Project Title: Advanced Handling and Safety Enhancement Studies R1

Project Description and Purpose:

Revision 1

The Idaho National Laboratory (INL) and the Republic of Korea (ROK) Korea Atomic Energy Research Institute (KAERI) both have active research programs in the utilization of civilian nuclear energy and the nuclear fuel cycle. This revision includes Phase 2 of the project, which is focused on nuclear safety enhancements through fundamental and lab-scale studies on molten salts and used nuclear fuel (UNF). Phase 2 will be divided into three coordinated efforts: (A) high-temperature molten salt behavior, (B) molten salt spill accident simulations, and (C) high-temperature UNF behavior. Across these efforts, the work will quantify and characterize volatilization, gas-phase release, aerosol formation, and subsequent transport and capture mechanisms to inform key source term parameters used in nuclear safety analyses, including airborne release fraction, respirable fraction, and, where applicable, leak path factor or decontamination factor.

Experimental activities will be conducted using multiple furnace systems. Salt and fuel samples will be heated in a simultaneous thermal analyzer (STA) located in the MSTEC hot cell at Fuel Cycle Facility. Additional experiments will utilize the SNIFF furnace system and remote distillation furnace in Hot Fuel Examination Facility (HFEF) to heat materials and collect volatilized species. A new experimental device will be designed and deployed in HFEF to enable spill configuration testing. Resulting samples will be analyzed at Analytical Reseach Laboratories (ARL) or Irradiated Materials Characterization Laboratory (IMCL) to support characterization and model development at the Materials Fuel Complex (MFC).

Phase 2A: Molten Salt at High Temperature

In Phase 2A fission product-bearing salts will be heated at varying scales with an aim of understanding the physical and chemical states of nuclear materials in molten salts (dissolved, suspended, or particulate forms) and the generation, transport, and capture of volatile radionuclides.

Task 2A1: Develop Phase 2A Experimental Plan

The first step in this phase will be to jointly develop a detailed experimental plan. This will include establishment of test scopes and matrices distinguishing normal and off-normal operating conditions for differing experimental environments and conceptual design of hot-cell-compatible experimental systems, considering radiological containment, remote operability, and measurement requirements. Experimental parameters may include temperature, pressure, and sweep gas flow.

Task 2A2: Perform fundamental volatilization measurements on molten salts

Thermophysical property instrumentation such as an STA will be utilized to evaluate mass loss as a function of temperature as well as molten salt stability at various temperature hold points. Additionally, isotopic composition will be analyzed of volatilized species from the salts. Information from these basic tests will inform the lab-scale investigations in the next tasks.

Task 2A3: Design hardware associated with lab-scale salt experiments

In this task, any hot cell furnace ancillary hardware required for holding samples, capturing aerosols, and performing off-gas analysis will be designed and implemented. This task includes any fabrication, testing, or qualification that may be required prior to implementing the newly designed parts.

Task 2A4: Perform lab-scale salt volatilization experiments

Existing hot cell furnaces will be utilized to perform volatilization experiments on molten salts. Such experiments can include using the (1) new HFEF SNIFF furnace that has filtration and off-gas analysis features, (2) remote distillation furnaces with vacuum capability, or (3) one of two multi-purpose furnaces.

Task 2A5: Perform sampling and analysis

Sampling and analysis may include ICP MS and OES on salt samples take before and after volatilization, SEM particle size analysis of captured particles/aerosols, gas mass spectroscopy and gamma spectroscopy on off-gases, ICP MS and OES on captured particulate/aerosols from filter media, and gravimetric analysis of samples before and after experiments to support interpretation of radionuclide release behavior and aerosol characteristics relevant to nuclear safety analysis.

Phase 2B: Molten Salt Spills

Molten salt spills are a representative accident scenario considered in nuclear safety analysis, often associated with equipment failure, handling errors, or seismic events. The purpose of this phase is to analyze actual releases under simulated spill conditions using an experimental setup.

