Idaho National Laboratory

April 24, 2015 CCN 235661

Mr. Jeffrey C. Fogg
DOE-ID Contracting Officer
U.S. Department of Energy
Idaho Operations Office (DOE-ID)
1955 Fremont Avenue
Idaho Falls, ID 83415-1221

SUBJECT: Contract No. DE-AC07-05ID14517 – Battelle Energy Alliance, LLC Response to Department of Energy, Idaho Operations Office Request for Information to Support Supplement Analysis of Proposed Commercial Fuel Research and Development Efforts

Reference: J. C. Fogg letter to D. M. Storms, Contract No. DE-AC07-05ID14517 – Request for Information to Support Department of Energy's Supplement Analysis of Proposed Commercial Fuel Research and Development Efforts (AS-CMD-INL-15-062), April 10, 2015, CCN 235586

Dear Mr. Fogg:

In response to the reference letter, Battelle Energy Alliance, LLC (BEA) is providing the following response to DOE-ID's request for information to support supplement analysis of proposed commercial fuel research and development efforts:

DOE-ID

1. Based on the proposed action, provide estimated quantities and disposition paths for all waste or materials, per the categories a-c below:

    a. Quantities of transuranic waste (TRU) generated.

    b. Quantities of low-level waste (LLW) generated.

    c. How much would remain in storage as spent nuclear fuel (SNF) on-site at INL? How long would this material remain in storage at INL? Provide a brief description of the storage of this SNF (e.g., location, type cask, etc.).

BEA

Under this contract, DOE-ID will acquire through an agreement for title transfer and ownership approximately 100 kgHM of commercially generated used nuclear (Light Water Reactor) fuel for laboratory-scale research purposes. Less than 10 kgHM of relevant, modern pressurized water reactor fuel may be selected and saved in a fuel library to enable future research activities regarding issues of fuel safety and performance. It is estimated that approximately 85 kgHM can be expected to be disposed as TRU wastes. Approximately 5 kgHM are expected to be disposed of as LLW. Both the TRU and LLW will be managed in accordance with DOE Order 435.1, *Radioactive Waste Management,* and to meet all disposal facility waste acceptance criteria.

Based on analysis of work scope and the type and quantity of the used nuclear fuel research materials, the total waste is expected to be approximately 7.4 cubic meters of TRU waste and 6.0 cubic meters of LLW.

DOE-ID

d. Are there any other potential waste streams that could be generated under the proposed action? If so, please describe.

BEA

It is not expected that any other waste streams will be generated.

DOE-ID

2. Has an intentional destructive acts analysis been conducted for the Materials and Fuels Complex? If so, please provide an unclassified summary of the results. If not, can INL provide a reference-able document to support an analysis regarding the effects of an intentional destructive act involving 80-100 kilograms of heavy metal in any INL facility where this material would be located?

BEA

Acts of intentional destructive have been analyzed in association with activities at the Materials and Fuels Complex (MFC). One such document is DOE/ID-10471 (1995), *Accident Assessments for Idaho National Engineering Laboratory Facilities*, which evaluates an aircraft crashing into the Hot Fuel Examination Facility (HFEF), among a number of other accidents.

Mitigation of intentional acts is addressed as follows:

- External terrorism is mitigated by protective forces and restricted access to the site.
- Internal sabotage is mitigated by the security clearance process for employees and subcontractors (background checks, etc.).

Documentation that addresses the types of accidents that would result from this type of destructive act, specifically in HFEF where the overwhelming majority of the work would be performed and where the fuel would be located, would be bounded by accidents analyzed in the HFEF safety analysis report, document DSA-003-HFEF-0.

DOE-ID

3. What is the bounding accident involving nuclear fuel for the Materials and Fuel Complex? Please provide a reference-able document to support an analysis of whether

the 80-100 kilograms of heavy metal would change the bounding accident in any INL facility where this material would be used?

BEA

Of the MFC facilities involved in commercial fuels work, HFEF has the bounding accident, which is a seismic induced release, that is considered extremely unlikely (HFEF Safety Analysis Report, document DSA-003-HFEF-0). The addition of the proposed material does not appreciably change the bounding source term nor the material considered at risk used in the analysis.

