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Test Plan Proposal for an International Joint Project for Transient Testing of Fast Reactor Fuels

K. Weaver, T. Pavey

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EXECUTIVE SUMMARY

An international joint project for transient testing of fast reactor fuels is proposed. Input from several countries is provided regarding their advanced/fast reactor programs, and the objectives and methods are given based on both historical projects and modern tools and methods. The full package would include irradiation testing using TREAT, use of existing facilities for non-destructive exams/evaluations (NDE) and post-irradiation exams (PIE), and integration of current state-of-the-art modeling and simulation (M&S) tools to aid in both experiment design, and prediction of experimental results.

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ACRONYMS

AFC	Advanced Fuels Campaign
ALIP	Annular Linear Induction Pump
ANL	Argonne National Laboratory
AOO	Anticipated Operational Occurrences
ART	Advanced Reactor Technology
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration (France)
ATR	Advanced Test Reactor
ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design Basis Accident
CEA	French Alternative Energies and Atomic Energy Commission
CEFR	Chinese Experimental Fast Reactor
CRBR	Clinch River Breeder Reactor
CSFR	Chinese Sodium Fast Reactor
DBA	Design Basis Accidents
DOE-NE	Department of Energy-Nuclear Energy
EBR-II	Experimental Breeder Reactor-II
EPMA	Electron Probe Microanalysis
ETR	Engineering Test Reactor
FBTA	Fuel Behavior Test Apparatus
FBTR	Fast Breeder Test Reactor (India)
FCCI	Fuel-Cladding Chemical Interaction
FCMI	Fuel-Cladding Mechanical Interaction
FCTT	Fuel Cladding Transient Tester
FFTF	Fast Test Flux Facility
FIB	Focused Ion Beam
FSRD	Fuel Safety Research & Development
GAIN	Gateway for Accelerated Innovation in Nuclear
GE	General Electric
GFR	Gas-cooled fast reactor
HFEF	Hot Fuels Examination Facility
IET	Integral Effects Testing
IFR	Integral Fast Reactor
IGR	Impulse Graphite Reactor

IMCL	Irradiated Materials Characterization Laboratory
INL	Idaho National Laboratory
JAEA	Japan Atomic Energy Agency
KAERI	Korean Atomic Energy Research Institute
LFR	Lead/lead-bismuth cooled fast reactor
LHGR	Linear Heat Generation Rate
LOF	Loss-of-Coolant Flow
LOHS	Loss-of-Heat-Sink
LWR	Light Water Reactor
MFF	Mechanistic Fuel Failure
MFC	Materials Fuels Complex
MOOSE	Multiphysics Object Oriented Simulation Environment
MOX	Mixed Oxide Fuel
MS	Mass Spectrometry
MSR	Molten salt cooled fast reactor
M&S	Modeling and Simulation
NEAMS	Nuclear Energy Advanced Modeling & Simulation
NRC	U.S. Nuclear Regulatory Commission
NTRD	Nuclear Technology Research and Development
ORT	Operational Reliability Testing
PBF	Power Burst Facility
PFR	Prototype Fast Reactor
PGSFR	Prototype Gen-IV Sodium-cooled Fast Reactor (Korea)
PIE	Post-Irradiation Examination
PIRT	Phenomena Identification and Ranking Table
PPS	Plant protective system
PRISM	Power Reactor Innovative Small Module
RBCB	Run Beyond Cladding Breach
SAFR	Sodium Advanced Fast Reactor
SAM	Systems Analysis Module
SEFOR	Southwest Experimental Fast Oxide Reactor
SEM	Scanning Electron Microscopy
SET	Separate Effects Testing
SFR	Sodium-cooled Fast Reactor
SHRT	Shutdown Heat Removal Tests

SLSF	Sodium Loop Safety Facility
TEM	Tunneling Electron Microscopy
TGA/DSC	Thermogravimetric Analysis/Differential Scanning Calorimetry
TOP	Transient Overpower
TREAT	Transient Reactor Test Facility
ULHOS	Unprotected Loss-of-Heat-Sink
ULOF	Unprotected Loss-of-Coolant Flow
UTOP	Unprotected Transient Overpower
VTR	Versatile Test Reactor
WPF	Whole-Pin Furnace

Test Plan Proposal for an International Joint Project for Transient Testing of Fast Reactor Fuels

1. INTRODUCTION

Several countries have been pursuing advanced nuclear reactor technologies in recent decades, including Japan, Korea, France, China, Russia, and others. The Generation IV International Forum began a "resurgence" in these new and advanced reactors, where several are based on fast, or fission spectrum, reactors.

Fast reactors have significant design work and experience throughout the world, beginning with the sodium-cooled fast breeder reactor programs in Russia, the U.K., and the U.S. during the late 1940s to mid-1950s. This expanded to other countries, and operation of fast reactors continued in France with the Phénix and SuperPhénix reactors; in Russia with BOR-60, BN-350, and BN-600; with JOYO and MONJU in Japan; with the Fast Breeder Test Reactor (FBTR) in India; and the Chinese Experimental Fast Reactor (CEFR) in China. The construction of additional fast reactors is planned in China, India, and Russia (where BN-800 was completed and put online recently), with tentative plans in France and Japan.

A key component to any reactor system is the fuel. For fast reactors, the fuel and in-core materials typically see a harsher environment than in their thermal reactor counterparts. Oxide, nitride, carbide, and metallic fuels have all been considered, both previously and currently, in advanced reactor programs. In addition to the steady-state irradiations needed for testing and qualification, fast reactor fuel systems will also need transient testing. There are four advanced reactor systems being considered for further development: sodium-cooled fast reactors (SFR), lead or lead-bismuth cooled fast reactors (LFR), molten salt fast reactors (MSR), and gas-cooled fast reactors (GFR). Most of the work and focus has been on SFRs, but there is some experience with LFRs.

Reestablishment of a fuel safety research and development (FSRD) program for fast reactor fuels is essential to meet the requirements for qualifying fuel and to meet the Department of Energy (DOE) Office of Nuclear Energy (NE) needs under the Nuclear Technology Research and Development (NTRD) program [1]. This is an especially opportune time as the Transient Reactor Test (TREAT) facility is again operational at Idaho National Laboratory (INL). The DOE FSRD program will provide the overall strategy, guidance and coordination of advanced fast reactor fuel R&D tasks related to performance under off-normal conditions. This directly supports both international and DOE-NE missions to improve economical and safety performance of advanced reactors and develop sustainable nuclear fuel cycles [2]. A well-executed program is also timely in that it can provide confirmatory experimental results for the Versatile Test Reactor (VTR) to be built by DOE and reduce the uncertainties in Modeling and Simulation (M&S) while supporting the development and qualification of an advanced fuel driver core option. Thus, the R&D outputs may directly result in improved understanding of operating margins resulting in enhanced performance potential of the VTR and similar reactor technologies in the private sector.

While metallic fuel is primary to U.S. nuclear R&D interests, oxide fuels remain a major interest to much of the international community. The U.S. experience with oxide fuels is substantial and the empirical and analytical knowledge base for oxide fuels is considerably greater than for metallic fuels. In particular, transient testing of modern metallic fuels has not been nearly as extensive as oxide fuels. The in-pile transient experimental database worldwide for oxides entails major programs within several countries for 30-50 years. For more than a decade, the only remaining in-pile safety experiments for SFR technology were performed at the Impulse Graphite Reactor (IGR) in Kazakhstan, primarily for oxide fuels supporting Japanese and, more recently, French SFR R&D, though with little capabilities to support testing irradiated fuels.

Despite little near-term U.S. interests in oxide fuels for fast reactors, the U.S. historical database of in-pile transient experiments for oxide fuels is rich from more than a decade of support for the FFTF, the Clinch River Breeder Reactor (CRBR), and the United Kingdom Prototype Fast Reactor (PFR) projects. In addition to this database, advanced oxide fuel forms with high burnup are stored at INL. The knowledge, materials, and capabilities developed through a test program will likely be of great interest to a broader international community, with crosscutting applications to oxide fuels.

Although most work and experience in fast reactors is with oxide fuels, metallic fuel is a fairly mature technology [3][4]. The U.S. experience and database with metallic fuels are substantial, comprised of over 130,000 irradiated metal pins in Experiment Breeder Reactor II (EBR-II), Fast Flux Test Facility (FFTF), and TREAT. They were demonstrated to reach fuel utilization at or above 20 at.%. To varying degrees, life-limiting phenomena have been investigated and there are no disqualifying safety-related behaviors of metallic fuels. A recent gap and Phenomena Identification and Ranking Table (PIRT) assessment concludes that "the current state of knowledge of Sodium-cooled Fast Reactor (SFR) fuel and structural material performance is sufficient for designing and licensing an SFR today within the envelope of a conservative database. Both the steady-state and off-normal irradiation database would be sufficient to support such a design" [4]. A separate study focused on accidents stated, "there are no major technology gaps in preparing a safety case…, so long as one stays with known technology" [5]. U-Pu-Zr alloy technology, although not demonstrated as large-scale as was U-Fs and U-Zr fuels in EBR-II, is believed to have sufficient information from the existing database to prepare and defend a safety case for its use. In fact, an extensive safety case was prepared near the end of EBR-II operations to convert the core to a Mark-V/VA core utilizing U-20Pu-10Zr fuel in HT9 or 316SS [6]. Still, great opportunity exists to reduce uncertainty factors incorporated in safety analysis with further fuel safety research and testing.

Metallic fuel technology is a key aspect of some SFR system designs, with implications for reactor safety, reactor operations, fuel processing technology, and overall system performance and economics. The inherent safety features of metallic fuels is the basis for passive safety mechanisms in SFRs such as the EBR-II, General Electric (GE) Power Reactor Innovative Small Module (PRISM), the Prototype Gen-IV SFR (PGSFR) from Korea, and the developing VTR designs. Modern Light Water Reactor (LWR) technologies have begun to capitalize on similar strategies which is distinct from the classical approach used by LWR designers to rely on active and redundant systems. To reduce uncertainties in transient fuel behaviors, which are usually accommodated with conservative application of uncertainties in analysis, a test program for fast reactor fuels coupled with the appropriate strategy can:

- ensure that performance of a given fuel system is adequately enveloped,
- establish fuel failure thresholds and reduce associated uncertainties,
- obtain additional data necessary to support the extension of the burnup limit beyond 10 at.% and fuel compositions containing minor actinides [4].

The restart of the TREAT facility provides a unique and timely opportunity for the advancement of an international test program. The TREAT facility will play a central role in supporting advanced fuel test programs as the only testing capability available in the world, guided and complemented by out-of-pile experiments, and integrated through M&S. Strategic development of separate effects experiments (both in- and out-of-pile) with simpler designs and well-defined boundary conditions in addition to more prototypic integral "loop" experiments will play a crucial role in the overall success of the program.

2. INTERNATIONAL COOPERATION ON FUEL SAFETY RESEARCH – TRANSIENT TESTING NEEDS AND REQUIREMENTS

As mentioned previously, several countries have operating, or operated, fast reactors, and are planning the next generation of advanced reactors based on the SFR technologies. These include:

- Japan – has operated JOYO (oxide fuel, including MOX) and MONJU (oxide fuel, including MOX), and is planning for the Japan Sodium Fast Reactor (JSFR)
- Korea – is planning the PGSFR (metal fuel)
- France – has operated Phénix and SuperPhénix (oxide fuel, including MOX), and is planning for ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration)
- Russia – has operated BOR-60, BN-350, BN-600, and BN-800 (all oxide fuel, including MOX), and is planning BN-1200 (oxide fuel, including MOX) and at least two lead/lead-bismuth cooled fast reactors (nitride fuel)
- China – has operated CEFR, and is planning the CSFR (Chinese Sodium Fast Reactor)
- India – has operated FBTR (carbide fuel, with plans to convert to metal fuel), and is constructing the Prototype Fast Breeder Reactor (PFBR) (oxide fuel to start, with eventual conversion to metal fuel).

While all of the countries above would benefit from performing transient testing in TREAT, only the first 3 (Japan, Korea, and France) are viable candidates for international cooperation due to current political and policy attitudes. As such, potential international users were contacted from: Japan, via the Japan Atomic Energy Agency (JAEA); Korea, via the Korean Atomic Energy Research Institute (KAERI); and France, via the French Alternative Energies and Atomic Energy Commission (CEA). They were asked for general interest in TREAT, and what their potential requirements would be given their respective programs.