Task 2B1: Develop design for molten salt spill apparatus

This phase will begin with experimental hardware design. Design of a hot cell spill system with integrated aerosol and off-gas capture features is potentially too complex and expensive for the size and duration of this agreement. As such, the project team will quickly approach this task with a target of defining a simple and creative approach to generating data that would be valuable to informing nuclear safety approaches. This activity will serve as a go-no-go gate for determining if further work should be pursued in this phase.

Task 2B2: Develop Phase 2B experimental plan

The experimental plan will be based on the extent of the spill fixture design and what data can be obtained. Experimental parameters may include drop height, sweep gas flow, and time depenardent aerosol generation after a spill.

Task 2B3: Perform molten salt spill experiments and analysis

The new experimental setup will be utilized to run a series of experiments that will provide data that can be used to calculate airborne release fractions and subsequent respirable fractions. This task will also include all analysis required on samples collected during the experiments.

Phase 2C: UNF at High Temperatures

In Phase 2C UNF will be heated at varying scales with an aim of understanding the physical and chemical states of nuclear materials and the generation, transport, and capture of volatile radionuclides.

Task 2C1: Develop Phase 2C experimental plan

The first step in this phase will be to jointly develop a detailed experimental plan. This will include establishment of test scopes and matrices distinguishing normal and off-normal operating conditions for differing experimental environments and conceptual design of hot-cell-compatible experimental systems, considering radiological containment, remote operability, and measurement requirements.

Task 2C2: Perform fundamental volatilization measurements on UNF

Thermophysical property instrumentation such as an STA will be utilized to evaluate mass loss as a function of temperature as well as UNF stability at various temperature hold points. Additionally, isotopic composition will be analyzed of volatilized species from UNF. Information from these basic tests will inform the lab-scale investigations in the next tasks.

Task 2C3: Design hardware associated with UNF experiments

In this task, any hot cell furnace ancillary hardware required for holding UNF, capturing aerosols, and performing off-gas analysis will be performed. This task includes any fabrication, testing, or qualification that may be required prior to implementing the newly designed parts.

Task 2C4: Perform lab-scale UNF volatilization experiments

Existing hot cell furnaces will be utilized to perform volatilization experiments on UNF. Such experiments can include using the (1) new HFEF SNIFF furnace that has filtration and off-gas analysis features, (2) remote distillation furnaces with vacuum capability, or (3) one of two multi-purpose furnaces.

Task 2C5: Perform sampling and analysis

Sampling and analysis may include ICP MS and OES on UNF samples take before and after volatilization, SEM particle size analysis of captured particles/aerosols, gas mass spectroscopy and gamma spectroscopy on off-gases, ICP MS and OES on captured particulate/aerosols from filter media, and gravimetric analysis of samples before and after experiments.

This project will generate a small amount of waste over its duration. The estimates include the following:

- Less than 1 cubic foot of industrial waste,
- Less than 1 cubic foot of PPE, wipes, and othe generic lab waste
- Less than 1 cubic foot of radioactive waste
- Less than 1 cubic foot of low-level waste. In addition,
- Less than 0.1 cubic foot of transuranic (TRU) waste will be produced.

All waste will be handled, processed, and disposed of in accordance with facility-specific requirements, with support from Waste Generator Services (WGS).

Samples will be taken during this project, but will be processed as waste at the ARL or IMCL at MFC.

Mass spectrometry equipment will be purchased to complete this work. After the project is completed, the equipment will be kept for future use.

Original

The Idaho National Laboratory (INL) and the Republic of Korea (ROK) Korea Atomic Energy Research Institute (KAERI) both have active research programs in the utilization of civilian nuclear energy and the nuclear fuel cycle. In this proposed collaboration, INL and KAERI will work together to further understand the voloxidation process with a focus on stabilization of damaged used nuclear fuel.