As part of the receipt process, BEA is preparing documentation that calculates the contribution of the proposed work scope to existing inventory and activities. This information will be compared to the bounding accident in the HFEF safety analysis report. Preliminary calculations show that the commercial fuel will not change the bounding accident or allowable material at risk for HFEF. This analysis is not yet ready to be referenced.

DOE-ID

4. Provide an estimate of the increased radiological emissions associated with the proposed action.

BEA

The dose calculation shown below quantifies estimated radiological emissions from the proposed action. BEA is seeking to have 50 fuel rods shipped to MFC/HFEF to support these activities. The amount of material received at MFC for all activities is approximately 100 kgHM. An Engineering Calculation and Analysis Report, document ECAR-2780, *Baseline ORIGEN-ARP Calculations for (the IRT) Used Fuel*, calculates bounding isotopics for one shipment. This inventory is presented in the Table below.

This inventory was multiplied by a factor of two to account for both shipments. The total inventory for each isotope was converted to a dose using CAP-88 modeling. A re-suspension factor of 0.001 was applied to all isotopes except gasses, which were assigned a re-suspension factor of 1. The estimated mitigated (i.e., assuming two sets of high-efficiency particulate Air [HEPA] filters for non-gaseous isotopes) dose for performing research and development on 50 fuel rods does not exceed 3.4 E-03 mrem/yr to the maximally exposed individual located south of MFC. (Note that H-3 contributes 60.4 % of the dose and I-129 contributes 39.6 % of the dose [gasses were assumed to have a re-suspension factor of 1 and no credit is taken for HEPA filters]).

Dose Calculation for Commercial Fuel Shipment to INL									
Nuclide	Total from ECAR-2780 Table 5	Multiplied by a factor of 2 to account for additional rods	Specific Activity	Dose Conversion Factors from CAP-88	Resuspension Factor (1 for gasses and 1.00E-03 for liquid and particulate)	Unabated Actual Dose	2 HEPAs at 99.97% Efficiency	Abated Dose	Major Dose Contributors Greater than 1%
	(grams)	(grams)	(Ci/gram)	(mrem/Ci)		(mrem/yr)		(mrem/yr)	(%)
Am-241	2.89E+01	5.79E+01	3.43E+00	1.50E+00	1.00E-03	2.98E-01	1.00E-08	2.98E-09	0.0%
Am-243[e]	1.87E+01	3.75E+01	2.00E-01	1.50E+00	1.00E-03	1.12E-02	1.00E-08	1.12E-10	0.0%
Ce-144	2.48E-01	4.96E-01	3.19E+03	2.60E-03	1.00E-03	4.11E-03	1.00E-08	4.11E-11	0.0%
Cm-243[e]	4.64E-02	9.28E-02	5.06E+01	1.20E+00	1.00E-03	5.63E-03	1.00E-08	5.63E-11	0.0%
Cm-244	7.71E+00	1.54E+01	8.10E+01	9.80E-01	1.00E-03	1.22E+00	1.00E-08	1.22E-08	0.0%
Cs-134	1.62E+00	3.24E+00	1.29E+03	3.40E-02	1.00E-03	1.42E-01	1.00E-08	1.42E-09	0.0%
Cs-137	8.54E+01	1.71E+02	8.69E+01	5.80E-02	1.00E-03	8.61E-01	1.00E-08	8.61E-09	0.0%
Eu-154	1.35E+00	2.71E+00	2.70E+02	5.00E-03	1.00E-03	3.66E-03	1.00E-08	3.66E-11	0.0%
Eu-155	1.97E-01	3.95E-01	4.85E+02	4.50E-04	1.00E-03	8.62E-05	1.00E-08	8.62E-13	0.0%
H-3	3.44E-03	6.88E-03	9.63E+03	3.10E-05	1.00E+00	2.05E-03	1.00E+00	2.05E-03	60.4%
I-129	1.32E+01	2.64E+01	1.76E-04	2.90E-01	1.00E+00	1.35E-03	1.00E+00	1.35E-03	39.6%
Np-239	1.61E-05	3.23E-05	2.32E+05	4.00E-05	1.00E-03	2.99E-07	1.00E-08	2.99E-15	0.0%
Pm-147	1.95E+00	3.91E+00	9.28E+02	2.50E-04	1.00E-03	9.06E-04	1.00E-08	9.06E-12	0.0%
Pr-144	1.05E-05	2.09E-05	7.56E+07	7.10E-07	1.00E-03	1.12E-06	1.00E-08	1.12E-14	0.0%
Pu-238	2.20E+01	4.40E+01	1.71E+01	1.70E+00	1.00E-03	1.28E+00	1.00E-08	1.28E-08	0.0%
Pu-239	2.94E+02	5.89E+02	6.21E-02	1.80E+00	1.00E-03	6.58E-02	1.00E-08	6.58E-10	0.0%
Pu-240	1.51E+02	3.01E+02	2.27E-01	1.80E+00	1.00E-03	1.23E-01	1.00E-08	1.23E-09	0.0%
Pu-241	6.30E+01	1.26E+02	1.04E+02	3.30E-02	1.00E-03	4.32E-01	1.00E-08	4.32E-09	0.0%
Ru-106	3.46E-01	6.93E-01	3.30E+03	3.60E-03	1.00E-01	8.23E-01	1.00E-08	8.23E-09	0.0%
Sb-125	1.21E-01	2.42E-01	1.04E+03	1.30E-03	1.00E-03	3.27E-04	1.00E-08	3.27E-12	0.0%
Sr-90	3.41E+01	6.81E+01	1.38E+02	9.40E-02	1.00E-03	8.83E-01	1.00E-08	8.83E-09	0.0%
U-234[e]	9.20E+00	1.84E+01	6.23E-03	1.40E-01	1.00E-03	1.60E-05	1.00E-08	1.60E-13	0.0%
U-235[e]	3.06E+02	6.11E+02	2.16E-06	1.20E-01	1.00E-03	1.58E-07	1.00E-08	1.58E-15	0.0%
U-238[e]	4.59E+04	9.18E+04	3.36E-07	1.20E-01	1.00E-03	3.70E-06	1.00E-08	3.70E-14	0.0%
Y-90	8.84E-03	1.77E-02	5.44E+05	5.80E-05	1.00E-03	5.58E-04	1.00E-08	5.58E-12	0.0%
						6.16E+00		3.40E-03	100.0%

