

Enclosure 2

US Ecology, Inc. Request for Exemptions from Requirements in
10 CFR 30.3, 10 CFR 40.3, and 10 CFR 70.3
for US Ecology Idaho, Grand View, ID

Provided in both Unredacted and Redacted Versions

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US Ecology, Inc. Request for Exemptions from Requirements in 10 CFR 30.3, 10 CFR 40.3, and 10 CFR 70.3 for US Ecology-Idaho, Grand View, ID

Based on:

Studsvik Processing Facility Memphis, LLC's Tennessee Agreement State Approved Alternate Disposal Authorization and Amendment No. 82 to License # R-79273-H16

Revision: 0

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US Ecology, Inc.
300 E. Mallard Dr., Suite 300
Boise, ID 83706
Phone: 208-331-8400 Fax: 208-331-7900
www.usecology.com

Studsvik Processing Facility Memphis, LLC
2550 Channel Ave PO Box 13143
Memphis, TN 38113
Phone: 901-775-0690 Fax: 901-775-0629
www.studsvik.com

1.0 INTRODUCTION

US Ecology, Inc. seeks approval to receive and dispose of low-activity radioactive wastes from Studsvik's Processing Facility in Memphis, TN (SPFM) at US Ecology Idaho (USEI), the company's RCRA subtitle-C hazardous and low-activity radioactive waste facility near Grand View, ID. USEI is regulated by the Idaho Department of Environmental Quality (IDEQ). Idaho is not a US Nuclear Regulatory Commission (NRC) Agreement State; however, Idaho regulations and the Grand View Resource Conservation and Recovery Act (RCRA) permit provide for the acceptance of this material if the appropriate NRC exemptions are received. The requested exemptions would allow SPFM to ship low concentrations of radioactive material to USEI resulting from a direct assay or sorting and segregation operations that occur within the Memphis, TN facility, which has had its operations previously approved by the Tennessee Department of Environment and Conservation (TDEC), Division of Radiological Health. The addition of the USEI facility as a disposal option for SPFM wastes would provide necessary flexibility for the facility's overall waste management and disposal program. USEI has a long and established record of safe and secure disposal of low-activity radioactive wastes at its Grand View, ID facility serving both commercial and government sectors.

A Safety Assessment for the proposed alternate disposal is attached, and it shows that the potential dose to a member of the public is consistent with the NRC's "less than a few millirem per year" criterion. Based on this assessment, SPFM received an amendment to their Tennessee Agreement State Radioactive Materials License #R-79273-H16 from the TDEC on September 12, 2012 pursuant to Section 0400-20-05-.121 of the Rules of the TDEC Division of Radiological Health – Method for Granting Approval of Alternate Disposal Procedures. This section of the regulation adopts the alternate disposal provisions in 10 CFR 20.2002. TDEC reviewed and approved Studsvik's amendment request package and issued Amendment 82 with new License Condition No. 44. License Condition No. 44 grants authorization for alternate disposal of SPFM waste at USEI provided that USEI receives exemptions from US NRC licensing requirements.

US Ecology is hereby requesting NRC review and approval of the said submittal for purposes of granting exemptions from the licensing requirements in 10CFR 30.3, 10CFR 40.3, and 10CFR 70.3 for purposes of disposing these wastes at our USEI facility. This framework is consistent with the scenario #3 for alternate disposals between Agreement States and Non-Agreement States outlined in NRC Agreement State Letter FSME-12-025 (March 13, 2012).

2.0 REGULATORY BACKGROUND

NRC Regulation §20.2002 describes a method for obtaining approval of radioactive waste disposal methods that are not otherwise authorized by regulation. Issuance of exemptions to USEI from all applicable Byproduct, Source, and SNM licensing regulations in §30.11, §40.13, and §70.17 would provide SPFM with another waste disposal outlet to utilize as part of its overall waste sorting and segregation processes. USEI is required to receive specific exemptions directly from the US NRC since Idaho is not an NRC Agreement State. Therefore, this Request for Exemptions submittal for USEI was prepared in accordance with 10 CFR 20.2002 and NRC's implementing guidance found in "Results of the License Termination Rule Analysis," dated May 28, 2004 (NRC, 2004) and reaffirmed in SECY-07-0060 (NRC, 2007) to ensure that doses to the public are maintained at "less than a few millirem" per year.

USEI is permitted as a RCRA Subtitle C hazardous waste landfill regulated by the IDEQ. USEI is authorized to dispose of low-activity radioactive materials exempt from regulation by the Atomic Energy Act of 1954, as amended, through regulatory authority provided by Idaho law and regulation. Radioactive material disposal limits, radiological performance assessment and source term reporting, environmental monitoring, limitations for potential exposure to radioactive material, and closure and post-closure requirements are published in USEI's RCRA Subtitle C permit and implemented through regulation by IDEQ. USEI has been granted several site-specific §30.11 Byproduct material and §70.17 SNM exemptions from the NRC for the purpose of disposing of various licensee waste streams. The operational

performance characteristics of the USEI site have been thoroughly reviewed by the NRC and determined to be protective within the NRC's published regulations and policies. A detailed description of the USEI facility is provided in Attachment 1.

USEI is permitted to receive waste shipments for treatment and disposal via truck or rail conveyance. There are no prohibitions against receipt of US Department of Transportation (DOT) Class 7 Radioactive Material shipments into USEI provided they meet USEI's waste acceptance criteria (WAC) and are manifested, marked, and labeled in accordance with all DOT regulations.

3.0 TECHNICAL DESCRIPTION

3.1 Purpose of the USEI Disposal Option

The purpose of the USEI alternate disposal option is to provide a safe and cost-effective disposal alternative for SPFM for waste contaminated with low-activity radioactive material.

There is currently only one remaining site in the nation offering access to a majority of Class A low level radioactive waste (LLRW) generators, including those in the State of Tennessee. The capacity of the remaining site is limited and would be better used for the disposal of waste that carries more significant risk. The USEI facility would provide SPFM with the option of disposing low-activity radioactive wastes in a safe and secure RCRA Subtitle C hazardous waste landfill that is designed to receive such waste.

3.2 Types of Waste for Disposal

The sorting and segregation processes utilized at SPFM are designed to apply to the most common types of waste produced by educational, industrial, commercial, research, and nuclear power operations. The primary waste forms are dry-active waste (DAW), resins, sludges, filter media, building rubble, piping, soil, metal scrap and other waste that meets the definition of a Solid Waste as defined by Tennessee Division of Solid Waste Management regulations. Liquid wastes that would be solidified or absorbed prior to placement in the disposal cell are also authorized for processing. DAW consists of paper, plastic, cloth, trash and small objects made of metal similar to ordinary household or commercial waste. Wastes accepted at SPFM are contaminated with small amounts of radioactive material. License amendment 82 to SPFM's License #R-79273-H16 would allow all of these waste types and forms to be disposed at USEI provided the required exemptions are granted and all waste meets USEI's WAC.

The debris wastes will exhibit an average bulk density of 0.44 grams per cubic centimeter, g/cc (27.5 pounds per ft³), based on SPFM's experience with similar waste streams. Radiological characterization of the waste stream is performed through SPFM's customers' 10 CFR 61 characterization programs as well as non-destructive assay techniques previously approved as part of other license amendments (details provided in Section 3.7). A summary of the radionuclides and concentrations evaluated as part of this request are discussed in Section 3.5.

This request does not include treatment or disposal of any hazardous wastes as identified in the RCRA regulations.

3.3 Environmental Concerns

Alternate disposal of SPFM wastes at USEI will result in minimal impact to members of the public from radiation exposure. This includes residents in the surrounding community, members of the public in close proximity to the waste during transportation activities, and workers at the disposal site. Impact is evaluated by determining the radioactive dose to each of these critical persons and determining the Maximally Exposed Individual (MEI), or the person who could be subjected to the highest potential dose from the proposed transportation disposal activities. Evaluation of SPFM wastes considered to be eligible for alternate disposal at USEI is based on a maximum annual MEI dose of less than 5 mrem/yr, which is used as an upper threshold for the "less than a few millirem" criterion discussed earlier. A comparison of this dose

threshold versus prevalent natural backgrounds and other acceptable dose levels for US NRC licensees is provided in Table 1.

