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November 5, 2021

SUBJECT: Request for Exemption Associated with Disposal of Specified Columbia Fuel Fabrication Facility Waste (Docket No. 70-1151)

Westinghouse Electric Company, LLC (Westinghouse) requests NRC approval of alternate disposal of specified low-activity radioactive materials from the Columbia Fuel Fabrication Facility (CFFF), License No. SNM-1107, for certain waste containing byproduct material and special nuclear material (SNM). The authority of 10 CFR 20.2002, and the exemptions requested herein from the requirements in 10 CFR 30.3 and 10 CFR 70.3 for byproduct material and SNM would allow Westinghouse to transfer the specific waste for disposal at the US Ecology Idaho, Inc. (USEI) RCRA Subtitle C disposal facility near Grand View, Idaho. Idaho is not an Agreement State; however, Idaho regulations and the USEI facility permit provide for acceptance of this material based upon the exemptions requested.

Enclosure 1 provides an evaluation in support of this request and was developed in coordination with USEI. This document summarizes the candidate waste material, the proposed manner and conditions of disposal, the nature of the environment, and estimates the doses to members of the public during transportation operations and to USEI workers during railcar and truck receipt, unloading, transport and disposal per the requirements of 10 CFR 20.2002.

The evaluation conservatively limits the volume of material to be shipped to USEI in a calendar year, meeting the standard in NUREG 1757 generally limiting alternate disposal exposures to not more than “a few mrem per year” to any member of the public. The enclosed evaluation also projects that the candidate waste will be several orders of magnitude below concentrations that would present a criticality concern and U.S. Department of Transportation criteria for fissile material.

To support shipment of UF6 cylinders prepared for disposal, Westinghouse is requesting approval of this submission no later than March of 2022.

Please contact me at (803) 331-9425 should you have questions or need any additional information.

Elise Malek
Elise Malek (Nov 5, 2021 09:52 EDT)

Elise Malek
Regulatory Affairs Manager
Westinghouse Columbia Fuel Fabrication Facility
Docket 70-1151 License SNM-1107

Enclosure 1: Exemption Request Evaluation
Enclosure 2: USEI Approval for Alternate Disposal Authorizations
Enclosure 3: USEI Part B Permit
Enclosure 4: Sanitary Lagoon Sludge Sample Test Results

cc:
Mr. Thomas Vukovinsky
Mr. David Tiktinsky

Enclosure 1

Columbia Fuel Fabrication Facility Evaluation
In Support of 10 CFR 20.2002 Request For Alternate Waste Disposal

1.0 INTRODUCTION

Westinghouse Electric Company, LLC (Westinghouse) requests U.S. Nuclear Regulatory Commission (NRC) authorization for alternate disposal of specified low-activity waste containing special nuclear material (SNM) from the Columbia Fuel Fabrication Facility (CFFF), License No. SNM-1107. The authority of 10 CFR 20.2002, and the exemptions requested herein from the requirements in 10 CFR 30.3 and 10 CFR 70.3 pursuant to 10 CFR 30.11(a) and 10 CFR 70.17(a) for byproduct material and SNM would allow Westinghouse to transfer the specific waste for disposal at the US Ecology Idaho, Inc. (USEI) disposal facility located near Grand View, Idaho. The USEI disposal facility is a Subtitle C Resource Conservation and Recovery Act (RCRA) hazardous waste disposal facility permitted by the State of Idaho to receive radioactive waste that is not licensed or exempted from licensing by the NRC.

This request supports Westinghouse's continued and future on-site remediation efforts of site lagoons, facility capital improvement projects and upgrades, and site operations waste management. To minimize the number of requests submitted to the NRC for authorization for alternate disposal of multiple individual specified wastes, Westinghouse is requesting approval to dispose of the volumetrically and surface contaminated wastes described within this request based upon bounding dose calculations with the corresponding volume limits based upon the annual USEI worker exposure limit. The dose evaluations and projected volume limits are provided on an annual basis as the implementation of this request will cover several years. The cumulative annual exposure will be determined by the sum of the fraction method of the exposure from each of the two waste streams.

The dose evaluations for this request for alternate disposal were performed using US Ecology's NRC-Approved *Site Specific Dose Assessment Methodology* (SSDA) for USEI. The SSDA provides a consolidated dose assessment framework for all occupational, transportation, and post-closure dose receptors required in 10 CFR 20.2002(d) – "*Analyses and procedures to ensure that doses are maintained ALARA and within the dose limits in this part.*" The information provided in this enclosure as well as the Technical Evaluation Report documents and Safety Evaluation Report produced by the NRC serve to satisfy the requirements in 10 CFR 20.2002(a), (b), and (c). The NRC approved the SSDA for use on August 24, 2015 (ADAMS Accession No. ML15125A364, provided as Enclosure 2).

Characteristics and operating parameters of the USEI disposal site are summarized in Section 2 of this Enclosure. Environmental conditions at the USEI site are well-documented in previous submittals to the NRC, including the Westinghouse Hematite Decommissioning Project (Docket #70-00036) and the Humboldt Bay Nuclear Power Plant Decommissioning Project (Docket #50-133).

A description of the material to be disposed is included in Sections 3 and 4. The material description includes physical and chemical properties of the material important to risk evaluation and the proposed conditions of waste disposal. Results of the SSDA dose evaluation are summarized in Section 7 for all occupational and transportation workers as well as postulated members of the public based on USEI's ResRad model (Ver. 6.5) and Inadvertent Intruder Scenarios described in NUREG-0782, "*Draft Environmental Impact Statement on 10 CFR Part*

61 Licensing Requirements for Land Disposal of Radioactive Waste" and NUREG/CR-4370, "*Update of Part 61 Impacts Analysis Methodology - Methodology Report.*" Enclosure 3 contains the Waste Acceptance Criteria (WAC) set forth in USEI's permit issued by the Idaho Department of Environmental Quality. The SSDA Data Input Screens with the project inputs for the CFFF waste is provided in Enclosure 4. The conclusion confirms doses to workers and members of the public will be below NRC limits.

2.0 DISPOSAL SITE CHARACTERISTICS

The USEI site is located in the Owyhee Desert of southwestern Idaho. It is at the end of Lemley Road, approximately 17 kilometers (10.5 miles) northwest of Grand View, (Owyhee County) Idaho. Grand View has a population of approximately 340. Owyhee County is a ranching and agricultural area of approximately 19,900 square kilometers (7,678 square miles). The county is sparsely populated, with an average population of 0.5 people per square kilometer (1.4 people per square mile per Reference 1).

This region has an arid climate with an average annual precipitation rate of 7.4 inches. The USEI site is located on a 1.6 kilometer (1 mile) wide plateau. Maximum surface relief on the facility is 27 meters (90 feet) and the mean surface elevation is 790 meters (2,600 feet) above sea level. The nearest residence is 1.6 kilometers (1 mile) southwest of the site. There are no other land uses in the immediate vicinity of the site.

The operational performance characteristics of the USEI site have been reviewed by the NRC and determined to be protective within the NRC's "less than a few millirem (mrem) per year" policy for Alternate Disposal Requests first stated in NRC Regulatory Issue Summary (RIS) 2004-08, "*Results of the License Termination Rule Analysis*," and reaffirmed in SECY-07-0060, "*Basis for Justification and Approval Process for 10 CFR 20.2002 Authorizations and Options for Change.*" The NRC has previously granted USEI 10 CFR 70.17 special nuclear material and 10 CFR 30.11 byproduct material exemptions for purposes of disposal of various licensee waste streams. Two key documents are referenced from previous NRC submittals:

- Hazardous Waste Facility Siting License Application for Cell 16 (American Geotechnics, dated June 30, 2006); This document describes USEI's environmental setting and was accepted by the Idaho Department of Environmental Quality (IDEQ) as part of the 2005 siting process, which resulted in IDEQ approval (December 6, 2006) of USEI's request to expand its landfill operations. (ADAMS Accession No. ML100320540 - Attachment 7)

- Summary of Hydrogeologic Conditions and Groundwater Flow Model for US Ecology Idaho Facility, Grand View, Idaho (Eagle Resources, dated January 13, 2010. This document provides a detailed description of USEI's site geology and hydrogeology. (ADAMS Accession No. ML101170554 - Exhibit B)

3.0 ANNUAL MAXIMUM DISPOSAL VOLUME

3.1 DETERMINATION OF ANNUAL MAXIMUM DISPOSAL VOLUME

The approach to determine the annual maximum disposal volume is predicated on the USEI transportation workers and USEI site workers being exposed to less than 5 mrem/year when transporting, handling and disposing of CFFF waste; and the USEI WAC.