Both Japan and Korea provided the additional information. However, as of the writing of this report, we had not received further information from France. When CEA provides the additional information, this report will be updated and a revision will be issued.

2.1 Japan

The Japan Atomic Energy Agency (JAEA) recognizes that a few programs between JAEA and DOE, like CNWG, may include possible proposals for TREAT tests. The JAEA-SFR fuel expert team provided general interest in TREAT tests in the following subsections.

2.1.1 Potential Needs for TREAT Experiments

- To obtain fuel performance data, including fuel melting and cladding deformation in transient overpower (TOP) events
- To evaluate the fuel pin breach limit in TOP events
- To assess power-to-melt in TOP events.

2.1.2 Candidate Test Fuels for TREAT Experiments

- High Pu content MOX fuel
- Minor actinide (MA) bearing MOX fuel
- Both solid and annular pellet types
- Several cladding types, including PNC316 / ODS.

Combinations of fuel-cladding will be down-selected according to the limitations/capabilities of experimental conditions.

2.2 Korea

The Korea Atomic Energy Research Institute (KAERI) provided general interest in TREAT tests in the following subsections.

2.2.1 Test Conditions

Transient conditions of interest are:

- Reactivity initiated accidents (RIA)
- Loss of coolant accidents (LOCA)
- Power ramps
- TOP
- Other conditions, including power (LHGR) change rate (W/sec), and initial state (power, temperature, pressure, etc.).

Depending upon the reactor type, the transient conditions will vary greatly. For example, LWR transients such as RIA and LOCA are different from those of SFR or LFR. In addition, transients in SFR with metal fuel are much different than those in SFR with oxide fuel for RIA due to different reactivity characteristics.

Previous TREAT tests of SFR metal fuel may not be completely representative of actual RIAs in SFR with metal fuel. Therefore, the list of transient conditions for tests need to be derived first, taking into account all potential reactor candidates to be covered including current LWRs, future fast reactors, and innovative SMRs. In addition, all the different fuels such as metal, oxide, nitride, carbide and other innovative fuel types and fuel compositions also need to be considered.

2.2.2 Fuel to be Tested

The size and volume of test fuel will affect the type of transient tests needed. Test fuel can be fuel specimens, small size rodlet, fuel rod, fuel bundle of several rods, and fuel assembly.

2.2.3 Measurements During the Test

Key measurements to take during the transient tests include:

- Power (LHGR, total power, neutron flux, axial and radial variations)
- Temperature (fuel centerline, bulk fuel, cladding, coolant)
- Fuel dimensional changes
- Pressure in and out of fuel
- Initiation time of fuel failure.

Key measurements after a test include:

- Fuel state as it is just after transient test
- Non-destructive examinations (NDE), including visual examination
- Sampling plan for post-irradiation examinations (PIE)
- PIE (microstructure, element/constituent migration, chemical reaction states, fission gas release [FGR], mechanical state [deformation, fracture toughness, ductility, etc.]).

2.3 France

Although CEA did respond to the initial request, they have not provided any additional information regarding the types of transient tests (including requirements) that would be needed for their advanced reactor programs (e.g., ASTRID). Any future response by CEA will be included as an update to this report.

3. SUGGESTED SCOPE FOR AN INTERNATIONAL FUEL SAFETY TEST PROJECT

3.1 Purpose/Objective

The primary objective of an international fast reactor FSRD program is to reconnect to historical knowledgebase, maintain, and develop the technical basis for fast reactor fuel performance in off-normal conditions. The specific objectives of the program are to:

- Perform and support investigations of performance limiting technical issues for reactor designs (i.e., JSFR, PGSFR, ASTRID, VTR, other commercial fast reactor designs, etc.),
- Provide experimental evaluation of fuel behavior under power-to-cooling mismatch scenarios for the development of advanced fuel designs and behavioral models,
- Identify fuel failure mechanisms and important failure thresholds while providing various levels of experimental characterization of these to develop, improve, and validate predictive modeling capability,
- Provide confirmatory and qualification experimental evaluations for fuel designs to establish margins and support validation of fuel and systems modeling tools.

3.2 Overview

The ultimate objective of reactor safety is the containment of core materials. This is achieved through main three barriers to release: the fuel cladding, the primary pressure vessel, and the primary pressure vessel containment. Maintaining the integrity of the fuel is the first priority. To this end, limits are established to ensure core damage is avoided during normal operational conditions. Some level of defects in core components must be anticipated and accommodated to ensure reliable and safe operation of the reactor. Since the 1980's, SFR designs in the U.S. using metallic fuel (mainly coming from the Integral Fast Reactor [IFR] program) have emphasized inherent safety features that provide benign characteristics in a passive response to upset events. This approach follows the more effective strategy of safety controls that rely less on administrative and engineering controls. With this approach, even Beyond Design Basis Accident (BDBA) conditions with very low occurrence probabilities can result in little or inconsequential fuel damage [7][8][9], depending on the core and plant design. Thus, both the probability and the potential consequences of off-normal conditions for a given reactor design must be evaluated. In describing performance in these conditions, the reactor core cannot be decoupled from the remaining system.

While the primary focus for a fuel safety program is understanding the behavior of the fuel system under off-normal conditions, other components and system behavior are not of explicit focus. A detailed and complex analysis of the complete system is required to describe the thermal-hydraulic envelope defining boundary conditions on the fuel system. A mechanistic study of specific plant transients alone does not define the full range of conditions that should be considered for fuel safety. Assessment of key fuel transient behaviors and damage thresholds to points of serious failure is very important to define margins that regulate allowable fuel performance limits. M&S plays a central role in the science-based strategy described in following sections. The fuel safety program will rely heavily on modern M&S tools to develop the technical requirements and objectives for experiments while, in turn, producing invaluable data for development and validation of models for fuel damage and its potential for propagation and consequences. Experimental developments in instrumentation and diagnostics can also provide opportunity to develop methods for detecting pin failures or incipient conditions with broader application that can influence operating strategies.

3.3 Conditions of Interest

Licensing of a fuel system requires identification of all degradation mechanisms and failure modes and definition of failure thresholds corresponding to each degradation mechanism. From the U.S. Nuclear Regulatory Commission (NRC), the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants requires [10]

1. the fuel system is not damaged as a result of normal operation and anticipated operational occurrences;
2. fuel system damage is never so severe as to prevent control rod insertion when it is required;
3. the number of fuel rod failures is not underestimated for postulated accidents; and
4. coolability is always maintained.

Most, if not all, international regulatory bodies maintain the same (or very similar) regulatory requirements for fuel system behavior under accident conditions.

Although other factors may be important to fuel damage, cladding temperature during reactor transients plays a key role in damage assessment as most cladding failure mechanisms are strongly temperature dependent. The cladding temperature is characteristic of transient events resulting in power-cooling mismatch conditions, which are classified as Anticipated Operational Occurrences (AOO), Design Basis Accidents (DBA), and BDBA [11]. AOOs are expected to occur at least once in the lifetime of a reactor while accidents are not expected to ever occur but are theoretically possible. Within-design-basis events are of strong interest in the design and development stages through licensing. In SFRs, common DBAs include Transient Overpower (TOP), Loss-of-Coolant Flow (LOF), and Loss-of-Heat-Sink (LOHS) accidents. 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 [12].

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 BDBAs is very low, ATWS events have been of significant interest in fast reactor safety. In part, this is due to the fact that 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 Transient Overpower (UTOP), the Unprotected Loss-of-Coolant Flow (ULOF), and the Unprotected Loss-of-Heat-Sink (ULOHS) accidents [9][11]. In all cases, which cover a wide range of thermal conditions, experimental programs are needed to understand the impact of associated temperature histories on cladding damage accumulation and margins and to validate the underlying key design assumptions regarding inherent passive safety mechanisms in SFRs. For reference, a summary of the range of cladding temperature response to various transients was given in [12] and is provided in Figure 1.

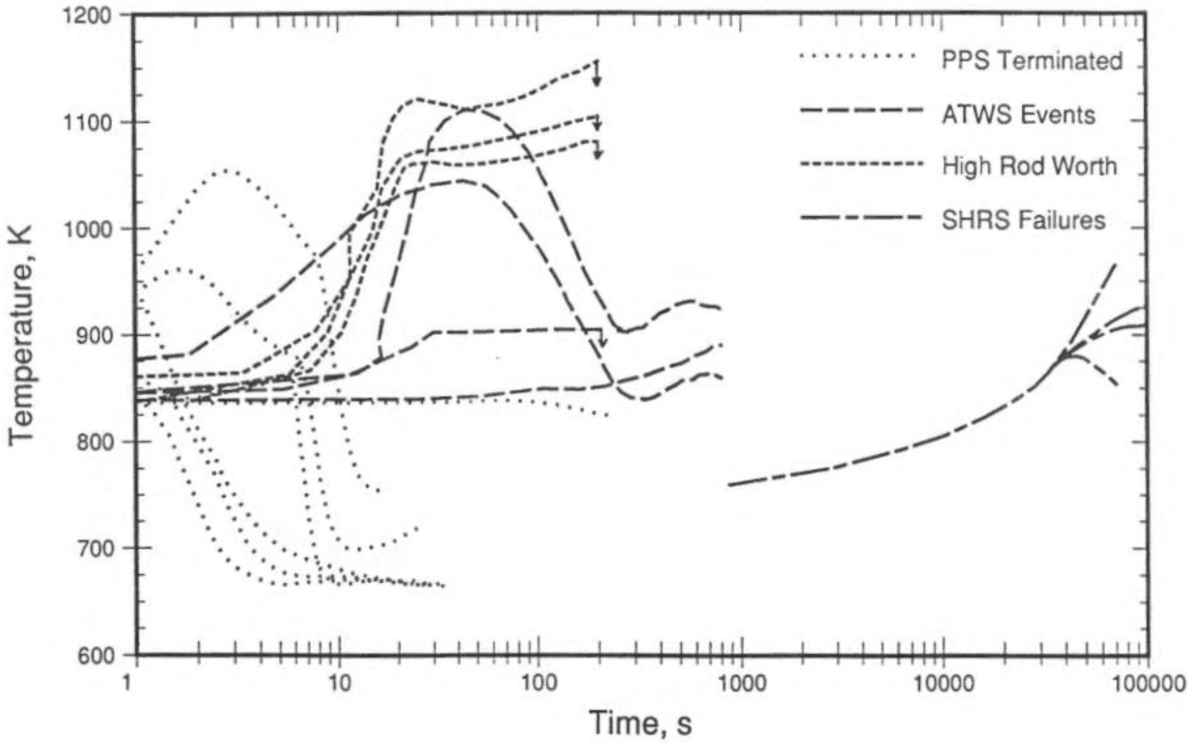

Figure 1. Overview of representative calculated peak cladding temperatures for metallic fueled SFR from [12] for a variety of events and specific conditions.

Reactor transient testing and much of the historical TREAT database may be divided into two categories distinguished by the point of cladding breach. Operational transient testing focuses on behavior during anticipated transients up to the point of cladding breach. Reactor safety testing studies the cause and nature of cladding rupture and post-failure fuel behavior during accident conditions [13]. Transient testing experiment programs may span into both categories of testing. For operational transient testing used to support FFTF and CRBR safety programs, the general objectives were to: 1) demonstrate core component performance allowed by the plant protective system (PPS), 2) establish the margin between PPS capability and component failure, and 3) confirm validity of design procedures (criteria, methods, etc.) [14]. The U.S. DOE and Power Reactor and Nuclear Fuel Development Corporation (Japan) Operational Reliability Testing (ORT) program is an example of operational transient testing, carried out in the 1980s for oxide fuel, largely using the EBR-II and TREAT facilities.

3.4 Science-based, Engineering-Focused Approach

The R&D approach comprising the core of this plan is based on the “solution-driven, goal-oriented, science-based approach to nuclear energy development” described by the DOE NE Roadmap [1]. This strategy aims at increasingly reliance on fundamental science and computational platforms to develop and qualify new technologies. The major elements of this strategy shown in Figure 2 require close coordination between experimental and M&S efforts, and, ultimately, lead to demonstrations where appropriate, based on technical merit and commercial commitment garnered in the R&D process.

