INL/RPT-23-75380

Nuclear Fuel Cycle and Supply Chain (NFCSC) Technical Monthly September FY-23

September 2023

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Idaho National Laboratory

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Idaho National Laboratory
Idaho Falls, Idaho 83415

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CONTENTS

FIGURES

Nuclear Fuel Cycle and Supply Chain (NFCSC) Technical Monthly September FY-23

1. ADVANCED FUELS CAMPAIGN (AFC)

1.1 Campaign Management and Integration

On September 27 and 28 INL hosted a workshop with General Atomics on the irradiation testing of SiC-SiC composite cladding materials with UO2 fuels in loop 2A at the Advanced Test Reactor. Topics discussed included modelling approaches to give reasonable confidence of fuel rod integrity as well as analyze impacts of a loss of hermiticity. Additionally post irradiation examination plans were discussed and tours given of the hot fuels examination facility and the irradiated materials characterization laboratory. The irradiation is planned to begin in January of 2025.

1.2 International Collaborations

Multiple meetings were held between INL, NEA, CEA, Westinghouse-Sweden, and other participants from FIDES on the final proposal for an advanced fuels testing campaign through FIDES. The meetings discussed testing of metal fuels, mixed oxide fuels, and advanced TRISO concepts. The purpose of this irradiation test (dubbed FAST-II for AFC) is to explore basic irradiation evolution of advanced fuels that are of interest to the broader FIDES community. The proposal will be presented to the FIDES Technical Advisory Group and Governing Board in October.

1.3 Industry FOA

1.3.1 WESTINGHOUSE ATF FOA

[INL] Completed destructive PIE on Westinghouse ATF-2 medium-burnup coated rodlets and issued performance report to Westinghouse. This report provided characterization information on Cold-Sprayed Cr-coated Optimized-Zirlo™ cladding. Three series of rodlets were tested, the W, IW, and W2 series rodlets. The W rods had standard UO_2 while the IW had only stainless-steel plugs. The W2 series contained either ADOPT™ or Standard UO_2 pellets.

1.3.2 FRAMATOME ATF FOA

[INL] Completed FY23 scope PIE on Framatome ATF-2 rodlets at 30GWd burnup and issued report to Framatome (Milestone). This report provided important characterization data on medium-burnup ATF-2 rods with both doped UO2 Fuel and Cr coated cladding, including mechanical property data, and H pickup data.

[INL] ATF-2C began irradiation April 26, 2023, in the Advanced Test Reactor 2A Loop Cycle 171A and reached 118 effective full power days of irradiation at the end of Cycle 171B on September 26, 2023.

1.4 ATF Lab Activities

1.4.1 ATF Fabrication Properties

[LANL] The L2 milestone M2FT-23LA020201011 titled "Handbook on ATF doped UO2 properties" was completed on September 8, 2023. This report presents a compilation of experimental reference datasets, including peer-reviewed journals, for the properties of Cr-doped UO2. Trends are provided when applicable as well as direct comparisons with undoped UO2. Recommendations for future data collection needs are also discussed throughout this document. This non-proprietary database that we will be submitting for publication later this year will be accessible through publicly available journals and will support modelling efforts currently in place for the development and qualification of ATF concepts. This milestone provides a living document to allow a repository for future test irradiations and PIE results currently on-going within the AFC.

The L4 milestone M4FT-23LA0202010111 titled "Progress report on coordination efforts towards advanced non-destructive pulsed neutron PIE fuel" The report details efforts toward pulsed neutron techniques and provides microstructural data such as phase fractions as well as crystallographic data such as lattice parameters from diffraction analysis as well as spatially resolved mapping of isotope densities from energy-resolved neutron imaging, in particular neutron absorption resonance imaging, and overall isotope assay with better sensitivity for minority isotopes from neutron absorption resonance spectroscopy. Furthermore, after characterization at ambient condition, heating of irradiated or spent fuel will allow to characterize differences of e.g., lattice thermal expansion compared to fresh fuel or disappearance of defects. To enable such characterization, a sample holder design is presented that is planned to be applied to a high-burnup fuel sample from the H.B. Robinson reactor. We present the design criteria as well as the design for the sample holder and assorted equipment to load the sample into the robotic sample changer for ambient condition characterization and a furnace capable of reaching up to 1150C for high temperature characterization. All sample handling is expected to be done by remote handling.

[LANL] L3 milestone M3FT-23LA020201024 titled "Development of advanced ceramic nuclear fuel composites", was submitted. The report focused on the development of advanced ceramic nuclear fuels for cross-cutting applications. This is important if advanced reactors operating at high temperatures and with long lifetimes without the need to refuel are to be established. UN emerged as the advanced fuel with the most cross-cutting potential as it has been identified as a long-term candidate for LWRs, has been proposed for microreactors, and has also been selected for commercial power generation in LMFRs. However, it is widely acknowledged in the field that enrichment to N-15 is beneficial and is often thought of as a requirement that hinders its implementation. Neutronic analysis to assess the benefits of N-15 enrichment have been previously carried out for LWRs and PWRs as well as for LMFRs. However, these assessments have not been previously performed for specialty microreactors. In this report, neutronic calculations were performed in collaboration with Brookhaven National Laboratory to assess the balance between N-15 versus U-235 enrichment for the case of a thermal spectrum heat pipe based microreactor. The challenges associated with formation of C-14, its waste disposal, and storage were also discussed. In addition, a thorough assessment of the existing data gaps when it comes to the fuel performance of UN was carried out and these gaps were related to needs for the development and validation of existing fuel performance models. The primary data gaps that were established are the fission and gas swelling behavior, the creep and fracture toughness characteristics, and the fracture/relocation behavior under transient operation. Finally, a prioritization of these data gaps for the three reactor categories assessed in this report was also performed. Depending on the application, UN can serve as the primary phase for engineered composites with enhanced performance characteristics. For example, secondary phases can be added to enhance thermo-mechanical characteristics or to increase cycle length by incorporating a burnable-absorber phase such as UN-UB_2. Proposed engineered composites and their performance characteristics must be assessed on a case-by-case basis depending on the reactor application they are proposed for.

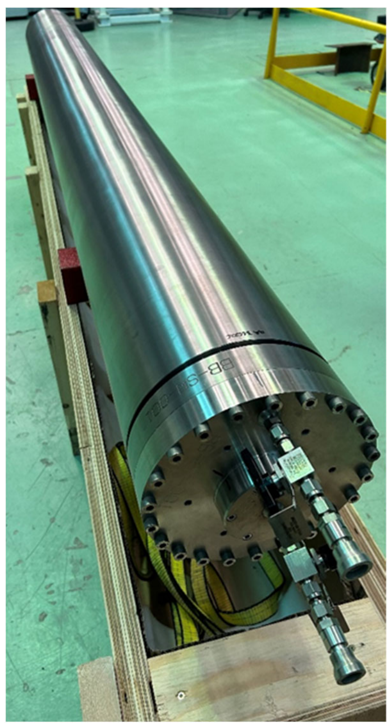
1.4.2 ATF Core Materials

[LANL] Milestone 4 (High temperature microscale mechanical testing at up to 400 °C of irradiated Cr coated Zircaloy) was completed and final report was submitted to the PICS:NE system. As an effort to develop accident tolerant fuel cladding after the Fukushima-Daiichi accident in 2011, Cr coating on the Zircaloy cladding has been proposed to prevent high temperature steam corrosion and oxidation. It is critical to test and understand the Cr coating performance under the reactor-relevant environment to predict its behavior. Through the milestones accomplished, it is now able to quantify the coating adhesion strength and the fracture behavior of each coating method via combined small-scale mechanical testing and ion irradiation at reactor operation temperature (~ 300 ℃). The results provide valuable database to take steps toward implementing this new accident tolerant cladding concept in the future reactors.

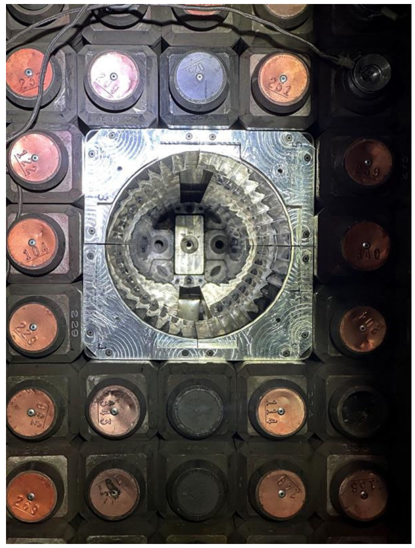
1.4.3 ATF Safety Testing

[INL] Safety analysis required to transition to the new irradiation containment vehicle continued. The third and fourth rodlets were assembled, welded, and leak checked. After welding, both rodlets were installed into their respective specimen holders in their capsules. The cladding thermocouples were welded onto the surface of the third rodlet.

[INL] Completed core reconfiguration to allow use of the new Big-BUSTER in TREAT, Figure below.



This was completed by removing the innermost 9 fuel drive segments from the TREAT core then placing the new moderator assemblies, Figure below, into the center of the core.



Completed core characterization in TREAT to successfully launch the large experiment capability in TREAT allowing for more room in the experiment capsules to enhance the experiments with new state-of-the-art instrumentation and allow for longer and multiple fuel pins.

