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GAIN Fuel Safety Research Workshop

Summary Report

September 2017

GAIN
Gateway for Accelerated Innovation in Nuclear

DISCLAIMER

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Acronyms

ANL	Argonne National Laboratory
ART	Advanced Reactor Technologies
AT	Applied Technology
ATR	Advanced Test Reactor, INL
BR2	Belgium Reactor 2
BWR	boiling water reactor
DOE	U.S. Department of Energy
EBR	Experimental Breeder Reactor (I and II, INL)
EFF	Experimental Fuels Facility, INL
EPRI	Electric Power Research Institute
FR	Fast Reactor
GAIN	Gateway for Accelerated Innovation in Nuclear
HFEF	Hot Fuel Examination Facility, INL
HTGR	High Temperature Gas Reactor
HTR	High Temperature Reactor
IMCL	Irradiated Materials Characterization Laboratory, INL
INL	Idaho National Laboratory
IRSN	Institute for Radiological Protection and Nuclear Safety - France
JAEA	Japan Atomic Energy Agency
LEU	low enriched uranium
LWR	light water reactor
MFC	Materials and Fuels Complex, INL
MSR	Molten Salt Reactor
NRC	Nuclear Regulatory Commission
NTD	National Technical Director
ORNL	Oak Ridge National Laboratory
PWR	pressurized water reactor
RIA	reactivity insertion accident
SCK-CEN	Belgian Nuclear Research Center
TREXR	TREAT Experiment Relational database
TWG	Technology Working Group (Advanced Reactors)
U.S.	United States

GAIN Fuel Safety Research Workshop

INL Energy Innovation Laboratory (EIL)
Idaho Falls, Idaho; May 1-4, 2017

The Gateway for Accelerated Innovation in Nuclear (GAIN) Fuel Safety Research Workshop was held in Idaho Falls, Idaho, at the Idaho National Laboratory (INL) meeting center in the Energy Innovation Laboratory (EIL). The purpose of the workshop was to gather fuel safety research needs from the broad range of industrial nuclear technology users and developers. The needs will be analyzed relative to the national laboratory capabilities to help inform program planning and funding. The workshop provided a forum to learn about transient testing from a research and regulatory perspective and tour participation of the Transient Reactor Test Facility (TREAT), Advanced Test Reactor (ATR), and Materials and Fuels Complex (MFC).

Prior to the workshop, the invitees received two surveys to submit information on fuel safety testing needs for light water reactor (LWR) and advanced reactor technologies. The data was provided to the Advanced Reactor Technology Working Group (TWG) chairs for molten salt, high temperature gas, and fast reactors and to the Electric Power Research Institute (EPRI) as the LWR stakeholder. The Advanced Reactor TWG summarized and presented the results during the workshop.

One of the many highlights of the workshop was the INL site tour on May 1, 2017. Thirty participants toured ATR, had lunch at the Experimental Breeder Reactor-I (EBR-I), learned about start up activities at TREAT, and spent the rest of the afternoon at MFC. The focus on fuels and materials research formed the tour, which included the Hot Fuel Examination Facility (HFEF), Irradiated Materials Characterization Laboratory (IMCL), and the Experimental Fuels Facility (EFF).

The workshop began on May 2, with a background presentation on transient testing capabilities in the United States (U.S.), presentations and discussions on LWR fuel safety research needs, and industry perspectives. A poster session, coordinated with lunch, featured various aspects of fuel safety studies and capabilities. This provided the participants a chance to connect with national laboratory experts in fuel safety research.

On Wednesday, May 3, the topic shifted to advanced reactor fuel safety research needs. Along with the TWG presentations summarizing the survey research needs, the Department of Energy (DOE) National Technical Directors (NTD) for molten salt reactors (MSR), fast reactors (FR), and high temperature gas reactors (HTGR) discussed the gaps between the needs and DOE capability.

Thursday, May 4, was filled with international presentations on fuel safety research and testing capability from the Halden Reactor Project, the CABRI International Program, Japan’s Nuclear Safety Research Reactor (NSRR), and the Belgium Reactor 2 (BR2).

The workshop provided connections between reactor technology developers, existing nuclear industry vendors, the Nuclear Regulatory Commission (NRC), DOE, and national laboratory scientists and engineers. The LWR and Advanced Reactor Survey Results are available in Appendices A and B, respectively. The Fuel Safety Research Testing Capability Matrix is in Appendix C. Appendix D contains the agenda, attendees, and poster session information. Presentations from the meeting are available on the GAIN website (gain.inl.gov).

Fuel Safety Research Impact on Advanced Reactor Technology Development

Dan Wachs, INL, presented an overview of current and historic fuel safety research activities conducted within the DOE complex with emphasis on preparations for transient testing at TREAT. The TREAT facility is a versatile irradiation test facility able to subject experimental specimens to various transient nuclear conditions of interest for fuel and reactor safety research. TREAT resides roughly one mile west of MFC. The reactor first achieved criticality in 1959 and operated successfully until 1994 when its operations were suspended following cancelation of the Integral Fast Reactor Program. During this operational period, the facility underwent various modifications and upgrades that maintained it in a 'state of the art' condition for more than 30 years.

With the re-emergence of interest in advanced reactors, DOE's Office of Nuclear Energy (DOE-NE) identified the need to reestablish transient testing capability in the U.S. to support deployment of these technologies. Resumption of operations at the TREAT facility was identified as the preferred option for meeting this need. The Resumption of Transient Testing Program is nearly complete and will culminate in reactivation of the TREAT facility in late 2017 (over one year ahead of the commitment date) and is expected to operate in support of fuel safety research for at least another 40 years. However, execution of transient tests that support fuel safety research requires simultaneous recovery and/or development of supporting scientific capabilities that either connect TREAT with other key facilities or supports the collection of critical test data from the experiments.

While TREAT provides the foundational capability to support a wide variety of mission types related to nuclear security and physical science, the primary mission focus is on the support of nuclear fuel technology to enable nuclear energy applications. This mission set includes:

- **Enabling the deployment of advanced reactor systems** by providing the means to identify and quantify fuel safety criteria for use in design, licensing, and regulation of these new reactor technologies. In most cases, TREAT will be the only transient testing facility in the world capable of supporting advanced reactor concepts (e.g., non-LWR system). The remaining transient testing facilities around the world are primarily suitable for conducting a limited range of tests to support LWR fuel systems. The unique attributes of the TREAT design enables reactor technology specific test devices to be inexpensively prepared, used, and replaced with alternate devices in a timeframe on the order of several weeks. In contrast, the CABRI facility in France recently took many years to convert from a semi-permanent Na test loop to a pressurized water test loop. Test devices to support reactor technologies based on cooling system using liquid metal (Na, Pb-Bi, or Pb), high temperature gas, pressurized water, or molten salt have all been conceptualized, and in most cases, implemented during TREAT's history. This versatility is critical to meeting the DOE's advanced reactor technology development objectives.

- **Optimization of LWR fuel technology** that is necessary to ensure continued economically competitive use of the existing fleet of commercial nuclear reactors. These reactors must be operated within the boundaries defined by existing fuel safety criteria that are based on a foundation of experimental data developed decades ago. This data is scarce and in some cases not prototypic of current nuclear fuel designs. Due to the empirical nature of tests conducted during the 1960–1980s, performance was characterized macroscopically and test data does not lend itself well to modern modeling and simulation techniques. Operating and regulatory margins may be significantly improved by modern experiments that better quantify the response of modern fuel systems to off-normal conditions. In addition, very little historic testing data extends into the very high burnup regime (>~65 GWD/MT). The lack of data significantly constrains the potential to extend the life of fuel to reduce both fuel consumption and disposal costs. It is also possible that entirely new fuel designs could be deployed to further enhance LWR performance. The development of accident tolerant fuel designs is a prime example of this type of technology. Capabilities being established at TREAT are required to develop the performance database required to design, license, and regulate this technology.

- **Fuel behavior science** studies that are required to mature and deploy the modern modeling and simulation based nuclear fuel technology development model will be greatly enhanced by the unique capabilities of TREAT. Fundamental understanding of nuclear fuels and materials behavior has long been impeded by the complexity and interdependence of phenomena that occur under irradiation. TREAT offers researchers both unprecedented access to the test sample during irradiation and remarkable control of the sample environment. The open layout of TREAT allows users to integrate novel instrumentation into experiments that will provide the opportunity to collect real-time, in-situ fuel behavior data. TREAT's ability to deliver a nearly infinite array of shaped nuclear transients to a sample located in an independent test vehicle (that is decoupled from the driver core) gives the user the ability to stimulate sample behavior in a highly controlled fashion. This ultimately allows researchers to meticulously map the material response to nuclear environments in ways that may uncover fundamental physical properties. When this combined understanding is subsequently integrated together it will provide the knowledge required to predict and design entirely new technologies.

