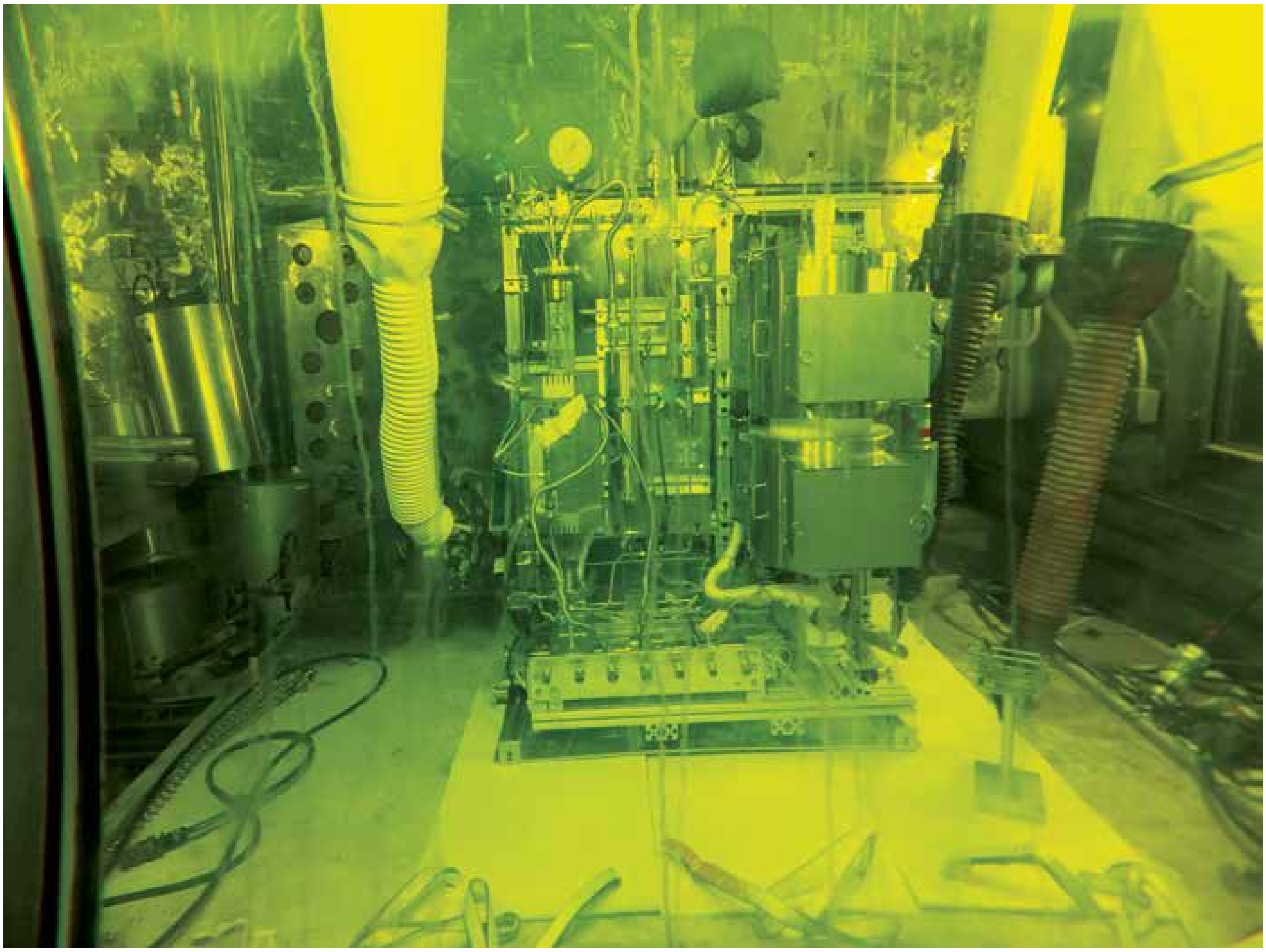


Figure 3. SATS in the main hot cell of the IFEL..

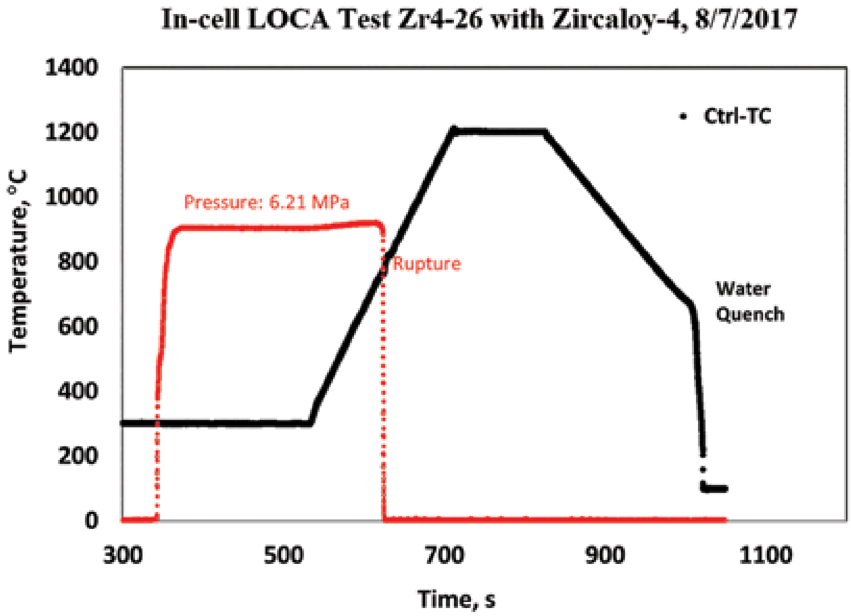


Figure 4. Pressure and temperature histories for in-cell LOCA integral test Zr4-26.

A Severe Accident Test Station (SATS) capable of examining the oxidation kinetics and accident response of irradiated fuel and cladding materials for design basis accident (DBA) and beyond design basis accident (BDBA) scenarios has been successfully installed and demonstrated in the Irradiated Fuels Examination Laboratory, a hot cell facility at Oak Ridge National Laboratory.

Nuclear Fuel Characterization with Pulsed Neutrons

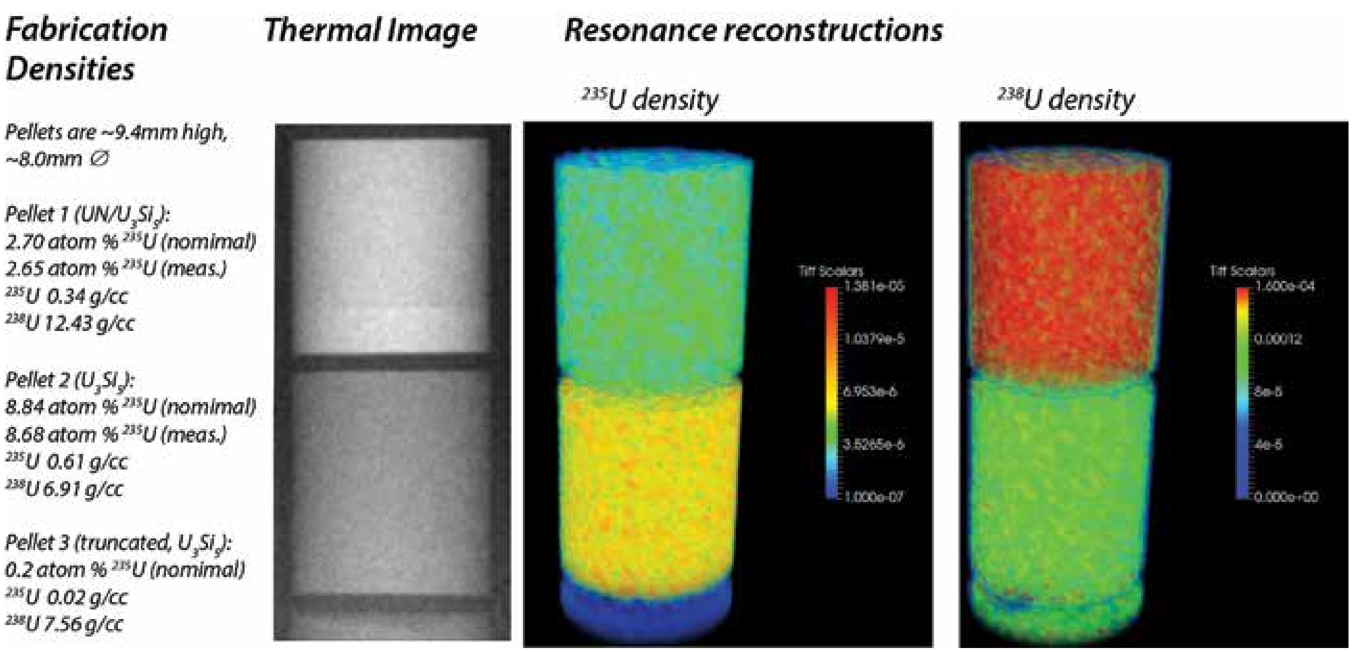


Figure 1. Neutron computed tomography (CT) of UN/U3Si5 and U3Si5 fuel pellets with different enrichment levels. While the thermal neutron tomography, as available at a reactor, does not distinguish the differences in enrichment levels of the middle and lower pellet (8.84% and 0.2% enrichment, respectively), the reconstruction of 235U and 238U densities clearly provides this information.

Neutrons are a unique probe to interrogate entire fuel pellets, providing rich information on parameters ranging from crystallographic parameters (e.g. atomic positions, thermal motion parameters, strains), to meso-scale parameters (e.g. dislocation densities), to micro-structural features (e.g. weight or volume fractions of phases, grain orientation distribution or texture), to macroscopic features (e.g. dimensions, voids, cracks, isotope densities). Furthermore, the ability to penetrate shielding containers offers the opportunity to conduct this characterization on highly radioactive materials, ultimately providing detailed characterization of the same specimen pre- and post-irradiation. Pulsed neutrons, such as provided by the LANSCE spallation neutron source, provide distinct benefits over the polychromatic neutron flux provided by a reactor source. The Advanced Post-Irradiation Examination (APIE) efforts at LANL have developed novel techniques, such as energy-resolved neutron imaging, and applied both novel and established techniques, such as neutron diffraction, to nuclear fuel forms ranging from ceramic fuels (oxides, nitrides, carbides, silicides) to metallic fuels (transmutation fuels, CONVERT fuels). In the future, availability of intense pulsed neutron sources may allow to apply the developed techniques pool-side at a research reactor. Here we summarize these efforts.

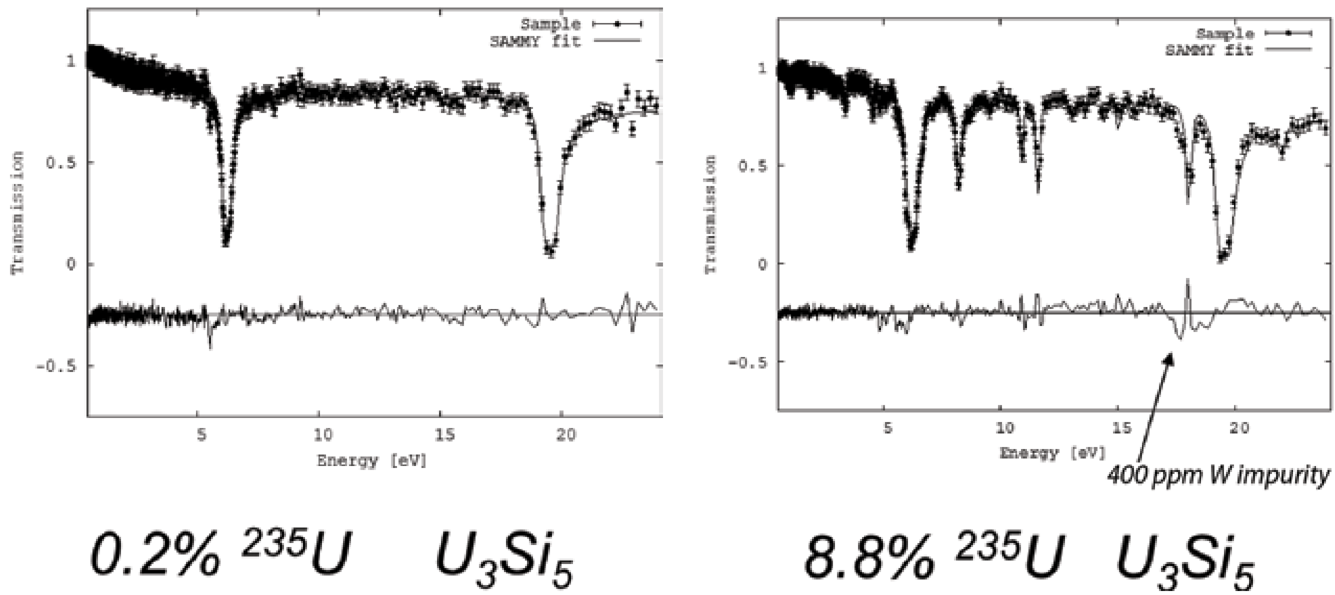


Figure 2. Measured cross-section of an energy-resolved neutron imaging measurements for pixels corresponding to volumes with different 235U enrichment levels. The measured data is fit with known cross-sections against areal densities of each isotope. The peak in the difference curve at 18.6 eV originates from a ~400 ppm contamination with W atoms, underlining the sensitivity of the method.

Project Description:

The APIE effort consists of three pillars: (1) Method development for fuel characterization with pulsed neutrons, (2) application of these techniques to actual fuels (in part witness samples for subsequent irradiation tests), and (3) development of pulsed neutron sources which could be deployed pool-side at a test reactor. Several milestone reports were submitted describing these efforts in detail. The application of the characterization tools to fuels guides development of synthesis routes, e.g. by detecting and quantifying the presence of contamination phases (e.g. UN in U-Si fuel forms) or by microstructural changes due to melting of a phase during sintering in a composite fuel (e.g. U_3Si_5 in a UN/U_3Si_5 composite fuel). Characterization by the multitude of parameters mentioned above provides a detailed parameters set for the pre-irradiation state of specimen to be inserted into a test reactor for irradiation. While X-rays can in principle provide similar parameters, for nuclear fuels they only interrogate a few micrometers on the surface, which is in many cases not representative of the bulk. Furthermore, these bulk properties are of value for modeling efforts, e.g. describing the effects of temperature and irradiation on microstructure evolution. When applied to an irradiated fuel, the bulk characterization of all pellets in an irradiated rodlet will guide destructive examination by identifying the most interesting regions of the entire volume based on a multitude of parameters. A science-based development of e.g. accident tolerant fuels or transmutation fuels will accelerate fuel development while at the same time providing a rigorous basis for licensing, the ultimate goal of fuel developments, and thus contributes to a reliable, and economic operation of the nation's current reactor fleet and next generation reactors.

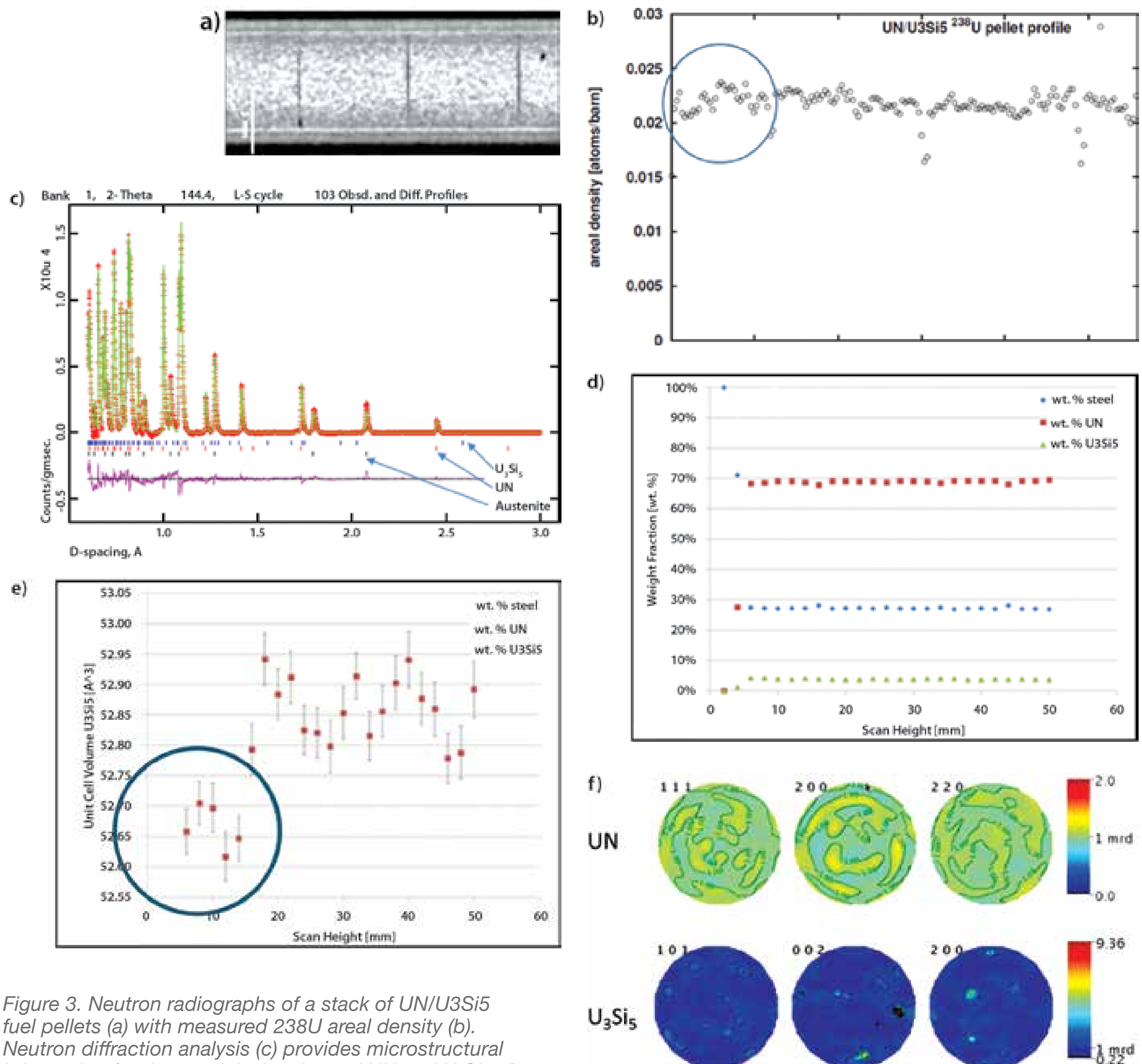


Figure 3. Neutron radiographs of a stack of UN/U3Si5 fuel pellets (a) with measured 238U areal density (b). Neutron diffraction analysis (c) provides microstructural information (such as weight fractions of UN and U_3Si_5, d) and crystallographic information (such as unit cell volume of U_3Si_5, e). The texture information, represented as pole figures (f), indicates that while the UN grains of a random orientation distribution, the U_3Si_5 phase has melted during sintering and re-solidified as large single crystals, forming an interconnected matrix.

Accomplishments:
As an example of the thorough characterization described above, investigations on UN/U-Si composite fuels are explained below. At a pulsed neutrons source in combination with a pixilated time-of-flight imaging detector (provided by Anton Tremsin, UC Berkeley), energy-resolved neutron imaging is possible by recording neutron transmission data for each pixel of the imaging detector. This allows to measure the areal density of isotopes for each pixel by fitting the data over a several 10 eV energy range, covering several absorption resonances, with the known neutron cross-sections. Data analysis is accomplished by utilizing the SAMMY code, developed at ORNL, and fitting data for each of ~100,000 pixels covered by a typical fuel pellet (Adrian Losko, LANL). Using tomographic reconstruction codes, the areal densities are converted to atoms per volume of a voxel in the CT, which can be converted to gravimetric density with the molar mass of each isotope. From the CT reconstruction of isotopes present in the samples, slices for isotope densities or other analysis can be conducted. Figure 1 gives an example of 3D isotope densities in UN/U3Si5 and U3Si5 pellets with different enrichment levels encapsulated in a stainless steel container (provided by Andy Nelson, LANL). The average enrichment levels measured by neutron resonance transmission analysis (NRTA) are within 0.1% of the nominal enrichment levels. While characterization with thermal neutrons, as shown on the left, does not allow to distinguish the highest (8.8%) and lowest (0.2%) enrichment level pellets, the reconstruction of the isotope density of 235U clearly allows that distinction. Figure 2 gives examples of the neutron transmission fits. A peak in the difference curve of the area where the 8.8% enriched material was interrogated originates from a 400 ppm contamination with tungsten, which has a resonance at 18.6 eV.

Neutron diffraction complements the characterization by energy-resolved neutron imaging (Sven Vogel, LANL). Figure 3 shows neutron radiographs of another set of UN/U3Si5 pellets (a) with the areal density for 238U derived from the data in (b). A somewhat higher scatter in the areal densities was observed for the data points corresponding to the left-most pellet. Neutron diffraction analysis of diffraction data from 2mm slices of the rodlet is conducted by fitting a model describing crystallographic and microstructural parameters to the measured diffraction intensities (c). While the UN and U_3Si_5 weight fractions are constant for all pellets (d, also showing the weight fraction of the steel container), the unit cell volumes of the hexagonal U_3Si_5 phase (e) are distinctly lower for the data points

The Advanced Post-Irradiation Examination (APIE) package provides fuel characterization advancing development of synthesis routes as well as pre- and post-irradiation examination to unprecedented levels of detail.

corresponding to the left most pellet. Such reduction in unit cell volume can be explained with incorporation of nitrogen atoms into the U_3Si_5 matrix. Measurement of the crystal orientation distribution (texture) using neutron diffraction furthermore provided a random orientation of the UN grains while in the 2mm probed volume for the data shown as pole figures in (f) the U_3Si_5 appears only in three grain orientations. Such strong texture, consistent with the presence of three single crystals in the interrogated volume, could be explained with melting of the silicide phase during sintering followed by resolidification as single crystals. The UN, with a higher melting point, remained solid, preserving the random grain orientation of the pre-sintered compacted material. The thermal expansion of U_3Si_5 along the crystallographic a and c axes is to the best of our knowledge unknown and will be measured in high temperature neutron powder diffraction experiments in the future. If significant anisotropy exists, i.e. the expansion along the a and c axes is substantially different, the resulting thermal stresses may lead to cracking of the sample, e.g. during an irradiation test. Since visual inspection of the pellet did not reveal the significantly different microstructure of this particular pellet, interpretation of an irradiation test would have been difficult. The presence of larger scatter of the areal density observed in the energy-resolved neutron imaging could then be explained by density variations due to melted and resolidified U3Si5 leaving behind voids.

In the future, we plan to design and build a shielding container that allows application of the aforementioned neutron characterization techniques to irradiated, highly radioactive samples. A schematic of the present design is shown in Figure 4a. A similar cask, shown in Figure 4b, was used at the Chalk River Reactor to characterize spent fuel from that facility, thus establishing precedence that this type of measurement is feasible. The authorization basis of the Lujan Center at LANSCE, where the measurements described above were performed, allows for up to 380 plutonium equivalent grams total, thus allowing handling of several e.g. U-20Pu-10Zr transmutation fuel slugs.

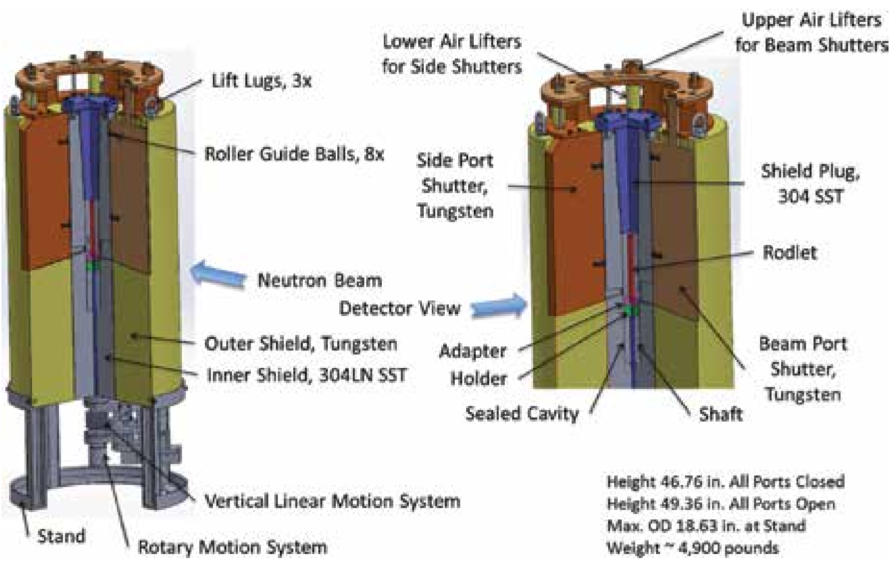


Figure 4. a) Initial design of a shielding cask allowing characterization of highly radioactive samples, e.g. irradiated or spent nuclear fuels. The device allows motion of the sample within the cask to enable tomography or diffraction texture measurements with the beam entering and exiting through removable windows. b) A cask used at the Chalk River, Canada, research reactor to characterize spent fuel from that facility by neutron diffraction.

High resolution optical microscopy in HFEF

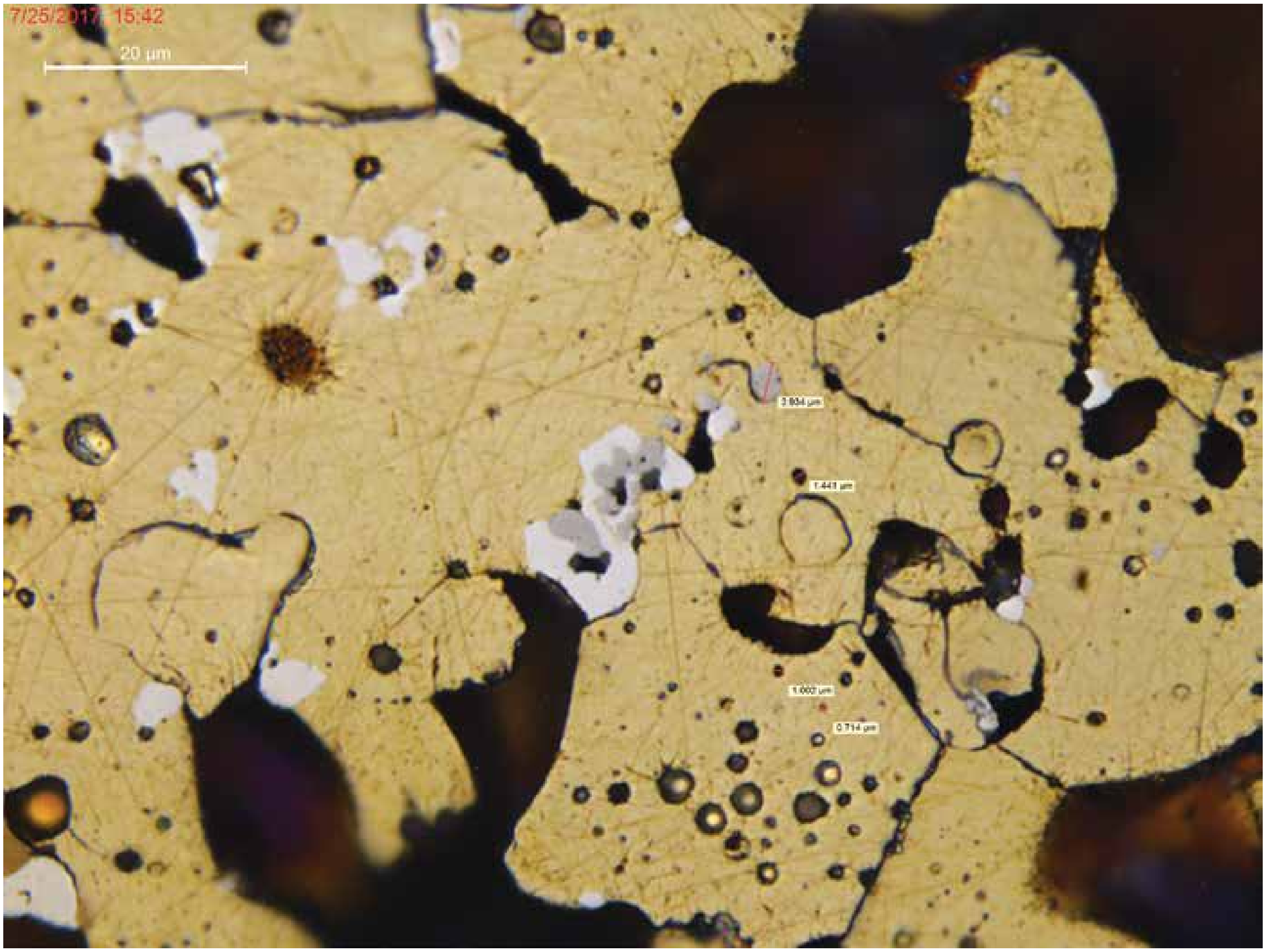


Figure 1. High resolution 1000x image with precipitates measurements down to the micrometer range in metallic fuel from AFC-4A.

Postirradiation examination of the Advanced Test Reactor (ATR) irradiated Accident Tolerant Fuel (ATF) concepts continues at the Idaho National Laboratory Hot Fuel Examination Facility (HFEF). The metallography hot cell station was updated with a new high resolution microscope.

Project Description:

This new capability will support both the Accident Tolerant Fuel portion and the Advanced Reactor Fuel portion of AFC. A variety of different irradiation tests designed to provide data on several different potential ATF and ARF concepts have already been examined with this new capability. Optical

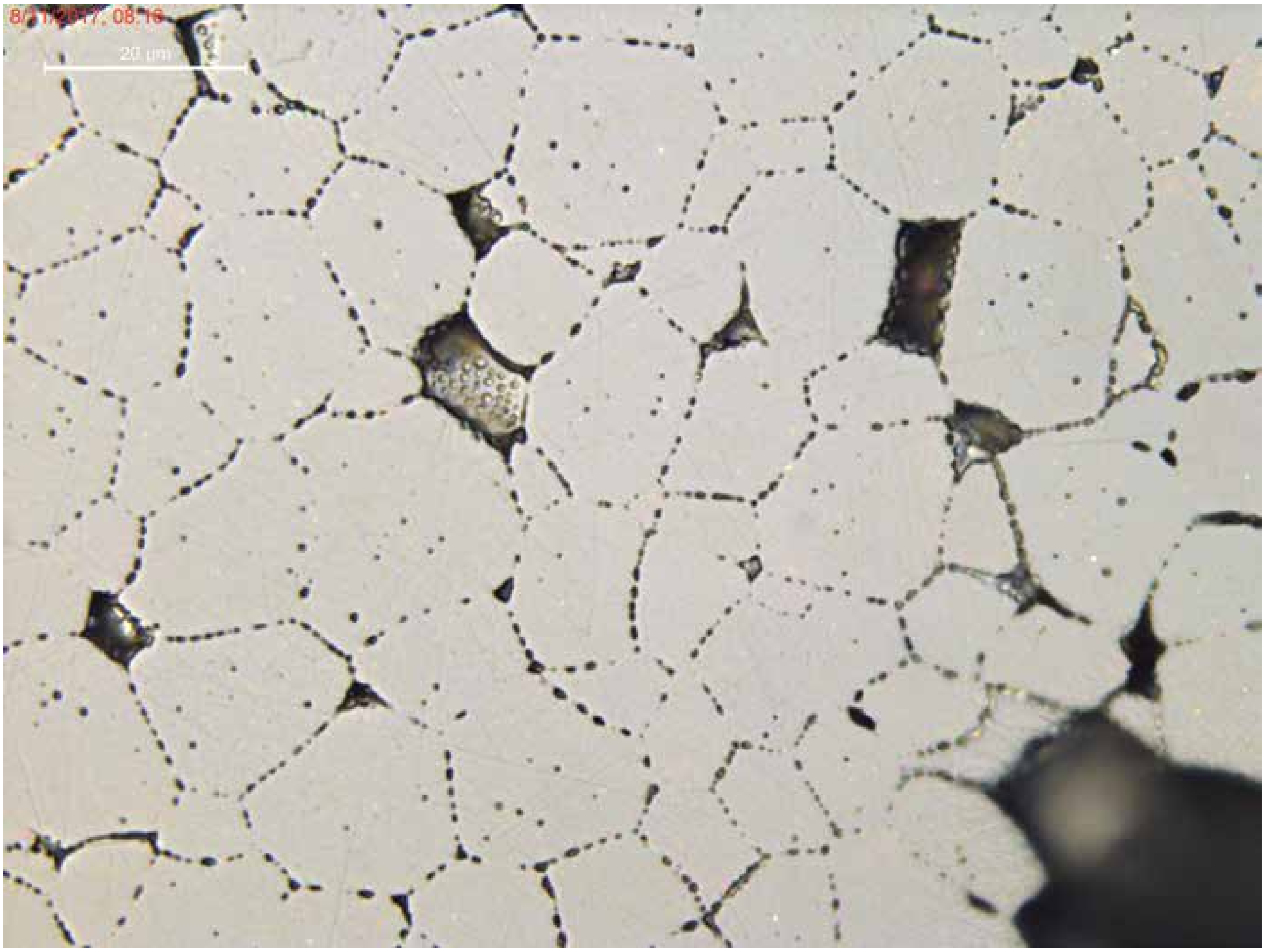


Figure 2. High resolution 1000x image showing very high porosity in irradiated UO_2 fuel from ATF-1.

microscopy is a cornerstone destructive examination that gives a first understanding of the microstructure of the fuel. High resolution optical microscopy is a state-of-art technique to appreciate secondary phases, metallic inclusion down to the micrometer range. Both the AFC-OA (Figure 1) and ATF-1 (Figure 2) irradiations are drop-in style tests a fuel cladding and fuel meat combination to various different burnups. The results of these screening irradiations support the development of new fuel forms to help light water reactors potentially operate with enhanced safety margins.

High resolution optical microscopy has enable AFC scientists to gain microstructure information down to the micrometer range

Accomplishments:
The high resolution optical microscopy was installed and qualified in the metallography hot cell station of HFEF. It was a successful effort of HFEF team supported by AFC scientists.

ADVANCED LWR FUEL SYSTEMS

2.1 Acciedent Tolerant Fuels

2.2 High Performance LWR Fuel Development

2.3 Analysis

2.4 ATF Cladding and Coatings

2.5 Irradiation Testing and PIE Techniques

2.6 Transient Testing

2.1 ACCIDENT TOLERANT FUELS

ATF Industry Advisory Committee

Committee Chair: Bill Gassmann, Exelon
Collaborators: Jon Carmack

The Advanced Light Water Reactor Fuel Industry Advisory Committee (IAC) was established in 2012 to advise AFC's National Technical Director (NTD) on the development and execution of a program focused on advanced fuels for light water reactors. The IAC is comprised of leaders from the commercial light water reactor industry. They represent the major suppliers of nuclear steam supply systems, owners / operators of U.S. nuclear power plants, fuel vendors, and the Electric Power Research Institute (EPRI) Members are selected on the basis of their technical knowledge of nuclear plant and fuel performance issues as well as their decision-making positions in their respective companies.

The IAC meets monthly via teleconference and in a face-to-face meeting once a year and is currently chaired by William Gassmann of Exelon Corporation. Additional members represent BWXT, Westinghouse Electric Company, AREVA, Global Nuclear Fuels, EPRI, Dominion, Duke Energy, and Southern Nuclear.

AREVA

Exelon

SOUTHERN COMPANY

BWXT
BWX Technologies, Inc.

EPRI
ELECTRIC POWER RESEARCH INSTITUTE

Westinghouse

Dominion Energy

GE imagination at work

DUKE ENERGY

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

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

Principal Investigator: Edward J. Lahoda
Collaborators: Westinghouse Electric Company LLC , General Atomics, Massachusetts Institute of Technology (MIT), Idaho National Laboratory (INL), Los Alamos National Laboratory (LANL), Southern Nuclear Operating Company and Exelon Nuclear, Argonne National Laboratory, Ceramic Tubular Products, Pennsylvania State University, University of Wisconsin and Argonne, National Nuclear Laboratory (United Kingdom) (NNL), Army Research Laboratory (ARL), Institute for Energy Technology (Norway), United Technologies Research Center

Figure 1. Process demonstration of SiC endplug sealing. A baseline end-plug is shown in the bottom left; the remaining end-plugs incorporate a local seal to facilitate rodlet pressurization.

The overall objective of this program is to introduce accident tolerant fuel (ATF) lead test rods and assemblies (LTR/LTA) for SiC and coated zirconium cladding with U_3Si_2 fuel an into commercial reactors by 2019 and 2022 respectively. The objective of the current Phase 1b work is to design, test and build using commercially scalable technologies test articles for up to 6 year-long exposure at PWR conditions of prototypical ATF fuel rodlets in ATR and Halden. The data from this 6-year test reactor exposure and test evaluation will be used as the basis to license and load LTRs into commercial reactors in 2019 and LTAs in 2022.

Figure 2. Cr coated zirconium alloy tubes before and after testing in the MIT reactor.

Project Description:
The technical objectives of this program are:

- The technical objectives of this program are:
- Fabrication of representative test fuel rodlets consisting of SiC and coated zirconium cladding with U_3Si_2 fuel
- Development of an ATF LTR/LTA project plan
- Generation of test data and modeling supporting test and commercial reactor feasibility
- Determine the expected behavior of the ATF options in LOCA/station blackout scenarios
- Evaluate the beyond design basis accident performance, expected plant licensing impacts and resultant economic savings for commercial plants
- Determine the feasibility of licensing a commercially feasible ATF

Accomplishments:
SiC cladding development has continued. Low corrosion rate, hermetic, closed end SiC tubes (Figure 1) have been produced by General Atomics and tested in autoclaves at the Westinghouse Churchill site and are undergoing testing in the Massachusetts Institute of Technology reactor. Cr coated cladding tubes have been produced by the University of Wisconsin and successfully tested in the autoclaves and in the Massachusetts Institute of Technology reactor

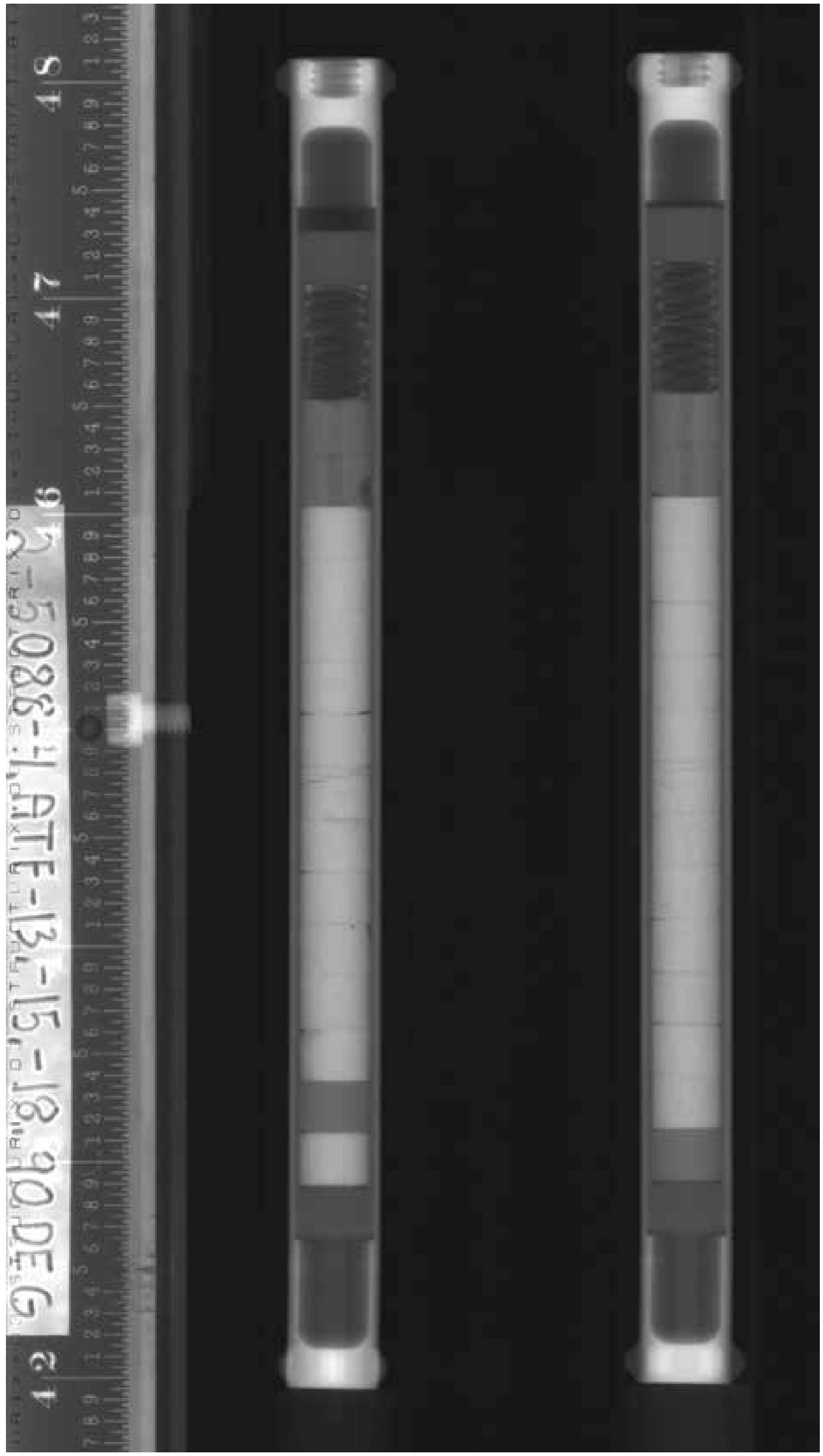


Figure 3. Neutron radiographs of 20 MWd/kgU U_3Si_2 Pins from ATR.

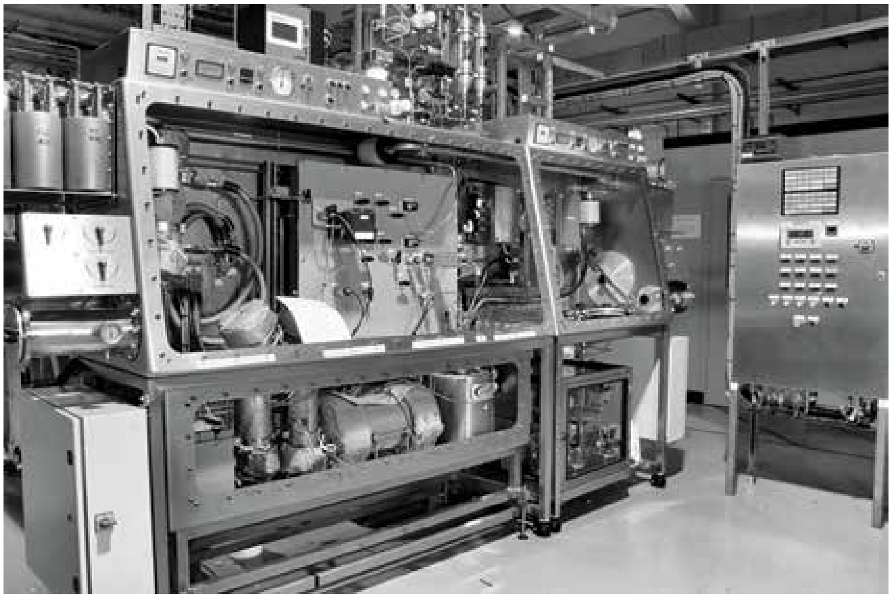


Figure 4. NNL Installation for UF_6 to U_3Si_2 Studies.

(Figure 2). Both SiC composite and Cr coated cladding are currently undergoing final irradiation testing and will be used to fabricate rodlets for exposure in the Advanced Test Reactor (ATR) and Halden reactor (Institute for Energy Technology) starting in 2018. Ceramic Tubular Products has developed a continuous process for doing chemical vapor deposition of SiC on the surface of SiC composite windings. United Technologies Research Center has been exploring methods for making composites utilizing SiC particles and SiC pre-polymer compounds to replace the CVI process.

Two, 6-inch U_3Si_2 fueled rodlets (Figure 3) produced by INL have been removed from the ATR after achieving burnups of ~20 MWd/kgU. Swelling was 0% +/- 1% and fission gas release was <3% for linear heating rates of 12 to 15 kw/ft. Test facilities on methods to produce U_3Si_2 from UF_6 were completed buy NNL (Figure 4) and testing commenced in September, 2017. LANL completed generation and documentation of U_3Si_2 properties and performed extensive oxidation and hydration studies of U_3Si_2 in simulated PWR environments.

Southern Nuclear Operating Company and Exelon Nuclear have been evaluating the economic gains from using ATF in current nuclear plants. Potentially significant gains have been identified and are being quantified using probability risk assessment techniques by Westinghouse and accident scenario modeling by Fauske and Associates. The significant economic potential offered by ATF has lead Exelon to accept LTRs for their Byron reactor in 2019. These LTRs will utilize Cr coated cladding produced by the Army Research Laboratory and U_3Si_2 produced by INL. Both Southern Nuclear Operating Company and Arizona Power Systems have also shown interest in testing Westinghouse ATFs.

Accident tolerant fuel will provide not only significant fuel cost savings, but also enable significant savings for current nuclear plant in operating costs due to the reduction or elimination of safety related equipment and testing requirements.

ATF Industry Teams - AREVA

Principal Investigator: Kiran Nimishakavi
Collaborators: University of Florida (UF) and Electric Power Research Institute (EPRI)

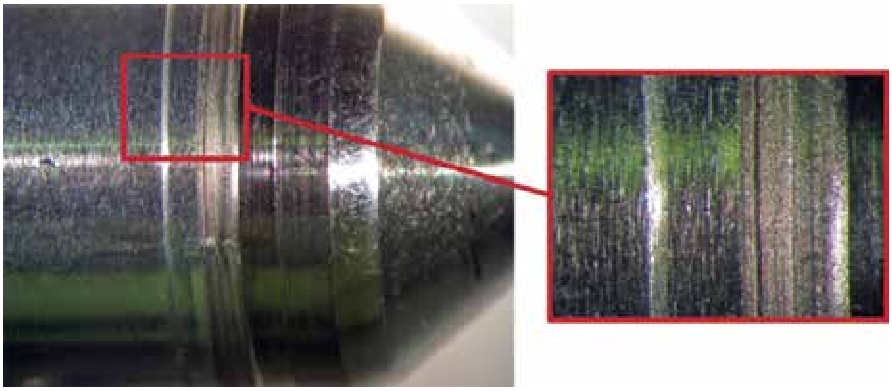


Figure 1. Cr-coated tubes welded to end plugs using AREVA's upset shape welding process.

Advanced Fuel concepts bring the potential to not only significantly enhance the reactors capability to withstand accidents but may provide the means to streamline and improve plant operational and maintenance costs in a very challenging economic market.

The work in Phase 1 has been successful in providing data that allowed the AREVA Team to assess several Enhanced Accident Tolerant Fuel (EATF) concepts with the goal of being able to deploy Lead Test Rods or Lead Test Assemblies (LTRs/LTAs) in a commercial reactor by 2022. As a result of several gate reviews and the subsequent DOE technical review committee recommendation, in Phase 2, AREVA team narrowed its focus on two EATF concepts 1) A near term solution based on a fuel rod design utilizing Chromium-Coated Zirconium-alloy Cladding with Chromia-Doped UO_2 Pellets, 2) A longer-term solution with a Silicon Carbide (SiC) Advanced Cladding Concept. So the work in 2017 is mainly centered on the development of these two concepts.

Project Description

In Phase 2, AREVA's plan is to perform the necessary irradiation testing, manufacturing integration, and analysis of chosen EATF concepts to support the fabrication and insertion of LTAs and fully demonstrate to utilities the value and performance under normal operation and accident conditions.

The key technical objectives of this research are:

1. Support development and characterization of the advanced fuel concept through testing and analyses.
2. Fully evaluate the performance of the concept and assess the economic impacts.
3. Support qualification and integration of manufacturing processes

Figure 2. Welded Cr-coated tubes after corrosion test in 360°C water for 1 day (ASTM G2).

4. Implementation of the LTA irradiation program (including all interactions with and submittals to the NRC)

The overall objective of the DOE EATF program is to develop an improved and more robust nuclear fuel design that will reduce or mitigate the consequences of reactor accidents and improve the economics of the reactors thermal output and reliability. The best way to achieve this is through a systematic approach along parallel fronts with design, development, testing and licensing of EATF concepts over the next five years. The data collected as part of this program will ultimately be used as a basis to support the DOE's near-term goal of LTAs insertions by 2022 and long-term goal of batch implementation by 2025.

Accomplishments:

As part of Cr-coated cladding development, a feasibility study was conducted to demonstrate the weldability of Cr-coated tubes to solid Zr-4 end plugs using AREVA's commercial upset shape welding (USW) process. The study showed that the USW process is compatible with welding Cr-coated tubes. The USW-welded Cr-coated tubes showed no defects at the weld region (see Figure 1). Subsequently, welded Cr-coated tubes were tested under

Figure 3. The prototype machine for coating full-length tubes.

a corrosive water environment at 360°C (ASTM G2). As shown in Figure 2, the Cr-coated surface exhibits a silver metallic color due to lack of corrosion whereas, the Zr-4 end-plugs appear black due to the formation of $ZrO2$. AREVA completed the successful qualification of the USW process for Cr-coated tubes at AREVA's fuel assembly manufacturing plant in Romans, France. The same qualification will be performed at the AREVA Horn Rapid Road (HRR) facility in Richland, WA, prior to the manufacturing of ATF-2 fuel pins.

The most promising results of GFY 2017 are from the PWR irradiations on Cr-coated tubes, which have completed a one-year irradiation cycle in a commercial reactor in Europe. Initial visual examinations on the Cr-coated samples during outage showed positive results. The Cr-coating is very much adherent to the underlying substrate, and no delamination was observed. The transition between the coated and uncoated part of the tubes was smooth, and no particular features were observed. The Cr-layer has a slight golden tint to it, which suggests a minimal oxide formation ($< 1\mu m$) on the surface of the cladding. This will be confirmed during post-irradiation examinations in early 2018. These initial on-site PIE results, after irradiation in a commercial PWR, confirm the excellent performance of the Cr-coated cladding, which complements the results obtained during the out-of-pile tests.

A significant milestone has been achieved to fabricate a full-length Cr-coating prototype machine to deposit Cr on full-length tubes with an improved PVD technique. All the components have been received and assembled. Figure 3 shows the

Figure 4: Cr_2O_3-doped UO_2 pellets (green) manufactured for the ATF-2 rodlets.

overview of the assembled prototype machine. The first test under vacuum was performed and did not reveal any leaks. Initial coatings performed on flat samples showed a uniform coating over the whole length of the prototype machine. Fabrication of sample holder to coat the first tubes is underway. The first tubes are planned to be coated during GFY 18, followed by a parametric evaluation of coating conditions to obtain the best quality coatings over the full-length. The prototype manufacturing and production of tubes is on schedule for the fabrication and insertion of lead test rods in a commercial reactor by 2019

Several activities are ongoing to implement the manufacturing line for chromia-doped pellets at the HRR fuel fabrication facility in support of the EATF LTA insertion into a commercial reactor in 2019.

A small batch of Chromia-doped UO_2 pellets were manufactured for AREVA's ATF-2 rodlets (see Figure 4). These pellets will be loaded into 6-inch Cr-coated rodlets for exposure in the Advanced Test Reactor (ATR) – ATF2 in early 2018.

In support of developing a stable SiC/SiC cladding design, many exploratory studies have been performed in GFY 2017. A SiC/SiC clad model was implemented in an AREVA NP fuel rod thermal-mechanical code. Preliminary calculations revealed that during normal operation significant thermal stresses exist along the thickness of the cladding, which are high enough to generate cracks in the cladding. To address this, various options are being considered for improving the thermal conductivity of the base SiC cladding.

ATF Industry Teams – General Electric

General Electric: Raul B. Rebak
Collaborators: Evan J. Dolley (GE Research), Kurt A. Terrani (ORNL), Yukinori Yamamoto (ORNL), Kevin Ledford (GNF), Russell E. Stachowski (GNF), Russ M. Fawcett (GNF), Jonas Gynnerstedt (Sandvik), William P. Gassmann (Exelon), John B. Williams (Southern Nuclear)

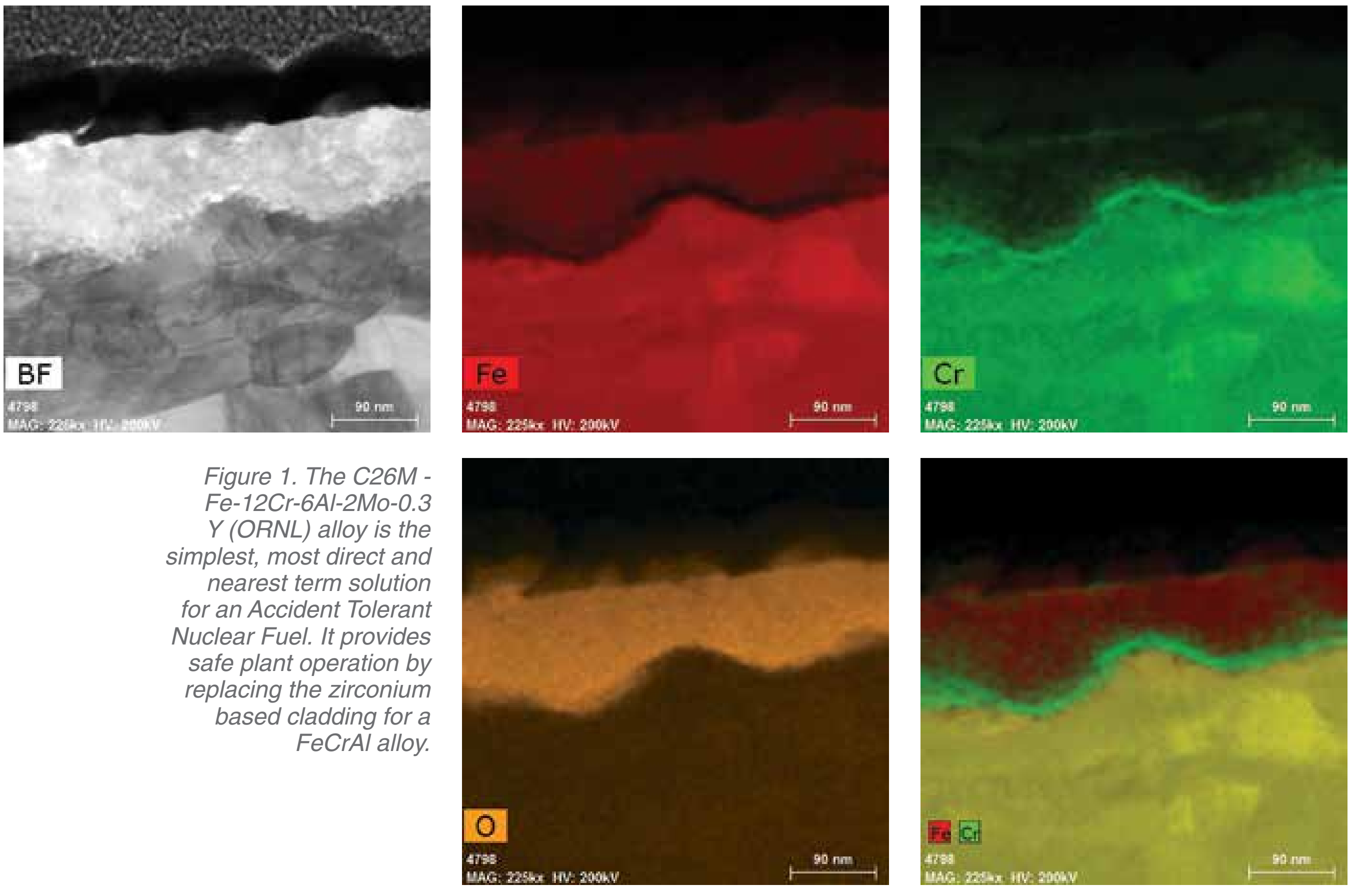


Figure 1. The C26M - Fe-12Cr-6Al-2Mo-0.3 Y (ORNL) alloy is the simplest, most direct and nearest term solution for an Accident Tolerant Nuclear Fuel. It provides safe plant operation by replacing the zirconium based cladding for a FeCrAl alloy.

The General Electric (GE) and Oak Ridge National Laboratory (ORNL) Accident Tolerant Fuel or Advanced Technology Fuel (ATF) concept utilizes a FeCrAl alloy material as fuel rod cladding in combination with uranium dioxide (UO_2) fuel pellets presently in use. This combination will increase the safety in the operation of light water reactors. The use of FeCrAl cladding is a simple near-term path for the current fleet of power reactors.

Project Description:

The overall goal of the U.S Department of Energy Advanced Fuel Campaign Accident Tolerant Fuel Program for LWRs is to identify alternative fuel system technologies to further enhance the safety, competitiveness and economics of the current fleet of commercial nuclear power. There are currently 99 power reactors operating in the USA, of which the 89% are at least 30 years old and 45% of the reactors are at least 40 years old. Therefore, there is an immediacy in the developing ATF that can have a significant impact to the safety of the remaining life of current fleet. The sooner the solution is found the better. The industry cannot afford to wait 30 years to find a viable ATF concept since no commercial reactor will be running to use the concept.

Under normal operation conditions in LWRs, the FeCrAl alloys offer outstanding corrosion resistance like current well known materials such as type 304SS or nickel alloys. Under accident conditions, FeCrAl alloys are orders of magnitude more resistant to reaction with superheated steam than zirconium, therefore generating negligible ignitable hydrogen and heat of reaction. FeCrAl alloys would keep their coolable geometry for longer time allowing for quenching measures during the coping time. On the less favorable side, the FeCrAl alloys are less transparent to neutrons than zirconium alloys which impacts fuel cycle cost and electricity generation costs, and the FeCrAl may release more tritium to the coolant. Both adverse attributes can be minimized or eliminated by design, fabrication and regulatory modifications.

Accomplishments:

GE and ORNL are characterizing two alloys side by side regarding all their nuclear properties, from mechanical to fabrication to environmental resistance. Kevin Field of ORNL developed a manual containing the properties of the FeCrAl alloys. The two alloys are Kanthal-Sandvik APMT produced via powder metallurgy and a traditionally melted alloy developed by ORNL (C26M).

- C26M - Fe-12Cr-6Al-2Mo-0.3Y (ORNL),
- APMT – Fe-21Cr-5Al-3Mo (Sandvik).