Using USEI's NRC-Approved SSDA, the maximum annual disposal volume was determined for each waste type. Bounding concentrations were used in each of the SSDA runs using the USEI WAC maximum concentration of 3,000pCi/g. The USEI WAC is the maximum concentration that is allowed for disposal of low level radiological waste at the USE landfill in Idaho. For volumetrically contaminated waste the maximum annual disposal volume, which results in an annual max dose of 4.98mrem is 322,000 ft^3. The maximum annual surface contaminated waste disposal volume, which results in an annual dose of 4.98mrem, is 122,000 ft^3.

3.2 DETERMINATION OF DISPOSAL VOLUME TO BE SHIPPED ANNUALLY

The volume of volumetrically contaminated waste and surface contaminated waste shipped during a year will be based upon not exceeding the "less than 5 mrem/year" criteria by utilizing a "sum of the fractions" calculation as follows;

For volumetrically contaminated waste - $\frac{5 \text{ mrem/year}}{322{,}000 \text{ ft}^3\text{/year}} = \frac{1.55\text{E-5 mrem}_{vc}}{\text{ft}^3}$

For surface contaminated waste - $\frac{5 \text{ mrem/year}}{122{,}000 \text{ ft}^3\text{/year}} = \frac{4.1\text{E-5 mrem}_{sc}}{\text{ft}^3}$

$$\sum \left[\left(\frac{1.55\text{E-5 mrem}_{vc}}{\text{ft}^3} \text{ x total}_{vc} \text{ ft}^3\right) + \left(\frac{4.1\text{E-5 mrem}_{sc}}{\text{ft}^3} \text{ x total}_{sc} \text{ ft}^3\right) \right] < 5 \text{ mrem}$$

4.0 WASTE DESCRIPTION

The waste descriptions below are intended to clearly define the waste material included in this request and bound the volume of each material allowed to be shipped offsite. This approach parallels the approved Humboldt Bay decommissioning project alternate disposal request as described in the associated safety evaluation report (ADAMS Accession No. ML12244A100). This section along with Section 5 describe the process by which Westinghouse will sample and analyze the waste to determine conformance with the USEI WAC.

4.1 VOLUMETRICALLY CONTAMINATED WASTE

Volumetrically Contaminated Waste for the purposes of this request is defined as:

- CaF_2 sludge dredged from site lagoons.
- Sanitary Lagoon sludge.

- Sanitary Lagoon liner.
- Contaminated soil including small stones, rocks and incidental amounts of vegetation from under and adjacent to the Sanitary Lagoon.
- Contaminated soil including small stones, rocks and incidental amounts of vegetation from the CaF_2 pad repairs.

Absorbent materials such as Portland Cement®, Power Pellets®, DrySorb® or equivalent material may be mixed with the above-mentioned materials to ensure no free-standing liquids are present within the shipping package as required. The volumetrically contaminated waste being considered under this request are contaminated with SNM (low enriched uranium {<5wt% U-235}) and Technetium-99 (Tc-99). The request is bound to the materials and volume limits as described below.

4.1.1 CALCIUM FLOURIDE SLUDGE

Calcium fluoride is a by-product of CFFF site operations. It is pumped in very low concentrations to lagoons where the solids settle and are periodically dredged from the lagoon bottom, dewatered and placed in a large pile for further drying prior to disposal. CFFF is taking steps to reduce the amount of Uranium and Tc-99 in the CaF_2 with the goal of reducing the activity concentrations to below the free release limit of 30-picocuries per gram. The CaF_2 currently in the lagoons is expected to contain similar activity concentrations as described in ADAMS Accession No. ML21039A719. The test results show the waste concentration is well under the assumed concentration in the dose assessment calculations as described in Section 7 below. A maximum total of 400,000 ft^3 of CaF_2 will be shipped from the North, South, West 1 and West 2 lagoons.

4.1.2 SANITARY LAGOON REMOVAL

Currently, CFFF site sanitary sewage is treated and then discharged to a polishing lagoon, known as the Sanitary Lagoon. To reduce environmental risk at the site, CFFF is in the planning stages of decommissioning the Sanitary Lagoon. Volumetric waste associated with removal of the Sanitary Lagoon has three main components:

- Lagoon sludge
- Lagoon liner
- Soil under and adjacent to the lagoon

Sludge samples were collected from the Sanitary Lagoon in 25 locations as identified in Enclosure 4. Three additional samples were collected which included an additional sampling point near the input pipe (SLS-B1) and two duplicate samples (SLS-B2 as a blind field duplicate for SLS-1, and SLSB-3, as a blind field duplicate for SLS-19). The results of the radionuclide analysis are provided in Enclosure 4. The sludge results are expected to bound the soil concentrations.

Up to 100,000 ft^3 of sludge and 300,000 ft^3 of soil may be generated from the closure of the Sanitary Lagoon. Soil excavation volume was calculated assuming approximately 4’ of soil will be removed under and directly adjacent to the lagoon. Waste volumes will be closely tracked to ensure that the calculated dose associated with disposal does not exceed the yearly aggregate waste volume limit. The lagoon liner is an industrial landscape fabric to limit vegetation growth that lines the upper rim of the lagoon and will be disposed of with the soil and sludge. Waste will include soil, rock, liner and incidental amounts of vegetation.

4.1.3 CaF_2 PAD REPAIRS

The CaF_2 pad has degraded over time and remediation efforts are planned in 2022. The repairs to the pad are expected to result in some soil and debris removed with contamination levels bound by the CaF_2 sludge results as described in ADAMS Accession No. ML21039A719, which was the last pile of CaF_2 on the pad and which had the highest recorded activity concentration. The soil will be treated as volumetric contaminated waste and the concrete debris will be treated as surface contaminated waste as described in Section 4.3.2. A maximum of 75,000 ft^3 of soil and debris will be generated from the pad remediation project.

4.2 EXAMPLES OF RADIOLOGICAL CHARACTERIZATION OF VOLUMETRICALLY CONTAMINATED WASTE

Typical waste survey methods will be either by analytical laboratory analysis, or gamma-spectroscopy combined with ISOCS modeling. Analytical laboratory analysis will determine Isotopic Uranium and Tc-99 concentrations. Gamma-spec analysis may be used when the Tc-99 values are known, or in conjunction with laboratory sampling.

The following examples of the radiological characterization of candidate volumetrically contaminated waste is provided to demonstrate that the candidate volumetrically contaminated waste will likely meet the USEI WAC of 3,000 pCi/g total activity (the sum activity of all radionuclides present). The characterization data provided below is the analytical laboratory average radiological concentration data collected from each potential waste stream that has been sampled for disposal at USEI to date.

Waste Description	**Gross Analyte Activity (pCi/g)**				**Total Activity (pCi/g)**
	U-234	**U-235**	**U-238**	**Tc-99**	
East Lagoon Sludge Mixture	605.6	29.8	117.0	4.0	756.4
CaF_2 generated in 2020	46.6	2.3	6.8	0	55.6
Soil under East Lagoon	13	0.7	3.4	2.6	19.7
Sanitary Lagoon Sludge Samples	1,136.8	67.4	242.8	87.1	1534.1

Radiological concentrations may be determined prior to or after packaging the waste as described in Section 5. Any packaged waste that is determined to have radiological concentrations of 3,000 pCi/g total activity or higher will not be shipped to USEI for disposal.

4.3 SURFACE CONTAMINATED WASTE

Site chemical operations undergo constant maintenance activities and configuration changes to maintain or improve safe plant operations. These activities generate surface contaminated waste. Surface contaminated waste for the purposes of this request is defined as:

- Obsolete UF6 Cylinders.
- CaF_2 pad demolition debris including concrete, drainage piping and incidental amounts of soil and rock.
- Sanitary Lagoon demolition debris including concrete, rebar, brick, mortar, piping, tanks and valves.

The surface contaminated waste being considered under this request is contaminated with SNM (low enriched uranium {<5wt% U-235}) and potentially Technetium-99 (Tc-99).

4.3.1 UF6 CYLINDERS

The UF6 Cylinders are transportation containers that are no longer in service. The UF6 Cylinders are solid form (steel), approximately 7 feet in length and 2.5 feet in diameter. The UF6 Cylinders are empty and had previously been through the UF6 Cylinder internal wash/rinse process prior to being placed into storage for pending disposal. The UF6 Cylinders will be downsized to eliminate void space prior to packaging for shipment off-site for disposal. While emptied and cleaned to the standard mentioned, the UF6 Cylinders are internally contaminated with SNM. The UF6 Cylinders will be transported to the USEI site by means of trucks. A maximum of 1,000 cylinders will be shipped under this request, conservatively totaling 50,000 ft^3 of waste volume.

