

DOE-ID NEPA CX DETERMINATION

Idaho National Laboratory

SECTION A. Project Title: Transient Reactor Test Facility (TREAT) Low-Enriched Uranium (LEU) Fuel Conversion Project

SECTION B. Project Description and Purpose:

The Transient Reactor Test (TREAT) Facility Low-Enriched Uranium (LEU) Fuel Conversion Project proposes to develop and qualify LEU fuel for the TREAT reactor. The TREAT reactor conversion project is managed by National Nuclear Security Administration's (NNSA's) Office of Materials Management and Minimization (MMM) that works to develop and qualify new fuels and technologies to support conversion efforts domestically and abroad.

In instances where suitable LEU fuels do not exist for particular reactors to convert, the Convert Program contributes to the development of new LEU fuels. As no suitable LEU fuel is currently available with which the TREAT reactor could convert, the proposed action is to research and develop a new LEU fuel and fabrication capability to allow for conversion. The proposed action uses existing facilities at the Idaho National Laboratory's (INL's) Materials and Fuels Complex (MFC) and TREAT to support research and development (R&D) activities in support of LEU fuel development for the TREAT reactor.

To determine the feasibility of converting the TREAT reactor from HEU to LEU fuel, LEU fuel element assemblies and a fabrication process for those assemblies need to be designed, engineered, developed, and tested. This environmental checklist (EC) covers fuel engineering and design feasibility studies and planning (including both fuel design and fuel fabrication) and LEU fuel design and testing needed to make such a determination.

Activities for the TREAT conversion project have been organized into four principle activities: (1) Fuel (LEU) Design, (2) Fuel Fabrication, (3) Fuel Qualification, and (4) Facility Conversion. Because an LEU fuel for the TREAT reactor has not been designed, tested, or developed, fuel production and the physical and operational changes to convert TREAT once a fuel is qualified are beyond the scope of this EC. If conversion of the TREAT reactor is determined to be feasible, and an LEU fuel is qualified, additional analysis in compliance with the National Environmental Policy Act (NEPA) will be performed to evaluate the physical and operational changes needed for full conversion and to disclose the environmental impacts associated with conversion.

The scope of this EC includes 1) fuel engineering and design feasibility studies and planning (including both fuel design and fuel fabrication), 2) modifications to building MFC-723, 3) LEU fuel design and testing, and 4) evaluating the performance of candidate LEU fuel element designs.

Tests would be developed for each fuel subsystem and would encompass three categories--out-of-pile testing, analysis and characterization; irradiation testing; and post-irradiation examination (PIE). Test articles would include small samples for material properties, small mock-up test assemblies, and full-scale fuel element assemblies.

Materials performance testing would be performed on samples rather than complete fuel elements. These tests would be brief steady-state irradiation using the hydraulic shuttle irradiation system "rabbit" at the Advanced Test Reactor (ATR). Full-scale element assemblies would undergo transient testing in the TREAT reactor.

Activities associated with designing, engineering, developing, and testing LEU fuel element assemblies and fabrication processes are described below:

LEU Fuel Design

LEU fuel element design options would be selected from trade studies and matured through more detailed design phases until the best option becomes clearly identifiable. Neutronics, thermal-hydraulics, and structural analysis of design concepts would be performed by Argonne National Laboratory (ANL) and Idaho National Laboratory (INL). Driving accident scenarios would also be analyzed. As the fuel element design matures, drawings and specifications and other system design documents and fabrication specifications would be drafted and released. Periodic design reviews would be held to evaluate adequacy of the LEU fuel element design and the ability to fabricate the fuel element assembly.

Fuel Fabrication

Fuel element fabrication activities include fuel block scale-up studies and associated materials characterization. Initial fabrication would use surrogate fuel material (ZrO₂) and UO₂. INL and Los Alamos National Laboratory (LANL) would perform characterization studies to evaluate fuel blocks to verify performance requirements. A fabrication development plan would be generated for manufacturing the Design and Demonstration Fuel Element (DDE), Lead Test Assemblies (LTA), and LEU core based on the results of the scale-up study. The DDE and LTAs would be full size fuel element assemblies and consist of 12 fuel blocks (UO₂ mixed in graphite blocks) surrounded by a single zircaloy cladding with a Zr-4 alloy top end fitting and a Zr-4 alloy bottom fitting assembly.