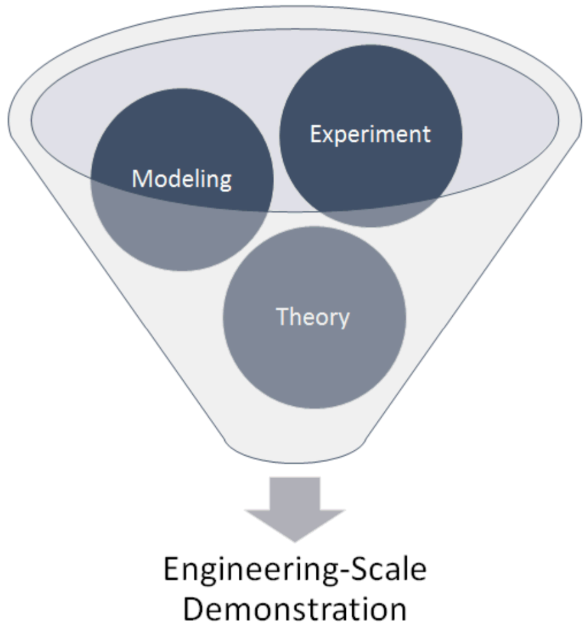

Figure 2. Major elements of science-based R&D (adapted from [1]).

Past research has relied heavily on empirical demonstrations of performance as a measure of readiness to move to the next tier of system integration. Such experience- and intuition-based approaches often have limited ability to predict performance and anticipate behavior that are usually governed by complex interactions of physical/chemical processes among integrated components. Therefore, multiple iterations of design, test, and evaluation are necessary, and the result of each interaction is highly unpredictable. Modern approaches often integrate advanced M&S with experimental validation to assess performance, provide understanding, and predict behavior, which is often termed as science-based, engineering focused approach. For this proposed fuel safety program, M&S serves as both a critical tool for experimental design and result interpretation, and a beneficiary that uses the experimental data to improve and validate sub-models in the codes.

3.4.1 Separate-Effects to Integral-Effects Testing

Taking advantage of Separate Effects Testing (SET) and Integral Effects Testing (IET) is central to achieving the science-based, engineering-focused approach. (Of course, the distinction between SET and IET is graded.) Separate effect testing (SET) plays a critical role in enhancing the sub-model used in simulation codes. Separate effects experiments provide unique opportunity for developing an understanding of the underlying theory related to a physical process with the goal of isolating phenomena of interest. SET design facilitates tightly controlled boundary conditions and focused measurement diagnostics to reduce uncertainty in measured results and forcing conditions. The implicit simplicity implied by SET requires less resources for a given test facilitating higher throughput and evolution. These conditions provide ideal input for developing theory and validating M&S tools [15]. For these reasons, SET has become a recognized foundation of technology development.

Early nuclear energy R&D relied heavily on large experiment and technology demonstrations, requiring substantial resources, though relatively affordable at the time. For nuclear fuels and materials behaviors, facilities like the TREAT facility were built to drive simulated boundary conditions using simplified, compact devices. In a resource constrained environment, IET experiments should generally serve two major purposes: 1) early phenomena identification experiments that elucidate fuel behaviors and failure mechanisms to be compared with modeling predictions and provide the basis for detailed testing strategies [12], and 2) select confirmatory or qualification testing that will provide ultimate validation of M&S tools and has been and is still embraced by the regulatory process. "Integral" M&S tools with improved sub-models developed by SET use IET experiments that approximate the actual service conditions are used for validation. This overall approach to code validation is illustrated in Figure 3 and described in literature [15][16].

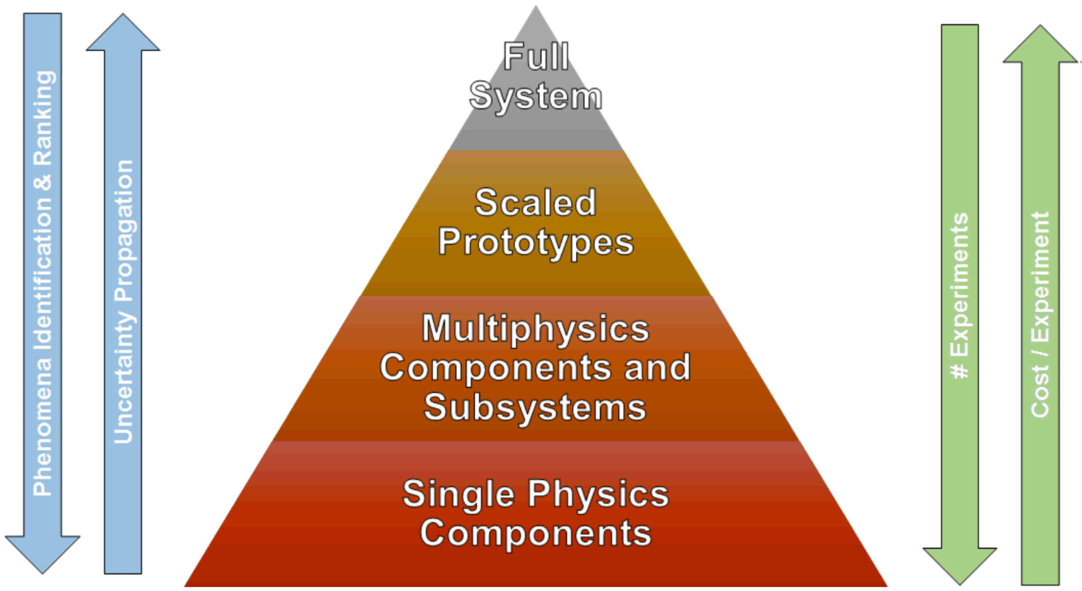

Figure 3. Relationship of Separate Effects Testing and Integral Effects Testing to experiment design and required resources (adapted from [16]).

Code developers have routinely attempted to use historic data for this purpose but find that the data collected are often insufficient to demonstrate the higher fidelity features that are provided and, thus, prevent realization of the full potential of a code. Consequently, new IET that implements modern data intensive experimental techniques based on modern sensors and instrumentation systems is required.

4. FUEL TESTING NEEDS

Fuel behavior and response to relevant off-normal conditions is the primary target of fuel safety R&D. To accomplish this goal, a science-based, engineering focused approach is being taken that capitalizes on the restart of the TREAT facility and nearly three decades of advancements in M&S and instrumentation since the last fuel experiments. The categories of required transient testing [38] correspond to the specific objectives of the fuel safety program as follows:

Developmental Testing. Developmental testing is a tool used primarily by fuel developers to inform fuel designs and specific features. The experiments are used to measure transient response and identify potential concerns for reactor operations. It is also used to perform detailed evaluation of specific fuel behaviors to reduce model uncertainties. Test conditions typically proceed from mild to aggressive energy depositions and ramp rates. Understanding intrinsic fuel response is the primary objective such that these transient tests can be conducted using simple, capsule-type with reduced fuel lengths experiments in a transient test reactor. These tests typically fall into the category of SET testing. In some cases, developmental testing may utilize IET to clearly identify relevant phenomena.

Limits Assessment Testing. Limits assessment testing focuses on characterizing fuel pin failure thresholds and post-failure consequences. These tests provide the basis for establishing operational limits for the SFR designs to maximize performance while ensuring safety. The results are of use to both fuel developers and reactor designers. This category of testing ultimately should be performed on full-size fuel pins under prototypic coolant flow and fuel/cladding temperature conditions (high level of integrality). These tests require careful design to specifically target final experiment conditions to just before and after failure. Measurement of online fuel motion is a primary objective for limits assessment testing.

Confirmatory/Qualification Testing. Confirmatory testing is used to confirm established thresholds and limiting conditions of operation for an established fuel and reactor design (including the specific characteristics of each). Therefore, it represents the final phase in the fuel qualification process producing results used in the preparation of the fuel/core safety evaluation that will be the subject of regulatory review. Given the high level of experimental confidence, these tests are performed on full-size fuel pins and/or small bundles of pins under prototypic conditions of energy deposition, coolant flow, and fuel/cladding temperature profiles. High quality assurance requirements are expected for all experimental inputs and outputs.

4.1 Metal Fuel Behaviors of Interest – An Example

The following provides an example, using metal fuel, of the behaviors of interest. Steady-state fuel performance provides the initial state of the fuel, which is required to predict transient fuel performance. A detailed overview of metallic fuel performance and approaches to modeling is provided by [39]. Figure 4 provides a summary of key metallic fuel phenomena under irradiation. A historical overview of the development of modern metallic fuels is provided by [20] and many other references therein. This section summarizes key transient phenomena and behavioral models needed to predict the transient performance of metallic fuels under postulated transient conditions to failure.

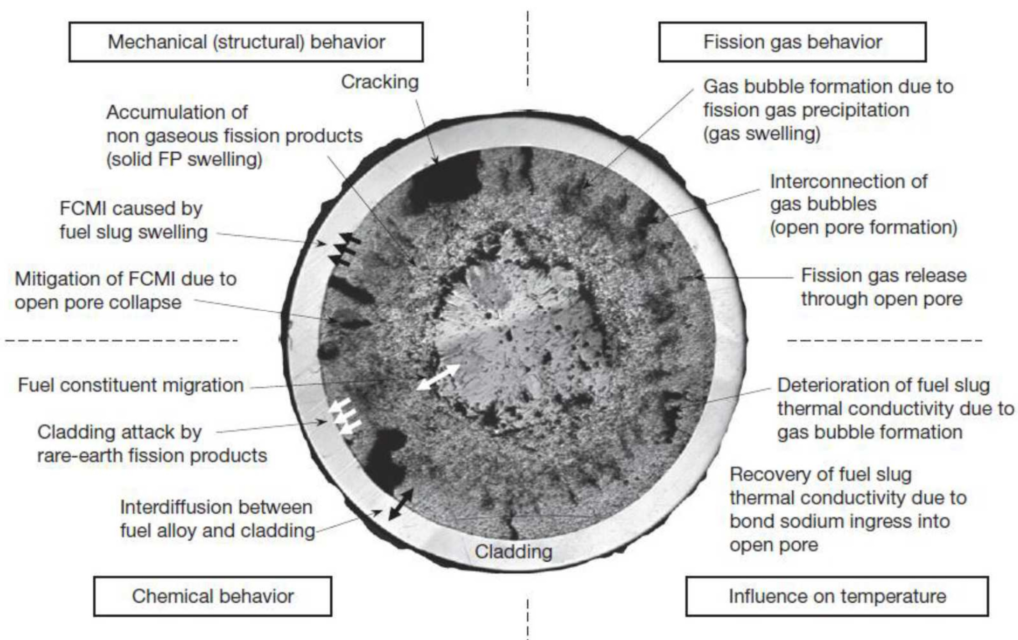

Figure 4. Summary of metallic fuel behaviors under irradiation (from ref [39]).

4.1.1 Behavior of Metallic Fuels Under Transient Conditions

Several reviews have been provided summarizing the performance of metallic fuels under transient conditions. A detailed review of the early transient testing evolution accomplished in the TREAT facility is provided in [17]. A summary of important transient testing experiments performed in the TREAT facility for SFR fuels during the 1980's is given by [32]. A detailed review of the M-series tests comprising the most relevant findings for modern metallic fuel is given by [18]. The safety implications for SFRs for metallic fuel performance under off-normal conditions have been summarized by [7][8][9].

4.1.2 Key Transient Fuel Behavioral Models

Power to melting

The linear power to melt, i.e., the linear power required to cause fuel centerline to melt, is an important indicator that characterizes safety margins. It is evaluated on the basis of calculations of the temperature distribution and solidus temperature of a fuel slug. Fuel melting is a complex function of fuel specifications, irradiation conditions, fuel behavior such as radial composition (constituent migration, solid fission product accumulation), smear density (swelling), axial linear power profile, sodium infiltration into fuel, and coolant conditions. To calculate radial temperature profile, the most important model is the effective thermal conductivity of the irradiated fuel, which is usually a simple model to account for the volume fraction of pores. The effective thermal conductivity of the fuel can decrease down to 40-50% of the unirradiated fuel [40]. The sodium infiltration into connected pores can recover part of the loss in effective thermal conductivity. Reported thermal conductivity uncertainty is up to 25%, depending on fuel burnup [41].