DOE-ID

5. Provide an estimate, if applicable, of any increased radiological doses to workers associated with the proposed action.

BEA

BEA controls worker doses to as low as reasonably achievable within administrative limits. The proposed work will not affect this approach or the limits.

DOE-ID

6. What are the estimated dose rates associated with a NAC-LWT cask loaded with 25 high-burnup fuel rods? Would those be different from dose rates that would be calculated if a transportation, aging, and disposal canister (TAD) was also used to transport the rods? If so, please provide an explanation.

BEA

The Certificate of Compliance (Certificate Number 9225) is the document that specifies the loading configurations and requirements as derived in the NAC-LWT Safety Analysis Report. Section 5.(b)(1)(viii) defines the type and form of material and section 5.(b)(2)(ix) gives the maximum quantity of pressurized water reactor material per package. The licensing process for a type B package requires that the licensee for the package submit analysis to the Nuclear Regulatory Commission demonstrating compliance with applicable Codes of Federal Regulation (10 CFR Part 71, *Packaging and Transportation of Radioactive Material*. The conclusions of the analyses demonstrate that the payloads will be under the 49 CFR 173.441(a) limits not to exceed 200 mrem/hr at any point on the external surface of the package and not exceeding 10 mrem/hr at 2 meters from the outer lateral surfaces of the vehicle. Although use of a different cask (such as a transportation, aging, and disposal canister) might result in different actual dose rates, dose rates would still be required to be below 49 CFR 173.441(a) limits.

DOE-ID

7. Provide as an attachment to this response a thorough description of the research and development work scope associated with each of the two proposed commercial fuel shipments.

BEA

See attached document, *Scope of Work for 2 Shipments of 25 Used Commercial Fuel Rods*, for a thorough description of the research and development work scope.

If you need additional information, please contact me at 526-1148 or Kemal Pasamehmetoglu at 526-5305.