Development of Transient Testing Experimental Capabilities

Transient testing in TREAT is envisioned to enable a wide range of fuel safety research activities that are essential to the development, design, and deployment of advanced nuclear fuel and reactor technologies. The extent to which TREAT can support this mission is driven by the availability of enabling technologies and techniques that define the envelope of experimental capability. These enabling technologies can be loosely expressed as three pillars as shown in Figure 1. The first pillar is defined by the range of nuclear transients the reactor can deliver to the test samples. The second pillar is defined by the range of the sample environments that the test sample can interact with during the transient. The third pillar is defined by the ability to characterize the test sample's response to the combined environment. The reactor's remarkable longevity as a core nuclear energy research tool is tightly coupled to the continuous investment in new experimental techniques and enabling technologies as well as TREAT's unique ability to accommodate upgrades in all three areas.

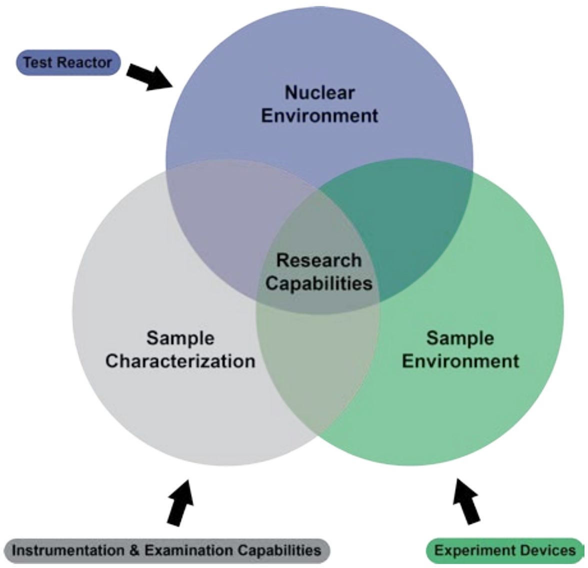

Figure 1. Pillars of transient testing capability.

Nuclear Environment

TREAT has the ability to generate a wide range of shaped transients that allow researchers to design experiments to simulate the conditions nuclear fuel may experience during either operational or off-normal events. This capability has been expanded over several decades of operation through installation of computer controlled reactor control systems and control rod drives with enhanced capabilities. While the range of transient shapes currently available at TREAT exceed those of any other test facility in the world, significant enhancements are still possible. These can be achieved through installation of increasingly dynamic control systems (i.e., higher speed rod drives or a He-3 injections system) or through enhanced reactor modeling that supports increased transient prescription definition (i.e., through asymmetric rod motions or improved knowledge of safety margins). Implementation of these technologies will ultimately be required to enable expansion of the TREAT mission space to support study of fuel response to shorter transients (i.e., prototypic reactivity initiated accident [RIA]) and to a larger fraction of longer transients (i.e., post-DNB phase of loss of coolant accident [LOCA]).

Sample Environment

The TREAT core allows users the flexibility to accommodate self-contained tests devices ranging from 10–25 cm square in cross section and roughly 3–4 m in length (with a 1.2 m long active core region). The test devices are largely decoupled from the reactor core and do not rely on its safety systems, thus, a wide variety of test environments can be created to simulate the thermal-hydraulic and mechanical environment of virtually any nuclear reactor system. This critical ability tightly coupled the nuclear transient fuel response to its specific environment. A wide variety of irradiation test devices have been developed and deployed at TREAT historically. The most noteworthy devices were used for integral system testing of sodium cooled fast reactor fuels. As the only transient testing facilities dedicated to nuclear energy technology development in the U.S. (and the only transient testing facility in the world capable of supporting most Gen IV reactors concepts), TREAT's ability to simultaneously support sample environments relevant to a variety of reactor types (through static capsules or recirculating loops) is unique and extraordinary.

Sample Characterization

The purpose of transient testing is to characterize the response of a sample or engineered system to a nuclear stimulus. At the most basic level, this is based on pre- and post-irradiation examination to evaluate the integrated change in sample condition. Characterization of this type is accomplished using the nuclear materials handling and scientific examination infrastructure currently installed and under development at MFC. However, the response of test articles to transient irradiations can be extremely complicated and multi-staged such that the behaviors leading up to the final observed state could not be de-convoluted using just post-test examination. Techniques to enable researchers to monitor real-time response are therefore crucial to effective transient testing. TREAT offers unparalleled access to an experiment during irradiation and instrumentation approaches can thus be deployed to support otherwise impossible in-situ observations of test response. TREAT experiments represent the cutting edge application for in-pile instruments and are strongly positioned to drive that development.

In the early 1990s, TREAT represented the state-of-the-art in transient testing worldwide. While in standby mode for the past 25 years, significant advances in relevant experimental capability have been made in ancillary technical fields. These advances are being gradually incorporated into the baseline capability of TREAT to align the U.S. transient testing program with modern nuclear technology development and deployment approaches. The prioritization of this capability development will be driven by the needs of the nuclear technology community.

NRC Perspective

Transient testing is an important part of fuel development and licensing. Significant testing will be needed to qualify new fuel and cladding for both LWRs and non-LWR advanced reactors. The regulatory requirements and guidance for LWR fuel exists and the path forward is generally well-known. The regulatory guidance for non-LWRs is under development. Advanced reactor design criteria (draft) available. Licensing these new concepts is feasible by utilizing the significant capability that exists at the DOE national laboratories. Early engagement and identification of issues will be needed.

Regulatory guidance with respect to research and fuel development will rely on:

- Understanding damage mechanisms
- Quantification of safety margins
- Validation of simulation tools.

Research for fuel licensing (as well as economic optimization) for new fuel, cladding, and range of operation will need to focus in these areas:

- Fuel damage mechanisms (initial damage through release of fission products)
- Fuel swelling and relocation
- Fission gas release
- Burnup effects on thermal and mechanical properties
- Cladding performance
- Benchmarks for 5% < enrichment <20%
- Defining “fuel qualification” (MSR).

Summary of LWR Industry Needs

Ken Yueh, EPRI, provided background information on the need to understand fuel behavior during a RIA or LOCA. The discussion included recent transient research activities focused on transient conditions and normal operations. The LWR survey results and gap analysis are included in Tables 1–4.

Existing or emergent issues during transient conditions include:

- RIA/LOCA performance (U_2Si_3 fuel clad with SiC or SiC-coated zirconium cladding, fission gas release, higher burnup, doped pellets [CR_2O_3, BeO_2, Al_2O_3/SiO_2, Gd_2O_3, etc.])
- Power ramp (no capability gap, ATR/TREAT/Halden)
- Power cycling – load follow (EDF has extensive experience, some data exist within the U.S. industry but will require extensive evaluation).

Existing or emergent issues during normal operation include:

- Crud phenomenon understanding and modeling (no gap, EPRI and CASL programs)
- Feed-water nickel concentration (no gap, EPRI program)
- Fuel degradation on failure (data gap on advanced fuel)
- Fission gas release (data gap on advance fuel)

Table 1. Current industry needs methodology.

What methodology do you currently use for				
Regulatory	Advanced fuel designs	New operating regimes	Optimizing reg/operating methods	Other, please explain
• Calculation • Fuel vendor • PWROG • CE reload methods as modified to use CASMO4/SIMULATE3	Calculation participating in EPRI, PWROG and CASL	• Halden test data with calculation • Licensed method • Vendors/EPRI/CASL • Talk to fuel vendors, transient thermal analyses	• Calculation • look for ways to gain process efficiencies while staying within our approved methods.	• Industry feedback with EPRI, reactor vendor • Using previous NRC guidance for LWRs
• fuel performance analysis using advanced computational techniques.				
• Studsvik codes coupled to T/H system codes (RELAP5/RETRAN/VIPRE) • We generally use calculations, based on previous testing when applicable/needed. When conditions are entirely new (e.g. advanced fuel designs) and testing has not been done in the past, we do testing			S3R core model integrated into a full scope simulator	

Table 2. Separate effects testing and priority.

Complementary Separate Effects Testing Capability	Highest Priority SE Testing
Electrically heated test train - Argonne National Laboratory (ANL), Oak Ridge National Laboratory (ORNL), Studsvik, etc.)	Steam cooling of high performance metallic fuel, 550°C peak fuel temperature assuming Zr cladding
SiC creep and burst test – SiC creep too slow, MBT at ORNL used to test SIC	U_3Si_2 swelling, melting point and conductivity as a function of burnup
U_3Si_2 interaction with water/steam	Modified burst testing on advanced cladding concept
Fuel annealing studies (unclad fuel) to measure fission gas release	
Fuel swelling measurement (U_3Si_2) – ATR	
Measure effect of irradiation on U_3Si_2 melting point and conductivity	
Modified burst testing on advanced cladding concept – Studsvik and ORNL Complementary Separate Effects Testing Capability	

Table 3. Source material and pedigree.

Source Material Required	Pedigree Need to Support Evaluation
Fresh and irradiated U_3Si_2/UO_2 fuel in SiC or SiC coated cladding	Fuel grain size distribution
Uranium nitride fuel	Cladding mechanical properties, hydrogen concentration, oxide thickness
	Power history, burnup, material composition and manufacturing process
	Complete characterization
	Standard pedigree per INL standard
	Late 2nd and 3rd cycle fuel from typical boiling water reactor (BWR)/pressurized water reactor (PWR)

Table 4. Post-irradiation examinations.