Fuel expansion and molten motion

Axial fuel extrusion is important in both fuel pin integrity in accidents such as unprotected transient overpower for reactor core safety. In a molten zone of a fuel slug, the coalescence of fission gas bubbles leads to the axial upward extrusion of the molten fuel, which was observed in TREAT experiments using the fast-neutron hodoscope. It occurs in the case of fuel slug melting by overheating in a transient overpower event, and also in a loss of flow event with fuel slug liquefaction by Fe diffusion resulting from FCCI. First, fuel melting does not induce significant FCMI, because melting usually starts near the top of the fuel pin and upward expansion occurs instantaneously upon melting. Furthermore, axial fuel expansion provides the reactor core with a large negative reactivity. Conversely, there was no shrinkage upon fuel cooling observed in TREAT tests, and, therefore, no consequent positive reactivity insertion. Fuel extrusion is larger for low burnup fuels as more fission gas is retained in the fuel as observed in U-Fs experimental data. Current models for estimating the amount of extrusion are based on the internal pressure of coarsened gas bubbles balancing the plenum gas pressure.

Liquefaction at the fuel-cladding interface

Liquefaction at the fuel-cladding interface directly impacts the safety margin to cladding breach. The liquefaction, i.e., FCCI during steady-state irradiation, is dominated by the eutectic reaction of lanthanides with the cladding. During some transient events, the cladding inner temperature can exceed certain threshold values, resulting in the formation of a liquid phase in the reaction zone between fuel slug and cladding. The liquefaction reaction promotes significant cladding wastage and Fe diffusion from cladding into the fuel. Out-of-pile tests, specifically FBTA tests, were used to characterize the penetration rate and models were proposed based on the experimental data. However, the dependence of liquid-phase cladding penetration on temperature, heating time, and burnup has not been quantitatively assessed.

Fuel pin failure mechanism

The cause for fuel pin failure in transient events can be a combination of the plenum gas pressure rise due to coolant temperature increase and FCCI from steady-state irradiation plus liquid-phase penetration. Because the thermal properties of the metallic fuel system allow peak temperatures to closely follow the axial coolant temperature profile (even under transient conditions), FCCI is more significant near the top of the active fuel column and the cladding strength is lower. All cladding breaches have been observed near the top of the fuel pin in TREAT M-series tests. The dominant factor in the fuel pin failure mechanism is dependent on the fuel burnup and the event type (i.e., power-to-flow and time dependencies). A high-burnup fuel pin will burst as a result of higher plenum gas pressure before liquid-phase eutectic penetration develops. A low-burnup fuel pin will take more time before the cladding bursts, and liquid-phase penetration will develop during that time.

Post-failure behavior

In TREAT M-series tests, the molten fuel alloy in failed fuel pins ejected rapidly through the breach site and dispersed in the coolant. As mentioned above, the breach always occurs at the top of the fuel column allowing for the fuel to be swept out of the core region, providing negative reactivity feedback to the core. The driving force of the fuel ejection was believed to be the expansion of trapped fission gas, bond sodium vapor pressure, and plenum gas flow toward the breach site. The non-reactive chemical interaction of Na coolant and metallic fuels and the relatively close melting point of the fuel to the Na boiling point results in a rather benign outcome. All M-series tests were done in individual flow tubes, so post-failure interaction of pins and event propagation was not evaluated. Out-of-pile molten fuel quench experiments have also been performed to study fuel-coolant interaction providing additional validation of large coolability margins in metallic fuel SFRs [25][41].

4.2 R&D Focus Areas

The design and safety analysis of metal fuel pins require an understanding of the irradiation behavior and a quantitative evaluation of its influence on the margin to fuel pin failure. The various phenomena in a fuel pin during irradiation, as shown in Figure 4 above, are interlinked and fundamentally based on fuel microstructure evolution. The comprehensive understanding of irradiation behavior needs multi-scale, multi-physics models that are based on respective phenomena (separate effects) and relationship among them (integral effects). To accomplish the objectives of fuel safety R&D for metallic fuels, three key R&D focus areas have been identified as outlined in Figure 5.

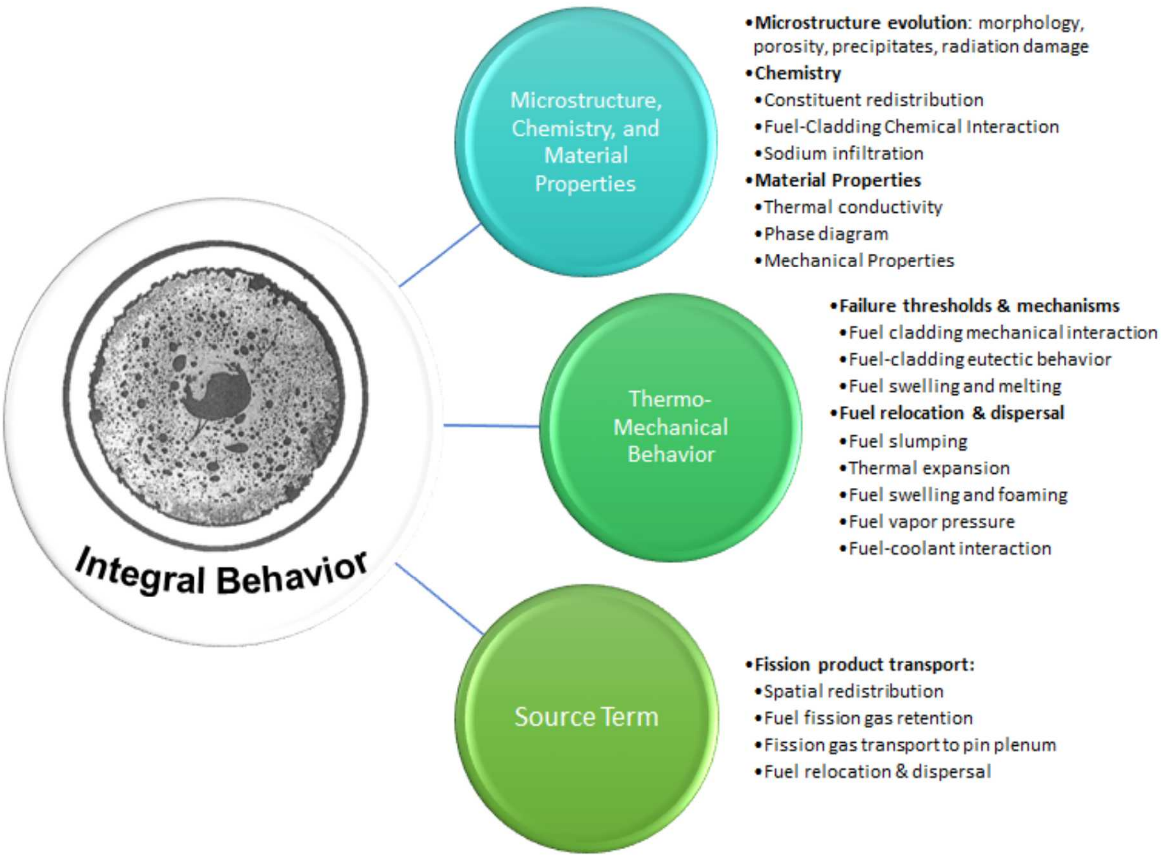

Figure 5. Three R&D focus areas for fuel safety research for metallic fuels.

The three focus areas for fuel safety R&D are:

Microstructural, Chemistry, and Material Properties. This category encompasses detailed understanding of the impacts of irradiated fuel microstructure on transient performance and potential consequences. Experimental approaches are typically SET in nature to develop an experimental basis for the key fundamental phenomena driving metallic fuel transient performance. This focus area provides direct input to the two remaining focus areas and is aligned with the development and validation of detailed mechanistic models. It is also interesting to note that the only reference available for thermal conductivity of irradiated U-xPu-Zr alloys comes from examination of experiments performed in TREAT [40].

Thermo-Mechanical Behavior. The thermomechanical response of metallic fuels is of greatest concern to fuel and reactor designers, regulators, and utilities. Typically, the thermomechanical models are based on integrated mechanistic behaviors from the first focus area. Thermo-mechanical areas correspond with the key transient models of interest. Along with IET experiments, SET experiments may be used to focus on particular transient models of interest such as power to melt, fuel axial expansion, relocation, etc.

Source Term. A recognized gap in the metallic fuel SFR database [42][43], source term studies will focus on capturing detailed mechanistic to integral understanding of fission product transport in metallic fuels during off-normal conditions. The availability of legacy irradiated materials of high relevancy provides opportunities to exploit advanced characterization facilities available today to understand long-lived fission product distribution and response to transient conditions. In particular, in-pile experiments on metallic fuels including experiments in sodium coolant will be emphasized to extract impactful data, including information on short-lived isotopes only accessible through in-pile experiments. The historical safety programs recognized the potential of leveraging the TREAT experiments, which were mostly focused on studying transient thermomechanical behaviors, for source term data during late planning phases of the program.

4.3 Modern Applications

4.3.1 DOE Advanced Reactor Fuel Development and Deployment

The FSRD is part of the NTRD Advanced Fuels Campaign (AFC). The initial testing strategy has a primary interest to connect with the historical database. The availability of fuel specimens for testing is another factor considered for test matrix selection. Fresh metallic fuel specimens are manufactured at INL and will be needed to support initial experiment commissioning tests that may also provide ideal model validation data. In some cases, advanced fuel forms present characteristics that may be more important during early fuel life, such as He-bonded fuel designs with more exotic geometries (e.g., annular). For example, will annular fuel collapse into the center during over temperature events introducing potential for reactivity insertion hazards? Or will it induce enhanced fuel-cladding mechanical interaction early in life due to reduced gap? Other advanced fuel forms could result in related questions. In any case, the priority test specimens are legacy irradiated materials from EBR-II and FFTF stored at INL. In particular, full length pins irradiated FFTF are presently invaluable as source material for continued evaluation. The Mechanistic Fuel Failure (MFF) fuel assemblies are U-10Zr pins in HT-9 cladding.

4.3.1.1 Methods to Accelerate Technology Development

Though important irradiated material stockpiles exist from EBR-II and FFTF, producing irradiated specimens of new fuel without the availability of a fast spectrum test reactor is a recognized challenge. Recent design efforts have looked at the possibility of scaling fuel pin diameters to accelerate burnup in testing at the Advanced Test Reactor (ATR) [44]. Results show that reducing the fuel diameter by one half can reduce the time to 30 at% burnup from approximately 12 years to 2-3 years. The report also highlights related interests in post-irradiation furnace testing to failure and some of the challenges associated with scaling cladding to represent prototypic cladding behaviors. Preparations are currently underway to perform experiments to evaluate this approach. As noted in the report [44], this approach has great potential in evaluating fuel failure at various burnups. This approach may play an important role in filling a potential gap in fuel safety R&D.

4.3.2 Fuel Specifications of Interest

A reference fuel form for R&D should be aligned with current applications and needs such as the VTR or existing commercial SFR designs. To align with these applications, the parameters are derived to be consistent with recommendations from [3] as shown in Table 1. A design with characteristics similar to the Mark-V/VA driver fuel prepared for EBR-II near the end of operations. In this regard, with a well-established experience base for this fuel concept, primary opportunities for testing support would fall into the category of performance extension testing through margin reduction and confirmatory testing to link to the historical database. A summary of historic EBR-II fuel specifications is given in Table 1 of [45].

Table 1. Reference fuel specifications for safety R&D based on preliminary metallic fuel specifications for VTR design.

Composition, wt%	Smear Density, % TD	Fuel Height, cm	Fuel-Cladding Bond Material	Cladding Material	Peak Linear Heat Rate (W/cm)	Burnup (at% HM)
U-xPu-10Zr, $x \leq 20\%$	75	~100	Sodium	HT9 or 20% cw 316SS	450	10

Advanced fuel forms beyond initial startup may also be of interest for the VTR and could take the form of a He-bonded fuel (lacking Na-bond) of exotic geometry (e.g., annular). Advanced concepts will likely require more testing support for characterization and validation of fuel transient response and failure behaviors.

