INL/RPT-23-74940-Revision-0

AFC Metallic Fuel Research and Development 5-Year Plan

March 2024

Changing the World's Energy Future

Colby B Jensen, Douglas L Porter

Idaho National Laboratory

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http://www.inl.gov

Prepared for the
U.S. Department of Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517, DE-AC07-05ID14517

INL/RPT-23-74940
Revision 0

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Advanced Fuels Campaign

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ABSTRACT

The mission of the U.S. Department of Energy (DOE) Advanced Fuel Campaign (AFC) program is conducting R&D on nuclear fuel technology that enables near- and long-term implementation of the reactor systems necessary to meet national nuclear energy objectives. Its primary goals align with goals of the DOE Office of Nuclear Energy in sustaining the current LWR fleet through Accident Tolerant Fuels program and Enabling Advanced Reactors. The latter goal is to be achieved through the following subgoals in order of logistical priority:

- Establish the qualification basis for reference metallic fuel designs for sodium-cooled fast reactors.
- Develop next generation metallic fuel fabrication and design for improved fissile utilization and management.
- Develop accelerated fuel development and qualification methodologies.
- Identify next generation fuel technologies.

The purpose of this document is to serve as a five-year research and development (R&D) plan to achieve the goals identified to support enabling advanced reactor deployment related to metallic fuels starting in 2025. The specific objectives of this document are to:

- align program R&D work across technical areas, national laboratories, and with stakeholder interests, and
- aid yearly and outyear scope and budgetary planning activities.

This plan lays the foundation of *databases, capabilities, expertise, and research-commercial-regulatory integration* for launching *next-generation initiatives in advanced fuel technologies (fuel and cladding) and improved methodologies for achieving accelerated qualification of next generation fuel technologies*.

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ACRONYMS

AFC	Advanced Fuels Campaign
ANL	Argonne national Laboratory
AOO	Anticipated operational occurrence
ARES	Advanced Reactor Experiments for Sodium fast reactor fuels
ASME	American Society of Mechanical Engineers
ATOMIC	Accelerated Testing of Materials in Capsules
ATR	Advanced Test Reactor
ATWS	Anticipated Transient Without Scram
BDBA	Beyond design basis accident
BEAST	Boosted Energy Advanced Spectrum Test
BAF	Blister Anneal Furnace
BSE	Backscatter electron
CDF	Cumulative damage fraction
DBA	Design basis accident
DISECT	Disc Irradiation for Separate Effects Testing with Control of Temperature
DOE	Department of Energy
DSC	Differential scanning calorimetry
EBR-II	Experimental Breeder Reactor-II
EOL	End of life
EPMA	Electron probe microanalysis
FAST	Fission-Accelerated Steady-state Testing
FBTA	Fuel Behavior Test Apparatus
FCRD	Fuel Cycle Research & Development
FCCI	Fuel-cladding chemical interaction
FCMI	Fuel-cladding mechanical interaction
FFTF	Fast Flux Test Facility
FIPD	Fuels Irradiation & Physics Database
FGR	Fission gas release
HALEU	High assay low enriched uranium
HFEF	Hot Fuel Examination Facility

IET	Integral effects test
IFR	Integral Fast Reactor
IMCL	Irradiated Materials Characterization Facility
IMIS	IFR Material Information System
IMPACT	Irradiated Materials Properties Accelerated Characterization Test
INL	Idaho National Laboratory
JAEA	Japan Atomic Energy Agency
LDRD	Laboratory-Directed Research and Development
LOF	Loss of flow
LOHS	Loss of heat sink
LTR	Lead test rod
LWR	Light Water Reactor
MOOSE	Multiphysics Object-Oriented Simulation Environment
MOX	Mixed Oxide
MSTL	Modular Sodium Test Loop
MTR	Material test reactor
NEAMS	Nuclear Energy Advanced Modeling and Simulation
NQA	Nuclear Quality Assurance
NRC	Nuclear Regulatory Commission
NSUF	Nuclear Science User Facility
ODS	Oxide dispersion multiphysics
PICT	Peak internal cladding temperature
PIE	Post-irradiation examination
PIRT	Phenomena identification and ranking table
PNNL	Pacific Northwest National Laboratory
QL	Quality level
RAI	Requests for additional information
R&D	Research and development
SATS	Severe Accident Test Station
SEM	Scanning electron microscopy
SET	Separate effects test

SFR	Sodium fast reactor
SQA	Software quality assurance
TEM	Transmission electron spectroscopy
THOR	Transient Heatsink Overpower Response
THOR-C	THOR-Commissioning
THOR-M	THOR-Metal
THOR-MOXTOP	THOR-Mixed Oxide TOP
TOP	Transient overpower
TR	Topical Report
TREAT	Transient Reactor Test Facility
TRU	Transuranic
WPF	Whole Pin Furnace

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AFC Metallic Fuel Research and Development 5-Year Plan

1 Introduction and Purpose

The mission of the U.S. Department of Energy (DOE) Advanced Fuel Campaign (AFC) program is conducting research and development (R&D) on nuclear fuel technology that enables near- and long-term implementation of the reactor systems necessary to meet national nuclear energy objectives. Its primary goals align with goals of the DOE Office of Nuclear Energy in sustaining the current Light Water Reactor(LWR) fleet through Accident Tolerant Fuels program and Enabling Advanced Reactors [1][2]. The latter goal is to be achieved through the following subgoals in order of logistical priority:

- **Establish the qualification basis for reference metallic fuel designs for sodium-cooled fast reactors** (SFR). The reference fuel design targeted by initial activities in the program is sodium bonded U-10Zr and U-20Pu-10Zr alloys with 75% smeared density cladded in HT9 and, as a second priority, advanced austenitic cladding (e.g. CWD9), for pool-type SFR applications. This selection is based on the U.S. experience with these fuels overviewed in [3]. While CWD9 is not currently a primary interest for future activities in the program, its potential utility in areas where only lower neutron exposure (void swelling) is required, and connection with other advanced austenitic claddings developed worldwide, justify capturing a performance basis for it during the planned work to develop a qualification basis from existing information. The highest priority during the next five years is establishing a fuel qualification basis from existing data and knowledge, while ongoing R&D will advance the state-of-the-art to capture new data in identified areas to expand the performance envelope.
- **Develop next-generation metallic fuel fabrication and design for improved fissile utilization and management**. The long-heralded demonstrated promise of fast reactor technology lies in its efficacy in efficient fuel cycle management. Near- and long-term strategies should maintain a focus on full fuel cycle considerations from fabrication economics to irradiation performance to backend considerations for fission product and remaining fissile management. Overarching objectives of longer-term plans in these areas remain under development in the AFC program. Specific planned examples of known R&D priorities in these directions include development of new fabrication approaches with improved economics along with assessment and targeted research of candidate fuel innovations to provide enhanced economic and/or performance benefits. Development of improved high-temperature strength cladding is a clear need to enable higher temperature applications including as launch point for enabling potential benefits of lead-cooled fast reactors and advanced ceramic fuels. Ideally, higher temperature performance would retain low void swelling characteristics for long-life applications. Further research on fabrication and performance of transuranic (TRU) fuels has clear ties to DOE fuel cycle goals, potentially including testing of materials produced in collaboration with the Materials Recovery and Waste Form Program. At the same time, non-traditional applications such as microreactors have applicational interests in potentially different operational windows such as larger fuel pins, lower power densities, and lower temperature applications.
- **Develop accelerated fuel development and qualification methodologies**. These efforts are embedded into R&D activities where applicable and, in some cases, are also under exploration to validate new techniques such as accelerated burnup accumulation testing like the ongoing Fission Accelerated Steady-state Testing (FAST)-01 experiment.
- **Identify next generation fuel technologies**. This goal addresses the need to nurture longer-term fuel technologies to enable next generation applications and potentially new reactor designs. Advanced ceramics and metal alloys and combinations of those (i.e. CER-CER, CER-MET), and other composite materials are examples of potential fuel types of interest.

The purpose of this document is to provide a five-year R&D plan to achieve the goals identified to support enabling advanced reactor deployment related to metallic fuels starting in 2025. The current focus of this plan is on the first program subgoal described above. The plan will be a living document that will likely include minor updates on about a yearly basis. Previously, the AFC program published a document describing the rationale and objectives of the metallic fuel R&D efforts [4]. Its conclusions and general rationale remain in effect for the work described in this plan. One exception is a reduced target on sodium-free fuel, as it should first be proceeded by recent and further techno-economical evaluations to define its prioritization in the program.

The specific objectives of this document are to:

- align program R&D work across technical areas, national laboratories, and with stakeholder interests, and
- aid yearly and outyear scope and budgetary planning activities.

1.1 Background on the Reference Metal Fuel Design

Metallic nuclear fuel materials (metallic fuel or "metal" fuel) have been studied since the foundational beginnings of the atomic age. Achieving maximal fissionable density in nuclear fuels naturally leads to selection of metal fissionable materials, especially important for modern requirements for low enrichments. To date, the most notable publicly available nuclear fuel applications have been in sodium-cooled fast reactors (SFR) in the U.S. However, a wide range of studies and applications have been explored over the past 70 years.

Application of metallic alloys fuels in liquid-metal cooled fast reactors and crucial performance information is well described in literature. Walters et al. performed a comprehensive review of metallic fuel technology in 1984, based largely on extensive experience with the metallic fuel driver core of Experimental Breeder Reactor-2 (EBR-II) as an immediate precursor to the U.S. Department of Energy (DOE) Integral Fast Reactor (IFR) program and the selected demonstration of the General Electric PRISM reactor. Later, Hofman et al. presented an extensive review in 1994 and 1997, summarizing metallic fuel performance in SFR systems based on extensive findings of the IFR program, which abruptly ended in 1994. Just earlier, the NRC published NUREG-1368 [5] and NUREG-1369 [6] providing a regulatory summary of the metallic fuel design assessment for those reactors with some crucial data gaps identified. In addition, the EBR-II Mk-V driver core fuel safety case was documented but never able to be implemented providing a comprehensive design basis for U-10Zr & U-20Pu-10Zr [7].

While DOE ended the IFR program in 1994, its influence has continued strongly over the past 30 years. International and domestic interests in metallic fuels have continued to sprout and grow, many based on the foundation of EBR-II driver core and IFR program work. Countries such as Japan, Korea, India, Russia, and China all have made significant investments in SFR metallic fuel R&D. Notable development has been achieved in Japan [8], Korea [9], and India [10] while Russia has reprioritized its R&D towards nitride fuel [11] and China has more recently taken first steps toward metallic fuel for SFRs [12]. While in the U.S., notable R&D activity was restarted by the U.S. DOE as part of the Advanced Fuel Cycle Initiative with a focus on burning minor actinide fission products about 20 years ago. This program has evolved into the modern Advanced Fuels Campaign with a contemporary focus towards supporting advanced reactor deployments with improved economics and performance in a once-through fuel cycle. Still, minor-actinide burning and fuel recycle applications will remain an important next phase priority for commercial interests and an important goal for the U.S DOE for managing the nuclear fuel cycle [4]. Also from this viewpoint, it is crucial to capture ternary fuel performance during initial phases of this R&D plan based on an extensive existing experience base and datasets.

Over the past 20 years, R&D in several topics has already provided unique insights that should be incorporated into a state-of-the-art assessment of metallic fuel performance. Some special developments of note include preservation and examinations of the prototypic length Fast Flux Test Facility (FFTF) MFF experiments never completed historically [13][14], experiments in the Advanced Test Reactor (ATR) leveraging the fast-thermal spectral effects, characterization done with the FUTURIX experiment in the Phenix reactor [15], restart of the TREAT facility and SFR testing capabilities [16], modern PIE applied to legacy and recent irradiated fuels [17], modern modeling & simulation development especially leveraging multi-scale capabilities [18], and the fast reactor database development documenting and qualifying irradiation of metallic fuels in EBR-II and FFTF [19]. Some of these efforts are ongoing and further described in sections of this document.

Over the past decade, multiple assessments have also been made to evaluate the state of metallic fuel assembly technology through Phenomena Identification and Ranking Table (PIRT) and expert based gap assessments. A PIRT study was published in 2011 addressing all aspects of the fuel concluding “an SFR could be designed and licensed based upon the technology base developed from the successful operation of EBR-II and FFTF,” constrained to a specified set of design limits [20]. More recently, multiple studies have provided analysis of metallic fuel research needs. The University of Florida developed a report of research needs from an expert-based workshop [21]. Williams et al. performed applied PIRT to the U-Zr system to prioritize future research directions specifically for the swelling and constituent redistribution phenomena and to identify the most influential source variables impacting these phenomena [22]. Beausoleil et al. used PIRT methodology to evaluate an annular geometry U-Pu-Zr design to provide recommendations for an R&D approach for mechanistic understanding [23]. In the past year, the AFC program has performed further gap studies, yet to be published, to identify research pathways that contribute to the descriptions found in this document. At the same time, metallic fuel data associated with irradiation testing performed in EBR-II and FFTF the IMIS database was developed but remained unfinished. Recently, a new database, Fuels Irradiation and Performance Database (FIPD) has been developed and data is being qualified and loaded into it [19]. Porter and Crawford developed a preliminary fuel design basis for the Versatile Test Reactor (VTR) using U-Pu-Zr in HT9 cladding, providing a first step towards identifying and quantifying a modern fuel qualification basis, also resulting in identification of data gaps to support near-term fuel deployments [24]. Finally, a report was created for the Nuclear Regulatory Commission (NRC) to assess metallic fuel [25] to exercise the recently published NRC NUREG document titled “Fuel Qualification for Advanced Reactors” [26].

Quality Assurance

All AFC R&D activities are performed consistent with the AFC program quality assurance program plan and respective national laboratory quality programs [27].

2 Technical Description

Research activities are summarized in this section based on subject topics to include a primary point of contact for each activity description containing background, objectives, approach, deliverables, and schedule.

2.1 Topical Report Development

Primary author(s): Colby Jensen

Background

The highest-level technical milestones for AFC R&D on metal fuels will be the creation of NRC topical reports (TR) to further reduce potential risk for metallic fuel technology users and strengthen NRC-DOE collaborations in metal fuel R&D. As described in [28], the TRs allow for "a single Nuclear Regulatory Commission (NRC) staff review of a safety-related topic that applies to multiple nuclear power plants." TRs "increase the efficiency of the licensing process and reduce the burden on licensees by minimizing the time and resources that both industry and the NRC staff expend on multiple reviews of the same topic."

The NRC has provided three criteria to be met for a topical report submission. The following provides the criteria and the justification for an AFC-led topical report on metal fuel technology for SFRs.

1. *"The report deals with a specific safety-related or other generic subject regarding a U.S. nuclear power plant that requires a safety evaluation (SE) by the NRC staff; for example, component design, analytical models or techniques, or performance testing of components and/or systems that can be evaluated independently of a specific license application." [28]*

- Nuclear fuel is a central component of nuclear safety. Metal-fuel technology is among the highest demand near term advanced reactor fuels systems in the U.S. as demonstrated by many reactor designers such as TerraPower, Oklo, ARC-100, GE, Toshiba, in addition to non-SFR applications such as LWRs by Lightbridge. The AFC program is primary steward of significant DOE data, models, analytical and experimental techniques that are and will be independent of specific license applications regarding metal fuels for SFRs.

2. *"Be applicable to multiple licensees, for multiple requests for licensing actions, or both. Examples of requested licensing actions include license amendment requests (LARs), relief requests, and other types of TR-based submittals that are not submitted pursuant to 10 CFR 50.90 or 50.55a." [28]*

- Many current, planned, and future potential metal-fueled reactor applicants could use DOE prepared topical reports for future licensing actions. Already, much of the information is likely to be submitted by current or near-term applicants. However, capturing this information in publicly available topical reports will ensure broader utility for future uses and potentially allow opportunity for expanded usage by first applicants.

3. *"Increase the efficiency of the review process for applications that reference the TR." [28]*

- Like item 2, the many planned and future potential applications of metal fuels will benefit from a well-constructed referenceable topical report.

Objectives

The objective of this activity is to develop NRC topical reports to document a metal fuel fabrication and performance basis to minimize risk to future licensing activities of reactor designers and provide greater efficiency in future metal fuel licensing activities.

An indirect objective met by the first is ensuring and prioritizing AFC R&D activities that are consistent with potential regulatory and licensee interests for the fuel system.

Approach

This activity will primarily be focused on 1) establishing an agreeable approach to Topical Report development with the NRC with a preliminary outline, 2) developing and defining fuel design limits based on existing documents, 3) collecting and refining data and analysis results with appropriate supporting documentation that define design limits, 4) final AFC review and submission, 5) responding to Requests for Additional Information (RAI) as needed. Development of topical report(s) will require close engagement with the NRC, especially in the near term to ensure development of a useful and effective product. It's also worth noting that the ability of the DOE AFC program to address RAI is unique and advantageous with the materials and tools at its disposal.

Much of the information needed to begin this effort could likely be adapted from existing documents [7][24][25] and a variety of other key documents. Additional data from recent testing or studies will be used where needed as well. R&D into technical questions should be carried out in their specific activity areas defined elsewhere in this document. Therefore, this activity will focus on coordination, analysis, and writing. Activities described elsewhere in this document will supply expertise and data to support synthesis of specific subjects in TRs.

The design limits developed in this effort will be used for reference to measure future fuel design enhancements and overall technology progression.

Deliverables and Schedule

- Year 1-2 – Final draft of Topical Report for Reference Design Metallic Fuel System: U-10Zr/U-20Pu-10Zr alloys in HT9 cladding. (Note ternary alloy fuel may merit a separate report in a following year.)
- Year 2 – Final draft of Topical Report for BISON code for Metallic Fuel Performance
- Year 5 – Extended Design Basis of Reference Metallic Fuel Design

2.2 Steady State Performance

Extensive steady state irradiation tests in EBR-II and in FFTF have demonstrated robust and reliable operation of U-Zr and U-Pu-Zr fuels up to burnups between 10 and 20 at.%.

Fuel design limits must consider all fuel degradation modes to avoid potential failure during normal application. These design limits include specific irradiation performance behaviors where t in U-Zr and U-Pu-Zr fuels exhibit constituent migration, fuel swelling, fission gas release, fuel-cladding chemical interaction (FCCI), and fuel-cladding mechanical interaction (FCMI). These phenomena depend on key features (i.e., fuel composition, porosity, fission gas content, and content of lanthanide fission products [Ce, Nd, La, Pm, etc.] in the fuel) and material properties (i.e., thermal, mechanical, thermodynamic, and kinetic).

In the following sections for each of the main fuel performance phenomena AFC experts have summarized the current understanding and knowledge gaps, identify objectives for the next 5 years, along with an experimental or modelling (or a combined) approach to achieve them.

2.2.1 Fuel Performance

2.2.1.1 Fuel-Cladding Chemical Interaction

Primary Author(s): Yachun Wang, Luca Capriotti, Geoffrey Beausoleil

Background

FCCI is an interdiffusion phenomenon that occurs on the metallic fuel-cladding interface, forming interaction layers and carbon loss in cladding, during irradiation. There are two primary types of FCCI of issue for the fuel based upon the controlling interdiffusion of elements between fuel and fission products, and cladding: Lanthanide fission products (Lns)-cladding-interdiffusion-dominated (e.g., Ce/Nd with Fe) [29] and fuel-cladding-interdiffusion-dominated (U/Pu with Fe).

The interdiffusion region, typically spanning from the inner cladding to the fuel periphery, has a two-fold effect on the metallic fuel performance generally corresponding to the two FCCI types. Firstly, FCCI acts to thin the effective cladding thickness for mechanical loading as the interaction layer is typically embrittled (the loss of ductility) and often cracked, and the carbon loss layer that creates reversion of the martensite to soft, low-carbon ferrite, and grain growth. Both are considered wastage [24]. FCCI increases the probability of stress rupture failure in cladding by causing both uniform fronts of material exchange and stochastic, localized regions of material exchange. These interactions cause a reduction in cladding lifetime and thereby limit the ultimate achievable burnup in metallic fuel under steady-state operation conditions. Secondly, due to the formation of (U, Pu)-Fe eutectic phases in the fuel, FCCI in the fuel increases the probability of localized fuel and/or cladding melting during some high temperature transient events [30]. This section pertains to “steady-state” relevant phenomenon of the first type. Eutectic effects are generally more relevant during off-normal or transient conditions and this FCCI type is discussed in section 2.3.1.1.

- Over the past decades, post-irradiation examination (PIE) has been performed for some Experimental Breeder Reactor (EBR)-II and Fast Flux Test Facility (FFTF) fuel pins which provides a moderate FCCI dataset of HT-9 cladding performance. Significant Ln-cladding-interdiffusion-dominated FCCI has been discovered in many EBR-II HT-9 clad U-Zr fuel pins irradiated to burnup levels of 5-10 at%, plus some EBR-II HT-9 clad U-Pu-Zr fuel pins with 3-10 at% burnup [31][32][30]. PIE of historic fuel pins have shown the most severe FCCI behavior to occur at the fuel element region exposed to the combination of high power and high temperature [31] and, of course, fuel burnup (concentration of Ln fission products). The dependence of FCCI on temperature is easy to understand as diffusion in solids are strongly temperature dependent and exhibit Arrhenius behavior [33]. Higher temperature generally renders higher kinetic energy to increase the interdiffusion rate. The supply of fission products to the fuel-cladding interface increases with the increase of power. Previous efforts have shown that FCCI in U-10Zr fuel with HT9 cladding is a complex, mostly localized phenomenon as FCCI depends not just on fuel type, fuel topography, and cladding material, but also on operating conditions such as power and irradiation temperature [31][32][30] .

