Fuel Cycle Technologies 2016 Achievements Report

Fuel Cycle Research and Development

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Prepared for
U.S. Department of Energy
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Fuel Cycle Technologies

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CONTENTS

FIGURES

TABLES

Fuel Cycle Technologies 2016 Achievements Report

1. INTRODUCTION

The U.S. Department of Energy (DOE), Office of Nuclear Energy (DOE-NE) established the Fuel Cycle Technologies (FCT) program to address challenges with growing stockpiles of used nuclear fuel (UNF) and high-level waste (HLW) and address the urgency to enhance accident tolerance of the existing reactor fleet. Resolution of these challenges involves developing systems that reduce waste while improving resource utilization and safety. Towards this goal, the FCT program implemented a research and development (R&D) science-based approach, integrating theory and experiment with high performance computing modeling and simulation.

The FCT program establishes and strengthens collaborations with experts at international agencies, universities, and with industry to develop the future nuclear fuel cycle. 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 the venue to integrate the expertise, experience, and ideas into commercial practice.

In 2016, DOE announced the Gateway for Accelerated Innovation in Nuclear (also referred to as GAIN), which was established to provide the nuclear energy community with 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 FCT program supports this important effort through R&D and provides innovative and impactful technologies and capabilities that advance commercialization of nuclear energy.

DOE recognized the shortage in the nuclear technology and science talent base, not only in the United States (U.S.), but also in many countries. It is necessary to attract, retain, and expand this knowledge base, as well as initiate education and training programs for innovative nuclear technologies. Whereas this is not a unique problem to any given country, the Nuclear Energy Association (NEA) developed the NEA Nuclear Education, Skills, and Technology (NEST) Framework to address gaps in nuclear skills capacity building, knowledge transfer, technical innovation, and look at long-term options to manage radioactive waste and spent nuclear fuel (SNF). The FCT program has initiated, in conjunction with the NEA NEST framework, an opportunity for students and young professionals in multiple countries to learn and develop technical skills and experience to solve grand challenges facing the nuclear community.

To achieve the goals and objectives of the FCT program, eight R&D campaigns are engaged in impactful research, development, demonstration, and deployment:

- Fuel Cycle Options 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.
- 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.
- Fuel Resources Program investigates the feasibility of recovering uranium (U) from seawater as an option for nuclear fuel resources.
- Used Fuel Disposition (UFD) Research and Development Campaign provides options to develop technology for storage, transportation, and disposal of used fuel and wastes.
- 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.
- Nuclear Fuel Storage and Transportation (NFST) Program develops options for integrated waste management systems and consolidated storage facilities via safe transportation of nuclear fuel.
- Advanced Fuels Campaign (AFC) performs science-based R&D on accident tolerant fuels (ATFs) in addition to advanced light-water reactor (LWR) fuel and transmutation fuel.

2. FUEL CYCLE OPTIONS CAMPAIGN

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2.1 Overview

2.1.1 Mission

The Fuel Cycle Options mission has evolved to reflect the shift in emphasis towards more general studies of strategic interest related to nuclear power use. Specifically, the campaign:

- Informs DOE-NE fuel cycle program development, planning, and budget formulation by conducting studies on the impacts of nuclear power deployment for both continuing with the current U.S. fuel cycle and transitioning to alternative sustainable fuel cycles (considering the entire integrated fuel cycle and using processes and tools developed by the campaign).
- Contributes to integration of, and the basis for, DOE-NE fuel cycle R&D activities and informs decisions on R&D and infrastructure needs in a systematic manner.
- Analyzes prospective and sustainable integrated fuel cycles to identify benefits and challenges that lead to the development of common fuel cycle technology goals.
- Supports the assessment of ongoing fuel cycle R&D programmatic risks by developing the processes and facilitating the technology readiness assessments for the fuel cycle R&D campaigns and development of associated technology development roadmaps.
- Communicates the results of campaign activities to support what the fuel cycle R&D program is doing and why the R&D is being done.

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. **Domestic and International Deployment of Nuclear Energy**: Develop an understanding of the potential role of nuclear power in the domestic and global energy market by analyzing possible future nuclear power deployment considering both current and alternative nuclear fuel cycles. The analyses include using market-driven scenarios and considerations of the effects of economics and other external issues such as CO_2 management, and any associated climate-change policies and emissions targets, to identify and what aspects of nuclear fuel cycles can influence the use of nuclear power.
2. **Costs and Economics of Nuclear Power**: Develop improved estimates for nuclear facilities costs and fuel cycle economics, and the potential for reducing costs with alternate and/or advanced technologies such as small modular reactors, molten-salt reactors, and sodium-cooled fast reactors (SFRs) for recycle, mainly by analyzing historical cost and economic data on nuclear facilities and using this information to either inform on projected costs of reactors and other fuel cycle facilities, or by developing models to allow applying the cost information obtained from the historical information to other types of nuclear fuel cycle facilities including those that have not been previously constructed or those using technologies that have not yet been deployed.

3. **Transition to an Alternative Sustainable Nuclear Fuel Cycle**: Perform analyses of transition from the current U.S. fuel cycle to an alternative sustainable fuel cycle, indicating the effects on nuclear power economics, waste generation, resource requirements, and other performance metrics such as greenhouse gas emissions, using a range of assumptions about growth in the use of nuclear power, the likely costs of such fuel cycles, the availability of the required fuel materials and supporting infrastructure, other uses for nuclear energy, and competition from other energy sources, using the results to develop basic understanding about beneficial fuel cycle characteristics beyond those identified in the Nuclear Fuel Cycle Evaluation and Screening study.
4. **Capabilities and Limitations of Technology-Specific Fuel Cycles**: Continue development of the understanding of the challenges and benefits associated with the current and alternative sustainable nuclear fuel cycles, including transition to an alternative sustainable fuel cycle, by analyzing technology-specific fuel cycles at both steady-state and transition using both existing technologies and technologies currently being developed in the DOE-NE R&D programs to determine the constraints associated with transition and the ability of the new technologies to provide the performance potential identified in the Nuclear Fuel Cycle Evaluation and Screening study.
5. **Novel Fuel Cycle Concepts from Industry, Universities, and Government**: Perform analyses of specific fuel cycle concepts or supporting technologies from industry, universities, and others, mainly to continue to identify any differences from the Nuclear Fuel Cycle Evaluation and Screening study results and seek to reconcile them, but also to explore the impact of such concepts or technologies on overall fuel cycle performance, including the use of ATFs, small-modular reactors, and high-temperature reactors.
6. **Technology Readiness Assessment**: Continue development and support of objective and verifiable processes and assessments of technology maturity, evolving the Technology Readiness Assessment process as needed, and working with the R&D campaigns to both facilitate assessments of technology readiness level (TRL) for all critical technologies and supporting development of Technology Development Roadmaps by the R&D campaigns to estimate the time, funding, facilities, and personnel required to bring technologies to maturity, leading to overall assessment of program risk.
7. **Communication**: Continue to contribute to a broader understanding of nuclear power potential to address issues such as greenhouse gas emissions while providing energy for electricity and other uses, the potential for reducing the costs of nuclear power leading to more attractive economics, and the urgency for performing R&D in order to meet external constraints such as retirement of current nuclear reactors, while also providing basic information about nuclear fuel cycles, maintaining efforts for the online Fuel Cycle Catalog to foster understanding of nuclear fuel cycles, their capabilities, and their limitations.

In fiscal year (FY) 2016, the activities of the campaign were organized into two technical areas: (1) integrated fuel cycle analysis and (2) development, deployment, and implementation issues, consistent with the Fuel Cycle Options Campaign objectives listed above.

2.1.3 Key FY 2016 Deliverables

Key deliverables for the campaign summarized results in each of the two technical areas. In the *Integrated Fuel Cycle Analysis* area, the campaign performed analyses of complete integrated fuel cycles under equilibrium conditions (i.e., after transition to an alternative fuel cycle has been completed). In the *Development, Deployment, and Implementation Issues* area, the campaign focused on performing analyses of transition from the current U.S. fuel cycle to an alternative fuel cycle, considering deployment and implementation options, economics, and impacts of modular facilities to inform on choices, decisions timing, and costs.

Key deliverables/reports in FY 2016 include:

- Analyses of Advanced Fuel Cycle Options (9/15/2016)
- FY 2016 Study on Fuel Cycle Impacts of Accident Tolerant Fuels (9/30/2016)
- Minor Actinide Study Report (10/12/2015)
- Advances in Developing Improved Cost Estimates (6/1/2016)
- Updated Sections of the Fuel Cycle Cost Basis Report (9/30/2016)
- Nuclear Fuel Cycle Options Catalog: FY 2016 Improvements and Additions (8/31/2016)
- Transition Analysis to Fast Reactor U/Pu Continuous Recycle (12/15/2015)
- Transition Analysis to Fast Reactor U/TRU Continuous Recycle (3/15/2016)
- Transition Analyses to U/Pu Continuous Recycle in Fast and Thermal Reactors (7/29/2016)
- FY 2016 Update to Transition Economics Assessment (8/15/2016)
- Technology and System Readiness Assessment Process for R&D Evaluation (6/30/2016)
- Regional and Global Impacts of Nuclear Energy (9/16/2016).

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

2.2 Integrated Fuel Cycle Analysis

2.2.1 Analyses of Advanced Fuel Cycle Options

Information on innovative fuel cycle and reactor concepts currently under development by industries, universities, and national laboratories was collected. The claimed fuel cycle performance benefits were reviewed and compared to the current U.S. fuel cycle. The primary purpose of the study was to place the concepts into the appropriate fuel cycle evaluation groups and check if the claimed improvements were consistent with the findings of the Nuclear Fuel Cycle evaluation and screening (E&S) study for potential fuel cycle benefit, a study completed in FY 2015 and available at www.fuelcycleevaluation.inl.gov. These assessments were supported as necessary by additional fuel cycle performance analyses.

The majority of the concepts use innovative reactors with higher thermal efficiency to replace the current LWRs for either achieving the resulting better fuel cycle performance associated with such an increase in thermal efficiency (but still using a once-through fuel cycle), or obtaining much more substantial waste reduction and increased resource utilization by using continuous recycle of UNF. A large fraction of the innovative concepts use small modular reactors with power output less than 300 MWe. Enhanced passive safety features, greater security, lower environmental impacts, and lower construction costs are among the claimed benefits that are common to many of the concepts utilizing the innovative reactors. Overall, the fuel cycle performance claims were found to be generally consistent with the results of the E&S study. However, in order to achieve the claimed fuel cycle performance benefits, some of the innovative concepts adopted or assumed the development of non-conventional technologies, such as fuels that vent fission gas during irradiation, recladding of UNF, remote reactor operation by satellite link based on a sophisticated reactor control system at a central location, advanced cladding materials, liquid fuel in fuel tubes, etc. Consequently, for most of the concepts, significant R&D is needed to develop and demonstrate the new technologies prior to any deployment of the nuclear energy systems using these innovative concepts.

2.2.2 Minor Actinide Study

This study, mostly completed in FY 2015 and finalized at the beginning of FY 2016, performed a further examination of the relative benefits and challenges of U/Pu and U/transuranic (TRU) fast reactor recycle fuel cycles (TRU is Pu and the minor actinide elements). This included options where the fast reactors provide fissile material to thermal reactors, which are the most promising fuel cycles from the E&S study. These fuel cycles have essentially the same potential for improvement in fuel cycle performance compared to the current U.S. fuel cycle, but with U/TRU fast reactor recycle fuel cycles appearing to require more R&D based on the relative immaturity of such concepts. Considerations beyond the scope of the E&S study were used to either confirm the similarity in potential performance benefit or identify any differences that could be used to support the need for the additional R&D required to mature U/TRU recycle technologies. The study determined that the performance benefits are still essentially the same for both U/Pu and U/TRU fast reactor recycle fuel cycles, even with the additional considerations, but highlighted that the main difference between U/Pu and U/TRU recycle was in the parts of the fuel cycle where R&D is required. For U/Pu recycle, the R&D challenges are in HLW development (and possibly disposal of HLW with higher minor actinide content), while for U/TRU recycle, the R&D would need to focus on recycle fuel fabrication, in-reactor fuel performance, separations technologies, and HLW development from reprocessing (i.e., the R&D challenges mainly occur in the parts of the fuel cycle that contain the bulk of the minor actinides). Given that both U/Pu and U/TRU recycle fuel cycles have R&D challenges, a choice for either U/Pu or U/TRU recycle would likely depend on the preferences for conducting R&D either on HLW (and disposal of HLW with larger minor actinide content), or on recycle fuel fabrication, operation, and reprocessing, respectively. At this time, no further assessment was made of the relative difficulty of the R&D choices, or the likelihood of success for either choice, and a strategy that pursued both U/Pu and U/TRU recycle options would appear to be the logical approach until such time that the results of R&D begin to differentiate the development and deployment challenge and the subsequent project risk, if that should occur, with all most promising groups from the E&S study currently as potential candidates for development.

2.2.3 Economics Evaluation of the Promising Options

This study continued the examination of historical data, which focused on LWRs in FY 2015, to improve the credibility of facility and fuel cycle cost estimates. In FY 2016, the study examined the historical construction and operational costs of major reprocessing facilities, with special focus on the Thermal Oxide Reprocessing Plant (THORP), La Hague, and Rokkasho facilities, including comparisons to identify consistencies and discrepancies in reported costs, and with sensitivities to discount rates and operational life. Levelized cost calculations were performed to convert capital costs to unit costs. It was found that the cost of reprocessing facilities is generally driven by the complexity necessary to ensure adequate safety and security standards, and by the amount of highly trained personnel required to operate and support such facilities. More data was available for THORP than for other facilities because of the extended debates about THORP since its inception. Only two large commercial facilities are in operation today in the Western world, one (La Hague) owned by Areva in France and the other (THORP) owned by the United Kingdom's Nuclear Decommissioning Authority. Both facilities started operations in the

1990s, and have similar reported construction and operational costs. In contrast, the Rokkasho reprocessing facility has experienced severe delays and cost overruns. The cost of the Rokkasho facility, as provided by Japan Nuclear Fuel Limited, is about twice the cost of THORP and La Hague/UP3 for the same annual nominal reprocessing capacity and using essentially the same technologies, but there are differences. Insights on how the high costs for Rokkasho could be explained were provided in the study report.

2.2.4 Fuel Cycle Impacts of Accident-Tolerant Fuel

This work continued the evaluation of the system performance impacts of ATF fuel and cladding concepts from FY 2015. In FY 2016, the benefit metrics from the E&S study were evaluated for four additional ATF concepts being considered for implementation in LWRs in the U.S., and were compared to metric values for the current UO_2 fuel / zircaloy-clad fuel assemblies currently used in commercial LWRs ("basis of comparison" or EG01 in the E&S study). One of the "most promising" options from the EG29 E&S study, which includes an SFR and an LWR, was also evaluated with an ATF cladding (FeCrAl) and standard U-Pu mixed-oxide fuel in the LWR, and the results compared to those from the original EG29 with U-Pu mixed oxide fuel (MOX) and zircaloy cladding for the LWR. The results for the benefit metrics for the once-through configurations were generally similar to those for EG01. The configurations with molybdenum-based cladding and metallic U-Mo fuel, which required higher enrichments than for EG01, resulted in several of the metric values being degraded relative to those for EG01. The metric value and bin results for EG29 with FeCrAl cladding were essentially identical to those for the case with zircaloy cladding since they are dominated by the fast reactor (larger fraction of the power generated), and the use of the FeCrAl cladding had only a minor impact on the power sharing. In this study, an initial assessment of the "Development and Deployment Risk" criterion was also conducted. Since the focus of implementation of ATF concepts is the current once-through commercial LWR fuel cycle with intended compatibility with currently operating reactors, many of the individual metrics are similar to, or only slightly "worse" than for EG01 with the degree depending on how much a particular concept deviated from the conventional UO_2 zircaloy fuel. The greatest difference is in "Development Time," which is driven primarily by the need to demonstrate performance under reactor operating conditions (temperature, irradiation, etc.) to have the requisite confidence to support licensing and actual implementation.

2.2.5 Fuel Cycle Catalog

Maintenance and update of the publicly-available online Fuel Cycle Catalog continued during FY 2016. Several fuel cycle options, reactors, and fuels were added to the catalog during the year. In addition, a new online process for entering data from data packages was implemented. The capability, called the *Option Manager*, is currently being tested with data for six fuel cycle options received from the Nuclear Energy University Program (NEUP). The remaining tasks to be completed include developing the corresponding review and approval procedure for use with the online data entry process, and developing instructions for users.

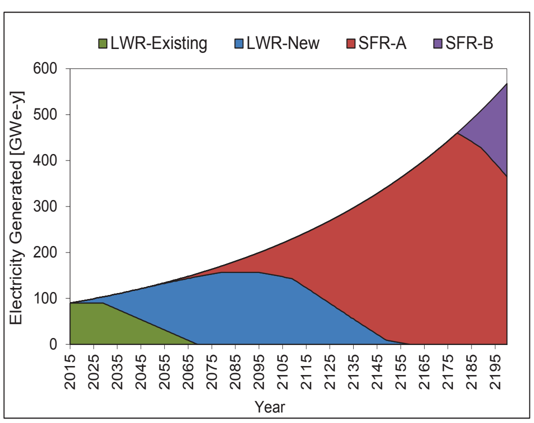
2.3 Development, Deployment, and Implementation Issues

2.3.1 Transition to an Alternative Uranium-Based Fuel Cycle

Performing a reasonably correct analysis of transition to a different fuel cycle is a highly complex analytical and modeling endeavor. One of the key challenges of performing transition analyses is the lack of robust tools. Therefore, as part of the work described in this section, the analysis capability is being enhanced and existing tools such as VISION and DYMOND are being improved.

While the recycle of U/Pu or U/TRU in fast reactors would have the potential fuel cycle performance benefits identified by the E&S study, changing to such a fuel cycle from today's use of LWRs using a once-through fuel cycle in the U.S. may introduce issues that need to be addressed in order to have a successful transition to the new fuel cycle. Transition scenario studies have been conducted for the most promising fuel cycle options to develop an understanding of the requirements for such a successful transition, especially identifying any issues or constraints that would inhibit or prevent such a transition. The scenario studies explored a range of effective transition strategies, with the goal of developing an understanding of transition issues, times, costs, and constraints to enable development of effective transition strategies, to identify robust transition pathways that consider economic conditions, energy demand, etc., and to identify the decisions that need to be made, the time frame for such decisions, and the effects of delaying decisions. This ultimately requires consideration of a broad range of possible implementing technologies and future conditions to inform decision-makers.

A major consideration during transition is having sufficient fissile material for starting the fast reactors before the fast reactor fleet is self-sustaining by recycling its own fuel. An example of a transition strategy is shown in Figure 1 for a transition to a fast reactor U/Pu recycle fuel cycle replacing all of the LWRs, using an energy growth of 1% per year. Following an initial period of slower deployment, mainly to simulate a case where a learning period is provided for the new fast reactors, the fast reactor fleet goes through a period of rapid growth as the fleet expands from just a few reactors to 100 or more. A source of fissile material for fast reactor fuel is required during this period, both for initial startup as well as to supplement the fissile material available from recycled fast reactor fuel, and can be either from recycled LWR fuel or low enriched uranium (LEU). As the fast reactor fleet expands, the fraction of fissile material that needs to be added from outside the fuel cycle decreases and by the end of transition, the fleet has become self-sufficient (i.e., all new fissile material required is produced by the fast reactor fleet that only requires a small ongoing input of natural U, but is much smaller than for continuing with the current U.S. LWR fleet as Figure 1 illustrates).



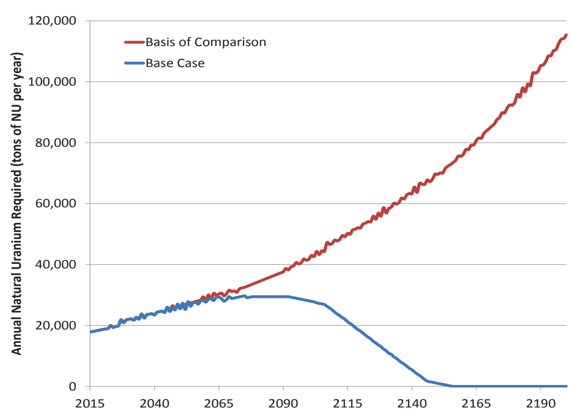


Figure 1. Transition to fast reactor recycle of U/Pu and the reduction needed for natural U compared to the current U.S. fuel cycle.

2.3.2 Nuclear Energy, Renewable Energy with Storage, and Climate Change Mitigation

Continuing work from previous years, in FY 2016, sensitivity scenarios of nuclear energy and renewable energy with energy storage were investigated for addressing global climate change. Both the impact of the proposed U.S. EPA Clean Power Plan (CPP) and global climate mitigation policy for limiting global warming to 2°C were assessed for the U.S. under alternative assumptions of nuclear capital costs, as well as energy storage availability and costs. The analysis employed an enhanced version of the global change assessment model that provides representations of energy storage systems and the integration costs of the use of intermittent energy generation sources. The analyses were conducted in two parts, with the first part showing that reductions in the nuclear capital cost have a tremendous impact on increasing the market share of nuclear power and on lowering power sector CO_2 emissions. Nuclear capital cost reductions alone, facilitating greater use of nuclear, have the potential to reduce U.S. power sector CO_2 emissions to a level comparable to the reduction projected for the proposed EPA CPP. Moreover, the combination of the CPP and low-cost nuclear allowed CO_2 emissions to even fall below the CPP final goals. Lowering nuclear capital costs increased nuclear energy shares, allowed easier compliance with the CPP, and resulted in overall greater use of electricity.