4.3.3 Needs of Private Industry

Multiple private companies are pursuing reactor designs that leverage the U.S. database for metallic SFR fuels. Transient testing experiments are of interest to support meaningful reduction of fuel behavioral uncertainties and database extension. In some cases, design requirements necessitate extension of the existing database. The experiments planned under the AFC program will aid in satisfying many desired fuel safety questions. Once again, the legacy materials from EBR-II and FFTF are of high interest and, in some cases, are desired for individual testing purposes. Due to the unique and irreplaceable nature of these specimens and the large historical investment of DOE, ideally these specimens are used for experimental purposes funded by federal money to provide outputs that can benefit all U.S. industry. Of course, the experiment objectives should be co-developed with industry partners to provide the greatest overall impact.

The facilities and expertise created and maintained for safety R&D will provide opportunities to be leveraged by U.S. laboratories, industry, academia, as well as international R&D institutions.

4.3.4 International Programs

Significant SFR technology programs exist in many countries around the world, as described earlier. The predominant fuel of focus in these programs is oxide. Only Russia, India, and China operate SFR technology today, each having geo-political barriers for close collaboration with U.S. institutions. The JOYO reactor in Japan is planned to be restarted but with an uncertain timeline. Countries with great interest in SFR fuels and materials and active R&D include France, Japan, and Korea. As mentioned earlier, a lack of existing experimental facilities places the TREAT facility as nearly the only facility capable of doing in-pile transient testing [31]. The TREAT facility is unique in the world, however, in its ability to work with irradiated materials due to the proximity of world-leading Post-Irradiation Examination (PIE) facilities and a steady-state material test reactor, the ATR (thermal reactor). International programs have expressed interest in using TREAT facilities to support their R&D efforts, and specific tests and fuel forms were described earlier. In addition to legacy metallic fuels, legacy oxide fuels of advanced annular designs with burnups to 20 at% are also available at INL and hold high potential for meeting needs of international programs. Discussions with the French SFR R&D programs have been underway since the restart of the TREAT facility was announced.

4.4 Modeling and Simulation

Consistent with the science-based, engineering-focused approach described earlier, M&S plays a fundamental role in fuel safety research. In fuel safety R&D, M&S tools are vital for designing impactful experiments; while M&S developers are also a primary consumer of experimental data for model development and validation. This concept is illustrated in Figure 6 below. The following sections provide an overview of primary relevant modeling tools used in the U.S.

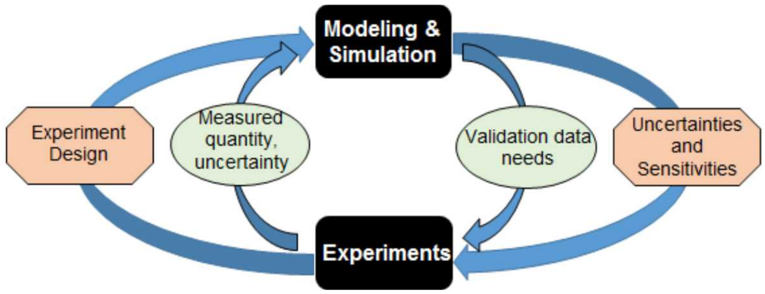

Figure 6. Relationship between modeling and simulation and experiments. Experiments provide data meeting validation needs while modeling provides design tools to explore system response.

4.4.1 Overview of Relevant Codes

Computer simulation has been an integral part of fast reactor design and safety analysis since the 1970's. In a recent review, approximately sixty available computer codes were identified for the safety analysis of fast breeder reactors [46]. Although capabilities have eroded due to cancellation of major SFR projects in the early 1990's, critical modeling and simulation capabilities have been maintained primarily at Argonne National Laboratory (ANL), including cross-section processing, core physics, fuel cycle and depletion, thermal-hydraulics, and reactivity feedback and transient safety analysis. Figure 7 shows the robust work-flow for design and safety analyses of fast reactor codes, summarized by [47].

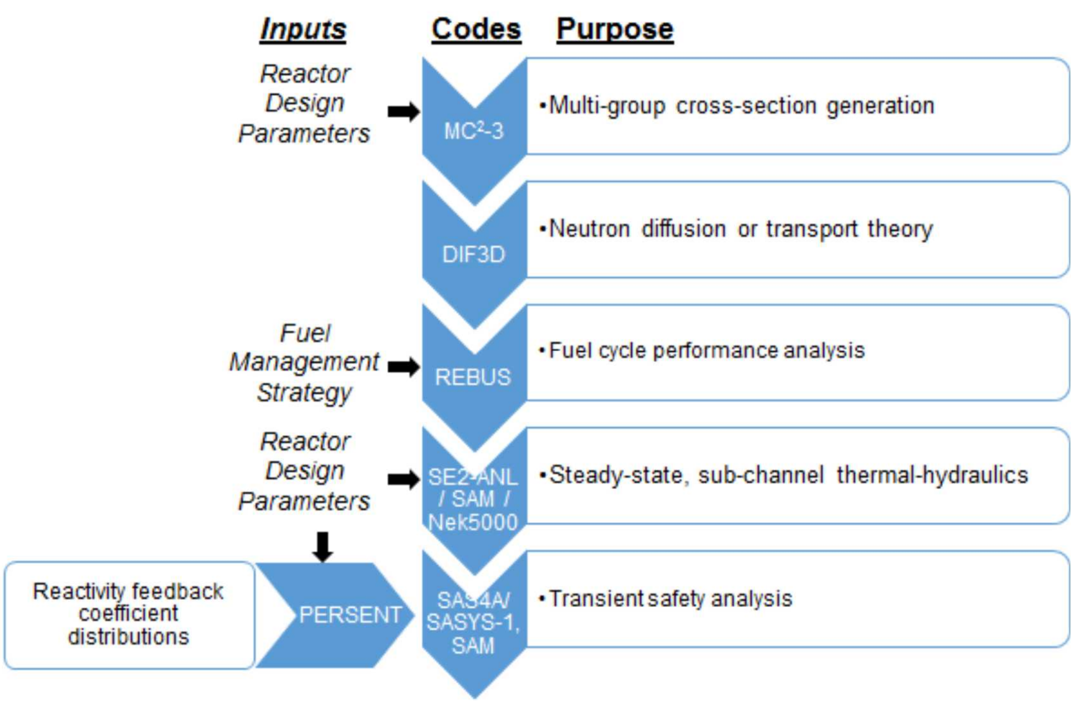

Figure 7. Overview of computer codes used for SFR analysis adapted from [50].

4.4.1.1 Important Fuel Performance and Systems Analysis Codes

SAS4A/SASSYS-1

More specifically for transient analysis, SAS4A/SASSYS-1 is the simulation tool to perform deterministic transient safety analysis for liquid-metal cooled fast reactors. The events that are analyzed include anticipated operational occurrences and design basis and beyond design basis accidents. A detailed user's guide for the code is found in [41]. The code is capable of multi-scale analysis, from a fuel pin to the whole plant. The code was originally developed to analyze fuel failure models specific to SFRs using basic core thermal-hydraulics, reactivity feedback, and system analysis features. The code has significant heritage supporting the licensing of FFTF, CRBR and MONJU. SAS4A was originally developed to analyze sever core disruption accidents with coolant boiling and fuel melting and relocation for oxide fuels. Models have been extended and adapted to metallic fuel analysis. SASSYS-1 allows detailed thermal-hydraulic analysis of the primary and secondary coolant circuits and the balance of the plant steam-water system. The "two" codes were merged in the 1980's to provide complete system transient analysis, where the whole-plant dynamics capability of SASSYS-1 enables prediction of passive safety feedback.

The specific capabilities and features of SAS4A/SASSYS-1 are summarized in the Table 2. The code validation is based on EBR-II shutdown heat removal tests, which includes unprotected loss of heat sink and protected and unprotected loss of flow; FFTF loss of flow without scram; PHENIX shutdown test; and MONJU turbine trip test. For fuel failure validation, the past experiments in TREAT and furnace tests of whole pins were used, which point to the need for future TREAT experiments to provide data for code validation. The identified capability gaps in SAS4A/SASSYS-1 for metal fuels include: thermal stratification (especially in loss of flow accidents), fuel failure models and accident progression (metal fuel models are being developed), and spatial kinetics. The codes are actively maintained under the DOE Advanced Reactor Technology (ART) program. Recent enhancements include coupling of Computational Fluid Dynamics (CFD) (StarCCM+), Dakota for transient Uncertainty Quantification/Sensitivity Analysis (UQ/SA), and a new fuel axial expansion model. The code is also used by US universities to support Nuclear Energy University Programs.

Table 2. Overview of SAS4A/SASSYS-1 code modules and their purpose [41].

Modules	Purpose
TSCLO	Single phase coolant thermal-hydraulics and fuel pin heat transfer
TSPK	Reactor point kinetics and reactivity feedbacks
PRIMAR-4	Primary and secondary coolant loops, thermal-hydraulics and heat transfer
CNTLSYS	Reactor and plant control and protection system simulation
BOP	Balance-of-plant systems and thermal-hydraulics and heat transfer
DEFORM-4	Oxide fuel pin - fuel and cladding mechanics
DEFORM-5	Metallic fuel pin – cladding mechanics
SSCOMP	Metallic fuel pre-transient characterization and mat. props.
FPIN2	Metallic fuel and cladding mechanics
TSBOIL	Two-phase coolant thermal-hydraulics and fuel pin heat transfer
CLAP	Molten cladding relocation and heat transfer
PLUTO2	Post-cladding-failure oxide fuel-coolant interaction with fuel-coolant thermal-hydraulics
PINACLE	Molten metallic fuel relocation and heat transfer prior to failure
LEVITATE	Post-failure oxide and metallic fuel relocation with fuel-cladding heat transfer

LIFE-METAL

Relevant to fuel performance, LIFE-METAL predicts the behavior of metallic fuel rods under normal operating conditions in fast reactor environmental as a function of reactor operating history [48]. The code can perform steady-state and design basis transient analysis for the thermal, mechanical, and irradiation behavior. Axial variations in operating conditions are accounted for by using powers and fast fluxes for up to nine fuel axial nodes and one plenum node, although axial heat conduction and mechanical coupling between axial nodes are not considered. The main predictions of interest include changes in fuel length and fissile content due to burnup and breeding, fuel temperature and margins to fuel melting and margins to low melting eutectic formation with cladding, FCMI and FCCI, cladding deformation, and margins to cladding failure due to fission gas pressure loading. The code has been extensively used for experiments at EBR-II. An important function of the code was to provide initial conditions for transient fuel performance codes.

BISON

Metallic fuel development began in BISON a few years after light water reactor development began (~ 2009). The initial development was funded by the NTRD AFC. With recent interest from industry, the U.S. NRC, and the VTR program, the Nuclear Energy Advanced Modeling & Simulation (NEAMS) program has increased funding for metallic fuel development and validation in FY18. The material models with the most mature development are binary uranium-zirconium and ternary uranium-plutonium-zirconium alloys for fuel and HT9 for cladding. The first material models were incorporated from open literature sources, which include mechanical and thermal material properties that are functions of temperature, porosity, and zirconium concentration, swelling, fission gas release, zirconium diffusion, creep, and sodium coolant channel boundary conditions. These material models have been utilized to simulate ERB-II fuel pins, and the results compared to measurements. The EBR-II experiment measurements mostly consist of cladding strain and zirconium redistribution. As such, comparisons to EBR-II measurements are currently limited to these two figures of merit. Simulations and comparisons to experiment measurements have also been done for transient testing of EBR-II fuel pins in TREAT, which included temperature measurements. Preliminary comparisons between BISON calculations and experiment measurements are favorable. These comparisons have also highlighted the importance of the fuel swelling and fission gas release models on cladding strain. Also, BISON metallic fuel simulations are currently being run in support of ATR experiments and VTR exploratory design calculations. Further development and evaluation of individual material models and fuel system models are planned for FY19 [49].