Sincerely,

Dana M. Storms, Manager
Contracts Management

KP:JHB

Attachment

cc: DOE-ID
P. K. Bowers, MS 1226
J. D. Depperschmidt, MS 1216
R. V. Furstenau, MS 1203
D. W. Kamerman, MS 1235
K. A. Lockie, MS 1235
S. M. Olson, MS 1221
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T. R. Allen, MS 3695
J. Alvarez, MS 3695
K. J. Beierschmitt, MS 3695
R. A. Crone, MS 6144
J. J. Grossenbacher, MS 3695
C. S. Mascareñas, MS 3402
D. R. Newby, MS 3898
K. O. Pasamehmetoglu, MS 3860

Attachment 1
April 24, 2015
CCN 235661
Pages 1 through 6, inclusive (excluding cover sheet)

Scope of Work for 2 Shipments of 25 Used Commercial Fuel Rods

1st Shipment

The proposed action is an evaluation of the technical and economic feasibility and nonproliferation acceptability of electrochemical recycling as a component within a nuclear fuel cycle. Fewer data are available regarding this recycling route than other potential options for future fuel cycles, and the study aims to acquire additional data to better model and understand the potential advantages or disadvantages of this technology versus other nuclear fuel cycle options. The overall study includes a laboratory-scale feasibility evaluation at gram-scale which was successfully completed with existing materials and equipment. The primary component of the overall study is an integrated, kilogram-scale test of electrochemical recycling using modern used nuclear fuel. This component is described as the Integrated Recycling Test (IRT) and allows the acquisition of information regarding how processes perform and interact in realistic operating situations and is key to understand and predict an integrated process behavior. A third arm of investigation is tasked to examine key technology risks or issues of longer-term economics or scalability; this branch is described as Critical Gap Research. The study is divided into three phases to facilitate planning, approval and execution processes. The laboratory-scale evaluation was completed as the primary focus of Phase I. Phase II-a of the study was focused on the design and fabrication of process equipment, and Phase II-b is focused on installation and operation of the equipment to recover fuel materials for fabrication of recycled fuel. The emphasis of Phase III is upon fabrication of recycled fuel, irradiation, and post-irradiation examination. Each Phase is further described below.

The Integrated Recycling Test is planned to be performed at the Hot Fuel Examination Facility (HFEF) of the Department of Energy's (DOE's) Idaho National Laboratory (INL). Irradiation of the recycled fuel is planned for INL's Advanced Test Reactor. Post-irradiation examination (PIE) of the recycled fuel when it is removed from the reactor is also planned for the Hot Fuel Examination Facility.

The acquisition of used fuel feedstock for the study has been developed in conjunction with a commercial nuclear fuel vendor who has interest in post-irradiation inspections of advanced cladding alloys. This teaming approach for acquisition of fuel has the potential to provide significant advantages to all parties. The workscope for this PIE is also described below.

Phase II-a Equipment Design, Fabrication and Fuel Procurement
Phase II-a focused on the design, construction, and mockup testing of kilogram-scale process equipment and has been performed in MFC-765, MFC-768B, MFC-785, MFC-787 and MFC-789. In addition, during Phase II-a, approximately 25 rods of modern used light water reactor (LWR) fuel would be sought for use as feedstock. This feedstock does not currently exist at MFC, and is not readily accessible at INL or in the DOE complex. Current plans call for the receipt of used pressurized water reactor (PWR) fuel from a commercial reactor with a burnup in the range of 55 GWD/MT and 5-10 year cooling time. The used fuel would be loaded at a commercial source and would be transported in a Nuclear Assurance Corporation legal-weight truck (NAC-LWT) cask. Upon arrival at INL, used fuel rods would be stored in fuel storage pits in the HFEF hot cell until taken for project activities described in this document.

Phase II-b Integrated Recycling Test
- Modular Workstations
A modular workstation approach will be used to support the kilogram-scale equipment for the IRT in the HFEF Argon Cell. Two large workstations and a smaller intermediate table would provide the operating space and power/instrumentation connections for replaceable equipment modules within the hot cell. The workstations and associated power/instrumentation feedthroughs would have the capability to support the expected kilogram-scale process equipment, and allow equipment to be placed in storage bins below the operating surface when not in active use. The workstations would include one or more integrated balances used for mass tracking purposes. The work scope also includes preparation of the in-cell area and the shifting of some existing functions to other hot cell locations.

- Fuel Decladding Equipment

Used nuclear fuel rods from light water reactors would be processed as feedstock for the IRT. These used nuclear fuel rods must be stored, handled, sectioned, and de-clad. The de-clad fuel would likely be sieved to remove fines. Some fines may be further manipulated to demonstrate or test processing methods. Some higher burnup cladding sections may also be processed to demonstrate approaches to reduce fuel holdup. Equipment, storage containers, and handling fixtures are required to perform these tasks.