For the second part of the analysis, a simplified model of the curtailed energy fraction of renewable energy arising from the mismatch in electricity load profiles, but with the capacity for large-scale energy storage, was included to analyze the potential for greater use of renewable energy when large-scale storage is available. The analyses, which explored renewable energy (wind and solar) deployment with energy storage under the 450-ppm atmospheric CO_2 emission constraint, showed that energy storage supports the greater use of renewable energy, but that the ultimate potential of renewable energy is hindered by their relatively shorter facility lifetime (25 years or less). The more frequent need for large amounts of capital to replace generation facilities using renewables due to the shorter facility lifetime, relative to nuclear or carbon capture and storage technologies, was shown to constrain renewable energy market penetration. The share of renewable energy with low cost energy storage is estimated to peak in 2050 at 33% of total electricity generation and fall to 27% by the end of century. Nuclear and carbon capture and storage shares are predicted to increase to 31% and 36%, by 2100, respectively. The dynamic interactions of technology costs, technology lifetimes, and deployment history were shown to affect the future deployment of technology options. In the U.S., the analyses showed that the significant historical deployment of nuclear energy and its near-future retirement have tremendous implications for the reduction of power sector CO_2 emissions and future technology choice, depending on which technologies replace the current U.S. LWR fleet.

2.3.3 Transition Economics Assessment – FY 2016 Update

The scenario for deployment of SFRs to replace LWRs had assumed that the initial fuel for the SFRs would be from the recovered fissile material from the reprocessing of the LWR used fuels. The impact of using LEU, but with higher assay of $5\% < {}^{235}U$ / total $U < 20\%$, has now been evaluated. The study found that the separations facility utilization could be improved through employment of LEU fuel for startup of SFRs and could result in cost reductions that more than offset any increased costs elsewhere in the fuel cycle. In particular, the resulting improvement in the utilization of the first separations plants for recycling used SFR fuel that significantly reduced unit costs for those facilities, with the unit cost for the first plant in an SFR-optimized case being reduced by 84%. These cost improvements could be very important for making the initial facility deployments cost effective. The study also found that the option of using a fuel for starting the SFRs that was not dependent on the products from reprocessing LWR spent fuel effectively eliminated the need for simultaneous development and deployment of technologies for both the fast reactors (and the Pu-containing fast reactor fuels) and SFR UNF reprocessing, while eliminating the need for LWR UNF reprocessing altogether. Such an approach allows fast reactor deployment with enriched-U fuel (still LEU as noted above, around 16–18% ^{235}U/total U), and allows optimizing the time for introducing SFR UNF reprocessing to maximize the capacity factors of both the reprocessing facilities and the U/Pu recycle fuel fabrication facilities. The economic value of this decoupling will be analyzed, but currently can be noted that such an approach also represents a significant risk reduction versus having the start of transition dependent on simultaneous development and deployment of SFRs, U/Pu fast reactor recycle fuel, LWR UNF reprocessing, and SFR UNF reprocessing. It also reduces the risk for individual facilities if there is a short-term mismatch between supply and demand at any stage of the process.

2.3.4 Technology Readiness Assessment Methodology

Development of the Technology and System Readiness Assessment (TSRA) Process for R&D Evaluation was completed in FY 2016 for informing the planning and decision-making processes for the R&D of advanced nuclear energy systems and the implementing technologies. The TSRA process was derived from DOE G 413.3-4A, “Technology Readiness Assessment (TRA) Guide,” and was informed by recent efforts on TRL and TRA process. The TSRA process is intended for use on any system, whether just a single technology, or a very large complex system with many interrelated technologies such as an entire nuclear energy system. The TSRA process can also be used to monitor the progress of technology development and to facilitate choices between competing technologies to reduce program risk. Periodic repetition of the TSRA process can be used to provide a quantitative assessment of the progress of R&D efforts, and as R&D stages or elements are completed, updating of the effort required for the remaining R&D.

3. MATERIAL RECOVERY AND WASTE FORM DEVELOPMENT

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3.1 Overview

The MRWFD Campaign is responsible for developing 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 the R&D efforts performed within the MRWFD Campaign in FY 2016. Each subsection contains a high-level overview of the activities and key results, produced during the fiscal year. More detailed accomplishments are available in the *Material Recovery and Waste Form Development FY-2016 Accomplishments Report*.

This section briefly outlines the campaign mission, objectives, and challenges and highlights key technical accomplishments made during FY 2016. 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 the International Atomic Energy Agency [IAEA]), integration of MRWFD Campaign activities with other FCT campaigns, (primarily Advanced Fuels, UFD, Fuel Cycle Options, and Material Protection, Accountancy, and Control Technology), and integration with DOE Offices of Environmental Management, Science , and the National Nuclear Security Agency.

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.

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.

3.1.1 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.

3.1.2 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 U/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 UNF.
- Develop advanced waste forms with greatly improved properties and cost and develop/demonstrate associated processes.

3.1.3 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 life-times measured in units of millions of years.

3.2 Reference Technologies and Alternatives

This activity supports development of on-line monitoring tools, evaluation of solvent degradation mechanisms, and development of tritium removal technologies (for open and closed fuel cycle applications). The focus in FY 2016 was on continuing a collaboration with the Commissariat à l' Énergie Atomique (CEA) regarding on-line monitoring. This collaboration focuses on the combination of a micro Raman probe with a microfluidic sample chip for implementation in aqueous reprocessing facility. Testing of a closed loop NO_2 oxidation of sim-fuel at low temperature was continued to determine the effects of process parameters on performance, in preparation for future proposed testing with actual fuel. The MRWFD Campaign continued a collaboration with the European Union Framework 7 SACSESS program (Safety of ACtinide Separation Processes) by participating in an international workshop, performing solvent degradation studies, and performing an Advanced TALSPEAK flowsheet test with simulated feed at Forschungszentrum Jülich. Investigation of separating tritiated water from normal water was performed using silicoaluminophosphate membranes. Degradation studies of the ALSEP and innovative-SANEX process solvents under representative process conditions were performed.

3.3 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 2016, co-precipitation of actinyl ions (U, Np, Pu, Am) continued to be investigated and showed promising results. Progress was also made on understanding the radiation stability of Am(VI) under radiolysis conditions, along with the extraction of Am(VI) in 3D printed centrifugal contactors. Significant progress on the development and understanding of the ALSEP solvent extraction process was made, including development of a process flowsheet proposed for testing with simulated feed in FY 2017.

3.4 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). Five major thrust areas were included in the activities for FY 2016: (1) iodine capture in which the impacts of penetrating organic forms of iodine were studied, (2) iodine immobilization in which scale-up testing of fused silica based waste form for silver functionalized aerogel was demonstrated, (3) tritium separations in which a process for separating tritium from irradiated hulls chlorination was developed, (4) 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, and (5) integrated off-gas treatment system development in which an engineering study to develop a reference off-gas treatment flowsheet was completed.

3.5 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 2016, the radiolysis of diglycolomide ligands was further investigated in support of the ALSEP process development efforts as well as the European Union SACSESS 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. Two research directions continued harvesting data to support f-element solution chemistry efforts of MRWFD campaign. One focus area collected f-element optical absorbance data to construct a cross-actinide matrix of extinction coefficients to ease f-element monitoring in solution mixtures. The second effort was geared towards structural modification of aminopolycarboxylate aqueous holdback reagents to facilitate liquid-liquid systems capable of fast equilibration and efficient An/Ln separation. Thermodynamic data were collected for the complexation of Np(V) with HEDTA in a wide pH region.

3.6 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 HLW raffinate and a durable waste form for radio-iodine. 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 2016, a direct comparison of multi-phase ceramic waste forms produced via melt processing and HIP methods was performed. A rheology study of glass ceramic waste forms was performed to determine the impact of crystallization on the melt rheology. Equipment and procedure modifications were completed and a first, large-scale chlorination test of irradiated UNF cladding was completed to demonstrate Zr recycle. Improved loading ceramic waste forms for electrochemical salt HLW were developed and tested at laboratory scale.

Studies of the long-term performance of glass were continued. A model for the mechanism of ion-exchange process was parameterized and the result published in a prestigious journal. An approach to implementing a model of corrosion acceleration was developed. Studies of the long-term performance of example multiphase radioiodine waste forms were initiated. 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. The results of those studies along with the current status of steel-based waste forms will be documented in FY 2017.

3.7 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 2016, testing of kg-scale U/TRU co-deposition using a high-current-density solid cathode was continued using rare earth elements as surrogates for TRU. Gram-scale U/TRU co-deposition tests were also completed using actual salt samples from the Mark IV electrorefiner at the INL Fuel Conditioning Facility. Investigations on electrorefining and monitoring of electrochemical systems utilizing fluoride salt systems (rather than chloride salt systems) were continued in FY 2016.

3.8 Key Fiscal Year 2016 Deliverables

- Issued the FY 2015 Annual Technical Accomplishments report. This report highlights the research accomplishments from the previous fiscal year and places the research activities in context of the overall program objectives.

- The MRWFD campaign had significant publications in impactful journals in FY 2016, including one article in the journal *Science* and (Figure 2) another article in *Nature Communication.* These articles demonstrate the significant impact that the campaign is having on the broader scientific community.
- Demonstrated the Advanced TALSPEAK process, for the separation of Am and Cm from TRUEX raffinate streams using simulated feeds Forschungszentrum Jülich (Figure 3).

Figure 2. Cover of *Science* journal.

Figure 3. Centrifugal contactor battery installed in the Julich Laboratories used for testing the Advanced TALSPEAK process.

- A flowsheet has been designed for an upcoming demonstration of the ALSEP process for separating Am and Cm from lanthanides and other fission products in a co-decontamination raffinate.
- A successful hot test of tandem Am(III) oxidation and extraction demonstrated the generation and immediate extraction of Am(VI) together with its recovery by reductive stripping.
- Successfully demonstrated scaled consolidation of iodine-loaded silver aerogel with hot isostatic pressing and spark plasma sintering (Figure 4).

Figure 4. Pucks of iodine loaded Ag^0-aerogel produced with hot isostatic pressing (left) and spark plasma sintering (right).

- Characterized the contents of legacy Kr-85 waste form capsules by scanning electron microscopy (SEM)-electron dispersive spectrometry and x-ray diffraction (Figure 5).

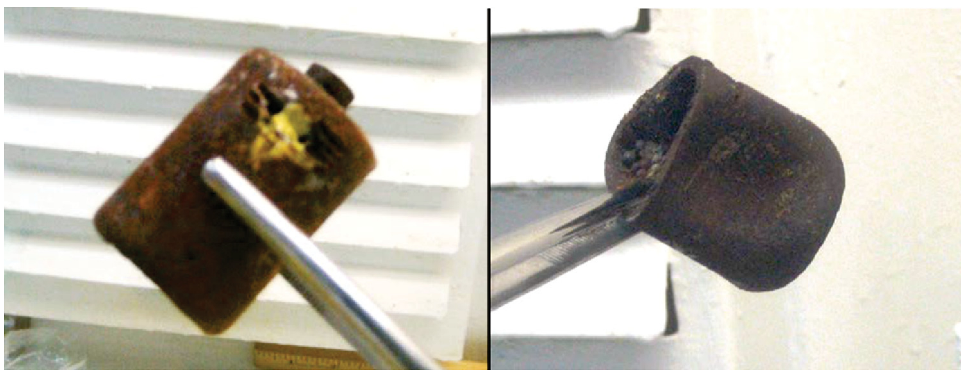

Figure 5. Kr-85 legacy waste form capsule 2 (left) and capsule 5 (right).

- Completed an initial large-scale test of Zr recycle technology using UNF. The test used 100 g of zircaloy-4 and M-5 cladding with burnups ranging from 9–50 GWd/MT (Figure 6).

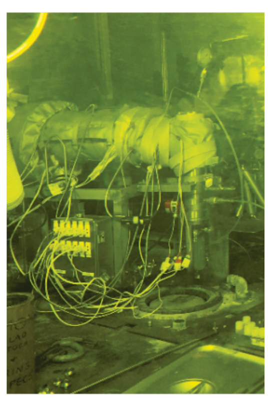

Figure 6. Zr recycle test equipment in hot cell.

3.9 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 world-wide and have resulted in a number of publications in prestigious journals and invitations to present at international conferences.

4. MATERIAL PROTECTION, ACCOUNTING, AND CONTROL TECHNOLOGIES PROGRAM

Mike Miller, Los Alamos National Laboratory (LANL), NTD

4.1 Overview

The MPACT Campaign mission is to develop 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 as well as the back end of the open fuel cycle.

Simply improving nuclear material measurement performance is not enough to meet timeliness detection goals for advanced fuel cycle options (Figure 7), instead an integrated systems approach is required that fully utilizes additional operational data streams. In addition, integration of safeguards and security considerations and technologies may provide economic benefit in addition to enhanced system performance for all fuel cycles (Figure 8).

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. Additionally, a Modeling and Simulation Roadmap was developed, as well as the *Used Fuel Extended Storage Security and Safeguards by Design Roadmap*.

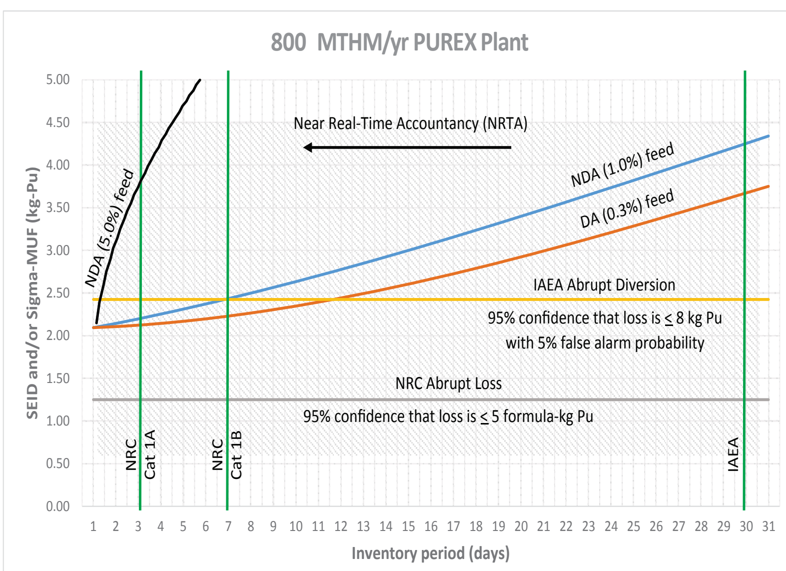

Figure 7. Standard error of the inventory difference/sigma material unaccounted for as a function of the inventory period for a commercial-scale reprocessing plant showing challenge to meeting timeliness goals even with the low uncertainty afforded by destructive analysis (0.3%).

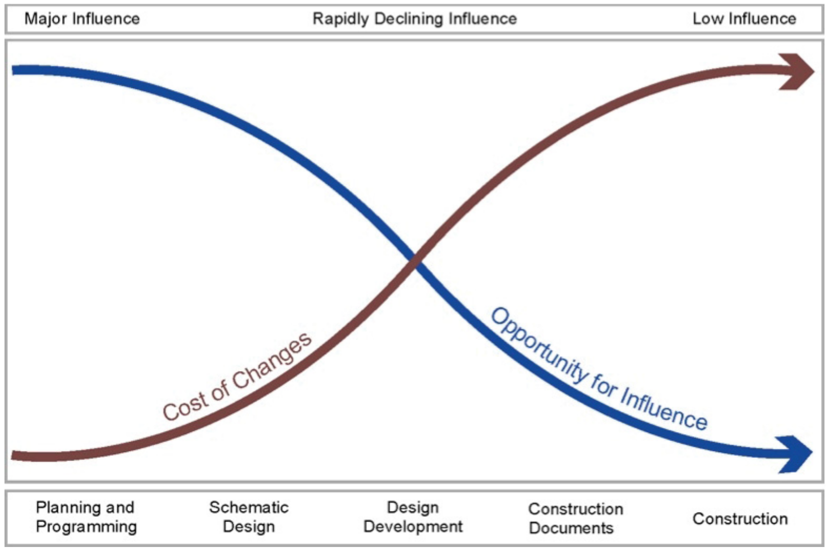

Figure 8. Notional relation between cost of changes and opportunity for influence as a function of project stage.

4.1.1 Objectives of MPACT Campaign

- Develop tools, technologies, and approaches in support of used fuel safeguards and security for extended storage, electrochemical processing, and other advanced nuclear energy systems.
- 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.

4.1.2 Challenges and Drivers for MPACT Campaign

- Storing used fuel for an extended time until an ultimate disposition pathway is available.
- 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.

- Testing technologies with high 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. nuclear energy systems.
- 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.

4.1.3 Key FY 2016 Deliverables

- Electrochemical Process Monitoring for Enhanced Safeguards
  - Analysis of actinide sensor tests with plutonium and preparation of material for a U sensor
  - Level/density probe fabricated, calibrated ,and installed for field test
  - Analysis of tests of microfluidic sampling and design of a high-throughput droplet generator
- Modeling and Simulation for Electrochemical Processing Safeguards
  - Issuance of Modeling and Simulation Roadmap
  - Integration of security scenarios with electrochemical Safeguards and Security Performance Model
  - Development of Advanced Integration example for electrochemical recycling and issuance of an Advanced Integration Roadmap
- Security Evaluations of Used Nuclear Fuel in Extended Storage
  - Security considerations for consolidated interim storage facility (ISF)
  - Issuance of *Used Fuel Extended Storage Security and Safeguards by Design Roadmap*
- Advanced Neutron and Gamma-Ray Instrumentation
  - Published review paper published characterizing microcalorimeter performance and assessment of microwave high throughput readout
  - High-dose neutron counter development and testing
  - Multi-isotope process (MIP) monitor testing (H-Canyon) during process campaigns.

4.2 Major Research and Development Activities

4.2.1 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 actinide) 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 SEM analysis of samples from the runs was completed. Fabrication, calibration and qualification of the level/density sensor planned in concert with the JFCS was also completed. The sensor has been placed into the hot cell for testing during the JFCS integrated recycle test (IRT) activities, and related out of cell hardware installed (Figure 9). In addition, a micro-analytic sampling system was designed and a water-based version tested. Finally, improvements to experimental and modeling techniques for molten salt solutions were validated and a robust voltrammetric sensor module designed, fabricated, and tested.

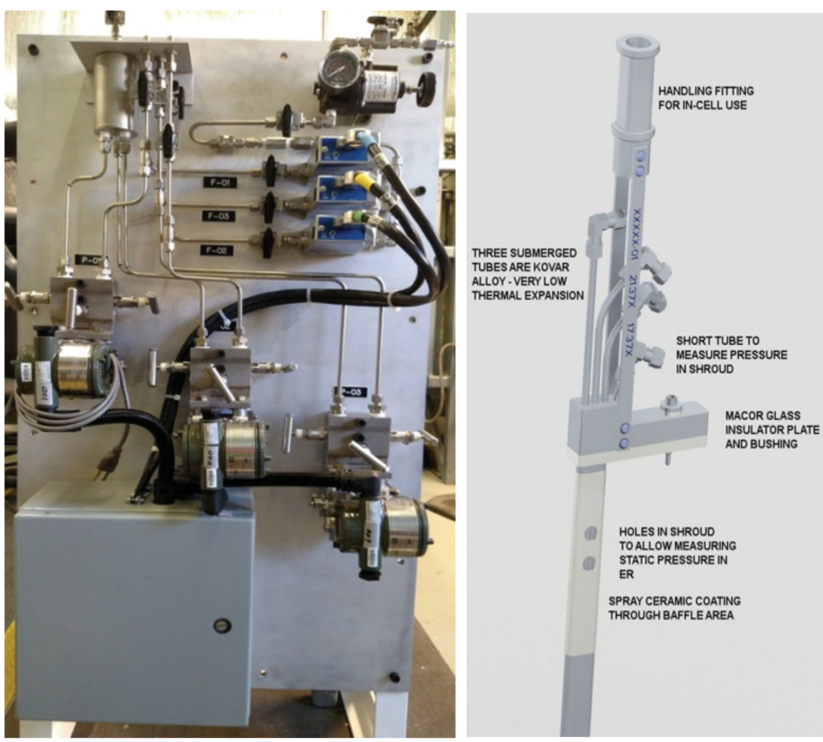

Figure 9. Gas panel for level/density sensor (left), drawing of sensor (right).

Modeling and Simulation for Electrochemical Processing Safeguards: Advanced radiation transport calculations (using Monte Carlo N-Particle (transport code) coupled to application-specific algorithms) have been performed for the Planar Electrode and Mark-IV electrorefiners to identify signatures for advanced monitoring instrumentation development, including dynamic models incorporating moving of materials. 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 this year was updated and validated through comparison with experimental data. At the facility level, the Argonne Model for Pyrochemical Recycling (AMPYRE), which calculates the mass balance of a complete electrochemical processing facility, was updated to include tracking at the isotopic level. We continued to develop the Safeguards and Security Performance Model, incorporating results from the other modeling activities, including integration of the Scenario Toolkit and Generation Environment (STAGE) to provide physical security modeling capability. In 2016, the *Modeling and Simulation Roadmap* was issued, outlining current approaches and gaps in the areas of radiation transport and sensors, process and chemical models and shock physics and assessments. Based on MPACT modeling, simulation, and other technologies, an example of advanced integration for electrochemical recycling was developed and included in the *Advanced Integration Roadmap* released in 2016 (Figure 10).