Table 1. Dose Comparison of Natural Background and Regulatory Limits Versus Proposed Dose Limit for SPFM Alternate Disposals at USEI

Dose Level or Regulatory Limit	Dose (mrem/yr)
Average Background Dose Due to Natural Sources – Worldwide	350
Average Background Dose Due to Natural Sources – State of Tennessee	115
Dose Limit to a Member of the Public from Licensee Operations (10CFR 20.1301)	100
License Termination Rule Dose Limit (10CFR 20.1402)	25
MEI Dose Limit for Alternate Disposal of SPFM Wastes at USEI	5

3.4 Potentially Affected Facilities

The USEI disposal facility is a Subtitle C RCRA hazardous waste treatment and disposal facility permitted by the IDEQ. The USEI site is located in the Owyhee Desert of southwestern Idaho in the town of Grand View, approximately 113 kilometers (70 miles) southeast of Boise. 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 kilometers (1.4 people per square mile).

This region has an arid climate. The USEI site is located on a 1.6 kilometers (1 mile) wide plateau. Maximum surface relief on the facility is 27 m (90 feet) and the mean surface elevation is 790 m (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. An aerial photograph of the USEI site taken in 2010 is shown in Figure 1.



Figure 1. Aerial View of the USEI site, looking South (circa 2010).

USEI currently has a permitted landfill capacity of over 11 million cubic yards (M yd³), which represents approximately 23 years of disposal capacity at current levels. USEI began construction on its newest disposal cell (Cell 16) in 2012, which will expand its available constructed hazardous and low-activity radioactive waste airspace to approximately 2M yd³ by the end of calendar year 2013. The remaining permitted capacity of Cell 16 is expected to be built out over the course of several years as need arises. An excerpt from USEI's RCRA part B permit with Radiological Waste Acceptance Criteria is provided in Attachment 2.

3.5 Radionuclides of Concern

This safety assessment considers 59 radionuclides of concern as possibly being present in an average year of waste processing at SPFM. These nuclides were chosen from actual waste receipts over the last two full years of operating history (2010-2011), with adjustments made to screen out nuclides with short half-lives or those found in a gaseous physical form. The list of nuclides evaluated as part of this assessment is presented in Table 2.

Table 2. List of Radionuclides and Average Concentrations Evaluated¹.

Nuclide	pCi/g	Nuclide	pCi/g	Nuclide	pCi/g
Ag-108m	5	Eu-152	7	Ra-226 ³	10
Ag-110m	25	Eu-154	2.5	Ru-103	1.5
Am-241	0.5	Eu-155	4	Ru-106	10
Au-195	3.5	Fe-55	1000	Sb-124	8
Ba-133	0.3	Fe-59	8	Sb-125	100
Be-7	20	H-3	325	Sn-113	2
C-14	25	I-125	0.1	Sr-89	17
Ce-139	1	I-129	0.3	Sr-90	14
Ce-141	11.5	I-131	18	Tc-99	4
Ce-144	180	Mn-54	80	Te-123	4
Cm-242	0.1	Na-22	0.2	Th-228	1
Cm-243	1.3	Nb-94	4	Th-232 ³	4
Cm-244	0.5	Nb-95	25	U-233 ²	8
Cm-245	5	Ni-59	100	U-234 ²	190
Co-57	12	Ni-63	925	U-235 ²	10
Co-58	200	Pu-238	5	U-238 ²	190
Co-60	650	Pu-239	0.3	Natural Uranium ^{3,4}	225
Cr-51	55	Pu-240	0.3	Zn-65	115
Cs-134	175	Pu-241	15	Zr-95	30
Cs-137	500	Pu-242	0.2		

1. The USEI WAC limit is 3,000 pCi/g total for all nuclides per shipment. The sum for all nuclides listed in Table 2 exceeds this value since it represents an average annual inventory. All individual shipments from SPFM to USEI must meet the <3000 pCi/g criterion.
2. The individual nuclides of Uranium are modeled in the concentrations provided by each generator.
3. These nuclides are assumed to be in equilibrium with all of their progeny nuclides: Ra-226 (8), Th-232 (10), U-235 (11), and U-238, 14)
4. Natural uranium total activity modeled as 48.6% U-238, 2.8% U-235, and 48.6% U-238.

3.6 Mixtures of Radionuclides

Since mixtures of radionuclides are expected within individual shipments, a method must be identified to ensure the sum of these concentrations (or calculated dose) from the mixture remains within set limits. A standard method to evaluate the effect of a mixture against a fixed limit is the sum-of-the-ratios (SOR) calculation. The contribution of each isotope to the total is evaluated by the formula:

$$SOR = \sum_{x=1}^{x=n} \frac{X_1}{ACL_1} + \dots + \frac{X_n}{ACL_n} < 1$$

where:

- SOR = the sum of the ratios for the mixture.
- x_1 = the measured or calculated waste activity concentration for isotope 1.
- x_n = the measured or calculated waste activity concentration for isotope n.
- ACL_1 = the activity concentration limit for isotope 1.
- ACL_n = the activity concentration limit for isotope n.

The result of this calculation represents the sum of the fractional contributions of each nuclide to the mixture so that the final disposal activity concentrations will not exceed the allotted limit. SOR will be used to evaluate SPFM waste shipments against USEI's WAC limits and other concentration (or dose) limits established as part of this program.

3.7 Waste Evaluation Methodology

Evaluations of the radioactive content of SPFM customer waste materials are performed using methods appropriate for the type and quantity of radioactive material sent for evaluation. SPFM will use both direct and indirect assay methods to identify the constituent isotopes radioactive material in batches of sorted and segregated waste material.

A necessary prerequisite to these evaluations is that the customer provides a detailed isotopic evaluation of the waste material that includes an analysis for difficult-to-detect nuclides. The analyses typically used to determine activity in support of waste classification per 10 CFR Part 61.55 and the NRC's Branch Technical Position (BTP) on Waste Classification provide adequate isotopic information. (NRC, 1983) Generator knowledge may also be adequate in the case of non-production facilities where the source of radioactive material is specifically known. The generator must document the basis for the isotope. Generators will be informed that their waste is a candidate for SPFM's programs and their waste streams' 10 CFR 61 analysis or equivalent results will be maintained annually and reported to SPFM. The information in this profile is used to pre-screen the waste materials for suitability for evaluation under the general Bulk Survey for Release (BSFR) process already in use at SPFM. This pre-screening process will also be utilized to determine suitability of waste streams for acceptance at USEI.

Shipments of incoming waste materials to SPFM must be accompanied by appropriate documentation for each shipment identifying the radioactive material in the shipment. For shipments of material classified as Low Level Radioactive Waste by the generator, the Uniform Waste Manifest described in 10 CFR Part 20, Appendix G provides adequate documentation. For shipments that meet the definition of radioactive material in 49 CFR 173.403, the shipping papers required by 49 CFR Part 172.200 will supply adequate documentation. For shipments that do not meet either of the above criteria, documentation containing equivalent information must be provided. As part of the waste evaluation process, SPFM will compare the radionuclide distribution identified on the shipment documentation to the generator's profile data to validate the consistency of the isotopic data.

Prior to the evaluation for radioactive material content, the physical form of the waste must be inspected to ensure it falls within the parameters of the radionuclide analysis method and of the disposal site model. The Microshield and RESRAD models used to evaluate the dose impact to the disposal site and its workers as well as the In-Situ Object Counting System (ISOCS) software used to evaluate waste containers assume that the waste material is essentially a homogeneous mixture with the radioactive material uniformly distributed throughout the waste.

The NRC's current BTP on Waste Form and Concentration Averaging provides some guidance on uniformity for waste classification purposes. (NRC, 1995) The BTP suggests that sample results that are within a factor of 10 of each other can be considered representative and therefore be used to calculate the average activity of the waste. Again, this BTP assumes the waste is of a homogeneous nature (resins, sludge's, etc.) and includes typical DAW in that description.

To verify homogeneity, containers are surveyed and an average contact radiation reading is established. All container contact dose rates should then be within a factor of 3 to be considered homogeneous.

Radioassay of incoming SPFM waste materials consists of one or a combination of three methodologies:

- Waste that does not contain any gamma emitting radionuclides must be directly sampled and the samples must be analyzed with equipment capable of detecting the suspected radionuclides. Waste materials in this category require an individual sampling and analysis plan to ensure the nuclides are properly quantified. The sampling plan must ensure representative samples are taken. Samples may be composited if appropriate and are typically sent to an outside laboratory approved by the SPFM Quality Assurance Program.
- Waste that contains only gamma-emitting radionuclides is evaluated using gamma spectroscopy instrumentation. SPFM uses the Canberra ISOCS and Genie 2k software to identify and quantify gamma activity. The ISOCS system consists of a high-purity germanium detector with a multi-channel analyzer and spectroscopy software to obtain positive identification of radionuclides. The ISOCS software is also capable of determining the quantity of radioactive material in the container.
- Waste that contains a mixture of gamma- and non-gamma-emitting nuclides may be evaluated with the ISOCS and the use of scaling factors. The profile provided by the customer identifies the radionuclides and their relative abundance in the waste. Once the waste is packaged, there is no process other than radioactive decay that can change this relationship. The relationship between the easily detectable nuclides (typically, the energetic gamma emitters) and the difficult-to-detect nuclides can be used to determine the isotopic activity of the entire batch of waste.