1.4.4 ATF Post Irradiation Examination

[INL] The equipment is currently being manufactured in Karlsruhe, Germany with a scheduled shipment to INL in the November timeframe. Concurrently, BEA engineers are working on Analytical Lab design modifications to accommodate the future equipment.

1.4.5 ATF Performance Assessment

[BNL] Completed M4 milestone report "Assessment of Core Performance of Advanced LWR Fuel Concepts in Normal and Accident Conditions – Summary Report on FY23 Activities" (M4FT-23BN020206012). The efforts in FY23 focused on the modeling of microreactors, coordinating with FT-23LA02020102 - Development of cross cutting advanced ceramic fuel concepts.

1.5 Advanced Reactor Fuels

1.5.1 AR Irradiation Testing

[INL] ATR 171B-1 FAST-1 irradiations remain on-going with expected cycle completion on 9/26/23. Preparations are on-going to support a 11/16/23 shipment to HFEF. Irradiation support hardware fabrication (cadmium-lined baskets & flux-wire monitors) is now concluding. Flux-wire monitors are done, and cad baskets are expected to be completed early next fiscal year.

FAST-1 destructive exams completed GASR operations, sectioning, and metallography of 2 rodlets (U-10Zr annular and solid fuel) supporting on-time completion of milestone. EBR-II legacy PIE is on-going. Additional FY2023 PIE work is leveraging MFF legacy fuel available in the hot cell to expand the database on FCCI, thermophysical properties, and swelling on prototypic U-10Zr fuel pins. This work is supported by on-going BISON modeling and sectioning of the first MFF pin. MFF sectioning of the first pin is now complete including preparation of SEM and thermophysical samples to be analyzed in the coming months.

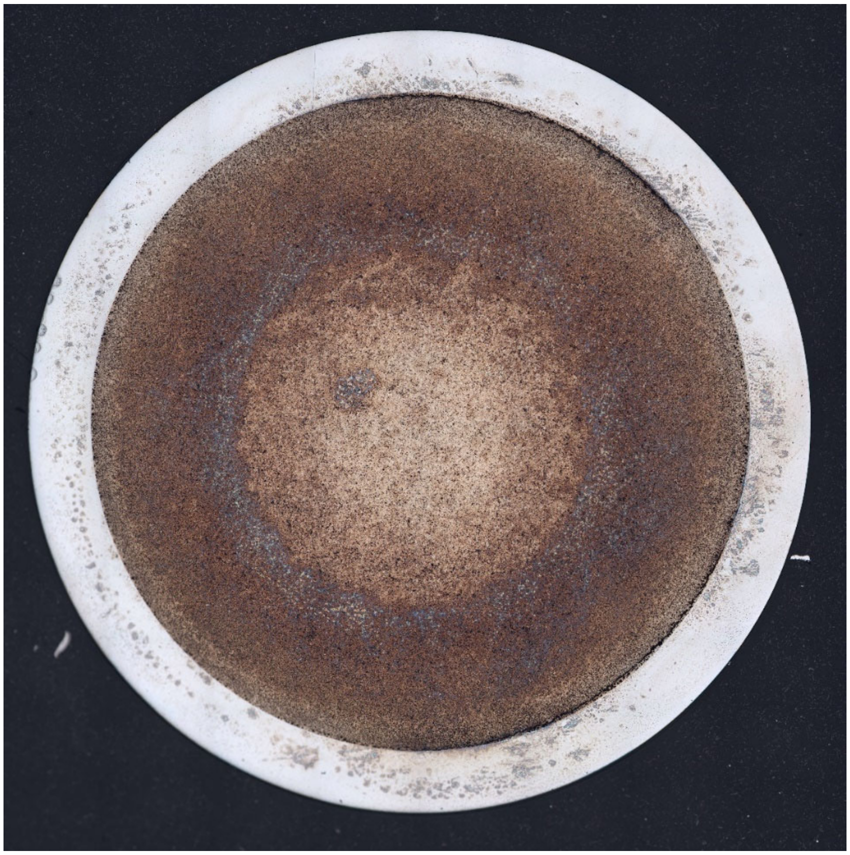


Figure 1. FAST-007 transverse cross-section from metallography.

1.5.2 AR SAFETY TESTING

[INL] The THOR Conax improvement investigation is finalizing its data analysis. The investigation aims to identify and address the root causes of the performance issues that affected the THOR-Conax helium leakage. Below is an image of some of the testing taking place.

The new one-piece heatinks that will be used for the THOR project has been ordered for FY24. This material will arrive soon and will allow us to start the fabrication process. The one-piece heatinks are expected to improve the performance and reliability of the THOR system.

[INL] The project has contacted Ajax/Tocco corporate office and have been told the sales manager no longer works there. Another point of contact is being provided, at which point BEA will ask for a quote to provide the necessary design services, followed by fabrication of a new furnace.

1.5.3 AR PERFORMANCE ASSESSMENT

[INL] Completed the purchase of a new glovebox and submitted the corresponding Level 2 milestone (M2FT-23IN020305031). INL and TerraPower developed an out-of-pile modular sodium test loop to support sodium fast reactor experiments. The modular sodium test loop (MSTL) is currently housed in the Idaho Engineering Demonstration Facility (IEDF). MSTL is used to calibrate sensors, evaluate test hardware, and perform hydraulic experiments in support of the Natrium fuel qualification and other sodium fast reactor experiments. The IEDF glovebox is necessary to safely handle sodium and sodium contaminated equipment in IEDF.

Continued designing the necessary IEDF supporting infrastructure.

1.6 Silicon-Carbide Cladding

1.6.1 SIC CLADDING LAB ACTIVITIES

[INL] Completed TEM and APT characterization for SiC cladding tested in TREAT, which met the level 3 milestone. This is the first of its kind characterization using high resolution techniques such as TEM and APT on TREAT irradiated SiC. The XCT data provided 3D representation of cracks and porosity in SiC CMC samples to help understand the PCMI behavior and crack propagation. The TEM results draw direct comparison of the grain structure and size of the fiber, CVI matrix, and CVD coating, which contributes to different mechanical and thermal properties of each component. The Atom Probe

Tomography reveals silicon segregation found at certain grain boundaries and defects, which will be correlated to TEM and EELS analysis. The data will be later used to construct multiscale BISON models for SiC, where anisotropic properties can be calculated and predicted. This work paves the way for mechanistic understanding of SiC CMC under PCMI and irradiation conditions.

[ORNL] The milestone M3FT-23OR020501024 titled "Comprehensive characterization on stress corrosion cracking of SiC composite tubes" has been completed. The milestone report describes results from the mechanical evaluation of unirradiated SiC fiber–reinforced SiC matrix composite tubes fabricated by General Atomics under a high-temperature steam environment. The experiments were aimed at identifying key material degradation behavior under environments relevant to loss-of-coolant accidents of light water reactors. The findings will help identify test parameters for future integrated loss-of-coolant accident tests using fueled and irradiated tubes.

Mechanical tests of the SiC composite tubes at 1,000°C under steam and inert environments were conducted using a unique test capability at Oak Ridge National Laboratory (Fig. 1a). The material tested was a duplex tube with a thick monolithic SiC layer on the outer surface. The tubes were subjected to preloading at stresses greater than the matrix-cracking stress and then exposure to the high-temperature steam with 50%–75% of the preload at a constant displacement (Fig. 1b). In the presence of matrix cracks, the steam exposure caused embrittlement of the SiC composite tubes and failure at a stress level below the matrix-cracking stress. The material degradation was explained by a fiber oxidation model, which can be applicable to various SiC cladding concepts. The embrittlement could be a limiting factor for using SiC cladding when subjected to loss of cooling accident conditions, and further investigation of the behavior of these materials under application-relevant conditions is warranted.

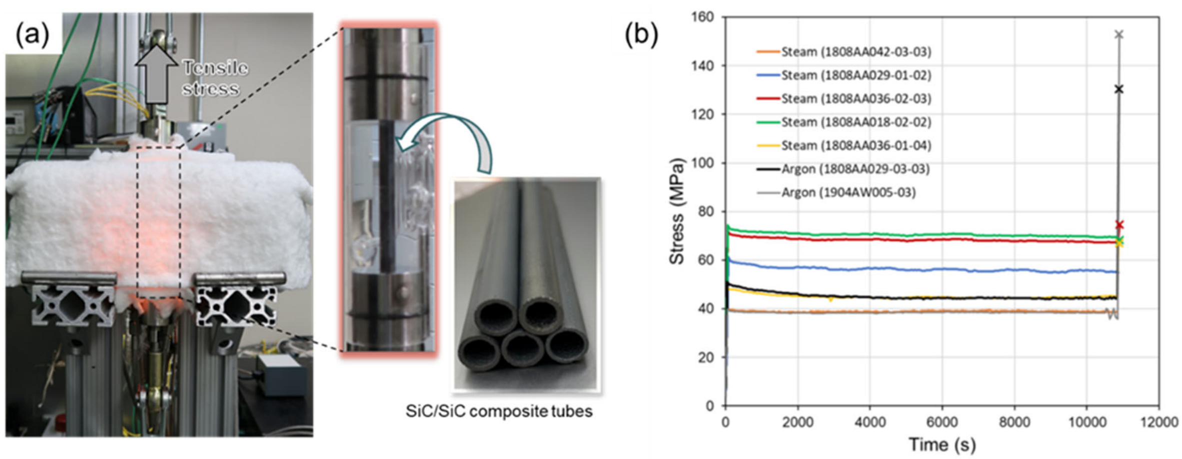


Figure 2. (a) Testing apparatus used for stress-rupture tests under high-temperature steam and (b) Time-dependent stress of SiC/SiC composites at a constant displacement condition at 1,000°C under steam and argon environments for 3 h. Cross marks indicate postexposure tensile strength.