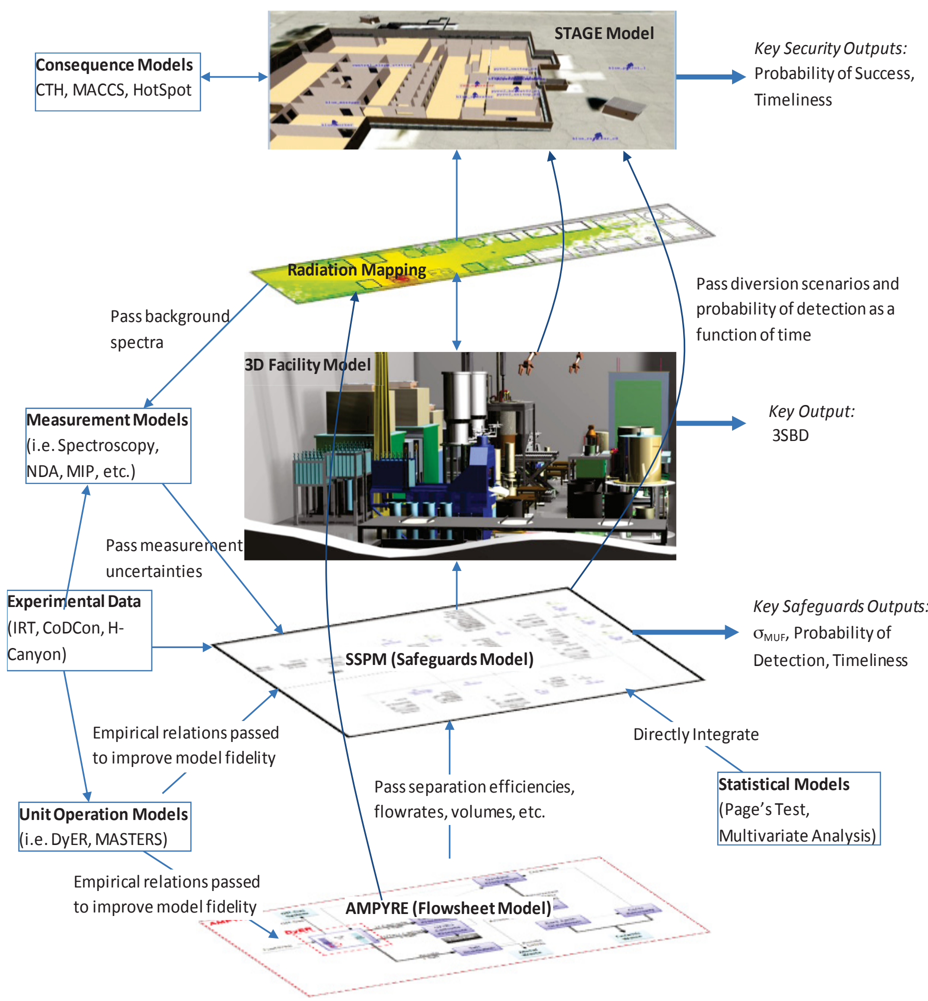

Figure 10.Example of advanced integration for electrochemical recycling.

4.2.2 Used Fuel Extended Storage

Concepts and approaches are being developed for integrated safeguards and security for used fuel extended storage. This includes the risk-informed security analysis (vulnerability and consequence), assessment of and addressing technology gaps, and providing leadership in the area of best practices for security of dry storage. This effort is coordinated with the UFD campaign and the NFST planning project.

Security Evaluations of Used Fuel in Extended Storage: Focused on a preliminary evaluation of a generic pilot storage facility design from NFST, continued numerical evaluation of the spent fuel ratio, consequence analysis for dry cask sabotage, and dynamic self-protection assessment capability. In FY 2016, high consequence bimodal attack modeling was performed, validating the source term for this type of attack with previous simulations. Preliminary security evaluations on a storage facility were also completed, with recommendations on augmentation to security forces issued in a classified report. Finally, the *Used Fuel Extended Storage Security and Safeguards by Design Roadmap* was released in 2016. This work is being transitioned to the new Office of Waste Management given maturity level is ready for implementation.

4.2.3 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, engineering-driven science-based R&D program is being conducted to improve precision, accuracy, speed, sampling and monitoring methods, and scope of nuclear material accounting and control. 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. Testing of this technology is planned as part of the JFCS IRT. Previous testing showed that the advantage of increased resolution can in principle lead to an improvement in determining plutonium isotopic composition by a factor of 10–60, thereby showing promise to break the current 1% uncertainty barrier.

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 (Figure 11). 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.

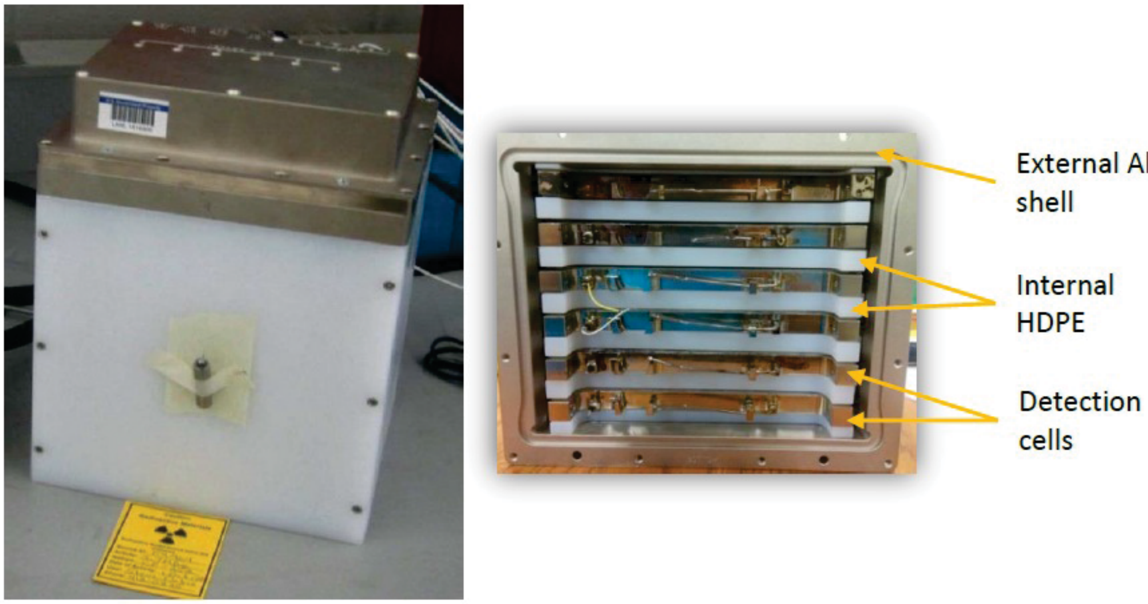

Figure 11. B-10 lined parallel plate detector comprised of six sealed corrugated cells with high density polyethylene in between cells.

The MIP monitor, a near real-time monitor for SNF reprocessing facilities, started a series of field tests at the Savannah River National Laboratory H-Canyon in 2015, with encouraging initial results. Six data collection campaigns were conducted in 2016, including first cycle solvent extraction preparation runs (i.e., cold runs) and one hot run with SNF (Figure 12**Error! Reference source not found.**).

Figure 12. MIP monitor test set-up in H-Canyon hot sample aisle.

In-situ measurement of Pu concentration was investigated in a U/TRU ingot using thermocouples in 2016 with successful preliminary surrogate testing. Testing of the approach is planned during the JFCS IRT experiments to determine Pu concentration in U/TRU output products (Figure 13).

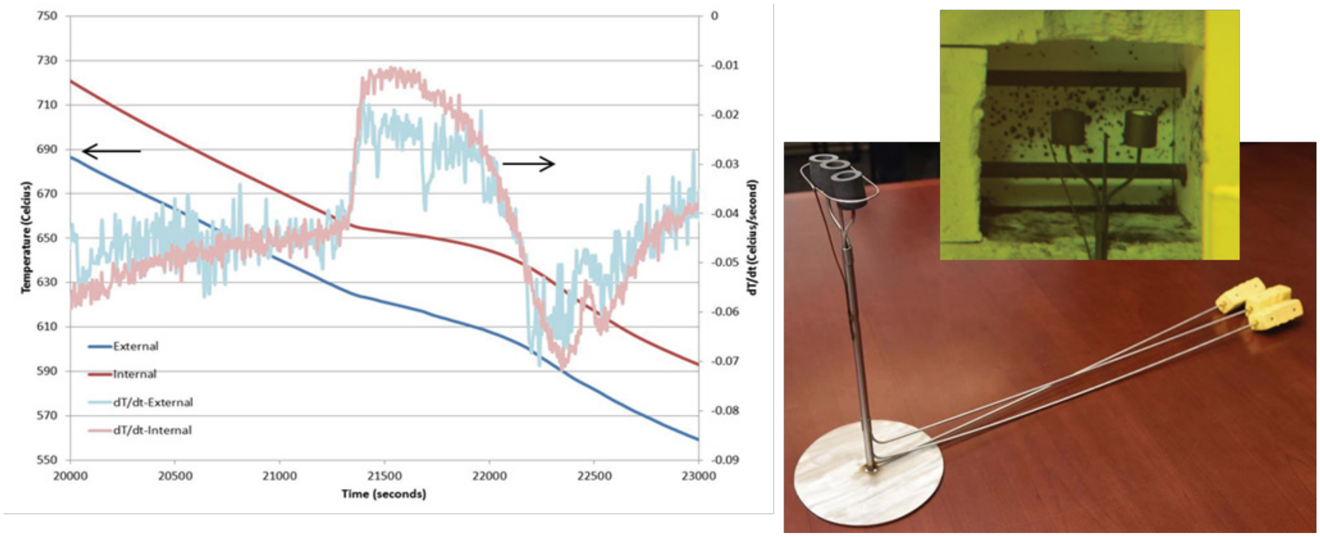

Figure 13. In-situ measurement experimental thermocouple configuration (right), surrogate testing data (left) from 8g Al 6061 alloy with heat of fusion similar to 100g U/TRU.

5. FUEL RESOURCES PROGRAM

Stephen Kung, DOE, NTD

5.1 Overview

The Fuel Resources Program seeks to identify and implement actions DOE can take to assure the long term availability of economical nuclear fuel. The program evaluates fuel resources and develops recovery technologies to increase resource accessibility to enable a sustainable fuel cycle. Priority attention in the near term is focused on developing the technology for extraction of U from seawater. The following activities are included in this effort: (1) Technical Support/Management provides technical coordination of R&D activities within the Fuel Resources Program and coordinates participation in working group and review meetings, as well as international cooperative activities. (2) Advanced Grafting focuses on the development of advanced adsorbent materials prepared by irradiation (electron-beam and gamma-ray) induced and chemical grafting methods to increase the U adsorption capacity and selectivity; (3) Advanced Nanosynthesis incorporates nanotechnology and nano-manufacturing techniques into the development of advanced adsorbent materials to provide increased selectivity and capacity for U recovery; (4) Ligand Design and Thermodynamics uses computational screening tools to rationally design and evaluate ligands for enhanced selectivity and capacity, followed by rational synthesis of promising ligands for subsequent experimental validation. (5) Marine Testing and Modeling conducts sorption and U recovery experiments in several distinct marine environments, while the modeling component provides data and adsorption models for scale-up and evaluation of marine deployment. (6) Cost Analysis conducts cost and energy analyses and develops cost/energy models for newly developed adsorbents and technologies. The purpose of this subtask is to aid in focusing R&D efforts on achieving evidence-based cost minimization strategies. (7) Durability and Recycle conducts material durability evaluation and degradation studies during the reuse of adsorbent to reduce the technology cost and performance uncertainties. Contributions to the aforementioned activities are made by researchers at Oak Ridge National Laboratory (ORNL), Lawrence Berkley National Laboratory, and PNNL, as well as by university collaborators under the NEUP.

Overall, the main accomplishment of the Fuel Resources Program in the past six years is revealed in Figure 14, where it is shown that adsorbents developed in this program have tripled the U extraction from seawater compared to adsorbents developed over the previous 50 years.[1] Specifically, in 2016, adsorbents developed by atom-transfer radical polymerization exceeded a capacity of 6 g U per Kg adsorbent, following contact with seawater for a period of 8 weeks. Two additional classes of materials also exceeded 5 g U/Kg adsorbent: adsorbents prepared by radiation-induced graft polymerization on high-surface area trunk polymers, and commercially-available polymers, which were surface modified without radiation through application of a simple, scalable chemical treatment. These remarkable achievements are the result of increased adsorbent density on trunk polymer fibers accompanied by an optimized ratio of amidoxime ligand and co-polymer organic acid. In addition to increased adsorbent U capacity, enhanced understanding of the functional activity of the amidoxime ligand has been demonstrated through advanced spectroscopy, computational chemistry, and adsorption modeling. Major accomplishments in all the Fuel Resources Program activities are described in this section.

[1] The adsorbents displayed in Figure 14 are not time normalized and it is worth noting that some samples reported previously in the literature had been deployed in seawater for up to one year – several times longer than the conditions used to investigate samples prepared by the Fuel Resources Program. Accounting for deployment duration would afford a greater than five times enhancement in time-normalized uranium uptake.

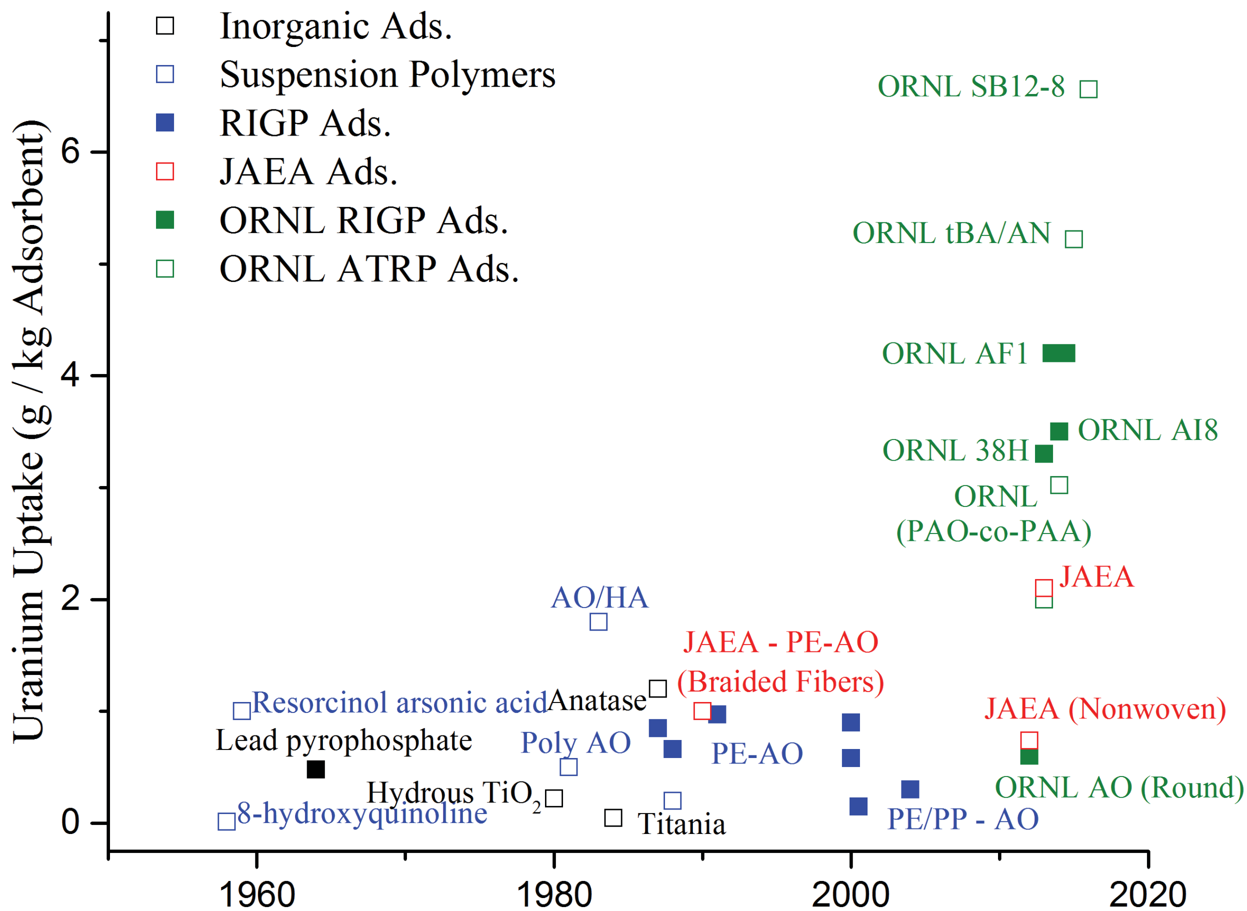

Figure 14. Review of U adsorbent capacity reported in the literature for adsorbents developed in the past six decades. Adsorbents developed by the Fuel Resources Program in the past six years demonstrated a three-fold increase in U capacity from seawater, as compared to adsorbents developed in the previous 50 years. Data are not normalized for deployment time.

5.2 Development of Novel Amidoxime-Based Polymeric Adsorbents

Various novel fiber adsorbents were synthesized in FY 2016 using radiation-induced graft polymerization, atom-transfer radical polymerization (ATRP), and through surface modification of commercially available acrylic fibers. The goal was to develop new materials of high U adsorption capacity and selectivity.

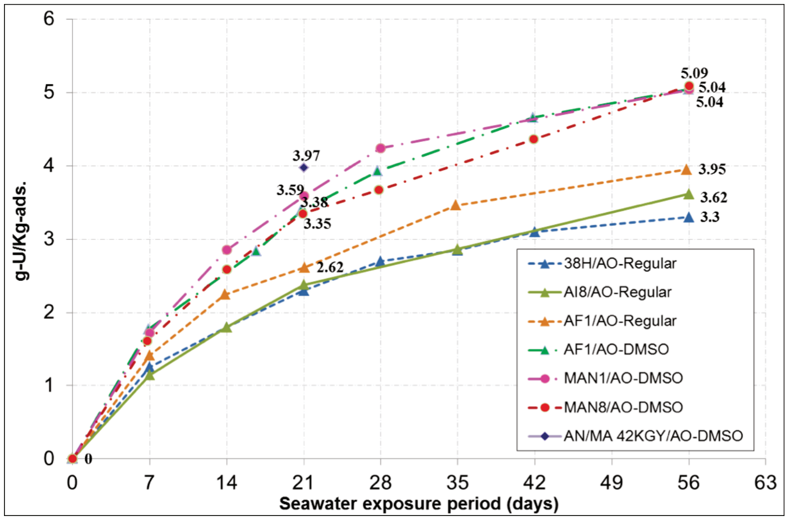

Figure 15. Comparison of U adsorption capacities for ORNL's new (2016) and old adsorbents (regular).

For the radiation-induced graft polymerization subtask, the focus has been on improving U adsorption capacity, kinetics, and selectivity by the incorporation of new hydrophilic co-monomers onto the high-surface-area polyethylene trunk fibers and optimization of a large number of synthesis and process conditions. Several new adsorbents were developed in FY 2016 that demonstrated U adsorption capacities ranging from 5.0 to 5.4 g-U/kg-adsorbent after 56 days of seawater exposure at 20°C in flow-through column experiments at PNNL (Figure 15). A key step in achieving these capacity improvements was the discovery of using dimethyl sulfoxide as the solvent in the amidoximation reaction, as opposed to using the conventional water-methanol solution. In addition, new co-monomers were grafted onto the high-surface-area polyethylene fibers that enhanced the adsorption capacities including methacrylonitrile, hydroxyethyl acrylate, and methyl acrylate. The adsorbent grafted with methyl acrylate and acrylonitrile, AN/MA-42kGy-AO-DMSO, yielded particularly high uptake kinetics and attained a U adsorption capacity of 4.0 g-U/kg-adsorbent after only 21 days of seawater exposure at 20°C in flow-through columns. This result translated to a greater than 50% increase in adsorption capacity as compared to the AF1 adsorbent. Promising results were also realized by several adsorbents that were tested at Broad Key Island, FL, in flumes containing ambient flowing seawater, and included 56-day capacities of 6.4 and 6.8 g-U/kg adsorbent for the AF1 and AF1-AO-DMSO adsorbents, respectively. Adsorbents having high U selectivity were also developed that contained hydroxyethyl acrylate and acrylonitrile. These adsorbents achieved relatively low V/U ratios of less than 1.85.

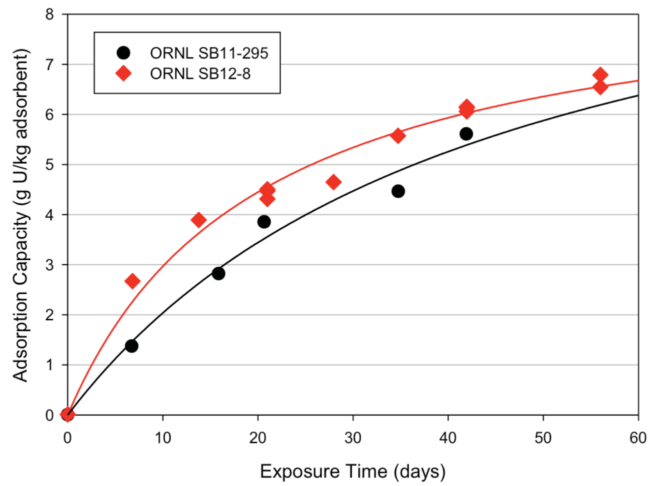

Figure 16. Kinetic data of U adsorption by ATRP synthesized adsorbent at ORNL.

ATRP-synthesized fiber adsorbents, including P(AN-co-HEA) on PVC-co-CPVC fibers, showed the highest performance so far, achieving 6.56 g U/kg adsorbent after 56 days exposure in natural seawater (Figure 17).