These DDE and LTA assemblies are the initial set of fuel element assemblies to test LEU design features prior to batch fabrication of 500 LEU core fuel elements. LTAs are irradiated to obtain test data and to confirm successful performance in the TREAT reactor environment. It is anticipated that if modifications are identified from DDE/LTA testing, only minor changes would be made to the LEU fuel element assembly design and specification used in fabricating the LEU core fuel element assemblies.

Several mockup fuel elements would be fabricated to demonstrate the manufacturability of cladding design and fuel element assembly with fuel blocks and end caps. BWX Technologies Inc. (BWXT) in Lynchburg, VA would fabricate the fuel blocks and ship them to INL for fuel element assembly. Suppliers of cladding and end caps would likewise ship these components for assembly.

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MFC-723 is the favorable location to perform assembly of TREAT test articles. The assembly process would include fuel block cleaning via ultrasonic cleaning bath, fuel block outgassing, and electron beam welding of cladding. Fuel blocks would be placed into the cladding, and end caps would be welded to the cladding to seal the fuel blocks inside the assembly. X-ray inspection, X-ray tomography, and a leak check of the fuel element assembly along with final quality assurance and control checks would be completed.

Materials Testing

Fuel Block

Fuel block material testing activities include qualification testing, irradiation testing and post irradiation examination (PIE), and fabrication studies. Specific activities related to fuel block material testing are described below:

Out-of-Pile Testing—Out-of-pile testing would consist of characterization tests on unirradiated fuel and would be conducted over a temperature range to include normal operating conditions and accident conditions.

Irradiation Testing and PIE—Testing in a radiation environment would be conducted on small samples to determine the effects on fuel block material behavior from exposure to neutron flux and temperature. Specimen cross sections would be examined at the microscopic scale to evaluate changes in microstructural features under irradiation, including material degradation due to fission or fluence damage, irradiation creep of materials, cross-sectional swelling profiles, and fission gas bubble morphology.

Fuel Block Fabrication—TREAT's original core was designed and fabricated from 4-in x 4-in blocks while the upgrade core was designed and fabricated from smaller blocks arranged in a 4-in x 4-in array. For TREAT conversion, fuel block scale-up studies need to make provision for both block arrays to be implemented. Fuel blocks for testing would be made using the full manufacturing process.

The fuel blocks would consist of three major components: graphite, carbon, and UO₂. Trace elemental analysis of the fuel blocks would be performed by inductively coupled mass spectrometry. X-ray diffraction of samples would be performed to verify crystallinity of the graphite. An electron microprobe would measure stoichiometry of the UO₂. To examine the material interfaces for evidence of chemical interactions, scanning electron microscopy with energy dispersive spectroscopy would verify elemental analysis.

Thermal diffusivity of the samples would be measured by laser flash analysis at temperatures ranging from room temperature to 900°C on samples machined to be 12.7-mm in diameter by 3-mm thick. A minimum of three samples per block would be required.

Outgassing of fuel blocks would be determined by heat treatment of a single 1-cm diameter by 1-cm tall sample at temperatures up to 900°C in a thermal gravimetric analyzer with mass spectrometry.

Samples for immersion density measurements using the Archimedes method would be machined to 1 x 1 x 1-in. cubes from five randomly sampled fuel blocks.

In order to measure particle and defect size and spacing using three-dimensional tomography, samples would be machined from 10 randomly sampled fuel blocks and machined to 1 x 1 x 1-in. cubes. Particle size and size distribution would be measured to determine the mean and variance of particle density within the sample volume. Analogous analysis would be performed for defects (e.g., cracks and voids).

The compressive strength of the fuel blocks would be measured on 1/4-in. diameter x 1/2-in. tall samples to be tested on a load frame under a fixed crosshead displacement speed until failure.

Tensile strength would be measured on 1/8-in. diameter dogbone samples from a minimum of 10 samples per block. Samples would be tested on a load frame under a fixed crosshead displacement speed until failure.

The thermal shock behavior of the fuel blocks would be measured on a minimum of five samples per block by heating 1/4-in. diameter x 1/2-in. tall samples to 800°C and quenching them on a chilled copper hearth. Subsequently, these samples would be tested for compressive strength.