Non-Destructive Examinations	Small Sample Examinations	Destructive Examinations
Chemical and isotopic analysis	Bow and length	Optical microscopy
Optical microscopy	Eddy current	Mechanical (tensile, bend, compression, micro-hardness)
Electron microprobe	Neutron radiography	Density
X-ray diffraction	Gamma scanning	Composition
Gamma scanner	Radiation mapping	Fission gas measurement
Alpha scanner	High resolution visual	High temperature furnace (accident conditions)
TEM	Large plate/element checker	Blister annealing testing
SEM with FIB	Metrology	

Historical Report and Additional Information

The LWR research community needs access to a complete set of experimental results and testing conditions used by Shimizu for U_3Si_2 testing (available literature has limited information). In addition, access to uranium nitride irradiation data in both thermal and fast spectrum would be valuable.

For current fuel, research needs to include a realistic examination of the phenomena to ensure a real-world concern, rather than a change to a decimal point value in an analysis that is already grossly conservative.

Table 5. Capability and data gaps.

PIE Capability Gaps	Operational Gaps
▪ There are no capability gaps on post-irradiation examination. Additional capability may need to be developed as needs arise. ▪ There are no capability gaps on normal operational issues.	The following are data gaps on operational issues: ▪ New fuel operational fission gas release ▪ New fuel degradation in contact with coolant ▪ New fuel / cladding power ramp performance ▪ Fuel load following limits/guidance.
RIA Capability Gap	**RIA Data Gap**
▪ Some tests could be conducted using pressurized static capsules. ▪ A pressurized flow loop that is representative of commercial reactor conditions is needed in most tests: – DNB potential at partial power – High temperature failure – Over pressure failure – Fuel-coolant interaction – Transient fission gas release – Prototypical commercial reactor RIA pulse width is between 25 and 65 ms - Cladding ductility is pulse width dependent.	▪ PCMI performance of doped and new fuel designs. ▪ PCMI performance of non-zirconium cladding – Mechanical tests could generate most of the test data – PCMI performance of zirconium-based cladding under some conditions is needed to verify mechanical characterization data. ▪ Fuel-coolant interaction. ▪ Transient fission gas release – TREAT pulse width is > 60 ms.
LOCA Capability Gap	**LOCA Data Gap**
▪ Fully realistic test conditions are lacking – 50% external electrical heating used in Halden LOCA tests – 100% external heating utilized by others – TREAT could provide 100% nuclear heating. ▪ A pressurized flow loop is desirable – To remove heat from initial conditioning at pre-transient power to close fuel-cladding gap – To test multiple rods to evaluate flow blockage due to balloon/burst and potential local temperature excursions. ▪ In situ characterization of fuel relocation / dispersal.	▪ Fuel fragmentation threshold and mechanisms – Fission gas and grain boundary role – Pellet stress contribution from operational temperature gradient. ▪ Fuel relocation and dispersal. ▪ Fuel cladding balloon and burst behavior. ▪ No data on new fuel designs – Pellet behavior – Limited cladding characterization.

Advanced Reactor Fuel Safety Research Needs

Molten Salt Reactor Technology Working Group Perspectives

The MSR TWG includes industries and utilities developing various molten salt technologies. Nick Smith, Southern Company, is the chair and Lou Qualls, ORNL, is the DOE NTD. See Figure 2 for the MSR TWG affiliates.

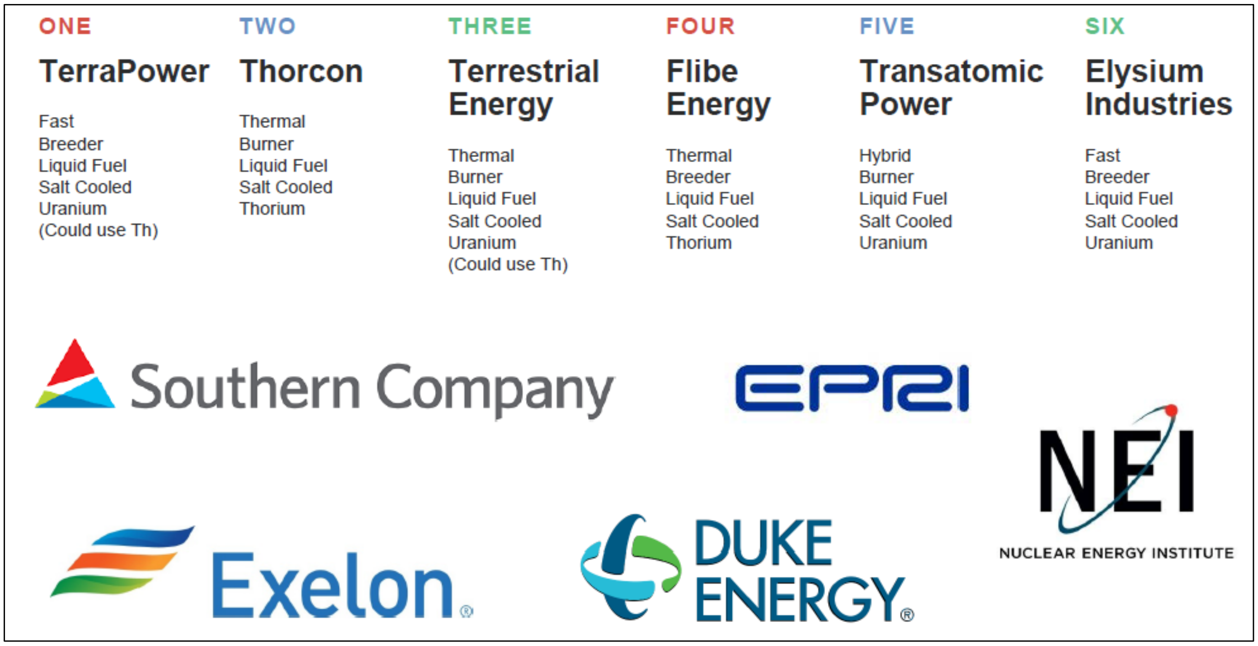

Figure 2. MSR TWG affiliates.

Molten Salt Reactor Fuel Safety Research Needs

Lance Kim, Southern Research, represented the MSR TWG during the workshop. Table 6 provides the questions and summary results of the pre-meeting survey.

Table 6. Pre-meeting survey results.

Fuel Safety Research Questions	Summary of MSR Survey Results
Describe the design basis transient events that have been developed for your reactor concept.	Full loss of flow, partial loss of flow, over fissile addition, cold fuel insertion, earthquake pressure wave, gas cycling, reactivity addition with circulating gas
What are the relevant fuel safety criteria and fuel design limits that have been defined that will be used to demonstrate compliance with 10CFR50 App A: General Design Criteria?	Fuel boiling, fuel freezing, lack of fuel draining, pressure pulse induced density increase, gas/volatile FP release fraction for leak, release after freezing, UCl4, UCl5, UCl6 production and release after freezing, mobility temperature to

Fuel Safety Research Questions	Summary of MSR Survey Results
	prevent radiological separation. Structural failure after exceeding creep limits over time.
What integral system tests are required to validate your fuel system failure modes and the efficacy of the defined fuel safety criteria?	Fuel dumping speed, heat capacity for decay heat minimizing temperature rise.
What capabilities should test devices have to meet your testing needs?	Static and flowing systems, corrosion control systems
What separate effects studies could be used to assess specific fuel system behavior of interest prior to or in parallel with integral systems tests?	Simulated fuel tests
What are the sources of fresh and irradiated fuel materials available to conduct relevant experiments?	Spent nuclear oxide fuel, U, Pu, DU, Unat, U_3O_8, UO_2SNF metallic fuel (but it is a bit harder)
If you need access to historical reports and/or data, please list them below.	Integral FR program handling reports and estimated test and production system cost, fuel product removal efficiencies for pyro-processing.

DOE Perspectives

Joel McDuffee, ORNL, provided the DOE NTD perspective on MSR research needs. It is difficult to down select to materials of interest with many competing MSR concepts. Fluoride salt reactor concepts can take advantage of extensive previous work and existing data. Chloride salt concepts require more testing.

A low-cost, rapid scoping irradiation screening is needed and existing irradiation facilities must be used to the extent possible. Fast reactor concepts will be the most difficult. Fast spectrum irradiation locations in existing thermal test reactors tend to be small and not well suited to flowing salt loops. Therefore, investment is needed in new capabilities for longer-term confirmatory irradiation investigations.

The next step is to evaluate what has already been done and what can be done. University reactors are already irradiating some salt samples. National laboratory reactors have existing capability. Identify what can be done with a modest investment, such as installing irradiation facilities for salt work. Plan and propose the larger investments required, including development of a flowing salt loop near or in a reactor core.

Fast Reactor Technology Working Group Perspectives

The FR TWG consists of a set of industries and utilities developing various fast reactor technologies. Jake DeWitte, Oklo, is the chair and Bob Hill, ANL, is the DOE NTD. FR TWG affiliates include:

• ARC	• GE
• Columbia Basin	• General Atomics
• Duke	• Oklo
• Elysium Industries	• Southern
• EPRI	• TerraPower
• Exelon	• Westinghouse

Robust fuel behavior can enhance the safety case. The fuel-changing phase is not necessarily fuel failure; it can be a safety benefit. The coolant system can be a barrier to radionuclide release. Fuel design variations are being considered to extend the operating envelop of metal fuels and next generation cladding materials.