Gamma spectroscopy is used to detect and quantify the gamma emitting nuclides. A single, easily detectable nuclide is chosen as the key nuclide against which all of the hard-to-detect isotopes will be scaled. The key nuclide is typically Co-60 or Cs-137 which emit high energy gamma radiation that easily penetrates the waste and container. The quantity of the scaled nuclide is determined by multiplying the activity of the key nuclide that has been detected by the ratio of the key nuclide to the nuclide of interest as established by the profile. The use of scaling factors is described as an acceptable method of activity determination in the Waste Classification BTP. This is the primary method of evaluation used by SPFM and is currently implemented in accordance with its license. SPFM would employ these same procedures for waste destined for USEI.

3.8 Use of the Canberra ISOCS for Quantitative Analysis

The ISOCS system is designed to identify gamma-emitting radioisotopes and calculate the activity within a container or object. Background radiation profiles and verification of energy calibrations are performed daily to ensure proper detector operation. Each container is counted for sufficient time to ensure that the lower limit of detection (LLD) for each reported isotope is no greater than 10% of the release limit. Low energy gamma nuclides for which the LLD cannot be achieved in a reasonable amount of time must be scaled to the key nuclide as described above. Efficiency calibrations are

calculated by the software using the container's physical dimensions, waste weight, type of waste, and measurements of the detector in relation to the container. The ISOCS software is capable of performing quantitative analyses over a wide range of container shapes, volumes, and material densities. The software is set to calculate the activity concentration of each identified isotope in pCi/g and also has the capability to calculate total activity (pCi), activity per area (pCi/cm²), and activity per length (pCi/cm). Activity due to background radiation is subtracted from the container evaluation prior to reporting the final specific activity. The final derived activity concentration is an average of all of the activity divided by all of the mass.

The ISOCS software has the capability to calculate specific activity of many different shapes and sizes. Examples are but not limited to:

- Simple Box
- Complex Box
- Rectangular Plane
- Circular Plane
- Simple Cylinder
- Complex Cylinder
- Pipe
- Tank
- Cone
- Etc.

The LABSOCS software has the capability to perform many laboratory analyses with varying geometric configurations such as:

- General Purpose Beaker
- General Purpose Marinelli Beaker
- Simplified Box
- Simplified Sphere
- Etc.

The ISOCS Calibration Software User's Manual provides a more thorough description of the ISOCS equipment and software. The system will be operated in accordance with a standard operating procedure which implements the manufacturer's recommendations and operating requirements. SPFM's operating procedure for the ISOCS system was revised on November 22, 2011 to meet the new licensing requirements of Studsvik's Radioactive Material License # R-79273-H16 Amendment 77.

All personnel operating the ISOCS system must be a properly trained and qualified Radiation Protection Specialists. Specific training and qualification requirements are identified in SPFM training procedures and include instruction on operation of the ISOCS equipment and basic gamma-spectroscopy. Training must be conducted directly by a Canberra Representative or by a Canberra-trained SPFM operator.

3.9 Waste Evaluation Process Summary

The following steps summarize the requirements of SPFM processes:

- 1 Verify Customer Part 61 data and valid certification (or equivalent) is on file for the shipment and waste stream to be evaluated.
- 2 Evaluate the waste container for radiological homogeneity. Survey the container and verify that all contact dose rate readings are within a factor of 3 of the average contact reading.
- 3 Inspect the container contents for physical homogeneity. Object sizes should be relatively small and uniform. Densities should be relatively uniform and within a factor of 10.

Evaluate container for activity concentration using the ISOCS in accordance with the operating procedures. Analyze ISOCS data and perform any necessary scaling for hard-to-detect nuclides.

- 1 Evaluate measured activity against release limits and calculate dose fractions and adjusted weights.
- 2 Any container that exceeds the established release limits may not be released. The waste may be sorted to identify and remove non-complying waste and the remainder re-evaluated.
- 3 Shred, destroy, disfigure, or otherwise render unrecognizable any radioactive material labels prior to final packaging or shipment for disposal.
- 4 Absorb or solidify any free-standing liquids.
- 5 Verify the total specific activity of each container does not exceed any of the limits in USEI's WAC (see Tables C.1 through C.4b of Attachment 2). All material that exceeds any of USEI's WAC limits or any other limit defined in the NRC exemptions for USEI must be disposed as Class A LLRW.

SPFM has developed standard operating procedures to implement and control these processes as part of their previously approved BSFR program. The most current versions of all procedures are referenced in Studsvik's Radioactive Material License #R-79273-H16 Amendment 77.

“Confidential Information Submitted Under 10 CFR 2.390; Withhold from Public Disclosure under 10 CFR 2.390.”

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Revision 0, June 2013*

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5.0 ANNUALIZED BASELINE DOSE ASSESSMENT

5.1 Dose Assessment Methodologies

The dose equivalent for a MEI is evaluated to ensure that it is consistent with the NRC standard of a "less than a few millirem per year" to a member of the public. Typically, all transportation and disposal workers who come in contact with radioactive waste shipments performed under an exemption are treated as 'members of the public' as defined by the NRC because the USEI site, while permitted under Idaho law to accept certain radioactive materials, is not licensed by the NRC or an Agreement State. This methodology will continue to be used for all USEI personnel or contractors involved with waste transportation or handling tasks associated with this ADR. However, workers employed by Studsvik's SPFM facility are not evaluated for comparison against the "less than a few millirem" criteria since SPFM is licensed by the State of Tennessee (License Number R-79273-H16) and all workers and truck drivers are trained, occupational radiation workers monitored under an approved and inspected Agreement State radiation protection program.

This dose assessment also includes long-term, post-closure evaluations to a potential future resident using RESRAD and a postulated inadvertent intruder at the USEI facility.

5.1.1 Transportation and USEI Site Workers

Doses to USEI workers and drivers are calculated by modeling various activities to estimate internal and exposure dose rates from acute exposure. External dose estimates are performed using the Microshield® software code, (Ver. 7.02) with standard geometries for the pathways required. All doses calculated in Microshield include buildup, with the media exhibiting the highest number of photon mean-free-paths chosen for estimation of the buildup factor. Two Microshield models were run for each job function due to the potential for different types of uranium to be present in SPFM wastes. To account for this potential, the individual uranium isotopes shown in Table 2 were modeled individually as part of the larger radionuclide list. Conversely, natural uranium (in secular equilibrium) was modeled separately in Microshield with a total activity of 225 pCi/g split between the U-238, U-235, and U-234 constituents using activity fractions of 0.486, 0.028, and 0.486, respectively. In addition, the natural uranium is assumed to be in complete secular equilibrium with all progeny in both the U-238 and U-235 decay chains. Details of the dose pathways are discussed in Sections 5.2 and 5.3. Results of the Microshield external dose evaluation for each transportation or USEI worker model are provided in Attachment 4. Calculations of internal doses are provided in Attachment 5.

Internal dose calculations from handling activities assume a portion of the radioactivity in the waste materials is re-suspended and available for inhalation. Dose conversion factors (DCF) from Federal Guidance Report 11 (FGR 11, EPA 1986) are applied to calculate dose rates for various handling and disposal activities. All USEI employees who work with any hazardous materials are required to participate in an Occupational Safety and Health Administration (OSHA) compliant respiratory protection program. Although respiratory protection is required for the above specified workers, no credit for protection factors is taken for USEI workers in these dose assessments.

5.1.2 Disposal Site Modeling

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 LLRW 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. The USEI RESRAD model has all dose pathways turned on except for ingestion of aquatic foods, since there is no credible source of aquatic foods for consumption at the USEI site. A number of default parameters in the Grand View model have been replaced with site specific parameters developed to support previous

USEI permit modifications (See Attachment 1). A summary of USEI's RESRAD Input Parameters is provided in Attachment 6.

Doses to a potential inadvertent intruder are performed using the methodology provided in NUREG-0782 along with updated dose conversion factors published in NUREG/CR-4370.