The proposed study between INL and KAERI is four years in duration and is divided into three phases with an overall emphasis and priority on voloxidation technology. All three phases include a range of activities and each phase has an area of emphasis. The phases overlap each other in duration in order to compress the study schedule. Phase A has a planned duration of two years and is focused on demonstration of voloxidation and stabilization on simulated fuel (simfuel) and development of related processes and hardware. Phase B is concurrent in duration with an emphasis on safeguards analysis and the understanding of material-unaccounted-for during key process steps. Phase C is two years in duration and is focused on validation of the previous phases through demonstrations of voloxidation and stabilization with used nuclear fuel in a hot cell facility.

The proposed four-year joint study focuses on exploring advanced technologies, particularly voloxidation, to eliminate safety concerns regarding used fuel management in highly radioactive environments. The collaborative research includes development of technologies for enhancement of safety during handling of damaged used nuclear fuels. Particular focus will be on process development for the initial step in stabilization of damaged used fuels by using a voloxidation process for oxidative decladding as an alternative to mechanical processes. All work will be performed at the Material and Fuels Complex. Expected facility utilization will include the Hot Fuel Examination Facility (HFEF), the Fuels and Applied

Sciences Building (FASB), Analytical Laboratory (AL), Engineering Design Laboratory (EDL), Fuel Manufacturing Facility (FMF) and the Experimental Fuels Facility (EFF).

The following subsections describe the proposed activities in detail.

Phase A: Demonstration of Voloxidation on Simulated Fuel

This task covers the development of key processes for damaged used fuel stabilization. The key processes include the following: (1) fuel material recovery, (2) fabrication of stabilized pellets, and (3) fabrication of fuel rods as a final product.

- Fuel Material Recovery:

For a damaged fuel rod, it is difficult to recover UO2 pellets using existing mechanical decladding methods, and thus an alternative strategy is desired. Simulated damaged fuel rods will be fabricated using depleted uranium and be used for a fuel material recovery test. The operating principle of the fuel material recovery equipment will be based on voloxidation, or oxidative decladding (500°C, oxidizing atmosphere). This process results in oxidation of UO2 to U3O8. An appropriate vibration method will be considered during thermal treatment for detaching fuel from the cladding. After fuel material recovery, oxidation-reduction treatment of recovered powder will be repeatedly carried out to obtain compactable and sinterable UO2 powder. Milling of oxidationreduction treated powder will be an optional step to prepare a more compactable and sinterable powder.

- Fabrication of Stablized Pellets:

This key process deals with mixing the recovered material with lubricants, pressing the powder mixture to produce green pellets, and sintering the green pellets to fabricate stabilized pellets. The lubricants, compacting pressure, and sintering temperature are major factors affecting stability of the final pellets.

- Fabrication of Fuel Rods:

In this key process, fuel rods will be fabricated with the stabilized pellets. Gas tungsten arc welding (GTAW) will be performed to weld fuel rods. The performance of the fabricated fuel rods will be evaluated by helium or pressure decay leak testing and radiography.

Task A1: Performance evaluation of key processes with simfuel

KAERI will investigate the performance of equipment for the fuel material recovery using simulated damaged fuel rods with a length of 10–20 cm. The fuel material recovery tests with the simulated damaged fuel rods will be carried out with an oxidizing atmosphere at 450–500°C in flowing air. In this step, fuel
material recovery efficiency can be estimated by measuring weight before and after oxidation and chemical analysis of cladding residues.

For fabrication of stabilized pellets, the compactable and sinterable UO2 powder will be prepared by repeating oxidation and reduction. In addition, the uranium powders will be mixed with a lubricant (Acrawax powder), and the mixed powder will be pressed into green pellets. Then, the green pellets will be sintered at about 1700°C in an atmosphere of 4%H2/Ar. The properties of the stabilized pellets will be evaluated by measuring density, compaction strength, and leachability.