Fast Reactor Fuel Safety Research Needs

General transient considerations include loss of flow, loss of heat sink, and overpower (reactivity insertion) leading to

- Peak fuel temperature-change of phase
- Peak cladding temperature
- Peak cladding strain (total strain and thermal creep)
- Peak structural temperatures and strain
- Radionuclide evolution and release.

Separate Effects Tests

- Furnace testing on pins and high temperature cladding and structural material creep tests)
- Fuel-cladding-coolant-structure compatibility and durability.

Integral Effects Tests

- Reactivity insertions
- Loss of flow
- Run beyond cladding breach
- Fuel movement effects.

Testing Capabilities

- Immersion configurations (static capsule, flowing loop)
- Geometry flexibility (single pin, multi-pin)
- Spectrum flexibility
- Chemistry and corrosion control
- Enhanced hodoscope capabilities.

Fuel Sourcing

- High assay low enriched uranium (LEU) supply and associated infrastructure development
- Fuel manufacturing/production; prototyping to commercial scale up
- Plutonium and actinide-bearing UNF materials.

Fast Test Reactor (as soon as possible to maximize benefit to developers)

- Accelerate new fuels and materials development
- Enable exploratory studies on fuel design improvements and next generation technologies
- Opportunity to support fuel supply chain with fuel fabrication for the test reactor.

Modeling and Simulation

- Existing tools (BISON, LIFE-METAL, SAS4A/SASSYS)
- Legacy data
- Lab development supporting verification and validation (industry owns design-specific validation for regulatory purposes).
- Mechanistic-driven, meso-scale tools to inform testing programs.

Historical reports and data are needed.

- Metal fuels reports and data, including supporting documentation of applicable metallic fuel transient tests, including as-built data packages, as-run conditions, PIE results, and supporting documentation.
- Legacy and modern fast reactor fuel experimental reports and data
  - Experimental data on UO_2, UN, UC, and advanced metal fuel irradiation performance
  - Experimental data on cladding materials
  - Pyroprocessing reports and data
  - Centralized Reliability Data Organization database of component reliability for liquid metal reactors.
- Improve the process on how U.S. companies' access applied technology documents.

Development of new fuels for fast reactors will require a full suite of supporting capabilities. New fuel development is achievable. This is good timing for fast reactor development with the support of TREAT and a new fast test reactor.

DOE Perspectives

Dave Grabaskas, ANL, provided the DOE NTD perspective on FR research needs. EBR-II and the Fast Flux Test Reactor have databases, legacy documents, and national laboratory expertise that can provide the FR TWG industries with the experimental data they need. There is a steady-state performance database that is being expanded. Transient tests included shutdown heat removal tests, cladding breach experiments, furnace tests, melt tests, etc.

Grabaskas provided two key references for consideration.

- There are no major technology gaps that would prevent the design and the development of a licensing case for a sodium-cooled fast reactor as long as one stays with known technology. (SFR Accident Initiator/Sequence Gap Analysis [FCR&D-REAC-2010-000126])
- The current state of knowledge of SFR fuel and structural material performance is sufficient for designing and licensing a SFR today within the envelope of the existing database. The boundaries of the existing database would be a fuel burnup of 10 at% or less, metallic or oxide fuel, a peak cladding temperature of 600°C or less, a peak dpa of 100 or less, and with fuel that has not been reprocessed. Both the steady-state and off-normal irradiation database would be sufficient to support such a design. (SFR Fuels and Materials Gap Analysis [SAND2011-6546]).

The closure of the following gaps would reduce uncertainties and/or extend the existing database:

Fuel Performance
Minor actinide-bearing fuel includes several dozen rodlets irradiated in Cd-shrouded ATR positions that are not a prototypic environment for integral fuel rod performance. Collected relevant fuel data for mechanisms are primarily dependent on fuel temperature. Minor actinide-bearing fuel also includes two pins in the FUTURIX-FTA experiment in the Phenix fast reactor.

High burnup/DPA includes about a dozen rodlets irradiated in Cd-shrouded ATR positions to burnups ≥20%. Not a prototypic environment for integral fuel rod performance. Collected relevant fuel data for mechanisms that are primarily dependent on fuel temperature.

Current AFC testing is continuing in Cd-shrouded positions in ATR. Limited by lack of domestic fast reactor.

Transient Behavior
Only severe unprotected transient overpower (UTOP) tests were completed at TREAT for metal fuel and severe loss of flow phenomena may be different (such as fuel movement during the transient, reactivity effects, fragmentation in coolant channel, movement of fuel out of the coolant channel, and fuel freezing/relocation). Severe loss of flow simulations exist but are lacking validation.

Source Term

- Migration of radionuclides during irradiation. Radionuclide migration to bond Na/fission gas plenum may be available for release with cladding failure. There are several gaps in current state of knowledge.
- Release of radionuclides during fuel pin failure due to pin depressuraization mechanics, formation of bubbles in coolant channel, and/or entrainment of radionuclides.
- Release of radionuclides during fuel melting. There is a lack of data on high burnup fuel melting in liquid sodium.

Conclusions
Substantial database exists, sufficient for licensing within historic database limits including steady-state irradiation history, transient tests, and source term information. TREAT restart presents opportunities to reduce uncertainties in loss of flow model validation and radionuclide behavior during severe accidents. Additional tests/experiments are possible and may extend database or reduce uncertainties. Many do not need irradiated fuel or radioactive elements. Further deviation from existing database may require new and repeat tests for new fuel types and coolants and new facilities or facility refurbishment.

High Temperature Gas Reactor Technology Working Group Perspectives

The HTGR TWG consists of a set of industries and utilities developing various high temperature gas-cooled reactor technologies. Farshid Shahrokhi, AREVA is the chair and Hans Gougar, INL, is the DOE NTD. Members of HTGR TWG are AREVA, BWXT, Duke Energy, StarCore Nuclear, and X-Energy.

High Temperature Gas Reactor Fuel Safety Research Needs

Fundamental enabling technology needed for HTGR to move forward are:

- TRISO coated particle fuel qualification
- Nuclear grade graphite characterization
- Integrated HTGR neutronics and thermo-hydraulics certified codes and methods (source term, radionuclides transport models, thermos-hydraulics).

The main components of this are DOE's Advanced Reactor Technology programs today. In addition, the U.S. needs a fuel supply chain (high assay LEU and commercial scale fuel manufacturing capability).

Development work required to address specific need of individual components during the detailed design

- Design specific – This work is harder to define, and often is variable depending on the designer's strategy.
- Before detail design is launched – tradeoff often results in whether it is more practical to do some research and development or design around it.
- During detail design – this will be a large fraction of the work for supporting laboratories in the coming years as HTGR deployment proceeds.

Long-term research and development to enable future, more advanced concepts. (This obviously includes VHTR, but it would also include direct Brayton cycle and advance fuel cycles.)

DOE Perspectives

Paul Demkowicz, INL, provided the following DOE perspective on HTGR fuel safety research needs.

Future safety testing needs and plans

Re-irradiation of fuel prior to out-of-pile safety testing in order to generate short-lived I-131 ($t_{1/2}$ = 8.02 d) and measure iodine release (I-131 is a major off-site dose contributor).

- Current plans will utilize the neutron radiography (NRAD) reactor at HFEF.
- Testing will commence as part of the AGR-2 safety testing campaign.

High temperature safety testing in atmospheres containing air or moisture to expand the range of reactor accident scenarios (e.g., air ingress, steam generator tube rupture).

- A key issue is the potential for oxidizing atmospheres to volatilize fission products in the fuel matrix and graphite.
- Furnace system is currently being designed and tested at INL with planned deployment in the Fuel Conditioning Facility air cell.
- Testing is planned for the AGR-5/6/7 fuel; earlier testing on AGR-2 and AGR-3/4 fuel will be performed if schedule allows.

Appendix: A
LWR Pre-Meeting Survey Results

1. What are your existing or emergent fuel transient performance issues that are not being adequately addressed?

- Effect of transients, design basis accidents such as LOCAs and RIAs on high burnup UO_2 and U_3Si_2.
- Better understanding of CRUD phenomena, and modeling of same; fuel performance codes for thermal conductivity calculations; improved thermal mixing without exacerbating grid fretting. Higher burnup fuel.
- High Nickel concentration values in feed water and jet pump riser indications due to FIV.
- Vibration effects, constrained thermal expansion.
- Improving power/lifetime, power cycling capability.
- For ATF concepts using a) U_3Si_2 fuel and coated Zr and b) U_3Si_2 fuel and SiC cladding, the following testing will be valuable (to be performed using both fresh fuel and fuel irradiated at different burnup levels):
    1) Reactivity Initiated Accident (RIA)
    2) LOCA
    3) Power ramp testing (100 W/cm-minute, i.e., much slower than RIA) to simulate Condition II transients, particularly evolution of fission gas release and swelling during such transients (transient behavior is more difficult to model analytically and testing would supplement data obtained at steady-state)
    4) Testing of the performance of a leaking fuel rod during operation, particularly to investigate water-U_3Si_2 interaction.
- At power RIA events leading to DNB failure.
- All existing fuel transient issues are currently addressed through existing methods or defined NRC licensing actions. Emergent transient issues center around industry needs to qualify ATF and evaluate fuel designs relative to higher than 5% enrichment and burnup beyond 62 GWd/mTU.
- Effects of two phase flow on pumping, ability to drop pressure to allow pumping without electricity, using steam only, either for pumping or water injection. Ability to test metallic fuel and Lightbridge fuel geometry and materials (Zr), or Lightbridge shape with stainless steel.
- Need to restart TREAT for ATF transient testing.
- Fuel rod integral testing, for at power RIA events, to investigate fuel rod failure from DNB.