4.3.2 DEMOLITION DEBRIS

Two planned site environmental improvement projects will require remediation of contaminated components. Specifically, the repairs to the CaF_2 pad and the removal of the Sanitary Lagoon. The CaF_2 pad work will generate concrete and drainage piping debris with contamination bound by the results of the CaF_2 sludge results and volume limits as specified in section 4.1.3. In addition, the Sanitary Lagoon removal will generate surface contaminated demolition debris including tanks, piping, valves, concrete, brick, mortar and rebar from removal of the lagoon itself and supporting system components that will require removal as part of the system redesign. Sanitary lagoon demolition debris volume will be limited to 75,000 ft^3.

4.4 EXAMPLES OF RADIOLOGICAL CHARACTERIZATION OF SURFACE CONTAMINATED WASTE

Typical waste survey methods will be either direct alpha scan survey, or gamma-spectroscopy combined with ISOCS modeling. Alpha scan survey data will be collected in disintegrations per minute (dpm) and converted to Uranium activity in µCi, the items total surface area will be used to determine total activity, and the items weight will be used to determine the radioactive concentration in pCi/g. Gamma-spec survey data will be collected in U-235 and/or U-238 total activity per item using ISOCS modeling software. Using the plant nominal enrichment, U-234 activity will be calculated, and the total U activity per item, and radioactive concentration will also be determined. These pCi/g concentrations will be compared to the USEI WAC of 3,000 pCi/g total activity to ensure compliance for disposal.

The following examples of the radiological characterization of candidate surface contaminated waste are provided to demonstrate that the candidate surface contaminated waste will likely meet the USEI WAC of 3,000 pCi/g total activity. The characterization data provided below are examples of radiological survey methodology that could be used to evaluate any potential surface contaminated waste that has been collected for disposal at USEI.

Waste Description	Alpha Scan Survey		
	dpm/100cm2	U activity (µCi)	pCi/g
UF6 Cylinder RBU2447	148,000	46.6	69.4
UF6 Cylinder NB15413	254,000	80.0	119.2
Waste Description	**Gamma-Spec Survey**		
	U-235 activity (µCi)	**U activity (µCi)**	**pCi/g**
UF6 Cylinder WEC1034	1.550	44.9	75
UF6 Cylinder CE0193	9.124	264.5	441

The alpha scan survey data was collected on the interior of UF6 cylinders after the cylinder had been cut and was open for scanning. The maximum result was conservatively used, and applied to the entire surface area of the cylinder, and then converted to Total U activity. This activity, divided by the weight of the cylinder determines the pCi/g concentration. The gamma-spec survey was also collected on the interior of UF6 cylinders after the cylinder had been cut and was open for scanning. ISOCS Modeling software was used to determine the total U-235 activity on the cylinder interior, then the plant nominal enrichment was used to determine total U activity. This activity, divided by the weight of the cylinder determines the pCi/g concentration.

After radiological survey, the candidate waste will be packaged for transportation and disposal. Multiple surface contaminated objects will likely be placed into the same waste container (e.g. super-sack, drum, B-25, or equivalent), and the package will be assigned an ID and manifested.

Any packaged waste that is determined to have radiological concentrations of 3,000 pCi/g total activity or higher will not be shipped and disposed of at USEI.

5.0 WASTE SURVEY, SAMPLE, AND ANALYSIS

The volumetric wastes described above may be collected individually or combined in any mixture. Individual waste streams, or waste mixtures, will be sampled representatively to determine radiological concentrations. If individual waste streams are sampled and then combined with other waste streams, a weighted average concentration will be determined, or conservatively the highest radiological concentrations from the individual waste streams may be applied to the total waste mixture. All volumetric waste will be subject to a minimum of 1 (one) radiological sample (Iso-U, and Tc-99) from each approximate 100 cubic yards of volumetric material. To ensure a representative sampling, at least one aliquot will be collected from each approximate 25 cubic yards of material, and each sample aliquot will be combined and homogenized for sample collection. Each sample will be analyzed by an offsite laboratory, and/or gamma-spec to confirm that the material meets the USEI WAC prior to shipment offsite.

All surface contaminated waste destined for disposal at USEI will be radiologically surveyed on-site prior to packaging for transportation. Radiological survey to determine total activity may be performed via direct alpha scan, in-situ gamma spectroscopy, or by other appropriate methods. In the case of direct alpha scan, an alpha measurement will be converted to activity, and conservatively applied over the objects surface to determine the total activity of the item. In the case of in-situ gamma spectroscopy, In-Situ Object Calibration Software (ISOCS) will be used to determine activity. After radiological survey, the surface contaminated wastes described above may be collected and packaged individually or combined and packaged in any mixture.

5.1 MATERIAL CONTROL & ACCOUNTING

The results of waste survey, sample and analysis as described above will be utilized for Nuclear Material Control, Inventory, and Accounting purposes such as completion DOE/NRC Form 741, Nuclear Material Transaction Report and NRC Form 540 Uniform Low-Level Radioactive Waste Manifest Shipping Paper.

6.0 RADIOLOGICAL CONTROLS OF ONSITE WORK ACTIVITIES

The CFFF Site has been granted NRC License SNM-1107. Chapter 5.2.42 of the associated license application states: “Adults likely to receive greater than 0.5 rem in a year, from sources external to the body, are monitored by personnel dosimeters”. This monitoring requirement applies to all occupationally exposed workers at the CFFF, including USEI personnel and other sub-contractors. A prospective As Low As Reasonably Achievable (ALARA) analysis is performed during pre-job activities, and a Radiation Work Permit (RWP) is developed to document the necessary personnel monitoring requirements and Health Physics oversight for performing the work.

The RWP covers all of the radiation safety controls; contamination control, air sampling, protective clothing, and any bioassay sampling requirements as applicable for the scope of work performed at CFFF. Personnel working under the RWP are included in the applicable monitoring programs governed by CFFF’s NRC license. Only personnel who have completed required safety training and are on the approved personnel list are assigned to work under an RWP.

7.0 RADIOLOGICAL ASSESSMENT

As described in the following exposure scenarios, the dose equivalent for the Maximally Exposed Individual (MEI) has been demonstrated to not exceed "a few mrem per year." The standard of a "few mrem per year" to a member of the public is set forth in NRC RIS 2004-08, "*Results of the License Termination Rule Analysis*." The NRC has clarified in the *Guidance For The Review Of Proposed Disposal Procedures And Transfers Of Radioactive Material Under 10 CFR 20.2002 and 10 CFR 40.13(a)* final draft that "a few mrem per year" should be understood as less than 5 mrem/year.

Version 3b of the SSDA was used in this assessment. In August of 2020 version 3a was amended to allow more job specific parameters to be used in the inadvertent intruder scenarios, primarily to the dilution factor. Version 3b has been reviewed and approved by the NRC, and was used in a recent alternate disposal request submitted by Vermont Yankee. External exposure assessments in the SSDA were performed using MicroShield Code, Version 7.02. Evaluations of potential external and internal dose hazards are discussed in the sections that follow while all inputs to the SSDA workbook are provided in Enclosure 4. A summary of total estimated doses for all transporters, as well as USEI workers performing surveying, handling, treatment and disposal tasks on the CFFF waste is provided in Table 7.1.

As mentioned in Section 3, total volumes for each waste type were determined in the SSDA runs by using bounding concentrations, which are consistent with USEI's WAC limits of 3,000pCi/g total activity.

It is expected that a majority of the surface contaminated material will be transported to USEI by means of trucks, such as 50 cubic yard / 22 ton capacity aluminum end-dump truck or Conestoga style aluminum trailers, while volumetric waste will be shipped by rail in gondola cars. In addition, there is potential for mixed material consisting of volumetric and surface contaminated waste. This type of waste will likely be shipped in gondolas cars similarly to volumetric waste. Two bounding SSDA models were run to account for the different generated waste streams, one for surface contaminated material shipped by truck, and another for the shipments of volumetric waste by rail. Since any loads containing a mixture of both volumetric and surface contaminated material would be shipped by rail, the SSDA run accounting for just volumetric material is used as the bounding volumetric limit for both volumetric and mixed material. This approach is conservative as mixed material loads will have a lower density than loads that just consist of volumetric material. With all material modeled at the same bounding concentrations, lower density material will lead to a higher number of shipments (larger calculated volumes) to achieve the same overall dose, when compared to the volumetric SSDA run. Dose results are summed for any job functions that are shared between the two models. In addition, post closure dose scenarios are summed between the two models to report the overall post closure dose. Results are discussed below.

7.1 TRANSPORT DOSE TO THE PUBLIC

All materials will be transported by truck or a combination of truck and rail to the USEI facility in Grand View, ID. All conveyances will be verified to comply with DOT external loose surface contamination limits prior to shipment. Therefore, transport will not pose the potential for internal dose to the drivers or other members of the public. All loads will meet the DOT requirements for packaging.