Fuel rod fabrication that includes remote end-cap welding technology using GTAW will be performed by INL. Equipment will be prepared for preliminary tests. The performance of equipment will be evaluated by welding of cladding and end caps. Optimum welding conditions will be determined based on microstructure observations and hardness tests on the weldments as funding allows. The fabrication of simfuel rods of 35 cm (~14 inches) in length and between 0.635 cm (0.25 inches) and 1.27 cm (0.5 inches) in diameter will be targeted. In this step, ceria (CeO2) or depleted uranium dioxide (DUO2) pellets will be used as stabilized simfuel pellets and loaded into fresh cladding tubes such as Zircaloy-4 or ZIRLO. Unlike commercial nuclear fuel rod fabrication, the fuel rods for the stabilization process do not need components such as plenum springs and axial blankets. The performance criteria for the fabricated fuel rods will be evaluated by helium or pressure leak testing and radiography.

Task A2: Derivation of improvements of key processes using simfuel

The design for improved remote operability of fuel material recovery and pelletizing equipment will be performed by KAERI. For fuel rod fabrication, the design including improved remote operability will be performed by INL.

Task A3: Optimization of operational conditions for key processes using simfuel

The optimized operating conditions for fuel material recovery and stabilized pellet fabrication will be derived by complementary tests. The performance of these processes and characteristic data for the stabilized pellets will be reported by KAERI. For the fuel rod fabrication process implemented at INL, the optimized operating conditions will be derived by complementary tests at INL. The performance of these processes and characteristic data for the fuel rod fabrication will be reported by INL. These optimized conditions will be applied to the verification tests in Task C2 and C3.

Task A4: Design of key process equipment for hot cell test

In this task, key process equipment (which may include fuel material recovery, stabilized pellet fabrication, and fuel rod fabrication) will be designed in a collaboration between INL and KAERI for hot cell operation.

Task A5: Manufacture of key process equipment for hot cell test

Key process equipment (which may include fuel material recovery, stabilized pellet fabrication, and fuel rod fabrication) for HFEF hot cell operation will be manufactured by INL as funding allows. Existing INL hot cell capabilities will be utilized to the maximum extent possible in lieu of newly manufactured equipment.

Phase B: Safegaurds Analysis -

The Safeguards-By-Design (SBD) concept recommends preemptive consideration for points or regions for safeguards measurement where access is limited such as in a hot cell. It is necessary to minimize Material Unaccounted For (MUF) during these key processes and to evaluate material balances in bulk handing facilities for safeguards effectiveness.

This task will focus on safeguards analysis for the damaged used fuel stabilization process, especially on MUF characteristics and material balances. During the damaged fuel stabilization process with simfuel, MUF (hold-up and process losses) will be measured and analyzed to investigate the origin of MUF occurrence. This activity can be performed by associating the simfuel test at KAERI with the hot cell tests at INL.

Based on the analysis of the origins of MUF occurrence, plans for MUF analysis and material balance evaluations in the hot cell operations will be established. Safeguards recommendations for damaged fuel stabilization processes will be derived and these will reflect analysis of the hot cell verification test. The safeguards recommendations will contain (1) methods to reduce MUF including equipment design and process procedures, (2) material balance evaluation techniques, and (3) safeguards measures such as in-cell radiation detectors for hold-up verification.

Task B1: Analysis of MUF origins and establishment of hot cell verification plan

A hot cell verification test plan will be established for MUF analysis and material balance evaluation for the stabilization process taking consideration for hot cell conditions by INL. The plan will include radiation measurement activities with in-cell radiation detectors for hold-up verification.

Task B2: Measurement and analysis of MUF in damaged fuel stabilization

Measurement of MUF (hold-up and process losses) will be conducted during the damaged fuel stabilization process in an INL hot cell. Mass of nuclear material in each unit process will be tracked to determine MUF. Taking samples for destructive analysis (DA) and non-destructive analysis (NDA) can be included for material balance evaluation, as well as in-cell radiation detection of the process equipment to verify hold-up and process losses.