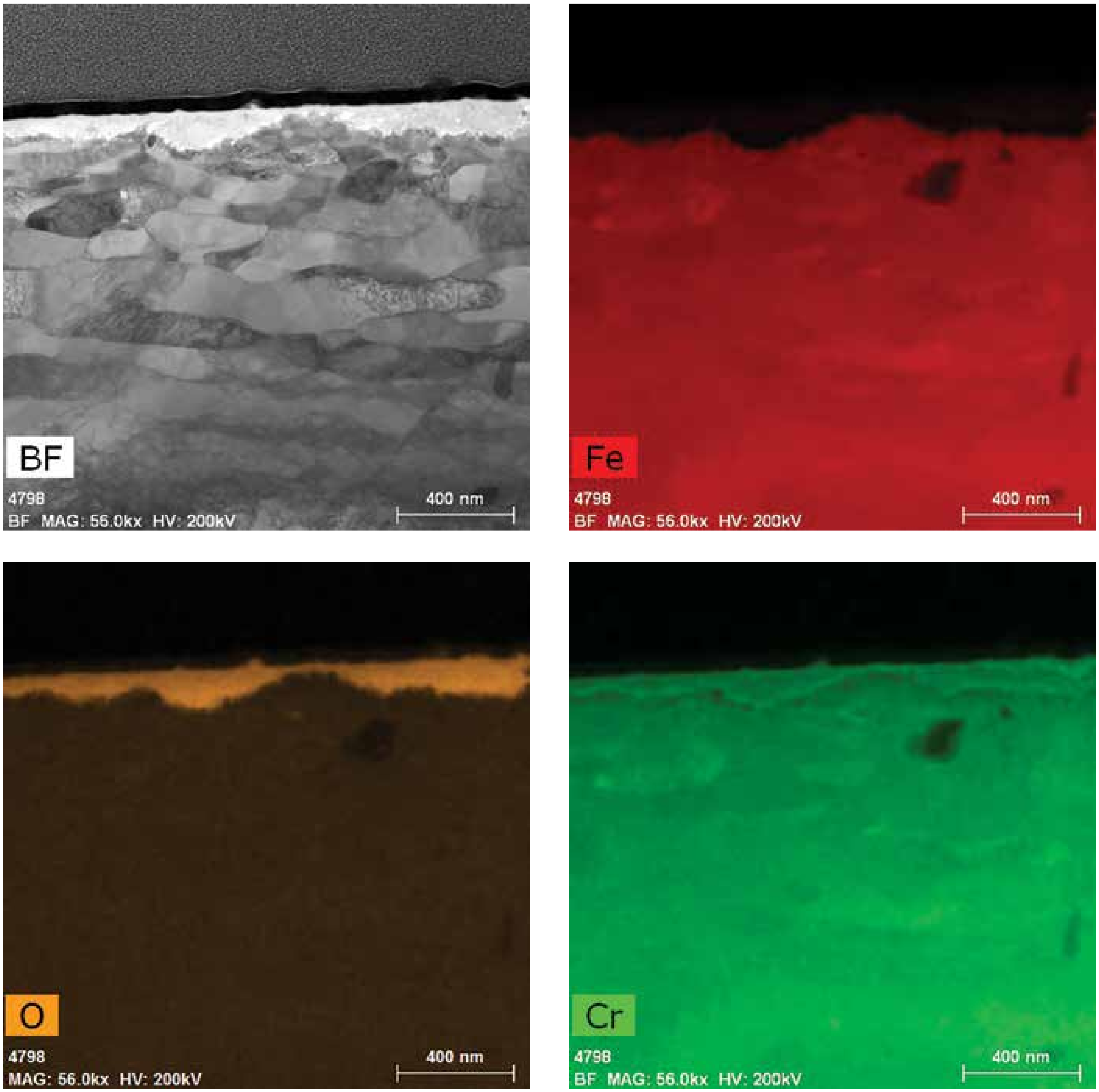


Figure 2. Cross sectional spectral images of the surface oxide formed on FeCrAl APMT coupon exposed for one year to pure water + 3.75 ppm hydrogen at 330°C. In the hydrogen atmosphere, there is a single layer oxide film, rich in Cr. In hydrogen the oxide film is less than 100 nm thick.

This is the simplest most direct and nearest term solution for an Accident Tolerant Nuclear Fuel. It provides safe plant operation by replacing the zirconium based cladding for a FeCrAl alloy.

Significant efforts were invested in the tube making process of FeCrAl alloys both at Sandvik and at US plants such as Superior Tube and Century Tube. Yukinori Yamamoto from ORNL was the lead in the success of the tube effort.

Two primary alloy producers (Metaltek in Wisconsin and Sophisticated Alloys in Pennsylvania) are fabricating ingots of C26M for tube production purposes.

Steam testing showed that C26M alloy is as resistant to attack by steam as APMT. High temperature (~300°C) hydrogenated and oxygenated water electrochemical testing showed that C26M and APMT behave in a similar manner as well-known reactor materials such as nickel alloy X-750 and stainless steel type 304.

GE/GNF is working with utilities such as Southern Nuclear to insert the first FeCrAl fuel prototype into a commercial reactor in the US. The plant selected for the first effort is SNC's Plant Hatch, Unit 1 during the Cycle 29 refueling outage (1Q18). It is planned to insert 2-8 Lead Fuel Rods (LFRs) into each of 2-4 GNF2 Lead Test Assemblies (LTAs) as segmented rods of the two FeCrAl alloys mentioned before (APMT and C26M).

GNF/GEH and Southern interfaced with the Nuclear Regulatory Commission (NRC) to obtain the authorizations to transport the fuel to the Georgia plant and to install the LTA into the reactors.

Development of Accident Tolerant Fuel Options For Near Term Applications

Principal Investigator: Jacopo Buongiorno (Massachusetts Institute of Technology)
Collaborators: K. Smith, K. Shirvan (MIT), K. Sridharan (UW), L. Shao (TAMU), M. Tonks (PSU) P. Broda (AREVA), W. Liu (ANATECH), J. Hales (INL).

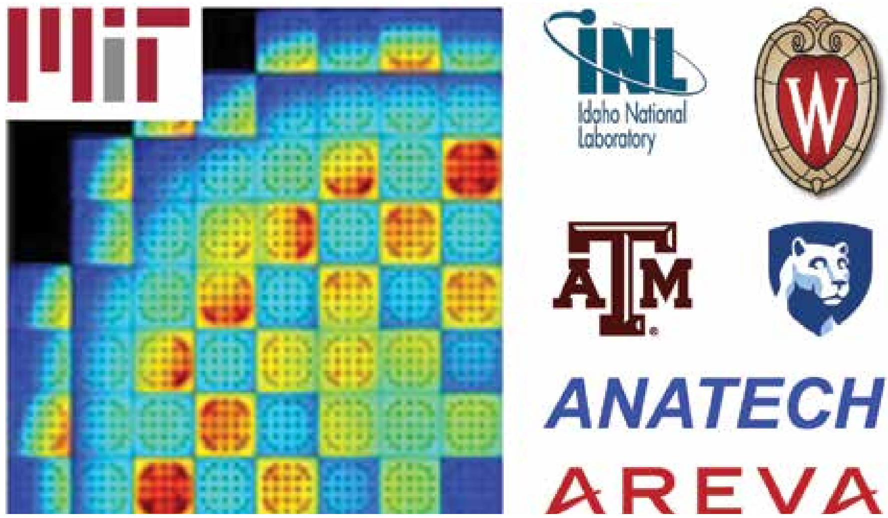


Figure 1. The IRP Team (From Left to Right): MIT, INL, UW, TAMU, PSU, ANATECH and AREVA

The project develops state-of-the-art simulation tools for new fuel cladding materials that will tolerate accident conditions without generating massive amounts of flammable hydrogen, thus greatly mitigating the consequences of Fukushima-like events at nuclear power plants worldwide.

This work focuses on development and evaluation of computational models under the NEAMS framework to estimate time to failure for near term accident tolerant fuel concepts. The Integrated Research Project (IRP) Team (Figure 1) builds upon strong university capabilities at the Massachusetts Institute of Technology (MIT) with its experience in fuel design and safety analysis, the University of Wisconsin (UW) with its experience in severe accident modeling and development of cladding coatings for the ATF industrial campaign, Texas A&M University (TAMU) with its material ion irradiation capability, and Pennsylvania State University (PSU) with its meso-scale fuel performance modeling experience. INL is a member of the team to allow for efficient implementation and integration of the models. The IRP also benefits from close collaboration with two industrial partners: ANATECH, engineering firm with state-of-art experience in fuel modeling under accident conditions and AREVA, one of the main nuclear fuel suppliers in the US.

Project Description:

Our work is focused on current operating light water reactors world-wide, and we aim to improve their safety performance with use of innovative accident tolerant fuels. The objective of this IRP is to develop computational tools to evaluate such ATF options for near term applications. The computational tools will be predominantly developed under the framework of

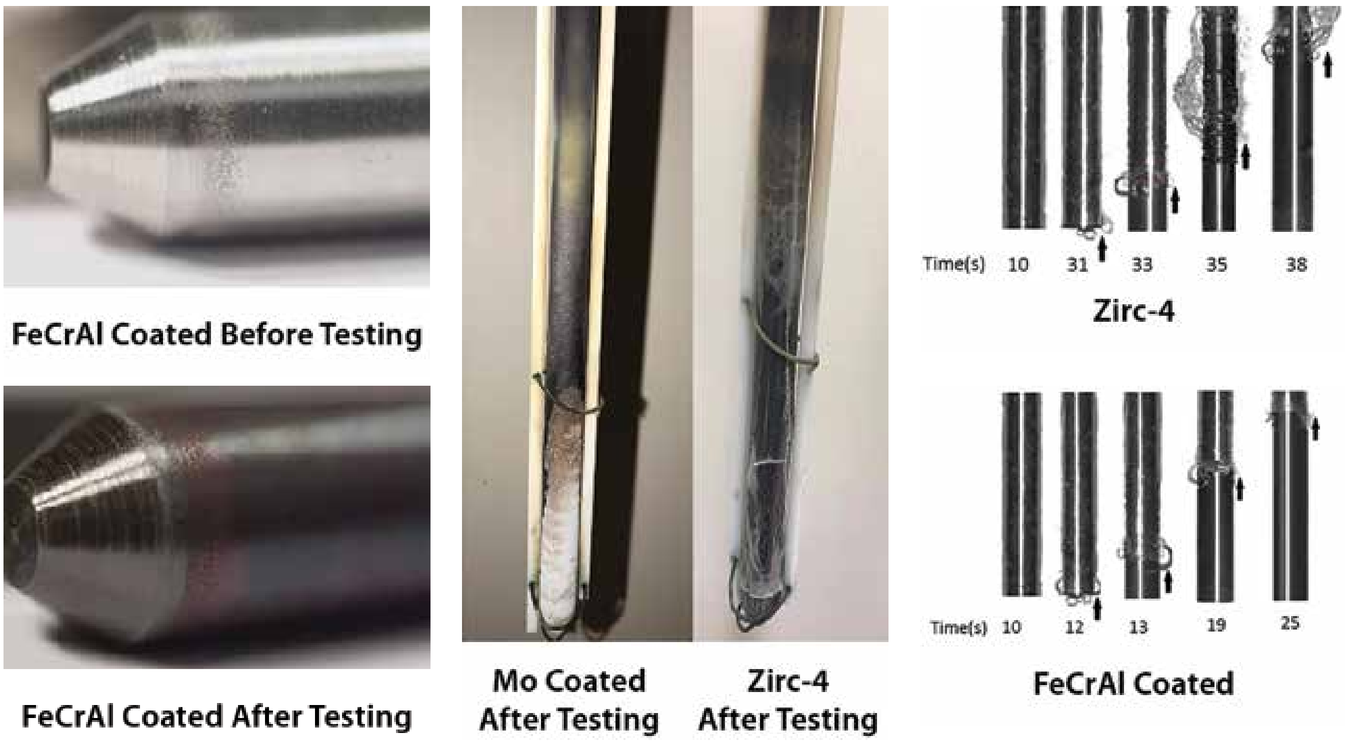


Figure 2. Specimen images of a FeCrAl coated rodlet after exposure to 320oC liquid water under both tension and compression before and after 40 days of testing (left), two rodlets exposed to 1200oC after 90 min (middle), two rods quench front after exposure to 1000oC (right).

the NEAMS such as MOOSE/BISON (engineering scale fuel performance). For proper validation, existing experimental data will be utilized and new low-cost experimental data will be generated within this IRP. For verification, industry standard and licensing tools for neutronics, thermal hydraulics and fuel performance will be used for model development of conventional fuel and to provide cross-code checking of ATF performance. The newly developed models will be evaluated at the core level under both steady state and transient conditions, including severe accidents, where time to failure of the fuel and core components is estimated. The project findings will advance the state-of-the-art tools for simulating the accident tolerance of current and innovative nuclear fuels. The project will play a strong role in the successful execution of DOE's accident tolerant fuel program, which was started in response to the Fukushima accident. It will also improve selected safety analysis tools used by the nuclear regulatory commission (NRC) by extending their ability to predict fuel failure during severe accident conditions.

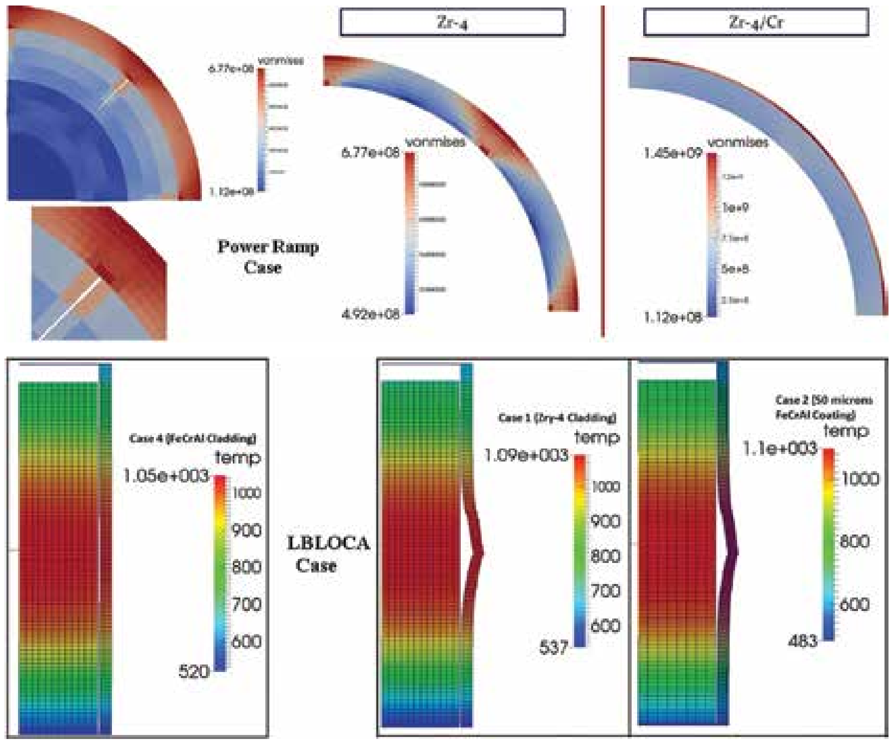


Figure 3. BISON predictions of the effective stress during a power ramp (top) and a Large Break Loss-of-Coolant Accident (LOCA; bottom) for Zircaloy-4, FeCrAl and Chromium coated claddings.

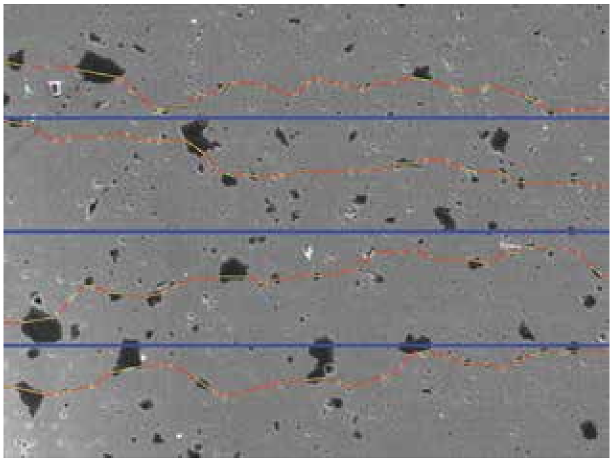
Accomplishments:

The cold spray coating technology has been used by UW to deposit, FeCrAl and Mo alloy coatings on various Zr4 flat and tubular geometries obtained from AREVA (Figure 2). The Mo is used as an inter-diffusion layer between FeCrAl alloy and Zr4 to prevent formation of low-melt eutectic. Both mechanical properties and wear resistance has been characterized for the coated materials by UW.

TAMU systematically studied the phase formation and diffusion kinetics at interfaces of FeCrAl coated Zr4 after prolonged annealing, and ion radiation responses. The subsequent mechanical property changes of each interface phase was studied using pillar compression tests.

MIT has performed low temperature and high temperature steam oxidation in addition to PWR prototypical corrosion/CRUD deposition and 4-point bend, pressurized tube and burst mechanical tests with the coated claddings to better inform the modeling and simulation work and understand new failure modes by the ATF concepts. MIT has also performed quench studies showing higher quench temperature but slower nucleate boiling rate for unoxidized FeCrAl surfaces compared to Zr4.

MIT and UW are engaged in a TRACE/MELCOR short term station blackout benchmark to gain confidence in each system code prediction of increases gains in additional coping time provided by the ATF concepts. Thus



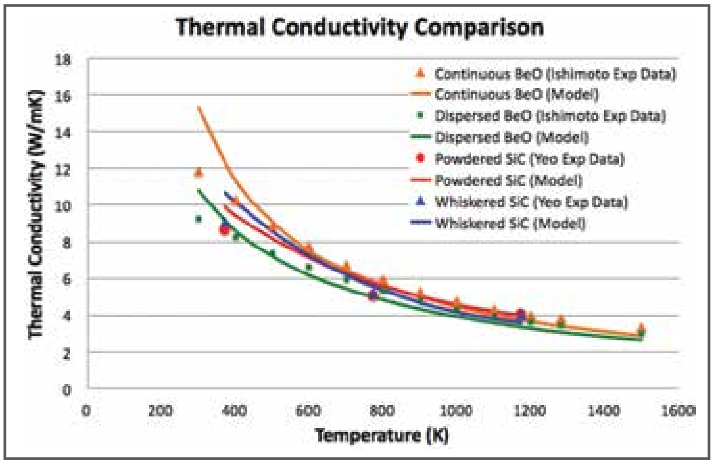


Figure 4. Summary of the thermal resistor model that predicts the thermal conductivity of UO_2 accident tolerant fuel with high thermal conductivity additives, where an illustration of the approach used to determine the connectivity of the additive is shown on the left, and a plot of the performance of the model is shown on the right.

far, the time to first melting point and time to onset of significant hydrogen generation has been used as a figure of merit by TRACE and MELCOR, respectively. The results thus far show notable but modest gains in coping time with FeCrAl cladding under various scenarios including unmitigated large break loss-of-coolant accident and short term station black out for a 3-loop PWR design. Both codes capability has also been extended to include ability to model cladding coatings.

At PSU, the MARMOT mesoscale fuel performance tool has been used to investigate the impact of high thermal conductivity additives BeO and SiC on UO_2 thermal conductivity using 2D and 3D simulations. These results informed the development of a macroscale model of the fuel thermal conductivity with high thermal conductivity additives that will be implemented in the BISON fuel performance tool (Figures 3 and 4, respectively). The impact of these additives on fission gas release will also be investigated using MARMOT simulations. Initial simulations by MIT in collaboration with AREVA and INL on chromia doped pellets against experimental data from Halden reactor showed a gap in mechanistic prediction of fission gas release that will be further investigated next year.

MIT in collaboration with ANATECH and INL has performed a detailed steady state and transient fuel performance modeling of a Zr4 cladding with FeCrAl, Cr, Mo/FeCrAl coatings using the BISON fuel performance framework. An in-house fuel performance tool along with finite element analysis tool, ABAQUS, were used to verify BISON calculations. It was found that plasticity models are typically required under steady state, especially during pellet-to-clad mechanical interaction due to thermal expansion and mechanical properties mismatch of each coating layer with Zr4 and each other. Under power ramp conditions, it was found that the coatings are loaded upon contact which may result in onset of cracks. During upcoming year, additional physics informed by generated data will be used to improve the coated cladding models.

Fundamental elastic and mechanical properties of high density fuels

Principal Investigators: U. Carvajal Nunez (LANL)
Collaborators: T.Saleh, N.A. Mara, A.T. Nelson (LANL)

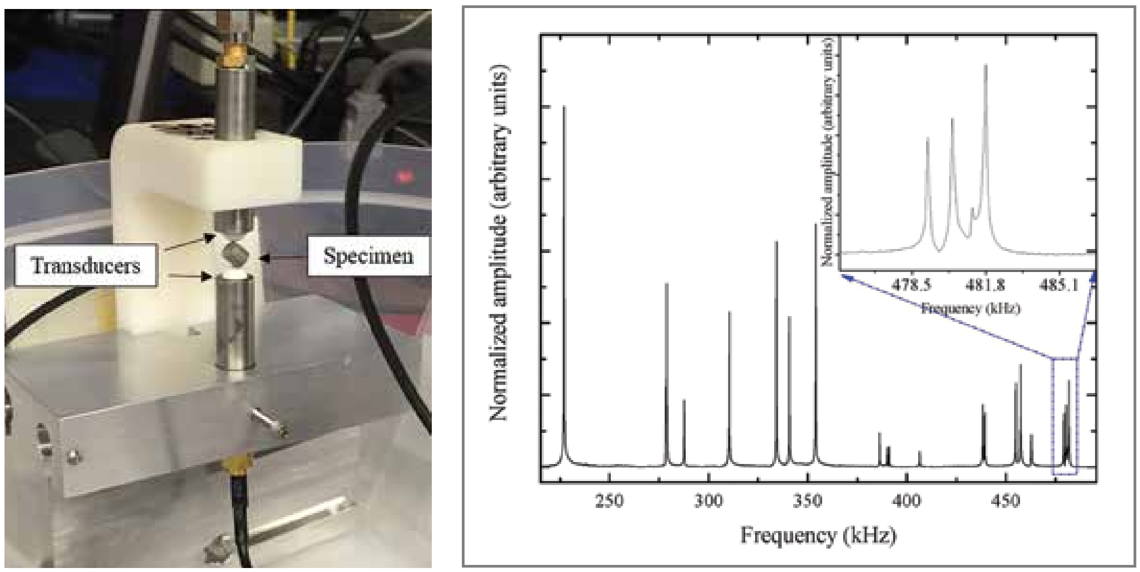


Figure 1. Experimental set up of a RUS cylinder sample sandwiched between hemispherical Al_2O_3 wear plates, glued to the transducers (left). Measured frequency response of a U_3Si_2 sample from 200 to 500kHz (right).

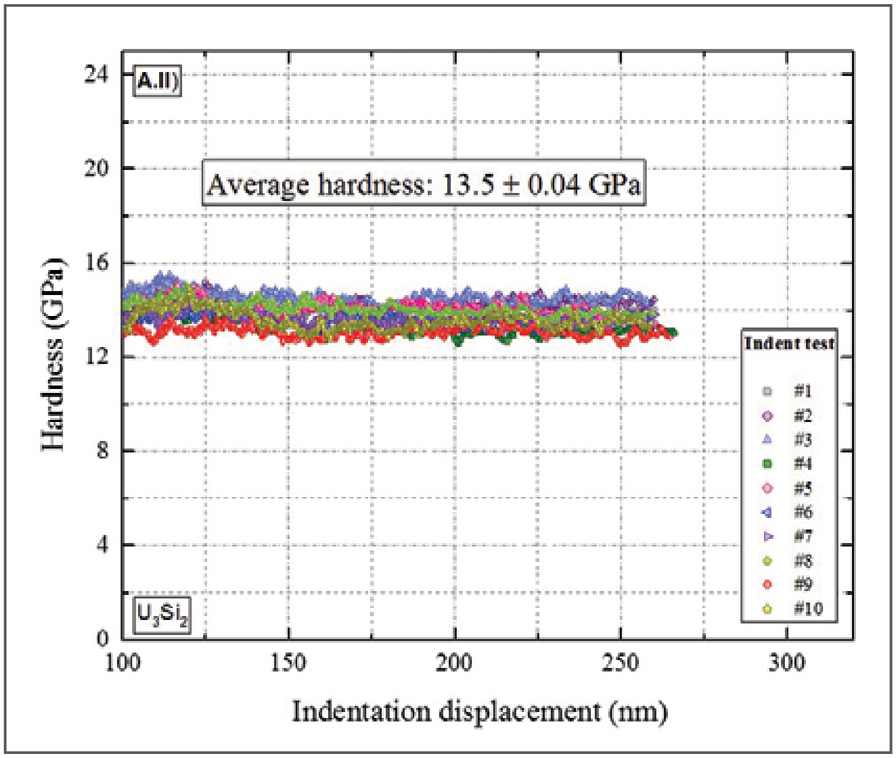
The mechanical properties of nuclear fuels are historically far less studied than thermodynamic or transport properties, but nonetheless play an important role in fuel behavior. For new or little understood fuel forms such as many presently considered as accident tolerant candidates, even basic mechanical properties are absent from the literature. Evaluation of mechanical properties is therefore necessary to support fuel performance modeling. Two methods have been deployed to meet this need and demonstrated on a series of U-Si compounds. Nanoindentation was used for measurement of hardness and Young's Modulus, and Resonant Ultrasound Spectroscopy was used to determine elastic constants C11, C44 and the bulk modulus; these can then be used to calculate Young's Modulus and Poisson's ratio. These data have now been determined for a U_3Si, U_3Si_2, USi.

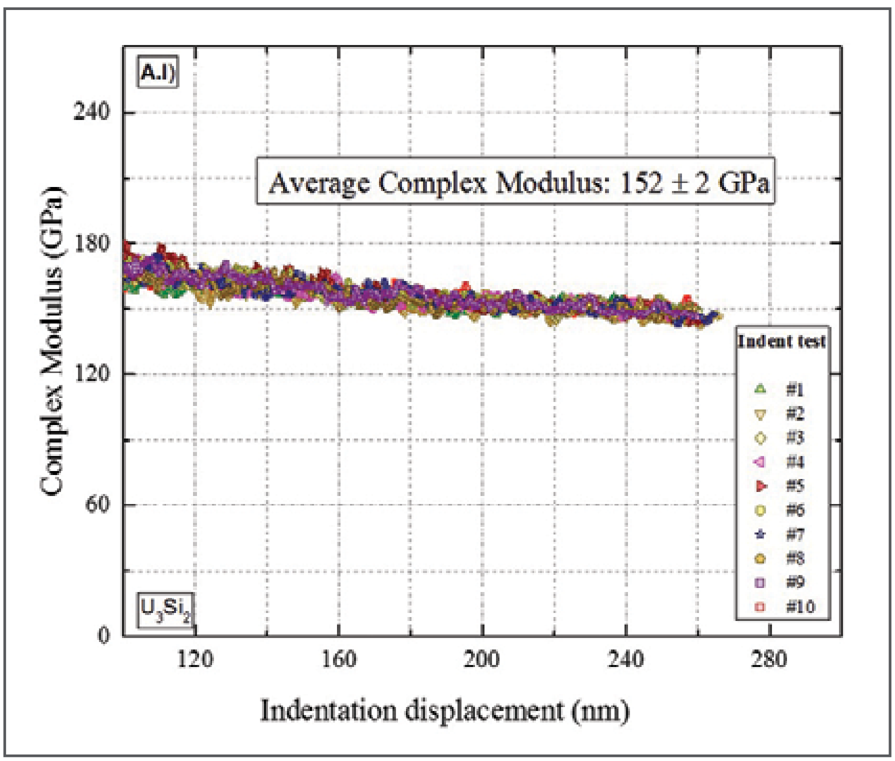
Mechanical property measurements of ATF fuel materials provide previously unmeasured data to facilitate more sophisticated fuel performance modeling and provide fundamental inputs to modeling and simulation efforts.

Project Description:

The Office of Nuclear Energy currently funds the investigation of novel ceramic nuclear fuel concepts. One of the primary performance objectives targeted for improved performance of nuclear fuels is minimizing fracture in fuel pellets. New materials or composite fuel architectures that offer greater resistance to cracking under the extreme environments encountered during nuclear reactor service would provide significant improvements to steady state (e.g. heat transfer, fuel redistribution) and transient (e.g. radionucleide release at elevated temperatures) conditions. Options currently under investigation include 'nontraditional' nuclear fuels designated around high uranium density. These include uranium silicides, uranium borides, and composite fuel materials constructed of these and uranium nitride or uranium dioxide. Preliminary screening of the thermophysical and thermodynamic properties of these concepts has provided confidence in their soundness, but evaluation of their mechanical properties at relevant temperatures must be executed in order to support further study. Development of novel and improved means of measuring the mechanical properties of uranium-bearing materials is a critical step in advancing toward fuels that are less susceptible to cracking under both normal operating conditions and possible transients.

Figure 2. Elastic modulus (A.I) and Hardness (A.II) for U_3Si_2 as determined through nanoindentation.





Accomplishments:
Sample fabrication took place in the Fuel Research Laboratory at LANL. The material chosen for initial investigation was U_3Si_2 given its prevalence across a range of ATF concepts. In addition, U_3Si and USi were included in the study given their presence in U_3Si_2 as common impurity phases. Samples of U_3Si_2 containing varying levels of oxygen were also included in the study to assess the impact of oxidation on mechanical properties. Resonant Ultrasound Spectroscopy (Figure 1) and both quasi-static and continuous stiffness nanoindentation measurements were performed at LANL. Results of nanomechanical testing (Figure 2) highlight properties determined as a function of depth in these materials. Use of both resonant ultrasound spectroscopy (RUS) and nanoindentation have provided a number of mechanical and elastic properties for these compounds that were previously unavailable. Hardness is a critical component in post-irradiation examination and provides insight into chemical and microstructural evolutions of fuel following burnup. Elastic and bulk moduli are essential parameters that link strain to stress; determining the impact that the temperature gradients present in operating fuel will have on stress and in turn fuel pellet fracture requires these data. Finally, the fundamental elastic constants as provided by RUS are essential inputs to density functional theory and other modeling efforts. The availability of these parameters for U_3Si_2 and other fuel materials will improve the ability of companion modeling and simulation efforts to capture and predict the behaviors of these materials. These measurements will be extended to fuel operating temperatures in the coming year.

Response of U_3Si_2 to steam and pressurized water at LWR operating temperatures

Principal Investigators: Andy Nelson
Collaborators: E.S. Wood, Justin White

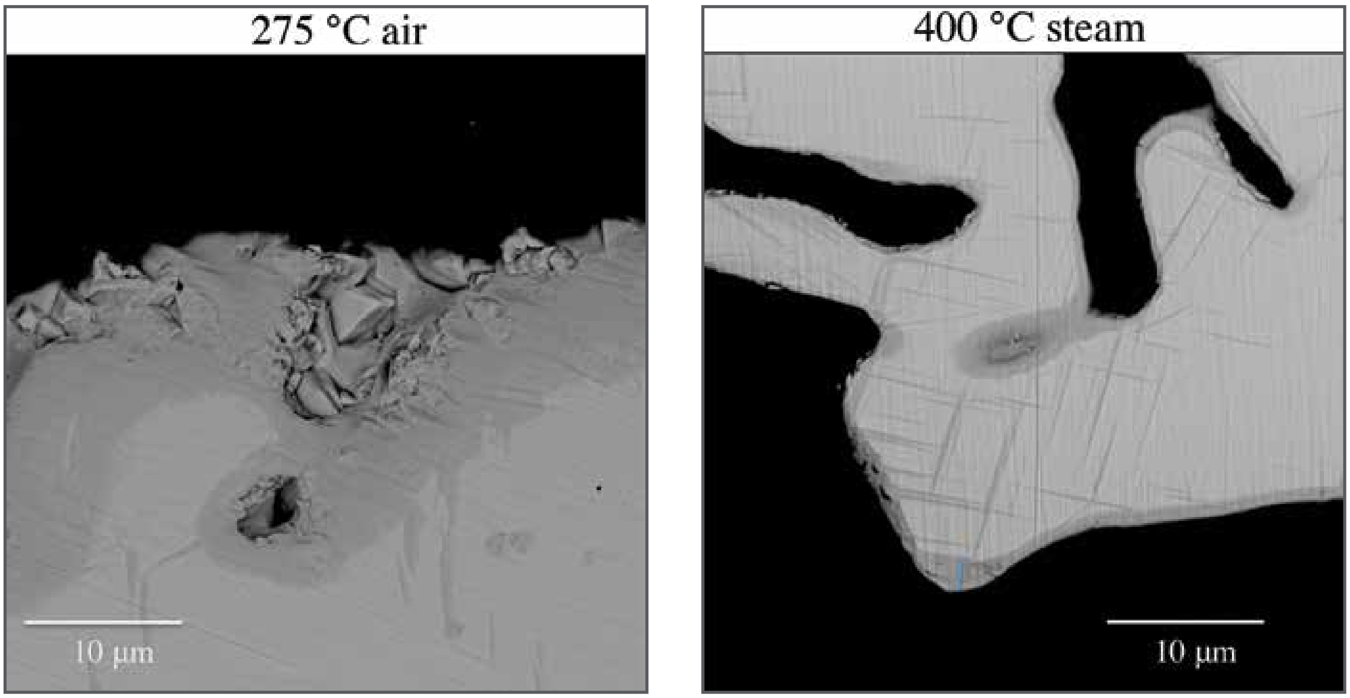


Figure 1. Compares the microstructures of U_3Si_2 oxidized under H_2O and air.

New Accident Tolerant Fuel (ATF) concepts such as U_3Si_2 offer potential benefits to reactor operation and performance, but also introduce uncertainties. The behavior of U_3Si_2 when exposed to pressurized water and steam at temperatures between 250 and 400C has established the susceptibility of U_3Si_2 to hydriding. This behavior has not been previously observed in U-Si compounds, and must be better understood to project its response under a leaker or Loss of Coolant Accident (LOCA) scenario.

Execution of tests to assess vulnerabilities of proposed ATF concepts is a key component of their qualification for commercial Light Water Reactor (LWR) deployment. Many proposed cladding or fuel systems possess properties or specific performance attributes that would be superior to the reference UO_2

/ zirconium systems in certain scenarios. However, uncertainties are inherent in the assessment of new technologies. Fuel behavior across a complete range of reactor conditions and possible operating occurrences must be established. The Advanced Fuels Campaign (AFC) has invested in development of test capabilities to study the behavior of candidate fuel materials in environments as they would be experience during operation. Autoclave testing in chemistry-controlled environments is a key component of qualification. Autoclave testing of cladding materials is available at multiple institutions, but assessment of actinide materials under similar conditions poses operational challenges; flow through autoclaves would have to accommodate testing of little known materials that may fully pulverize and result in distribution of uranium powder throughout the loop.

A buffered autoclave capability for uranium materials was therefore developed to facilitate assessment of new proposed fuel materials. Testing can be performed to characterize the stability of fuels under exposure to pressurized water at chemistries that can be matched to either Pressurized Water Reactor (PWR) or Boiling Water Reactor (BWR) conditions. A range of uranium silicide, uranium nitride, and ceramic fissile composite materials have been assessed using this technique. While results have found that U_3Si_2 is more stable than other fuel materials tested with the exception of UO_2, it does suffer hydriding which governs pulverization under these conditions. Efforts are underway to better understand this behavior and assess its impact on use of U_3Si_2 as an LWR fuel.

The basic behavioral investigations of U_3Si_2, U_3Si_5, UN, UN/U_3Si_2 composites, and UO_2 were previously completed at Los Alamos National Laboratory (LANL). Uranium dioxide was included as a reference material, as its behavior under pressurized water is relatively well known and can therefore be used to validate system performance. These studies proved that, while still inferior to UO_2, U_3Si_2 performed the best of the above materials as it largely resisted pulverization through over 30 days of exposure to 300C water. Testing of U_3Si_5 found rapid pulverization after only two days and led to the termination of work on UN composites shielded by U_3Si_5.

Work in FY17 centered on evaluation of the specific microstructural mechanisms that govern failure. This work was partially prompted by observations that testing of U_3Si_2 in PWR water chemistries (~5 ppm H_2) resulted in improved stability early in testing (negligible weight change was found through 30 days), but rapid pulverization beyond this time. Characterization of the pulverized material through x-ray diffraction (XRD) found negligible sign of oxide compounds. This contrasted to results obtained for testing in either neutral or slightly oxidizing water chemistries, where considerable evidence of oxidation-induced pulverization is observed.

Assessment of the microstructures of U_3Si_2 exposed to autoclave conditions and following H_2O oxidation at atmospheric pressure revealed that the microstructure resulting from H_2O oxidation differs considerably from that found following oxidation by O_2 alone. During oxidation of U_3Si_2 in high-oxygen environments (e.g. air), UO_2 formation occurs rapidly and is followed by further oxidation of UO_2 to U_3O_8. Above approximately 350C, oxygen diffusion along U_3Si_2 grain boundaries results in rapid pulverization of the sample. However, oxidation of U_3Si_2 in H_2O either at atmospheric pressure or during autoclave testing drives an additional phenomenon.

Figure 1 compares the microstructures of U_3Si_2 oxidized under H_2O and air. The tests were executed at different temperatures to match the kinetics at the length scale shown. Testing performed in air results in development of three surface structures: U_3O_8 can be observed spalling off the sample surface. A layer of UO_2 appears and is more strongly adhered, and a region of USi_3 can be found in the subsurface regions as formed by the Si rejected from the U_3Si_2 bulk as the uranium oxides form. The same structures are formed during oxidation in H_2O, but at a lesser degree. This is caused by the greatly reduced oxygen activity of H_2O compared with O_2 at these temperatures. However, an additional feature is observed. Needle-like

This work highlights ongoing efforts to mitigate oxidation of high uranium density fuel powders, which should relax the air handling constraints of these fuels in a commercial environment.

precipitates can be found in the sample interior. These have been identified as a hydride phase. Recent work at LANL has established the ability of hydrogen alone to fully pulverize U_3Si_2 samples at these temperatures. It is believed that the strain induced by the formation of U-Si-H phases (precipitation of the only reported structure, $U_3Si_2H1.8$, would result in a volume increase of roughly 10%), leads to pulverization of the material once its concentration exceeds a critical threshold.

Better understanding of this behavior is needed to assess the importance of this finding to LWR deployment of U_3Si_2. The impact of hydrogen as resulting from H_2O oxidation following a cladding breach during normal operation is the direct motivation for the above work, but at higher temperatures and pressures hydrogen will remain important as a major chemical actor during a LOCA. Ongoing work will examine hydrogen solubility, kinetics, thermodynamics, and structural implications of hydriding in the U_3Si_2 system.

Transmission Electron Microscopy of Native Oxides of U-Si Compounds

Principal Investigators: P. Edmondson
Collaborators: A. Nelson, Q. Smith, D. Skitt, K. Terrani

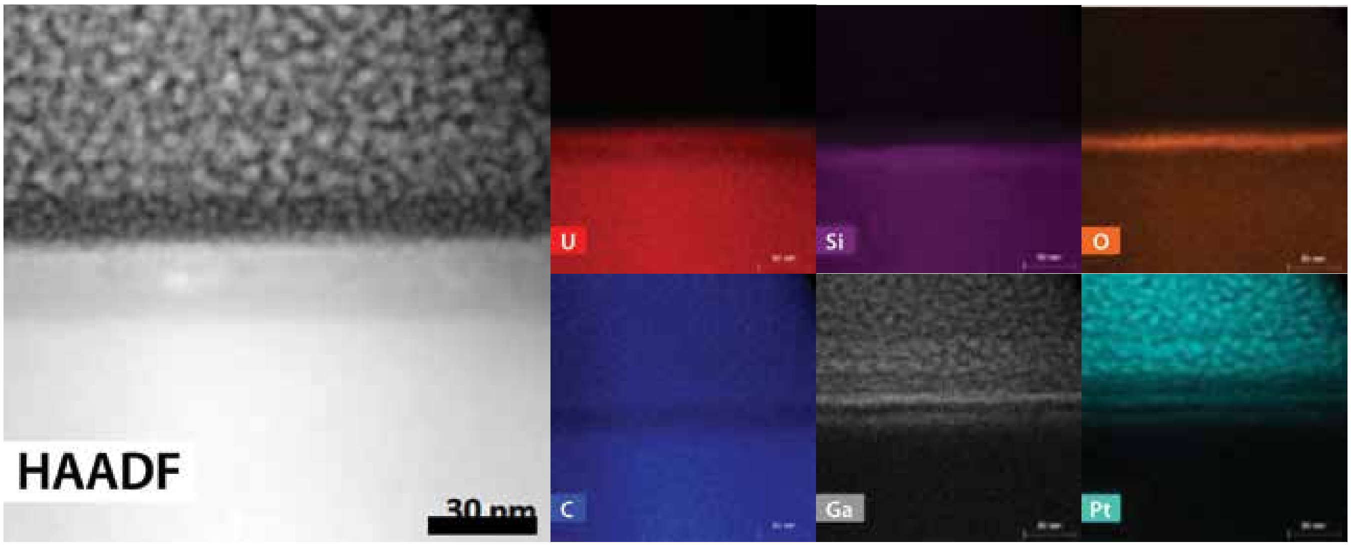


Figure 1. STEM-EDS maps recorded for U_3Si_2 sample showing chemical structure of native oxide layer.

Various uranium-silicide compounds are of interest as ATF concepts and as constituents in fissile ceramic composites, but remain far less understood with respect to their basic structure and chemistry compared with more established fuels. One particular area that is not understood is the native oxides formed on these materials following fabrication. Oxidation is a known vulnerability of these fuels, and further oxidation of feedstock has been demonstrated to impair fabrication. A collaborative investigation executed by ORNL and LANL utilized advanced characterization methods to assess the structure of the surface oxides of U-Si compounds. Unprecedented detail on chemistry and structure of the native oxide of these materials on the nanometer scale was provided by this work. This information is key for interpretation of degradation mechanisms relevant to light water reactor deployment of these materials.

Project Description:

Execution of basic studies to understand the fundamental processing-structure-property-performance linkages for proposed ATF concepts is a key component of their deployment. High density fuels such as U_3Si_2 provide advantages, but are far less understood than UO_2. Execution of a science-based fuel development campaign to advance the technical readiness level of these systems requires understanding of their basic structure and chemistry.

Surface oxidation is a fundamental challenge to deployment of U_3Si_2. While previous studies of elevated temperature oxidation behavior have been performed, the structure of the as-fabricated surface oxide (i.e. native oxide) of U_3Si_2 and other compounds in the U-Si system has not been previously investigated. This is of particular relevance to development of industrial processing methods for these systems. Oxidation of powder feedstocks has been found to significantly retard sintering kinetics; storage of U_3Si_2 powders in even very low oxygen environments (e.g. inert gloveboxes) will result in oxidation, and degrade the product densities achieved through sintering operations. As the oxide structure formed at ambient temperatures was not previously understood, it was not possible to develop more robust processing steps or other mitigation techniques. The results of this work address this gap and demonstrate the potential for state of the art characterization at lower length scales to address similar challenges in fuel development.

Accomplishments:

Specimens of U_3Si_5 and U_3Si_2 were prepared using reference fabrication processes at LANL and shipped to ORNL. At ORNL, the samples were polished and subsequently imaged using electron backscatter diffraction (EBSD). The results of the EBSD examination for U_3Si_5 provided important insights into another previously unknown aspect of U_3Si_5 behavior, as the micro cracking observed in U_3Si_5 samples was correlated to a specific crystallographic orientation. This data will assist in future studies of the elevated temperatures phase transformations in this portion of the U-Si phase diagram.

Following EBSD, focused ion beam (FIB) lift outs were performed to prepare foils for transmission electron microscopy (TEM). Both TEM and scanning TEM (STEM) was performed in the Low Activation Materials

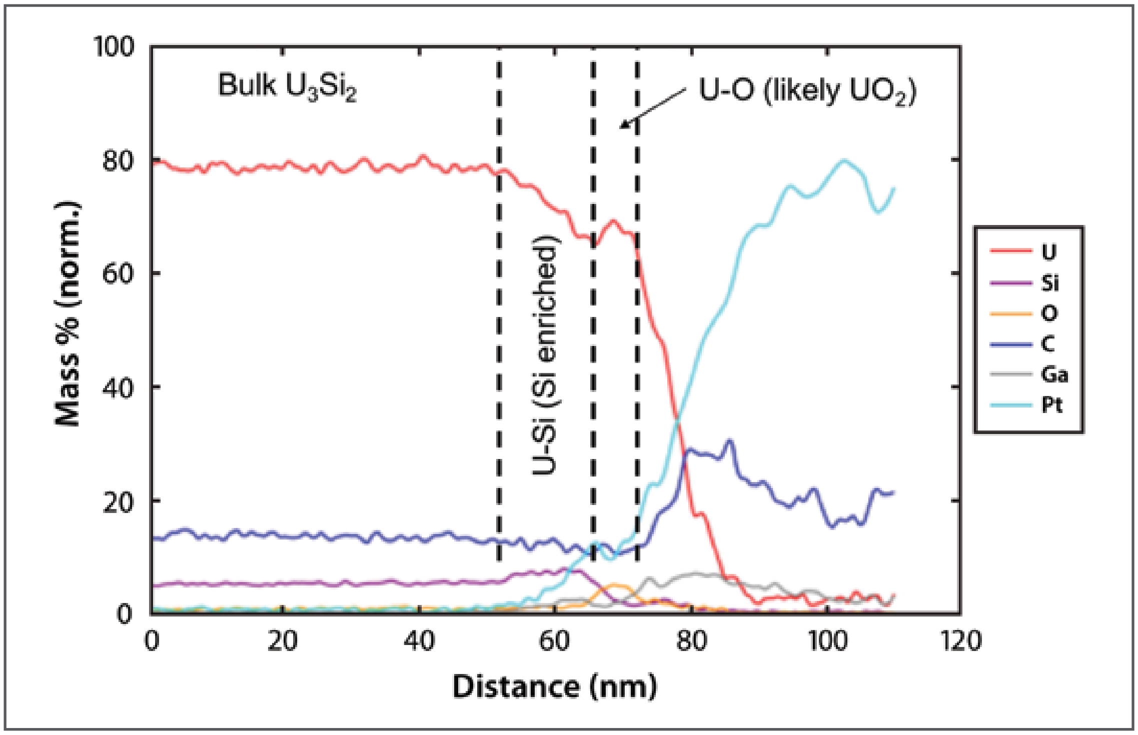


Figure 2. Representative EDS line scan obtained from a U_3Si_2 sample highlighting the native oxide structure. Results highlight the ~10 nm thick oxide (likely UO_2) and U-Si region of slight Si enrichment relative to the U_3Si_2 bulk in the subsurface region.

Development & Analysis (LAMDA) Laboratory using the FEI F200X Talos instrument. Characterization performed using the Talos consisted of high resolution STEM and energy dispersive X-ray spectroscopy (EDS). Multivariate statistical analysis was used to analyze the large EDS datasets and assess the chemical composition.

Results of the EDS scans are highlighted in Figure 1. The elemental maps show two main surface features. First, a region of O enrichment and absence of Si can be seen at the sample surface. A subsurface region of U depletion and Si enrichment is found next before transition to bulk U_3Si_2 of relatively constant U and Si content. This chemical behavior, although on a vastly smaller length scale, confirms that the native oxide structure present on U_3Si_2 samples is similar to the structure found following elevated temperature oxidation. An important finding was the limited thickness of the affected region. The surface U-O

Synergy of ORNL characterization methods and LANL fabrication of high density fuels leads to new insights regarding native oxide structure of U-Si compounds

structure (identified as crystalline and likely UO_2 but not independently confirmed in this study) is roughly 10 nm thick. The region of Si-enrichment is roughly 15 nm thick. These behaviors are show in Figure 2, a single line scale of the EDS data. The three regions are identified in the figure.

The native oxide structure of U_3Si_5 was characterized in the same manner. While U_3Si_2 and U_3Si_5 exhibit similar chemical evolutions following oxidation at high temperature, the native oxide structure observed in U_3Si_5 was distinct. No crystalline oxide was found. Instead, an amorphous U-Si-O region roughly 20 nm in thickness was observed. This result was unexpected and suggests that Si content may play a larger role in the formation of surface oxides for compositions in the U-Si binary than previously believed based on thermodynamics alone.

The results of this study will be expanded to include other U-Si binary compounds in the near future to better understand the transition between the well-ordered surface oxide observed in U_3Si_2 to the disordered and chemically homogeneous behavior of U_3Si_5. These findings will help improve our understanding of how oxidation progresses in the U-Si system and may lead to methods of mitigating oxidation during fabrication or service for high density U_3Si_2 fuels.

Development of high density fissile composite fuels: UO_2/UBx systems

Principal Investigator: Erofili Kardoulaki
Collaborators: Andy Nelson, Darrin Byler, Kenneth McClellan - LANL; Tiankai Yao, Bowen Gong, Spencer Scott, Jie Lian - RPI

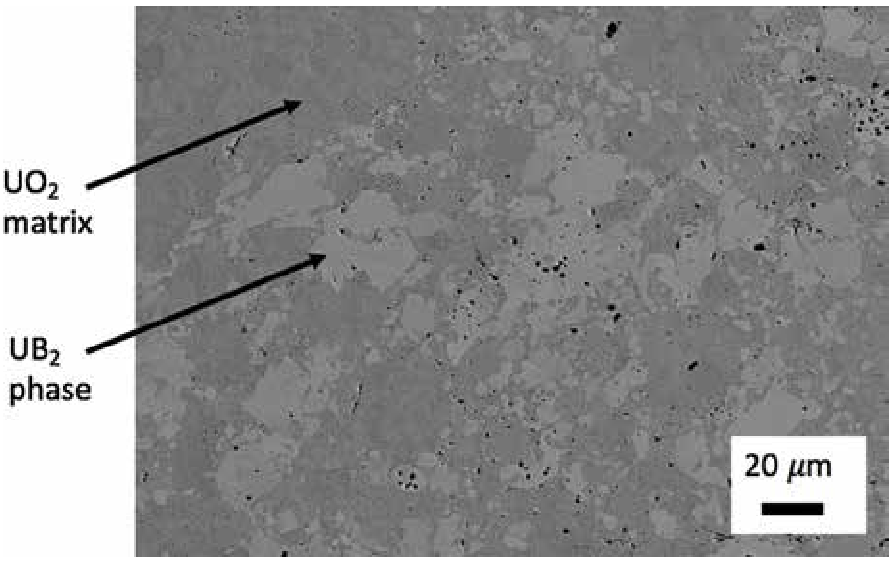


Figure 1. SEM backscatter image showing the typical microstructure of a UO_2-30%wt UB_2 sample.

Developing accident tolerant fuels (ATFs), for use in conventional light water reactors (LWRs) is a major goal of the nuclear energy community. Improved thermal conductivity, oxidation resistance, high fissile density and high melting point are all desirable properties for ATFs. Recent research has aimed to improve the thermal conductivity of ATFs via the addition of a second phase, with better thermal conductivity, in existing UO_2 fuels.

Difficulties exist during conventional fabrication of some UO_2-based ATFs due to phase interactions between the composite constituents at temperatures above those of in-service LWRs. Application of field assisted sintering (FAS) techniques to the development of a novel fabrication route for these ATFs is emerging as an important goal. FAS techniques include spark plasma sintering (SPS) and flash sintering (FS), both variations that use electric field and current to sinter powders at reduced times and temperatures. The operating conditions during FAS, therefore, enable a viable and cost-effective fabrication route for ATFs that cannot be fabricated conventionally.

Project Description:

Composites of UO_2 with phases of uranium borides, namely UB_2 and UB_4, can be proven to increase thermal conductivity thus providing better safety margins in an accident scenario. Since ^{10}B is a neutron absorber, tailoring the $^{10}B/^{11}B$ enrichment ratio would enable a built-in burnable poison. An important parameter when assessing the feasibility of UO_2-UB_x composites is the required phase fraction of UB_x to provide a significant increase in thermal conductivity, compared to UO_2. If large phase fractions are required, then the necessary enrichment process may make this fuel too costly to implement. Nonetheless, UB_x can still be included as a third phase in a different ATF composite system where the required volume fraction would be small enough that no enrichment would be necessary. In this case also, the volume fraction can be tuned to act as an efficient burnable absorber.

Uranium boride phases can be added to composite fuels to improve thermal conductivity and provide a built-in burnable poison.

Accomplishments:

In this work, UO_2-UB_2 and UO_2-UB_4 composites and the end members UB_2 and UB_4 have been sintered by SPS at Rensselaer Polytechnic Institute (RPI) and the conditions necessary to produce high density pellets (>90% TD) have been assessed.

Arc-melting was used to produce the UB_2 and UB_4 buttons that were then crushed, milled and sieved to prepare the UB_2 and UB_4 feedstocks used in this work. X-ray diffraction (XRD) characterization showed that phase-pure materials were obtained through this method. These feedstocks were blended with $UO_{2.00}$ feedstock to create UO_2 composites with 5, 15 and 30%wt of both UB_2 and UB_4 phases. Pure UB_2 and UB_4 samples were also prepared alongside the UO_2-UB_2 and UO_2-UB_4 composites.

Various heating profiles were tested to optimize the sintering parameters during SPS. It was found that the sample composition heavily influenced the sintering temperature with pure borides requiring higher temperatures to sinter to high densities. While the UO_2-based composites sinter to high densities at 1773 K in 5 minutes, the pure UB_2 and UB_4 samples had to be sintered at 2023 K in order to achieve densities larger than 90% TD. A typical microstructure, taken in backscatter mode, from a polished UO_2-30%wt UB_2 sample is shown in Figure 1. The islands of UB_2 are rounded indicating that it is possible the UB_2 phase was partially melted during SPS. The distribution of the UB_2 phase is shown to be fairly uniform and no additional phases are currently observed.

Thermal diffusivity values collected from LFA measurements performed at LANL, for UO_2-UB_2 (squares) and UO_2-UB_4 (circles) composites, sintered at 1773 K are shown in Figure 2, together with reference data from Fink [1] for UO_2 and from White [2] for UB_2 as a function of temperature for the range of 300-873 K. The thermal diffusivity for all composite samples was increased, when compared to the diffusivity of UO_2, and sample UO_2-30%wt UB_2 showcased the largest increase with respect to UO_2.

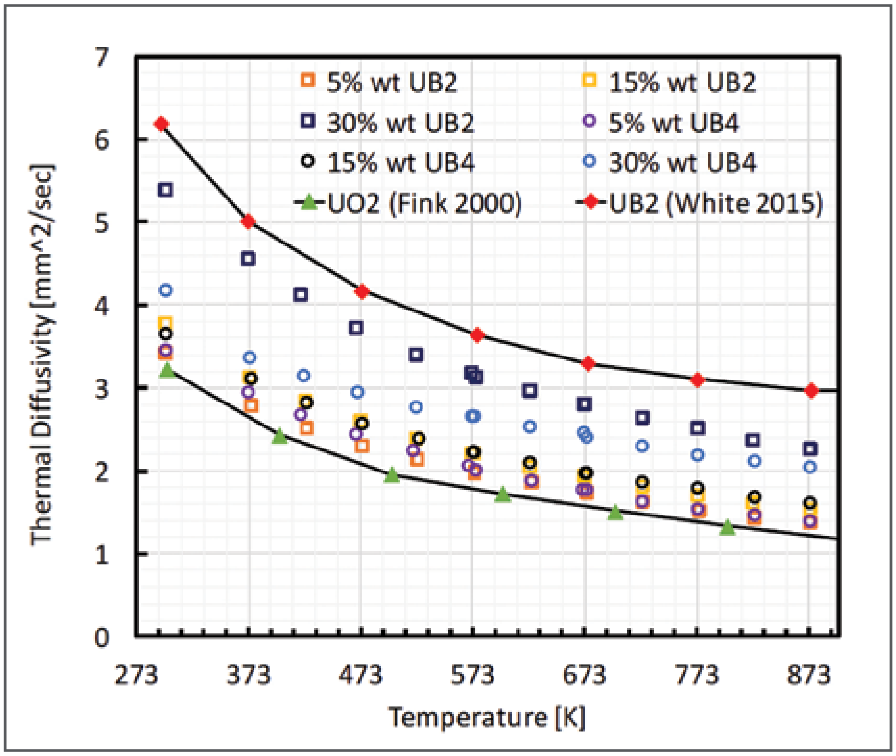


Figure 2. Thermal diffusivity as a function of temperature for UO_2-UB_2 (squares) and UO_2-UB_4 (circles) composites along with reference thermal diffusivity data on UO_2 [3] and UB_2 [2].

Thermal conductivity values have been calculated for the UO_2-UB_2 composites from measured thermal diffusivity values and from calculated, based on the rule of mixtures, specific heat and thermal expansion values and are show in Figure 3. The specific heat values for the composites were calculated based on UB_2 data from Fredrickson et al. [3] while the thermal expansion of UB_2 was taken from White [2]. Equivalent data for UO_2 were taken from Fink [1]. The thermal conductivity of these samples was calculated to 873 K. As expected, based on the thermal diffusivity data, the thermal conductivity of UO_2-UB_2 samples were shown to increase as the weight fraction of the UB_2 phase increased. As with the diffusivity data, the UO_2-30%wt UB_2 sample was calculated to have the highest increase in thermal conductivity, namely 50%, compared to UO_2.

An initial screening of the thermo-physical properties of the produced UO_2-UB_2 composites was carried out from 300 to 873 K and a significant increase in the thermal conductivity, compared to UO_2, was found for a UO_2-30%wt UB_2 sample. This deems the UO_2-UB_2 composite system as a potential candidate for ATF, although the economic impact due to enrich-

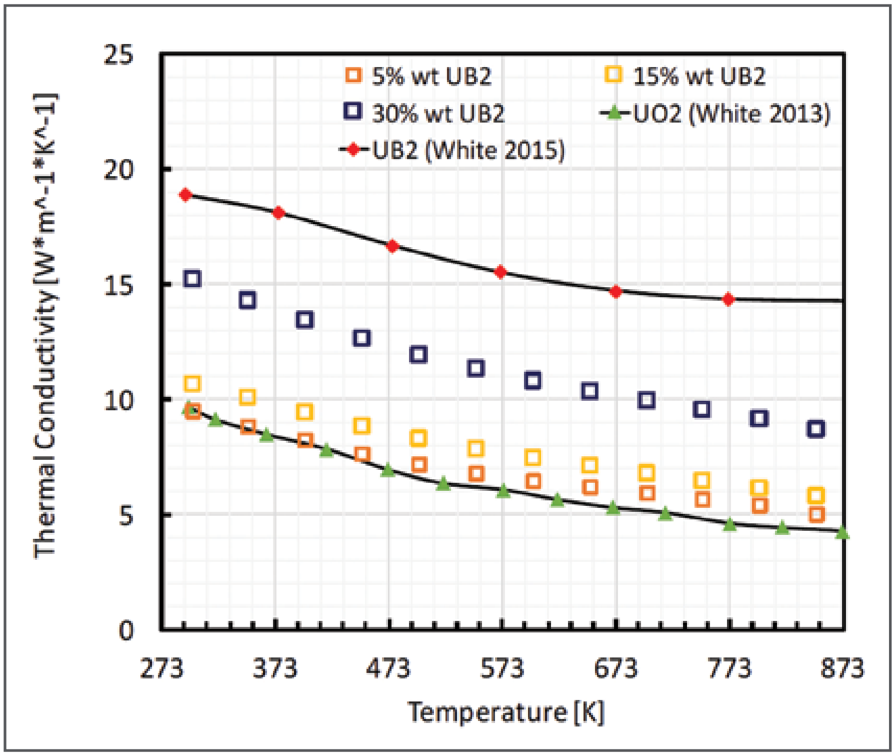


Figure 3. Calculated thermal conductivity values as a function of temperature for UO_2-UB_2 composites (squares) together with reference thermal conductivity data on UO_2 [1] and UB_2 [2].

ment of ^{10}B to ^{11}B will have to be taken into consideration. The results presented here will serve as a basis for the assessment of the UO_2-UB_x fuel forms. Further high temperature evaluation of the thermophysical properties of this fuel are necessary to effectively map out its accident tolerant capabilities.

[1] J. Fink, "Thermophysical properties of uranium dioxide," J. Nucl. Mater., vol. 279, no. 1, pp. 1–18, 2000.

[2] J. T. White, "Report on the basic chemistry, microstructure and physical properties of high uranium density boride compounds," 2015.

[3] D. R. Fredrickson, M. G. Barner, R.D., Chasanov, R. L. Nuttall, R. Kleb, and W. N. Hubbard, "The Enthalpy of Uranium Diboride from 600 to 1500 ° K by Drop Calorimetry," HIgh Temp. Sci. 1, vol. 80, pp. 373–380, 1969.

[4] J. T. White and A. T. Nelson, "Thermal conductivity of UO2+x and U4O9-y," J. Nucl. Mater., vol. 443, no. 1–3, pp. 342–350, 2013.