The recent PIE studies of U-10Zr fuel found that the diffusion of lanthanide fission products to the fuel/cladding interface facilitated the lanthanides-cladding interdiffusion dominated FCCI [29]. The knowledge gaps to be investigated in this research area are:

1) Determining the diffusion mechanisms for Ln diffusion throughout the fuel. There is debate between whether the diffusion behavior is dominated by concentration gradients (Fickian effects), temperature gradient diffusion (Soret effects), or how these two combines in various operating conditions. The presence of sodium-filled pores in the fuel may also influence transport.
2) Determining diffusion coefficients within the cladding to adequately represent the ingress of Lns into the cladding.
3) Determining characteristics of the fuel-cladding interface that leads to the apparently stochastic onset of FCCI.
4) Incorporating the above results into an updated FCCI model within BISON for Ln transport within the fuel (fuel composition model), Ln-Fe interactions, U/Pu-Fe interactions, and the FCCI thickness in the fuel-cladding interface. Additionally, the development of a model to represent the stochastic nature of FCCI onset would be beneficial to modelling steady-state behavior.
5) There is both a lack of FCCI data for U-Zr fuel, as well as an explanation of how it was measured in the past [7]. More data by SEM examination needs to be performed on available representative ‘legacy' irradiated fuel pins.
6) There is less FCCI dataset for U-Pu-Zr fuel compared to the U-Zr fuel.
7) Meanwhile, the carbon loss layer in the HT-9 cladding is known to be softened and compromises the creep-resistance of cladding, especially at high temperature regime. However, very limited experimental study has reported and/or examined the carbon loss layer in the HT-9 cladding, including two EBR-II U-10Zr fuel pins [35] and one MFF (acronym unknown) fuel pin) [14].

Objectives

- Complete a modernized summary of FCCI data across the range of irradiation conditions testing in EBR-II and FFTF via new and existing measurements.
- Identify any potential remaining data gaps beyond the existing legacy metal fuel materials library.

Approach

The following key activities will be performed to accomplish activities in this area:

1) Surveying the available EBR-II fuel pins that have been irradiated to high burnup levels (approaching 20 at%, such as X425 assembly pins). The selected pins can be sectioned at various heights to understand the role that temperature plays in FCCI at high burnup.
2) Surveying the FCCI experiment database for previous EBR-II and recently examined FFTF (MFF) fuel pins to establish the database of FCCI thickness in relationship with temperature and power (burn up) for U-10Zr and U-Pu-Zr fuel.
3) In order to better evaluate/simulate the effect of carbon loss layer on the creep behavior of cladding through BISON modelling, it is necessary to examine/characterize/measure potential carbon loss layer thickness and the hardness change in the carbon loss layers in some high-temperature irradiated MFF fuel pins.

Scanning electron microscopy (SEM) imaging and elemental analysis is the best approach (in terms of time and cost efficiency) to determine, examine, and quantify FCCI thickness in various fuel samples; therefore, SEM will serve as the primary approach to characterize FCCI in selected fuel pins. The carbon loss layer seemed to occur at high operating temperature (above ~615°C PICT), forming in the deeper clad matrix beyond the interdiffusion layer. Therefore, it is important to pay attention when examining/characterizing fuel sample irradiated at high temperature (above ~615°C PICT) to determine if carbon loss layer formed and its thickness. Backscatter electron (BSE) imaging through SEM instrument will be capable to determine the carbon loss region if dramatic microstructure evolution (phase change and grain growth) happened during irradiation. Otherwise, any characterization approach (for instance Electron Energy Loss Spectroscopy) that can provide accurate carbon quantification results is recommended to determine carbon loss in the cladding. Small-scale mechanical testing techniques at MFC, including nanoindentation, compression creep (or tensile creep) testing, are versatile and promising to examine very localized regions like cladding wastage (FCCI layer and carbon loss layer) and the clad matrix to provide thermo-mechanical properties in the cladding.

Deliverables/Milestones

Year 1-3 – Conduct examination/characterization of FCCI for high burnup fuel pin from previous EBR-II irradiation and some hotter MFF fuel pins.

Year 1-3 – A survey dataset of FCCI thickness in relationship with temperature and power (burn up) for U-10Zr and U-Pu-Zr fuel.

Year 2-4 – Identify representative U-Pu-Zr fuel pin for FCCI characterization and measurement.

Year 3-5 – Conducting diffusion study to measure thermo-physical property of Ln in the fuel, data analysis and interpretation, report the diffusion mechanism.

Year 3-5 – Incorporate experimentally measured mechanical properties for BISON thermomechanics model development/validation.

2.2.1.2 Fission Gas Release

Primary author(s): Colby Jensen, Doug Porter

Background

Fission gas release (FGR) in nuclear fuels plays an important role in overall fuel performance with impacts on important fuel performance limits. Fuel designs for SFRs must accommodate fission gas generation and release over the life of the pin as well as during certain transients. FGR dominates pin plenum pressure and is certainly a dominant loading on cladding, driving thermal creep, including during transients.

In SFR metallic fuel, the fuel smeared density, as defined by the as-fabricated outer diameter of the cylindrical fuel slug divided by the inner diameter of the cladding, was chosen in the reference fuel design to be 75% because of theoretical predictions of when fission gas bubbles would interconnect and form pathways to the gas plenum of the fuel pin [34]. FGR has been quantified via measurements on many integral scale fuel pins from EBR-II and FFTF. Detailed studies of fuel swelling, gas porosity development and evolution, and release to the fuel plenum have been subject of study in several instances. As the fuel swells into contact with the cladding, pin average fission gas release increases to near 60-80% where it typically levels off for the duration of its life. Regarding fuel swelling, solid fission product accumulation eventually further increases the fuel/cladding mechanical stress.

Data from FFTF, longer fuel pins, have generally shown consistency with the more extensive EBR-II database. Extensive model development has been performed to date by a variety of researchers, with good ability to predict pin average FGR for reference fuel design. Applications of these models have not been thoroughly vetted against a wide range of fuel design variations, but some data has indicated differences in total release for different fuel alloys, smeared densities, and geometries. Still, the underlying fundamental behaviors between these are evidenced to be similar. FGR from fuel during transient over temperature conditions has been measured during historical hot-cell furnace experiments but little of that data has not been quantified to date.

Objective

Provide an experimental basis and means of calculating FGR from reference design metallic fuel.

Approach

Primary activities to be undertaken include:

- Collect additional FGR data from integral FFTF MFF fuel pins stored at INL.
- Evaluate range of operating conditions corresponding to available FGR data and identify potential materials to extend database range to perform additional measurements.
- Develop experimental plan to collect data for fission gas release experiments (considering terminal temperature, ramp rate, and potentially overpressure including transient conditions.) Planning may be integrated with needs of Section 2.2.1.3.
- Collect data from transient testing experiments for FGR where possible.

Deliverables

Year 0-2: Report MFF fuel pin fission gas release data.

Year 1-2: Develop experimental plan for FGR experiments.

Year 4-5+: Perform FGR separate effects experiments.

2.2.1.3 Fuel-Cladding Mechanical Interaction

Primary Author(s): Pavel Medvedev

Background

Metallic fuel qualification relies on demonstrating that end-of-life (EOL) cladding strain meets a certain design criterion. Cladding strain occurs due to FCMI, pin pressure and cladding swelling. Cladding strain affects fuel pin lifetime (cladding breach) by thermal creep. In addition, bundle tightness, stress on the duct and flow restriction is affected by cladding swelling and both thermal and irradiation-induced cladding strain. Experimental observations only show integral impact of FCMI, pin pressure and cladding swelling on cladding strain, but are unable to quantify individual contributions of FCMI and pin pressure. Fuel swelling and fission gas release are the phenomena that drive FCMI and pin pressure. Therefore, a methodology to accurately calculate cladding stress arising from FCMI is critical for metallic fuel qualification as it would determine the remaining unknown in the force balance of the fuel pin.

Objectives

Provide experimental basis and calculation of cladding stress arising from FCMI.

Approach

An approach to calculate cladding stress arising from FCMI will be developed during the performance period by teams comprised of nuclear fuel and solid mechanics theorists and experimentalists. Separate effect furnace and/or irradiation tests could allow to isolate FCMI stress by relieving pin pressure and eliminating associated cladding stress. Testing unconstrained fuel specimens could be used to eliminate fuel FCMI-induced fuel hot-pressing.

Deliverables/Schedule

Year 1: Develop formal approach and test plans to determine cladding stress arising from FCMI (may be integrated with needs of Section 2.2.1.2.

Year 2: Test design and fabrication

Year 2-3: Irradiation

Year 4: PIE

Year 5: Data analysis

2.2.1.4 Fuel Constituent Redistribution

Primary Author(s): Jake Hirschhorn, Luca Capriotti, Boone Beausoleil

Background

Metallic nuclear fuels contain multiple crystalline phases at temperatures ranging from room temperature to that of normal operation. The radial temperature gradient within the fuel during normal operation produces differences in chemical potential across multiphase regions, which promotes migration of fuel constituents in a process known as *constituent redistribution* [36]. The material properties and irradiation behaviors of individual phases govern the fuel's local response to its thermomechanical and irradiation environment, and the combined effects of these local responses dictate the fuel's engineering scale performance. Local melting point in the fuel is an example of key fuel performance consideration, that is especially important for high-Pu content alloys. Key observables impacted by phase and constituent composition include local fission rate, melting temperature, heat conduction, thermomechanical properties, porosity development and evolution, and more.

For fuel compositions of interest (U-10Zr and U-20Pu-10Zr) operated at prototypic EBR-II temperatures, constituent redistribution promotes interdiffusion of U and Zr at inner and intermediate fuel radii. When fuel temperatures are high enough to promote formation of a single-phase BCC region, zirconium migrates toward the fuel centerline from intermediate radii to form what is frequently referred to as a *Zr bathtub* [36]. Until recently, experimental data collected to characterize constituent redistribution was limited. A handful of electron probe microanalysis (EPMA) scans of cross-sectioned fuels irradiated in EBR-II showed the relative compositions of U, Pu (if applicable), and Zr [37]. Optical micrographs of irradiated fuel cross sections showing spatially distinct pore structures have also been used to indirectly infer temperatures and the locations of boundaries between phase regions [38]. These data have been used to develop and validate models for constituent redistribution. More recently, a substantial number of additional EPMA, microscopy, and phase identification data have been collected [39].

Numerous constituent redistribution models have been developed. The most recent effort employs a CALPHAD (Computer Coupling of Phase Diagrams and Thermochemistry) thermodynamic description and kinetic data from Diffusion Controlled Transformation assessments [18]. Calibration and validation efforts have historically been conducted using simplified thermomechanical models or by coupling to outdated thermomechanical assessment cases. Example simulation results are shown alongside a micrograph of irradiated ternary fuel in Figure 1.

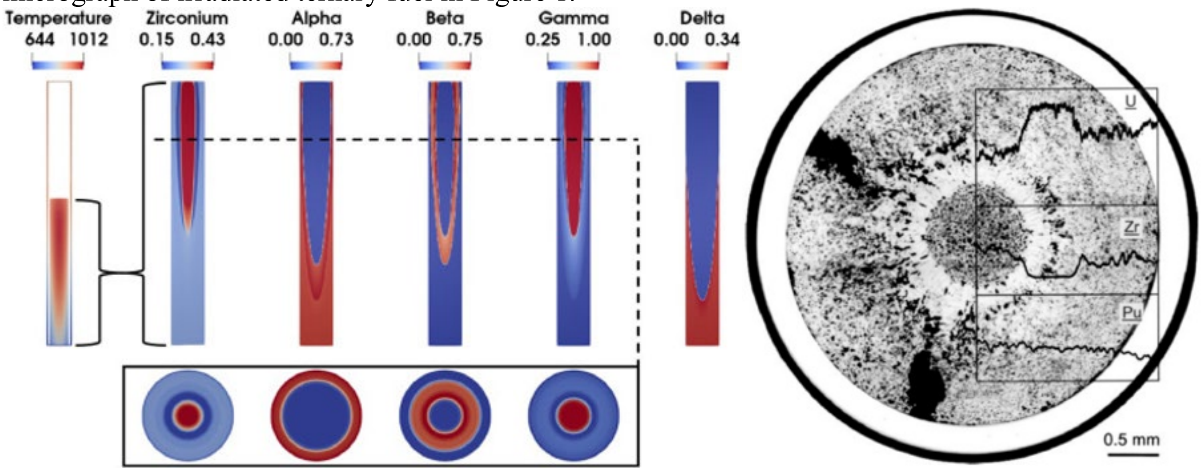


Figure 1. Fully coupled thermomechanics and constituent redistribution results from BISON showing formation of concentric phase and constituent rings (left) and a micrograph of irradiated metallic fuel with a similar ring structure (right) [18][40].

Constituent redistribution is not considered a lifetime-limiting phenomenon for commercialization of first-generation metallic fuel technology, which is likely to be very similar to the metallic fuels used in EBR-II and FFTF experiments. However, understanding and predicting constituent redistribution is important for next generation metallic fuel technology, which is likely to differ from historical metallic fuel designs in terms of geometry, operating conditions, etc. An example of this is the proposed sodium-free annular fuel rod designs [41]. Improved understanding of constituent redistribution and predictive capabilities thereof would improve confidence in these regions, expediting design and qualification of next generation metallic fuels. Furthermore, improved understandings of phase stabilities and constituent redistribution would allow incorporation of phase-specific material properties into fuel performance codes like BISON. Currently, thermomechanics models like thermal conductivity, thermal expansion, elasticity, creep, etc. are parameterized as functions of temperature and/or burnups. Collecting phase-specific material properties and applying them within a mechanistic constituent redistribution framework would greatly improve predictive capabilities and confidence in unexplored regions of the metallic fuel design space.

Objectives

High-impact topics for future work have been outlined in the literature [18]. These focus on reducing uncertainties in ternary thermodynamics, reducing uncertainties in binary and ternary kinetics, and collecting phase-specific material properties. Key observations are provided below.

- Agreement between model predictions and experimental data is exceptional for binary fuels.
- Model predictions for ternary fuels exhibit reasonable trends but are less accurate than desired.
- Adjustments to ternary transition temperatures on the order of 60°C is sufficient to improve the accuracy of ternary model predictions.

Approach

Improving the accuracy of the systems' thermodynamic descriptions, incorporating those data into existing models, and applying those models within realistic thermomechanics simulations would enable improved calibration of kinetic parameters. Diffusion couple studies could also be performed to further reduce kinetic uncertainties. These improvements could be committed back to the BISON repository and leveraged to perform fully coupled thermomechanics and constituent redistribution simulations, which could be validated against multiphysics datasets. These efforts should leverage synergies with ongoing

FAST experiments and recently awarded NEUP projects to assess and account for differences between solid and annular metallic fuel behaviors to support design and deployment of next generation metallic fuels.

Specific tasks designed to meet the objectives above are listed below. The proposed work involves tightly coupled experimental and modeling tasks. The work scope is intended to leverage past experience to the extent possible and apply fundamental nuclear engineering and materials science principles to obtain high impact data and predictive tools. Priority is given to work that (1) reduces uncertainties in first generation commercial metallic fuel applications, supporting qualification, licensing, and operation; and (2) enables design, development, and deployment of next generation commercial metallic fuel applications.

- Conduct experiments to confirm the location of transition temperatures and solvus lines in regions of the ternary phase diagram relevant to U-20Pu-10Zr, reassessing the system CALPHAD database as necessary.
- Conduct diffusion couple experiments to reduce uncertainties in poorly characterized binary and ternary phases (particularly β and ζ).
- Incorporate the above data into the existing constituent redistribution model in BISON.
- Apply the updated BISON model within assessment cases powered by the EBR-II FIPD to recalibrate and validate it using historical and newly acquired data.
- Coordinate with ongoing FAST experiments and NEUP projects to apply the refined model to sodium-free annular fuel rod designs.
- Characterize phase-specific material properties and apply them in conjunction with the refined constituent redistribution model.

Deliverables

- New experimental data and refined models published in high-impact scientific journals.
- Reduced uncertainties in thermodynamics and kinetics (updated phase diagrams, diffusivities).
- Phase-specific material properties and associated thermomechanics models.
- Refined models and validation cases committed to the BISON repository.

Schedule

- Year 2-3: Phase stability experiments
- Year 2-3: Kinetics experiments
- Year 2-4: Phase-specific property experiments
- Year 4: Update BISON model
- Year 4-5: Study and adjust for annular fuel effects
- Year 4-5: Recalibrate and validate BISON model
- Year 4-5: Apply phase-specific material properties in BISON

2.2.2 Fuel Irradiation Experiment Development

Primary Author(s): Nick Woolstenhulme, Colby Jensen, Caleb Massey, Boone Beausoleil, Ben Eftink, Stu Maloy

Irradiation testing of fuel specimens in material test reactors (MTRs) has been a crucial endeavor in the development of every mainstream nuclear fuel system. The indispensability of irradiation testing arises from the effects of neutron-nuclide interactions, thermal gradients from nuclear self-heating, and effects of ionizing radiation which only exist together in a reactor environment. Furthermore, the probability of all neutron-nuclide interactions (e.g., actinide fission, atom displacement, and capture transmutation) depends on the target nuclide and energy of the incident neutron. This situation is particularly problematic for modern research on fast reactor fuels since all operational test reactors currently available to the United States are thermal spectrum water-cooled MTRs. Creative solutions are needed to make progress in the near term, and foundation-setting efforts to succeed in the long term. This is one of the challenges that underlies the development of irradiation experiments for metallic fuels.

A batch of nuclear fuel can provide energy for years. Accordingly, irradiation tests can also last for years, especially when considering the effort needed to select worthy fuel concepts, manufacture specimens, develop irradiation test designs, manage logistics, and perform examinations in shielded facilities. Herein lies the second challenge that influences development of irradiation experiments for metallic fuels. There is a dichotomous relationship in the community's yearning to mature nuclear fuel technologies rapidly while also craving to understand physical phenomena more mechanistically. Historic fuel development projects, the best known of which achieved success with remarkable haste, did not expend years of effort to understand and develop models for behaviors at every size scale knowable, they did not even have the computational power to do so. Instead, they irradiated numerous specimens at end-use scale in representative nuclear environments and quickly abandoned ideas that performed poorly. Yet the successful fuel systems that emerged have continued to be studied decades after they were first deployed to develop more mechanistic understandings. New optimization potentials have emerged as a result. Balancing the imperative of accelerating fuel development while ensuring that behaviors are well understood is a crucial tension point in realizing sustainable fuel technologies. As a venue that is both time consuming and crucial in creating data, the field of irradiation testing is often at the forefront in managing this tension.

Objectives

The main goals of AFC metallic fuel research were outlined in the introduction of this plan. It is important to understand how these goals propagate to irradiation testing work. The list below overviews these overarching goals in the context of irradiation testing:

- **Establish the qualification basis for reference metallic fuel designs for sodium-cooled fast reactors:** Most of the irradiation performance data for the reference metallic fuel designs originate from historic irradiations. There are opportunities, however, to augment historic data to unlock the full potential of the reference design. These opportunities revolve around analytic experiments targeted at certain phenomena as well as filling gaps in transient testing data.
- **Develop next-generation metallic fuel fabrication and design for improved fissile utilization and management:** This goal is influential on the irradiation testing plan as it drives the need for irradiations that can assess the effect of fabrication technologies at an appropriate scale. This area includes modified fabrication techniques and investigations into permissible tolerances/defects.
- **Develop accelerated fuel development and qualification methodologies:** This goal also influences the metallic fuel irradiation testing plan primarily towards developing accelerated fission rate experiments and tests with in-situ data collection techniques.
- **Identify next-generation fuel technologies:** This goal impacts irradiation plans by driving the need for capabilities able to expose next-generation fuel specimens to relevant conditions. Present consensus is inexact about which fuel innovation areas are most worth pursuing. This goal's influence on near term plans mostly involves experiments which compare the performance of candidate fuel concepts. The following list of fuel innovation areas illustrates some potentials:
    - **FCCI Mitigations:** This area includes investigations into solutions for mitigating FCCI to improve tolerance to burnup, temperature, and minor actinide addition. Examples include fuel-cladding interlayers and lanthanide-arresting additives.
    - **Cladding Enhancements:** This area includes studies on cladding alloys and processing techniques that could improve the high temperature performance of metallic fuel cladding such as modified ferritic/martensitic steels to increase creep rupture resistance.

- **Broadened Applications:** This area focuses on expanding the application envelope for metallic fuels into new venues such as small reactors. Examples include alloy adjustments with higher uranium loading, larger diameter pins, and filling FCCI data gaps for extended time-at-temperature performance.

2.2.2.1 Prioritizing Pursuits with Current Capabilities

Background

There are a few existing irradiation capabilities which will be utilized for their value in metallic fuel testing. The first is the Advanced Fuel Campaign (AFC) capsule which can house full diameter, short-length, SFR rodlets in high-flux inner core positions in the Advanced Test Reactor (ATR). These capsules are placed in cadmium lined baskets (Cd-basket) to filter thermal neutrons. This approach reduces self-shielding effects to flatten the radial power gradients and helps reduce spurious effects from thermal neutron capture transmutation in cladding. Custom enrichments are used to achieve SFR-representative heating rates. A gas gap between rodlet cladding and the capsule wall elevates the fuel temperature to represent SFR conditions well, although the manufacturing tolerances on these gaps increases uncertainties on actual specimen irradiation temperatures.

An evolution of the AFC design, referred to as FAST, essentially employs the same approach, but reduces the diameter of rodlets by a factor of two. Using bespoke isotopic enrichments maintains the same linear heating rate as full diameter pins but with roughly four times the volumetric fission rate to accelerate burnup accumulation. The smaller diameter reduces self-shielding effects, so FAST capsules can be used without Cd-basket, while providing extra volume for an additional sodium-filled inner capsule, making the thermal resistance network less sensitive to manufacturing tolerances on the gas gap between inner and outer capsule. Compared burnup-to-cladding-fast-fluences ratios to the AFC design, FAST creates conditions even less representative of true SFRs. Thus, cladding-centric test objectives in FAST rodlets pertain better to fuel-cladding interactions, rather than behaviors belonging to the cladding alone.

