2017 NE-4 Nuclear Technologies Research and Development (NTRD) Achievements Report

March 2018

INL/EXT-18-44372

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy

2017 NE-4 Nuclear Technologies Research and Development (NTRD) Achievements Report

March 2018

INL/EXT-18-44372

Compiled and edited by:
Bonnie Hong (bonnie.hong@inl.gov)
Lisa Aldrich (lisa.aldrich@inl.gov)

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

CONTENTS

FIGURES

Fuel Cycle Technologies 2017 Achievements Report

1. INTRODUCTION

Fiscal year (FY) 2017 initiated new and reestablished presidential and departmental nuclear energy priorities for the United States (U.S.) Department of Energy (DOE), Office of Nuclear Energy (DOE-NE) that will:

- Support the Administration's priorities to achieve U.S. energy dominance and economic prosperity through U.S. technology leadership
- Support DOE-NE mission priorities and goals for the existing nuclear fleet, advanced reactor pipeline and national fuel cycle infrastructure
- Focus on early stage research, and effective cooperation with industry, academia, and other government agencies
- Propose investments to maintain and improve capabilities and infrastructure to conduct leading-edge science, research, and technology development.

DOE-NE's mission priorities will focus on the existing nuclear fleet, advanced reactor pipeline, and fuel cycle infrastructure. Therefore, the DOE Office of Nuclear Technology Research and Development (NE-4) mission aligned to (1) support the research and development (R&D) of innovative reactor technologies that may offer improved safety, functionality, and affordability and (2) conduct research to reduce long-term technical barriers for advanced nuclear energy systems (NES), including space fission systems. Key activities to execute this mission include:

- R&D planning to support the building of a versatile advanced test reactor by 2026.
- R&D of advanced nuclear fuels covering the entire spectrum of existing and advanced NES.
- Analysis of fuel cycle system options, assessment of overall performance under various scenarios, and improvement of the understanding of complex interdependencies.
- R&D of advanced material recovery and waste form technologies that improve current fuel cycle performance through minimizing processing, waste generation, and potential for material diversion.
- Assessment of security vulnerabilities of advanced fuel cycles and development of management and safeguards technologies and systems to address the risks.

To achieve the NE-4 mission priorities, two major programs with nine R&D campaigns are engaged in impactful research, development, demonstration, and deployment:

Fuel Cycle Technologies (FCT) Program

- Fuel Cycle Options (FCO) Campaign evaluates complex fuel cycle options and provides performance metrics on various systems and scenarios to improve the knowledge of interdependencies between technologies and different systems.

- Advanced Fuels Campaign (AFC) performs science-based R&D on accident tolerant fuels in addition to advanced light-water reactor (LWR) fuel and transmutation fuel.
- Joint Fuel Cycle Studies (JFCS) is a collaboration with the Republic of Korea to assess the feasibility and nonproliferation of electrochemical recycling to manage used fuel.
- Material Recovery and Waste Form Development (MRWFD) Campaign researches advanced fuel cycle material recovery and waste management capabilities to improve fuel cycle performance with less processing and waste generation.
- Material Protection, Control, and Accountability Technologies (MPACT) Program develops tools and capabilities to secure the next generation nuclear materials management and safeguards for nuclear fuel cycles.

Advanced Reactor Technologies (ART) Program

- Fast Reactors Campaign targets benefits in either resource utilization or waste management and possesses favorable features for small reactor applications and plutonium management.
- Gas-Cooled Reactors Campaign seeks to raise the technological readiness of high temperature gas-cooled reactors (HTGRs) and other reactor concepts utilizing coated particle fuel, graphite, or helium coolant by performing high risk, enabling research and development that commercial developers are unable or unwilling to undertake in the current energy market.
- Molten Salt Reactors (MSR) Campaign is a new campaign within ART that pursues potential lower cost energy, flexible fuel loadings, and positive impact for advanced fuel cycles.
- ART Energy Conversion (EC) Project provides solutions to convert heat from an advanced reactor to useful products that support commercial application of the reactor designs.

Fuel Cycle Technologies Program

2. FUEL CYCLE OPTIONS CAMPAIGN

Temitope Taiwo, Argonne National Laboratory (ANL), National Technical Director (NTD)
Brent Dixon, INL, Deputy NTD

2.1 Overview

2.1.1 Mission

The FCO Systems Analysis and Integration Campaign performs analyses and studies, interacts with all Nuclear Technology Research and Development (NTRD) campaigns, and provides an integrated NES view that will:

- Support strategic planning for NE-4 by helping to refine research goals and focus R&D activities of the NTRD program
- Determine the technical and economic feasibility of advanced NES to inform NE-4 program development, planning, and budget formulation
- Contribute to integration of, and the basis for, NE-4 activities by identifying technology maturation needs and pathways and informing decisions on R&D infrastructure needs in a systematic manner.

This mission also supports national energy security needs for a diversified energy portfolio, which includes nuclear power for the long-term.

2.1.2 Campaign Objectives

The campaign has the following primary objectives.

1. Facilitate integration of the NE-4 R&D portfolio and strategy by working with NTDs and federal managers and directors. Inform R&D prioritization, including documentation that supports transparent decision making for R&D investments.
2. Analyze specific NES to identify benefits and challenges - leading to development of common technology R&D goals. Contribute to a broader understanding of the required characteristics of NES that are capable of providing substantial improvements with respect to the current U.S. once-through fuel cycle option.
3. Develop tools, models and processes to maintain leading-edge capability in the world and ensure top-notch analyses of steady-state and dynamic behaviors of NES, including quantifying impact of performance improvements, and studying issues associated with transition and deployment.
4. Support assessment of programmatic risks of ongoing R&D by developing and facilitating the application of the Technology and System Readiness Assessment (TSRA) process by the R&D campaigns.
5. Develop improved fuel cycle economic analysis capabilities, providing results of greater credibility mainly by analyzing historical economic data to inform on projected costs of reactors and other fuel cycle facilities, increasing modeling capabilities, and continuing improvement of supporting economic data.

6. Develop understanding of the potential role of nuclear power in the domestic and global energy market considering alternative energy sources, the effects of economics and other external issues such as energy policies and structure of the unregulated market.
7. Communicate the results of campaign activities - to help explain what DOE NE-4 is doing and why the R&D is being performed.

In FY 2017, the campaign activities were organized into three general areas, namely Campaign Management, Equilibrium System Performance (ESP) and Development, Deployment and Implementation Issues (DDII), consistent with the Fuel Cycle Options Campaign objectives listed above (see Figure 1).

The campaign management area provides national technical leadership for the FCO Campaign activities, by: (1) planning and managing the activities in the campaign, (2) facilitating integration of R&D activities with the other DOE NTRD program campaigns, and (3) providing support to the Federal managers in managing and executing activities to achieve program objectives. The technical analyses within the FCO campaign were conducted under the ESP and DDII work areas. The features and performance of alternative end-state NES are studied in the ESP work area. The issues and challenges of transitioning to and deploying the end-state systems, along with the competitiveness of nuclear in the future energy fix are studied in the DDII work area.

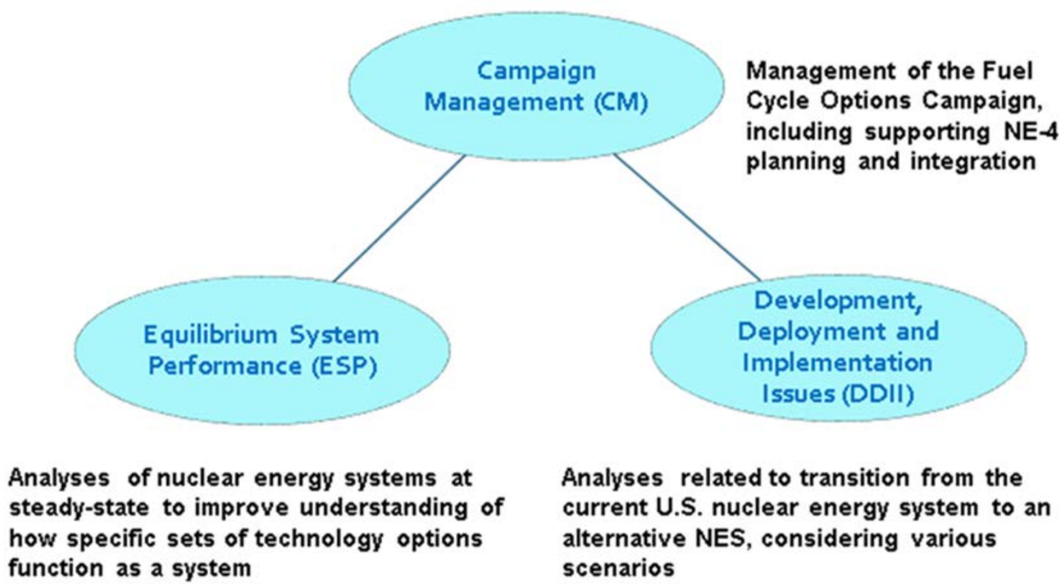


Figure 1. FCO Campaign organization by work areas.

2.1.3 Key FY 2017 Deliverables

Key deliverables for the campaign summarized results in each of the two technical areas. In FY 2017, the key deliverables/reports are:

ESP Deliverables

- Fuel Cycle and Nuclear Technology Seminars for DOE-NE – Molten Salt Reactors (3/31/2017)

- Technology-Specific Performance of Most Promising Fuel Cycles (8/1/2017)
- Compatibility of Nuclear Technologies with Variable Grid Demand (9/1/2017)
- Fuel Cycle Concepts - FY 2017 Update (9/15/2017)
- Cost Estimation Algorithm for Advanced Nuclear Reactor Concepts (9/30/2017)
- Advanced Fuel Cycle Cost Basis - 2017 Edition (9/30/2017).

DDII Deliverables

- Transition Analyses to Uranium/Transuranic (U/TRU) Continuous Recycle in Fast and Thermal Reactors (4/28/2017)
- Report on Transition to a Second Generation Nuclear Fuel Cycle (9/29/2017)
- Updated Analysis Results for the Most Promising Fuel Cycle Options (Evaluation Groups 23, 24, 29, and 30) (6/30/2017)
- Market Penetration of Nuclear Power under Various Technology and Climate Change Policy Scenarios (8/31/2017)
- Lessons Learned from Trial Application of the TSRA Process to Example Metallic Fuel and Aqueous Reprocessing Systems (9/29/2017).

The following sections provide representative highlights of Fuel Cycle Options Campaign work completed during the fiscal year.

2.2 Equilibrium System Performance

In the *Equilibrium System Performance* work area, campaign personnel perform analyses of NES to improve the understanding of how specific sets of technology options function as a system. This includes working with other campaigns to consider the effects of a range of specific implementing technologies for all parts of the nuclear energy system. It also includes the enhancement of the tools and capabilities for such analyses, and development and maintenance of analysis data and libraries. Highlights from these studies are provided in the following subsections.

2.2.1 Analyses of Advanced Fuel Cycle Options - FY 2017 Update

Data on fuel cycles utilizing innovative fuel and reactor concepts developed by industries, universities, and national laboratories, and fuel cycles pursued by several foreign countries are being evaluated for the purpose of informing DOE-NE and to check if the claimed improvements are consistent with the findings of the fuel cycle evaluation and screening (E&S) study, www.fuelcycleevaluation.inl.gov. Information on 15 additional fuel cycles was collected in FY 2017, which included eleven fuel cycles utilizing innovative fuel and advanced reactor concepts, and four foreign national fuel cycles.

It was observed that advanced fuel concepts are under development for the current commercial LWR or HWR fleets. The innovative HWR fuel concepts mainly target recycle of uranium recovered from LWR used fuels and improvement of HWR core performance. Examples include the *directly recycled uranium* and *natural uranium equivalent* approaches. The major purpose of the innovative LWR fuels reviewed is the recycle of Pu in the current LWR fleet without major changes to fuel management schemes and hardware. The four LWR fuels reviewed in FY 2017 were REMIX (Regenerated mixture), CONVERT, TOP-MOX, and $(Th,Pu)O_2$. As a result of enriched uranium in the REMIX fuel mix, the Pu content in the fuel is saturated lower than 2% and the core safety features are comparable to those with conventional UO_2 fuel. With increasing recycle stages, the Pu content in the three other fuel concepts increases with potential safety implications, because of the use of depleted uranium in the original fuel mix.

The information on fuel cycle options being proposed by developers in France, South Korea, Japan, and the United Kingdom was reviewed. It was found that the proposed national nuclear fuel cycles are dependent on the energy policies of the countries and that the fuel cycles have been developed prior to the current political administrations in those countries. Thus, revision of some of the fuel cycles is possible if the energy policy of a current administration is different from that of a previous administration. Finally, it was found that the claimed fuel cycle performance characteristics of the advanced fuels and fuel cycle concepts are consistent with the findings of the E&S study.

Under this activity, the campaign supported the DOE NE-4 program office by developing white-papers in quick turn-around studies, e.g., (1) supported the response to DOE S-1 office request on the potential roles of molten salt reactors for consuming LWR used fuel, (2) developed information on U.S. advanced reactors for a brochure used at the 2017 International Atomic Energy Agency (IAEA) General Conference (see Figure 2), (3) developed a white-paper on the benefits of nuclear energy R&D.

Figure 2. FCO Campaign supported development of data for brochure used at the IAEA.

2.2.2 Fuel Cycle and Nuclear Technology Seminar on Molten Salt Reactors

The FCO Campaign developed presentation material and held an informational seminar on MSRs in Germantown, MD, February 16, 2017, to provide detailed technical background on molten salt reactors, the history of molten salt reactor development, the current status of the technologies, and the identified issues requiring R&D. The objectives of the seminar were to provide:

- Information on the concepts, technologies, fuel cycles, and challenges associated with MSRs, including a review of various domestic and international, historic and current MSR technologies and the reasons behind the recent interest in MSR technologies, and,
- Summary of the current status of MSRs and the major challenges to development and deployment.

The seminar was organized into a series of modules, with each module providing information on specific aspects of MSRs. The seminar was attended by about 50 DOE-NE and U.S. Nuclear Regulatory Commission (NRC) personnel.

2.2.3 Compatibility of Nuclear Technologies with Variable Grid Demand

The mix of deployed electricity generation technologies in the U.S. is evolving from baseload and dispatchable peaking generation to also include more non-dispatchable intermittent generation like wind and solar. This work discussed the changes expected on U.S. grid markets as the fraction from variable renewable energy (VRE) production is increased, and the options for nuclear energy to adapt to an increasingly variable grid demand. Nuclear power is almost exclusively a baseload electricity generation technology in the U.S., but is used for load-following in some other countries. While nuclear energy is the largest reliable source of low-emission electricity that is also extendable, it is currently under economic stress in the U.S. due to the low price of natural gas and to the penetration of subsidized and low-marginal cost variable renewable electricity.

The study confirms that the grid market will tend to see an increasing number of both small and fast variations and large and slow variations as wind and solar VREs are installed. These variations are caused by the intermittent characteristics of electricity generation from wind and solar renewables. The increasing installed capacity of VREs cannot fully replace the existing installed capacity of dispatchable production, including nuclear, as those will still be needed in the absence of variable renewable production, but with lower average load factors (as operational backup).

The technical compatibility of advanced nuclear technologies to conduct load following operations was considered in this work. When comparing with the proven load following performance of pressurized water reactors (PWRs), qualitative trends were identified for different advanced nuclear reactor design options. Most nuclear reactor technologies have the potential to allow some load-following operation and could be optimized to increase their maneuverability. Due to differences in physics, some advanced reactor technologies like fast reactors, could be particularly well suited to provide higher flexibility. While load-following for existing nuclear reactors is a way to increase revenue in an electricity market with fluctuating electricity prices due to the penetration of VREs, the new revenue levels will still likely be lower than that enjoyed with low variable renewables and higher-priced natural gas, which may make new reactor builds more difficult to justify economically. Reducing the power load of a nuclear reactor for stabilizing the power grid could be avoided by implementing electricity storage or with hybrid systems. The availability of storage would further increase the profitability of the major low greenhouse gas emitting energy producing technologies, whether it is nuclear or variable renewables. Separately, hybrid systems coupling industrial processes and nuclear reactors can be designed to provide better performances than that of each subsystem separately, while having a stabilizing effect on the price of electricity.

One question not addressed in this study is whether expanded market share for VREs backed up with large investments in flexible dispatchable generation is the most cost-effective and reliable approach to achieving broader goals such as lower carbon emissions and increased sustainability, or if other market mix solutions may achieve the same goals without the added market volatility.

2.2.4 Cost Estimation Algorithm for Advanced Nuclear Reactor Concepts

An algorithm to estimate the capital cost of any nuclear reactor design was development in FY 2017, and its application to the construction cost of an advanced reactor concept was provided as example of its use. First, a reference design (a standard PWR) was adopted for which detailed and defensible cost information were found based on historical data. Then, the individual components of the reference PWR were sorted in decreasing order of importance, in terms of the fractional contribution of each to the total direct cost of the plant; and cost models were developed for each of the most expensive components, including all those that provide a contribution to the total direct cost larger than 2%, with a cumulative contribution to the direct cost of about 70%. By focusing primarily on the most expensive components, it is possible to tailor the algorithm to the desired degree of fidelity, at the expense of larger efforts for more robust estimates. For several components, detailed cost models were developed in this work. For example, a detailed bottom-up cost models for the containment building was developed which is based on extracted unit costs for the labor and material required for the installation of all the major structures, from formwork to rebar and cadwelds, and on the geometrical parameters of the building. This approach can be used to perform cost estimates of reactor buildings of any size and shape. Similarly, detailed analyses of actual construction data (both nuclear and non-nuclear) were performed to establish cost-estimating methods for other components, such as the steam turbine generator. Finally, an approach to derive the total overnight costs based on the calculated direct costs was developed, based on the known historical relationships between direct costs and the other large categories of construction expenditures: namely indirect, owners' and contingencies costs.

The cost model and the associated algorithm were tested on the ABR1000 reactor design as an example. The results showed that the ABR1000 is expected to have a total overnight cost about 30% lower than that of the reference PWR. However, since the power level of the ABR1000 is substantially smaller than that of the reference PWR, its unit overnight cost is expected to be higher. The algorithm allowed the identification of which cost components are likely to be more expensive and of which are likely to be less expensive, and by how much, for each alternative design studied, thus potentially providing insight into the cost drivers of various reactor technologies. This work, for example, allowed the identification of the vessel head as a likely major cost driver for fast reactors with designs similar to the ABR1000, potentially informing R&D decision makers on the most effective areas of R&D for potential reduction of the construction cost of advanced reactor designs.

This work is expected to continue with several objectives, including the extension of the model to a larger set of components, and to other components not in LWRs, but present in other advanced reactor designs. It will also be extended for other nuclear fuel cycle facilities, such as reprocessing plants or remote fuel fabrication facilities, and for small modular plants, for which a high degree of standardization and possibly factory fabrication may reduce the need for indirect costs as compared to standard plants. The work can also be extended with the purpose of evaluating the uncertainties in the cost estimating process, both for the cost estimating relationships and unit costs, to obtain a quantification of the uncertainties in total capital investment cost and project risk.

2.3 Development, Deployment, and Implementation Issues

In the *Development, Deployment, and Implementation Issues* work area, the campaign conducts analyses related to transition from the current U.S. NES to an alternative NES, considering various deployment and implementation scenarios. This includes implementation and application of the TSRA process on NES technologies currently being developed in the NTRD program, the analyses of nuclear energy in the context of the overall energy system, and the enhancement of tools and capabilities for NES analyses.

2.3.1 Transition to a Second Generation Nuclear Fuel Cycle

While the reactor portion of the nuclear energy system is now in its third technical generation, the total system including the rest of the fuel cycle is still in its first generation, with some of the initial infrastructure approaching end-of-life. This first generation *once-through* fuel cycle has inherent inefficiencies as it involves only using the fuel material once, then disposing of it as waste. Less than 1% of the energy potential in the fuel material is utilized in this fuel cycle, which limits the sustainability of nuclear energy from both a natural resources perspective (uranium) and an environmental perspective (radioactive waste). Each of the most promising fuel cycles identified during the E&S study use fuel recycling and advanced reactor technologies to dramatically improve nuclear energy system efficiency, reducing uranium usage by two orders of magnitude while also reducing the long-term hazard from radioactive waste by similar amounts. If implemented, such a fuel cycle would firmly establish the technical sustainability of nuclear energy.

The transition study examined options for and constraints on implementing these second-generation fuel cycles, including the technologies needed and the timeframes for transitioning the nuclear energy system infrastructure (reactors, fuel facilities, recycling facilities, etc.). Some key findings are:

- The U.S. reactor fleet is aging and will need to be retired and replaced over the next 15-30 years (35-50 with additional life extension). This replacement cycle presents a unique opportunity to transition to a much more sustainable second-generation fuel cycle based on fast reactors or a combination of fast and thermal reactors that produce their own fissile material via continuous recycling of their used fuel.
- To transition to the most promising second-generation fuel cycles requires replacement of the current infrastructure and generation of an inventory of plutonium to fuel the new reactors. Two primary approaches are available to produce the required plutonium, starting and continuing to fuel new thermal reactors with low-enriched uranium (LEU, ~5% ^{235}U) until enough Pu is produced for use in the fast reactors, or starting fast reactors on LEU (up to 19% ^{235}U), then continuing their operation on recycled materials (no additional LEU). In either case, the used reactor fuel would be reprocessed to recover U/Pu or U/TRU. Starting fast reactors with enriched uranium (up to 19% ^{235}U) would effectively decouple the deployment of the fast reactor technology from the LWR LEU used nuclear fuel separations development and from current questions related to the management of spent nuclear fuel.
- Matching the changing demand for recycling during transition with the deployment of additional recycling facilities is necessary to optimize system costs.
- Plutonium is a valuable resource during transition so it should be used in Pu-producing reactors such as the fast reactors as early as possible instead of having it consumed in Pu-consuming reactors such as PWRs loaded with MOX fuel.

- For all scenarios envisioned, the fast reactor technology is needed as early as possible in order to maximize the benefits of transition; the earlier that LEU fuel is phased out, the less total uranium that needs to be mined and enriched.
- Life extensions that keep existing LWRs operational until after the fast reactor technology is available will lead to better transition performance. This is because the earlier the LWRs retire, the more likely they will be replaced by other LWRs that will operate and require enriched fuel for another 60–80 years.

2.3.2 Market Penetration of Nuclear Power under Various Technology and Climate Change Policy Scenarios

A broad range of analyses have been conducted by the FCO Campaign for investigating the current and future role of nuclear energy in the context of the evolving U.S. regional and global energy system. These analyses have been conducted with a long-term time horizon (hundred years) suitable for addressing energy technology R&D needs and interactions with climate change mitigation efforts. Modeling simulations of the nuclear energy system and pathways with alternative technology and climate change mitigation assumptions in this work explore the scale, timing and value of the nuclear energy system in contributing to the U.S. and global demand for energy. Several topics are explored spanning analyses conducted over multiple years. They include the regional and global impact to nuclear energy of climate change mitigation scenarios and availability of carbon capture and sequestration, renewables and energy storage technologies. Focus on nuclear energy issues include sensitivity assessments of nuclear capital costs, lifetime extensions and nuclear energy deployment and availability, traditional and non-traditional roles of nuclear energy, NES lifecycle analysis, and assessment of alternative nuclear fuel cycle and reactor technologies. The nuclear energy contribution is significant in meeting the future global demand for energy and nuclear energy has a particularly important and prominent role in supporting climate change mitigation efforts.

2.3.3 Lessons Learned from Trial Application of the TSRA Process

The TSRA process for R&D evaluation was developed in FY 2016 for the purpose of informing the DOE-NE planning and decision-making processes for the research and development of advanced NES and the implementing technologies. Follow-on work was conducted in FY 2017 to mature the TSRA process and conduct initial implementation to one or more technologies in the Advanced Fuels and MRWFD Campaigns. The systems used for the trial implementations of the TSRA process were different: (1) metallic LEU-10Zr fuel with HT-9 or SS-316 cladding incorporated in a fuel assembly for a fast reactor – very mature with a long history so no critical technology element was identified, and (2) dissolution component of an aqueous reprocessing system – still under development with several critical technology elements identified and assessed as an example. Despite the significant differences in the two examples, there was consensus on the following key characteristics of the TSRA process:

- Confirmed the value of the systematic/structured approach for performing TSRA.
- General agreement that the questionnaires for the technology readiness levels (TRLs) are comprehensive.
- General agreement that the process is useful/applicable to both mature systems and those under development (i.e., still in the R&D phase).

- Some clarification would be useful to better differentiate between TRLs when there appears to be overlap, and a complementary guide should be developed with definitions/criteria appropriate for the specific technology/system. The latter may be best done at a system/program level and could foster consistency and integration across research teams.

3. ADVANCED FUELS CAMPAIGN

Jon Carmack, INL, NTD

3.1 Overview

3.1.1 Mission

AFC performs research, development, and demonstration (RD&D) activities for advanced fuel forms (including cladding) to enhance the performance and safety of the nation's current and future reactors; enhance proliferation resistance of nuclear fuel; effectively utilize nuclear energy resources; and address the longer-term waste management challenges. This includes development of a state-of-the art RD&D infrastructure to support the development of advanced fuel systems using a goal-oriented science-based approach.

3.1.2 Objectives

AFC has been given the responsibility to develop advanced nuclear fuel technologies for the Fuel Cycle R&D program. The current focus is on the following:

- **Advanced LWR Fuels** with enhanced accident tolerance, improved performance, and increased utilization.
- **Advanced Reactor Fuels** with emphasis on actinide transmutation with enhanced proliferation resistance, resource utilization, and waste minimization in future fuel cycles.
- **Capability Development** in tools and techniques, such as advanced in-pile instrumentation, characterization, post-irradiation examination (PIE) and separate effects testing to generate data for advanced modeling and simulation activities.
- **Advanced Fuel Performance Modeling and Simulation** is essential to the development of advanced fuel systems. AFC interfaces with the Nuclear Energy Advanced Modeling and Simulation program to develop multi-scale, multi-physics fuel performance codes.

3.1.3 Challenges

- **Major Increase in Fuel Burnup, Performance, and Utilization over the Current Technologies.** An increase in fuel burnup is desired for all the fuel cycle options. However, the quantitative goals for burnup depend on the reactor type and, more importantly, selected fuel cycle options. In some cases, there are practical and economic limitations to burnup beyond the fuel cycle efficiency and technology limitations. Burnup in once-through nuclear fuel cycles is limited by the initial enrichment constraints and cladding material properties. Burnup for fuels under full recycle may be limited by reactor physics, storage, and/or disposal constraints after the discharge of spent fuel. Another important consideration in increasing the burnup is to ensure zero-failure, a standard for which industry strives at its current, moderate burnups. Quantitative limits for the burnup grand-challenge under various fuel cycle scenarios will be developed as the program progresses and fuel cycle scenarios are defined.
- **Low-loss Fuel Fabrication Processes.** The challenge for fabrication is to substantially lower the irretrievable losses from the current levels, which are typically on the order of 1%. This requires the development of cleaner and more efficient fabrication processes without imposing an economic

penalty on fuel fabrication. The objective is to generate less waste during the fabrication process and increase resource utilization. Such improved fabrication processes may also contribute to increased safety of plants and enhance safeguards and materials accounting.

- **LWR Fuels with Increased Performance and Enhanced Accident Tolerance.** Improvements will be measured by increased margin to fuel failure, increased response time during an accident to prevent severe damage to the core, and reduced hydrogen generation when the core is uncovered and the fuels and cladding are in contact with steam.

3.1.4 Major R&D Activities

The following are the major RD&D activities for next-generation LWR fuels, adjusted to the FY 2017 budget and program direction.

- **Advanced LWR Fuels with Enhanced Accident Tolerance**. Fundamental RD&D activities continued on accident-tolerant fuels (ATF) concepts; Phase 2, year 1 was completed and research activities were coordinated between DOE laboratories, industry funding opportunity announcement teams, university integrated research project teams and Nuclear Energy University Programs (NEUP).
- **Advanced Reactor Fuels**. Primary RD&D areas included advanced fabrication technology development; fabrication and characterization of minor actinide- and lanthanide-bearing fuels; fundamental property measurements and fuel cladding chemical interaction testing; and irradiation performance testing.
- **Capability Development**. Primary RD&D areas included advanced modeling and simulation (M&S) of fuel performance and fabrication processes; characterization technique development; and unique in-pile and out-of-pile material property measurements. Experimental transient testing capabilities were established for the Transient Reactor Test (TREAT) facility.
- **International Coordination and Collaboration**. AFC researchers are active in international collaborations with Korea, France, Japan, China, European Atomic Energy Community (EURATOM), and OECD-NEA. These interactions and collaborations are managed through a combination of participation in Generation IV Global International Forum (GIF) projects, International Nuclear Energy Research Initiative (INERI) projects, and participation in bilateral and trilateral government-to-government agreements. Active work has been executed in collaboration with the Korean Atomic Energy Research Institute (KAERI) and with the Halden reactor operated by the Institute for Energy Technology laboratory of Norway.

3.1.5 Major Achievements

The research and development staff of the Advanced Fuels Campaign have made significant impactful progress in the accident tolerant fuel and in the advanced reactor fuels programs. Major accomplishments and progress on key understanding of advanced nuclear fuels have been made and the development and establishment of key resources needed to generate data for qualification and licensing have been made.

3.1.5.1 Accident-Tolerant LWR Fuel

Accident-tolerant LWR fuel research made significant progress in FY 2017. The industry led projects have been accelerating their activities to support development and deployment of ATF technologies in active commercial reactors within the next 1 to 2 years. Laboratory led research and development of ATF technologies has been targeted to developing the material, physical, chemical, and thermal property data needed by the industry and regulatory bodies to qualify and license ATF technology in active commercial nuclear power reactors.

The AFC has produced initial data on steady state separate effects irradiation testing through the ATF-1 experiment series in the Advanced Test Reactor (ATR). In addition, in FY 2017, the AFC has developed significant in-pile irradiation testing capabilities and initiated LWR prototypic condition testing capabilities in the DOE Advanced Test Reactor. The sensor qualification test (SQT) was inserted into the ATR in a PWR prototypic condition chemistry loop, representing a major reduction in risk for the ATF-2 fueled test planned for the spring of 2018. Figure 3 is a photograph of the ATF-2 SQT experiment engineering and assembly team with the assembled SQT experiment. Data from the SQT experiment will be reported in FY 2018 following completion of the irradiation cycles.

Figure 3. Sensor qualification test engineering and assembly team.