Assessment and Improvement of Handling for High Density Fuel Materials

Principal Investigators: J.T. White
Collaborators: S.S. Parker, A.T. Nelson

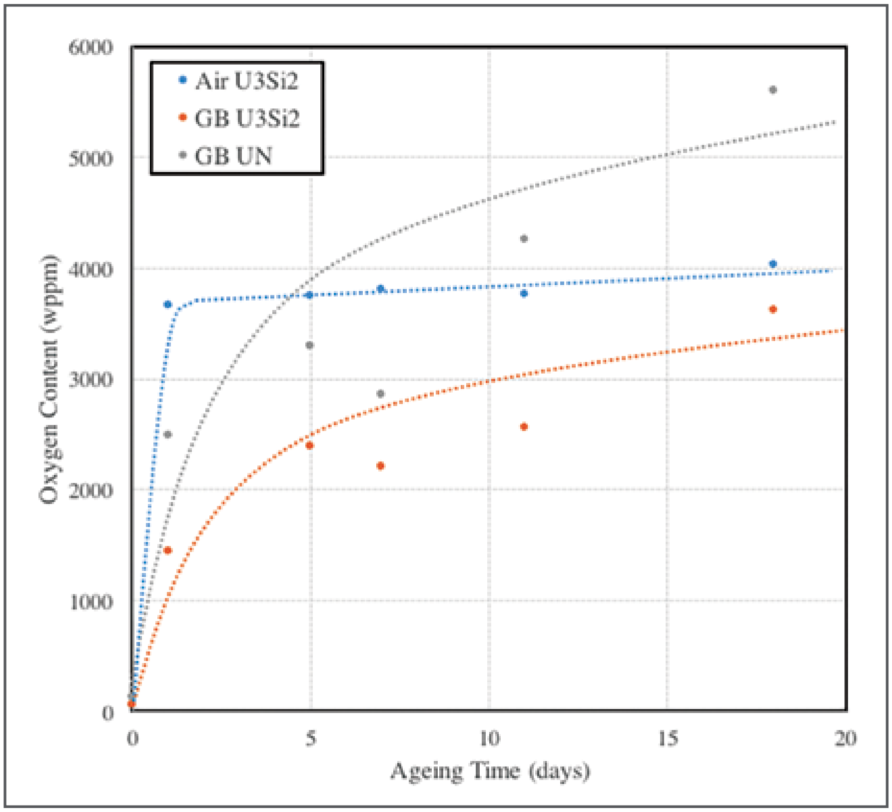


Figure 1. Oxygen uptake measurements conducted on powders of U_3Si_2 and UN in an atmosphere controlled glove box or in air.

Processing of high uranium density fuel forms (e.g. U_3Si_2, UN) in a commercial setting presents challenges because of the propensity of the comminuted powders to adsorb oxygen when exposed to air. The formation of oxides on the surface of the powders is detrimental to the sintering behavior of the powder compact and in the worst-case scenario can even lead to metal fires if the particle size of the feedstock is small enough. To prevent this, feedstock powders were combined with a large volume fraction loading of polymeric binders, which aimed to coat the powders, mitigating surface oxidation and allowing air-handling prior to sintering in a low oxygen-environment furnace.

Further study and assessment of this phenomenon will have significant implications in the ability to control sintering and microstructure of mixed oxide transmutation fuels. In addition to the effect on sintering rate, O/M control during densification was observed to strongly affect the final density and microstructure of both $(U,Pu)O_2$ and $(U,Ce)O_2$ pellets.

Sintering under hypostoichiometric conditions was observed to limit grain growth; this allows porosity to remain located on grain boundaries where their elimination is possible. Excessive grain growth as occurs at higher O/M instead was observed to trap porosity at grain boundary interiors, thus limiting ultimate attainable densities.

Project Description:

This project was divided into four parts: 1) examine the rate at which oxygen contaminates the high uranium density fuels in various processing environments, 2) investigate different methodologies to incorporate high binders loadings to UO_2 and develop methods to carefully debind the organic component from the green pellet without introducing cracking, 3) apply the debinding strategies to U_3Si_2 and allow exposure of the powders to air prior to sintering, and 4) compare the thermophysical properties of processed under different sintering atmospheres to pellets fabricated using traditional routes.

To understand to influence of oxygen adsorption in high density fuel forms, powder distributions typically used to sinter U_3Si_2 and UN were subjected to air and glove box environments and subsequently analyzed for oxygen impurities and phase formation as a function of time over a period of multiple months. Research on the high binder loaded systems initially focused on characterization a suite of prospective binders, which were down-selected for the UO_2 study. The chosen binder formulation was systematically incorporated to UO_2 powders to understand how binder distributions could be controlled and carefully removed from the green compact using a thermogravimetric analyzer (TGA) under different sintering atmospheres without the concern of oxidation impacting the results. The debinding strategies under the various atmospheres were then applied to U_3Si_2 in a TGA and in a low oxygen content metal furnace with and without exposure to ambient air. Finally, the thermal diffusivity of U_3Si_2 was characterized on pellets sintered in H_2 and vacuum environments.

The findings of this study have provided new insights into avenues for large scale processing of U_3Si_2 and similar fuel materials to help enable commercialization of these ATF concepts.

Accomplishments:

High purity (<400 wppm O) UN and U_3Si_2 was prepared that was subsequently milled and sieved to a -325 mesh size and exposed to a 40 vppm O glove box atmosphere and air. UN material that was exposed to energetically reacted with air forming U_3O_8 and UO_2. The other three powder sets were analyzed for oxygen content using the inert gas fusion technique and XRD for phase content over the course of two months, the results of which are shown in Figure 1. UN exposed to the glove box environment did not saturate over

This work highlights ongoing efforts to mitigate oxidation of high uranium density fuel powders, which should relax the air handling constraints of these fuels in a commercial environment.

the course of the experiment reaching a maximum of 5700 wppm O, likely from the smaller particle size distribution. U_3Si_2 rapidly adsorbed oxygen ~4000 wppm O when exposed to air and gained similar quantities albeit over a slower rate in the glove box atmosphere.

Research on the high binder additions initially focused on characterization of a suite of polymeric materials in a simultaneous thermal analyzer (STA) to determine the respective melt points and decompositions of each compound. The commercial polymers were selected to have minimal oxygen content to prevent the detrimental oxide contamination during the debinding stages. Binders were then down-selected and incorporated into UO_2 feedstock at 30 vol% loading using a variety of methods to improve the distribution of polymeric materials in the UO_2 matrix. Through the aid of TGA, a thermal debinding schedule was developed by controlling the mass loss rate on the high binder loaded pellets yielding pellets in the ~91% dense range. The semi-optimized debinding and sintering cycle was ~62 hours in length. When thermal debinding was not controlled, large voids/cracks formed and pellet densities were in the ~84-86% range.

High binder loaded U_3Si_2 green pellets were prepared in a glove box and either exposed to air or sintered in the as fabricated state. Debinding was performed using the same profile developed using the TGA. The results for the glove box prepared specimen yielded a 91% dense pellet, while the green pellet exposed to air for one hour resulted in an 89% dense pellet. If the binder is not included, U_3Si_2 pellets exposed to air typically are in the 80% theoretical density range. This initial result highlights the potential of this approach to facilitate air handling of U_3Si_2 powders without sacrificing final density.

Finally, the effect of sintering under vacuum and hydrogen, without high binder additions, on thermal transport properties was characterized against the standard reference sintering approach. Vacuum sintering runs have been used within the campaign as an alternative to sintering in a tungsten metal furnace. Hydrogen-containing atmospheres are a traditional means of mitigating deleterious effects of oxygen on sintering. However,

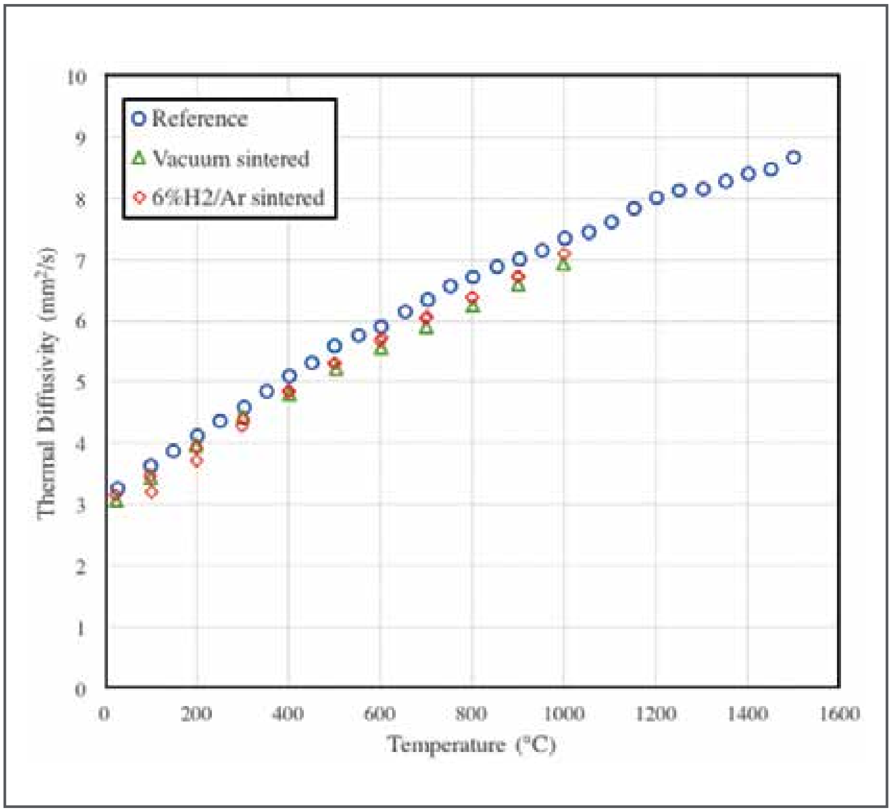


Figure 2. Characterization of the thermal diffusivity for pellets of U_3Si_2 sintered in Ar (reference), vacuum, and 6%H_2/Ar.

preliminary efforts in the synthesis of U_3Si_2 and other U-Si compounds found that hydrogen atmospheres produced poor results. It is now known that this was caused by hydriding of the U_3Si_2 feedstock at low temperatures. Work here used standard argon at low temperatures before switching to a 6%H_2/Ar mix at 1000°C. Hydrogen was removed from the furnace prior to cooling.

The results are summarized in Figure 2, where laser flash analysis data is presented. Both vacuum and hydrogen pellets are slightly less dense than the reference data (approx. 90% theoretical density) which accounts for much of the difference in thermal diffusivity. The vacuum sintered material showed similar diffusivity to the reference Ar sintered pellet when corrected for density, while the sample sintered in 6%H_2/Ar displays hysteresis at temperatures below 400°C. This is presently attributed to microcracking induced by minor hydride formation caused by incomplete removal of hydrogen from the system, but further work is needed to substantiate this hypothesis.

Separate Effects Investigation on the Role of Oxygen on Fresh Fuel Properties of U_3Si_2

Principal Investigators: J.T. Dunwoody
Collaborators: J.T. White and A.T. Nelson

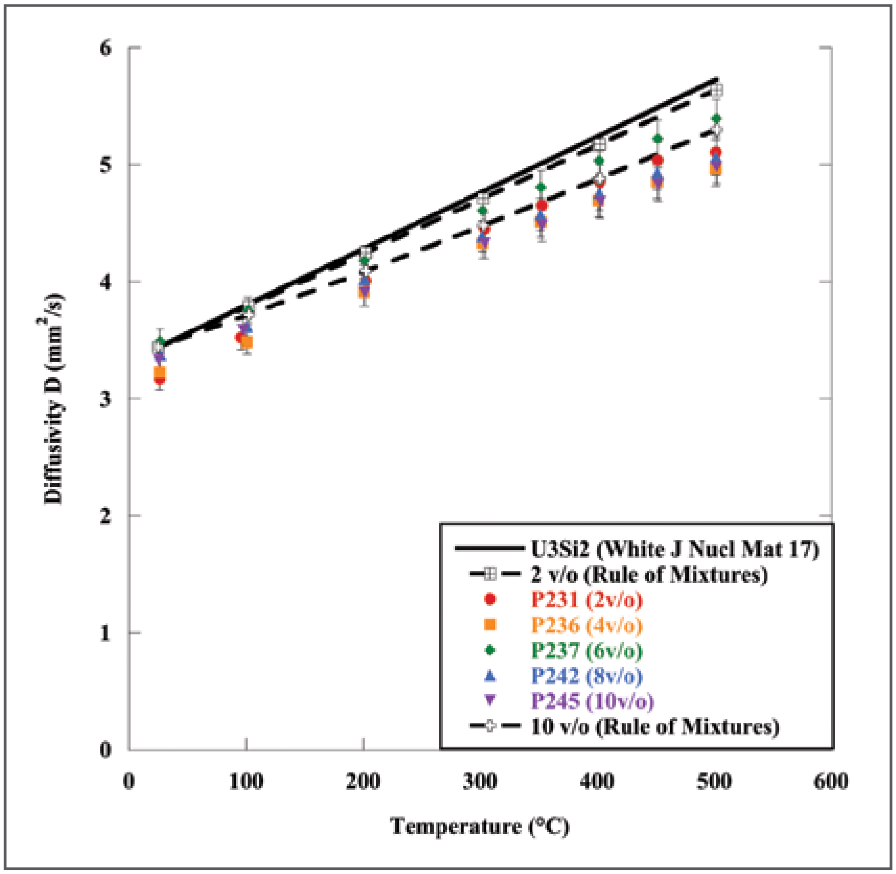


Figure 1. Thermal diffusivity collected for U_3Si_2 pellets containing varying levels of oxygen. Five sample sets were synthesized with feedstocks containing oxygen at levels sufficient to produce 2, 4, 6, 8, and 10 volume percent UO_2. The 2% and 10% trend lines are the degradation expected for the respective volume fractions UO_2 in a U_3Si_2 matrix as calculated from a rule of mixtures approximation. Data for high purity, high density U_3Si_2 is also shown for reference.

A major challenge presented by numerous ATF concepts is the scale of fabrication necessary for commercialization. Industrial fabrication of U_3Si_2 fuel will likely require modifications from the controls of the laboratory as presently employed. One likely outcome of adaptation to higher throughput production will be increased oxygen content. A significant unknown for this fuel system is how this oxygen affects the properties of both fresh fuel and irradiation behavior. A 'separate effects' study was conceived to synthesize U_3Si_2 fuel pellets with controlled levels of oxygen in order to assess the impact on fresh fuel properties.

Project Description:

A number of factors (e.g. feedstock synthesis, storage, possible air exposure, sintering environment) have been found possible of increasing the oxygen content in U_3Si_2 feedstock. Presently U_3Si_2 and other U-Si compounds are synthesized from metallic uranium; while arc melting is suitable for the laboratory setting and is presently employed to fabricate enriched material at the 100g to even kg scale, it is not economically viable at quantities necessary for commercial fuel production. A number of approaches are currently under investigation as means of direct synthesis of U_3Si_2 from a U-F precursor as well as means of relaxing the handling constraints currently employed to fabricate U_3Si_2. Any of these proposed routes would result in oxygen levels in U_3Si_2 at likely higher levels than are found in U_3Si_2 as produced in the laboratory. Further, process controls will eventually be needed to help constrain proposed industrial processes with regard to oxygen content.

The 'separate effects' approach to fuel development is well suited to address this problem, as U_3Si_2 samples can be produced with controlled oxygen levels and limited variations to other microstructural or chemical variables. This facilitates assessment of oxygen's impact, first on the properties of fresh fuel but additionally to irradiation testing. Success in this work demonstrates the value of a science-based approach to fuel qualification as employed by AFC for executing work to support commercialization of ATF concepts.

Accomplishments:

A series of U_3Si_2 feedstocks were synthesized to contain a range of oxygen contents at LANL. Nominally pure U_3Si_2 high density pellets contain approximately 500 wppm oxygen. The thermodynamics of the U-Si-O system dictate that oxygen introduced into the material either by feedstock synthesis, ageing, fabrication, or other means is quickly incorporated as UO_2. Feedstocks were produced containing nominally 2000, 4000, 6000, 8000, and 10,000 weight percent oxygen. For the U_3Si_2+O system, these levels of oxygen correspond to volume fractions of approximately 2, 4, 6, 8, and 10% UO_2 within the uncertainty of the synthesis method (+/- 2%).

The oxygen-containing feedstocks were then fabricated into pellets using the standard U_3Si_2 sintering procedure. This yielded property measurement samples of better than ninety percent theoretical density sized to accommodate a range of techniques. The initial focus of investigation was thermal conductivity. The thermal conductivity of U_3Si_2 is far superior to UO_2; this provides one of the main advantages for a U_3Si_2 fuel as centerline temperatures and stored energy is significantly reduced. However, the incorporation of oxygen as UO_2 into U_3Si_2 pellets poses a concern for reactor operation as the UO_2 would be expected to degrade thermal conductivity through either presence of the oxide phase or by internal cracking that could be caused by thermal expansion mismatch between the two materials.

The results of laser flash analysis examination of a number of U_3Si_2 samples containing varying oxide contents are plotted in Figure 1. Thermal diffusivity is a measure of a number of intrinsic material behaviors that affect heat transfer. Perturbations to thermal diffusivity correlate directly to thermal conductivity when material composition is not significantly

Use of a separate effects approach shows potential to isolate effect of industrial fabrication on properties of unirradiated U_3Si_2.

altered, as is the case in this study. Figure 1 plots results for samples synthesized in this study alongside data for high purity U_3Si_2 and also the thermal diffusivity that would be expected for U_3Si_2 containing 2 and 10 volume percent UO_2 based upon a rule of mixtures calculation. Results highlight that while oxygen contents up to 10 volume percent (approximately equal to 10,000 wppm oxygen in U_3Si_2 feedstock) are found to have an overall minimal effect on thermal conductivity, the degradation appears to be slightly more than would be expected due to UO_2 alone. It is hypothesized that this effect may be caused by micro-structural evolutions more complex than captured by a simple model of a UO_2-U_3Si_2 composition. Further analysis of the microstructures of these samples is ongoing to better understand this behavior and its implications on use of U_3Si_2 as a commercial LWR fuel.

Advanced LWR Fuel Concept Analysis

Principal Investigators: M. Todosow
Collaborators: L.-Y. Cheng, A. Cuadra

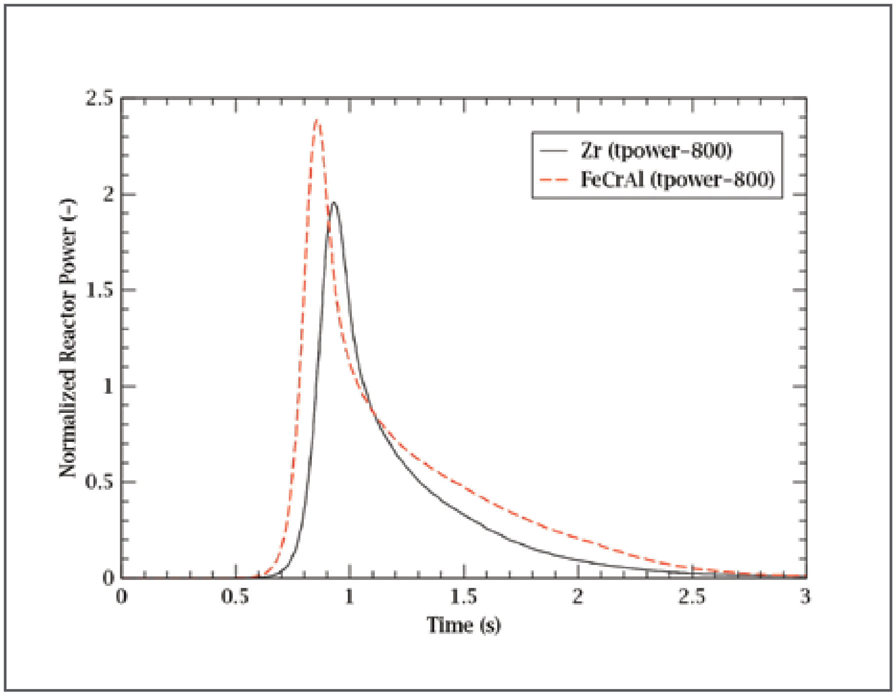


Figure 1. Normalized reactor power in a $1.05 RIA at HZP.

In the aftermath of Fukushima, a focus of the DOE-NE Advanced Fuels Campaign has been the development of advanced nuclear fuel and cladding options with the potential for improved performance in an accident; these are referred to as Accident Tolerant Fuels (ATF). An assessment of the impacts of advanced LWR fuels/cladding on reactor performance and safety characteristics is needed to identify potential issues associated with their viability/desirability for potential implementation in commercial reactors.

Project Description:

The analysis effort provides support to identify promising advanced fuels/ATF candidates for further development. The project performs analytical assessment that includes neutronics analyses to evaluate the impact on performance parameters (e.g., cycle length/burnup, power distributions, etc.), safety-related characteristics (e.g., reactivity and control coefficients/worths, kinetics parameters, etc.), and performance in a broad spectrum of transient and accident scenarios. The key components of

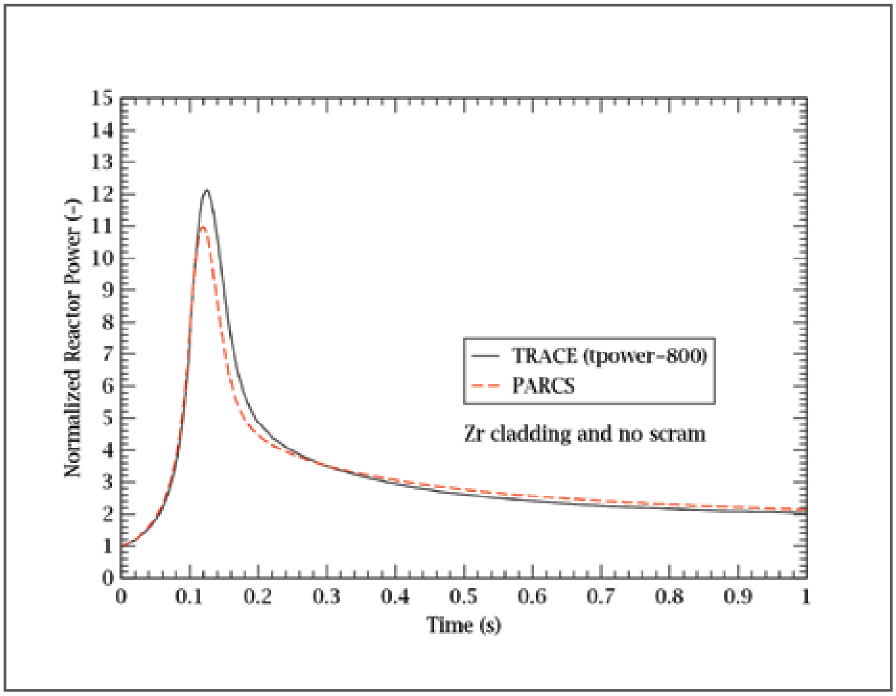


Figure 2. Comparison of normalized reactor power in a \$1.05 RIA at HFP.

the assessment are initial screening analyses, three-dimensional core analyses, and reactor system analyses. The integrated analytical approach starts with an initial screening using infinite lattice physics calculations. Three-dimensional core analyses then provide fuel cycle performance data and reactor kinetics parameters for time-dependent accident analysis. Reactor system analyses finally provide estimates of safety margins and "coping time", i.e., time to various limiting/failure conditions, under various transient accident scenarios. The search for attractive and viable advanced fuels/ATF candidates requires consideration of a broad spectrum of potential fuel and cladding options. Evaluation of the performance of these fuel and cladding options is vitally important, because it identifies whether the options have at least equivalent performance to the present UO_2-Zr fuel-cladding system under nominal conditions and improved performance in accident scenarios. Both of these characteristics are necessary in order for any ATF concept to be adopted by industry and utilities.

4.9%-enriched			
Fuel/Clad	UO_2-Zr	UO_2-Mo/Zr	U+10 w/o Mo-Zr
Discharge burnup (GWd/t)	61.6	50.2	53.6
Cycle length in effective full-power days (EFPDs).	533	518	465

Table 1. Three-Batch Cycle Length for Composite UO_2-UN Fuel Plus Zr Cladding

Accomplishments:

Activities in FY17 continued the assessment of ATF concepts and assembly-level calculations focused on composite UO_2-UN fuels with Zircaloy cladding. Results for the UO_2-UN/Zircaloy configurations based on a detailed, explicit model of a 17x17 pressurized water reactor (PWR) assembly with 4.9 w/o enrichment are shown in Table1. for discharge burnup (Gigawatt-days per metric-ton) and cycle length (Effective-Full-Power-Days). Two "bounding compositions for the nitride fuel were considered, i.e., 10% and 40%. For each composition, two cases were evaluated assuming 100% N-14 and 100% N-15, respectively to quantify the poisoning impact of N-14. Based on the results in Table 1, a 10% volume fraction of UN would have a relatively minor impact, while a 40% fraction would result in a significant penalty unless the nitrogen was fully enriched in N-15.

Based on detailed TRITON assembly calculations for UO_2/Zr and UO_2/FeCrAl, PARCS Beginning-of-Life (first core) models were used to calculate kinetics parameters and power distributions (including pin powers) for Hot-Zero-Power (HZP) and Hot Full Power (HFP) for an AP1000-like core. A $1.05 reactivity insertion accident (RIA) was analyzed using conventional point kinetics with PARCS and TRACE, for both HZP and HFP conditions. Figure 1. shows the normalized reactor power for the RIA. The FeCrAl-cladded fuel with its shorter neutron lifetime reached a higher peak power at a slightly earlier time than the Zr-cladded fuel. In both cases the negative fuel temperature reactivity coefficient was responsible to limit the increase in reactor power. The reactor scram on power occurred after the peak power was reached. Figure 2. shows a comparison of the RIA at the HFP condition as predicted by the TRACE and PARCS code respectively. The treatment of feedback in the PARCS point kinetics model differs from TRACE in the way core parameters are averaged: square

This project evaluates the performance of advanced fuel/ATF options and identifies whether the options have at least equivalent performance to the present UO_2-Zr fuel-cladding system under nominal conditions and improved performance in accident scenarios.

power-weighted average for TRACE, and adjoint- and fission cross-section-weighted average for PARCS. Results indicate the two codes compare favorably for the initial phase of the transient when the reactor power is increasing and the final phase when the reactor approaches a higher steady power. TRACE appears to predict a higher and broader power peak than PARCS and is thus more conservative.

Additional loss-of-coolant accidents (LOCA) have been analyzed to evaluate the impact of burnup-dependent fuel pellet thermal conductivity on the peak cladding temperature. Results of a small-break LOCA analysis indicate that the increased stored energy due to the lowering of the thermal conductivity has little impact on the peak clad temperature (PCT). According to the transient results the PCT occurs some 80 s after the initiation of the accident and by then most of the stored energy has already been released from the fuel. It is anticipated that the thermal conductivity degradation (TCD) effect due to burnup will be more significant for faster transients such as a large-break LOCA.

An effort was initiated to benchmark results produced by the TRACE 3-loop PWR plant model against design basis accidents presented in the Safety Analysis Report (SAR) for a typical 3-loop plant. The comparison provides guidance on the modeling of the emergency core cooling systems (ECCS), especially in regard to their performance in terms of initiation and duration of the injection and the rate of injection.

2.4 ATF CLADDING AND COATINGS

ATF wrought FeCrAl tube production for Hatch-1 field test

Principal Investigator: Yukinori Yamamoto, Kevin Field (ORNL)
Collaborators: Kurt Terrani, Kory Linton, Maxim Gussev, Zhiqian Sun (ORNL), Raul Rebak, Russ Fawcett (GE)

Figure 1. Edwin I. Hatch Nuclear Power Plant (Baxley, GA) (Ref.: https://www.georgiapower.com/about-energy/energy-sources/nuclear/hatch.cshtml).

First mass production of ATF wrought FeCrAl alloy tubes, performed partially through commercial manufacturers and partiality at ORNL, was completed in FY2017. A Gen. II wrought FeCrAl alloy consisting nominally of Fe-12Cr-6Al-2Mo-0.2Si-0.05Y, in weight percent, was down-selected for the production. Seamless, thin-wall tube production process was conducted through commercially available multiple process steps including vacuum induction melt, hot isostatic press, hot-extrusion, gun-drilling, and multiple tube drawing process with optimized inter-pass annealing, to obtain sufficient deformability during production process and control the quality of the final tube products (e.g. microstructure, size tolerance, surface conditions, etc.). The tubes are to be used in lead test rods in Edwin I. Hatch Nuclear Power Plant, Baxley, GA (Figure 1), to assess the feasibility of newly developed, accident tolerant FeCrAl alloy cladding.

This project achieved the successful mass production of the seamless, thin-wall nuclear-grade FeCrAl tubes for the first time.

Project Description:

Technical objectives of this project are to evaluate the various process parameters and conditions of the tube production steps for Gen. II ATF wrought FeCrAl seamless tubes (Figure 2) produced through commercially available manufacturing processes, suggest optimized processing routes and parameters for mass production of the ATF wrought FeCrAl alloy tubes with the improved production quality and reasonable material yield, and deliver the tube products for field testing in a commercial nuclear power reactor (Edwin I. Hatch Nuclear Power Plant, Baxley, GA) to assess the feasibility of newly developed, accident tolerant FeCrAl alloy cladding in late 2017. Because of excellent steam oxidation resistance of FeCrAl alloys at elevated temperatures, the replacement of the current Zr-based fuel cladding with the ATF FeCrAl alloy tubes is expected to increase safety margin in LWRs during severe accidents. The present project revealed a high potential to transfer the ATF FeCrAl tube production technology into commercial manufacturers and discuss/develop more economical pathway to fabricate such tube products with sufficient quality to meet with the required nuclear grades.

Accomplishments:

Technical goal of the project is to deliver the ATF wrought FeCrAl alloy tubes with the improved production quality and sufficient lengths requested for the field testing. Based on the knowledge obtained regarding process conditions described below, the ATF wrought FeCrAl tube production with optimized process parameters was carried out. The seamless, thin-wall tubes were successfully delivered in June 2017 with the size of ~10 mm outer diameter, ~0.4 mm wall thickness (with a tight size tolerance), and total length of more than 200 ft.

Figure 2. Mock-up of ATF wrought FeCrAl alloy fuel cladding assembly.

The efforts initiated from the evaluation of commercially available tube production process steps by divining into three different sections; (1) the heat production process, (2) the master bar production process, and (3) the tube drawing process. For heat production, it was found that not only chemical composition control but also elimination of solidification defects in the vacuum induction–melted ingot is critical for improving the quality of the final products as well as the processability of the materials during the tube reduction process. A hot-isostatic press helped in reducing and eliminating the internal, pre-existing cracks in the ingot, which could be a source of premature failure (e.g., crack initiation from oxide particles) during the tube drawing process. The master bars were produced by a hot-extrusion

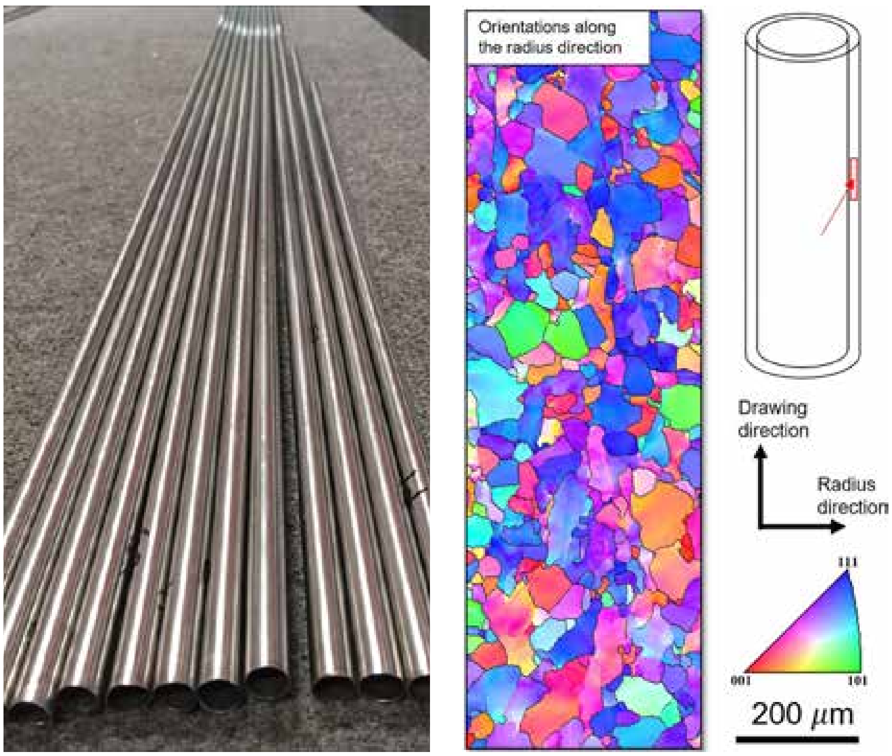


Figure 3. As-drawn ATF FeCrAl alloy tubes and microstructure characterization results.

process. The area reduction ratio and the extrusion temperature are key to controlling the quality of the master bars and the grain structure (Figure 3). Because of the limited deformation ability of FeCrAl alloys, especially at ambient temperatures, a controlled, uniform grain structure with relatively small grains (<100 μm) is required prior to the tube reduction process. A relatively small amount of area reduction per pass, combined with controlled inter-pass annealing, is necessary to achieve a reduction of premature failure events, a highly uniform wall thickness, concentricity, and a high-quality surface finish.

Dual-Purpose Coating Development for SiC-Based LWR Cladding

Principal Investigator: Dr. Yutai Katoh (ORNL)
Collaborators: Dr. Caen Ang (ORNL), Dr. Stephen Raiman (ORNL), Dr. Xunxiang Hu (ORNL), Dr. Kurt Terrani (ORNL)

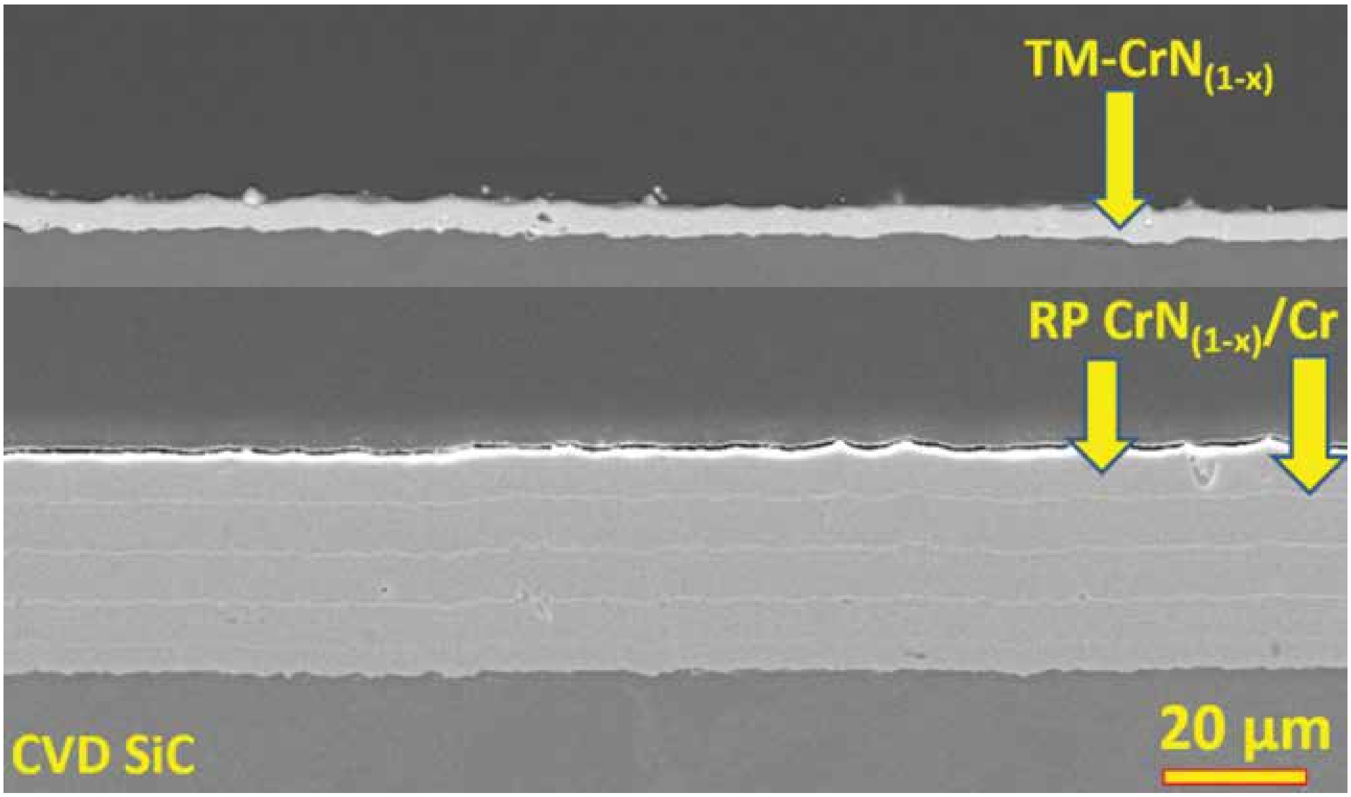


Figure 1. Cr multilayer coatings on SiC substrates deposited by physical vapor deposition process. RP and TM represent vendors that collaborated with the program.

Silicon carbide (SiC) continuous fiber-reinforced SiC matrix composite (SiC/SiC composite)-based fuel cladding is a concept that potentially enables an LWR core with ultimate accident-tolerance. For this concept to demonstrate the technical feasibility, chemical compatibility with the radiolytic water and containment of gaseous fission products are the two leading issues. Coating with a material that provides the corrosion barrier and the hermetic sealing is among the promising approaches to address these critical challenges. Technology development for this dual-purpose coating was conceptualized and pursued by a team of Oak Ridge National Laboratory researchers.

Project Description:

As the potential accident-tolerant fuel cladding material, SiC/SiC composites are known to combine

Significant progress was shown for dual-purpose coating approach to overcome the most critical technical feasibility issues for silicon carbide composite-based enhanced accident-tolerance fuel cladding concepts

the highest levels of high temperature strength, radiation stability, neutron transparency, and resistance to high temperature steam oxidation. However, a few critical technical issues remain to be addressed before these materials are considered truly viable for the proposed application: the hydrothermal corrosion of SiC and the microcracking of SiC matrix. The hydrothermal corrosion is intrinsic to SiC with an oxidizing water chemistry and is known to be accelerated by the increasing oxygen activity and the presence of ionizing radiation. The susceptibility of SiC matrix composites to develop the microcrack network implies there is the potential for compromised fission product gas retention. Presently there is no technology established to manufacture gas-tight SiC/SiC composite fuel cladding of full length. Moreover, the unique stress state evolutions during operations of the reactor impose a serious concern for additional microcracking.

Among various strategies proposed to date to mitigate these threats, the corrosion barrier – hermetic seal dual-purpose coating is a promising approach that presents potential to address both challenges. Development of the barrier coatings on LWR fuel cladding has actively been pursued for metallic substrates including zirconium alloys, stainless steels, and molybdenum with the purposes of enhanced corrosion resistance during normal and/or off-normal conditions. Some of the coatings have demonstrated promises; for example, physical vapor-deposited chromium nitride coating on Zr-alloy tubes proved to remain intact during testing in both PWR and BWR water loops of Halden reactor up to 287 full power days. Since the coating technologies are highly dependent on the substrate materials, significant effort will be required to develop adequate coatings for SiC/SiC composite substrate.

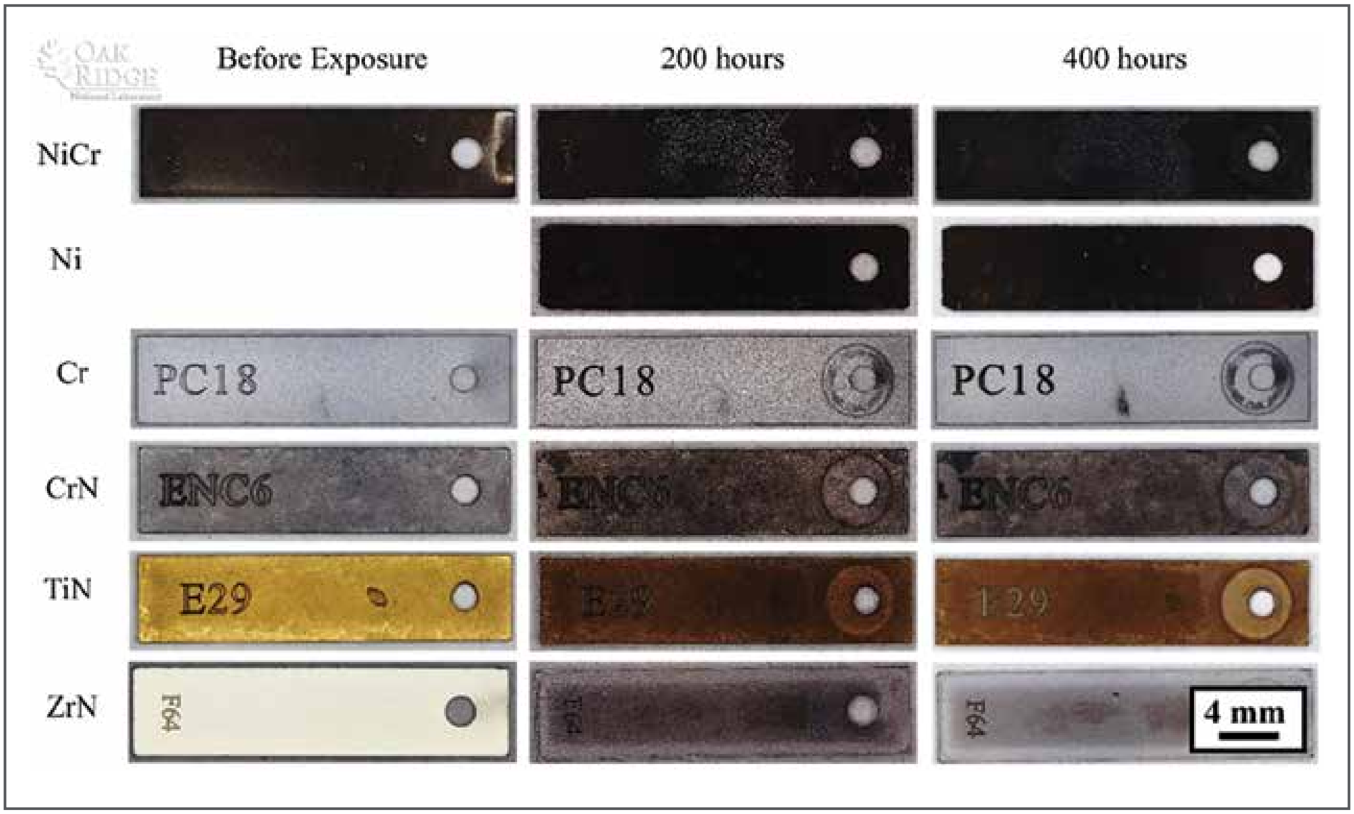


Figure 2. Optical micrographs of coated SiC coupons imaged before exposure to 300°C water, after 200 hours, and after 400 hours of exposure. Six different coupons are shown, each with a different coating applied to a SiC substrate. The Ni coated sample was not imaged prior to exposure.

Successful development of the dual-purpose coating technologies will be a critical step toward establishing the SiC/SiC composite-based cladding for the enhanced accident-tolerance LWR fuels.

Accomplishments:

To address the above-mentioned critical challenges, a dual-purpose barrier coating for SiC/SiC cladding is proposed by Oak Ridge National Laboratory. Identification of the appropriate coating materials, technologies and evaluation for baseline properties are the focus of the initial phase of the research and development. The coating materials selected for evaluation are of compositions that previously demonstrated promising performance in LWR environments when applied to metallic cladding substrates. However, the technologies to deposit these materials require significant effort to integrate with a SiC substrate. Three approaches at varying technological readiness levels were pursued: namely, electrochemical deposition (ECD), vacuum plasma spray (VPS), and cathodic arc physical vapor deposition (PVD). While the PVD technology required minimal adaptation for any substrate, ECD and VPS coatings needed extensive development for compatibility with the SiC substrate (Figure 1).

The initial effort led by Dr. Ang focused on the development of first-generation coatings through the three approaches. The ECD coatings are yet to prove of value due to unacceptable level of cracking after processing. The VPS coatings failed to deposit an intended phase of the coating material on the SiC substrate. Therefore, these two technologies were determined to require significant more efforts toward demonstrating the technical feasibility. On the other hand, the PVD technology resulted in apparently intact coatings with various materials. Therefore, selected PVD coatings of Cr, CrN, and TiN were further evaluated for mechanical properties, helium permeability, and hydrothermal corrosion.

The mechanical properties evaluation was also performed by Dr. Ang (Figure 2). All of the PVD coatings on the SiC/SiC composite substrates exhibited adequate tensile debond strength per ASTM D4541. However, the scratch testing according to ASTM E2546 indicated the highest delamination resistance for the metallic Cr coating and significant variations in delamination resistance for CrN coatings depending on the condition of deposition. The helium permeability testing performed by Dr. Hu revealed the gas tightness for all of Cr, CrN, and TiN coatings applied to small diameter SiC/SiC tube substrates as measured at room temperature up to slightly more than atmospheric pressure. Finally, the autoclave testing for hydrothermal corrosion performed by Dr. Raiman indicated that Cr and CrN are promising compositions for coatings.

The future works include additional characterization of as-coated and corroded materials, examination of the coated specimens following neutron irradiation in inert and PWR water loop environments, and the second round of coating optimization based on feedback from the present evaluation.

ASTM Round-Robin Testing for SiC/SiC Cladding Tube Strength

Principal Investigator: Dr. Yutai Katoh (ORNL)
Collaborators: Dr. Gyanender Singh (ORNL), Dr. Edgar Lara-Curzio (ORNL), Dr. Christian Deck (General Atomics), Dr. Stephen Gonczy (Gateway Materials Technology, Inc.)

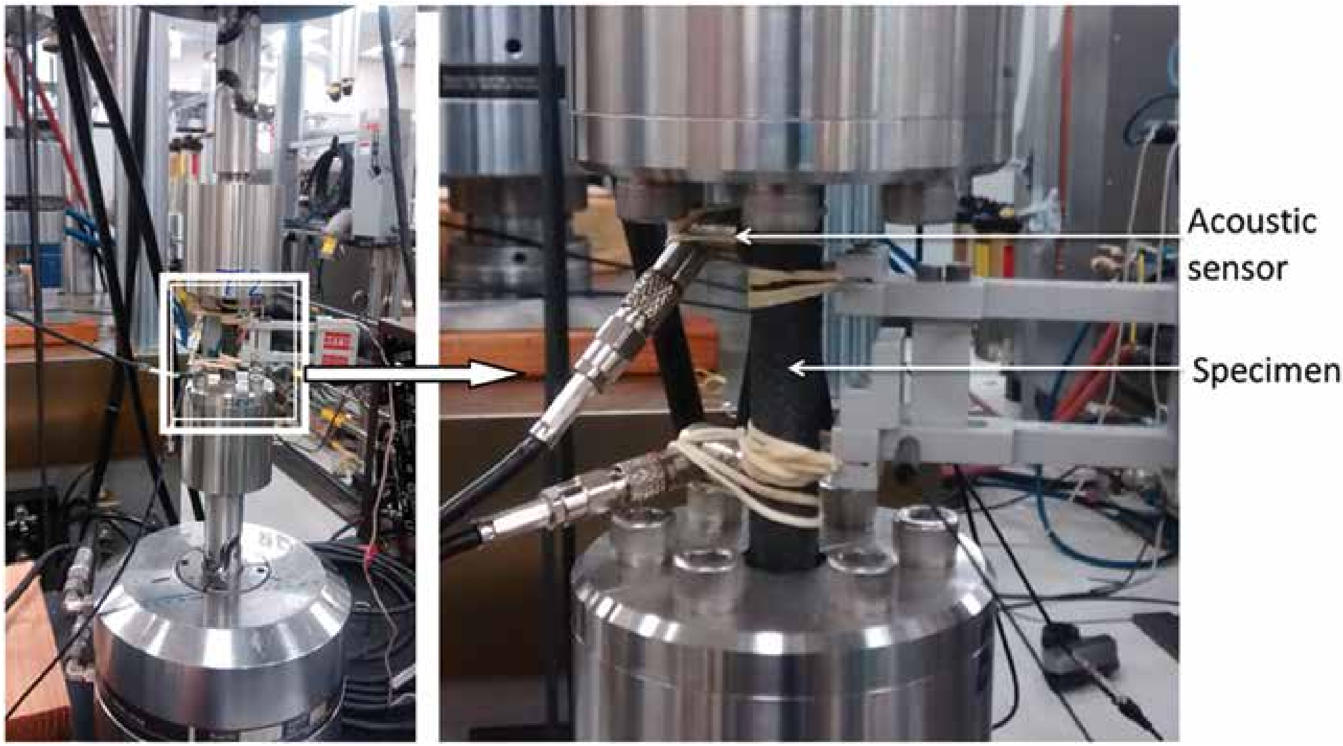


Figure 1. Setup for axial tensile testing of tubular geometry SiC/SiC composite test article at Oak Ridge National Laboratory. Arms from extensometer and transducers for acoustic emission detection are attached to the test specimen.

Silicon carbide (SiC) continuous fiber-reinforced SiC-matrix composites (SiC/SiC composites) are promising candidate materials for enhanced accident-tolerance fuel cladding. ASTM test standards are required for evaluating the properties of SiC/SiC composite tubes representative of fuel cladding. Thus, ASTM standards are necessary for assessing and certifying the adequacy of SiC/SiC composite material for cladding application. Moreover, in order to establish the failure probability of SiC/SiC cladding, it is essential to understand the statistical distribution of mechanical properties of SiC/SiC composite tubes of cladding-relevant architecture. The present work was intended to 1) determine statistics, 2) establish the precision and bias statements for the ASTM C-1773 test standard and 3) help establish database for the axial tensile properties of SiC/SiC composite tubes.

Project Description

SiC/SiC composites offer a set of properties that make these materials highly attractive for cladding applications in light water reactors with enhanced accident tolerance. However, the deployment and commercialization of components for nuclear applications manufactured using SiC/SiC composite require several important considerations. Of particular, a qualified database of the properties of SiC/SiC composite will be needed for performing rigorous design qualification. Such qualified database is one which is generated using procedures that comply with standards associated with the design code of the component of interest. The current database of mechanical properties for nuclear grade SiC/SiC composite is limited and inadequate for the purposes stated above.

Although various properties of nuclear grade SiC/SiC composites have been measured, tubular test specimens were not used in those studies. The ASTM standard test method for axial tensile properties (ASTM C1773-13) of continuous-fiber reinforced ceramic composite tubes has been developed only recently. This ASTM standard currently lacks the precision and bias statements which convey important information to the users of the standard about the quality of their test results.

The present work fills these gaps in the development process of SiC nuclear fuel cladding technology through coordinating an interlaboratory round robin study on the tensile properties of SiC/SiC tubular specimens with participation of laboratories from government institutions, academia and industry. This interlaboratory study served multiple objectives: 1) understand the statistical properties of prototypical SiC/SiC composite tubes for fuel cladding applications, 2) develop a precision statement for the ASTM C1773 Standard Test Method (Monotonic Axial Tensile Behavior of Continuous Fiber-Reinforced Advanced Ceramic Tubular Test Specimens at Ambient Temperature), and 3) expand the limited database of mechanical properties of SiC/SiC composite tubular structures.

Accomplishments

An interlaboratory round robin testing program was carried out for the axial tensile properties of SiC/SiC cladding tubes with the architecture prototypical to LWR fuel cladding. The tubular specimens were specifically designed for this interlaboratory round robin

Round-robin testing campaign across leading national laboratories helped complete an ASTM standard and establish properties database for SiC/SiC composite fuel cladding.

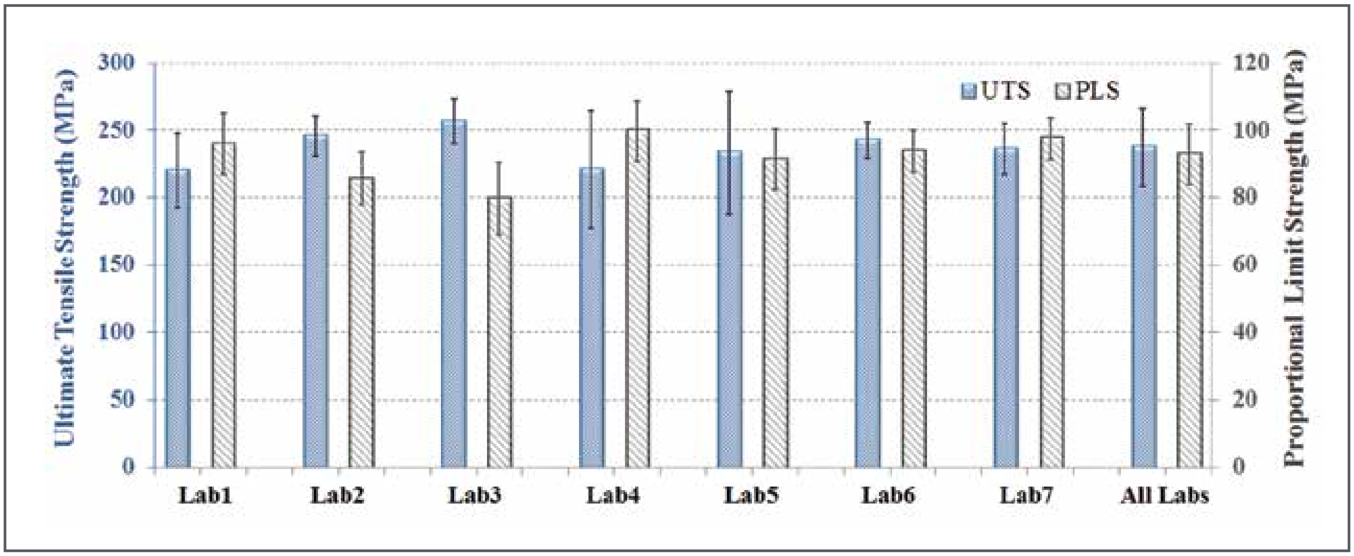


Figure 2. Ultimate Tensile Strength (UTS) and Proportional Limit Stress for each laboratory obtained from interlaboratory round robin testing (Error bar: ±1 Standard deviation about the mean).

testing and fabricated by General Atomics with the tri-axially braided Hi-Nicalon™ Type S fibers, nominally 150 nm pyrocarbon interphase, and the chemical vapor-infiltrated SiC matrix. Following the test procedure development and completion of the lead test led by Dr. Singh at Oak Ridge National Laboratory (Figure 1), a large number of test specimens were tested by seven member laboratories of ASTM Committee C28 on Advanced Ceramics. The participating laboratories include NASA Glenn Research Center, Southern Research Institute, General Atomics, Westinghouse Electric Company, United Technology Research Center, and General Electric.

The data sets (Figure 2) show reasonable consistency across the laboratories, verifying that the current C1773-13 ASTM standard is adequate for testing ceramic fiber reinforced ceramic matrix composite tubular specimens. Limited statistical variation in the mechanical properties demonstrates the adequate quality of the SiC/SiC material employed in these tests. The distribution of ultimate tensile strength data was best described with a 2-parameter Weibull distribution with shape and scale parameters of 10.1 and 249.1 MPa, respectively. A lognormal distribution, with log mean and log standard deviation of 4.52 and 0.096,

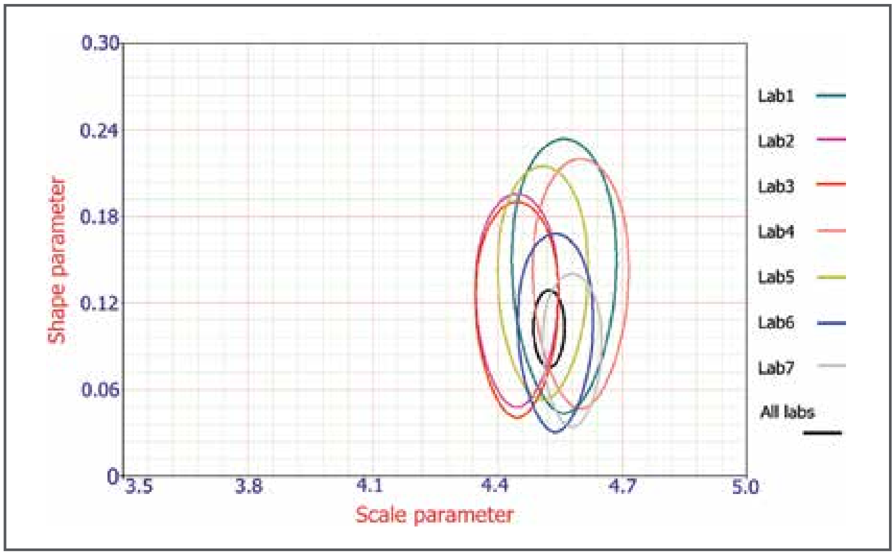


Figure 3. An example analysis for inter-laboratory round-robin test results – statistics for proportional limit tensile stress are compared across laboratories using lognormal distribution-based 95% confidence bounds.

respectively, provided a good description of the distribution of proportional limit stress data (Figure 3).

The study also led to the development of the precision statement for the ASTM C1773-13 standard.The precision statementwill be incorporated in the next revision of the ASTM standard The precision statement conveys the important information about the repeatability and reproducibility of the test results. Repeatability refers to the accuracy and random variability of the results obtained with the same method on identical test items in the same laboratory by the same operator using the same equipment within short intervals of time within a single laboratory. Reproducibility refers to the accuracy and random variability associated with successive measurements of the same property carried out by operators working in different laboratories, each obtaining single results on identical test material when applying the same method Thus, precision of a test method provides an estimate of the variation the user of the test method can expect, which in turn reflects the utility of the standard test method.

Combined Experiment and Modeling for Beyond Design Basis Accident Analysis

Principal Investigator: Kevin Robb
Collaborators: Bruce Pint, Kurt Terrani, Larry Ott

Concurrent high-temperature testing and accident modeling are informing FeCrAl ATF assessment.

To guide the development and prioritization of ATF concepts, there is a need to assess their performance under accident conditions. To perform these assessments, experimental data such as oxidation kinetics and material failure points are required. Iron-chromium-aluminum (FeCrAl) alloys are under active development to serve as core materials with enhanced accident tolerance. To accelerate development and deployment, concurrent efforts have been underway to optimize the alloy, generate the required test data, and conduct assessments of the ATF concept performance.

Project Description

The first-generation alloy, B136Y (Fe-13Cr-6Al), developed at ORNL has more favorable irradiation properties than that of the commercially available APMT alloy. However, unlike the APMT alloy, the oxidation kinetics for B136Y have not been developed. The B136Y alloy will be used in an upcoming QUENCH test at the Karlsruhe Institute of Technology. The QUENCH test will provide integral test data as to the performance of the alloy under conditions it could experience during an accident. The oxidation kinetics of B136Y need to be developed to: 1) inform and interpret the QUENCH test, and 2) provide more accurate material properties for use in modeling accidents during concept assessments.

Accomplishments

Sections of B136Y FeCrAl tubing were exposed to a steam environment in the Severe Accident Test Station High Temperature Furnace at ORNL (Figures 1 and 2). The furnace temperature was varied during two series of tests to simulate: 1) the anticipated conditions during the upcoming QUENCH test; and 2) the predicted prototypic conditions based on previous station blackout simulations. Through 15 tests, the oxidation kinetics rate constant for B136Y was found to be 3-5 times higher than that of the commercial APMT alloy. The tests have also provided confidence that the cladding will retain a coolable geometry up to 1500°C under temperature ramp rates characteristic of station blackout scenarios.

The MELCOR code was used at ORNL to perform integral simulations of BWR beyond design basis accidents. Based on the oxidation test results, the oxidation rate constant was varied from 1x-10x that of AMPT (Figures 3 and 4). Since the oxidation rate of FeCrAl is so low compared to zirconium based alloys, increasing the FeCrAl oxidation rate constant over this range was found to have only secondary or tertiary impacts on the predicted accident progression. The predicted gains afforded by the FeCrAl ATF concept were found to be consistent with previous year's work.