The thermal expansion behavior of the fuel blocks would be measured by dilatometry performed at temperatures ranging from room temperature to 900°C on three 1/4-in. diameter by 1/2-in. tall samples.

Reflector Block

The current TREAT core uses graphite as the reflector material in both the fuel element upper and lower reflectors and permanent reflectors surrounding the core. The base length of the fuel element assembly is 96 in., and the center of the fuel section is 48 in. from the top of the grid plate surface. Dummy fuel elements would be filled with graphite blocks made from the same graphite material as the reflector block. The base length and cross-sectional properties would adhere to the grid plate, or associated grid plate insert, and geometric parameters so proper cooling is maintained.

For each selected graphite vendor, data relating to geometric stability, neutronic configuration, and thermal-hydraulic analysis would be needed.

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The need for substantial irradiation testing is not anticipated, because irradiation data was obtained from the Very High-Temperature Reactor graphite qualification campaign. However, the following parameters and any anisotropic performance variations would be included in analyses:

- Dimensional stability (temperature and under irradiation)
- Strength
- Coefficient of thermal expansion
- Heat capacity
- Thermal conductivity
- Oxidation tests for impurities.

Fuel Encapsulation

Fuel blocks are entirely encapsulated within a hermetic boundary (i.e., the fuel element assembly cladding). This protects the materials within from the oxidizing effects of TREAT's air environment and contains fission products and gases produced during reactor operation. TREAT successfully operated using Zr-3 cladding. Zirconium metal would provide the best corrosion resistance for cladding materials, but other considerations (such as structural strength and creep resistance, metal growth due to irradiation damage, or thermal cycling) would be evaluated.

Besides zircalloys, other alloys utilizing niobium or niobium and tin have been developed and used for nuclear cladding in light water reactors, and M5 (Zr-1%Nb) was developed for improved hydrogen embrittlement resistance and high burnup applications at low temperatures. These alloys would be tested as part of the proposed action.

Steady-state irradiation testing with PIE examination measurements similar to out-of-pile examination would be completed. Specimen cross sections would be examined at the microscopic scale to study microstructural features after irradiation including material degradation due to fission/fluence damage, irradiation creep of materials, cross-sectional swelling profiles, and fission gas bubble morphology.

For oxidation testing of cladding materials, specimens would be lightly polished to remove any residual oxides on the surface deposited from machining and then degreased in acetone, ethanol, and high purity water. Oxidation exposure would then be performed using a thermogravimetric analysis (TGA) with air flow. The specimens would be brought to temperature in ultra-high purity helium. After temperature is reached, the gas would be switched to air to start the test. At the end of the test, specimens would be stored in labeled containers.

The original TREAT fuel elements were built in segments using Zr-3 to envelope the fuel section. The upper and lower reflector graphite blocks were not protected from the oxidizing environment in the core and were surrounded by aluminum instead. For these reasons, the upper and lower reflectors were protected from the fuel meat's temperatures with corrugated zircaloy spacers at the ends of each fuel section. Because zircaloy and aluminum cannot easily be joined by welding, these segments were joined by rivets. The rivets occasionally failed during TREAT fuel element handling.

The TREAT upgrade core was designed with a single one-piece cladding that encompassed the fueled section and the upper and lower non-fueled reflectors. This design mitigates mechanical failure of the riveted regions observed in the existing TREAT fuel elements. The primary drawback pertains to the length at which fuel cladding must be manufactured, including stricter control during welding processes (especially electron beam welding processes) to exacting tolerances. Corrosion performance of the welded structures would be evaluated.

TREAT LEU Structural Fuel Analysis

The structural integrity of the TREAT LEU fuel elements during transport are driven by gravity loading and the loading applied to both end fittings, the cladding, and the upper and lower welds attaching the fittings to the cladding. Geometric fuel designs and methods would be evaluated to verify structural integrity and function of the assembly is maintained.

To evaluate the TREAT LEU fuel elements for structural integrity while in the reactor, gravity, pressure, vibration, and thermal loadings would be considered. Modification of the cladding design from the original or upgrade designs and implementation of a new material, requires the buckling capacity of the graphite be evaluated. The ASME, Section III, Division 5, Subsection HH standard directs the means by which structural calculations are measured for material properties of the graphite.