Volumetric and mixed material waste will be loaded into IP-1 bags and staged for transport. Once the bag is ready to be shipped, it will be lifted into a standard 50 cubic yard / 22 ton capacity aluminum end-dump truck, or Conestoga style aluminum trailers. Two bags will be placed in each truck. The truck will then be tarped and proceed 5 miles to the railyard where the bags will be lifted into lined gondola railcars. Accounting for the estimated volume of material, per the SSDA model, approximately 604 truckloads per year will be required to haul bagged material to the railyard. Modeled doses to the truck drivers for this process are reported in Table 7.1.

Transportation dose with respect to the gondola cars is expected to be very low. Calculated exposure rates 1 meter from the surface of the rail car would be 2.70E-3 mrem/hr. In order for a member of the public to receive a dose greater than a few mrem, they would have to stand within 1 meter of the car for nearly 2,000 hours. This is a very unlikely scenario and not considered to be credible.

The backend-dray portion of the transportation takes approximately 45 minutes from the Rail Transfer Facility (RTF). Truck transport is shared between 8 drivers. The dose model assumes the driver sits 0.6 meters from the material. Dose results are reported in Table 7.1.

Surface contaminated waste will be transported to USEI by lined 50 cubic yards/23 ton capable aluminum end-dump trucks, or Conestoga style aluminum trailers. As supported by the SSDA model for Surface Contaminated waste, up to 122,000 cubic feet, with concentrations ≤ 3,000 pCi/g can be shipped and maintain doses below a "few" millirem. If applicable, the material, e.g. UF6 Cylinders, will be cut open to eliminate voids before being loaded into the IP-1 bags and then loaded onto the trucks. The distance from CFFF to the USEI disposal facility is approximately 2,520 miles. Assuming an average speed of 55 miles per hour, the trip is estimated to take 46.4 hours. Five drivers are assumed to transport this material for purposes of dose modeling in the SSDA. Since approximately 158 truckloads would be required to transport the estimated 122,000 cubic feet per the SSDA model, more drivers/trucks may be used as needed. If more than five drivers are used, the doses reported in Table 7.1 will go down. The SSDA model assumes the driver sits approximately 0.6 meters from the edge of the contaminated load with 0.25 inches aluminum shielding between him and the surface contaminated material. The shielding accounts for the aluminum end of the trailer. The external dose rate to the truck drivers is calculated to be very low (3.46E-3 mrem/hr), and as a result, the dose to other members of the general public can reasonably be concluded to be minimal and below the required limit of a few mrem.

The MEI for transportation dose to the public as described above is the Back-End Dray driver with a max calculated dose of 1.36E-01 mrem/year.

7.2 USEI WORKER DOSE ASSESSMENT

External dose rates in the SSDA are calculated using dose-to-source ratios (DSR) developed with the Micro Shield Code, Version 7.02. A total dose rate for all nuclides present is calculated by summing the contributions from the individual nuclides. Specifics for these templates used for each job function can be found in the Technical Basis Document for the SSDA. In addition, below are summaries of each USEI worker function and the assumptions used in performing the dose calculations.

Internal Doses are calculated using Dose Conversion Factors from Federal Guidance Report 11 for all of the radionuclides present. The SSDA uses a dust loading fraction of 2.3E-04 g/m^3, and a standard man breathing rate of 1.2 m^3/hr for light work. (ICRP, 2004). A total dose rate for all nuclides present is calculated by summing the contributions from the individual nuclides.

Based on the SSDA dose modeling performed, both the long-haul truck driver and the RTF Excavator Operator are the MEI's with calculated doses of 4.98 mrem/year. These doses are within the few millirem requirement. Results for the below described functions are reported in Table 7.1 below.

Gondola Railcar Surveyor
Upon receipt at USEI's RTF, the gondolas will be surveyed and screened prior to transloading the material to trucks and transporting to USEI Site 2 for direct disposal. Approximately 10 minutes is required to perform a survey of each gondola. Based on current practice, a surveyor is assumed to stand at a distance of one meter from the gondola during the survey, with four surveyors sharing the task.

RTF Excavator Operator
All transloading of material are done within a containment building employing a 24,000 cubic feet per minute (cfm) filtration system. An excavator positioned on top of a bridge platform above the railcar will transfer the material into end-dump trucks. During off-loading operations, the excavator operator remains in the cab that pulls air through a filtration system.

For dose modeling it is assumed off-loading of a gondola car can take up to 45 minutes. The operator sits approximately 2 meters from the material. Two excavator operators share these activities.

Gondola Railcar Cleanout
Once a railcar is off-loaded, USEI personnel will remove any residual material inside of the railcars with shovels and brooms. This operation normally takes 10 minutes to complete. Four personnel share this task. The dose rate is modeled at 30 cm from a ½ layer of waste material.

RTF Truck Surveyor
Once trucks are loaded, surveys will be performed and screened prior to the material being sent to the disposal site. Truck surveys take 5 minutes to perform. Surveyors are assumed to stand one meter from the truck or trailer during the survey. Four surveyors share this task.

Disposal Site Truck Surveyor
Since the surface contaminated materials are being transported directly to the disposal site, surveys will be performed there and not at the RTF. Modeling assumptions are the same for this function as they are for the RTF truck surveyor.

Cell Operator

After delivery to the disposal cell, a bulldozer operator wearing a respirator within an enclosed cab, spreads and compacts the waste. For this dose scenario the deposited material is based on the volume of one gondola car. It is assumed that 15 minutes is needed to spread and compact the volume of material, equivalent to one gondola car. Two personnel share this responsibility. It is important to note that this function will be shared between the two SSDA models, specifically. Doses from both models are therefore summed to calculate total dose for the project.

Table 7.1
Results of SSDA Dose Evaluation for CFFF Waste Project

Function	Minimum Number of Workers	Waste Contact Time (hr)	External Exposure Rate (mrem/hr)	Internal Dose Rate (mrem/hr)	Distance (m)	Total No. of Repetitions	Total External Dose per Worker (mrem)	Total Internal Dose per Worker (mrem)	Total Project Dose per Worker (mrem)	% of Max Annual MEI Dose
Volumetric/Mixed Material										
Front-End Dray Truck Drivers	4	0.09	4.06E-03	0.00E+00	0.6	604	5.58E-02	0.00E+00	5.58E-02	1.1%
Gondola Railcar Surveyors	4	0.33	2.81E-03	0.00E+00	1.0	121	2.80E-02	0.00E+00	2.80E-02	0.6%
Bulk/IMC Truck Surveyors (RTF)	4	0.08	3.18E-03	0.00E+00	1.0	356	2.27E-02	0.00E+00	2.27E-02	0.5%
RTF Excavator Operator	2	0.75	2.14E-03	1.08E-01	2.0	121	9.70E-02	4.88E+00	4.98E+00	99.6%
Gondola Railcar Cleanout	4	0.16	2.70E-03	1.08E-01	0.3	121	1.31E-02	5.21E-01	5.34E-01	10.7%
Back-End Dray Truck Drivers	8	0.75	4.06E-03	0.00E+00	0.6	356	1.36E-01	0.00E+00	1.36E-01	2.7%
Landfill Cell Operators	2	0.25	7.91E-04	1.08E-01	1.0	242	2.39E-02	3.25E+00	3.28E+00	65.6%
Surface Contaminated Material										
Long-Haul Direct Truck Drivers - Drive Time	5	45.45	3.46E-03	0.00E+00	0.6	158	4.98E+00	0.00E+00	4.98E+00	99.5%
Bulk/IMC Truck Surveyors (disposal site)	4	0.08	2.81E-03	0.00E+00	1.0	158	4.45E-03	0.00E+00	4.45E-03	0.0%
Landfill Cell Operators	2	0.25	7.94E-04	1.08E-01	1.0	38	3.77E-03	5.13E-01	5.16E-01	10.3%
Project Dose-Totals										
Long-Haul Direct Truck Drivers - Drive Time							4.98E+00	0.00E+00	4.98E+00	99.5%
Front-End Dray Truck Drivers							5.58E-02	0.00E+00	5.58E-02	1.1%
Gondola Railcar Surveyors							2.80E-02	0.00E+00	2.80E-02	0.6%
Bulk/IMC Truck Surveyors (RTF-highest dose reported)							2.27E-02	0.00+00	2.27E-02	0.5%
RTF Excavator Operator							9.70E-02	4.88E+00	4.98E+00	99.6%
Gondola Railcar Cleanout							1.31E-02	5.21E-01	5.34E-01	10.7%
Back-End Dray Truck Drivers							1.36E-01	0.00E+00	1.36E-01	2.7%
Landfill Cell Operators (summed)							2.78E-02	3.76E+00	3.79E+00	75.9%

7.3 POST CLOSURE DOSE TO THE GENERAL PUBLIC

USEI's RCRA permit requires that it demonstrate that no person will receive an annual dose exceeding 15 mrem for 1,000 years after closure of the facility. This standard is more restrictive than the annual 25 mrem total effective dose equivalent (TEDE) stated in 10 CFR 20.1402 for NRC license termination, as well as the limits for near surface disposal of low-level radioactive waste set forth in 10 CFR 61. RESRAD code Version 6.5 was used for modeling the Grand View site for potential long-term post-closure doses. A number of default parameters in the Grand View model have been replaced with site specific parameters consistent with the facility's 2005 permit modification and a report prepared by its consultant (previously submitted to the NRC as part of a Request for Additional Information response for the exemption request for the Westinghouse Hematite project, Docket #070-00036, ML12135A301).