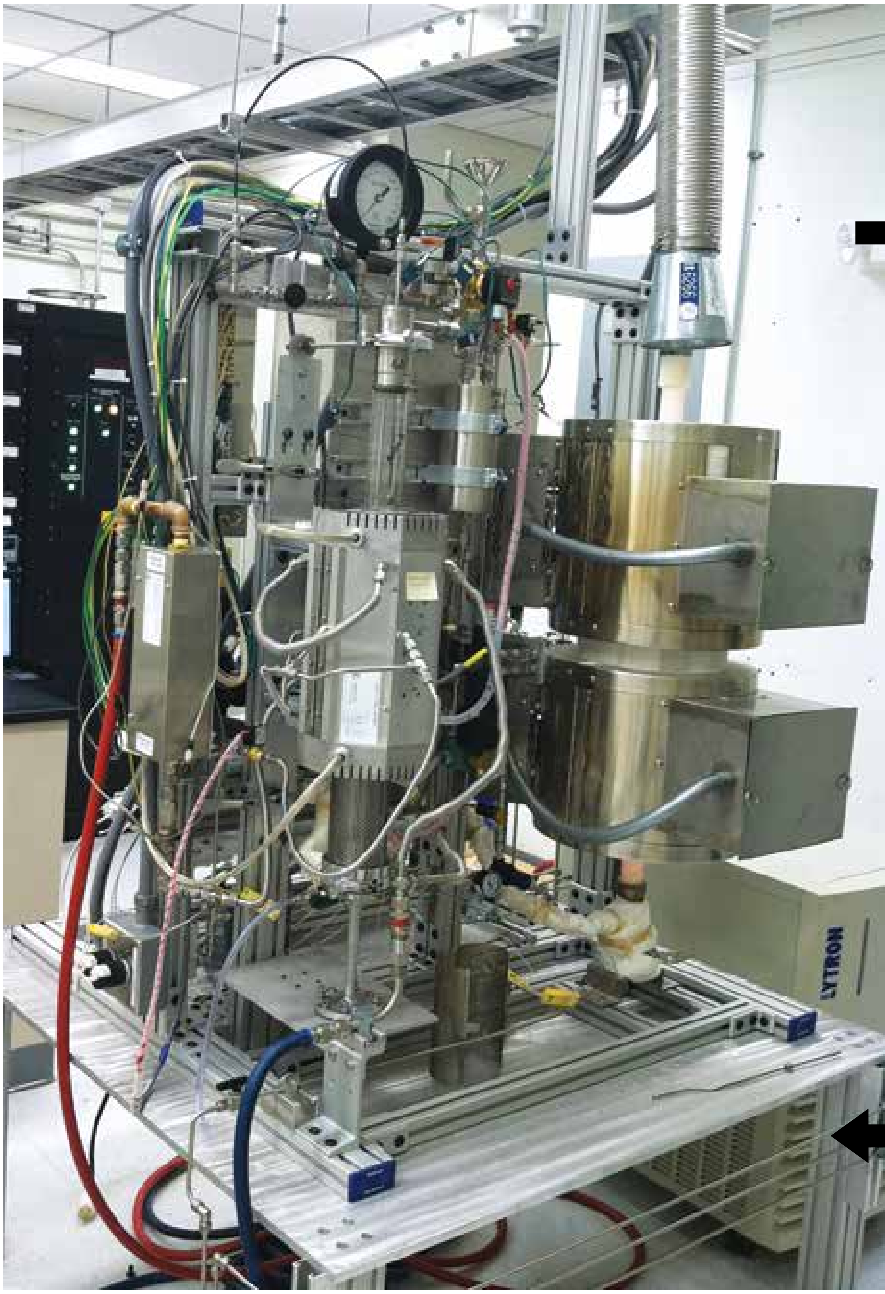


Figure 1. High Temperature Furnace

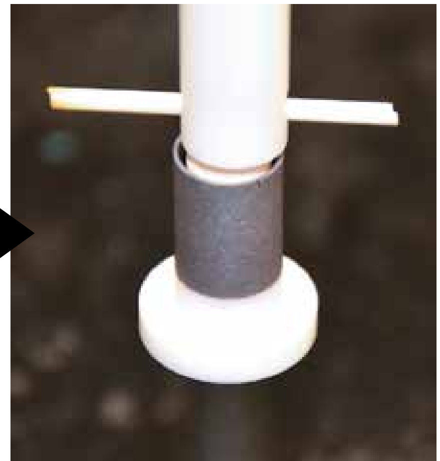


Figure 2. FeCrAl Oxidation Test Tube

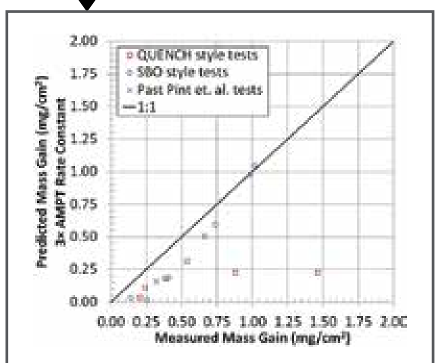


Figure 3. Oxidation Kinetics Test and Modeling

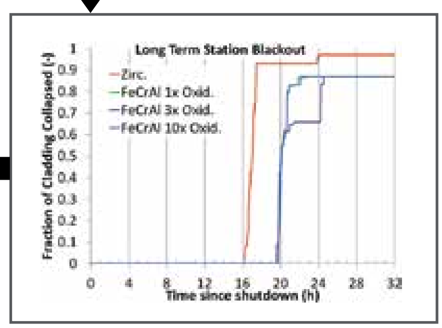


Figure 4. MELCOR LTSBO Accident Simulation

FeCrAl ODS Alloy Development for Fission Platforms

Principal Investigator: Sebastien Dryepondt
Collaborators: Caleb Massey, Philip Edmonson and Maxim Gussev

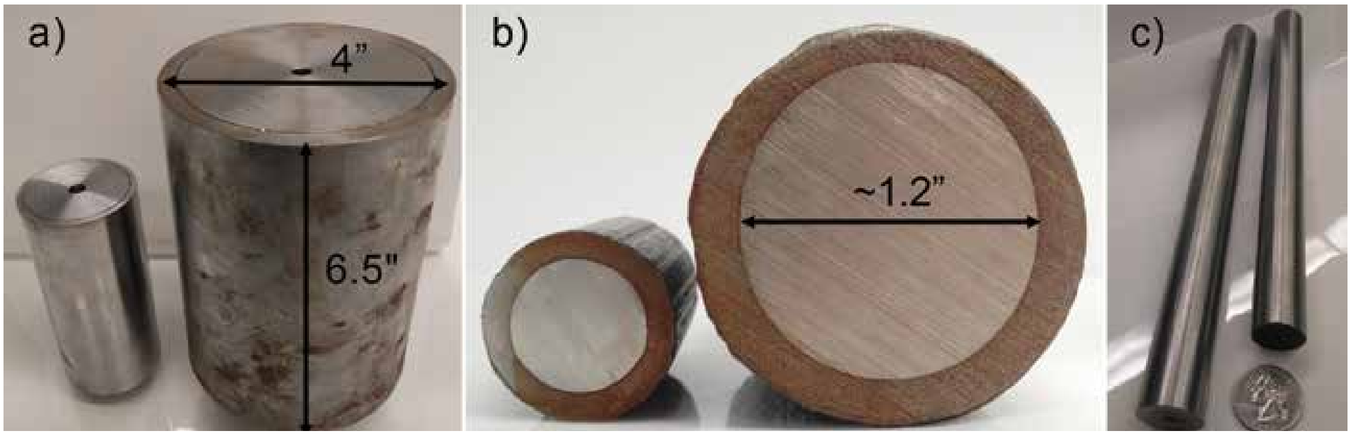


Figure 1. a) Extrusion cans, b) Cross section of the extruded bars, c) Final ODS 106ZY rods. a) and b) highlight the difference between previous small extrusions bars and the larger bars required for the production of the two rods presented in c).

Advanced wrought and oxide dispersion strengthened (ODS) FeCrAl alloys are among the leading candidates for next generation accident tolerant fuel cladding due to their high strength and excellent oxidation behavior at temperatures up to 1450°C. The superior mechanical strength of ODS FeCrAl alloys up to ~800°C allows for the use of thinner cladding, thus limiting the neutronic penalty from the replacement of Zr-based alloys by Fe-based alloys. The ODS FeCrAl high strength comes, however, with limited ductility making the fabrication of thin tubes very challenging.

Project Description:

The overall project goal is to develop new low-Cr ODS FeCrAl alloys exhibiting great oxidation resistance in steam at T>1400°C, good mechanical strength up to at least 800°C, superior irradiation resistance at 200-500°C, and sufficient ductility at low temperature to allow for the production of tubes less than 500 micrometers thick. Extensive microstructure characterization, tensile testing at 20-800°C, and steam oxidation testing at 1200-1450°C of several ODS FeCrAls has led to the selection of an ODS Fe-10Cr-6Al-0.3Zr+0.3Y_2O_3 alloy (106ZY). Previous work on small powder quantities also revealed that careful control of the powder ball milling and extrusion steps allows for the optimization of the ODS FeCrAl microstructure and properties. In particular, increasing the extrusion temperature decreased the alloy strength and hardness but increased the alloy ductility. The project is now focusing on the fabrication of large batches of ODS FeCrAl and the search for industrial partners to produce thin tubes. Understanding the effect of tube processing on the ODS FeCrAl microstructure is also key to optimize the final tube properties. The expected project outcome is the production of ODS FeCrAl cladding with enhanced safety margin for the current and next generation reactors.

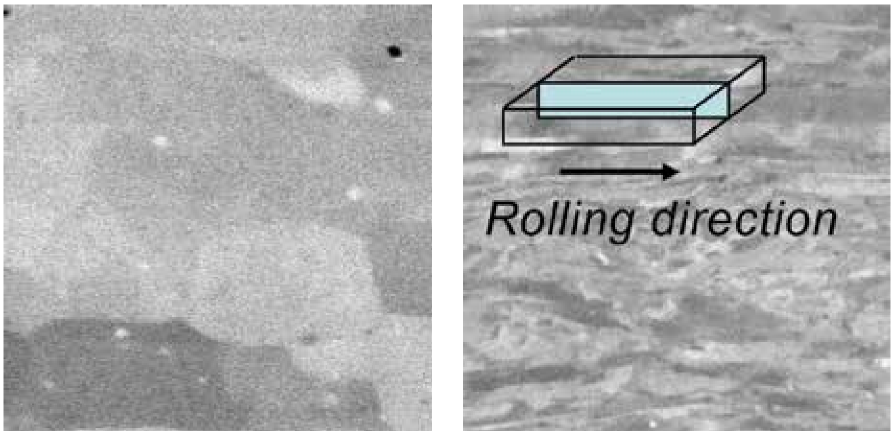


Figure 2. Microstructure of the 106ZY large rods, a) As extruded rod, b) After, four cold rolling steps, with a thickness reduction of 40% at each step. A very fine and elongated grain structure formed after cold rolling.

Accomplishments

A large CM08 ball milling machine was used at ORNL to produce ~four kilograms of 106ZY powder. Two master ODS rods of this material (250mm in length and ~22mm in diameter) were prepared by extrusion at 1100°C, with the extrusion can, extruded bars and final ODS 106ZY rods are shown in Figure 1. The two rods exhibited very similar properties, with hardness values ~300HV, and good tensile strength and ductility up to 600°C. Industrial pilgering was applied to the two rods to initiate tube production.

In addition, thin 0.4mm foils were fabricated by cold rolling from 3mm thick 106ZY plates to assess the alloy deformability and determine the impact of cold rolling on the alloy microstructure. It was determined that the material hardness values had to be less than 350HV to fabricate a tube by pilgering (cold rolling).

As can be seen in Figure 2, a very fine and elongated grain structure was observed for the 106ZY foils, ~100nm in size, to compare with the micrometer grain size for the as extruded plates. This nano-grain structure led to very high strengths, with yield strengths close to 1400 MPa, but very limited ductility. These results indicate that for tube fabrication, high temperature anneals will likely be required after each cold pilgering step to avoid detrimental hardening of the 106ZY alloy.

Finally, atom probe tomography (APT) analysis of alloy 106ZY, and a similar alloy without Zr addition (106Y) was conducted to determine the impact of Zr on the ODS FeCrAl alloy microstructure. The presence of a large number of (Y,Al,O) nanoclusters, ~2 to 10nm in diameter was observed for both alloys, and larger Zr carbonitrides were also detected for alloy 106ZY. The beneficial effect of Zr is related to the entrapment of impurities such as C and N, but Zr addition does not seem to affect the formation of (Y,Al)-rich nano oxides.

Large Fe-10Cr-6Al-0.3Zr ODS rods with excellent steam oxidation resistance, mechanical strength and ductility were fabricated at ORNL to be used for production of accident tolerant fuel cladding.

2.5 IRRADIATION TESTING AND PIE TECHNIQUES

ATF-1 Postirradiation Examination Highlights for 2017

Principal Investigator: Jason M. Harp
Collaborators: Luca Capriotti, Areva, Westinghouse

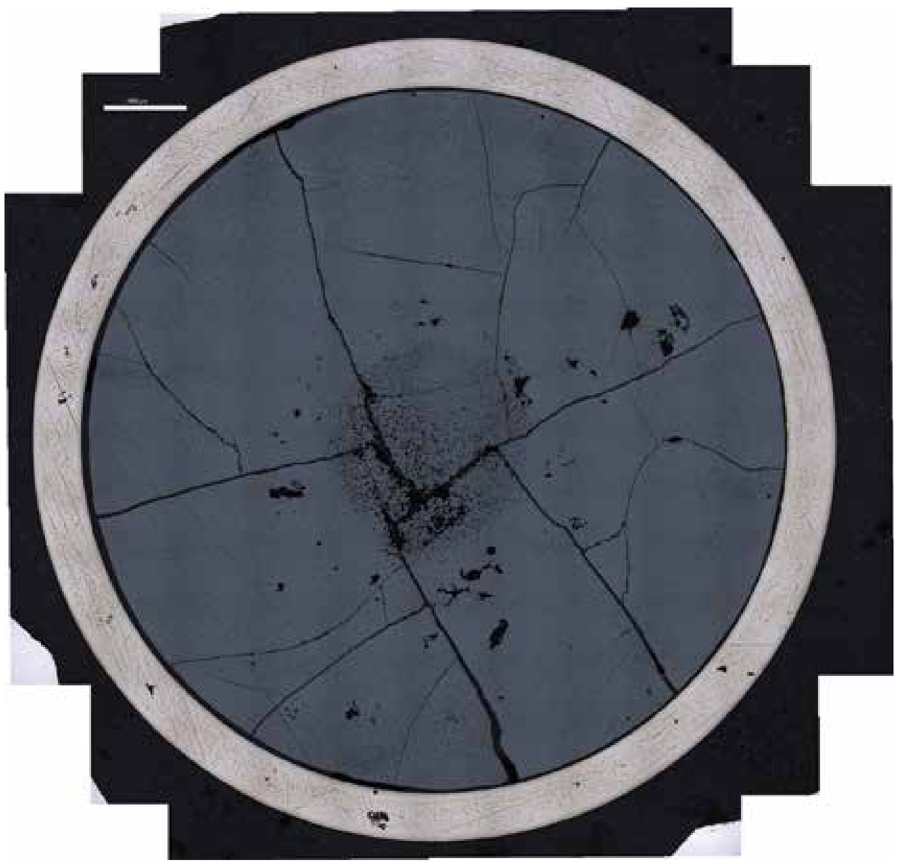


Figure 1. Optical Microscopy cross section from ATF-1A R1 (UO_2-Zircaloy-4)

Postirradiation examination of the Advanced Test Reactor (ATR) irradiated Accident Tolerant Fuel (ATF) concepts continues at the Idaho National Laboratory Hot Fuel Examination Facility (HFEF). This year the first destructive tests were performed on ATF-1 concepts, and the first examinations were performed on uranium silicide (U_3Si_2) irradiated at prototypic light water reactor conditions.

Project Description:

These irradiations are designed to provide data on several different potential ATF concepts. The ATF-1 irradiations are drop-in style tests that test a fuel cladding and fuel meat

The postirradiation examination of ATF-1 represents the first results from the collaboration of industry and DOE to research new fuels to enhance the accident tolerance of current commercial light water reactors.

combination to various different burnups. The small fuel rods are encapsulated to prevent release into the ATR primary coolant in the event of a cladding breach. To date, UO_2 plus additives clad in Zircaloy-4 (Areva concepts) and U_3Si_2 clad in Zirlo have been examined postirradiation. The linear heat generation rate is set by varying enrichment and axial location of rodlets in the irradiation positions. The temperature on the cladding surface is controlled by the gap between the cladding and the capsule. This configuration approximates light water reactor irradiation conditions inside the fuel pin so that fuel behavior and fuel-cladding interactions can be evaluated. The results of these screening irradiations support the development of new fuel forms to help light water reactors potentially operate with enhanced safety margins. PIE of these concepts consists of several different examinations that are used to determine different fuel performance properties such as fuel swelling, cracking, fission product distribution, burnup, radiation induced dimension changes, changes in microstructure, and fuel-cladding interactions.

Accomplishments:

Baseline PIE was completed on 3 ATF-1 rodlets from the Areva concept that tested UO_2 with several additives that are expected to enhance the thermal conductivity of UO_2. These samples are now available for additional PIE exams as desired by the vendor sponsor. Neutron radiography, gamma spectrometry that reveals fission product distribution, dimensional inspections, fission gas release data, burnup analysis, and optical microscopy has been produced for these samples. An example of optical microscopy performed on ATF-1A R1 (UO_2-Zircaloy) is shown below. As expected at this low burnup there is no contact between the fuel and the Zircaloy. The fission gas bubble morphology changes radially in the fuel pin as the local irradiation temperature increases towards the center of the fuel pin. This sort of microstructure is very valuable for

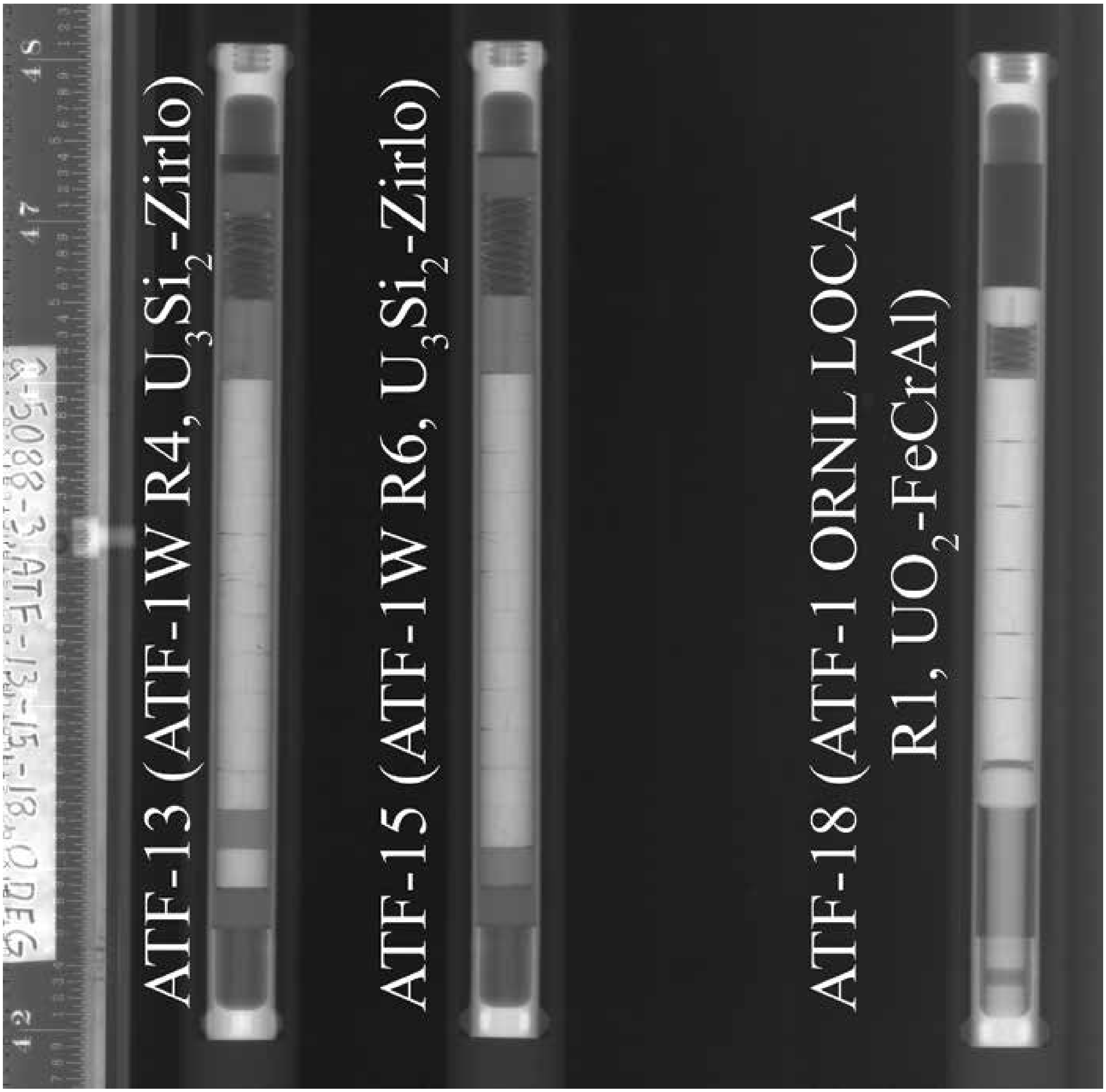


Figure 2. Capsule neutron radiography of 3 encapsulated rodlets from ATF-1

the modeling community, and these samples will likely be of great interest to modelers both in the Advanced Fuels Campaign and in NEAMS.

PIE was initiated on a UO_2-FeCrAl concept fabricated by ORNL. This sample is destined for the ORNL severe accident testing station. Some fission gas was detected in the capsule of this test indicating that there may be a small leak in the rodlet. No visual defects were observed on the rodlet. An eddy current system was applied to the rodelt and no defects were observed. This was the first use of this eddy current system on irradiated fuel, so the lack of findings may be due to a less than optimized system. This sample is now ready for transfer to ORNL when an approved shipping method can be determined.
PIE was also initiated on 2 ATF-1 rodlets from the Westinghouse concept (U_3Si_2-Zirlo). These rodlets were calculated to have reached a burnup of approximately 20 GWd/mtU which is about 1 typical cycle in a light water reactor. The neutron radiography from these capsules and the ORNL UO_2-FeCrAl concept are shown in the figure below. This radiography was performed while the rodlets were still in their stainless steel capusles. The difference in contrast between the Zirlo cladding (nearly neutron transparent) and the FeCrAl cladding (neutron absorbing) is quite obvious in this radiography. Cracking is observed in the U_3Si_2, but not more than would be expected from UO_2 at similar burnups. Also quite significant is the lack of axial swelling in the U_3Si_2. There is no observable change in the axial fuel stack height in the U_3Si_2 between the as-fabricated condition and the height observed in neutron radiography. This indicates the axial fuel swelling is $< 1\%$ based on the limit of the accuracy of measurements that can be ascertained from neutron radiography (~100 µm). Gamma spectrometry did not reveal any significant migration of fission products in the uranium silicide. At this time fission gas release results are still preliminary, but the release is $<<3\%$ and $<1\%$. Both the swelling and fission gas release behavior results are critical results for this fuel and will support the case for placing U_3Si_2 in light water reactors in the near future.

Characterization of transmutation fuels with energy resolved neutron imaging

Principal Investigator: S. C. Vogel/LANL
Collaborators: A.S. Losko/LANL, R. Fielding/INL

Transmutations fuels consist of a combination of high Z (U, Pu) atoms that are also strong absorbers for thermal neutrons (239Pu). Tomographic methods based on X-rays or thermal neutrons are therefore not suitable to characterize the bulk (mm-cm) of these materials. However, utilizing the ability to pick certain neutron energies in time-of-flight imaging at which the neutron cross-section of the isotopes contained in the samples is low, these sample can be penetrated and characterized by tomographic methods. Using areal density analysis of the isotopes in the beam path of the transmission signal from each pixel 2D areal densities and ultimately 3D gravimetric densities can be also derived from the same measurement. This was demonstrated at LANL on two U-20Pu-10Zr samples with and without Am/Np produced at INL.

Project Description:

Characterization methods applicable to the bulk, i.e. cm sized volumes, of nuclear fuels, are sparse. Neutron CT provides characterizations of dimensions and macrostructural features such as voids and cracks. However, for enriched fuels or fuels containing strong neutron absorbers, such as 239Pu, thermal neutrons do not work as even layers of <1mm fully attenuate the neutron beam. The energy-resolved neutron imaging (ERNI) developed at LANSCE circumvents this problem by allowing to conduct tomographic reconstruction with neutron radiographs collected at neutron energies for which the neutron cross-sections are small and several ten percent of the incident intensity is transmitted through even several mm of sample material. Enabling technologies for this technique are the LANSCE pulsed neutron source, emitting neutron pulses at 20 Hz, and a pixilated neutron time-of-flight imaging detector. Same as thermal neutron CT, this method is non-destructive and allows characterization of encapsulated materials, even through cm of Pb to shield gamma radiation to characterize irradiated fuels in the future. Assessment of the entire fuel volume pre-irradiation and ultimately post-irradiation will enable identification of regions of interest for destructive post irradiation examination that will

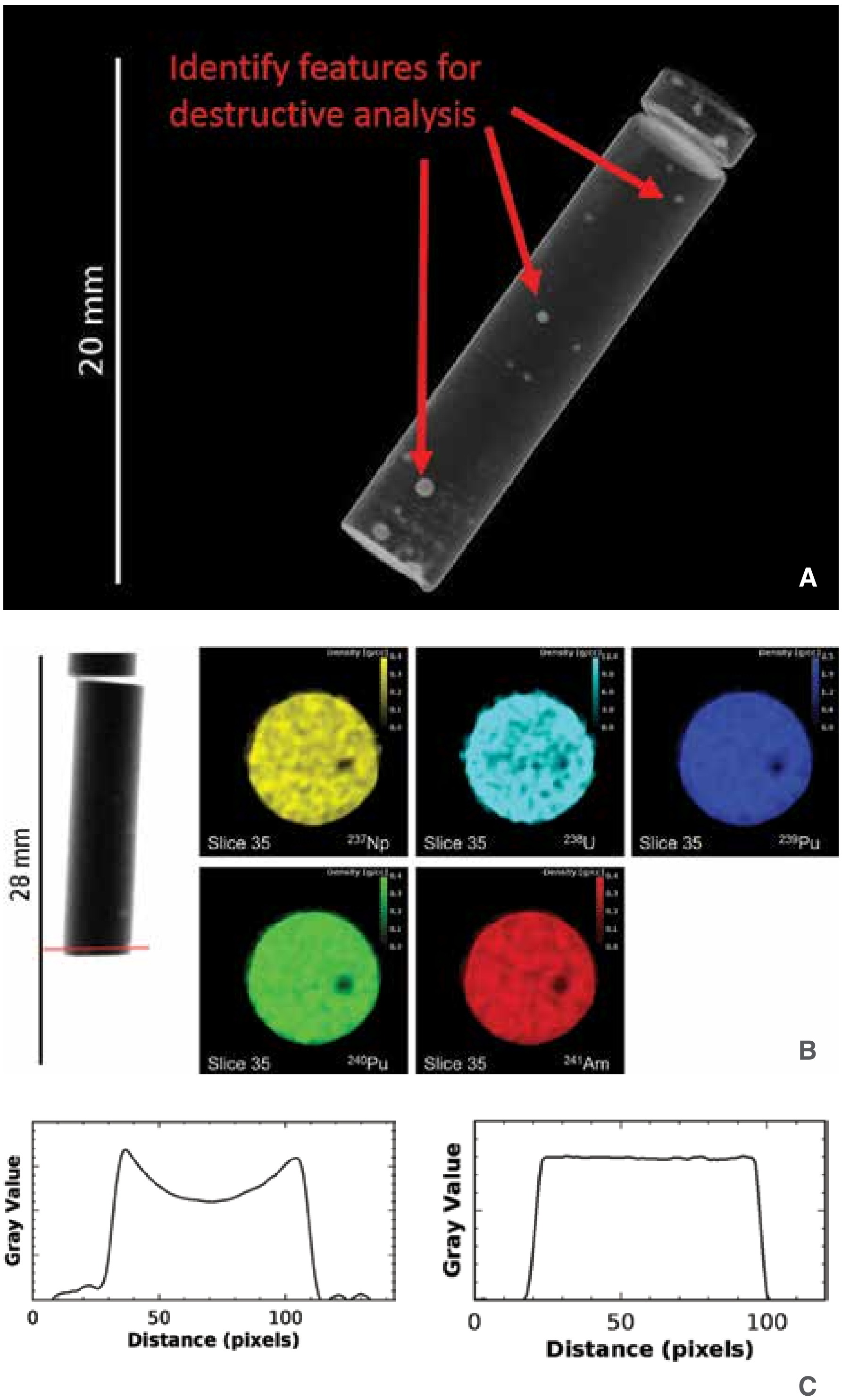


Figure 1: (a) Tomographic reconstruction of a U-20Pu-10Zr transmutation fuel slug, identifying several inclusions. (b) 2D isotope density maps for a slice including one of the features, indicating a decrease in density at the location of the inclusion for all isotopes characterized. (c) Scan through the reconstructed density in the CT datasets obtained from thermal neutrons (top) and epithermal neutrons (bottom), showing density variations outside and inside the fuel in the density derived from the thermal neutron data, which are artifacts of the CT reconstruction due to the partially complete attenuation of the thermal neutron beam.

Energy-resolved neutron imaging was applied to U-20Pu-10Zr fuel slugs to demonstrate the characterization of fuel forms opaque to X-rays and thermal neutrons as well as 3D measurements of isotope densities with this technique and producing for the first time XRF-like isotope maps obtained non-destructively for the bulk of the sample.

allow characterization of features hitherto hidden from conventional characterization methods.

Accomplishments:
One of the technical goals of this research was to demonstrate energy-resolved neutron imaging and neutron diffraction characterization, successfully demonstrated to oxide, nitride, silicide fuels in the past, also to transmutation fuels. For this research, four slugs of metallic fuels with a nominal composition of U-20Pu-10Zr (wt. %) were produced at INL (R. Fielding) and loaded in two rodlet/capsule assemblies. Material in one capsule also contained Am (~2 wt. %) and Np (~3 wt. %). Tomographic imaging and diffraction measurements were performed to characterize these samples at the Flight-Path 5 and HIPPO beam lines at LANSCE/LANL. Due to the presence of Pu, the penetration of the material by thermal neutrons is limited to ~1.3 mm (e-1 penetration depth) and since diffracted neutrons are also attenuated, the volume probed by thermal neutrons is only a layer of ~650 micrometer. However, access to transmission data in the epi-thermal neutron range, provided by the pulsed neutron spallation source at LANSCE and captured with a 512x512 pixel time-of-flight imaging detector, allows characterization of sample dimensions, cracks, chemical and isotopic inhomogeneities for these samples. Several inclusion and shape irregularities were identified with this unique tool. The diffraction data complements these results, but cannot provide bulk characterization as the penetration of thermal neutrons, suitable for diffraction, is limited to the outer layers of the material. The bulk composition described by partial densities of the main isotopes 237Np, 238U, 239Pu, 240Pu, and 242Am agrees to better than 0.1 g/cc with the partial densities determined by mass spectrometry except for 238U, which was overestimated. These finding underlines the substantial benefits of pulsed neutrons over a continuous neutron source,

such as a reactor, or hard X-rays for characterization of specimen containing fissionable material with large thermal neutron attenuation. The goal of the Advanced Non-destructive Fuel Examination work package is the development and application of non-destructive neutron imaging and scattering techniques to ceramic and metallic nuclear fuels. Data reported in a detailed report were collected in the LANSCE run cycle that started in September 2016 and ended in February 2017. Data analysis for the tomography data is completed and this report provides an overview of the characterized samples (Figure 1). The identified features in the bulk of the double encapsulated material on these demonstration fuels illustrates the value of this characterization for pre-irradiation examination, e.g. to reject samples with flaws, as well as to post-irradiation examination, e.g. to guide destructive characterization by preparing regions of interest identified by the techniques presented here.

Key results include:

- 3D characterization of cracks, flaws and isotopic distributions are determined non-destructively in complete slugs to spatial resolutions of at least 100 microns. The tomographic reconstruction of the data from the epi-thermal neutron energy range allowed to identify several spherical features within all slugs. One slug showed a deviation from the cylindrical geometry.
- Using detailed analysis of the transmission data, using the SAMMY code and known neutron absorption cross sections for interpretation of neutron transmission data, isotope specific density maps through arbitrary slices of the reconstructed volume can be displayed. The same analysis provides partial densities for isotopes averaged over the entire sample, which are except for 238U in excellent agreement with results from other methods.
- The density maps of the major isotopic constituents of the slugs, computed for slices containing the spherical features, allowed to identify voids as the most likely nature of the features
- 3D characterization of isotopic distributions are possible in four slugs in three to five days with the neutron flux available at LANSCE
- Diffraction data of crystallographic phase, texture and spatially resolved measurements were collected but are of limited value as the penetration depth is low and only ~0.65 mm of material is interrogated.

The potential of laser-driven neutron sources for pool-side characterization

Principal Investigator: S. C. Vogel/LANL
Collaborators: A.S. Losko/LANL, M. Roth/TU Darmstadt, Germany

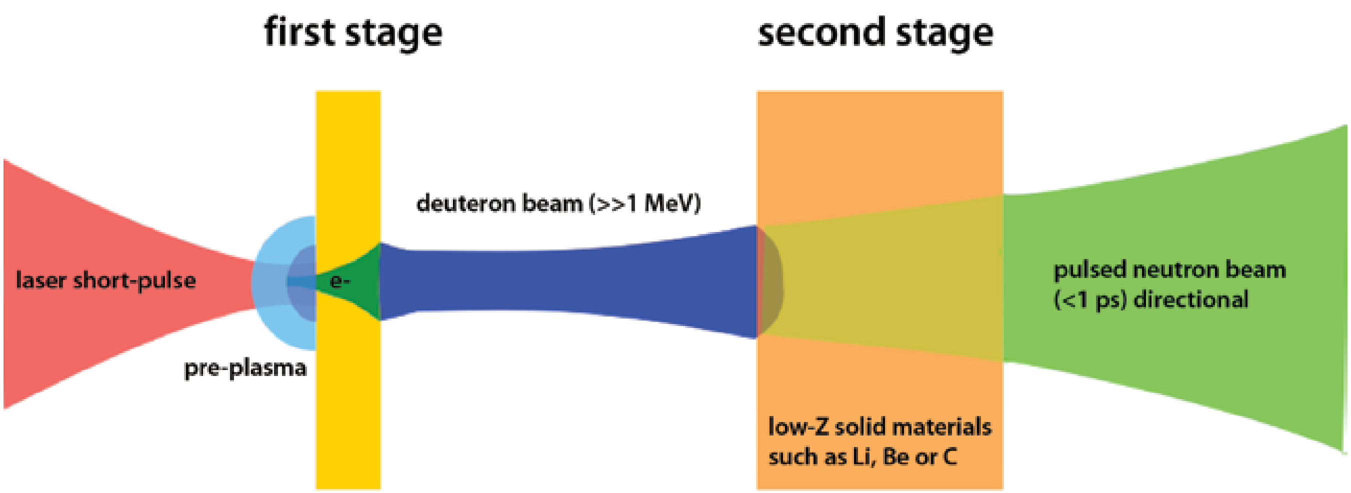


Figure 1. Schematic of the principle of laser-driven neutron production – an energetic (>1 J), high contrast (>10E-8 between peak and pre-pulse), short (few tens of fs) laser pulse hits a target of appropriate thickness (<1 μm), the pitcher, and accelerates the electrons and nuclei (protons, deuterons) in the target material towards the second target, the catcher. The nuclei are converted to neutrons in the second target.

Unique characterization capabilities were developed for NE-mission relevant characterization and applied to fresh nuclear fuels at LANSCE. Length scales covering atomic distances (unit cell parameters, chemical & residual strains), microstructure (phase composition, dislocation densities, texture), and macro-scale (gaps, voids, inclusions, mapping of isotope densities) are covered. A major disadvantage of the techniques is the requirement of an intense pulsed neutron source as a driver to avoid costly and time consuming shipment to existing neutron facilities such as LANSCE.

Laser-driven neutron sources have the potential to provide these capabilities e.g. at ATR/INL. Recent developments in high-power (PW), high repetition rate (1-10 Hz) laser system bring this approach, demonstrated on single shot laser systems, within reach of deployment. The reduction of cost for LANSCE-like capabilities by two orders of magnitude makes this approach very attractive.

Project Description:

The technical objective to of the research was to assess the current state-of-the-art of utilizing lasers to produce intense neutron and hard X-ray pulses that are suitable for characterization applications. Intense neutron sources, such as LANSCE at LANL, provide excellent characterization capabilities which can significantly accelerate development and licensing of new nuclear fuel forms, such as accident tolerant fuels. Being able to characterize several pellets on length scales ranging from the atomic distances to macroscopic cracks and voids allows to characterize fuels pre- and post irradiation. The latter will guide destructive examination. However, at present the fuel, especially irradiated fuel, would have to be transported to LANSCE for characterization, which is both cost intensive as well as time consuming. Laser-driven neutron sources have the potential to provide such capability pool-side, both during development of fuels, e.g. by irradiation tests at ATR, as well as at power plants to provide enhanced characterization of fuel failures or as a quality control tool during fabrication. Besides characterization of LWR fuel pellets, pulsed neutrons have the potential to become in-process diagnostics, e.g. for the enrichment level of the fuel feed, in next generation reactor types such as molten salt reactors. Assessing the current state-of-the-art of laser-driven neutron sources provides the NE mission to participate in technology maturation and a pathway to possible deployment within the next decade.

Accomplishments:

The LANL PI (S. Vogel) of this effort brought together world-leading researchers in the field of laser driven neutron sources, covering fields such as laser-plasma interaction (M. Roth, Technical University of Darmstadt, Germany), Juan Fernandez (LANL), high power lasers(W. Leemans, LBL, C. Haefner & C. Siders/LLNL), laser target design (M. Roth, S. Glenzer/SLAC), and neutronics of target/moderator systems (M. Mocko/LANL) with experts in neutron-based fuel characterization (Vogel & Bourke/LANL) to compile a comprehensive report on the potential of laser-driven neutron sources for pool-side

Laser-driven neutron sources have the potential to bring LANSCE-like pulsed neutron characterization capabilities to irradiation sites such as ATR at a fraction of the investment and operation cost and providing an excellent tool to characterize irradiated fuels.

characterization. The report covers for the first time the basics of high power lasers, particle acceleration, neutron production from the accelerated particles and the demonstrations of the technique thus far.

In a laser-driven neutron source the laser is used to accelerate deuterium ions into a beryllium target where neutrons are produced (Figure 1). At this time, the technology is new and their total neutron production is approximately four orders of magnitude less than a facility like LANSCE. However, recent measurements on a sub-optimized system demonstrated >10E10 neutrons in sub-nanosecond pulses in predominantly forward direction. The compactness of the target system (1cm diameter, 1 cm length) compared to a spallation target (10 cm diameter, 20 cm length) may allow exchanging the target during a measurement to e.g. characterize a highly radioactive sample with thermal, epithermal, and fast neutrons as well as hard X-rays, thus avoiding sample handling. Furthermore, a spallation target emits neutrons isotropically, while the neutrons produced by deuteron break-up reactions using a laser a predominantly traveling forward. This insight allows to embed the laser neutron target into the moderator, allowing >90% of the neutrons to interact with the moderator, while at best only a few percent of the spallation neutrons have the potential to interact with the moderator. For the flux on sample, this insight brings down the difference of a laser-neutron source and LANSCE by at least two orders of magnitude. Since spallation also produces neutrons with >100 MeV neutron energies, while the neutrons produced by deuteron breakup are limited to less than 50 MeV (depending on deuteron energy), the shielding requirements for a laser driven neutron source are much less than for a spallation source. For the application, this translates to several meters closer sample positions (e.g. 1m vs. 10m), allowing to re-gain another two orders of magnitude for flux on sample. This brings LANSCE-like fluxes on sample within reach with today's laser technology.

At this time several groups are working on laser-driven neutron production and are advancing concepts for lasers, laser targets, and optimized neutron target/moderator systems. Advances in performance sufficient to enable poolside fuels characterization with LANSCE-like fluence on sample within a decade may be possible. The report describes the underlying physics and state-of-the-art of the laser-driven neutron production process from the perspective of the DOE/NE mission. It also discusses the development and understanding that will be necessary to provide customized capability for characterization of irradiated fuels.

Development of the Multi-SERTTA Irradiation Vehicle for ATF Testing in TREAT

Principal Investigator: N. Woolstenhulme

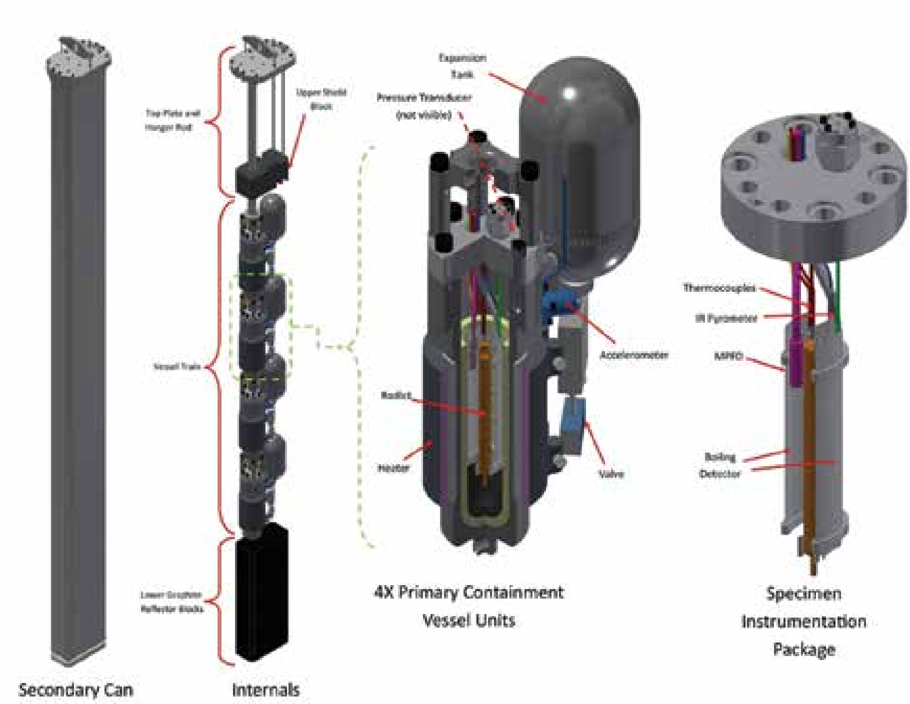


Figure 1. Multi-SERTTA Overview.

The Transient Reactor Test facility (TREAT) is graphite-based transient test reactor able to safely provide flexible power maneuvers simulating nuclear-heating for fuels and materials safety research. While TREAT's design is both brilliant and basic, its primary role is only to provide neutrons to test specimens. Various irradiation vehicles (e.g. capsules, loops) are placed in the core to provide desired specimen boundary conditions, safely contain hazards, and support test instrumentation. Historic TREAT irradiation vehicle designs focused on sodium-cooled systems. A sizeable effort is needed to design new water-environment capabilities for testing Accident Tolerant Fuels (ATF) and other water-reactor technologies.

Project Description:

Three water-bearing irradiation vehicles are planned. The first of which, and the current project focus, is a multiple capsule design Static Environment Rodlet Transient

Test Apparatus (Multi-SERTTA). Multi-SERTTA is capable of static liquid water conditions representing pressurized water reactors (280°C, 15.5 MPa), enables high throughput testing, and offers a respectable instrumentation package suitable for Reactivity Initiated Accident (RIA) type testing of ATF fresh fuel samples. A larger single-vessel version (Super-SERRTA) will offer enhanced natural convection, increased instrumentation options, and structural capability for RIA and Loss of Coolant Accident (LOCA) testing as needed for pre-irradiated fuel specimens. Finally, full PWR forced convection capabilities, in addition to all of Super-SERTTA's capabilities, will be available in the TREAT Water Environment Recirculating Loop (TWERL). These water-bearing irradiation vehicles will enable a full suite of safety testing capabilities on water-reactor fuels at any point in their development life cycle (Figure 1). The Multi-SERTTA design has been the primary focus of recent design and engineering efforts in order to enable timely testing of ATF specimens under the ATF-3-1 campaign. Owing to Multi-SERTTA's novel layout, a full size out-of-reactor prototype was needed to qualify its design and ensure its success. A modified Multi-SERTTA design was created to enable nuclear calibration measurements (referred to as -CAL). Since TREAT had no historic examples of core-to-specimen power coupling factors (PCF) measured in the presence of PWR-condition water, this first-of-a-kind -CAL design had to both provide equivalent reactivity worth for trial core operations, and enable PCF measurements on fissionable dosimeters surrounded by hot pressurized water. The principle design modification performed for -CAL involved placing a small inner tube through each Multi-SERTTA vessel so that dosimeter could be surrounded by the water, but not immersed in it, thus enabling timely extraction, dosimeter radiation measurement, and radiochemical analysis in order to determine PCFs (Figure 2).

The Multi-SERTTA irradiation vehicle will enable high throughput nuclear-heated transient safety research on ATF specimens, surrounded by hot pressurized water, in the TREAT facility.

Accomplishments:

In order to support the Multi-SERTTA prototype fabrication and testing effort, a detailed prototype testing plan was developed and complete to accomplish a level 3 milestone and prototype fabrication work commenced. Simultaneously, the -CAL design underwent detailed engineering, performance predictions, and safety analyses in preparation for a design review. This design review was performed in mid FY17

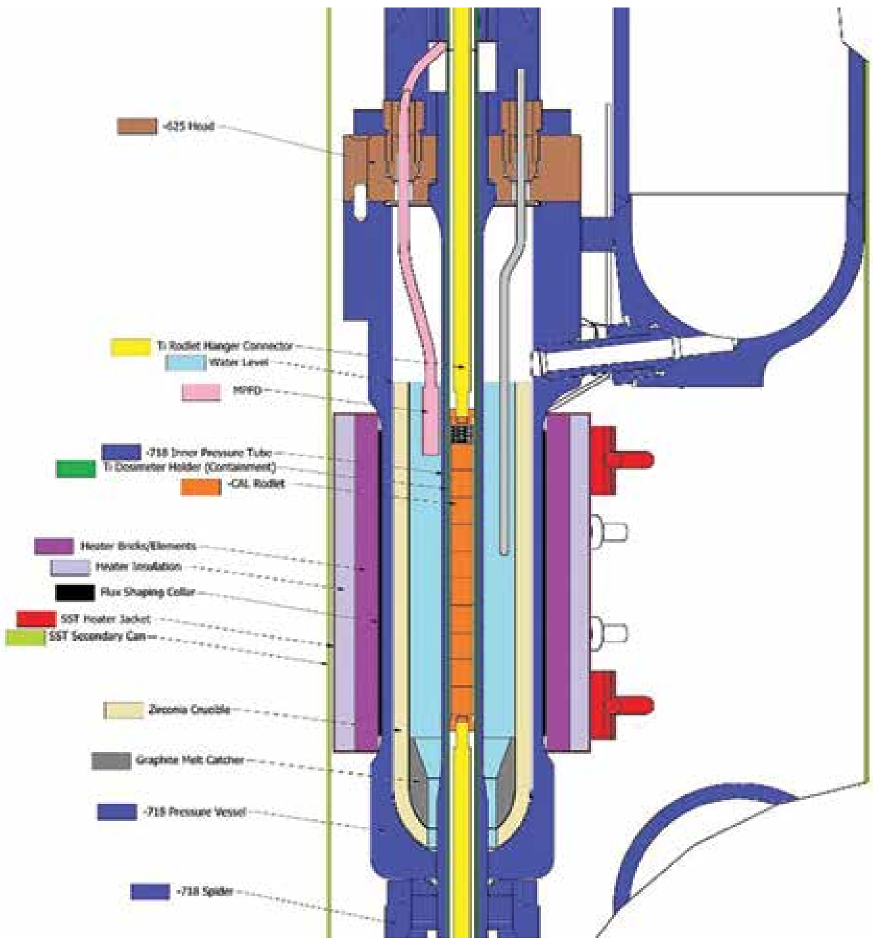


Figure 2: Multi-SERTTA-CAL Single Vessel Cross Section.

in accomplishment of a level 2 milestone. Several notable accomplishments occurred within both of these main tasks.

The process for brazing instrumentation leads into an Inconel braze plug was demonstrated and qualified in accordance with INL welding program in order to enable assembly of the prototype vessels. An equivalent-weight Multi-SERTTA handling mockup, which consisted of a representative sheet metal enclosure with a steel ballast bar within, was fabricated, delivered to TREAT, and test fit successfully within storage hole interfaces. All of the large metallic containment prototype containment components were successfully fabricated and dimensionally inspected (Figure 3). Out of tolerance conditions discovered were useful in determining where future fabrication development is needed, and in determining which design tolerances need to be relaxed. A specialized hydrostatic pressure testing rig was designed and constructed to enable qualification of Multi-SERTTA prototype vessels. The outcomes from several Multi-SERTTA-related efforts were provided to projects funded by other sources including development of in-pile instrumentation and design of a low-activity experiment handling bench. All of these activities have helped transition the Multi-SERTTA and related projects from the design phase into the deployment phase. This transition is timely considering the imminent restart of TREAT and data needs for ATF developments.

This work was primarily performed by INL's experiment design and analysis departments including the following contributors: C. Baker, J. Bess, C. Biebel, L. Burke, D. Chapman, C. Davis, D. Dempsey, C. Hill, C. Jensen, N. Jerred, L. Nelson, R. O'Brien, J. Palmer, J. Parry, E. Rosvall, J. Schulthess, S. Snow, D. Wachs, J. Wiest, N. Woolstenhulme.

Figure 3: Highlights from Prototype Fabrication.

In-Pile Fuel-Coolant Interaction (FCI) Studies Supporting Transient Testing Experiments

Principal Investigators: Colby Jensen
Collaborators: Nicolas Woolstenhulme (INL), Daniel Wachs (INL), Michael Corradini (UW-Madison)

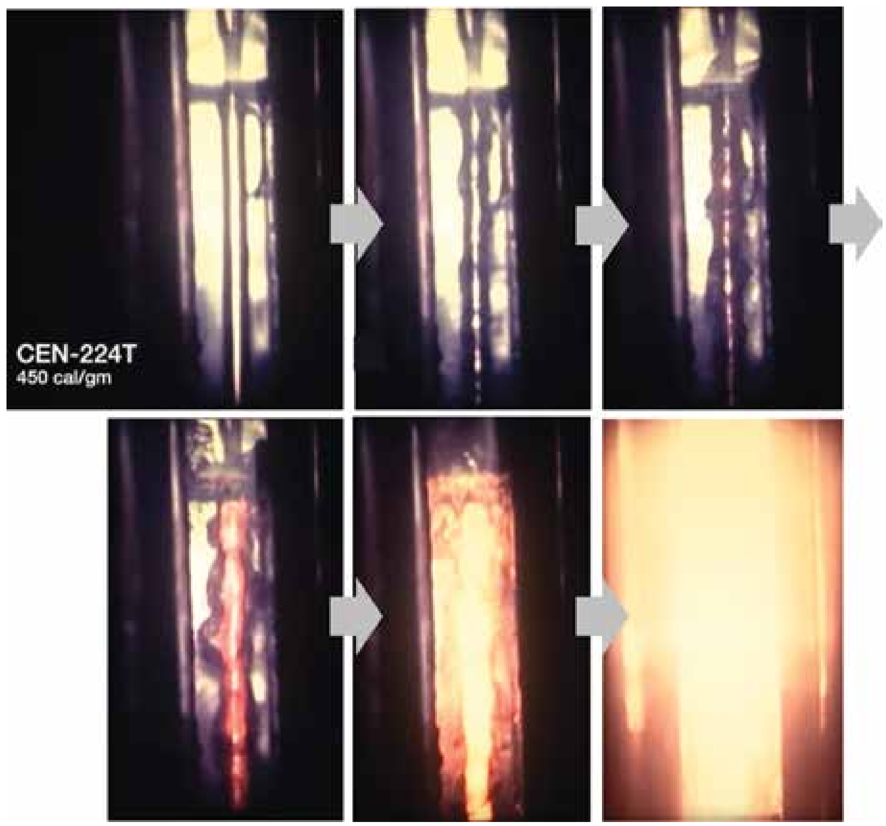


Figure 1. Still-frame progression from high-speed video of high energy RIA experiments performed in TREAT in the mid-1960s (https://youtu.be/h0o4P_F4s9s).

Fuel-coolant interaction (FCI) has long been a subject of interest in severe-accident reactor safety applications. Early transient testing in Idaho focused heavily on providing insight to the phenomenological behavior and consequences of such events. Since then, several out-of-pile experiments have studied these interactions, though effective analytical treatment has never been accomplished thus requiring significant margins from the thresholds of such events. In high energy events in water, significant cladding-coolant chemical interaction and resulting thermal energy may play a role in further propagation of the energetic event. High energy FCI events (molten

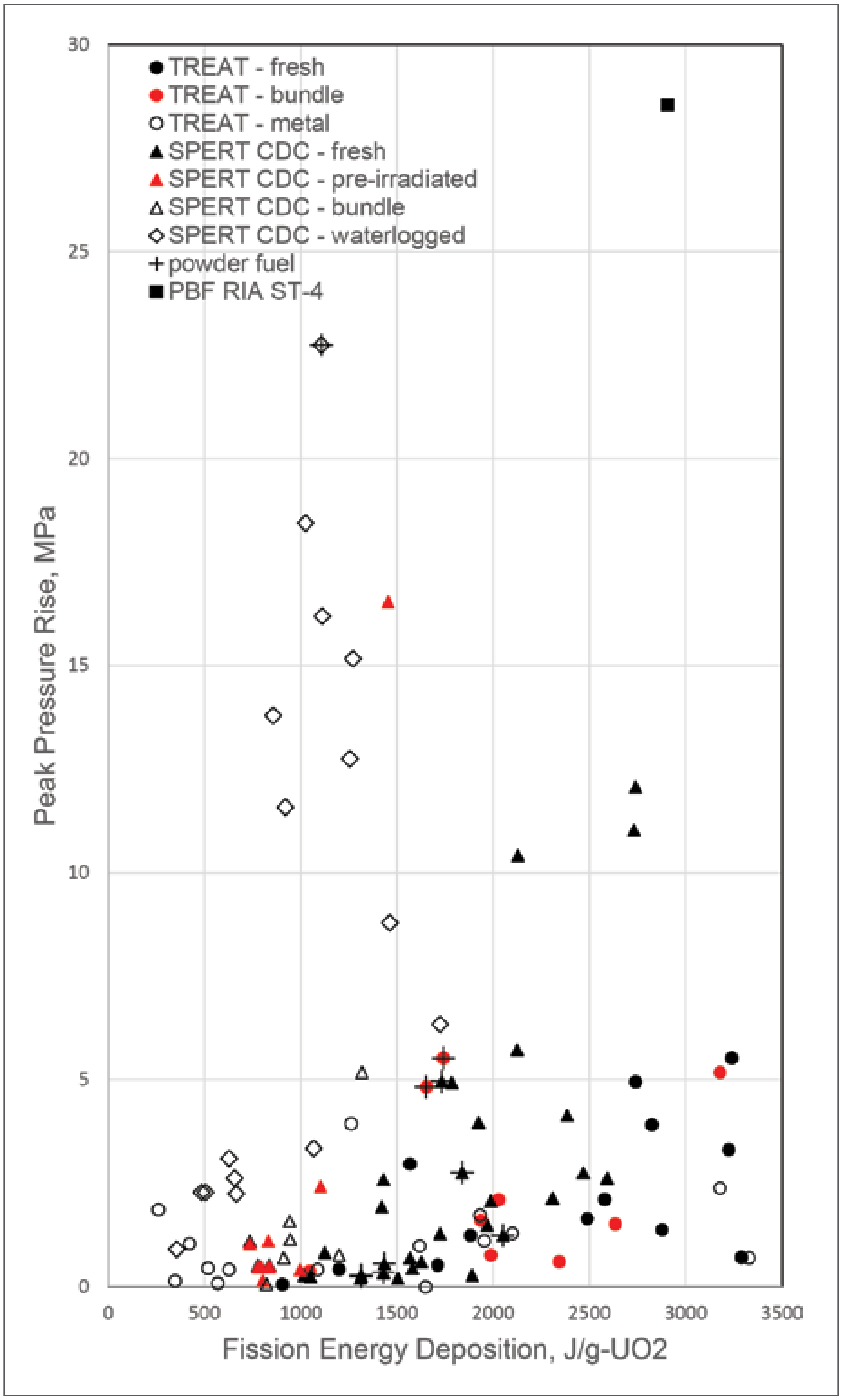
The potential for energetic interaction between fuel ejected at high temperature and the surrounding coolant is a critical fuel safety issue and that has evaded effective analytical treatment for 50 years.

fuel coolant interaction) is not a credible scenario for any deployed nuclear reactor technology, though dispersal of high temperature fuel particulate is of interest for accident progression involving high burnup fuels. In transient testing such as will be performed at the Transient Reactor Test (TREAT) facility, FCI events represent the limiting condition for experiment-induced loading on experiment containments. This research intends to renew expertise and develop analytic capabilities to evaluate FCI and possible thermal-mechanical energy conversion.

Project Description:

Fuel-coolant interactions (FCI) play a critical role in the safety evaluations of nuclear systems and have been the subject of considerable research, including beyond the nuclear industry. FCI events occur when cladding fails allowing direct interaction of fuel and coolant. With an extremely large temperature gradient driver, FCI may result in highly energetic behavior with rapid pressurization and thermal-to-mechanical energy conversion. This research is focused on developing technical expertise and analytic tools to evaluate FCI behaviors. For the purposes of this discussion, FCI events in nuclear systems are classified as the result of two different power-to-cooling mismatch conditions: (1) rapid, high energy overpower events (reactivity initiated accident) with possible cladding failure resulting in fuel dispersed directly into a range of coolant conditions; and (2) loss of sufficient cooling (loss of coolant accident) leading to cladding failures and possible fuel dispersal. In the most severe propagation of events, fuel melting and molten fuel coolant interaction (MFCI) may occur. The primary focus of this research is developing a capability required to design and perform transient

Figure. 2. Compilation of peak pressure rise measured during in-pile RIA experiments at SPERT, TREAT, and PBF. The results distinguish between various experimental parameters.



experiments, which oftentimes includes considering consequences of fuel failure well-beyond limits imposed on commercially deployed technologies. While establishing needed experimental capability, this research will also provide opportunity to begin further investigation of related FCI behaviors for: 1) potentially reducing uncertainties and impacting limits used to guide fuel usage including for high burnup fuels, which have a high propensity for fragmentation and dispersal into coolant, conducive of energetic FCI behaviors; and 2) advanced fuel concepts that have little to no in-pile experimental FCI exploration.

Accomplishments:

The primary goal of this work is to develop capabilities to analyze FCI events with an initial focus on reactivity initiated accident (RIA) related phenomenology. Initial work at Idaho National Laboratory (INL) has focused on compiling information from historical in-pile experiments. Several historical facilities located on the present-day INL site have contributed to the experimental understanding of FCI physics and resulting nuclear fuel system behavior. Testing in these facilities provided much of the foundational knowledge of fuel system response, data for model validation (some still unique in the world), and set a precedent for the experimental approaches used to study FCI phenomena. In particular, reactor tests performed at BORAX and Special Excursion Reactor Test (SPERT) reactor facilities provided much early insight into large-scale system behaviors. In the 1960's, capsule type experiments were started in the TREAT and SPERT Capsule Driver Core (CDC) experiments with a focus on in-water behavior up to very high energy depositions. By the mid-1970's, experiments began in the Power Burst Facility (PBF) utilizing a prototypic water loop capable of operation to Pressurized Water Reactor (PWR) conditions.