The MiniFuel design uses a small capsule in various reflector positions in the High Flux Isotope Reactor (HFIR). MiniFuel uses a similar approach to FAST via accelerated fission rates but differs in that its specimens are much smaller such as unclad kernels and discs. These small specimens exhibit little self-shielding while gamma heating in the hardware comprises the majority heat source which, unlike fission heating, does not create large thermal gradients or deplete during irradiation. MiniFuel is well suited to controlled separate-effects tests where a specimen represents a small region of fuel behaving in isolation from the integral system. This approach can be useful for developing and benchmarking models of lower length scale phenomena. As simple drop-in capsules, active instrumentation cannot be used in the AFC, FAST, or MiniFuel capsule designs, but passive thermometry such as melt wires and SiC monitors can be used.

The neutron irradiation capabilities discussed above employ thermal neutron filtering and/or sub-size specimens to manage fission rate gradients, but they do not increase the fast neutron population to a level which represents atom displacement damage in SFR core materials. This is particularly problematic for observing cladding behaviors where data needs revolve around mechanical properties changes and void swelling at high fast fluence. Presently ion beam irradiations can be performed to help assess some of these behaviors. Normally ion beam methods would go unmentioned in a neutron irradiation plan. Unlike neutrons, ions have an electrical charge and thus cannot penetrate nearly so far samples, making them useful for assessing micron-scale microstructural effects, but ineffective for measuring changes in engineering-scale properties. Still, without a true fast spectrum test reactor available presently, ion beam irradiations represent some utility, especially noting the modest cost and schedule to perform such tests.

Approach

The AFC capsule, being the only existing design capable of housing full diameter fuel rodlets, will be used to irradiate rodlets produced by fabrication techniques which are candidates for enhancing the economics of metallic fuel fabrication. This test series will be referred to as AFC-5. This work will also assess whether the AFC-5 rodlet design can be lengthened somewhat while still retaining its value as a near term and cost-effective irradiation strategy. This test will be irradiated and examined in two batches to represent two burnup states.

The ongoing FAST-1 irradiation experiment will continue as planned to include PIE and comparison to BISON models. This test is discussed more in the next section as it is primarily an exercise in developing methodologies for accelerated testing, but FAST-1 also carries some specimens with modified fuel design features which will be compared and assessed to help prioritize future fuel development pursuits. These specimens include lower smear density fuel slugs, annular fuel slugs, alternate fuel alloys including lanthanide-arresting additives, and claddings with liners.

Present plans do not include a near term MiniFuel irradiation, but a couple concepts that have been considered for potential needs considering fuel free swelling behavior and maybe even studies on FCCI if particular needs are identified. These applications could leverage MiniFuel's ability to create more isothermal specimen conditions to conduct FCCI studies between fuel discs of candidate alloys and cladding/liner discs. Such a test, conducted in concert with AFC capsule or FAST rodlet irradiations, would help sort out the effects of thermal gradients and mechanical constraints.

Prior data from HT9 irradiations will be synthesized as a function of irradiation parameters and new specimens from various heats, including wrought and welded samples, will be prepared for ion irradiations. High-energy ion irradiations will be performed to establish the swelling incubation period and swelling rates at peak swelling temperature ranges (400-500C). Micromechanical testing will be performed on irradiation samples and compared to models of cavity nucleation/growth. New data and assessments from these ion irradiations will be useful in refining understanding and maintaining active expertise in this field while awaiting fast-neutron irradiations in a true SFR.

Deliverables

- Year 2 – Completion of AFC-5 Design
- Year 3 – Commencement of AFC-5 Irradiation
- Year 1 – Completion of FAST-1 Irradiation
- Year 2 – Completion of past data review and new ion beam test/specimen preparation
- Year 3 – Completion of AFC-5 Irradiation Medium Burnup Specimens, Begin PIE
- Year 4 – Completion of AFC-5 PIE for Medium Burnup Specimens
- Year 4 – Completion of AFC-5 Irradiation Higher Burnup Specimens, Begin PIE
- Year 4 – Completion of HT9 ion beam irradiation tests and micromechanical tests
- Year 5 – Completion of AFC-5 PIE for High Burnup Specimens

2.2.2.2 Furthering Competencies in Analytic Experiments

Background

The value of irradiation experiments, and thus their very nature, necessarily includes the combination of multiple physical phenomena. The practical aspects of planning and preparing for such experiments also often drive toward batch processing groups of specimens representing different experiment objectives. It can be difficult to clearly categorize a given irradiation campaign clearly into the oversimplified taxonomy of basic research, applied research, and development work. At any rate, the phrase "analytic experiments" is put forth here to refer to irradiation tests where the objectives are prioritized toward analyzing and understanding certain phenomena more than other purposes such as comparing the performance of candidate fuel designs or simulating the end-use environment with high fidelity. It is recognized, however, that some portion of these three objectives is present in nearly every irradiation test.

Approach

The principal purpose of the FAST-1 experiment is to provide data to assess the scaled-rodlet accelerated fission rate method. This method is expected to distort the response of certain fuel performance phenomena and their interactions with each other. In some cases, these behaviors may be amplified (e.g., the role of thermal gradients in species diffusion in the fuel), and in other cases effects may be minimized (e.g., the role of time at temperature). It is expected that FAST-1 will hint at unique analytic experimental opportunities to help distinguish interesting effects. Completion of FAST-1 irradiation and PIE, combined with comparison to predictions from the Bison code, will all help to identify the most appropriate analytic experiments that can be supported. FAST-1 PIE and Bison modeling are already underway for lower burnups specimens. The knowledge gained from this effort will be used to develop the successor to FAST-1, the Accelerated Testing of Materials in Capsules (ATOMIC) irradiation test. ATOMIC will include fuel specimens across a broad range of reactor applications, some of which will be metallic fuel alloys. The ATOMIC test will be developed to further reveal behaviors in metallic fuel and support analytic experiments with cost-effective capsules and accelerated schedules.

The first Irradiated Materials Properties Accelerated Characterization Test (IMPACT-1) is a novel and approach to facilitate instrumented lead-out tests in ATR using a newly installed top head closure plate. This approach will use a new type of probe embedded in metallic fuel specimens to measure fuel thermal conductivity evolution during its early life as it swells to the point of interconnected porosity. The IMPACT-1 test will obtain this first-of-A-kind data and exhibit the value of in-reactor measurement for accelerating fuel research. This test was developed and constructed under an INL Lab Directed Research and Development (LDRD) project. Due to delays in the reactor operation schedule, LDRD resources will not be able to support its installation and irradiation. Completion of this test will be supported by AFC to harvest data from this important analytic experiment.

Another example of this type of test is the Disc Irradiation for Separate Effects Testing with Control of Temperature (DISECT). This experiment is presently under construction and will be irradiated in the Belgian Reactor -II (BR2). DISECT will contain thin samples of metallic fuel alloys and be irradiated in a device able to monitor and control specimen temperature, thus enabling assessment of the effects of fission rate, burnup, and temperature on important fuel behaviors. This test is planned to be supported to completion by the Nuclear Science User Facility (NSUF) program, rather than the AFC metallic fuel program, but is worth mentioning here as it will be an important irradiation to aid understanding of metallic fuel behavior. Further PIE evaluations may be considered to support AFC specific needs.

The planned analytic experiments discussed above are primarily focused on behaviors of metallic fuel alloys, but there are also special data needs regarding cladding behavior. An experiment designed to separate the effects of irradiation-assisted creep from thermally driven creep, and the effects of fission gas pressure vs. solid fuel swelling on cladding, are needed to develop a better understanding of these important influences. This experiment is in the early phases of conceptualization, and it is presently unknown whether drop-in capsules or instrumented lead-out tests will be used. As resources become available, likely after some tasks are completed such as IMPACT-1 irradiation, this cladding creep effects will be designed and constructed for irradiation.

Deliverables

- Year 1 – Completion of ATOMIC Design
- Year 1 – Commencement of IMPACT-1 Irradiation
- Year 2 – Commencement of ATOMIC Irradiation
- Year 2 – Completion of IMPACT-1 Irradiation
- Year 3 – Completion of Creep Effects Test Design

- Year 3 – Completion of IMPACT-1 PIE
- Year 4 – Completion of ATOMIC Irradiation
- Year 4 – Commencement of Creep Effects Test Irradiation
- Year 5 – Completion of ATOMIC PIE
- Year 5 – Completion of Creep Effects Test Irradiation

2.2.2.3 Setting the Stage for Qualification Tests

Background

Much good work can be done with the irradiation capabilities described thus far, but some new approaches will be needed in order to progress new fuel technologies to a level of maturity needed to qualify them and deploy them. Specifically, the ability to irradiate full length pins (at least the length of EBR-II pins) in a higher fast flux environment will be a crucial development. These specimens will be needed to feed into PIE and transient testing to develop the data needed to support lead rod irradiations in future commercial SFRs under license amendments. Naturally, this endeavor is challenging without a fast spectrum test reactor available to the United States, and the plan for how to proceed in this situation is somewhat obstruse, but there are some concrete steps that will likely lead to a path which, while far from ideal, can help to bring new metallic fuel technologies to a reasonable level of maturity.

Approach

The Joyo reactor in Japan is the only existing SFR that is available to the United States under present geopolitical circumstances. Joyo has not operated for several years but plans exist to resume operation in the next few years. Joyo's safety basis is not agile for supporting experimental fuel irradiations but can support materials-only tests without undue concern. Joyo's long term fuel supply and operational plans are uncertain, but there does appear to be a unique near-term opportunity to provide some materials specimens for irradiation in a true SFR. The amount of specimen volume allocated to AFC for this opportunity is expected to be modest, so AFC will prioritize and prepare important specimens of HT9 and 14YWT alloys to be irradiated in Joyo.

The Boosted Energy Advanced Spectrum Test (BEAST) will use rings of fuel plates in an ATR flux trap to multiply incident thermal neutrons into fast neutrons while a multi-hole cadmium basket will help achieve high fast-to-thermal flux ratio on several large SFR pins. Once commissioned, BEAST will require at least a few years occupation of an ATR flux trap and yearly replacement of the booster fuel assembly. BEAST represents the best possible simulation of a fast reactor environment that can be achieved in existing thermal spectrum test reactors but will be a relatively expensive endeavor compared to previous tests. Early efforts for BEAST will include design and safety analysis work so that specifications can be issued, and booster fuel assemblies procured. The optimal timing of the BEAST schedule will need to be managed carefully so that appropriate full-size specimens can be prepared in sync with periods of flux trap availability and within overall project budget constraints. The near-term detailed design work will be crucial in maturing assumptions and estimates in order to manage the plan for BEAST irradiation.

Despite being the best possible capability in current test reactors, BEAST will not likely be able to achieve more than about half the end-of-life cladding fast fluence of a true fast reactor. Unless a massive investment is also made to put a flowing liquid sodium loop in BEAST, then it will not be able to irradiate pins in highly representative fluidic and mechanical boundary conditions either. Full qualification of new metallic fuel designs, or even of reference fuel designs fabricated under different specifications, will ultimately require irradiation in a true fast reactor. Joyo currently operates using mixed oxide fuel but has limited fuel supply and may modify its safety bases to allow for metallic fuel driver assemblies. If this were to happen, then the regulatory hurdle may be reduced to permit lead test rods (LTRs) of experiment metallic fuels in Joyo. Unlike materials-only specimens, however, international logistics for irradiated fuel specimens are significant obstacles that would need to be considered.

TerraPower plans to build the Natrium SFR not far from INL in western Wyoming. Oklo plans to build the Aurora SFR on INL property where irradiated specimens could probably be shipped for PIE without even using public roads. Presuming successful construction, these two reactors look like attractive candidates for hosting lead test rod (LTR) irradiations. Still, many things need to be considered such as what facility infrastructure is needed to install/remove experimental assemblies, how willing are plant operators/utilities to accept experimental pins, and what level of data is needed from ATR and the Transient Reactor Test Facility (TREAT) irradiations to gain LTR permission under an NRC license amendment request.

Of course, the most ideal option would involve construction of a fast spectrum test reactor in the United States. Given that the VTR project has yet to receive funding, one should expect that this option is many years from fruition. Continuing to do the next best thing in the meantime is a strategy that will help the community move forward in meaningful ways while catalyzing research needs to justify a fast spectrum test reactor that does what other reactors cannot. Many thoughts need to be followed to completion in order to formulate the best end game strategy for qualifying progressive metallic fuel designs. A task is thus planned to support stakeholder discussions and development of this ultimate strategy.

Deliverables

- Year 2 – Completion of BEAST Design
- Year 2 – Material Specimens Ready for Joyo Irradiation
- Year 3 – Completion of BEAST Design
- Year 4 – Stakeholder review of LTR License Amendment Strategy
- Year 5 – Commencement of BEAST Irradiation

The overall irradiation plan described in this section is illustrated graphically in Figure 2 below.

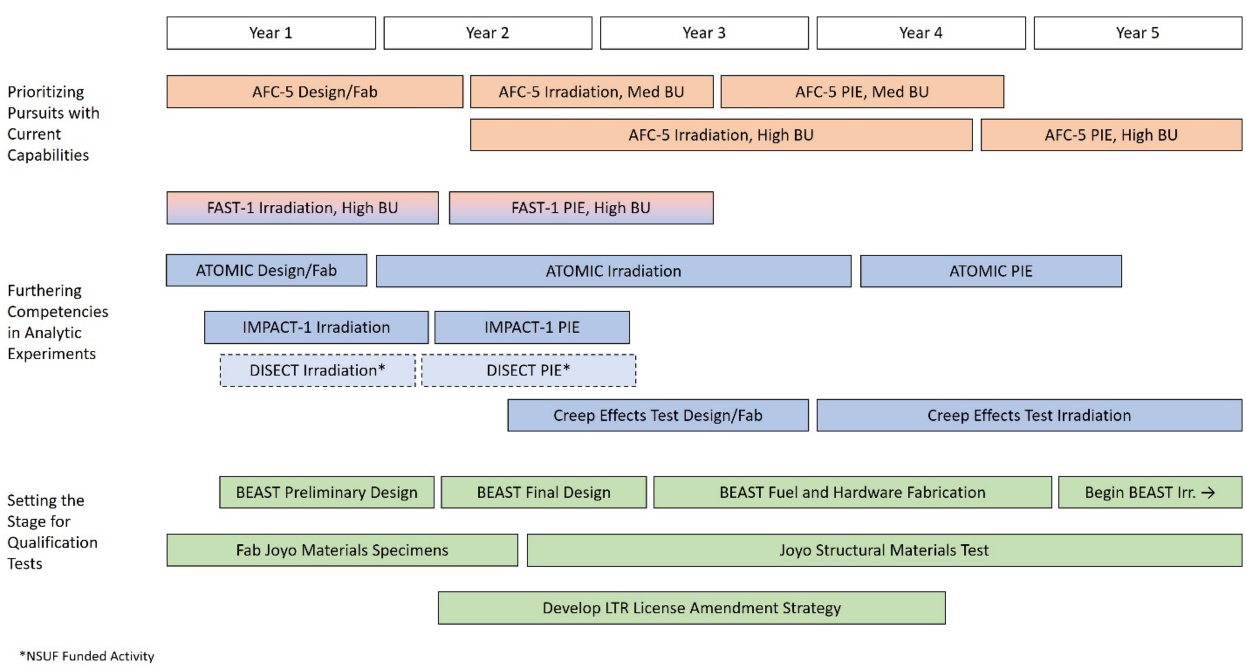


Figure 2. Overview of Irradiation Testing Plan

2.3 Transient Performance

2.3.1 Fuel Performance

Transient performance of metallic fuel is one its most desirable attributes, being compatible and conducive towards passive reactor design strategies. Reducing uncertainties in fuel behavior during hypothesized transient conditions provides an opportunity to increase confidence for relying on passive design and probabilistic-risk-assessment-based licensing strategies. Establishing clear knowledge of all fuel degradation mechanisms and failure thresholds corresponding to relevant transient scenarios is crucial to fuel qualification. Although other factors may be important to fuel damage, cladding temperature during reactor transients is a dominant factor in damage assessment as most cladding failure mechanisms are strongly temperature dependent. The Preliminary Safety Evaluation Reports of the IFR era reactors pointed to plans to do more transient testing to complete understanding to support licensing [5][6].

The restart of the TREAT facility is a notable opportunity, restoring transient irradiation testing capability in the U.S. Combined with a rich material library of EBR-II- and FFTF-irradiated fuel pins available at INL, it is imperative that transient testing these materials happens at an expeditious pace to allow full harvest of priceless materials before they or needed facilities could become inaccessible (no immediate expectations for that). It also allows for a cleaner transition to studying next generation technologies.

The cladding temperature is characteristic of transient events resulting in power-cooling mismatch conditions, which are classified as Normal (startup, shutdown, power ramps, etc.), Anticipated Operational Occurrences (AOO), Design Basis Accidents (DBA), and Beyond Design Basis Accidents (BDBA). AOO are expected to occur at least once in the lifetime of a reactor while accidents are not expected to ever occur. DBA events are of strong interest in the design and development stages through licensing. In SFRs, common DBA includes Transient Overpower (TOP), Loss-of-Coolant Flow (LOF), and Loss-of-Heat-Sink (LOHS) accidents.

A special class of BDBA includes Anticipated Transient Without Scram (ATWS), where automatic scram systems are assumed to fail, and only passive reactivity feedback effects drive the response of the reactor. Although the probability of occurrence of BDBA is very low, ATWS events have been of significant interest in fast reactor safety. In part, this is because inherent safety mechanisms can be used to prevent or mitigate serious potential outcomes. Still, the significant potential threat that the consequences of very low-probability BDBA events pose to public health and safety also drives this interest. The main concerns being the SFR core is not in its most reactive configuration, the large fission product and plutonium inventory available, and the large volume of liquid sodium. Generic ATWS events that have been the focus of study are double fault events including the Unprotected TOP, the Unprotected LOF and the Unprotected Loss-of-Heat-Sink (LOHS) accidents. For reference, a summary of the range of cladding temperature response to various transients is provided in Figure 3. Longer duration transients such as shutdown heat removal system accidents are not included on the figure but can extend to several hours.

In addition to whole plant transients, local faults are also of great interest to fuel safety and performance. Examples of these include accumulated low-level effects, coolant blockages, fabrication defects, distorted geometries, and gas release into a subassembly. These effects are not explicitly planned to be studied experimentally.

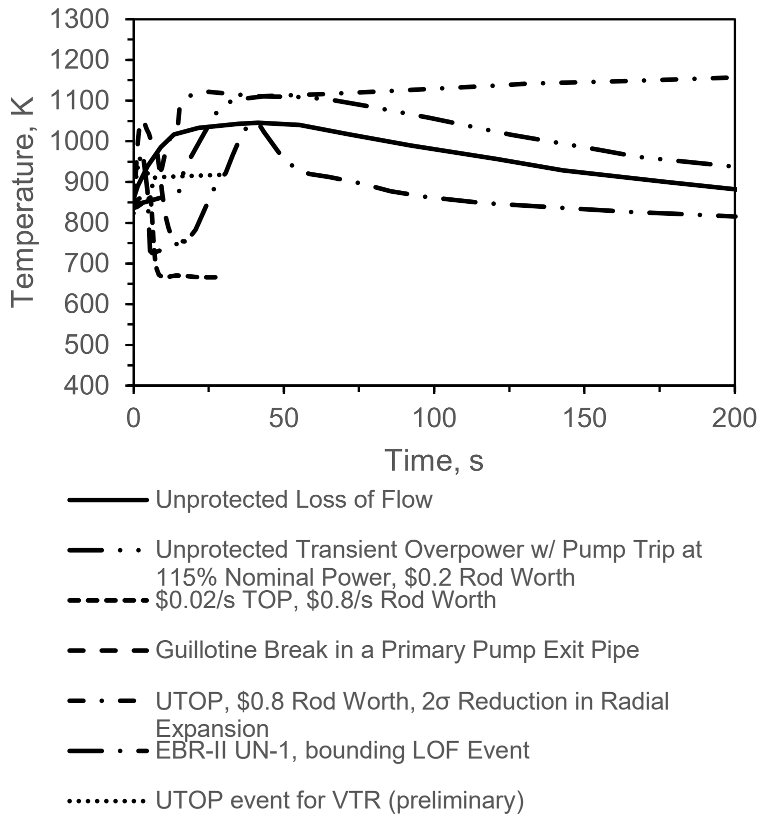


Figure 3. Overview of representative calculated peak cladding temperatures for metallic fueled SFR for a variety of events and specific conditions, adapted from [42][43][44].

2.3.1.1 Fuel-Cladding Eutectic Interaction

Primary Author(s): Colby Jensen, Luca Capriotti, Cynthia Adkins, Michael Benson

Background

Fuel-cladding eutectic liquefaction is a form of FCCI described by the interdiffusion of Fe and U across the fuel-cladding interface, that is significantly more dominant at temperature thresholds corresponding to transient events. It is distinguished from the steady-state FCCI phenomena caused by migration of lanthanide fission products to the cladding, which also diffuses into the cladding to cause loss of ductility, typically considered strengthless in the affected zone, as described in Section 2.2.1.1. It should be noted that depending on operating conditions some Fe-U interdiffusion can occur prior to transients, which can impact eutectic behavior during subsequent transients. This can happen at lower temperatures in Pu-bearing fuel. The steady-state FCCI behavior can also influence the kinetics of eutectic formation and is typically accounted for as preexisting loss in cladding thickness in calculating eutectic penetration in the cladding. In the case of transient impacts, chemical interaction is explained via the Fe-U phase diagram, ignoring the effects of other constituents such as Zr, Pu, or other fission products, revealing two key minima in the liquidus for regions near 715°C and 1080°C. As Porter and Crawford discuss, the lower temperature eutectic is formed at a relatively high concentration of uranium while the latter corresponds to a high concentration of iron [24]. While the main concern related to eutectic formation is weakening and eventual cladding breach, the eutectic may also form within the fuel meat to an appreciable level of fuel melting, again shown by figures in [24][45] where previously melted fuel zones are visible by large pores yet cladding remains well intact.