The SSDA contains a screening RESRAD model to assess the impact of the CFFF waste on the USEI site. The model is consistent with USEI's post-closure dose model included in the Part B RCRA permit, which assumes that all of the CFFF waste is distributed evenly within the contaminated zone (area = 88,221 m^2, depth = 33.6 m). 'Screening' in the SSDA means that ALL nuclides are evaluated at their peak dose-to-source ratio regardless of when it occurs. The radionuclide concentrations are automatically adjusted in the SSDA Workbook to reflect aggregation into the entire landfill volume. All other RESRAD code parameters remain the same. The results of the screening model show a maximum annual dose of 1.04 mrem. Due to the very low dose projection from the screening model, a separate project-specific dose model was not necessary.

Three post-closure inadvertent intruder scenarios were also conducted using the framework from NUREG-0782, "*Draft Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste*," and NUREG/CR-4370, Volume 1, "*Update of Part 61 Impacts Analysis Methodology*" built into the SSDA. These scenarios include:

- Intruder Construction Scenario – An inadvertent intruder may excavate or construct a building on a disposal site following a breakdown in institutional controls. Under these circumstances, dust will be generated from the application of mechanical forces to the surface materials (soil, rock) through tools and implements (wheels, blades) that pulverize and abrade these materials. The dust particles generated may be then entrained by localized turbulent air currents and can thus become available for inhalation by the intruder. The intruder may also be exposed to direct gamma radiation resulting from airborne particulates and by working directly in the waste-soil mixture. The Construction Worker scenario uses the Air Uptake and Direct Gamma Exposure pathways to estimate a total dose to the intruder.

- Intruder Well Drilling Scenario – An intruder accesses the site and develops a well. The intruder is exposed to contaminated drill cuttings spread over the ground surface and contaminated airborne dust. The scenario presented in NUREG/CR- 4370 was modified to exclude consideration of exposure to cuttings in a mud pit due to the standard practices in the area around the waste site. The assumption that drill cuttings are spread over the ground will result in higher dose estimates than if the cuttings were assumed to be in a mud pit because of the decrease in the shielding factor. The driller is assumed to work on site for a period of 40 hours and it is assumed that the contaminated layer is drilled through in 8 hours. As such, the driller is assumed to be exposed to the undiluted cuttings for 8 hours and to diluted material for the balance of the exposure duration. The dilution is calculated based on the ratio of the depth of the waste layer to the total well depth. No dilution in the USEI landfill is assumed. The Well Driller scenario includes contributions from Internal and External dose to the intruder.

- Intruder Driller Occupancy Scenario - An inadvertent intruder occupies the site upon which a well had been drilled through waste materials. The Driller Occupancy Scenario uses the same concentrations in the exhumed well cuttings as the Well Driller scenario. The Driller Occupancy scenario uses the Air Uptake and Direct Gamma Exposure pathways to estimate a total dose to the intruder.

To be more complete with respect to post closure dose modeling, additional intruder scenarios were considered. These inadvertent intruder scenarios are not held to the same post closure dose standard as USEI's RCRA permitted RESRAD model as the NRC allows up to a 500 mrem/year dose limit (NUREG-2175). With relation to the material of interest, the estimated inadvertent intruder doses for the three above scenarios were calculated to be 4.35 mrem for the Construction Scenario, 3.65E-01 mrem for the Well Driller Scenario, and 7.65E-01 mrem for the Driller Occupancy Scenario, as reported in Table 7.2. Even though a higher dose is allowed for inadvertent intruder post closure scenarios, each of these estimated doses meet USEI's RCRA permit post closure dose limit of 15 mrem. These models are very conservative by design as to have flexibility to be used as a general tool for various types of sites. As mentioned previously, the SSDA version 3b was used in this analysis. Version 3b allows more realistic dilution factors to be used in each of the intruder scenario. Annual requested volumes contained within this document were compared to USEI's overall average annual volume receipts to calculate the dilution factor (f_d). For volumetric waste the f_d was 0.064, where the value used for surface contaminated waste was 0.03. USEI's total average annual volume receipts from 2015-2020 was 4.99+06 cubic feet. Even with using a more realistic dilution factor, these estimates are conservative and likely not realistic, which is especially the case for the construction scenario. For example, USEI has a requirement that all radiological waste must be placed no closer than 3.6 meters from the top of the constructed cap. The Intruder Construction scenario assumes excavation for constructing a building up to 3.0 meters below the surface with the lower 1.0 meter consisting of waste. Realistically in this scenario, the waste will not even be disturbed by the construction activities.

Table 7.2
USEI SSDA Post Closure Results for CFFF Waste Project

	Volumetrically Contaminated Waste (mrem/yr)	Surface Contaminated Waste (mrem/yr)	(mrem/yr)
USEI RESRAD Post-Closure Screening Dose	1.01E+00	2.60E-02	1.04E+00
Inadvertent Intruder Doses			
1. Construction Scenario	3.64	7.07E-01	4.35E+00
2. Well Driller Scenario	2.50E-01	1.15E-01	3.65E-01
3. Driller Occupancy Scenario	5.41E-01	2.24E-01	7.65E-01

8.0 CRITICALITY SAFETY

A Criticality Safety Assessment for the USEI site was performed as part of a prior alternate disposal application by the Westinghouse Hematite site. The "Nuclear Criticality Safety Assessment of the US Ecology Idaho (USEI) Site for the Land Fill Disposal of Decommissioning Waste from the Hematite Site, Rev. 2 (NSA, 2011)" verified that wastes containing U-235 may be sent to the USEI site for disposal since very large margins of safety had been incorporated into the normal operating conditions associated with these wastes and the probability for serious abnormal conditions is acceptably small. A maximum fissile concentration of 0.1 gram U-235 per liter of media was developed as an inherently safe concentration of SNM for the exhumed Hematite waste materials. This converts to an equivalent activity concentration of 216 pCi/g U-235 in soil (assuming a soil density of 1 g/cc).

To achieve the average activity concentration, the candidate waste will be aggregated as described in Section 4.0. It is intended to only utilize the waste described in this submittal for aggregation of waste. To ensure the activity of the candidate waste as packaged for shipment does not exceed an average activity concentration of 216 pCi/g U-235, the waste will be sampled to verify the average activity concentration is acceptable for disposal at USEI. USEI personnel will review the sample data to ensure acceptability of the waste for disposal prior to shipment to the USEI site.

Considering the characterization results of the candidate waste, USEI's WAC is the limiting factor as it would be exceeded before the 0.1 gram of U-235 per liter of media safety limit is reached.

9.0 CONCLUSIONS

Westinghouse has described the volumetrically contaminated and surface contaminated waste, including the physical and chemical properties important to risk evaluation. Westinghouse proposed the manner and conditions of waste disposal, the analysis and evaluation of pertinent information on the nature of the environment, and the nature and location of other potentially affected licensed and unlicensed facilities. Specifically, how the material will be transported via US Department of Transportation (DOT) regulations to US Ecology Idaho, Inc. (USEI), which is a Subtitle C Resource Conservation and Recovery Act (RCRA) hazardous waste disposal facility permitted by the State of Idaho to receive radioactive waste that is not licensed or exempted from licensing by the NRC. As such, the material is authorized to be removed per state and local regulations and will be shipped per existing federal regulations to a location approved by the state of Idaho to receive the material. Therefore, the request is authorized by law pending NRC approval of the exemption.