Figure 17. Example of an LCW adsorbent material prepared using the polyacrylonitrile fiber K1 provided by ORNL. The fiber was shaped into a form factor for exposure in the PNNL flume for time series capacity testing.

PNNL staff collaborating with LCW Technologies and the University of Idaho have developed an amidoxime-based polymeric adsorbent (Figure 4) using commercially available and inexpensive acrylic fibers. An adsorption capacity of 5.28 ± 0.16 g U/kg adsorbent (LCW-10 adsorbent, Table 1) was achieved after 56 days seawater contact. The LCW adsorbent has a half-saturation time (11.2 ± 1.3 days) that is about half of the AF1 adsorbent (22.9 ± 1.7 days) and displays greater U selectivity; the LCW-10 adsorbent has a V/U mass ratio of nearly 1, compared to ORNL adsorbents which typically have V/U mass ratios between 1.2 and 4. Because of the inexpensive starting material and simple production chemistry (no radiation exposure or surface polymerization is necessary) this adsorbent is predicted to have a production cost around $330/kg, which makes it competitive with current U mining technologies. PNNL and LCW Technologies jointly submitted an invention disclosure to the U.S. Patent and Trademark Office on this new technology entitled: "Converting acrylic fibers to amidoxime-carboxylate containing polymer adsorbents for sequestering U and other elements from seawater," Application number 15/179,766.

5.3 Summary of Adsorption Capacity Studies Using Flow-Through Columns

Significant advances in U adsorption capacity have been achieved in FY 2016 by adsorbents synthesized under the Fuel Resources Program. The new adsorbents are compared in Table 1 to earlier adsorbents developed in this program. The adsorbents tested in prior years include the 38H, AI8, and AF1 adsorbents that were developed at ORNL. All the others listed in Table 1 were developed and tested in FY 2016. There is nearly a doubling in adsorption capacity over the approximately 4 years of testing that this table covers (2012–2016), from the 38H adsorbent with a 56-day capacity of 3.30 g U/kg adsorbent to the more recent SB12-8 adsorbent with a capacity of 6.56 g U/kg adsorbent.

Table 1. Summary of flow-through column testing of amidoxime-based polymeric adsorbents developed by the Fuel Resources Program using filtered Sequim Bay seawater at PNNL.

Adsorbent	N	Saturation Capacity[a,b] (g U/kg adsorbent)	56-day Adsorption Capacity[a,b] (g U/kg adsorbent)	Half-saturation Time[a,b] (days)
38H	4	4.29 ± 0.24	3.30 ± 0.18	16.9 ± 2.8
AI8	1	5.17 ± 0.18	3.54 ± 0.12	25.8 ± 2.1
AF1	5	5.56 ± 0.15	3.91 ± 0.11	24.0 ± 1.5
AF1FR2	1	7.05 ± 0.21	5.00 ± 0.15	22.9 ± 1.7
MAN1-AO/DMSO	1	6.70 ± 0.22	5.04 ± 0.16	18.5 ± 1.6
MAN8-AO/DMSO	1	7.75 ± 0.37	5.09 ± 0.24	29.2 ± 3.0
AN/MA/42kGY-a1	1	8.43 ± 0.72	5.13 ± 0.44	36.0 ± 6.1
LCW-MSL-10	1	6.34 ± 0.19	5.28 ± 0.16	11.2 ± 1.3
SB12-8	1	8.90 ± 0.45	6.56 ± 0.33	20.0 ± 2.6

a. Determined at a temperature of 20°C using one-site ligand saturation modeling.
b. Normalized to a salinity of 35.

5.4 Effect of Current Velocity of Adsorption Capacity

A collaborative study into the effect of current velocity on amidoxime-based polymeric U adsorbent performance was conducted by PNNL in collaboration with ORNL and Georgia Tech. Markedly different results were obtained depending on whether the exposure was conducted using a flow-through column or a recirculating flume (Figure 18). There was a minor difference in U adsorption capacity as a function of the linear velocity for the seawater exposure using flow-through columns, but a very significant increase in adsorption capacity was observed with increasing linear velocity in the recirculating flume studies. The 56-day U adsorption capacity at a linear velocity of 0.48 cm/s was 2.02 ± 1.08 g U/kg adsorbent, while the 56-day U adsorption capacity at a linear velocity of 8.24 cm/s was 4.71 ± 0.20 cm/s, more than a two-fold difference.

Modeling results showed that the mass-transfer coefficient increased mostly linearly with seawater velocity in the flume studies, while remaining flat for the column studies. The difference in adsorbent performance between the columns and the flume can be attributed to two features: (1) flow resistance provided by the adsorbent braid in the flume, which significantly reduces the seawater velocity through the braid and (2) enhancement in braid movement (i.e., fluttering) as linear velocity increases. Based on the flume studies, we suggest that when ocean currents are greater than approximately 6 cm/s, adsorption capacities will be maximized for a given adsorbent braid of a certain fiber density and form-factor.

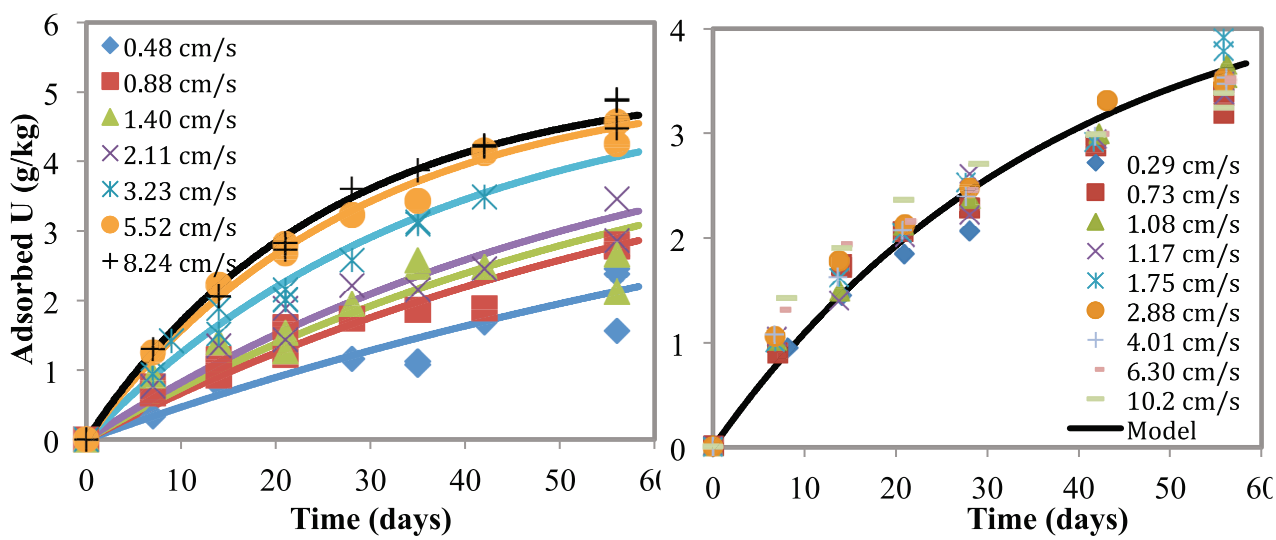

Figure 18. Time series measurements of U adsorption capacity as a function of the linear velocity of the ambient seawater exposure. Left panel: Exposure in a recirculating flume. Right panel: exposure in a flow-through column.

5.5 Marine Testing at the University of Miami's Broad Key Island Research Station

Marine testing at Broad Key Island, FL, was conducted to validate adsorption capacity and adsorption kinetics results obtained in Sequim Bay, WA, and to assess the effect of different oceanographic and water quality conditions (e.g., temperature, dissolved organic carbon, salinity and trace element content) on U uptake. Several formulations of the ORNL amidoxime-based polymeric adsorbents were investigated. Marine testing at Broad Key Island offers the opportunity to test adsorbent performance under warmer ambient and more saline conditions than those exist at the marine test site on Sequim Bay off the Washington coast. This is particularly important since the U capacity of amidoxime-based adsorbents responds strongly to temperature; the higher the temperature, the higher the U adsorption capacity.

Flow-through column and recirculating flume experiments were conducted on five different amidoxime-based adsorbent materials, four produced by ORNL (AF1, AI8, AF8, and AF1-DMSO) and one by LCW technologies (LCW-10), using ambient filtered seawater and identical exposure systems. All exposures were conducted at ambient seawater temperatures in order to provide results consistent with a natural seawater deployment in Florida coastal waters. The ORNL adsorbents AF1, AI8, and AF1-AO-DMSO all had fairly similar adsorption capacities (6.0 to 6.6 g U/ kg adsorbent) after 56 days of exposure at ambient temperature (26 to 31°C) and salinity (35.7 to 37.4), while the AF8 adsorbent was considerably lower at 4.4 g U/kg adsorbent (Table 2). All adsorbents tested at Broad Key Island had higher capacities than those observed at PNNL, with the higher temperatures likely a major factor contributing to this difference.

Table 2. Comparison of 56-day U adsorption capacities and half-saturation times with ORNL adsorbents for flume experiments conducted at Broad Key Island and at PNNL.

	PNNL		Broad Key Island			
Adsorbent	56-day Adsorption Capacity[a] (g U/kg adsorbent)	Half Saturation Time (days)	Average Temperature (°C)	Average Salinity	56-day Adsorption Capacity[b] (g U/kg adsorbent)	Half Saturation Time (days)
ORNL AF1	3.86 ± 0.18	23 ± 1	30.1 ± 0.7	35.7 ± 0.9	6.35 ± 0.10	25 ± 1
ORNL AI8	3.54 ± 0.17	26 ± 2	26.6 ± 1.4	35.9 ± 0.7	5.96 ± 0.24	21 ± 2
ORNL AF8[c]			26.6 ± 1.4	35.9 ± 0.7	4.43 ± 0.81	51 ± 2
ORNL AF1-AO-DMSO	5.04 ± 0.16	18 ± 2	30.6 ± 1.1	36.4 ± 0.7	6.77 ± 0.56	21 ± 2

a. Determined from one-site ligand saturation modeling of time series data obtained at a temperature of 20°C and normalized to a salinity of 35.
b. Determined from one-site ligand saturation modeling of time series data obtained at the ambient temperature and ambient salinity given in the table.
c. Characterization of AF8 at PNNL is ongoing.

In general, the elemental distribution (expressed as a relative percentage) on all the adsorbents agreed well, including good agreement with the elemental distribution pattern for AF1 adsorbent exposed at PNNL (Figure 19). The most notable exception to a uniform elemental distributional pattern occurs with V. The relative mass percentage for vanadium retained by the adsorbents ranged from a minimum of 13% for the AF8 formulation to a maximum of 29% for the AI8 formulation; expressed in terms of a V/U mass ratio, it varies from a low of 1.2 to a high of 2.7 for AF8 and AI8 adsorbents, respectively. All V/U mass ratios at Broad Key Island are lower than those observed for the AF1 adsorbent at PNNL, with temperature likely playing a significant role. As U has a higher adsorption capacity at higher temperatures, one would expect that warmer exposures would favor a lower V/U mass ratio, which could explain why the V/U mass ratio for the PNNL exposures are higher than observed for the Broad Key Island exposures.

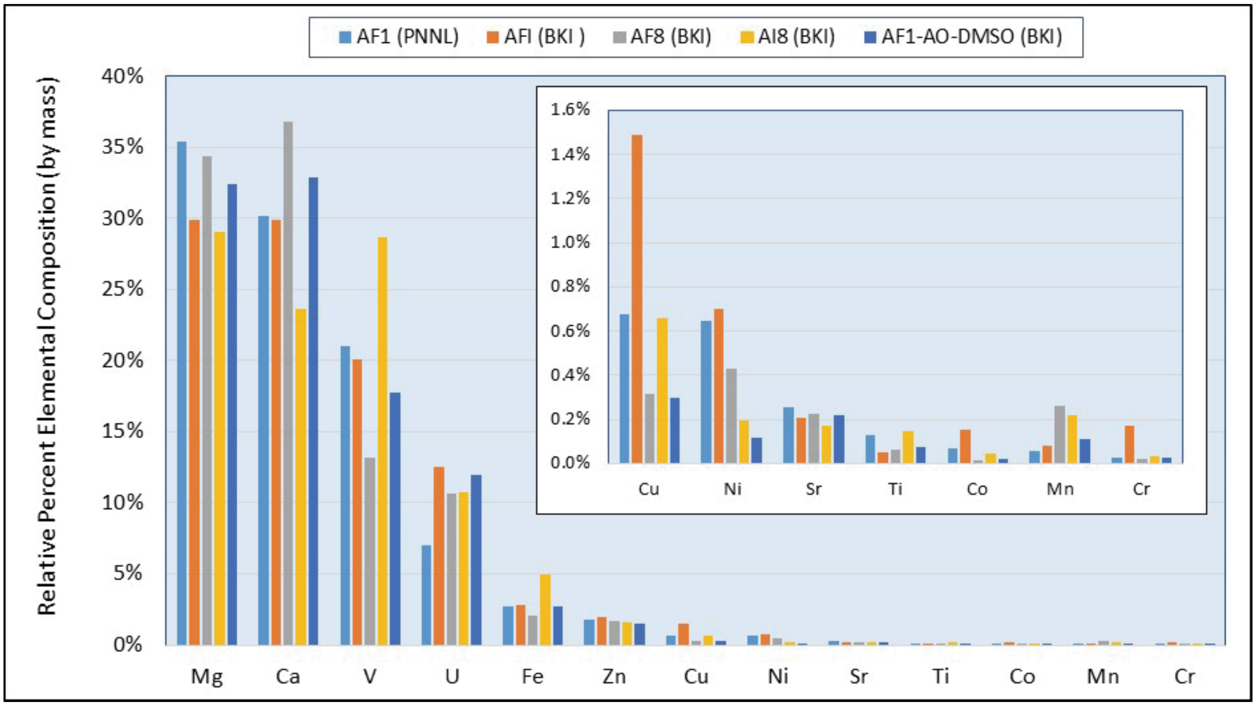

Figure 19. Relative percentage of the major elements adsorbed onto four different ORNL formulations of amidoxime-based adsorbents from 56-day exposures at Broad Key Island, FL. Also included for comparison is the AF1 adsorbent elemental distribution determined at PNNL in Sequim Bay seawater.

5.6 Ligand Design, Characterization, and Thermodynamic Studies

5.6.1 Ligand Design Modeling

A key step in predicting ligand selectivity and efficiency at sequestering U is the ability to accurately predict the stability constants (i.e., K_1 values, commonly reported in logarithmic form) for the uranyl and major competing VO_2^+ and VO^{2+} ions. A computational protocol has been developed based on density functional theory calculations to accurately predict the log K_1 for UO_2^{2+}, VO_2^+ and VO^{2+} complexes (Figure 20). This protocol was used to elucidate the main factors influencing the selectivity of the current generation of amidoxime-derived sorbents. As follows from our results, the cyclic imide dioxime (H_2IDO) affords a more preferable configuration for sequestration of U from seawater than the acyclic amidoxime (HAO). At the same time, however, IDO^{2-} shows stronger binding affinity and higher selectivity for VO_2^+ over UO_2^{2+} and is likely responsible for the higher sorption of vanadium

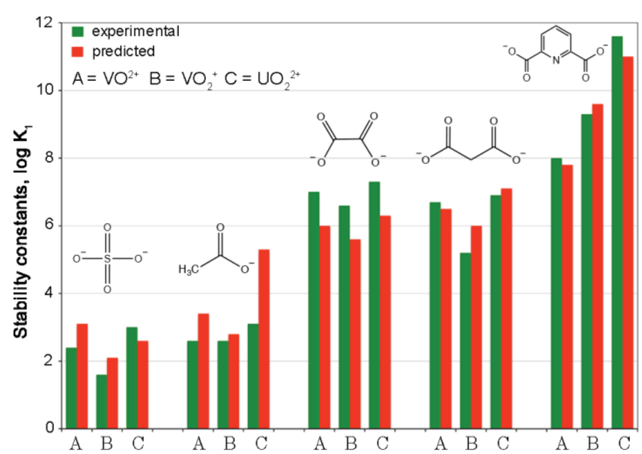

Figure 20. Comparison of experimental (green) and predicted (red) log K_1 values for A) VO_2^+, B) VO^{2+}, and C) UO_2^{2+} complexes with identical ligands.

ions in marine tests, while AO^- does not appear to bind the VO_2^+ ions at all under seawater conditions. Thus, selectivity of poly(acrylamidoxime) adsorbents toward UO_2^{2+} vs VO_2^+ could be improved by minimizing the formation of the cyclic imide dioxime. It was also found that simple dicarboxylic functional groups possess low binding affinity and selectivity for uranyl because they are poorly organized for the UO_2^{2+} complexation, which is consistent with experiments. Moreover, the obtained data enabled us to propose the utilization of the ligand design principles based on structural preorganization to achieve a dramatic enhancement of carboxylates in UO_2^{2+} ion binding affinity and selectivity. This concept was exemplified through the investigation of the complexes of the UO_2^{2+}, VO_2^+, and VO^{2+} ions with the highly preorganized ligand PDA (1,10-phenanthroline-2,9-dicarboxylic acid), which was found to be very selective for uranyl.

5.6.2 Characterization

Efforts in the characterization activity involved the continued application of X-ray Absorption Fine Structure (XAFS) spectroscopy to determine how the adsorbent polymers actually bind U. Due to possessing high sensitivity and atomic specificity, XAFS is the only technique capable of directly investigating the coordination environment of metals extracted from environmental seawater by amidoxime-functionalized polymers. Work completed during the previous year revealed small molecule surrogates do not adequately represent how the more structurally and chemically complex adsorbents bind U, which was published in the journal *Energy and Environmental Science* (Impact Factor 25.4). This article was also highlighted on the rear cover of the journal, as well as in "DOE Pulse," the webpage of the Advanced Photon Source, and energy.doe.gov. XAFS investigation of an adsorbent copolymer composed of amidoxime and phosphonic acid groups revealed similar behavior. This constitutes an instance of emergent phenomena, macroscopic behavior arising from the interaction of molecules that individually do not display such properties, and is expected to be critical for rational development of adsorbents possessing the desired U uptake and selectivity. XAFS data regarding the vanadium binding environment have also been collected, but in contrast to the aforementioned U-binding mode, preliminary analysis indicates vanadium is bound in a consistent fashion between the adsorbent polymer, the small molecule standards, and the computationally-predicted binding model. These results are in the process of being published. Ongoing efforts involve more detailed interrogation of emergent phenomena through investigation of U-binding as a function of polymer chain length and morphology by application of XAFS and small angle neutron scattering.

5.6.3 Thermodynamic and Structural Studies

A rare, non-oxido V(V) complex with glutaroimide-dioxime (H_3L), $Na[V(L)_2]\cdot 2H_2O$(cr), was crystallized from aqueous solution and characterized via x-ray diffraction. The complex was found to contain two fully deprotonated L^{3-} ligands bound to the bare V^{5+} cation via two oxime oxygens and the imide nitrogen. An intermediate complex, $Na[VO_2(HL)]$(cr), was also isolated and found to contain the typical VO_2^+ moiety present in many V(V) complexes. Further characterizations using ^{51}V, ^{17}O, ^{1}H, and ^{13}C nuclear magnetic resonance spectroscopy demonstrated the unprecedented stepwise displacement of the oxido oxygens to form the bare V(V)-glutaroimide-dioxime complex. ESI-MS studies of V(V)-glutaroimide-dioxime solutions allowed the identification the intermediate 1:1 M:L complex as well as the bare $V(L)_2$ complex at m/z = 330.8.

Structural insights into the much higher sorption of V(V) to amidoxime-based sorbents relative to U(VI) and Fe(III) were gained by comparing the structural parameters of the V(V)-glutaroimide-dioxime complex with the analogous U(VI)- and Fe(III)-glutaroimide-dioxime complexes. For these complexes, the degree of protonation of the ligand was found to decrease from U(VI) to V(V). In conjunction with the substantially shorter bond lengths observed for the V(V) complex relative to the other complexes, this implies stronger bonding in the V(V) complex and higher thermodynamic stability. In fact, the trend in binding strengths parallels the observed trend in sorption of these cations to poly(amidoxime) sorbents in marine tests.

5.7 Publications and Patents

5.7.1 Publications in Peer-Reviewed Journals

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5.7.2 Publications in Conference Proceedings

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6. USED FUEL DISPOSITION RESEARCH AND DEVELOPMENT CAMPAIGN

Peter Swift, SNL, NTD

6.1 Overview

6.1.1 Introduction and Objectives

The UFD Campaign identifies alternatives and conducts scientific research and technology development to enable storage, transportation, and disposal of UNF and wastes generated by existing and future nuclear fuel cycles. An overarching objective of the campaign is to support the Administration's 2013 *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste*.

Near-Term Objectives (2017–2021)

- Support the DOE-industry high-burnup fuel full-scale storage demonstration project.
- Develop understanding of how temperature and pressure affect cladding integrity in high-burnup UNF through experimentation and predictive modeling.
- Develop understanding of how corrosion and stress corrosion cracking affect performance of stainless steel dry storage canisters through collection of material and environmental data and predictive modeling.
- Characterize external loadings on UNF during normal conditions of transport.
- Field a deep borehole test, with drilling beginning in 2017 and testing complete by 2021.
- Complete evaluation of the direct disposal of dual-purpose canisters.
- Develop the experimental and modeling basis for understanding long-term performance of disposal systems in clay/shale, salt, and crystalline rock.
- Initiate technical work in support of DOE's plans to develop a separate mined geologic repository for high-level radioactive waste from defense programs and some DOE-managed SNF from defense and research activities.