2. What methodology do you currently use to resolve fuel safety issues associated with the following?

a. Emergent regulatory questions
   - Our fuel vendor and/or the PWROG the methodologies currently used to address emergent issues.

– Studsvik codes coupled to T/H system codes (RELAP5/RETRAN/VIPRE).
– CE reload methods as modified to use CASMO4/SIMULATE3.
– Ask experts, not the NRC due to cost and bias.
– Calculations, based on previous testing when applicable/needed. When conditions are entirely new (e.g., advanced fuel designs) and testing has not been done in the past, testing is done.
– Fuel performance analysis using advanced computational techniques.
– Historically emergent issues have been addressed through adaptation of existing methods and the development of AREA and ARITA advanced methods using existing databases.

b. Advanced fuel designs
– Calculation.
– For ATF, methods are not yet approved. We are participating in various industry working groups (EPRI, PWROG, and CASL).
– Studsvik codes coupled to T/H system codes (RELAP5/RETRAN/VIPRE).
– The above methods for fuel design upgrades are not applicable at this time for advanced fuels.
– Talk to fuel vendors, transient thermal analyses.
– Fuel performance analysis using advanced computational techniques.
– Specific NRC licensing actions on GAIA and Atrium 11 fuel designs.
– Just starting a company, so a primer is needed to resolve these issues.

c. Introduction of new operating regimes
– Test reactor data from Halden with calculations.
– Current licensing methods. Exploring alternate/advanced method in development by the fuel vendors, EPRI and CASL.
– Studsvik codes coupled to T/H system codes (RELAP5/RETRAN/VIPRE).
– Ability to keep Rx operational through natural disasters for better disaster power availability.
– Specific NRC licensing actions adapting methods using existing data.
– Operation through electricity blackout or earthquakes, etc.

d. Optimizing current regulatory/operating methods
– Calculations.
– Identify ways to gain process efficiencies while staying within our approved methods.
– S3R core model integrated into a full scope simulator.
– CE reload methods as modified to use CASMO4/SIMULATE3
– Fuel performance analysis using advanced computational techniques.
– Specific NRC licensing actions adapting methods using existing data.

e. Other, please explain.
– Industry feedback with EPRI, reactor vendor.
– Need lower cost, better performance fuel, that also improves safety, but safety improvements without performance AND cost improvements are a NO GO. Actual public safety cannot be improved, above the current zero public deaths. Only plant damage/cost of cleanup after an accident can be improved, so increased cost or worse performance, and kills more people by making nuclear more expensive, thus promoting more deadly fossil fuels.
– Using previous NRC guidance for LWRs.

3. **What additional access to in-pile transient testing capability do you need to support timely resolution of issues?**

- Low, mid, high burnup U_3Si_2 fuel in SiC and coated cladding- hopefully TREAT can provide the RIA and perhaps LOCA conditions for this fuel.
- We rely on our fuel vendor(s) to have access to this capability to support updates/changes to methods/codes, as the need arises.
- Since CFE, a government utility doesn't have testing capabilities of its own, we contract research labs or the fuel vendor when need it.
- Pellet degradation effects on pin conduction/convection.
- Testing to show that esoteric NRC concerns against the 1980s licensing basis (Standard Review Plan) are not a real concern. Examples of this are control rod ejection, thermal conductivity degradation, grid growth impact on spent fuel pool criticality.
- Faster qualification of new more economic, better performing ATF. GE, Westinghouse/GA, AREVA ATF are not better performance AND lower cost.
- See answer to question #1.
- Flowing coolant RIA testing capability.
- Current vulnerabilities to potential or emergent testing needs are the limited number of facilities and their availabilities worldwide.
- Ability to run test cell with attached pump either in loop, or at core inlet, both for water flow and two phase, and superheated steam flow. For both PWR/BWR/Lightbridge fuel shapes.

4. **What capabilities should test devices have to meet your testing needs?**

- PWR environment; static loop okay for RIA but will need some flow control of coolant over the time scale of a LOCA; Single pin is probably okay.
- Test devices should be able to do loop tests; multiple pin assemblies; power increases and cyclical power operation (e.g., FPO); and power pulses (RIA/LOCA). They should also be able to handle debris testing and fluctuation in boron concentrations.
- Thermal hydraulic environment.
- More measurements in conventional or near-conventional operating regimes. The other regimes are either too rapid or too severe to be of additional practical value compared to existing data.
- Need space for small bundle Lightbridge or similar fuel with both water and steam cooling availability. Space for a below cell linear pump, with saturated water or steam line input to pump.
- Prototypical PWR conditions, in terms of temperature, pressure and water chemistry.
    1) Static conditions sufficient for some testing (e.g., RIA) while flowing conditions needed for others (e.g., performance of leaking fuel during operation).
    2) Testing device should handle both fresh fuel and irradiated (at various BU levels) fuel.
    3) Single pin testing is okay for some testing (e.g., RIA) and for starting LOCA evaluations. The latter will however need to be supplemented with mini-bundle testing soon after, to capture lattice effect.
    4) Testing devices must be adequately instrumented not only to accurately capture performance indicators but also to provide data for M&S code benchmark
- BWR/PWR flowing loops with multi pin configuration to investigate DNB propagation.
- LWR pressure, temperature and flow conditions are necessary to capture benefits of advancement through precise determination of uncertainties. Application of measurement advancements to

achieve internal pin data such as fuel and cladding temperatures and dimensional characteristics as well as internal gas pressure and fission products are key.

- Test BWR/PWR test cells or Cylindrical, or hexagonal Lightbridge (Zr) or SS Lightbridge touching fuel pins constrained on outside. Continuous positive displacement flow, water phase, two phase, and steam phase, and in core transitions for decay heat removal test for BWR/PWR cells or power operation or decay heat for Lightbridge fuel.

5. What complementary separate effects testing capability would be useful to first evaluate issues at a phenomenon level before committing to integral scale, in-reactor testing?

- Baseline UO_2 in coated cladding.
- Boron transport loop modeling data; debris transport; RIA from shutdown conditions.
- Fission gas migration. Sub channel flow measurements.
- Electrically heated version of above pumped cell performance.
- And separately steam cooling test.
- Mechanical testing for irradiated SiC (creep and burst especially).
    1) Test interaction between U_3Si_2 and water/steam, with fuel irradiated at different levels.
    2) Perform fuel annealing studies (uncladded fuel in a furnace and subject to gradual heatup to measure extent of fission gas release vs temperature, again for both fresh and irradiated fuel.
    3) Measure swelling of U_3Si_2 as a function of burnup and temperature.
    4) Measure effect of irradiation on melting point and thermal conductivity of U_3Si_2.

- MBT testing on hydrided advanced cladding concepts.
- Thermal conductivity, fission gas production, and specific influence of irradiation on new materials.
- Much of the above testing can be performed in separate effects testing.
- Existing ATF-2, ATF-3, Halden, and commercial reactor testing will likely be adequate for ATF in the short term.

6. What is the highest priority candidate separate effects study(ies) that could be used to isolate phenomena of interest to improve codes and better design in-core testing?

- Fuel performance data. How cladding and fuel assembly materials respond to thermal fatigue due to power (thermal and neutron) or temperature cycling;
- Sub channel flow measurements.
- Steam cooling of high performance metallic fuel, or similar 550°C peak fuel temperature fuel, assuming Zr cladding.
- Measurement of swelling of U_3Si_2 as a function of burnup and temperature, and measurement of effect of irradiation on melting point and thermal conductivity of U_3Si_2
- MBT testing on hydrided advanced cladding concepts
- Specific influence of irradiation on new materials
- PWR/BWR test cell flow testing for decay heat removal.

7. **What source material is required to conduct these experiments (fresh fuel, pre-irradiated fuel)?**

- Fresh and pre-irradiated fuel; U_3Si_2 and UO_2 in coated cladding, and U_3Si_2 in SiC cladding.
- Electrically, heated rods.
- Fresh fuel, high assay LEU.
- U_3Si_2 in SiC and coated Zr claddings, both in fresh and in already irradiated conditions (at different BU levels). Also need a UO_2/Zr baseline in the same condition. It is desirable to test UN.
- Fresh fuel and components such as cladding.
- Electrically heated PWR/BWR fuel forms, and then new fuel, then pre-irradiated fuel. Hollow twisted cruciform tubes for separate effects test, Lightbridge fuel.