Technical staff at the Advanced Test Reactor and at the Hot Fuel Examination Facility (HFEF) have successfully irradiated new accident tolerant fuel technology for light water reactors and provided initial material and physical property data on the irradiation performance of new and novel ATF technologies in the ATF-1 experiment test series. Specifically, the initial measurements of fission gas release and swelling of U_3Si_2 have been made in the HFEF on fuel irradiated in the ATR ATF-1 experiment series and has provided some of the first irradiation performance data on this fuel system in LWR conditions. This initial data has provided impact on critical design parameters for General Atomics and Westinghouse ATF fuel designs currently under development as light water reactor accident tolerant fuel. Initial data obtained from irradiation of fuel technology in the ATF-1 experiment series continues to provide data guiding design of ATF technology and has been incorporated into the design of ATF-2 fuel experiments. The ATF-2 SQT test train will generate critical performance data on in-pile instrumentation, thermocouples, linear variable differential transducers, and fuel extensiometers for the AFC ATF program. Direct comparison with data generated in the Halden test reactor will now be possible and applicability to future ATF technology development assessed. The facility modifications needed at ATR to support this test have been executed in a short time frame supporting the accelerating ATF program and are significant in that this loop will be the first LWR PWR condition coolant loop experiment performed in ATR.

The Severe Accident Test Station was installed in a hot cell at ORNL to allow the capability for loss of coolant accident (LOCA) condition testing of irradiated ATF technologies. The Severe Accident Test Station system is pictured in Figure 4 installed in the ORNL hot cell.

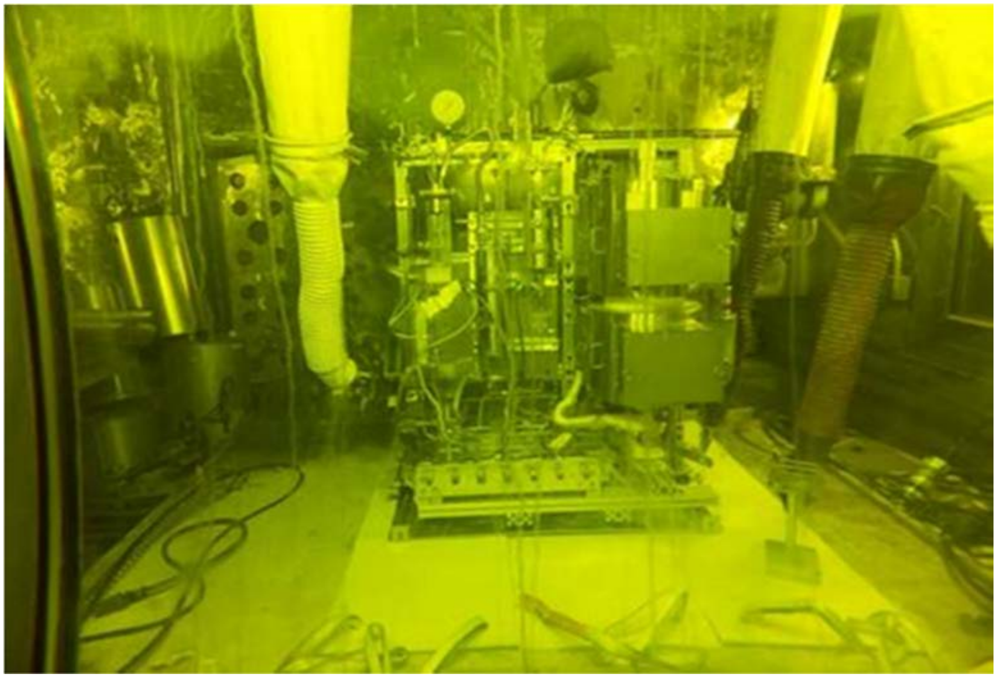


Figure 4. Integral loss of coolant accident test facility in shielded hot cell at ORNL.

The ATF program industry teams have made significant progress in FY 2017 towards insertion of lead fuel rods of ATF in active commercial nuclear reactors. All three teams plan on LFR insertions in the 2018 to 2019 timeframe. Uranium silicide (U_3Si_2) fuel pellets have been fabricated for Westinghouse-General Atomics irradiations at INL and FeCrAl C_26M alloy cladding has been fabricated for GE lead fuel rod insertions at ORNL. U_3Si_2, silicon carbide cladding, and FeCrAl cladding were all subjected to significant material property characterization testing and the results of such testing has been published in the first edition of material property handbooks for each of these material systems. Figure 5 is a photograph of the FeCrAl cladding fabricated at ORNL for insertion in the Hatch commercial nuclear reactor by GE. Figure 6 is a neutron radiograph of U_3Si_2 fueled experiments following irradiation in the ATR on behalf of Westinghouse. Figure 7 is a photograph of Cr_2O_3 additive UO_2 fuel pellets fabricated by AREVA for insertion in the ATR ATF-2 fueled experiment. All of these accomplishments are evident of the strong government-industry efforts needed to complete the research, development, and deployment of ATF technologies in active commercial nuclear power plants.

Figure 5. GE FeCrAl alloy cladding development at ORNL.

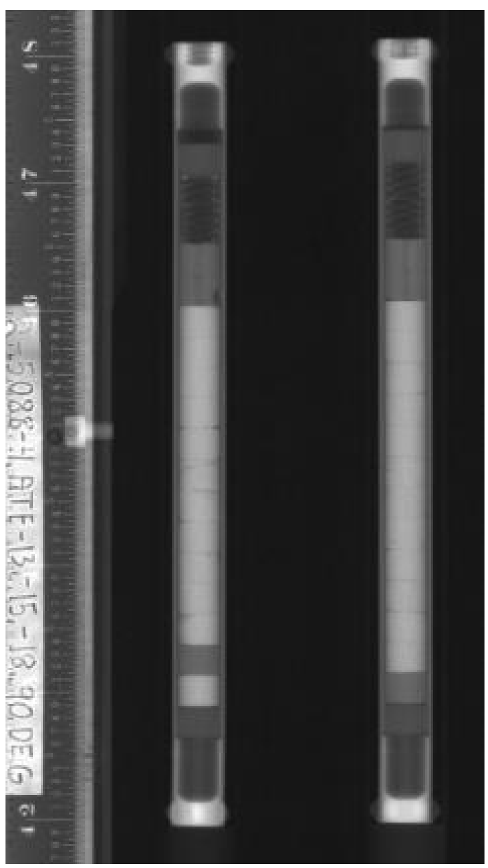


Figure 6. Westinghouse U_3Si_2 fueled rodlets fabricated by INL and removed from ATR.

Figure 7. AREVA Cr_2O_3-doped pellets for ATF-2 rodlets.

Scientists at Los Alamos National Laboratory are analyzing the responses of U_3Si_2 to steam and pressurized water at LWR operation temperatures as part of ongoing efforts to mitigate oxidation of high uranium density fuel powders, which should relax the air handling constraints of these fuels in a commercial environment (see Figure 8).

400°C steam 275°C air

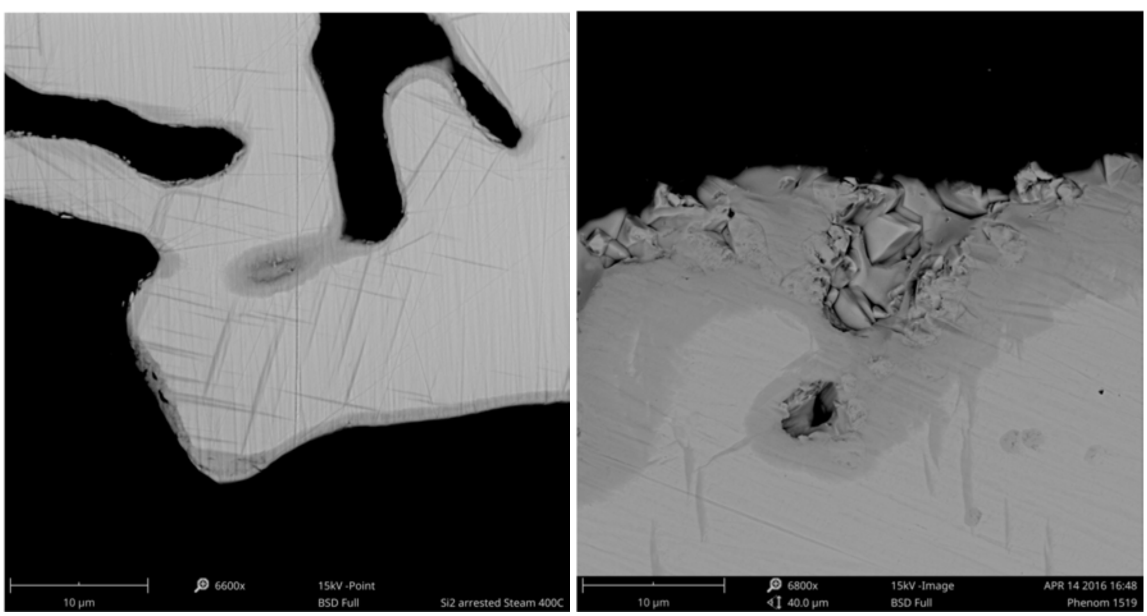


Figure 8. Comparisons of the microstructures of U_3Si_2 oxidized under H_20 and water.

3.1.5.2 Advanced Reactor Fuels

The Advanced Reactor Fuels program realized major achievements in FY 2017 in the areas of fabrication development, characterization of actinide-bearing composition fuels, irradiation testing of candidate fuel systems, and PIE of metallic fuel experiments.

Advanced fabrication development accomplishments include fuel cladding chemical interaction barrier development and the optimization of the fabrication process using casting fabrication methods. Fabrication of fuel incorporating integral zirconium fuel cladding chemical interaction barriers has been optimized to the point where irradiation testing is feasible.

In addition, demonstration of metallic fuel fabrication with integral Zr liner using extrusion was completed. INL has a complete engineering scale extrusion line, capable of extruding bare uranium alloys on the kilogram scale. Figure 9 is a photograph of the metallic fuel extrusion line at INL and an example starting billet prior to extrusion.

Figure 9. Photograph of metallic fuel extrusion line at INL with extruded fuel pin on the runout table (inset photo of starting billet and Zr can).

Major progress was made through the demonstration of remote casting of metallic fuels in the HFEF at INL. A DU-10Zr pin was remotely cast using the Casting and Sampling Furnace. Figure 10 is a photograph of a U-10Zr metallic fuel slug cast remotely in the HFEF Casting and Sampling Furnace.

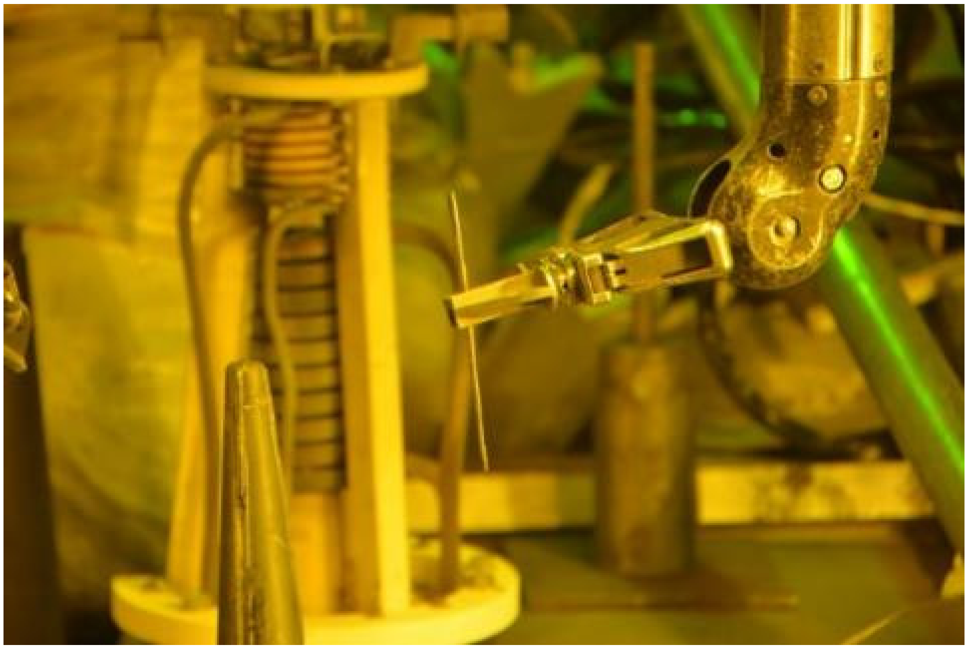


Figure 10. Remotely cast U-10Zr pin.

The first distillation run using Am/Pu feedstock was completed in the Fuel Manufacturing Facility at INL. Improving the efficiency of the neptunium oxide reduction process will provide vial neptunium metal feedstock and the Transuranic Breakout Glovebox provides the capability to access quantities and types of TRU material that were previously inaccessible. Figure 11 is a photograph of an INL research performing the reduction of neptunium oxide in a glovebox in the Analytical Laboratory at INL.

Figure 11. Neptunium oxide reduction at the Analytical Laboratory at INL.

A major step forward in microstructural characterization of irradiated nuclear fuel samples was accomplished through the first electron probe microanalyzer (EPMA) examination of the FUTURIX-FTA experiment was completed in Irradiated Materials Characterization Laboratory (IMCL). This analysis allows detailed quantitative analysis of constituent distribution and migration at the microstructural level. The Spectrum Comparison Report was issued documenting a comprehensive comparison of proto-typicality of thermal test reactor experiments with historical fuel performance data and knowledge. Staff at the IMCL have successfully installed, calibrated, and performed the first microstructural characterization of chemical speciation in a highly irradiated nuclear fuel sample. Installation of the EPMA instrument in a shielded environment with associated radiological containment has allowed research staff at the IMCL to generate the first radial analysis of elemental speciation on a high dose metallic fuel sample. Figure 12 provides example data from the first EPMA examination in IMCL of the FUTURIX DOE-1 sample. This type of data has not been available to researchers in the U.S. since 1994. Key phenomenological fuel behavior can now be discerned. As an example, the data gained through analysis of irradiated fuel on the EPMA will aid in understanding the difference between fuel irradiated in fast spectrum environments and fuel irradiated in thermal spectrum environments. This is a key issue for the research, development, and testing of nuclear fuel systems in the Advanced Test Reactor and applicability of data to prototypic advanced nuclear reactors of the future.

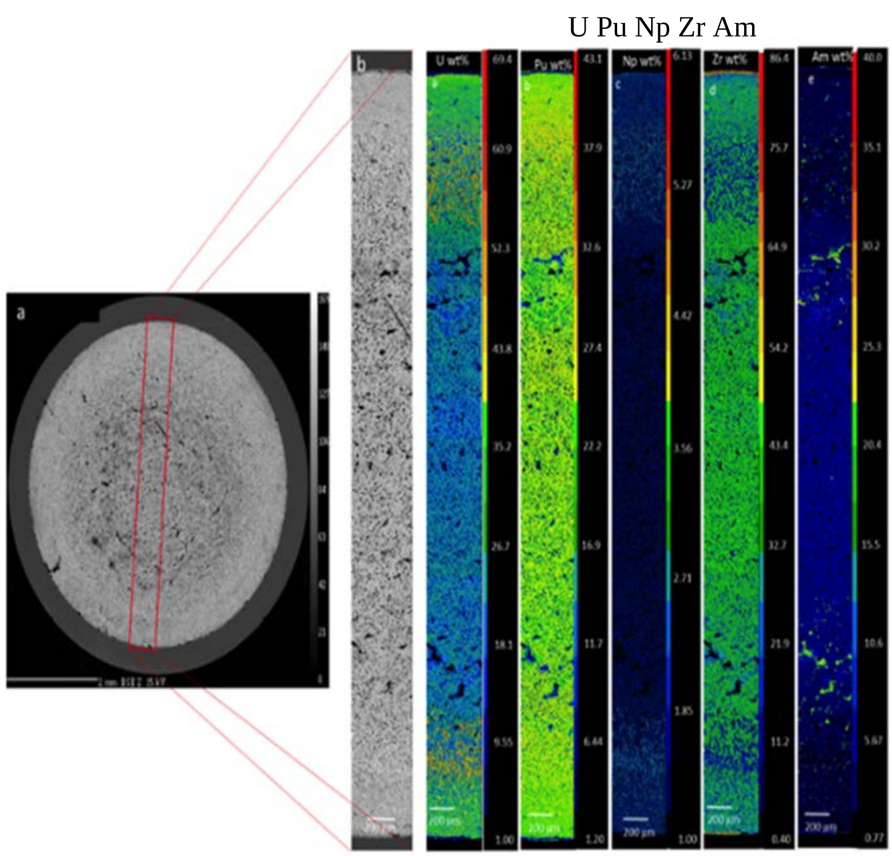


Figure 12. First EPMA examination in IMCL of FUTURIX DOE-1 sample.

Figure 13 shows a high-resolution photomicrograph of metallic fuel irradiated in the AFC-4C irradiation experiment. Evident in the photomicrograph are small precipitates formed between lanthanide fission products and additive elements. It is hoped that these precipitates will limit the mobility of fission products and result in reduction of fuel cladding chemical interaction.

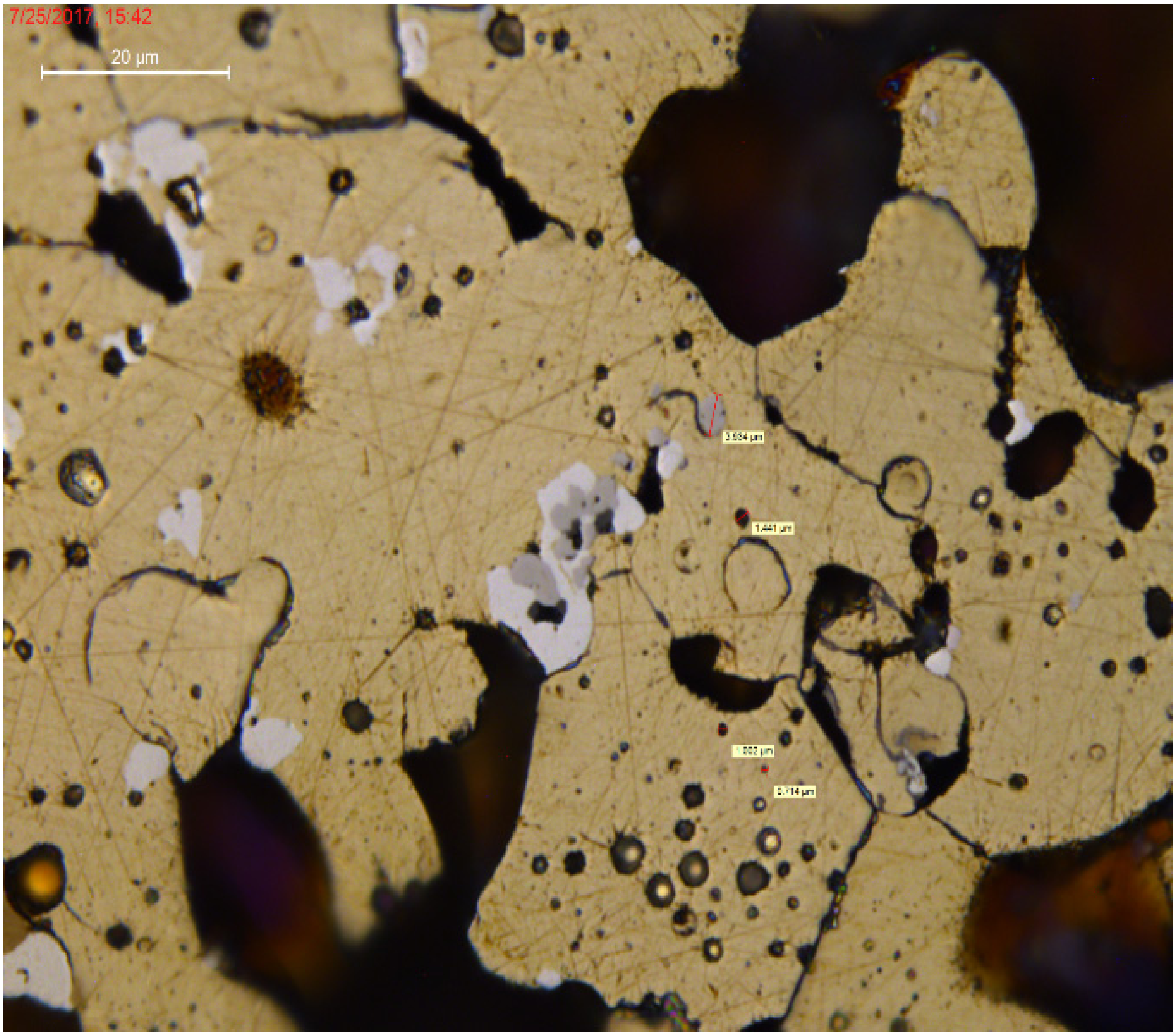
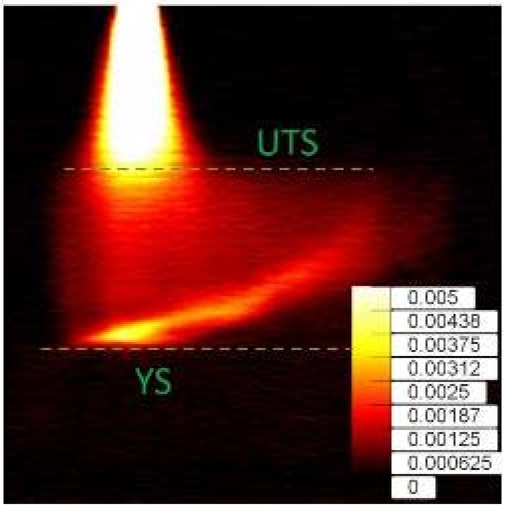


Figure 13. High-resolution image with measurements at the micrometer range in metallic fuel from AFC-4A.

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



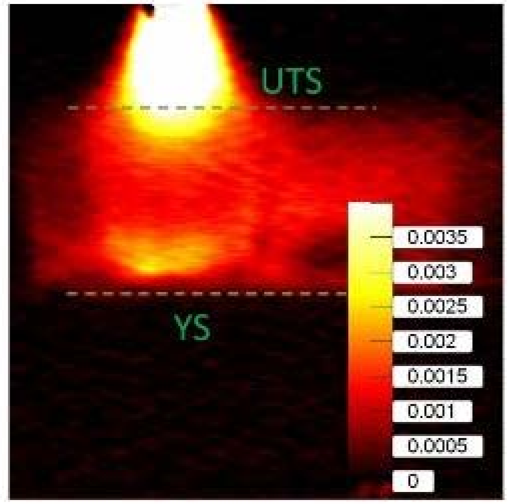


Figure 14. Strain rate maps from the stir zone and from the unaffected zone.

A more comprehensive list and description of FY 2017 accomplishments can be found in the Advanced Fuels Campaign 2017 Accomplishments report (INL/EXT-16-40127 Revision 2) available on the DOE Office of Science and Technology (OSTI) site.

3.1.6 Major Activities Planned for FY 2018

The coming year will be a challenging year. Fuel and cladding will be fabricated and inserted in the ATR ATF-2 PWR condition loop facility. Transient testing activities will be initiated in the TREAT transient test facility. The material property handbooks will be updated to reflect the material property characterization completed to date and the Nuclear Data Management and Analysis System (NDMAS) database will be finalized so that data generation from prototypic experiments will be archived and made available to needed design and assessment organizations. A full list of the major activities planned for FY 2018 is provided below:

- Issue a report documenting the development of the NDMAS ATF-2 database and verification of data quality.
- Fabricate ATF mini-fuel samples for High Flux Isotope Reactor irradiation.
- Initiate ATF miniature fuel irradiation in support of separate effects testing.
- Generate a report on the development of a characterization methodology for separate effects testing of irradiated ceramic fuels.
- Report on in-cell re-fabrication of irradiated rods and LOCA testing.
- ATF-2 assembled and quality inspected test train ready for transfer to ATR for insertion.
- Generate a report on the determination of the feasibility to waterproof UN for LWR use.
- Issue update to LWR SiC/SiC cladding handbook of properties.
- Issue update LWR FeCrAl cladding handbook of properties.
- Report documenting ATF PIE results obtained to date.

- Extrude annular U-Zr fuel.
- IRT-1 fabrication complete and approved for ATR Cycle 164B.
- PIE Report on AFC-3C, 3D, and 4A.
- Evaluate alternative AFC irradiation capsule designs in ATR.
- Report on BISON fuel performance code development and application to fast reactor fuels.
- Fuel Safety Research Plan for metallic fuels (i.e., transient testing).
- Report on status of Am and Np purification processes and future directions.
- Report on tensile testing of high dose irradiated AR cladding materials.
- Complete Phase II qualification of thermal conductivity microscope for IMCL.

4. JOINT FUEL CYCLE STUDIES

Mike Goff, INL, NTD
Ken Marsden, INL, Deputy NTD

4.1 Overview

JFCS is chartered to investigate issues important to determination of the technical and economic feasibility and nonproliferation acceptability of electrochemical recycling as one option for the management of used nuclear fuel. Electrochemical recycling utilizes dry processes that allow collection of U and group collection of a U/TRU product in compact fuel cycle facilities. The technology offers head-end processes that allow recycling of actinides from oxide fuel, such as used LWR fuel.

The JFCS is a schedule-driven activity of 10-year duration and is divided into three phases. All phases include a range of activities, but each has an area of primary emphasis. The JFCS began in 2011, and the first phase was 2 years in duration and focused on the Laboratory-Scale Feasibility Study (LSFS) to verify the scientific feasibility of electrochemical recycling at small scale. The second phase is 5 years in duration (2013–2017) and has a primary emphasis to demonstrate reliable and reproducible integrated process operations as well as recover sufficient fuel material for recycled fuel fabrication. The third phase is 3 years in duration (2018–2020) and will be focused on validation of recycled fuel fabrication processes, recycled fuel irradiation, and PIE. This overview provides highlights of the accomplishments of the JFCS Campaign through FY 2017.

Key FY 2017 outcomes:

- Installation and operational testing of six new pieces of remote equipment and nine instrumentation and power feedthroughs into the HFEF.
- Successful completion of process experiments with fourteen irradiated Fast Flux Test Facility (FFTF) fuel elements
- Completion of a suite of experiments investigating simplified U/TRU recovery techniques
- Side by side studies demonstrating extended durability of advanced electrode materials for oxygen production
- Joint selection of waste form processes for remote evaluation following fuel processing in the Integrated Recycling Test (IRT).

4.2 Laboratory-Scale Feasibility Study

The LSFS purpose was to evaluate the technical feasibility of the electrochemical process at laboratory scale. This was accomplished through a small-scale study with irradiated materials and existing equipment. An experiment with approximately 100 grams of irradiated fuel was performed in the Hot Fuel Dissolution Apparatus in the INL HFEF.

Some of the prescribed operations in the LSFS had been performed previously at laboratory scale with used fuel, and some operations were performed for the first time. The operations were performed in a linked manner to allow assessment of an integrated process. The LSFS confirmed the feasibility of electrochemical recycling of used LWR fuel. The LSFS also provided integrated operating experience to inform decisions about equipment and process development for kilogram scale studies.

4.3 Integrated Recycling Test

An integrated testing activity at kilogram scale is underway to test electrochemical recycling flowsheets and provide material balance information for an integrated process model. IRT is a critical component for the overall goals of the JFCS. The IRT includes decladding of irradiated fuel, electrochemical processes to reduce and collect U/TRU material, and the fabrication, irradiation, and PIE of metal fuel rodlets produced from recycled oxide fuel. Most operations for the IRT will be performed within the HFEF hot cell of the Materials and Fuels Complex at INL. During Phase I (approximately calendar year [CY] 2011–2012), the design of process equipment began and fabrication initiated on some components. During Phase IIA (CY 2013–2014) equipment design and fabrication continued. Phase IIB (CY 2015–2017) focused on the design completion, equipment fabrication, testing, installation, and operation of equipment to recover re-usable materials from irradiated fuel. The objective by the end of Phase IIB is the casting of fuel slugs including recycled material for irradiation in the ATR. Figure 15 provides a diagram illustrating the process operations of the IRT. The following paragraphs provide additional description of primary equipment components which will be operated during the IRT.

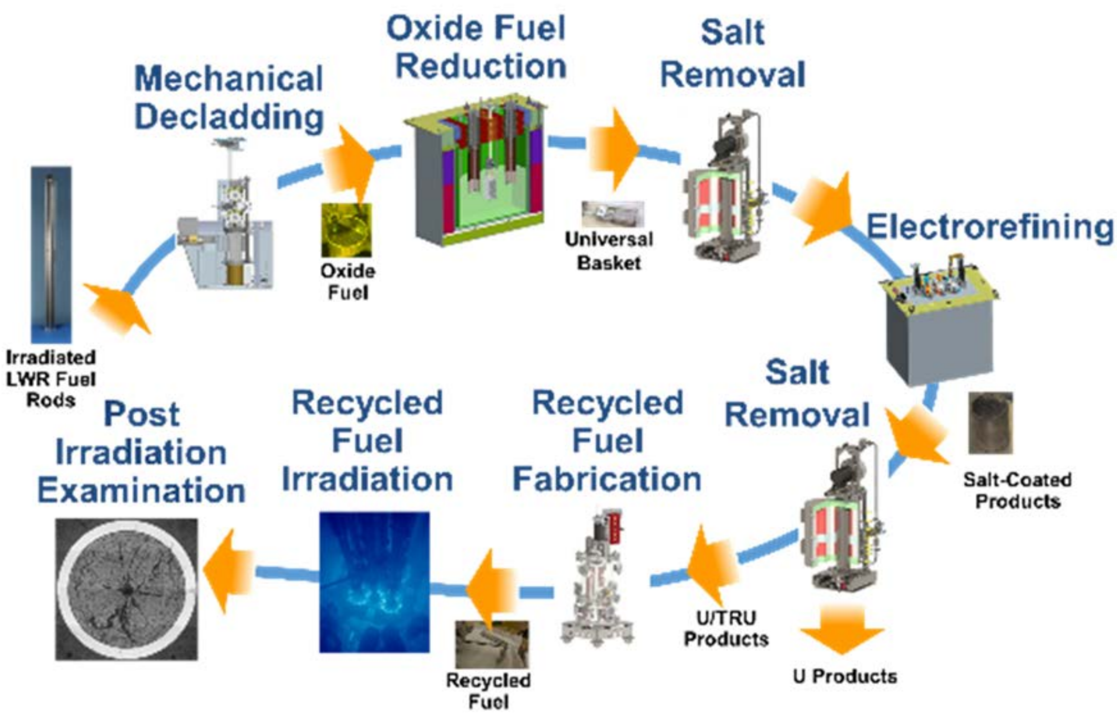


Figure 15. Workstation and equipment layout for the JFCS IRT.

Modular Workstations: The modular workstations will serve as the work platforms for IRT process equipment in HFEF. These include expansion of an existing table at window 11M in HFEF, a new table installed at window 12M, and a smaller intermediate table installed between these two tables. These workstations include integrated balances as well as storage space for tools, equipment, and archived samples. Figure 16 depicts the workstation and equipment layout for the IRT in HFEF.