5.2 Transportation Dose Assessment

All waste materials will be packaged in either 20 yd³ and 30 yd³ intermodal containers (IMC) and transported from SPFM to the USEI facility in Grand View, ID using either truck or rail transport. A total of 561 shipments are expected for an average year's total volume from SPFM of approximately 489,000 ft³.

For direct truck transport from SPFM to USEI, IMCs will be placed onto a flatbed or chassis trailer and driven as exclusive use shipments. All truck drivers will be trained radiation workers administered under SPFM's agreement—state approved radiation protection program, and actively monitored for external exposure using thermo-luminescent dosimeters (TLD). Therefore, no external dose calculations are provided for this task. It is anticipated that up to six drivers will be assigned to this task.

Rail transport will entail a combination of short-haul truck transport (called front-end dray [FED]) of approximately five miles on chassis trailers to a local rail spur where the IMCs will be trans-loaded onto six-position articulated flatcars. Delivery time from the SPFM to the rail trans-load facility is estimated to take five minutes. It is anticipated that up to six drivers will be assigned to this task with each driver making approximately 94 trips in a given year. For modeling purposes in Microshield, the FED driver sits approximately 4 meters from the surface of an IMC with approximately 0.2 inches of aluminum shielding between the driver and the waste. Only external dose is evaluated for the truck drivers since all waste containers are sealed during transportation activities. For the list of nuclides and concentrations in Table 2, each FED driver is estimated to receive 0.75 mrem/yr or ~15% of the 5 mrem/yr limit.

Once the loaded railcars reach USEI's rail transfer facility (RTF) in Mayfield, ID, the IMCs would be lifted off the flatcars onto a waiting truck with a chassis trailer to perform the back-end dray (BED) to the USEI disposal facility in Grand View, ID (see Figure 2). Each BED trip from Mayfield to Grand View is conservatively estimated to take approximately 45 minutes to complete at an average speed of 55 miles per hour. A minimum of 10 BED drivers are assigned to this task, with each driver expected to make 56 trips from the RTF to the USEI disposal facility in a given year. For modeling purposes in Microshield, the same model as the FED drivers was used. Only external dose is evaluated for the truck drivers since all waste containers are sealed during transportation activities. 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. For the list of nuclides and concentrations in Table 2, each BED driver is estimated to receive 3.73 mrem/yr or ~75% of the 5 mrem/yr limit. Details of the transportation dose evaluation are provided in Table 3.



Figure 2. Photo of Taylor industrial forklift placing a loaded IMC onto a BED truck.

5.3 USEI Worker Dose Assessment

5.3.1 Shipments Received via Truck

Regardless of how the IMCs are shipped to USEI, many of the receipt, handling, and disposal tasks required remain the same. For direct shipments by truck, IMCs will be delivered to USEI's Grand View, ID disposal facility where they will be received, inspected, and surveyed prior to being emptied into the disposal cell. Upon receipt at USEI, the IMCs will be surveyed while still on the trailer. Five minutes is required to perform a survey of each IMC. Based on current practice, a surveyor is assumed to stand at a distance of 1 meter from the container during the survey, with six surveyors sharing the task. A 0.2 inch thick aluminum shield is used to simulate the wall of the IMC between the surveyor and the waste in the Microshield model. Each of the eight surveyors will perform approximately 70 surveys per year on SPFM shipments. For the list of nuclides and concentrations in Table 2, external dose (i.e., total dose) to each surveyor is estimated to be 4.33 mrem/yr or ~87% of the 5 mrem/yr limit. The 'Truck/IMC Surveyor' task has been identified as the MEI for this ADR.

Once the receiving and survey activities are completed, the truck driver transports it to the disposal cell for burial. Each IMC is then tipped directly into the disposal cell and compacted by the Landfill Cell Operator. These bulldozer operators, wearing full-faced respirators within an enclosed cab, spread and compact the waste into standard 'lifts' within the landfill. The average time to spread and compact 50 tons of material (which is the capacity of 2.5 IMCs) is 15 minutes. A modest internal dose is assumed for these operators since no credit is taken for published protection factors of the full-face respirators. For the list of nuclides and concentrations in Table 2, the total dose (internal + external) to each Landfill Cell Operator is estimated to be 1.18 mrem/yr or ~24% of the 5 mrem/yr limit, with 0.977 mrem from the external dose pathway and 0.202 mrem from the internal dose pathway.

A summary of expected USEI worker doses from transportation of IMCs from SPFM via truck is provided in Table 3.

5.3.2 Shipments Received via Rail

If SPFM chooses to ship IMCs via rail, shipments will come into USEI's rail transfer facility (RTF) located in nearby Mayfield, ID. Surveys of IMCs on trains will be performed using the same techniques and modeling information described for truck shipments above. The estimated dose to the IMC Surveyor (via rail) is the same as for the truck shipment scenario, 4.33 mrem/yr or ~87% of the 5 mrem/yr limit.

After completion of the incoming surveys, each IMC will be off-loaded from the flatbed railcars to an awaiting truck hauling a chassis trailer. The offload will be accomplished using a Taylor Industrial Forklift equipped by a spreader bar attachment that 'picks' each IMC off the train from the top and places it directly onto the chassis trailer. A photograph of the Taylor loading an IMC on a waiting truck is provided in Figure 2.

The USEI workers performing this task are referred to as the 'Rail Transfer Equipment Operators.' The operator sits in the elevated cab at a distance of 16 feet (4.9 meters) from the nearest wall of an IMC. The remainder of the standard IMC Microshield model is kept constant from the surveyor model, with 0.2 inches of aluminum between the operator and the waste material. No credit is taken in the model for any additional shielding that may be provided by the mast or other structures of the forklift. For the list of nuclides and concentrations in Table 2, external dose (i.e., total dose) to each RTF Equipment Operator is estimated to be 3.33 mrem/yr or ~66% of the 5 mrem/yr limit.

The BED truck drivers will then transport the IMCs to USEI's Grand View disposal facility where the loads will be offloaded and compacted in the disposal cell as described previously for the truck shipments. The total number of IMCs received at USEI will be the same regardless of whether they are transported by truck or rail, so the dose estimate for the Landfill Cell Operator will be the same for both

scenarios; 1.18 mrem/yr or ~24% of the 5 mrem/yr limit, with 0.977 mrem from the external dose pathway and 0.202 mrem from the internal dose pathway.

A summary of expected worker doses from transportation of IMCs from SPFM via truck is provided in Table 4.

Table 3. Summary of Annualized Dose Assessment for IMC Transportation by Truck

Job Function	No. Workers	Waste Contact Time (hr)	External Exposure Rate ¹ (mR/hr)	Internal Dose Rate (mrem/hr)	Dist. (m)	No. Required Trips or Reps	Total External Dose (mrem)	Total Internal Dose ² (mrem)	Total Project TEDE (mrem)
FED Truck Drivers	6	0.09	8.9E-02	0.0E+00	4.0	0	0.0E+00	0.0E+00	0.0E+00
Long-Haul Truck Drivers ³	6	32.73	8.9E-02	0.0E+00	4.0	561	0.0E+00	0.0E+00	0.0E+00
RTF Equipment Operator	4	0.25	9.5E-02	0.0E+00	4.9	0	0.0E+00	0.0E+00	0.0E+00
Truck/IMC Surveyors	8	0.08	7.7E-01	0.0E+00	1.0	561	4.3E+00	0.0E+00	4.3E+00
BED Truck Drivers	10	0.75	8.9E-02	0.0E+00	4.0	0	0.0E+00	0.0E+00	0.0E+00
Landfill Cell Operators	4	0.25	1.2E-01	2.4E-02	2.0	135	9.8E-01	2.0E-01	1.2E+00

1. The external exposure rates in Table 3 are the sum of two Microshield runs as described in Section 5.1.1.
2. Calculation of the Internal Dose Rate using the FGR11 DCFs is provided in Attachment 5.
3. TEDE is not calculated for Long-Haul Truck Drivers since they will be trained and monitored radiation workers.