In the TREAT reactor, one experiment device provided access for high speed videography of RIA experiments in water. Many of these resulted in FCI events with an example given in the video frames shown in Figure 1. Figure 2 shows the compiled peak pressure rise measured in experiment coolant as a function of energy deposition in the fuel. The results demonstrate increased severity of the resulting pressure spikes with waterlogged fuel rods and some indication of increased pressure pulse severity for powder form fuels and pre-irradiated fuels. Overall, the data obtained for measured peak pressure also illustrates a stochastic nature for FCI events. With a few exceptions, it is notable that the majority of the experiments performed to simulate high energy

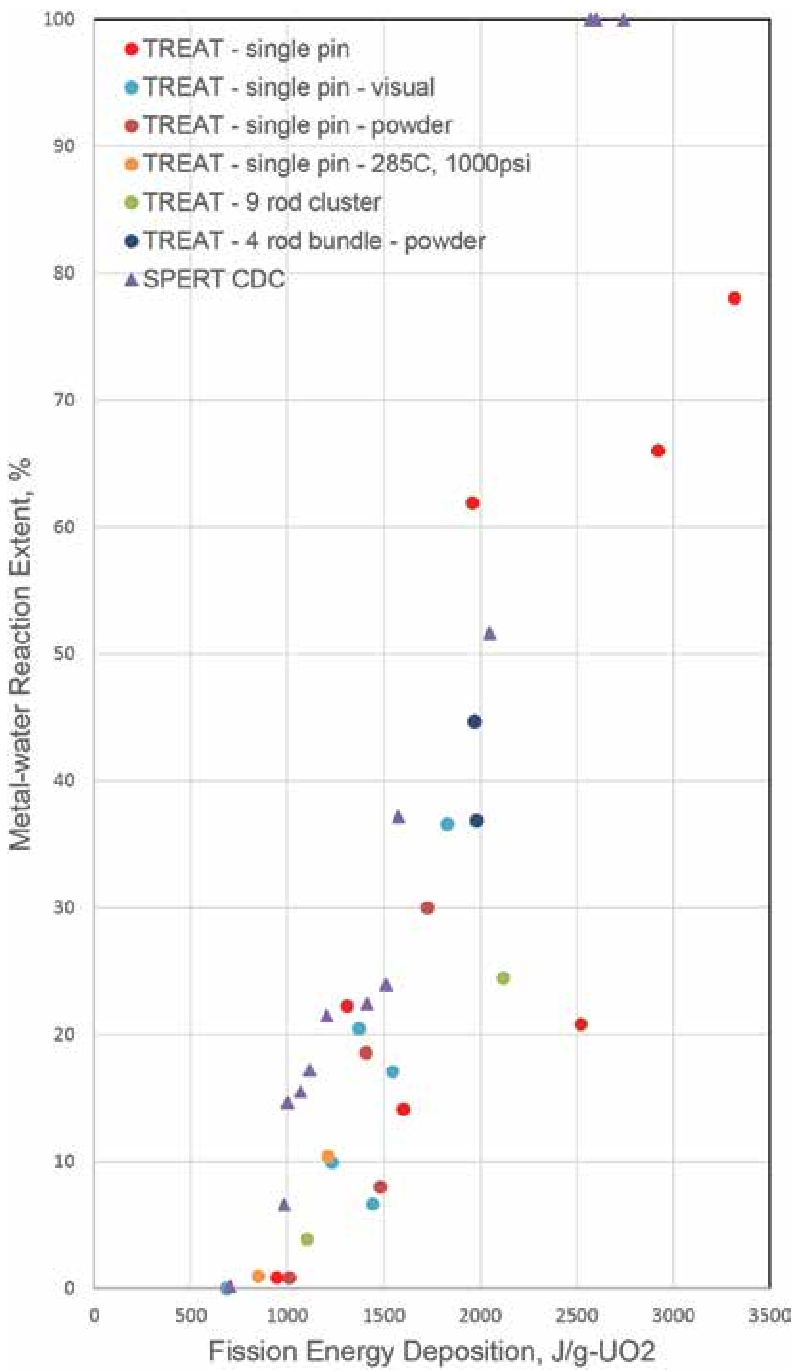


Figure. 3. Metal-water reaction extent results for experiments performed in SPERT and TREAT with UO_2 fuels in Zircaloy-2 cladding. Metal-water reaction extent is defined as the total percent-cladding consumed by chemical reaction in the experiment. The SPERT reports specify the total cladding as that next to the active fuel zone.

RIA events have been performed at low temperatures and pressures. The extent of metal-water reaction for several tests is shown in Figure 3 for tests performed in TREAT and SPERT for several test conditions. The SPERT data defines the extent of metal-water reaction as the percent of the mass of the cladding encompassing the active fuel length. The general trends of metal-water reaction vs. energy deposition provide useful information as to the contribution of chemical energy to the system during these transient events.

In addition to compiling existing experimental data, a new collaborative project with the University of Wisconsin-Madison (UW) has been initiated to predict FCI pressure pulse generation using an FCI predictive tool developed at UW several decades ago. These efforts are focused on using the historical experimental results as benchmark data for predictive capabilities while compiling historical data into a usable form. Work is planned to expand this benchmarking effort and related expertise while beginning to apply these analysis techniques to modern TREAT experiment designs. This effort has already gained the attention of international transient testing researchers to provide a gateway to broader developing collaboration in this area.

ADVANCED REACTOR FUELS SYSTEMS

3.1 AR Fuels Development

3.2 AR Computational Analysis

3.3 AR CORE Materials

3.4 AR Irradiation Testing & PIE Techniques

3.5 Capability Development

3.1 AR FUELS DEVELOPMENT

Metallic Americium Feedstock Production

Principal Investigator: Leah Squires
Collaborators: James King, Scott Wilde, Cory Brower, James Newman, Ryan Johnson, Dru Charles, Steven Monk, Mark Lounsbury, Blair Grover

A method of distilling americium metal from a mixture of americium and neptunium metals has been developed. Multiple parameters were investigated to optimize the process. The goal for this year was to distill approximately 10g of pure americium in support of a collaboration with the French. Over 15g of pure americium metal was produced.

Project Description:

The disposal of spent nuclear fuel is one of the main challenges facing the nuclear power industry due to the long term health and safety hazards of storing the material once it is removed from the reactor. A key factor in reducing the hazards associated with long term storage is the reduction or elimination of minor actinides in the fuel such as neptunium, americium and curium, which have relatively long half-lives. Transmutation is one method currently under investigation to resolve this issue. Transmutation aims to incorporate the long lived actinides into new fuel which can be placed in a fast reactor where the elements of concern will fission into products which have shorter half-lives. In order to develop the process of transmutation it is necessary to first fabricate small quantities of fuel with minor actinide additives on which to perform thorough characterization and irradiation testing. Since americium is a key component in the fabrication of transmutation fuels and, currently, there is very little of this material in existence as pure metal, it is necessary to develop methods to isolate it from the feedstock material that is available. The majority of the americium bearing feedstock available is in oxide or mixed metal form. Due to the numerous types of americium bearing feedstock available, many of which are in small quantities, methods for isolating the material must be continuously developed.

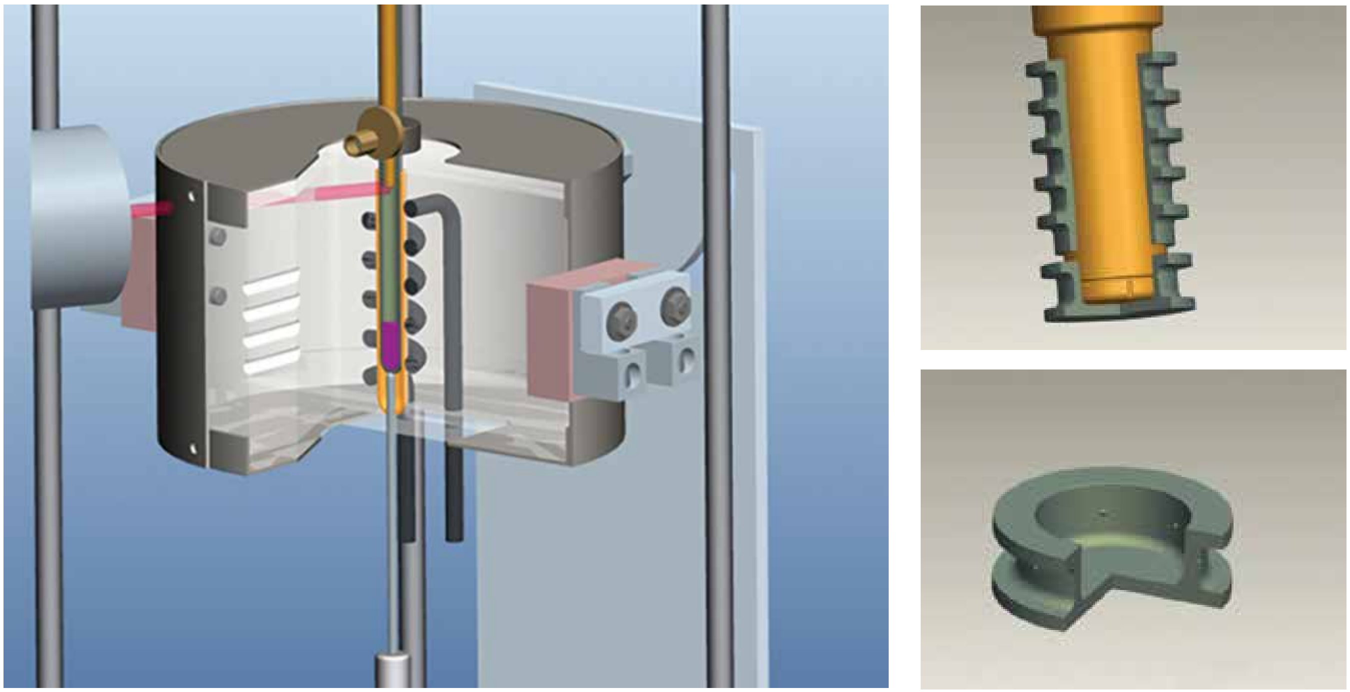


Figure 1. (Left) The inside of the furnace containing a tantalum crucible (grey) loaded with starting material (purple) and the copper cold finger. (Top Right) The tantalum endcap on the copper cold finger. (Bottom Right) The tantalum endcap.

The feedstock work package has purified all of the americium metal necessary to support AmBB and also made enough to begin fabrication of AFC fuels.

Accomplishments:

Work in cooperation with France required the production of 10g of pure americium metal in support of the americium bearing blanket (AmBB) initiative during FY2017. It is the goal of the French project to add americium to fuel which is to be placed in the blanket of a fast reactor in order to transmutate the long lived actinides such as americium. The US obligation to this agreement includes the production of americium and fabrication of the fuel. In order to support the fabrication needs approximately 10g of pure americium metal were needed by the end of FY17. This goal was met and an additional 5g of material were purified. There were a number of process improvements made in FY17 that made it possible to obtain the necessary amount of material. It was discovered that the material below the surface of the distillation mixture was not being allowed to vaporize and therefore some sort of agitation or stirring mechanism was necessary in order to break this surface tension and improve process yields. This was accomplished by agitating the furnace (Figure 1) backward and forward every fifteen minutes while it was being held at temperature. This significantly improved the yield by allowing material trapped below the surface to come to the surface and vaporize and ultimately condense higher up on the walls of the crucible. In addition to this improvement, the method for retrieving the material was also improved in FY17. Although a cold finger is placed inside the crucible to collect material it has been determined from numerous previous experiments that the material plates out on the walls of the crucible below the cold finger rather than on the cold finger itself. Thus, the material must be retrieved after cooling using a drill bit. In order to improve this process as well as to limit the dose received by the operators, who in the past have hand drilled the material out, a hobby lathe (Figure 2) was purchased and retrofitted for use in the glovebox. This has allowed for the much more efficient material retrieval (Figure 3) and has also decreased the finger dose to the operators.

Also during FY17, it was discovered that the purified americium metal is highly susceptible to oxidation even in an inert atmosphere glovebox with tightly controlled oxygen levels. In order to ensure the metal does not oxidize the shavings removed from the crucible walls using the lathe are now reconsolidated into a larger volume ball to reduce the surface area exposed to oxygen.

Figure 2. The modified hobby lathe with the addition of a catch pan for collection of retrieved metal shavings and a stand to allow it to stand upright when not in use and conserve glovebox floor space.

Figure 3. (Left) Americium metal as it is retrieved by the lathe. (Center) Americium metal that was sealed in a Swagelok container and kept inside the inert atmosphere glovebox for approximately one month. (Right) Consolidated americium metal ball

Neptunium Feedstock Reduction Experiments

Principal Investigator: Leah Squires
Collaborators: Scott Wilde, James King, Randy Scott, Dru Charles, Mark Lounsbury, Cory Brower, Ryan Johnson, James Newman, Steven Monk, Ryan Pitt, Blair Grover

Figure 1. The neptunium oxide reduction setup in the Hot Uniaxial Press Furnace insert.

A method of reducing neptunium oxide to neptunium metal using calcium metal and calcium chloride salt was previously developed. Unfortunately, some conditions have changed and the process is no longer working efficiently. Much of this year's work has focused on understanding what has caused this process change and upgrading the equipment necessary to increase the efficiency of neptunium production in the future. In addition, the remaining starting material from the original batch was exhausted during FY17. A large batch of new starting material was obtained and broken out into quantities that can be handled in the facility where the reduction is performed.

Project Description:

The disposal of spent nuclear fuel is one of the main challenges facing the nuclear power industry due to the long term health and safety hazards of storing the material once it is removed from the reactor. A key factor in reducing the hazards associated with long term storage is the reduction or elimination of minor actinides in the fuel such as neptunium, americium and curium, which have relatively long half-lives. Transmutation is one method currently under investigation to resolve this issue. Transmutation

Figure 2: Neptunium oxide reduction takes place in the casting laboratory glovebox in the analytical laboratory. Due to the hazard categorization of the facility and limited glovebox shielding only small quantities of neptunium oxide can be handled at any one time in this glovebox.

aims to incorporate the long lived actinides into new fuel which can be placed in a fast reactor where the elements of concern will fission into products which have shorter half-lives. In order to develop the process of transmutation it is necessary to first fabricate small quantities of fuel with minor actinide additives on which to perform thorough characterization and irradiation testing. Since neptunium is a key component in the fabrication of transmutation fuels and, currently, there is very little of this material in

Improving the efficiency of the neptunium oxide reduction process will provide vital neptunium metal feedstock for the Advanced Fuels Campaign and the Transuranic Breakout Glovebox provides INL with the capability to access quantities and types of transuranic material that were previously inaccessible.

existence as pure metal, it is necessary to develop methods to isolate it from the feedstock material that is available. The majority of the neptunium bearing feedstock available is in oxide form. A direct chemical reduction process was developed at INL to reduce the neptunium oxide to neptunium metal.

Accomplishments:

The process developed at INL to reduce the oxide to metal utilizes a mixture of calcium metal and calcium chloride. The calcium metal acts as the reducing agent to remove the oxygen from the neptunium. The calcium chloride salt absorbs the calcium oxide that results from the reduction of neptunium and oxidation of calcium. The process is performed in the Hot Uniaxial Press furnace (Figure 1) in which it can be heated to approximately 900 degrees centigrade and stirred. The furnace is located in the Casting Laboratory glovebox of the Analytical Laboratory (Figure 2). The furnace underwent some upgrades during FY17 that included complete change out of all thermocouples including the temperature override control thermocouples. A thorough cleaning process was also performed on the furnace since it was used for other processes and the build up from these processes and the reduction process seemed to be hampering reduction efficiency. The stock of neptunium oxide starting material was exhausted during FY17. Due to the hazard categorization of the facility where the work is performed and the limited glovebox shielding, only 40-50g of this material can be handled at any one time in the glovebox. For this purpose (and others) the shielded Transuranic Breakout Glovebox was brought on line in the Fuel Manufacturing Facility last year. This glovebox gives INL a significant capability for the breakout and repackaging of transuranic materials from a large variety of container types. The glovebox was put to use for the first time in FY17 to breakout approximately 6kg of neptunium oxide material (Figure 3) that was obtained from the Space Battery program. Much effort was put into additional shielding and process development to allow this breakout to happen with minimal personnel extremity and whole body dose. The material was broken down into quantities that can be handled in the Casting Laboratory glovebox where the reduction runs occur.

Figure 3. The process of breaking out 6kg of neptunium oxide into smaller batches for use in the process involved specialized tools and radiological shielding.

Preparation of Pu, Am and Np Standards for EPMA Analysis

Principal Investigator: Karen Wright
Collaborators: Cindi Papesch, Rob O'Brien

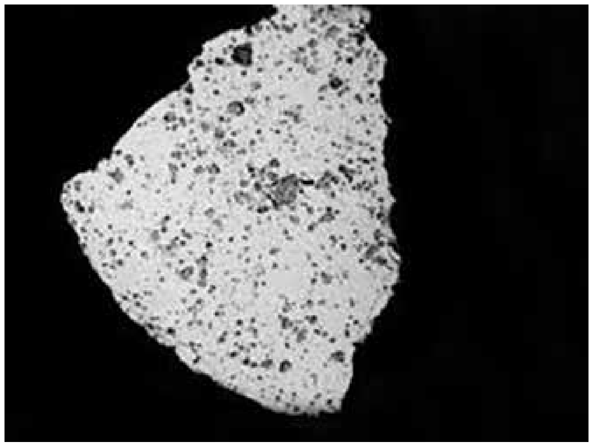


Figure 1. Typical heating and pressure profile from a fluidity test.

Robust, homogeneous, and well-characterized analytical standards are required to perform accurate quantitative analysis (Figure 1) using electron probe microanalysis (EPMA). While typical elemental standards (e.g. iron, silicon, chromium, etc.) can be procured, actinide materials (e.g. plutonium, neptunium) must be fabricated by INL for use. This research involved sintering oxides of uranium, neptunium, and americium, mounting them for use, and testing them for homogeneity.

Project Description:

The ability to use EPMA for quantitative analysis of irradiated fuels is predicated on the existence of well-characterized analytical standards for all relevant analytes, including fuel components such as neptunium, plutonium and americium. By fabricating standards for fuel components (e.g. Fig 1, NpO_2), we now have the ability to quantitatively analyze irradiated fuels, thus improving our ability to model fuel performance in the reactor. The ability to analyze and model fuel performance is key to building more robust and innovative fuels for future reactor development.

Accomplishments:

1. Np, Am, Pu oxide powders were pressed (Performed at FMF by FMF operations personnel)
2. Pressed pellets were sintered and sectioned in the MFC analytical lab by Cindi Papesch
3. Aliqouts of the pellets were mounted and polished at MFC EML by Nick Bolender
4. MFC analytical lab tested aliquots of the pellets to ascertain the actinide content
5. EPMA analysis for homogeneity was performed by Karen Wright at IMCL

The ability to use actinide analytical standards in EPMA analysis enables us to quantitatively analyze irradiated fuel.

Fabrication of Metallic Fuel Specimens for Laser Flash Analysis

Principal Investigator: Randall Fielding
Collaborators: Cynthia Papesch, Leah Squires, FMF Operations Crew

Table 1. Fabricated compositions and as cast mass.

Fabricated Compositions	As Cast Mass
90Pu-10Zr	9.901 g
70Pu-30Zr	9.951 g
70U-20Pu-10Zr	9.930
34U-29Pu-3Np-4Am-30Zr	9.917
65U-20Pu-3N p-2Am-10Zr	10.024 g

In support of the FCRD Transmutation Fuels Handbook several transuranic (TRU) bearing alloys were produced for subsequent laser flash diffusivity characterization. The previous year's work focused on producing legacy AFC-1 and AFC-2 series alloys suitable for differential scanning calorimetry. This year's work follows on producing the same alloy compositions in a size that is applicable to laser flash diffusivity measurements.

Project Description:

Over the years the Advanced Fuels Campaign and its predecessors have irradiated a number of alloys. Many of these alloys were characterized at the time of irradiation, however, techniques and equipment have been substantially improved since the initial characterization, and also due to budget and time restrictions some alloys received only minimal characterization. This is compounded for TRU bearing alloys due to the difficulties associated with characterization these alloys. Literature reviews have also shown that much of the historical data for these alloys and similar alloy compositions is either of lesser quality or contradictory in nature. In order for these data deficiencies to be filled, additional legacy material is needed for further characterization.

The transmutation fuel package has supplied fuel samples for thermal and microstructural characterization of legacy AFC-1 and AFC-2 using recently recovered minor actinide feedstocks of americium and neptunium.

The newly produced alloys were arc cast in much the same manner as previously, with the exception of the mold material. The original irradiation alloys were cast using a quartz mold, however, it has been found that a re-usable copper mold produces less waste, is more repeatable, and because molds are easily machined give greater diameter flexibility. Also, in some of the previous alloys, silicon contamination was found during the microstructural analysis, by using the copper molds silicon is removed from the process. Although copper contamination is also possible, due to its high thermal conductivity, and the large mass of the arc hearth and mold the copper is not heated to a high enough temperature to allow detrimental interactions with the mold and hearth. Previously, laser flash diffusivity samples have been cast using a 10-12 mm mold. However, this size of mold necessitates using large amount of limited TRU feedstocks, and is much larger than is needed for other characterization processes. Because of this it was decided that the FY17 characterization alloys would be cast at 5 mm diameter, which allow for additional laser flash analysis technique development. In this way a single casting batch can produced samples suitable for differential scanning calorimetry, microstructural analysis, chemical analysis, and laser flash analysis.

Accomplishments:
The fuel compositions were developed by Cynthia Papesch, the Fuel Characterization work package manager to support the Metal Fuel Handbook. In recent years many of the TRU bearing characterization and irradiation alloys consisted of only plutonium alloys, even though many of the AFC-1 and AFC-2 series alloys also contained the minor actinides americium and neptunium. A driver for the plutonium only compositions was the lack of available metal minor actinide feedstocks. However, through the contributions of Leah Squires in the Feedstock Preparation/Purification work package usable quantities of metallic minor actinides have become available and were provided for characterization alloy production.

Samples were produced at 5 mm diameter in order to facilitate laser flash diffusivity technique development using smaller than standard samples. Table 1 shows the composition, in weight percent, and the total mass cast.

All of the alloys produced are legacy AFC alloys or are related to these alloys. 90Pu-10Zr and 70Pu-30Zr were not irradiated but are related to two binary alloys that were included in the AFC-1B/1D tests. The phase data of compositions in this range is conflicting in the literature and should be studied further. 70U-20Pu-10Zr was irradiated in the AFC-2E test and more recently in the AFC-3F test. Also, this alloy composition was one of the standard compositions proposed for the Integral Fast Reactor concept and is very similar to other proposed ternary metal fuel cycle compositions. 34U-29Pu-3Np-4Am-30Zr was included in both the AFC-1F/1H tests and the FUTRURIX-FTA test. Further characterization of the alloy is needed to clarify historical data collected under less desirable conditions, for example, in an air atmosphere glovebox allowing air to ingress into the instrument during some of the measurements. The final alloy, 65U-20Pu-3Np-2Am-10Zr, was irradiated in AFC-2E, and much like the other alloys needs further data collected to verify or clarify earlier produced data. Figure 1 shows the as cast samples.

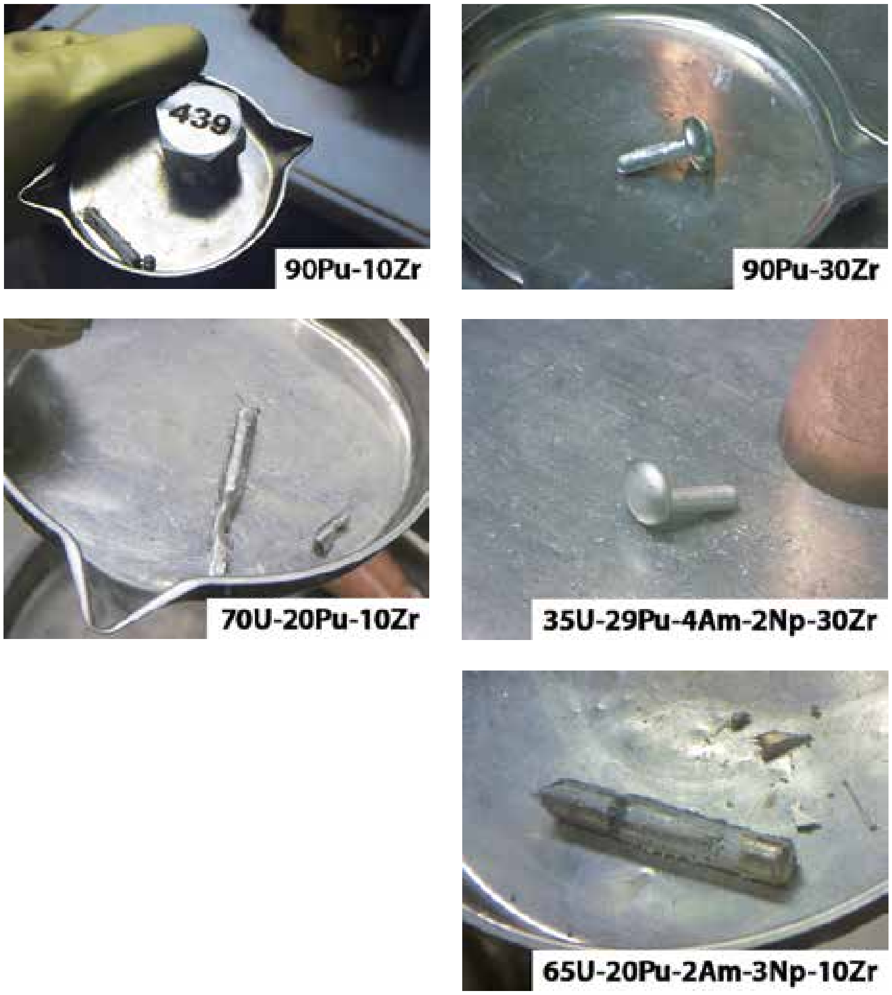


Figure 1. As cast characterization alloys

FCCI Barrier Development

Principal Investigator: Randall Fielding
Collaborator: Laura Sudderth

Sample ID	Charge Mass	Wt% Zr	Foil Thickness	Foil wt%	Pin Length (mm)
F7	11.8	10.0	0.0005	0.54	49.5
F8	11.82	9.64	0.005	6.52	53.3
F9	14.76	9.98	0.001	1.26	63.5
F10	14.736	9.96	0.002	2.44	63.5

Table 1. Table showing the foil thickness and contribution to the overall zirconium percentage used for the first series of tests.

Despite the robustness of metal fuels, they are susceptible to fuel cladding chemical interactions (FCCI). FCCI is caused by fission product rare earth elements interacting with the iron based cladding. This reaction effectively reduces the cladding thickness, which is major limiting factor for high burnup metal fuels. If this FCCI could be mitigated, possible burnup limits could also be increased. A method of mitigating FCCI in metallic fuels is to incorporate an integral FCCI barrier material, such as zirconium, between the fuel and cladding. The initial feasibility of integral fuel FCCI barrier was shown previously during the Integral Fast Reactor program. However, during post irradiation examination, it was seen that as the fuel swelled, the sheath was split, and the fuel extruded out. In areas where the fuel was still contained in the sheath, there was no evidence of FCCI. A possible strategy to accommodate fuel swelling is to incorporate a foil wrapped around the fuel with an overlapping section that corresponds to the amount the fuel will swell. As the fuel swells, the foil will unwrap while maintaining complete coverage of the fuel.

Project Description:

2016 work showed the feasibility of arc casting a zirconium wrapped sample suitable for AFC-OA style irradiation tests. 2017 work focused on optimizing the process to determine the optimal zirconium foil thickness, minimize the amount of fuel between the overlapped zirconium layers, and to characterize the foil/fuel alloy bond. Several casting trials were performed. The first trial used 4.3 mm diameter glass molds wrapped with four zirconium foil thicknesses; 12 µm, 25 µm, 50 µm, and 125 µm. By varying the foil thickness not only is the minimum amount of non-fissile zirconium determined, but also the physical limitations associated with handling of the foil were better determined. After the optimal thickness was determined based on which pin held its shape better, a foil oxidation study was performed. During the minimum foil thickness study it was observed that fuel alloy would often infiltrate between the overlapping sections of foils and evidence of interaction (metallurgical bond formation) was seen. In order for the foil to unwrap as the fuel swells a metallurgical bond is not desired. To impede this bonding a series of castings were done using zirconium foil that was heat treated in an air atmosphere at 650°C for 1 hour to form an oxide layer. Both 25 µm and 50 µm foils were heat treated. An added benefit of heat treatment and oxidation is the heat treatment stiffened the zirconium foil. When the foil is stiffer, it better maintains its structural integrity and is less susceptible to physical damage from the molten fuel. This prevents potential pathways for the molten fuel to escape the foil from developing. Based on the results of casting into the pre-oxidized foils, a final set of experiments was performed in which 50 µm foil was formed, heated to 650° C for 1 hour in air, and the interior surface coated with a ZrO_2 slurry to further prevent interaction.

Accomplishments:

Four U-10Zr pins were cast using different thickness of zirconium foil. Table 1 shows the mass of the cast material, initial zirconium weight percent, thickness of the foil, the weight percent of zirconium liner (essentially raising the zirconium content of the fuel), and finally the total fuel length cast. As seen in Table 1, the thickness of the foil can be a substantial contributor to the overall amount of zirconium, therefore, minimizing the foil thickness is desirable. During the assembly of the molds it was noted that the 12 µm foil is much harder to load into the quartz mold, because it deforms easily

Figure 1. Archival X521 fuel slug that was used to supply the injection cast U-10Pu-10Zr fuel slug for AFC-3F.

Fabrication of fuel incorporating integral zirconium FCCI barriers has been optimized to point where irradiation testing is feasible.

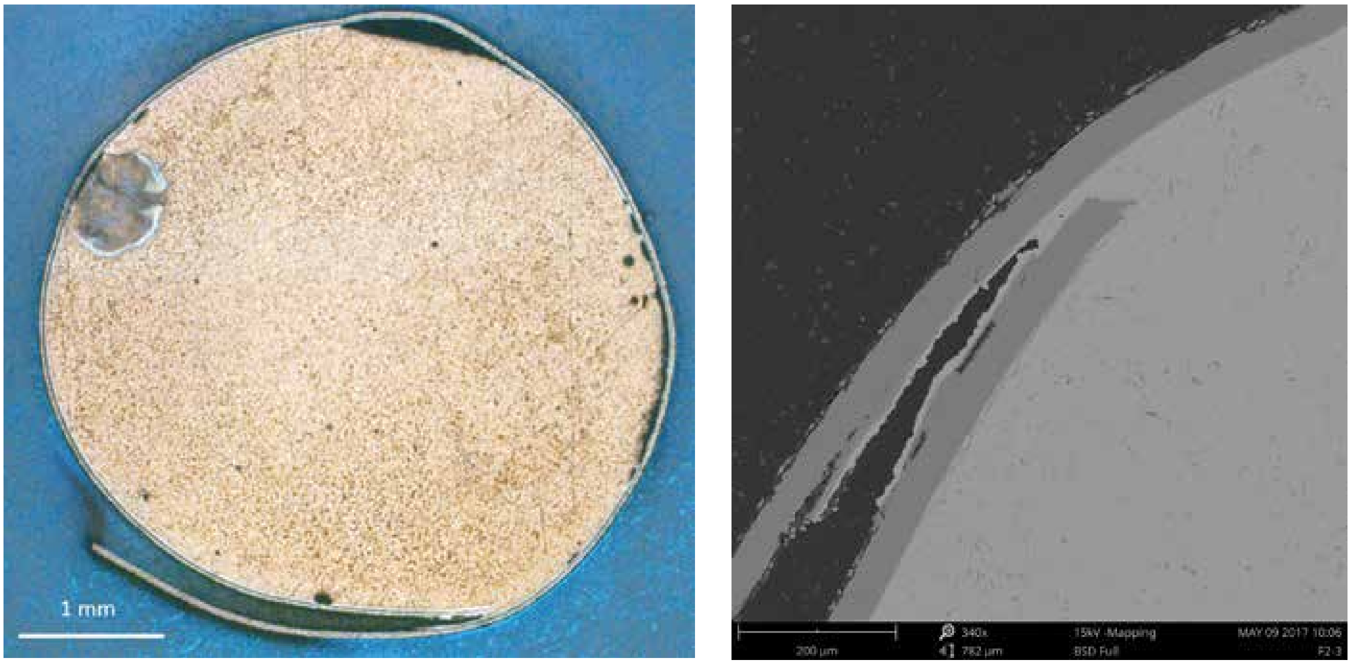
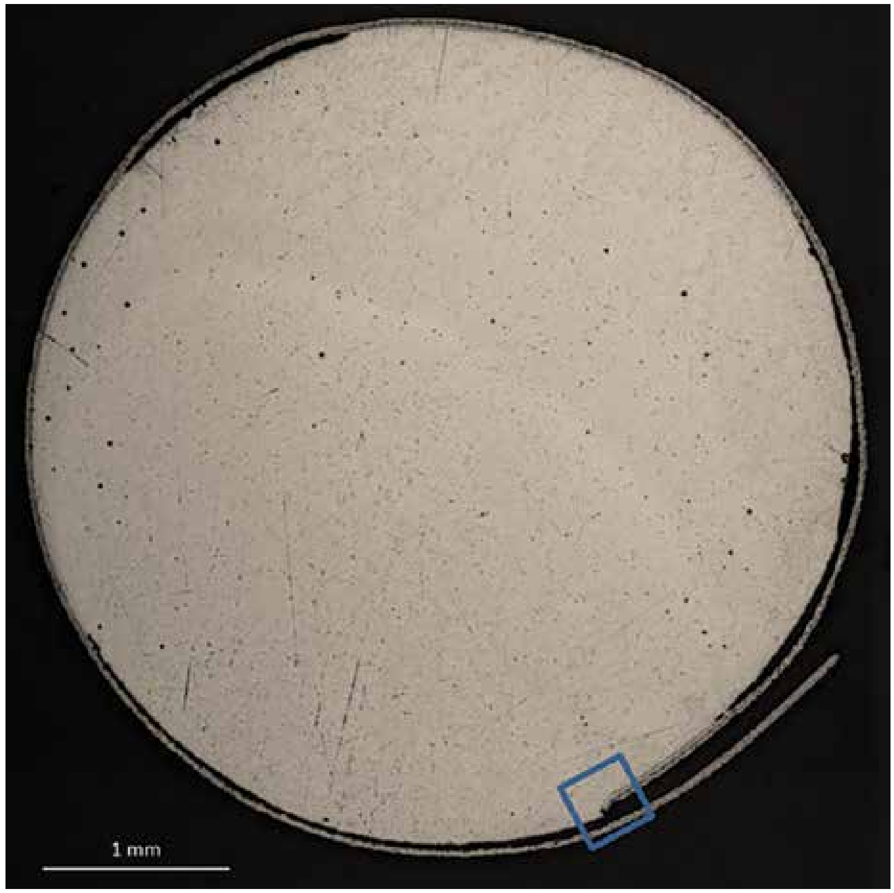


Figure 2. Middle section of heat treated 50 µm foil. SEM image (Right) shows minor infiltration and brittle nature of a possible interaction product.

and lacks the stiffness to maintain its desired shape, giving a loose fit in the quartz mold. This loose fit in the mold allows the melt to flow around the foil, and the lack of structural integrity of the foil can allow portions of the foil to be pushed into the center of the fuel. On the other hand, the 125 µm foil is quite stiff. Due to its stiffness there was some difficulty with mold assembly, but the ability of this foil to retain its shape during casting produced cast pins with only a little infiltration of the foil near the top of the pin. Figure 1 shows a cross section of the middle of the 125 µm foil pin. Notice the foil fully wraps around the fuel with no infiltration. The 25 µm foil had similar handling difficulties to the to the 12 µm foil, leading to fuel alloy flowing around the foil fully encapsulating the zirconium foil in some areas. The behavior of the 50 µm foil more closely resembled that of the 125 µm foil, but still presented some defects. Although the 50 µm foil did not contain the fuel as desirably as the 125 µm foil, it handled easier and provided a more desirable zirconium content in the fuel pin.

Based on the above results another set of experiments was performed in which the 25 µm and 50 µm foils were heat treated in an air atmosphere to produce an oxide layer on the foil and to impart additional stiffness. Two samples of each thickness were heat treated, one sample was rolled and inserted in the glass mold prior to heat treatment and the other sample was heat treated while flat. After heat treating, both samples were a uniform dark color. Casting results showed pre-rolled samples performed better. The samples rolled post heat treatment did not hold the rolled shape as well as the post heat treatment rolled samples, and in some cases, the foil was interior to the fuel. While the 25 µm foil continued to behave poorly, the 50 µm foil rolled pre-heat treatment appeared to better maintain integrity and shape through the casting process. Although some minor infiltration and evidence of interaction was also observed, its overall performance was improved. Figure 2 shows an example of the minor infiltration, and an overall view of the fuel.

The final round of experiments used a slurry coating of ZrO_2 on the interior surface of the 50 µm foil after heat treatment to further mitigate interactions and to possible seal any openings for infiltration. Casting and heat treatments were performed in a similar manner to previous tests. The resulting castings showed a gap between the fuel foil wrap. Figure 3 shows a cross section of the final pin. This was the desired result to ensure the foil would not bond to the fuel and not be able to unwrap during irradiation. Therefore, this preparation technique with the 50 µm foil was identified as the preferred method for fabrication via arc-melting, and was shown to be capable of producing the desired results.



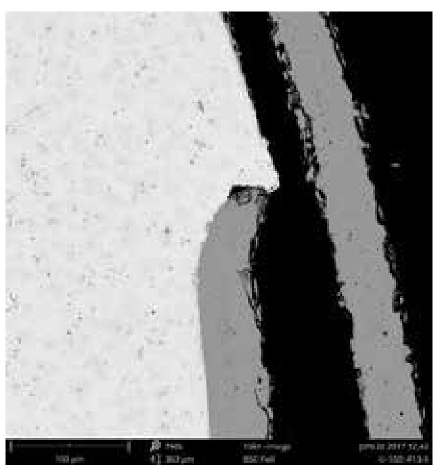


Figure 3. Heat treated 50 µm foil with ZrO_2 coating. SEM image (Top) shows area highlighted in left image. Notice no discernable infiltration.

Alloy Optimization Casting and Characterization Studies

Principal Investigator: Michael T. Benson
Collaborators: James King, Robert Mariani

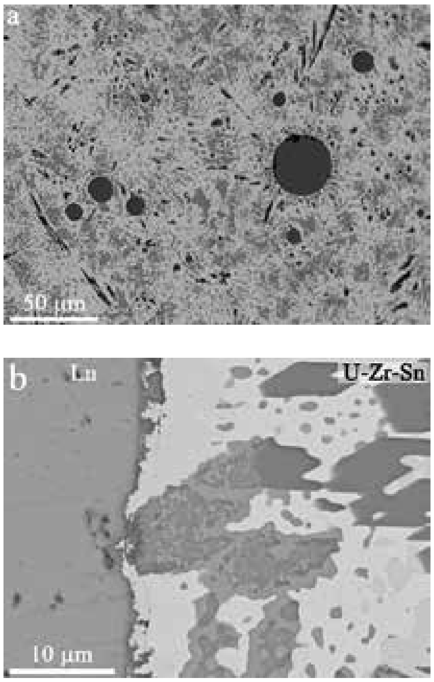


Figure 1. a) SEM backscatter image of U-10Zr-4.3Sn-4.7Ln; b) SEM backscatter image of U-10Zr-4.3Sn/Ln diffusion couple interface.

Fuel-cladding chemical interaction (FCCI) occurs when the nuclear fuel or fission products react with the cladding material. A major cause of FCCI in U-Zr and U-Pu-Zr fuels during irradiation is fission product lanthanides (Ln), which migrate to the fuel periphery, coming in contact with the cladding. The result of this interaction is degradation of the cladding, and will eventually lead to rupture of the fuel assembly. Tin and palladium are being investigated as minor component additives to control FCCI in metallic fuels specifically due to lanthanides. The role of the additive is to prevent FCCI by forming very stable intermetallic compounds with the lanthanides, thus preventing interaction with the cladding. Studies are underway to characterize the effects of these additives in a metallic fuel.

Project Description:

The technical objectives of this research are to investigate additives to metallic fuels to improve the performance. Previous work on palladium has shown promising results for controlling FCCI. The current work is a continuation of that work with palladium, as well as expanding the investigation to include tin. The objective of this work is to characterize the microstructure of the metallic fuels with these additives, and to evaluate performance in out-of-pile tests. This work was carried out using U-Zr based fuels as well as U-Pu-Zr based fuels.

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

Accomplishments:

In FY17, investigations began using tin as an additive to control FCCI. Alloys analyzed include U-10Zr-4.3Sn-4.7Ln, as well as alloys with a high or low lanthanide loading, U-10Zr-2Sn-4.3Ln and U-10Zr-4.3Sn-2Ln. Figure 1a shows the scanning electron microscope (SEM) image of U-10Zr-4.3Sn-4.7Ln. EDS analysis indicates the round precipitates are Ln_5Sn_3. The long, dark precipitates are Zr_5Sn_3, while the matrix is comprised of alpha-U (light areas) and delta phase (UZr_2, dark grey areas). To be considered a viable fuel additive, the additive has to preferentially bind the lanthanides over other fuel constituents. The as-cast fuel indicates a preference to bind the lanthanides. A small amount of Zr-Sn precipitates are still present due to a small excess of Sn over the lanthanides.

To better investigate and understand the interaction between the additive and the lanthanides, a diffusion couple was run between U-10Zr-4.3Sn and the lanthanides, shown in Figure 1b. In the as-cast and annealed alloys, high temperature phases generated in the arc melter are possible; phases that won't form under reactor temperature, i.e. 650°C. In the diffusion couple, the temperature never goes beyond 650°C. In Figure 1b, the lanthanides are on the left, and the alloy on the right. The large, dark precipitates in the upper right hand corner are Zr_5Sn_3. The lanthanides are aggressively reacting with the precipitates near the interface. Ln_5Sn_3 is forming, releasing Zr from the precipitate. Uranium is coming into what was once the Zr_5Sn_3 precipitate and forming UZr_2 with the excess Zr. This result shows conclusively the preference for Sn to bind the lanthanides over other fuel constituents. This result, along with the stability of Zr_5Sn_3 in the as-cast fuel, show tin to be an excellent additive material for controlling FCCI.

To continue the investigation of tin, more diffusion couples are underway. In addition, two alloys, U-20Pu-10Zr-4.3Sn and U-20Pu-10Zr-4.3Sn-4.7Ln, were fabricated in FMF in FY17, to explore tin interactions in an alloy containing Pu. Characterization and diffusion couples will be run in FY18.

In FY16, 2 alloys with Pd included were fabricated in FMF. During FY17, those alloys were cut for analysis, and transferred to the appropriate facility for characterization. The alloy compositions (in wt %) are U-20Pu-10Zr-3.86Pd, U-20Pu-10Zr-3.86Pd-4.3Ln (where Ln = 53Nd-25Ce-16Pr-6La). A third alloy, U-19Pu-0.7Zr-4.3Ti-5Mo, was also fabricated and sectioned for analysis. This is follow-on work to the U-0.7Zr-4.3Ti-5Mo alloy investigated at Colorado School of Mines. All three alloys have been characterized by SEM, and the Pd alloys have also been characterized with transmission electron microscopy (TEM). Diffusion couples have been run between the 3 alloys and iron. Those samples will undergo SEM and possibly TEM analysis in FY18. In addition to analyzing the diffusion zone, the bulk material represents the annealed microstructure, and will be investigated as well.

Fuel-cladding chemical interaction due to fission product lanthanides can be prevented using additives to react with the lanthanides.

MOX Fabrication Trials

Principal Investigator: Seongtae Kwon
Collaborators: Randall Fielding

The key fabrication processing capability of substoichometric MOX fuel by fine controlling oxygen partial pressure has been established.

The fabrication capability of MOX fuel in INL has been inactive for years after several discontinuations in activity. Atmosphere control is one of the key factors in successful fabricating MOX (mixed oxide) fuel for high burnups. Regaining MOX fabrication capability in INL is attempted through fine controlling oxygen partial pressure in the sintering furnace for substoichiometric compositions required for high burnups.

Project Description:

Precise stoichiometry control of MOX fuel is achieved by control of oxygen partial pressure during the sintering of the oxide fuel. For this fine tuning, the gas panel for the sintering furnace in FMF is required to be upgraded for the stoichiometry control after consulting with Kenneth McClellan to utilized LANL's experience in MOX fabrication. Main modifications include

- replacing current analog mass flow controller to digital controllers
- installing oxygen monitors for the gas flows in and out of the furnace
- removing extra thread–sealing points including desiccant filter.

Other improvements required for the modification of the SCS (substoichiometric control system) include, establishing methods for gas mixture using lecture bottles, upgrading and modifying controlling software according to modified procedures, and developing means for a wide range of oxygen partial pressure for various stoichiometry.

The main compositions to be explored for this initial stage of MOX fabrication will be (U0.8Pu0.2) O_2-x composition with x 0~0.08 range. Processing condition map for the target stoichiometry will be developed for the stable fabrication of MOX fuel. The project will be further developed to produce stable microstructure of the fuels to mitigate the dimensional change along with retaining porous structure to accommodate fission products.

Accomplishments:

Digital mass flow controller and oxygen monitor recommended by LANL were located, procured and installed. Due to the discontinuation of the digital mass flow controller model recommended, MKS 647C, a newer version, MKS 946, was selected. Rotameters for dry Ar, wet Ar, hydrogen and nitrogen was installed to visually monitor the flow of gases. Software has been revised to accommodate procedure and component changes.

For the fine tuning of low oxygen partial pressure, a wet Ar with 300 ppm water content in a lecture bottle has been located and purchased. Wet Ar was mixed with hydrogen to produce oxygen partial pressure desired for the sintering atmosphere. A table indicating oxygen partial pressure for specific water/hydrogen ration has been developed as shown in Table 1. The amount of oxygen partial pressure known to be required to achieve 1.92 oxygen subtoichiometry is known to be 5 E-11 at 1600°C. The current setting of gas mixture is expected to produce low stoichiometry range compositions, but actual outcome needs to be verified since the calculation does not count for the contribution of furnace components to the oxygen partial pressure.

Acceptance test plan (ATP) is being conducted to verify the function of the new components, new software and the furnace. The furnace has been gradually heated up to 1800°C (yet plan to be done by early Sept.). During the heating steps oxygen sensing capability from oxygen monitor, oxygen partial pressure manipulation by changing mixed gas ratio between the amount of hydrogen and moisture content from the moisture sensor was monitored along with flow of each gas. Achieving a stable furnace temperature control by optimizing proportional band and integral reset have been performed during ATP.

A practice pressing using depleted uranium oxide which will be a matrix phase of the MOX fuels to be fabricated has been performed to ensure the pressing capability in the glove box. Modification of pressing mold with tighter tolerance and verifying the powder characteristic are planed based on the result of the practice pressing.

H_2/H_2O	PO_2	
	1600ºC	650ºC
100	1.39E-12	6.93E-27
50	5.54E-12	2.77E-26
30	1.54E-11	7.70E-26
20	3.46E-11	1.73E-25
16.644	5.00E-11	2.50E-25
10	1.39E-10	6.93E-25
5	5.54E-10	2.77E-24
1	1.39E-08	6.93E-23

Table 1. Modified SCS (substoichiometric control system) gas panel

Demonstration of Remote Casting of Metallic Fuels

Principal Investigator: Randall Fielding
Collaborators: Blair Grover

A DU-10Zr pin was remotely cast using the Casting and Sampling furnace located in the Hot Fuel Examination Facility.

	Mass (g)	Wt. %
Uranium	176.221	89.9
Zirconium	19.70	10.1

Table 1. Charge used for the U-10Zr gravity casting run.

Although many of the fuel pins fabricated for the Advanced Fuels Campaign are produced using fresh actinide feedstocks which can be safely handled in a glovebox, the overall goal of many of the advanced reactor fuels is to use transmutation fuels produced using recycled feedstocks. Recycled transuranic material, especially electro-metallurgically recycled material, contains fission product contamination which can drive remote handling for safety. Therefore a major goal for advanced fabrication is to develop fabrication techniques suitable for remote deployment.

Project Description:

The Casting/Sampling furnace can be configured to cast metallic fuel using either the gravity method or the counter gravity injection (suction) method. In the gravity casting method, the feedstock is loaded into an yttria coated graphite crucible. As the material is heated to a molten state a stopper rod is pulled out of the crucible which allows the alloy to flow out of the crucible into the permanent graphite mold coated with zirconium oxide. Counter gravity casting uses the same furnace, although the molten material is drawn up into a zirconium oxide coated quartz mold using a pressure differential between the mold and the furnace chamber. Counter gravity casting is similar to the process which was used for the casting of EBR-II fuel although on a much smaller scale.

Although suction casting has been found to be less sensitive to operational parameters such as atmosphere contamination and alloy composition as compared to gravity casting, it also has a high required recycle stream and utilizes single use molds, creating additional waste, and a route for fuel losses. Gravity casting, on the other hand is more sensitive to operational parameters and mold design, but uses a permanent re-usable mold and has a higher melt utilization rate compared to counter gravity injection casting. Under the AFC program the main focus of development has been gravity casting, because of the smaller recycle and waste streams, reduced fuel losses, and the ability to cast at either ambient or increased pressure, despite the increased parameter sensitivity.

Accomplishments:

The Casting/Sampling furnace was installed in the Hot Fuel Examination Facility at window 10M. Initial casting attempts resulted in the charge not fully consolidating and the crucible and stopper rod adhering together, thereby not allowing material to flow out of the crucible. The charge consolidation issue was solved through repeated bake-outs and runs of the furnace. During mock-up and qualification activities the furnace chamber was not maintained under a vacuum because it was necessary to move and furnace system multiple times. Equipment qualification activities in cell were performed during a hot cell maintenance outage. During this time the cell atmosphere deteriorated, allowing more moisture to build up than normal. As the furnace was open and closed and moved around on the table due to other operations moisture was adsorbed into the furnace refractory materials which were not released until several high temperature runs, including earlier casting attempts. During the mock-up phase of qualification it was seen that often the crucible stopper rod and loaded crucible would adhere together at high temperatures, despite lifting the stopper rod. In order to reduce this adherence the stopper/crucible interface was modified to reduce the amount on contact area between the two parts.

After the appropriate modifications had been made a yttria coated crucible was loaded with depleted uranium and zirconium pieces and placed in the Casting/Sampling furnace while it was configured for gravity casting. Table 1 shows the masses of the original charge material. The furnace was heated to 1500°C and held for approximately 10 minutes. Following the hold time the stopper rod was lifted allowing the fuel to drain from the crucible into the mold. During the last two minutes of the heating cycle, prior to lifting of the stopper rod, the furnace chamber was evacuated to ensure material could flow into the mold. Figure 1 shows the heating cycle. The resulting pins are shown in Figure 2. Although the pins were not full length, this experiment shows that the concept is feasible and equipment functions while some optimization is necessary. An example of the optimization includes a slight modification to the mold. Despite being evacuated for the two minutes prior to casting, it appears from the morphology of the fuel slugs that gas remained trapped in the mold cavities. This has been verified using a very similar mold design, the Glovebox Advanced Casting System (GACS), and the same heating profile. The mold modification will include a vent path out of the bottom of the mold cavity. As the material flows into the cavity, any residual gas is pushed out the vent. The vent will be made small enough that any molten material will quickly freeze before it can flow out of the cavity.

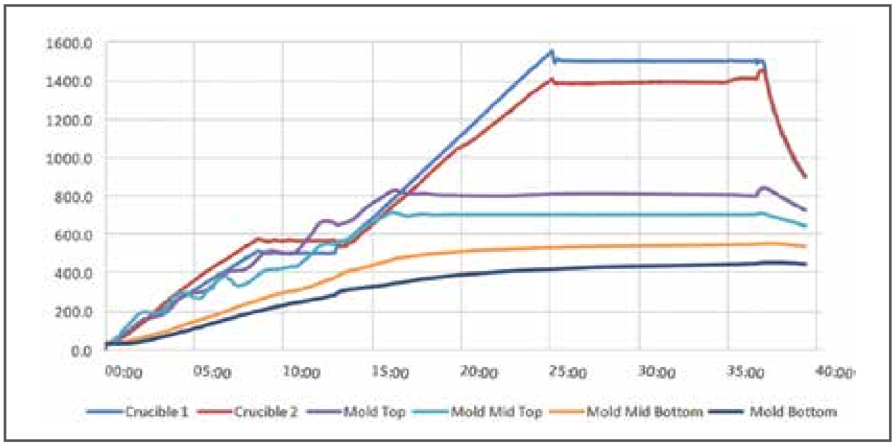


Figure 1. Temperature (^{0}C) versus run time (mm:ss) plot of the remote casting run.

Figure 2. One of the resulting remotely cast U-10Zr pins.

Thermal and transport properties of U_3Si_2

Principal Investigator: Krzysztof Gofryk
Collaborators: D. Antonio, K. Shrestha, C. Papesch, Y. Zhang, J. Harp, and Jon Carmack

Understanding of thermal and electrical transport of nuclear materials

U_3Si_2 is considered as a new fuel for the existing LWR fleet. While nuclear fuel operates at high to very high temperatures, measurements of thermal conductivity and other materials properties lack sensitivity to temperature variations and to material variations at reactor temperatures, especially those related to heat transport, scattering interactions, and electronic correlations. These variations need to be characterized as they will afford the highest predictive capability in modeling and offer best assurances for validation and verification at all temperatures.

Project Description:

Nearly 20% of the world's electricity today is generated by nuclear energy. Since the thermal properties, especially thermal conductivity, of the fuel governs the conversion of heat produced from fission events into electricity; it is an important parameter in reactor design and safety. Therefore detailed understanding of electronic properties and the way the nuclear materials transport heat are of paramount interest of nuclear energy research and one of the DOE missions. Uranium-silicon compounds have been extensively investigated for use as nuclear fuels in new generation reactors. In particular, U_3Si_2 has become a material of interest for its potential as an accident tolerant fuel used in commercial light water reactors (LWR). Several factors make U_3Si_2 appealing, including a higher uranium density than UO_2 (currently the most commonly used commercial fuel). An improved thermal conductivity over the comparatively poor UO_2 is also of particular note, as better thermal properties can contribute to smaller temperature gradients during reactor start up, which could reduce cracking of fuel pellets. A higher thermal conductivity is also beneficial during some accident scenarios. Although much effort has been dedicated to studying the thermo-physical properties of U_3Si_2 at high temperatures, there is only limited data available at lower temperatures. The operating temperatures in nuclear reactors are high (~1000 K), however, many important physical characteristics such as the effect of electronic correlations and/or impact of defects and other degrees of freedom on the electrical and heat transport in nuclear materials are all emphasized at moderate or low temperatures. Therefore, in order to better understand the nature of the 5felectrons and mechanisms that govern electrical and heat transport in this important technological material, and to accurately model this compound at all relevant temperatures, these effects must be quantified.

Accomplishments:

In order to understand the electronic and thermal properties we have studied U_3Si_2 by means of the heat capacity, electrical resistivity, Seebeck and Hall effects, and thermal conductivity. Polycrystalline samples of U_3Si_2 were prepared by arc-melting stoichiometric amounts of elemental U and Si. The arc-melted ingots were then comminuted into powder, pressed and sintered into pellets. Samples for this work were sectioned from these pellets. Powder diffraction confirmed that the structure was tetragonal (space group P4/mbm) with lattice parameters similar to those previously reported in literature. A sketch of the crystal structure of U_3Si_2 is shown in the inset of Figure 1. The thermal conductivity, resistivity, Hall effect, and heat capacity measurements were done in a DynaCool Quantum Design Physical Property Measurement System. All the results obtained, especially small magnetoresistivity, large low-temperature heat capacity, and characteristic dependence of the Seebeck coefficient, point to delocalized nature of 5f-electrons in this material. The electrical resistivity is reduced with lowering temperature, characteristic of metals (see Figure 1). Also the magnitude and temperature dependences of Hall and Seebeck effect are typical for correlated metals. The low temperature heat capacity is enhanced and shows an upturn in Cp/T (T), characteristic of spin fluctuations (see Figure 2). The thermal conductivity of U_3Si_2 is ~8.5 W/m-K at room temperature and we show that the lattice part of the total thermal conductivity is small, with electrons dominating heat transport above 300 K (see Figure 3). As shown in the inset of Fig. 5, at about room temperature the results are very close to previous high temperature studies, and the trends with respect to increasing temperature are nearly identical. In this inset we also include a high temperature laser-flash measurements performed on the same samples as used in the low temperature studies and previous laser flash measurements performed at LANL by White at al [J.T. White et al.,"Corrigendum to "Thermophysical properties of U_3Si_2 to 1773 K, J. Nucl. Mater. 484 (2017) 386–387]. This knowledge of the details of the heat transport in U_3Si_2 will be useful for researchers working on modeling and simulations of this new advanced fuel.

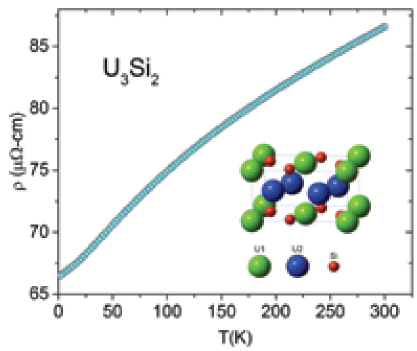


Figure 1. The temperature dependence of the electrical resistivity of U_3Si_2. The inset shows the tetragonal unit cell of U_3Si_2.

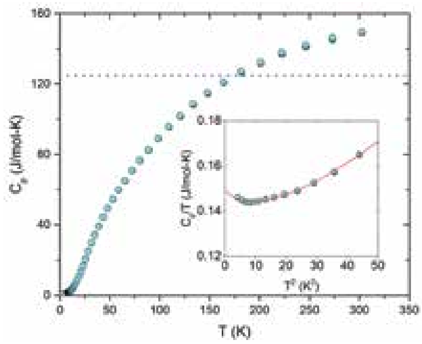


Figure 2. The temperature dependence of the heat capacity of U_3Si_2. The dotted line marks the theoretical Dulong-Petit value. Inset: the low temperature part of the heat capacity of U_3Si_2.

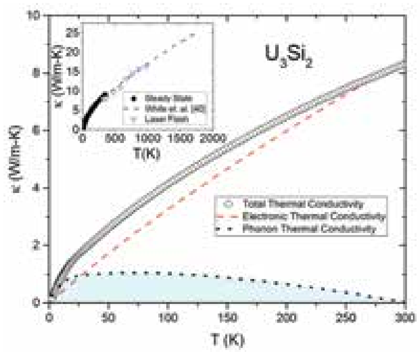


Figure 3. The temperature variation of the thermal conductivity of U_3Si_2. The dashed line is the electronic component while the dotted line represents the lattice part of the thermal conductivity of U_3Si_2. Inset: the thermal conductivity of U_3Si_2 from 2 up to 1800 K.

Update of the Metallic Fuels Handbook

Principal Investigator: Dawn E. Janney
Collaborators: Cynthia A. Papesch


Metallic Fuels Handbook,
Part 1:
Alloys Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr, Including Those with Minor Actinides (Np, Am, Cm), Rare-earth Elements (La, Ce, Pr, Nd, Gd), and Y

Dawn E. Janney

With special assistance from Cynthia A. Papesch and contributions from Douglas E. Burkes, James I. Cole, Randall S. Fielding, Steven M. Frank, Thomas Hartmann, Timothy A. Hyde, Dennis D. Keiser, Jr., J.Rory Kennedy, Andrew Maddison, Robert D. Mariani, Scott C. Middlemas, Thomas P. O'Holleran, Bulent H. Sencer, and Leah N. Squires

August 2017

Figure 1. A screenshot of Part 1 of the updated Metallic Fuels Handbook


Metallic Fuels Handbook,
Part 2:
Elements and Alloys not Based on U-Zr, Pu-Zr, U-Pu, or U-Pu-Zr

Dawn E. Janney

With special assistance from Cynthia A. Papesch and contributions from Douglas E. Burkes, James I. Cole, Randall S. Fielding, Steven M. Frank, Thomas Hartmann, Timothy A. Hyde, Dennis D. Keiser, Jr., J.Rory Kennedy, Andrew Maddison, Robert D. Mariani, Scott C. Middlemas, Thomas P. O'Holleran, Bulent H. Sencer, and Leah N. Squires

August 2017

Figure 2. A screenshot of Part 2 of the updated Metallic Fuels Handbook

Transmutation of minor actinides such as Np, Am, and Cm in spent nuclear fuel is of international interest because of its potential for reducing the long term health and safety hazards caused by the radioactivity of the spent fuel. One important approach to transmutation involves incorporating minor actinides into U-Pu-Zr alloys, which may also include rare-earth elements (La, Ce, Pr, Nd) as a result of previous reprocessing as part of a closed fuel cycle.