End Caps

The current TREAT fuel element assemblies contain Zr-4 top and bottom fittings welded to their respective ends of the assembly's cladding. Weld testing and verification would be performed on the end caps.

Fuel Element Assembly Testing

Once the fuel element assembly has been welded, a helium leak test would be performed to verify integrity and design specifications are met. If the cladding fails the helium leak test, two-dimensional radiography would be performed to determine leak location and whether or not it could be fixed. If the leak cannot be located or fixed, the cladding would be scrapped and recycled by the program for reuse.

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Subassembly tests (i.e., an out-of-pile and steady-state irradiation) would be conducted to obtain information on dimensional stability, structural integrity, interface behavior, potential corrosion characteristics, thermal hydraulics, and neutronic performance of the assembly. These tests would be followed by several transient irradiations per subassembly to obtain the same information.

LEU Design and Demonstration Fuel Element Assembly

The LEU design and demonstration fuel element assembly test is needed to demonstrate suitable irradiation performance to provide data regarding expected emissions during irradiation. To demonstrate conformance with fuel element assembly requirements, PIE would include and evaluate the following:

- Visual examination
- Dimensional characteristics, including length and width, to quantify the magnitude of specimen swelling and thermal/irradiation creep
- Thickness of the surface oxide layer
- Mechanical integrity of entire cladding element
- Interactions between fuel block and cladding, UO₂ particles and matrix, and cladding and insulation material (if used)
- Specimen cross sections examined at the microscopic scale with regard to evolution of microstructural features under irradiation, including material degradation due to fission/fluence damage, irradiation creep of materials, cross-sectional profiles, and fission gas bubble morphology
- Post-irradiation mechanical properties
- Post-irradiation residual stress state
- Outgassing behavior of graphite fuel composite specimens
- Dimensional stability of graphite fuel composite specimens
- Thermal conductivity degradation of graphite fuel composite specimens
- Fission product diffusion
- Element gamma scanning.

Transient Testing

A final demonstration of the fuel element assembly in a prototypic geometry (i.e., full-scale fuel element assembly) is required prior to seeking approval for operational use of the LEU fuel element assemblies in TREAT. To accomplish this objective, multiple irradiation tests would be performed in the TREAT reactor using standard TREAT LEU lead test assemblies, control rod element and access slot elements. The standard TREAT LEU lead test assemblies would be equipped with additional instrumentation. Each TREAT LEU lead test assembly would be fabricated in accordance with the TREAT LEU element assembly specification and TREAT LEU conversion quality assurance requirements. The test will require an approved Experimental Safety Analysis Package along with applicable input to the Core Safety Analysis Package prior to testing.

A range of three to six lead test assemblies would be designed to the performance capabilities of the fuel element assembly with multiple tests (e.g., transients) run on each assembly. Transient testing of the LEU lead test assemblies in the HEU core would evaluate the LEU assemblies up to the established LEU fuel element assembly Safety Analysis and Review (SAR)/technical specification requirement safety limits. The current TREAT safety basis documentation (SAR-420) allows for placement of LEU fuel element assemblies in TREAT within the confines of the experiment process as described in Chapter 10 ("A minimum of one level of containment (the fuel can) will be required and designed to retain its integrity during all planned test and credible accident conditions as defined in FSAR Chapter 15 for a new form of a TREAT driver fuel element being tested in TREAT (i.e. LEU conversion lead test assembly)."); impacts are covered and consistent with the TREAT SAR and Environmental Assessment for the resumption of transient testing in the TREAT reactor. The test assemblies will require an approved Experimental Safety Analysis Package along with applicable input to the Core Safety Analysis Package prior to testing. No more than 2 assemblies will be simultaneously tested.

Facility Modifications

In order to accommodate these activities and manufacturing of test articles, upgrades and modifications to MFC-723 are needed and new equipment needs to be installed. Modifications include providing additional power, replacing light fixtures with more efficient and reliable fixtures, installing new rollup doors, constructing an Assembly Room, and adding a new heating, ventilating, and air conditioning (HVAC) unit, new water line via the fire sprinkler main, safety equipment, and suspect exhaust system.