1.6.2 SiC Cladding (FOA)

[INL] On-site meeting was held for GA visitors at INL. The meeting was productive and covered discussions on modeling and simulations for SiC CMC cladding, ATR irradiation, and planned PIE activities. INL also received the LFA samples and 173a cycle reference rods. The pre-irradiation characterization and testing will follow in the next few months. The shipment of non-fuel SiC rodlets irradiated at ATR for 171a to HFEF will be delayed to March 2024.

1.7 Capability Development

1.7.1 ATR Loop Installation

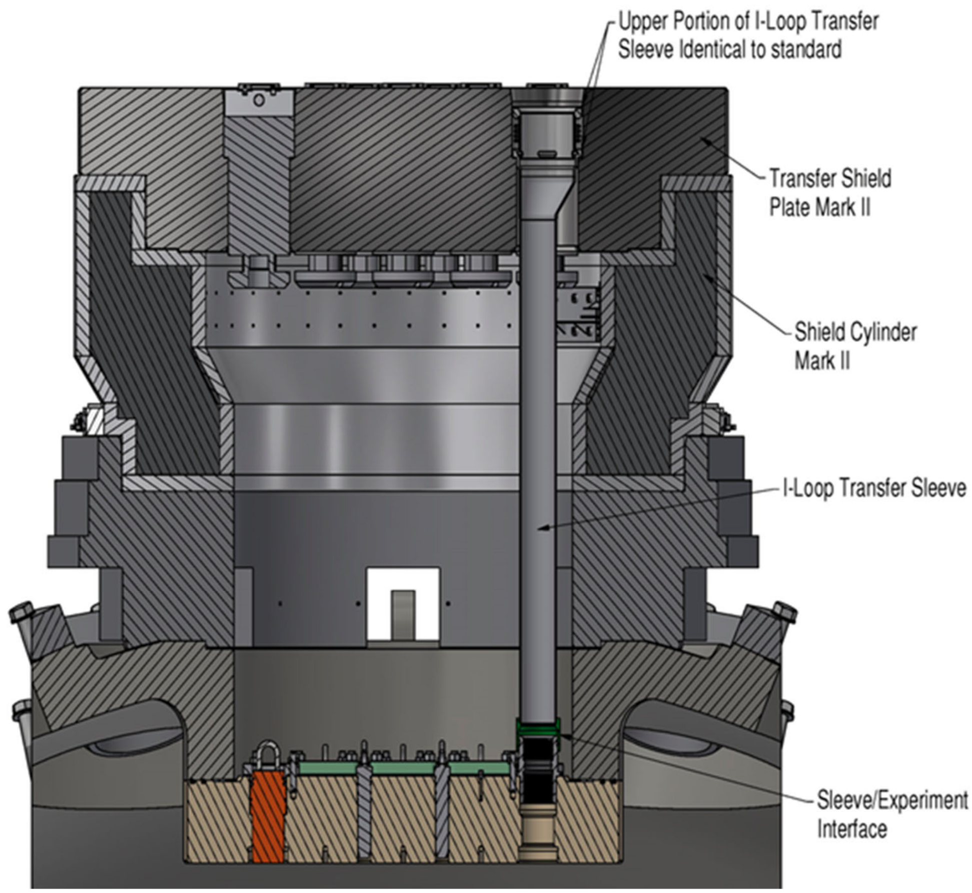
[INL] Complete Conceptual Design of Cubicle & Cleanup System (C&CS) Report - FY23 Level 2 Milestone (INL M2FT-23IN020601019).

Completed Conceptual Design for HDW Spool Pieces and initiated Preliminary Design for HDW Spool Pieces.

Completed Conceptual Design for Nozzle Trench Piping Shielding and initiated Preliminary Design for Nozzle Trench Piping Shielding.

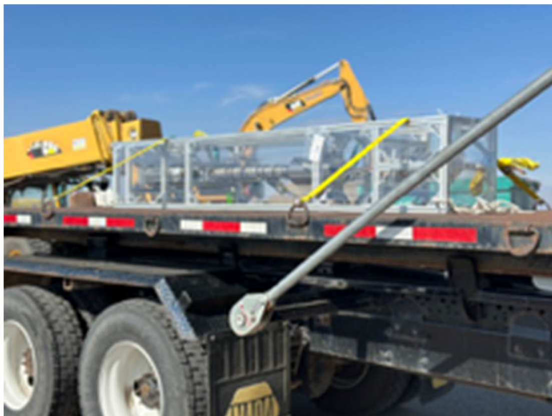
Completed SST prototype lab modifications for autoclave flow testing.

Released the Transfer Shield Plate and Shield Cylinder fabrication subcontract RFP - FY 24 Level 2 Milestone (INL M2 M3FT-24IN020601011).



1.7.2 TREAT LOCA Testing Infrastructure

[INL] TWIST LOC-C-1 was transported to TREAT where final assembly was completed, including x-ray verification of placement, transfer of experiment to vertical position, filling the capsule with water, leak testing, inserting into Big-BUSTER and final leak test.



TREAT irradiation of the first TWIST experiment (LOC-C-1) was successfully completed. meeting the PEMP Notable Outcome milestone. All instrumentation performed as expected and the data analysis is being completed.

Assembly of TWIST LOC-C-2 was completed at Measurement Science Lab (MSL) and transferred to the Advanced Fuels Facility (AFF) at MFC.

The TWIST LOC-C-2 and LOC-C-3 rodlets were assembled, welded, and leak checked. The LOC-C-2 rodlet has been placed into the holder.

1.7.3 Refabrication and Instrumentation Capability

[INL] Completed the year end accomplishments report and submitted the Level 3 milestone M3FT-23IN020603014. The accomplishments report demonstrates the progress toward building an advanced refabrication and reinstrumentation capability that can support future instrumented experiments. Critically important in this accomplishments report is documenting the progress toward developing capabilities to weld thermocouples on cladding exterior surfaces, and to perform centerline drilling of irradiated UO2 fuel to allow for centerline instrumentation to be utilized. Highly instrumented experiments are a need for future experiments as in-situ data is necessary to measure material performance during experiments.

Completed a draft presentation of the different remote welding system capabilities at INL. The summary of the existing welding systems and planned expansion of those capabilities is an aid to communicating both capabilities and limitations in performing welding on experiments including both end cap welding, seal welding, and welding of thermocouples to cladding exterior surfaces. This is significant to help experiment designers understand the current limitations of the systems, and planned future capabilities so that experiments can be planned and designed accordingly.

For more information on Material Recovery and Waste Forms Development contact Ken Marsden (208) 533-7864.

2. MATERIAL RECOVERY & WASTE FORM DEVELOPMENT CAMPAIGN

2.1 Accelerated EBR-II Processing

[INL] Completed the final FY23 treatment batch. In total, 134 kgs of HEU EBR-II driver fuel was treated in 11 processing batches at the MK-IV electrorefiner.

Completed 11 HALEU regulus production runs via the cathode processor in FCF.

Completed preparations for receipt of the HFEF-14 stainless steel shipping cask at FCF and proceeded with receipt of the 1st shipment of EBR-II driver fuel for FY24.

[INL] Completed the Multi-Function Furnace post installation adjustments and continued with a cladding hull distillation run.

Salt removed from the cladding will be returned to the MK-IV ER to help maintain the conditions necessary for future treatment.

Continued with HALEU pyropolishing experiments. Three of the smaller HALEU reguli from the FCF production process have been transferred to HFEF, with two of the three being electrorefined in the in the ER of the Scalable Pyrochemical Recycling testbed (SPYRE). The resulting dendrite was collected and sampled for further analysis to determine the impact on the chemical and isotopic composition.

Completed the transfer of the Scraped Cathode Rod Assembly Prototype (SCRAPE) into the FCF argon cell. The SCRAPE is an advanced cathode configuration that will test the ability to remotely recover uranium dendrite from the cathode mandrel and compact it while still within the electrorefiner. If successful, efficiency gains in the electrorefining process are anticipated.

[INL] Completed repairs to the cracked hot cell window at workstation #10 in the HFEF argon cell. Upon completion of the window repair, HALEU previously produced in the May production run were packaged for removal from the hot cell.

2.2 Zircex

[INL] Short communication completed in fulfillment of deliverable commitments. Overall, a new high dose rate and temperature cell holder has been designed, built, and successfully tested in support of advanced low temperature chlorination technologies for used nuclear fuel reprocessing. This capability is now available to the Material Recovery and Waste Form Development (MRWFD) campaign for future irradiation studies.

[PNNL] PNNL has completed scope for FY and the Raman instrument is ready to ship back to INL.