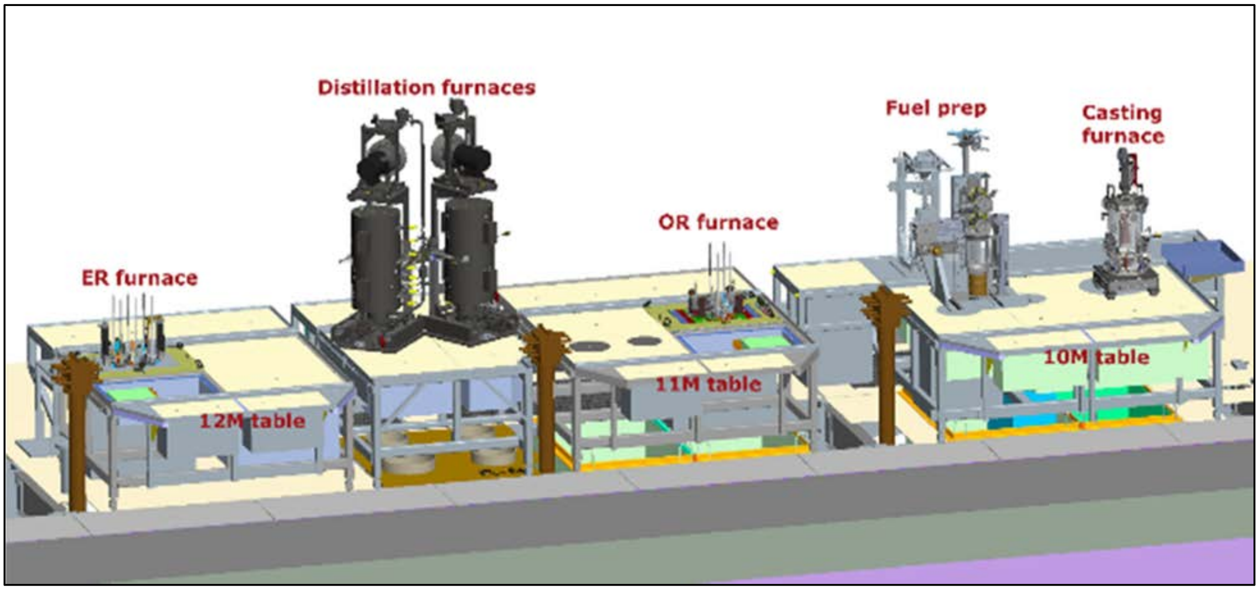


Figure 16. Workstation and equipment layout for the JFCS IRT.

Head-End Equipment: The equipment necessary to prepare irradiated fuel for IRT processing is described as head-end equipment. This includes equipment for decladding, sieving, handling of fines, and material storage. In FY 2017, head-end equipment was used to process fourteen irradiated FFTF fuel elements. The declad elements were processed in two separate batches of seven rods each. The head-end equipment performed as designed, and overall fuel recovery was close to 99% with approximately 96% of the recovered fuel greater than 45 µm. Figure 17 shows the vibratory decladding system during decladding operations. Figure 18 shows the fuel product collected following decladding.

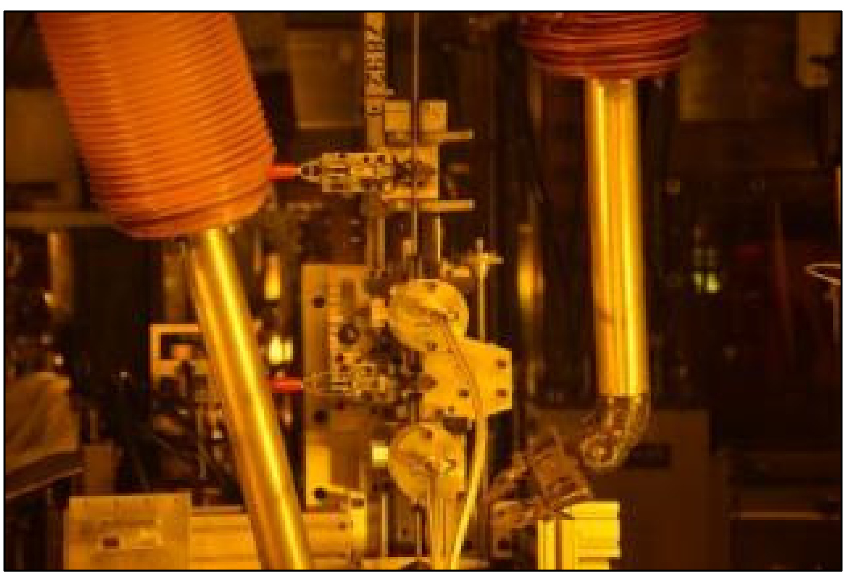


Figure 17. Vibratory decladding system during testing with irradiated fuel in HFEF.

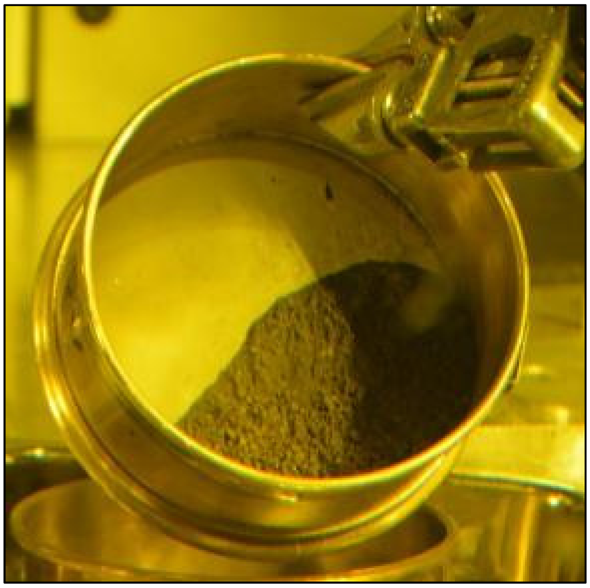


Figure 18. Photograph of typical fuel product collected from one FFTF fuel element.

Oxide-Reduction System: The oxide reduction system (Figure 19) electrolytically reduces oxide fuels to produce a metallic product that is suitable for further electrochemical recycling. In FY 2017, oxide reduction was performed on two batches of irradiated declad FFTF fuel particulate in separate universal baskets. The objective of the experiments was to deliver reduced fuel to the electrorefiner module for subsequent uranium electrorefining. Pre- and post-run salt and fuel samples were obtained and elemental and isotopic analyses are underway to determine the extent of fuel reduction and the distribution of fission products between the salt and fuel phases.

Figure 19. Oxide reduction system at HFEF window 11M.

Electrorefiner System: The purpose of the electrorefiner is to allow collection of TRU elements and fission products from the reduced fuel and accumulate those elements and products in the electrorefiner salt. Purified U is collected, and periodically group recovery will be performed to recover a mixed U/TRU product by use of a liquid cadmium cathode (LCC). Following setup, a batch of depleted uranium metal and a batch of depleted uranium oxide reduced in the oxide reduction system were processed for process parameter development prior to experiments with irradiated oxide fuel. Two batches of declad irradiated FFTF fuel, which had previously been reduced in the IRT oxide reduction module, were then used for electrorefining experiments. Initial tests indicate the desired accumulation of TRU materials in the electrolyte in preparation for upcoming U/TRU recovery. Analysis of product and salt samples from these experiments is currently in progress.

Distillation System: The distillation module is used to remove residual salts and/or cadmium from products that originate from the oxide reduction or electrorefining modules. Two separate distillation systems are installed to accommodate the variety of operation experiments. Experiment operations in FY 2017 included salt removal from the surrogate and irradiated fuel experiments.

Fuel Fabrication Equipment: Recovered actinide materials will be used to fabricate fuel rodlets for irradiation in ATR. In FY 2017, the Casting and Sampling Furnace (Figure 20) was tested for operability using several surrogate casting with uranium and uranium-zirconium alloys. Surrogate testing demonstrated that the Casting and Sampling Furnace is ready for operations with recovered U/TRU products and is staged for casting operations in FY 2018.

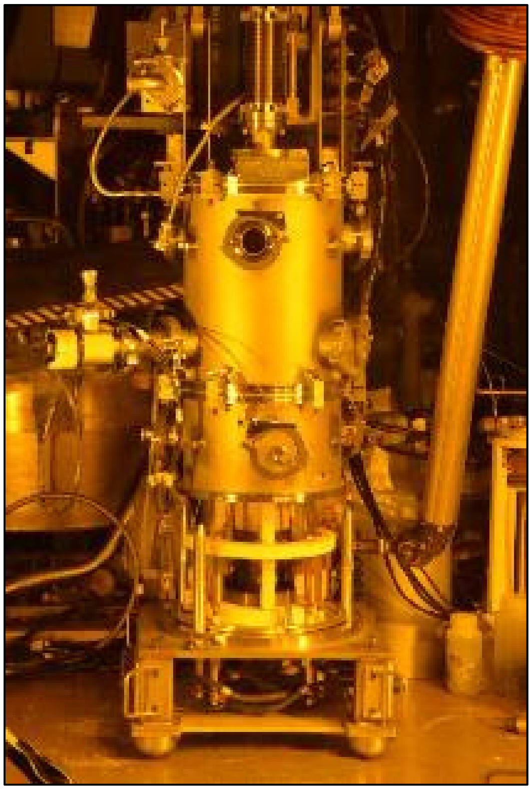


Figure 20. Remote casting system at HFEF window 10M during remote qualification.

4.4 Critical Gap Research and Development

A range of research and development was performed in FY 2017 to provide fundamental knowledge necessary to perform the IRT, close process modeling gaps, and to develop and demonstrate improved processes that are critical to confirm the feasibility of a commercial-scale process. Significant accomplishments from each research activity are briefly summarized below.

Lithia Monitoring: In the oxide reduction process, Li_2O serves as the oxide ion transport species. Experience has demonstrated that successful performance of the electrolytic reduction process requires the Li_2O concentration be maintained within a particular range. A real-time method for monitoring the concentration of Li_2O is being developed, and will be a valuable element to deployment of electrochemical recycling. A Li_2O sensor based on the yttria stabilized zirconia was studied to evaluate its accuracy, reliability, and feasibility for engineering application in an oxide reduction system. When the LiCl–Li_2O salt contains no dissolved lithium metal, the Li_2O sensor can promptly detect the Li_2O concentration changes. The presence of dissolved Li metal in the LiCl–Li_2O salt can affect the OCP measurements, although this influence may be mitigated by a using a “Li-equilibrated” yttria stabilized zirconia membrane.

Oxide Reduction: The IRT oxide reduction system components were upgraded to address long-term corrosion concerns and improve remote operability, and a series of experiments were performed with depleted UO_2 to tune process parameters.

Anode Material Selection and Design: Platinum has served as the primary anode material for much of the development of the electrolytic reduction process for used UO_2 fuel. Iridium alloys and graphite are two alternative materials that could function in place of platinum, and testing continued in FY 2017 to examine their performance. A series of four electrolytic reduction runs was performed with platinum and iridium anodes operating simultaneously during the conversion of uranium oxide to uranium metal. In these tests, iridium exhibited superior performance characteristics over platinum during the electrolytic reduction of uranium oxide in LiCl – 1 wt% LiCl at 650°C.

Graphite is also being explored as an alternative material to replace the platinum. A parametric investigation of the oxide reduction process using graphite anodes was conducted to verify the efficacy of this approach, study operational parameters associated with the use of graphite anodes, and identify further engineering necessary for larger scale application of the graphite in the electroreduction process. Specific process operations that were studied included the batch size, electrode polarization, current densities, and the optional use of current and voltage control for the process. Bench-scale (25-100 g UO_2) experiments yielded a reduced product that was typically greater than 95% metallic uranium, such as that shown in Figure 21. These studies demonstrated that the electrolytic reduction of UO_2 and by inference used oxide fuel can be achieved using graphite anodes, but controlling anode polarization and the chemistry of the electrolyte are important to successful operations.

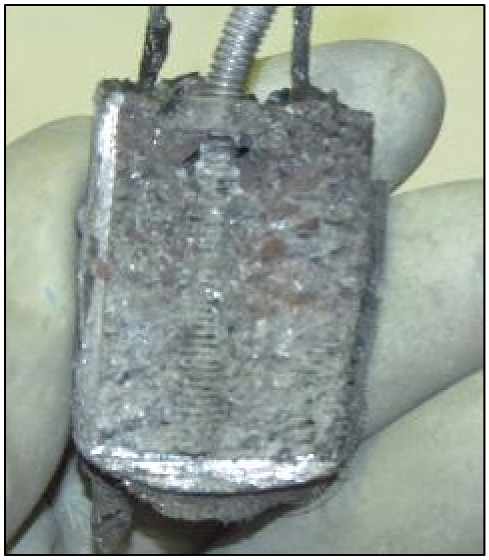


Figure 21. Photograph of metallic uranium product recovered from bench-scale UO_2 electroreduction experiments using a graphite anode.

Low U/TRU LCC Operations*:* In order to recover U/TRU rare earth products in the necessary rapid time frame for the IRT, it is desired to perform LCC operations at low concentration of transuranium elements in the salt. Few data are available for LCC operations at such low concentrations and challenges may occur. In addition to supporting the near-term IRT, elucidation of LCC performance at low TRU concentrations is important to manage the TRU inventory for long-term electrorefiner operations. A series of LCC experiments were completed and samples analyzed to explore these issues.

Electrochemical Kinetic Parameters for Np via AC Electrochemical Methods: In electrochemical separation processes where the kinetics become the dominant factor, characterization of the interfacial electrochemical kinetics parameters, particularly the standard rate constant, exchange current density, and transfer coefficient are required to fully understand the process. We applied AC voltammetry to molten LiCl-KCl eutectic solutions containing $NpCl_3$ at several concentrations and over a range of temperatures to characterize the interfacial electrochemical kinetics of Np using equations we previously developed for analysis of AC voltammetric data specifically for the electrodeposition of an insoluble metallic product.

High Temperature Materials: A number of distillation and melting operations are required for the IRT. One challenge is that current crucible materials for melting operations do not allow the recovery of actinoid holdup. As an example, ZrO_2-based drosses react with Li_2O in the oxide reductions system to form a ternary oxide. Yttria (Y_2O_3) drosses will quantitatively react with UCl_3 in the electrorefiner to increase YCl_3 concentration in the salt. Materials with potentially improved characteristics are under development. One material being investigated for containment of U/TRU products with low product loss is hafnium nitride. Another material is urania (UO_2) formed into dense crucibles or coatings. An experiment with UO_2 coatings is shown in Figure 22. These urania-based materials should allow for recycling of the spent urania crucible, along with any material holdup, back into the process to recover the material holdup.

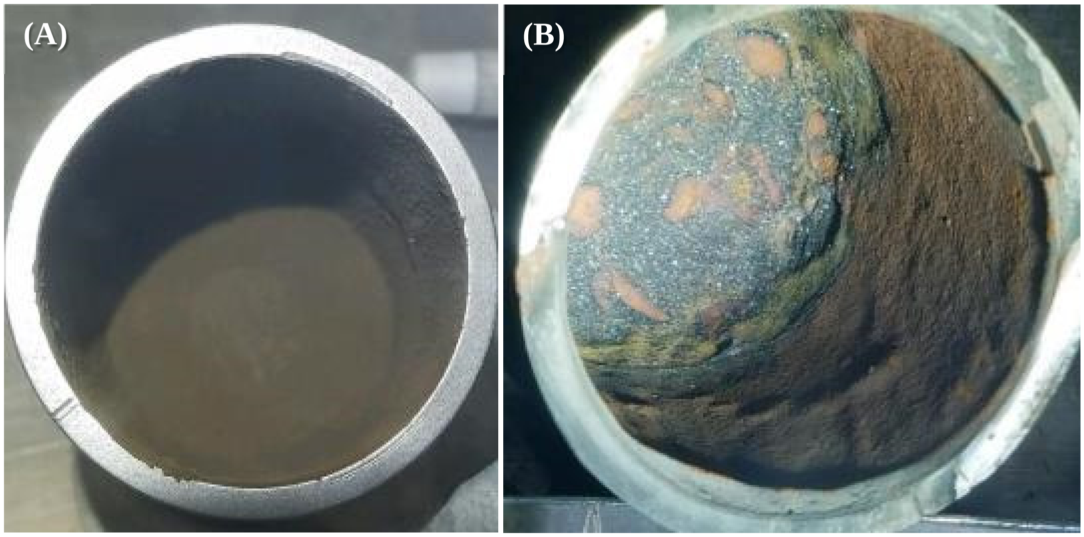


Figure 22. UO_2-coated crucible (A) before and (B) following experiments with molten U.

Electrochemical Waste Forms and Processes: The identification and demonstration of appropriate waste forms and associated production processes are critical to the overall demonstration of the feasibility of electrochemical recycling. In FY 2017, treatment options for the IRT were down selected to include the metal waste form, advanced ceramic waste form, silica-alumina-phosphate composite, and lanthanide borosilicate glass. These will be demonstrated with fission products during Phase III of the IRT. Waste form research activities continued waste form performance testing and characterization for the nano-silica-alumina-phosphate composite, Zr-Cr alloy, and mixed oxide/alloy waste forms as well as iodine sorbent development.

Fuel Fabrication: Fuels critical gap activities focused in the areas of fuel casting technique development, fuel cladding chemical interaction, and thermal characterization. In the casting technique development, a DU-20Pu-10Zr rod was successfully cast using the gravity casting methods utilizing the JFCS casting mold. Experiments with a solid Y_2O_3 crucible liner showed good results including low dross formation, and future melt experiments at this scale will use this approach. Fuel cladding chemical interaction testing was performed using diffusion couples of U-20Pu-10Zr-3RE fuel against Cr coated iron and TiN coated iron alloys. Chromium-coated samples showed this element to be an effective diffusion barrier against rare earth elements, even at 750°C. The TiN diffusion couples are awaiting characterization. Finally, differential scanning calorimetry was performed on the U-20Pu-10Zr-3RE fuel. Differential scanning calorimetry testing showed that the as-cast microstructure of the fuel is not an equilibrium phase, and multiple phase transitions were removed and others developed upon heating the sample to 1200°C.

5. MATERIAL RECOVERY AND WASTE FORM DEVELOPMENT CAMPAIGN

T. A. Todd, Idaho National Laboratory, J. D. Vienna, Pacific Northwest National Laboratory

The MRWFD Campaign develops advanced separation and waste processing technologies to support the various fuel cycle options defined in the *DOE Nuclear Energy Research and Development Roadmap, Report to Congress, April 2010*. Although research is performed to support a range of potential fuel cycles, the focus is on the most promising fuel cycles evaluated in the Nuclear Fuel Cycle Evaluation and Screening – Final Report, October 2014; which entail actinide recycle. This section provides a highlight of the results of R&D efforts performed within the MRWFD Campaign in FY 2017. Each subsection contains a high-level overview of the activities and key results, produced during the fiscal year.

This section briefly outlines the campaign mission, objectives and challenges and highlights key technical accomplishments made during FY 2017. The campaign continued to utilize an engineering driven-science-based approach to maintain relevance and focus.

MRWFD Campaign management and integration activities included international collaboration activities (primarily focused on bilateral and multilateral collaborations with France, China, Japan, European Union, and IAEA), integration of MRWFD Campaign activities with other FCT campaigns, (primarily Advanced Fuels, Used Fuel Disposition, FCO, and MPACT), and integration with DOE Offices of Environmental Management, Science, and the National Nuclear Security Agency (NNSA).

Technical accomplishments are reported under the following R&D categories:

- Reference Technologies and Alternatives
- Sigma Team for Advanced Actinide Recycle
- Sigma Team for Off-Gas Capture and Immobilization
- Fundamental Science and Methods, Modeling, and Simulation
- Advanced Waste Form Development and Performance
- Domestic Electrochemical Separation Technologies
- Process Demonstrations.

> ***Mission***
>
> **Develop advanced fuel cycle separation and waste management technologies that improve current fuel cycle performance and enable a sustainable fuel cycle, with reduced processing, waste generation, and potential for material diversion.**

5.1 Mission

MRWFD, formerly Separations and Waste Forms, applies expertise and technical capabilities to a wide array of applications. This campaign now also leverages its expertise by working with others in areas such as environmental remediation, national security missions, as well as civilian nuclear applications. The mission of the MRWFD Campaign is to:

Develop advanced fuel cycle separation and waste management technologies that improve current fuel cycle performance and enable a sustainable fuel cycle, with minimal processing, waste generation, and potential for material diversion.

Mission implementation is outlined in the *Campaign Implementation Plan*, issued in November 2012. A revision will be made following issuance of a new Nuclear Energy Roadmap, and associated implementation plans.

5.2 Objectives

- Develop technologies that support the current once-through fuel cycle and have potential near-term application.
- Develop a fundamental and practical understanding of methods for the separation of uranium and TRU elements from used fuel.
- Develop a fundamental and practical understanding of the factors affecting performance of advanced waste forms over geologic time-scales.
- Develop and demonstrate enabling technologies to separate and immobilize gaseous fission products from used nuclear fuel.
- Develop advanced waste forms with greatly improved properties and cost and develop/demonstrate associated processes.

5.3 Challenges

- Separation of minor actinides from lanthanides in both aqueous and molten salt media.
- Capture and immobilization of off-gas constituents of used fuel, including iodine, krypton, tritium and potentially carbon in a cost-effective manner.
- Development of separation technologies and waste forms is very interrelated to the types of fuels being processed, the types of fuels being fabricated, and the reactors used to burn recycled fuels.
- Measuring waste form lifetimes in a laboratory is impossible, considering they are on the order of hundreds of thousands to millions of years.
- Achievement of advanced separation and immobilization processes in a cost effective manner.
- Predict performance of waste forms with lifetimes measured in units of millions of years.

5.4 Reference Technologies and Alternatives

This activity supports evaluation of solvent degradation mechanisms and development of tritium removal technologies (for open and closed fuel cycle applications). Testing of a closed loop NO_2 oxidation of fuel at low temperature (for tritium and possibly iodine removal) has progressed to the point of demonstration with actual used fuel. Some issues with facility availability, waste management and funding have delayed the demonstration, but efforts continued to prepare for the hot demonstration in FY 2017. The MRWFD Campaign had productive collaboration with the European Union Framework 7 Safety of Actinide Separation Processes program (Safety of Actinide Separation Processes) and has now been involved in the follow-on program GENIORS (Generation IV Integrated oxide fuels recycling strategies). Investigation of separating tritiated water from normal water was performed using inorganic membranes. Degradation studies the ALSEP and innovative-SANEX process solvents under representative process conditions were performed.

5.4.1 Sigma Team for Advanced Actinide Recycle

This activity is developing more robust and simplified approaches for separating actinides to enable future fuel cycles that transmute actinides for improved resource and waste management. There is a large international effort in nearly every fuel cycle country working on this difficult chemical separation and the FCT program is making significant progress on the development of cost-effective methods of separating the minor actinides from used fuel. In FY 2017, the primary activities were focused on understanding the radiation stability of Am(VI) under radiolysis and solvent extraction conditions, and flowsheet extraction testing of Am(VI) in 3D printed centrifugal contactors. Significant progress on the development and understanding of the ALSEP solvent extraction process was made, including the first countercurrent flowsheet test using a simulated feed. Development of mixed donor ligands as selective trivalent actinide extractants also continued to make good progress.

5.4.2 Sigma Team for Off-Gas Capture and Immobilization

This activity is needed to enable any new fuel treatment facility to meet current regulations. The capture of iodine at very high decontamination factors is required and iodine has a very long half-life, so immobilization is important to reducing the source term in a geologic repository. Krypton (Kr) capture will be needed if processing fuel less than roughly 30 years old. Tritium may also require capture if removed from fuel at the headend. It is very important to understand the behavior of the entire off-gas system, to avoid cross-contamination of sorbents (e.g., iodine on tritium or krypton sorbents). Four major thrust areas were continued in FY 2017: (a) iodine capture in which the impacts of penetrating organic forms of iodine were studied, (b) iodine immobilization in which scale-up testing of fused silica based waste form for silver functionalized aerogel was demonstrated, (c) tritium separations in which a process for separating tritium from irradiated hulls chlorination was developed, and (d) krypton separations and storage in which desorption process for mordenite sorbents was studied and a new higher-capacity near room temperature sorbent was developed and tested.

5.4.3 Fundamental Science and Methods Development Modeling and Simulation

This activity is utilizing new tools and research methods to understand the fundamental properties of extraction systems. These fundamental properties are the basis for understanding any separation process from a science-based approach rather than an empirical approach, which has been the typical approach used in the past. A greater understanding of the fundamental properties (such as thermodynamics, kinetics, effects of radiation on chemistry) will enable the development of more robust processes and also support future models that allow for a predictive capability of process performance. In FY 2017, the radiolysis of diglycolomide ligands was further investigated in support of the ALSEP process development efforts as well as the EU GENIORS collaboration. As part of the DOE-CEA bilateral collaboration, monoamide radiation chemistry was also further investigated to better understand how these compounds and their degradation products will behave under process application. Structural modification of aminopolycarboxylate aqueous holdback reagents to facilitate liquid-liquid systems capable of fast equilibration and efficient An/Ln separation were developed.

5.4.4 Advanced Waste Forms Development and Performance

These activities are necessary for the immobilization of waste streams from the advanced separation processes, including advanced aqueous and electrochemical processes. These waste forms are designed to improve the performance over current waste forms, such as borosilicate glass, over geologic time frames. Higher performance can be achieved by utilizing glass ceramic or ceramic waste forms for high-level waste raffinate and a durable waste form for radioiodine. Any new waste form must be processed in production-scale continuous processing equipment. Ceramic containing waste forms must be processed at higher temperatures than glass waste forms; therefore standard joule-heated melters are not adequate, so new process technology is needed (e.g., cold-crucible induction melters). In FY 2017, characterization of multi-phase ceramic waste forms produced via melt processing and HIP methods was performed. The first large-scale chlorination test of irradiated used nuclear fuel cladding was completed to demonstrate the potential for zirconium recycle from used fuel. Improved loading ceramic waste forms for electrochemical salt high-level waste were developed and tested at laboratory scale.

Studies of the long-term performance of glass were continued. An approach to implementing a model of corrosion acceleration was further refined. Studies of the long-term performance of example multiphase radioiodine waste forms were continued. Initial electrochemical and solution emersion test result on the AgI phase were completed. Studies on the performance of electrochemical metal waste form performance were completed.

5.4.5 Domestic Electrochemical Separations Technologies

This activity is developing technologies to potentially enhance performance and reduce waste volumes in the treatment of fast reactor fuels. This technology is suited to treatment of metallic fuels for TRU recycle. In FY 2017, refinement of the equipment design and testing procedures for kg-scale U/TRU codeposition using a high-current-density solid cathode was continued using rare earth elements as surrogates for TRU. Studies were performed to identify solutions for areas where technical gaps in electrochemical recycle technology have been identified.

5.4.6 Process Demonstrations

Two process demonstration activities were in progress during FY 2017. The first demonstration, the Co-Decontamination Project (CoDCon) demonstration, was established to demonstrate the ability to control the uranium and plutonium (stripped together from a loaded solvent) at a specified ratio (such as 70:30). The second demonstration will demonstrate the chlorination of zirconium (as a possible headend treatment for naval fuel) to recovery high-enriched uranium, which could be downblended for use in tritium production or as advanced reactor startup core fuel.

5.4.6.1 CoDCon Flowsheet Demonstration

The CoDCon flowsheet demonstration was established to quantify, using laboratory-scale equipment, the accuracy and precision to which a specific uranium-to-plutonium (U/Pu) ratio can be achieved in the Pu-containing product from a tri-butyl phosphate based nuclear fuel recycling flowsheet. For the purpose of this project, the target U/Pu mass ratio is $^7/_3$ (2.33), in both the mixed U/Pu nitrate stream from the solvent extraction and in the final MOX product. The uncertainty associated with achieving a specific target U/Pu ratio will be established. Another major objective of the project is to demonstrate optical spectroscopic techniques for real-time monitoring of the concentrations of key components (e.g., Pu, U, and HNO_3) in the process solutions. The monitoring capability is viewed as critical to achieving the first stated objective of a product with U/Pu mass ratio of $^7/_3$. In order to establish the uncertainty in the U/Pu ratio, four CoDCon tests are planned. For each CoDCon test, the following key steps are to be performed:

1. Prepare simulated dissolved fuel solutions to be used as feed for the tests
2. Perform solvent extraction to separate the desired U/Pu product
3. Convert the resulting U/Pu nitrate solution to oxide
4. Characterize the U/Pu product.

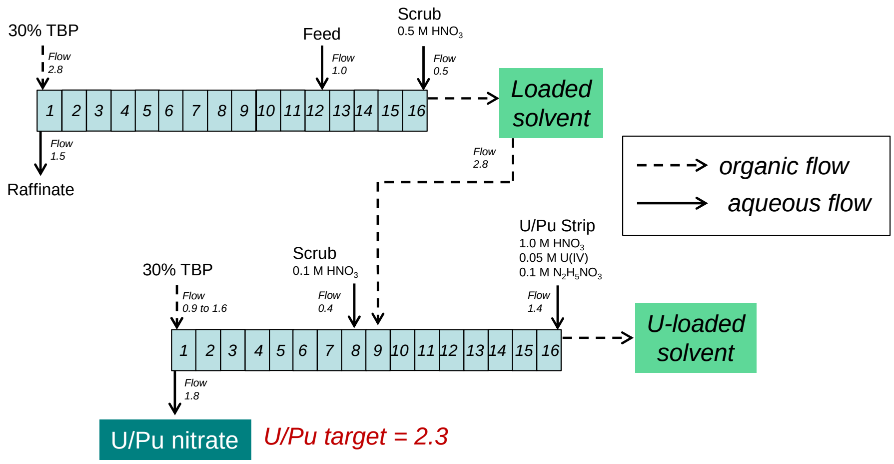


Figure 23. CoDCon flowsheet schematic.

5.4.6.2 ZIRCEX Demonstration

The DOE is interested in recovery of HEU from naval fuel as a potential source of material to support advanced reactor start-up cores, which require high-assay low enriched uranium. DOE-NE has been researching a new head-end process that would greatly reduce waste volumes over the methods used previously to process naval fuel. Previous processing of naval fuels (prior to 1991) was done to recovery HEU for defense purposes (Pu and tritium production reactor driver fuel) and after halting the processing, a final batch of HEU product was downblended for use in LWR reactors.