Equipment anticipated to be installed in MFC-723 include a glove box, direct fed additive manufacturing system, ultrasonic cleaning bath(s), degassing furnace, fume hood an electron beam (EBeam) welder, x-ray/three-dimensional computerized tomography (3D CT) scan station, vacuum leak test station and a tungsten inert gas (TIG) welding station. Work in MFC-723 would include the use of LEU, graphite, and zircaloy.

A ventilation stack would be added to the warehouse wall through a drilled hole. Minimal disturbance of wall materials is anticipated.

After PIE at INL, the irradiated sample segments and PIE remnants would be stored with other similar Department of Energy (DOE)-owned irradiated materials and experiments at MFC, most likely in Hot Fuel Examination Facility (HFEF) or the Radioactive Scrap and Waste Facility (RSWF) in accordance with DOE's Programmatic Spent Nuclear Fuel (SNF) Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement (FEIS) and Record of Decision (ROD) (DOE/EIS-0203, 1995) and supplemental analyses (DOE/EIS-0203-SA-01 and DOE/EIS-0203-SA-02) and the Amended Record of Decision (February 1996). Ultimate disposal of the irradiated sample segments and PIE remnants would be along with similar DOE-owned irradiated materials and experiments currently at MFC.

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Low-level waste (LLW) would be shipped to the Nevada National Security Site (NNSS) for disposal. The environmental impacts of transferring low level waste from the INL to the Nevada National Security Site were analyzed in the 1996 Nevada Test Site EIS (DOE/EIS-0243) and supplemental analysis (SA) (DOE/EIS-0243-SA-01) 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 mixed low level waste (MLLW) disposal sites. The SA considers additional waste streams, beyond those considered in the 1996 Nevada Test Site (NTS) EIS that may be generated at or sent to the Nevada National Security Site for management.

Packaging, repackaging, transportation, receiving, and storing used nuclear fuel and research and development for used nuclear fuel management is covered by DOE's Programmatic Spent Nuclear Fuel (SNF) 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 (February 1996). The analysis includes those impacts related to transportation to, storage of, and research and development related to used nuclear fuel at the INL (see Tables 3.1 of the SNF Record of Decision (May 30, 1995) and Table 1.1 of the Amended Record of Decision [February 1996]).

Additionally, in order to complete proposed work activities, it is necessary for the project to use the HFEF hot cell which contains both defense and nondefense related materials and contamination. Project materials will come into contact with defense related materials. It is impractical to clean out defense related contamination, and therefore, waste associated with project activities may be eligible for disposal at the Waste Isolation Pilot Plant (WIPP).

National Environmental Protection Act (NEPA) coverage for the transportation and disposal of waste to WIPP are found in 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 transuranic (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.

Projected Start Date: October of 2013

Projected End Date: September of 2025

Estimated Cost: Approximately \$166 million

SECTION C. Environmental Aspects or Potential Sources of Impact:

Air Emissions

MFC-723 would be considered a new release point, and an Air Permitting Applicability Determination (APAD) would be required for the proposed activities. LEU fuel block particles (graphite and UO₂) and minor quantities of CH₄, CO₂, H₂, CH₂CH₂, and CO may be generated during fuel block outgassing, fuel block machining operations, and during fuel block handling and assembly into fuel element assemblies. It is anticipated that small amounts of UO₂ particulates would be generated during approximately two years of fabrication of fuel blocks and fuel element assemblies (rough estimate is approximately 60 grams of UO₂). These processes would be performed in a negative pressure fabrication and assembly area where fume hood(s) exhaust and general ventilation exhaust would be channeled into a high-efficiency particulate air (HEPA) filter bank prior to exhaust stack release.

Experiment irradiation will be performed at the ATR. Air emissions would include minor amounts of radionuclides and toxic air pollutants. The work ATR is encompassed by the operating requirements/processes identified in SAR-153 and the source term in SAR-153 Chapter 12 "Radiation Protection." ATR radionuclide emissions are sampled and reported in accordance with Laboratory Wide Procedure (LWP)-8000 and 40 Code of Federal Regulation (CFR) 61 Subpart H. All radionuclide release data (isotope specific in curies) directly associated with this experiment will be calculated and provided to ATR Programs Environmental Support organization.