[PNNL] A study was completed to prepare high-assay low-enriched uranium (HALEU) aqueous polishing high-activity waste raffinate for vitrification. The study included glass formulation and testing studies completed in June and melter feed processing completed in September. Combined, these studies show the feasibility and supply data for the cost analyses of a vitrification process using either in-canister melting or GeoMelt® in-container vitrification. The spray drying of melter feed with a sugar addition to destroy nitrates was found to be a reasonable approach to prepare free flowing particulate feed to the melter (Figure 3). Two reports were issued, one summarizing the glass formulation and testing results and the second summarizing the melter feed preparation studies.

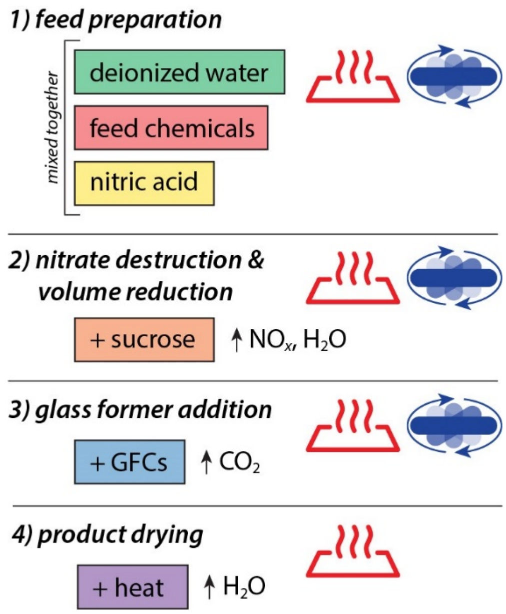


Figure 3. General process used for preparing the simulated HAW feed tests.

2.3 Waste Forms & Off-Gas Capture

[INL] A prototype conceptualization of a salt waste furnace was completed (Figure 4 and Figure 5, below). The device is designed to remove halogens from salt-based waste streams so that the remaining salt cations can be converted to oxides and immobilized in a glass-based waste form. It allows a single system to perform both the dechlorination and the vitrification steps required to treat salt wastes. The chlorine is removed either as HCl, NH4Cl, or a mixture thereof and these byproducts can either be captured directly or reacted with uranium metal to create UClx within a separate chamber within the device. Several additional ideas are captured within the report, "Conceptual Design of a Salt Dechlorination and Vitrification Apparatus" INL/MIS-23-72319, including charging dry or wet chemicals into the crucible, stirring the melt, and actively cooling the melt.

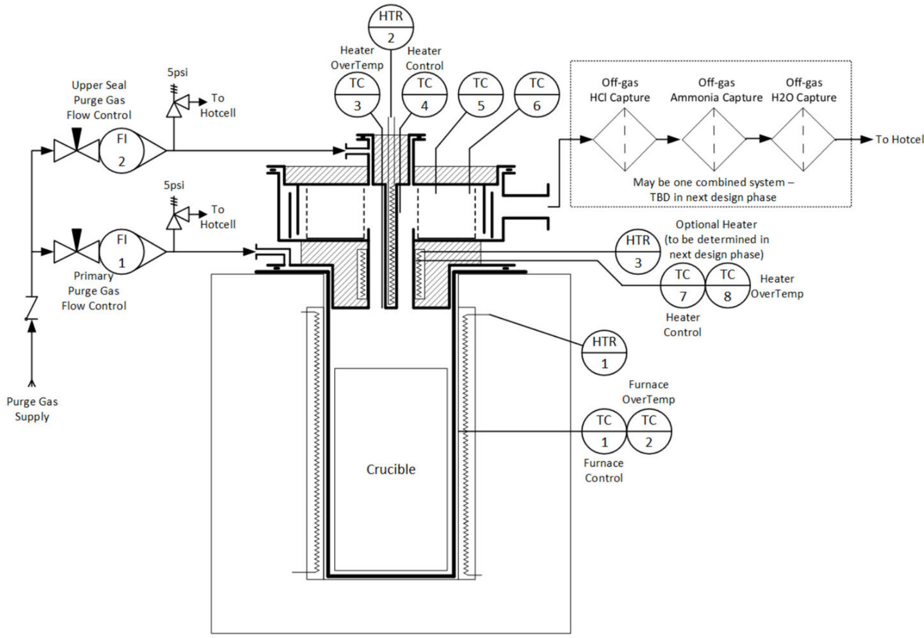


Figure 4. Conceptual process and instrumentation diagram.

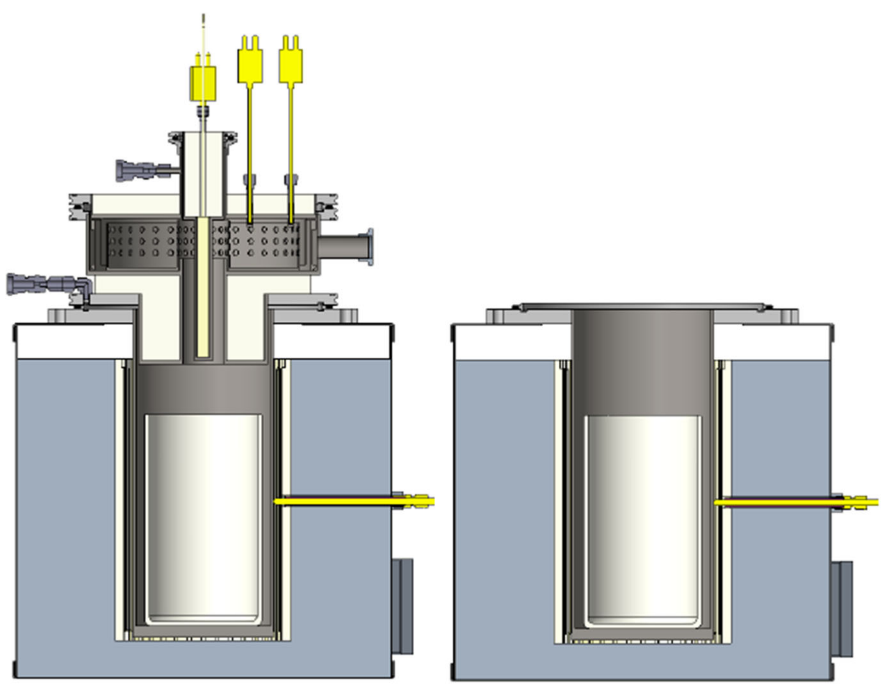


Figure 5. Conceptual model section with the condenser (left) and without (right)

Kr/Xe Cost Sensitivity Study: INL's Integrated Energy & Market Analysis team, supported by researchers from INL and PNNL, conducted a cost sensitivity analysis for krypton and xenon capture looking at current market prices and projections for the baseline cyrogenic distillation for capture and separation. They concluded that xenon sales would not be financially viable at today's market price but may have potential as demand for xenon grows. Krypton, on the other hand, would not be worth capturing strictly for sale. However, since regulatory requirements drive the need for krypton capture, sales could offset a very small portion of the sunk cost.

Prioritizing Off-Gas Metrics: INL hosted a workshop to prioritize solid sorbent testing parameters for UNF reprocessing off-gas treatment. The result of that workshop is, “Prioritizing Off-Gas Metrics: A Guide for Comparable Off-Gas Capture Testing,” to be used by researchers across multiple institutions to ensure consistent, comparable results. It represents a collaborative effort among four national laboratories and DOE NE to set standards for off-gas research.

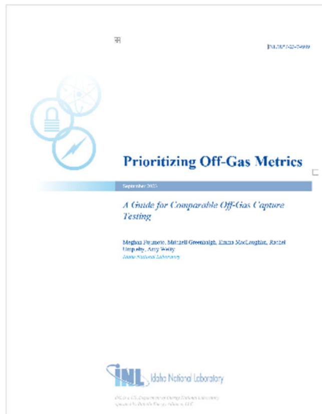


Figure 6. INL Workshop Report.

[PNNL] Milestone M4FT-23PN030104045 was completed as a PNNL report (PNNL-34917) with the title “Complete studies of crystalline vs glassy Fe-P-O waste forms,” which summarizes work done at PNNL on different cooling curves for the DPF5-336 iron phosphate waste form reference material. A joint PNNL-INL report (PNNL-34755) was completed on conceptual designs for implementing a dehalogenation and vitrification device for processing iron phosphate waste forms in a hot cell. A joint INL-PNNL report (PRS/RPT-23-03838) between the Systems Analysis (SA) Campaign and MRWFD was sent to the SA and MRWFD NTDs for review regarding a noble gas cost sensitivity study.