A promising approach to HEU recovery is to remove the zirconium from naval fuel with a chloride volatility process (ZIRCEX), leaving only the HEU fuel components, and most fission products to be dissolved for recovery of the HEU through a single solvent extraction process. This would greatly reduce the size of the extraction and waste processing equipment needed, because the extraction flowsheet would be concentrated and would generate 250 times less liquid high-level waste (for comparison, the previous process generated 3 million liters of HLLW per MTU, while the process with a ZIRCEX headend would only generate 12,000 L per MTU). Extensive work on this process was completed in the 1960s but was never demonstrated on this type of fuel. The ZIRCEX process utilizes chlorination of the fuel at 350-450°C, which volatilizes Zr as $ZrCl_4$ (g) and other constituents of Zircaloy from the remainder of the fuel. The uranium and most of the fission products do not form volatile compounds and are left in the chlorination vessel. This material is then oxidized (to enable removal from the chlorination vessel) and in the actual process would then be sent to dissolution and solvent extraction. The proof-of-concept project is only for the chlorination and oxidation steps, as the downstream processes (dissolution, extraction, waste treatment) are mature and ready for implementation. The process would also be amenable to processing aluminum clad HEU fuel, such as ATR fuel as the aluminum will form a volatile $AlCl_3$ complex, similar to the $ZrCl_4$ with zirconium-clad fuel. Laboratory-scale tests with irradiated naval fuel coupons were successfully completed and a 6-inch (1/4 –scale vessel diameter) pilot plant has been constructed and will begin testing with zirconium rods in early 2018.

5.5 Key Fiscal Year 2017 Deliverables

The following FY 2017 deliverables were completed.

- Preliminary testing of a CoDCon solvent extraction flowsheet and conversion of nitrate products to oxides were completed in preparation for the first demo test with simulant solutions. This testing will be performed in a shielded glovebox at Pacific Northwest National Laboratory (PNNL) with simulated feeds containing actual concentrations of uranium and plutonium.
- Co-precipitation of U, Np, Pu, and Am (all hexavalent) was demonstrated with simulant solutions Demonstrated co-crystallization of americium (VI) with uranyl nitrate for the first time. This activity is exploring the co-precipitation of actinyl ions (U, Np, Pu, Am) from aqueous media as a group actinide separation concept.
- The first ever flowsheet test of the ALSEP (Actinide-Lanthanide) separation process was successfully performed using a simulated feed spiked with radiotracers. The test was performed at ANL in 1.25 cm centrifugal contactors that were made via additive manufacturing techniques. The flowsheet test was successful in demonstrating the separation of Am and Cm from lanthanides in a single process (as opposed to two processes –TRUEX and TALSPEAK) which were demonstrated under the AFCI and GNEP programs.
- Extraction of Am(VI) with monoamide extractants in centrifugal contactors was demonstrated with simulant solutions. This approach was previously demonstrated using phosphate-based extractants, but it appears there may be advantages to using a nitrogen-based monoamide extractant.
- Demonstrated tritium recovery from aqueous solutions using SAPO-34 and LTA zeolite membranes. Results show an improvement from previous testing and that the process may be competitive with current baseline approaches such as CECE.
- Completed an engineering analysis on an integrated off-gas capture system for a fuel recycle facility. This analysis is instrumental to understanding the global off-gas problem and identifying high-priority research needs.
- Investigated the radiation stability of the ALSEP solvent under gamma radiolysis

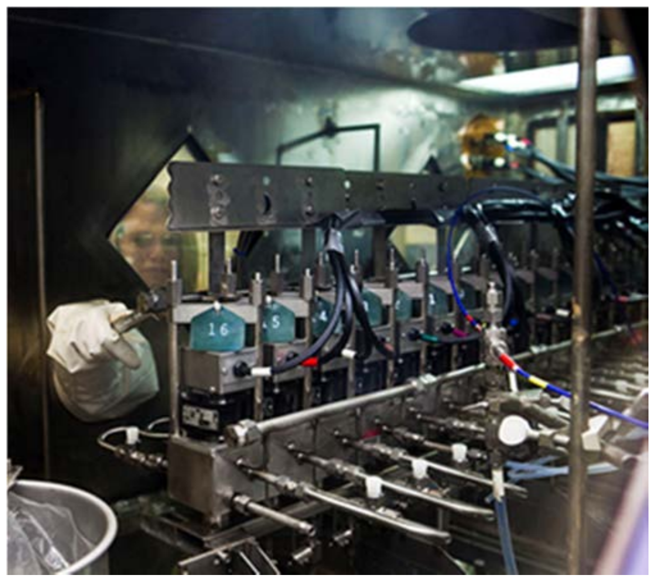


Figure 24. Centrifugal contactor equipment is shielded glovebox at PNNL.

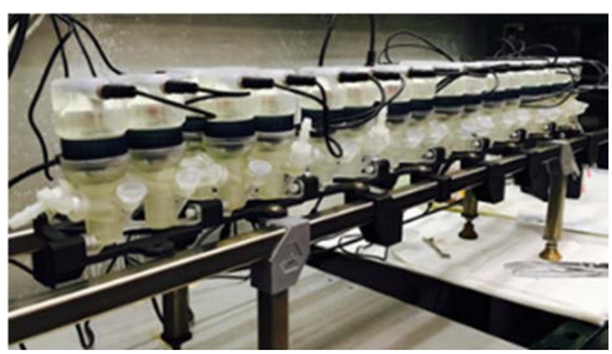


Figure 25. 3D-printed centrifugal contactors used for ALSEP flowsheet test at ANL.

and relevant flowsheet conditions for extraction and scrub sections of flowsheet. This work was done in the INL solvent test loop, a unique testing capability for testing solvent extractant stability in realistic process conditions.

- Completed joint radiation experiments with CEA on monoamide extractants as part of the bilateral DOE-CEA collaboration. Joint journal manuscripts were published on this work.
- Completed zirconium chlorination (recycle) tests using North Anna cladding. While this testing was successful, the zirconium product was not pure enough for recycle applications, therefore, focus of FY 2018 research activities will be on further purification of the zirconium product.
- A new module for the MASTERS model to incorporate the ability to model TRU recovery on solid cathodes was completed. The MASTERS model is a comprehensive mass balance model for electrochemical recycle based on the MATLAB platform.
- A joint workshop on modeling and simulation between MPACT and MRWFD was held and a plan for standing up this joint M/S effort was development and submitted to DOE in March 2017.
- Presentations on glass corrosion R&D were made to the Nuclear Waste Treatment Review Board pre-briefing and public meeting, and the results were lauded by the board.
- A ceramic waste form test was completed in a cylindrical induction melter using SYNROC waste form composition.
- Significant progress was made in understanding and modeling Stage 3 glass corrosion. This is a region where, after relatively stable glass dissolution rate, an unexplained acceleration in the dissolution rate occurs.

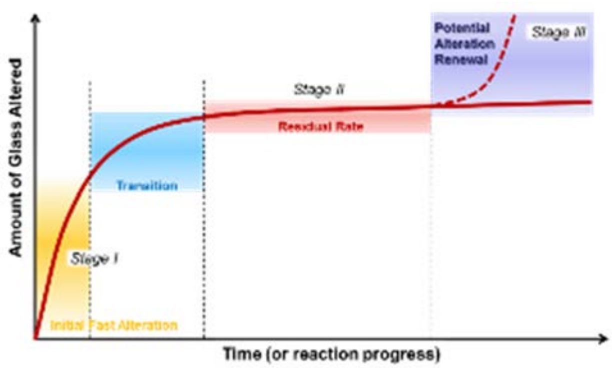


Figure 26. Glass corrosion stages versus time.

5.6 Summary

The MRWFD Campaign continues to make significant progress toward the development and understanding of nuclear materials recovery, waste form development, waste form performance, and nuclear materials processing. These contributions are recognized worldwide and have resulted in a number of publications in prestigious journals and invitations to present at international conferences.

6. MATERIAL PROTECTION, ACCOUNTING, AND CONTROL TECHNOLOGIES PROGRAM

Mike Miller, INL, NTD

6.1 Overview

The MPACT Campaign develops innovative technologies and analysis tools to enable next-generation nuclear materials management for existing and future U.S. nuclear fuel cycles, to manage and minimize terrorism risk, and to enhance confidence in and acceptance of nuclear energy.

The existing and future nuclear energy enterprise must prevent, deter, and detect misuse of nuclear materials and associated fuel cycle technologies for both national and global security. While a mature nuclear materials management infrastructure is in place for the existing nuclear energy system, research is needed to support new or improved fuel cycle options. The MPACT Campaign works with the fuel cycle technology development campaigns to optimize nuclear materials management for the benefit of both the facility and regulator. As such, the campaign has engagement with stakeholders in NNSA (Defense Nuclear Nonproliferation Office of Research and Development and Office of Nonproliferation and Arms Control), NRC and IAEA, as well as with industry and universities.

Simply improving nuclear material measurement performance is not enough to meet timeliness detection goals for advanced fuel cycle options, instead an integrated systems approach is required that fully utilizes additional operational data streams. In 2016, a series of roadmaps for the campaign were developed, including an Advanced Integration Roadmap outlining methods for integrating modeling and simulation, advanced technologies and analysis to provide enhanced system performance; a Modeling and Simulation Roadmap; and the Used Fuel Extended Storage Security and Safeguards by Design Roadmap. In 2017, we built on these roadmaps to enhance the research and development required to meet our 2020 milestone “Lab scale demonstration of an advanced safeguards and security system.” With the reorganization of the Office of Nuclear Energy, the used fuel security analysis work has been moved to the Office of Spent Fuel and Waste Disposition.

6.1.1 Objectives of MPACT Campaign

- Develop tools, technologies, and approaches in support of electrochemical and aqueous processing, advanced fuel fabrication, and associated advanced NES.
- Develop, test, and demonstrate advanced material control and accounting technologies that would, if implemented, fill important gaps in existing MPACT capabilities.
- Develop, test, demonstrate, and apply MPACT analysis tools to assess effectiveness and efficiency of MPACT systems, guide R&D, and support advanced integration capabilities.
- Perform technical assessments in support of advanced fuel cycle concepts and approaches.
- Develop guidelines for safeguards and security by design and apply to new facility concepts.

6.1.2 Challenges and Drivers for MPACT Campaign

- Future advanced fuel cycle facilities may be larger, more complex, and more widespread.
- Insider and outsider threats may continue to become increasingly sophisticated and capable.

- Achieving stringent goals for detection timeliness and sensitivity in advanced fuel cycle facilities will be difficult and expensive.
- Satisfying stringent physical protection requirements in advanced fuel cycle facilities will be expensive.
- Addressing stakeholder concerns will require positive assurance that risks of nuclear proliferation and terrorism are minimized.
- Demonstrating a lab-scale advanced safeguards and security system in the early 2020 time frame.
- Maximizing use of all information- traditional nuclear materials accounting, physical security, and process knowledge.
- Testing technologies with higher TRLs as opportunities arise to bridge gaps necessary for practical use.

Technical challenges for the MPACT campaign include:

- Improving the accuracy and precision of nuclear material accountancy measurements, while improving their timeliness and cost-effectiveness.
- Expanding the scope of detection to include more indicators, taking advantage of existing data where possible and new sources of data where appropriate.
- Expanding and strengthening assessment algorithms to exploit larger data sets and provide results in near-real time in an integrated manner that quantitatively takes into account uncertainties and correlations.
- Modeling and simulating MPACT performance against a wide spectrum of assumed threats and rigorously demonstrate MPACT effectiveness and efficiency in future U.S. NES.
- Integrating safeguards and security into the design of future nuclear fuel cycle facilities from the earliest stages of the design process.
- Raising technology maturation to a TRL appropriate for useful field testing.

6.1.3 Key FY 2017 Deliverables

- Campaign Management and Integration
  - Implementation plan for MPACT 2020 milestone (M2)
- Electrochemical Process Monitoring for Enhanced Safeguards
  - Test micro-analytic droplet generator with An-bearing salt (M2)
  - Expansion of AMPYRE to allow for operation sequencing, equipment sizing, and sensor interface
  - Potentiometric/actinide sensor testing with uranium
  - OR Voltammetry probe readiness (M2)
  - Advanced ER Voltammetry probe design and testing
  - Time-dependent radiation field mapping simulations

- Advanced Integration
  - Advanced integration use case analyses
  - Safeguards and Security Performance Model (SSPM) and Scenario Toolkit and Generation Environment (STAGE) model integration
- Advanced Instrumentation and Field Tests
  - High-bandwidth array readout demonstration for microcalorimeter
  - High-dose neutron counter performance test (M2)
  - Thermocouple test with U-Pu alloys.

6.2 Major Research and Development Activities

6.2.1 Safeguards and Security by Design – Electrochemical

Safeguards and Security by Design is a methodology and discipline for integrating next generation MPACT considerations into the design of nuclear facilities from the very earliest stages. The goal is to identify innovative process and facility design features that maximize the effectiveness and efficiency of safeguards and security, and to work with the design team throughout the design process to introduce such features as appropriate, minimizing the need for costly retrofits. Electrochemical processing is being used as the test case for application to advanced fuel cycle technologies, in coordination with the MRWFD campaign and JFCS. Advanced concepts and approaches, analysis tools, and instrumentation are being developed and applied in an integrated manner to optimize the overall system effectiveness.

Electrochemical Process Monitoring for Enhanced Safeguards: Advanced process monitoring instruments (level/density, voltammetry, and direct actinide measurement) are being developed for electrochemical processing as part of the safeguards and security by design effort. Actinide sensor initial ion exchange runs were completed with Pu in 2015 and this year the concept was demonstrated with uranium. Results from actinide sensor tests with U, Pu, and surrogates in molten salt environments have now demonstrated proof-of-concept. Fabrication, calibration and qualification of the second level/density sensor planned in concert with the JFCS was also completed. Having demonstrated performance in molten salts, tests were conducted to address systematic uncertainty reduction in aqueous environments. Results of these tests have shown accuracies of 0.3% or less for density and depth (journal paper in press). An engineering-scale voltammetry probe was designed, fabricated, and tested in LiCl and LiCl-Li_2O salts in preparation for field testing under the JFCS. Finally, the micro-analytic sampling system was demonstrated with actinide bearing salt (UCl_3-LiCl-KCl) and analyzed using XRF, demonstrating the viability of this technique (Figure 27).

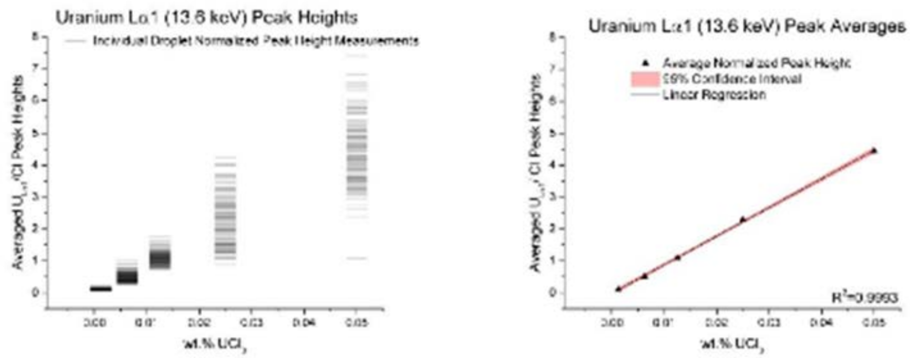


Figure 27. XRF measurements of U-bearing salts using micro-analytic sampler showing reduction in uncertainty for multiple droplets (right) versus single droplet (left).

Modeling and Simulation for Electrochemical Processing Safeguards: Advanced radiation transport calculations (using the MCNP code coupled to application-specific algorithms) have been performed for the for a generic electrochemical processing facility to identify signatures for advanced monitoring instrumentation development, including dynamic models incorporating moving of materials. This year an end-to-end simulation was performed demonstrating the ability to simulate materials moving from the electrorefiner through the process (Figure 28). Additional fidelity in radiation transport simulations can be enabled by mass flow models under development using chemical process models that include dynamics. The Dynamic Electrorefiner (DyER) code is currently being developed for this purpose, and continued to be made more flexible and validated through comparison with experimental data and coupled at the facility level, to the Argonne Model for Pyrochemical Recycling (AMPYRE), which calculates the mass balance of a complete electrochemical processing facility. Given the synergies between modeling and simulation needs for the MPACT and MRWFD campaigns, a joint workshop was held to help identify common needs and areas of collaboration. A number of potential collaboration areas were identified:

- Process flowsheet design and optimization
- Facility design with safeguards and security considerations included
- Training operators and inspectors
- Evaluation of maloperations – intentional and unintentional
- Evaluation of in-process materials hold up
- Evaluation of forensics signatures
- Application to licensing
- Providing an integrated and holistic view of facility operations including uncertainty quantification.

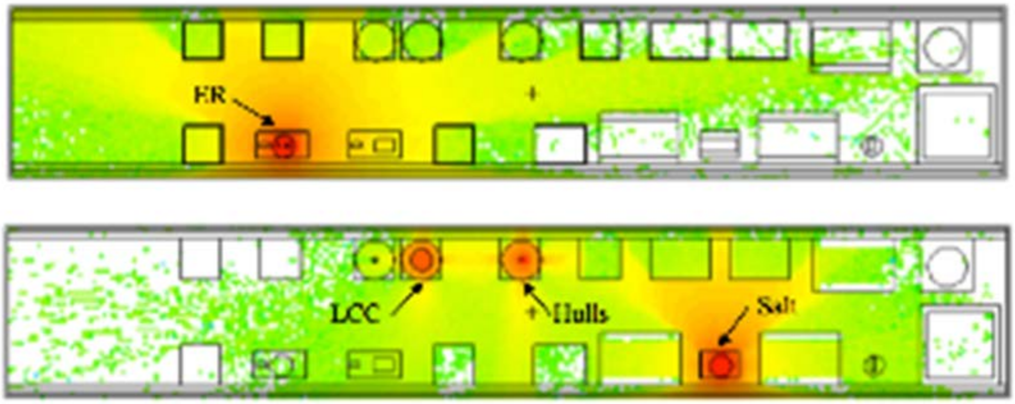


Figure 28. Dynamic radiation signature maps showing material movement effects.

6.2.2 Advanced Integration

A key focus area in FY 2017 was building on and refining the advanced integration roadmap that was completed in 2016, providing a framework for integrating results of the instrumentation and analyses developed in the MPACT campaign over the last several years and ultimately leading to the completion of the 2020 milestone – Lab-scale demonstration of an advanced safeguards and security system (documented in the Implementation Plan for the MPACT 2020 Milestone). These planning activities helped focus R&D efforts in the areas of (1) method development, (2) facility models, and (3) process monitoring. Figure 29 shows the interplay between high fidelity capabilities, system level models, and associated key metrics that form the backbone of the advanced integration approach.

The SSPM, incorporating results from the other modeling activities, including integration of the STAGE to provide physical security modeling capability is a key tool in the advanced integration approach and is relatively mature. Additional inputs and outputs are being defined, on a common process basis, which in turn provides focus for SSPM refinement. Underlying the integration capability is extensible logic modeling, where Bayesian fusion approaches are employed as one way of propagating uncertainties and capturing correlations within the large can complex data sets. One advantage to this approach is the ability to parse and identify contributions to uncertainty, refine as additional data is available, and being able to quantify the overall impacts of assumptions based on subject matter expertise elicitation.

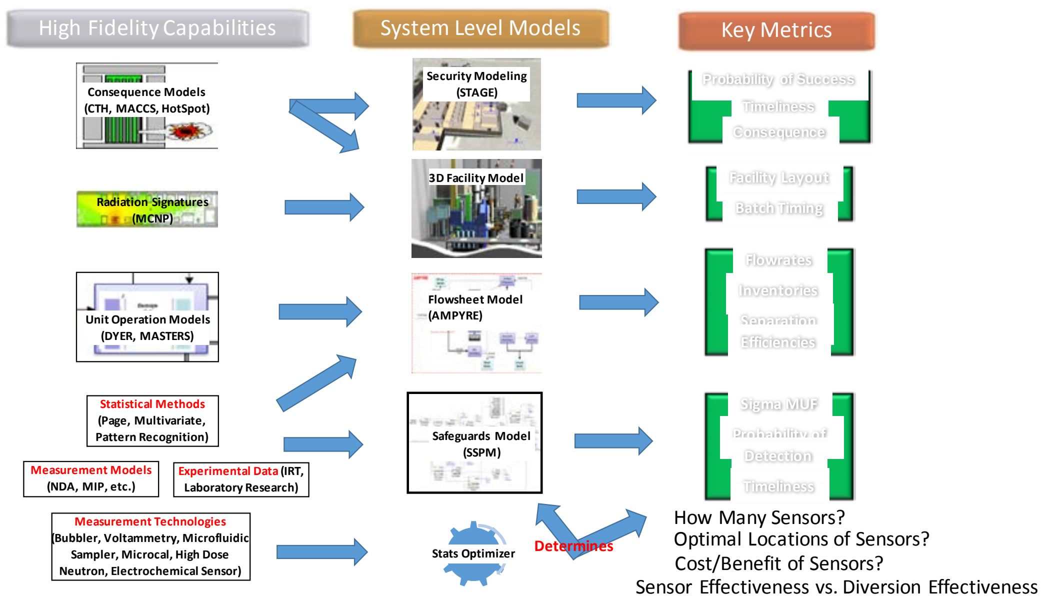


Figure 29. Advanced integration construct using electrochemical processing facility example.

6.2.3 Exploratory Research/Field Tests

Advanced instruments are being developed with new capabilities that will significantly advance the state of the art in nuclear material accounting and control. A focused, innovative, R&D program is being conducted to improve precision and accuracy of traditional methods as well as development of new sensors and methods. As the technical readiness level of these technologies increases, we are planning and executing field tests in fuel cycle facilities to obtain operational experience and demonstrate their effectiveness.

Development of the super-high-resolution gamma spectrometer based on microcalorimetry continued this year, focusing on the assessment of microwave technologies for high throughput readout of the thousands of miniature pixels needed to make a practical detector. Full systems testing of a prototype readout (tens of pixels) with U and Pu materials was conducted in FY 2017, demonstrating the basic concept (collaboration with University of Colorado). This technology is being prepared for field testing in the near future.

A neutron counter that can withstand very high gamma dose is under development using a ^{10}B-lined parallel plate technology originally developed for replacement of the IAEA standard High-Level Neutron Coincidence Counter. In addition to adaptation to high-dose applications, this technology allows for the extraction of average neutron energy, important for complex sample matrix applications such as assay of the product ingot from electrochemical processing. Testing of improved counters in gamma-ray radiation fields greater than 100R/hr this year showed stable neutron counting performance with minimal reduction in neutron detection efficiency. Testing of this technology is planned as part of the JFCS IRT.

In-situ measurement of U/TRU ingot Pu concentration using thermocouples in 2017 showed the ability to perform accurate cooling curve measurements on the outside of the crucible, a key aspect of practical use of the method. Figure 30 shows results for a 29g Ag sample (surrogate for 100g, 50-50 U-Pu) with measurement probes internal and external. Given the simultaneous response of the external probe compared to the internal one, plans are being made to test this method during the JFCS IRT.

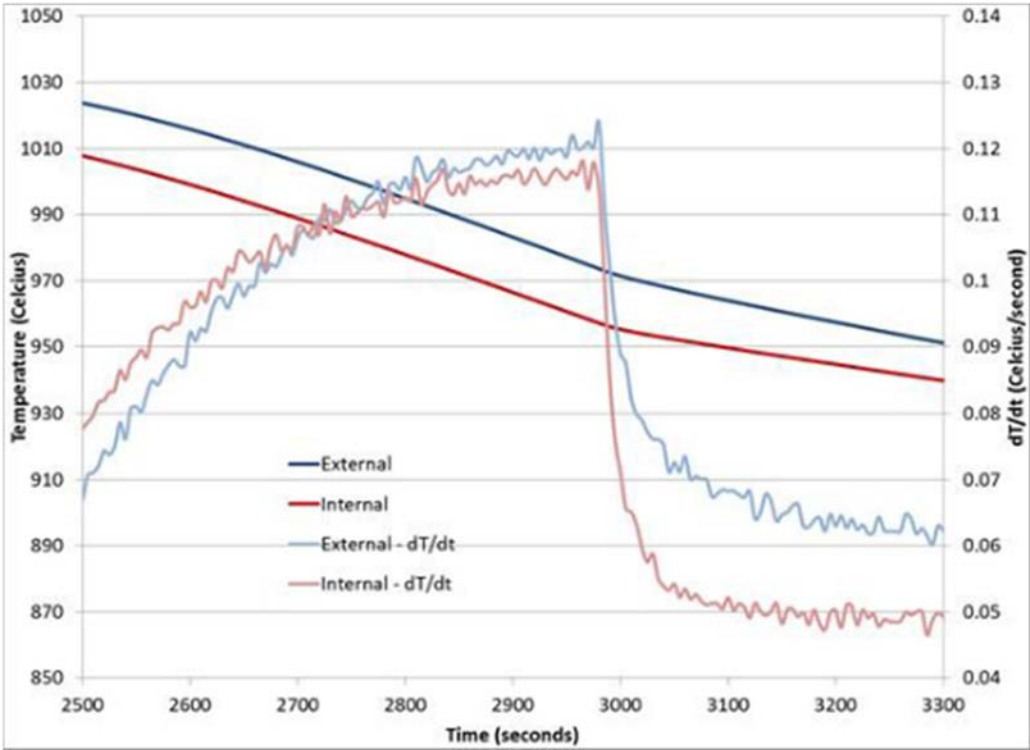


Figure 30. Cooling curve for a 29g Ag sample showing equivalence in sensitivity for external measurement.

Advanced Reactor Program

7. ADVANCED REACTOR TECHNOLOGIES – FAST REACTORS

Robert Hill, ANL, NTD

7.1 Overview

7.1.1 Mission

Advanced reactors face significant technical hurdles to commercialization due to unique, innovative features and lack of a licensing infrastructure for new technology options. The general mission of the ART Program is to identify and resolve the technical challenges to enable transition of advanced non-LWR reactor technologies and systems to support detailed design, regulatory review, and deployment by the early 2030s.

Fast reactors are a key technology for advanced fuel cycles that target benefits in either resource utilization or waste management; favorable features for small reactor applications and plutonium management have also been identified. For the commercial deployment of fast reactor technology, two recurring challenges are identified: (1) cost reduction, given that reactor capital investment is the dominant cost in advanced fuel cycles, and (2) establishment of a licensing pathway.

7.1.2 Objectives

- Research, develop, and demonstrate <u>innovative cost reduction and performance enhancing technologies</u>. Some promising examples include improved system and component design, advanced structural materials, advanced energy conversion, advanced modeling to optimize performance, and improved fuel performance.
- Clarify <u>fast reactor licensing</u> criteria and science-based <u>approach</u> for demonstration of regulatory compliance. This work includes the validation of safety tools to assure margins, refinement of techniques to address key issues (e.g., mechanistic source term), and the qualification of fast reactor fuels.
- <u>Collaborate with industry</u> to identify and conduct essential research to reduce technical risk associated with advanced reactor technologies.
- Develop and sustain the <u>domestic infrastructure and knowledge base</u> within national laboratories and universities to perform needed research. This includes the purposeful training of next generation engineers and scientists by engaging them in advanced reactor concept design and analysis, and NEUP awards that support fast reactor R&D.
- Engage with standards developing organizations to <u>address gaps in codes and standards</u> to support advanced reactor designs.
- Utilize <u>international collaborations</u> to leverage and expand R&D investments through targeted bilateral and multi-lateral R&D agreements.

7.1.3 Challenges

- Investments are needed to re-establish the U.S. infrastructure (facilities and expertise) to support the testing of technology innovations for fast reactor applications.

- Efforts are needed to preserve and manage at risk data, knowledge, and experience related to past U.S. DOE fast reactor design, operations, tests, and component technology.
- Existing fast reactor testing data needs to be captured into modern databases for archival, with details and pedigree appropriately described.
- Fast reactor design and safety methods need to be modernized and validated.
- Extension of ASME code qualification of materials to the fast reactor operating regime needs to be renewed with proper testing, modeling, and new materials for modern applications.

7.1.4 Major R&D Activities

The current scope of the ART fast reactor R&D is structured into three technical areas:

- Technology Development
- Methods, Modeling, and Validation
- Advanced Materials.

Specific activities in each technical area are described in the following sections. Many of the ART fast reactor R&D activities are leveraged through international collaborations, and the broad scope of multi-lateral and bilateral collaborations is separately described in Section 7.5.

A separate project management work area is funded to lead and coordinate the R&D activities which include scientific research and development at national laboratories and universities as well as collaborations with industry and international partners. Working directly with DOE-NE, the NTD will prioritize, coordinate, and oversee execution of the diverse R&D tasks. The Advanced Materials technology area lead has a similar role for materials R&D within the ART Program. Specific project management achievements in FY 2017 included coordination of R&D efforts with the newly organized industry Fast Reactor Technology Working Group, and the creation of a Liquid Metal Fast Reactor (SFR and LFR) Technology Roadmap.

7.2 Technology Development and Demonstration

The ART fast reactor technology objectives are: evaluation and understanding of system integration and performance benefits of various innovations; preserving previous fast reactor R&D knowledge; researching new primary reactor component, sensor, and reliability monitoring technology options; basic experiments to enable and improve the design of compact heat exchangers; creating and sustaining the R&D infrastructure (both personnel and hardware) needed for long-term research, development and demonstration of fast reactor components, instrumentation, and validation testing.

7.2.1 Mechanisms Engineering Test Loop (METL)

METL is an intermediate scale sodium facility designed for the testing of systems and components at prototypic thermal conditions in reactor-grade sodium. In FY 2017, significant progress was made on the construction of the METL facility in Building 308 at Argonne National Laboratory; an overhead view of the installed test vessels is shown in Figure 31. The following activities were completed this year:

- The fabrication and installation of the piping system was completed.
- Electrical resistance heating was installed on the vessels, piping, and components

- Control power was installed in the facility and connected to an outside transformer
- Wiring for the installed electrical resistance heaters was pulled and the heaters were connected to their power supply
- Helium leak checks were performed to ensure appropriate leak tightness
- Insulation was installed on the vessels, piping and components
- The control system was programmed for manual operation
- Preparations were made for the initial heat-up of METL in FY 2018
- An M2 METL Status Report was prepared and delivered to DOE in September 2017

Figure 31. View of METL test vessels.

7.2.2 Gear Test Assembly (GTA)

The GTA is an experimental apparatus designed to test mechanical components used in the fuel handling systems of liquid-sodium cooled fast-spectrum nuclear reactors. The assembly will be placed in a test vessel of the METL which will provide liquefied sodium at controlled temperatures and flow rates for testing. The GTA is designed for maximum flexibility by accommodating various sizes of normal and parallel helical spur gears and roller bearings. The system can also be modified to test helical spur gears, worm gears, and straight or spiral bevel gears as well as hydrostatic or roller bearings with minimal replacement of parts inside the liquid sodium testing area. Resulting data will be taken using vibration probes, torque sensors, tachometers, thermocouples, level sensors, etc. and compared with data recorded by the METL system on sodium flow rates, purity and temperatures. There will also be extensive pre- and post-test metallurgical analysis of the gears to determine the onset and evolution of mechanical failure.