8. **What fuel pedigree/pre-characterization do you need to support evaluation?**

- For fuel - grain size distribution, density.
  For cladding - mechanical properties; for coated cladding - H levels, ZrO_2 thickness, and remaining Zr metal thickness for low, mid and high burnup U_3Si_2.
- Power history, burnup, material composition, manufacturing process.
- Complete characterization.
- Standard pedigree per INL standards is sufficient.
- Late 2nd cycle and 3rd cycle fuel from typical BWR/PWRs operated in the U.S.
- Full dimensional, micro-structural, and mechanical properties.

9. **What post-irradiation examinations (PIE) are required to collect the desired data for non-destructive examinations?**

- Bow and length with contact Profilometer (ECP) examination
- Eddy current (EC) examination
- Neutron radiography (NRAD) examination
- Precision gross and isotopic gamma (PGS) scanning
- Radiation mapping
- High Resolution visual (VEM) examination
- Large plate/element checker
- Metrology

10. **What PIE is required to collect the desired data for destructive examination in a hot cell?**

- Leica Microscope/Leitz Metallograph/Leco Microhardness
- Optical metallography/ceramography and other test sample preparations
- Sample Preparation
- Tensile, compression, and bend (Instron) testing
- Density measurement analysis
- Element, fuel and/or hardware test section cutting or disassembly
- Fuel element fission gas (GASR) sampling
- High temperature furnace testing

- Hot Fuel Dissolution Apparatus (HFDA)
- Blister annealing testing
- Oxide reduction furnace
- Fuel accident conditions simulation furnace (FACS) testing
- Metal Waste Form Furnace (MWFF)
- Element, fuel and/or hardware test section cutting or disassembly

12. What small-sample PIE is required to collect the desired data?

- Chemical and Isotopic Analysis
- Optical microscopy
- Scanning electron microscope (SEM)
- Shielded electron probe micro (EPMA) analysis
- Shielded mechanical testing
- Shielded micro X-ray diffraction (ÂµXRD) analysis
- Shielded sample preparation (SSPA)
- Counting (gamma scan, alpha spec, gas proportioned counter)
- Transmission electron microscope (TEM)
- Dual beam focused ion beam (FIB) with SEM
- FIB/SEM (Quanta)
- Gas mass spec
- Micro x-ray diffraction / X-ray diffraction
- Nano scale electrical and mechanical testing
- Particle fuel specialty processes (leach burn leach, particle picking, kernel cracking and picking, hardware leaching, condensate plate analysis)
- Metallographic optical analysis
- Micro hardness testing
- TEM (Titan ChemiSTEM)

12. If you need access to historical reports and/or data, please list them below.

- Complete set of experimental results and testing conditions used by Shimizu for U_3Si_2 testing. Also, access to UN irradiation data in both thermal and fast spectrum is valuable.

13. Please add any additional information and/or questions below.

- For current fuel, research needs to include a REALISTIC examination of the phenomena to ensure that it is of real world concern, rather than a change to a decimal point value in an analysis that is already grossly conservative.
- Need to standardize format where experimental data are provided with data in tabular format to be included in addition to plots (this would allow avoiding conversion of plots into tables later).
- Eddy current measurements to be added to the answer to Question #9, “What post-irradiation examinations (PIE) are required to collect the desired data for non-destructive examinations?”

Appendix B:
Advanced Reactors Pre-Meeting Survey Results

1. **Describe the design basis transient events that have been developed for your reactor concept.**

   - We have not fully developed the DBAs for our Steam Cycle HTGRs. Initially we will assume a standard set of accidents (e.g., medium and small de-pressurized conduction cool-down, pressurized conduction cool-down, small reactivity insertion accidents, loss of load, loss of feed-water, etc. A loss of reactor cavity cooling system (our redundant passive Decay Heat Removal) would be a beyond design bases accident (we do not use severe accident terminology). As our design effort progresses to preliminary design phase; the DBA set will be finalized using our PRA tool. This tool is now too primitive (not enough design data) to be of any value above and beyond our deterministic DBA selection.
   - Worst case: large hole, loss of helium leading to partial depressurization (to containment pressure) of reactor.
   - Loss of forced flow (molten salts or lead), reactivity insertion accidents (unattended control/shutdown rod removal or withdrawn), loss of heat sinks.
   - Liquid fuel - Full loss of flow, partial loss of flow, over fissile addition, cold fuel insertion, earthquake pressure wave, gas cycling, reactivity addition with circulating gas.
   - Events for the LFR are grouped into categories such as Loss of Flow (LOF), Loss of Heat Sink (LOHS), Overpower Transient (OT), etc., type of events. Each is analyzed with various initiating events, e.g., LOF can be initiated by loss of power to the pumps, by assembly flow blockage etc. Each is analyzed in protected (shutdown occurs) and unprotected mode. Unique to LFR, relative to other LMRs, is Steam Generator Tube Rupture as Steam Generators can be immersed directly into the primary pool. Grouping of events into DBA and BDBA is being performed.
   - Design basis transients , double primary sodium pump trip, dingle primary sodium pump seizure, leak in primary sodium pump outlet or core inlet pipe, Withdrawal of all control rod assemblies to rod stops, local blockage in fuel assembly (6 sub-channel) severe accidents, unprotected loss of off-site power, unprotected loss of heat sink, unprotected transient overpower, unprotected operating basis earthquake, instantaneous full blockage of an assembly, loss of all decay heat removal systems, and unprotected safe shutdown earthquake.
   - Similar to the metal fuel historical fast reactor transient data transient overpower is the most challenging. Tests previously conducted in TREAT.
   - Steady state fuel effects, decay overheating effects on fuel and reactor structure of fast chloride liquid fuel.

2. **What are the relevant fuel safety criteria and fuel design limits that have been defined that will be used to demonstrate compliance with 10CFR50 App A: General Design Criteria?**

   - We have not yet defined our PDCs based on Appendix A GDCs. We are working with the NRC on review of DG-1330 and the subsequent RG1.232 to define mHTGR-DC. We will then develop SARRDLs (specified acceptable radiological release design limit). These are acceptable fuel radiological releases during normal operations and accident conditions that would allow release of circulating activities to the environment and still meet the site parameter does limits.
   - TRISO fuel temperatures remain below 1500°C.

- Fuel boiling, fuel freezing, lack of fuel draining, pressure pulse induced density increase, gas/volatile FP release fraction for leak, release after freezing, UCl_4, UCl_5, UCl_6 production and release after freezing, mobility temperature to prevent radiological separation. Structural failure after exceeding creep limits over time.
- Main criteria are fuel pellet melting and cladding failure, for example due to high-temperature creep.
- Peak fuel temperature, peak clad temperature, peak clad strain-total, peak clad strain-thermal creep.
- Assuming this refers to the NLWRDCs that are near final, definitions include cladding failure with fractional radionuclide release (still in development).
- Partial loop freezing, overheating effects on structure as a factor of time and temperature.

3. **What integral system tests are required to validate your fuel system failure modes and the efficacy of the defined fuel safety criteria?**

- Completion of the AGR program which completely characterizes TRISO particle fuel is what we need. Codes and methods will then be used to support our safety case and the actual safety testing during the first three years of module one of the FOAK commercial scale demonstration plant will validate our fuel.
- We plan to use the HTTR facility in Japan because it already exists and is designed for the fuel we intend to use.
- Fission product release beyond design basis.
- Fuel dumping speed, heat capacity for decay heat minimizing temperature rise.
- Valuable integral tests to be performed on both fresh and irradiated fuel/cladding samples, are
  - Rapid reactivity insertion, to simulate RIA and check fuel melting and cladding behavior
  - Loss of flow (to simulate 1) assembly blockage and 2) pump failure, with the latter looking at subsequent natural circulation and thus requiring an integral system test)
  - Run-beyond-cladding-breach testing (to test performance of a rod that has a hole, to look especially at the interaction between lead coolant and fuel/cladding and eventually interaction between failed rod and adjacent ones)
  - Gradual power ramp testing, to monitor (irradiated) cladding behavior upon fission gas release evolution as power/temperature is increased.
- Already performed for initial fuel design. Follow on fuel designs will likely require similar testing capabilities under challenging transient overpowers.
- Fuel does not fail. Need to evaluate when structure fails, whether core structure, internal components creep failure, pump failure, or heat exchanger.

4. **What capabilities should test devices have to meet your testing needs?**

- Not applicable for TRISO particle fuel, assuming these are fuel test devices not component testing devices. Our fuel is being characterized and tested in ATR under AGR program.
- See previous response concerning use of HTTR facility in Japan.
- Thermal hydraulic environment and transient types.
- Static and flowing systems, corrosion control systems.
  - (Note that the question refers to testing needs in general and as such testing to assess long-term performance of the fuel/cladding system and not exclusively accident-type of events, is included in the discussion below).