- Oxide Reduction System

De-clad used fuel would be electrochemically reduced to metallic form in the oxide reduction system. The reduction process would have the capability to test the 'universal' basket concept, where a loaded basket can be processed through the oxide reduction, distillation, and electrorefining systems without unloading. The reduction system vessel is anticipated to be rectangular in design, in order to allow the testing of features important to scalability, such as multiple electrodes and variations in basket thickness or electrode spacing. The anode systems would include the flexibility to test a variety of materials, geometries, immersion depths, and off-gas capture settings. An oxide reduction system will be designed, constructed, tested, installed, and operated as a part of the Integrated Recycling Test.

- Electrorefining System

Processed fuel from the oxide reduction system will be electrorefined in order to recover a purified low-enriched uranium product. Transuranic elements are accumulated in the electrorefining salt and extracted as a uranium/transuranic/rare earth product using a liquid cadmium cathode (LCC) in order to acquire feedstock to produce fuel rodlets. The electrorefining vessel is planned to be rectangular in design, in order to allow the testing of features important to scalability.

- Distillation Systems

Vacuum distillation would be employed to separate salts from metallic products between and following the oxide reduction and electrorefining processes, and also to separate cadmium and salt from the uranium/transuranic/rare earth products. Two remote distillation systems are planned in order to maintain the IRT processing schedule. Two distillation systems will be designed, constructed, tested, installed, and operated to remove salt from metallic products.

- Sampling/Casting Furnace

It is expected that a variety of metallic products will require sampling, and the casting of transuranic fuel slugs is an important objective of the IRT. Material losses, remote reliability, and scalability are important issues for long-term success of electrochemical recycling processes. A remote sampling/casting furnace will be designed, constructed, tested, installed, and operated in order to produce homogeneous metallic samples and cast transuranium-bearing fuel slugs. Fuel sampling/casting will be performed in HFEF with fuel fabrication performed in a shielded glovebox.

- Used Fuel Feedstock

Less than 100 kgHM of used LWR fuel is necessary for feedstock. It is desirable to acquire this feedstock through the development of a collaborative arrangement with a fuel vendor and commercial PWR owner. INL will assist DOE and a commercial U.S. power plant fuel vendor/utility in negotiation of an Agreement for Title Transfer and Ownership of the used LWR fuel. Preparations for handling of a loaded cask in HFEF will be accomplished with a dry run. Following this, the used fuel will be loaded at the reactor site, transported, and unloaded at HFEF.

- Nuclear Safety

The fuel must be received and handled within HFEF following documented nuclear safety regulations. A range of preparations are necessary.

<u>Phase II-b Critical Gap Research and Development</u>

- Head End Processes

The method by which used fuel would be prepared for processing is important to the subsequent process operations. This task would perform studies to evaluate and select the fuel preparation methods which would be used in the IRT. Mechanical design development including testing, cutting methods, and handling of fuel pieces and fines would be performed. Exploration of potential measurement methods to determine input accountancy is also an important issue for the determination of nonproliferation acceptability. Additional studies of processing methods for fuel fines which could be employed for the IRT, such as agglomeration or sintering, or the examination of the off-gasses that may result from processing of fines, would be identified. Prior research has indicated that high burnup fuels may display increasing fuel hold up in the cladding. Studies may also be performed of processing methods that would be used in the IRT to reduce this fuel hold up.

Current technology for electrolytic reduction utilizes platinum anodes, however, other materials, such as iridium or conductive ceramics may provide cost and durability advantages. Testing of anode materials would be performed, with focus on testing the performance of iridium electrodes versus surrogate salt systems.

Experiments will be performed to support system design and troubleshoot oxide reduction system operations in the IRT. These research items will include activities such as construction materials and system behavior and characteristics during different operational scenarios. This testing will be performed with non-radioactive surrogates and/or depleted uranium.

- Electrorefining & Liquid Cadmium Cathode Operations
This activity seeks to test a prototype electrorefining system with molten salt and uranium dendrites at appropriate scale. This would allow challenges or performance limitations to be identified prior to remote equipment fabrication and installation and allow trouble-shooting of remote process operations. This testing may include such issues as the impact of process parameters, system design, and materials of construction. These tests will be performed with cold surrogates and/or depleted uranium.