In FY 2017, the GTA (Figure 32) underwent initial testing in distilled water, although galling occurred between a hydro-dynamic journal and a thrust bearing due to low speed “jogging” of the mechanism. The use of hydrodynamic bearings has been abandoned and the components have been modified for use with tapered roller and cylindrical pin thrust bearings. All mechanical components required for the GTA initial testing phases have been manufactured. Upon completion of testing in distilled water, the gear test assembly will be disassembled, cleaned, reassembled and tested in a qualification station to measure its leak tightness before transfer to METL in Building 308. An end-of-year milestone report was prepared and submitted to DOE regarding the status of the Gear Test Assembly development and testing.

Figure 32. Gear test assembly.

7.2.3 Under Sodium Viewing (USV) Technology

Currently there is no reliable inspection/monitoring method for sodium-cooled fast reactors in-vessel components due to challenges associated with liquid metal conditions (opaque coolant, high temperature, high radiation, and corrosive environment). The USV technology being developed will play a critical role in safe operation of advanced reactors. It could also be applied for other I&C applications of SFRs, such as in-situ, real-time passive or active steam generator leak detection/location, and sodium boiling detection and/or location for fuel-pin failure and reactor core monitoring.

During the past decade of R&D work on ultrasonic viewing, a variety of ultrasonic waveguide transducers (UWTs), including single UWTs, linear UWT arrays, brush-type UWTs (BUWTs), and brush-type ultrasonic waveguide phased array (BUWT-PA) transducers, were developed. A USV test facility was constructed at Argonne for automated in-sodium test, signal/image processing, and defection detection. The ART Program has successfully demonstrated UWT techniques with real-time detection resolutions of 0.5 mm in both width and depth up to 343°C in sodium. Submergible high-temperature transducers with real-time detection resolutions of 0.5 mm in depth and 1 mm in width up to 343°C in sodium have also been successfully demonstrated.

In FY 2017, a submergible high-temperature prototype was developed and evaluated using $LiNbO_3$ transducer elements with different backing materials for better detection resolution, faster sodium wetting, and higher operating temperature (>500°C). We also successfully tested single-element, multiplexing, and electronic beam steering of an 8- and 32-element BUWT-PA in water. By integrating BUWT and phased array (PA) techniques, Argonne developed a BUWT-PA that showed real-time detection resolutions of 0.5 mm in depth and 0.5–2 mm in width with mechanical scanning method in water mockup tests, as shown in Figure 33. Additional water mockup tests were conducted to optimize the beam steering to achieve higher beam intensity, better beam focusing, enhanced detection resolution, and faster inspection. It was demonstrated that electronic steering would reduce inspection time and more signal averaging would improve signal-to-noise ratio but with longer inspection time. An integrated final report of the ultrasonic USV systems developed at PNNL and ANL was completed and submitted. A technical paper, entitled “Development and demonstration of ultrasonic USV system for SFRs,” was submitted to the International Conference on Fast Reactors and Related Fuel Cycles (FR17).

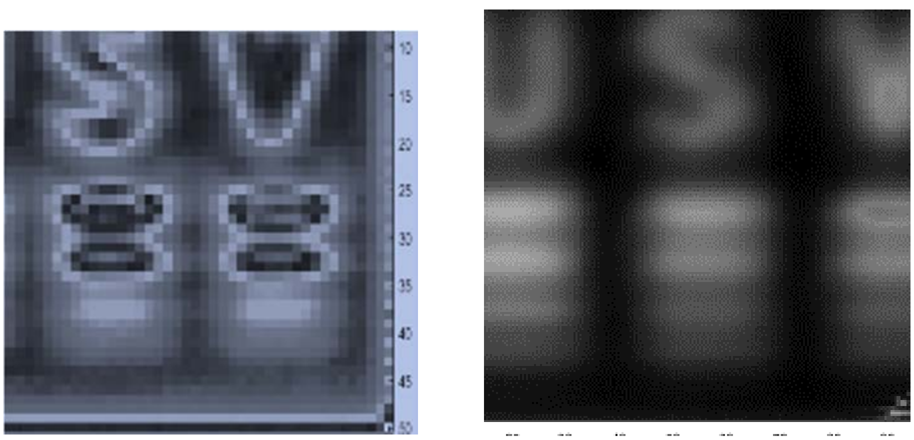


Figure 33. Intensity image generated by a 32-element BUWT-PA using 5-element focusing and mechanical scanning with signal average: 20 (left) and 20,000 (right).

USV experts collaborated with Westinghouse to evaluate the potential use of similar ultrasonic devices at higher temperature (>400°C) for lead-cooled fast reactor. We are also collaborating with Westinghouse to leverage the USV facility for evaluation and validation of thermoacoustic power sensor for potential use in monitoring temperature and neutron flux in SFR core.

7.2.4 Heat Exchanger Phenomena

To reliably design compact sodium heat exchangers, fundamental data is needed on the draining and refilling of sodium from compact heat exchanger channels should the intermediate sodium circuit be drained so that sodium is not retained inside of the heat exchanger where it could result in unwanted stresses from sodium freezing and thawing or unwanted oxidation of sodium forming high melting temperature plugs. A fundamental understanding of the potential stresses resulting from inadvertent freezing and remelting of sodium is also needed. Fundamental data is needed on the plugging of narrow sodium channels due to the precipitation of dissolved oxygen from sodium when the sodium temperature decreases from an initial value to a nonvarying final value. Oxide precipitation and plugging could occur at the cold end of a sodium heat exchanger, if the sodium becomes contaminated with oxygen due to ingression of air with failure of the cold trapping purification circuit.

In FY 2017, assembly of the sodium draining and refilling experiment was continued, and shakedown and draining tests using water were conducted. In early FY 2018, remaining assembly of the experiment facility has been completed and additional water draining tests are being conducted and observed with high speed high definition video. Initial draining and refilling tests with sodium will be conducted in wetted stainless steel circular tubes. In FY 2017, assembly of the Sodium Freezing and Remelting Experiment (see Figure 34) was completed and the first sodium freezing test was conducted. Measured strains showed that the sodium pulled inward on the stainless steel test section wall as the sodium froze and cooled down. A new concept was developed for renovating the Sodium Plugging Phenomena Loop incorporating past experience and lessons learned to avoid inadvertent plugging.

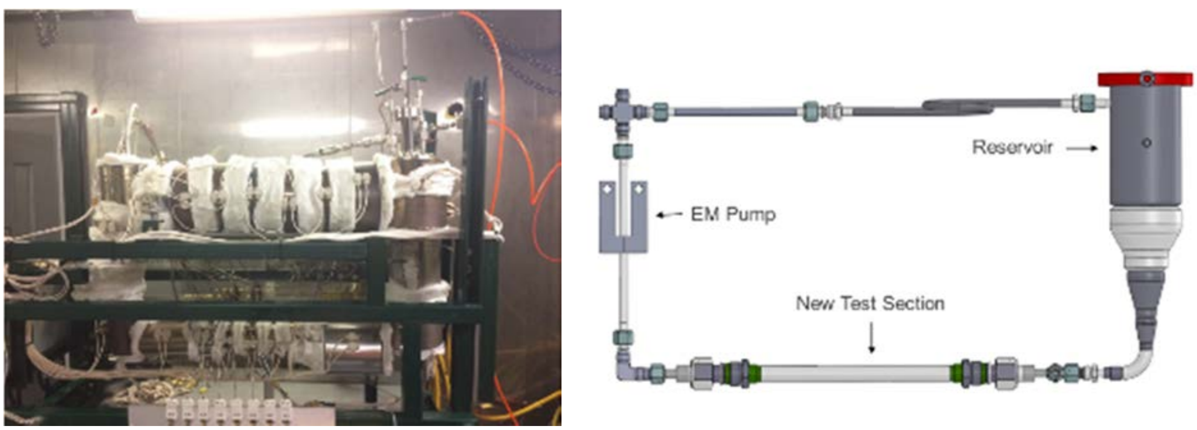


Figure 34. Sodium freezing and remelting experiment.

7.2.5 Fast Reactor Knowledge Preservation

The Fast Reactor area of the DOE ART Program has been leading an effort to preserve information on fast reactor technology for the past 10 years. This effort is essential to re-establishing the technology base for fast reactors going forward in the U.S. The recent effort has included recovery and conversion of hard copy information located in OSTI and the preparation of lessons learned from the operations and maintenance of the FFTF. A milestone report was prepared (3rd in a series) on the FFTF lessons learned that included:

1. Reactor Physics Startup Testing

2. Control Rod Absorber Assemblies and Boron Carbide Pellets in FFTF
3. Integrated Leak Rate Testing
4. Sodium Spill and Fire Testing at the Hanford Site
5. Recording, Archival and Recovery of FFTF Data
6. Designing for Ease of Decommissioning
7. Removing Non-Drainable Sodium from the FFTF Reactor Vessel.

7.3 Methods, Modeling, and Validation

In the Methods area, the objectives are: addressing high priority modeling capabilities to improve existing design capabilities and sustain U.S. tools for utilization by industry and international partners; conducting benchmark and analysis studies to verify and validate fast reactor design and safety analysis codes and simulation techniques; addressing key fast reactor regulatory gaps, as identified in the DOE-NE Regulatory Technology Development Plan; leveraging modeling and validation R&D and promoting inherent safety approach through international collaborations.

7.3.1 Fast Reactor Database Development

The ART Fast Reactor Methods, Modeling, and Validation R&D Program supports the development of several databases to archive and organize data needed for metallic fuels qualification and validation of the state-of-the-art codes and advanced methods for design and analysis of fast reactors:

1. The EBR-II test database encompasses archival of data from nearly 100 tests and calibration runs performed during the 1984-87 timeframe with considerable influence on inherent safety of today's advanced fast reactor concepts. The spectrum of tests includes loss-of-flow with scram to natural circulation, scram with delayed loss-of-flow to natural circulation, reactivity feedback characterization, loss-of-flow without scram, loss-of-heat-sink without scram, dynamic frequency response, and steam-drum pressure reduction assessments.
2. The FFTF Passive Safety Testing Database initially aimed to archive the data from thirteen tests that demonstrated the safety margins during unprotected accidents. Of particular interest was a series of loss-of-flow without scram tests from power levels up to 50% to demonstrate the effectiveness of a reactor self-shutdown device called the Gas Expansion Module. More recently, the PNNL team's effort expanded the database to include FFTF start-up tests to measure various reactivity feedback coefficients and thermal-hydraulic response of the reactor.
3. TREAT Test Database is an archive of data from hundreds of experiments conducted at the TREAT facility during its 35 years of operation from 1959 to 1994. The experiments in TREAT were performed to support the development of fuel for numerous types of nuclear reactors. Specifically, the experiments investigated the response of various nuclear fuels and fuel element designs to off-normal and accident-related transients, with and without the presence of various types of coolants. The responses ranged from minor fuel damage to gross fuel meltdown.
4. Fast Reactor Fuels Irradiation and Physics Database aims at preserving the data from fuels irradiation tests during the past U.S. fast reactor R&D programs, in particular the metal-alloy fuels irradiated in the EBR-II and FFTF reactors. The database is intended to support fuels qualification program during a future fast reactor license application as well as development and validation of fuel performance

codes. The information maintained in the database includes detailed pin-by-pin data fabrication data, operational history, PIE information, and documentations related to specific experiments. The selected experiments cover wide range of fuel performance information, including prototype fuel behavior and failure mode during operational modes, fabrication parameters, lead tests, high temperature swelling behavior, and fuel-cladding mechanical interaction.

5. Finally, the SFR Component Reliability Database development activity aims to re-establish the database on structures, systems and components failure rates. Lost after termination of the IFR program, this activity resurrects the old CREDO database and expands its scope using additional records from operational logs of EBR-II, FFTF, Sodium Reliability Experiment, and Fermi-I reactor.

7.3.2 Fast Reactor Code Modernization

The focus in FY 2017 was on the SAS4A/SASSYS-1 safety analysis software which is used to perform deterministic analysis of anticipated events as well as design-basis and beyond-design-basis accidents for advanced fast reactors. It plays a central role in the analysis of U.S. DOE conceptual designs, proposed test and demonstration reactors, and in domestic and international collaborations.

During FY 2017, modern software quality assurance procedures were implemented to support rigorous configuration management. DOE-NE authorized the extension of copyright for an additional five years, which was essential so that Argonne can continue developing, distributing, and supporting the software for both domestic use and international collaborations. SAS4A/SASSYS-1 Version 5.2 was completed in March 2017 and released to users in May. Following the release of Version 5.2, additional extensions to the void and cladding reactivity feedback models were implemented, and Control System capabilities have been improved through a new virtual data acquisition system for plant state variables. In addition, a new Block Signal for a variable lag compensator was added to represent reactivity feedback for novel shutdown devices.

The importance of SAS4A/SASSYS-1 maintenance reflects its relevance to a number of U.S. Department of Energy programs as well as domestic and international collaborations. External collaborators have also funded improvements in the SAS4A/SASSYS-1 code, and these are shared with all users. SAS4A/SASSYS-1 has a growing user base that continues to strengthen the promotion of advanced non-LWR reactor concepts. Additional users will help solidify DOE’s leadership role in fast reactor safety both domestically and in international collaborations.

7.3.3 FFTF Benchmark Development

Recent IAEA Coordinated Research Projects have provided an important international context to evaluate fast reactor safety behavior, validate safety analysis codes, and highlight inherent features to prevent severe accidents. For this purpose, the U.S. has contributed EBR-II passive safety test data. In FY 2017, a new benchmark was created based on the FFTF passive safety tests.

FFTF at Hanford, Washington was a 400 MW-thermal, oxide-fueled, sodium-cooled test reactor, built to assist development and testing of advanced fuels and materials. In July 1986, a series of unprotected transients were performed in FFTF as part of the passive safety testing program. Among these tests were thirteen unprotected loss of flow without scram tests. The goals of this program included demonstrating liquid metal reactor safety margins, providing data for computer code validation, and demonstrating the inherent safety benefits of specific design features.

The most severe of these tests, LOFWOS Test #13 which was initiated at 50% power and 100% flow with the pump pony motors left off, has been proposed (and in process of being approved) as an IAEA-Coordinated Research Project. In preparation for this project, Argonne and PNNL teams have undertaken a major effort in FY 2017 to complete the benchmark specification for this test. This benchmark specification is intended to support collaborative efforts with international partners on the validation of simulation tools and models in the area of sodium fast reactor passive safety. Validated tools and models are needed to verify the safety of SFR designs and assess a reactor's ability to incorporate the passive safety features into a system's inherent response to accident initiators. Comparisons with experimental data and other safety code predictions create unique opportunities to improve SFR computational codes and methods. The conditions of the subject test, along with the reactivity feedbacks from the Gas Expansion Module and FFTF's limited free bow core restraint system, create the conditions for a challenging benchmark exercise. The IAEA coordinated research project is expected to kick-off in October 2018 as a four-year program with wide participation of international R&D organizations.

7.4 Advanced Materials

The objective of the ART advanced materials R&D is to provide the technical basis needed to support the regulatory requirements for structural materials required for fast reactors that could be deployed in the near-to-mid-term. Activities include the qualification of higher performance structural materials for fast reactor construction to expand reactor design envelopes, and the extension of qualified lifetimes and usage temperatures of structural materials already approved within the ASME Code for construction of fast reactors. The activities also include improving high temperature design methodologies for use of the qualified materials under elevated temperature cyclic service of fast reactors.

7.4.1 Alloy 709 Procurement and Testing

Alloy 709 is derived from NF709 (Fe-20Cr-25Ni-1.5Mo-Nb,B,N), which was a commercial heat- and corrosion-resistant austenitic stainless steel developed by Nippon Steel Corporation in Japan in seamless tubing product form for power boiler applications. Alloy 709 provides time dependent strength nearly double that for conventional 304 and 316 stainless steels at sodium fast reactor relevant temperatures and very long design lifetimes.

During the advanced alloy down selection and subsequent intermediate term testing program for fast reactors, hot-rolled and hot-forged Alloy 709 plates from small heats (50 to 400 lbs), melted with electro-slag remelt process followed by homogenization heat treatment, were produced by specialty fabricators in the U.S. and Japan to support testing. The hot working and heat treatment conditions used by Nippon Steel for seamless tubing were optimized by the ART Program to maintain the creep strength of Alloy 709 plates but with improved creep-fatigue resistance. Alloy 709 is intended for sodium fast reactor applications that include reactor vessel, core supports, primary and secondary piping, and possibly intermediate heat exchanger and compact heat exchanger. Hence, development of processing conditions and fabrication scale up for different product forms such as plates, pipes, bars, forgings and sheets, in addition to seamless tubing, are required.

A U.S. commercial vendor in Pennsylvania was contracted to melt an Alloy 709 heat using industrial melt practice. A master heat was electric-arc-melted, refined by the argon-oxygen-decarburization process, and then bottom-poured into two rectangular ingots and two round ingots, with a total weight of about 45,000 lb. Some of these ingots were further processed by electro-slag remelt and homogenization heat treatment. The chemical compositions of the ingots were all within the specification and met the chemistry aims provided by the ART Program.

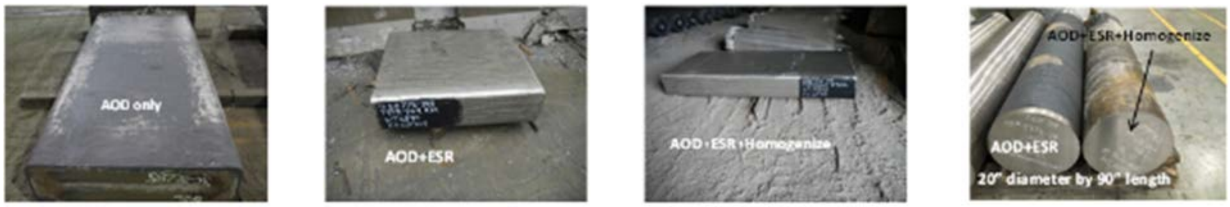
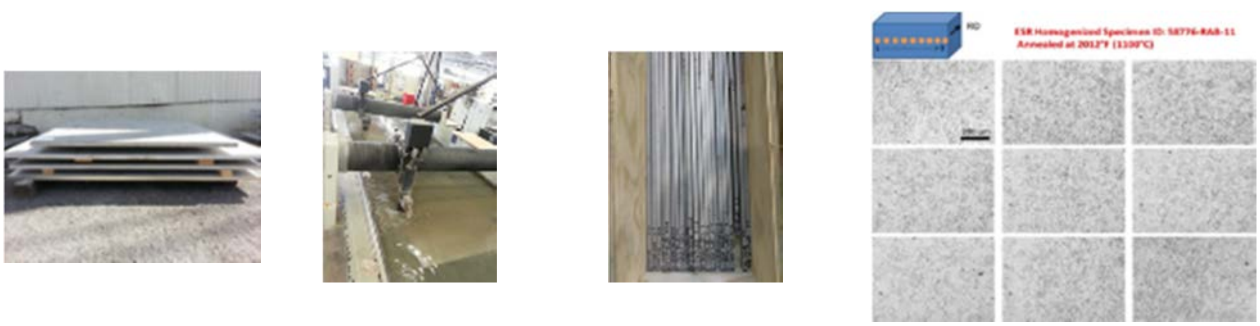


Figure 35. Alloy 709 rectangular and round ingots at vendor shop floor after fabrication.

While a uniform hot working temperature of 1100°C was found to produce the desired microstructure in the plates from small heats, the rectangular ingots with different melting conditions were hot rolled from a start temperature of 1200°C to finishing temperatures in the range 986 to 1029°C, using standard mill rolling practice. The as-rolled plates were then cut into three sections and were solution annealed at 1050, 1100 and 1150°C, respectively, to provide test materials to study the effects of post-fabrication heat treatment. Mechanical properties and microstructure screening of the Alloy 709 plates were performed. The room temperature mechanical properties determined from the rolling and transverse directions were above the specification minimums and the grain size distributions were found to be more uniform than those from the small heats previously procured. Elevated temperature creep, fatigue and creep-fatigue property screenings have been initiated.

The ART Program has also stood up the creep testing capabilities at ANL, INL, and ORNL to support the Alloy 709 ASME Section III Division 5 Code Case effort. The design of a sodium materials test loop with large exposure vessels to accommodate ASTM standard size test specimens was completed and some long lead components have been purchased. Loop construction can proceed immediately once funding ($1.3M) can be identified.



Some of the delivered plates (a)	Waterjet cutting at ORNL (b)	Strips for microstructure screening (c)	Micrographs showing good microstructures (d)

Figure 36. (a) Set of Alloy 709 plates delivered to ORNL, (b) plates being waterjet cut at ORNL to obtain strips for microstructure screening, (c) ss-waterjet-cut strips, and (d) sample micrographs showing good microstructures.

7.4.2 Extension of Grade 91 Design Life to 60 years in ASME Division 5

The intermediate heat exchangers (IHXs) occupy a substantial amount of space within the reactor vessel for pool-type plants. In past sodium fast reactor designs, the IHX construction material was Type 304 stainless steel. Grade 91 steel was specifically developed for high temperature heat transfer apparatus taking advantage of its high yield strength to minimize ratcheting and strain accumulation, low thermal expansion coefficient to minimize thermal stresses and good creep rupture strength, particularly for very long design lifetime, to minimize required height and wall thickness. These favorable Grade 91 properties would lead to a much more compact IHX design as compared with Type 304 stainless steel at the same heat transfer rate. Construction cost would be reduced with smaller IHXs. A smaller IHX footprint would in turn allow the deployment of a smaller reactor vessel and further driving down construction cost. Similar benefit applies to the construction of air dump heat exchanger using Grade 91 steel.

The current design lifetime of Grade 91 in the ASME Section III, Division 5 Code is 300,000 h. Extension of the design lifetime of Grade 91 steel to 500,000 h would permit designers to effect a 60-year fast reactor designs that would reduce levelized energy cost. Efforts from the base program of the Fast Reactors Campaign, as well as international collaboration and contributions from ASME Code committees, are leveraged for the Grade 91 design lifetime extension.

1. Progress in FY 2017 includes the development of microstructural evolution models to predict thermal aging effects on tensile properties. Some U.S. archived Grade 91 materials that were exposed to thermal aging conditions ranging from 50,000 to close to 150,000 hours were recovered from ORNL. In-situ and ex-situ tensile tests and microstructural characterization were performed on these materials to generate long-term data to validate the predictive thermal aging models.
2. The development of micro-mechanical finite element modeling of high temperature creep fracture mechanisms of Grade 91 to understand the potential for mechanisms change under long term operating conditions was continued. The grain boundary cavitation and sliding mechanisms were integrated with the grain deformation mechanisms to simulate the deformation and damage of Grade 91 under creep loads. New mechanics insight was obtained relating to the change from notch strengthening to notch weakening behaviors under creep conditions.
3. Testing of key feature test articles and numerical analyses were initiated in an effort to extend the elastic perfectly-plastic (EPP) strain limits and creep-fatigue evaluation procedures in the ASME Code to Grade 91 structural components. The extension is necessary as the EPP methods were qualified for cyclic hardening materials such as stainless steels and not for cyclic softening materials such as Grade 91. Testing and model formulation to support the development of engineering scale, finite element-based inelastic analysis method were also initiated. The inelastic analysis method is used in the ASME Code to address “hot spots” in structure components that experience high strain accumulations and creep-fatigue damages due to the severe thermal transients in sodium fast reactor systems.

7.4.3 Other ASME Code Qualification Activities

Efforts to extend the design lifetimes from 300,000 to 500,000 h for the other qualified materials in ASME Division 5 Code are also being pursued. The materials include 304H and 316H stainless steels, Alloy 800H austenitic steel, and 2¼ Cr-1 Mo ferritic steel. The stainless steels and ferritic steel are the reference construction materials for the versatile test reactor and other sodium fast reactor designs (e.g., GE-Hitachi’s PRISM design and TerraPower’s Traveling Wave design). Engineering scale, finite element-based inelastic constitutive models are also being developed for these materials to support inelastic design analyses following the procedures of the ASME Division 5 Code.

7.5 International Collaborations

7.5.1 IAEA Coordinated Research Project on EBR-II Benchmark

During 2013–2017, Argonne hosted an IAEA-coordinated research project on benchmark analyses of sodium-cooled fast reactor inherent safety demonstration tests performed at the EBR-II. The project involved analysis of the most severe protected loss of flow and an unprotected station blackout tests conducted during an extensive testing program to demonstrate the inherently safety features of a pool-type, sodium-cooled fast reactor.

The four-year project was executed in three phases involving the blind analyses, model refinements based on comparison with the experimental data, and sensitivity studies with participation of 20 R&D organizations from 12 countries as the largest IAEA coordinated research project on fast reactors. Argonne's role included preparation of the benchmark specifications, detailed analyses utilizing U.S. modeling tools, coordination of the participants' contributions, and lead technical role in preparation of the final IAEA technical document.

7.5.2 IAEA Coordinated Research Project on Sodium Properties

Argonne National Laboratory continued to lead the drafting of two sodium property handbooks for the IAEA Coordinated Research Project (CRP) on Sodium Properties (NAPRO). A total of 11 organizations from 10 countries participated in this CRP. The two property handbooks are entitled: "Sodium Coolant Handbook: Physical and Chemical Properties" and "Sodium Coolant Handbook: Thermal-Hydraulic Correlations." At the end of FY 2017, these two handbooks total 756 pages not including references or front matter. A total of 893 references were collected and documented. Several publications were produced in FY 2017 including a conference paper at ICAPP-2017 and two conference papers at FR-2017. The two handbooks are undergoing final IAEA internal review.

Despite the assumption that sodium properties were considered "established" by many experts, the CRP identified and documented in these handbooks that there were a number of inconsistences and gaps that are relevant to both computation and experimental applications. In particular, further development of best-estimate and high fidelity/physics based simulation codes require accurate, complete, consistent, precise and reliable sodium property data sets. Natural circulation and source term modeling are particularly sensitive to thermodynamic properties and thus uncertainty quantification will be important to mature our understanding of these phenomena.

7.5.3 Generation-IV SFR System Steering Committee and System Integration Project

The System Steering Committee (SSC) provides coordination of the sodium-cooled fast reactor R&D collaborations within Generation-IV. Member countries include China, EURATOM, France, Japan, Korea, Russia, and the U.S. The SSC also handles the interface with the GIF Policy and Expert Groups, GIF Task Forces, GIF Methodology Working Groups, and other SSCs. The System Integration and Assessment Project includes the same members and provides technical oversight of the GIF SFR technical projects. In addition, Generation-IV SFR design options are identified (e.g., BN-1200 was recently added as a design track) and system trade studies are shared. In FY 2017, the U.S. contributed a joint U.S.-Japan summary report based on the recent metal fuel alternatives study.

7.5.4 Generation-IV SFR Safety and Operations Project

The GenIV Safety and Operations Project Arrangement focuses on computational methods, simulations, experiments, and studies of innovative designs that have direct impact on the safety of sodium-cooled fast reactors. During 2017, the US DOE contributed deliverables in two areas: 1) the development of mechanistic source term capabilities, and 2) the development of an advanced, reduced-order three-dimensional modeling capability to represent thermal flow phenomena such as thermal stratification.

The mechanistic source term calculations were performed for a metal fuel, pool-type sodium fast reactor. Calculations were performed utilizing best-estimate models and data to identify potential gaps in the current knowledge base. The results of the analysis predicted small offsite doses and demonstrated that a mechanistic source term calculation is possible utilizing current modeling tools and data. However, gaps in available data and tools result in uncertainties or the use of conservative assumptions that could make it difficult for future SFR vendors to reduce site boundaries and emergency planning zones.

7.5.5 Generation-IV SFR Component Design and BOP Project

Argonne participated in two Project Meetings; the first held at CEA in Cadarache, March 7-10, 2017, and the second at Argonne, September 12-15, 2017. Current Project members are CEA (France), JAEA (Japan), KAERI (Republic of Korea), and DOE/ANL (U.S.). Research and development areas include the supercritical carbon dioxide Brayton cycle, nitrogen gas Brayton cycle, in-service inspection, repair and upgrading, leak before break methodology development, sodium-heated steam generators, and sodium facilities. At the meeting in September, Argonne provided an update on validation of the ANL Plant Dynamics Code with data from the Naval Nuclear Laboratory Integrated System Test. A major accomplishment in FY 2017 was extension of the Project Arrangement for another ten years until October 11, 2027, with a signed letter from Raymond V. Furstenau of DOE on September 20, 2017. China is expected to join the project in the near future.

7.5.6 Generation-IV SFR Safety Design Criteria Task Force

The GIF task force was established in 2011 to develop Safety Design Criteria (SDC) and Safety Design Guidelines (SDGs) for Sodium-cooled Fast Reactors. The main objective was to prepare reference criteria and guidance for the design of structures, systems and components to achieve the safety goals of the Generation-IV SFR system design tracks. The task force mission was to bridge the gap between the high-level safety fundamentals and the more detailed national codes and standards by developing the reference criteria and guidelines. In 2013, the SDC report was completed and distributed to various international organizations and national regulators for review and feedback. Based on comments received during the following two-year period, the report underwent a significant revision reflecting the feedback received from IAEA, U.S. NRC, IRSN (France), and NNSA (China).

In parallel to the SDC report update, the task force has also initiated an effort on the development of safety design guidelines to provide recommendations on how to comply with the safety design criteria and present examples of good practices to help the designers to achieve high level of safety in specific topical areas. The first SDG report on “Guidelines on Safety Approach and Design Conditions of Generation-IV SFR systems” addressed the reliance on inherent/passive safety features and the design measures for prevention and mitigation of severe accidents, and was completed in 2016. The second report on “Safety Design Guidelines of the Key Structures, Systems and Components” is an ongoing effort intended to address the neutronic, thermal, hydraulic, mechanical, chemical and irradiation considerations that are important for the safe design of the reactor core, coolant systems, and containment systems for Generation-IV SFRs.

7.5.7 Japan Bilateral Collaboration (CNWG) - Advanced Materials

A collaborative effort with JAEA in the fast reactor materials area under the Civil Nuclear Energy Research and Development Working Group (CNWG) of U.S.-Japan Civil Nuclear Cooperation (Bilateral Commission) was initiated in 2013. The objective was to conduct R&D on advanced materials in support of Code qualification and codes and standards development required to apply the materials to sodium-cooled fast reactors. Grade 91 steel and its associated weldments are the focus. The Phase I program was completed in 2016 and the collaboration was extended for another five years for a Phase II program.

This collaboration has been leveraged to support the extension of Grade 91 design lifetime to 500,000 h. The technical basis of the Japan Society of Mechanical Engineers (JSME) temperature dependent fatigue design curves was jointly investigated by the Japan Atomic Energy Agency (JAEA) and the ART Program and the recommendation to replace the single fatigue design curve in the ASME Code by the temperature dependent JSME curves was formulated. This would expand the creep-fatigue design envelopes for Grade 91 structural components.

7.5.8 Japan Bilateral Collaboration (CNWG) - Metal Fuel

A core design study was conducted to compare the core performance characteristics of metal-fueled Sodium-cooled Fast Reactors (SFRs) that are developed with different design preferences: JAEA prefers a loop-type primary system with high coolant temperature (550^{o}C), while ANL targets a pool-type primary system with a conventional coolant temperature (510^{o}C). The comparative core design study is conducted based on the 3530 MWth Japan Sodium-cooled Fast Reactor (JSFR) metallic-fuel core: the core configuration is redesigned for accommodating the design preferences.