Therefore, fuel design must consider limits on temperatures at the cladding-fuel interface to manage eutectic formation (not considering advanced designs with engineered barriers). A reasonable approach used historically has been to limit the amount of eutectic penetration into the cladding to avoid any potential damage during mild transients and limit penetration. For some scenarios, an additional limit(s) may need to be considered to prevent potential relocation and its corresponding reactivity effects due to eutectic melt formation inside intact cladding. While melted fuel has been shown to be tolerable in cladding with much higher melting points, the approach used by EBR-II precluded a fraction of fuel liquefaction by also limiting the amount of cladding erosion to maintain interaction to an experimentally verified acceptable fraction of the total fuel. The primary reason for this is to avoid necessarily predicting fuel relocation effects.

Historically, evaluation of eutectic penetration rates was primarily performed using the Fuel Behavior Test Apparatus (FBTA) in addition to more integral experiments such as the Whole Pin Furnace and TREAT M-series experiments. Cohen presented an overview of measured eutectic penetration behavior in U-20Pu-10Zr in HT9 showing the typical Arrhenius-type relationship of penetration rates with temperature and a legacy correlation provides conservative prediction [45]. The paper also reports lower thresholds for eutectic formation with higher burnup fuels, associated with increasing fission product concentrations that alter the chemical makeup, lowering the activation energy for eutectic formation. Denman presented a comprehensive survey of most available data from irradiated fuel testing to refine eutectic penetration models accounting for the variety of specimen and test parameters used in historical experiments [46]. More recently, further evaluations of historical data have resulted in an additional proposed modeling approach [47].

Objectives

- Establish a validated model for eutectic penetration rates for the reference design metallic fuel system with recommended fuel design limits based on available data.
- Measure eutectic penetration via modern techniques for confirmatory purposes and to fill any to-be-identified data gaps.

Approach

Data Synthesis and model comparison. Compile all existing data for cladding wastage rates from available heating tests performed, like work by Denman [46] with potential updates to fuel irradiation histories based on modern database information. The focus will be the reference fuel design compositions. Existing eutectic penetration rate models will be used to evaluate performance and identify additional potential data gaps or confirmatory data needs. Compositional dependent temperature thresholds for the onset of eutectic formation will be identified with estimates of uncertainty. A recommendation for a modeling approach will be provided early in these evaluations with updated recommendations provided later when additional information may merit it.

A detailed experimental test matrix will be developed as described in the previous paragraph. Experimental data should come from separate effects tests (SET), like the historic FBTA setup [45], and integral experiments as are planned in the TREAT facility. SET experiments can be diffusion couples for fresh fuel materials and heating small segments of irradiated fuel under controlled temperature conditions, followed by characterization of the fuel cross section using metallography and SEM. The eutectic formation may be comprised of several eutectic compositions of combined U, Pu, and Fe with and without lanthanide elements. Each of these possible eutectic compositions will have a rate of formation and an activation energy that can be determined using differential scanning calorimetry (DSC). From these data a diffusion model and mechanism can be proposed that could be used in determining the time needed for a particular penetration depth for a metal fuel design. It should be noted that evaluation of potential fuel relocation behaviors may benefit from test geometries that are based on fuel design and segments long enough to ensure adequate interaction of melt phase with its boundaries. In the case of previously irradiated materials, comparable pre-test characterization should be performed or leverage existing data if available.

Segment heating should be done using a setup providing low uncertainty for specimen temperature. Testing irradiated fuel samples can be accomplished using a variety of furnace systems available in HFEF and IMCL. Two primary candidates for segment/small specimen testing include the Blister Anneal Furnace (BAF) for fuel segments and the DSC instrument for smaller fuel-cladding pieces. More details on the specific testing systems will be provided with development of a detailed test matrix after year 1.

Deliverables

- Year 1 and Year 4 – Report on evaluation of existing available eutectic penetration data and models, specifically focused on reference fuel design conditions, with recommendation for modeling approach and associated uncertainty.
- Year 1 – Development of test matrix to address any potential experiment gaps and providing confirmatory evaluation of key data points including specific materials to be updated in this plan or added as a separate reference.
- As available – Evaluation of integral experiments or additional data sets.
- Year 4 – Completion of test matrix experiments on reference fuel compositions.

2.3.1.2 Transient Stress Rupture

Primary author(s): Colby Jensen, Ryan Sweet, Caleb Massey, Jason Schulthess

Background

With increasing burnup and corresponding fission gas plenum pressure buildup and FCMI, fuel failure during off-normal events is increasingly driven by cladding overpressure and creep rupture. During transients, relatively high-temperature, short duration, and high stress conditions may be experienced by the cladding. Creep rupture models must be able to predict cladding behavior spanning steady-state to transient conditions.

In metallic fuel, FCMI effects tend to be mitigated by characteristics including similarity in thermal expansion coefficients between the fuel and cladding combined with relatively low temperature differences between the fuel and cladding (compared to oxide fuel) due to good thermal conductivity (and thermal diffusivity) of the fuel. In addition, the high conductivity of the fuel drives peak cladding temperatures towards the top of the fuel column, thus making the location of primary cladding degradation near the fuel plenum where fuel stresses will equilibrate with the plenum pressure. Therefore, hydrostatic forces from the plenum pressure (due to the initial fill gas and fission gas release) will dominate cladding loading.

Kramer et al. provides a summary of the state-of-the-art for creep rupture predictions for HT9 showing a disparity in creep predictions developed using data from long-duration, low-temperature conditions, and those from data from short-duration, high-temperature conditions [48]. Transient stress rupture data was collected using the Fuel Cladding Transient Tester at Hanford comparing the effects of material lots, temperature ramp rates, hold temperatures, and irradiation effects using "burst-test" style experiment techniques. DiMelfi hypothesized the differential effect to be caused by a microstructural change induced at high temperatures [49]. These effects are important since the timescales of interest correspond to many of the more relevant transient events, and they directly impact rupture predictions. Historically, some SET was performed using constant-rate, high-temperature tensile testing in effort to measure these effects [50]. This issue appears to remain unresolved in public literature.

Objective

Establish a data-validated model for transient stress rupture for claddings considered reference design with recommended fuel design limits based on available data.

Approach

The first step in this task area will be to establish a database of all relevant data and models to perform independent evaluation of the performance of existing models and especially confirm issues of time-dependent effects on material strength. Available data is primarily available from testing done in the FCTT using segments of fresh, irradiated, and irradiated defueled cladding tubes [51][52]. Other important data include whole pin furnace experiments (include more integral effects like cladding wastage) and the tensile test data from [50]. These activities will be coordinated with the creep model development described in Section 2.4.1.3. The second step will be the development of a recommended test matrix based on findings from existing data and models. In the event of further merited investigation of microstructural impacts on transient properties, separate effects testing may be performed to evaluate time/rate property dependencies on fresh materials. Additional rupture testing could be performed using existing laboratory cladding burst facilities. Hot cell testing options will be evaluated but would likely include the BAF in HFEF or the cladding burst system at INL, the Severe Accident Test Station (SATS) systems (in and out of cell) or modified burst system at Oak Ridge National Laboratory, and/or potentially new capability corresponding with Section 2.3.2.2. The experimental plan could include testing fresh materials from existing material lots, including those from historic programs, and testing defueled irradiated segments from EBR-II and/or FFTF. Additional testing may be suggested to better identify the effect of the heating rate on the changes in the microstructure and subsequent impact on the creep strain rate.

Deliverables/Schedule

- Year 0 and Year 5 – Report on evaluation of existing available data and models, specifically focused on reference fuel design conditions, with recommendation for modeling approach and associated uncertainty.
- Year 1 – Development of a test matrix and testing strategy to address any potential experiment gaps and providing confirmatory evaluation of key data points including specific materials (to be updated in this plan or added as a separate reference) with recommendations of future experiments to address data gaps.
- Year 2 – Evaluation and incorporation of additional data sets for creep and other constitutive properties investigated across the program.
    - Implementation of resulting model into fuel performance tool compare against experimental data.
    - Statistical analysis of data in coordination with ongoing modelling efforts can identify the most sensitive material parameters for expected cladding environmental conditions and the highest sources of uncertainty to the material failure calculation.

2.3.1.3 Fuel Melting and In-Pin Relocation

Primary Author(s): Colby Jensen

Background

Fuel melting is a form of degradation of nuclear fuels that is traditionally precluded by design criteria. Normal and off-normal conditions may be affected by fuel melting considerations. Power-to-melt is a common relationship used to define margins and can be limiting in some circumstances. In this case, fuel melting refers to bulk melting of the fuel, and not localized low melt eutectic phases. In metallic fuels, fuel compositions reach liquidus at temperatures below that of the cladding so the negative consequence to the cladding are generally much less severe as compared to more traditional ceramic fuels. It is also notable that due to compositional variation across the fuel radius, melting may not first appear at the centerline. An additional factor that can promote bulk fuel melting is described in Section 2.3.1.1 where primarily Fe in the cladding can propagate into the fuel above ~715°C to form low melting eutectic compositions that can induce significant bulk melting under specific time at temperature conditions.

During severe accident conditions, the in-pin fuel motion of metallic fuels has been observed as extrusion of molten fuel into plenum, driven by fuel density changes and expansion of closed porosity in the fuel (and maybe sodium vapor), providing beneficial negative reactivity effects. Quantifying these relocation effects has been done effectively using the TREAT hodoscope in the six historic M-series experiments. Additional exploration of these processes could provide higher confidence models to support plant safety analysis.

The margin to melting is governed by input power, thermal conductivity, volumetric heat capacity in transients, the coolant boundary condition, and the liquidus temperature of the fuel composition. The specific fuel performance issues of interest are the thermal properties of the fuel corresponding with local material phase, composition, and temperature.

Objectives

Establish data and analysis tools to evaluate pertinent fuel melting and in-pin fuel relocation effects.

Approach

The material properties characterization and fuel-cladding eutectic studies described in Sections 2.4.2 and 2.3.1.1, respectively, will provide needed data to support fuel melting evaluations. Historic TREAT M-series experiments provide useful reference data to evaluate integral performance simulations. Integral fuel experiments will also be used where possible to extract data regarding fuel melting thresholds to compare with models. In addition, TREAT experiments will provide in-situ fuel motion data to support validation of in-pin fuel motion under severe accident conditions. TREAT experiments could also be designed to specifically target these behaviors if ongoing experiments are not enough.

Deliverables/Schedule

- (covered in Section 2.4.2) Establish requisite thermal properties and melting points with quantified uncertainties.
- Year 1 – Establish a threshold for fuel melting quantity to avoid fuel relocation based on existing data or new experiments.
    - Year 1 – Develop and implement fuel melting model framework, which includes enthalpy of fusion, to simulate melt volume and distribution in simplified experiments.
    - Year 2 – Implement accumulated thermal properties for melted fuel into the BISON melting model and compare simulations against M-series TOP test measurements.
- (as part of ongoing experiments described in Section 2.3.1) – Demonstrate and validate fuel melting under integral physics experiments in TREAT and/or furnace experiments.

2.3.1.4 Post-Failure Consequences - Thermomechanical

Primary Author(s): Colby Jensen

Background

Metallic fuel is fully compatible with sodium coolant with no propensity for chemical interaction. This characteristic is probably most notably demonstrated by the historic Run Beyond Cladding Breach experiments in EBR-II. In this case, the fuel was not subjected to power-cooling mismatch as in a transient but was artificially compromised to cause cladding breach while continuing extended operation of the fuel with primary consideration being fission gas released into the primary coolant. During severe accident conditions, breached fuel could have important thermal interaction with the coolant and concerns about compromising primary containment, recriticality, and coolant blockage are considered.

The historic M-Series experiments in the TREAT Mk-III sodium loop and experiments in the CAMEL facility are examples of notable experimental approaches used to characterize post-failure behaviors in full sodium flow condition representing conditions of a severe accident. Metallic fuel breach in these conditions has typically proven favorable results, with fuel breach at the top of the fuel column (the hottest location during irradiation lifetime and during the transient due to good thermal transport properties.) Molten fuel has been observed to be driven out of the breach location and carried away from the core to a less reactive configuration, without significant tendency for blockage formation. The general favorable response of the fuel in these conditions is in part due to the small temperature difference between the coolant and the cladding in conditions where fuel failure might occur. This also mitigates fuel-coolant thermal interactions and potential for generating mechanical forces that might threaten neighboring rods or primary vessel containment.

Some of the key opportunities for study include evaluation of fuel relocation especially for prototypic length fuel pins (> EBR-II length), propensity for fuel blockage under limiting scenarios, fission product transport measurements, and measurement of molten alloy thermal transport properties.

Objectives

Perform experiments and analysis to evaluate post-failure fuel behaviors that reduce uncertainties in metallic fueled SFR safety analysis.

Approach

The AFC program does not plan to address this topic in a comprehensive manner. The general strategy is to leverage other activities to provide additional data and results that support relevant objectives. Among other topics, this activity area is one that could likely provide benefit from collaboration between the AFC program and the Fast Reactor Program.

Fuel alloy property measurements described in Section 2.4.2 will target measurement (and development of measurement techniques) of fuel alloy thermohydraulic properties above alloy liquidus temperatures. Section 2.3.1 provides an overview of planned additional TREAT experiments that could provide additional insights on post-failure behaviors on prototypic length fuel pins and potential small fuel bundle experiments.

Deliverables

None specified here at this time – see sections referenced above.

2.3.1.5 Post-Failure Consequences - Source Term

Primary Author(s): Fidelma Di Lemma

Background

Source term is defined as the amounts and types of radioactive material released to the environment following a severe reactor accident [53]. More specifically, this includes the characterization of the radioisotope quantity and form migrating from the fuel to the coolant, into the containment, and finally to the environment. Various interconnected phenomena and scales play a role in accidents leading to radioactive release. Source term studies provide the means to characterize the consequence of low-probability accident events and govern the response actions after an accident (such as restrictions and decontamination measurements). Moreover, source term determination is required for licensing and construction of nuclear power plants, as it determines the extension of the evacuation and exclusion zones after an unplanned transient event.

While substantial work exists for LWRs and guidelines for their source term determination are provided in NUREG-1465 [54], limited work exists for SFRs. Current knowledge on source term and uncertainties are summarized in [55][56]. Uncertainties on fission product release and further research topic needing investigation are summarized in the following as described in [55]: 1) several radionuclides from pin (I, Cs, Sr, Eu) need further studies on their transport and release, especially in high burn up fuels and near melting; (2) the behavior of molten fuel and debris dispersed in sodium (focusing on realistic sodium chemistry); (3) fission products transports, solubility and release in bubbling sodium and during vaporization to the cover gas or spray fire; (4) fission products chemistry and interaction with structural materials and sodium, (5) and finally sodium-concrete interaction [53][57]. Moreover, [58] reported that fission product and actinide partitioning between liquid, vapor/gas and aerosol phases during evaporation of contaminated liquid sodium into air is not well understood.

Other transient testing plans, described in detail in session 2.3, provide a unique opportunity to accomplish this work as a much smaller effort compared to specific dedicated task.

Objectives

The proposed studies will investigate fission products off-normal behavior and their release after a transient. This will support the development and reduction of uncertainties in predicting source determination. The data will be especially applicable to model source term code, which is a vital tool for the future licensing of SFRs.

Approach

The proposed studies will respond to the knowledge gap described previously by integrating SET (separate effect testing) and IET (integrate experiment testing). SET can provide a deep mechanistic understanding of fission product behavior in accidental conditions; and thus provide the knowledge basis to understand more complex phenomena occurring during severe accidents. SET can also help to improve and validate mechanistic models thanks to their separate variable approach and the possibility of studying simple systems of importance for modelling. On the other hand, IET provides validation of understanding derived from SET especially potential interdependencies between individual phenomena and aid in identifying specific effects that become targeted by SET. In the near term, this R&D plan focuses on SET leveraging existing capabilities and materials, including performing PIE on archive samples and in-situ tests on fission products chemical form and transport. These studies will provide the science-based understanding needed to further develop SET and IET plans. IET will plan to test high burn up irradiated fuels using the TREAT facility in coordination with the other R&D focus areas. Specifically, these tests should include a focus on fission product transport in sodium requiring online monitoring of fission product release from in-pile experiments. The proposed studies are summarized in Figure 4 with a short description of key topics following the figure.

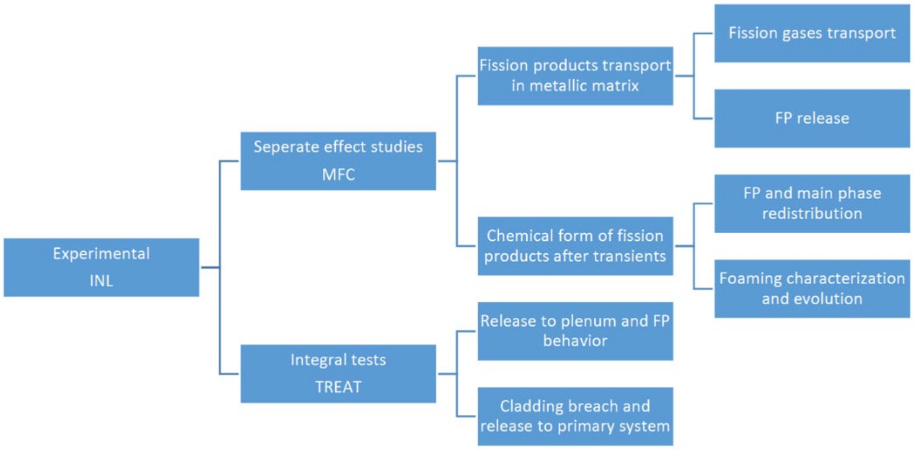


Figure 4. Overview of the proposed research plan for fission product transport and source term determination.

1) **SET on fission gases transport in metallic fuel.** Although integral tests on fission gas release exist, a mechanistic understanding of fission gas behavior in metallic fuel is still lacking. This study aims to provide insight into fission gas diffusion by SET studies. In situ tests monitoring fission gas diffusion and bubble behavior in irradiated fuel are proposed. The data obtained from these tests could provide insight for further a mechanistic microstructural modeling of fission gas behavior in metallic fuel. See also Section 2.2.1.2.

2) **SET on fission products release from metallic fuel.** Limited knowledge exists on the release during accident conditions and on plenum pressurization. This study aims to explain these behaviors, integrating different scale testing, from the nano to the macro scale. The experimental plan to be developed will integrate different available techniques, such as TEM studies of small irradiated material (lamella), to evaluate fission product behavior during in situ heating, micro testing of FP release from fuel fragment by thermogravimeter-differential scanning calorimeter (TG-DSC), mass spectroscopy (MS), Knudsen Effusion Mass Spectrometer (KEMS) measurement, and fuel pin failure testing in-furnace coupled with monitoring of the release by mass spectrometer. These studies will support further IET focus on online fission product monitoring of irradiated pins in TREAT, such as the experiments described in Section 2.3.

3) **Fluff evolution.** On the top of sodium-bonded metallic fuel pins, there is a particular microstructure named "fluff". In the "fluff" region, fission gases (Xe and Kr) and volatile elements (Cs, and possibly I) are believed to accumulate in a highly porous fissile matrix (containing U and Pu). This could be relevant for source term analyses. Evolution of the foam under thermal transients could be an investigation in a transient furnace. This will also provide information of possible ejection to higher axial position of fuel materials, influencing reactivity. This work is currently being performed under a parallel effort with participation of AFC team members.

4) **SET on Chemical form of fission products after transients.** Understanding the chemical form of fission products before and during transients is of paramount importance to evaluate their transport and release. While data on FP in normal operation exist from various PIE campaigns, some gaps exist in understanding their behavior at high temperatures [10]. Archive samples could be analyzed using advanced microscopy techniques to understand fission product chemical form after a moderate power ramp in a transient furnace, to be compared with future examinations of irradiated samples that will undergo transients in TREAT. Pending these initial results, further advanced characterization is proposed

to include focused ion beam (FIB) preparation for transmission electron microscopy (TEM) analyses and in situ micro testing.

5) **IET Release to plenum and FP behavior.** This proposed experiment aims at characterizing fission product release during transients. Irradiated samples to be identified will undergo transients in TREAT. These studies will focus on plenum pressurization, and PIE will be performed on the samples after transient to study FP migration. These studies will be linked to the planned SET furnace testing, described in session 2.3. They will also focus especially on high-burnup fuel up to partial melting, as these have been identified as the necessary area of further investigation.

6) **Cladding breach and release to primary system.** Further studies will focus on online monitoring of fission products and fuel release after fuel pin failure, which will be initiated by reactor transient (in TREAT). These tests will utilize panned integral tests on irradiated materials with plans to fail pins and will necessitate a sodium loop as described in detail in session 2.3. A new online system would need to be designed and tested for fission gas monitoring. Such work will connect this experimental plan to the work started in the M-test and will aim to advance the knowledge on fission product and debris release from fuel to primary system and their interaction. This data will be fundamental as input for the M&S code for source term determination.