RELAP5-3D

RELAP5-3D has been primarily developed to simulate thermal-hydraulic transients in reactor systems that use light water as the working fluid. However, the code has a generalized capability to simulate a wide range of working fluids other than light water, including various liquid metals, such as sodium and the eutectics of lead-bismuth and sodium-potassium, and liquid salts. RELAP5 series of codes, although extensively used and validated for thermal-hydraulic analysis of reactor cooled by light water, have not been validated as thoroughly for sodium. RELAP5-3D was evaluated for the analysis of a sodium-cooled fast reactor in 2006 [50]. The applicability evaluation consisted of several steps, including identifying the important transients and phenomena expected in a fast reactor, identifying the models and correlations that affect the code's calculation of the important phenomena, and evaluating the applicability of the important models and correlations for calculating the important phenomena expected. The result showed that the existing models were adequate or that relatively minor changes were required to improve the code's representation of the important phenomena. The effort identified improvements and additional models needed and evaluated the accuracy of the sodium properties used in the code.

SAM

The Systems Analysis Module (SAM) is being developed as a practical plant-level system analysis tool for sodium, lead or salt fast reactors that can be used with SAS4A and SASSYS-1 and CFD codes [51]. SAM uses flexible multi-physics integration, and it is built on Multiphysics Object Oriented Simulation Environment (MOOSE) framework and other software libraries. For core modeling options, it can perform single channel, multi-channel and flexible full core modeling, with hex-lattice core component built in for SFR core modeling.

5. EXPERIMENTAL INFRASTRUCTURE

Historically, both in-pile and out-of-pile experimental facilities were developed to cover the range of experimental conditions required by the spectra of off-normal/transient events. A 2011 report from the NEA concluded that the most important and top tier R&D need for SFR projects is fuel safety and severe accident issues due to knowledge gaps on new pin design and materials, followed by thermal-fluids and reactor physics, followed by sodium risks and structural integrity [31]. The same report provides a survey of available facilities worldwide. The findings highlight few available facilities for fuel safety and severe accident studies at a time when restart of the TREAT facility was still being considered. The FBTR in India, the JOYO reactor in Japan, and the IGR in Kazakhstan are the only available in-pile facilities, each with important limitations. TREAT restart is truly timely to provide a unique and vital resource for DOE and U.S. programs, but also holds great potential to serve international needs as well.

Since temperature and time are the primary drivers for fuel failure phenomena, historical capabilities were mapped according to temperature/time regimes to fill testing needs as shown in Figure 1. For reference, a range of representative transient cladding temperature responses for a variety of SFR transients are given in Figure 8. The figure is also overlaid with a plot of predicted failure thresholds of HT9 and D9 cladding materials. The figure well-illustrates the need for a transient test reactor (TREAT facility), out-of-pile furnaces, and a steady-state reactor such as EBR-II. The figure also illustrates the important overlap existing between different facilities that provide direct linkage between different facilities. This overlap provides opportunity to evaluate potential biases induced by the constraints of a particular facility.

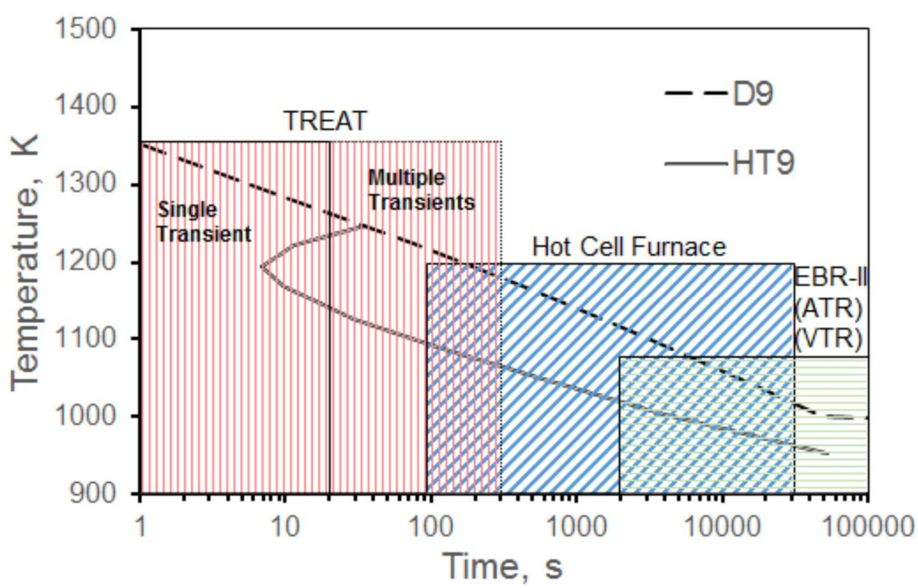

Figure 8. Test regimes for key capabilities needed for safety R&D on metallic fuels overlaid with predicted failure thresholds of cladding materials with 10 MPa plenum pressure. Adapted from [12].

5.1 In-Pile, Out-of-Pile, and Hot Cell Experimental Capabilities

While out-of-pile testing provides only uniform temperature conditions, the ability to cover wide ranges of temperature/time regimes shown in Figure 8 gives it a marked advantage. Some in-pile testing is necessary to provide spot checking of out-of-pile damage results. Out-of-pile testing will provide representative form, consequence, and detection of cladding breach. Tube-burst, eutectic penetration (i.e., FBTA), and furnace testing of irradiated fuel pins (i.e., WPF) are primary tools for addressing basic technical issues out-of-pile. Historically, in-pile tests at longer times and lower temperatures would be done in EBR-II, while higher temperature and shorter time ranges would be performed in the TREAT facility (M-series tests). References for these facilities were given previously.

The design of the TREAT facility limits the duration of any single experiment to tens of seconds at full power. As seen in Figure 8, the time-temperature range provided by TREAT is outside realistic temperature ranges of prototypic transients of approximately 800°C. Still, the failure behaviors achieved in prototypic thermal-hydraulic and neutronic environment at non-realistic peak temperatures proved to be of high value to model development and validation of detailed fuel failure, consequences, and detection. Program plans indicate interest and potential to design TREAT experiments that could be run multiple times to develop accumulated fuel damage to the failure point [12]. This testing concept was never implemented, yet, it was believed with very rapid power startup and shutdown between runs, the resulting fuel behavior would approximate continuous behavior. This “multiple transient” regime of testing is also shown in Figure 8. The lack of a longer time, lower temperature in-pile facility make this type of “out-of-the-box” thinking more relevant to current experiment development.

The out-of-pile facilities, including the FBTA and the WPF, that were used historically for testing metallic fuels are no longer available. At INL, several facilities are available to begin detailed studies of off-normal behaviors. Further evaluation of existing facilities to fill this recognized gap is still required. Preliminary investigation of existing facilities show high potential for existing furnaces and diagnostics equipment in the Hot Fuels Examination Facility (HFEF) to provide detailed studies of whole irradiated pins or segments. Advanced characterization equipment is available or actively being installed in the Irradiated Materials Characterization Laboratory (IMCL) including Scanning Electron Microscopy (SEM), Tunneling Electron Microscopy (TEM), Focused Ion Beam (FIB), Electron Probe Microanalysis (EPMA), thermal conductivity measurement systems, Thermogravimetric Analysis/Differential Scanning Calorimetry (TGA/DSC), and more. The latter toolset represents unique opportunities to study irradiated materials using state-of-the-art approaches, enabling detailed understanding of fuel microstructure and key behaviors such as FCCI/eutectics, thermal transport, and fission product transport. A recognized gap includes the addition of a Mass Spectrometry (MS) system to the TGA/DSC system that will provide important information regarding fission product release behaviors. Separate plans will detail strategies to utilize these capabilities to support the fuel safety R&D needs.

Evaluations of needed in-pile infrastructure have already been underway [52] to design crosscutting experimental infrastructure supporting liquid metal cooled reactor R&D. A detailed overview of the evolution of SFR experimental devices used in TREAT is provided in [30][52]. The conceptual designs of current TREAT devices are summarized in Figure 9. A modular test design will support simple adaptations for separate effects studies. This device is intended to provide opportunity for low radiological activation of hardware to facility post-irradiation handling. These designs also provide options for temperature control and flexible instrumentation strategies. The heat-sink capsule design is meant to serve as a SET to semi-IET platform. The device can provide near prototypic heat rejection from the test specimen under a range of conditions that could possibly be enhanced with active heat removal system. This device is in conceptual form and a design variant for LOF studies is planned to be evaluated.

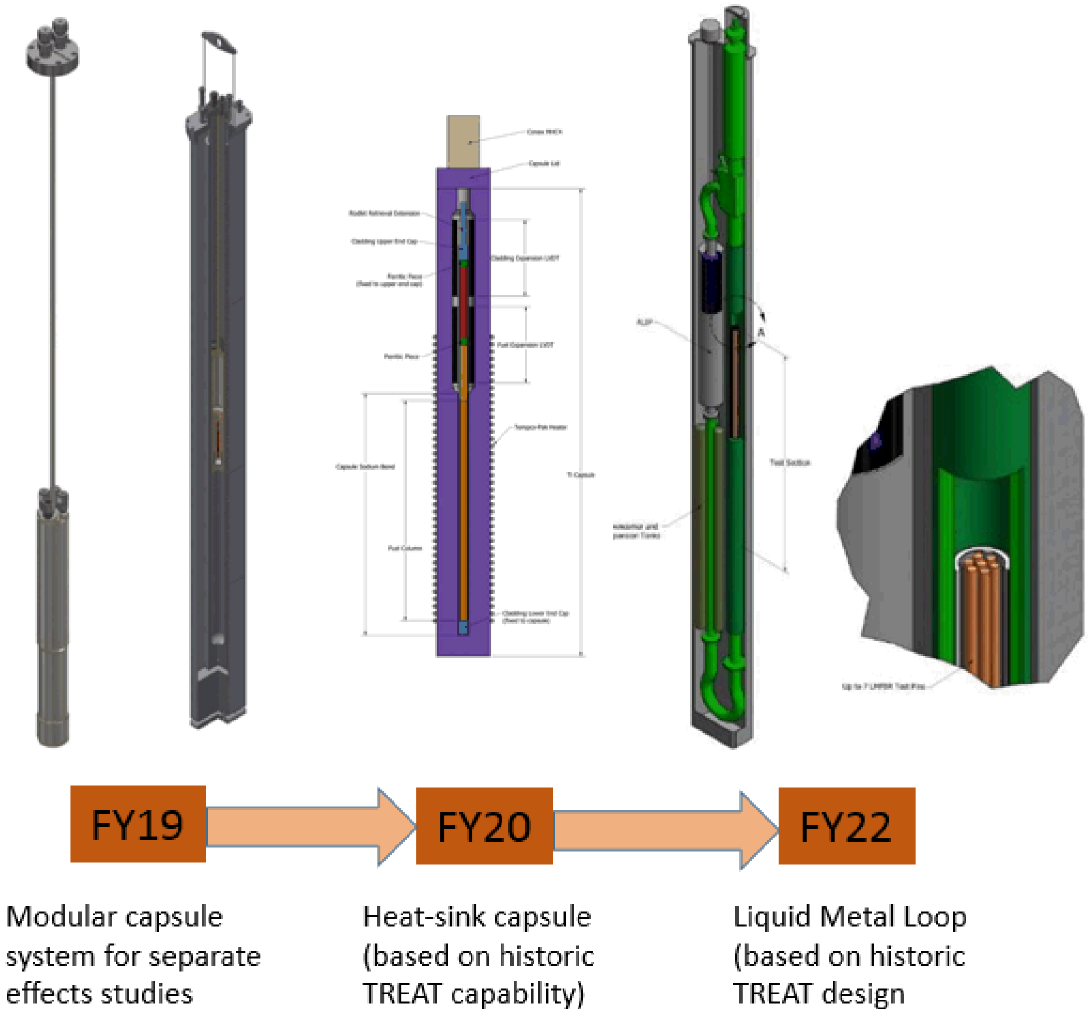

Figure 9. In-pile experimental platforms needed to support fuel safety R&D for metallic fuels with planned development schedule.

The current liquid metal loop concept is based on mature Na loops designs that were the foundation to efficient success of historic TREAT reactor experiments. While the historic Mk-III Na loop demonstrated great performance [53], a design update is planned to ensure technical requirements of modern testing needs may be achieved. In fact, near the end of the IFR program, R&D plans show recognized needs beyond the M-Series to adapt the Mk-III design to accommodate longer fuel pins from FFTF and small fuel bundles, perform LOF tests, and provide greater pump head to drive representative post-failure response [35]. These requirements will be developed as part of design efforts. A source for the compact Annular Linear Induction Pump (ALIP) used in the Mk-series TREAT loops still needs to be identified. In all cases, integration of these devices with hot-cell facilities is a high priority to provide access to the library of irradiated materials for experiment assembly and disassembly for PIE.