Task B3: Overall evaluation of MUF in damaged fuel stabilization process

Task B3 will be focused on evaluation of MUF characteristics and material balance evaluation from the verification test results for the damaged fuel stabilization process. A final evaluation report by INL will include MUF characteristics, material balance evaluations, and safeguards recommendations on the damaged fuel stabilization in terms of Safeguards-By-Design (SBD).

Phase C: Validation Tests with Used Nuclear Fuel

In this task, using LWR used fuel, stabilized pellets and fuel rods as the final products of a stabilization effort will be fabricated in an INL hot cell. The performance data of at least one of three key processes and the characteristics of any stabilized products will be evaluated.

Task C1: Qualification tests for the manufactured equipment -

The qualification tests for key process equipment (which may include fuel material recovery, stabilized pellet fabrication, and fuel rod fabrication) will be carried out at INL to demonstrate fabricated equipment readiness to be operated in a hot cell environment. A management self-assessment (MSA) may be required following the qualification tests if mandated by the nuclear facility.

Task C2: Recovery verification test with used fuel

After completion of the qualification testing and/or MSA, the verification test with the hot cell process equipment will be implemented with used fuel stored at the INL. The fuel material recovery (voloxidation) will be verified using the optimized operational conditions. The fuel material recovery efficiency will be analyzed by measuring weight and chemical analysis of cladding residues.

Task C3: Fabrication of stabilized pellets and fuel rods using spent fuel

This task requires the completion of Task C2 since the product of Task C2 becomes feed material for the fabrication of stabilized pellets. The voloxidation step will be prioritized for design, fabrication, and hot cell testing. If demonstrated, the stability of fabricated pellets will be evaluated by measuring the density of the pellet and the compaction strength at INL. Moreover, the leachability of the fabricated pellet will be measured, and it will be compared to that of used fuel pellets.

If funding allows for fuel rod fabrication, the fuel rod fabrication process will be verified at INL using the stabilized fuel pellets under optimized operating conditions. The prototype fuel rods will be of ~35 cm (~14 inches) in length and between 0.635 cm (0.25 inches) and 1.27 cm (0.5 inches) in diameter. The performance of the fabricated fuel rod will be evaluated by measuring helium or pressure leak rate and radiography, if radiography is feasible based on radiation levels.

Task C4: Evaluation of verification tests and derivation of improvements in damaged fuel stabilization

Based on the results of Task C2 and C3, improvements for the equipment and associated operating conditions will be reported for future use by INL. At this time the only expected equipment purchases are welders and ancillary hardware for existing furnaces.

Environmental Aspects or Potential Sources of Impact:

Air Emissions

Experiment irradiation and PIE will be performed at the ATR and MFC facilities. Air emissions would include minor amounts of radionuclides and toxic air pollutants. The irradiation in the ATR is not a modification in accordance with Idaho Administrative Procedures Act (IDAPA)

58.01.01.201 and 40 Code of Federal Regulation (CFR) 61 Subpart H. ATR radionuclide emissions are sampled and reported in accordance with Laboratory Wide Procedure (LWP)-8000 and 40 CFR 61 Subpart H.

The irradiated specimens will be delivered to the MFC HFEF for disassembly and then undergo routine PIE before being sent to the AL for analysis. The PIE examination in HFEF and the analysis completed in the Analytical Lab is not a modification in accordance with Idaho Administrative Procedures Act

(IDAPA) 58.01.01.201 and 40 Code of Federal Regulation (CFR) 61 Subpart H.

Packaging in HFEF is not a modification in accordance with Idaho Administrative Procedures Act (IDAPA) 58.01.01.201 and 40 Code of Federal Regulation (CFR) 61 Subpart H.

Releases of radioactive airborne contaminants from the proposed action are not expected to increase to the annual dose to the Maximum Exposed Individual (MEI).

Discharging to Surface-, Storm-, or Ground Water

NA

Disturbing Cultural or Biological Resources

CULTURAL RESOURCES: Pursuant to the 2023 Programmatic Agreement as amended in 2025, the proposed action does not meet the threshold of a federal undertaking with the potential to affect historic properties and will have no effect to historic properties.