It is, therefore, important to understand not only the properties of U-Pu-Zr alloys but also those of U-Pu-Zr alloys that also include minor actinides (Np, Am) and rare-earth elements (La, Ce, Pr, and Nd). However, the existing experimental data is widely scattered, and much of it was published before ~1975. The Metallic Fuels Handbook summarizes the available experimentally based knowledge of several key properties of U-Pu-Zr alloys with minor actinides and rare-earth fission products.

Project Description:

Transmutation of minor actinides such as Np, Am, and Cm in spent nuclear fuel is important because of its potential for reducing long-term health and safety hazards caused by the radioactivity of the spent fuel. One important approach to transmutation (currently being pursued by the DOE Fuel Cycle Research & Development Advanced Fuels Campaign) involves incorporating the minor actinides into U-Pu-Zr alloys, which can be used as fuel in fast reactors. These fuels are well suited for electrolytic refining, which leads to incorporation rare-earth fission products such as La, Ce, Pr, and Nd. It is, therefore, important to understand not only the properties of U-Pu-Zr alloys but also those of U-Pu-Zr alloys that include minor actinides (Np, Am) and rare-earth elements (La, Ce, Pr, and Nd) in concentrations relevant for transmutation fuels.

Previous revisions of this Handbook included information about elements, binary alloys, and ternary alloys in the U-Np-Pu-Am-La-Ce-Pr-Nd-Zr system. This version revises and updates this information, and incorporates data about alloys with four or more elements (much of it originally generated at INL). It also adds information about alloys investigated by the METAPHIX Programme, a collaboration between the Central Research Institute of Electric Power (CRIEPI) in Japan and the Joint Research Centre-Institute for Transuranium Elements of the European Commission (JRC-ITU).

The handbook is intended to serve several audiences, including project leadership and management, researchers, and modelers.

Accomplishments:

The majority of the research and writing for this revision of the Handbook was done by the P.I., who is an INL employee. Some information was obtained from INL publications (including some that are not generally available). Authors of these publications are Douglas E. Burkes, James I. Cole, Randall S. Fielding, Steven M. Frank, Thomas Hartmann, Timothy A. Hyde, Dennis D. Keiser, Jr., J.Rory Kennedy, Andrew. Maddison, Robert D. Mariani, Scott C. Middlemas, Thomas P. O'Holleran, Cynthia A. Papesch, Bulent H. Sencer, and Leah N. Squires. Although some of these people are no longer employed at INL, all were INL employees when they performed the work used in the Handbook. Personnel from the INL Research Library provided invaluable help in finding obscure references, Dr. Pavel Medvedev assisted with Russian-language translation, and Dr. Steven Hayes provided insight and valuable guidance.

It is expected that the Metallic Fuels Handbook will be a multi-year project with new versions available periodically. Each version will update the data in the previous versions, as well as expand the scope of the Handbook. The original Handbook (in 2015) included information about elements and alloys in the U-Pu-Zr system. The first revision (in 2016) contained information about the elements, binary alloys, and ternary alloys in the U-Np-Pu-Am-La-Ce-Pr-Nd-Zr system. This Handbook, the second revision, adds information about alloys whose nominal compositions include at least four elements. Each of the revisions includes information about phases and phase diagrams, heat capacities, thermal expansion, and thermal conductivities and diffusivities for the materials it covers. The Handbook has a strong bias in favor of experimental data when available, but supplements the experimental data by modeling results as needed. The Handbook contains numerous references to other modeling papers.

The organization of the 2017 revision of the Handbook is significantly different from that of earlier versions, and presents all of the available information about each material or group of related materials together. This organization is intended to make it easier to compare materials with similar compositions (for example, to assess whether addition of minor actinides significantly changes the properties of U-Pu-Zr alloys, or to understand any changes caused by incorporation of rare-earth elements into these alloys).

Because of its size, the 2017 revision has been split into two documents: Part 1 contains information about U-Zr, Pu-Zr, U-Pu, and U-Pu-Zr alloys with and without minor actinides (Np, Am, Cm), rare-earth elements (La, Ce, Pr, Nd, Gd) and Y. Part 2 contains information about elements and other alloys in the U-Np-Pu-Am-La-Ce-Pr-Nd-Zr system.

References considered for the 2017 revision of the Handbook were published from the late 1930s through 2017.

Some information was obtained from INL publications whose authors are Douglas E. Burkes, James I. Cole, Randall S. Fielding, Steven M. Frank, Thomas Hartmann, Timothy A. Hyde, Dennis D. Keiser, Jr., J. Rory Kennedy, Andrew. Maddison, Robert D. Mariani, Scott C. Middlemas, Thomas P. O'Holleran, Bulent H. Sencer, and Leah N. Squires. Although some of these people are no longer INL employees, all were employed at INL when they did the work included in the Handbook. The staff of the INL Library provided indispensable assistance in finding often obscure publications. Dr. Pavel Medvedev (INL) translated a paper from Russian to English. Dr. Steven Hayes (INL) and Dr. Steven Frank (INL) provided helpful discussions.

Provides a single source of information about important properties of materials in the U-Np-Pu-Am-La-Ce-Pr-Nd-Zr system and a critical assessment of the information based on an extensive survey of widely scattered, often obscure, sources.

Modeling Restructuring in MOX Fuels

Principal Investigator: Pavel Medvedev
Collaborators: Steven Hayes, Samuel Bays, Stephen Novascone, Luca Capriotti

This is an important step in establishing the relevancy of fuel performance data generated for fast reactor fuels using ATR cadmium-shrouded experiments.

The present lack of a domestic fast neutron flux irradiation capability combined with continued development of fast reactor fuels in the U.S. motivated an innovative engineering solution to utilize a unique neutron flux tailoring capability in the Advanced Test Reactor (ATR) at the Idaho National Laboratory. To achieve the objectives of the fast reactor fuel irradiation tests, the incident neutron flux was hardened substantially by placing fueled irradiation capsules inside specially designed cadmium shrouds. Use of cadmium prevents thermal neutrons from reaching the fuels being tested and alleviates the plutonium self-shielding that would normally arise during irradiations of high density, highly enriched fuels in a thermal neutron spectrum.

Project Description:
This work illustrates the profound effect this engineered solution has on the efficacy of the experiments. Based on the comparison of postirradiation measurements of the columnar grain region in fast reactor, mixed oxide fuels with fuel performance calculations, it is demonstrated that thermal conditions achieved in these cadmium-shrouded fuel experiments are substantially prototypic of a sodium fast reactor and are suitable for concept-screening tests supporting development of new fast reactor fuels. It is also shown that if the experiments were conducted in an unmodified ATR neutron spectrum, gross plutonium self shielding would cause a strong depression of the fission power at the fuel centerline preventing fuel restructuring, a hallmark feature of MOX fuel behavior under fast reactor conditions.

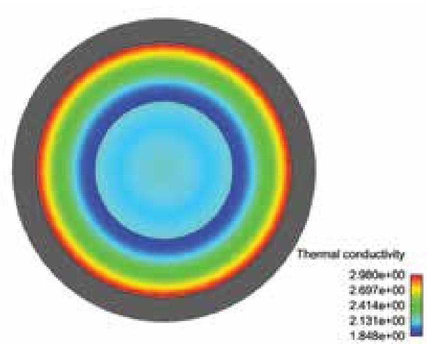


Figure 1. Radial distribution of fuel thermal conductivity

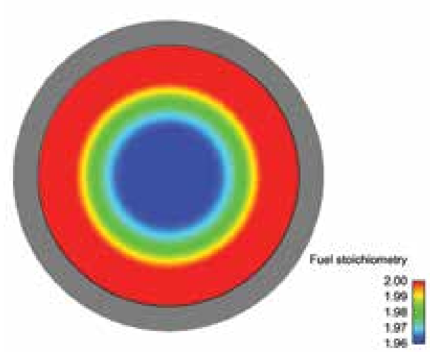


Figure 2. Radial distribution of fuel stoichiometry

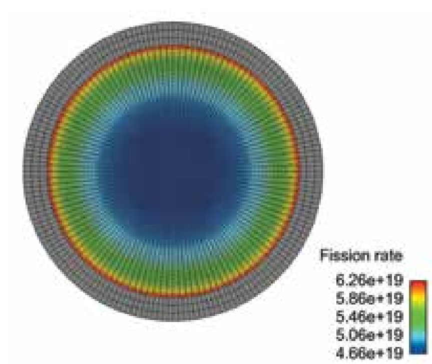


Figure 3. Radial distribution of the fission rate

Accomplishments:

In this study radial power density distributions were utilized to evaluate corresponding radial temperature distributions in the cadmium-shrouded irradiation vehicles used for irradiation testing of transmutation fuels and perform comparisons to irradiations in a fast reactor. Formation of the columnar grain region in fast reactor oxide fuels can be easily quantified during postirradiation examination (PIE), allowing determination of the fuel temperature on the outer boundary of the columnar grain region. Therefore, modeling of the MOX fuel restructuring and comparison of model results to the PIE is an established practice to validate thermal analyses of fast reactor oxide fuel. This is an important step in establishing the relevancy of fuel performance data generated for fast reactor fuels using ATR cadmium-shrouded experiments.

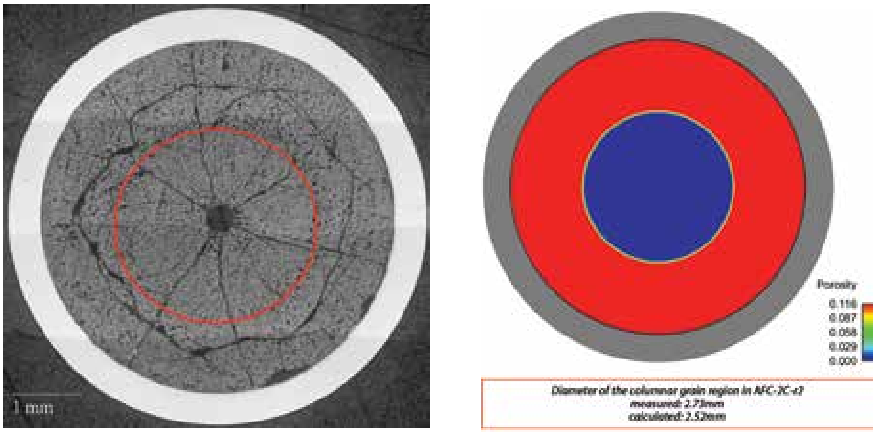


Figure 4. Comparison of measured and calculated dimensions of the columnar grain region in MOX fuel

A BISON capability for the Modeling and Analysis of Advanced Metallic Fuels

Principal Investigator: Cetin Unal and Christopher Matthews
Collaborators: Garrison Stevens, Jack Galloway, Naveen Prakash, Daniel Versino, Ohio State University Collaborators (NEUP): Jinsuo Zhang, Xian Li, Jeremy Isler,

A BISON capability enabling coupled thermal-mechanical-chemical-diffusion analysis of uranium, plutonium and zirconium metallic fuels is necessary and required for the development of advanced fuel concepts, and licensing of advanced reactor concepts. Such a capability provides a tool not only for design and licensing but inexpensively test new simulation-aided concepts prior expensive experimentation, optimize the fuel and reactor performance, and develop adequate safety basis and its requirements.

The goal of this project is to develop a predictive BISON capability enabling coupled thermal-mechanical-chemical-diffusion analysis of uranium, plutonium and zirconium metallic fuels. The fully-coupled capability will model all key aspects of metallic fuel performance under irradiation conditions of advanced reactors, such as species transport, fuel-clad-mechanical interaction including crackling, fuel-clad chemical interactions, and swelling with appropriate phase dependent properties. Advanced calibration methodologies developed by NEAMS program are used to calibrate the capability against available data. Gaps in modeling are identified and solutions are being developed, and experimental needs are continually determined.

Project Description:

The use of metallic fuels in these new nuclear technology concepts has been gaining renewed attention. Metallic fuels are attractive since they have higher thermal conductivity with a highly conductive gap that enables the fuel to operate at lower temperatures with reduced stored energy. Additional benefits of metallic fuels include a more favorable neutron economy, higher fuel densities, and easier fabrication and reprocessing.

Generally speaking, the thermo-mechanical-chemical-diffusion behavior of a nuclear fuel pellet or rod involves a complex system of interdependent processes as a result of the high thermal-power densities and irradiation effects. All of the physics are driven by processes occurring at the microstructure level (i.e., at the grain or subgrain scale). The migration of porosity, fuel constituents, and fission products cause fuel restructuring in metal fuels. Irradiation induced effects such as fission product generation, as well as chemical interactions, change the material properties of the fuel and cladding. Swelled fuel causes mechanical interaction between fuel and cladding that can produce stresses and deformation in addition to the stress caused by internal pressure in the fuel.

Despite the fact that many researchers have developed multiple computer codes, a robust predictive capability for quantifying fuel behavior and constituent distribution in metallic fuels and the associated uncertainty is still unavailable; large uncertainties and scatter still exist in predictions.

Accomplishments:
We have improved BISON capability for metallic fuels significantly this year. The validated specifies distribution kernel for U-Zr fuel is implemented. Short-term solutions to numerical problems encountered in stress aware swelling model are developed. A mechanistic gas release model is implemented. A critical review of fuel-clad-chemical interactions along with modeling requirements is published. The mechanism of lanthanide transport is studied experimentally (NEUP collaboration) and using simulations. A new staged Bayesian calibration approach allowing us to see deficiencies in modeling and developing state aware mitigation strategy is used to calibrate the capability. We apply the current capability to the fuel rod T179 (U-19Pu-10Zr) to demonstrate how an advanced simulation aided metallic fuel can be designed.

Two cases were run to compare with the baseline T179 simulation (1, 2). The "axial shift" case utilizes a power profile that is shifted from the bottom of the core to the top. The second case represents changing the fabrication of the fuel rod to create a linearly varying the concentration of plutonium as a function of axial height.

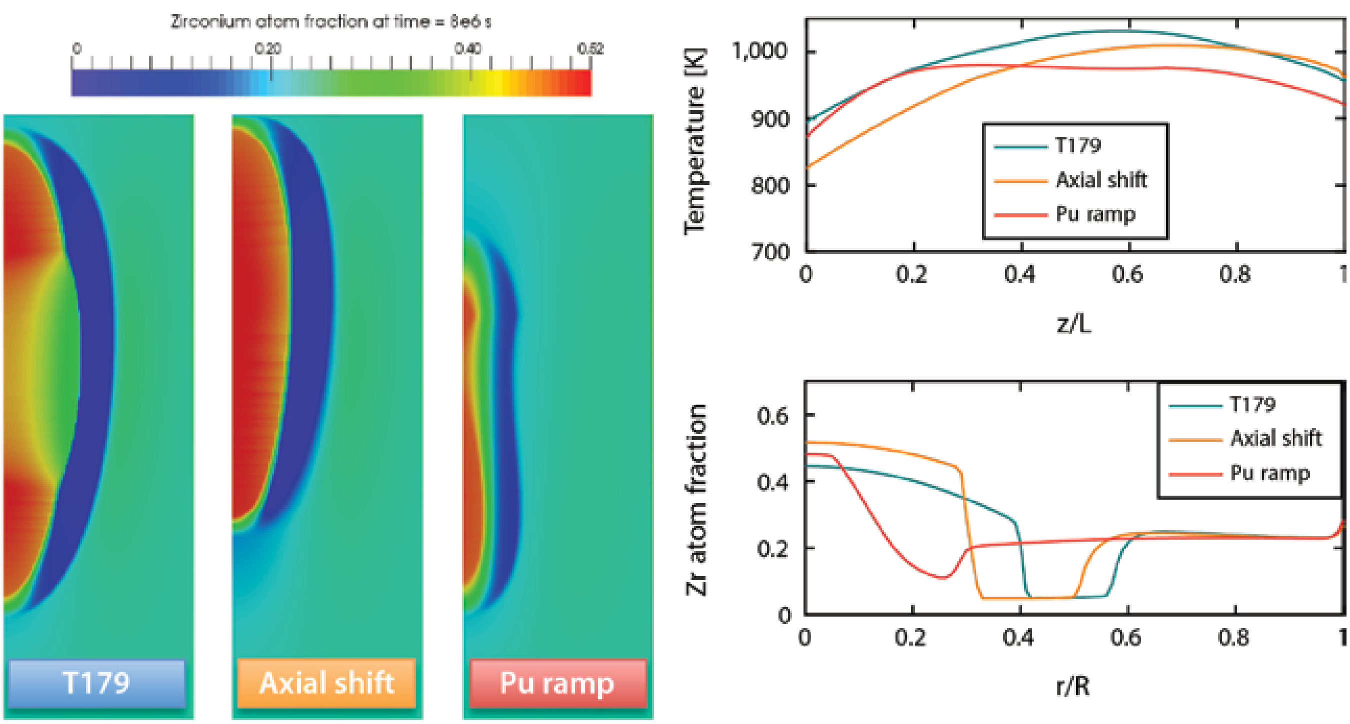


Figure 1. Simulation results at end-of-life showing Zr redistribution in a 2D-RZ slice of the fuel (Left), centerline temperature (Right Top), and centerline Zr atom fraction (Right Bottom).

In general, the behavior of the "axial shift" case performs similar to the baseline T179 case in terms of centerline temperature (Fig. 1). At the end of the irradiation, the location of maximum zirconium redistribution tends to follow the axial power profile.

The plutonium ramp case results in a unique centerline temperature profile due to the change in thermal conductivity as a function of temperature; In general, a decrease in plutonium results in an increase in diffusivity coefficients. As a result, the increasing surface temperature is compensated by the increased thermal conductivity such that temperature profile at the temperature centerline remains flat for nearly two-thirds of the rod. This leads to cooler overall temperatures which in turn slows zirconium redistribution, resulting in less phase disparity throughout the pin.

A critical review of previous work, the physics of FCCI can be separated into individual phenomena so that targeted models can be developed for each (3). Through examination of experiments conducted both in- and out-of-reactor, the behavior of lanthanides provides a natural separation of models by tracking their behavior through (1) production and transport in the fuel to the clad, (2) interaction with macroscopic changes in fuel topography including cracking and swelling, and finally (3) inter-diffusion at the fuel-cladding interface.

Our collaboration with Ohio State University generated valuable data that we used in our initial modeling of the lanthanide transport. The diffusion coefficient of cerium in liquid sodium cerium was calculated to be on the order of 10-5 cm2/s (4) while the diffusivity in cesium was slightly lower.

The experimental results indicated that the solubility of cerium, praseodymium, and neodymium in liquid sodium varied from 1×10-5 to 3×10-5 at.% in the temperature range of 723 to 823 K (5). The time dependence of solubility data showed that the solubility limit was reached within 30 minutes, indicating the dissolution rate is likely high relative to diffusion, thus suggesting lanthanide transport in liquid metals would be diffusion limited. The solubility of lanthanides in liquid cesium varied from 2x10-5 to 9x10-3 at.% in the temperature range of 473 - 723K (5). The dissolution rate was also fast in cesium.

We developed a conceptual pore model to better describe the lanthanide transport behavior through the fuel (7, 8). We apply the conceptual model to a single idealized sodium filled pore configuration as well as idealized sodium filled interconnected pore network. The precipitation kinetics in both the fuel and the pore considers a driving force due to the difference between oversaturated dissolved lanthanide concentration and solubility limit.

As long as oversaturated conditions occur, then precipitation in isolated or interconnected porosity are possible outcomes. The dissolved lanthanides diffuse to the cold side of the pore, exceeding the solubility limit in the liquid, resulting in precipitation on the cold side of the pore. The mechanism at the fuel-pore interface based on the Ln chemical potential causes isolated pores to act like a lanthanide sink as opposed to a pump. Experimental results of high solubility in liquid cesium also support the assumption that liquid cesium filled pores attracts the lanthanides and act like a sink. In the case of sodium filled interconnected pores, lanthanides are able to diffuse to inner clad surfaces and precipitate in the vicinity due to lower solubility of lanthanides in sodium. Initial results show that lanthanide transport from fuel to the cladding can be dramatically accelerated only when the pores are interconnected or there are sodium filled cracks present.

Fig. 2 shows the current results of staged Bayesian calibration of diffusion coefficients along with phase transition temperatures for U-Pu-Zr and U-Zr metallic fuels (9). Comparison of coefficients for three U-Zr and two U-Pu-Zr fuels (see table in Fig. 2) provide insight to the effects of plutonium on the fuel's phase diagram. Findings indicate change in the gamma diffusion region, where the multiplier increases by a factor of four and has a larger uncertainty when no plutonium is present. Transition temperature form beta to gamma phase also appears to be affected, as it increases for U-Pu-Zr. The transition region from delta to beta, however, remains nearly the same meaning the gap between the two regions increases with plutonium content.

Implementation of the statistical calibration approach over the previous manually adjusted coefficients has enabled rapid assessment of several thousand model evaluations that would have been impractical with the previous approach. Inclusion of expert judgement to the calibration process by weighting the error metric has been shown to accurately describe the phase transition temperatures and thermal conductivities. Furthermore, the statistical approach provides a better understanding of standard variations in the model and presents a path forward for reducing model errors, such as missing dependence of porosity changes to the fuel's thermal conductivity model. Progressive calibration completed to date has supported the development of a more advanced, better informed model calibration capability for improving the model's predictive capability. The final calibration of BISON capability with state aware parameters and predictive maturity index including the gas release model is in progress.

□□	48.5	30.6	51.2	27.9
□□	8.52	5.28	8.47	5.76
□□	54.7	29.8	47.5	28.2
□□	55.4	28.6	12.2	7.62
Transitional Temperatures [K]				
A-B	944.3	14.1	943.4	9.86
C-D	986.9	9.25	1030	31.7
Remaining Model Error				
RMS Error	0.078	0.007	0.101	0.012

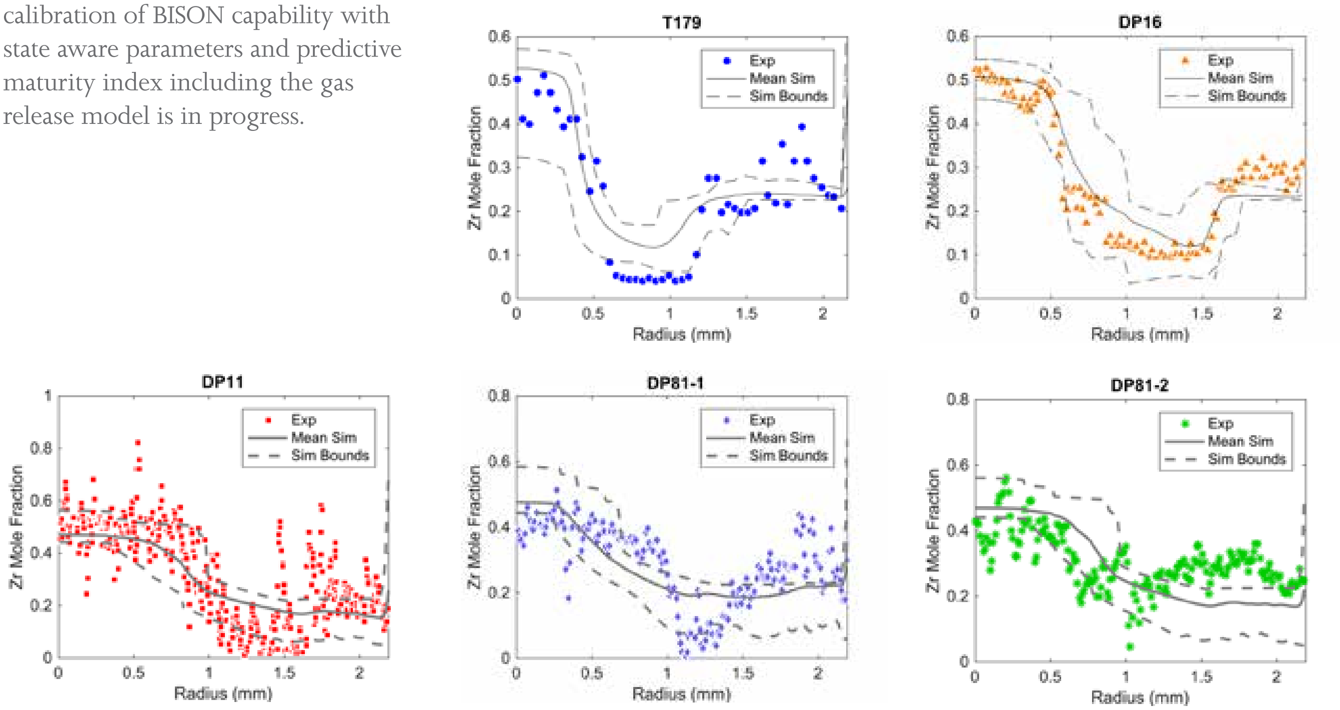


Figure 2. Current results from staged Bayesian calibration approach of diffusion coefficients along with transition temperatures for U-Zr and U-Pu-Zr metallic fuels.

Metallic Fuel Modeling and Simulation Using BISON

Principal Investigator: Pavel Medvedev
Collaborators: Al Casagranda

Metallic Fuel Modeling and Simulation Using BISON is pusued with the following objectives

a. Predict cladding failure

b. Justify and extend peak fuel burnup

c. Justify and extend peak fuel temperature

d. Simulate LOCA and RIA

e. Investigate impact of engineering changes on fuel performance, especially during transients

f. Coatings, liners, annular fuel

g. Justify fast reactor fuel testing in a thermal spectrum with Cd shrouds

h. Design better irradiation experiments

This project is important because it transforms AFC into a goal-oriented science based program

Project Description:

BISON is a finite element-based nuclear fuel performance code applicable to a variety of fuel forms. It solves the fully-coupled equations of thermomechanics and species diffusion, for either 1D spherical, 2D axisymmetric or 3D geometries. Fuel models are included to describe temperature and burnup dependent thermal properties, fission product swelling, densification, thermal and irradiation creep, fracture, and fission gas production and release. Plasticity, irradiation growth, and thermal and irradiation creep models are implemented for clad materials. Models are also available to simulate gap heat transfer, mechanical contact, and the evolution of the gap/plenum pressure with plenum volume, gas temperature, and fission gas addition. BISON has been coupled to the mesoscale fuel performance code MARMOT, demonstrating fully-coupled multiscale fuel performance capability. BISON is based on the MOOSE framework and can therefore efficiently solve problems using standard workstations or very large high-performance computers.

Accomplishments:

The primary technical goal for the BISON team related to metal fuel modeling was to evaluate and provide a preliminary capability for anisotropic swelling in order to simulate the EBR2 X447 experiment. A literature search was conducted to ascertain the prior work on anisotropic swelling behavior in metal fuels. A number of experimental observations on the extent of swelling anisotropy in both UZr and UPuZr fuels are well established in the literature. However, an analytical model to account for anisotropic swelling in metal fuel was not found. An empirical model implemented in the ALFUS code based on a small set of experimental results is the current state-of-the-art. This model has been implemented in BISON using the 1.5D capability in tensor mechanics. A 1.5D version of the X447 model is being used to evaluate the effects of anisotropic swelling on fuel and cladding deformation.

Since the BISON code is transitioning the approach for computing mechanical behavior to provide a more general framework for future developments (such as anisotropic materials, cracking, 1.5D, etc.), the metal fuel specific material models were modified to make use of these new features. These included UPuZr fuel (elastic, thermal, creep and swelling), HT9 (elastic, thermal and irradiation creep) and SS316 (elastic, thermal and irradiation creep) cladding materials. In addition, a new model for Na infiltration of UPuZr fuel was added to account for the changing conductivity once the fuel swelling reaches a threshold and the Na fills in the porosity.

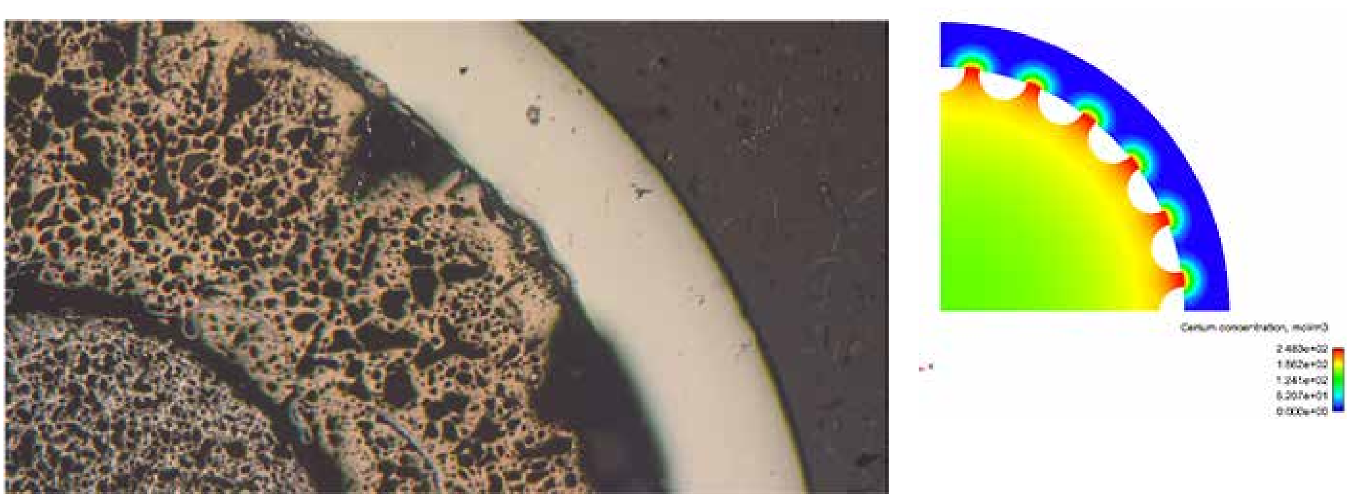


Figure 1. Comparison of the calculated cerium concentration in the fuel and cladding calculated using BISON with a PIE image showing evidence of the FCCI

Assessment of the Impacts of Annular Fuel on Reactor Performance

Principal Investigator: T. K. Kim
Collaborators: G. Aliberti, N. Stauff

The transmutation fuel package has supplied fuel samples for thermal and microstructural characterization as well as the fuel and assembly of the AFC-3F irradiation test.

An annular metallic fuel is one the advanced fuel concepts that are under development by the Advanced Fuel Campaign. The primary motivation of the annular fuel is to reduce fuel smeared density (< 75%) for accommodating swelling of ultra-high burnup fuel without major fuel-cladding mechanical integration. The annular fuel in contact with cladding could eliminate the need for sodium bond, which provides various impacts on reactor performance characteristics as depicted in Figure 1.

Project Description:

The primary objective of this study is to assess the impacts of annular fuel on reactor performance characteristics. As long as fuel composition and fuel smeared density are retained, neutronics performance (i.e., flux level, burnup, power distributions, etc.) of the annular fuel are comparable with those with a solid fuel. However, absence of bond-sodium in the annular fuel allows relocation of fission gas plenum below active fuel rather than above, which is typical location in a solid fuel. Since the gas plenum is located at low temperature position, the annular fueled SFR has different reactor performance characteristics.

For instance, internal gas pressure, Cumulative Damage Factor (CDF), irradiation damage (DPA) of load-bearing internal structure could be reduced because of lower temperature and further distance from the active fuel. These impacts are all positive to increase fuel residence time (i.e., burnup) and reactor lifetime. However, as tradeoffs, additional shielding is needed to protect upper internal structures and reduce activation of secondary sodium in intermediate heat exchanger. In addition, extra transient tests are needed because the annular fuels have different failed fuel symptoms and melting behaviors during transient scenarios.

Accomplishments:

Impacts of the annular fuel on internal gas pressure, cumulative damage factor (CDF), and irradiation damage of load-bearing grid plate were evaluated using the 1000 MWth Advanced Burner Reactor (ABR-1000). In this study, fuels were replaced with the annular fuels and the gas plenum pf each fuel pin was located below the active fuel. However, other design parameters such as pin diameter, fuel composition and smeared density, discharge burnup, etc., were retained. The relocation of gas plenum below active fuel reduced fission gas temperature by 155K, which reduced hoop stress by 33% and increased time to cladding-rupture (i.e., CDF=1) by a factor of 12.4. Thus, the annular fuel can reside in the core longer than solid fuel, which is good for ultra-high burnup fuel. It is noted that the irradiation damage (DPA) of load-bearing internal structures such as a grid plate is one of the factors to determine reactor lifetime. In the annular fueled SFR, the grid plate could be further away from active fuel because of a lower gas plenum, which is good to reduce the irradiation damage. In order to compare the irradiation damage of the grid plate, Monte Carlo (MCNP) models were developed using full ABR-1000 assemblies using annular and solid fuels. The resulting DPA with annular fuel is about ~5% smaller than that with the solid fuel.

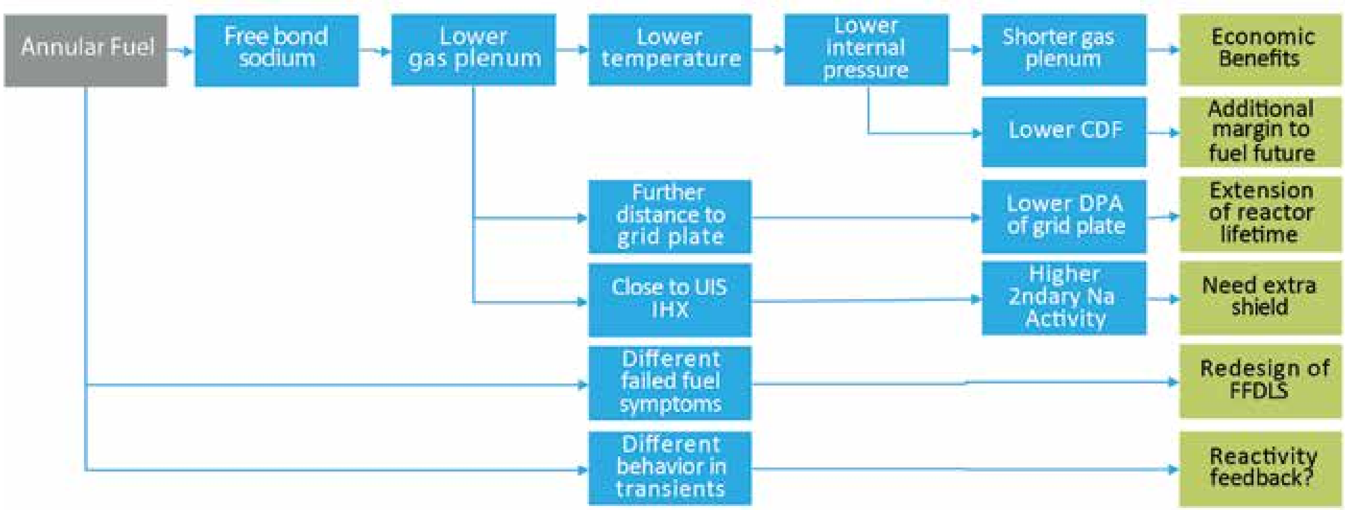


Figure 1. Schematic overview of the Impacts of Annular Fuel on Reactor Performance

3.3 AR CORE MATERIALS

ODS Tube and Rod Process Development

Principal Investigator: Stuart Maloy
Collaborators: Eda Aydogan (LANL), Dave Hoelzer/ORNL, Curt Lavender/PNNL, Bob Odette/UCSB, John Lewandowski/CWRU

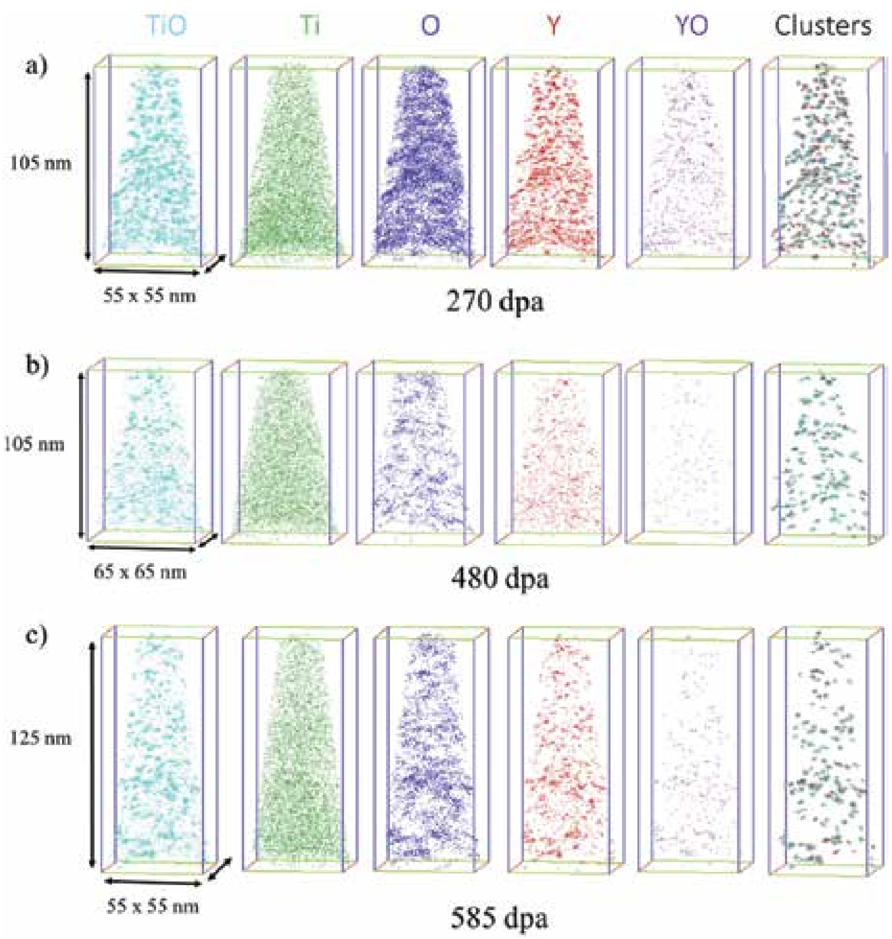


Figure 1. Analysis of oxide particles in 14YWT using Atom Probe Tomography after irradiation with Fe ions to extreme doses show excellent stability even to the highest dose of 585 dpa. (Aydogan, E., Almirall, N., Odette, G.R., Maloy, S.A., Anderoglu, O., Shao, L., Gigax, J.G., Price, L., Chen, D., Chen, T., Garner, F.A., Wu, Y., Wells, P., Lewandowski, J.J., Hoelzer, D.T)

The Fuel Cycle R&D program is investigating options to transmute minor actinides. To achieve this goal, new fuels and cladding materials must be developed and tested to high burnup levels (e.g. >20%) requiring cladding to withstand very high doses (greater than 200 dpa) while in contact with the coolant and the fuel.

Project Description:

To develop and qualify materials to a total fluence greater than 200 dpa requires development and testing of advanced alloys and irradiations in fast reactors. Specimens of previously irradiated HT9 specimens are being irradiated in a fast reactor to high doses (>200 dpa). In addition, improvements in the radiation tolerance of HT-9 are being made through minor changes in the composition. Advanced radiation tolerant materials with fine oxide dispersion strengthening are also being developed to enable the desired extreme fuel burnup levels. This fine microstructure provides an alloy with high strength at high temperatures and excellent radiation tolerance (e.g. reduced void swelling and ductility retention at low temperatures) but also increases the difficulty of producing engineering parts (e.g. thin walled tubes) from these advanced materials. Thus, in this project, research is underway to produce tubes using techniques such as high temperature hydrostatic extrusion, intermediate temperature plug drawing and low temperature pilger processing.

Accomplishments:

Through a collaborative effort between LANL, UCSB and ORNL a large heat (50 kg) of a nanostructured ferritic alloy (14YWT) was produced named FCRD-NFA1. Significant progress has been made this year at characterizing the microstructure of the consolidated materials and performing mechanical testing including tensile testing and fracture toughness testing. Characterization was performed on plates and hydrostatically extruded tubes. Plates showed microcracks produced from an in plane [001] texture produced during deformation but hydrostatically extruded tubes did not show any microcracks. The tubes produced from this heat of material using hydrostatic extrusion were made at Case Western Reserve University. High dose ion irradiations were performed on the hydrostatically extruded tubes and TEM and Atom Probe Tomography after irradiation showed that the particles were stable up to doses of 585 dpa (see Figure 1). Future work is aimed at producing tubing using pilger processing at PNNL/Sandvik and determining weld procedures at ORNL.

This research is critical to the application of advanced ODS steels to engineering applications as it addresses one of the most difficult tasks which is to produce tubing from these radiation tolerant, high strength steels and the stability of these steels at very high doses.

Complete Status Report Documenting Development of Friction Stir Welding for Joining Thin Wall Tubing of ODS Alloys

Principal Investigator: D.T. Hoelzer
Collaborators: M.N. Gussev, J.R. Bunn / ORNL

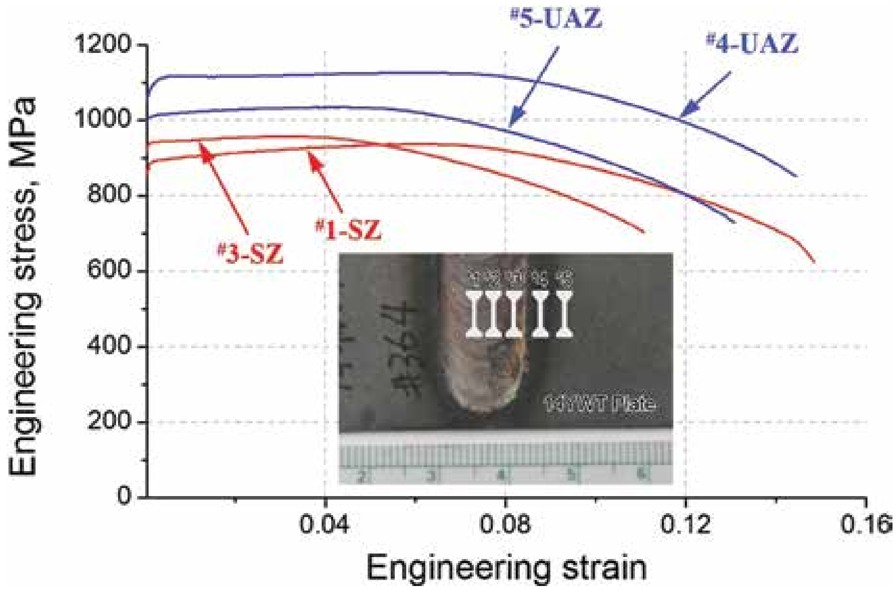


Figure 1. Engineering stress-strain curves of the SS-Mini specimens showing their location relative to the stir zone in the 1 mm thick plate of 14YWT.

Advanced oxide dispersion strengthened (ODS) ferritic alloys, such as 14YWT, possess excellent high-temperature mechanical properties that are resistant to radiation damage at high dose and temperature exposures, but are considered impossible to weld. Fusion welding relies on forming a melt pool for joining materials, which causes significant coarsening and agglomeration of the nano-oxide dispersion and destroys the ultra-small grain structure of ODS alloys. Friction stir welding (FSW) is a solid-state joining technique that has recently been investigated for welding ODS alloys since it relies on frictional heat generated by a moving pin tool rotating at high speeds and the ODS material for joining. For this reason, the NTRD program began exploring FSW for joining thin plates and ultimately thin wall tubing of 14YWT. This project focuses on developing FSW conditions that avoid defects such as porosity, worm-holes, grain growth and tensile residual stresses that can degrade the joint quality.

Project Description:

The advanced ODS 14YWT ferritic alloy is being investigated for thin wall fuel cladding in fast nuclear reactor technologies. The goals of this program are to develop fabrication methods to produce thin wall tubing and welding methods to join thin wall tube sections of 14YWT. The latter goal on joining thin wall tubes of 14YWT presents a major challenge due to issues with joining ODS alloys and limited experience with joining ODS alloys by FSW. The main task involves development and optimization of FSW process parameters to ensure weld joints that are defect free and do not degrade the salient microstructural features of 14YWT responsible for the excellent high-temperature mechanical properties and radiation resistance. The objective of this research is the successful joining of plate and tubing of 14YWT with thicknesses less than 1 mm. The knowledge base obtained for joining thin plate and tubing of ODS alloys will be novel since few publications exist on joining ODS alloys by FSW and all these studies used thick plate geometries. This research includes tasks on characterizing the microstructure and assessing the mechanical properties of specimens prepared from the stir zone produced by FSW in a 1 mm thick plate of 14WT.

Since the microstructure produced in the stir zone is very complex, a novel approach based on optical extensometry is being explored for studying the local deformation along the gage of miniature tensile specimens from the stir zone during tensile testing. A recent task on measuring residual stresses in hot rolled plates and FSW joints of 14YWT was initiated. The results of this research support the main goal of the fuel cycle R&D program on developing fabrication methods for producing thin wall fuel cladding from advanced ODS alloys and together will help maximize nuclear fuel utilization of next generation nuclear reactors.

Accomplishments:

The research goal was to conduct detailed microstructural and mechanical properties characterization studies of the bead-on-plate stir zone produced by friction stir welding (FSW) in a 1 mm thick plate of 14YWT for developing FSW process parameters that produce high quality welds.

The microstructural analysis determined that the FSW conditions for producing the bead-on-plate stir zone (SZ) on the 1 mm 14YWT plate caused slight coarsening of the grain size and the Ti-, Y- and O-enriched oxide particles and formation of several horizontal cracks. The slight coarsening of the oxide particles was

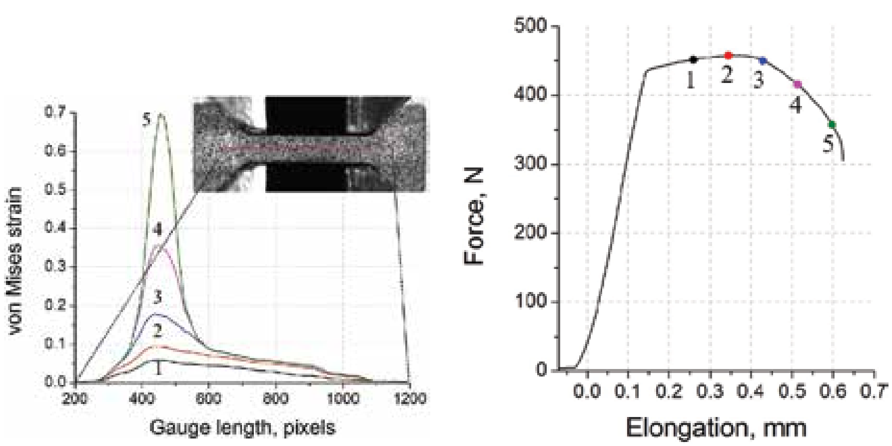


Figure 2. Specimen #1 from stir zone. (a) Strain distribution (von Mises strain, Green-Lagrange strain tensor) along the specimen gauge at different time (1-5). (b) Experimental tensile curve with force and elongation values for every image. Insert shows superimposed line of DIC analysis on specimen gauge.

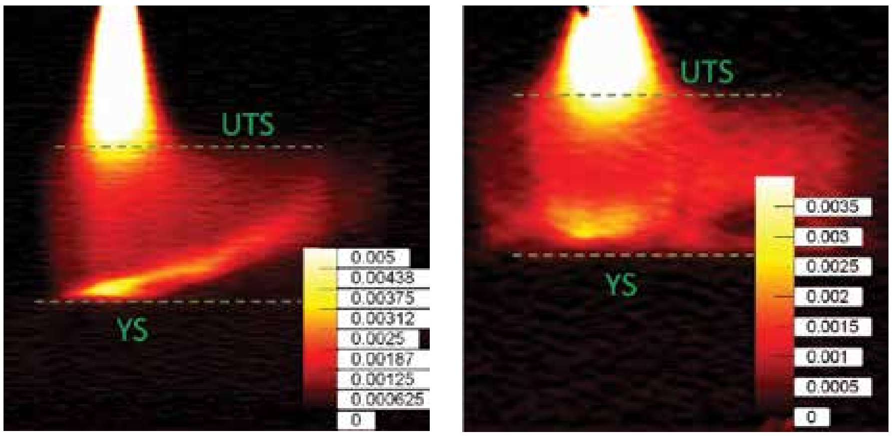


Figure 3. Strain rate maps of (a) specimen #1 from the stir zone and (b) specimen #4 from the unaffected zone.

not consistent with bulk diffusion mechanisms of constituent oxide elements, but enhanced diffusion rates due to the high density of dislocations formed during FSW could account for their coarsening during the short-time that the spinning pin tool advanced through the thin plate. The horizontal cracks observed on the advancing side of the SZ may have formed because of residual stresses caused by FSW and their propagation direction was consistent with the {100}<110> delamination fracture mechanism.

The tensile properties and deformation behavior of specimens fabricated from the unaffected zone (UAZ) and SZ of the thin 14YWT plates were studied using digital image correlation (DIC). With this approach, the surface of a miniature tensile specimen (SS-Mini) is painted with speckles of black and white paints for measuring localized strains along the gage during tensile testing and acquiring digital images as a function of time. Figure 1 shows stress-strain curves of four SS-Mini specimens and their location relative to the stir zone. Both the tensile properties and stress-strain curves were slightly affected by FSW. The room temperature yield and ultimate tensile stresses of SZ specimens were ~13% lower than that of the UAZ specimens, but the uniform and total elongations were unaffected by FSW. The distribution of strain observed along the four SS-Mini specimens is illustrated in Figure 2 for specimen #1 from the SZ. The distributions of strain along the specimen gage for 5 points (Fig. 2a) observed in the force-elongation curve (Fig. 2b). The strain distribution is homogeneous after yielding, but becomes localized in the gage during plastic deformation leading to necking until failure occurs. Strain rate maps for specimens #1 (SZ) and #4 (UAZ) indicated localized high strain rates formed and propagated along the gage during the tensile test before dissipating, which was consistent with deformation bands. The strain rates increased in localized regions of the gage after reaching the ultimate tensile stress. The localized strain rate intensity increased rapidly as necking proceeded until the specimen failed.

Residual stress experiments were conducted on 4 hot-rolled 14YWT plates having different deformation magnitudes. Data was collected from neutron scattering scans in 3 orthogonal directions on each plate to obtain residual stress data in the longitudinal (extrusion), traverse (rolling) and thickness directions. Preliminary results indicated that residual stresses increased with deformation magnitude and rolling direction may affect the residual stresses. The residual stresses in the SZ caused by FSW will be determined in the future using the data from these experiments.

Friction stir welding successfully produced a bead-on-plate stir zone in a 1 mm-thick plate of the advanced ODS 14YWT alloy that has been well characterized, which is promising for joining thin wall fuel tubing of advanced ODS alloys for next generation nuclear reactor technologies.

Mechanical Properties of Original and N-doped HT9 Steels after Modified Thermomechanical Processing

Principal Investigator: T.S. Byun
Collaborators: J.P. Choi, K. Mattlin/ PNNL, S.A. Maloy/LANL

The 12Cr-1MoVW (HT9) steels with tempered-martensitic structures have been used as core structure materials for the sodium-cooled fast reactors as those exhibit high resistance to irradiation damage and good compatibility with liquid sodium coolant. Although the core components of typical fast reactors were designed for use in the 350–550°C temperature range, ongoing development efforts for higher performance reactors require significantly improved capabilities of core materials under high dose (i.e., high burnup) and high temperature conditions. In particular, therefore, the core materials for future fast reactors have to demonstrate excellent creep and swelling resistance in high temperature (> 500°C) and long term irradiation as well as to retain good mechanical performance to avoid embrittlement induced by low temperature (< ~430°C) neutron irradiation. In this research, new processing routes and nitrogen doping are applied to improve the high temperature performance of HT9 steels.

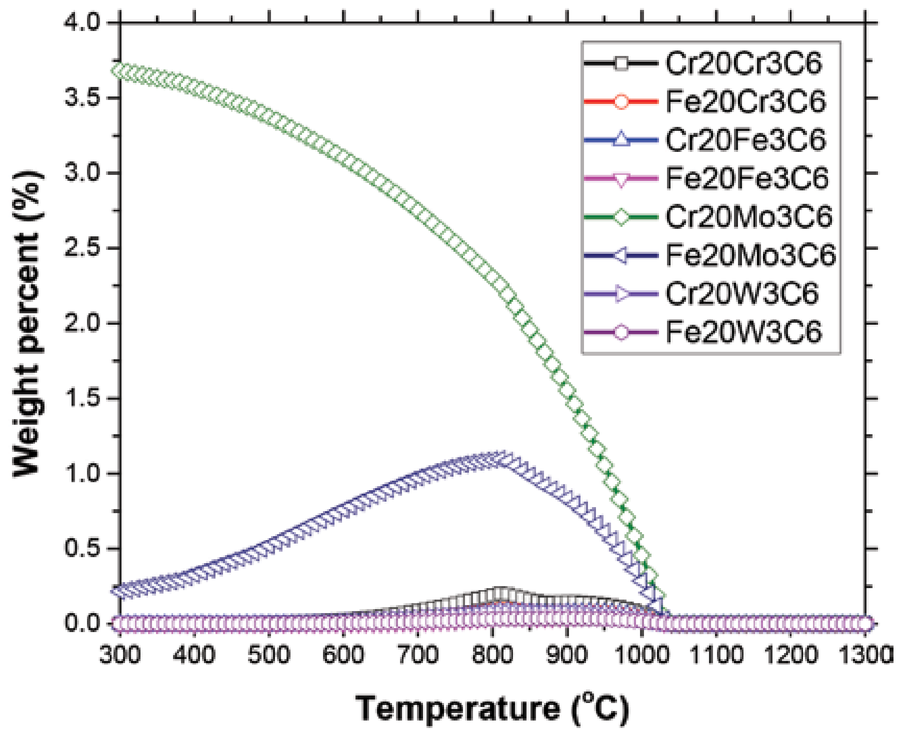
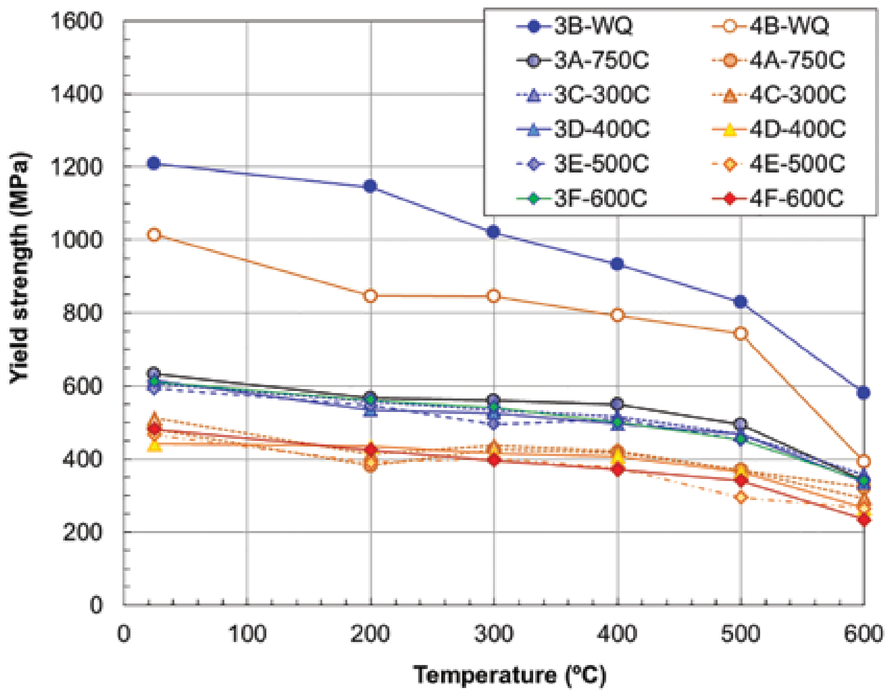


Figure 1. Temperature dependence in formation of M23C6 carbides in the N-doped HT9 steel.

Project Description:
This work particularly aims to develop an improved thermomechanical processing (TMP) route for HT9 steels as they are currently the most probable candidates for the core materials of advanced fast reactors. These structural materials will have better resistance to radiation-induced damage if controllable microstructural features and their size scales can directly involve radiation damage processes such as formation of nanometer scale defect clusters, voids, and helium bubbles, creep deformation, elemental segregation, and phase changes. A principle used in this process improvement is the largely accepted idea that more defect recombination sites or higher defect sink density is required for higher radiation resistance. Therefore, refinement of lath structure and precipitates is the key method in the processing development of HT9 steels. In this fiscal year, new TMP routes were explored using two HT9 steels (one with and one without nitrogen addition) provided by the LANL team led by Stuart A. Maloy. The research activities specifically focused on testing a series of new processing routes to determine the microstructures of HT9 steels that lead to improved mechanical properties including high-temperature fracture toughness. First, a comprehensive thermodynamics simulation using Thermo-Calc Software was performed for the two HT9 steels to obtain key metallurgical information on equilibrium phase formation and elemental distributions within phases. Second, the first round of new heat-treatment routes, including both the typical normalization plus tempering and the normalization plus newly designed two-step tempering, have been applied to the two steels; the treated materials were then evaluated by tensile property testing over a wide temperature range of room temperature to 600°C.

Significant improvement of HT9 steel performance via an innovative thermomechanical treatment will help achieve much higher burnup in fast reactor fuels.

Figure 2. Temperature dependence of yield strength (YS) in HT9 steels with and without nitrogen doping (4 and 3 series, respectively, in the legend) after pre-tempering at 300–600°C and final tempering (750°C) treatments, which are compared with the YS data after water quenching (WQ) and 750°C tempering only.



Accomplishments:

(1) Completion of comprehensive thermodynamics simulation on the formation and composition of equilibrium phases in the proposed thermomechanical processing (TMP) conditions is the first main achievement for the processing development scope in FY 2017. The calculated phase diagrams for HT9 alloys show that the $M_{23}C_6$ carbide, Sigma, Laves, and body centered cubic (BCC) phases form in the low-temperature region at < ~800°C. Fe and Cr are the major elements within face centered cubic (FCC) phases above the upper transformation temperature (A3) (~880°C). Small amounts of FCC phases remain at lower temperatures. The temperature-dependent compositions within individual phases also comprise key reference data for the processing development. Figure 1, for example, displays the temperature dependent stabilities of various $M_{23}C_6$ carbides, showing that the most stable or

major phase is Cr20Mo3C6. The amount of $Cr_{20}Mo_3C_6$ decreases with temperature, and indeed different temperature dependencies are observed for the different $M_{23}C_6$ compounds. The origin of the significantly different mechanical properties depending on the addition of nitrogen was explained by thermodynamic calculation results, which indicate that the N-doped and N-free HT9 alloys have almost the same precipitation behaviors except for nitride precipitation and pure element solution. In the HT9 with higher N content, the FCC phases are predicted to exist below A3 line, which consist of pure Cr, Ni, and CrN. The amount of CrN phase decreases with decreasing temperature, while the Cr and Ni contents in FCC increase with decreasing temperature. These residual austenite phases should be the cause for the relatively lower strength and higher ductility found in the N-doped HT9 steel.

(2) The second main achievement was the completion of mechanical tensile testing for the two HT9 steels after taking various TMP routes guided by the thermodynamics calculation. The two HT9 steel plates, heat-3 (standard) and heat-4 (N-doped), were received from LANL and rolled to 1.6 mm thick strips. Coupons were cut from the strips and heat-treated to produce as-quenched, single-tempered, and double-tempered conditions. The first treatment in the two-step tempering was made at four different temperatures (300, 400, 500, and 600°C) and followed by a 30 min long tempering at 750°C. As seen in Figure 2, the yield strength (YS) of the two HT9 steels monotonically decreases with test temperature, regardless of different alloy compositions and TMP routes. Over the whole test temperature range the high-N HT9 steel heat-4 (0.044 wt.% N) has noticeably lower strength than the low-N steel heat-3. Compared to the monotonic temperature dependence in strength, the ductility parameters show more complex behavior. Uniform ductility almost linearly decreased with test temperature, except for a few cases such as the as-quenched steels. The total elongations at room temperature and 600°C for both HT9 steels in all tempered conditions are in the ranges of 12–20% and 16–20%, respectively. In all tempering conditions a ductility minimum was measured at about 400°C, which is believed to be caused by dynamic strain aging (DSA).