Without EBR-II capabilities and until VTR is available, associated testing gaps will rely on the ATR and possibilities to extend the functional range of the TREAT facility, along with similar hot cell furnace approaches of the IFR program. Further evaluation of available options and needs is required.

5.2 In-Pile Instrumentation

Modern in-pile instrumentation development holds great potential to provide access to data streams that were not historically available. A strategic plan for instrumentation development for in-pile transient testing is found in [54]. The design of the TREAT facility provides great flexibility for instrumentation due to low fluence in experiments, a unique attribute that provides great value. Key measurement parameters of interest in metallic fuel transient response include: peak fuel temperature, peak cladding temperature, peak cladding strain, fuel deformation, melting, and relocation, radionuclide evolution and release. Historic instrumentation strategies included advanced systems to measure fuel pin deformation, melting, and relocation. These systems included high speed video devices in separate effects experiments and the novel hodoscope design [57]. The latter has been refurbished and is now operational to support all testing approaches with the unique ability to monitor fuel mass location real time through opaque structures. Instrumentation included in historic Na loop experiments included thermocouples welded to flow tube structures, pressure transducers connected to well characterized taps to remove them further from the active core, and magnetic flow meters.

- Along with fuel deformation and motion, system temperatures are the other key measurement parameters for evaluating performance of metallic fuels. Excellent thermal transport provided by metallic coolants provide reasonable uncertainty in fuel temperatures from thermocouples placed on flow tube exterior walls. Opportunities exist to provide spatially focused temperature measurements in metallic fuel experiments. Measurement of fuel meat and or cladding was not done historically and could help reduce uncertainties in data that directly impact fuel performance operational margins.
- Fuel deformation and movement is provided by the TREAT hodoscope, but opportunities for improved characterization of irradiated fuel expansion effects could provide important data for M&S. Quantification of the resolution of detectable material mass from the hodoscope device is an important target for leveraging its advantageous measurement configuration to support a wide range of experiments.
- During the M-Series tests, a prototype online fission product monitoring system was tested in TREAT. The device was based on gamma spectrometry in the void space above the core reflector, monitoring the plenum region of the loops. The results were not satisfactory due to high gamma interference encountered during the transient. This effort was a component of the late plan to better capitalize on transient experiments to supply data relevant to source term studies. This recognized gap in the late program should not be overlooked today.
- Experiment instrumentation also played an important role in executing the objectives of experiments. During the last decade of operations, the reactor control system allowed feedback provided by experimental measurements to trigger transient control rod ejection or scram. The former feedback approach was used to initiate highly controlled prototypic combined LOF-TOP conditions simulating positive void reactivity insertion with the control rod ejection when voiding was detected in the sodium coolant. This condition has more relevance to oxide fuels due to its higher stored thermal energy and temperatures. Experimental coupled scram capability is another unique capability that is important to testing objectives. Frequently, the goal was to halt a transient at a desired point in fuel damage evolution (pre-, at-, post-failure) to preserve (freeze) the final state for post-irradiation examination.

Currently, a remanufacturing system that allows for instrumentation installation is being developed to allow resizing irradiated fuel pins, inserting instruments into fuel and on the pin, and resealing the rod with end caps. This approach has never been attempted before on metallic fuel experiments but may provide important opportunities for unique experiment designs and data collection. With oxide fuels, this process has been well matured through sophisticated means to ensure little impact on fuel behavior through an intrusive process. For metallic fuel, the process would need to be adapted and qualified. The potential offered by this capability is worthwhile to provide technical requirements to the ongoing refabrication system design.

6. CONCLUSIONS

With the restart and resumption of operations for the TREAT facility, an additional and important tool has been added to the set of testing facilities for nuclear fuel. In particular, this one-of-a-kind facility will not only fill an important void for domestic testing programs, but will also allow our international partners to resume fuel limit testing for their advanced reactor programs. In addition to transient testing capabilities, the INL is well positioned to provide the full suite of capabilities for fuel testing – from fabrication, to irradiation (both steady-state and transient), to PIE using modern instrumentation, to modern M&S integration.

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Appendix A

Historical Fuel Safety Program and Gaps

Historically, metal fuels found successful use and experience of more than 13,000 driver fuel rods with a 10 at.% burnup limit in fuel designs from initial U-5Fs driver fuel in EBR-II with later successful conversion of the core to U-Zr fuel. The addition of Pu to form the ternary U-Pu-Zr alloy did not change the mechanisms that control fuel element lifetime. Although a limited number of rods have been evaluated, the work performed was believed to be sufficient to determine the behavior of life-limiting and safety-related phenomena. In general, these phenomena are found to be the same as those for the U-Fs and U-Zr fuel, which has substantial reliability, a significant database, and experience. A detailed overview of the metallic fuel experience is provided in [3].

Briefly, the historical database reveals interesting characteristics for metallic and oxide SFR fuels. Several papers highlight transient performance findings [3][17][18][19][20]. Both fuels have demonstrated robust performance with overpower to failure on the order of four times nominal Linear Heat Generation Rate (LHGR). The difference in thermal conductivity between the two fuel forms is the key to understanding the main differences in transient performance [9][11]. Metal fuel has thermal conductivity of approximately 2-10 times that of oxide fuel over the range of operating conditions. The following are key observations of the performance of metal and oxide fuels.

- Metal Fuel
  - Fuel-cladding interface temperature follows coolant axial temperature distribution.
  - Failure occurs near the top of the fuel column coinciding with peak coolant temperature.
  - Little Fuel-Cladding Mechanical Interaction (FCMI) occurs, failure is by stress rupture of thinned cladding due to Fuel-Cladding Chemical Interaction (FCCI) driven by internal pin gas pressure.
  - Fuel damage temperatures may occur below the coolant saturation point. As a bit of side note, it is interesting to consider that stronger cladding may not provide better safety characteristics, i.e., it may be beneficial to fail prior to reaching coolant saturation temperature in the fuel.
  - The fuel is chemically nonreactive with the metal coolant.
  - The stored thermal energy in the fuel is low.
- Oxide Fuel
  - Failures from fuel column top to core midplane with slow to fast power ramps.
  - Failure occurs by plastic strain in cladding due to FCMI, internal pin gas pressure, and cladding melting and ballooning.
  - Fuel temperature is high relative to the coolant.
  - The fuel has a benign chemical reaction with coolant.
  - The stored thermal energy in the fuel is high.

Historically, both in-pile and out-of-pile experimental facilities were developed to cover the range of experimental conditions required by the spectra of off-normal/transient events.

- Several out-of-pile test facilities were established to support transient testing. Separate effects facilities included the Fuel Cladding Transient Tester (FCTT) for stress rupture behavior of oxide fuel claddings [21] and the Fuel Behavior Test Apparatus (FBTA) for eutectic penetration of cladding due to fuel-cladding chemical interactions in metallic fuels [22]. To understand the correlated roles of fission-gas pressure and fuel-cladding chemical interactions, the Whole-Pin Furnace (WPF) system was developed and installed in the Alpha-Gamma Hot Cell Facility at ANL [23][24]. The WPF allowed for testing heating rates up to 30°C/s and peak temperatures of 1100°C for time durations up to several days. This facility has since been fully decommissioned, but some hot cell furnace systems are available at INL. Many other such out-of-pile facilities existed in historical programs including thermal hydraulic facilities such as the CAMEL facility at ANL used to investigate fuel-coolant interaction behaviors [25].

- Operational transient testing was performed in EBR-II at the former ANL-West site in Idaho, present-day INL Materials Fuels Complex (MFC), in addition to extensive steady-state irradiation testing [3][26]. Transient testing programs included the ORT program for operational transient testing of oxide fuel using the metal driver fuel of EBR-II [27] and the Shutdown Heat Removal Tests (SHRT), which successfully demonstrated the ability of the metallic-fueled fast reactor to survive unprotected ULOF and ULOHS without core damage [28] as whole reactor tests. The Run Beyond Cladding Breach (RBCB) tests demonstrated benign impacts of failed cladding. In the overpower transients, test ramp rates covered ranges of 0.1 to 10% $\Delta P/P_0$ per second up to 100% overpower. Peak temperature achieved in these tests was 890 °C. The upper level of transient rates available in EBR-II overlapped the lower bound of possible ramp rates in TREAT.
- The Sodium Loop Safety Facility (SLSF) in the Engineering Test Reactor (ETR) at INL was designed to address whole core accidents related to reactor design, licensing, and operation. The ETR provided prototypic fuel conditioning and operation for up to 35 days. The SLSF provided prototypic thermal-hydraulic conditions for up to 37-pin test configurations. A total of seven experiments were performed in SLSF, six of which had the purpose of whole core accident simulation. The final experiment investigated a local fault event of propagation dynamics accompanying molten fuel release into the coolant [29]. A similar testing facility was designed to a significantly mature level for the TREAT facility as part the TREAT upgrade project but was never completed.
- For intense, shorter duration transients, the TREAT facility was used to perform transients covering a wide range of events from mild transients to hypothetical core-disruptive accidents. A detailed review of the evolution of experimental infrastructure is given in [30]. A brief description of the TREAT experiments on metallic fuels follows.

Historically, EBR-II played a crucial role in transient evaluation of metallic fuels. It was used to assess the effect that the transient would have on its driver fuel (Mark-II design) lifetime and to qualify the driver fuel for transient operation. Out of 56 low-ramp-rate transients (1.6% power increase per second and 13 high-ramp-rate tests (4 MWt per second), no discernable damage was found. The results were extended to U-Zr (Mark-IIIA) driver fuel, which experienced EBR-II transient-overpower tests during and after the core conversion. The ULOF, SHRT, and ULOHS tests in EBR-II demonstrated its ability to withstand those transients without core damage. A high temperature (up to 800°C) irradiation test of an assembly in EBR-II showed that, in post-irradiation examination of the fuel rods, molten phase attacked the cladding but the attack was not sufficient to induce cladding breach. Subsequent irradiation testing of the fuel rods that endured the transient did show cladding rupture caused by stress beyond the end of life of 8 at% design. The absence of a domestic test SFR is a recognized gap in modern U.S. R&D. Thus, the VTR is planned to fill that gap.

SFR Mission of the TREAT Facility

In 2010, the OECD NEA performed an assessment of SFR experimental facilities finding that in-pile safety and severe accident facilities are sparse in the international community [31]. At the time, the TREAT facility was recognized as having important potential, yet the future of the facility was still undetermined. Resumption of operations at the TREAT facility fills an international gap in capability for transient testing SFR technology.

TREAT has a rich history of experimentation describing off-normal reactor conditions. The unique flexibility of the facility has allowed for supporting a wide variety of experiment goals. Over the 35 years of operation, more than 900 experiments were performed targeting a wide range of technical issues. Primary issues addressed by a variety of experimental programs included phenomenology of fuels and materials, transient performance, operational safety, accident consequences, and validation of analytical and computational models. The primary mission of the TREAT facility comprising most of its operation was supporting SFR programs as the principal in-pile test facility in the U.S. In this mission, TREAT experiments supported several reactor programs including EBR-II, Fermi I, the FFTF, CRBR, the Southwest Experimental Fast Oxide Reactor (SEFOR), the IFR as well as the Power Burst Facility (PBF) and other water reactor and space power programs.

Over the course of more than 30 years, TREAT performed an expansive array of SFR fuel experiments following the design evolutions of both metallic, oxide, and some carbide fuels. In-pile reactor safety testing studies on oxide fuel have been carried out in various test reactors around the world, including extensive studies performed in TREAT. For metallic fuel, in-pile studies to failure have only been performed in TREAT. Summaries of the last decade of experiments performed in TREAT, most relevant to modern fuel systems, are found in [32][3]. A description of the evolution of TREAT fuel experiments since its beginnings is found in [33][30].