Synthesized several MOFs predicted by AI/ML approach. These were characterized by using various spectroscopic techniques to evaluate the phase purity, surface area and morphology. However, based on the initial experiments, none of the MOFs show any xenon nor krypton adsorption at different temperatures. Next steps will include evaluating the ZIFs as alternate sorbents for Kr separation. (B. Riley)

2.4 Aqueous Separations

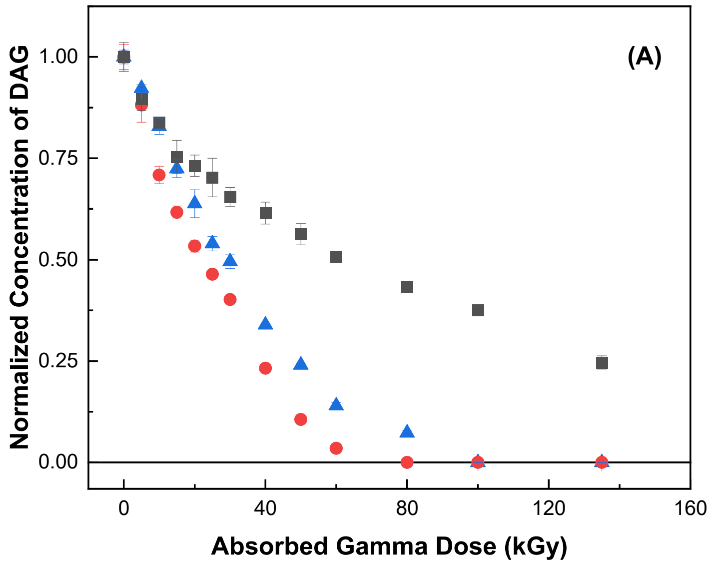
[ANL] A milestone report summarizing the development of DEHiBA D-value models for HNO3, U(VI), Pu(IV), Np(IV), Np(V), and Np(VI) and their incorporation into AMUSE v5 was completed in fulfillment of a Level 4 milestone. The developed models have been implemented in a compiled version of the code and used to develop a flowsheet based on the reference flowsheet developed for PNNL using user-defined D values. Am(III) and Li have also been implemented as constant D values.

A second milestone report summarizing the work on the on the development of microfluidic flow cell and initial results collected for kinetics data on the extraction of Ce(IV) by 30 v/v% TBP and 1 M DEHiBA from HNO3 was completed in fulfillment of a Level 4 milestone.

[INL] One FY23 research activity explored molecular constructs based on the guanidinium moiety, seeking affinity for a pertechnetate anion. The unsubstituted guanidinium cations showed evidence of pertechnetate recognition, yielding a modest impact on Tc distribution in mixtures containing a thousand-fold excess of nitrate anion. Guanidiniums containing an amino substituent are oxidized via a Tc-catalyzed redox mechanism which reduces technetium to its tetravalent state. This transformation affords TcO^{2+} species, which are excluded from the DEH*i*BA uranyl solvate in the organic phase. As a result, over 90% of Tc-99 can be efficiently scrubbed when equilibrated with moderately acidic mixtures of diaminoguanidine. The results were summarized in a recent report (INL/RPT-23-74656) submitted to complete M2FT-23IN030101026 milestone.

Another FY23 research activity pursued opportunities for improvement of hydrodynamic properties of solvents based on monoamide extractant DEH*i*BA. Four strategies to lower the viscosity of monoamide solvent formulations were examined. First, the characterization of novel extractants sought to modify the extractant geometry. Second, the addition of co-solvating *n*-octanol was studied. Third, a survey of alternative diluents was performed. Fourth, two extractants were combined. The first effort characterized 7 novel monoamide structures, identifying ligands E and F as a promising alternative to DEH*i*BA. When loaded with 100 $g \cdot L^{-1}$ uranium, the viscosity of ligand E formulations is 16% lower, relative to the equivalent reference flowsheet solvent. The addition of *n*-octanol as co-solvent did not lower the solvent viscosity. A variety of diluents was studied to explore the third option. The most promising outcome resulted from a progressive lowering of molecular weight of a linear paraffinic alkane. The replacement of *n*-dodecane with *n*-decane in the reference solvent yields a 31% reduction in viscosity when containing 100 $g \cdot L^{-1}$ uranium. This modification did not significantly lower the uranium loading properties. The effects of isoparaffinic branching and the presence of aromatic hydrocarbons were detrimental. For the fourth option, an equimolar mixing of DEH*i*BA and ligand E positively influenced solvent which contained the highest quantities of uranium but did not show evidence of synergy. After the examination of all four approaches, a 50% reduction of solvent viscosity was demonstrated for a novel monoamide F when utilized as 1.5 $mol \cdot L^{-1}$ / *n*-decane formulation. The results were summarized in a recent report (INL/RPT-23-74835) submitted to complete M3FT-23IN0301010210 milestone.

[INL] Completed milestone M4FT-23IN030101028 with a draft manuscript submitted to the NTD. Said manuscript is in the process of being submitted to Physical Chemistry Chemical Physics (Impact Factor = 3.676, 2023) for publication. The final data acquired by this project focused on the impacts of steady-state gamma irradiation on the radiolysis of DAG in water, aqueous 2.0 M nitric acid (HNO_3), and in a biphasic solvent system composed of aqueous 2.0 M HNO_3 in contact with 1.5 M *N,N*-di-(2-ethylhexyl) isobutyramide (DEH*i*BA) dissolved in *n*-dodecane. The DAG molecule was found to exhibit significant reactivity with the hydroxyl ($^{\bullet}OH$) and nitrate ($NO_3^{\bullet}$) radicals, indicating that oxidation would be the predominant degradation pathway in radiation environments. This is consistent with its role as a pertechnetate redox control reagent, suggesting similar products are formed. Steady-state gamma irradiations demonstrated that DAG is readily degraded within a few hundred kilogray, the rate of which was found to increase upon going from water to HNO_3 containing solutions and solvents systems, as shown in Figure 7A.



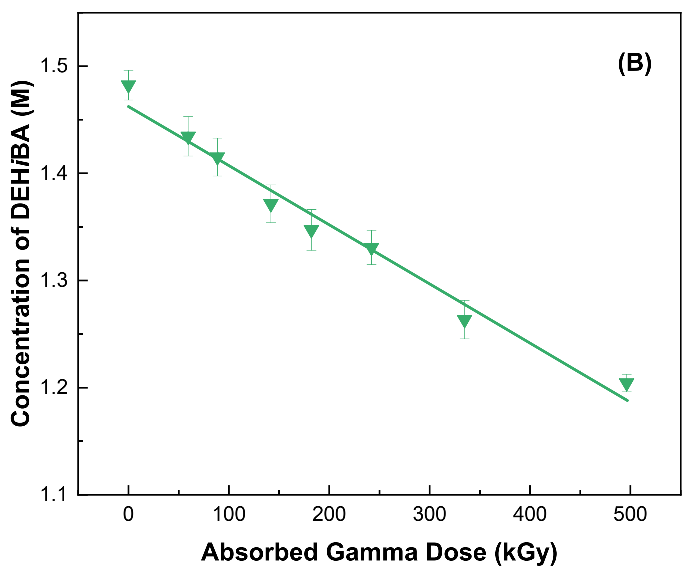


Figure 7**. (A)** Normalized concentration of DAG as a function of absorbed dose from the gamma irradiation of formerly: 50 mM DAG in water (■); 50 mM DAG in 2.0 M HNO_3 (●); and 100 mM DAG in 2.0 M HNO_3:1.5 M DEHiBA/n-dodecane (▲). **(B)** Concentration of DEHiBA (▲) as a function of absorbed gamma dose from the gamma irradiation of 100 mM DAG in 2.0 M HNO_3 in contact with 1.5 M DEHiBA in n-dodecane.

This behavior was attributed to a relatively slow reaction between DAG and the predominant HNO_3 radiolysis product, nitrous acid (HNO_2). Although no evidence was found for the radiolysis of DAG altering the radiation chemistry of the contacted DEH*i*BA/*n*-dodecane phase in the investigated biphasic system (Figure 7B), the utility of DAG as a redox control reagent will likely be limited by significant competition with HNO_2.

[PNNL] The report *DEHiBA-Based Uranium Recovery Flowsheet with 50% Reduction of Feed Volume* (PNNL-34964) was completed and issued to fulfill milestone M2FT-23PN030101041. A flowsheet was developed for recovery of uranium (U) from spent nuclear fuel (SNF) in a single solvent extraction cycle. The flowsheet uses *N,N*-di-(2- ethylhexyl)isobutyramide (DEHiBA) as a selective extractant for U. The Argonne Model for Universal Solvent Extraction (AMUSE) code was used to simulate the flowsheet, which resulted in the following key predicted performance metrics: (1) production of 8.4 L of HLW raffinate per kilogram of U processed (versus the set goal of ≤ 8.6 L of HLW raffinate per kilogram of U), and (2) 15.2 L solvent used per kilogram of U processed (versus the set goal of < 24 L/kg U). These metrics represent a significant intensification of the process over previously published flowsheets using DEHiBA, which would result in reduced plant footprint and operating costs. A series of batch contacts were performed to verify the flowsheet conditions. Strategies for routing Np and Tc to the HLW raffinate were demonstrated during the batch contact experiments.

[LANL] Results on the Gaussian Process regression (GPR) performance in predicting U(VI) distribution coefficients for the branched and straight monoamides have been presented in previous reports (see June and July 2023 reports). Due to the success in our methodology, we moved on to using GPR for the prediction of *M*(IV) distribution coefficients based on the same experimental factors. Results for this are presented in Figure 8. Again, observe that our selected kernel performs fairly well in predicting the distribution coefficients based on the varying input parameters. Note that the small distribution coefficients for *M*(IV) extraction with the branched monoamides, as all D values are < 2.0. This is an expected result at branching the of the R chain in monoamide ligands has been shown to suppress the extraction of *M*(IV) cations.