Long-Term Objectives (2021–2026)

- Support the implementation of a full-scale, Nuclear Regulatory Commission (NRC)-licensed storage confirmatory data demonstration project via significant collaboration with industry.
- Develop the technical basis necessary to support eventual transportation of high burnup UNF.
- Develop the technical basis for the Deep Borehole Disposal concept with Deep Borehole Field Test data.
- Support implementation of integrated storage, transportation, and disposal concepts.

6.1.2 Campaign Challenges

A key campaign challenge is to provide a sound technical basis for supporting the DOE strategy for managing the back end of the nuclear fuel cycle, which includes identifying and evaluating safe and secure options for storage, transportation, and permanent disposal of radioactive wastes resulting from existing and future fuel cycles.

6.2 Major Research and Development Focus Areas

Storage and Transportation R&D supports development of the technical bases to inform management and licensing decisions regarding storage and transportation of SNF. Current activities in this area are focused on three topics: storage, transportation, and security. Storage R&D focuses on closing technical gaps related to extended storage of UNF, including uncertainties with high-burnup UNF cladding performance and long-term canister integrity. Transportation R&D focuses on ensuring transportability of UNF following extended storage, addressing data gaps regarding nuclear fuel integrity and retrievability, and understanding stresses and strains on fuel during normal conditions of transport. Security R&D focuses on questions related to the consequences of a potential terrorist attack and how to mitigate attacks in current and future storage facility designs.

The UFD R&D campaign has extensive international collaborations to leverage expertise and research performed in other used fuel communities around the world, including participation in the international Extended Storage Collaboration Project led by the Electric Power Research Institute (EPRI) with input from the DOE, NRC, and programs in multiple other nations. The campaign also has collaborations with Germany, Spain, the Republic of Korea and the United Kingdom as well as the IAEA and Euratom. In addition, there are ongoing domestic collaborations with the Nuclear Energy Institute, nuclear power plant site operators, fuel and storage-system vendors, and the NRC. In order to leverage research ongoing within the U.S. academic community and to continue a pipeline of students and young professionals interested in solving nuclear waste problems, the campaign collaborates through the NEUP with numerous universities, including Penn State, University of South Carolina, and the Colorado School of Mines.

Most R&D in the Storage and Transportation areas is aligned with one of three focus areas in 2016: (1) the EPRI/DOE High Burnup Spent Fuel Data Project, (2) the multi-modal international transportation test undertaken jointly by the DOE and Equipos Nucleares S. A. (ENSA) of Spain, and (3) understanding the potential for the formation of stress corrosion cracks in stainless steel canisters used for dry storage. Each of these focus areas is a continuation of a long-term project.

1. **EPRI/DOE confirmatory high-burnup used fuel data project** is led by EPRI under contract to DOE and is in the third year of a multi-year test to collect data from an instrumented UNF dry storage system containing high-burnup fuel located at the North Anna Nuclear Power Plant in Virginia. The primary goals of the test are to provide confirmatory data on the long-term storage behavior of high-burnup fuel for model validation and potential improvement, provide input to future dry storage cask designs, support NRC license renewals and new licenses for Independent Spent Fuel Storage Installations, and support transportation licensing for high-burnup used fuel. Plans call for loading the test storage cask with representative fuel assemblies in 2017, and for conducting separate effects testing on sister pins in parallel with the storage test. These tests will be conducted at ORNL, PNNL, and ANL. Non-destructive testing will be conducted in 2017 and destructive testing will begin in 2018.
2. **The ENSA/DOE multi-modal normal conditions of transport test** is a collaboration between ENSA and the DOE. The purpose of the test is to quantify the shocks and vibrations experienced by surrogate fuel during heavy-haul truck, ocean-vessel, and rail including transfer between these modes

of transportation. Results will be compared to past tests as well as fuel failure limits to support transportation cask design and licensing, and to help validate existing models.

3. The third area of focus is to better understand the potential for **stress corrosion cracking** on dry storage canisters by providing scientific data and analysis relevant to multiple storage environments around the country. Ongoing work in this area includes collaboration with EPRI, industry participants, the NRC, and programs in other nations.

Disposal R&D focuses on identifying multiple viable geologic disposal options, addressing technical challenges for generic disposal concepts in various host media (e.g., mined repositories in salt, clay/shale, and granitic rocks, and deep borehole disposal in crystalline rock). R&D goals at this stage are to reduce generic sources of uncertainty that may impact the viability of disposal concepts, to increase confidence in the robustness of generic disposal concepts, and to develop the science and engineering tools needed to select, characterize, and ultimately license a repository. International collaborations are a significant component of the disposal R&D portfolio, and include: DECOVALEX (Development of Coupled Models and their Validation against Experiments, with participation from multiple nations in Europe and Asia); the Mont Terri underground research laboratory (Switzerland); Colloid Formation Migration (Switzerland); SKB Task Force (Sweden), Salt R&D (Germany); and crystalline disposal R&D with Korean Atomic Energy Research Institute (KAERI) Underground Research Tunnel. The deep borehole disposal concept has not been demonstrated anywhere in the world, and the campaign has therefore identified support for a DOE-managed field test using nonradioactive surrogate waste as one of its highest priorities for the coming years. Drilling for the deep borehole field test is planned to begin in the fall of 2017.

DOE-managed High-Level Waste and Spent Nuclear Fuel Research Activity. Work in this area was initiated in FY 2015 following the Administration's decision in March 2015 to move forward with the planning for a consent-based defense waste repository (DWR) for DOE-managed HLW and some DOE-managed SNF. Activities planned for FY 2017 will include identification of the inventory of waste potentially suitable for disposal in a separate DWR, development of preliminary DWR design concepts in multiple geologic media, and planning of the organizational and procedural framework necessary to implement a separate repository program within the DOE's existing authority under the Atomic Energy Act of 1954, licensed by the NRC and consistent with the requirements of the Nuclear Waste Policy Act of 1982, as amended.

6.3 Storage and Transportation Accomplishments

Storage and Transportation R&D occurs in five work areas: field demonstration, experiments, analysis, transportation, and security. Selected work is highlighted in the following subsections and in Figures 22–24.

6.3.1 Field Demonstration

The High Burnup Confirmatory Data Project continues to progress according to plan. Accomplishments in 2016 resulted in inspecting and preparing the cask for the storage of the high burnup fuel, extracting the 25 sister rods, shipping them to ORNL, and starting non-destructive analysis on those rods. To date, approximately 40% of the gamma scans of the sister rods have been completed. Additional non-destructive analysis will be completed in FY 2017.

Accomplishments for the demonstration cask include good progress on obtaining the storage license from the NRC. Requests for Information from NRC staff are being addressed with positive feedback from both Dominion and the NRC. In FY 2017, the cask will be loaded with the 32 assemblies presently in the North Anna Nuclear Power Plant storage pool and moved to the North Anna Independent Spent Fuel Storage Installation. Data will be collected on the canister internal gas composition to understand the effectiveness of the drying process and the integrity of the fuel cladding. Ongoing temperature measurements at 72 locations within the cask will be collected to understand peak cladding temperatures and temperature decay rates within the cask. This information will be important to understand the potential for radial hydride formation in the cladding, which is a potential mechanism for cladding failure.

6.3.2 Experiments

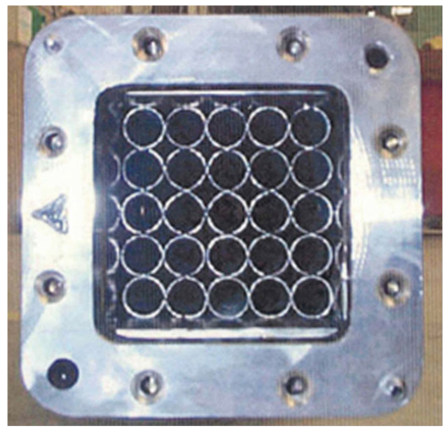

Figure 21. End-view of the Nuclear Assurance Corporation legal weight truck basket used for shipping irradiated fuel rods from the North Anna Nuclear Power Plant to ORNL.

Storage Canister Stress Corrosion Cracking: The major FY 2016 accomplishment in the area of stress corrosion cracking was quantification by direct measurement of a full-diameter mockup of through-wall tensile stresses at canister welds, about 1 inch on each side of the welds and at weld repair areas. The remainder of the cask does not experience through-wall tensile stress and is therefore at a much lower risk of experiencing through-wall cracks. In addition, both Savannah River National Laboratory and the Southwest Research Institute (work at the Southwest Research Institute is not funded by DOE) did not find chloride-induced stress corrosion cracks in bounding experimental conditions. These results indicate that the risk of storage canister through-wall cracks is still a possibility, but less of a risk than thought one year ago. FY 2017 work will focus on better understanding the environmental differences in canister dust deposits within the country to better identify areas of higher and lower corrosion risk.

Figure 22. Full-diameter mockup of a representative used fuel storage canister, prior to sectioning for residual stress measurement.

6.3.3 Transportation

Multi-Modal Transportation Test: FY 2016 has been a year of preparing for the multi-country, multi-mode transportation test. The purpose of this test is to quantify the shocks and vibrations seen by the fuel during normal conditions of transportation. This test will involve heavy-haul truck transport from a nuclear power plant in central Spain to the northern coast of Spain, transport by barge to Belgium, transport by ocean liner to Baltimore, and then to Colorado by train. Once in Colorado, the transportation system will undergo extensive, controlled normal conditions of transport testing. The transportation system will then return to Spain via the same route. Data will be collected during all out-bound legs as well as the transfer between modes (e.g., transfer from boat to rail). Accomplishments during FY 2016 included developing the detailed test plan between DOE and ENSA in Spain, obtaining the transportation cask system, determining the placement of accelerometers and strain gages on the surrogate assemblies through modeling at PNNL, obtaining and testing the sampling equipment and batteries to ensure they will collect data during the duration of the trip. This is a one-time test, so it is vital that all the equipment is thoroughly tested before the test begins.

Figure 23. Instrumented surrogate fuel assembly on seismic shaker table.

6.3.4 Analysis

FY 2016 accomplishments in the analysis control account area were in the area of developing a model for storage canister stress corrosion cracking and thermal analysis of storage canister.

Thermal analysis of three loaded canisters was completed and the results indicated that actual fuel and canister temperatures are significantly lower than thought a year ago. Analysis focused on reducing the conservatisms within the input data and the codes. Results indicated that the demonstration cask peak clad temperatures should not exceed 271°C, compared to the NRC regulatory limit of 400°C. This has important implications for potential radial hydride formation in cladding. Radial hydrides are not expected to form at cladding those cladding temperatures. On the other hand, stress corrosion cracking may be more of a concern on cooler canister surfaces. In FY 2017, the program desires to obtain data from additional in-service sites to further verify this thermal data.

Additional analysis accomplishments were in the focus area of stress corrosion cracking where a model is being developed to combine temperature, humidity, salt, and other data to determine the areas of the country where storage canister stress corrosion cracking is of high or lower risk and to focus future R&D efforts.

6.3.5 Security

Security accomplishments helped refine and alter the design of consolidated ISF design to mitigate potential security risks. Work also analytically quantified potential radiation releases from worst-case terrorism events. This work is classified and is not discussed further here.

6.4 Disposal Research Accomplishments

Disposal R&D has been performed in multiple areas, including analysis of generic mined repository concepts in salt, crystalline rock, and clay/shale rock, as well as deep borehole disposal. Related R&D had examined the performance of engineered barriers, including the waste form, in various geologic environments, and the feasibility of direct geologic disposal of existing dual-purpose (storage and transportation) canisters currently used for dry storage of SNF. Selected work is highlighted here.

6.4.1 Generic Mined Repository R&D

In FY 2016, the Generic Disposal System Analysis group continued work developing a system modeling capability for probabilistic evaluations of nuclear waste disposal options in deep geologic media. Model development focused largely on the source term and flow and transport in a fractured host rock. Advances included a new canister degradation model, a fully-integrated waste form process model, an expanded selection of waste form degradation models, an improved calculation of decay and ingrowth, a new isotope partitioning model, and a new discrete fracture network simulation capability. Integration with other UFD R&D work continued at a strong pace, especially in the area of fracture flow and transport. The new source term and discrete fracture network capabilities were demonstrated by developing a multi-million cell, three-dimensional reference case for a mined repository in crystalline rock. Probabilistic thermal-hydrologic simulations indicated that radionuclide concentrations at monitored locations in the model domain are particularly sensitive to waste package degradation rates, waste form dissolution rates, sorption coefficients, and fracture distribution. Importantly, the results showed that a relatively large fracture density is necessary for significant flow and transport in a crystalline formation.

6.4.2 Deep Borehole Field Test

In FY 2016, the Deep Borehole Field Test activities focused on revising the field test project plan and conducting research to further develop the technical basis for a successful field test.

6.4.3 International Collaborations in Disposal Research

Activities in FY 2016 included collaborative work investigating field tests from underground research laboratories in Switzerland (Mont Terri and Grimsel Test Site), Sweden (Äspö), Japan (Horonobe and Mizunami), the Czech Republic (Bedrichov Tunnel), France (Bure), and the Republic of Korea's KAERI Underground Research Tunnel. A highlight of FY 2016 was DOE's participation in the dismantling and analyses of the Full-scale Engineered Barriers EXperiment (FEBEX) experiment at Grimsel Test Site, an *in situ* full-scale heater test conducted in a crystalline host rock that was in operation for 18 years. After dismantling the test site, a detailed post-mortem analysis was conducted to evaluate the integrity of both engineered and natural barrier components (see Figure 24). The project provided a unique opportunity for better understanding the performance of barrier components that underwent continuous heating and natural resaturation for a significant time period. FY 2016 work also included the first full year of monitoring and modeling of the multi-nation full-scale emplacement heater test in clay/shale host rock at the Mont Terri URL, where heaters were turned on February 15, 2015. This multi-year test continues to add to our understanding of the thermal-hydrological-mechanical response of near-field host rock and engineered barrier components, demonstrate emplacement technologies for bentonite buffer materials, and provide data to help validate coupled thermal-hydrological-mechanical models used in repository analysis.

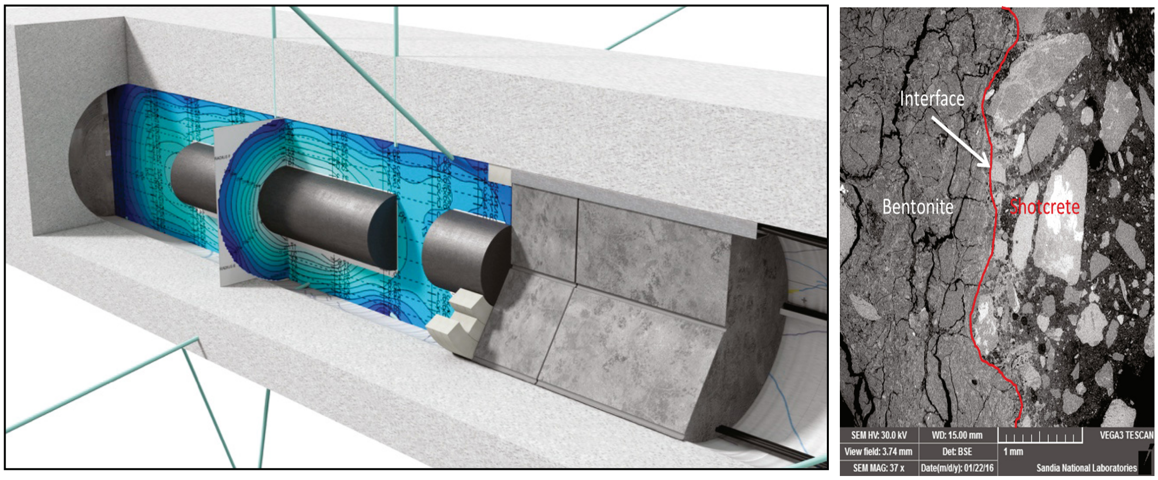

Figure 24. Buffer saturation measured from post-mortem analysis of FEBEX text at Grimsel Test Site (left). Microscopic image of sample at the interface between bentonite backfill and host rock (right).

6.5 DOE-Managed High-Level Waste and Used Nuclear Fuel Research Accomplishments

Research in the DOE-managed HLW and UNF focus area was initiated in FY 2016 for wastes that are potentially eligible for disposal in a separate repository developed under the DOE's existing authority under the Atomic Energy Act of 1954. R&D activities associated with the development of a DWR were conducted in the following areas:

1. **Inventory and Waste Characterization** to address the technical elements necessary to delineate the inventories of waste forms for disposal and their expected behavior in various disposal concepts.
2. **Preliminary Repository Design** to address the technical elements necessary to evaluate the preliminary design concepts for the inventory within select media.
3. **Safety Analysis/Performance Assessment** to address the technical elements necessary to establish the safety case associated with select repository sites.

Selected work from these areas is highlighted here.

6.5.1 Inventory and Waste Characterization

FY 2016 DWR activities included: (1) developing a preliminary inventory for engineering/design/safety analyses; (2) assessing the major differences of this inventory that used in other repository systems and the potential conceptual impacts to various disposal systems; (3) designing and developing an on-line waste library to manage the inventory information of relevant wastes and waste forms; and (4) constraining post-closure waste form degradation performance for safety assessments. The preliminary inventory for the analyses of a DWR for FY 2016 and includes both HLW and DOE-managed SNF waste canister counts and thermal information. A prototype on-line waste library database (with user's guide) was developed with data for the Cs/Sr capsule wastes from the Hanford Site. Degradation rate models for both UO_2 and HLW glass were constrained for both granite and salt repository concepts, and are being used within the current safety assessments. Each DWR waste form was mapped into those performance models based on its expected degradation behavior. For waste forms expected to have short waste form lifetimes, an instantaneous degradation rate is used. Note that in all cases the waste form degradation is the initial, kinetic step, and the dissolved radionuclides are evaluated against solubility limits based in part on the geologic environment.

6.5.2 Preliminary Repository Design

In FY 2016, the *Preliminary Design Concepts Work Package* focused on design concepts and thermal analysis for crystalline and salt host media. FY 2016 work concludes that thermal management of defense waste, including the relatively small subset of high thermal output waste packages, is readily achievable. To ensure engineering feasibility, the design concepts are based upon established and existing elements and/or designs. The multipack configuration options for the crystalline host media, pose the greatest engineering challenges as these design involve large, heavy waste packages that pose specific challenges with respect to handling and emplacement. Some DOE-managed SNF presents issues for post-closure criticality control, and a key recommendation made herein relates to the need for special packaging design that includes neutron absorbing material. Overall, FY 2016 work finds that the preliminary design options for defense waste derived from other published and well-studied repository designs are potentially suitable for both operational and post-closure safety (Figure 25).

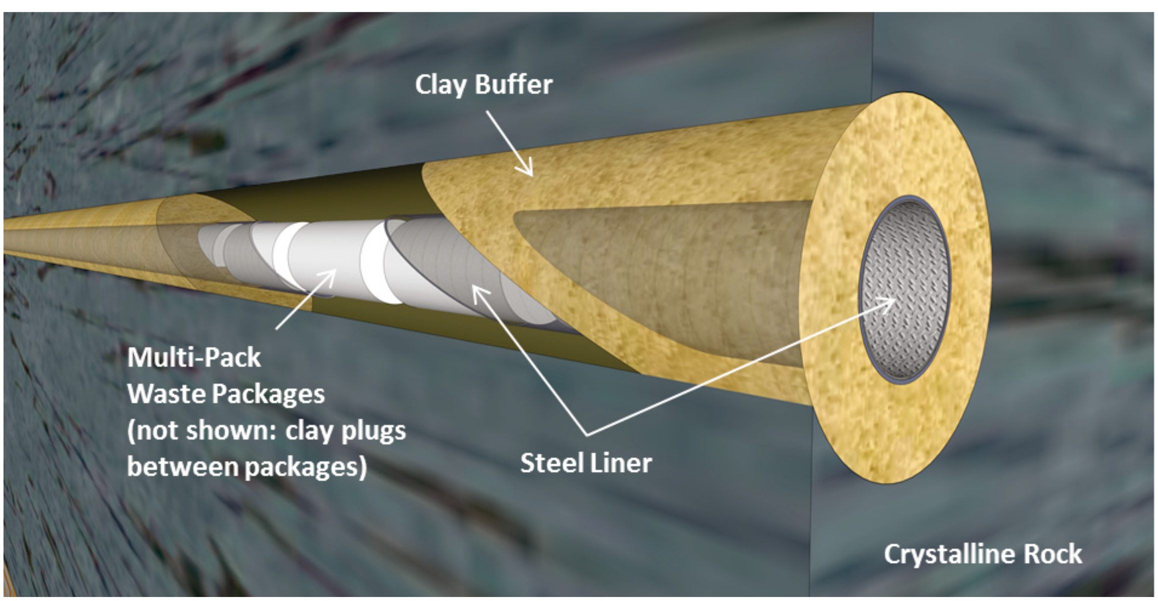

Figure 25. Schematic of defense waste repository multipack emplacement design for crystalline host media.

6.5.3 DWR Safety Analysis/Performance Assessment

One of the main components of a comprehensive DWR Safety Case will be a post-closure safety analysis or performance assessment. R&D progress in this area during FY 2016 addressed four major tasks:

- Development of generic reference cases (i.e., knowledge or technical bases for “generic” or “non-site-specific” deep geologic repositories) for two primary host rocks under consideration for a DWR: crystalline (granite) and bedded salt.
- Features, events, and processes analyses/screening to support the technical bases and performance assessment.
- Performance evaluation of alternative engineered concepts for the layout of a repository and the design of an engineered barrier system, corresponding to the given host rock.
- Post-closure safety assessment of the repository system under consideration.