Both metal-fueled SFR core concepts developed display similar core performance: comparable breeding ratio, fuel discharge burnup, neutron spectrum, peak fast flux fluence, and reactivity feedback coefficients. The safety impact originating from different design preferences are not found to be significant since similar nominal peak cladding temperatures and satisfactory inherent safety behaviors are demonstrated. The resulting differences in thermal efficiency and core compactness between these two core concepts impact the reactor economics, but it was difficult to quantify the economic tradeoffs. As a consequence, this study confirms that both metal fueled SFR core concepts developed by ANL and JAEA based on different design preferences and approaches are viable options.

7.5.9 Japan Bilateral Collaboration (CNWG) on Advanced Modeling and Simulation

Advanced modeling and simulation tasks under the U.S.-Japan bilateral collaboration mainly focus validation of SFR plant dynamics analysis capabilities during postulated accidents. The tasks are carried out by sharing experimental data, mutually complementary exchange or joint development of simulation tools, joint benchmark exercises, and sharing technical knowledge to contribute to the development of comprehensive and well-validated advanced predictive analytical capabilities for SFR plant dynamic behavior including in-vessel/ex-vessel event evaluations.

One of the objectives of the advanced modeling and simulation projects is the V&V of coupled system+CFD analysis capabilities to support the use of a plant system analysis code in conjunction with the higher-fidelity methods to address multi-dimensional mixing and heat transfer in SFR lower and upper plena. In FY 2017, the efforts focused on analysis of JAEA's PLANDTL test and initial assessments of MONJU plant turbine trip tests at Argonne. In return, JAEA team focused on the analysis of EBR-II SHRT-17 test using their respective coupled system-CFD analysis capabilities.

Another focus area was the enhancement of the ex-vessel accident analysis program CONTAIN-LMR that was originally developed in Sandia National Laboratories (SNL) and subsequently refined by JAEA. The continued CONTAIN-LMR validation efforts at Sandia and JAEA focused on V&V of the sodium spray and pool fire analysis capability and sodium-debris-concrete interaction model based on experimental data available at JAEA and Sandia.

7.5.10 I-NERI Collaboration between U.S. and ROK

Started in FY 2014 as a multi-year collaboration and completed in FY 2017, the objectives of this project were to build a physics validation database based on good quality data from ZPPR-15, EBR-II, BFS experiments, and to validate the Argonne fast reactor design code suite (MC2-3/DIF3D/REBUS-3) using the database. The initial U.S. work scope involved building MCNP models for ZPPR-15 physics experiment and EBR-II depleted fuel data to perform uncertainty evaluations on gamma dose, neutron spectrum, and foil reaction rate. KAERI's initial work scope involved building MCNP models for BFS-109-2A and BFS-76-1A physics experiments (as the most recent tests conducted in 2010 and 2012 at IPPE, Russia and paid by KAERI) and to perform uncertainty evaluations. In FY 2017, both Argonne and KAERI teams completed verification & validation studies using the Argonne fast reactor design code suite (MC2-3/DIF3D/REBUS-3) based on the ZPPR-15, EBR-II, and BFS databases. Generally, excellent code-to-code agreements achieved, and the agreement with the test data varied in the range of 5-30% depending on the type of feedback effect measured and the location of the measurements.

7.5.11 France Bilateral Collaboration on ASTRID Fast Reactor Concept

Under a framework agreement between the U.S. and France for cooperation on low carbon energy technologies, Argonne National Laboratory performed coupled systems-CFD transient benchmark analyses of the preliminary ASTRID design. Coupled simulations performed during FY 2017 demonstrate the effect of temperature variations within large volumes on natural circulation flow rates.

CEA defined two protected loss of flow transient scenarios for the collaboration. For the first scenario, both the primary and intermediate sodium loops transition to natural circulation and large margins to sodium boiling are maintained during both the standalone and coupled simulations. For the second scenario, the intermediate sodium loop is assumed to drain resulting in a sudden and complete loss of heat rejection for the primary loop. So far, only stand-alone calculations have been completed because boiling occurs in the primary loop due to a lack of natural circulation.

8. ADVANCED REACTOR TECHNOLOGIES – GAS-COOLED REACTORS

Hans Gougar, INL, NTD

8.1 Overview

The ART program raises the technological readiness of promising advanced (non-Light Water Reactor) reactor concepts. The types of R&D performed under ART supports broad families of reactor concepts rather than specific designs. These activities require expertise and resources, such as materials test reactors and hot cell testing facilities that are generally not available in private industry and would be too risky for any one company or consortium of companies to develop. Towards this goal, the ART program works with laboratories, universities, and vendors to identify, prioritize and conduct R&D that builds the technological platform upon which advanced reactor designs can be developed and commercialized.

The ART program establishes and strengthens collaborations with experts at international agencies, universities, and industry to develop future nuclear power systems. DOE-NE national laboratories and their international counterparts provide extensive expertise, experience, and access to critical R&D facilities. Universities provide a wellspring of innovative ideas, while industry provides design, engineering, and construction expertise and resources needed to turn the base technology into a viable commercial product.

In 2016, DOE announced the Gateway for Accelerated Innovation in Nuclear (GAIN) program, which was established to facilitate nuclear energy community access to the technical, regulatory, and financial support necessary to advance nuclear designs toward commercialization, while ensuring the continued safe, reliable, and economic operation of the existing nuclear fleet. The ART program supports this important effort with its R&D program, working closely with vendors via the Advanced Reactor Working Group and concept-specific Technology Working Groups that help to establish priorities and R&D goals.

8.1.1 Mission

The Gas-Cooled Reactors Campaign (ART-GCR) seeks to raise the technological readiness of HTGRs and other reactor concepts utilizing coated particle fuel, graphite, or helium coolant by performing high risk, enabling research and development that commercial developers are unable or unwilling to undertake in the current energy market.

The campaign focuses on helium-cooled, high temperature reactors that are generally moderated by graphite and cooled by helium. This reactor concept is distinguished by its lower power density, high thermal inertia, and unique coated particle fuel form that enable an unprecedented level of inherent safety. The base technologies underlying gas-cooled reactors were demonstrated in the 1980s, but commercial efforts faded by the mid-1990s. Today's modular GCR designs will require higher levels of performance from fuels, materials, and analytical methods in order to be licensed and to be economical in a challenging energy landscape. Both pebble bed and prismatic high temperature gas-cooled reactor concepts, as well as some molten salt-cooled concepts are supported by the campaign.

8.1.2 Objectives

The objectives of the ART- GCR program fall into the following four primary technology areas.

- Fuel Qualification – Production scale fabrication, irradiation testing, and post-irradiation safety testing and analysis of tristructural isotropic (TRISO) coated particle fuel.
- Graphite Qualification – Baseline characterization, irradiation, and post-irradiation testing of available nuclear-grade graphites are performed so that these materials can be used in modern GCRs.
- High Temperature Alloys – To realize the very high efficiency and to serve existing process heat industrial consumers, new alloys are being tested and added to the American Society of Mechanical Engineers (ASME) nuclear section of the Boiler and Pressure Vessel Code. Existing alloys in the code are limited in the long-term operating temperatures at which they can operate.
- Core Nuclear Design and Analysis Methods and Validation– Simulation methods and codes developed for the first generation HTGR program rely on assumptions and approximations that lead to significant uncertainty in analytical results. Newer methods and modern simulation capabilities will reduce these uncertainties and allow designers to boost performance and economics, while maintaining adequate safety margins.

All of these research activities generate tremendous amounts of data that must be qualified, analyzed, properly archived and made available to regulators, vendors, and other stakeholders. NDMAS was developed under the Next Generation Nuclear Plant (NGNP) program to facilitate this important task. NDMAS is an important enabling technology that is now being applied the ART-GCR program.

8.2 AGR Fuel Qualification Program

8.2.1 Overview

The Advanced Gas Reactor (AGR) program develops and tests TRISO-coated particle fuel that can be deployed in various high temperature reactor designs currently under development by commercial vendors. The program has performed three separate fuel irradiation tests to date, designed to assess fuel performance and investigate fission product transport behavior to support calculation of radionuclide source terms. The program is currently focused on PIE of the second and third irradiations (AGR-2 and AGR-3/4) and preparation for the final irradiation (AGR-5/6/7). This final irradiation represents final fuel qualification, using fuel fabricated entirely at the industrial vendor (BWXT).

8.2.2 Objectives

- To provide the data needed for a GCR vendor to license a design using TRISO fuel
- To establish a commercial vendor for this fuel.

8.2.3 Key Deliverables

- Completed delivery of AGR-5/6/7 Fuel Compacts to INL
- Fabricated, assembled, and loaded the AGR-5/6/7 Test Train for insertion into the Advanced Test reactor
- Completed AGR-5/6/7 Irradiation Test Plan
- Completed safety testing of two AGR-2 compacts (Oak Ridge National Lab)
- Completed and published the AGR-1 PIE Subsite on the NDMAS SharePoint Site.
- Completed 60% Design Review of the in-cell Air/Moisture Ingress Furnace.

8.2.4 Summary of Activities

8.2.4.1 Fuel fabrication and irradiation

A major deliverable for FY 2017 was the fabrication of the fuel compacts to be used in the AGR-5/6/7 fuel qualification irradiation. TRISO particles (Figure 37) and fuel compacts with two different packing fractions (25 and 40%) were fabricated at BWXT in the first half of the fiscal year. A portion of the fuel compacts, sufficient to populate the irradiation test train as well as provide adequate spares, were delivered to INL in March 2017. Characterization of the compacts was also completed at BWXT and the final fuel compact certification package was submitted to INL in August 2017. While the fuel product generally is of very high quality, quality control (QC) measurements indicated that a few of the specifications were not met. Spare compacts from the AGR-5/6/7 fabrication campaign were also sent to ORNL for confirmatory QC measurements. Technical evaluation of the fuel properties relative to the specifications and the objectives of the AGR-5/6/7 irradiation experiment indicate that there is still reasonably high confidence that the experiment objectives can be met with the existing fuel. Several lessons were learned as a result of the AGR-5/6/7 fuel fabrication effort, including: (a) interruptions in the fuel process development at the industrial vendor (partly a result of programmatic funding constraints) and loss of "operational rhythm" can have detrimental effects on fuel quality; and (b) additional work is needed to convert the current TRISO fuel fabrication process into a production scale process and realize a "steady state" fuel product quality (work that was eliminated from the AGR program with the termination of the NGNP program).

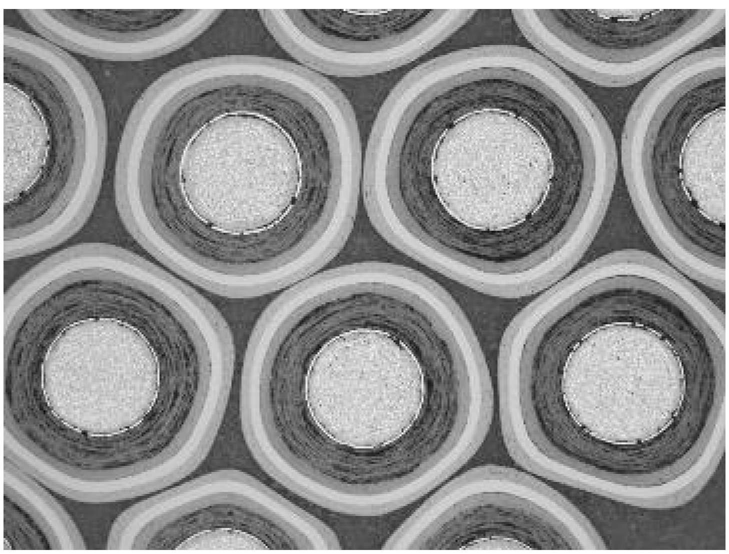


Figure 37. Cross-section micrographs of AGR-5/6/7 coated particles.

Fabrication of the AGR-5/6/7 irradiation test train was completed, and the experiment was declared ready to insert into the ATR reactor. This included final fabrication of all test train components, dimensional measurements of selected graphite parts (to support post-irradiation measurements to be performed at a later date), selection and loading of the fuel compacts into each capsule, and welding each capsule together into the final test train. The test train contains a total of 194 fuel compacts (compared to 72 and 66 in the AGR-1 and AGR-2 irradiations, respectively) in 5 different capsules (Figure 38). Four of the capsules constitute the AGR-5/6 portion of the irradiation, with irradiation conditions tailored to approximately represent the distribution of temperature, burnup, and fast neutron fluence that would be experienced by fuel in a high temperature reactor core during normal operation. The center capsule (Capsule 3) constitutes the AGR-7 portion of the irradiation. AGR-7 is a fuel performance margin test, designed to irradiate the fuel at temperatures far in excess of those expected in the reactor during normal operation, with the ultimate of objective of exploring upper thresholds for satisfactory fuel performance and the causes of fuel failure. The fuel in this capsule will be irradiated to burnup as high as 18% FIMA and time-averaged peak temperatures of approximately 1500°C. The test train is now awaiting insertion into ATR to commence irradiation in Cycle 162B. In addition, the formal Test Plan for the AGR-5/6/7 irradiation was prepared (PLN-5245).

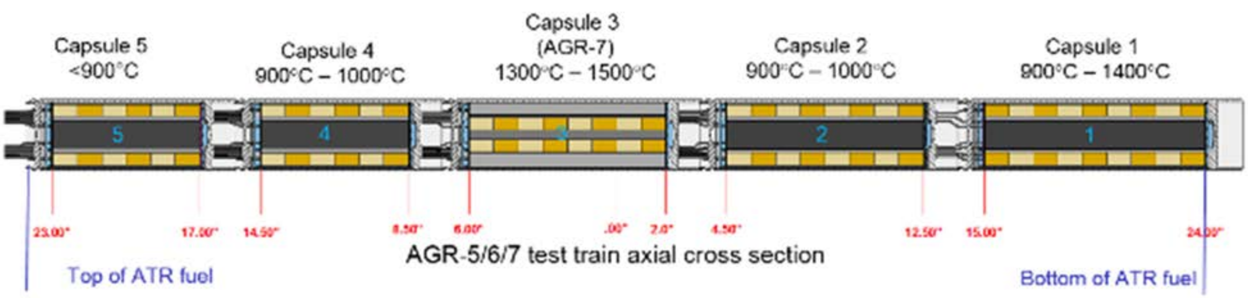


Figure 38. Axial cross section of the five capsules of the AGR-5/6/7 irradiation test train.

8.2.4.2 Post-irradiation examination

The AGR program is currently supporting PIE on two separate irradiation experiments: AGR-2 and AGR-3/4. While the majority of the initial PIE on the irradiation capsules for AGR-2 has been completed at INL, destructive examination and safety testing of fuel compacts is ongoing at ORNL. AGR-3/4 PIE is being performed at INL.

Three high-temperature safety tests were completed on AGR-2 compacts at ORNL. These tests are performed to assess fission product retention and coating behavior at high temperatures (1600 to 1800°C) similar to or beyond those expected during a depressurized loss of coolant accident in an HTGR. The results of the safety test generally continue to demonstrate the excellent performance of uranium carbide/uranium oxide (UCO) TRISO fuel (including relatively low release of fission products and low incidence of coating failure) and the comparatively higher release of cesium from uranium oxide (UO_2) TRISO fuels, which is indicative of a limited number of particles experiencing SiC layer failure.

In addition to the AGR-2 safety tests and compact destructive exams discussed above, ORNL has also set up a separate furnace system for long-duration (thousands of hours) testing on loose, irradiated AGR-2 particles. The Furnace for Irradiated TRISO-particle Testing (FITT) is located in a glovebox and will be used to heat loose particles at modest temperatures (1000-1600°C) in order to assess silver retention. Particles will be gamma counted before and after thermal exposure in the furnace in order to determine total silver release. This capability augments the AGR program’s two furnace systems at INL and ORNL for high-temperature testing of whole fuel compacts.

PIE conducted on fuel months or years after completion of the irradiation neglects measurement of key short-lived fission products that important for reactor safety considerations. In particular, iodine-131 is predicted to be one of the dominant sources of off-site dose during reactor accidents. Measurement of I-131 release during PIE requires re-irradiation of the fuel immediately prior to performing safety tests. At INL, the capability to re-irradiate individual particles in the neutron radiography (NRAD) reactor in the Hot Fuel Examination Facility (HFEF) has been developed. In FY 2017, two tests were performed which involved re-irradiation of several AGR-2 particles and subsequent transfer of the particles to the Fuel Accident Conditions Simulator (FACS) furnace in the HFEF main cell for heating in helium to measure fission product release, including I-131. Additional tests of this type are planned, as well as developing the ability to insert whole AGR fuel compacts into the NRAD core for re-irradiation.

Advanced microscopy methods developed during AGR-1 are being deployed to better understand the links between microstructure of the fuel kernel and TRISO layers and fission product mobility. Precession electron diffraction (PED) has been employed on AGR-2 fuel to characterize SiC layer grain boundary characteristics that may tend to promote or restrict fission product (e.g., Ag or Pd) movement. Transmission electron microscopy (TEM) and selected area diffraction (SAD) have been used in new studies of irradiation-induced defects in the SiC layer of a pair of TRISO particles. As the layer most important for retaining metallic fission products, the thermodynamic phases of the SiC layer has been studied in both AGR-1 and AGR-2 particles. Palladium has been implicated in corrosion of SiC layers. SAD analyses have shown that alpha-phase precipitates may form in the bulk beta-phase under irradiation, and data have been gathered, which suggest alpha phase may promote intra-granular transport and precipitation of palladium silicides.

The highly complex PIE of the AGR-3/4 experiment has continued this fiscal year at INL. One of the key AGR-3/4 activities is determining the distribution of fission products in the matrix and graphite rings that surrounded the fuel compacts. During the irradiation, fission products from the fuel compacts migrated through the rings, and analysis of this migration behavior is a primary objective of the experiment. Gamma scanning of the matrix and graphite rings from the AGR-3/4 capsules using the Precision Gamma Scanner (PGS) in HFEF was completed. A system for taking physical samples from the cylindrical matrix and graphite rings was developed and deployed in the HFEF main hot cell. The system uses an end mill and a vacuum to take very thin (< 1mm thickness) circumferential specimens from a rotating ring, in order to measure the radial profile of fission products in the ring. A total of eight rings were sampled (inner and outer rings from four AGR-3/4 capsules), and the specimens were packaged and shipped off-site for analysis of fission product inventory. Figure 39 shows a ring mounted in the sampling apparatus in the HFEF hot cell (left) and the cesium-134/137 concentration profile measured within the ring (right).

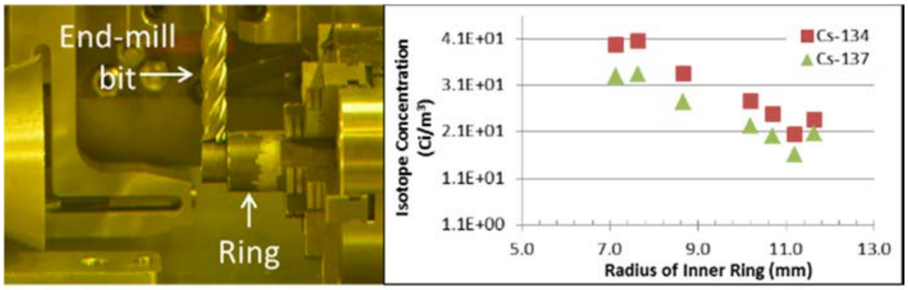


Figure 39. AGR-3/4 Capsule 7 inner ring mounted for sampling (left) and measured cesium concentration profile from inside of ring to outside of ring (right).

Figure 40.A shows a compact mounted to the deconsolidation rod prior to the start of deconsolidation, and Figure 40.B – Figure 40.D shows the reduction in compact diameter after the first, second, and third periods of deconsolidation, respectively. This approach is necessitated by the presence of the designed-to-fail particles at the center of the AGR-3/4 compacts. A total of three AGR-3/4 compacts were successfully subjected to the radial deconsolidation procedure, as documented in the report "Radial Deconsolidation and Leach-Burn-Leach of AGR-3/4 Compacts 3-3, 12-1, and 12-3" (INL/EXT-17-43182).

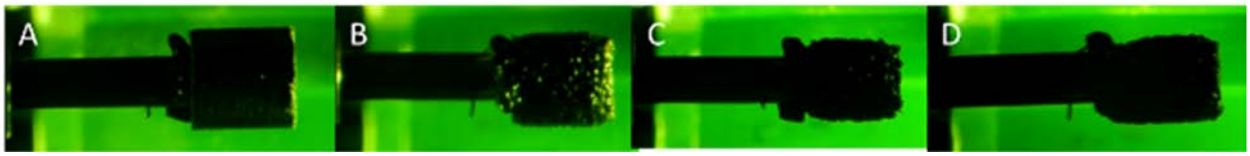


Figure 40. (A) AGR-3/4 Compact 3-3 mounted in apparatus prior to start of radial deconsolidation. (B-D) sequence of material removal from compact.

A key data need for the AGR program continues to be assessment of fuel performance at high temperatures under oxidizing conditions, representative of accidents that involve air or moisture entering the core. To meet this need, INL is developing a furnace system for heating irradiated fuel and graphite specimens in atmospheres containing O_2 and H_2O, while measuring release of fission products as a function of time. The system will be installed in the Fuel Conditioning Facility (FCF) air cell. Work has included specification of required testing conditions, assessment of facility suitability, and design of the furnace and supporting subsystems. Conceptual and 60% design reviews were completed this year. In addition, a benchtop system has been established in order to perform testing of various concepts and subsystems.

8.2.5 International Collaborations

The AGR program currently supports two separate collaborative activities as part of the GIF Very High Temperature Reactor (VHTR) Fuel and Fuel Cycle (FFC) Project Management Board (PMB). The first of these collaborations is a round robin test campaign designed to assess the ability of various laboratories to perform leach-burn-leach (LBL) analysis on TRISO fuel. The participants are China, Korea, and the US. The US took on the majority of the preparatory work for this activity, with ORNL preparing specimens for each country to analyze. This involved creating TRISO particles with two types of coating defects intentionally generated on each particle, and then characterizing each particle to confirm that the generated defect meet requirements. The two types of defects are designed to be identified during pre-burn leaching and post-burn leaching of the particles during LBL analysis. Particles were then mixed with a well-characterized coal standard powder containing metallic impurities that will also be analyzed in the LBL process. ORNL completed fabrication and characterization of the defect TRISO particles this year and sent the specimens to Korea (issues with documentation have delayed shipment of specimens to China), and also performed all of the necessary LBL work on the US specimens as part of the round robin experiment. It is expected that the remainder of the work on this activity will be completed next year, with results documented in a dedicated report.

The second collaborative activity as part of GIF is a TRISO fuel accident test modeling benchmark involving Japan, Korea, and the US. Each participant performed computational modeling of fission product release from TRISO fuel specimens that were recently heated in helium to temperatures of 1600-1800°C in the US and the EU. All participants have completed the computations, and comparison of all predictions with empirical data were discussed at the annual PMB meeting in June 2017. A final report is currently being prepared to present and discuss all of the predictions.

8.3 Graphite Development and Qualification Program

8.3.1 Overview

The Graphite Development and Qualification Program seeks to characterize the behavior of graphitic materials so that they can be qualified for use in high temperature gas-cooled reactors. Through measurement, analysis, and material science studies, graphite material behavior is determined for both irradiated and unirradiated nuclear core component applications.

8.3.2 Objectives

Baseline Characterization of Commercially Available Graphite Grades

Baseline (virgin) material properties for a number of commercially available graphite grades are being determined to provide unirradiated material response for nuclear graphite core components and to provide a baseline for material property changes due to irradiation.

Irradiation Testing and PIE

Irradiation material properties are measured and analyzed after irradiation of graphite to levels comparable to 15-20 years of irradiation lifetime in a nuclear reactor. Of specific importance is the irradiated creep response of the various graphite grades to temperature and dose, since the amount of creep controls the safe lifetime limits of nuclear graphite components. Finally, extensive analysis, fundamental material science studies, and degradation behavior is required to predict the specific and overall response of the graphite core components during normal and off-normal conditions in the nuclear reactor.

Understanding of Mechanisms and Development of Models

The measured irradiated and unirradiated material properties are analyzed and interpreted through an understanding of the basic material science studies in order to predict the graphite behavior for current and future grades of nuclear graphite.

8.3.3 Key Deliverables

- Baseline Property Testing
  - Completed AGC-5 specimen pre-IE unirradiated testing.
  - Completed baseline material property testing and characterization of all remaining partial graphite billets (roughly 5 remaining billets).
  - Irradiation Testing and PIE
  - Completed AGC-3 specimen PIE testing (irradiated at 800°C).
  - AGC-4 Test Series capsule (irradiated at 800°C) was reinserted into the ATR to complete its irradiation cycle. AGC-4 irradiation began in FY 2015 but was removed from ATR due to conflicts with an adjacent experiment.

  - Completed analysis for AGC-2 irradiation material property changes. This completes the irradiation, testing, and analysis of all 600°C irradiated graphite data.
  - Completed irradiation annealing studies on AGC specimens to determine the extended defect structure within irradiated graphite. This will assist in defining the extent of thermal property damage recovery for normal and accident conditions.
- Mechanisms and Models
  - Completed initial thermal creep studies to ascertain the underlying internal stresses within the as-machined and mechanically stressed graphite. This data is assisting in interpretation of irradiation creep analysis and the determination of microstructure versus irradiation damage effects on material property changes.
  - Completed chronic oxidation studies for fine grained graphite grade 2114 to assist in the development of an improved oxidation model.
  - Completed the first IAEA Technical Document (TecDoc) volume detailing known behavior of irradiation creep in graphite "Improving the understanding of Irradiation Creep Behavior in Nuclear Graphite: Part 1 Current models and mechanisms."
  - Completed an IAEA Technical Document discussing graphite oxidation principles in graphitic core components in high temperature reactor designs. Currently under review by IAEA for final distribution.

8.3.4 Summary of Activities

- AGC Material Property Testing: This activity covers the primary post-irradiation measurement and testing of the AGC specimens. Non-destructive thermal and physical tests that were performed on the AGC specimens before irradiation are repeated after irradiation to determine the effects of irradiation on the graphite grades. After the non-destructive tests are complete, the final part of material property testing activity is mechanically testing the irradiated specimens and directly compared to unirradiated material properties determined from as-received graphite billets in Baseline activity (see below). Some initial analysis of the PIE data from these initial measurements is necessary to determine data trends and potential outliers in the data.
- AGC Analysis: Once the data has been gathered from material property testing, the trends and results from the data must be interpreted. This analysis and interpretation of the data is required to understand the graphite behavior and predict eventual graphite component performance.
- Baseline Activities: The unirradiated, as-received virgin material properties for all graphite within the HTGR program are measured in order to provide a Baseline for all graphite material properties in the AGC program. While the AGC program measures the effects of irradiation on the material properties, the baseline program provides the statistically valid material property values that can be used to provide the "true" irradiated material property values. In other words, the changes to the materials properties will be applied to the Baseline values to provide statistically valid material properties to the graphite grades.
- Mechanisms Development Activities: In order to analyze and interpret the data from the AGC irradiation activities (as well as the trends seen in the Baseline activities), a strong understanding of the unirradiated and irradiated graphite response is required. This understanding is achieved through fundamental studies, sophisticated modeling, and extensive collaborators with graphite researchers throughout the world. General accomplishments within this area include continuing studies in irradiation damage mechanisms, studies on underlying mechanisms for irradiation creep,

investigation of microstructure evolution during irradiation and oxidation, determination of graphite oxidation principles, and investigation of the microstructure effects on fracture and strength.

- Analysis and Model Development: The development of material behavior models, based on the fundamental understanding of the material property changes, is necessary to predict the performance of the graphite components during service within a HTR. Irradiated and unirradiated graphite performance will be necessary to predict any issues that may occur during nuclear service. Accomplishments include irradiation creep model development, oxidation model development, creep mechanisms Technical Document development (IAEA TecDoc), biaxial fracture response, and licensing activities.

8.4 High Temperature Alloys

8.4.1 Overview

Alloy Testing and Qualification focuses on two specific areas in support of the near-to-mid-term deployment of high temperature gas-cooled reactors. The first addresses the cross-cutting issues of materials behavior during extended service beyond the scope of the current ASME Code and the high temperature design and materials issues identified by the NRC and its Advisory Committee on Reactor Safeguards (ACRS) in support of the licensing process. The second area corresponds to the incorporation of Alloy 617 into ASME Section III, Division 5.

8.4.2 Objectives

- Perform testing and analysis to address cross-cutting issues of materials behavior during extended service beyond the scope of the current ASME Code and the high temperature design and materials issues identified by the NRC and its Advisory Committee on Reactor Safeguards (ACRS) in support of the licensing process
- Incorporation of Alloy 617 into ASME Section III, Division 5 of the Boiler and Pressure Vessel Code

8.4.3 Key Deliverables

Updated the Draft ASME Boiler and Pressure Vessel Code Case for Use of Alloy 617 for Construction of Nuclear Components for Section III, Division 5, (INL/EXT-17-42999)

8.4.4 Summary of Activities

8.4.4.1 Short Term Notch Testing of Alloy 617 Base Metal

Experiments have been carried out on the behavior of creep specimens containing two notches with U-shaped profiles such that one notch ruptures and the other remains intact for metallographic analysis. Tests have been completed on specimens with a pair of either small or large radius notches over a range of temperatures from 750 to 900°C, where data from smooth bar specimens are available. No detrimental effects from the multi-axial stress states (imposed by the notches) have been observed. To the contrary, the small radius U-notch tests exhibit notch strengthening with a rupture life ~3 times longer than straight gauge specimen tests for similar conditions. The large radius U-notch tests have rupture lives of similar magnitude as the straight gauge tests at similar conditions. Metallographic examination using electron back-scatter diffraction indicates that in the small radius notched specimens the strain is localized very close to the root of the notch, while for the large radius notches the deformation is uniformly distributed across the gage section (similar to smooth bar test specimen).

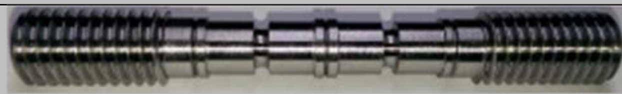
	V-notch
	U-notch, small radius
	U-notch, large radius

Figure 41. Notch specimen geometries employed in the base metal test program

8.4.4.2 Integrated Elastic Perfectly-Plastic and Simplified Model Test Methodology for Creep-Fatigue Damage Evaluation

The goal of the proposed integrated EPP and simplified model test (SMT) methodology is to incorporate an SMT data based approach for creep-fatigue damage evaluation into the EPP methodology. This avoids the separate evaluation of creep and fatigue damage and eliminates the requirement for stress classification in current methods; thus, greatly simplifying evaluation of elevated temperature cyclic service. The purpose of this methodology is to minimize over-conservatism, while properly accounting for localized defects and stress risers. Computational analyses and experimental studies have been conducted to support the development of this integrated methodology. Results from the verification and validation simulations of two-bar ratcheting experiments and scaled nozzle-to-sphere tests, conducted previously at ORNL, have provided added confidence that the EPP methodology can be successfully used as a screening tool to evaluate designs for compliance against the ASME Section III.