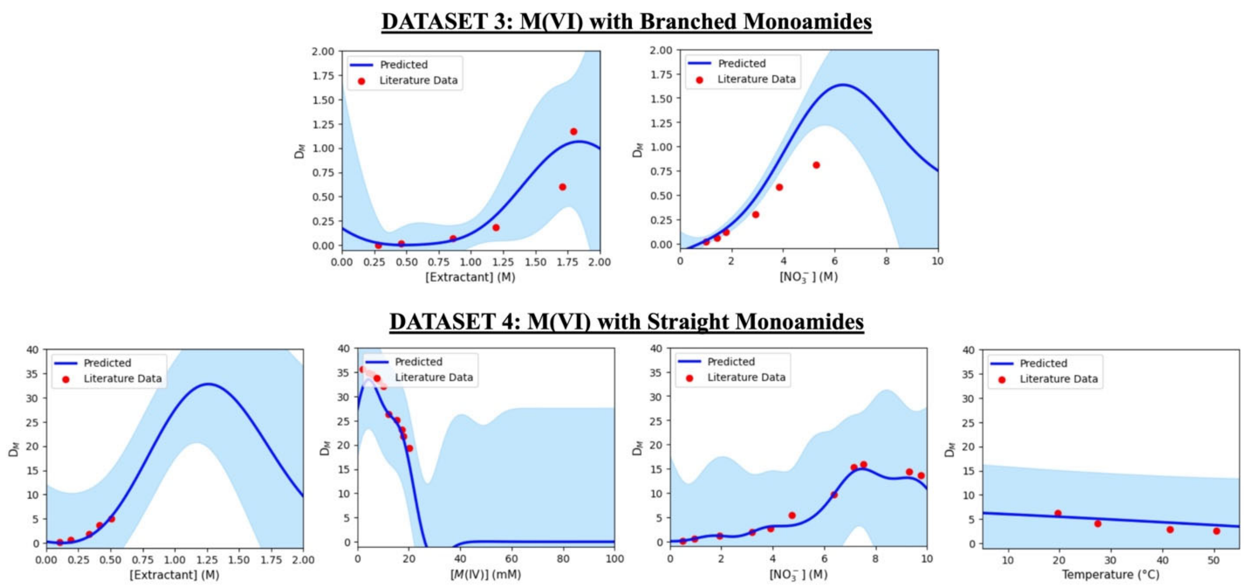


Figure 8. Gaussian Process regression modeling of *M*(IV) distribution coefficients based on varying experiment conditions. Blue lines show the model prediction, with the shaded area representing the 95% confidence interval (1.96*σ). Red dots are experimental data points published in literature.

Based on these above results and knowing that branched monoamides can show selectivity for the U(VI) cation, we pursued GPR on mixed DEHBA/DEHiBA systems to understand their ability to separate U(VI) for *M*(IV) cations. Here, we only had 25 total data points for training, making this a fairly crude model. The input parameters we elected to train on were: DEHiBA percentage, total extractant concentration, U(VI) and Pu(IV) concentrations, and nitrate concentration. We see that large separation factors (>50) are predicted for a mixed DEHBA/DEHiBA system when the total DEHiBA percentage is high and at low acidity solutions. We are currently working on refining the models to improve accuracy.

On the experimental side, we have automated the collection of Th, as the surrogate for Pu, distribution ratios with either DEHBA or DEHiBA extractants in the past months. The chosen experimental conditions are at the extremes of the possible ranges for each variable, which aided in the training of the ML model. In general, we found that temperature had little effect on the distribution ratio of Th, and that DEHBA was better at extracting Th than DEHiBA, following what has been previously reported in literature and the predictions from the Gaussian Process regression. Even with our "top performing" experimental condition, the highest distribution ratio achieved was only 3.3 (experimental conditions: 16 mM Th, 5 M HNO_3, 30 mM DEHBA), indicating modest extraction. These experimental data points were used to train the ML model and predict new experimental conditions that could further improve the extraction of Th into the organic phase.

Upon training with the dataset shown in the previous report, the ML model suggested a set of 18 experimental conditions would be the most impactful in refining the ML model, and therefore help the model get closer to predicting the optimal extraction of our Pu(IV) surrogate, Th(IV). The experimental data from these ML-predicted conditions were then fed back into the ML model, marking a "closing-of-the-loop" between computation and experiment. This cycle was conducted three times, resulting in finding an experimental condition with a Th distribution ratio twice as high as the previously observed highest value, 6.6 (experimental conditions: 4 mM Th, 5 M HNO_3, 40 mM DEHBA). This is a significant accomplishment and indicates that our approach of combining an ML model with automated separations has promise to transform how separation experiments are optimized. This accomplishment marked the completion of the FY23 Milestone.

2.5 Pyro/Molten Salt Processing

[INL] Small group meetings continue to update progress on Nuclear Technology Transfer sheets strategy and CRADA approvals. It is expected that the bi-weekly small group meetings will continue for the foreseeable future. The second-generation Mechanical Cutting and Decladding System (MCADS) was received in August and a receipt inspection was completed in September. It is expected that the MCADS begin qualification testing in Q1 of FY24.

2.6 Innovative Aqueous Separations

[PNNL] The report Dissolution of Voloxidized Fuel Simulants into 1.5 M DEHiBA (PNNL-34862) was completed and issued to fulfill milestone M2FT-23PN0304020214, Recommend conditions for head-end based on voloxidation and direct fuel dissolution into DEHiBA solvent. In this work, the ability to directly dissolve voloxidized fuel simulants into 1.5 M N,N-di-(2-ethylhexyl)isobutyramide (DEHiBA) dissolved in n-dodecane was examined. The uranium species investigated included U_3O_8, UO_3 and $UO_2(NO_3)_2{\cdot}nH_2O$; the latter being a formed by decomposition of the voloxidized product $[NO][UO_2(NO_3)_3]$. The extent to which fission products, particularly the lanthanide elements, dissolve in 1.5 M DEHiBA was determined to be much less than with 1.1 M TBP which demonstrates that a preliminary separation can be achieved during the dissolution step in a DEHiBA based flowsheet. One notable exception is Re which was used as a surrogate for Tc. The added Re dissolved to a much greater extent in DEHiBA based tests than in TBP based tests, highlighting the potential need to develop effective Tc management strategies for DEHiBA based flowsheets. Based on this work, the following recommendations are made:

- Direct dissolution of $[NO][UO_3(NO_3)_3]$ in DEHiBA should not be pursued for further technology development at this time.
- Direct dissolution of epsilon-UO_3 should be further developed, including characterizing the reaction stoichiometry and mechanism by which the dissolution occurs.
- Laboratory scale dissolution of epsilon-UO_3 should be demonstrated.

- A continuous flow separation of undissolved solids from a triphasic system (two liquid phases and one solid phase) needs to be developed tangentially to process scale-up.
- A paper study should be performed to determine if the radiolytic dose that the solvent will receive will allow for a practical operation.
- A process flowsheet needs to be developed for direct uranium extraction with DEHiBA, including routing of Np and Tc to the HLW.

The report *Carbonate/peroxide processing report and recommendations* (PNNL-34812) was completed and issued to fulfill milestone M3FT-23PN0304020215. In this work, an alternate Used Nuclear Fuel (UNF) reprocessing technology using alkaline dissolution in carbonate was investigated. This technology has many advantages over the traditional nitric acid dissolution, including reduction of toxic emissions, low energy requirements, suppression of organic solvents, and conceptual ease of separation of uranium from most fission products. The main objective of this first-year research was to design a dissolver solution capable of rapidly dissolving uranium oxides and maximize U concentration at the levels which would render UNF reprocessing economical. The dissolver solution also needed to have a high tolerance to pH changes and prevent the premature precipitation of the dissolved U(VI). Several dissolver solutions were tested on UO_2, including different concentrations of sodium and ammonium carbonates. Urea and guanidine were also tested as buffers and to prevent the precipitation of the dissolved uranium. Hydrogen peroxide was used as the oxidant for UO_2 and U_3O_8 and to form complexes with uranyl ion enhancing U(VI) solubility. Its stability was tested, and its concentration optimized. Raman spectroscopy and ICP-MS were used to assess the dissolver solution quality for all forementioned parameters. The dissolver solution was developed using UO_2 as a model uranium oxide and its optimized composition of 1.25 – 1.5 M ammonium carbonate / 0.75 – 1.0 M guanidine carbonate / 2 M hydrogen peroxide solubilized 0.5 M U(VI) and maintained in solution for several days without precipitation despite fast conversion of uranyl peroxo and peroxocarbonate complexes to tris-carbonato-uranium(VI). Performance of the optimized dissolver solution was tested for dissolution of different uranium oxide materials including alpha-U_3O_8, epsilon-UO_3 and uranium oxide doped with selected fission product metals and subjected to heat treatment simulating voloxidation (SimFuels). The dissolution of epsilon-UO_3 resulted in at least 0.5 M U(VI) while dissolution of alpha-U_3O_8 achieved only about 0.2 M U(VI) solution. This was attributed to the highly crystalline nature of alpha-U_3O_8 and its preparation method requiring high temperature. The preliminary results with SimFuel samples indicate that the dissolver solution can achieve at least 0.3 – 0.45 M U(VI) concentration depending on the composition of the starting material.