Section 7.0, Radiological Assessment, describes exposure scenarios for the Maximally Exposed Individual, describes how the material will meet DOT regulations for transport and confirms that a person's annual dose will not exceed 15 mrem for 1,000 years after closure of the USEI facility. The expected dose is a small fraction of the NRC decommissioning limits for exposure to any member of the public of 25 mrem/year TEDE, and is within the "few mrem per year" criterion that the NRC has established in RIS 2004-08. Section 8.0, Criticality Safety, identifies that the USEI Waste Acceptance Criteria would not be exceeded before the safety limit of 0.1 gram of U-235 per liter of media would be reached. This analysis demonstrates the calculated radiation dose is ALARA and within NRC limits

CFFF maintains an NRC approved Physical Security Plan and shipments of the waste material for disposal at USEI will be conducted in accordance with applicable regulations which provide reasonable assurance the requested exemption will not endanger life, property or the common defense and security.

Finally, this request is directly related to CFFF's ongoing efforts to remediate legacy environmental concerns at the site. Closing lagoons with aging liners, remediating contaminated soil and calcium fluoride, and properly disposing of retired contaminated equipment is all in the best interest of the public. In addition, the waste concentrations of the shipped material will be a fraction of the calculated concentration making the dose assessment extremely conservative. The assumed debris volumes provided are also conservative to assure this request bounds the actual need.

Westinghouse has provided the aforementioned information to support, in accordance with 10 CFR 70.17(a), an NRC determination that the request for disposal of specific materials from the CFFF site at the USEI site is in accordance with applicable laws and regulations, will not endanger life or property or the common defense and security, and is in the interest of the public.

10.0 REFERENCES

10.1 American Geotechnics, "Hazardous Waste Facility Siting License Application Cell 16," Project No. 06B-C1202, June 30, 2006 (ML100320540 - Attachment 7)

10.2 Eagle Resources, Inc. "Summary of Hydrogeologic Conditions and Groundwater Flow Model for US Ecology Idaho Facility, Grand View, Idaho." January 13, 2010 (ML101170554 - Exhibit B)

10.3 US Ecology Idaho, Inc. USEI Site B Permit No. IDD073114654 (2004)

10.4 U.S. Nuclear Regulatory Commission, Regulatory Issue Summary 2004-08, "Results of the License Termination Rule Analysis." Office of Material Safety and Safeguards, May 28, 2004

10.5 U.S. Nuclear Regulatory Commission, "Basis and Justification for Approval process for 10CFR20.2002 Authorizations and Options for Change." SECY-070060. Division of Waste Management and Environmental Protection, March 27, 2007

10.6 U.S. Nuclear Regulatory Commission, "Guidance For The Reviews Of Proposed Disposal Procedures And Transfers Of Radioactive Material Under 10 CFR 20.2002 And 10 CFR 40.13(a)", Division of Uranium Recovery, Decommissioning, And Waste Programs Guidance Document, October 16, 2017

10.7 Nuclear Safety Associates "Nuclear Criticality Safety Assessment of the US Ecology Idaho (USEI) Site for the Land Fill Disposal of Decommissioning Waste from the Hematite Site, Rev. 2" NSA-TR-09-14

10.8 NUREG-2175, "Guidance for Conducting Technical Analysis for 10 CFR Part 61, Nuclear Regulatory Commission, Washington, DC, March 2015

10.9 Federal Guidance Report No. 11: Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, EPA 520/1-88-020 September 1988

10.10 NRC License SNM-1107, Westinghouse Electric Company, LLC

10.11 Westinghouse Columbia Fuel Fabrication Facility Fundamental Nuclear Material Control Plan

10.12 Safety Evaluation Report; Docket 70-1151; License SNM-1107; Request for 10 CFR 20.2002 Alternate Disposal Approval and Exemptions From 10 CFR Part 30 and 10 CFR Part 70 for Disposal of Columbia Fuel Fabrication Facility Waste at the US Ecology Idaho Facility (ML 20302A085)

Enclosure 2

Copy of Letter from L. Camper to J. Weismann approving use of USEI SSDA for 10 CFR 20.2002 Alternate Disposal Authorization Requests, August 24, 2015 (ML15125A364)

August 24, 2015

Mr. Joseph J. Weismann, CHP
Vice President of Radiological Programs
and Field Services
US Ecology, Inc.
Lakepointe Centre I
300 East Mallard Dr., Suite 300
Boise, ID 83706

SUBJECT: US ECOLOGY, INC. – TECHNICAL EVALUATION REPORT OF US ECOLOGY IDAHO'S PROPOSED METHODOLOGY SUPPORTING ALTERNATE WASTE DISPOSAL PROCEDURES IN ACCORDANCE WITH 10 CFR 20.2002

By letter dated June 14, 2013, US Ecology, Inc. (USEI) requested an exemption to receive and dispose of low-activity radioactive waste from Studsvik's Processing Facility in Memphis, TN at USEI, a Resource Conservation and Recovery Act Subtitle-C hazardous and low-activity waste facility near Grand View, ID. USEI also requested that the U.S. Nuclear Regulatory Commission (NRC) review a newly developed Site-Specific Dose Assessment Methodology (SSDA). In a letter dated March 10, 2014, USEI withdrew the request to dispose of low-activity waste from Studsvik Processing Facility; however, USEI requested that the NRC continue to review the SSDA. USEI stated that this process provides a streamlined methodology for preparing and reviewing future 10 CFR 20.2002 alternate disposal requests (ADR) from USEI.

This Technical Evaluation Report (TER) documents the NRC staff's technical review of the proposed methodology. Similar to a review of a 10 CFR 20.2002 exemption request, the NRC staff performed a technical review of the methodology and associated documents and evaluated the technical basis and assumptions incorporated into the calculations used by USEI. The NRC staff also used the methodology to evaluate a previously evaluated exemption request and compared the conclusions. Based on this review, the NRC staff considers the use of USEI's SSDA to be an appropriate method for evaluating future proposed disposals. The SSDA methodology can be used to satisfy the criteria in § 20.2002 (d); however, individual 20.2002 requests by USEI, or other licensees wishing to ship to USEI, must address the criteria in § 20.2002 (a), (b), or (c) separately.

In response to your initial request, the SSDA, the technical basis document, and the NRC's detailed TER are considered proprietary and will not be available for public review. However, a second, publicly-available TER was also developed to demonstrate how this process will satisfy the NRC's mission of protecting public health, safety, and the environment. The NRC would note that specific parameter values, in the necessary form, that have not always been included with historical submittals may need to be included in future submittals in order for the SSDA methodology to be used.

In accordance with 10 CFR 2.390 of the NRC's "Agency Rules of Practice and Procedure, "a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

Copies of both TERs are enclosed. Please contact Mr. Maurice Heath if you have any questions concerning the above. He can be reached at (301) 415-3137 or via email at Maurice.Heath@nrc.gov.

Sincerely,

/RA/

Larry W. Camper, Director
Division of Decommissioning, Uranium Recovery,
and Waste Programs
Office of Nuclear Material Safety
and Safeguards

Enclosures:
Technical Evaluation Report (Proprietary Version)
Technical Evaluation Report (Public Version)

Enclosure 3

USEI Part B Permit EPA ID. No.: IDD073114654
Revision Date: July 28, 2016
Part C.3.2 WASTE ACCEPTANCE CRITERIA

C.3.2 Radioactive Material Waste Acceptance Criteria

The following waste acceptance criteria are established for accepting radiological contaminated waste material that is not regulated under the Atomic Energy Act of 1954 ("AEA"), as amended. This may be accomplished by the following regulatory mechanisms; use of a general or specific exemption from regulation by the Nuclear Regulatory Commission (NRC) or an Agreement State; a Release from Radiological Control declaration by the Department of Energy (DOE); or a determination that 91(b) radioactive material is no longer regulated by the Department of Defense (DoD). Material may also be accepted if it is not regulated or licensed by the NRC or Agreement State or has been authorized for disposal by the IDEQ and is within the numeric waste acceptance criteria. Waste acceptance criteria are consistent with these restrictions.

The following five tables establish types and concentrations of radioactive materials that may be accepted. These tables are based on categories and types of radioactive material not regulated by the NRC, an Agreement State, the DOE, or the DoD for alternate disposal. The criteria are consistent with these restrictions and detailed analyses set forth in *Waste Acceptance Criteria and Justification for FUSRAP Material*, prepared by Radiation Safety Associates, Inc. (RSA) as subsequently refined, expanded and updated in *Waste Acceptance Criteria and Justification for Radioactive Material*, prepared by USEI.

Material may be accepted if the material has been specifically exempted from regulation by rule, order, license, license condition, letter of interpretation, or specific authorization under the following conditions: Thirty (30) days prior to intended shipment of such materials to the facility, USEI shall notify IDEQ of its intent to accept such material and submit information describing the material's physical, radiological, and/or chemical properties, impact on the facility radioactive materials performance assessment, and the basis for determining that the material does not require disposal at a facility licensed under the AEA. The IDEQ will have 30 days from receipt of this notification to reject USEI's determination or require further information and review. No response by IDEQ within thirty (30) days following receipt of such notice shall constitute concurrence. IDEQ concurrence is not required for generally exempted material as set forth in Table C-4a.