	
Pressurized SMT specimen before testing	Failed pressurized SMT specimen

Figure 42. SMT specimens before and after creep-fatigue.

8.4.4.3 Alloy 617 Code Qualification

The Code Case qualifying Alloy 617 for nuclear construction up to 425°C has been fully approved. Balloting of the Code Case for qualifying the material for use up to 950°C and 100,000 hours is proceeding according to a plan that was developed by the Alloy 617 Code Qualification Task Group. Since a new material has not been approved for nuclear construction in many years, the technical committees are not generally familiar with the detailed analysis and background required to develop the sections of the Code Case for elevated temperature construction. As a result, the preferred path forward is to ballot individual topical sections of the Code Case with the appropriate technical groups that have specific expertise in that area, while providing the remainder of the Code Case for review and comment. Sections dealing with time-dependent allowable stresses, physical properties, fatigue and creep-fatigue, and calculation of strain limits have all passed at the Working Group level and are now being balloted at the Subgroup or Committee level.

8.5 Design Methods and Validation

8.5.1 Overview

Simulation methods used for designing and licensing gas-cooled reactor cores (neutronics and thermal-fluidics) are developed and experimentally validated. Data from the experimental validation effort will be used by vendors and the NRC.

8.5.2 Objectives

- Conduct integral and separate effects tests under prototypical or scaled conditions to provide NQA-1 qualified data for the validation of computer simulations.
- Develop computational methods for the neutronic and thermal fluidic analysis of pebble bed and prismatic high temperature GCRs.

8.5.3 Key Deliverables

- Formal completion of the Natural convection Shutdown heat removal Test Facility (NSTF) air-based Reactor Cavity Cooling System testing campaign
- Documentation of the top-level program objectives, testing requirements, and unique considerations for the NSTF water-based Reactor Cavity Cooling System test assembly
- Completion of an NQA-1 audit of the High Temperature Test Facility (HTTF) quality assurance program.
- Completion of an IAEA report comparing results obtained for Phase I of the prismatic reactor design cases, based on the 350 MW General Atomics Modular High Temperature Gas-Cooled Reactor.
- Completion of a simulation of the first Organization for Economic Cooperation and Development – Nuclear Energy Agency (OECD-NEA) High Temperature Reactor Loss of Forced Cooling Experiment conducted in Japan in 2010.

8.5.4 Summary of Activities

8.5.4.1 Experimental Validation

Two large integral facilities have been constructed to re-create accident conditions and to characterize passive heat removal systems in modular HTGRs. The NSTF program at Argonne was initiated to perform large scale experimental testing and generate validation data for passive decay heat removal concepts. The HTTF was built at Oregon State University to simulate the conditions in a HTGR during severe loss of forced cooling accidents.

NSTF

In early FY 2017, after 33-months of active air-based testing and completion of stated air-based program objectives, the project saw a formal conclusion the air-based testing campaign. The phase was concluded with an archival-style disassembly and long-term storage of related components. Electronic copies of testing records were mirrored across two hard drives, disconnected from service, and placed into locked dry-storage in separate building at the Argonne site. Additionally, a formal internal audit was performed and reaffirmed the programs compliance to NQA-1. In total, 27 tests were conducted, 16 of which were accepted, and included studies of multiple baseline repeats, prototypic accident scenario, blocked risers channels, power variations, azimuthal and cosine skews, chimney roles, and meteorological variations.

By mid-year FY 2017, the program began conversion of the test facility to water-based cooling. Design of the test facility was approved in previously hosted meetings attended by key players including DOE, NRC, industry, and other national laboratories. The design has prioritized safe and reliable operation; including addressing all applicable ASME codes for pressure components. Additionally, the design retains all aspects common to a fundamental boiling water thermosiphon and thus is well-poised to provide necessary experimental data to advance basic understanding of natural circulation phenomena and contribute to computer code validation.

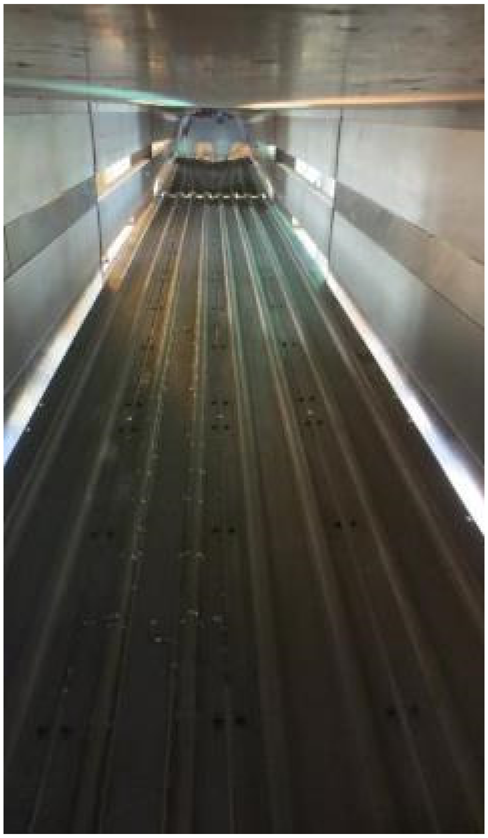


Figure 43. Installed 8-riser cooling panel test section.

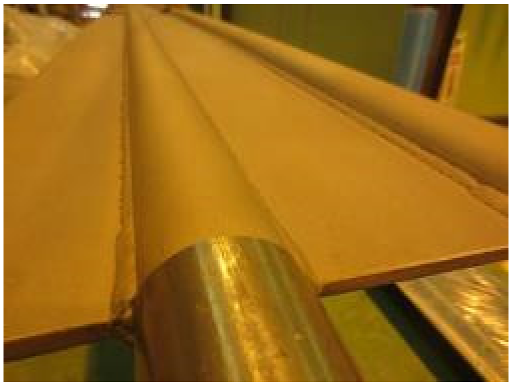


Figure 44. Detail of weld and bead blasting on panel.

Figure 45. 1,200-gallon storage tank.

The accomplishments also included development of the top-level program objectives, testing requirements, and unique considerations for the boiling water test assembly. A discussion of the proposed 6-year testing program is given, which outlines the specific strategy and testing plan for facility operations. The deliverable report also includes a detailed description of facility design, including as-built dimensions and specifications of the various mechanical and liquid systems, design choices for the test section, water storage tank, network piping, and instrumentation.

HTTF

An NQA-1 audit of the HTTF quality assurance program was performed. The quality assurance documents were revised to address the two findings and six recommendations from the audit.

Repairs to the HTTF were completed, shakedown and system characterization tests were run, and matrix testing began; four matrix tests were performed. All of the graphite heater banks failed again, as the contact area between adjacent rodlets was reduced when thermal expansion shifted the stack of rodlets from a perfectly vertical orientations, resulting in electrical arcing. The arcing led to some very high local temperatures, damaging both the rodlets and the core ceramic blocks. New rodlets were redesigned, as shown in Figure 46, and ten were ordered for testing.

Figure 46. Graphite heater rodlets redesigned to maintain uniform electrical continuity under thermal expansion of HTTR core blocks.

The RELAP5-3D code was assessed using data from the first HTTF matrix test, PG-22, which was a gas exchange test between the primary coolant system and the reactor cavity simulation tank.

Core Simulation

The OECD modular HTGR benchmark is being used at INL to drive the development of full-core and transient models and for code-to-code verification against other methods. The IAEA CRP on HTGR Uncertainty Analysis in Modeling (UAM) is likewise being developed to compare the propagation of uncertainties in lattice and full-core modeling.

The OECD/NEA benchmark is a multiyear project (2013-2018) that has already yielded a set of reference steady-state and lattice problems that can be used by DOE, NRC, and vendors to assess their codes. This benchmark activity is now being completed with the comparison of the data submitted for Phases I and III. An example of the good agreement achieved for the eigenvalues of the stand-alone neutronics exercise 1 is shown in Figure 47 for the various transport and diffusion codes.

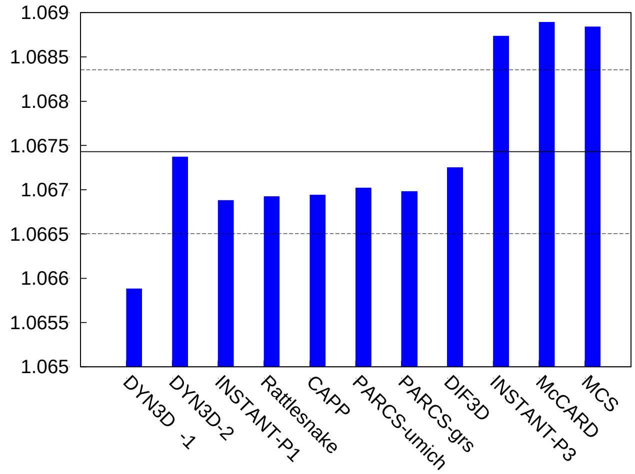


Figure 47. Comparison of core eigenvalues generated for the MHTGR-350 core for Phase I (neutronics) of the OECD/NEA benchmark project.

The IAEA launched the CRP on the HTGR UAM in 2013 to study uncertainty propagation in the HTGR analysis chain. Two benchmark problems are defined, with the prismatic design represented by the General Atomics (GA) MHTGR-350 and a 250 MW modular pebble bed design similar to the HTR-PM (INET, China). INL is leading the prismatic reactor problem specification of Phases I-III.

A detailed report was completed at the end of August on the comparison results obtained for Phase I of the prismatic reactor design cases, based on the 350 MW General Atomics Modular High Temperature Gas Cooled Reactor. Phase I includes the data sets received for the neutronics cell (Exercise I 1), lattice (Exercise I 2a, b), and supercell (Exercise I 2c) exercises; the steady-state (Exercise I 3) and transient (Exercise I 4) thermal fluids exercises; and the experimental validation case based on the Very High Temperature Test Reactor Critical (VHTRC) facility. An earlier report released as part of this activity detailed the specifications for the Ex. II-1 depletion cases and included the preliminary INL results obtained with the SCALE/SAMPLER code.

Both these deliverables were submitted to the DOE and IAEA for review and approval. An example of the typical data produced by this benchmark is shown in Figure 48. The 1σ standard deviation of the eigenvalues determined by GRS/INL and KAERI using the indicated covariance data libraries are compared here for all the exercises of Phase I and vary between 0.45% - 0.8%. These findings can typically be used by industry to determine the margins available in their HTGR designs, and to assist with the NRC licensing requirement of performing best estimate plus uncertainties analysis.

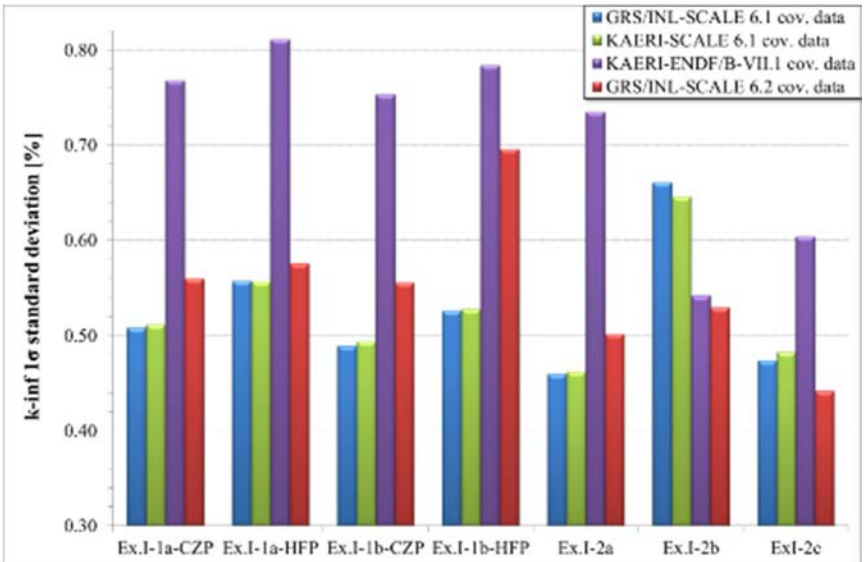


Figure 48. 1σ standard deviation of the eigenvalues for Phase I Exercises 1 and 2. The GRS/INL data used either SCALE 6.1 or SCALE 6.2 covariance data with the SCALE 6.2 SAMPLER/NEWT sequence, while KAERI used either SCALE 6.1 or pure ENDF/B-VII.1 covariance data.

International Collaborations

A collaborative effort between JAEA and INL, as part of the Civil Nuclear Energy Working Group (CNWG) - is underway to simulate the High Temperature engineering Test Reactor (HTTR) loss of forced cooling (LOFC) transient performed in December 2010. Additionally, the simulation approach for the modeling of an HTTR power transient performed as part of a tritium experiment was also discussed. The INL-coupled code Parallel and Highly Innovative Simulation for INL Code System (PHISICS)/Reactor Excursion and Leak Analysis Program (RELAP5-3D) was utilized to simulate the LOFC transient and a controlled power ramp used for the Tritium Code Validation run. JAEA developed the RELAP5-3D model with guidance from INL during the last quarter of 2015 and measured data from the HTTR LOFC was provided by JAEA for validation of the model results.

8.6 Data Management and Analysis

A critical requirement of the ART program is storage and accessibility of data in a robust database. This requirement conforms with DOE’s goal to increase access to the results of the research that it supports, in accordance with the principle that:

> “Sharing and preserving data are central to protecting the integrity of science by facilitating validation of results and to advancing science by broadening the value of research data to disciplines other than the originating one and to society at large.” (DOE Public Access Plan 2014)

The ART Gas-Cooled Reactor Campaign has established the NDMAS to address this requirement. NDMAS collects experimental data with chain-of-custody documentation, supports data qualification, stores data in a secure relational database, and provides data analysis and controlled data access. For user convenience, NDMAS creates a wide array of summary tables that combine related data sets and provides access to those data via graphs and files located on the NDMAS website. There, program members can review tabular and graphical data summaries, examine statistical analyses, download data for advanced analysis, or request specialized data queries or analysis.

The NDMAS team is housed in INL's Human Factors, Controls, and Statistics department. Statisticians on the NDMAS team are available to provide advanced statistical analysis of experimental data and related data sets. Temperature control during irradiation experiments, for example, routinely incorporates statistical quality control of thermocouple behavior, based on relationships between reactor configuration and thermal response at measurement locations. Uncertainty quantification critical to those thermal calculations, used for temperature control and post-irradiation analyses, was recently published in Nuclear Engineering and Design (Pham et al. 2017).

In FY 2017, NDMAS continued to expand to meet the needs of ART and related nuclear R&D programs, and is now actively supporting, or has agreed to support, the following additional programs and experiments:

- Tritium-Producing Burnable Absorber Rods Materials Irradiation Separate-Effects Test
- ATF Program
- High Performance Research Reactors
- Advanced Fuels Campaign - Fuel Systems Handbook Program (planned)
- Transient Reactor Test Facility Program (planned).

This continued expansion benefits all users by providing additional opportunities to include new features in the data management and delivery system. Within the ART program, NDMAS recently completed development of the necessary database tables, summary tables and webpages to incorporate data from the baseline graphite characterization program, AGR-1 PIE, and the AGR-5/6/7 experiment. The continued expansion of experimental data allows users to access characterization data measured during fabrication, irradiation monitoring, and in PIE, via SQL calls to the array of databases managed in the system.

9. ADVANCED REACTOR TECHNOLOGIES – MOLTEN SALT REACTORS

Lou Qualls, Oak Ridge National Laboratory, NTD

9.1 Mission

Advanced reactors face significant technical hurdles to commercialization due to unique, innovative features and lack of a licensing infrastructure for new technology options. The general mission of the Advanced Reactor Technologies Program is to identify and resolve the technical challenges to enable transition of advanced non-LWR reactor technologies and systems to support detailed design, regulatory review, and deployment.

Molten salt reactors are a new campaign within ART with potential for lower cost energy, flexible fuel loadings, and positive impact for advanced fuel cycles. For the commercial deployment of MSRs, two challenges are identified: 1) understanding of the chemistry related MSR operation and full salt life cycles, and 2) establishment of a licensing pathway for a unique, liquid-fueled reactor concept.

9.1.1 Objectives

- Clarify licensing criteria and science-based approach for demonstration of regulatory compliance. This work includes the validation of safety tools to assure margins, refinement of techniques to address key issues (e.g., mechanistic source term), and the qualification of MSR fuels.
- Collaborate with industry to identify and conduct essential research to reduce technical risk associated with MSR technologies.
- Develop and sustain the domestic infrastructure and knowledge base within national laboratories and universities to perform needed research. This includes the purposeful training of next generation engineers and scientists by engaging them in advanced reactor concept design and analysis, and NEUP awards that support MSR R&D.
- Engage with standards developing organizations to address gaps in codes and standards to support advanced reactor designs.
- Utilize international collaborations to leverage and expand R&D investments through targeted R&D agreements.

9.1.2 Challenges

- Investments are needed to establish the U.S. infrastructure (facilities and expertise) to support the testing of technology needed for MSR applications.
- Efforts are needed to preserve and manage at risk data, knowledge, and experience related to past U.S. DOE MSR design, operations, tests, and component technology.
- Existing MSR testing data needs to be captured into modern databases for archival, with details and pedigree appropriately described.
- MSR design and safety methods need to be defined, developed, and validated.
- Extension of ASME code qualification of materials to MSR operating regimes needs to be initiated.

9.1.3 Major R&D Activities

The current scope of the ART MSR R&D is structured into four main technical areas:

- Salt Chemistry
- Technology Development
- Methods, Modeling, and Validation
- Advanced Materials.

A separate Project Management work area leads and coordinates the R&D activities which include scientific research and development at the National Laboratories and Universities as well as collaborations with industry and international partners. Working directly with DOE-NE, the NTD prioritizes, coordinates, and oversees execution of the R&D tasks. The Advanced Materials Technology Area Lead (TAL) has a similar role for materials R&D within the ART Program, including materials specific to MSRs. Specific Project Management achievements in FY 2017 included engagement with the newly organized industry Molten Salt Reactor Technology Working Group, and the creation of a MSR Technology Roadmap intended to guide future research activities.

9.2 Technology Development and Demonstration

The MSR technology objectives are: evaluation and understanding of system integration and performance benefits of various innovations; preserving previous MSR R&D knowledge; researching new primary reactor component, sensor, and reliability monitoring technology options; creating and sustaining the R&D infrastructure (both personnel and hardware) needed for long-term research, development and demonstration of MSR components, instrumentation, and validation testing.

9.2.1 Forced-Flow Fluoride Salt Testing Loop

A small (<10 liter) forced flow loop for fluoride salts was preliminarily designed in FY 2017. The loop was originally planned for tritium behavior and mitigation studies. However, due to limited budgets, the loop has been repurposed for corrosion testing, scheduled to begin in FY 2019. Equipment and materials are on-site with the exception of the electric pump. A schematic of the loop is shown in Figure 49. The following activities were completed in FY 2017:

- The design of the piping system was completed.
- The control system was procured.
- A ventilated hood was procured for installation into the facility.
- A status report that included budget and schedule estimates for tritium testing was prepared.

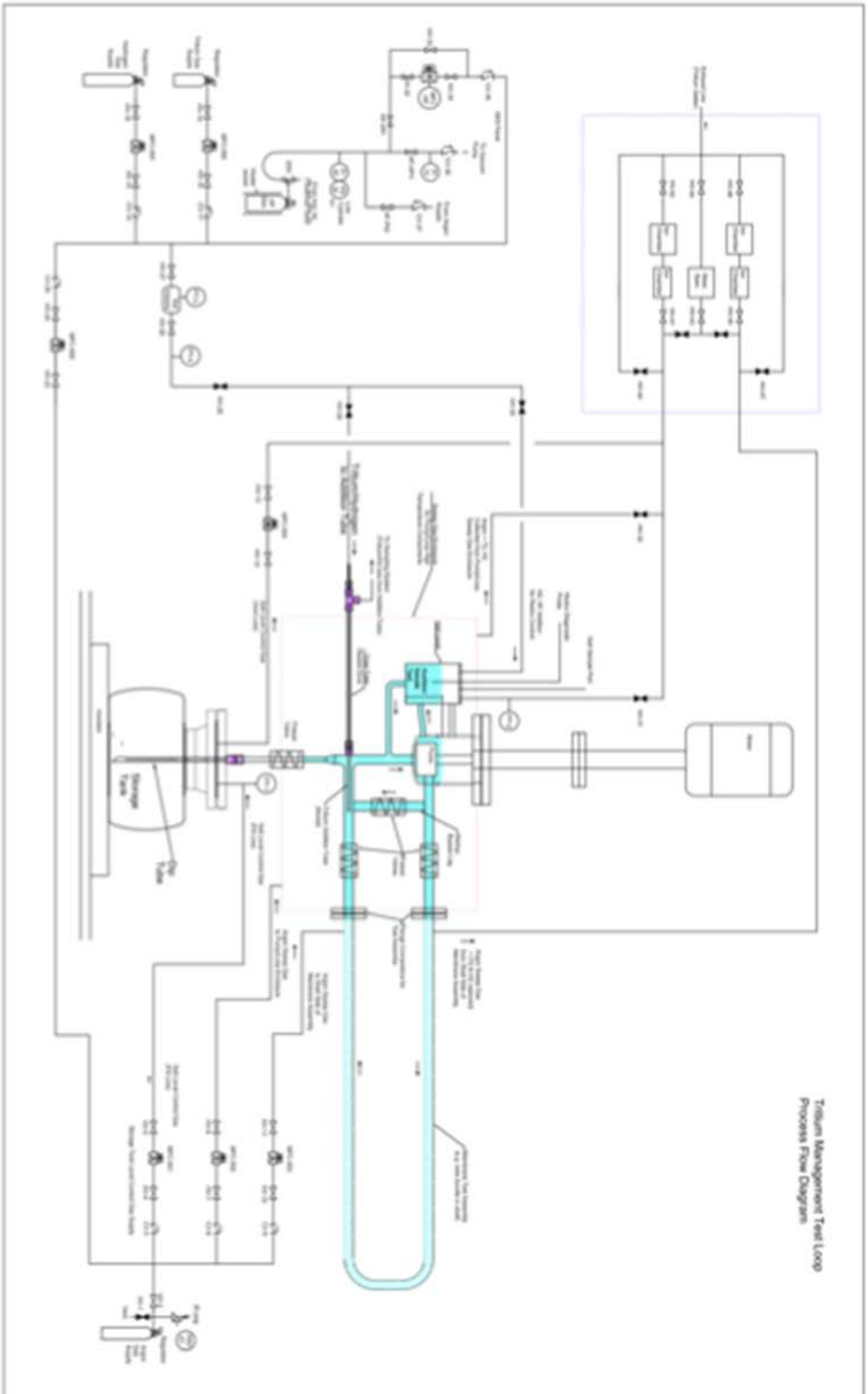


Figure 49. Schematic of forced-flow fluoride salt test loop.

9.2.2 High Temperature Fission Chamber

Currently there are limited traditional reactor control instrumentation that can operate at the higher temperatures needed for advanced reactors or in contact with new reactor coolant fluids. The High Temperature Fission Chamber is a project to modify a mature neutron detection technology to operate at temperatures representative of advanced reactors (700°C nominal operation and 800°C accident conditions).

Design of a system with higher temperature materials was completed and a prototype system was fabricated. A test facility was installed at the Ohio State University Research Reactor to house the fission chamber and maintain it at a target temperature of 700°C while within the flux field of the reactor.

In FY 2018, the instrument, shown in Figure 50, will be tested at the reactor within the dry furnace. After operation of the experiment, the design will be modified if needed, and the electronics package design will be industrialized for installation as an instrument for a test reactor installation. An effort to identify candidate advanced, high temperature reactors for testing of a future commercial prototype of the instrument will occur in FY 2018.

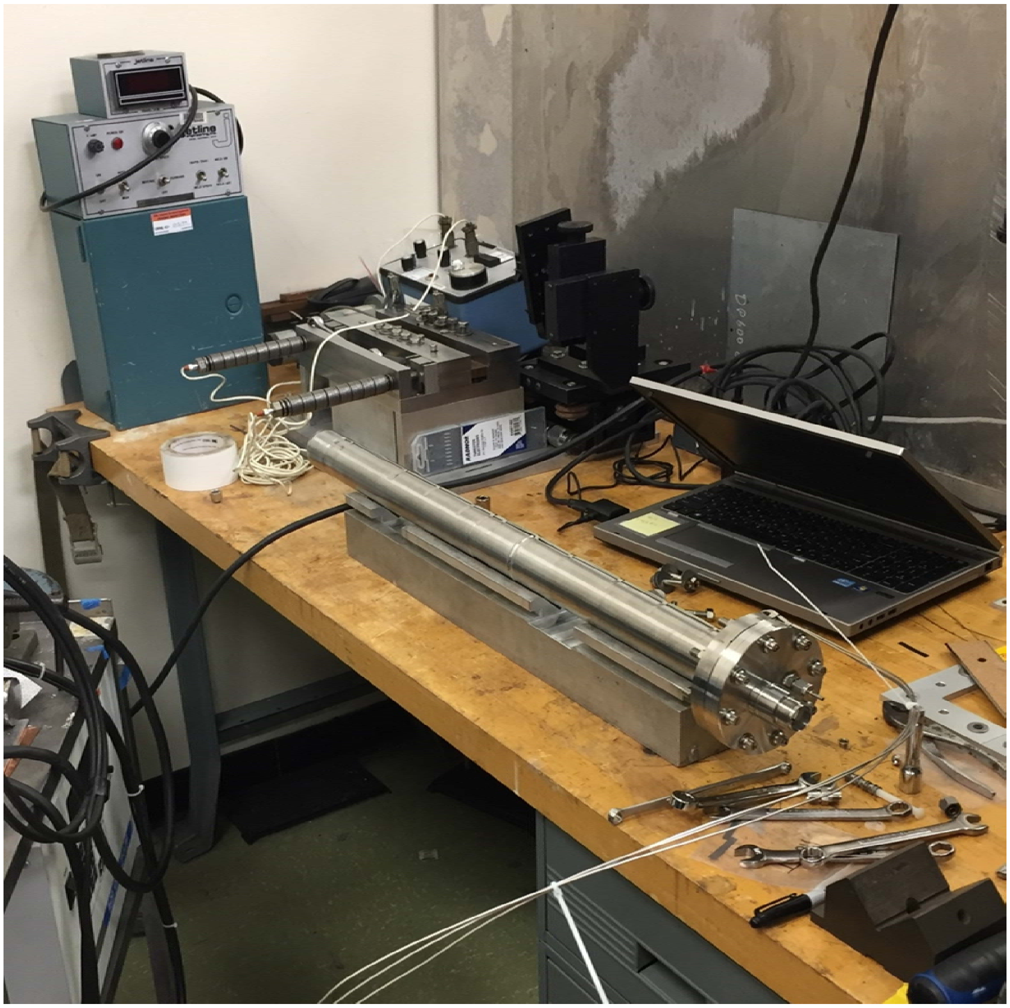


Figure 50. High temperature fission chamber and electronics during benchtop testing.

9.2.3 Fluoride Salt Medium Size Loop

An existing medium-sized (~75 liters) forced flow loop with FLiNaK salt, shown in Figure 51, is operational at ORNL. This loop is large enough to support thermal hydraulic and heat transfer data generation and instrument and small component testing are possible on the loop. This loop will be re-started in FY 2018 and preliminary testing will occur. The loop is currently configured to provide thermal hydraulic data for a pebble-bed configuration and initial operation is expected to produce some relevant data. Plans to use the loop for corrosion testing or instrument testing will occur within those areas of the project.

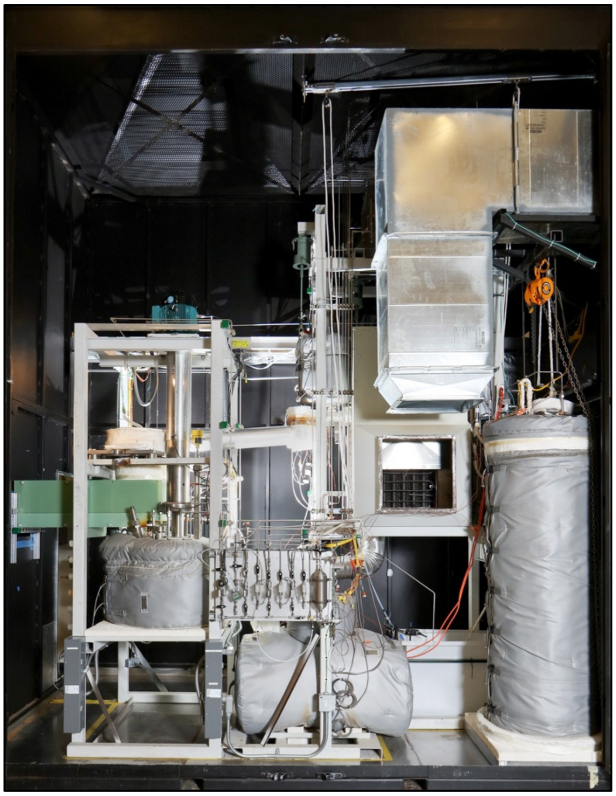


Figure 51. Thermal image of the medium-sized fluoride salt loop.

9.2.4 MSRE and MSBR Knowledge Preservation

Documentation from the Molten Salt Reactor Experiment exists and requires identification, digitization, and cataloguing. A limited effort has been initiated in this area. However, a significant amount of data remains to be examined for both the MSRE, which was built and operated, and the Molten Salt Breeder Reactor, which was developed as a concept but never built. Export control review and classification review of documents is required prior to release.

9.3 Methods, Modeling, and Validation

In the Methods area, the objectives are:

1. Addressing high priority modeling needs to improve existing design capabilities, including the integration of dynamic chemistry modeling with neutronic and thermal hydraulic modeling;
2. Conducting benchmark and analysis studies to verify and validate MSR design and safety analysis codes and simulation techniques to the degree possible with the limited data sets;
3. Addressing key MSR regulatory gaps.

9.4 Advanced Materials

The objective of the ART advanced materials R&D is to provide the technical basis needed to support the regulatory requirements for structural materials required for MSR that could be deployed in the near-to-mid-term. Activities include the qualification of structural materials for MSR construction, establishment and the extension of qualified lifetimes and usage temperatures of structural materials already approved within the ASME Code for construction of MSRs if applicable.