Deliverables/Schedule

- Year 2 – Report/paper on coupling modeling and in situ study
- Year 2 – Report/paper on fission products behavior after a 30% overpower transient from X512
- Year 2 –Report/paper on fission gas release in TREAT and SET (see Section 2.2.1.2).
- Year 5 – Report on fission product behavior during severe TOP and LOF experiments based on IET and SET experiments planned mostly in other sections.

2.3.1 Transient Experiment Series

2.3.1.1 ARES Project

Primary Author(s): Colby Jensen

Background

The Advanced Reactor Experiments for SFR fuels (ARES) project is a current major initiative in safety testing metallic fuels [59]. It is a collaboration between Japan Atomic Energy Agency (JAEA) and the DOE AFC program facility sharing initiative aimed at testing transient performance of fast reactor fuels. The DOE portion of that collaboration primarily focused on capability development in the Temperature Heatsink Overpower Response (THOR) test device for TREAT and metallic fuel performance. The capability also includes development of remote assembly and disassembly equipment in HFEF for previously irradiated test pins.

Objectives

- Establish a capsule testing device in TREAT for testing SFR fuels.
- Support the JAEA objectives to evaluate the behavior of high-burnup, advanced-design MOX fuel during slow TOP conditions.
- Evaluate the transient performance of fresh U-10Zr fuels, measure the creep rupture influence in a high burnup metallic fuel pin, and establish in-pile LOF performance to compare with WPF experiment results. The experiment results will add to a limited database for fuel performance code development and validation.

Approach

The TREAT facility will be used to impose necessary heating to test specimens housed in the THOR capsule described in Section 2.3.2.1. The ARES project consists of three distinct experiment test series including:

- The THOR-C experiments are being used to commission the THOR capsule while characterizing the behavior of fresh metallic fuels during a range of transient conditions from TOP- to LOF-like conditions in an SFR. Table 1 provides the THOR-C test matrix.

Table 1. Experiment test matrix for THOR-C series. All test specimens are fresh fuel whose actual lengths may vary slightly from the given values.

Test ID	Composition/ % U_{235}/ Geometry/ Cladding	Fuel OD/Length Cladding OD/ID (mm)	Internal Pin Pressure at RT (MPa)	Initial Fuel Temp (K)	Peak Cladding Temp (K)	Time to Peak Cladding Temp (s)	Notes
THOR-C-1	U-10Zr/69.6/ Solid/HT9	4.3/343 5.8/4.9	0.1	500	<900	N/A	Calorimetric calibration to satisfy facility safety requirements, gamma spectrometry for detailed axial power characterization
THOR-C-2			1.5	500	1400	25	Test pin failure detection diagnostics, explore fuel failure for relatively slower power transient
THOR-C-3 (A,B)	U-10Zr/69.6/ Annular/HT9	2.46 ID/ 4.88 OD/ 305-343 5.8/4.9	0.1	500	900	N/A	Annular fuel, oscillating power, thermal conductivity trial
				~~500~~	~~1400~~	~~25~~	~~Annular fuel, programmatic calibration, fuel overheating stability~~
THOR-C-4*	U-10Zr/69.6/ Solid/SS316	4.3/343 5.8/4.9	0.1	500	1400	8	Hodoscope qualification, tieback to historical TREAT M-series
THOR-C-5*					1400	100	Hodoscope qualification, LOF simulation
THOR-C-6*			1.5		1200	100	Hodoscope qualification, LOF simulation

- THOR-M experiments will test previously irradiated metallic fuel pins from EBR-II. Table 2 provides the THOR-M test matrix.

Table 2. Experiment test matrix for MTOP/MLOF series using EBR-II irradiated fuel pins.

Test #	Fuel Composition/ Burnup (at%)/Cladding	Initial Temperature (K)	Peak Temperature Target (K)	Time to Peak Temperature (s)	Notes

MTOP-1 (A,B)	U-20Pu-10Zr/ ~10-15%/HT9	500	Low-Energy Calibration		Verify specimen power coupling
			1400	25	Targeting conditions similar to historically planned M8 test
MLOF-1 (A,B)	U-10Zr/ ~7.5%/HT9	500	Low-Energy Calibration		Verify specimen power coupling
			1150	100	Targeting thermal conditions similar to historic furnace test

- THOR-MOXTOP experiments will test high burnup MOX pins irradiated in EBR-II. The details of these tests are protected under a CRADA agreement. The test matrix may be found in [59].

Deliverables/Schedule

- Year "-3" – Complete final design of all experiments.
- Year "-2" – Complete necessary commissioning experiments.
- Year "-1" – Complete remote assembly capability for THOR device and first irradiated MOX test in TREAT
- Year 0 – Complete first irradiated metallic and MOX experiments
- Year 1 – Complete second irradiated metallic fuel experiment
- Year 2 – Complete PIE on first irradiated MOX and metallic fuel experiments
- Year 3 – Complete analysis and documentation of results

2.3.1.2 Future Na Loop and Furnace Testing

Primary Author(s): Colby Jensen

Background

The behavior of metallic fuels under TOP and LOF events in "integral" physics conditions has been accomplished using the historic TREAT sodium loop in the M-series and in the WPF in the FM-series tests. The historic M-series tests included six primary experiments with a total of 15 EBR-II fuel pins. One pin with U-10Zr in HT9 cladding with 2.9 at% burnup was tested in the final test with more plans to test HT9-cladded pins. The tests simulated severe TOP conditions with great success in identifying fuel degradation behaviors, fuel failure thresholds, fuel relocation effects, and some indication of post-failure blockage potential. In addition to these, more TREAT tests were at various stages of planning for LOF conditions, small bundle effects, and prototypic length pins irradiated in FFTF in the MFF/IFR series. The WPF FM experiments included seven total experiments with six having HT9 cladding with maximum burnup of 13.5 at%. The WPF was used to simulate integral behaviors consistent with LOF conditions (longer duration temperature exposure with near negligible internal heat generation) to the point of failure.

Much of the planned testing was focused on the IFR design, which has similar characteristics as modern metallic-fueled SFR designs. Therefore, some of the planned experiment objectives are still applicable today. In particular, testing programs that shore up passive safety strategies are important to better arm licensees and the regulator with data that reduces uncertainties associated with the fuel/core behaviors. For example, the inherent behavior of metallic fuel during an unprotected LOF event has potential for more credit to shut down a reactor prior to sodium boiling that can lead to further reactivity insertion. Another example is testing of prototypic length fuel pins from FFTF can provide first experimental validation of fuel degradation to failure to better quantify failure location and post-failure dispersal consequences.

Based on review of the historical DOE planning, literature, and modern data needs, the following experimental goals are of interest using the TREAT sodium loop, THOR capsule, and a furnace capability:

- First prototypic length fuel testing (FFTF MFF pins)
- First in-pile LOF testing (beyond EBR-II SHRT tests)
- Higher burnup (>10 at%)
- Modern/Novel fuel designs (He-bonded, fuel-cladding barrier, etc.)
- MA-bearing fuel
- Bundle testing for pin-pin interaction and post-failure behavior.

Objectives

Develop experiments and tools to close remaining data gaps and reduce fuel performance and failure uncertainties that aid passive safety reactor studies.

Approach

The TREAT sodium loop is under development now to first support testing under the Natrium™ reactor demonstration project at INL (see Section 2.3.2.1). Significant infrastructure to support testing, assembly, hot-cell handling, PIE is now underway at INL. The AFC program is directly establishing capability to do out-of-reactor sodium loop testing using the Modular Sodium Test Loop (MSTL) undergoing installation in the Idaho Engineering Demonstration Facility at INL. The TREAT sodium loop will be available to begin testing around the end of 2025 but with significant non-AFC usage in the first couple years. Preliminary discussions have begun to start developing a joint test program with international interests to do transient testing of metallic fuels in addition to MOX fuels. These plans will continue to develop in the coming years targeting 2026 to begin.

Transient furnace testing is also under evaluation as described in Section 2.3.2.2. These experiments should be performed in the same timeline starting around 2026.

Deliverables/Schedule

- Year 0 – Complete operational preparations to begin testing in MSTL
- Year 0-1 – Develop joint sodium loop test plan
- Year 2 – Begin TREAT testing and furnace testing
- Year 3 – Complete prototypic length fuel pin testing in TREAT and furnace
- Year 4 – First small bundle experiment in TREAT on metallic fuel
- Year 5 – Summary of reference fuel design transient performance

2.3.2 Transient Testing Capabilities

TREAT uniquely provides nuclear heating options for testing fuels under a range of relevant conditions, with primary focus on accident conditions. The total energy available in a single transient is a primary constraint to TREAT's application to testing fuels for SFR applications. ATR could potentially be used for longer duration transient tests, in particular, operational power ramp testing if it is deemed necessary, although a fast spectrum test reactor would be more ideal. As shown in Figure 5, these facilities provide capability that spans the applicable range of fuel failure thresholds for reference cladding types. In any case, all near term metallic fuel transient tests are formulated around use of TREAT and furnaces.

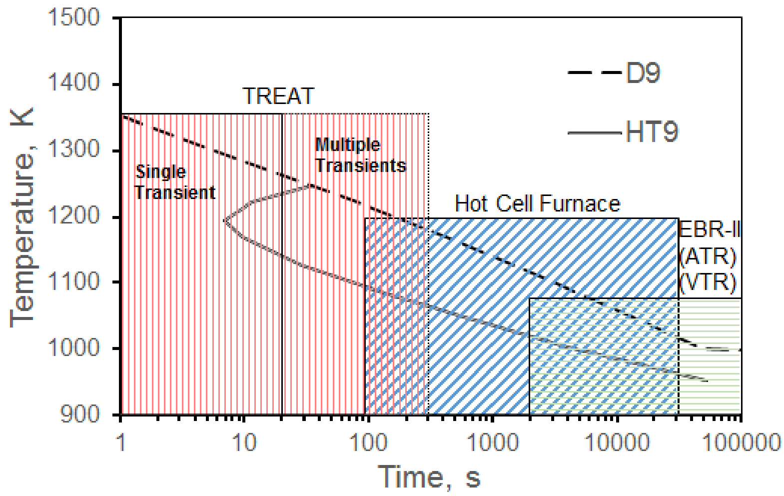


Figure 5. Relevant cladding temperature failure thresholds and testing facilities that can provide corresponding capability (adapted from [42]).

2.3.2.1 In-Pile Devices

Primary Author(s): Nicolas Woolstenhulme

Background

The Temperature Heat-sink Overpower Response capsule (THOR) is currently the only sodium-environment transient test capability in world and is useful for single pin transient tests (up to EBR-II length). Even when the Mk-IIIR sodium loop is fully deployed, THOR will still remain in use for some tests since its construction and assembly will cost less. THOR is not large enough to support transient studies on bundles or longer fuel pins such as the FFTF-MFF pins. THOR is a static sodium capsule which removes heat from test pins through conduction to a solid metal heat sink. This approach is adequate for some types of test objectives, but the Mk-IIIR flowing sodium loop will be needed to support long pins, bundle tests, and more prototypic thermal hydraulic conditions. The Mk-IIIR sodium loop is a modernized version of the historic Mk-III sodium loop. This loop will recirculate sodium for prototypic heat transfer conditions and can support long pins (FFTF MFF pins) in single flow tubes or as small bundles to evaluate pin-to-pin interactions. The Mk-IIIR sodium loop design represents the most prototypic SFR transient testing device, but preparation of a given test will be more costly owing to its specialized features and magnitude of hardware.

Objectives

Establish capability to study fuel behaviors under a range of power-cooling mismatch conditions corresponding to operational to accident transients, including to fuel failure.

Approach

At present the THOR capsule has been deployed and used in a handful of transient tests with good results in most cases. As with all first-of-a-kind systems, these commissioning tests revealed some design refinements that will be needed to facilitate assembly and improve reliability. These modifications are underway, and the updated-design THOR capsules will experience first use in HFEF and TREAT shortly. Following these successes, THOR will be considered fully qualified and commissioned for use.

The Mk-IIIR sodium loop is under development and nearing the final phases towards its deployment in TREAT. Much of the Mk-IIIR design, supporting infrastructure, and inaugural irradiations have been or will be undertaken as a TerraPower collaboration with INL. AFC will develop a second design package of test trains, instrumentation, and transient condition designs for use in the Mk-IIIR sodium loop to make it more broadly useable and accessible for research sponsored directly by DOE. A selection of these tests will be conducted to commission the sodium loop and to fill key data gaps described elsewhere in this document.

Deliverables

- Year 1 – Completion of THOR Design Updates
- Year 3 – Complete AFC-sponsored sodium loop test design

2.3.2.2 Transient Furnace

Primary Author: Colby Jensen

Background

From the chart shown in Figure 3, some relevant time-at-temperature regimes are out of reach of both the TREAT facility and steady-state test reactors. Fortunately, these domains also correspond to relatively long duration events compared to the thermal-time constants of metallic fuels so that the impacts of internal heat generation and temperature gradients are usually small (exception being some power ramps with active cooling). These conditions lend themselves to non-nuclear-heated testing in furnaces. Additionally, thermal effects on cladding performance are commonly studied using furnaces, including for transient heating and pressurized conditions to understand the effects of ramp rates on fission gas release on cladding tubes.

The whole pin furnace (WPF) was used historically to study effects on integral fuel pins and the FCTT was used on pressurized cladding segments. These tests allowed more prototypic temperature conditions corresponding to many of the most relevant transient events for pool-type SFRs where time-at-temperature durations span mins to hours. Currently, available potential facilities to study these behaviors include the BAF and SATS facilities. However, due to the logistical hurdles of testing fast reactor materials at INL in SATS and the limited capability of the BAF, a custom transient furnace may be needed to efficiently address data gaps.

Objectives

- Establish a heating capability to evaluate integral irradiated-fuel segment (pressurized) behavior during postulated transients beyond TREAT’s reach.

Approach

Recent evaluations of transportation logistics and potential testing requirements have led to pre-conceptual design of a transient furnace capability at INL for testing pressurized fuel segments [ref]. The current design strategy is to contract the furnace design to a private company and then perform necessary qualifications to install in a hot cell. Additional potential cross-cutting applications of this system are also under consideration for justifying this capability including fission gas/volatile product release and advanced deformation measurement capability. Development of this design is planned to proceed until one of the following potential conclusions is reached: an alternative option is found, identified data needs change, resource limitations, or completion.

Deliverables/Schedule

- Year 0 – Completed design of a furnace system to do pressurized fuel segment testing in a hot cell.
- Year 2 – Pending first deliverable results – complete out-of-cell fabrication of furnace system

- Year 3 – Phase III qualification of furnace system in a hot cell

2.4 Material Properties and Performance

2.4.1 Cladding Properties and Testing

2.4.1.1 Mechanical Properties

Primary Author(s): Doug Porter, Mychailo Toloczko, Caleb Massey, Ryan Sweet

Background

Fuel cladding is highly important to the success of most all fuel designs as the cladding is used to contain the fuel to an expected geometry and to prevent the release of fuel and fission products to the primary coolant of the reactor. To date, the cladding materials have been Fe-based, or, in some cases, Ni-based for sodium-cooled fast reactors. In a few cases Zr-based alloys were used but proved not to be adequate for the operating temperatures. Nb-based alloys were also tried. In recent years Fe-based steels, most of them stainless steels (Cr content ≥ 10.5 wt.%, C < 1.2 wt.%), either in austenitic or ferritic (body-centered) crystal structures. Type 304, 316, and advanced 316 or 15Cr-15Ni based alloys (D9 type) have dominated the austenitic alloys, and HT-9 and 9Cr-1Mo the ferritic alloys (actually, ferritic/martensitic steels). The latter alloys were chosen in many instances because of their known resistance to void swelling caused by neutron irradiation. The advanced austenitic alloys will also be considered, because these alloys are at least as strong to operating temperatures proposed for many reactors and may have adequate swelling resistance. They are also not so sensitive to low-temperature embrittlement. For the current AFC program work, HT-9 has been the alloy of choice due to its large database of properties in a reactor environment. However, if the data gained from irradiation of the advanced austenitic alloys is examined, the swelling resistance of some of these may be adequate for use in many SFR designs.

As the database for other alloys, such as the oxide dispersion strengthened (ODS) types, have shown that operating temperatures can exceed those formerly thought challenging to cladding material strengths and stress-rupture characteristics, there may be reasons for further examination of these advanced materials.

A problem for design and fuel pin performance modeling, even for those cladding materials that had a large database, is that newer data and modeling techniques have shown that some of the design equations for these properties have room for improvement. A full review of the equations and their bases should be done. The last time a complete review was conducted was for the last revision of the FCRD Materials Handbook in September 2014. Recently, a new AFC report [60] expands on the Pacific Northwest National Laboratory (PNNL) reports and brings forth a plan for future work to close the gaps. These findings will be the subjects of ongoing R&D.

Objectives

Generate a testing program, analyses, and design equations useful for design, engineering, and model validation of HT9 and CWD9 materials.

Approach

Performance data will be prioritized to be obtained for realistic combinations of fuel cladding irradiation temperature and dose. Creep data should emphasize realistic hoop stresses. Will need to rely on historical data and materials for highest dose data. Low dose reactor conditions (bottom and top of core) can potentially be evaluated with new neutron irradiations.

Deliverables/Schedule

- Year 1: Identification and procurement of a large-batch campaign-specific heat of HT9 for experiments to close identified data gaps
- Year 2-4: standard-sized isothermal creep tests for model development/validation

- Year 2-5: Comparison of tensile, fracture toughness, and creep properties of current (HT9, T91, CWD9) and next generation (Gr92, I, ODS) cladding concepts

2.4.1.2 Testing of High Dose Irradiated Cladding Materials

Primary Author(s): Stuart Maloy, Caleb Massey, Tarik Saleh, Ben Eftink

Aside from FCCI and creep being the predominant failure modes for fuel/cladding combinations in fast reactor environments, irradiation-induced degradation to representative end-of-life displacement damage levels must be assessed for both the leading cladding candidates (HT9/D9). This data is used to develop improved radiation tolerant cladding in the future and also for developing models for predicting the effects of irradiation on mechanical properties and microstructure in these materials. Specimens are available from previous irradiations in FFTF, BOR-60 and the Phenix reactor to doses ranging from 100 to 250 dpa at irradiation temperatures from 350 to 650C.

A recent milestone report has highlighted various gaps in the current understanding of HT9 degradation as a function of composition, irradiation temperature, and irradiation dose. Specifically, for operating temperatures representative of the cold legs of current sodium-cooled fast reactors (~300-350C), irradiation-induced hardening and embrittlement is still a concern for ferritic/martensitic alloys such as HT9. Although it is well known that the defect concentration at lower irradiation temperatures saturates at doses exceeding ~10 dpa, it is not yet known whether additional compounding effects including irradiation-enhanced segregation and precipitation worsen hardening/embrittlement effects at extremely high doses (> 80 dpa). This gap is highly synergistic with the need for post-irradiation irradiation hardening data on various heats of HT9, of which there are 8 available heats of HT9 that have been irradiated to doses up to 180 dpa in prior FFTF irradiations.

In addition to understanding the limits to low-temperature hardening behavior, additional higher dose irradiation data (>200 dpa) at typical void swelling temperatures (400-500°C) is needed to investigate in reactor steady state swelling rates and possible issues from radiation induced segregation at extreme doses. At these intermediate temperatures, irradiation-induced segregation and precipitation may also affect cavity nucleation and growth, necessitating the need for detailed microscopic investigations on prior FFTF/MOTA and BOR60 irradiated samples of HT9. Testing data is also needed to fill gaps in knowledge on radiation effects in CWD9. Significant studies have shown the improved swelling resistance in D9 but a better understanding is needed on the effect of additions of silicon and phosphorus on the incubation period.

Deliverables/Schedule

- Year 1 – retrieve samples from BOR-60 and CEA hot cells and develop an inventory of samples available for testing.
- Year 2-3 – perform mechanical testing and microstructure characterization on the samples filling gaps in previous data on irradiation effects in HT-9 and obtaining new data on advanced ferritic/martensitic steels and ODS steels.
- Year–4 - Summarize data and perform additional tests if needed. Collect data to add to handbook and provide input for model development.
- Year 5 – Based on the results from testing initial materials from high dose irradiations select additional testing to improve the knowledge base or suggest future irradiations to obtain this data in the future.

2.4.1.3 Cladding Creep

Primary Author(s): Doug Porter, Laurent Capolungo, Yachun Wang, Ryan Sweet

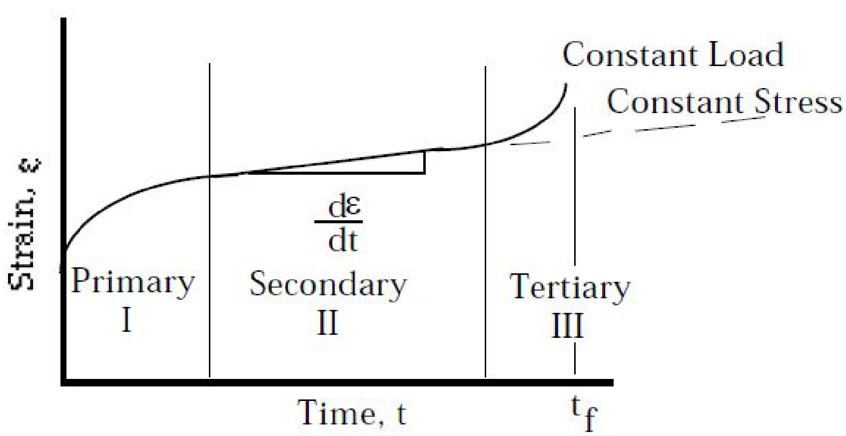
Background

Design bases for the fuel in recent SFR designs (e.g., EBR-II/Integral Fast Reactor [7], VTR [24]) have chosen cladding strain limits (thermal creep strains) or Cumulative Damage Fraction (CDF). CDF is an attempt to use stress rupture data to predict when a pressurized tube will fail. Typically, the thermal creep limit is 1-2%, and the CDF limit is 0.05 CDF=1 indicates failure). Statistics are then applied to show that only one fuel rod in a core loading will fail at the exposure (burnup) chosen as a limit for a given rod design and operating conditions.