[SRNL] A reactor (Figure 9) for studying the direct extraction of solid solutions of uranium containing Pu, Np, and lanthanide fission product elements in a tributyl phosphate (TBP) solvent was fabricated by the SRNL Glass Shop which completes milestone M4FT-23SR030402038. The reactor was designed to test the use of air sparging rather than mechanical stirring to mix the surrogate spent fuel and TBP solvent. The reactor is currently being tested and will subsequently be installed in a radiochemical hood where a UV-Visible spectrometer will be used to monitor the concentrations of the actinide and lanthanide elements.

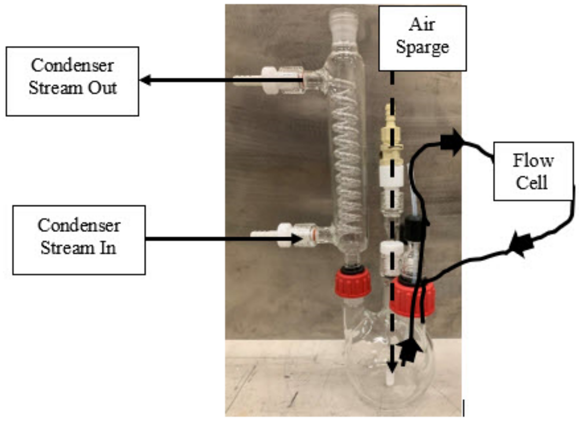


Figure 9. Reactor for Direct Extraction Experiments

A progress report for the direct extraction experiments performed with lanthanide elements during FY-23 was completed. The report completes milestone M4FT-23SR030402032 (Issue progress report on kinetic measurements for lanthanide oxide dissolution in a TBP solvent).

[ORNL] The company that is integrating the instruments into the robotics system (JKEM) is coming to the laboratory for a final walkthrough of the facility that will house the autonomous laboratory. We will show them the laboratory since they are at the stage that requires a more detailed procedure for the robotics. We will also discuss size/weight and facility requirements. We also submitted an order for an automated pipette system with shaker ability and that procurement is through and will be delivered by the end of October.

2.7 Innovative Salt Systems

[INL] In September, the research completed a manuscript on the results of the crystallization work and submitted the manuscript for publication. Experiments drew to a close on all scope and research teams focused on planning for FY24.

[ORNL] We continue to test the most recent iteration of the chlorine reference electrode. In addition, we synthesized solid solutions of UO_2 with La_2O_3, Nd_2O_3, and Ho_2O_3 at 1,300 °C. However, these mixtures were not solid solutions. They are mostly the mixtures of individual oxides. Solubility measurements of the mixtures in $MgCl_2$-KCl salt were exactly the same solubility as observed for the individual oxides. We will adjust the synthesis conditions to get the solid solutions. Finally, solid solutions of several RE elements will be synthesized to test their mutual effect on the solubility when the uranium is not present in the system.

We are submitting an allocation request for Summit supercomputer time (SummitPLUS call) for FY24 to request 150,000 node hours. A large portion of the work will be done to perform the work for the NE molten salt project, including the development of machine-learning potentials and methods to predict molten salt thermodynamic and kinetic properties.

[PNNL] An optical probe customized for different solution conditions and characterized for performance in acidic solution was demonstrated. Specifically, the aim of this work is to develop a probe that utilizes an attenuated total reflectance (ATR) approach to characterize very dark solutions. The milestone report titled "Demonstrate probe pathlength modularity to customize for different solution conditions and characterize performance in acidic solution" (document number PNNL-34779), was completed early (9/7/2023), for milestone # M4FT-23PN030401042.

[BNL] New postdoc Insung Han came up to speed on the molten salt dynamic light scattering instrumentation and has been doing experiments on zeolite nanoparticle sedimentation and redispersion in the two eutectic mixtures over a range of temperatures. Collaborators in the Dai Group are preparing a new set of samples for Insung. We are working with beam line scientists at NSLS-II to obtain preliminary SAXS data on our samples for a beam time proposal to be submitted in the next cycle. Reliable sources of monodisperse metal nanoparticles are being sought for the next round of investigations.

For more information on Material Recovery and Waste Forms Development contact Ken Marsden (208) 533-7864.

3. MPACT CAMPAIGN

3.1 Campaign Management

3.1.1 NTD & Management Support

[LANL] Federal Program Manager, Control Account Manager, Deputy, and National Technical Director worked with LANL web developers to develop the landing page and secondary pages of the MPACT website. FY24 planning efforts continued.

[BNL] Participated in weekly program review meetings with others on the management team. Reviewed M2 and M3 milestones. Worked on MPACT branding and outreach initiatives, including reviewing options for an MPACT logo and an MPACT website. Participated in outreach meeting with SHINE to determine how MPACT might support them in the future. Reviewed and evaluated NEUP pre-proposal.

3.1.2 MPACT Website Development

[LANL] Website developers have been working with the MPACT management team to structure the website. The landing and secondary pages have been drafted and the team is working on continued revisions.

3.2 Front-End Domestic Safeguards

3.2.1 Enrichment Plant

[SNL] The initial SNL gas centrifuge enrichment plant model is being adjusted using the information from Centrus. A preliminary calculation of the SEID has been performed using a simplified/reduced set of information. Transients and compressible fluid activities are neglected in the preliminary calculation.

3.2.2 Fuel Fabrication - Holdup

[ORNL] The facility level testing is being extended because battery life for the prototypes has greatly exceeded expectations. It was initially thought that the units would work for about 2 weeks but performance at NFS in facility testing is more in the 3–4-week range and still going. The current plan is to let them run as long as possible.

3.2.3 Fuel Fabrication - Standards

[ORNL] Completed development of HEPA filter waste measurement standards allowing for a safer method to measure this process waste stream.

3.3 Back-End Domestic Safeguards

3.3.1 Electrochemical & Aqueous spike-based Reprocessing NMAC

INL] FY-23 data has been analyzed. Prepared the final milestone report. The overall conclusion is that he Na-22 based radioactive tracer dilution technique is feasible for determining the total salt mass of molten salt systems for pyroprocessing.

3.3.2 FIELD TEST SUPPORT

[LANL] The updates to the HV boards under the PSMC project have been completed. The boards were shipped to INL. LANL staff traveled to INL and performed the installation and testing of the equipment. The system is operating as expected.

3.3.3 MOLTEN SALT PM/NMA

[ANL] The annual report was sent to the NTD and federal program manager in fulfillment of the end-of-year milestone. Preparations for FY24 are underway including project planning and equipment procurement tasks.

[INL] Coordinated with LANL and ANL on PM/NMA tasks/planning for next year. Single bubbler test in HFEF went well and data was collected.

[LANL] Completed report documenting assessment of data utilizing Principle Component Analysis Techniques: M3FT-23LA040103072-Report of data analysis in support of the testing of an electrochemical recycling electorefiner voltammetry probe and molten salt process monitoring.

3.3.4 REPROCESSING SAMPLER

[ANL] The end-of-year report was sent to the NTD and federal program manager in fulfillment of Milestone M2FT-23AN040103082 (Ship Molten Salt Sampler Components to INL). Since submission of the report, preparatory activities related to the FY24 sampler deployment at INL have been underway.

3.3.5 SAFEGUARDS MODELING SUPPORT

[SNL] Continuing to develop the F3M tool. Zircex report is being finalized.

3.3.6 MOLTEN SALT SAMPLING & ANALYSIS (WITH MRWFD)

[ANL] A report fulfilling the project milestones was submitted to the NTD. Updates on sampling practices will be adopted in the coming year to address many of the shortcomings identified in the report.

3.3.7 PSMC – KM200 COMMERCIALIZATION

[INL] LANL SMEs spent several days at INL refitted the PSMC with new electronic components. The refit/repair was successful and the PSMC performed well through initial tests with sources.

[LANL] Staff traveled to INL to install and test the updated HV boards. They are working as intended.

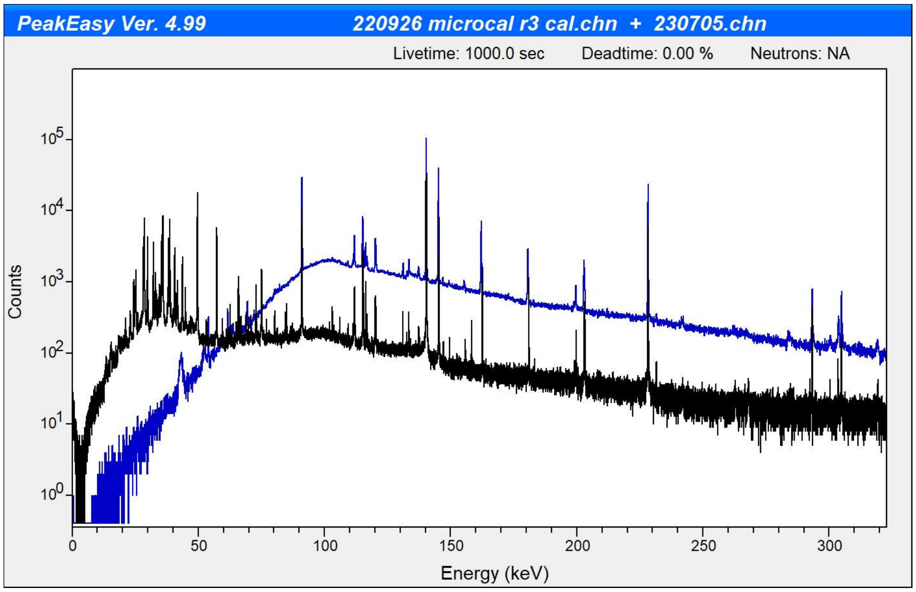
3.3.8 ECHEM SSBD (WITH MRWFD)

[ANL] The final report for this project was provided to the NTD in fulfilment of the success criterion. The report included details on electrorefiner SSBD design approaches that were developed during a series of coordinated meetings between the safeguards and pyroprocessing groups at Argonne. The content in the report will be used to inform future electrorefiner designs and operational practices.