Generating and Managing Waste

Rev 2: At INL facilities, activities may generate industrial, hazardous, radioactive, and mixed waste. The BEA Waste Management Program's (WMP) Waste Generation Services (WGS) subcontractor assists projects in characterizing and managing waste. WGS brokers handling and shipping of hazardous and radioactive waste.

Releasing Contaminants

When chemicals are used during the project there is the potential for spills that could impact the environment (air, water, soil).

Using, Reusing, and Conserving Natural Resources

All materials will be reused and recycled where economically practicable. All applicable material will be diverted from disposal in the landfill where conditions allow.

Determination:

References:
10 CFR 1021, Appendix B to subpart D, items B3.6, "Small-scale research and development, laboratory operations, and pilot projects"

Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement and Record of Decision (DOE/EIS-0203, 1995) and supplemental analyses (DOE/EIS-0203-SA-01 and DOE/EIS- 0203-SA-02) and the Amended Record of Decision (1996)

Final Environmental Impact Statement for the Waste Isolation Pilot Plant (DOE/EIS-0026, October 1980) and Final Supplement Environmental Impact Statement for the Waste Isolation Pilot Plant (SEIS-I) (DOE/EIS-0026-FS, January 1990)

Final Waste Management Programmatic Environmental Impact Statement [WM PEIS] (DOE/EIS-0200-F, May 1997) and Waste Isolation Plant Disposal Phase Supplemental EIS (SEIS-II) (DOE/EIS-0026-S-2, September 1997)

Final Site-Wide Environmental Impact Statement for the Continued Operation of the Department of Energy/National Nuclear Security Administration Nevada National Security Site and Off-Site Locations in the State of Nevada (DOE/EIS-0426, December 2014).

Justification: The proposed R&D activities are consistent with CX B3.6 "Siting, construction, modification, operation, and decommissioning of facilities for small-scale research and development projects; conventional laboratory operations (such as preparation of chemical standards and sample analysis); small-scale pilot projects (generally less than 2 years) frequently conducted to verify a concept before demonstration actions, provided that construction or modification would be within or contiguous to a previously disturbed area (where active utilities and currently used roads are readily accessible). Not included in this category are demonstration actions, meaning actions that are undertaken at a scale to show whether a technology would be viable on a larger scale and suitable for commercial deployment."

NEPA coverage for the transportation and disposal of waste to WIPP are found in the Final Waste Management Programmatic Environmental Impact Statement [WM PEIS] (DOE/EIS-0200-F, May 1997) and Waste Isolation Plant Disposal Phase Supplemental EIS (SEIS-II) (DOE/EIS-0026-S-2, Sept. 1997), respectively. The 1990 ROD also stated that a more detailed analysis of the impacts of processing and handling TRU waste at the generator-storage facilities would be conducted. The Department has analyzed TRU waste management activities in the Final Waste Management Programmatic Environmental Impact Statement (WM PEIS) (DOE /EIS-200-F, May 1997). The WM PEIS analyzes environmental impacts at the potential locations of treatment and storage sites for TRU waste; SEIS-II addresses impacts associated with alternative treatment methods, the disposal of TRU waste at WIPP and alternatives to that disposal, and the transportation to WIPP.

The environmental impacts of transferring LLW from the INL Site to the Nevada National Security Site were analyzed in the 2014 Final Site-Wide Environmental Impact Statement for the Continued Operation of the Department of Energy/National Nuclear Security Administration Nevada National Security Site and Off-Site Locations in the State of Nevada (DOE/EIS-0426) and DOE's Waste Management Programmatic EIS (DOE/EIS-200). The fourth Record of Decision (ROD) (65 FR 10061, February 25, 2000) for DOE’s Waste Management Programmatic EIS established the Nevada National Security Site as one of two regional LLW and MLLW disposal sites.

Approved by Robert Douglas Herzog, DOE-ID NEPA Compliance Officer on: 7/1/2026