- Fuel/cladding system in the test devices must be immersed in liquid lead environment at prototypical pressure (~atm) and temperature (~500°C, but also higher), and be tested in both fresh and irradiated conditions.
  - With the exception of very rapid transients (e.g., RIA), the majority of tests in support to the LFR fuel safety case (especially those looking at cladding long-term performance) should be performed in flowing lead conditions with controlled lead chemistry.
  - As for the lattice configuration, for transient testing such as loss of flow the test section should be able to accommodate a 19-rod, hex array assembly, with rod OD and pitch in the ranges $9 \leq OD \leq 13.5$ mm; $12 \leq P \leq 18.5$ mm (which corresponds to a hexagon with a ~5 cm side length).
  - However, tests looking at rapid transients can start with single-pin configuration but soon move to the mentioned mini-assembly configuration in order to capture lattice effect, including potential propagation of failure from one pin to adjacent ones, subchannel blockage, etc.
  - Testing device power capability should include both very rapid power excursions (to simulate RIA) and more gradual power ramps (to be used for assessing fuel system behavior as a function of fission gas release for example).
- Both static capsule and flowing Na loops would be of interest for overpower transients. Single pins are expected to be adequate, but there may be interest in multiple pin configurations.
- Multi-pin equivalent configurations.
- Power ramps.
- Temperatures from 500°C to 1000°C, initially (Gen IV-1) to eventually 1500°C for Gen IV-2 designs. Flows from static to high flowing loops, no pins/liquid fuels. Fast ramp from at least 5%/minute. Flow failure testing and heat up rates, flow decay rates.

5. What separate effects studies could be used to assess specific fuel system behavior of interest prior to or in parallel with integral systems tests?

- TRISO particle fuels already have years of separate effects tests. Manufacturing quality control is the key to the performance TRISO fuel. B&W has mastered this capability and must maintain its capabilities until a commercial market develops following the FOAK testing and validation.
- Model simulation of thermal/hydraulic characteristics of proprietary fuel sleeve design.
- Effects of irradiation on properties such as thermal conductivity, interaction between sheath and fuel.
- Simulated fuel tests
  - High-temperature creep of cladding tubes.
  - Chemical interaction between hot liquid lead and fuels such as UO_2, UN, and metal
  - Corrosion testing in flowing lead at prototypical pressure (~atm), temperature (between 450 and 750°C) and with coolant chemistry.
- Whole pin furnace tests on irradiated fuel pins of interest for loss of flow/heat shield conditions; cut fuel segment fuel tests to assess eutectic penetration rates; differential scanning calorimetry on irradiated fuel pin segments to assess solidus temperatures and fuel/clad eutectic penetration rates.
- Out of pile thermal diffusion studies to evaluate fuel evolution during transients.
- Some of these pump failure and heat up tests can be performed by pipe heating to simulate volume heating tests and flow coast down.

6. **What are the sources of fresh and irradiated fuel materials available to conduct relevant experiments?**

   - We need high assay LEU up to 20% enriched uranium. The US government from existing down blending effort must supply the enriched uranium for the first core of the FOAK plant until such time that a market for high assay fuels develops and a commercial viability can be demonstrated for subsequent cores.
   - See response on using HTTR facility in Japan.
   - Canadian Nuclear Laboratories.
   - SNF oxide fuel, U, Pu, DU, Unat, U_3O_8, UO_2; and SNF metallic fuel, but it is a bit harder.
   - Non-irradiated cladding (metallic and SiC) and UO_2 can be provided by Westinghouse and its collaborators. Advanced fuels under consideration for the LFR (UN, advanced metal fuel) could be provided through domestic and international labs. Sources of irradiated samples are to be determined.
   - Retained irradiated MFF pins, fresh fuel pins can be readily manufactured, small number of new irradiated fuel pins could be available for testing.
   - This is the single most limiting issue for advanced reactor deployment. There needs to be a bridge supply of greater than 5% LEU.

7. **If you need access to historical reports and/or data, please list them below.**

   - Not so much for fuel but for reactor systems we need access to NPR (new production reactor - HTGR) design work that was performed in the late 1980s and 1990s.
   - Will need access to work on TRISO fuel experiments.
   - Integral Fast Reactor program handling reports and estimated test and production system cost, FP removal efficiencies for pyro-processing.
   - Experimental data on irradiation performance of UO2 fuel in D9 cladding, or of D9 cladding alone. Experimental data on UN irradiation performance. Experimental data (if available) on manufacture and performance of advanced metal fuel (i.e., a design compatible with LFR operating conditions, e.g., w/o sodium bond). Not related to fuel specifically: Centralized Reliability Data Organization database of component reliability for liquid metal reactors (currently being assembled by ANL).
   - Supporting documentation of applicable metallic fuel transient tests (M-series, whole pin furnace tests, etc.) including as-built data packages, as-run conditions, PIE results, etc.
   - The Applied Technology (AT) reports developed as part of historical fast reactor research and development are some of the best resources. However, the process for companies accessing AT needs to be re-evaluated. US companies should not have to wait several months to get this data.

8. **Please add any additional information and/or questions below.**

   - Fuel production coaters at B&W are idle at risk for detrition, obsolesces and dismantling. DOE should provide the means to maintain this capability for HTGRs otherwise we have to recreate it at additional costs.

Appendix C:
Fuel Safety Research Testing Capability

Applicable INL Post-irradiation Examination Capability

HFEF (Non-destructive Examination)	HFEF (Destructive Examination – Hot Cell)	IMCL – PIE on Irradiated Materials	EML	Analytical Lab
Metrology	Optical metallography /ceramography and other test sample preparations	Plasma FIB (Helios)	Dual beam focused ion beam (FIB) with SEM	Micro x-ray diffraction / X-ray diffraction
Neutron radiography (NRAD) examination	Leica Microscope Leitz Metallograph Leco Microhardness	TEM (Titan ChemiSTEM)	Metallographic optical analysis	Counting (gamma scan, alpha spec, gas proportioned counter)
Radiation mapping	Blister annealing testing	FIB/SEM (Quanta)	Micro hardness testing	Micro-gamma scan
Bow and length with contact Profilometer (ECP) examination	Element, fuel and/or hardware test section cutting or disassembly	Optical microscopy	Nano scale electrical and mechanical testing	Particle fuel specialty processes (leach burn leach, particle picking, kernel cracking and picking, hardware leaching, condensate plate analysis)
Eddy current (EC) examination	Density measurement analysis	Shielded mechanical testing	Scanning electron microscope (SEM)	Gas mass spec
Precision gross and isotopic gamma (PGS) scanning	Fuel element fission gas (GASR) sampling	Shielded electron probe micro (EPMA) analysis	Transmission electron microscope (TEM)	Chemical and Isotopic Analysis
High Resolution visual (VEM) examination	High temperature furnace testing	Shielded micro X-ray diffraction (µXRD) analysis	Sample Preparation	
Large plate/element checker	Fuel accident conditions simulation furnace (FACS) testing	Shielded sample preparation (SSPA)		

HFEF (Non-destructive Examination)	HFEF (Destructive Examination – Hot Cell)	IMCL – PIE on Irradiated Materials	EML	Analytical Lab
	Metal Waste Form Furnace (MWFF)			
	Hot Fuel Dissolution Apparatus (HFDA)			
	DEOX Furnace			
	Oxide reduction furnace			
	Electrorefiner Furnace			
	Tensile, compression, and bend (Instron) testing			

Presenter Information

DOE/National Labs

Kemal Pasamehmetoglu, Associate Lab Director, Nuclear Science & Technology, INL

Rita Baranwal, GAIN Director, INL

Daniel Wachs, Scientific Lead-Transient Testing, INL

Paul Demkowicz, (rep for HTGR NTD), INL

Dave Grabaskas, (rep for FR NTD), ANL

Joel McDuffee, (rep for MSR NTD), ORNL

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Nuclear Regulatory Commission

Jeffrey Schmidt, Office of New Reactors, NRC

Steven Bajorek, Senior Technical Advisor for Thermal-Hydraulics, NRC

INL Tour Agenda

Monday May 1, 2017

INL Tour of ATR, TREAT, and MFC: Tour Guide – Ryan Weeks

7:30 Meet at the Willow Creek Building, 1955 Fremont, Idaho Falls, Idaho, for Badging

8:00 Board Bus - Depart INL - Travel to the Advanced Test Reactor (ATR)

9:00 ATR - Check In, Tour, Presenter - TTAF Staff/TBD

11:15 Board Bus - Travel to Experimental Breeder Reactor No. 1 (EBR-1)

11:45 Lunch at EBR-1: The first power plant to produce electricity using atomic energy

12:45 Board Bus - Travel to the Transient Reactor Test (TREAT)

1:15 TREAT - Check in, Tour, Presenter - Dave Broussard

2:15 Board Bus - Travel to the Materials and Fuels Complex (MFC)

2:30 Check In, Tour

Group 1

2:30 Hot Fuel Examination Facility (HFEF)
Presenter - Jason Harp

3:15 Irradiated Materials Characterization Laboratory (IMCL)
Presenter - Brandon Miller