Development of an enhanced performance assessment capability for geologic disposal of SNF and HLW has been ongoing for several years in the U.S. repository program. This enhanced performance assessment capability, called the Generic Disposal System Analysis modeling and software framework, has now been applied to a generic DWR in two potential host media: bedded salt and crystalline rock. Two types of emplacement concepts were examined, including single-canister vertical-borehole emplacement for the hotter DOE-managed SNF waste and multi-canister horizontal emplacement for HLW. Additional R&D related to safety assessment included regional geologic evaluation focused on the geologic and hydrologic environment.

7. JOINT FUEL CYCLE STUDIES

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

7.1 Overview

JFCS is chartered to investigate issues important to determination of the technical and economic feasibility and nonproliferation acceptability of electrochemical recycling for the management of UNF. Electrochemical recycling utilizes dry (non-aqueous) 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 post-irradiation examination. This overview provides highlights of the accomplishments of the JFCS Campaign through FY 2016.

Key FY 2016 outcomes:

- All six pieces of planned process equipment have been successfully installed into the Hot Fuel Examination Facility (HFEF), including nine instrumentation and power feedthroughs, work tables and balances.
- Initial operations will be performed with irradiated material from the INL Fast Flux Test Facility reactor. Irradiated fuel elements which will be processed have been selected and are staged inside HFEF.
- The oxide reduction system and the distillation systems have successfully completed surrogate testing with depleted U to qualify the equipment and fine tune process parameters. The remote decladding system has successfully demonstrated decladding of irradiated Fast Flux Test Facility material.
- A study on the fundamental properties of actinoid species in molten chloride using alternating current techniques was completed.

7.2 Laboratory-Scale Feasibility Study

The purpose of the LSFS 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 in a DOE facility. Approximately 100 grams of heavy metal were processed 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 inform equipment and process development for kilogram scale studies.

7.3 Integrated Recycling Test

An integrated testing activity at kilogram scale is planned 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. IRT includes the fabrication, irradiation, and post-irradiation examination of metal fuel rodlets produced from recycled LWR 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) is focusing on the completion, testing, installation, and operation of equipment to recover re-usable materials from used LWR fuel. The objective by the end of Phase IIB is the casting of fuel slugs including recycled material for irradiation in the Advanced Test Reactor (ATR). The following paragraphs provide additional description of primary equipment components which will be operated during the 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 26 depicts the workstation and equipment layout for the IRT in HFEF.

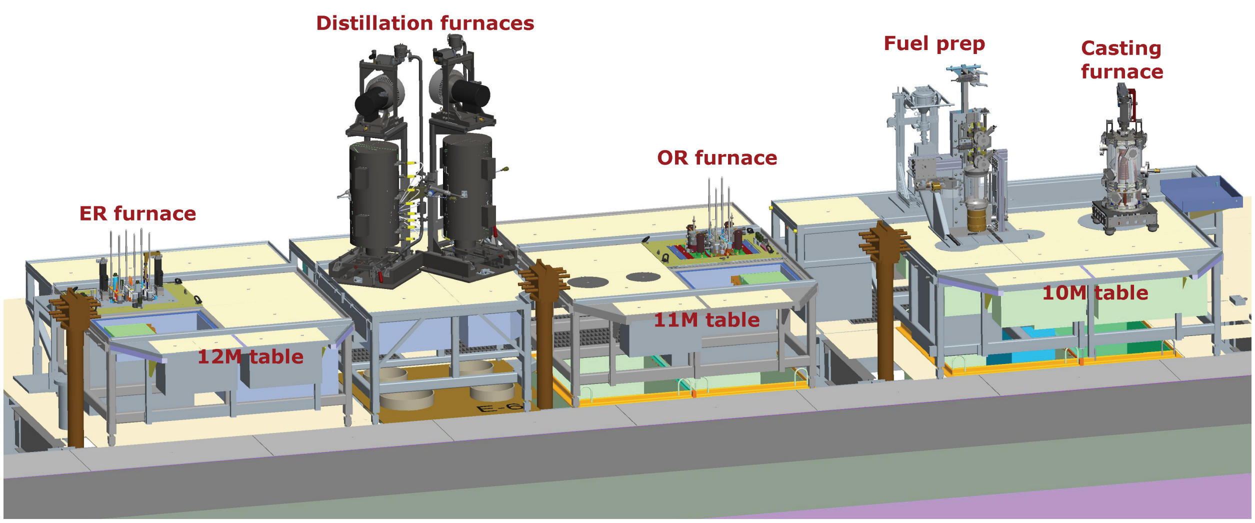

Figure 26. Workstation and equipment layout for the JFCS IRT.

Head-End Equipment: The equipment necessary to prepare UNF for IRT processing is described as head-end equipment. This includes equipment for decladding, sieving, handling of fines, and material storage. The equipment was installed and tested with irradiated Fast Flux Test Facility fuel elements in late FY 2016. Figure 27 shows the vibratory decladding system in checkout testing in HFEF.

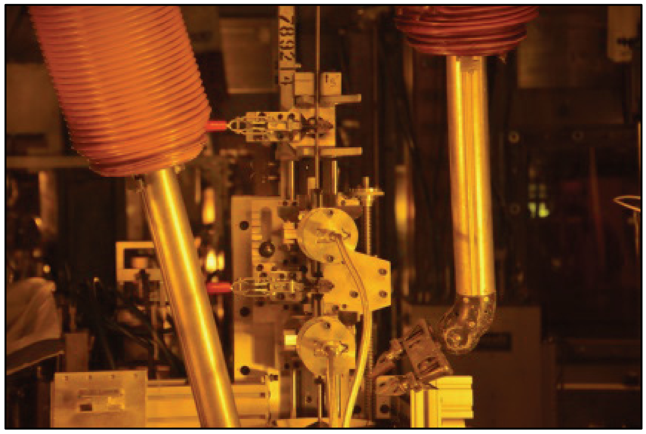

Figure 27. Vibratory decladding system during testing with irradiated fuel in HFEF.

Oxide-Reduction System: The oxide reduction system electrolytically reduces oxide fuels to produce a metallic product that is suitable for further electrochemical recycling. The oxide reduction system was installed and tested with depleted U in FY 2016. The system layout on the HFEF 11M table is shown in Figure 28.

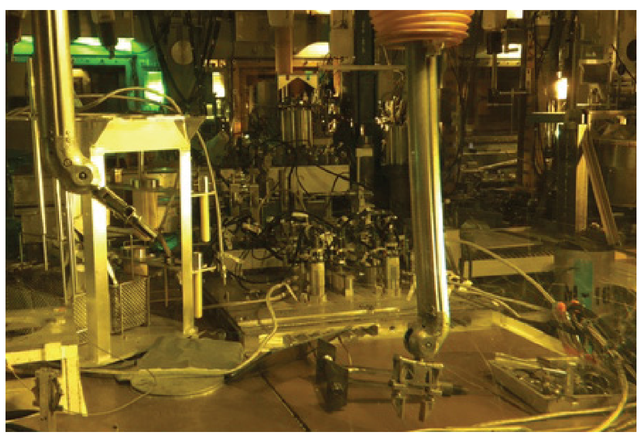

Figure 28. The oxide reduction system at HFEF window 11M.

Electrorefiner System: The purpose of the electrorefiner is to separate TRU elements and fission products from the reduced fuel and accumulate those elements and products in the ER salt. Purified U is collected, and periodically group recovery is performed for U/TRU elements through a liquid cadmium cathode (LCC). In FY 2016, the electrorefiner was installed into HFEF and staged for initial experiments.

Distillation System: The distillation module is used to distill 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 operations. Significant remote testing with the distillation systems has been completed, including experiments to distill salt from depleted U products.

Fuel Fabrication Equipment: Recovered actinide materials from the JFCS will be used to fabricate fuel rodlets for irradiation in ATR. A casting furnace was installed into HFEF and remote testing initiated in FY 2016.

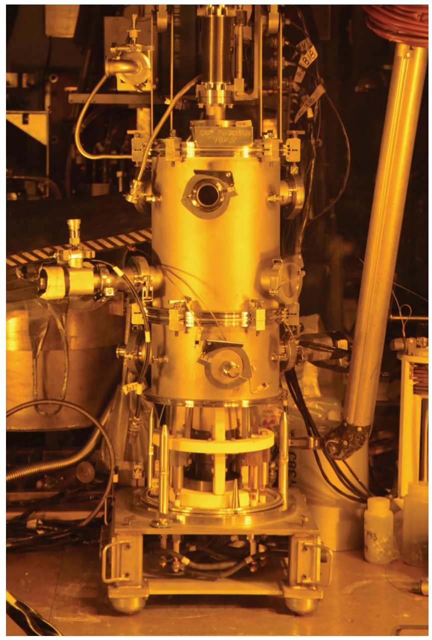

Figure 29. The remote casting system at HFEF window 10M during remote qualification.

7.4 Critical Gap Research and Development

A range of research and development was performed in FY 2016 to provide fundamental knowledge necessary to prepare for 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 that 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.

Oxide Reduction: The IRT oxide reduction system components were upgraded to better address corrosion issues 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 2016 to examine their performance. One particular issue of concern is behavior of these materials in the presence of tellurium, selenium, and iodine, fission products with potentially deleterious effects.

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.

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. 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 UO_2 formed into dense crucibles or coatings. An experiment with UO_2 coatings is shown in Figure 30.

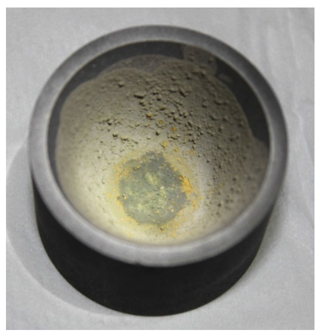

Figure 30. UO_2-coated crucible following experiments with molten U.

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. These will be demonstrated with fission products during Phase III of the IRT. Activities in FY 2016 continued the investigation and down-selection of waste forms and waste form process equipment.

Fuel Fabrication: Experiments to explore casting parameters with the new remote casting system were performed in FY 2016. Experiments continued toward cladding coatings and liners to mitigate fuel-cladding chemical interaction which may occur due to interactions with lanthanoid fission products. Experiments were also performed to evaluate the possibility to obtain representative solid samples of U/TRU products.

8. NUCLEAR FUELS STORAGE AND TRANSPORTATION PROGRAM

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8.1 Overview

In January of 2012, the Blue Ribbon Commission (BRC) issued a report to the Secretary of Energy that included a number of recommendations. In January of 2013, the Administration released its *Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste* (*Strategy*), which serves as a statement of Administration policy regarding the importance of addressing the disposition of SNF and high-level radioactive waste (HLW); it lays out the overall design of a system to address the issue; it outlines the reforms needed to implement such a system; and it presents the Administration's response to the BRC recommendations.

In conjunction with the development of the Administration's *Strategy*, DOE established the Nuclear Fuels Storage and Transportation (NFST) Planning Project on October 1, 2012. The mission of the NFST is to lay the groundwork for implementing interim storage, including associated transportation, per the Administration's *Strategy*, and to develop a foundation for a new nuclear waste management organization. The purpose of the NFST is to make progress on this important national issue, within existing legislative and budgetary authorizations, while the Administration and Congress work together on legislative changes. An over-arching goal is to develop options for decision-makers on the design of an integrated waste management system.

The objective of the NFST project is to identify and begin implementation of activities to (1) plan for implementing interim storage; (2) improve the overall integration of storage as a planned part of the waste management system; (3) prepare for the large-scale transportation of UNF and HLW, with an initial focus on removing UNF from the shutdown reactor sites; and (4) develop foundational information, resources, and capabilities needed to support the aforementioned objectives and future implementation decisions and actions.

8.2 Near-Term Objectives

Near-term NFST Program objectives include:

- Developing and maintaining an integrated plan to accomplish the strategy goals.
- Improving integration of storage as a planned part of the waste management system, including evaluating standardization of dry cask storage systems.
- Developing and evaluating design options for an integrated waste management system.
- Developing and applying systems analyses to provide quantitative estimates of system impacts of utility actions and inform future decisions.
- Preparing for large-scale transportation of UNF and HLW with an initial focus on removing UNF from the shutdown reactor sites.
- Establishing and maintaining a unified and integrated UNF database and analysis system to characterize the input to the waste management system.

8.3 Long-Term Objectives[2]

Long-term NFST Program objectives include:

- Siting, designing and licensing, constructing, and starting operations of a pilot ISF with an initial focus on accepting UNF from shutdown reactor sites.
- Developing transportation infrastructure and capabilities to facilitate the acceptance of UNF at a pilot ISF.
- Siting and licensing a larger ISF with sufficient capacity to provide flexibility in the waste management system and allowing for acceptance of enough UNF to reduce expected government liabilities.

NFST activities are aligned with the key principles of the BRC recommendations and provide a foundation for a new nuclear waste management organization. NFST activities are divided into four major areas: (1) consent-based siting of an ISF, (2) storage, (3) transportation, and (4) strategic crosscuts.

8.4 Consent-Based Siting Accomplishments

NFST is laying the groundwork for a consent-based siting process for nuclear waste management facilities that reflects input received from interested parties at the local, state, and tribal levels.

- To launch the consent-based siting effort, on December 23, 2015, DOE issued an Invitation for Public Comment in the Federal Register soliciting input on important considerations in designing a fair and effective process for siting. The comment period ended on July 31, 2016.
- DOE hosted a "kick-off meeting" for the consent-based siting effort in Washington D.C. on January 20, 2016.
- DOE held a series of eight public meetings around the country to hear from the public, communities, states, Tribal Nations, and all interested stakeholders on what matters to them as DOE moves forward in developing a consent-based process. These meetings included presentations from 40 diverse panelists and participation of about 600 people via webinar and in person. The meetings were held in:
  - Chicago, Illinois on March 29th.
  - Atlanta, Georgia on April 11th.
  - Sacramento, California on April 26th.
  - Denver, Colorado on May 24th.
  - Boston, Massachusetts on June 2nd.
  - Tempe, Arizona on June 23rd.
  - Boise, Idaho on July 14th.
  - Minneapolis, Minnesota on July 21st.

[2] Per the Administration's *Strategy*, legislation is required to enable full implementation of the longer-term objectives.

- DOE hosted a meeting in Washington D.C. on September 15, 2016 to summarize the input received in the initial phase of public engagement on consent-based siting and discuss next steps in designing a process.
- Comments received through the Invitation for Public Comment and eight public meetings were summarized in the draft report *Designing a Consent-Based Siting Process: Summary of Public Input*, that was released on September 15, 2016.
- NFST performed an analysis of a nationwide survey of public preferences conducted annually by the Center for Energy, Security & Society (CES&S), a joint research collaboration of the University of Oklahoma and Sandia National Laboratories. The 2016 survey focused on public preferences and support for different spent fuel management options, including continued on-site storage, interim storage, deep geologic repositories, and an integrated systems approach. Additionally, the survey measured public preferences for a repository for only defense-related waste versus a repository that would co-mingle both defense and commercial wastes.

8.5 Storage Accomplishments

NFST has initiated a number of activities to help lay the groundwork for implementing interim storage per the Administration's *Strategy for the Management and Disposal of Used Nuclear Fuel and High Level Radioactive Waste.* The *Strategy* calls for an initial pilot ISF with an initial focus on accepting UNF from shutdown reactor sites. The *Strategy* also describes a larger ISF with sufficient capacity to provide flexibility in the waste management system and allow for acceptance of enough UNF to reduce the expected government liabilities. Recognizing that the consent-based siting process would ultimately define the facility and its capabilities to be deployed at one or more sites, there are multiple alternatives for deploying interim storage capacity to meet the goals in the Administration's *Strategy*.

8.5.1 Storage Design

Storage design activities supported the future design and licensing of one or more ISFs. Consistent with the BRC recommendations and the *Strategy*, it is envisioned that a future ISF would be deployed in phases, utilizing modular design concepts for expanded functional capabilities, capacity, and throughput as appropriate. NFST is developing a range of design information for use in evaluating ISF deployment as part of an integrated waste management system:

- Made significant progress on the issuance of a contract for the development of a generic pilot ISF design and Topical Safety Analysis Report. It is anticipated that the contract will be awarded in early FY 2017.
- Developed cost estimates for a multi-pack canister carrier disposal concept utilizing the grouping of 4-PWR size canisters for convenience of storage, transportation, and possibly disposal (should the disposal concept permit larger packages).
- Completed pre-conceptual design for a facility that provides for receipt and packaging of 1,500 metric tons of fuel stored in dry casks at reactor sites or an ISF, as well as individual fuel assemblies shipped directly from reactor fuel pools (Figure 31).
- Participated on ASME code case development. As prompted by NRC in a request to the chair of Section XI of the ASME BPV code, rules for In-Service-Inspection of the canisters/casks are

being developed via a code case. NFST staff are leading development of the code case section 3000 (Acceptance Criteria).

- Issued a document that establishes an initial set of functions and requirements for storage and transportation portions of the waste management system, provides bases for planning future activities (e.g., alternative analyses), and identifies interfaces between the Storage and Transportation Systems (http://www.energy.gov/ne/downloads/nuclear-fuels-storage-and-transportation-requirements-document).

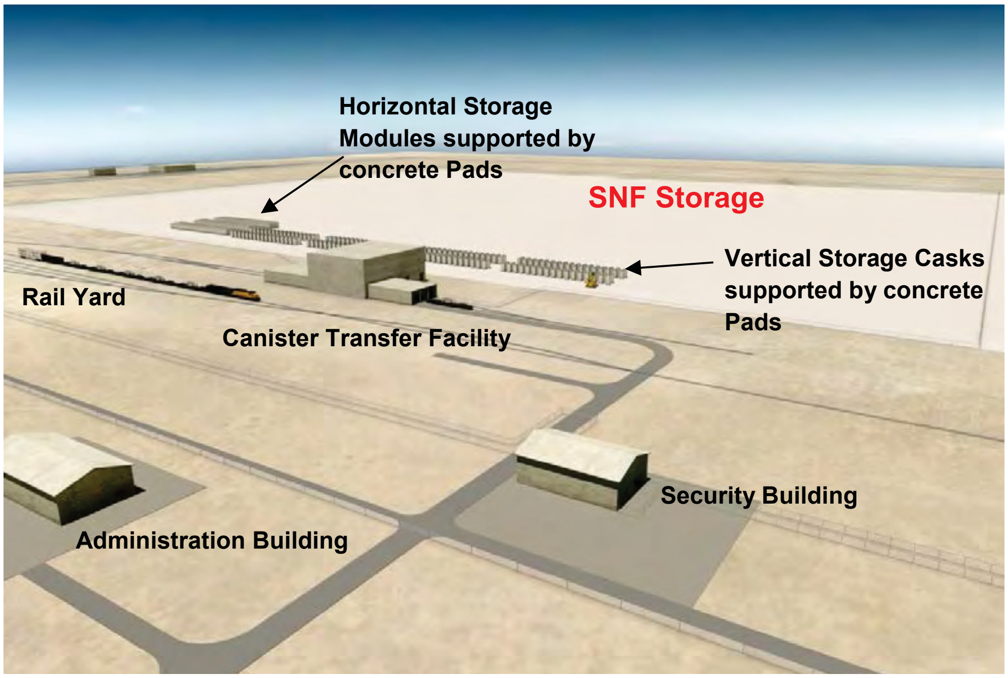

Figure 31. Artist concept of a pilot interim storage facility.

8.5.2 Regulatory

Regulatory activities supported the future demonstration of compliance with all regulations, in particular Title 10 of the *U.S. Code of Federal Regulations,* “Energy,” Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste and Reactor-Related Greater than Class C Waste” (10 CFR 72), and Title 10 of the *U.S. Code of Federal Regulations,* “Energy,” Part 71, “Packaging and Transportation of Radioactive Material,” (10 CFR 71).

- Investigated potential strategies for licensing a pilot ISF and later, a full-scale ISF under 10 CFR 72.
- Monitored the licensing activities of the privately-funded potential ISF sites at Andrews, Texas and Eddy-Lea counties in New Mexico.

8.5.3 National Environmental Policy Act

National Environmental Policy Act (NEPA) activities supported future efforts to meet the Council on Environmental Quality (CEQ) *Regulations for Implementing the Procedural Provisions of the National Environmental Policy Act* (40 CFR Parts 1500-1508) and, possibly, the DOE *National Environmental Policy Act Implementing Procedures* (10 CFR Part 1021).

- Investigated potential strategies for meeting NEPA requirements.
- Compiled information and identified data gaps relevant to environmental considerations for a future proposal for the transport and consolidated storage of SNF and related radioactive wastes.

8.6 Transportation Accomplishments

Transportation activities supported preparing for the eventual large-scale transportation of UNF and HLW to facilitate the acceptance of SNF at a pilot ISF, with an initial focus on accepting SNF from shutdown reactor sites. The primary focus is currently on making progress on long lead-time, destination-independent aspects of the transportation infrastructure.

8.6.1 Institutional

Institutional activities furthered established relationships with federal, state, and tribal entities to develop policies and agreements on transportation system operations and responsibilities.

- Nuclear Waste Policy Act Section 180(c) requires that DOE provide technical assistance and funds to States and Native American Tribes to train public safety officials of the appropriate units of local government through whose jurisdictions SNF would be transported. DOE issued a revised proposed policy for implementing Section 180(c) of the Nuclear Waste Policy Act for public comment on October 31, 2008 (73 FR 64933). NFST continued to collaborate with stakeholders on DOE's policy for implementing Section 180(c) through the development and performance of a Section 180(c) Policy Implementation Exercise to evaluate the efficacy of the proposed policy put forth in 2008. The exercise was completed and what the participants learned were captured to 1) foster additional discussion regarding issues that were identified, and 2) serve as a basis for recommendations on future Section 180(c) policy development and program implementation decisions.
- Continued engagement with states, Tribes, and rail carriers in the development of a standardized methodology for the selection of routes meant to ship SNF and HLW.