9.4.1 Alloy N Assessment Procurement and Testing

Alloy N is the structural alloy in the MSRE. Assessment of the cost and schedule to qualification for use in a “first” MSR is needed along with an assessment of the utility of the alloy to the industry if it were to be qualified. These assessments are to be compared to other options in order to prioritize allow development for MSRs. Other candidates include modified Alloy-N and structural materials with protective cladding that would contact the salt. Structural material is needed for MSR applications that include reactor vessel, core supports, primary and secondary piping, and s heat exchangers. Hence development of processing conditions and fabrication scale up for different product forms such as plates, pipes, bars, forgings and sheets, in addition to seamless tubing, are required.

9.4.2 Graphite

Graphite is used as a moderating material in thermal-spectrum fluoride-salt reactors. Existing materials programs within ART are examining graphite materials for use in high temperature gas reactors. This material research will be extended and adapted to MSRs to the degree possible.

9.4.3 Silicon Carbide

High temperature materials that are resistant to salt reactions are needed to hold tension in MSR cores. Silicon Carbide materials are potential in-core structural material candidates. Existing research is underway to evaluate the use of SiC materials in reactors and develop design rules for their use. Compatibility experiments of SiC materials with molten salts is required to determine their ultimate feasibility as in-core MSR structural materials.

9.5 International Collaborations

9.5.1 Coordinated Research Projects on Salt Properties

Salt properties for MSRs were extensively studied in the 1960’s and 1970’s, however, the data is incomplete and in some cases inconsistent. In particular, further development of best-estimate and high fidelity/physics based simulation codes require accurate, complete, consistent, precise and reliable property data sets. Natural circulation and source term modeling are particularly sensitive to thermodynamic properties and thus uncertainty quantification will be important to mature understanding of these phenomena. In a proposed FY 2018 INERI, comparison of property measurements from different organizations via different means would be performed to help establish the overall validity of the property measurements. The JRC has the most credible molten salt thermophysical property measurement laboratories and the U.S. would like to share information with them to the extent possible.

9.5.2 Generation-IV System Steering Committee and System Integration Projects

The SSC provides coordination of reactor R&D collaborations within Generation-IV. Member countries include China, EURATOM, France, Japan, Korea, Russia, and the U.S. The SSC also handles the interface with the GIF Policy and Expert Groups, GIF Task Forces, GIF Methodology Working Groups, and other SSCs. The System Integration and Assessment Project includes the same members and provides technical oversight of GIF technical collaborations.

The U.S. has been an observer at the GIF MSR provisional SSC meetings since the committee was formed. A U.S. delegate has been assigned to the activity for the past decade, but only became an official member once DOE signed the participation MOU in 2017.

The GIF MSR provisional system steering committee currently has two meetings planned in FY 2018. April 10–11 will be the steering committee meeting, which will be followed by an MSR technology workshop (not sponsored by GIF). In the fall, the U.S. will host the next provisional SSC committee meeting in association with ORNL’s annual MSR workshop (likely the first week of October).

10. ADVANCED REACTOR TECHNOLOGY/ENERGY CONVERSION

Gary E. Rochau, SNL, Technical Area Lead

10.1 Purpose

The EC Project provides solutions to convert the heat from an advanced reactor to useful products that support commercial application of the reactor designs.

10.2 Description

The EC Project in FY 2017 was focused on the application of heat from an SFR to a supercritical carbon dioxide (sCO_2) Brayton power cycle. Maximum turbine inlet temperature is 550°C and expected thermal efficiency is 45%. The current state of the sCO_2 technology is a laboratory scale demonstration of a sCO_2 Recompression Closed Brayton Cycle (RCBC) at 250 kWe. To achieve commercial interest, R&D is necessary to reduce the technological and economic risks for: materials that can handle the temperature and pressures without serious corrosion from the sCO_2; component technology that demonstrates scalability to commercial levels; system level testing to demonstrate performance and develop operating procedures; TRL risk management, systems engineering and economic models guided by a technology roadmap to support early engagement with industry to leverage knowledge and transfer technology.

10.3 Strategy

Perform basic and applied R&D to develop the sCO_2 RCBC to support the energy conversion of the SFR by 2030 at commercial off-the-shelf scale. To that end, the EC projects are:

- Leading energy conversion collaboration efforts between the offices of NE, EE, and FE; supporting concurrent engineering and integrated project management.
- Developing and testing materials suitable to supporting the sCO_2 RCBC for the SFR for the lifetime of the plant.
- Developing heat exchangers for the SFR coupling the sodium to the sCO_2 and understanding the consequences of sodium and sCO_2 interactions.
- Developing the sCO_2 components for the RCBC demonstration: including turbines, compressors, recuperators, seals, bearings, control valves, control systems, high-speed electrical generators, and waterless heat rejection.
- Testing sCO_2 components within a complete system to investigate performance trades, assess effects of design features such as fluid additives, and develop operating procedures.
- Developing and maintaining technology roadmaps, system engineering and economic models, science based steady state and dynamic models for specifying requirements and developing operating procedures, tracking results, and planning futures.

10.4 Goal Setting

During FY 2014, SNL set a goal to help establish a technology roadmap for the commercialization of the sCO_2 power cycle:

"By the end of FY 2019, Sandia National Laboratories shall develop, with industry, a fully operational 550°C 10 MWe R&D demonstration sCO_2 Brayton Power Conversion System that will allow the systematic identification and retirement of technical risks and testing of components for the commercial application of this technology."

This goal established a focus for the activities of the ART/EC Project and was implemented into the "sCO_2 Brayton Cycle: Roadmap to Product Commercialization" in FY 2016. The roadmap is a key deliverable of the management and integration work package, and is a living document that captures innovations and constantly evaluates technological readiness levels. The "systems engineering model for ART Energy Conversion" was developed to incorporate project management planning, system engineering modeling, and technology readiness assessment in a concurrent approach to the commercialization goal.

The Roadmap helped define the work packages of the ART/EC project, funded by a cross-cutting initiative Supercritical Transformational Electrical Production (STEP). The STEP program defined a 5-year plan for a 10 MWe pilot facility constructed by industry to fulfill the Goal. The facilities of the ART/EC program (at SNL and ANL) were transformed into component development platforms to reduce the technical risks of the pilot facility.

The FY 2017 focus of the EC project continued with the SFR application and the following activities:

- Technology Roadmap/project management plan/system engineering model completed
  - Systematic risk identification and retirement from components to system configuration.
- Commercialization of the sCO_2 system components to a higher TRL level to support sCO_2 system commercialization by 2030
- Operation RCBC at a turbine inlet of 550°C
  - Brayton Development platform working with industry to achieve high TRL components for system integration.
- Development of intermediate sodium to CO_2 heat exchanger (primary heat exchanger)
  - Sodium drain, fill, plug in printed circuit heat exchangers (PCHE)
  - Sodium CO_2 interactions.

These activities attempted to address and advance the TRL of components of the RCBC to 550°C and begin to move to 750°C turbine inlet temperature by:

- Engaging in federal business opportunities to Cooperative Research and Development Agreement processes, yielding laboratory/industry collaborations with patents, copyrights, and national awards
- Sandia procuring the world's first 1 MW turbo compressor at 750°C though design/build process

- Standing up eight additional test configurations in addition to the RCBC development platform to address:
  - Heat exchanger (SEARCH© design tool), optimized PCHE design
  - Heat exchanger test rig
  - Particle imaging velocimetry, flow distribution measurement
  - sCO_2 seals test rig
  - sCO_2 bearings
  - Turbocompressor test rig
  - Dry heat rejection (tall loop), waterless power production
  - Parallel compression, combining compressor fluid output under different conditions.

Another key deliverable in FY 2017 supported the "Industry Days" workshop at ANL. Workshop documentation identifies:

- There is a range of opportunity for collaboration in component development across a range of applications.
- The biggest challenge for technology development is in economics. Top research priorities were identified.
- The primary challenges towards commercialization are:
  - Proving reliability of the system.
  - Meeting standards of utility.
  - Understanding all the risks associated with writing a contract between vendor and utility.
- Industry interest is tangible, but engagement needs to be supported both financially and operationally. Some areas to consider include:
  - Easing collaborations between industry and the national laboratories,
  - Facilitating information sharing that protects commercial IP,
  - Supporting technology development economically and via policies and programs

The top R&D needs were identified and prioritized as shown in Figure 52. Figures 53, 54, and 55 summarize the technical findings in FY 2017 for the components of the RCBC.

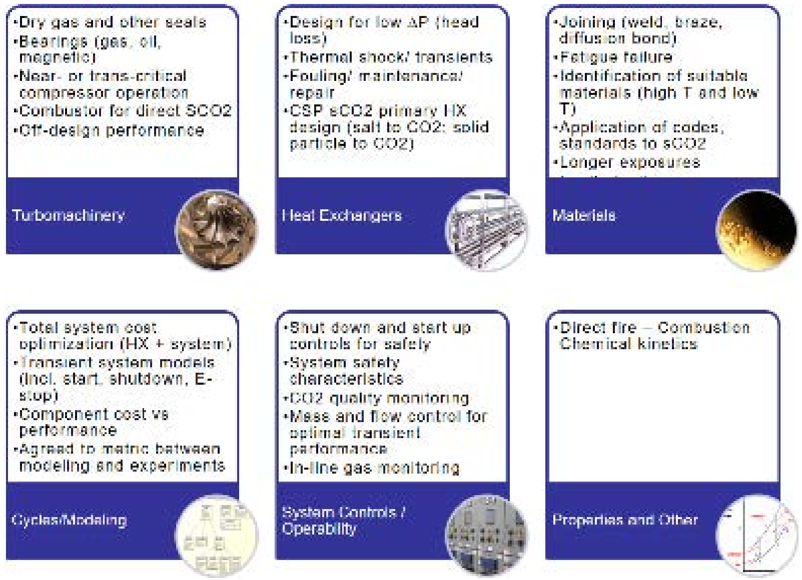


Figure 52. Industry Days top R&D needs.

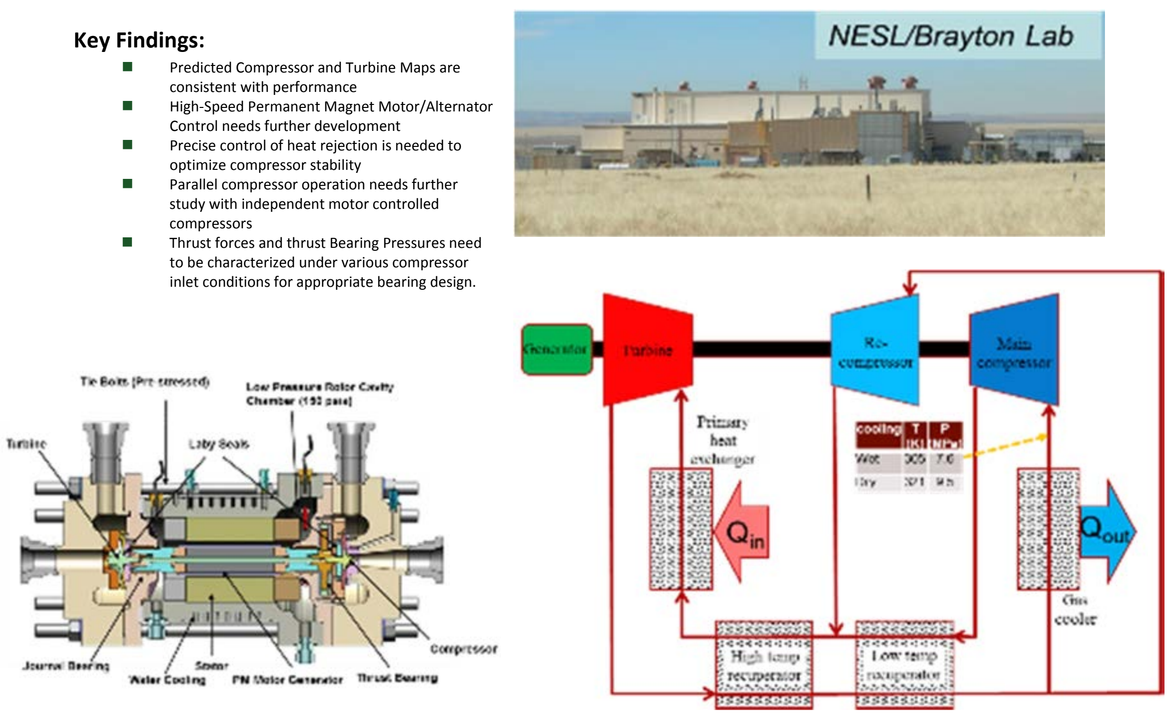


Figure 53. Key findings of R&D on RCBC testing.

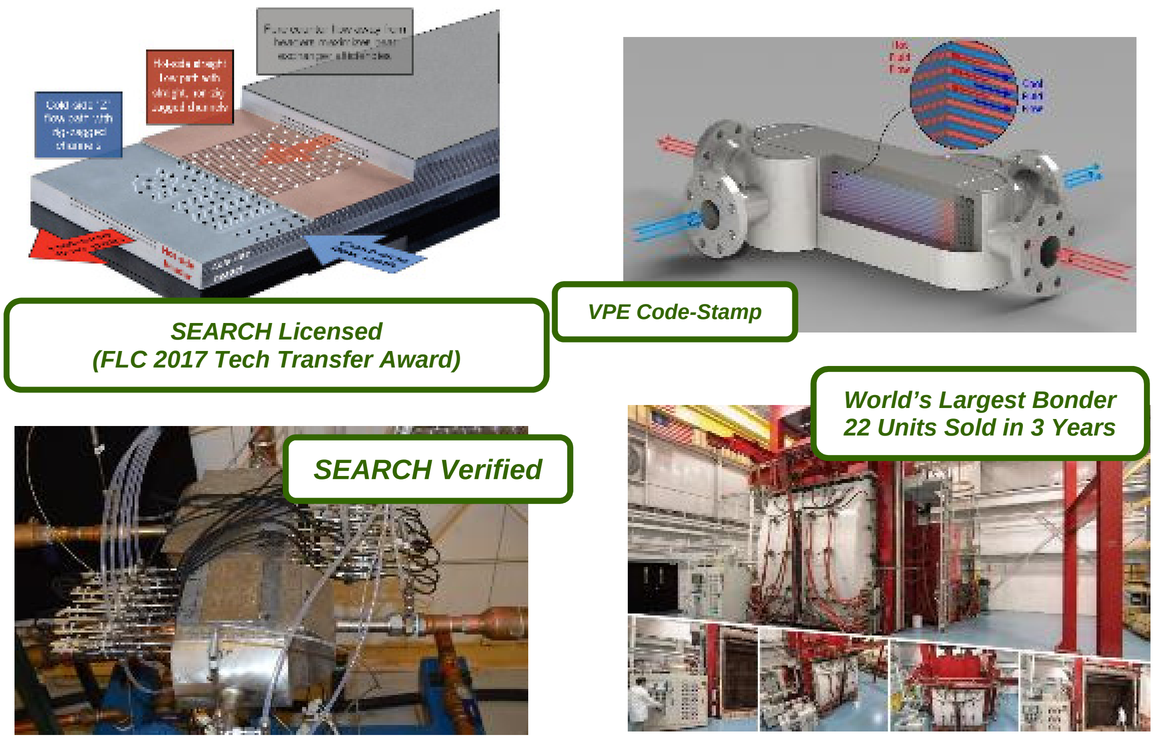


Figure 54. Key results of PCHE R&D.

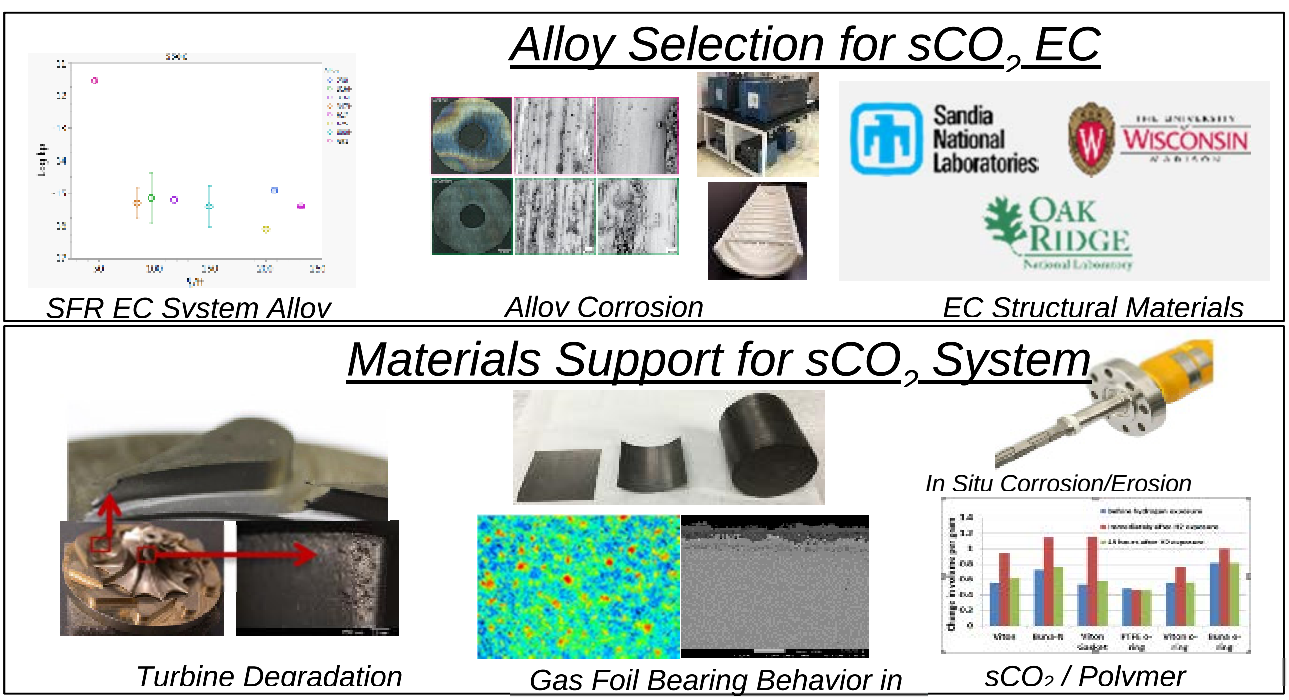


Figure 55. Key results of sCO_2 materials R&D.

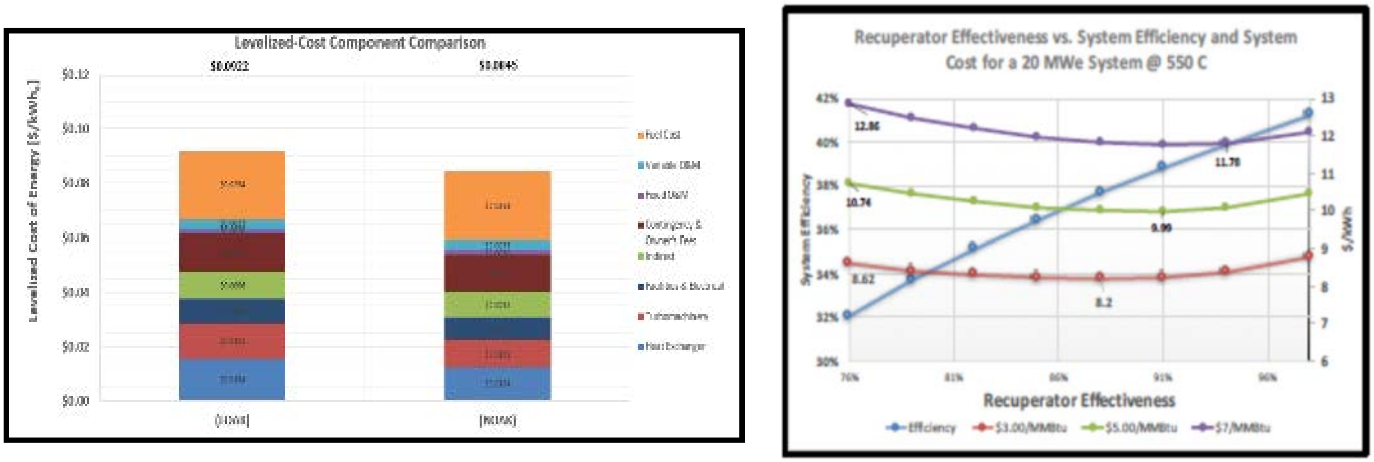


Figure 56. Results of Brayton LCOE economics model.

Figure 56 demonstrates the results of the Brayton LCOE economics model. FY 2017 work is focused on the integration of an existing RCBC Evaluation and Trade Studies (RETS) modeling tool with the Brayton LCOE economic tool to provide a much more rigorous estimation of the likely LCOE for Brayton systems. Preliminary results show how the new tool can provide insights. The example above shows the trade-offs between increased recuperator cost, system efficiency, and LCOE and suggests that optimal recuperator effectiveness varies based on fuel prices.

A schematic of the Plant Dynamics Code being developed at ANL is shown in Figure 57. This code has been specifically developed for both steady-state and transient performance analysis for sCO_2 cycles as well as cycle control strategy as well. The Plant Dynamics Code targets specific features of the cycle: operation near the critical point of sCO_2 in the compressor and incorporated real CO_2 properties in the heat exchangers and turbomachinery without ideal gas assumptions.

In FY 2017, the Plant Dynamics Code was used to characterize innovative features of the cycle with dry heat rejection, and cascaded cycles. The code was validated using the ANL PCHE heat transfer testing facility, the SNL Simple Brayton Loop, the SNL RCBC, and the Bechtel Marine Integrated Small-scale Test.

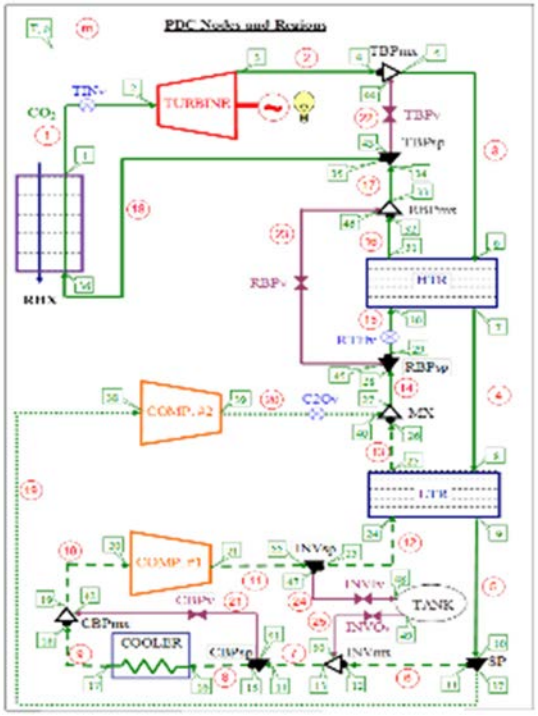


Figure 57. Schematic of the ANL Plant Dynamics Code.

sCO_2-Na reaction experiments continued in FY 2017 in open pool and in a semi-circular channel mock-up to investigate the self-sealing phenomena previously observed. At temperatures above 300°C, a small leak of high temperature sCO_2 into liquid sodium self-plugs in a very short time. The onset of the leak can be detected acoustically and stops when the leak plugs. Twelve sCO_2-Na tests were conducted, including in open pool and the channel mock-up. Prototypic bounding temperatures and pressures were recorded to support self-plugging model development.

Also studied in FY 2017 was nitrogen-sodium and inert gas-water tests in support of alternative power cycles. The apparatus is shown in Figure 58.

Figure 58. sCO_2-Na heat exchanger integrity R&D.

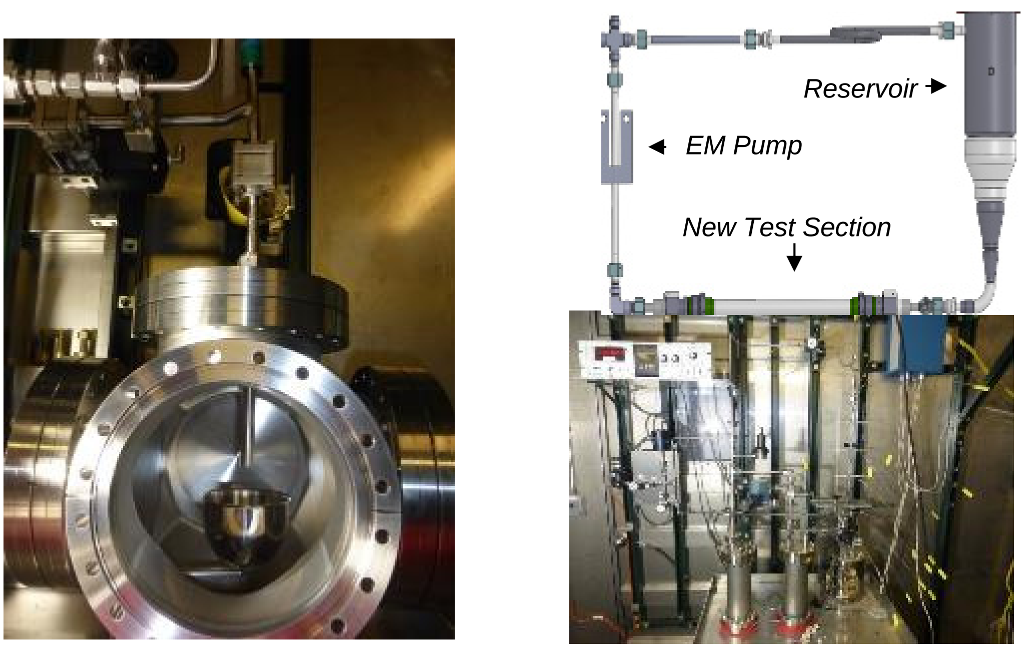


Figure 59. SFR intermediate heat exchanger development R&D.

In FY 2017, three experiment facilities (Figure 59) were assembled to obtain data essential to the reliable design of intermediate sodium-to-CO_2 heat exchangers, a critical requirement for the sCO_2 Brayton cycle to be successfully deployed together with SFRs.

Complete assembly of sodium Draining and Refilling facility was needed to conduct sodium-draining tests. The test aimed to determine lower limit on sodium channel size in compact heat exchangers to drain sodium in the required short timescale, prevent sodium bridges across the channel from being left inside of the sodium channels, and provide data for model validation. For the SFR, in the event of detection of sodium leakage or sodium fire, deliberate shutdown and draining of sodium -from intermediate sodium loop into dump tank- is required to minimize released sodium mass. The requirement is to drain intermediate sodium loop in fifteen minutes implying that sodium-to-CO_2 heat exchanger must be drained even more quickly. This requires orientation of the heat exchangers with sodium channels vertical or having significant vertical component.

If air ingresses the drained loop through the leakage site, then sodium remaining inside of the heat exchanger channels could be oxidized forming sodium oxide, Na_2O, which remains solid up to 1275°C. Worst case would be the retention of sodium bridges heat exchanger sodium channels and oxidizes forming solid plugs. Channels would be blocked and cannot be refilled with sodium. Sodium oxide plugs can be dissolved out by repetitively washing with hot purified sodium but that process can take a long time (weeks or months) making it impractical. Thus, there is a need to assure that heat exchanger design promotes efficient draining, and precludes retention of sodium bridging heat exchanger sodium channels.

Activities in FY 2017 were limited to reconstruction of the apparatus to allow plugging to occur only in the test sections, control of oxygen content in the sodium using cold traps, and new electromagnetic pumps and flow meters to handle the larger non-test sections.

11. REFERENCES

INL/EXT-16-40127, *Advanced Fuels Campaign 2017 Accomplishments*, Revision 2, October 2017.

INL/EXT-17-42999, *Updated Draft ASME Boiler and Pressure Vessel Code Case for Use of Alloy 617 for Construction of Nuclear Components for Section III, Division 5*, Revision 0, August 2017.

INL/EXT-17-43182, *Radial Deconsolidation and Leach-Burn-Leach of AGR-3/4 Compacts 3-3, 12-1, and 12-3*, Revision 0, September 2017.

U.S. Department of Energy, *Public Access Plan*, July 2014.

Appendix A
Acronyms

AFC	Advanced Fuels Campaign
ANL	Argonne National Laboratory
ATF	accident-tolerant fuel
ATR	Advanced Test Reactor
BUWT-PA	brush-type ultrasonic waveguide phased array
CEA	French Alternative Energies and Atomic Energy Commission
CoDCon	Co-Decontamination Project
CRP	coordinated research project
CY	calendar year
DOE	Department of Energy
DOE-NE	DOE Office of Nuclear Energy
E&S	evaluation and screening
EC	Energy Conversion (Project)
EPMA	Electron Probe MicroAnalyzer
EPP	elastic perfectly-plastic
EURATOM	European Atomic Energy Community
FCT	Fuel Cycle Technologies
FFTF	Fast Flux Test Facility
FY	fiscal year
GIF	Generation-IV International Forum
HFEF	Hot Fuel Examination Facility
HTGR	high temperature gas-cooled reactor
IAEA	International Atomic Energy Agency
IHX	intermediate heat exchangers
IMCL	Irradiated Materials Characterization Laboratory
INERI	International Nuclear Energy Research Initiative
INL	Idaho National Laboratory
IRT	Integrated Recycling Test
JAEA	Japan Atomic Energy Agency
JFCS	Joint Fuel Cycle Studies
KAERI	Korean Atomic Energy Research Institute
LCC	liquid-cadmium cathode
LEU	low-enriched uranium
LOCA	loss of coolant accident
LSFS	Laboratory-Scale Feasibility Study
LWR	light-water reactor

METL	Mechanisms Engineering Test Loop
MOX	mixed oxide fuel
MPACT	Materials Protection, Accounting, and Control Technologies
MRWFD	Material Recovery and Waste Form Development
MSR	molten salt reactor
NE-4	DOE Office of Nuclear Technology Research and Development
NEA	Nuclear Energy Association
NES	nuclear energy systems
NEUP	Nuclear Energy University Program
NNSA	National Nuclear Security Agency
NRC	Nuclear Regulatory Commission
NTD	National Technical Director
NTRD	Nuclear Technology Research and Development
OECD-NEA	Organization for Economic Cooperation and Development Nuclear Energy Agency
ORNL	Oak Ridge National Laboratory
OSTI	Office of Science and Technology
PCHE	printed circuit heat exchangers
PNNL	Pacific Northwest National Laboratory
PWR	pressurized water reactor
RCBC	Recompression Closed Brayton Cycle
R&D	research and development
RD&D	research, development, and demonstration
sCO_2	supercritical carbon dioxide
SFR	sodium-cooled fast reactor
SNL	Sandia National Laboratories
SQT	sensor qualification test
SSC	System Steering Committee
SSPM	Safeguards and Security Performance Model
STAGE	Scenario Toolkit and Generation Environment
TREAT	Transient Reactor Test facility
TRL	technology readiness level
TRU	transuranic
TSRA	Technology and System Readiness Assessment
U.S.	United States
USV	under sodium viewing
UWT	ultrasonic waveguide transducer
VRE	variable renewable energy