Table 4. Summary of Annualized Dose Assessment for IMC Transportation by Rail

Job Function	No. Workers	Waste Contact Time (hr)	External Exposure Rate ¹ (mR/hr)	Internal Dose Rate (mrem/hr)	Dist. (m)	No. Required Trips or Reps	Total External Dose (mrem)	Total Internal Dose ² (mrem)	Total Project TEDE (mrem)
FED Truck Drivers	6	0.09	8.9E-02	0.0E+00	4.0	561	7.5E-01	0.0E+00	7.5E-01
Long-Haul Truck Drivers	6	32.73	8.9E-02	0.0E+00	4.0	0	0.0E+00	0.0E+00	0.0E+00
RTF Equipment Operator	4	0.25	9.5E-02	0.0E+00	4.9	561	3.3E+00	0.0E+00	3.3E+00
Truck/IMC Surveyors	8	0.08	7.7E-01	0.0E+00	1.0	561	4.3E+00	0.0E+00	4.3E+00
BED Truck Drivers	10	0.75	8.9E-02	0.0E+00	4.0	561	3.7E+00	0.0E+00	3.7E+00
Landfill Cell Operators	4	0.25	1.2E-01	2.4E-02	2.0	135	9.8E-01	2.0E-01	1.2E+00

1. The external exposure rates in Table 3 are the sum of two Microshield runs as described in Section 5.1.1.
2. Calculation of the Internal Dose Rate using the FGR11 DCFs is provided in Attachment 5.

The dose assessments for the truck and rail shipments scenarios have been done independently, under the assumption that all shipments would be performed using either one or the other. In reality, shipments will likely be made using a combination of both throughout a given year, at SPFM's discretion. The breakdown of truck and/or rail shipments has no impact on the MEI dose (Truck/IMC Surveyor at USEI).

5.4 Post-Closure Dose to the General Public

Two RESRAD models were run to assess the impact of the SPFM waste on the USEI site. The first model is consistent with USEI's post-closure dose model included in the Part B RCRA permit. This model assumes that shipments from SPFM are received and distributed evenly within the entire USEI landfill contaminated zone (area = 88,221 m², depth = 33.6 m). To distribute the annual average radionuclide concentrations in Table 2 over this contaminated zone, the concentrations are adjusted to reflect even aggregation into USEI's entire landfill mass from the dimensions above and a compacted density value of 1.5 g/cm³. The average annualized SPFM waste mass of 6.1E+09 g is divided by the USEI landfill mass of 4.45E+12 g resulting in a mass dilution factor of 1.37E-03. All other RESRAD code parameters for the USEI site remain constant. The results of this 'Baseline' model show a maximum annual dose of 0.242 mrem at approximately 326.1 years following closure of the facility. The RESRAD output report for this Baseline case is provided in Attachment 6.

The second RESRAD model is a postulated "concentrated burial" scenario, where all of the SPFM waste mass is buried within a much smaller portion of the landfill, resulting in less radioactive waste dilution from masses of other non-radioactive waste received concurrently at USEI. All of the SPFM waste (activity) in this scenario is assumed to arrive at USEI in a four month period, rather than over a full calendar year. The assumed annual volume from the SPFM facility (489,000 ft³) is converted to tons (6,724) using the average density value of 27.5 lb/ft³. Assuming that this entire mass of waste arrives at USEI in a 4-month period, it would be aggregated into one-third of the average annual waste receipts at USEI, or 184,756 tons (554,267/3), based on a five-year rolling average from 2008-2012. Dividing the SPFM total mass by the pro-rated USEI burial mass results in a concentrated scenario dilution factor of 3.64E-02. The results of the concentrated model show a maximum annual dose of 6.42 mrem at approximately 326.1 years following closure of the facility. The RESRAD output report for the 'concentrated' case is also provided in Attachment 6.

Several post-closure inadvertent intruder scenarios were also evaluated using the guidance provided in 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." Six inadvertent intruder cases, using three defined scenarios, were evaluated as part of this safety assessment:

1. Intruder Construction Scenario – Baseline: 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. Disposal of debris at annualized average radionuclide concentrations from SPFM is evaluated.
2. Intruder Construction Scenario – Concentrated: Same scenario as the Baseline Construction case with the exception that it is assumed that the same total activity from the SPFM is received at the USEI facility in a four-month period instead of over an entire year. All other parameters and assumptions were held constant.
3. Intruder Well Drilling Scenario – Baseline: 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 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 hrs and it is assumed that the contaminated layer is drilled through in 8 hrs. 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 baseline case uses the average annual shipped concentrations for both the waste (C_w) and well cuttings (C_w').

4. **Intruder Well Drilling Scenario – Concentrated:** Same scenario as the Baseline Well Drilling case with the exception that it is assumed that the same total activity from the SPFM is received at the USEI facility in a four-month period instead of over an entire year. The intruder driller drills directly through this postulated ‘concentrated’ waste pocket and brings the cuttings to the surface. All other parameters and assumptions were held constant.
5. **Intruder Driller Occupancy Scenario - Baseline:** An inadvertent intruder occupies the site upon which a well had been drilled through waste materials. The baseline case uses the average annual shipped concentrations for the exhumed well cuttings (C_w') in all calculations, which are the same as in the Baseline Well Driller scenario.
6. **Intruder Driller Occupancy Scenario - Concentrated:** Same scenario as the Baseline Driller Occupancy case with the exception that it is assumed that the same total activity from the SPFM is received at the USEI facility in a two-month period instead of over an entire year. The inadvertent intruder that occupies the site is exposed to the elevated well cuttings, which are assumed to be three-times higher than the baseline case (100% of annual activity shipped in a four-month period to USEI).

The results of all evaluated inadvertent intruder scenarios are provided in Table 5. Details of the assumptions, methods, and calculations for each inadvertent intruder scenario are provided in Attachment 7.

Table 5. Summary of Inadvertant Intruder Dose Evaluations for SPFM Waste at USEI

Scenario	Dose Pathway	Calculated Dose (mrem/yr)
Construction – Baseline	Air Uptake	1.98
	Direct Gamma	14.7
	Total:	16.7
Construction – Concentrated	Air Uptake	5.9
	Direct Gamma	44.1
	Total:	50.0
Well Driller – Baseline	Internal	0.84
	External	5.68
	Total:	6.53
Well Driller – Concentrated	Internal	2.53
	External	17.1
	Total:	19.6
Driller Occupancy – Baseline	Air Uptake	0.03
	Direct Gamma	0.95
	Total:	0.98
Driller occupancy - Concentrated	Air Uptake	0.08
	Direct Gamma	2.86
	Total:	2.94

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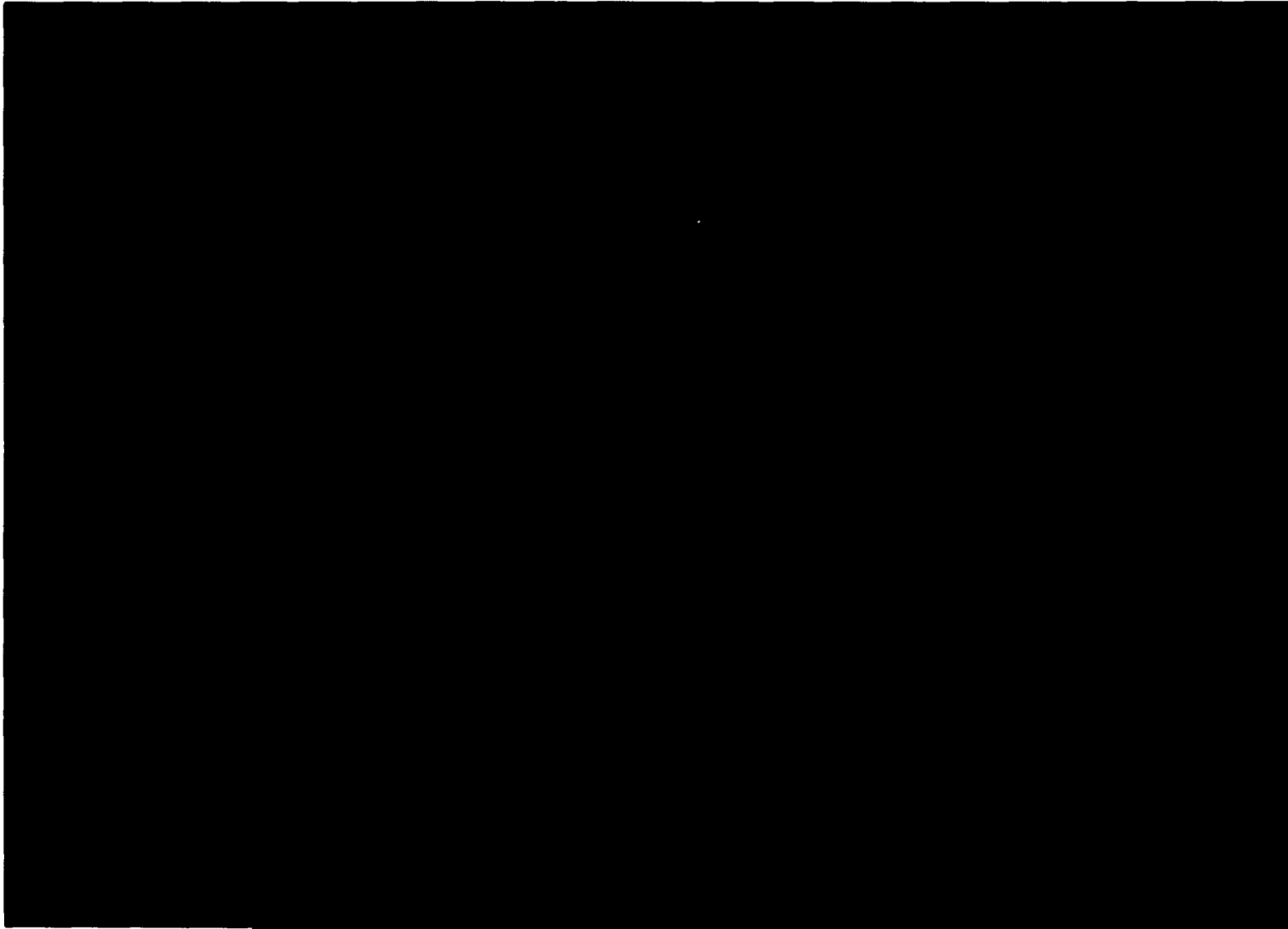
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US Ecology, Inc.
300 E. Mallard Dr., Suite 300
Boise, ID 83706
Phone: 208-331-8400 Fax: 208-331-7900
www.usecology.com