8.6.2 Operational

Operational activities supported the identification and initial development of the necessary operational functions needed to prepare for the eventual large-scale transport of UNF, greater than class c low-level radioactive waste, and HLW from origin sites.

- Completed initial site-specific SNF de-inventory studies for the shutdown Humbolt Bay, Maine Yankee, and Trojan nuclear plant sites.
- Further revised the Preliminary Evaluation of Removing Used Nuclear Fuel from Shutdown Sites (The latest version is available at http://energy.gov/ne/downloads/preliminary-evaluation-removing-used-nuclear-fuel-shutdown-sites).
- Released Version 2.0 of the Stakeholder Tool for Assessing Radioactive Transportation (START) and a user manual. START is a transportation decision-support tool enabling users to represent and evaluate a wide range of transportation routing scenarios to assist with stakeholder communications and information sharing.

- Completed a review of the safety record of SNF shipments that showed that the transportation of SNF is an activity that has been accomplished routinely and safely in many countries around the world, including the U.S., for decades. A review of publicly available information on SNF transportation worldwide indicates that since 1962, likely more than 44,400 cask shipments have been made, and likely more than 109,000 metric tons of heavy metal have been shipped, without injury or loss of life due to the radioactive nature of the material transported. An accident involving a SNF cask that occurred in Tennessee in 1972, believed to be the most severe transportation accident involving SNF ever documented, was examined. Documents published by the NRC assessing the risks in transporting SNF were also reviewed. A review of the four NRC studies conducted between 1977 and 2014 showed that each has concluded that the radiological risks of transporting SNF are low compared to the risks inherent in truck and rail transportation, and that regulations on SNF casks are adequate in protecting the health and safety of the public in the event of a transportation accident.
- Developed transportation operations process flowcharts that illustrate what is necessary for implementation of a functional transportation system for SNF and HLW. The flowcharts are a starting point to describe the concept of operations for the transportation system within the integrated waste management system.

8.6.3 Hardware

Hardware activities supported the future acquisition of transportation casks and ancillary equipment for truck and rail shipments, specialty rail cars, intermodal transfer equipment, and monitoring and maintenance equipment.

- A contract for Cask and Buffer Railcar Prototype Development was awarded to a team led by AREVA. The contract included design, analysis, and fabrication of the cask and buffer railcars for testing to meet Association of American Railroads Standard S-2043, "Performance Specification for Trains Used to Carry High-Level Radioactive Material." The end result of this effort will be development of the final cask and buffer railcar designs, including associated analysis, and fabricated railcar prototypes ready for testing. The new cask railcars have been named *ATLAS.*(Figure 32)
- Phase 1 conceptual designs were completed, including preparation of planning documents for project management, requirements, and engineering, as well as development of cask loading procedures and conceptual design of cask cradles and preparation of conceptual designs for the cask and buffer railcars. Phase 2 was initiated by performing computer modeling for the preliminary cask and buffer railcar designs. During the design analysis and modeling of the eight-axle railcar conceptual design, it was determined that hunting was occurring and that the damping in the trucks was not sufficient to damp out this vibration. To address this issue, the railcar designer made the decision to change to a twelve-axle railcar design.
- Developed a framework for future analysis of transportation system hardware operations alternatives and establishes a process for a future, definitive analysis of alternatives and resulting procurement decisions.
- Identified the operational and maintenance activities expected for the transportation casks and specialized railcars necessary for the shipment of SNF in dry storage at the shutdown sites to an ISF.

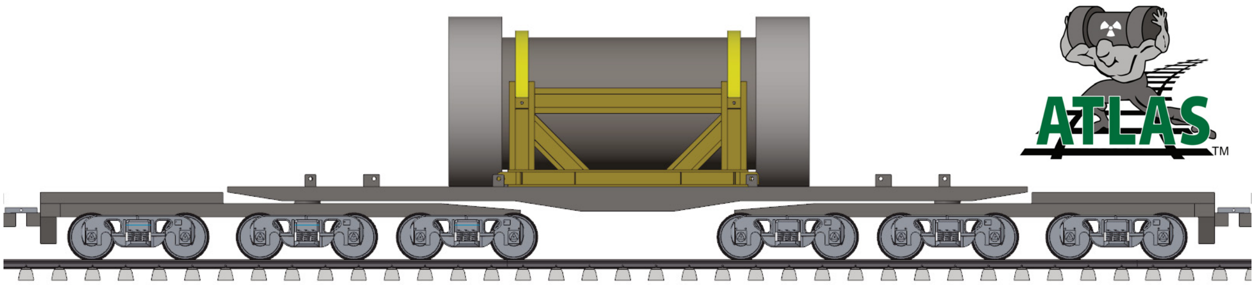

Figure 32. ATLAS railcar.

8.7 Strategic Crosscut Accomplishments

The strategic crosscutting activities support the consent-based siting, storage, and transportation objectives discussed above.

8.7.1 Project Management

NFST project management activities provided program direction, coordination, management, and integration to ensure work was planned and performed consistently with programmatic guidance and requirements, was of high quality, was appropriately coordinated with other related DOE activities, and was informed by relevant activities in other agencies (e.g., the NRC).

8.7.2 Waste Management System Architecture Analysis

Integrated waste management system architecture analyses were conducted to support the future deployment of a comprehensive system for managing UNF from shutdown and operating reactors, greater than class c low-level radioactive waste generated during the decommissioning of nuclear power plants, and DOE-managed UNF and HLW.

- Summarized the advantages and disadvantages of including an ISF as part of an integrated waste management system, based on a review of numerous analyses of the impacts of an ISF (previously discussed as monitored retrievable storage) by the DOE and independent groups.
- Continued to analyzed and evaluate alternative waste management system architectures, guided by and augmenting analyses that were previously completed. FY 2016 activities focused on identifying the extent to which commercial nuclear reactor sites would be affected by potential waste management system architecture configurations and operational concepts.

8.7.3 Next Generation System Analysis Model

NFST continued developing the Next Generation Systems Analysis Model (NGSAM) that will be more readily sustainable and maintainable in the future as compared to legacy integrated waste management system analysis tools and be flexible for use by a broader set of users.

- Initiated benchmarking of the Next Generation System Analysis NGSAM to assess its current set of capabilities as compared to legacy software currently being used. The FY 2016 benchmarking activities focused on evaluating the algorithms and logic for SNF management operations at the fleet of U.S. commercial reactors and strategies for clearing SNF from those reactor sites. Modified NGSAM to implement feedback from the benchmarking effort.

- Continued the phased development of NGSAM by including algorithms and logic to model multiple ISFs having the capability to received SNF and place it into a dry storage configuration (vertical and/or horizontal configurations).

8.7.4 Execution Strategy Analysis

The Execution Strategy Analysis capability continued to be enhanced to provide an approach and a tool for ongoing performance assessment of the evolving project execution plan that takes into account significant assumptions, risks, and uncertainties throughout the project life cycle.

- Enhanced the Execution Strategy Analysis model to simulate alternative strategies for deploying multiple ISFs that could have different functional capabilities and could be deployed either by the Federal government, or as private initiatives, or in combination.

8.7.5 Multi-Objective Evaluation Framework

The NFST continued establishing a multi-objective evaluation framework for identifying and evaluating alternatives for an integrated nuclear waste management system. Multi-objective evaluation framework uses accepted decision analysis techniques to identify the information needed to be obtained during planning efforts to support future decisions and to evaluate information currently being developed and collected by other NFST activities.

- Completed a multi-objective evaluation of several potential waste management system architectures, with a focus on understanding the trade-offs associated with alternative allocation and acceptance strategies.

8.7.6 Data and Document Access

The NFST continued the development and maintenance of a collaborative UNF document and data access system, the Centralized Used Fuel Resource for Information Exchange (CURIE; curie.ornl.gov). CURIE provides ready access and use of UNF data, reports, and tools to support NFST activities.

- Implemented a capability where CURIE users can access an interactive SNF map using the “Map” link on the homepage (Figure 33). The interactive map provides geographic and time-dependent information on the inventory of commercial SNF in the U.S.

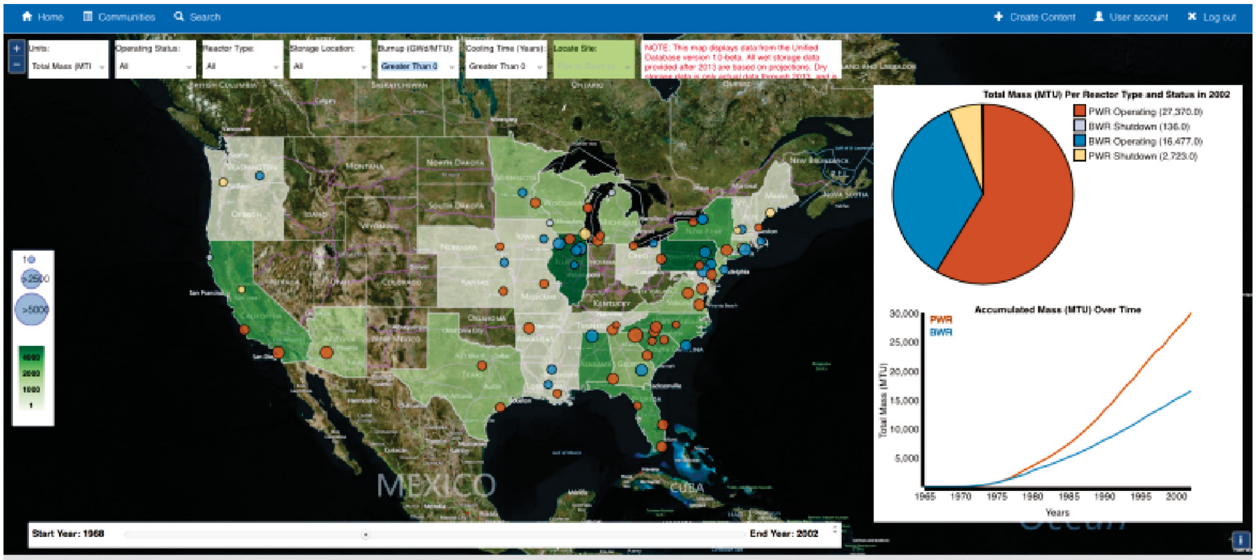

Figure 33. SNF inventory map available on CURIE.

8.7.7 Standardization and Integration

The NFST continued evaluating opportunities for standardization within the nuclear waste management system to establish the basis for any future policy decision-making regarding potential benefits and impacts of deviating from current SNF management practices.

- Concluded a three-year systematic evaluation of including standardized canisters into the waste management system. The FY 2016 evaluation, building on FY 2014 and FY 2015 work, focused on SNF that is loaded into standardized canister systems at reactors or that is transported to an ISF in a reusable, bolted-lid transportation cask and subsequently loaded into standardized canister systems (Figure 34). The goal of this effort was to gain a better understanding of the impacts of leaving spent fuel pools open for extended periods of time to accommodate access to the fuel and how updated packaging for disposal concepts and associated costs impact the system-wide evaluation.
- Developed a recommended path-forward to further develop the standardized canister concept, including (1) demonstration of technologies that would be required to efficiently load SNF into smaller-capacity standard canisters, (2) development of a more detailed conceptual design for potential future applications for certificates of compliance, if standardized canisters were to be incorporated into the commercial waste management system, and (3) continued integrated waste management system analyses using the results of the demonstration and conceptual design efforts.

Figure 34. Potential standardized transportation, aging, and disposal canister concepts.

8.7.8 SNF Characterization and Assessment

The NFST continued establishing a unified, comprehensive UNF database and integrated analysis system, referred to as Used Nuclear Fuel Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS), to characterize the input to the waste management system; provide a credible, controlled data source for key information; assess issues and uncertainties related to the extended storage and transportability of loaded canisters; support safety confidence and R&D prioritization; and provide a foundational data and analysis capability resource for the future.

- Released and distributed Version 3 of UNF-ST&DARDS that includes an updated Unified Database with SNF discharges through June 2013 (from EIA GC-859 data) from the U.S. fleet of nuclear power reactors, dry storage inventory of ~1,800 loaded casks, decay analysis of all the discharged SNF fuel assemblies, projected SNF assembly discharges, evaluation of the dose rates at 1 m from discharged SNF, criticality, shielding, thermal, and containment analyses of loaded dry storage casks (Figure 35).
- Continued an effort to update and supplement important UNF information previously documented in the old Office of Civilian Radioactive Waste Management Characteristics of Potential Repository Wastes Data Base and import it into the Unified Database.
- Updated information on the inventory of commercial SNF to incorporate recent dry storage data and future commercial SNF projections.

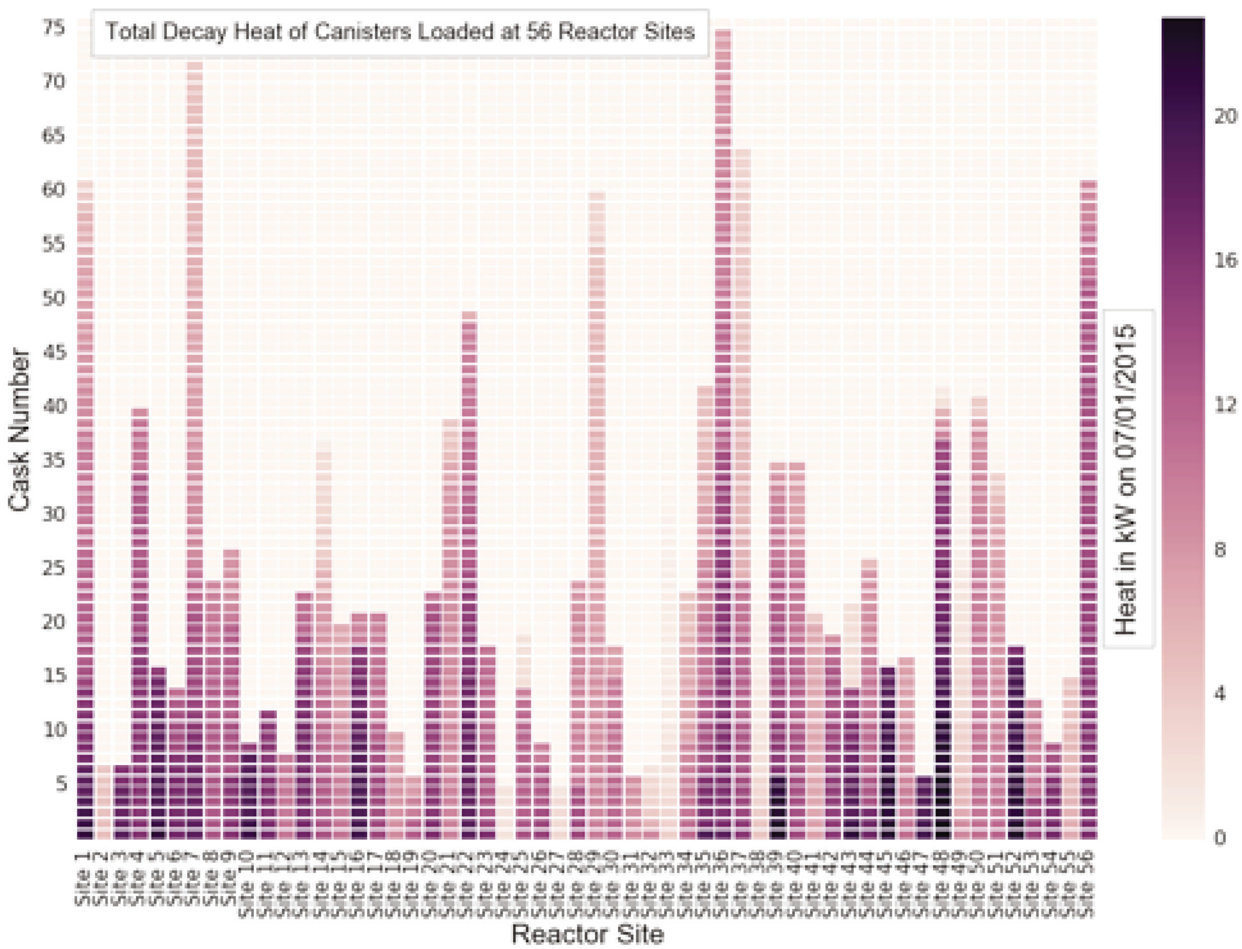

Figure 35. UNF-ST&DARDS overview of cask decay heat.

9. ADVANCED FUELS CAMPAIGN

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9.1 Overview

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

Supporting the Fuel Cycle Research and Development (FCRD) program, AFC is responsible for developing advanced fuel technologies to augment the various fuel cycle options defined in the DOE *Nuclear Energy Research and Development Roadmap*, Report to Congress, April 2010.

AFC pursues a goal-oriented, science-based approach aimed at a fundamental understanding of fuel and cladding fabrication methods and performance under irradiation, enabling the pursuit of multiple fuel forms for future fuel cycle options and reducing the time needed for fuel technology. This approach includes fundamental experiments, theory, and advanced modeling and simulation. The modeling and simulation activities for fuel performance are carried out under the Nuclear Energy Advanced Modeling and Simulation program, which is closely coordinated with AFC.

AFC initiatives in FY 2016 included management and integration of the advanced fuel and development activities supported by DOE through industry-led projects, national laboratory-executed research and development, and activities funded through DOE's NEUP. The campaign management staff also is responsible for developing and executing international collaborations on nuclear fuel research and development, primarily with France, Japan, the European Union, Republic of Korea, and China, as well as various working groups and expert group activities

In the Organization for Economic Cooperation and Development Nuclear Energy Agency (OECD-NEA) and the IAEA, three industry-led funding opportunity announcements, and university-led Integrated Research Projects (IRPs) funded in 2015, made significant progress in fuels and materials development. All are closely integrated with AFC and ATF research. The key FY 2016 technical area outcomes are highlighted in the following subsections.

9.2 International Coordination and Collaboration

Bilateral agreements are supported in place and active with France, Japan, the European Union, the Republic of Korea, and China. The emphases of these agreements crosses the activities of the AFC and include; advanced LWR fuels with enhanced accident performance, metallic fuel development, irradiation testing and data analyses, and development of characterization and PIE techniques. Three joint irradiation projects have been developed with the Halden Reactor Project (Norway) in advanced LWR fuels, an instrumentation qualification test in ATR in advance of the ATF-2 loop test, a bilateral loop irradiation test of ATF concepts, and a creep test of FeCrAl and silicon carbide (SiC) samples in the Halden reactor. Activities are supported under four multinational agreements and arrangements: the Gen IV Sodium Fast Reactor project arrangement, the OECD-NEA, the European Atomic Energy Community (EURATOM), and coordinated research projects under IAEA. These multi-national agreements allow the review and coordination of fuel development activities worldwide.

9.3 Advanced LWR Fuels with Enhanced Accident Tolerance

LWR fuel with enhanced accident tolerance is another R&D area under AFC. These fuel systems are designed to achieve sufficiently higher fuel and plant performance to allow operation to significantly higher burnup, and to provide enhanced safety during design basis and beyond-design-basis accident conditions. The overarching goal is to develop advanced nuclear fuels and materials that are robust, have high- performance capability, and are more tolerant to accident conditions than traditional fuel systems.

The primary focus in FY 2016 was to continue fundamental research, development and demonstration on several promising ATF concepts; complete Phase I of the industry-led projects; establish screening attributes and metrics for ATF concepts; establish the needed infra- structure for testing and evaluation of candidate technologies; and coordinate research activities between DOE laboratories, industry funding opportunity announcements teams, university IRP teams and NEUP investigators. The needed capabilities and infrastructure are primarily in place for execution of Phase II activities. This includes high-temperature steam oxidation testing (recently developed specifically for ATF), material property measurements, and irradiation testing.

Also included in FY 2016 was the establishment of experimental transient testing capabilities for the Transient Reactor Test Facility at INL. Three principal test modes are currently under development: a static capsule test capability, a water test loop, and a sodium test loop. The static capsule test capability is being prepared for initiating testing of candidate ATF technologies.

9.4 Evaluation of ATF Concepts

The overall goal of ATF development is to identify alternative fuel system technologies to further enhance the safety, competitiveness, and economics of commercial nuclear power. The complex multi-physics behavior of LWR nuclear fuel in the integrated reactor system makes defining specific material or design improvements difficult; as such, establishing desirable performance attributes is critical in guiding the design and development of fuels and cladding with enhanced accident tolerance.

The proposed technical evaluation approach and associated metrics were compiled and released in 2014 in the "Light Water Reactor Accident Tolerant Fuel Performance Metrics" report. A summary of the ATF metrics was published in a technical journal article in FY 2016,[3] including addition of proposed weighting factors for each performance regime and fuel system attribute. These weighting factors were developed via coordination with the ATF Industry Advisory Committee and were reviewed by the Independent Technical Review Committee (TRC) that was convened in FY 2016.

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[3] S.M. Bragg-Sitton, M. Todosow, R. Montgomery, C.R. Stanek, and W.J. Carmack, "Metrics for the Technical Performance Evaluation of Light Water Reactor Accident Tolerant Fuel," *Nuclear Technology*, 195(2), p.111–123, August 2016.