HFEF is a hot-cell complex designed for the preparation and examination of irradiation experiments in support of a wide variety of programs and process demonstrations. It is anticipated that the potential radiological releases to the Main Cell from this project would be consistent with other in cell processes performing macro- and microanalysis. This work is encompassed by the HFEF source term that was derived from DSA-003-HFEF Rev 6, Chapter 3 "Radiological Inventory". Irradiated specimens will be delivered to the MFC HFEF for disassembly and then undergo routine PIE. All radionuclide release data associated with the PIE portion and analysis of this activity will be recorded as part of the HFEF continuous stack monitors and provided to the program's Environmental Support organization in accordance with Laboratory Wide Procedure (LWP)-8000 and 40 CFR 61 Subpart H.

Fuel element irradiation would be performed at the TREAT facility. TREAT radionuclide emissions are sampled and reported in accordance with Laboratory-wide Procedure (LWP)-8000 and 40 CFR 61 Subpart H. Emissions would be controlled under APAD INL-15-001, "Resumption of Transient Testing of Nuclear Fuels and Materials at TREAT."

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Up to 20 g of unirradiated LEU heated to above 100C at the Idaho National Laboratory Research Complex (IRC). No HEPA filter would be used. Emissions will be evaluated against the radiological National Emission Standards for Hazardous Air Pollutants (NESHAP) limits for IRC.

Generating and Managing Waste

The proposed action would generate uranium bearing waste, LLW, industrial waste, and TRU waste. Estimated volumes for each type of waste are as follows:

- Uranium bearing:
 - ~300g U-235 embedded in ~7100 cubic inches of carbon/UO₂ fuel blocks
- LLW:
 - Zr-4 sheet metal (~360 cubic inches)
 - Zr-4 metal (~520 cubic inches)
 - Graphite blocks (~7000 cubic inches)
- Industrial:
 - <1 cubic foot (e.g., wipes)
 - ~20 gallons fluids (cleaning of fuel blocks; cooling fluid for e-beam welder; wipe-down of Zr-4 sheets)
- TRU:
 - ~1200g of U-238 contained in ~7100 cubic inches of fuel block material

Wastewater would be generated from 1) the ultrasonic cleaning bath for fuel blocks, 2) the cooling fluid (e.g., glycol material) used by e-beam welding equipment, vacuum furnace, and alternative manufacturing equipment (e.g., chiller), 3) condensation from the air conditioning system, and 4) personnel wash sink. Contaminated water (from chemicals or rad) would be collected and disposed according to regulation.

If wastewater from fuel block cleaning contains UO₂, it would be considered LLW. Wastewater streams that come from the cooling water of the vacuum furnace and the electron beam welder would only be generated when maintenance is performed on the cooling systems of the respective equipment and it is determined the cooling system water needs to be changed out or drained. This wastewater would then be disposed according to regulation.

When dispositioned as waste, irradiated sample debris and PIE material will likely be categorized as TRU and potentially MTRU waste. Categorizing this material as waste is supported under DOE O 435.1, Att. 1, Item 44, which states "...Test specimens of fissionable material irradiated for research and development purposes only...may be classified as waste and managed in accordance with this Order..." Project personnel would work with Waste Generator Services (WGS) and/or BEA waste management staff to characterize and properly disposition the waste.

DOE's Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement, DOE/EIS-0203-F covers the repackaging, transportation, storage, and shipment of the HEU fuel element assemblies. That analysis includes impacts related to transportation to, and storage of the fuel element assemblies at the INL (See Tables 3.1 of the SNF Record of Decision [May 30, 1995] and Table 1.1 of the Amended Record of Decision [February 1996]).

Non-contact cooling water will be discharged at IRC. All discharges must comply with the discharge limits for the City of Idaho Falls sewer system.

Releasing Contaminants

As described in the air emissions section above, radioactive air emissions are anticipated as a result of irradiation activities associated with this project.

Clad cleaning and passivation fluids are general purpose industrial cleaners that would be generated and disposed in accordance with INL procedures.