A combined product of uranium, transuranium, and residual rare earths will be recovered during the IRT using the liquid cadmium cathode (LCC) approach. Studies will be performed to explore the impact of process parameters, system design and materials of construction. These studies will be performed with cold surrogates, depleted uranium, and potentially transuranium elements utilizing equipment and materials currently on site at the Materials and Fuel Complex.

- Product Conditioning
In the recovery of the uranium/transuranium/rare earth product via LCC technology, the rare earth contamination may be higher than the desired concentration in the metal fuel. Jointly-planned experiments will be performed to explore the feasibility of various approaches to reduce the concentration of rare earths in uranium/transuranium/rare earth products. These studies will be performed with cold surrogates, depleted uranium, and potentially transuranium elements utilizing equipment and materials currently on site at the Materials and Fuel Complex. Resulting wastes are included in the estimates provided within this document.

Four distillation operations are potentially required for the IRT. Distillation system design issues require confirmation, including the performance of stainless steel for the pressure boundary for distillation up to 1200°C. Investigations will be performed to confirm the design and materials for the distillation system and test the performance of advanced crucible materials for distillation operations. Initial screening would be performed with uranium, and final testing would be performed with uranium/transuranium/rare earth products.

A high-temperature distillation furnace is currently used to separate salt and consolidate the dendritic uranium product into a dense ingot. Initial scoping experiments have been successful for an approach in which salt and metals are continuously separated at atmospheric pressure by a porous bed, with a short residence time. These characteristics and system compactness potentially provide improved product purity, process monitoring, and safeguards opportunities. These tests will be performed with cold surrogates and/or depleted uranium.

- Fuel Fabrication
Engineering-scale fuel fabrication is a critical element to commercialization of electrochemical recycling. Weld inspection is a challenging technology that must be perfected for remote application. Testing of ultrasonic and laser weld inspection systems will be performed. An alternative approach to qualify the welding process based on statistical analysis of welding process parameters will also be evaluated. These tests utilize only non-radioactive materials.

The properties of some metal fuel alloys have not been thoroughly characterized. Fabrication and analysis of fuel alloys may be performed in the Casting Laboratory of the Materials and Fuels Complex. These studies will be performed with cold surrogates, depleted uranium, and potentially transuranium elements utilizing equipment and materials currently on site at the Materials and Fuel Complex.

Another area of effort is in-reactor cladding performance. One issue that may limit the integrity of metal fuel is the interaction of fuel constituents and fission products with the cladding, commonly described as fuel cladding chemical interaction (FCCI). Studies of barrier materials to mitigate FCCI are needed to verify performance, especially in the case of TRU and rare-earth-bearing metal fuel. Promising candidates may be tested for irradiation in Phase III of this effort.

- Fundamental Properties and Waste Forms

Activities will be performed to increase the state of fundamental knowledge regarding relevant molten salt systems. These efforts may include studies such as the electrochemical or thermophysical characteristics of molten solutions, and technology relevant to monitor process conditions inside molten salt electrochemical systems. These studies will be performed with cold surrogates, depleted uranium, and potentially transuranium elements utilizing equipment and materials currently on site at the Materials and Fuel Complex. Resulting wastes are included in the estimates provided within this document.

The identification and demonstration of appropriate waste forms is critical to the overall demonstration of the feasibility of electrochemical recycling. The most technologically feasible and cost effective options for fission product concentration and immobilization from the electrochemical recycling of used LWR in the IRT would be determined. These evaluations would involve waste experts from multiple laboratories. The selected waste process approaches are those that would eventually be remotely demonstrated in the IRT in Phase III. Once processes have been defined, laboratory-scale waste forms may be fabricated, if necessary, from fission product streams for characterization and testing. Fission product concentration and waste form processes would be optimized for application to the IRT.

In the IRT, some iodine will enter the oxide reduction vessel with used fuel. Experiments have shown that I_2 will likely be released with O_2 during the reduction process. Improved understanding of how to capture this iodine, either as a gas or by capture within the molten salt is critical to understand the mass balance of this important fission product in a real process. Methods to better understand I_2 release during the reduction process and quantitatively capture the iodine from the off-gas stream would be evaluated in a surrogate salt system. Methods to extract iodine and tellurium from the molten salt would also be examined. The feasibility of getter materials that selectively absorb reactive fission products would be tested in a surrogate salt system.