The proposed technical evaluation methodology is intended aid in the optimization and prioritization of candidate ATF designs. Detailed evaluation of each concept will gauge its ability to meet performance and safety goals relative to the current UO_2 – Zr alloy system and relative to one another. This ranked evaluation will enable the continued development of the most promising ATF design options given budget and time constraints, with a goal of inserting one (or possibly two) concepts as an lead fuel rod or assembly in a commercial LWR by 2022.

The TRC was organized to provide an independent assessment of the technology feasibility for near term research and development of candidate ATF design concepts and prioritization of those concepts. Established in late 2015, the TRC was comprised of technology experts selected based on their knowledge of the technologies under review, reactor operations, and fuel fabrication plant operations. The cross-section of experts includes experience in the areas of materials (metals and ceramics), neutrons', thermal- hydraulics, and severe accidents to enable assessment of the technology feasibility for near-term development of the ATF design concepts.

The review of ATF concepts proposed by industry and national laboratories was held January 2016 in Washington D.C. The TRC was tasked with independent assessment of technology feasibility for near term research and development of candidate ATF design concepts and prioritization of those concepts but will also provide input to prioritization of concepts requiring longer-term development.

Input from the TRC was provided to DOE to provide input to selection of industry teams and concepts for Phase II research and development work.

9.5 ATF Industry Advisory Committee

The Advanced LWR Fuel Industry Advisory Committee (IAC) was established in 2012 to advise AFC's NTD on the development and execution of a program focused on advanced fuels for light water reactors. The IAC is comprised of leaders from the commercial light water reactor industry. They represent the major suppliers of nuclear steam supply systems, owners / operators of U.S. nuclear power plants, fuel vendors, and the Electric Power Research Institute. Members are selected on the basis of their technical knowledge of nuclear plant and fuel performance issues as well as their decision-making positions in their respective companies. The IAC meets monthly via teleconference and in a face-to-face meeting once a year. The IAC met in November 2015 in Washington D.C. at the Tennessee Valley Authority offices. Progress toward the TRC prioritization of successful concepts was discussed at the November 2015 meeting. In particular, the IAC provided input to the weighting factors recommended for use in applying the technical performance metrics to evaluation of ATF concepts.

Following the TRC prioritization in January 2016, committee work focused on more detailed discussions of technical aspects and funding associated with the concepts to be pursued going forward. Specifically, future needs for Phase 2 test reactor irradiation were considered, and utility input was sought on the funding challenges which are expected to arise following the TRC prioritization of concepts and selection of Phase 2 industry awards by DOE. Contacts were established with the NRC and Institute of Nuclear Power Operations regarding the IAC charter and planned near term activities. Both organizations are standing by to become more involved with the IAC at the appropriate time. In addition, new utility members joined the IAC in FY 2016 following entry into Phase 2. In general, utility leaders have begun to take on a greater level of interest in ATF as commercialization discussions become more detailed.

9.5.1 ATF INDUSTRY TEAMS – Westinghouse Electric Company

The overall objective of this program is to introduce ATF lead test rods and assemblies for SiC and coated Zr cladding with U_3Si_2 fuel into a commercial reactor by 2022. The objective of the current Phase 1b work is to design, test and build using commercially scalable technologies test articles for up to 6 year-long exposure at PWR conditions of prototypical ATF fuel rodlets. The data from this 6-year test reactor exposure and test evaluation will be used as the basis to license and load lead test rods into commercial reactors in 2019 and lead test assemblies in 2022. Figure 36 is a photograph of a Westinghouse ATF-1 experimental rodlet containing U_3Si_2 fuel. It is currently under irradiation in the ATR.

Figure 36. U_3Si_2 fueled rodlet now in ATR.

9.5.2 ATF INDUSTRY TEAMS – AREVA

AREVA has established a focus team to develop promising technologies for improved fuel performance. Composite fuel pellets designed for higher thermal conductivity were fabricated by University of Florida using spark plasma sintering process. Several types of protective metallic and ceramic coatings were developed and tested by University of Wisconsin and Savannah River National Laboratory. Molybdenum cladding with an outer coating was evaluated by EPRI. In the fourth quarter of 2016, AREVA integrated its global EATF concepts into the DOE Program. These include chromia doped pellets, chromium coated cladding and silicon carbide sandwich cladding (SiC-SiC). Figure 37 is a photograph of Cr_20_3-Gd_20_3-Chromia doped pellets manufactured at the AREVA facility in Erlangen, Germany.

Figure 37. Cr_2O_3-Gd_2O_3-Chromia doped pellets manufactured in Erlangen.

9.5.3 ATF INDUSTRY TEAMS – General Electric

The General Electric ATF design concept utilizes a FeCrAl alloy material as fuel rod cladding in combination with UO_2 fuel pellets currently in use, resulting in a fuel assembly that leverages the performance of existing/current LWR fuel assembly designs with improved accident tolerance. The use of FeCrAl cladding is a direct near term path to improve the safety of operating commercial light water reactors. Figure 38 is a photograph of an APMT tube manufactured by Sandvik.

Figure 38. Sandvik Materials Technology made APMT tubes 9.5 mm OD and 0.8 mm wall thickness.

9.5.4 UNIVERSITY-LED TEAMS – UNIVERSITY OF ILLINOIS

Zr alloys found in current U.S. reactors exhibit extremely poor resistance to high temperature steam present in loss of coolant accident conditions. The rapid oxidation of these metals degrades their structural properties, introduces large amounts of exothermic heat, and produces a significant quantity of hydrogen gas. Modifying an alloy by applying an oxidation resistant coating is a technique that has been successfully used in many industries to mitigate this type of adverse reaction. Applying this to zircaloy will allow for the inheritance of its well tested bulk material properties, while enhancing the environment facing surface properties. One candidate material for a protective outer coating is alumina forming FeCrAl alloys, which are well known for their oxidation resistance. This research focuses on the development and testing of a FeCrAl coating on zircaloy. Figure 39 is a graph showing the correlation between Al concentration and reduced kinetics.

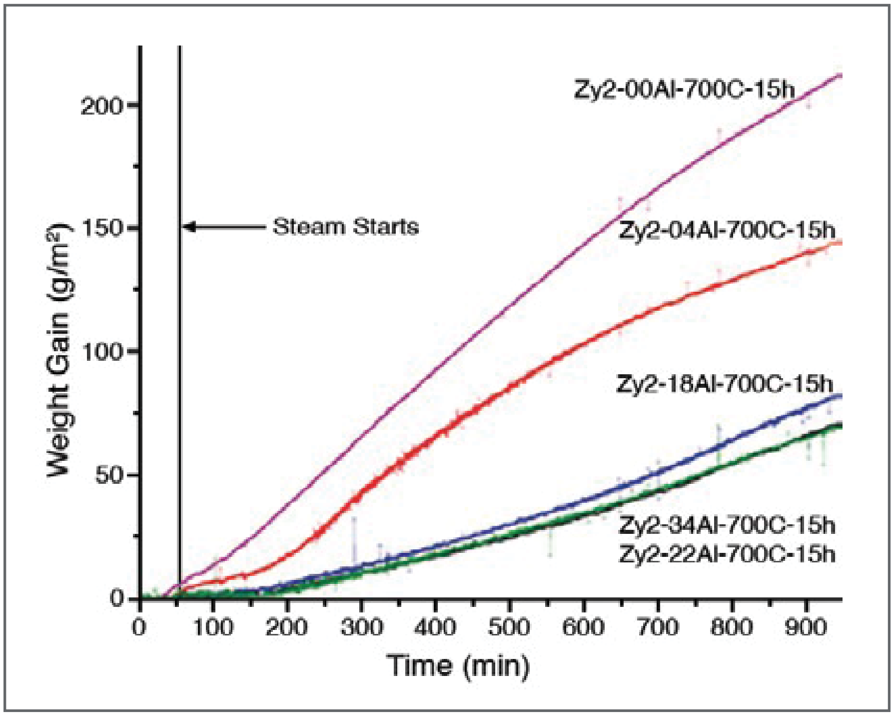

Figure 39. Normalized 700°C steam weight gain of various FeCrAl coatings deposited on Zr-2. Increasing the aluminum concentration is seen to reduce the kinetics.

9.5.5 UNIVERSITY-LED TEAMS – UNIVERSITY OF TENNESSEE

The goal of this NEUP-IRP project is to develop a fuel concept based on an advanced ceramic coating for Zr-alloy cladding. The coated cladding must exhibit demonstrably improved performance compared to conventional Zr-alloy clad in the following respects:

- During normal service, the ceramic coating should decrease cladding oxidation and hydrogen pickup (the latter leads to hydriding and embrittlement).
- During a reactor transient (e.g., a loss of coolant accident), the ceramic coating must minimize or at least significantly delay oxidation of the Zr-alloy cladding, thus reducing the amount of hydrogen generated and the oxygen ingress into the cladding.

The objective of this project is to produce durable ceramic coatings on Zr-alloy cladding. If successful, this research will have a very substantial impact on novel materials developing, especially on the development of advanced materials for applications in extreme environments. If it can be demonstrated that a ceramic coating resists oxidation in high temperature, aqueous water environments, as well as in the presence of energetic radiation, this will be a major advance in the development of corrosion-resistant materials for nuclear applications. Figure 40 is a photograph of Kurt Sickafus, Principle Investigator, and undergraduate students at the University of Tennessee.

Figure 40. Principal Investigator Kurt Sickafus discusses the microstructure of a multilayer TiN/TiAlN coating on a Zr-alloy substrate with prospective undergraduate students at The University of Tennessee. Kurt is describing both an image obtained using a SEM and the associated chemical analysis obtained using energy dispersive X-ray spectroscopy.

9.6 Advanced Reactor Fuels

The Advanced Reactor Fuels area achieved major objectives in FY 2016 in the fabrication development, characterization of actinide-bearing fuels, irradiation testing, and PIE of metallic fuel experiments. The campaign is currently pursuing the investigation of fuel alloy additions to immobilize the lanthanide fission products, sodium-free annular fuel concepts, and cladding coatings and liners for mitigation of fuel-cladding chemical interaction as advanced fast reactor technologies. A new TRU breakout glovebox was activated at INL with the capability to handle larger quantities of actinide feedstock materials. Figure 41 is a photograph of the inside of this new glovebox.

Figure 41. TRU Breakout Glovebox with 3013 can opener installed.

A significant accomplishment was demonstrating the reduction of americium oxide to americium metal in a new distillation process. Figure 42 is a photograph of metallic americium recovered from this distillation process.

Figure 42. Metal retrieved from the last distillation run using the lathe.

In FY 2014, the FUTURIX-FTA experiment was returned from the Phenix fast reactor in France, and in FY 2016 baseline PIE of these fuels was completed. The FUTURIX-FTA fuel experiment contained four pins of TRU-bearing metallic and nitride fuels. Figure 43 is an optical photomicrograph of a polished cross section of FUTURIX-FTA DOE1 (U-28.3Pu-3.8Am-2.1Np-31.7Zr) irradiated in the Phenix reactor. In the area of cladding development, a major goal is to develop fast reactor claddings capable of withstanding high irradiation doses (>250 dpa). In collaboration with industry and international partners, the program has successfully fabricated candidate ferritic-martensitic materials with improved material properties and has irradiated selected cladding samples to high doses.

Figure 43. Optical microscopy montage collected from a polished cross section of FUTURIX-FTA DOE1 (U-28.3Pu- 3.8Am-2.1Np-31.7Zr).

Figure 44 is a photograph of multiple tubes of an advanced oxide dispersion strengthens steel (14YWT) produced by hydrostatic extrusion at Case Western Reserve University. In addition, a first set of material samples irradiated in Russia's BOR-60 reactor has been returned to the U.S. for materials property characterization.

Figure 44. Multiple tubes produced by hydrostatic extrusion at CWRU on heat of 14YWT (FCRD-NFA1).

9.7 AFC International Collaborations

AFC researchers are very active in international collaborations with Korea, France, Japan, China, Russia, EURATOM, and OECD-NEA. These interactions and collaborations are managed through a combination of participation in Generation IV Global International Forum projects, International Nuclear Energy Research Initiative (INERI) projects, and participation in bilateral and trilateral government-to-government agreements. The ceramic fuels areas have collaborations primarily under the headings of Advanced Fuels within the U.S./Japan bilateral and the Gen IV SFR. There is also collaboration on Field Assisted Sintering of Nuclear Fuels under a US/EURATOM INERI arrangement.

9.7.1 Gen IV Sodium Fast Reactor Arrangement on Advanced Fuels

The SFR Advanced Fuel arrangement started in 2007 with a targeted duration of 10 years within the frame of the Generation IV Sodium Fast Reactor program. The primary objective is to investigate high burn-up Minor Actinide bearing fuels as well as cladding and wrapper materials capable of withstanding high neutron doses and temperatures. The project has been structured in 3 steps: evaluation of advanced fuels and materials options, minor-actinide bearing fuels evaluation, and assessment of high burn-up capability of advanced fuel(s) and materials. Participants in the arrangement include the DOE, CEA, Japan Atomic Energy Agency (JAEA), KAERI, EURATOM, China, and Russia with the latter two having joined in December 2015.

In FY 2016, the program management board completed the advanced SFR fuel type recommendation milestone which confirmed the prior Advanced Fuel Comparison report on fuel types and noted that the final SFR fuel type selection for each member country is dependent upon multiple domestic factors. The specifics of each country's experience, infrastructure and policies are critical determining factors in addition to the technical aspects in determining a preferred fuel type; the country-specific recommendation along with the reasoning was presented for each member country. Changes in the representatives and/or alternate representatives were made for EURATOM, France, and Russia in 2016.

9.7.2 U.S./Japan Civil Nuclear Energy R&D Working Group Collaboration on Advanced Fuels

Cooperative research under the Advanced Fuels area of the FCRD and Waste Management Sub-Working Group is performed under the general areas of proper- ties, performance and analysis. The goal of this effort is to perform collaborative R&D for evaluation of basic properties and irradiation behavior of advanced fuels. The objectives of the collaboration are to expand the basic properties and performance data and to improve understanding of advanced fuels with an emphasis on employment of advanced experimental techniques. Through incorporation of new minor actinide-MOX irradiation data the effort will also enable development and application of advanced modeling and simulation tools for design and performance analysis of oxide fuels.

In FY 2016, technical expert meetings were held in Japan and in the U.S. at INL to advance specific tasks on basic properties of fuels, development of PIE data, and modeling and simulation of irradiated transmutation MOX fuel.

Several joint publications from the fuel properties activities were prepared during the period. A key accomplishment in FY 2016 was negotiation of a BISON license for JAEA that will allow the collaboration to advance in jointly developing a minor actinide -MOX BISON model for fuel performance.

Another key aspect of the collaboration was a visiting JAEA scientist, Shinya Nakamichi, working at LANL on basic fuel properties. A highlight of the research by the current visiting scientist follows. For the near term, Mr. Nakamichi will be the last visiting scientist at LANL as the next Civil Nuclear Energy R&D Working Group visitors will reside at INL to support the BISON model development.

9.7.3 U.S.-France Advanced Nuclear Fuels R&D Collaboration

Conceptual design work continued on the Am-bearing blanket experiment planned for irradiation in ATR at INL. The concept proposes to investigate the possibility of transmuting Americium in the breeder blankets of future sodium fast reactors, and would put 10–15% Am into either depleted UO_2 or depleted U-Zr blanket rods. Such Am-bearing blanket rods would operate in low power and low temperature regimes for extended periods of time where no performance data currently exists. At year end, the DOE-CEA agreement for this experiment in ATR had not been signed, so final design and fabrication activities have been deferred until FY 2018.

9.7.4 OECD-NEA Expert Group on Accident Tolerant Fuels for LWRs

The Organization for Economic Cooperation and Development / Nuclear Energy Agency (OECD-NEA) Nuclear Science Committee approved the formation of an Expert Group on ATF for LWRs in 2014. Chaired by Kemal Pasamehmetoglu, INL Associate Laboratory Director for Nuclear Science and Technology, the mandate for the Expert Group on ATF for LWRs defines work under three task forces: (1) systems assessment, (2) cladding and core materials, and (3) fuel concepts.

Scope for the Systems Assessment task force (TF1) includes definition of evaluation metrics for ATF, TRL definition, definition of illustrative scenarios for ATF evaluation, and identification of fuel performance and system codes applicable to ATF evaluation.

The Cladding and Core Materials (TF2) and Fuel Concepts (TF3) task forces are working to identify gaps and needs for modeling and experimental demonstration; define key properties of interest; identify the data necessary to perform concept evaluation under normal conditions and illustrative scenarios; identify available infrastructure (internationally) to support experimental needs; and make recommendations on priorities. Where possible, considering proprietary and other export restrictions (e.g., International Traffic in Arms Regulations), the Expert Group will facilitate the sharing of data and lessons learned across the international group membership.

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

The original Expert Group mandate was established for June 2014–2016. In April 2016, the Expert Group voted to extend the mandate one additional year to June 2017 in order to complete the task force deliverables; this request was subsequently approved by the Nuclear Science Committee. All three task forces are expected to publish their respective deliverable reports in summer 2017.

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

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

CRPs are typically initiated with a technical workshop, followed by a solicitation for proposals on potential projects under the CRP. Each CRP runs approximately 4 years, with a joint plan for the work established based on proposals submitted by various member institutes/organizations. Studies under that joint plan are typically managed through a series of consultants meetings and small contracts.

A technical meeting on ATF for LWRs was initially held October 2014 at ORNL to launch the ACTOF CRP. Focused on nuclear fuel performance and safety, the objective of ACTOF is to support options for the development of nuclear fuel with improved tolerance of severe accident conditions through experiments to acquire data on new fuel and cladding materials and modeling of new fuel designs using ATF materials. ACTOF is expected to provide information to IAEA Member States to support decision making on ATF choices and to provide data, analyses, and advanced techniques to understand and predict the integral performance of ATF designs under normal, transient, and severe accident conditions.

The first research coordination meeting on ACTOF was held in November 2015 and was attended by 14 organizations across 11 countries, including Westinghouse and Battelle Energy Alliance (BEA, with INL as the participating laboratory) in the U.S. The Westinghouse contribution to the CRP will include information associated with the design and development of U_3Si_2 and Un-U_3Si_2 composite fuel, SiC composite cladding, Ti_2AlC-coated Zr cladding, and SiC wrapped Zr cladding. The INL contribution will provide implementation of material models and properties for FeCrAl and U_3Si_2 in INL's fuel performance code BISON, validate models against experiments, perform simulations of fuel rod behavior with ATF cladding and/or fuel (under normal and accident conditions), and perform sensitivity studies on critical material properties using BISON interfaced with DOE uncertainty quantification tools. Additional participants currently include Karlsruhe Institute of Technology in Germany, VTT Technical Research Center in Finland, A.A. Bochvar Institue (VNIINM) in Russia, Bhabha Atomic Research Centre in India, and KAERI in Korea. Proposals will be accepted until the next research coordination meeting, tentatively scheduled for spring/summer 2017.

Appendix A
Acronyms

ACTOF	Accident Tolerant Fuels for LWRs
AFC	Advanced Fuels Campaign
ANL	Argonne National Laboratory
ATF	accident-tolerant fuel
ATR	Advanced Test Reactor
ATRP	atom-transfer radical polymerization
BRC	Blue Ribbon Commission
CEA	French Alternative Energies and Atomic Energy Commission
CPP	U.S. EPA Clan Power Plan
CRP	coordinated research project
CY	calendar year
DOE	Department of Energy
DOE-NE	DOE Office of Nuclear Energy
DWR	defense waste repository
EPRI	Electric Power Research Institute
ENSA	Equipos Nucleares S. A.
EURATOM	European Atomic Energy Community
E&S	evaluation and screening
FCRD	Fuel Cycle Research and Development
FCT	Fuel Cycle Technologies
FY	fiscal year
HFEF	Hot Fuel Examination Facility
HLW	high-level waste
IAC	Industry Advisory Committee
IAEA	International Atomic Energy Agency
INERI	International Nuclear Energy Research Initiative
INL	Idaho National Laboratory
IRP	Integrated Research Project
IRT	Integrated Recycling Test
ISF	interim storage facility
JAEA	Japan Atomic Energy Agency
JFCS	Joint Fuel Cycle Studies

KAERI	Korean Atomic Energy Research Institute
LANL	Los Alamos National Laboratory
LCC	liquid-cadmium cathode
LEU	low-enriched uranium
LSFS	Laboratory-Scale Feasibility Study
LWR	light-water reactor
MIP	Multi-Isotope Process
MOX	mixed oxide fuel
MPACT	Materials Protection, Accounting and Control Technologies
MRWFD	Material Recovery and Waste Form Development
NEA	Nuclear Energy Association
NEST	Nuclear Education, Skills, and Technology
NEUP	Nuclear Energy University Program
NFST	Nuclear Fuels Storage and Transportation (project)
NGSAM	Next Generation Systems Analysis Model
NRC	Nuclear Regulatory Commission
NTD	National Technical Director
OECD-NEA	Organization for Economic Cooperation and Development Nuclear Energy Agency
ORNL	Oak Ridge National Laboratory
PNNL	Pacific Northwest National Laboratory
R&D	research & development
SACSESS	Safety of Actinide Separation Processes
SEM	scanning electron microscopy
SFR	sodium-cooled fast reactor
TRL	Technology Readiness Level
THORP	Thermal Oxide Reprocessing Plant,
TRC	Technical Review Committee
TRU	transuranic
TSRA	Technology and System Readiness Assessment
UFD	Used Fuel Disposition Research and Development (campaign)
UNF	used nuclear fuel
UNF-ST&DARDS	Used Nuclear Fuel Storage, Transportation & Disposal Analysis Resource and Data System
U.S.	United States
XAFS	X-ray absorption fine structure