Advanced Reactor Cladding Tube Pilgering Process Development (PNNL)

Principal Investigator: Curt Lavender and Ron Omberg
Collaborators: Wendy Bennett and Marie McCoy at PNNL and Bob Johnson at Sandvik Specialty Metals

Cladding developed with these processes will produce: (1) a more radiation tolerant fuel cladding on high strength materials that cannot be processed by conventional means, (2) a fuel that can be used to a reduce of the half-life of isotopes in spent fuel that will need to be disposed of in the Yucca Mountain Repository, and (3) a fuel that can be used to improve the economics of Sodium Fast Reactors.

Radiation effects at displacement doses greater than 400 displacements per atom (dpa) in a reactor environment create a challenging environment to materials used for cladding and ducts. The activity is investigating processes for fabricating promising radiation tolerant materials for cladding, specifically using extrusion and pilgering processes in a sequential mode.

Project Description

Extrusion and pilgering are promising techniques for fabricating advanced alloys such as MA956 and 14YWT suitable for producing tubing for reactor use. This project is using the extrusion press located at the Pacific Northwest National Laboratory (PNNL) to perform sequential extrusions followed by pilgering using the pilger mill located nearby as Sandvik Special Metals. The goal and objective is to produce tubing samples with dimensions for in-reactor use as specified by Los Alamos National Laboratory (LANL).

With respect to the next generation of reactors, success in this activity will produce: (1) a more radiation tolerant fuel cladding, (2) a fuel that can be used to reduce the half-life of isotopes in the spent fuel that will need to be disposed of in Yucca Mountain, and (3) a fuel that will improve the economics of Sodium Fast Reactors.

Accomplishments

Accomplishments this fiscal year include extruding annular thick wall annular tubes with both MA956 and 14YYWT. The MA956 thick wall tube extrusions were then pilgered at the pilger mill located at nearby Sandvik and are shown in Figure (1) below. Two 14YWT thick wall annular tubes were extruded in the extrusion press at PNNL and are staged for subsequent pilgering at Sandvik. These are shown in Figure (2) below.

Figure 1. MA956 Tubing after Pilgering

Figure 2. 14YWT Thick-Wall Annular Tubing Extrusions Staged for Pilgering

Advances in the Effective Use of Ion Irradiations for Core Materials Research

Principal Investigator: Mychailo Toloczko, PNNL
Collaborators: Jing Wang, PNNL

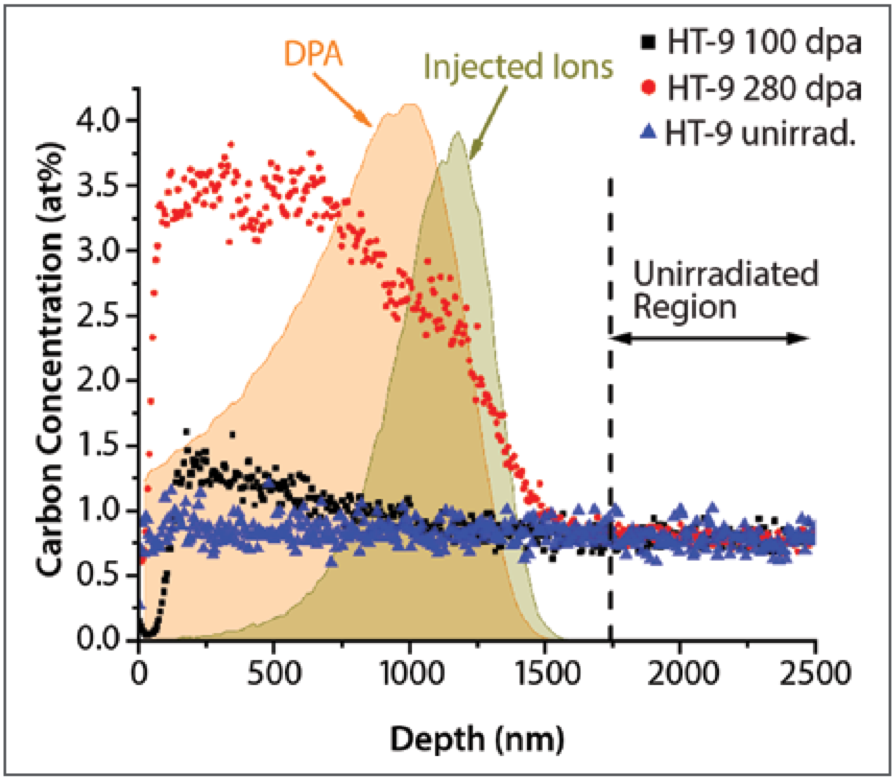


Figure 1. Depth profile of carbon concentration in unirradiated HT-9, HT-9 irradiated to 100, and to 280 dpa at 400°C.

Accelerator-based ion beam irradiation techniques have been used to study radiation effects in materials for decades. The popularity of this technique was initially quite high due to the ease and speed of performing irradiations. However, popularity eventually waned as many people cast doubt on its effectiveness for assessing neutron irradiation effects. However, in recent years as access to neutron irradiation facilities capable of providing high dose has greatly diminished, and recent advanced reactor concepts have called for very high burnup fuels, a need for ion irradiations has grown. To make best use of this technique, the AFC is carefully examining and altering ion irradiation techniques to improve their ability to provide useful irradiation effects information. Presented here are two recent insights into the ion irradiation method and consequent changes to the method.

Project Description:

The work described here supports the AFC advanced reactor core materials campaign that seeks to design materials for clad and duct that can survive to very high burnup. This work requires assessing the effects of irradiation on the mechanical properties and dimensional stability of candidate materials to determine their value. In the last five years, the AFC has started using heavy ion irradiations to assess material performance out to very high dose (while continuing to use neutron irradiations to low doses). The effective use of ion irradiations for materials studies will greatly speed up the research cycle, thus adding substantial benefit. Thus, while ion irradiations are already actively being used, a parallel activity exists to improve its ability to provide information on neutron irradiation performance.

Accomplishments:

Two areas of advancement this year in the understanding of ion irradiation processes are on the effect of sputtering and the effect of carbon contamination. Heavy ion irradiations with beam energies in the 1-5 MeV range that are commonly used for structural materials studies produce irradiation damage only to a depth of ~1 μm. While sputtering is known to occur during heavy ion irradiations, it was found that its effect on the size of the irradiated region and the distribution of damage within the irradiated region has been completely overlooked. It was found that for beam energies below 1 MeV, very large amounts of sputtering occur. This causes a convolution of the damage accumulation and injected atoms as a function of depth, thereby altering the irradiation damage profile and injected atom distributions in an undesirable

The AFC is not only making extensive use of ion irradiations to meet research goals but is actively working to understand and improve the method to obtain the best possible radiation effects information on candidate advanced reactor clad and duct materials.

manner and to a large extent. Our work found that in order to avoid this issue for high dose ion irradiations, the beam energy should be at least 1.8 MeV. Ion irradiation programs within AFC have always been using at least 1.8 MeV energy beams.

The second issue that was found and assessed was a strong propensity for materials to become contaminated with carbon during ion irradiations. This is a very important issue in the AFC because all the core materials of interest are ferritic alloys whose mechanical properties and microstructure are strongly affected by small changes in carbon content. As with the sputtering issue, it was found that the structural materials ion irradiation community was largely unaware of this issue. One of the accelerators used by the AFC was found to not produce carbon contamination while another was found to have intermittent issues. Analysis of the accelerator revealed that for what is considered a clean beamline, there were still hydrocarbons in the vacuum chamber, and these were getting pulled down the beamline to the specimen where the ion beam would then break down the hydrocarbon and release carbon. This carbon is then either ballistically injected into the specimen or enters through radiation enhanced diffusion processes. The amount of carbon contamination can greatly exceed the amount of carbon that is meant to be in the material causing substantial changes in the material properties and radiation response. An example of the level of carbon contamination that occurred in tempered martensitic steel specimens irradiated to low or moderate dose levels is presented in Figure MBT1. The addition of these high amounts of carbon causes the formation of precipitates in the material (Figure MBT2) that wouldn't normally occur and will affect the properties of the material and how it responds to further irradiation. Several means to mitigate the possibility of carbon contamination have been devised. Some accelerator operators are fitting additional equipment that prevent the hydrocarbons from reaching the surface. Within the AFC, coatings are being developed that can be applied to specimens to prevent carbon ingress.

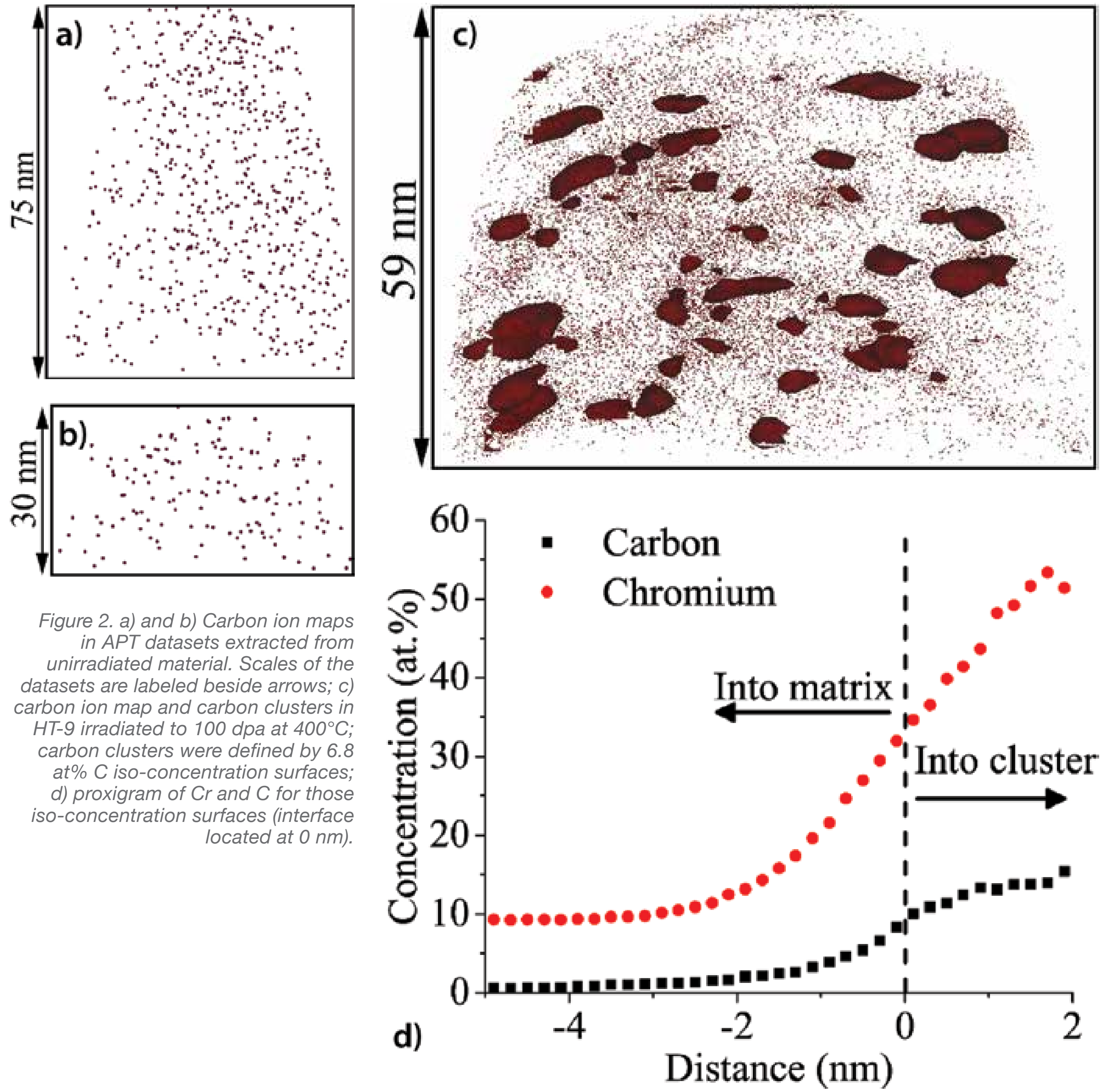


Figure 2. a) and b) Carbon ion maps in APT datasets extracted from unirradiated material. Scales of the datasets are labeled beside arrows; c) carbon ion map and carbon clusters in HT-9 irradiated to 100 dpa at 400°C; carbon clusters were defined by 6.8 at% C iso-concentration surfaces; d) proxigram of Cr and C for those iso-concentration surfaces (interface located at 0 nm).

Mechanical Properties of HFIR Irradiated FeCrAl Alloys

Principal Investigators: Dr. Tarik A. Saleh, Los Alamos National Laboratory
Collaborators: Dr. Kevin G. Field, Oak Ridge National Laboratory, Mr. Matthew E. Quintana, Los Alamos National Laboratory, Mr. Tobias J. Romero, Los Alamos National Laboratory

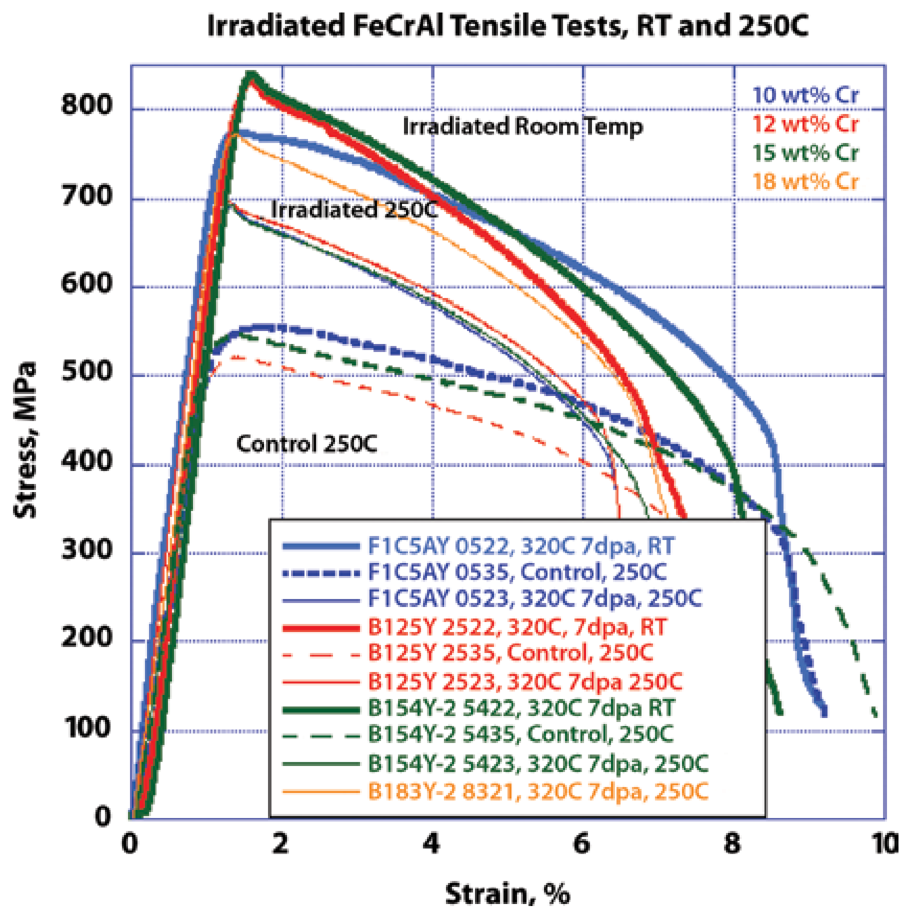


Figure 1. All Tensile Tests for HFIR Irradiated GenI FeCrAl Alloys at Room Temperature and 250C.

collected on neutron irradiated FeCrAl alloys and provided control tensile data from GenII FeCrAl alloys to use to compare to future irradiations.

Due to their excellent performance in high temperature oxidizing steam environments, Iron-Chromium-Aluminum (FeCrAl) alloys are a strong candidate material for Accident Tolerant Fuel (ATF) cladding applications. A number of these alloys, GenI FeCrAl alloys with varying Cr (10-18 wt. %) and Al (3-5 wt %) content, irradiated to 7dpa at 320C in the High Flux Isotope Reactor (HFIR) were tested in tension at room temperature and 250C. A second group of GenII FeCrAl alloy controls, with 6 wt%Al, and 10 to 13 wt % Cr, were also tested in tension to use as baseline tensile data for ongoing irradiation experiments in HFIR.

Project Description:

Due to the Fukishima accident of 2011 interest in the accident tolerant fuels and cladding materials have increased. Research into claddings that are resistant to the high temperature steam oxidation conditions seen in loss of coolant accidents (LOCA) have led studies of iron-Chromium-Alumium (FeCrAl) alloys, which form a protective aluminum oxide layer under these conditions. Irradiations of these alloys are underway in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). Alloys of various Cr content are being studied to understand the effects of irradiation produced Cr rich alpha' phase on the mechanical properties of these alloys. This report presents mechanical testing on four first generation (GenI) FeCrAl alloys of varying Cr (10-18 wt%) and Al (3-5 wt%) content that were irradiated to 320C and 7 dpa in the HFIR reactor, as well control samples of two second generation (GenII) FeCrAl alloys of varying Cr content (10 and 13 wt%) that will be used for comparison to samples that are currently being irradiated in the HFIR reactor.

Accomplishments:

Ship irradiated samples from ORNL hot cells to complete these tensile experiments and provide samples for futureirradiated material mechanical testing (shear punch and high temperature tensile tests) (Quintana, Saleh, LANL, Field ORNL)

Uniaxial tension of HFIR irradiated FeCrAl alloys at additional temperatures and conditions to compliment and expand exisiting work. (Saleh, Romero, LANL)

Uniaxial tension of GenII FeCrAl control specimens to create a database of unirradiated data to compare to ongoingirradiations in HFIR. (Saleh, Romero, LANL)

Microstructural analysis of HFIR irradiated FeCrAl Alloys (Field, ORNL)

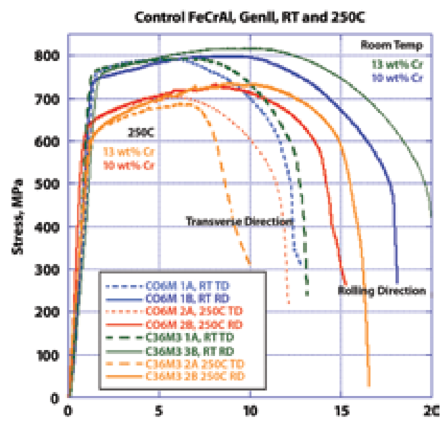


Figure 2. Room and elevated temperature (250C) tensile tests of control GenII FeCrAl alloys.

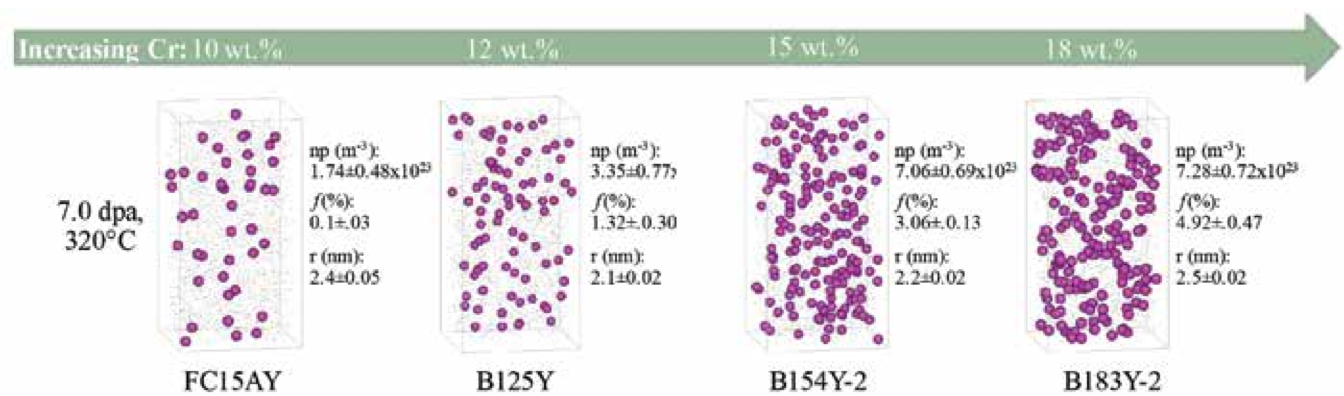


Figure 3. Synthetic Microstructures of alpha' ingrowth in GenI FeCrAl alloys irradiated at 320C and 7dpa, generated from SANS data.

3.4 AR Irradiation Testing & PIE Techniques

Progress and Status of Irradiations in the AFC Series in ATR

Principal Investigator: Steven Hayes/Gary Povirk
Collaborators: Doug Dempsey

Transmutation of the long-lived transuranic actinide isotopes contained in spent nuclear fuel into shorter-lived fission products would dramatically decrease the volume of material requiring disposal and the longer-term radiotoxicity and heat load of high-level waste sent to a geologic repository.

The Idaho National Laboratory (INL), through the Department of Energy (DOE) Idaho Operations Office, has been assigned the responsibility of fabricating and irradiating advanced nuclear fuels and materials in the Advanced Test Reactor (ATR). The AFC-OA series irradiation experiments will evaluate fuels and materials performance for advanced nuclear systems. Irradiation experiments may include a range of fuel and material forms and compositions from prototypic miniature fuel rods design to evaluate integral performance to small samples designed to evaluate separate effects. Fuel forms currently under investigation include metallic, oxide, and other ceramics. Fuel compositions may include historic binary and ternary metallic alloys and conventional oxide and mixed oxide (MOX) fuels, minor actinides to study transmutation, lanthanide additions to simulate reprocessing effects, and alloying additions to influence constituent redistribution.

Accomplishments:

Two AFC experiments are currently in the irradiation phase of the experiment, AFC 3F and AFC-4C. Their current irradiation status is summarized in the following tables:

Experiment ID and Position	Fuel Composition ID	Neutron Flux ($n/cm^2/sec$)	HGR (W/g)	LHGR (W/cm)	Fission Density (fissions/cc)	HM Burnup (atom %)	U-235 Burnup (atom %)
	EOC	60.1 Days			60.1 Days		
AFC-3F A-11 @23.0 MW	3F-1	4.99E+14	162.15	365.53	7.896E+20	2.12%	4.09%
	3F-2	7.95E+14	155.22	343.68	7.488E+20	2.06%	8.34%
	3F-3	8.85E+14	172.78	378.08	7.821E+20	2.19%	9.60%
	3F-4	8.57E+14	160.16	361.15	7.577E+20	2.10%	9.31%
	3F-5	6.46E+14	186.97	416.72	8.577E+20	2.36%	5.41%
	MAX	**8.85E+14**	**186.97**	**416.72**	**8.577E+20**	**2.36%**	**9.60%**

Experiment ID and Position	Fuel Composition ID	Neutron Flux ($n/cm^2/sec$)	HGR (W/g)	LHGR (W/cm)	Fission Density (fissions/cc)	HM Burnup (atom %)	U-235 Burnup (atom %)
	EOC	60.1 Days			60.1 Days		
AFC-4C A-10 @25.9 MW	4C-1	5.31E+14	197.62	332.31	1.333E+21	3.82%	7.36%
	4C-2	8.98E+14	152.33	354.28	1.159E+21	2.92%	13.47%
	4C-3	1.04E+15	159.36	352.01	1.110E+21	3.11%	16.39%
	4C-4	9.68E+14	161.17	331.00	1.051E+21	3.23%	15.33%
	4C-5	6.95E+14	157.60	355.27	1.129E+21	3.10%	9.79%
	MAX	**1.04E+15**	**197.62**	**355.27**	**1.333E+21**	**3.82%**	**16.39%**

Update to the Advanced Reactor Fuels Irradiation Test Plan

Principal Investigator: Gary Povirk
Collaborators: Doug Dempsey, Jason Harp, Steven Hayes

The plan outlines the different activities required to support the development of advanced metallic fast reactor fuel systems. To achieve this goal, the recommended approach is to continue programs in irradiation testing, post-irradiation examinations, and fuel element fabrication, but also increase the emphasis placed on mechanistic fuel system modeling, literature reviews, validation of models against historic fuel performance data, and evaluation of the impact of prospective fuel designs on reactor plant performance (see Figure 1). In this manner, once an irradiation test program is designed and initiated, there is not only an improved likelihood that the fuel system will behave as expected but there is also a better chance, if the testing is successful, that the fuel design will result in substantial improvements to reactor plant economics and/or safety.

Project Description:

Potential customers to the Advanced Fuels Campaign (AFC) include the Versatile Test Reactor (VTR) and the burner reactor concepts being pursued by the Department of Energy, breed-and-burn reactor designs (TerraPower), and fuel systems for more conventional sodium-cooled fast reactors (GE-Hitachi, Advanced Reactor Concepts, OKLO). Given the broad array of potential applications, the approach will be to explore technically relevant fuel designs but also to develop an improved understanding of fuel performance fundamentals so that specific fuel designs for a given reactor plant can be developed with reduced time and cost. To make matters more specific, two broad classes of fuel designs will be pursued: (1) once-through fuel systems; and (2) fuels for enhanced actinide utilization that would require recycling. For a once-through design, a major focus will be on the elimination of bond sodium, which should facilitate the direct disposal of the fuel in a geological repository. This work will build upon existing efforts to fabricate and characterize the behavior of sodium-free annular fuel, but it will also emphasize fuel alloys that are likely to be employed for the VTR. Fuel designs for enhanced actinide utilization, however, have significantly different constraints and objectives than a once-through fuel system. The focus here will be on fuel shapes that can be fabricated within a hot cell and fuel element designs that can achieve high energy and power density.

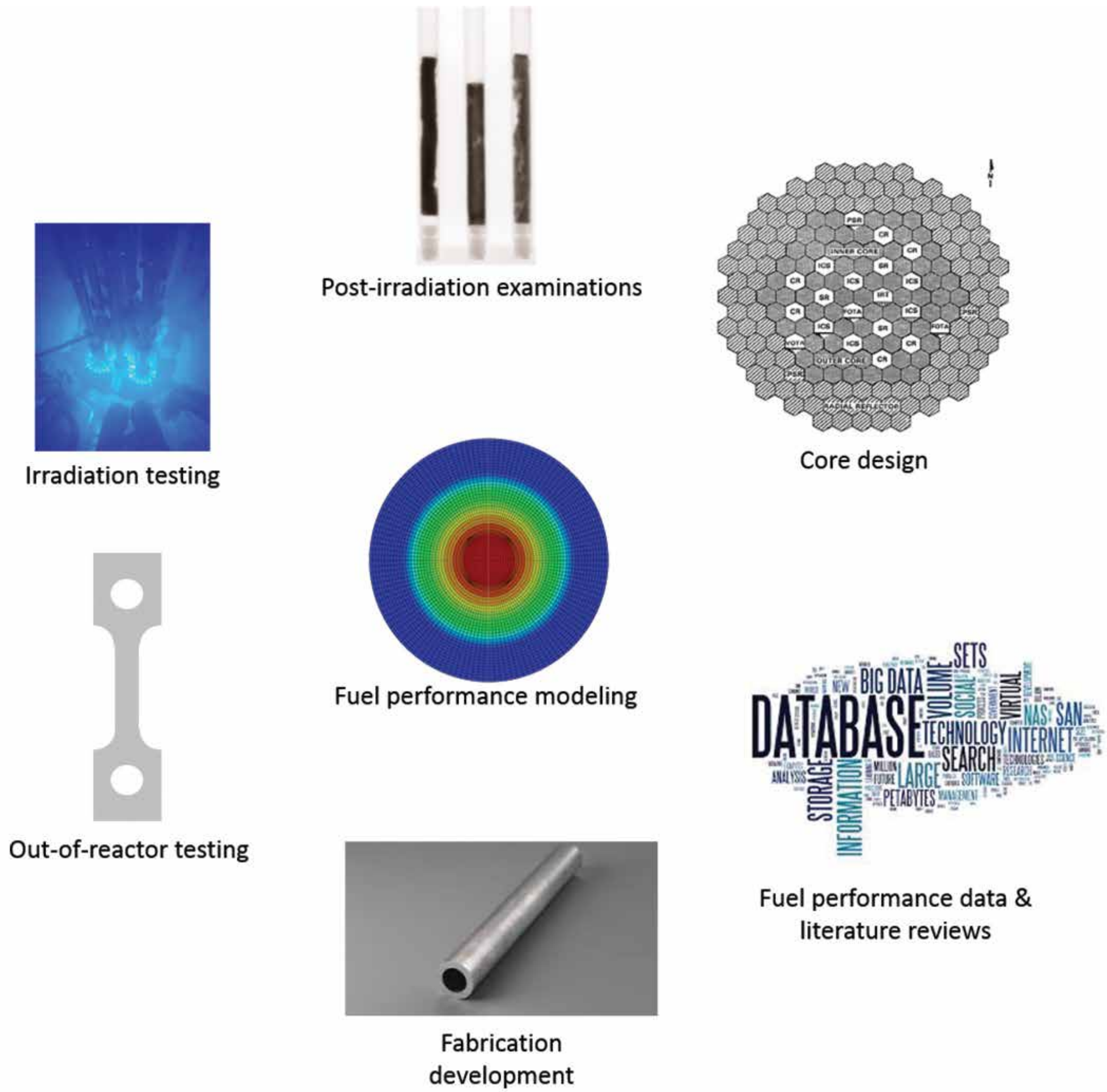


Figure 1. A schematic showing the components of a fuel system development program. A successful program will integrate all of the activities to produce a compelling design at minimum cost, both in terms of financial resources but also with respect to time.

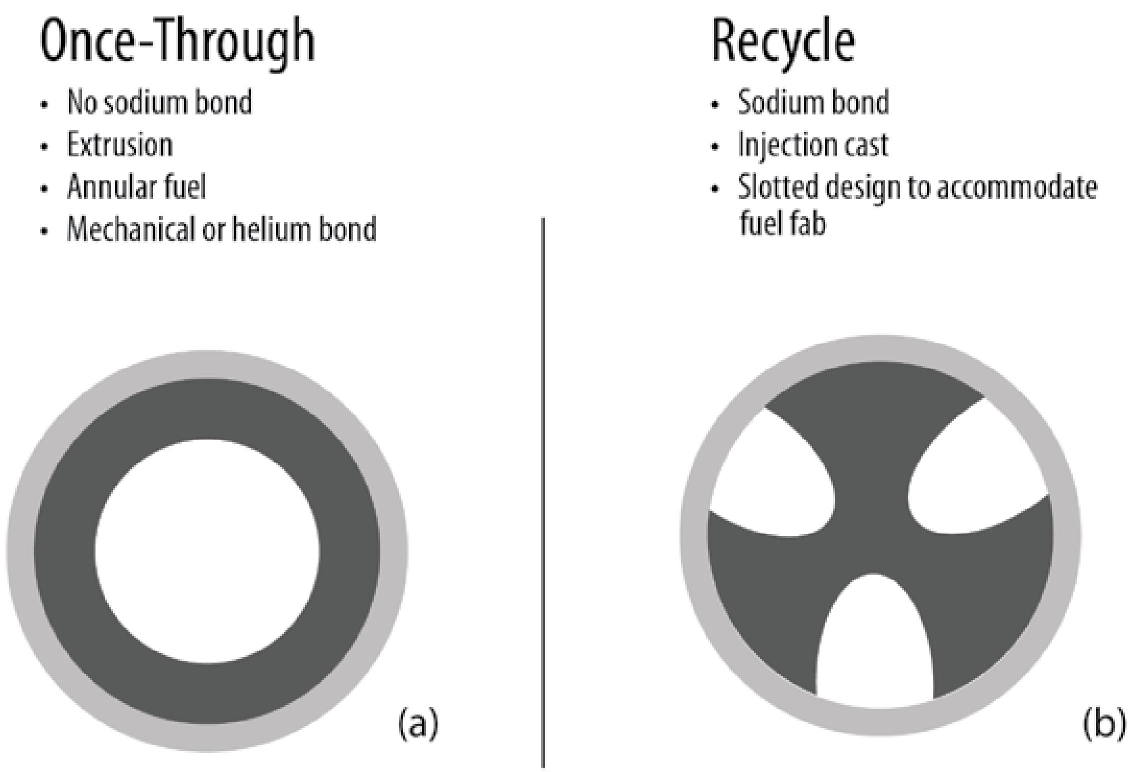


Figure 2. Example fuel designs for (a) a once-through fuel system; and (b) a design when fuel recycling is anticipated.

Accomplishments:

The major accomplishment was to refocus the test program on once-through fuel systems and fuel designs for enhanced actinide utilization. For a once-through fuel system, the elimination of bond sodium suggests that an annular fuel geometry is preferred with either a mechanical bond or with a narrow helium-filled gap between the fuel and cladding to assure the transport of thermal energy from the fuel into the coolant. The focus of development will be on designs appropriate for the second generation, sodium-free fuel system associated with the Versatile Test Reactor. For this application, burnup requirements are expected to be on the order of 10 at.%, so designs that use well-studied fuel alloys of either U-10Zr or U-20Pu-10Zr combined with a smear density of 75 percent are expected to be sufficient.

For a design that involves recycling of the fuel, high burnup designs provide substantial economic benefits, which in turn implies low smear densities to accommodate the accumulation of fission products. In this case, the presence of bond sodium provides significant benefit in terms of heat transfer and results in little additional cost to recycle the fuel. Unfortunately, the fabrication of annular fuel in a hot cell is judged to be difficult, while slotted fuel designs [1] can likely be manufactured by casting (see Figure 2 for a schematic). Concerns associated with slotted fuel designs include distortion of the fuel pin because of the non-uniform pressure exerted by the fuel onto the cladding and the potential for increased axial growth when compared to an annular design. However, once the fuel has swelled into the available void space (typically by 1-2 atom% burnup), the effect of the initial fuel geometry should be minimal. For a given fuel alloy composition, decreasing smear density also necessarily reduces the inventory of fissile material contained within the fuel rod. To offset this reduction, the behavior of pure uranium or uranium-plutonium alloys will be explored. However, given that a major benefit of U-10Zr is its increased resistance to fuel-clad chemical interaction, these fuel designs will require the development of liner or coating technologies to protect the clad from the fission products that accumulate as the fuel is burned.

An alternative to both annular and slotted fuel designs would be to fabricate porous metallic fuel. If sodium bond was used in conjunction with fuel with elongated porosity, one potential advantage is that bond sodium could potentially be infiltrated into the fuel during fabrication and thereby reduce the uncertainty regarding the amount of sodium in the fuel during the course of irradiation. For the present time, however, annular fuel is the default geometry for once-through fuel systems and slotted designs are assumed to be more appropriate for the recycled fuel designs. Fabrication and testing of these fuel systems will be the focus of the program in the coming years.

The testing program has been re-focused to support the Versatile Test Reactor and on fuel designs that will be able to achieve substantially higher burnups without a reduction in heavy metal inventory.

ATF-1 Postirradiation Examination Highlights for 2017

Principal Investigator: Jason M. Harp
Collaborators: Luca Capriotti, Gary Povirk, Steven Hayes, Douglas Porter, Pavel Medvedev

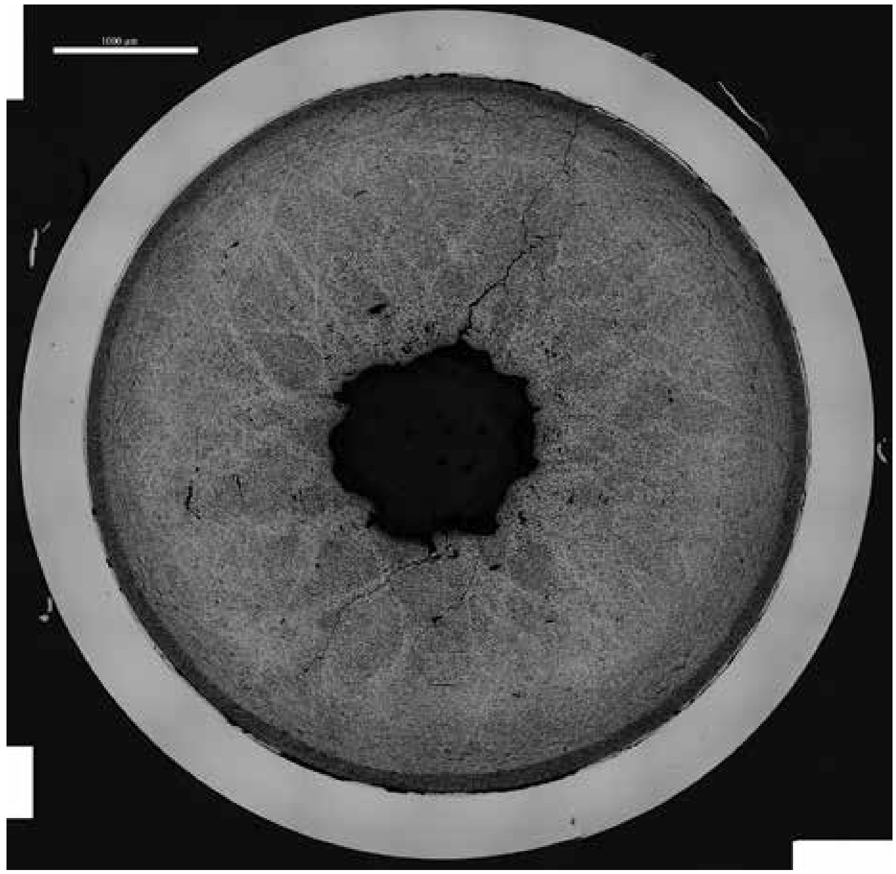


Figure 1. Gamma tomography showing the relative distribution of Ce-144 in (a) AFC-3A R4 and (b) AFC-3A R5. The black circles in the plot represent the cladding around the fuel. Intensity beyond the cladding is an artifact of the technique.

Postirradiation examination of transmutation fuel concepts is on-going at both the INL HFEF hot-cell and other INL PIE facilities. Currently 14 rodlets from AFC-3C, AFC-3D, and AFC-4A are actively being examined to complete baseline PIE characterization, and several other rodlets with composition and forms of high interest are being examined further. Advances in PIE techniques, including enhanced access to electron microscopy, continue to enrich the phenomenological understanding of nuclear fuel performance under irradiation.

Project Description:

The AFC irradiations are designed to explore the fuel performance of candidate transmutation and fast reactor fuel concepts in a simulated pseudo-fast neutron environment. The fast neutron environment is created using a cadmium filter while irradiating fuel in the Advanced Test Reactor which is a thermal neutron spectrum materials test reactor. The proper temperature is achieved by a control gap between the test fuel cladding and a capsule that isolates the test specimen from the ATR primary coolant. The current set of tests under examination consist of a variety of alloys and forms that have shown some potential in out-of-reactor testing to enhance certain fuel performance properties. The potential for eliminating sodium bonding between the fuel alloy and cladding is also explored in these fuels. Spent sodium bonded fuel currently requires additional treatment prior to disposal which is undesirable. Out-of-reactor tests cannot fully capture all the impacts of irradiation, so these alloys and forms must be tested through irradiation. Successful candidate fuels will be carried forward for additional testing. The eventual goal is to test a fuel candidate that can out perform the historic standard fuel alloy (U-10Zr) and form and allow burnup beyond 20 atom percent and cladding temperatures in excess of 600°C. Additionally, exams continue on fuel concepts that include minor actinide (Np, Am) additions to well studied fuels. Understanding the effect minor actinides have on fuel performance (if any), may allow for the destruction of minor actinides in fast reactors which in turn would reduce the long term radiotoxicity of a geological repository.

Accomplishments:

Baseline PIE was completed on the 4 rodlets from AFC-4A. This test included some novel alloy compositions, an annular form, and a control solid U-10Zr rodlet. This test includes examination of the novel Molybdenum, Titanium, Zirconate alloy with and without Pd as an additive. Neutron radiography, gamma spectrometry, fission gas release, optical microscopy and chemical analysis is now complete on these pins, and they will be available for further exams if desired in the future. The AFC-4A microscopy samples were some of the first samples examined on the new optical microscope available at HFEF. This new microscope is capable of resolving micrometer scale features in fuel. In addition to

This project continues to provide data on the performance of fast reactor and transmutation fuel forms in the absence of a fast test reactor. Information learned from this project will support future development of a fast spectrum test reactor and research options to improve long term repository safety.

the microstructure data that can be ascertained from optical microscopy, it is now possible to examine many more samples and larger samples than can be examined using electron microscopy. This scale of microscopy can also better screen many samples and better identify the key samples to be sent to electron microscopy. Full cross section microscopy of AFC-4A R1 (annular U-10Mo) is shown in the figure below. Efficiencies gained by this microscope allow for full cross section data to be collected at twice the magnification previously available in less than half the time. The 10 rodlets from AFC-3C and AFC-3D were received at HFEF and began the examination process. Removal of the rodlets from the capsules and some non-destructive exams have been completed.

Several scanning electron microscopy (SEM) exams were completed on a variety of different fuel forms including U-10Zr irradiated in FFTF, minor actinide bearing mixed oxide fuel from AFC-2D, and U-Pd-Zr alloys irradiated in AFC-3. Specific sample preparation is often required to successfully examine these fuels. The dose rate must be reduced not only to protect the workers loading samples into the microscope but also to limit the amount of radiation from the sample that can overwhelms the detectors in the microscope. A good example of the specialized sample preparation required for fresh fuel microscopy is shown in the backscatter electron microscopy shown in the image below.

The first issue of "Testing Fast Reactor Fuels in a Thermal Reactor: A Comparison Report" was issued this year. This report documents the multi-year effort to demonstrate the viability of testing fast neutron spectrum reactor fuel concepts in the thermal neutron spectrum Advanced Test Reactor. The report demonstrates the modeling effort needed to demonstrate that the radial power profile in this testing is representative of fast reactor irradiations. It also highlights a comparison between the irradiation results from like composition tests that were carried out in ATR (AFC-1) and the true fast spectrum reactor Phenix (FUTURIX-FTA). This report presents evidence that use of Cd-filtering creates irradiation conditions in the ATR that are relevant to, and suitable for, fast reactor fuel testing to be achieved.

Electron Microscopy of FUTURIX-FTA Metallic Fuel

Principal Investigator: Jason M. Harp
Collaborators: Luca Capriotti, Steve Hayes, Douglas Porter

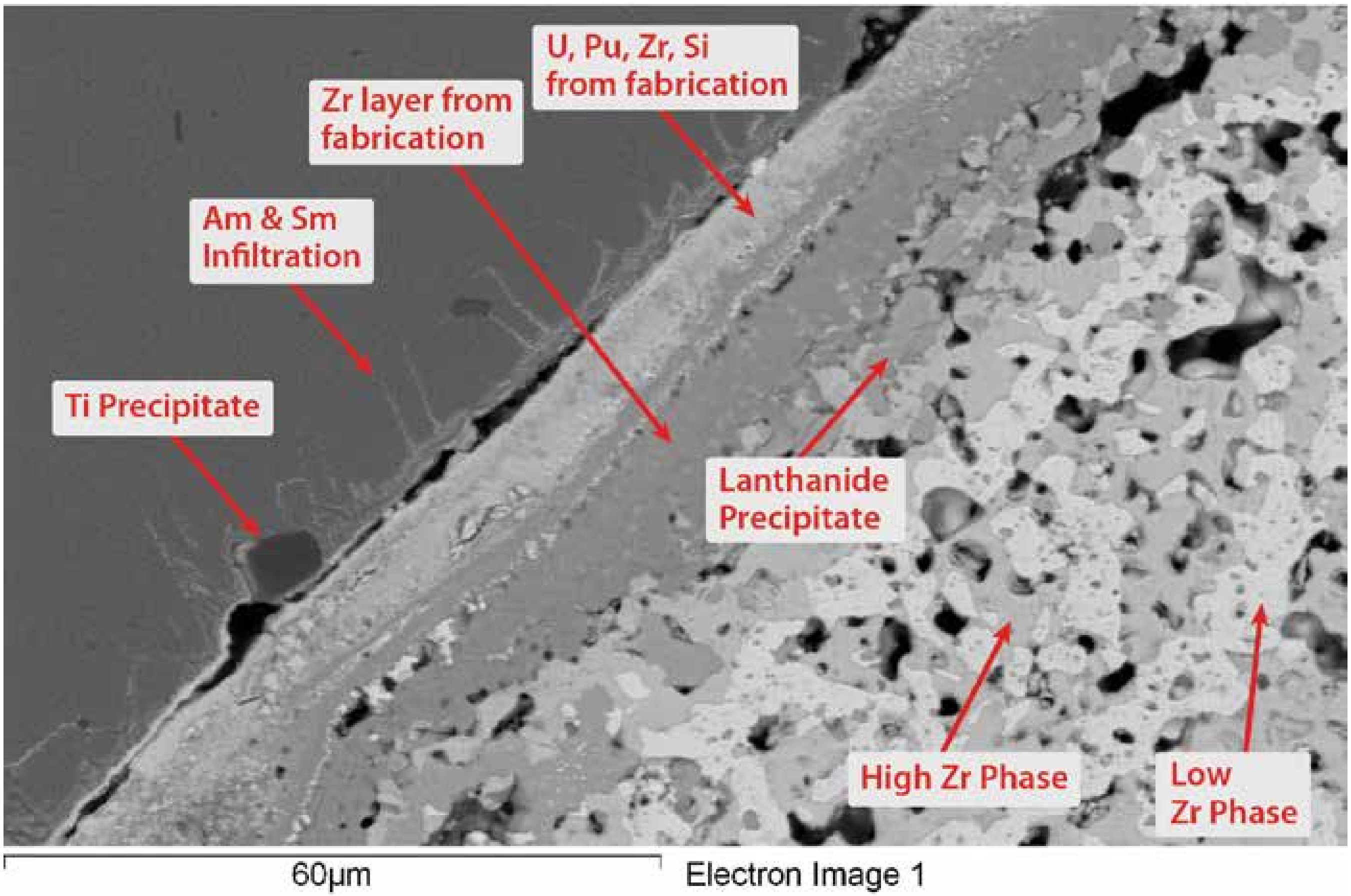


Figure 1. Backscatter electron imaging of the full cross section from FUTURIX-FTA DOE1 (34.1U-28.3Pu-3.8Am-2.1Np-31.7Zr) irradiated fuel

Scanning electron microscopy (SEM) was performed on a full transverse cross section sample from FUTURIX-FTA DOE1 (34.1U-28.3Pu-3.8Am-2.1Np-31.7Zr) which is the low-fertile, metallic composition from that test. This exam preceded electron probe micro analyzer (EPMA) work on the same sample. The data gathered from SEM exams which is semi-quantitative helps to guide quantitative EPMA exams. A great deal of information can be gathered from the SEM exams on the microstructure of the fuel post-irradiation that is not available to optical microscopy. Most notably information on the distribution of different chemical phases can be gathered. The general trends in microchemistry across the fuel cross section can also be rapidly evaluated using data from SEM exams.

Project Description:

Electron microscopy exams performed on this sample are a continuation of the postirradiation examination (PIE) of FUTURIX-FTA. This irradiation test was performed at the Phénix Reactor in France which was a fast spectrum sodium cooled reactor. The compositions irradiated in Phénix were also irradiated in the Advanced Test Reactor (ATR) as part of the AFC-1 irradiation test. These two irradiation tests are designed, in part, to show that fast reactor fuel performance can be explored in the absence of fast spectrum test reactor testing by utilizing a Cd-shrouded pseudo-fast spectrum irradiation position in the ATR. The comparing the microstructure of this fuel to sister AFC-1 tests is part of this comparison. There are several key microstructure properties that can be evaluated with the SEM. The redistribution of Zr in the fuel pin was evaluated and found to exist in this system, but the change in local Zr concentration was not as extreme as is seen historically in U-10Zr and U-20-10Zr fuel. The nature and extent of fuel cladding chemical interaction if any was also evaluated in this exam. The performance of the composition irradiated in FUTUIRX-FTA DOE1 is also of interest as it is quite unique especially because of the high Pu content and minor actinide (Np, Am) additions. Minor Actinides and Pu in spent nuclear fuel are key contributors to the radiotoxicity of long-term geological repositories. Fast spectrum reactor fuels such as the compositions irradiated in FUTURIX-FTA could be used to destroy Pu and minor actinides improving the safety of a geological repository. Because of this goal, it is important to know if the minor actinides have adversely impacted fuel performance perhaps by enhancing fuel cladding chemical interaction or by creating a local phase that has a lower melting temperature reducing the margin to melt in a portion of the fuel.

Accomplishments:

Scanning electron microscopy was performed on a full cross section from FUTURIX-FTA DOE1. This sample presented a challenge for the SEM due to the significant amount of radioactivity in the sample. Although the sample came out of the reactor in 2009, the dose rates from the sample were quite high for personnel loading the instrument. There was also a substantial amount of internal conversion x-rays being emitted from the sample that result from alpha decay in the sample. This created significant deadtime in the EDS detector and lengthened the time needed to collect EDS spectra. The internal conversion x-rays also limited the usefulness of the characteristic L line x-rays of the major constituents. In spite of these challenges it was possible to collect

Scanning electron microscopy has been performed on a full cross section of irradiated fuel that contains minor actinides.

high magnification backscatter electron images of the different microstructural features of this sample. The overall microstructure of the sample is shown in the first image below. This image is a composite of about 150 images stitched together to reveal the general state of the sample. In this sample, Zr redistribution is not as pronounced as is seen with U-10Zr samples, but the central region from the center of the fuel out to about a third of the fuel radius is single phase and has more Zr than the as fabricated fuel. Beyond this region there is some large porosity then the fuel transitions to predominately a 2 phase microstructure. The contrast of these phases and local EDS analysis of these phases show that the brighter phase contains more U and less Zr than the as-fabricated composition and the darker phase contains less U and more Zr than the as fabricated composition. The size of these phases varies radially with the phase size shrinking towards the outer radius of the fuel. At the outer radius of the fuel is a very pronounced Zr rind that was formed during fabrication. This same rind was present in the as-fabricated SEM performed on this fuel. At higher magnifications it is possible to observe some very small, novel interactions between the fuel and the cladding. A backscatter image of one of these regions is shown in the second figure which details the near cladding microstructure observed throughout the fuel. Several features are indicated in the figure as key features. The cladding is a variant of stainless steel 316, AIM1, which contains ~0.5 wt.% Ti. It appears the Ti has precipitated during irradiation. Next to the cladding is a layer of U, Pu, Zr, Si and possibly some Am. This layer of Si is an artifact from fabrication where Si from the quartz mold enters the fuel and was observed in fabrication. The Zr layer is also an artifact from fabrication where Zr from the wash used on the quartz enters the fuel and was observed in fabrication also. There are precipitates of Lanthanides (La, Ce, Nd, and others) near the Zr layer. However there does not appear to be any attack of the cladding from the major fission product lanthanides (La, Ce, Nd). The accumulation of lanthanides at the Zr rind has been observed historically. The higher Zr, lower U phase and the lower Zr, higher U phase mentioned previously are also indicated. The most significant feature observed in the figure is an infiltration of Am and Sm into the cladding. Both of these elements have high vapor pressures. It is suspected that this infiltration is assisted by these high vapor pressures, but more research into the phenomenon is needed.

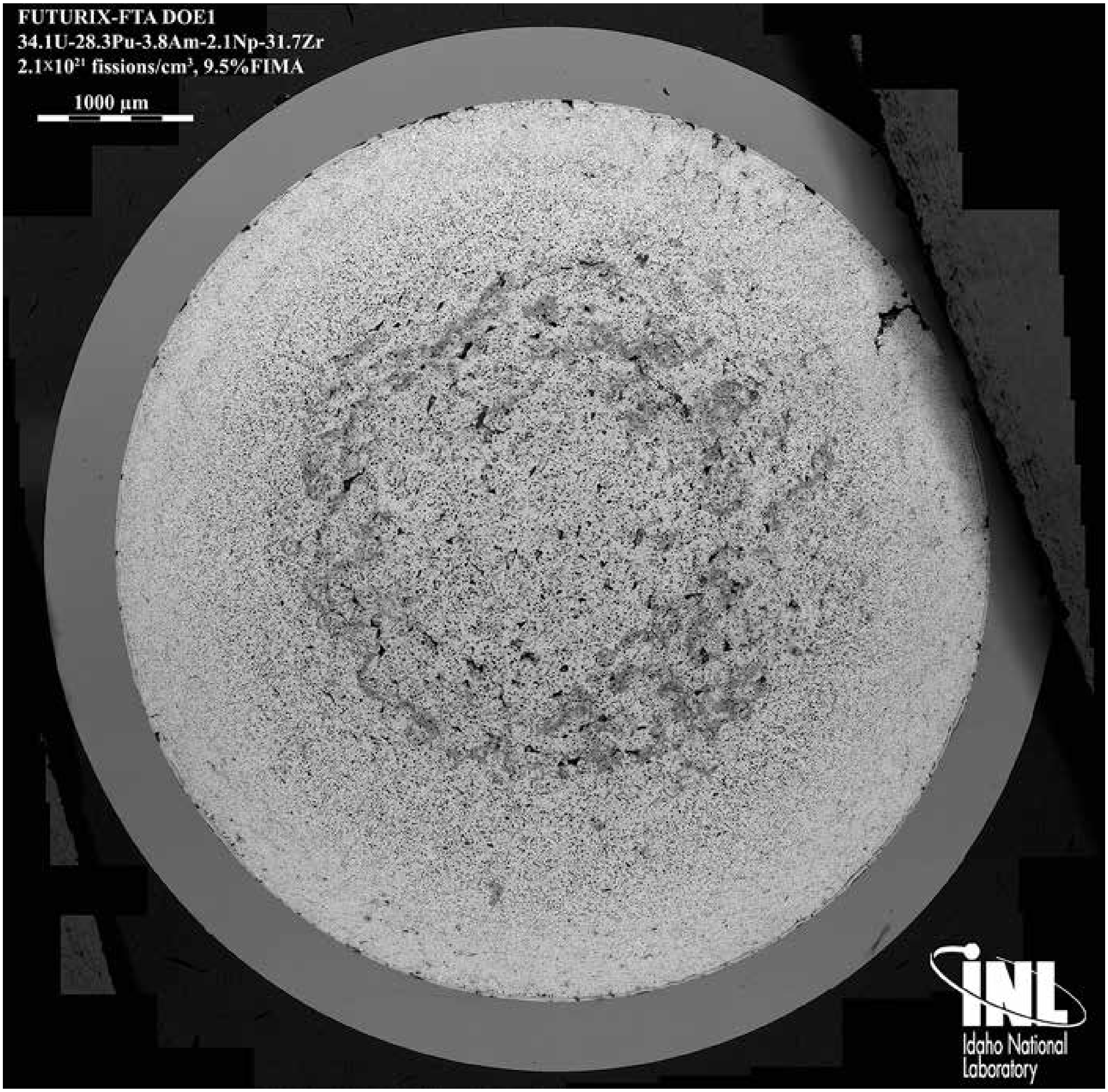


Figure 2. High magnification backscatter electron image detail of the FUTURIX-FTA DOE1 microstructure near the cladding.

Initial Electron Probe Microanalysis Examination of FUTURIX-FTA Sample DOE-1

Principal Investigator: Karen Wright
Collaborators: Jason Harp, Cindi Papesch

Idaho National Laboratory is the only DOE laboratory to have the capability to quantitatively chemically analyze highly irradiated nuclear fuels on a micrometer scale using EPMA.

Fast reactor irradiation of transmutation fuels is capable of converting long-lived actinide elements into much shorter lived fission products. This can result in a dramatic reduction to the long-term radiotoxicity and heat load of spent nuclear fuels, which could potentially lead to a greatly reduced timeframe for which a geologic repository would need to be licensed.

This work presents initial results of the first electron probe microanalysis (EPMA) performed in the United States on a metallic transmutation fuel specimen from the FUTURIX-FTA experiment irradiated in the Phénix fast reactor in France.

Project Description:

Idaho National Laboratory is the only DOE laboratory capable of conducting quantitative EPMA analysis on highly irradiated nuclear fuels. It joins a few government laboratories from Europe and Asia who have this capability, thus worldwide, this capability is rare, but highly significant. First, DOE's Advanced Fuels Campaign (AFC) is currently developing a new mechanistic model describing constituent redistribution in metallic fuels, and data obtained from this and subsequent EPMA examinations of other already irradiated fuels experiments (e.g., AFC-1, FUTURIX-FTA) is needed to begin validation of this model. Second, it is important to compare how irradiation performance data obtained from the cadmium-shrouded tests of fast reactor fuels conducted in the Advanced Test Reactor compare to identical (or very similar tests) conducted in fast spectrum test reactors (i.e., EBR-II, FFTF, and Phénix). The chemical redistribution information that will be obtained using the EPMA over the next several years is vital data needed for that comparison; in fact, that comparison, which is directly related to the radial temperature profile experienced by the test pins, cannot be considered comprehensive or complete without it.

Accomplishments:

The objectives of this research are several-fold. They include: 1) Construction of a radiologically shielded EPMA suite, capable of handling and analyzing highly radioactive (2 Ci of 60Co) irradiated fuel samples; 2) Development of the EPMA capability to be able to analyze quantitatively irradiated fuel samples, to include actinide elements. This includes fabrication and testing of neptunium, plutonium, and americium standards; and 3) Demonstration of the EPMA capability by quantitatively analyzing a cross section of irradiated nuclear fuel.