For metallic fuels, TREAT experiment results on 15 pins showed that failure thresholds around four times nominal power under relatively fast transient-overpower conditions (~8s period) [18]. Comparatively, hundreds of MOX fuel rods were tested in TREAT. It was observed that MOX fuel rods tested at faster ramp rates exhibited higher-breach margins, with breach just above the core midplane, where fuel temperatures would be higher. Slower ramp tests induced breach near the top of the fuel column, where cladding temperatures would be higher. The reason for the behaviors was thought to be fuel melting leading to cladding penetration in faster ramp tests, whereas the slower ramp allows cladding temperatures to increase such that the weakened cladding is more likely to breach by stress rupture. A more detailed review of the metallic fuel experiments follows below.

Historical Testing, Plans, and Potential Gaps

Transient testing of metallic fuels was performed extensively on early metal fuels designs until the U.S. fast breeder reactor programs (and most of the international nuclear community) shifted to concentrate on oxide fuels; however, metal fuels continued to be used as the EBR-II driver fuel. By the early to mid-1980’s, with the large safety tests performed in EBR-II demonstrating inherent passive safety (and with the events at Chernobyl and Three Mile Island), the U.S. program again shifted back to focus entirely on metallic fuels due to the demonstrated advantages in safety. Modern metallic fuel tests performed at the TREAT facility were termed the M-series experiments. A detailed review of the early metallic fuels experiments including at the TREAT facility, is found in [17].

After the initial M-series (M2-M7) experiments were completed in the mid-1980’s, significant R&D progress was made through the end of the IFR program in relation to maturing the metal fuel technology. Many advancements were achieved in fuel fabrication, material science understanding, and fuel performance during normal and off-normal conditions through extensive work in developing M&S tools complemented by ongoing experiment results from EBR-II and out-of-pile furnace testing (and using M-series for model development and validation). During the last years of the IFR program, the completed M-series tests proved invaluable and remain experimental highlights of the TREAT facility legacy. Results from the FBTA and particularly from the WPF experiments provided experimental basis for validated LOF models. Through the last years of the program, a continuation of the M-series tests was planned to investigate specific experimental knowledge gaps, including LOF behavior. Noted in historical documents, the close relationship among all TREAT experiments was one of the most valuable aspects of the testing [8]. This allowed distinguishing individual effects in a parametric manner.

The TREAT experiments M2 through M7 were focused on TOP behaviors [18]. Tests M1-M4 were performed to study cladding failure mechanisms margins to cladding failure, and prefailure fuel motion during TOP events. These tests were performed on U-5Fs fuel due to the unavailability of targeted fuel designs for the initial tests. Tests M5-M7 were tests of IFR program fuel designs in conditions similar to M2-M4. The M1 experiment was designed as a SET experiment using a laser-illuminated segment of irradiated metallic fuel imaged by a high-speed video system to investigate fuel expansion behavior (so called "fuel foaming effects"). This experiment largely proved unsuccessful due to gaseous release from the specimen that impeded visual. The concept of this test is still of interest today. The following six "integral" experiments were designed to study the behavior of fuel and cladding near the cladding-failure threshold, for a range of burnups and for several fuel pin compositions.

The first six M-series experiments were designed to use the slowest transient possible, starting from nominal SFR operating conditions (LHGR 40 kW/m, inlet temperature of 630°K, and an axial temperature rise across the pin of 150 ΔK). In 6 experiments, a total of fifteen pins were tested with 5 reaching cladding breach, in all cases targeting overpower levels of approximately four times nominal. In all M-series experiments, the pins were housed in individual flow shrouds to increase to create "independent" pin test configurations. Though many oxide tests were performed in similar Na loops devices in small bundles ranging from 3-7 pins, no metal fuel tests in bundle configuration were ever performed. Table A1 shows a table summarizing the results of these tests [18][3]. Cladding failure occurred when fuel-cladding temperature reached approximately 1080°C, where a sharp increase of fuel-cladding chemical interaction and eutectic liquefaction occurs.

The primary objective of the TREAT tests was evaluation of fuel failure behavior with test termination happening just prior to calculated failure thresholds or in response to flow fluctuations measurements made in the test train upon cladding failure, triggering scram in the experiment to "freeze" the state of the fuel. The latter approach is a unique capability allowed by the TREAT facility to control reactor through experimental feedback. The overpower conditions of these tests were much more severe than predicted IFR transients or any modern metal fuel SFR designs. The TREAT facility design limits transients to several tens of seconds, thus limiting the overpower conditions to achieve failure (thus, the 8 second ramp of M2-M7). Still, the value of these experiments was found in the high level "prototypicality," or maybe better described as "integrality," in these tests coupled with excellent instrumentation (including the hodoscope) to diagnose, understand, validate, and measure important failure phenomena [12]. These in-pile results coupled with out-of-pile furnace testing are designed and understood through the lens of theoretical and computational M&S.

Table A-1. Summary of TREAT M-Series experiments including specimen, overpower, and results. Values in parentheses are from first of two individual overpower transients performed (M5).

Experiment	**Fuel/Cladding**	**Fuel Design (EBR-II)**	**Burnup, at%**	**Test Overpower , * indicates cladding failure**	**Calculated Breach Threshold Overpower (Normalized)**	**Maximum Fuel Axial Expansion, %**	**Maximum Pin Pressure, MPa**
M2	U-5Fs/316SS	Mark-II	0.3	4.1	4.7	16	0.6-0.8
	U-5Fs/316SS	Mark-II	4.4	4.2*	4.5	-	7-9
	U-5Fs/316SS	Mark-II	7.9	4.1*	3.6-4.0	3	17-20
M3	U-5Fs/316SS	Mark-II	0.3	4.1	4.8	18	0.6-0.8
	U-5Fs/316SS	Mark-II	4.4	4.0	4.4	4	7-9
	U-5Fs/316SS	Mark-II	7.9	3.4	3.6-4.0	4	17-23
M4	U-5Fs/316SS	Mark-II	0.0	3.8	4.3	4	0.6
	U-5Fs/316SS	Mark-II	2.4	4.1*	4.4	7	2-6
	U-5Fs/316SS	Mark-II	4.4	3.8	4.3	4	7-9

Table A-1. (continued).

Experiment	Fuel/Cladding	Fuel Design (EBR-II)	Burnup, at%	Test Overpower, * indicates cladding failure	Calculated Breach Threshold Overpower (Normalized)	Maximum Fuel Axial Expansion, %	Maximum Pin Pressure, MPa
M5	U-19Pu-10Zr/D9	X419, X420, X421	0.8	4.3(3.4)	5.1(4.6)	1(1)	1(1)
	U-19Pu-10Zr/D9	X419, X420, X421	1.9	4.3(3.4)	5.1(4.6)	2(0.5)	3(3)
M6	U-19Pu-10Zr/D9	X419, X420, X421	1.9	4.4	4.6	2-3	3
	U-19Pu-10Zr/D9	X419, X420, X421	5.3	4.4*	4.5	3	10
M7	U-19Pu-10Zr/D9	X419, X420, X421	9.8	4.0*	4.4	3	19
	U-10Zr/HT9	X425	2.9	4.8	4.4	2-4	6

As was introduced earlier, the M-series tests, hot cell experiments and M&S efforts of the IFR program set the stage for follow-on experiments. Several years after M7, preparations were well underway for the M8 experiment with modifications to the TREAT facility and a significant TREAT facility core reconfiguration to provide greater energy capacity for M8. The M8 experiment was never completed due to increasing budget and political pressures pushing the decommissioning of EBR-II and placing TREAT into “operational standby.” An extensive calibration campaign comprising 17 irradiations was performed to develop a detailed understanding of the new reactor core configuration and provide crucial quantification of fuel power in the TREAT facility.

The M8 experiment was planned to investigate cladding failure under conditions in which cladding creep was expected to play a more dominant role [34]. Several characteristics would drive this targeted behavior:

- High pin plenum pressure corresponding to EBR-II Mark V fuel at ~15 at% burnup,
- Slower power transient compared to previous tests utilizing increased energy available in the “half-slotted” TREAT facility core (“full-slotted” in earlier tests).

The M8 experiment would provide a rapid rise to an LHGR of near 100 kW/m, just short of onset of fuel melting, followed by a power ramp over ~30 s period. The expected goal would provide cladding failure well before the onset of rapid cladding penetration by melt formation, as in the first six M-series tests. Other than these differences, the M8 tests were strategically planned to replicate the previous experiments as much as possible. The fuel pins planned for M8 were EBR-II Mark-V design with U-19Pu-10Zr in HT-9 cladding with 12 at% burnup. Two pins would be irradiated in individual flow tubes with one slightly overcooled. Thus, one pin would be planned to fail while the other would remain intact, just prior to failure. The test train was designed for use in a TREAT Mk-III Na loop.

Beyond M8, test M9 was designed to begin addressing questions for LOF conditions to investigate in-pin fuel expansion, axial location and relative timing of cladding breach, quantity of fuel ejection from the pins, and in-channel fuel dispersal [34][35]. In particular, the interest was to investigate the relationship of fuel melting, axial expansion, and reactivity feedback in a high-void-worth core. Initial analysis of the M9 test using the SAS4 computer code showed that such these tests were feasible, but it would require an accelerated sequence of events relative to predicted reactor event progression. The main uncertainties in analysis of such events that would be addressed would include the molten fuel fraction before failure, failure time to the onset of fuel motion at the fuel top, and the cladding conditions required to cause failure at this high heating rate. Experimental hardware from the previous M-series still needed evaluation of suitability for these objectives.

Though M8 was thought to be the final test on EBR-II length fuel, the investigation of LOF behavior would require several tests beyond M9. Driven by SAS4A validation priorities, distinct pin geometries were still desired to be tested including longer pins (from FFTF and more relevant to modern designs) and three or seven pin bundle geometries [34][35].

Furthermore, the preapplication safety evaluation report performed by the U.S. NRC for the GE PRISM and Rockwell International Sodium Advanced Fast Reactor (SAFR) designs in 1994 provides important insights to technology gaps needed for licensing [36][37]. A significant review of the GE PRISM design describes planned transient testing that would be performed in addition to the established database. This transient testing included:

- Cladding failure mechanisms and margins to failure as a continuation of the M2-M7 experiments to obtain data specific to the PRISM design including: reprocessed fuel and/or higher Pu (26 wt%), longer fuel, and higher burnup, to show insensitivity of performance from the existing database.
- To address fuel disruption and post-failure dynamics in pin-bundle geometry for LOF and LOHS ATWS events.
- To investigate the release and transport of fission products during hypothetical fuel disruption sequences with two general classes of transport mechanisms: 1) by sodium flow through the above-core sodium pool, and 2) fission gas-driven transport in large gas bubbles. Two tests using seven-pin bundles were planned to simulate TOP and LOF conditions, respectively.

Since the IFR program and with decades of no available transient testing capability, R&D testing needs have evolved but likely still include many of the objectives left behind by the IFR program. For two decades, metallic fuel researchers and SFR analysts had little experimental capabilities to investigate transient fuel behaviors. The restart of the TREAT facility represents an opportunity to gain new insights, provide answers and validation to long standing questions, and explore new territory altogether. In 2011, a high level PIRT study and gaps analysis was carried out by leading metallic fuel experts [4]. The final statement of the study is an expression of concern for extending the technology database without the availability of test facilities such as EBR-II, FFTF, and TREAT. This statement is indicative that any current or new SFR designs would be limited to the existing historical database. It should be noted that even the physical state (i.e., quality assurance concerns) of the historical database to provide licensing input was a primary concern. Specific technology gaps identified as high priority would require a transient testing facility for the following:

- Extending the burnup limit from 10-12 at% (pins are available at INL from EBR-II and FFTF at INL that could be used to study up to near 20 at% burnup),
- Investigating advanced fuel designs such as those with no Na bond, lower fuel smear densities, and advanced claddings,
- Transient response of recycled fuels,
- Transient response of high-content minor actinide bearing fuels,
- Investigating transient response of FCCI preventative measures such as fuel additives or cladding liners/coatings.

The now operational TREAT facility provides the opportunity to address these potential developments. Recently, the Gateway for Accelerated Innovation in Nuclear (GAIN) program hosted a workshop on fuel safety with representatives from research and industry (May 2017). The conclusions from the meeting were similar to those described here: metallic fuel is well established within the existing database, but opportunities exist to expand the database for new fuel forms and to provide more experiments for M&S development and validation.