3.3.9 MICROCALORIMETRY

[LANL] Significant progress in optimizing operating conditions for an INL detector assembly being tested at LANL has been made. A successful vibration isolation strategy based on kevlar suspension has led to notably improved energy resolution. Planning to install a similar suspension system at INL in early FY24.

[PNNL] Troubleshooting continues comparing measurement results from a fission source conducted in September of 2022 and the most recent measurements to understand differences in noise level and instrument sensitivity (see attached, black is Sept 2022 and blue is from measurements in June of 2023). Evaluation of these two measurements continues to determine the measured detection efficiency of the two different microcalorimeters SOFIA and SLEDGEHAMMER that were used for these measurements in 2022 and 2023, respectively.



3.4 Safeguards Education

3.4.1 NMAC Training

[BNL] Worked with the NMA training team to review the results of the July 2023 implementation and incorporate that experience into the next revision of the course materials. Discussed plans for a Spring 2024 NMA course implementation. Worked with the NMA training team to identify other areas where NMA training could assist industry.

3.4.2 NMAC Training & Safeguards Practitioner Development

[SNL] The Safeguards Practitioner Guidance Document was put into the correct template and awaiting further guidance on distribution.

For more information on MPACT contact Mike Browne at (505) 665-5056.

4. SYSTEMS ANALYSIS AND INTEGRATION (SA&I) CAMPAIGN

4.1 NUCLEAR ENERGY SYSTEM PERFORMANCE (NESP)

4.1.1 US ENERGY SECURITY AND SELF-SUFFICIENCY ANALYSES

[INL] Staff continued drafting the report on technology road mapping for the fuel cycle facilities needed to support the ARDP concepts. The draft was sent out for review and reviewer comments returned. Staff started working to address the review comments.

[ORNL] Collaborated with ANL to develop a report on the HALEU 1 task to address shortage options required for enriched uranium for large-scale reactor deployment. In addition to this report, ORNL led and delivered a report on a second HALEU task that analyzed HALEU facility capacities and production rates needed for projected nuclear growth out to mid-century for different reactor deployment and energy projection scenarios. The two reports are being revised to incorporate feedback from the federal program manager. A third report on the impacts of energy price and demand on US nuclear fuel cycle capacity sufficiency was led by ORNL and this report is also currently being revised to incorporate an additional section based on feedback from the NTD and Deputy NTD.

4.1.2 NUCLEAR ENERGY CONTRIBUTIONS TO ADDRESS CLIMATE CHANGE AND DECARBONIZATION

[**ANL, BNL, INL**] Submitted the report entitled “Potential of Nuclear Energy to Mitigate Severe Weather-induced Energy Stresses” by N. Mann, et al. This report is the deliverable in the fulfillment of the Level 3 milestone, M3FT-23AN120102014 under the work package of “FT-23AN12010201 - Nuclear Energy System Performance (NESP) - ANL.” The following is the summary of the report,

- The potential impacts of climate change on future power grid dispatch were simulated using the ERCOT grid and future capacity and weather scenarios with hazardous events. Even with large shares of wind and solar PV, adequate clean-firm capacity (like from nuclear power plants) and dispatchable capacity were shown to limit unserved energy (a measure of scarcity) during periods of low wind speed, low solar irradiance, and/or when temperatures were very high or low. Long-term planning models should be improved by incorporating simulated future hazardous weather events alongside more reliability metrics to avoid underinvesting in clean-firm capacity.

[**ANL, INL**] The DOE-NE blog on the report “Assessment of Nuclear Energy to Support Negative Emission Technologies” was released. Several news articles in the report were developed (ANS NewsWire, CNBC, Politico, etc.).

[**ANL, INL**] Staff updated C2N guidebook contribution following feedback from federal manager and submitted chapters to lead PI. Two members of the C2N team visited the Coronado coal power plant in St. Johns, Arizona. The visit was a useful opportunity to validate assumptions about infrastructure at coal power plants that are part of the C2N modeling effort. The two staff attended a community meeting to learn concerns on coal power plant closures and considered repurposing options. Additionally, staff drafted the milestone report for coal-to-nuclear analysis, which is the stakeholder guidebook with scheduled delivery Oct 15, 2023. This entailed completing the section on socioeconomics, workforce transition, and aspects of policy relevant to coal transitions.

[**BNL**] Finalized BNL contribution to NESP#7.

4.1.3 ASSESSMENT OF EMERGING NUCLEAR ENERGY TECHNOLOGIES

[SNL] For Task #10, the issue identified in the previous month regarding the speed at which the database was returning information was identified but no solution has yet been found. In addition, the rejection of an invitation to a DOE colleague to access the Nuclear Fuel Cycle Options Catalog identified a problem with the system that allows non-SNL personnel to access the catalog. The issue has not yet been resolved but a workaround has been implemented and communicated to the DOE colleague.

4.1.4 QUICK TURN-AROUND STUDIES

[ANL] Estimated re-enrichment requirements for a gas-cooled microreactor (GCMR) using the recovered uranium from the used nuclear fuel. Ran several SERPENT simulations to evaluate uranium vector at EOC, and enrichment requirements in a 2ndloading after re-enrichment of 1st core, using natural uranium as external feed.

[ANL, BNL, INL, ORNL] Evaluated the values of recovered uranium (RU) from high assay low-enriched uranium (HALEU) used nuclear fuels. It was observed that RU having a residual U-235 content higher than ~7% would cost less than the fresh low-enriched uranium, and the affordability increases as the residual U-235 content in RU increases.

[BNL] Performed Serpent calculations to support the report on Value of Recovered Uranium from HALEU UNF: calculated mixing ratios and equivalent LEU enrichments for Design-A Heat Pipe Reactor, Holos-Quad, PBMR-400, SFR-100, SFR-150.

4.1.5 COLLABORATION WITH OTHER CAMPAIGNS AND PROGRAMS FOR NUCLEAR PERFORMANCE

[INL] Staff worked on drafting the report on R&D maturation for silver mordenite in iodine capture.

4.2 ECONOMIC AND MARKET ANALYSIS FOR NUCLEAR ENERGY SYSTEMS (EMANES)

4.2.1 NUCLEAR COST MODELING AND TECHNO-ECONOMIC ANALYSIS

[INL, SNL] Staff started drafting the update report which documents the annual updates of the Advanced Fuel Cycle Cost Basis Report. This year's report will include updates to address inflation and escalation, as well as publication of the cost modules on fuel fabrication. Additionally, a request was made to a colleague to perform a technical review of the draft report that examines areas of storage, transport, and disposal of HALEU spent nuclear fuel that could affect the economics of these activities, compared to storing, transporting, and disposing of LEU spent nuclear fuel generated by a typical LWR. The technical review was completed, and comments are in the process of being addressed. The intent is to include this information in the next update of the Advanced Fuel Cycle Cost Basis report.

4.2.2 COLLABORATION WITH OTHER CAMPAIGNS AND PROGRAMS FOR NUCLEAR ECONOMICS

[INL] Staff completed the M4 milestone entitled "Cost-Benefit Assessment of Krypton and Xenon Recovery from Aqueous Reprocessing" then delivered the milestone to the NTDs of the SA&I and the MRWFD Campaigns.

- This report provides an understanding of the costs and benefits of capturing krypton (Kr) and xenon (Xe) from the existing process of aqueous reprocessing of used nuclear fuel (UNF). The market assessment shows that the average price of Xe is about $60/L while the cost assessment finds the unit cost of Xe to range from $71.50/L to $131.13/L. Kr capture is a regulatory requirement for aqueous reprocessing, meaning that the costs are "sunk costs." This research estimates a range of potential capture costs from $830/L to $1,523/L. Because these costs are sunk, unless there are any additional costs associated with the sale of Kr, it can be sold for additional revenue. Xe capture, however, requires additional capital expenditures and, as such, is not a sunk cost. While the expected costs of capture and storage of Xe are slightly higher than current market prices, the market experiences volatility and changes that make this opportunity valid for further research. Specifically, an investigation of the expected output purity of Kr and Xe from UNF aqueous reprocessing is warranted due to this factor's impact on the sale price point.

[INL] Staff addressed comments from collaborators in the MRWFD Campaign on the M4 evaluating the cost-benefit of capturing Kr/Xe. Then staff finalized the report and submitted it.

[ORNL] Participated in SAI/IWM collaboration meetings and started documenting the report for the deliverable.

4.3 SA&I Advanced Nuclear Fuel Availability (ANFA) Support

4.3.1 SA&I HALEU Support

[**ANL, ORNL**] Revisited the HALEU task #1 and task #2 reports and updated report outlines to avoid the overlaps between the two reports.

For more information on the Systems Analysis and Integration contact Brent Dixon (208) 526-4928.