Using, Reusing, and Conserving Natural Resources

All material would be reused and/or recycled where economically practicable. All applicable waste would be diverted from disposal in the landfill when possible. Project personnel would use every opportunity to recycle, reuse, and recover materials and divert waste from the landfill when possible. The project would practice sustainable acquisition, as appropriate and practicable, by procuring construction materials that are energy efficient, water efficient, are bio-based in content, environmentally preferable, non-ozone depleting, have recycled content, and are non-toxic or less-toxic alternatives. New equipment will meet either the Energy Star or Significant New Alternatives Policy (SNAP) requirements as appropriate (see <https://sftool.gov/greenprocurement>).

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SECTION D. Determine Recommended Level of Environmental Review, Identify Reference(s), and State Justification: Identify the applicable categorical exclusion from 10 CFR 1021, Appendix B, give the appropriate justification, and the approval date.

For Categorical Exclusions (CXs), the proposed action must not: (1) threaten a violation of applicable statutory, regulatory, or permit requirements for environmental, safety, and health, or similar requirements of DOE or Executive Orders; (2) require siting and construction or major expansion of waste storage, disposal, recovery, or treatment or facilities; (3) disturb hazardous substances, pollutants, contaminants, or Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA)-excluded petroleum and natural gas products that pre-exist in the environment such that there would be uncontrolled or unpermitted releases; (4) have the potential to cause significant impacts on environmentally sensitive resources (see 10 CFR 1021). In addition, no extraordinary circumstances related to the proposal exist that would affect the significance of the action. In addition, the action is not "connected" to other action actions (40 CFR 1508.25(a)(1) and is not related to other actions with individually insignificant but cumulatively significant impacts (40 CFR 1608.27(b)(7)).

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

Final Environmental Assessment for the Resumption of Transient Testing of Nuclear Fuels and Materials (DOE/EA-1954, February 2014)

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 Nevada Test Site and Off-Site Locations in the State of Nevada (DOE/EIS-0243) and supplemental analysis (SA) (DOE/EIS-0243-SA-01).

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, Sept. 1997).

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."

The current TREAT safety basis documentation (SAR-420) allows for placement of LEU fuel element assemblies in TREAT. Impacts from the proposed action are covered and consistent with the TREAT SAR and the Final Environmental Assessment for the Resumption of Transient Testing of Nuclear Fuels and Materials (DOE/EA-1954, February 2014).

DOE's Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement, DOE/EIS-0203-F covers the repackaging, transportation, storage, and shipment of the SNF. That analysis includes impacts related to transportation to, and storage of SNF at the INL (See Tables 3.1 of the SNF Record of Decision [May 30, 1995] and Table 1.1 of the Amended Record of Decision [February 1996]).

DOE/EIS-0200 made the Nevada National Security Site available to all DOE sites for low-level waste disposal, and DOE/EIS-0243 and ROD (65 FR 10061, February 2000) analyzed the impacts of transportation from the INL and disposal at the Nevada National Security Site. The environmental impacts of transferring low level waste from the INL to the Nevada National Security Site were analyzed in the 1996 Nevada Test Site EIS (DOE/EIS-0243) and supplemental analysis (SA) (DOE/EIS-0243-SA-01) and DOE's Waste Management Programmatic EIS (DOE/EIS-200). The SA considers additional waste streams, beyond those considered in the 1996 NTS EIS that may be generated at or sent to the Nevada National Security Site for management.

The impacts of transporting and disposing of waste resulting from defense activities that was placed in retrievable storage pursuant to a 1970 Atomic Energy Commission policy (see Section 1.2) and TRU waste that was reasonably expected to be generated by ongoing activities and programs was analyzed in DOE/EIS-0026 (October 1980) and the Final Supplement Environmental Impact Statement for the Waste Isolation Pilot Plant (SEIS-I) (DOE/EIS-0026-FS, January 1990).

NEPA coverage for the transportation and disposal of waste to WIPP are found in 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. DOE has analyzed TRU waste management activities in DOE /EIS-200-F (May 1997). The WM PEIS analyzes environmental impacts

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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. (SEIS-II also includes potential transportation between generator sites.)

Is the project funded by the American Recovery and Reinvestment Act of 2009 (Recovery Act) Yes No

Approved by Jack Depperschmidt, DOE-ID NEPA Compliance Officer on: 9/12/2016