Studsvik Processing Facility Memphis, LLC
2550 Channel Ave PO Box 13143
Memphis, TN 38113
Phone: 901-775-0690 Fax: 901-775-0629
www.studsvik.com

7.0 CRITICALITY SAFETY

A Criticality Safety Assessment for the USEI site was performed as part of a prior Alternate Disposal Request submittal to the NRC for 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. 6 (NSA, 2012)" 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 1.0 g/cc). The U-235 activity concentration for this submittal (U-235 = 10 pCi/g as shown in Table 2) is at a significantly lower concentration than that previously shown to be inherently safe from a criticality perspective. Therefore, it is not necessary to complete a new Criticality Safety Assessment for U-235 waste shipped from the SPFM facility.

To extend this concept to accommodate the addition of Pu-239, a comparison to another previously published inherently safe value is referenced. Mr. Herbert Cember published an inherently safe mass concentration for both U-235 and Pu-239 in reference 7.7, "Introduction to Health Physics, 3rd edition," of 11.94 g per liter (g/L) and 6.9 g/L, respectively, in aqueous solutions. These values are quoted for aqueous solutions, as they are the most favorable geometry for potentially achieving an inadvertent criticality event. They also represent a conservative application with respect to soils and debris waste streams that have far more void space in the shipping containers. The 6.9 g/L Pu-239 concentration translates to an activity concentration in soil (at 1 g/cc) of 4.35E+08 pCi/g Pu-239 (applying a specific activity value of 0.063 Ci/g). The activity concentration in Table 2 (4 pCi/g) is less than 0.000001% of the inherently safe Pu-239 concentration.

The SNM potentially present in this request for alternate disposal (U-235 + Pu-239) is at significantly lower concentrations than those analyzed to be inherently safe from a criticality perspective for disposal at USEI. Therefore, it is not necessary to implement criticality safety procedures for any shipments of SPFM wastes proposed for disposal at USEI.

8.0 REFERENCES

- Code of Federal Regulations, Title 10 Part 20, Standards for the Protection Against Radiation.
- Code of Federal Regulations, Title 49, Part 172, Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information and Training Requirements.
- Code of Federal Regulations, Title 49 Part 173, Shippers General Requirements for Shipments and Packaging's.
- U.S. Environmental Protection Agency, 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. US EPA Office of Radiation Programs. September 1988.
- Rules of the Department of Environment and Conservation, Division of Radiological Health Chapter 1200-2-5, Standards for the Protection Against Radiation.
- Rules of the Department of Environment and Conservation, Division of Radiological Health Chapter 1200-2-10, Licensing and Registration.
- Rules of the Department of Environment and Conservation, Division of Solid Waste Management Chapter 1200-1-7, Solid Waste Processing and Disposal.

International Council on Radiation Protection, Publication No. 66. "Human Respiratory Tract Model for Radiological Protection." Ann. ICRP 24 (1-3). 1994.

NRC Regulatory Guide 1.86. Termination of Operating Licenses for Nuclear Reactors, June 1976.

NUREG 1575, Multi-Agency Radiation Site Survey and Investigation Manual.

NUREG-1761, Radiological Surveys for Controlling Release of Solid Materials (DRAFT), July 2002.

NUREG 1757, Consolidated NMSS Decommissioning Guidance.

NUREG/CR-5512, Residual Radioactive Contamination from Decommissioning (DRAFT), April 1996.

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. NSA-TR-09-14, Rev. 6. July 2012.

NRC Branch Technical Position on Radioactive Waste Classification, May 1983.

Nuclear Regulatory Commission, "Draft Environmental Impact Statement on 10CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste. Volume 4." NUREG-0782. Office of Nuclear Materials Safety and Safeguards. September 1981.

U.S. Nuclear Regulatory Commission, "Update of Part 61 Impacts Methodology." EnviroSphere Company, on behalf of US Nuclear Regulatory Commission. NUREG-CR/4370, Volume 1. January 1986.

Nuclear Regulatory Commission Branch Technical Position on Waste Form (Revision 1), January 1991.

Nuclear Regulatory Commission Branch Technical Position on Concentration Averaging and Encapsulation, January 1995.

Nuclear Regulatory Commission, "Revised Policy Statement on Volume Reduction and Low-Level Radioactive Waste Management." Draft-Final. SECY-12-0003, ML113400160. January 9, 2012.

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

User's Manual for RESRAD Version 6. Argonne National Laboratories, July 2001.

Canberra Model S573 ISOCS Calibration Software User's Manual.

CRC Handbook of Environmental Radiation, Kement, Alfred W. Jr (Editor), Brodsky, Allen (Editor in Chief). 1982.

Data Collection Handbook To Support Modeling Impacts Of Radioactive Material In Soil by C. Yu, C. Loureiro*, J.-J. Cheng, L.G. Jones, Y.Y. Wang, Y.P. Chia,* and E. Fail lace, Environmental Assessment and Information Sciences Division Argonne National Laboratory, Argonne, Illinois, April 1993.

“Confidential Information Submitted Under 10 CFR 2.390; Withhold from Public Disclosure under 10 CFR 2.390.”

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US Ecology, Inc. Request for Exemptions
Revision 0, June 2013*

9.0 ATTACHMENTS

Attachment 1 - USEI Site Description

Attachment 2 - USEI Part B Permit EPA ID IDD073114654, Latest Revision Date July 18, 2012, Part C.3
– Waste Acceptance Criteria

Attachment 3 - SPFM Annualized Dose Assessment Data Summaries
a. IMC Shipments by Rail
b. IMC Shipments by Rail

Attachment 4 - Microshield Reports for Annualized SPFM Dose Assessment

Attachment 5 - Internal Dose Rate Calculation Results

Attachment 6 - Summary of RESRAD Input Parameters for USEI

RESRAD Parameter Input Summary and Summary Reports for USEI Baseline and Concentrated Burial Scenarios

Attachment 7 - Inadvertent Intruder Dose Assessment Results

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Attachment 9 - Nuclear Safety Associates, “Nuclear Criticality Safety Assessment of the US Ecology Idaho (USEI) Site for the Landfill Disposal of Decommissioning Waste from the Hematite Site, Rev. 2.” NSA-TR-09-14. December 2011.

Attachment 10 – Canberra Industries, Inc. “Model S573/S574 ISOCS/LabSOCS Validation and Verification Manual. Ver. 4.0. 2002.