These studies will be performed with cold surrogates, depleted uranium, and potentially transuranium elements utilizing equipment and materials currently on site at the Materials and Fuel Complex.

- ANL Activities

Supporting or linked activities will also be performed at Argonne National Laboratory in Illinois. These activities will include supporting research toward fundamental properties and waste forms, and head-end processes, with similar workscope to that described above. These activities will only be performed with cold surrogates, depleted uranium, or very limited amounts of transuranium materials allowed in radiological facilities at ANL.

Phase III Fuel Rodlet Fabrication and Irradiation

Phase III activities would focus on the production of one or more transuranium-bearing fuel rodlets, their irradiation in the Advanced Test Reactor, and then post-irradiation examination (PIE) at MFC. Phase III activities would also include continued flow sheet testing in HFEF and studies of waste processes. Waste process activities may include fission product concentration/separation demonstration for both oxide reduction and electrorefining salts, demonstration of glass and ceramic waste form fabrication from both waste salts, and demonstration of cladding recycle and alloy waste form fabrication for the immobilization of undissolved solids.

The anticipated scope of work includes several staged tasks, including performing non-destructive and destructive exams of four rods, analysis of data gathered during PIE, with interim reporting and a final report with conclusions from the PIE and analysis.

Post-Irradiation Examination Work of Commercial PWR Fuel Elements

Four fuel rods included with the initial 25 fuel rods shipped will be selected for examination. Non-destructive exams will occur at INL's HFEF and destructive exams will be performed at INL's HFEF, Electron Microscopy Laboratory (EML), Analytical Laboratory (AL), and the Center for Advanced Energy Studies (CAES). INL is proposing cladding axial and ring tests be completed at another facility. As requested by the fuel vendor, fluence measurements will be completed at Pacific Northwest National Laboratory (PNNL). The requested PIE work consists of fifteen items. Test material will be selected in accordance with the provisions of the vendor proposal and in consultation with the vendor. INL will be responsible for all aspects of the contracts including packaging and transportation.

Prior to starting PIE activities, a PIE Plan will be prepared which will provide details regarding testing activities as well as collection and transmittal of data. The PIE Plan will contain a schedule for Vendor review and approve the individual procedures that will be used for PIE inspections.

The PIE examinations to be performed include the following:

1. High magnification visual examinations (at least 10x magnification) to benchmark the visual appearance of the rods with the poolside data. The visual exam will be performed using a camera system inside the hot-cell in four azimuthal orientations (~90 degrees).
2. Rod puncture and gas sampling; pressure and volume determination
3. Fission Gas Release (FGR) analysis from gas extracted above and FGR % calculation
4. Eddy Current (EC) oxide testing
5. Profilometry
6. Axial metallography for the end plug weld area and Heat Affected Zone (HAZ)
7. Metallography, including hydride distribution imaging, measurements of pellet diameter, cladding oxide and bonding layer thickness
8. Microhardness of metallographic cross section
9. Cladding hydrogen content by hot vacuum extraction including specimen preparation
10. Cladding axial tensile tests including photos of deformed samples
11. Cladding ring tensile tests including photos of deformed samples
12. Transmission Electron Microscope (TEM) analysis including Second Phase Particle (SPP) (imaging, density, size histograms and Energy Dispersive X-ray Spectroscopy (EDS) analysis) and dislocation characterization (imaging, density, size histograms)
13. Gamma scanning (pointwise spectra) including burn-up (BU) calibration, 1 m segments
14. Fluence measurement. Cut cladding ring samples for dosimetry measurements by PNNL
15. Disposal of spent fuel and secondary waste.