Based on categories of waste described in the waste acceptance criteria, the concentration of the various radionuclides in the conveyance (e.g., rail car gondola, other container etc.) shall not exceed the concentration limits established in the WAC without the specific written approval of the IDEQ unless generally exempted as set forth in Table C-4a. Radiological surveys will be performed as outlined in Exempt Radiological Materials Procedure-01 (ERMP-01) to verify compliance with the WAC. If individual "pockets" of activity are detected indicating the limits may be exceeded, the RSO or RPS shall investigate the discrepancy and estimate the extent or volume of the material with the potentially elevated radiation levels. The RPS or RSO shall then make a determination on the compliance of the entire conveyance load with the appropriate WAC limits. If the conveyance is determined not to meet the limits, USEI will notify IDEQ's RCRA Program Manager within 24 hours of a concentration based exceedance of the facility WAC to evaluate and discuss management options. The findings and resolution actions shall then be documented and submitted to the IDEQ.

The radioactive material waste acceptance criteria, when used in conjunction with an effective radiation monitoring and protection program as defined in the USEI *Radioactive Material Health and Safety Plan* and *Exempt Radioactive Materials Procedures* provides adequate protection of human health and the environment. Included within this manual are requirements for USEI to submit a written summary report of all radioactive material waste receipts showing volumes and radionuclide concentrations and total activities disposed at the USEI site on a quarterly basis. The 4th quarter report of each year will also include an updated analysis of the cumulative impact on the facility performance assessment based upon the previous year's waste receipt.

These criteria and procedures are designed to assure that the highest potential dose to a worker handling radioactive material at USEI shall not exceed 400 mrem/year TEDE dose, and that no member of the public is calculated to receive a potential post closure dose exceeding 15 mrem/year TEDE dose, from the USEI program. TEDE is defined as the "Total Effective Dose Equivalent", which equals the sum of external and internal exposures. The public dose limit during operation activities is limited to 100 mrem/yr TEDE dose. An annual summary report of environmental monitoring results will be submitted to IDEQ by June 1st for the preceding year.

Materials that have a radioactive component that meets the criteria described in Tables C-1 through C-4c and are RCRA regulated material will be managed as described within this WAP for the RCRA regulated constituents.

Table C-1: Unimportant Quantities of Source Material Uniformly Dispersed* in Soil or Other Media**

	Status of Equilibrium	Maximum Concentration of Source Material	Sum of Concentrations Parent(s) and all progeny present
a	Natural uranium in equilibrium with progeny	<500 ppm / 167 pCi/g (^{238}U activity)	≤ 3000 pCi/g
	Refined natural uranium	<500 ppm / 167 pCi/g (^{238}U activity)	≤ 2000 pCi/g
	Depleted Uranium	<500 ppm / 169 pCi/g	≤ 2000 pCi/g
b	Natural thorium	<500 ppm / 55 pCi/g (^{232}Th activity)	≤ 2000 pCi/g
	^{230}Th (with no progeny)	0.1 ppm / <2000 pCi/g	
	Any mixture of Thorium and Uranium	Sum of ratios ≤ 1****	≤ 2000 pCi/g

*Refined Uranium includes ^{238}U, ^{235}U, ^{234}U, ^{234}Th, ^{234m}Pa, ^{231}Th

Table C-2: Naturally Occurring Radioactive Material Other Than Uranium and Thorium Uniformly Dispersed* in Soil or Other Media**

	Status of Equilibrium	Maximum Concentration of Parent Nuclide	Sum of Concentrations of Parent and All Progeny Present
a	^{226}Ra or ^{228}Ra with progeny in bulk form [1]	500 pCi/g	≤ 4500 pCi/g
b	^{226}Ra or ^{228}Ra with progeny in reinforced IP-1 containers [1]	1500 pCi/g	≤ 13,500 pCi/g
c	^{210}Pb with progeny(Bi & ^{210}Po)	1500 pCi/g	≤ 4500 pCi/g
	^{40}K	818 pCi/g	N/A
	Any other NORM		≤ 3000 pCi/g

[1] Any material containing ^{226}Ra greater than 222 pCi/g shall be disposed at least 6 meters from the external point on the completed cell.

Table C-3: Particle Accelerator Produced Radioactive Material

Acceptable Material	Activity or Concentration
Any particle accelerator produced radionuclide.	All materials shall be packaged in accordance with USDOT packaging requirements. Any packages containing iodine or volatile radionuclides will have lids or covers sealed to the container with gaskets. Contamination levels on the surface of the packages shall not exceed those allowed at point of receipt by USDOT rules. Gamma or x-ray radiation levels may not exceed 10 millirem per hour anywhere on the surface of the package. All packages received shall be directly disposed in the active cell. All containers shall be certified to be 90% full.

*Average over conveyance or container. The use of the phrase "over the conveyance or container" is meant to reflect the variability on the generator side. The concentration limit is the primary acceptance criteria.

**Unless otherwise authorized by IDEQ, other Media does not include radioactively contaminated liquid (except for incidental liquids in materials). See radioactive contaminated liquid definition (definition section of Part B permit).

*** $\frac{\text{Conc. of U in sample}}{\text{Allowable conc. of U}} + \frac{\text{Conc. of Th in Sample}}{\text{Allowable conc. of Th}} < 1$

Table C-4a: NRC Exempted Products, Devices or Items

Exemption 10 CFR Part*	Product, Device or Item	Isotope, Activity or Concentration
30.15	As listed in the regulation	Various isotopes and activities as set forth in 30.15
30.14, 30.18	Other materials, products or devices specifically exempted from regulation by rule, order, license, license condition, concurrence, or letter of interpretation	Radionuclides in concentrations consistent with the exemption
30.19	Self-luminous products containing tritium, ^{85}Kr, ^{3}H or ^{147}Pm	Activity by Manufacturing license
30.20	Gas and aerosol detectors for protection of life and property from fire	Isotope and activity by Manufacturing license
30.21	Capsules containing ^{14}C urea for *in vivo* diagnosis of humans	^{14}C, one µCi per capsule
31.12	General License for certain items and self-luminous products containing Radium 226	As set forth in 31.12 and see #4 under Additional information below
40.13(a)	Unimportant quantity of source material: see Table C-1	$\leq$0.05% by weight source material
40.13(b)	Unrefined and unprocessed ore containing source material	As set forth in rule
40.13(c)(1)	Source material in incandescent gas mantles, vacuum tubes, welding rods, electric lamps for illumination	Thorium and uranium, various amounts or concentrations,

		see rules
40.13(c)(2)	(i)Source material in glazed ceramic tableware (ii)Piezoelectric ceramic (iii) Glassware not including glass brick, pane glass, ceramic tile, or other glass or ceramic used in construction	≤20% by weight ≤2% by weight ≤10% by weight
40.13(c)(3)	Photographic film, negatives or prints	Uranium or Thorium
40.13(c)(4)	Finished product or part fabricated of or containing tungsten or magnesium-thorium alloys. Cannot treat or process chemically, metallurgically, or physically.	≤4% by weight thorium content.
40.13(c)(5)	Uranium contained in counterweights installed in aircraft, rockets, projectiles and missiles or stored or handled in connection with installation or removal of such counterweights.	Per stated conditions in rule.
40.13(c)(6)	Uranium used as shielding in shipping containers if conspicuously and legibly impressed with legend "CAUTION RADIOACTIVE SHIELDING – URANIUM" and uranium incased in at least 1/8 inch thick steel or fire resistant metal.	Depleted Uranium
40.13(c)(7)	Thorium contained in finished optical lenses	≤30% by weight thorium, per conditions in rule.
40.13(c)(8)	Thorium contained in any finished aircraft engine part containing nickel-thoria alloy.	≤4% by weight thorium, per conditions in rule.

Table C-4b: Materials Specifically Exempted by the NRC or NRC Agreement State

Exemption	Materials	Isotope, Activity or Concentration*
10 CFR 30.11**	Byproduct material including production particle accelerator material exempted from NRC or Agreement State regulation by rule, order, license, license condition or letter of interpretation may be accepted as determined by specific NRC or Agreement State exemption.***	Byproduct material at concentrations consistent with the exemption
10 CFR 40.14**	Source material exempted from NRC or Agreement State regulation by rule, order, license, license condition or letter of interpretation may be accepted as determined by specific NRC or Agreement State exemption.***	Source material at concentrations consistent with the exemption.