Recent unpublished analysis has shown the two limits, creep strain and CDF only coincide well for pressures and temperatures not representative of operating conditions of typical SFR fuels. Also, to demonstrate that only thermal creep causes cladding damage, early papers were located showing results for experiments using irradiated pressurized tubes. These experiments, using CW316 (130 dpa) [61] and HT-9 (200 dpa) [62] creep at low temperatures (~400 °C), reveal that creep at these temperatures (only irradiation-induced creep) does not cause tubing rupture. The tubes demonstrated diametral strains of >12% for CW316 and >5% for HT-9 without rupture. For this reason, our team has chosen to look closer at the thermal creep strain limits as there is much more data available on which to verify accuracies of design equations.

However, the current design equations for thermal creep are not suitable for prediction of strain under the changing operating conditions experienced by a typical fuel rod. The design equations are based upon data taken using pressurized tubes (constant stress, constant temperature), while fuel rods begin operating at maximum temperature and minimum stress (no fission gas, so absolute pressure = 1 atm). As the fuel burns out, the internal pressure due to the release of fission gas increases, but the operating temperature decreases as the fuel's fissile content decreases.

The thermal creep equation, for a pressurized tube looks like,



The primary creep for the pressurized tube experiment only occurs once, in the tubes based on the beginning microstructure of the material. The sample is heated rapidly, and the temperature and stress are now engaged. The temperature thermally activates the ease of motion of dislocations and precipitation may occur and/or precipitates are coarsened, Precipitation-related events can effectively pin the dislocations, restricting motion or coarsening could unpin them, allowing them to create strain in the material. As the dislocations become mobile, or unpinned, the primary creep strain is realized and then stops. Primary creep is effectively complete.

Once secondary creep begins, the process becomes self-evolving as defects/dislocations are generated and create strains linearly with time, the rate depending on stress, temperature and the creep mechanism involved. Tertiary creep begins and the tube quickly strains to failure. One can quickly see that primary and secondary creep in the cladding of a fuel rod, do not follow this simple pattern. Primary creep is perhaps the most difficult to predict as the stress and temperature conditions change throughout the life of the fuel rod. The cladding is also being irradiated, creating more defects. The point of all this discussion is that the cladding of an operating fuel rod is subject to much different conditions than is a simple pressurized tube in a furnace, or even in a reactor. Currently, design equations used within fuel behavior models may not be expected to provide accurate results, as they are generated from idealized conditions.

Recently, researchers at Los Alamos National Laboratory have been developing a thermal creep equation for HT-9 cladding based upon the changing microstructure of the material [63][64]. It was shown to work for in-reactor creep tests [65]. This model could be built to contain the changing creep mechanisms associated with the changing operating conditions in a fuel pin. The model is corrected and verified as the microstructure is traced throughout the operating history of the fuel rod.

Objectives

The goals are to:

- Continue to develop and validate the thermal creep model to make it applicable to steady-state and transient fuel rod operating conditions.
- Gather transmission electron microscopy (TEM)-based microstructural data (dislocation types, locations, and densities; precipitate types, chemistry, location and densities) for unirradiated HT-9, thermally crept HT-9 (Section 2.4.1.1) and for selected MFF (FFTF metallic fuel experiments) samples based upon temperature history, time and fuel burnup. If possible, microhardness measurements will be taken near the locations of where the TEM samples are taken. The purpose is to gather qualitative information concerning level of hardening, from irradiation damage, providing confirming evidence supporting the TEM-related information.
- Replace the current design equation for creep in a known fuel rod model, like that being developed in BISON.
- Validate the model against measured cladding profilometry of irradiated test fuels from FFTF and EBR-II.

Approach

The approach for experimental work (data gathering) echoes the listing of goals. The data are being gathered from unirradiated HT-9 taken from the cladding of an archive MFF fuel pin as well as strategic points of irradiated pins which demonstrated substantial thermal creep strain. The cladding strain profile of one of those pins is shown in Figure 6.

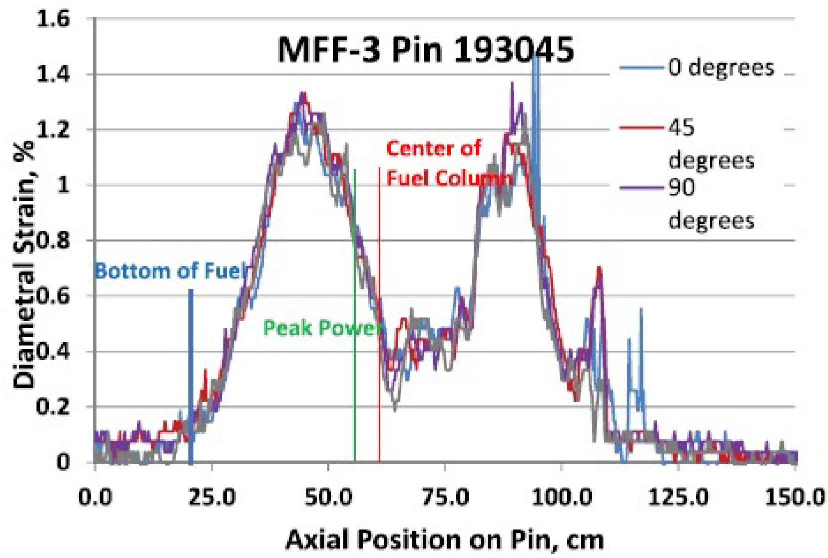


Figure 6. Example of cladding strain profile taken from an MFF pin [14].

Note that the lower peak in strain is near core midplane, but operating too cold for much thermal creep, but in the location where irradiation-induced creep is most likely observed. The upper peak in strain is most likely dominated by thermal creep. It is these locations where microstructural data can be gathered and used to develop the thermal creep model. Then, knowing the operational history of this pin, the model can be validated by comparison to the profilometry data.

Because the thermal creep deformation under the high gas pressures resulting from fission gas release is important for determining the margin to creep rupture, additional comparisons will be made between the implemented thermal creep model and burst testing experiment data. This work will augment the steady-state profilometry comparison by targeting experiments which have data within the temperature and stress ranges of the developed model.

Deliverables/Schedule

Year 1- Develop base creep model for HT-9, applicable to fuel rod operation.

Year 1 – Complete all proposed microstructure characterization and data analysis to support the thermal creep model development.

Year 2-3 – Implement and validate the thermal creep model using MFF fuel rod profilometry and complex operating histories with a fuel performance code.

Year 3-4 – Support corrections with any additional microstructure characterization needed. Provide initial assessment of cladding creep failure data.

Year 4-5 – Support corrections with any creep testing at bulk or mesoscale needed.

2.4.1.4 Next Generation Cladding Materials

Primary Author(s): Caleb Massey, Stuart Maloy, Mychailo Toloczko, Ben Eftink

Background

Next generation SFR fuel cladding materials should have improvements over the near-term cladding material while not giving up performance for any of the other properties. Desired areas of improvement include:

- Improved high temperature irradiation creep resistance
- Good fracture toughness after irradiation at all temperatures
- Very high dose swelling resistance

Because HT-9 and other already existing tempered martensitic steels provide very good performance, and are readily fabricated, a nearer-term cladding candidate could be an advanced tempered martensitic steel. This is bolstered by the fact that the fossil fuel community has shown that significant improvements in creep performance can be obtained with more advanced tempered martensitic steels. Another advantage of using an advanced tempered martensitic steel is that it is easily weldable and can be processed into thin-walled tube forms. Significant data exists on development and irradiation testing on advanced ferritic/martensitic steels. One of the alloys that shows exceptional promise in improved high temperature creep strength and improved radiation tolerance is T92 or NF616.

ODS ferritic steels are the current prime long-term candidate material because this class of material is known to have excellent high temperature creep resistance, and the available data indicate better swelling resistance than HT-9 or any other tempered martensitic steel. However, ODS ferritics are currently a challenge to fabricate and are expensive. Thus, a back-up candidate should be co-investigated.

Approach/Schedule/Deliverables

Year 0-1 – Development of AFC program Advanced Cladding Roadmap

2.4.2 Fuel Alloy Properties and Characterization

Primary Author(s): Cynthia Adkins, Geoffrey Beausoleil, Luca Capriotti, Randall Fielding, Colby Jensen

Background

Properties of metallic fuels have been measured and reported in many sources. Fuel properties are needed to cover the range of fuel cycle lifetime from fresh conditions for processing and fabrication, to performance in irradiation, to post-irradiation handling, storage, and follow-on processing. Recently, gap assessments have been underway to understand the applicational completeness of available data for the reference fuel alloys [21][22][23]. In this discussion, properties needs are described corresponding with fresh fuel, with implications for fabrication and pre-irradiation conditions, and irradiated fuels, for performance during irradiation and transient conditions (also has some implications for post-irradiation applications).

Fresh metallic fuel property measurements have focused on thermal properties, particularly thermal conductivity since fuel temperature is a primary dependency for reactor and fuel performance. Some mechanical properties have been investigated as well to a lesser extent, particularly regarding swelling behavior. For the reference metallic fuel alloys, fresh fuel thermal and mechanical properties and how these properties are linked to microstructure and in-reactor performance has not been well studied. The reason for this reduced emphasis historically is likely traced to the singular fabrication approach used for driver and experimental fuels using a similar vacuum casting process, which is the performance basis of those programs. Current and future deployment goals for these fuels are and will look to new/improved fabrication routes, where processing effects on performance need to be evaluated. Additionally, modern modeling capabilities increasingly demand for better material property inputs to reduce uncertainties in development and validation. An underlying goal for the program, also in the vein of accelerated fuel development, is to start supporting the process of linking material processing to properties to performance. This is an ambitious goal that is not proposed comprehensively but as we draw conclusions on reference fuel design performance and fill some data gaps in material properties, it is a relatively small effort to capture corresponding microstructural information in fresh fuel alloys.

The thermal conductivity of an SFR metal fuel in-pile is affected by several factors with primary impacts from porosity evolution and resulting sodium infiltration (for sodium bearing fuel). With increasing burnup fission product accumulation is believed to have an increasing role. Temperature effects during operation induce radial dependency of fuel alloy redistribution and material phases that also influence porosity morphology. In addition, the active process of irradiation/fission can also modify many of these behaviors. Post-irradiation examination requires preparations that can have varying levels of impact on fuel structure and properties. Recent AFC studies have been conducted on irradiated U-Zr alloy from FFTF MFF3 experiments that contain data to verify the other factors affecting thermal transport in SFR fuel.

Specific material properties on fresh and irradiated fuel to be measured include thermal conductivity and volumetric heat capacities over the applicable range of each parameter. Thermal transport across the fuel-cladding gap is also a measurement target. Hydrodynamic property measurements of melt phase should include surface tension and viscosity. Mechanical properties affect not only fabrication routes and methods but also in-reactor cladding strain. Some of the properties of interest include ultimate tensile and yield stress, elastic modulus, plastic hardening constants, Poison’s ratio, dynamic modulus, critical resolved shear stress, creep and hardness. Some data needs to span beyond liquidus points to support R&D in fuel fabrication and some extreme accident conditions.

Objective

The objective of this task is to provide data and understanding of relevant fuel properties throughout the fuel life cycle, with emphasis on properties supporting fabrication technology development and fuel performance prediction. A priority near-term goal is to identify applicable data and measure any potential properties identified as gaps for reference fuel alloys to support engineering scale analysis.

Approach

<u>Property data identification and quality assurance evaluation</u>: Extensive effort has been spent pulling together collections of fundamental metallic fuel properties such as crystal structures, thermal expansion, phase diagrams, thermal conductivities, and heat capacities on unirradiated fuels [66][67]. However, these data are insufficient to meet the goal of developing improved fuel for the purpose of qualification for use in advanced reactors. Additional effort is needed to cull and evaluate the quality of the data sets with the end goal of determining properties that need further quality assurance pedigree. These data collections would also be more useful to modelers and fuel developers in a database format rather than a handbook format. Suggested activities include:

- Review of fundamental properties with a focus on the measurement method and reported accuracy as well as the sample preparation and pre- / post-test microstructural characteristics. This action will help to determine relevant data sources for users.
- Design and populate a database format based on current standard user interface that allows for a centralized repository for the results of property and characterization analyses.

<u>Fresh fuel measurements</u>: Many advanced reactor designs use the EBR-II experience with metallic fuel as the basis for performance predictions. A quality-assured baseline characterization of EBR-II and FFTF metallic fuel properties will be developed that may be used for comparison with contemporary and future fuels development. This baseline will be produced by measuring defined characteristics of legacy EBR-II and FFTF fuels. This effort will include microstructural characterization, collection of mechanical property data, starting with hardness, and confirmation of thermal properties, if needed, based on previously reported data.

<u>Irradiated fuel measurements</u>: The microstructure of U-Zr alloy evolves during irradiation through constituent redistribution, porosity growth, and phase changes. These changes have an impact on the engineering performance of a fuel system by altering the physical properties, such as thermal

conductivity, hardness, solidus temperature, etc. The past ability of fuel engineers to provide mechanistic explanations and validating computational models have been hindered by the complexity of the reactor operating environment, the difficulty of controlling and isolating operational parameters in prototypic irradiations, and multiple microstructural phenomena that concurrently exist and evolve within the fuel during irradiation (e.g. constituent redistribution, swelling, fission product generation, phase transformations, and irradiation-induced defect formation) [22].

Microstructural characterization of irradiated metallic fuels from EBR-II and FFTF experiments along with additional out-of-pile neutron and ion irradiated metallic fuel samples has been ongoing in recent years however, thermal and mechanical property measurements on the same irradiated metallic fuels have only just begun to produce results. The AFC program has recently begun the process of direct measurement of thermal properties, particularly thermal conductivity and specific heat capacity, in a remote hot cell environment that has led to the start of a knowledge base of post-irradiated properties. A summary of fuel compositions and properties is given in Table 2.1.8.1. Many additional irradiated fuel samples are available to continue this property investigation. These data in conjunction with the microstructural and porosity characterization will lead to well-informed calculations of effective bulk thermal conductivity and models of related fuel performance.

Table 2.1.8.1 Current List of Irradiated Metallic Fuel Thermal Property Investigations

Composition	Property	Temperature Range (°C)	Approximate Burn-up (%)
U-10wt.%Zr	Bulk thermal conductivity	50 – 900	12
	Local Radial thermal diffusivity	25	12
	Specific heat capacity	25 - 1000	11
U-19wt%Pu–10wt%Zr	Local Radial thermal diffusivity	25	14
	Local Radial thermal diffusivity	25	0.001*

*TREAT irradiated at 700°C

<u>In-situ/non-destructive property measurements</u>: New irradiation test designs are targeting in-situ (during irradiation) measurement of properties as described in Section 2.2.2.2 with the IMPACT experiment designed under INL LDRD and the NSUF experiment, EPIC, using the THOR capsule described in Section 2.3.2.1. These measurements will expose predictive in-reactor behavior considering material changes due to irradiation. As these tests become available, data will be reported and utilized in model development and evaluation.

Fuel alloy mechanical properties effect cladding strain which is an important limiting factor in reactor performance. To accurately predict cladding strain using fuel performance codes a series of mechanical testing, starting with tensile testing at both room and elevated temperatures will be performed. Based on fabrication parameters. To support novel and advanced fabrication development and link it to possible fuel behavior effects, a system of annealing/heat treatment studies will be performed. These experiments will be used to simulate possible thermal cycles fuels may undergo during processing in a manner other than traditional injection casting and followed by sodium bonding. Samples produced during the thermal treatment studies will undergo characterization to document microstructure, thermal, and mechanical properties. The culmination of this work will be to link those properties that have been identified as affecting irradiation properties or in some cases processing during the fabrication stage with microstructures and thermos-mechanical history of the fuel. Alongside this work, studies will be initiated on the effects of fabrication or assembly parameters on the performance of the assembled fuel pin. Such parameters would include the process of sodium bonding and resulting flaws in the sodium bond. These tests will determine the importance of bonding prior to reactor insertion, and at what time and temperature these benefits are realized (i.e., bond processing temperature and time).

Deliverables

Over the 5-year period of this plan several deliverables are necessary to document the work performed. Generally, deliverables will be in the form of final and interim reports and peer-reviewed publications. Below are the two major objectives of this work:

- FFTF and EBR-II fuel baseline characterization report documenting microstructure and mechanical properties (i.e., minimum hardness value, also tensile properties up to reactor temperatures) and thermal properties. This work can be incorporated into a repository metal fuels database.
    - Reports will be updated as additional data is generated.
    - Input into final report documenting which describes physical properties effect on irradiation behavior.
- Report linking specific fabrication processes and resulting properties.
    Interim reports documenting steps and experiments to develop thermomechanical history-microstructure-properties report.

Schedule

- Year 1- FFTF and EBR-II baseline reports including room temperature thermomechanical properties.
    - Updates will be issued as additional data is generated.
- Year 1- Initiate heat treatments and subsequent characterization simulating possible fabrication paths. Continue characterization of irradiated fuel samples from FFTF and EBR-II experiments creating a correlation of thermal /mechanical properties for legacy metallic fuels.
- Year 2- Continue characterization of different processes or simulated process paths including some elevated temperature properties and mechanical testing of alloys of interest linking results to properties which affect irradiation performance.
- Year 3-4- Summarize results from processing routes into a metal fuels database with corresponding properties. Complete thermal property measurements of legacy irradiated metallic fuels.
- Year 5- Final report linking fabrication paths to microstructures, to mechanical properties and irradiation effects.

2.5 Code Development and Assessment

Matthews et al published a summary of current fuel performance simulation software capabilities [68] where code assessment and behavioral models in BISON are described in detail. The salient points from that manuscript are repeated or summarized in the following Code Assessment and Behavior Models sections.

2.5.1 Code Assessment

Primary Author: Steve Novascone, Pavel Medvedev, Alex Swearingen, Ryan Sweet, Jake Hirschhorn

Background

The Multiphysics Object-Oriented Simulation Environment (MOOSE) and MOOSE-based applications such as BISON have a well-designed SQA plan that, when implemented, allows the code developers to continuously maintain a complete set of NQA-1 documentation, meeting the standard for software serving a safety function with nuclear energy systems. The process relies heavily on a robust automated testing tool, the CIVET, and an in-code documentation system, MooseDocs, to implement the SQA process where developers must follow the correct processes to make additions to the code. The associated documentation is always current and accurate for every change to the framework and/or applications. The BISON code underwent a Nuclear Quality Assurance (NQA-1) audit, performed by software quality assessors from ASME's NQA-1 committee, and received a rating of 'Effective'. This ensured that the MOOSE framework and BISON meet Department of Energy and ASME NQA-1 requirements. MOOSE and BISON can be used in safety software applications (QL-1 and QL-2), including at ATR, which requires an NQA-1 pedigree. Because of this SQA plan, input files and models implemented in the BISON fuel performance code are continually maintained. This would allow routine updating to assessment cases which are developed to test metallic fuel performance as constitutive models are improved.

Objectives

Best practices recommend using separate experimental data sets for tuning and training fuel performance models and for code assessment. The culminating point of BISON development by the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program is to develop behavioral models relevant to reactor fuels and materials using a comprehensive mathematical interpretation of the underlying physical process which is derived from nuclear fuel performance data and fundamental simulations. Following this, AFC will perform blind assessment of BISON using an EBR-II datasets that have been thoroughly reviewed for accuracy from the original document sources. The following task proposes that integral fuel rod data will be used to provide independent validation of BISON capabilities, provided that this data has not been used for previous calibration efforts. This work builds on efforts of the NEAMS and the former VTR programs which established a methodology for metallic fuel benchmarks and the physics and simulation requirements to enable their development [36].

Approach

This approach will follow established processes for the evaluation of metallic fuels benchmark cases, as shown in Figure 8. Several subassemblies from legacy EBR-II testing will be simulated using the best available information without any model calibration to determine the baseline BISON code performance for metallic fuel. These simulations will be compared with PIE data and documented. Additional analyses will be performed to identify systemic uncertainty in constitutive models and potential calibration. The developed inputs will be sanitized of protected EBR-II data and archived in the BISON code repository for future assessments within the AFC program. Postulated transient conditions will be utilized along with available M-series transient fuel pin testing. Similarly, developed inputs will be documented and archived for future use.

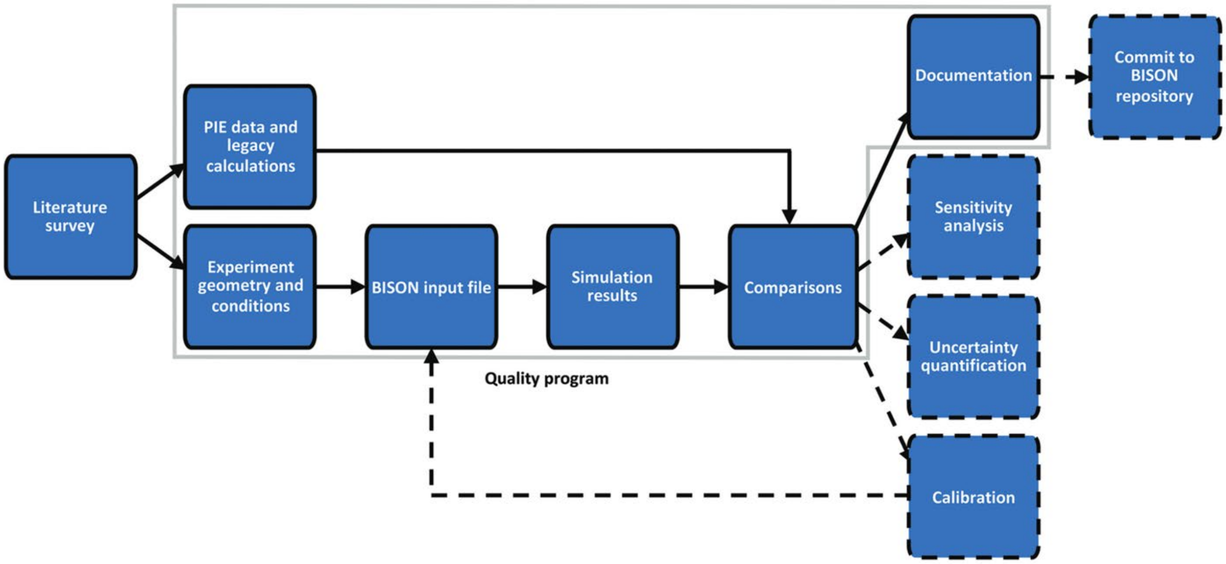


Figure 7. Flow diagram of the process to design and evaluate metallic fuel benchmark cases based on EBR-II legacy data (reproduced from [36]).