4:00 Experimental Fuels Facility (EFF)
Presenter - Tim Hyde

Group 2

2:30 Experimental Fuels Facility (EFF)
Presenter - Tim Hyde

3:15 Hot Fuel Examination Facility (HFEF)
Presenter - Jason Harp

4:00 Irradiated Materials Characterization Laboratory (IMCL)
Presenter - Brandon Miller

4:45 Tours Conclude

5:00 Board Bus - Return to Idaho Falls

5:45 Bus Arrives at Willow Creek Building

Poster Session

Tuesday, May 2, 2017

	Title	*Author(s)*	*Presenter*
1	INL Real Time Fuel Monitoring Capabilities	*Dawn M. Scates, Edward Reber*	Dawn Scates
2	The Fuel Accident Condition Simulator (FACS) Furnace for High Temperature Performance Testing or Irradiated Fuel	*Paul A. Demkowicz, Dawn M. Scates, Edward L. Reber, Les Scott, David V. Laug*	Paul Demkowicz
3	TREAT Transient Shaping	*Nicolas Woolstenhulme*	Nicolas Woolstenhulme
4	Static Capsules for TREAT Irradiations	*Nicolas Woolstenhulme*	Nicolas Woolstenhulme
5	ATR Transient Testing Capabilities	*Nicolas Woolstenhulme*	Nicolas Woolstenhulme
6	TREAT Real-Time Fuel Motion Monitoring System (FMMS)	*David Chichester, James Johnson, Scott Watson, Jay Hix, Sam Bays, Scott Thompson*	David Chichester
7	Ultra-High Temperature Thermocouples for In-Pile Applications	*Joe Palmer, Richard Skifton, Joshua Daw*	Joe Palmer
8	The Transient Reactor Test Loop (TRTL) Facility	*Daniel P. LaBrier, Wade R. Marcum*	Daniel LaBrier
9	Mechanical Properties on Fuel and Structural Materials	*J.L. Schulthess, K. Wachs, M. Heighes, R Lloyd*	Jason Schulthess
10	Transient Testing of Advanced Fuel and Cladding Material Concepts Under Conditions Relevant to DBA and BDBA Scenarios	*K. Linton, M.N. Cinbiz, N. Brown, Y. Yan, K. Terrani*	Kory Linton
11	In-Pile Instrumentation for Fuel Safety Testing	*Colby Jensen*	Colby Jensen
12	In-Pile Instrumentation for Fuel Safety Testing	*Colby Jensen*	Colby Jensen
13	Post Irradiation Blister Testing	*Adam Robinson, Francine Rice, David Sell*	Adam Robinson

Agenda

Tuesday, May 2, 2017

7:30	*	**Registration / Coffee**	**EIL Conference Room**
8:00	*1*	Welcome, INL Overview	Kemal Pasamehmetoglu
8:30	*2*	Safety Share, Introductions, GAIN Overview	Rita Baranwal
9:00	*3*	Status of Transient Testing Capabilities in the US	Dan Wachs
9:30		**Break**	
		LWR Fuel Safety Research Needs	
10:00	*4*	LWR Stakeholder Survey Results and Gap Analysis	Ken Yueh
11:00		Discussion	
12:00		**Working Lunch: Poster Session**	
		Industry Perspectives	
1:30	*5*	Regulatory	Jeffrey Schmidt
3:00		**Break**	
		Fuel Vendor Safety Research Needs to Support Current LWR Fuel Designs	
3:30	*6*	Westinghouse – Webcast	Paolo Ferroni
4:00	*7*	AREVA	John Strumpell
4:30	*8*	Lightbridge	James Malone
5:00		Wrap-up Discussion	Dan Wachs / Lori Braase
5:30		**Adjourn**	

***Presentation number**

Agenda

Wednesday, May 3, 2017

7:30	*	**Registration / Coffee**	**EIL Conference Room**
		Advanced Reactors Fuel Safety Research Needs	
8:00	*9*	Advanced Fuels Campaign Overview	Jon Carmack
8:45	*10*	Molten Salt Reactor Survey Results and Q&A	Lance Kim
9:30		**Break**	
10:00	*11*	Fast Reactor Survey Results and Q&A	Jake DeWitte
10:45	*12*	High Temperature Gas Reactor Survey Results Q&A	Farshid Shahrokhi
11:30	*13*	**Working Lunch: INL Demonstration and Deployment Capabilities**	**Corey McDaniel**
1:00	*14*	MSR/FHR Gap Analysis, Path Forward, and Q&A	Joel McDuffee
1:45	*15*	FR Gap Analysis, Path Forward, and Q&A	Bob Hill
2:30	*16*	HTGR Gap Analysis, Path Forward, and Q&A	Paul Demkowicz
3:15		**Break**	
3:45	*17*	Regulatory Requirements for Transient Testing	Steve Bajorek
4:15	*18*	TREAT Experiment Database (TREXR)	Art Wright
4:45		Discussion: Q&A, Short-term Actions	Dan Wachs Lori Braase
5:30		**Adjourn**	

***Presentation number**

Agenda

Thursday, May 4, 2017

7:30	*	**Registration/Coffee**	**EIL Conference Room**
		International Fuel Safety Research and Testing Capability	
8:00	*19*	Halden Reactor Project - Webcast	Wolfgang Wiesenack
9:00	*20*	CABRI International Program	Mark Petit, IRSN
10:00		**Break**	
10:30	*21*	Nuclear Safety Research Reactor (NSRR) status, capabilities, and hot cell facilities	Takeshi Mihara, JAEA
11:30	*22*	Belgium Reactor 2 (BR2)	Brian Boer, SCK-CEN
12:30		Collaboration Opportunities / Discussion	Dan Wachs / Lori Braase
1:00		**Adjourn**	

***Presentation number**

Attendees

First Name	Last Name	Company
John	Alvis	ANATECH
Stephen	Bajorek	Nuclear Regulatory Commission
Michelle	Bales	Nuclear Regulatory Commission
Rita	Baranwal	Idaho National Laboratory
Samuel	Bays	Idaho National Laboratory
Brian	Boer	SCK-CEN Belgian Nuclear Research Centre
Jeffrey	Borkowski	Studsvik Scandpower
Robert	Boston	Department of Energy - ID
Lori	Braase	Idaho National Laboratory
Jon	Carmack	Idaho National Laboratory
Alison	Conner	Idaho National Laboratory
James	Cook	Idaho National Laboratory
Paul	Demkowicz	Idaho National Laboratory
Jacob	DeWitte	Oklo
Timothy	Drzewiecki	Nuclear Regulatory Commission
Janelle	Eddins	Department of Energy - HQ
Darryl	Gordon	AREVA
Hans	Gougar	Idaho National Laboratory
Dave	Grabaskas	Argonne National Laboratory
Michael	Hanson	Elysium Industries
Steven	Hayes	Idaho National Laboratory
Brenden	Heidrich	Idaho National Laboratory
Bruce	Hilton	TerraPower
Jay	Hix	Idaho National Laboratory
Mark	Holbrook	Idaho National Laboratory
David	Hummel	Canadian Nuclear Laboratories

First Name	Last Name	Company
John	Jackson	Idaho National Laboratory
Colby	Jensen	Idaho National Laboratory
James	Johnson	Idaho National Laboratory
Dennis	Kaiser	Idaho National Laboratory
David	Kamerman	Department of Energy - ID
Rory	Kennedy	Idaho National Laboratory
Hussein	Khalil	Argonne National Laboratory
Lance	Kim	Southern Research
Teresa	Krynicki	Idaho National Laboratory
Dan	LaBrier	Oregon State University
Kory	Linton	Oakridge National Laboratory
Charles	Maggart	Department of Energy - ID
James	Malone	Lightbridge Corp.
Cristian	Marciulescu	Electric Power Institute
Joel	McDuffee	Oakridge National Laboratory
Takeshi	Mihara	Japan Atomic Energy Agency
Wayne	Moe	Idaho National Laboratory
Rob	O'Brien	Idaho National Laboratory
Joe	Palmer	Idaho National Laboratory
Pete	Pappano	X-energy
Kemal	Pasamehmetoglu	Idaho National Laboratory
Marc	Petit	IRSN
Edward	Pheil	Elysium Industries
Ian	Porter	Nuclear Regulatory Commission
Jorge	Ramos Herrera	CFE
Robin	Rickman	Terrestrial Energy USA
Adam	Robinson	Idaho National Laboratory
Dawn	Scates	Idaho National Laboratory

First Name	Last Name	Company
Jeff	Schmidt	Nuclear Regulatory Commission
Jason	Schulthess	Idaho National Laboratory
Suibel	Schuppner	Department of Energy - HQ
Farshid	Shahrokhi	AREVA
Sandra	Sloan	BWXT Technologies, Inc.
John	Strumpell	AREVA
Catherine	Thiriet	Canadian Nuclear Laboratories
Scott	Thompson	Idaho National Laboratory
Dan	Wachs	Idaho National Laboratory
William	Watson	Department of Energy - ID
Scott	Watson	Idaho National Laboratory
Joe	William	Nuclear Regulatory Commission
Nicolas	Woolstenhulme	Idaho National Laboratory
Arthur	Wright	Argonne National Laboratory
Ken	Yueh	Electric Power Institute