10 CFR 70.17	Special Nuclear Material (SNM) exempted from NRC regulation by rule, order, license, license condition or letter of interpretation may be accepted as determined by specific NRC or Agreement State exemption.***	SNM at concentrations consistent with the exemption.

*Sum of all isotopes up to a maximum concentration of 3,000 pCi/gm.
** Alternate disposals authorized by Agreement States also require an NRC exemption for the purposes of disposal in the State of Idaho.
*** Similar material not regulated or licensed by the NRC may also be accepted. Sum of all isotopes up to a maximum concentration of 3,000 pCi/gm. IDEQ shall be notified prior to the receipt of Special Nuclear Material not regulated or licensed by the NRC.

Table C-4c Material Released by Other Government Agencies

Exemption	Materials	Isotope, Activity or Concentration*
US DOE	Radioactive materials that have been released or cleared from radiological control	Radioactive materials at concentrations consistent with the Release**
US DoD	Radioactive materials determined not to be regulated under the AEA under authority granted to the DoD in Section 91(b) of the AEA of 1954, as amended	Radioactive materials at concentrations consistent with the Authorization**

*Sum or all isotopes up to a maximum of 3,000 pCi/gm.
**May include byproduct materials, source materials and special nuclear material as defined in the AEA of 1954 as amended. NORM and Particle Accelerator Produced Radioactive Material may also be accepted under Tables C.2 and C.3, as part of these Releases and Authorizations.

Additional Information for USEI's Waste Analysis Plan

1. US Ecology Idaho, Inc. (USEI) may receive contaminated materials or other materials as described in Tables C-1 - C-4b above. USEI may not accept for disposal any material that by its possession would require USEI to have a radioactive material license from the Nuclear Regulatory Commission (NRC).

2. Unless approved in advance by USEI and IDEQ, average activity concentrations may not exceed those concentrations enumerated in Tables C-1 and C-2. Additionally, for Tables C-1 and C-2, individual pockets of material may exceed the WAC for the radionuclides present as long as the average concentration of all radionuclides within the package or conveyance remains at or below the WAC and the highest dose rate measured on the outside of the unshielded package or conveyance does not exceed those action levels enumerated in ERMP-01.

3. Other items, devices or materials listed in Table C-4a, which are exempted in accordance with 10 CFR Parts 30, 40 or equivalent Agreement State regulations or 10 CFR Part 70 may be accepted at or below the activities (per device or item) or concentrations specified in those exemptions.

4. 10CFR20.2008 authorizes disposal of certain byproduct material as defined in Section 11.e(3) and 11.e(4) of the Atomic Energy Act, as amended, at disposal facilities authorized to dispose of such material in accordance with any Federal or State solid or hazardous waste law, as authorized under the Energy Policy Act of 2005.

5. The generator of particle accelerator produced waste must specify that the waste meets applicable acceptance criteria.

6. In accordance with permit requirements, notification of any exceedance of the WAC will be provided to the RCRA Program Manager within 24 hours, in accordance with the permit.

Enclosure 4

Sanitary Lagoon Sludge Sample Test Results

Enclosure 4

Sanitary Lagoon Sludge Sampling Results

Radionuclide Sum of Fractions

Sampling Event: Sanitary Lagoon Sludge Characterization   Total Sample Count: 28

	Analyte (pCi/g)				SOF	SOF
	U-234	U-235	U-238	Tc-99	Residiential	Industrial
Minimum Result:	12.6	0.7	3.0	0.0	1.4	0.0
Average Result:	**1,136.8**	**67.4**	**242.8**	**7.1**	**113.6**	**3.4**
Maximum Result:	2,870.0	189.0	703.0	50.2	295.8	9.6

#	Sample ID	Gross Analyte Activity (pCi/g)				SOF	SOF
		U-234	U-235	U-238	Tc-99	Residiential	Industrial
1	SLS-01	99.9	4.5	20.8	2.9	**9.9**	0.3
2	SLS-02	43.4	2.7	9.4	1.3	**4.4**	0.1
3	SLS-03	105.0	5.8	24.3	5.6	**10.8**	0.3
4	SLS-04	48.4	2.6	12.0	2.0	**5.0**	0.1
5	SLS-05	351.0	17.3	80.3	50.2	**37.5**	1.0
6	SLS-06	14.7	0.8	3.0	0.0	**1.5**	0.0
7	SLS-07	1,390.0	62.1	250.0	2.8	**132.7**	**3.4**
8	SLS-08	1,880.0	146.0	389.0	4.2	**190.9**	**6.5**
9	SLS-09	1,660.0	76.5	391.0	23.6	**166.4**	**4.6**
10	SLS-10	12.6	0.7	3.2	2.4	**1.4**	0.0
11	SLS-11	309.0	15.0	51.2	1.0	**29.4**	0.8
12	SLS-12	2,210.0	121.0	418.0	3.3	**215.2**	**6.1**
13	SLS-13	1,840.0	91.9	435.0	7.9	**184.5**	**5.3**
14	SLS-14	2,870.0	189.0	703.0	22.8	**295.8**	**9.6**
15	SLS-15	33.4	2.3	6.5	0.2	**3.3**	0.1
16	SLS-16	192.0	11.1	35.2	1.0	**18.7**	0.5
17	SLS-17	2,580.0	141.0	467.0	1.8	**249.5**	**7.0**
18	SLS-18	2,540.0	147.0	516.0	11.2	**251.2**	**7.4**
19	SLS-19	2,180.0	121.0	538.0	10.2	**221.8**	**6.8**
20	SLS-20	1,790.0	120.0	350.0	3.0	**177.9**	**5.6**
21	SLS-21	299.0	14.1	61.8	11.4	**29.8**	0.8
22	SLS-22	1,380.0	96.6	295.0	3.0	**139.5**	**4.5**
23	SLS-23	1,300.0	107.0	316.0	3.3	**136.1**	**4.9**
24	SLS-24	1,940.0	144.0	348.0	1.3	**192.2**	**6.2**
25	SLS-25	2,490.0	124.0	546.0	6.6	**246.4**	**7.0**
26	SLS-B1	42.5	1.7	8.4	1.2	**4.1**	0.1
27	SLS-B2	50.0	2.4	9.6	1.4	**4.9**	0.1
28	SLS-B3	2,180.0	119.0	511.0	12.9	**219.7**	**6.6**

% Moi stur	Moisture Corrected Activity (pCi/g)				SOF	SOF
	U-234	U-235	U-238	Tc-99	Residiential	Industrial
31	68.5	3.1	14.3	2.9	**6.8**	0.2
24	32.9	2.1	7.1	2.9	**3.4**	0.1
24	79.5	4.4	18.4	1.3	**8.0**	0.2
17	40.2	2.2	10.0	5.6	**4.4**	0.1
37	220.8	10.9	50.5	2.0	**22.1**	0.6
21	11.6	0.7	2.4	50.2	**3.8**	0.0
95	69.5	3.1	12.5	0.0	**6.6**	0.2
94	118.4	9.2	24.5	2.8	**12.2**	0.4
88	202.5	9.3	47.7	4.2	**20.4**	0.6
19	10.3	0.6	2.6	23.6	**2.3**	0.0
73	84.0	4.1	13.9	2.4	**8.1**	0.2
96	97.2	5.3	18.4	1.0	**9.5**	0.3
93	134.3	6.7	31.8	3.3	**13.6**	0.4
89	312.8	20.6	76.6	7.9	**32.5**	**1.1**
29	23.8	1.6	4.6	22.8	**3.6**	0.1
38	120.0	6.9	22.0	0.2	**11.7**	0.3
97	90.3	4.9	16.3	1.0	**8.8**	0.2
94	149.9	8.7	30.4	1.8	**14.9**	0.4
92	165.7	9.2	40.9	11.2	**17.4**	0.5
95	93.1	6.2	18.2	10.2	**9.8**	0.3
33	199.1	9.4	41.2	3.0	**19.6**	0.5
95	71.8	5.0	15.3	11.4	**7.8**	0.2
93	88.4	7.3	21.5	3.0	**9.4**	0.3
96	75.7	5.6	13.6	3.3	**7.7**	0.2
93	181.8	9.1	39.9	1.3	**18.0**	0.5
27	31.1	1.2	6.1	6.6	**3.3**	0.1
25	37.7	1.8	7.3	1.2	**3.7**	0.1
92	167.9	9.2	39.3	1.4	**16.9**	0.5


Sanitary Lagoon Proposed Sample Locations
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Legend
Proposed Systematic Sample Location
Sanitary Lagoon Grid
Sanitary Lagoon Boundary
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S
0 25 50 Feet
Source: Esri, Maxar, GeoEye, Earthstar Geographics, CNES/Airbus DS, USDA, USGS, AeroGRID, IGN, and the GIS User Community