Deliverables

As indicated above, completion of behavioral model development is not a prerequisite for the commencement of the initial code assessment and can easily serve as a gauge for needed future developments.

- Year 1-2 – Assessment of steady state code capability
- Year 3 – Assessment of transient capability

2.5.2 Behavioral Models Development

Primary Author(s): Pavel Medvedev, Steve Novascone

Background

Predictive fuel performance models require detailed understanding of the phenomena that takes place during irradiation of metallic fuel. This will be gained by conducting post irradiation examination of legacy and new irradiation experiments. It is expected that PIE will postulate theoretical mechanisms behind FCCI, swelling, densification, Zr redistribution, sodium logging and fission gas release. Computer models based on theoretical mechanisms elucidated by the PIE program can be developed.

Objective

Develop behavioral models that support fuel performance predictions of metallic fuel performance relative to fuel design limits.

Approach

Work in this area will require close integration with the NEAMS program, also working on model developments that should benefit AFC goals. In most cases, phenomenological model development should be included as tasks within individual activity areas provided in this document.

Phenomenological description of fuel behavior will be developed by conducting PIE. PIE images will be used to quantify mechanical and chemical changes in the fuel. Calculations will be performed to determine volumes of different phases generated in the fuel and quantitative explanation of observed changes of overall fuel volume will be provided. For FCCI, chemical composition of the layer will be determined and validated against the fuel depletion data. For swelling, fission gas inventory will be conducted and validated against depletion data and measured plenum inventory. To understand fuel densification, irradiation of unconstrained fuel will be performed to determine whether FCMI causes reduction of fuel porosity. Development of Zr migration theory requires conducting of Zr inventory in PIE samples. Sodium logging research requires development of PIE and sample preparation techniques compatible with sodium logged samples and conducting sodium inventory in the fuel pin.

Deliverables

See phenomena-based research areas in this document.

2.5.3 Metallic Fuel Database Development and Qualification

Primary Author(s): Doug Porter, Kun Mo

Background

During the IFR program, there was an effort to build the IFR Material Information System (IMIS) to allow researchers to easily access the data generated by the IFR program. The goal was to populate a database in an ORACLE-based system. This was cut short due to funding issues (the IFR program stopped). The existing database at that date was placed into an ACCESS format in 1995. That was the interim database, IMIS. The database included operating history and PIE data from IFR experiments on U-xPu-yZr ($0<x<28$ wt.%; $2<y<14$ wt.%) fuel rods irradiated in EBR-II.

Now there is a new effort to make this data available to support the qualification of generation IV reactors. This required migrating IMIS to a new system, and to make updates to IMIS. Some issues with the database structure have been identified as well as minor data holes that would affect models based on these data. Most of the existing documentation that supports these data, and which still exist, has been found.

This next generation of IMIS was later started by ANL to replace IMIS. It was the SFR Metallic Fuels FIPD database [19]. As with IMIS, FIPD is also an organized collection of EBR-II test pin data and documentation. The database includes pin operation conditions calculated using a collection of Argonne analysis codes developed during the IFR program, including axial distributions for power, temperatures, fluences, burnup, and isotopic densities. Improvements have been made since that time based on information collected for the subsequent spent fuel conditioning programs. FIPD also contains pin measured data from PIE, including pin fission gas release and gas chemistry measurements, and axial distributions from profilometry, gamma scans, and some neutron radiography. There is also an extensive collection of documents associated with different pins and experiments, including raw PIE data, design descriptions, safety analysis, and operational reports. FIPD inherited all available PIE data in IMIS database, with continuous effort to add more PIE data, such as lab notes with original micrographs and memos for experiments examined at the Alpha-Gamma Hot Cell Facility at Argonne, and the high-resolution neutron radiography data (NRAD) data performed at INL's Hot Fuel Examination Facility (HFEF). To ensure the data quality, the IMIS/FIPD data are being validated following the quality assurance (QA) plan already reviewed and supported by the NRC.

Objectives

The database may need to be redesigned by database normalization, and the data reviewed to make as much data as possible NQA-1 compliant. In addition, an archive search needs to be conducted to find critical burnup and fluence data for approximately 2% of the fuel in the database. The database will only need to undergo a small redesign in order to become reactor agnostic, and capable of hosting additional data provided by FFTF.

The FFTF experiments (namely the IFR-1 and MFF experiments) need to be put into the database including in-reactor operation information as well as the available PIE data. Fuel rods/pins from four 169-pin MFF experiments, all with peak burnups of 9-14 at.%, with peak cladding temperatures ranging from 588 °C to 651 °C were saved for examination. All are U-10Zr fuel in HT-9 cladding. There were an initial set of ~44 fuel rods which have been examined non-destructively and are not being saved for transient testing. Some have been destructively examined with data available to put into FIPD.

Further research is needed to determine if missing documentation from the EBR-II experiments is recoverable. The goal is to find as much as can be found. Some of these might require processing, such as neutron radiography films needing to be scanned digitally.

Approach

ANL will maintain the FIPD database and continue to organize it to be capable of utilization by fuel developers for use in validation of fuel performance codes. The data should also be made NQA-1 compatible where possible. These efforts are already underway.

Soon (starting in FY24), one goal is to use existing available operating and PIE data on the FFTF, MFF experiments, to incorporate it into FIPD. INL will assist in those efforts providing existing PIE. Experiment operating information, supplied by PNNL, will also be utilized. All of this data must be in formats available to fuel performance codes, such as the MOOSE/BISON formats.

The AFC program is also actively generating significant new data from experiments performed in the ATR, TREAT, and advanced PIE on legacy materials. The approach to managing those data will also be considered and a plan developed to be implemented.

Deliverables/Milestones

Year 1 – Prepare FIPD for use of FFTF (MFF) operating conditions to assess pin-by-pin operating conditions (cladding and fuel temperatures, axially and radially, burnup and burnup rates, radially and axially, neutron exposures, etc.).

Year 1 – Acquire and load MFF PIE data for initial eight pins (profilometry, precision gamma scanning, neutron radiography, metallography, etc.).

Year 2 – Acquire and load available MFF PIE data for additional pins.

Year 3 (early in FY) – Make decision on potential other fast reactor related fuel data that could/should be added to FIPD or a satellite database.

Year 4 – Complete qualification of available FFTF (MFF) PIE and operating data qualifications by implementing the NRC approved Quality Assurance Program Plan.

Year 5 – Qualify potential other fast reactor related fuel data added to FIPD or a satellite database.

2.6 Technology Development and Independent Analysis

Although a few important characteristics such as smeared density, length, and fuel alloy evolved over several decades, in many ways, the historic EBR-II programs utilized the same sodium-bonded metallic fuel design [69]. Additionally, other than incremental optimizations and technological improvements, the fuel fabrication process of injection casting of fuel slugs, fuel pin assembly followed by sodium bonding, also remained relatively constant throughout the duration of EBR-II operation. Although this fuel specification (meaning linkage to its fabrication) was extensively shown to be robust, advancements are still possible that could improve fuel cycle economics via simplifications, reduced time, improved fissile utilization, fuel design innovations, etc. Many perceived weaknesses of metallic fuel have been highlighted by pushing the fuel to higher burnup and temperature limits. Several design solutions have been considered for many years to address these performance ambitions. Some design changes can be seen as evolutionary, such as diffusion barrier application to mitigate fuel cladding chemical interaction (FCCI) and smeared density. Larger changes include concepts such as alloy changes, either minor additions or changing of major alloying components, fuel geometry, and fuel assembly changes. Some of these changes will affect fabrication processes.

As such, designs and fabrication methods must both progress hand in hand to ensure a fuel design can be fabricated for testing and if successful, eventual scale-up. As with design changes, fabrication processes can also benefit from smaller, more evolutionary changes. These evolutionary changes will include waste reduction, higher melt utilization, and process optimization to increase yield of acceptable fuel slugs. Similarly, changes in the fuel pin assembly process are important for improvement of the economics of the metal fuel cycle. In all these areas, improvement of fuel cycle economics should be a primary driver. Economic improvement may be realized in areas such as waste minimization to lower disposal costs, higher fabrication yields, reduction in process times, and uranium scrap capture and recycle.

2.6.1 Fuel Fabrication

This section of the report will cover work proposed to be done to help optimize the current fabrication technique to allow large-scale fabrication of metallic fuel pins with increased scale, more efficient flow, and reduction of waste. Gaps will be identified to achieve the objectives of this section. A model will be developed to analyze the portions of the fabrication process to include a time study and economic assessment. Some fuel property measurements will be suggested to fill gaps expected to occur in these models. These things will be covered in Section 2.7.1.1.

Secondly, there will be efforts to assess several alternative fabrication techniques, amongst many mentioned here. Two of them, extrusion and continuous casting were chosen for their maturity and imagined use for proposed future fuel designs, as well as alternatives to the injection casting method. The proposed work scope (see 2.7.1.2) will include production similar codes to assess economic likelihood, as well as experiments to test ancillary techniques. This section will be more speculative (requiring bench-top testing) and subject to availability of funding.

2.6.1.1 Casting Technology Gaps

Primary Author(s): Randall Fielding, Doug Porter

Background

The counter gravity injection casting process, also known simply as injection casting, used in the EBR-II fuel fabrication campaigns proved to be a robust and successful casting process producing tens of thousands of fuel slugs for irradiation. However, the process was largely built on empirical relationships and a wealth of operator experience. Despite the success of injection casting improvements remain to be made. These improvements include higher charge yields, shorter processing cycle times, and waste reduction. By improving the charge yields, less material requires recycling, either directly or with additional processing, which improves the overall economics of the process. Improved cycle times are important because metallic fuel charge sizes are necessarily limited in size due to criticality concerns. Past scale-up designs, using the traditional fabrication methods, required multiple parallel fabrication lines to feed the reactor with adequate fuel while keeping charge sizes within established limits. Decreasing waste will decrease total disposal cost and increase uranium scrap recovery.

Uranium recovery during EBR-II campaigns was approximately 95%. This can be increased through reduction of and further processing the materials formerly referred to as 'fine-fines' and 'glass and dust' which includes materials generated when cast slugs were separated from the quartz molds and crucibles, in which the fuel alloy was melted, were cleaned. Separation of the uranium alloy from the quartz mold waste was previously investigated during the IFR program [70]. Electromagnetic and electrostatic separation seemed to have the most promise. Others have just suggested that dissolution of these waste products followed by further processing is the best route, but it is likely to be costly. A direct method to eliminate this waste could also be reusable molds. This has also been studied but a solution for mass production was not found.

Increasing charge yields requires the casting process to be optimized or modified. To make these improvements some experimentation and system development will be needed but experimentation generally requires expensive casting systems and evaluations and can often be time consuming. To improve this development process numerical simulations of the fabrication processes are needed. Currently, little has been done in the area of fuel casting simulations. Through these simulations design changes can be tested and current processes can be better evaluated to determine improvements, with much less experimentation. For example, the VTR project optimized the crucible design to produce a smaller heel after casting, but testing was not possible without a furnace. To support model simulations of the fabrication processes better material properties, including both high temperature and liquid properties will be needed. These simulations may be able to increase the overall cycle time of the casting process, thus reducing the number of parallel fabrication lines needed to fuel an advanced reactor.

Objectives

The main objective of this work is to improve the overall economics of the metallic fuel cycle, and to support further efficient scaling of the technology for advanced reactors. Economics will be improved by improving charge yield and reducing times needed for the casting process, in other words, increase the amount of cast material that is converted to usable fuel slugs when compared to historic data. Economics will also be improved by reducing the mass of waste produced and ensuring all usable material is recycled. The data generated from this work will support DOE and commercial programs as metallic fuel cycles are implemented.

Approach

Current advanced reactor designs are based on the use of high assay low enriched uranium. Criticality issues have made it necessary for a smaller batch size compared to most metal industries. The EBR-II Mk-III fuel was enriched to approximately 69%. Due to criticality concerns batch sizes were limited to approximately 20 kgs. Advanced metallic fueled fast reactor designs are generally based on a high assay low enriched uranium (HALEU). It can be safely assumed that due to the lower enrichment levels batch sizes can be safely increased, perhaps to approximately 50-100 kgs, however, detailed analysis of the specific process equipment will be needed to determine this. Although injection casting has been used, larger batch sizes may present opportunities to optimize the processes to gain efficiency and improved economics. Analogous industries, such as specialty metal fabrication, may be a source of ideas for improvement.

Production information for these analogous processes, as well as production experts will be consulted to determine possible improvements in the fabrication process for the reduction of waste products and to gain efficiencies. As part of this advanced reactor and fuel vendors will be consulted to ensure applicability to current industry interests. These interactions will be used to build the model simulations. This effort may require fuel material properties (melt viscosity, surface tension, etc.) needed for accurate simulation as noted in Section 2.4.2. The results of this work will enable simulation of fabrication processes for efficient optimization. Experimental work, testing and developing modified fabrication techniques will support computational simulations for optimizing the processes used for EBR-II style fabrication or novel methods (Section 2.6.1.2).

Deliverables/Schedule

- Year 1 – Define and document gaps in the traditional fabrication process that need optimization.
- Year 1 – Define properties needed for accurate fabrication process model simulations.
  Investigate commercial model simulation methods and available software.
- Year 2 – Perform parametric and/or feasibility studies on casting process.
  Initiate liquid property measurements.
  Validate results generated by fabrication-properties-microstructure linkage development activities discussed in Section 2.5.1
- Year 3 – Final report detailing optimization data for traditional fabrication and opportunities for advanced fabrication techniques.

2.6.1.2 Advanced Fuel Fabrication Development

Primary Author(s): Randall Fielding, Doug Porter

Background

Advanced fuel forms may require novel and advanced fabrication methods (e.g., removal of the in-pin sodium, fuel cross-section geometry alterations, etc.). Fabrication of traditional fuel designs may also benefit from new fabrication methods by improving economics through process improvements and reducing the amount of scrap and waste produced. Possible methods include processes such as continuous casting, re-usable molds, gravity, or pressure assisted casting with re-usable molds, extrusion, powder metallurgy approaches, or additive manufacturing. Of these, continuous casting and extrusion are considered most promising and have been chosen to be studied in the near term [71][72]. Continuous casting for fuel fabrication is at a low technical readiness level. Extrusion is already being developed for several start-up reactor concepts showing a degree of feasibility [73]. Others have examined continuous casting for uranium alloys; however, it has not been implemented beyond a laboratory environment [74][75].

In addition, in support of Na-free fuel designs, the issue of fuel pin fabrication using tight-fitting fuel/cladding is a challenge. Methods to allow ease of fuel loading into the cladding will be studied.

Objectives

The overriding objective of this work is to improve the efficiency of the metallic fuel cycle and enable advanced fuel forms to progress beyond lab scale non-prototypic fabrication routes to higher levels of technical readiness levels. The economics of the proposed fabrication processes, including the current injection casting techniques, need to be assessed and compared to the optimized traditional fabrication method calculated as part of 2.7.1.1.

Approach

Computational simulations will be developed and used to evaluate process designs. Simulations will be used for initial feasibility, aid in design efforts before advanced systems are fabricated, and as a tool to evaluate advanced fabrication processes for further development is found or created. In other instances, simulations and experimentation will progress concurrently to validate the models and properties used in the simulations.

Besides model development, the main approach of the proposed work will be experimental testing of fabrication concepts and advanced systems. As funding permits, experimental work will be supported, and in some cases, guided by system computational simulations. As the ultimate objective of this work is to increase both material and economic efficiency of the fabrication process, experimentation will be documented and compared to the "standard" sodium bonded case to show fabrication enhancements. Continuous casting and extrusion have been selected as likely methods to improve the fabrication process, particularly for advanced fuel designs. Because continuous casting is at a lower technical readiness level, additional resources may be needed to progress this concept beyond initial feasibility testing. Initial work on continuous casting will consist of a small amount of work showing feasibility, a larger effort will be used to simulate a continuous casting process to determine system requirements for a lab scale casting system. As the best processes are selected, they will be further developed, while working with fuel property measurement efforts, to determine the optimal process that will provide a consistent product commensurate with or better than baseline metallic fuel fabrication, which will then be irradiation tested to fully evaluate performance.

Deliverables/Schedule

- Year 1 – Examine potential fuel pin/cladding gap closure methods (swaging, drawing, etc.) for use in easing fabrication of fuels where the Na bond has been removed.
  Based on a fabrication technology gap study, proposed systems, and fabrication concepts to mitigate identified technology gaps will be prioritized for potential success. For example, previous work on continuous casting of fuels has identified some of these gaps (fabrication rates, problems associated with high melting range alloys).
- Year 2 – Initiate simulation efforts on continuous casting designs to show feasibility and define system requirements.
  Initiate extrusion simulation and benchmarking efforts. Some initial extrusion development activities have already been done [73] and can be used to verify simulations.
  Initiate feasibility testing of processes identified in fabrication gap study.
- Year 3 – Design and initiate fabrication of continuous casting system.
  Down select the most productive within budget constraints.
- Year 4 – Initiate testing of continuous casting system.
- Year 5 – Computational evaluation of feasibility of re-usable molds at engineering or commercial scale.

2.6.2 Cladding Weld Qualification

Primary Author(s): Caleb Massey, Stuart Maloy, Tarik Saleh, Ben Eftink

Background

A major challenge for fuel cladding qualification is the demonstration of scalable prototypic fabrication pathways for fuel/cladding combinations. Fortunately, recent efforts by industry have revived supply chains for critical materials such as alloy HT-9 as demonstrated by a large batch heat of HT9 (plate and thin-walled tube variants) procured by TerraPower as part of their Natrium demonstration project. Thus, there is not a critical need for the AFC program to demonstrate tube-processing methods as part of a 5-yr qualification strategy for conventional HT-9 cladding. However, it is well known that for many materials, differences in microstructure and chemical segregation in weld and heat affected zones can alter irradiation degradation phenomena, including irradiation hardening, irradiation-induced precipitation, and cavity swelling.

Objectives

The primary objective of this work is to reduce anticipated fuel element failures at end cap weldments through a detailed comparison of weld microstructures, mechanical properties, and environmental degradation as a function of processing parameters, methodologies, and post-weld heat treatment conditions.

Approach

Historical weld approaches, including tungsten inert gas and laser welds, will be compared to capacitance discharge and pressure resistance welds. Weld properties will be evaluated using subsized tensile, fracture toughness, and creep tests in temperature ranges from 23-600C to cover the range of handling and operational temperatures expected for current fast reactor designs. Finally, neutron and ion irradiation campaigns will be developed and deployed to provide quantitative comparisons between ductility loss and microstructure evolution as a function of welding methodology.

Deliverables/Schedule

- Year 1-2 – Identify HT9 material supply chain and procure necessary for tube/end cap weldments
- Year 2-3 – Perform welds and characterize time-independent properties
- Year 3-4 – Perform targeted subsized creep tests on welds to establish time-dependent parameters
- Year 3-4 – Design ion/neutron irradiation experiments
- Year 4-5 – Perform irradiation experiments consistent with section 2.2.2.1

2.6.3 Transuranic-Bearing Fuel

Primary Author(s): Randall Fielding, Luca Capriotti, Colby Jensen, Nicolas Woolstenhulme

Background

The AFC Metallic Fuel program supports DOE’s goals to enable deployment of advanced reactors as well as development of advanced nuclear fuel cycles. Metallic fuel for SFRs have long been considered for application to closing the fuel cycle through burning minor actinides, supporting fuel recycle strategies, and improved management of national fissile resources. More than 600 U-Pu-Zr fuel pins have been irradiated in EBR-II providing a large database for U-Pu-Zr behavior. However, continued R&D is needed to study all aspects of fuel development related to minor-actinide and Pu bearing fuels in order to support an eventual commercial application of a transuranic bearing fuel. Throughout this plan, R&D activities will naturally extend from binary alloy to ternary alloy composition as a selected reference design described in Section 1.1. This section addresses the impacts of minor-actinide constituents in fabrication and performance.

The AFC program has a long history of studying technologies to transmute long-lived transuranic actinides contained in spent nuclear fuel into shorter-lived fission products. More than twenty years ago, candidate fuel alloys were selected for irradiation in a cadmium shrouded position at the INL ATR. The cadmium shroud creates a pseudo-fast spectrum irradiation environment in this thermal test reactor that emulates the radial power profile typically experienced in a fast reactor. In addition to ATR testing, the FUTURIX-FTA irradiation was performed at the Phénix reactor in France with the same transmutation performance objectives and to confirm that behavior observed in ATR testing was representative of a true fast neutron reactor spectrum [15]. Finally, an earlier irradiation of minor actinide bearing metallic fuel was the goal of the X501 experiment irradiation in EBR-II.