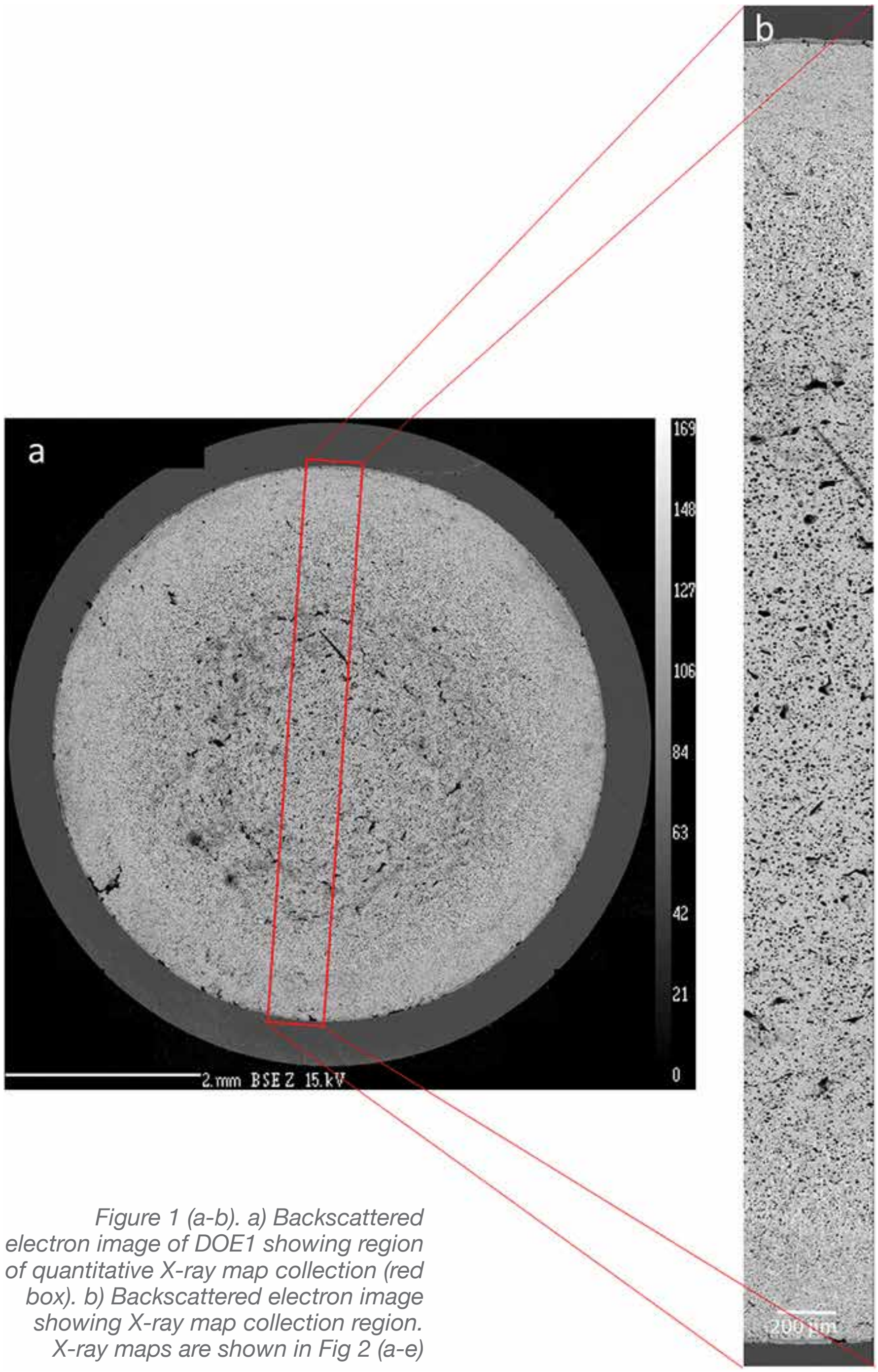


Figure 1 (a-b). a) Backscattered electron image of DOE1 showing region of quantitative X-ray map collection (red box). b) Backscattered electron image showing X-ray map collection region. X-ray maps are shown in Fig 2 (a-e)

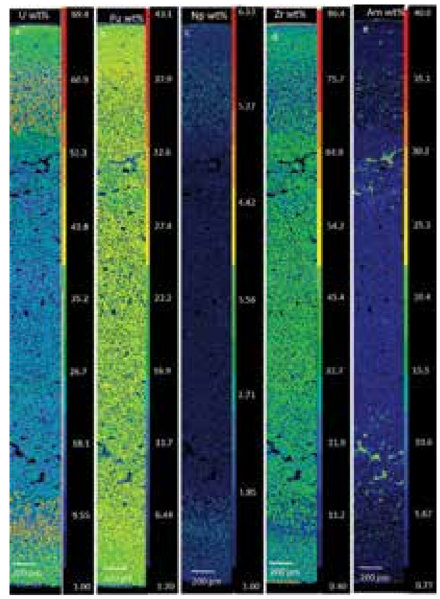


Figure 2 (a-e). a) U X-ray map, b) Pu X-ray map, c) Np X-ray map, d) Zr X-ray map, e) Am X-ray map. All maps are shown in weight %

Comparison Report on Fast vs. Filtered-Thermal Spectrum Irradiations

Principal Investigator: Jason Harp
Collaborators: Steven Hayes, Pavel Medvedev, Douglas Porter, Luca Capriotti

> ***According to one of the external peer reviewers of this work, "the data comparisons and analysis show that the ATR irradiations with the Cd shroud system is sufficiently prototypic that it can be used to develop fast reactor fuels with confidence."***

The testing of fast reactor fuels in the thermal spectrum of the Advanced Test Reactor using cadmium-filtering has been underway since 2003. The objective of this experimental approach is to create a thermal profile inside of test fuel rodlets that is nearly prototypic of corresponding fast reactor conditions. By doing so, this experimental approach should allow for the study of fast reactor fuel performance phenomena that depend primarily on the conditions of temperature and/or temperature gradient inside the fuel to be possible using a thermal test reactor.

Project Description:

Validation of the experimental approach has been undertaken by comparing fuel performance phenomena observed in fuels irradiated in cadmium-filtered positions in ATR to similar or identical fuels irradiated under similar conditions in genuine fast reactors. The results of these comparisons and associated analyses were documented in a Comparison Report.

Accomplishments:

The Comparison Report documents a preliminary comparison of limited fuel performance data available from the testing of metallic, oxide, and nitride fuels in cadmium-filtered positions in the ATR to the performance of identical or similar fuels irradiated in genuine fast reactors. Although a variety of fuel performance metrics were presented, particular attention was paid to those phenomena that are primarily, or significantly, dependent on conditions of temperature and temperature gradient with the fuel. Comparisons show how formation of a columnar grain region in oxide fuels is impacted by the radial power distribution in the fuel. In an unfiltered ATR neutron spectrum, significant power is shifted to the fuel pellet periphery, requiring a much higher power to achieve formation of the columnar grain region. Use of cadmium-filtering results in removal of the vast majority of thermal neutrons seen by the experimental test fuels, which tends to level out the radial power distribution similar to the essentially flat radial power distribution that would exist in a fast reactor. This effect was demonstrated by modeling formation of the columnar grain region in both cadmium filtered ATR and typical fast reactor neutron spectra, which confirmed the similarity of oxide fuel behavior in these two conditions.

Additionally, data obtained from postirradiation examinations performed on metallic and nitride fuels irradiated in cadmium-filtered positions in ATR as part of the AFC-1 test series were compared with similar fuels irradiated in genuine fast reactors. Results from neutron radiographs that imaged the general conditions of the fuel columns after irradiation revealed good qualitative agreement. Results from cladding profilometry agreed in no measurable diametral strain. Results from gamma ray spectrometry and tomography revealed no essential differences, and appear to confirm that the transport of important fission products like cesium and fuel constituents such as zirconium behave the same in both test environments. Results from fission gas/helium release measurements, when acknowledging the considerable variation observed in this phenomenon even from identical pins irradiated under identical conditions, show very good agreement. And finally, results from burnup analyses show good agreement under both test environments.

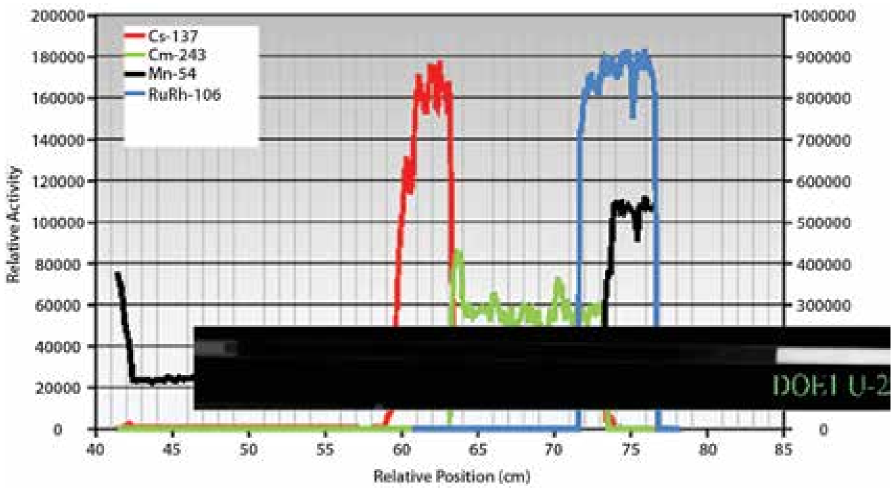


Figure 1 Axial distribution of important radionuclides in U-29Pu-4Am-2Np-30Zr metallic fuel irradiated in the Phénix fast reactor.

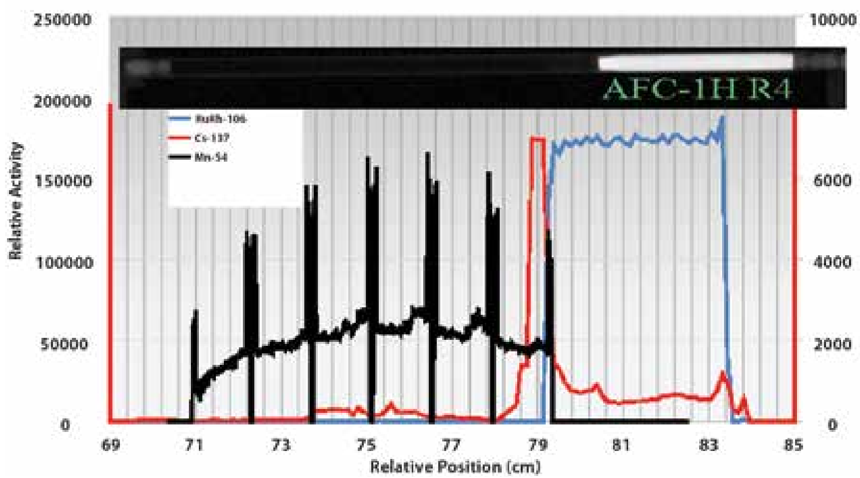
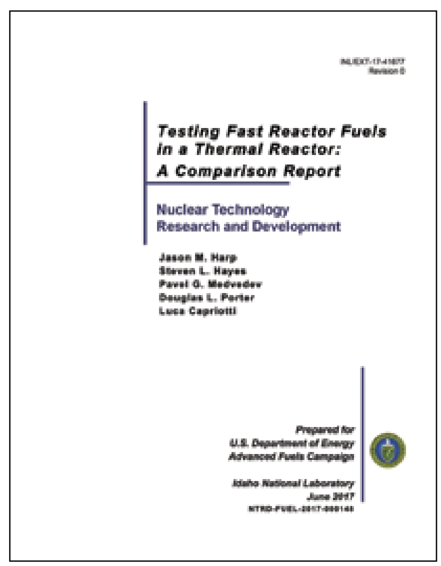


Figure 2 Axial distribution of important radionuclides in U-29Pu-4Am-2Np-30Zr metallic fuel irradiated in a cadmium-filtered position in ATR.



Testing Fast Reactor Fuels in a Thermal Reactor: A Comparison Report

Nuclear Technology Research and Development

Jason M. Harp
Steven L. Hayes
Pavel G. Medvedev
Douglas L. Porter
Luca Capriotti

Prepared for
U.S. Department of Energy
Advanced Fuels Campaign

Idaho National Laboratory
June 2017

Figure 3 The Comparison Report documents preliminary data and analyses that validate the cadmium-filtered testing approach used in ATR to test fast reactor fuels.

Advanced Instrument Development: Thermal Conductivity Microscope

Principal Investigator: David Hurley
Collaborators: Robert Schley

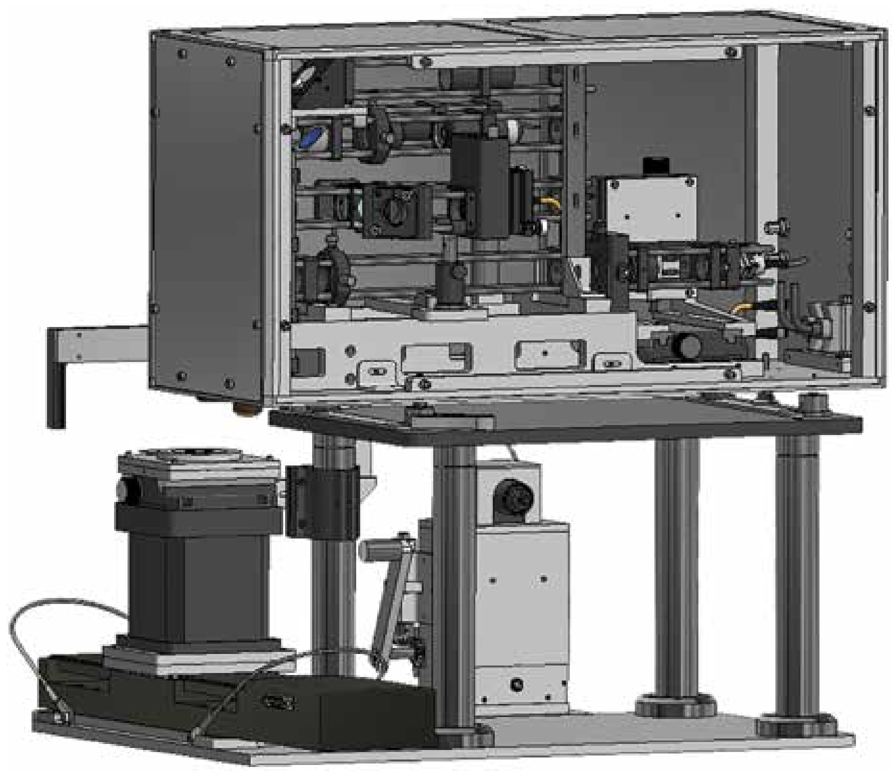


Figure 1. 3D model of Thermal Conductivity Microscope in cell system

The Thermal Conductivity Microscope (TCM) is an instrument designed to measure thermal property of irradiated samples. The TCM simultaneously measures the thermal diffusivity and conductivity using modulated thermoreflectance. This measurement approach involves measuring the temperature field spatial profile of samples excited by an amplitude modulated continuous wave laser beam. A thin gold film is applied to the samples to ensure strong optical absorption and to establish a second boundary condition that introduces an expression containing the substrate thermal conductivity. The diffusivity and conductivity are obtained by comparing the measured phase profile of the temperature field to a continuum based model. The TCM has been designed to operate in a radiation hot cell environment. It can be controlled remotely via the software interface and sample loading is compatible with hot cell manipulators.

Project Description

The Thermal Conductivity Microscope (TCM) will provide a new capability for the measurement of thermal properties of irradiated samples. Thermal conductivity and thermal diffusivity are among the most important properties of a nuclear fuel, but they can also be one of the most difficult to measure. The TCM provides micron-level thermal property information that is commensurate with microstructure heterogeneity. The development of the TCM connects closely with INL's larger PIE effort to provide new validation metrics for fundamental computational material science models.

A prototype TCM instrument was designed and tested during FY16. Work for FY 17 centered on developing ancillary equipment to support standalone remote operation in preparation for installation in the Irradiated Materials Characterization Laboratory (IMCL) and initiation of equipment qualification. The primary tasks included, 1- development of cell feedthroughs for both optical and electrical signals, 2 – design and construction of a film thickness monitor, 3 – development of an equipment qualification plan, 4 – develop an integrated equipment rack to house lasers and instrumentation, and 5 – develop coding instrumentation software for controlling the TCM.

Accomplishments

Several accomplishments were realized in the development of the TCM during FY17. Using the standard glove box panel, a layout for the electrical and optical feedthroughs was developed. Feedthroughs were incorporated into standard vacuum fittings to meet leak rate requirements and for ease of installation and replacement if necessary. Vendors for the feedthroughs were identified and the fittings were procured and tested.

The ability of the TCM to directly measure thermal conductivity requires a thin gold film to be applied to the sample. The film thickness is an important parameter that must be measured. A unique film thickness monitor was developed which allows film thickness to be determined based on light transmission. This device was designed and fabricated for use in a hot cell using remote manipulators for sample placement.

The TCM provides micron-level thermal property information that is commensurate with microstructure heterogeneity. The TCM will allow for a direct comparison between microstructure and material properties without the need to upscale modeling results to length scales associated with current experimental capability. The development of the TCM connects closely with INL's larger PIE effort to provide new validation metrics for fundamental computational material science models.

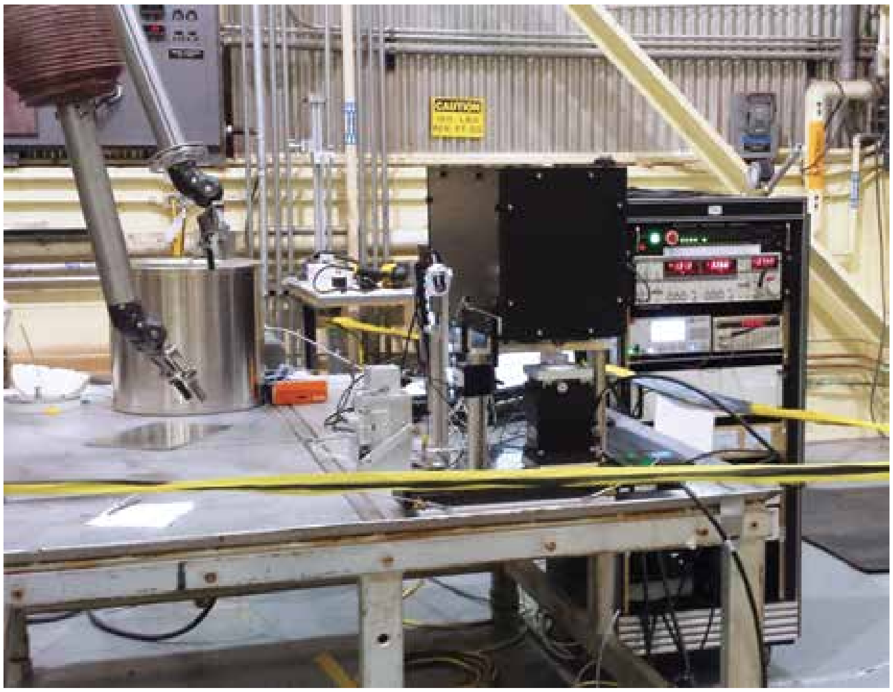


Figure 2. Thermal Conductivity Microscope testing at the Materials and Fuels Complex

An equipment qualification plan (EQP) for stage I/II mockup was also completed in FY17. This document contains basic instructions for the proper assembly of the TCM in the Mockup Shop located at the Materials and Fuels Complex (MFC) at Idaho National Laboratory (INL). Phase I: The TCM will be setup on the base plate, the electro-mechanical components positioned, and all electrical wiring will be connected. Phase II: After the initial system operation is verified, remote qualification at the MFC Mockup wall will be performed to verify the equipment can be operated in a hot cell environment with manipulators.

In addition to the thickness monitor and EQP, an equipment rack and software were also developed. The equipment rack incorporates all of the instrumentation and controls for remotely operating the TCM. An interface box in conjunction with a newly developed power box allow automation of signal switching and powering of the detectors, sensors, cameras and LEDs necessary for TCM operation. A method for sample height determination which will prevent sample to objective interference was also developed and incorporated. A graphical user interface (GUI) was developed using MATLAB to provide point-and-click control of the TCM and thickness monitor, as well as data analysis.

The TCM and associated equipment were entered into stage one mockup at MFC on 9/6/2017. This included the TCM, the equipment rack and all ancillary equipment as well as the film thickness monitor. Mockup testing is underway and includes manipulator compatibility and performance verification test using thermal conductivity standards.

APPENDIX

4.1 Publications
4.2 FY-16 Level 2 Milestones
4.3 AFC NEUP Grants
4.4 Acronyms

4.1 PUBLICATIONS

Author (s)	Title	Publication
M. E. Alam, S. Pal, K. Fields, S. A. Maloy, D. T. Hoelzer and G. R. Odette	Tensile deformation and fracture properties of a 14YWT nanostructured ferritic alloy	Materials Science and Engineering a-Structural Materials Properties Microstructure and Processing 675: 437-448(2016)
M. E. Alam, S. Pal, S. A. Maloy and G. R. Odette	On delamination toughening of a 14YWT nanostructured ferritic alloy	Acta Materialia 136: 61-73 (2017)
G. Aliberity, T. K. Kim, N. Stauff	Assessment of AmBB Performance and Tradeoff of Advanced Fuel Concepts – FY 2017	NTRD-FUEL-2017-00012, September 30, (2017)
C. Ang, Y. Katoh, C. Kemery, J. Kiggans, K. Terrani	Chromium-based mitigation coatings on SiC materials for fuel cladding	Transactions of the American Nuclear Society, 114 (2016) 1095
C. Ang, J. Kiggans, C. Kemery, S. O'Dell, K. Terrani, J. Burns, Y. Katoh	Characterization and mechanical properties of coated SiC fuel cladding	Transactions of the American Nuclear Society, 115 (2017)
C. Ang, S. Raiman, J. Burns, X. Hu, Y. Katoh	Evaluation of the First Generation Dual-purpose Coatings for SiC Cladding	ORNL/SR-2017/318, Oak Ridge National Laboratory, Oak Ridge, TN (United States), 2017
C. Ang, K. Terrani, J. Burns, Y. Katoh	Examination of Hybrid Metal Coatings for Mitigation of Fission Product Release and Corrosion Protection of LWR SiC/SiC.	ORNL/TM-2016/332, Oak Ridge National Laboratory, Oak Ridge, TN (United States), 2016
Daniel J. Antonio, Keshav Shrestha, Jason M. Harp, Cynthia A. Papesch, Yongfeng Zhang, Jon Carmack, and Krzysztof Gofryk	Thermal and Transport Properties of U_3Si_2	Journal of Nuclear Materials - submitted, under review
E. Aydogan, N. Almirall, G.R. Odette, S.A. Maloy, O. Anderoglu, L. Shao, J.G. Gigax, L. Price, D. Chen, T. Chen, F.A. Garner, Y. Wu, P. Wells, J.J. Lewandowski, D.T. Hoelzer	Stability of Nanosized Oxides in Ferrite Under Extremely High Dose Self Ion Irradiations	Journal of Nuclear Materials 486, (2017), 86-95
E. Aydogan, S. A. Maloy, O. Anderoglu, C. Sun, J. G. Gigax, L. Shao, F. A. Garner, I. E. Anderson and J. J. Lewandowski	Effect of tube processing methods on microstructure, mechanical properties and irradiation response of 14YWT nanostructured ferritic alloys	Acta Materialia 134: (2017), 116-127
M. T. Benson, J. A. King, R. D. Mariani, M. C. Marshall	SEM characterization of two advanced fuel alloys: U-10Zr-4.3Sn and U-10Zr-4.3Sn-4.7Ln	Journal of Nuclear Materials 494 (2017), 334-341

Author (s)	Title	Publication
T. M. Besmann, M. J. Noorhoek, T. Wilson, A. T. Nelson, E. S. Wood, J. W. McMurray, D. Shin, E. J. Lahoda and S. C. Middleburgh	Uranium Silicide-Nitride Fuels: Thermochemical Behavior and Compatibility	ICACC, Daytona Beach, FL, January, 2017
John D. Bess, Connie M. Hill, Nicolas E. Woolstenhulme, Colby B. Jensen	Analyses Supporting Design Review of TREAT Multi-SERTTA Experiment Test Vehicle	M&C 2017 Conference, April 16-20, 2017, Jeju, Korea
J.Bischoff, C.Vauglin, P. Barberis, C. Roubeyrie, D. Perche, D. Duthoo, F.Schuster, J-C. Brachet, K. Nimishakavi	AREVA NP's Enhanced Accident Tolerant Fuel Developments: Focus on Cr-Coated M5TM Cladding	2017 Water Reactor Fuel Performance Meeting, Korea, September 2017
P.A. Burr, D. Horlait and W.E. Lee	Experimental and DFT Investigations of (Cr,Ti)3AlC2 Solid Solution Stability	Materials Research Letters 5 [3] 144-157 (2017)
Lu Cai, Peng Xu, Andrew Atwood, Frank Boylan and Edward J. Lahoda	Thermal Analysis of ATF Fuel Materials at Westinghouse	ICACC, Daytona Beach, FL, January, 2017
U. Carvajal Nunez, T.A. Saleh, J.T. White, B. Maiorov, and A.T. Nelson	Determination of Elastic Properties of Polycrystalline U_3Si_2 using Resonant Ultrasound Spectroscopy	Journal of Nuclear Materials Accepted (2017)
U. Carvajal-Nunez, J.T. White, E.S. Wood, N.A. Mara, and A.T. Nelson	Nanoscale mechanical behavior of uranium silicide compounds	Top Fuel 2016: LWR Fuels with Enhanced Safety and Performance , pp. 1435 - 1441
P. Domitr, L-Y. Cheng, P. Kohut, J. Ramsey	Development of TRACE Model Based on TRAC-M Input for PWR LOCA Analysis	NURETH-17, Xian, China, September 3-8, 2017
P. Doyle, S. Raiman, R. Rebak, K. Terrani	Characterization of the Hydrothermal Corrosion Behavior of Ceramics for Accident Tolerant Fuel Cladding	Proceedings of the 18th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors (2017)
Sebastien Dryepondt, Caleb Massey, Maxim N. Gussev	Production and Characterization of large Batch FeCrAl ODS Alloy	ORNL/TM-2017/445 (2017)

Author (s)	Title	Publication
Juan C. Fernández, D. Cort Gautier, Chengkung Huang, Sasikumar Palaniyappan, Brian J. Albright, Woosuk Bang, Gilliss Dyer, Andrea Favalli, James F. Hunter, Jacob Mendez, Markus Roth, Martyn Swinhoe, Paul A. Bradley, Oliver Deppert, Michelle Espy, Katerina Falk, Nevzat Guler, Christopher Hamilton, Bjorn Manuel Hegelich, Daniela Henzlova, Kiril D. Ianakiev, Metodi Iliev, Randall P. Johnson, Annika Kleinschmidt, Adrian S. Losko, Edward McCary, Michal Mocko, Ronald O. Nelson, Rebecca Roycroft, Miguel A. Santiago Cordoba, Victor A. Schanz, Gabriel Schaumann, Derek W. Schmidt, Adam Sefkow, Tsutomu Shimada, Terry N. Taddeucci, Alexandra Tebartz, Sven C. Vogel, Erik Vold, Glen A. Wurden, Lin Yin	Laser-plasmas in the relativistic-transparency regime: Science and applications	Physics of Plasmas, Vol. 24, (5), 2017 pp. 056
Kevin Field, Mary Snead, Yukinori Yamamoto, Kurt Terrani,	Handbook of the Materials Properties of FeCrAl Alloys For Nuclear Power Production Applications	ORNL/TM-2017/186, August 2017
K. Gofryk	Unusual thermal behavior of uranium dioxide fuel	Invited talk at the 47èmes Journées des Actinides, Karpacz, Poland, March 2017. KG also chaired one of the scientific sessions during the meeting.
Yu. M. Golovchenko	Some Results of Developments and Investigations of Fuel Pins with Metal Fuel for Heterogeneous Core of Fast Reactors of the BN-Type	Energy Procedia 7 (2011)
Jason M. Harp, Steven L. Hayes, Pavel G. Medvedev, Douglas L. Porter, and Luca Capriotti	Testing Fast Reactor Fuels in a Thermal Reactor: A Comparison Report	INL/EXT-17-41677 (NTRD-FUEL-2017-000148), Rev. 0, September 30, 2017
Jason M. Harp, Douglas L. Porter, Brandon D. Miller, Tammy L. Trowbridge, William J. Carmack	Scanning electron microscopy examination of a Fast Flux Test Facility irradiated U-10Zr fuel cross section clad with HT-9	Journal of Nuclear Materials, Volume 494, 2017, Pages 227-239, ISSN 0022-3115, http://dx.doi.org/10.1016/j.jnucmat.2017.07.040

Author (s)	Title	Publication
Lingfeng He, Jason M. Harp, Rita E. Hoggan, Adrian R. Wagner	Microstructure studies of interdiffusion behavior of USi/Zircaloy-4 at 800 and 1000 °C	Journal of Nuclear Materials, Volume 486, 2017, Pages 274-282, ISSN 0022-3115, http://dx.doi.org/10.1016/j.jnucmat.2017.01.035[dx.doi.org]
Connie M. Hill, John D. Bess, and Nicolas E. Woolstenhulme	Neutronics Analysis of TREAT Multi-SERTTA Calibration Test Vehicle (Multi-SERTTA CAL)	ANS Meeting June 2017, San Francisco
Rita Hoggan, Jason Harp, Lingfeng He	Interdiffusion Behavior of U_3Si_2 and FeCrAl via Diffusion Couple Studies	Transactions of the American Nuclear Society, vol. 116, 2017
X. Hu, C. Ang, G. Singh, Y. Katoh	Technique development for modulus, microcracking, hermeticity, and coating evaluation capability characterization of SiC/SiC tubes	ORNL/TM-2016/372, Oak Ridge National Laboratory, Oak Ridge, TN (United States), 2016
J. Isler, J. Zhang, R. Mariani, C. Unal	Experimental Solubility Measurements of Lanthanides on Liquid Alkalis	Submitted to Journal of Nuclear Materials, August 2017.
D. E. Janney and B. H. Sencer	Microstructure changes caused by annealing of U-Pu-Zr alloys	Journal of Nuclear Materials 486: 66-69, 2017(This publication was intended in part to provide a convenient way to cite previously unpublished data in the Handbook.)
Colby B. Jensen, Nicolas E. Woolsten-hulme, Daniel M. Wachs	Fuel-Coolant Interaction Results for High Energy In-Pile LWR Fuels Experiments	Proceedings of Water Reactor Fuel Performance Meeting 2017, Sept. 10-14, 2017, Jeju Island, South Korea.
Zeses E. Karoutas, Raymond Schneider, N. Reed LaBarge, Javier Romero and Peng Xu	Westinghouse Accident Tolerant Fuel Plant Benefits	2017 Water Reactor Fuel Performance Meeting, September 10 -14, 2017, Jeju Island, South Korea
Hesham Khalifa, Christian Deck, George Jacobsen, Jon Sheeder, Jiping Zhang, Carlos Bacalski, Josh Stone, Chunghao Shih, Gokul Vasudevamurthy, Herb Shatoff, Xinyu Huang, Jingjing Bao, Richard Jacko, Edward Lahoda and Christina Back	Development of SiC-SiC Composite Accident Tolerant Fuel Cladding	ICACC, Daytona Beach, FL, January, 2017
T. Koyanagi, Y. Katoh, G. Singh, M. Snead	SiC/SiC Cladding Materials Properties Handbook	ORNL/TM-2017/385, Oak Ridge National Laboratory, Oak Ridge, TN (United States), 2017
Xiang Li, Adib Samin, Cetin Unal, and Robert Mariani	Ab-initio molecular dynamics study of lanthanides in liquid sodium	Journal of Nuclear Materials, 484, 2017 98-192, July 2017

Author (s)	Title	Publication
C. Matthews, J. Galloway, C. Unal	Advanced Simulation Aided Metallic Fuel Design	ANS 2017 Summer Meeting, San Francisco, June 11-15, 2017, LA-UR-17-2044, January 13, 2017
C. Matthews, J. Galloway, C. Unal, S. Novascone, R. Williamson	BISON for Metallic Fuels Modeling	International Conference on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development (FR17), Yekaterinburg, Russian Federation, 26 – 29 June 2017, IAEA-CN-245-366
C. Matthews, C. Unal, J. Galloway, D. D. Keiser, Jr., S. Hayes	Fuel Clad Chemical Interaction in U-PU-ZR; A Critical Review	Submitted to Nuclear Technology, September, January 2017.
Pavel Medvedev, Steven Hayes, Samuel Bays, Stephen Novascone, Luca Capriotti	Testing Fast Reactor Fuels in a Thermal Reactor	Submitted to Nuclear Engineering and Design, June 2017
Yinbin Miao, Jason Harp, Kun Mo, Shaofei Zhu, Tiankai Yao, Abdellatif M. Yacout	Bubble morphology in U_3Si_2 implanted by high-energy Xe ions at 300 °C	Journal of Nuclear Materials, Available online 2 August 2017, ISSN 0022-3115, https://doi.org/10.1016/j.jnucmat.2017.07.066 [doi.org].
Yinbin Miao, Jason Harp, Kun Mo, Sumit Bhattacharya, Peter Baldo, Abdellatif M. Yacout	In-situ TEM ion irradiation investigations on U_3Si_2 at LWR temperatures	Journal of Nuclear Materials, Available online 21 November 2016, http://dx.doi.org/10.1016/j.jnucmat.2016.11.020[dx.doi.org].
Simon Middleburgh, Ed Lahoda, Karol Luszck, Robin Grimes, David Andersson, Chris Stanek and Ted Besmann	Ongoing work on modelling of UN-U_3Si_2 fuel	ICACC, Daytona Beach, FL, January, 2017
Robert Oelrich, Sumit Ray, Zeses Karoutas, Ed Lahoda, Frank Boylan, Peng Xu, Javier Romero, Hemant Shah	Overview of Westinghouse Lead Accident Tolerant Fuel Program	2017 Water Reactor Fuel Performance Meeting, September 10 -14, 2017, Jeju Island, South Korea
S. Raiman, P. Doyle, C. Ang, K. Terrani	Hydrothermal Corrosion of SiC Materials for Accident Tolerant Fuel Cladding with and Without Mitigation Coatings	Proceedings of the 18th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors (2017)

Author (s)	Title	Publication
R. B. Rebak, W. P. Gassmann, and K. A. Terrani	Managing Nuclear Power Plant Safety with FeCrAl Alloy Fuel Cladding	Paper A0042, IAEA, Top Safe 2017, 12-16 February 2017, Vienna, Austria
R. B. Rebak, Y.-J. Kim, J. Gynnerstedt, K. A. Terrani, and R. E. Stachowski	Fabrication of FeCrAl Cladding for Accident Tolerant Fuel	Top Fuel 2016, September 11-15, 2016, Boise, ID
R. B. Rebak, M. Larsen, and Y.-J. Kim	Characterization of oxides formed on iron-chromium-aluminum alloy in simulated light water reactor environments	Corrosion Reviews 2017 DOI 10.1515/corrrev-2017-0011
R. B. Rebak, K. A. Terrani, W. P. Gassmann, J. B. Williams, K. L. Ledford	Improving Nuclear Power Plant Safety with FeCrAl Alloy Fuel Cladding	MRS Advances, 2017 Materials Research Society, DOI: 10.1557/adv.2017.5
J. Romero, W. A. Byers, G. Wang, A. Mueller, Z. Karoutas	Simulated Severe Accident Testing for Evaluation of Accident Tolerant Fuel	2017 Water Reactor Fuel Performance Meeting, September 10 -14, 2017, Jeju Island, South Korea
Markus Roth, Sven C Vogel, Mark Andrew M Bourke, Juan Carlos Fernandez, Michael Jeffrey Mocko, Siegfried Glenzer, Wim Leemans, Craig Siders, Constantin Haefner	Assessment of Laser-Driven Pulsed Neutron Sources for Poolside Neutron-based Advanced NDE–A Pathway to LANSCE-like Characterization at INL	Publication date: 04/19/2017 Issue: LA--UR-17-23190
Saleh, T.A., Romero, T.J, Quintana, M.E, Field, K.J.	Mechanical Properties of HFIR Irradiated FeCrAl Alloys	NTR&D milestone report NTRD-FUEL-2017-000006, LA-UR-17-28992
Raymond Schneider, N. Reed LaBarge, Hans Van De Berg, Martin Van Haltern, Edward Lahoda and Zeses Karoutas	Estimating the Benefits of Accident Tolerant Fuel (ATF)	PSA 2017, September 24-28, 2017, Pittsburgh, PA
M. Schuster, C. J. Crawford, R. B. Rebak	Thermal Shock Resistance of FeCrAl Alloys for Accident Tolerant Fuel Cladding Application	Paper NACE-2017-8900, Corrosion/2017, 26-30 March, New Orleans, Louisiana, USA
H. Shah, J. Romero, P. Xu, B. Maier, G. Johnson, J. Walters, T. Dabney, H. Yeom, K. Sridharan	Development of Surface Coatings for Enhanced Accident Tolerant Fuel	2017 Water Reactor Fuel Performance Meeting, September 10 -14, 2017, Jeju Island, South Korea
G. Singh, S. Gonczy, E. Lara-Curzio and Y. Katoh	Interlaboratory Round Robin Axial Tensile Testing of Tubular SiC/SiC Specimens	ORNL/SR-2017/397, Oak Ridge National Laboratory, Oak Ridge, TN (United States), 2017
R. E. Stachowski, R. B. Rebak, W. P. Gassmann, J. Williams	Progress of GE Development of Accident Tolerant Fuel FeCrAl Cladding	Top Fuel 2016, Boise, ID, September 11-15, 2016

Author (s)	Title	Publication
N. Stauff, T. K. Kim, S. Hayes	Tradeoff Study of Advanced Transmutation Fuels in Sodium-cooled Fast Reactors	International Conference on Fast Reactors and Related Fuel Cycles: FR-17, June 2017, Yekaterinburg, Russian Federation
G.N. Stevens, C. Unal, J. Galloway, C. Matthews	Progressively informed calibration of BISON nuclear fuel models	2017 ASME V&V Workshop, Las Vegas, NV, May 3-5, 2017, LA-UR-17-23571
Zhiqian Sun, Xiang Chen, Yukinori Yamamoto,	Examination of Powder Metallurgy vs. Induction Melting for FeCrAl Alloy Production	ORNL/TM-2017/381, July 2017
Zhiqian Sun, Yukinori Yamamoto,	Processability evaluation of a Mo-containing FeCrAl alloy for seamless thin-wall tube fabrication	Materials Science and Engineering A. Vol. 700 Pages 554 - 561(July 2017)
Zhiqian Sun, Yukinori Yamamoto,	Microstructural control of FeCrAl alloys using Mo and Nb additions	Materials Characterization, Vol. 132 Pages 126 - 131(August 2017)
C. Unal, C. Matthews, X. Li, J. Isler, J. Zhang, J. Galloway	A potential mechanism for lanthanide transport in metallic fuels	ANS 2017 Summer Meeting, San Francisco, June 11-15, 2017, LA-UR-17-20083
C. Unal, L. Xiang, J. Isler, C. Matthews, S. Abid, J. Zhang, J. Galloway, R. Mariani	Modeling of Lanthanide Transport in Metallic Fuels: Recent Progresses	International Conference on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development (FR17), Yekaterinburg, Russian Federation, June 26-29 2017, IAEA-CN-245-350, LA-UR-17-20106
J. Wang et al.	Accident tolerant clad material modeling by MELCOR: Benchmark for SURRY Short Term Station Black Out	Nuclear Engineering and Design 313 (2017) 458-469
J. Wang, M.B. Toloczko, N. Bailey, F.A. Garner, J. Gigax, L. Shao	Modification of SRIM-calculated dose and injected ion profiles due to sputtering, injected ion buildup and void swelling	Nuclear Instruments and Methods in Physics Research B, 387 (2016), 20-28.
J. Wang, M.B. Toloczko, K. Kruska, D.K. Schreiber, D.J. Edwards, Z. Zhu, J. Zhang	Carbon Contamination During Ion Irradiation - Accurate Detection and Characterization of its Effect on Microstructure of Ferritic/Martensitic Steels	Scientific Reports, 2017, In Press.

Author (s)	Title	Publication
Yuzhou Wang, David H. Hurley, Erik P. Luther, Miles F. Beaux II, Douglas R. Vodnik, Reuben J. Peterson, Bryan L. Bennett, Igor O. Usov, Pengyu Yuan, Xinwei Wang, Marat Khafizov	Ultralow Thermal Conductivity Pyrolytic Carbon Coating for Thermal Management Applications	Submitted to Carbon on July 25th
E. Sooby Wood, J.T. White, and A.T. Nelson	Oxidation behavior of U-Si compounds in air from 25 to 1000C.	Journal of Nuclear Materials 484 (2017) 245-257
E. S. Wood, J.T. White, and A.T. Nelson	The effect of aluminum additions on the oxidation resistance of U_3Si_2	Journal of Nuclear Materials 489 (2017) 84-90
Peng Xu, Ed Lahoda, Richard Jacko, Frank Boylan, and Robert Oelrich	Status of Westinghouse SiC Composite Cladding Fuel Development	2017 Water Reactor Fuel Performance Meeting, September 10 -14, 2017, Jeju Island, South Korea
Peng Xu, Ed Lahoda and Yun Long	Westinghouse Accident Tolerant Fuel Program Update on SiC Composite Cladding Development	ICACC, Daytona Beach, FL, January, 2017
Yukinori Yamamoto, Zhiqian Sun	Quality Optimization of Commercial FeCrAl Tube Production	ORNL/TM-2017/338, June 2017
E. Zapata-Solvas, S.R.G. Christopoulos, N. Ni, D.C. Parfitt, D. Horlait, M.E. Fritzpack, A. Chroneos, W.E. Lee, “,”	Experimental synthesis and DFT investigations of Zr3(Al1-x,Six)C2 MAX phases	Journal of the American Ceramic Society 100, 1377 (2017)
E. Zapata-Solvas, M.A. Hadi, D. Horlait, D.C. Parfitt, A. Thibaud, A. Chroneos, W.E. Lee	Synthesis and physical properties of (Zr1-x,Tix)3AlC2 MAX phases	Journal of the American Ceramic Society 100, 3393 (2017)

4.2 FY-17 LEVEL 2 MILESTONES

Work Package Title	Site	Work Package Manager	FY-16 Level 2 Milestone
AFC Campaign Management - INL	INL	Beverly, Ed	Establish a uniform framework for fuel system handbooks
ATF-2 ATR Loop Design - INL	INL	Hoggard, Gary	ATF-2 loop final design approval (design review closeout)
ATF-1 Irradiation Testing in ATR - INL	INL	Core, Greg	Ship First Laboratory-led ATF-1 concept capsule(s) from ATR to MFC to Support PIE initiation
Static Capsule Irradiation Device Prototype - INL	INL	Schulthess, Jason	Conduct Multi-SERTTA-CAL final design review to support prototype fabrication and testing
ATF-2 ATR Loop Design - INL	INL	Hoggard, Gary	Sensor Qualification Test ready to begin ATR Installation process
ATF-1 Irradiation Testing in ATR - INL	INL	Core, Greg	Issue Final report documenting irradiation status of all ATF-1capsules
ATF-2 ATR Loop Design - INL	INL	Hoggard, Gary	ATF-2 fueled test train ready for final assembly
Static Capsule Irradiation Device Prototype - INL	INL	Schulthess, Jason	Multi-SERTTA baseline module engineering package qualified for use in TREAT .
ATF-2 ATR Loop Design - INL	INL	Hoggard, Gary	ATF-2 fueled test train ready for insertion in ATR Cycle 164-A
Feedstock Preparation/Purification - INL	INL	Squires, Leah	Complete Preparation of Metallic Am Feedstock Needed for AmBB
Fuel Characterizations - INL	INL	Papesch, Cynthia	Update Advanced Reactor Fuels Handbook
Advanced Fabrication Development - INL	INL	Fielding, Randy	Demonstrate Remote Casting of Metallic Fuels (in synergy with JFCS)
PIE and Analyses - INL	INL	Harp, Jason	Initiate Advanced PIE Studies on Metallic Fuels (EPMA of irradiated fuel in IMCL)
PIE and Analyses - INL	INL	Harp, Jason	Issue Rev 0 Comparison Report on Fast vs. Thermal Spectrum Testing (FUTURIX-FTA compositions)
Irradiation Testing in ATR - INL	INL	Dempsey, Doug	Complete Conceptual Design of Advanced Coatings/ Liners Experiment (AFC-4F),
Advanced Instrument Development - INL	INL	Hurley, Dave	Enter TCM into stage one mockup and complete initial testing
Advanced High Density Fuel Development and Qualification (Advanced Ceramic Fuel Development) - LANL	LANL	Nelson, Andy	Issue Draft U_3Si_2 Fuel Property Handbook

Work Package Title	Site	Work Package Manager	FY-16 Level 2 Milestone
Advanced High Density Fuel Development and Qualification (Advanced Ceramic Fuel Development) - LANL	LANL	Nelson, Andy	Development of Strategies to Relax Handling Constraints for High Density Fuels
SE testing and High Density Fuel Fabrication for Halden (ATF-1 Fabrication) - LANL	LANL	Nelson, Andy	Separate Effects Study of Oxygen on Fresh Fuel Properties of U_3Si_2
LWR Neutron Irradiated Materials Testing - LANL	LANL	Saleh, Tarik	Report on microstructure and mechanical properties of HFIR irradiated Gen II FeCrAl alloys
High Dose Materials Testing for FR - LANL	LANL	Saleh, Tarik	Test and Report on Ring pull testing of BOR-60 Irradiated ODS tube material
Fuel Modeling Support - LANL	LANL	Unal, Cetin	Report documenting assessment of the coupled thermal-mechanicaldiffusion models in BISON for metallic fuel constituent redistribution
ATF SiC Cladding Development - ORNL	ORNL	Katoh, Yutai	PIE of nuclear grade SiC/SiC flexural coupons irradiated to 10 dpa at LWR temperature
ATF SiC Cladding Development - ORNL	ORNL	Katoh, Yutai	Determination of He and D permeability of neutron-irradiated SiC tubes
Microstructure Analysis and Evolution in Ceramic Fuels - ORNL	ORNL	Terrani, Kurt	Complete detailed post-fabrication characterization of uranium bearing ceramics
Transient testing Coordination and testing - ORNL	ORNL	Linton, Kory	Insert Severe Accident Test Station into the hot-cell and perform demonstration test

4.3 AFC NEUP GRANTS

Active Projects Awarded in 2012

Nuclear Energy University Project Grants

Lead University	Title	Principle Investigator
Ohio State University	Testing of Sapphire Optical Fiber and Sensors in Intense Radiation Fields, when subjected to very high temperatures	Raymond Cao
University of Tennessee	Better Radiation Response and Accident Tolerance of Nanostructured Ceramic Fuel Materials	Yanwen Zhang

Active Projects Awarded in 2013

Nuclear Energy University Project Grants

Lead University	Title	Principle Investigator
University of California, Berkeley	Developing Ultra-Small Scale Mechanical Testing Methods and Microstructural Investigation Procedures for Irradiated Materials	Peter Hosemann
University of Florida	Innovative Coating of Nanostructured Vanadium Carbide on the F/M Cladding Tube Inner Surface for Mitigating the Fuel Cladding Chemical Interactions	Yong Yang (Finished but no final report)
University of South Carolina	U_3Si_2 Fabrication and Testing for Implementation into the BISON Fuel Performance Code	Travis Knight
Utah State University	Optical Fiber Based System for Multiple Thermophysical Properties for Glove Box, Hot Cell and In-pile Applications	Heng Ban (Finished but no finalreport)
Iowa State University	In-pile Thermal Conductivity Characterization with Single-laser Heating/Time Resolved Raman	Xinwei Wang
Arizona State University	Mechanical Behavior of UO_2 at Sub-grain Length Scales: Quantification of Elastic, Plastic and Creep Properties via Microscale Testing	Pedro Peralta

Active Projects Awarded in 2014

Nuclear Energy University Project Grants

Lead University	Title	Principle Investigator
Ohio State University	Studies of Lanthanide Transport in Metallic Nuclear Fuels	Jinsuo Zhang
Texas A&M University	Development of High-Performance ODS Alloys	Lin Shao (September 30, 2017)
University of Arkansas	Computational and Experimental Studies of Microstructure-Scale Porosity in Metallic Fuels for Improved Gas Swelling Behavior	Paul Millett (September 30, 2017)
University of Notre Dame	Assessment of Corrosion Resistance of Promising Accident Tolerant Fuel Cladding Under Reactor Conditions	David Bartels
University of Tennessee	Enhanced Accident-tolerant Fuel Performance and Reliability for Aggressive iPWR/SMR Operation	Ivan Maldonado
University of Wisconsin, Madison	Development of Self-Healing Zirconium-Silicide Coatings for Improved Performance of Zirconium-Alloy Fuel Cladding	Kumar Sridharan
Virginia Polytechnic Institute and State University	Thermal Conductivity in Metallic Fuels	Celine Hin
Virginia Polytechnic Institute and State University	SiC-ODS Alloy Gradient Nanocomposites as Novel Cladding Materials	Kathy Lu

Active Projects Awarded in 2015

Nuclear Energy University Project Grants

Lead University	Title	Principle Investigator
Northwestern University	Electrically-Assisted Tubing Processes for Enhancing Manufacturability of Oxide Dispersion Strengthened Structural Materials for Nuclear Reactor Applications	Jian Cao
Massachusetts Institute of Technology	Multilayer Composite Fuel Cladding for LWR Performance Enhancement and Severe Accident Tolerance	Michael Short
University of Wisconsin, Madison	Radiation-Induced Swelling and Micro-Cracking in SiC Cladding for LWRs	Izabela Szlufarska
University of California, Berkeley	Developing a Macro-Scale SiC-cladding Behavior Model Based on Localized Mechanical and Thermal Property Evaluation on Pre- and Post-irradiation SiC-SiC Composites.	Peter Hosemann
Massachusetts Institute of Technology - IRP	Development of Accident Tolerant Fuel Options for Near Term Applications	Jacopo Buongiorno
University of Tennessee at Knoxville	Radiation Effects on High Thermal Conductivity Fuels	Steven Zinkle

Active Projects Awarded in 2016

Nuclear Energy University Project Grants

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

Active Projects Awarded in 2017

Nuclear Energy University Project Grants

Lead University	Title	Principle Investigator
University of Wisconsin-Madison	Extreme Performance High Entropy Alloys (HEAs) Cladding for Fast Reactor Applications	Adrien Couet
University of Wisconsin-Madison	Critical Heat Flux Studies for Innovative Accident Tolerant Fuel Cladding Surfaces	Michael Corradini
Colorado School of Mines	Development of Advanced High-Cr Ferritic/ Martensitic Steels	Kester Clarke
Massachusetts Institute of Technology	Determination of Critical Heat Flux and Leidenfrost Temperature on Candidate Accident Tolerant Fuel Materials	Matteo Bucci
University of New Mexico	An Experimental and Analytical Investigation into Critical Heat Flux (CHF) Implications for Accident Tolerant Fuel (ATF) Concepts	Youho Lee
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	Sama Bilbao y Leon
University of New Mexico	Nanostructured Composite Alloys for Extreme Environments	Osman Anderoglu

4.5 ACRONYMS

ABR .... Advanced Burner Reactor

ACTOF .... Accident Tolerant Fuels for LWRs

AFC .... Advanced Fuels Campaign

AmBB .... Americium Bearing Blanket

APIE .... Advanced Post-Irradiation Examination

ANL .... Argonne National Laboratory

APM .... Advanced Powder Metallurgy

APS .... Arizona Power Station

APT .... Atom Probe Tomography

Ar .... Argon

ARF .... Advanced Reactor Fuel

ASTM .... American Society for Testing and Materials

ATF .... Accident Tolerant Fuel

ATP .... Acceptance Test Plan

ATP .... Accident Test Plan

ATR .... Advanced Test Reactor

BARC .... Bhabha Atomic Research Centre

BCC .... Body Centered Cubic

BDBA .... Beyond Design Basis Accident

BEA .... Batelle Energy Alliance

BWR .... Boilin Water Reactor

BNL .... Brookhaven National Laboratory

CDC .... Capsule Driver Core

CDF .... Cumulative Damage Factor

CEA .... Commissariat à l' Énergie Atomique

CNWG .... Civil Nuclear Energy Research and Development Working Group

CRIEPI .... Central Research Institute of Electric Power

CRP .... Coordinated Research Prokject

CT .... Computerized Tomography

CWRU .... Case Western Reserve University

CVI .... Chemical Vapor Infiltration

DBA .... Design Basis Accident

DIC .... Digital Image Correlation

DOE .... Department of Energy

dpa .... Displacements Per Atom

DPA .... Irradiation Damage

DSA .... Dynamic Strain Aging

EATF .... Enhanced Accident Tolerant Fuel

EBR .... Experimental Breeder Reactor

EBSD .... Electron Backscatter Diffraction

ECCS .... Emergency Core Cooling System

ECD .... Electrochemical Deposition

EdF .... Electricite de France

EDS .... Energy Dispersive X-Ray Spectroscopy

EGATFL .... Expert Group on Accident Tolerant Fuel

EML .... Electron Miscroscopy Laboratory

EPMA .... Electron Probe Microanalysis

EPRI .... Electric Power Research Institute

EQP .... Equipment Qualification Plan

ERNI .... Energy-resolved Neutron Imaging

FAS .... Field Assisted Sintering

FCC .... Face Centered Cubic

FCCI .... Fuel Cladding Chemical Interaction

FCI .... Fuel-Coolant Interaction

FCRD .... Fuel Cycle Research and Development

FFTF .... Fast Flux Test Facility

FIB .... Focused Ion Beam

FMF .... Fuel Manufacturing Facility

FPTTWG .... Fuel Performance and Technology Technical Writing Group

FS .... Flash Sintering

FSW .... Friction Stir Welding

GA .... General Atomics

GACS .... Glovebox Advanced Casting System

GE .... General Electric

GEH .... GE-Hitachi

GNF .... Global Nuclear Fuel

GUI .... Graphical User Interface

HFEF .... Hot Fuel Examination Facility

HFIR .... High Flux Isotope Reactor

HFP .... Hot Full Power

HIPPO .... High Pressure Preferred Orientation

HRR .... Horn Rapids Road

HZP .... Hot Zero Power

IAC .... Industrial Advisory Committee

IAEA .... International Atomic Energy Agency

IFE .... Institute for Energy Technology

IFEL .... Irradiated Fuels Examination Laboratory

IMCL .... Irradiated Materials Characterization Laboratory

INL .... Idaho National Laboratory

IRP .... Integrated Research Project

JAEA .... Japan Atomic Energy Agency

JRC-ITU .... Joint Research Centre-Institute for Transuranium Elements of the European Commission

KAERI .... Korean Atomic Energy Research Institute

KIT .... Karlsruhe Institute of Technology

LAMDA .... Low Activation Materials Development and Analysis

LANL .... Los Alamos National Laboratory

LBNL .... Lawrence Berkeley National Lab

LED .... Light-Emitting Diodes

LFA .... Lead Fuel Assembly

LFR .... Lead Fuel Rods

LLNL .... Lawrence Livermore National Laboratory

LOCA .... Loss of Coolant Accident

LTA .... Lead Test Assemblies

LTR .... Lead Test Rod

LWR .... Light Water Reactor

MA .... Minor Actinide

MA-MOX .... Minor Actinide – Mixed Oxide

MATLAB .... Matrix Laboratory

MCNP .... Monte Carlo

MFC .... Materials and Fuels Complex

MFCI .... Molten Fuel Coolant Interaction

MIT .... Massachusetts Institute of Technology

MOX .... Mixed Oxide

MPa .... Megapascal

Multi-SERTTA............................Multiple Capsule Design Static Environment Rodlet Transient Test Apparatus

NE..Nuclear Energy

NEA .... Nuclear Energy Agency

NEAMS ....Nuclear Energy Advanced Modeling and Simulation

NEUP.. Nuclear Energy University Project

NNL...National Nuclear Laboratory

NRC...Nuclear Regulatory Commission

NTD...National Technical Director

NTRD ....Nuclear Technology Research & Development

NRTA .... Nuclear Resonance Transmission Analysis

ODS ....Oxide Dispersion Strengthened

OECD... Organization for Economic Cooperation and Development

ORNL .... Oak Ridge National Laboratory

PBF .... Power Burst Facility

PCF .... Power Coupling Factors

PCT...Peak Clad Temperature

PI ....Principal Investigator

PIE .... Post-irradiation Examination

PNNL...Pacific Northwest National Laboratory

PSU..Pennsylvania State University

PVD ....Physical Vapor Deposition

PWR ....Pressurized Water Reactor

R&D...Research and Development

RIA ....Reactivity-Initiated Accident

RCM .... Research Coordination Meeting

RPI..Rensselaer Polytechnic Institute

RUS .... Resonant Ultrasound Spectroscopy

SAR .... Severe Accident Test Station

SCS .... Stoichiometric Control System

SEM .... Scanning Electron Microscopy

SERTTA .... Static Environment Rodlet Transient Test Apparatus

SFR .... Sodium Fast Reactor

SiC .... Silicon Carbide

SLAC .... Stanford Linear Accelerator Center

SPS .... Spark Plasma Sintering

STA .... Simultaneous Thermal Analyzer

STEM .... Scanning Transmission Electron Microscopy

Super-SERTTA .... Single-Vessel Version Static Environment Rodlet Transient Test Apparatus

SZ .... Stir Zone

TAMU .... Texas A&M University

TBG .... Transuranic Breakout Glovebox

TCM .... Thermal Conductivity Microscope

TEM .... Transmission Electron Microscopy

TCD .... Thermal Conductivity Degradation

TCM .... Thermal Conductivity Microscope

TD .... Theoretical Density

TF .... Task Force

TMP .... Thermomechanical Processing

TGA .... Thermogravimetric Analyzer

TREAT .... Transient Reactor Test Facility

TRU .... Transuranic

TWERL .... TREAT Water Environment Recirculating Loop

UAZ .... Unaffected Zone

UCB .... University of California Berkeley

UCSB .... University of California Santa Barbara

UF .... University of Florida

USW .... Upset Shape Welding

UW .... University of Wisconsin

VNIINM .... A.A. Bochvar Institute

VPS .... Vacuum Plasma Spray

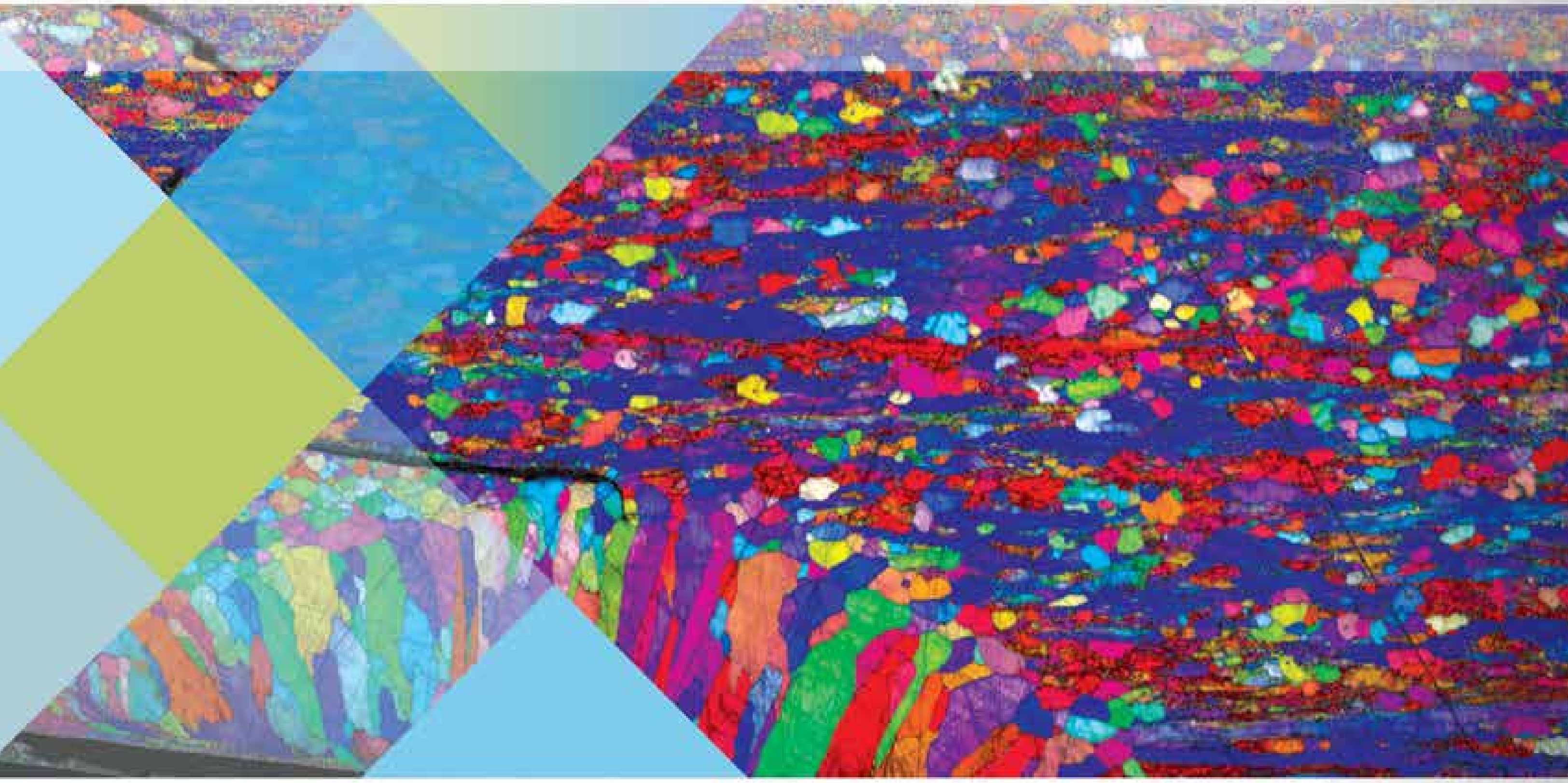
VTR .... Versatile Test Reactor

WQ .... Water Quenching

XRD .... X-Ray Diffraction

YMR .... Yucca Mountain Repository

YS .... Yield Strength



Advanced Fuels Campaign

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