ATTACHMENT 1
USEI SITE DESCRIPTION

Exhibit A: US NRC SERs for WEC Hematite Project (ML1114410870, ML13039A208)

Exhibit B: *Hazardous Waste Facility Siting License Application for Cell 16* (American Geotechnics, June 30, 2006); This document describes US Ecology Idaho's environmental setting and was accepted by the Idaho Department of Environmental Quality as part of the 2005 siting process, which resulted in IDEQ approval (December 6, 2006) of USEI's request to expand its landfill operations. (Provided in electronic format only on CD-ROM)

Exhibit C: *Summary of Hydrogeologic Conditions and Groundwater Flow Model for US Ecology Idaho Facility, Grand View, Idaho* (Eagle Resources, 2010). This document provides a detailed description of site geology and hydrogeology. (Provided in electronic format only on CD-ROM)

Exhibit D: *Site-Specific RESRAD Water Pathway Parameters for the Contaminated Soil, Vadose Zone, and Saturated Zone*. US Ecology Idaho, Grand View, Idaho (Eagle Resources, 2005). This document established site-specific RESRAD parameters that have been incorporated into USEI's RCRA Part B permit. (Provided in electronic format only on CD-ROM)

SAFETY EVALUATION REPORT

**REQUEST FOR ALTERNATE DISPOSAL APPROVAL AND
EXEMPTIONS FOR SPECIFIC HEMATITE DECOMMISSIONING
PROJECT WASTE AT
US ECOLOGY'S IDAHO FACILITY**

October 28, 2011

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1. INTRODUCTION

On May 21, 2009, Westinghouse Electric Company, LLC (WEC) requested that the U.S. Nuclear Regulatory Commission (NRC) approve alternate disposal (ML091480071), in accordance with 10 CFR §20.2002, of specified low-activity radioactive materials from the Hematite Decommissioning Project (HDP), for certain waste containing source material, byproduct material, and special nuclear material (SNM). WEC also requested a specific exemption from 10 CFR §30.3 and 10 CFR §70.3 pursuant to 10 CFR §30.11(a) and 10 CFR §70.17(a). Unimportant quantities of source material are exempted from 10 CFR Part 40 requirements pursuant to 10 CFR §40.13(a). The NRC's approval of the 10 CFR §20.2002 request, along with the requested exemptions, would allow WEC to transfer the specific waste for disposal at the US Ecology Idaho, Inc. (USEI) Resource Conservation and Recovery Act (RCRA) Subtitle C disposal facility.

The US Ecology Idaho facility is a RCRA Subtitle C hazardous waste disposal facility permitted by the Idaho Department of Environmental Quality (IDEQ), and is not an NRC licensee. It is located near Grand View, Idaho in the Owyhee Desert. The HDP material would be disposed of in Cell 15, which has an area of 88,220 m² (21.7 acres) and a depth of 33.6 m. The most important natural site features that limit the transport of radioactive material are the low precipitation rate (i.e., 18.4 cm/y (7.4 in. per year)) and the long vertical distance to groundwater (i.e., 61-meter (203-ft) thick on average unsaturated zone below the disposal zone).

As is usual with a RCRA Subtitle C site, a number of engineered features are present to enhance confinement of contaminants over the long term. These features include an engineered cover, liners and leachate monitoring systems. Operations at the site include a number of systems that minimize the potential for exposure of workers to any waste handled by the facility. These systems include a closed facility with filtered ventilation exhaust for transfer of incoming waste material from the shipping conveyance to trucks for transport to the cell, mechanized equipment for disposition of waste material in the cell, and the application of an asphaltic spray at the end of each day's operations. The site is permitted to receive non-Atomic-Energy-Act material or exempted radioactive material that meets site permit requirements.

The NRC reviews the safety implications of disposing unimportant quantities of material at disposal facilities that are not licensed by the NRC or an NRC Agreement State, as would be authorized by the NRC's approval of WEC's §20.2002 request.

The 10 CFR Part 20 dose limit for individual members of the public is 100 mrem/yr (1 mSv/yr) (10 CFR §20.1301). The NRC's practice is to approve §20.2002 requests that result in a dose not exceeding a few millirem per year because it is a fraction of the natural radiation dose (approximately one percent of the radiation exposure received by members of the public from background radiation), a fraction of the annual public dose limit, and an attainable objective in the majority of cases (see SECY-07-0060 and NUREG-1757). The NRC has approved one §20.2002 request that exceeded a few millirem per year, but was less than 25 millirem per year.

The NRC's review of a 10 CFR §20.2002 request for disposal of low-activity waste in a RCRA facility covers protection of individuals, inadvertent intruders, and the public. The period of performance is 1,000 years after the expected date of license termination of the facility, consistent with 10 CFR 20.1401 (the License Termination Rule in Subpart E of 10 CFR Part 20). Given the quantity of material being disposed and the nature of the disposal facility, a performance period of 1,000 years is considered adequate.

Because this 10 CFR §20.2002 disposal request included SNM, the staff assessed nuclear criticality safety, material control and accounting, and physical security aspects. These assessment areas are atypical for 10 CFR §20.2002 requests. The following SER sections address these aspects of the staff's review in addition to the staff's review of WEC's dose assessment.

2. BACKGROUND

The Hematite site was used for the manufacture of low-enriched, intermediate-enriched, and high-enriched materials during the period of 1956 through 1974. In 1974, the production of intermediate- and high-enriched material was discontinued and all associated materials and equipment were removed from the facility. From 1974 to cessation of manufacturing operations in 2001, the Hematite facility produced nuclear fuel assemblies for commercial nuclear power plants. In 2001, fuel manufacturing operations terminated and the facility license was amended to reflect a decommissioning scope of operations.

Activities at the Hematite site generated a large volume of process wastes contaminated with uranium of varying enrichment. Based on historic documentation, 40 unlined pits were excavated and used for the disposal of contaminated materials generated by fuel fabrication processes at Hematite between 1965 and 1970. The primary waste types expected in these pits are trash, empty bottles, floor tile, rags, drums, bottles, glass wool, lab glassware, acid insolubles, and filters. The recorded total uranium mass associated with the burial items ranged from 178 g of U-235 to 802 g of U-235 per burial pit, with a maximum amount associated with any single burial item of 44 g U-235. The U-235 enrichment of the material ranged between 1.65% and 97%. Based on available documentation, WEC determined that it is unlikely that the burial pits contain an unsafe mass of U-235. However, WEC estimates that a total of 20-25 additional burial pits might exist for which there are no records.

The excavated material from these burial pits could be shipped to the USEI facility if the material meets criteria established by WEC and approved by the NRC for this §20.2002 disposal request. Highly enriched uranium (HEU) will not be shipped to the USEI facility.

3. DOSE EVALUATION

WEC supplied information on the source term of the waste and a description of the job functions to evaluate different possible exposures for various members of the public. These scenarios included the doses to the transportation workers and USEI workers, post-closure dose to the general public, and dose to an intruder. For §20.2002 reviews, all the scenarios treat exposed individuals as members of the public because the material is proposed to be sent to a facility that is not licensed by the NRC or an NRC Agreement State.

3.1. Source Material

WEC estimates the volume of the waste that will be a candidate for disposal at USEI to be 22,809 cubic meters at a waste density of 1.69 g/cm³ (e.g., approximately 50,000 tons). Since the dose assessment calculations assume this amount as a limit, 50,000 tons will be an upper bound on the amount that WEC is permitted to send to USEI under its requested exemption. The waste is soil and debris with low concentrations of both SNM and byproduct material contaminants. WEC determined the radionuclides of concern based on studies in the Hematite Historical Site Assessment (ML092870417, ML092870418), which are summarized in Chapter 4 of the Decommissioning Plan (ML092330136).

WEC makes the following assumptions regarding the concentration of the source material: (1) the expected average concentrations of the radionuclides that would be shipped from the Hematite site to USEI and (2) the dilution that would occur when it is deposited along with other waste streams arriving at USEI. These source term estimations are reproduced in Table 3-1.

Table 3-1 Source Term Concentration of Radionuclides

Radionuclide of Concern	Expected Average Concentration transported from Hematite Site to USEI (pCi/g)	Concentration at USEI accounting for dilution during disposal (pCi/g)
Radium-226 (Ra-226)	1	2.12E-02
Thorium-232 (Th-232)	1.2	2.55E-02
Technicium-99 (Tc-99)	27	5.73E-01
Uranium-234 (U-234)	113	2.40E+00
Uranium-235 (U-235)	5.5	1.17E-01
Uranium-238 (U-238)	18	3.82E-01

The source term concentrations assumed to be transported from Hematite are derived from characterization measurements taken from samples at three depth ranges in the soil obtained within the contours illustrating the soil volumes expected to require excavation. The data are summarized in the Soil Contour Data Set, which was supplied to the staff in WEC's letter dated March 31, 2010 (ML100950397). WEC's letter was in response to a March 3, 2010 conference call with the NRC, which discussed WEC's December 29, 2009 response (ML100320540) to the staff's request for additional information (RAI) (ML093360222).