Irradiation tests designated AFC-1 (subgroups denominated AFC-1B, AFC-1D, AFC-1F, AFC-1G, AFC-1H) and FUTURIX-FTA (DOE1 and DOE2) contained sibling pins with low-fertile and non-fertile actinide bearing metallic alloy fuel compositions. Compositions of FUTURIX-FTA and AFC-1 rodlets are listed in Table 1.

Table 3. Irradiation experiments test matrix for AFC-1, FUTURIX-FTA and X501

NAME	FUEL TYPE	COMPOSITION	REACTOR IRRADIATION
AFC-1	Metallic low & non-fertile	U-[25-34]Pu-[3-7]Am-[0-2]Np-[20-40]Zr Pu-[0-12]Am-[0-10]Np-[40-60]Zr	ATR
FUTURIX-FTA	Metallic low & non-fertile	U-29Pu-4Am-2Np-30Zr Pu-12Am-40Zr	Phenix
X501	Metallic fuel	U-20.2Pu-9.1Zr-1.2Am-1.3Np	EBR-II

More recently, the IRT-1 experiment utilized recycled transuranic material from used FFTF MOX fuel. The test included 5 rodlets at two different levels of simulated carry over fission products. Fuel compositions were nominally U-20Pu-10Zr-1RE or U-20Pu-10Zr-3RE and included one rodlet with a chromium coating and another rodlet with a TiN coating. The rodlets were irradiated to reach target burnups of approximately 5%.

Objective

Support DOE goals to develop advanced nuclear fuel cycles. To accomplish this, improved efficiencies and economics in HALEU material utilization should be evaluated and material developments performed to establish needed technologies in all parts of the fuel cycle.

Approach

The first proposed activity is to coordinate and launch a multi-program effort to evaluate fuel cycle options combined with fuel design considerations to define the resulting tradeoffs of potential R&D avenues. This activity will be most successful with engagement of SFR reactor systems support and fuel cycle technoeconomic evaluation. In recent years, some unique capabilities used for processing TRU-bearing fuels have atrophied at the laboratories. With renewed emphasis and a potential definition of a clear mission, these capabilities could be recovered to address these important issues again.

Deliverables/Schedule

Year 0: Formulate appropriate trade study problem statement building on previous studies [77] and coordinate with potential collaborative DOE programs.

Year 1: Perform technoeconomic evaluation of fuel cycle options based on fuel design options.

3 Schedule and Risks

The preparation of summary technical reports for regulatory review and approval are the main ties across the broad program. Therefore, activity scheduling must be based on supporting delivery of those reports. Acquisition of new characterization and testing data and potential significant development in phenomenological understanding will necessarily require significant resources and time. Figure 8 shows a high-level view of the projected 5-year time frame of activities and high-level deliverables. Some lines contain multiple activities for simplification. The chart shows the primary active activities assumed part of the current baseline. The orange lines represent several key activities that impact the nearer term qualification basis goals. Transient testing is a specific area of concern that likely is likely to experience delays in completing needed activities on legacy FFTF and EBR-II materials. Longer-term oriented activities associated with AFDQ and next generation fuel development also is likely to be underfunded for several years under current budget assumptions. While delays in these areas could be viewed as acceptable, the cost will be significant lost time in setting up for launching a next phase R&D effort that could have dramatic impacts on emerging SFR demonstrations and markets. The activities listed are crucial to building a proper foundation for next generation qualification efforts.

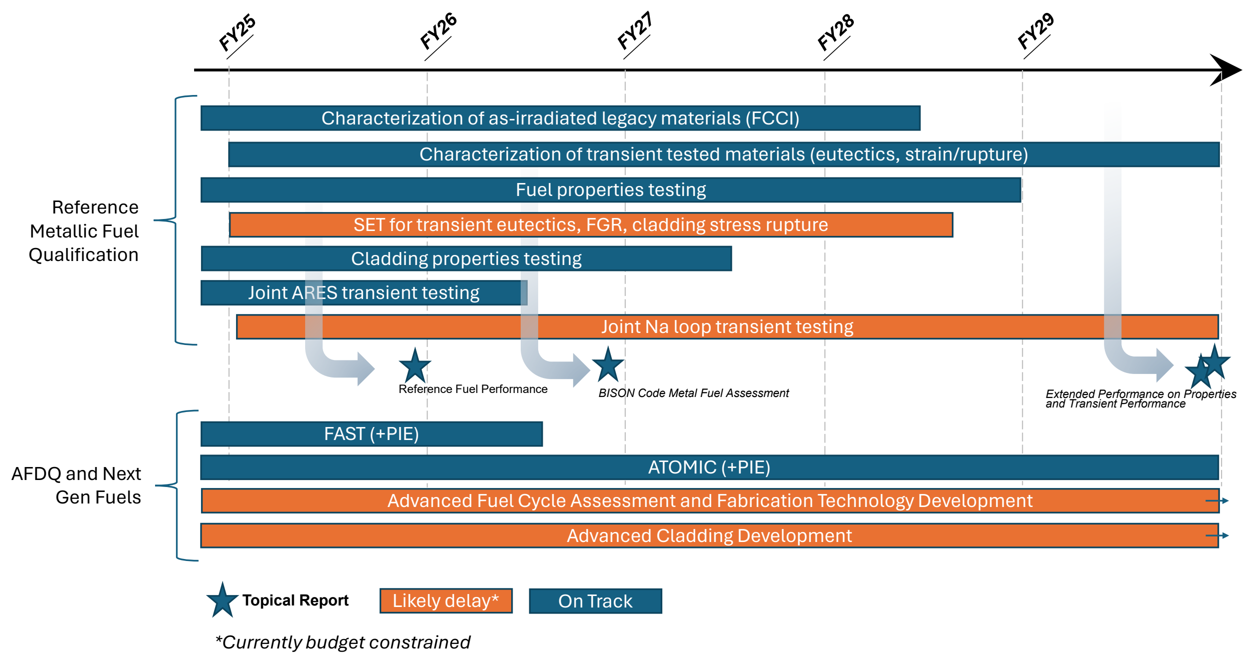


Figure 8. Estimated high-level schedule of integrated activities across the program showing topics currently under funded to meet displayed timeline.

Primary schedule and budgetary risks to the success of this R&D plan are as follows:

- **Budget.** Project progress is currently limited by budgetary resources. Current budgets have increased in the last few years after major loss in and resulting major shift in program direction in 2020. Budget levels have remained less than targeted levels for several years now. Without reaching initial target levels, some areas will take longer to complete such as transient testing of legacy materials. FY24 budget marks indicate potential for positive budgetary gains. This ties into the next bullet point.
- **Achieving balanced near- and long-term prioritization.** The near-term objectives of the program require significant resources while several activities are already aimed at

long-term objectives as strategic initiatives to reduce longer-term timelines. Examples include technoeconomic assessments of fuel design and fuel cycle implications, advanced fuel fabrication technology development, and accelerated testing development and evaluations.

- **Human and capital resources.** Competition of resources at including test materials, facilities, technical staff. The nuclear facilities at the national laboratories are in high demand. Overall capacities can have inherent limits and unexpected issues can arise such as equipment issues, staffing changes, etc. Careful planning and coordinating helps provide mitigation.
- **Uncertainty in experimental requirements.** Uncertainty of compatibility of potential new experimental approaches with existing facilities and procedures. New experimental work requires evaluation of hazards and supporting work controls with varying levels of required response. In some cases, the timeline for work can be impacted by such preparations depending on specific details that may not be defined yet. Development and refinement of data needs and testing requirements should be a primary early goal to continue as needed, especially after revision 0 of this plan. In some cases, more detailed test plans should be developed to further establish these.

4 Summary

The DOE AFC program is aggressively pursuing R&D of advanced metallic fuels to support demonstration and deployment of SFR technology to support several national goals in clean energy and improved fuel cycles. This document provides a detailed overview of the program's planning for the next five years. Yearly revisions are planned for this plan to address evolving needs, planning details, opportunities, constraints, and feedback from metallic fuel stakeholders. Near-term objectives are focused on establishing the qualification basis of SFR metallic fuel based on extensive legacy R&D programs that never completed some key research objectives that will likely be very useful to current and future licensing interests.

This plan lays a solid foundation of *databases, capabilities, expertise, and research-commercial-regulatory integration* for launching *next-generation initiatives in advanced metallic fuel technologies (fuel and cladding) and improved methodologies for achieving accelerated qualification of next generation fuel technologies*.

5 References

[1] "Advanced Fuels Campaign Execution Plan," INL Document INL/EXT-10-18954 Rev 8, July 2023.

[2] "Advanced Fuels Campaign Strategic Plan," INL Document unreleased draft.

[3] Crawford, D., et al., J. Nucl. Mater. 371 (2007) 202, doi.org/10.1016/j.jnucmat.2007.05.010.

[4] Povirk, G., et al., "Developmental Objectives for Advanced Reactor Fuels," INL/EXT-16-39923 Rev 1, August 2017, www.osti.gov/biblio/1599357.

[5] "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," NRC Document NUREG-1368, February 1994, www.nrc.gov/docs/ML0634/ML063410561.pdf.

[6] "Preapplication Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR) Liquid-Metal Reactor," NRC Document NUREG-1369, December 1991, www.nrc.gov/docs/ML0634/ML063410547.pdf.

[7] Briggs, L., et al., "Safety Analysis and Technical Basis for Establishing an Interim Burnup Limit for Mark-V and Mark-VA Fuel Subassemblies in EBR-II," ANL-NSE-1, March 1995, publications.anl.gov/anlpubs/2018/08/144381.pdf.

[8] "Status Report – 4S," IAEA Advanced Reactor Information Systems Report, October 2019, aris.iaea.org/PDF/Toshiba-4S_2020.pdf.

[9] Yoo, J., et al., Nucl. Eng. Technol. 48 (2016) 1059, doi.org/10.1016/j.net.2016.08.004.

[10] Kale, R., Prog. Nucl. Energy 122 (2020) 103265, doi.org/10.1016/j.pnucene.2020.103265.

[11] Zabudko, L., et al., "Fuels for advanced sodium cooled fast reactors in Russia: state-of-the-art and prospects," International Conference Fast Reactors and Related Fuel Cycles (FR09) December 7-11, 2009, Kyoto, Japan, inis.iaea.org/search/search.aspx?orig_q=RN:41070006.

[12] "Fuel Dispatched for China's CFR-600 fast neutron reactor," (September 30, 2022) World Nuclear News, world-nuclear-news.org/Articles/Fuel-despatched-to-China-for-CFR-600-fast-neutron.

[13] Pahl, R., J. Nucl. Mater. 188 (1992) 3.

[14] Carmack, J., et al., J. Nucl. Mater. 473 (2016) 167.

[15] Harp, J., et al., "Testing Fast Reactor Fuels in a Thermal Reactor: A Comparison Report," INL Report INL/EXT-17-41677 Rev 0, September 2017, www.osti.gov/servlets/purl/1458766.

[16] Jensen, C., et al., "Development of a Fuel Safety Research Program for Metallic Fast Reactor Fuels," Proc. Global 2019, Seattle, WA, USA, September 22-26, 2019.

[17] Harp, J., J. Nucl. Mater. 494 (2017) 227, doi.org/10.1016/j.jnucmat.2017.07.040.

[18] Hirschorn, J., J. Nucl. Mater. 544 (2021) 152657, doi.org/10.1016/j.jnucmat.2020.152657.

[19] ART Fast Reactor Databases, frdb.ne.anl.gov/.

[20] Walters., L., et al., "Sodium Fast Reactor Fuels and Materials: Research Needs," SNL Report SAND2011-6546, September 2011, www.osti.gov/servlets/purl/1044975.

[21] Aitkaliyeva, A., et al., "Research Needs for Uranium-Zirconium-Based Metallic Fuels," INL Report INL/EXT-20-58719 Rev 0, June 2020, inldigitallibrary.inl.gov/sites/sti/sti/Sort_25979.pdf

[22] Williams, W., et al., Ann. Nucl. Energy 136 (2020) 107016, doi.org/10.1016/j.anucene.2019.107016.

[23] Beausoleil, B., et al., Nucl Technol., 207 (2021) 1491, doi.org/10.1080/00295450.2020.1826272.

[24] Porter, D. and Crawford D., Nucl. Sci. Eng. 196 (2022) 110, doi.org/10.1080/00295639.2021.2009983.

[25] "Metallic Fuel Qualification: Fuel Assessment Using NRC NUREG-2246," NRC Document NUREG/CR-7305 August 2023, www.nrc.gov/docs/ML2321/ML23214A065.pdf

[26] "Fuel Qualification for Advanced Reactors," NRC Document NUREG-2246, www.nrc.gov/docs/ML2206/ML22063A131.pdf.

[27] "INL Nuclear Fuel Cycle and Supply Chain (NFCSC) Quality Assurance Program Plan," INL Document PLN-3708 Rev. 6, May 2021.

[28] "Topical Reports," U.S. Nuclear Regulatory Commission, March 2022, www.nrc.gov/about-nrc/regulatory/licensing/topical-reports.html.

[29] Harp, J., et al., J. Nucl. Mater 528 (2020) 151840, doi.org/10.1016/j.jnucmat.2019.151840.

[30] Matthews, C., Nucl. Technol. 198 (2017) 231, doi.org/10.1080/00295450.2017.1323535.

[31] Keiser, D., "Fuel-cladding interaction layers in irradiated U-Zr and U-Pu-Zr fuel elements," ANL Report ANL-NT-240, publications.anl.gov/anlpubs/2006/01/54974.pdf.

[32] Keiser, D., J. Nucl. Mater. 514 (2019) 393, doi.org/10.1016/j.jnucmat.2018.09.045.

[33] Zhang, J. and Taylor, C., "Studies of Lanthanide Transport in Metallic Fuel," NEUP Project Report, www.osti.gov/servlets/purl/1432451.

[34] Barnes, R., J. Nucl. Mater. 11 (1964) 135, doi.org/10.1016/0022-3115(64)90002-9.

[35] Pahl, R., J. Nucl. Mater. 204 (1993) 141, doi.org/10.1016/0022-3115(93)90210-P.

[36] Hirschhorn, J., et al., Nucl. Sci. Eng. 196 (2022) 123, doi.org/10.1080/00295639.2022.2043539

[37] Hofman, G., et al., J. Nucl. Mater. 227 (1996) 277, doi.org/10.1016/0022-3115(95)00129-8.

[38] Yacout, A., et al., Nucl. Technol. 115 (1996) 61, doi.org/10.13182/NT96-A35275.

[39] Rahn, T., et al., J. Nucl. Mater. 557 (2021) 153301, doi.org/10.1016/j.jnucmat.2021.153301.

[40] Kim, Y.S., et al., J. Nucl. Mater. 327 (2004) 27, doi.org/10.1016/j.jnucmat.2004.01.012.

[41] Miao, Y., et al., Nucl. Eng. Des. 352 (2019) 110157, doi.org/10.1016/j.nucengdes.2019.110157.

[42] Bauer, T., et al., "A Program to Resolve the Safety Implications of Fuel Damage in the Operation of Advanced Metal-Fueled Reactors," ANL Document ANL-IFR-103, February 1989.

[43] Liu, Y., et al., J. Nucl Mater. 204 (1993) 194, doi.org/10.1016/0022-3115(93)90217-M.

[44] Sumner, T. and Fei, T., Nucl. Sci. Eng. 196 (2022) 309, doi.org/10.1080/00295639.2021.2009982.

[45] Cohen, A., J. Nucl. Mater. 204 (1993) 244, doi.org/10.1016/0022-3115(93)90223-L.

[46] Denman, M., "Probablisitic Transient Analysis of Fuel Choices for Sodium Fast Reactors," MIT Dissertation, June 2011, dspace.mit.edu/handle/1721.1/76578.

[47] Shu, S., et al., Nucl Eng. Des. 417 (2024) 112819, doi.org/10.1016/j.nucengdes.2023.112819.

[48] Kramer, J., J. Nucl. Mater. 204 (1993) 203, doi.org/10.1016/0022-3115(93)90218-N.

[49] Di Melfi, R., et al., "Microstructural Evolution in a Ferritic-Martensitic Stainless Steel and Its Relation to High-Temperature Deformation and Rupture Models," ANL Report ANL-IFR-176, August 1992.

[50] Gruber, E. and Hughes, T., "Experimental Determination of the High-Temperature Intermediate-Time Mechanical Properties of Alloy HT9," ANL-IFR-236, August 1994.

[51] Cannon, N., et al., "Simulated transient behavior of HT9 cladding," Proc. 14th American Society for Testing and materials International Conference3, Andover, Massachusetts, June 27-29, 1988, www.osti.gov/servlets/purl/6710925.

[52] Huang, F., et al., "Transient failure behavior of HT9," Westinghouse Hanford Report WHC-SA-2513-FP, July 1994, www.osti.gov/servlets/purl/10173288.

[53] Middleton, B., et al., "The Development of a Realistic Source Term for Sodium-Cooled Fast Reactors: Assessment of Current Status and Future Needs," SNL Report SAND2011-3404, June 2011, www.osti.gov/servlets/purl/1018468.

[54] Soffer, L., et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NRC Document NUREG-1465, February 1995, www.nrc.gov/docs/ML0410/ML041040063.pdf.

[55] Grabaskas, D., et al., "A Mechanistic Source Term Calculation for a Metal Fuel Sodium Fast Reactor," Proc. International Conference on Fast Reactors and Related Fuel Cycles (FR17), Yekaterinburg, Russia, June 26-29, 2017, www.osti.gov/servlets/purl/1365801.

[56] Di Lemma, F., et al., "Fuel Safety Research Plan for Metallic Fast Reactor Fuel R&D Focus Area: Fission Product Transport for Source Term Determination," INL Report INL/INT-19-53083, January 2019.

[57] Fay, J., et al., J. Nucl. Mater. 567 (2022) 153813, doi.org/10.1016/j.jnucmat.2022.153813.

[58] Herranz, L., et al., Prog. Nucl. Energy 66 (2013) 52, doi.org/10.1016/j.pnucene.2013.03.002.

[59] Jensen, C., et al., Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) Project: Transient Irradiation Experiments for Metallic and MOX Fuels," Proc. International Conference on Fast Reactors and Related Fuel Cycles (FR22), April 19-22, 2022, Vienna, Austria, conferences.iaea.org/event/218/contributions/18794/contribution.pdf

[60] Shivprasad, A., et al., "Current research and technology gaps related to candidate fast reactor cladding materials HT9 and cold worked D9: A summary of HT9 and CWD9 cladding properties, design equations, and data gaps," DOE Report LA-UR-23-XXXXX, September 2023.

[61] Garner, F., and Porter, D., J. Nucl. Mater. 155-157 (1988) 1006, doi.org/10.1016/0022-3115(88)90458-8.

[62] Toloczko, M., et al., J. Nucl. Mater. 212-215 (1994) 604, doi.org/10.1016/0022-3115(94)90131-7.

[63] Wen, W., et al., Int. J. Plasticity 126 (2020) 102633, doi.org/10.1016/j.ijplas.2019.11.012.

[64] Tallman, A., et al., JOM-J. Min. Met. Mat. S. 73 (2021) 126, doi.org/10.1007/s11837-020-04402-2.

[65] Toloczko, M., et al., "Comparison of Thermal Creep and Irradiation Creep of HT9 Pressurized Tubes at Test Temperatures from~ 490° C to 605° C." Effects of Radiation on Materials: 20th International Symposium. ASTM International, 2001, www.astm.org/stp10557s.html.

[66] Janney, D., "Metallic Fuels Handbook, Part 1 and Part 2," INL Document INL/EXT-15-36520, inldigitallibrary.inl.gov/sites/sti/sti/Sort_6065.pdf.

[67] Kim, Y.S., and Hofman, G., "AAA Fuels Handbook," ANL Report, March 2003, publications.anl.gov/anlpubs/2003/03/45803.pdf.

[68] Matthews, C., et al., J. Nucl. Mater. 578 (2023) 154343, doi.org/10.1016/j.jnucmat.2023.154343.

[69] Burkes, D., et al., J. Nucl. Mater. 389 (2009) 458, doi.org/10.1016/j.jnucmat.2009.02.035.

[70] Westphal, B., and Benedict, R., "Actinide recovery techniques utilizing electromechanical processes". In: Proc. Conf. Actinide Processing. Methods and Materials. San Francisco, CA, USA (1994), 157, inis.iaea.org/collection/NCLCollectionStore/_Public/25/043/25043095.pdf.

[71] Woolstenhulme, N., et al., "A Design and Fabrication Options Trade Study for Metallic Fuel without Internal Sodium Bonding," Nucl. Eng. Des. draft paper (2022).

[72] Silva, A., and Fielding, R., "Historical Review of EBR-II Metallic Fuel Fabrication," INL Report INL/RPT-23-74932, September 2023.

[73] Pace, D., and Mackowiak, B., "Extrusion Development Report," INL Report INL/EXT-18-44551, March 2018, www.osti.gov/servlets/purl/1476746.

[74] Lee, Y.S., et al., "Development of vacuum continuous casting technology for uranium," KAERI Report KAERI/TR--1755/2001, 2001, inis.iaea.org/collection/NCLCollectionStore/_Public/32/065/32065615.pdf?r=1.

[75] Rawat, M., et al., Mater. Today-Proc. In Press (2023), doi.org/10.1016/j.matpr.2023.02.452.

[76] Capriotti, L., and Harp, J., J. Nucl. Mater. 539 (2020) 152279, doi.org/10.1016/j.jnucmat.2020.152279.

[77] Wigeland, R., et al., “Nuclear Fuel Cycle Evaluation and Screening – Final Report,” INL Report INL/EXT-14-31465, October 2014, fuelcycleevaluation.inl.gov/Shared%20Documents/ES%20Main%20Report.pdf