2nd Shipment

The Department of Energy (DOE) has initiated a project titled the "High Burnup Dry Storage Cask Research and Development Project (HDRP)." This project focuses on long-term aging issues important to the performance of the structures, systems, and components of the dry cask storage systems for high burnup spent nuclear fuel (SNF). The Electric Power Research Institute (EPRI) leads the project and would conduct the following actions:

1. Design, License (through the NRC), and implement a modification to the TN-32B cask and cask lid to allow for instrumentation
2. Select 32 high burnup SNF assemblies from Dominion Virginia Power's North Anna Power Station that show various characteristics of high burnup fuel
3. Identify 25 sister rods to be pulled from the 32 assemblies
4. Load the 25 sister rods into a NAC-LWT cask and ship them to the Idaho National Laboratory for post-irradiation examination (PIE) and testing
5. Load the 32 high burnup assemblies into the TN-32B cask and transport the cask onto the North Anna Independent Spent Fuel Storage Installation (ISFSI)
6. Monitor the cask for a period of 10 years.

The work INL would conduct supports items 2, 3, and 4 from the list above. A 'sister rod' is a fuel rod that has similar characteristics to those that would be stored in the Research Project Cask (the TN-32B storage cask). There are two potential donor fuel assembly sources for sister rods: assemblies having similar operating histories to those assemblies that are chosen for storage in the Research Project Cask or actual fuel assemblies selected for storage. Properties that must be similar in order to be considered a 'sister' include same fuel cladding type (e.g., Zircaloy-4, Zirlo, M5), same initial enrichment, same relative reactor core location, and the same reactor operating history. The sister rods are approximately 12’ 9” (~390 cm) in length and collectively contain approximately 44 kg heavy metal.

The current plan is for DOE to accept ownership of the sister rods at the North Anna site boundary. EPRI would be responsible for all logistics for the shipment from the North Anna site to the INL Site boundary. Upon arrival of the sister rods at INL, the NAC-LWT cask would be transferred to the Materials and Fuels Complex's (MFC's) Hot Fuels Examination Facility (HFEF). The sister rods would be unloaded into the HFEF hot cell, and placed in a safe storage condition until PIE could be performed. The exact PIE to be performed, as well as, the timing for the PIE is yet to be determined. It is anticipated the rods would be subjected to examinations and testing. The timing of the PIE work would be dependent on funding provided by DOE-HQ.

The sister rods are currently planned to be loaded and shipped to Idaho National Laboratory in January 2016. The approximate distance from the North Anna Power Station in Virginia to the Idaho National Laboratory is approximately 2,300 miles.

Initial Characterization

The rods would be subjected to high-resolution visual examination and eddy current evaluation for oxide thickness and defects. The rods would then be measured for length and diameter. Next, the rods would be gamma scanned to determine the burnup profile of the rods and any potential fission product relocation. Neutron radiography would be conducted on some rods to determine the transition from non-bonded to bonded fuel pellets and cladding. After data interpretation, many rods would be examined for fission gas pressure, fission gas sampling, and free volume determination. Once those data are gathered, decisions would be made regarding further analyses.

Analyses

Samples will be taken and subjected to optical microscopy at HFEF. Additional samples would be prepared for electro-optical examination and testing at the Electron Microscopy Laboratory. Mechanical properties of microstructural features would be measured using a Focused Ion Beam equipped with a compressive strength apparatus. Other samples would be sent to the Analytical Laboratory for isotopic analysis, fission gas analysis, and hydrogen content (cladding samples). Some samples would be transferred to the Irradiated Materials Characterization Laboratory for mechanical properties testing (e.g., bare cladding).

Special Testing

Five (5) rods have been identified for a special heat treatment that would mimic the vacuum drying process used to dry the large TN-32B storage cask. Rods have been selected that have “sisters” in locations where the cask temperature would be measured. The five rods can be heat treated in the HFEF using a furnace (to be built) that would replicate the heat profile to be measured in the TN-32B cask. These five rods would be replicates of the initial condition of the fuel in the storage cask at the beginning of the storage period. These rods would then be subjected to the analyses described above for comparison of the properties pre- and post-fuel cask drying.

Logistics

Each fuel rod would have an individual test plan detailing the characterization and testing proposed for that rod. These documents would be prepared when more information is available.

In order to complete proposed work activities, it is necessary for the project to use the HFEF hot cell that contains both defense and nondefense related materials and contamination. Project materials would come into contact with defense related materials. It is impractical to clean out defense related contamination, and therefore, waste associated with project activities could be eligible for disposal at the Waste Isolation Pilot Plant (WIPP). Low-level waste (LLW) generated by the project would likely be transferred to the Nevada National Security Site. It is expected that all generated wastes would be managed under existing BEA waste management plans and procedures.