The three depth ranges for the soil samples are Surface (0 m to 0.5 m), Root (0.5 m to 1.5 m), and Deep (1.5 m to 6.7 m). The median value of the concentrations measured in each stratum is weighted by the volume of waste within the stratum. Because Th-232 and Ra-226 were found in limited areas, the volumes of soil associated with these radionuclides are factored in as separate areas from that associated with the larger areas containing Tc-99 and uranium. WEC applied the weighted median value, instead of the weighted mean, as an expected value due to the presence of some high-concentration samples of Tc-99, specifically samples labeled EP-08-00-SL and EP-10-00-SL near the evaporation ponds.

The Soil Contour Data Set has a smaller volume of soil (8,710 m³) than the projected amount to be sent to USEI (22,809 m³) because it does not contain some of the lower activity additional material that will inevitably be mixed during excavation. WEC estimated that a larger volume of waste would be shipped than what WEC believes will be required to allow for uncertainty in the volume of material that might eventually be sent to USEI.

The staff has reviewed the source term characterization data and has concluded that there is a high level of variability within the Soil Contour Data Set. While the staff does not agree with the application of the weighted median as opposed to the weighted mean, the staff agrees that a statistically based sampling and characterization plan can be used to verify the assumptions applied in this analysis.

The staff expressed the importance of properly characterizing the waste to the licensee in teleconferences on March 3, 2010 and on April 19, 2010. In response to these discussions, WEC identified the elevated samples in the dataset, including the adjacent biased samples that aided in defining the extent of each area of elevated activity (ML100950386). WEC estimates that a total volume of 5 m³ bounds the extent of the highly contaminated soil around the two samples labeled EP-08-00-SL and EP-10-00-SL near the evaporation ponds.

In reviewing the provided information and the samples in the Hematite Radiological Characterization Report (ML092870496, ML092870506), the staff notes that no subsurface samples were taken below one of the samples (EP-10-00-SL) with highly elevated Tc-99 levels. The nearest samples are approximately 25 ft away from this sample. The staff concluded that, in the case where the 5 m³ does not adequately bound the volume of highly contaminated soil, the additional activity limits that WEC is placing on key radionuclides will be adequately protective, and will also restrict the volume of waste that WEC is able to send to USEI.

On May 24, 2010, WEC submitted its plan for further characterizing the waste as it is being prepared for shipment to USEI (ML101450240). A revised and final plan was submitted on February 18, 2011 (ML110530153). In the plan, WEC commits to characterizing the soil in the immediate vicinity of the highly contaminated samples separately from the remainder of the material, and states that this plan ensures the activity limits for the key radionuclides will remain within the scope of this analysis. The specific limits for key radionuclides are discussed in the Dose Assessment Results Section below (Section 3.3). The staff assessment of the adequacy of the characterization plan is contained in the Health Physics Evaluation Section below (Section 4).

The concentrations of the material shipped from the Hematite facility are reduced to reflect the intermixing of the 22,809 m³ (approx. 42,425 tons) of Hematite waste with the two million tons total waste arriving at USEI by multiplying by a dilution factor of 0.0212 (42,425 tons / 2.0E+6 tons). The projected shipment schedule for the Hematite site waste ranges from 18 months to 3 years. The annual disposal rate for USEI averaged over the past five years was 711,000 tons per year (13,673 tons/wk). Applying this average annual disposal rate, 2.0E+6 tons would be shipped over a period of about 2.8 years. In response to additional information requested during a June 21, 2010 phone call, WEC analyzed the impact on variations in the shipping volumes to be sent from Hematite, and the rates at which such volumes would be shipped and provided the information to the NRC on June 25, 2010 (ML110560334). WEC's bounding scenario assumes a maximum shipping rate of 20 railcars per week, and a minimum amount of

waste of 5,702 m³. This bounding scenario sends a smaller amount of waste at a higher concentration over a shorter period of time, which reduces the dilution with other non-radiological waste that would be arriving at USEI. In all scenarios, WEC assumes that USEI is receiving and disposing of a total amount from all waste sources of 13,673 tons/wk.

The staff found that WEC needed to justify its dilution assumptions for the homogeneous mixing (711,000 tons/yr i.e., 13,673 tons/week). WEC 's provided its justification in its March 31, 2010, submittal (ML100950397), in which the waste disposal amounts for USEI over the last five years were provided. Based on this five year average, the staff finds the assumption of a 711,000 tons/yr disposal rate for USEI to be reasonable.

3.1.1. Radiological Dose Scenarios

3.1.2. Transportation and Worker Doses

WEC analyzed doses to transportation workers and USEI exposure groups, including the gondola surveyor, excavator operator, gondola cleanout worker, truck surveyor, truck driver, stabilization operator, and cell operator. The dose to the transportation workers is bounded by the dose to the USEI worker groups due to the amount of distance and shielding that occurs throughout transportation from Hematite to USEI. (There is no internal dose to transportation workers since the gondola cars are covered.) WEC calculated that in order for a bystander to receive equal or greater dose than the maximally exposed USEI worker of 0.49 mrem, the individual would have to spend 408 hr at 1 meter from the gondola (490 μ R/1.2 μ R/hr@1m), or 326 hr at 1 foot away (490 μ R/ 1.5 μ R/ hr@1ft). WEC does not consider either of these to be credible exposure scenarios during transport. In WEC's March 31, 2010 submittal, WEC applied several methods described in the relevant literature to calculate the exposure time for a single transportation worker. WEC conservatively assumed the same individual inspected all 400 projected railcars (ML100950397). The longest exposure time estimated was 20 hours, which is significantly less than the amount of time required to receive a dose equivalent to the Maximum Exposed Individual. Twenty hours is the amount of time associated with a person inspecting the train as it is coming inbound. It is not for the transportation worker (engineer) who is operating the train. The staff finds this assumption acceptable given that there is sufficient distance and shielding for the engineer.

The analysis for the USEI employees assumes a specific number of workers per year will be available to carry out each of the job functions, and the dose is divided equally among all workers within a job function group. Job functions are not shared among employees of the excavator operator, truck driver, stabilization operator, and cell operator groups because the work crews are not assumed to overlap. However, the groups of gondola surveyors, gondola cleanout crews, and truck surveyors may include the same individual employees. WEC estimates that even if one individual carried out all the tasks for all three functions for the entire project (an impossible scenario), the hypothetical individual would receive 2.096 mrem. Table 3-2 summarizes the job function scenario assumptions. The minutes assigned is the amount of time for one person to perform each function one time.

Table 3-2 Job Function Scenario Assumptions

Job Function	Number of Workers in Group	Minutes to Perform Task	Type of Conveyance (count)
Gondola Surveyor	8	20	Gondola (400)
Excavator Operator	4	45	Gondola (400)
Gondola Cleanout	8	10	Gondola (400)
Truck Surveyor	8	5	Truck (1200)
Truck Driver	14	45	Truck (1200)
Stabilization Operator	6	45	Gondola (20)
Cell Operator	2	15	Gondola (400)

The waste will arrive at USEI's rail transfer facility in gondolas. The gondola surveyor will survey each gondola prior to the gondola being unloaded. The excavator operator transfers the material from the gondola into dump trucks. After the gondola is emptied, it is swept out using brooms and shovels by the gondola cleanout worker.

Once the dump truck has been loaded, it is surveyed by the truck surveyor before the truck driver transports the material from the rail transfer facility to the disposal site. WEC estimates that approximately 5% of the contaminated material contains heavy metals that will require stabilization prior to disposal. If the material has been identified as part of the 5% requiring treatment for hazardous material, it will be taken to a treatment building where it will be transferred into a RCRA-compliant treatment tank. A stabilization operator will then wet and mix the waste with the appropriate reagents. Stabilized waste will then be placed back into a dump truck and transported to the disposal site. At the disposal site, the disposal cell operators will spread and compact the waste that is deposited from the dump truck.

For the purposes of calculating worker doses, WEC conservatively assumes that all the work is completed in a single year, although the schedule allows for the project to be carried out over 18 months to 3 years. Furthermore, no credit is taken for USEI's respiratory protection program, including negative airflow in the stabilization building and commercial HEPA filtration systems in the cabs of the trucks.

3.1.3. Post-Closure and Intruder Dose

In addition to evaluating worker scenarios, WEC included a long-term post-closure analysis assuming a resident farmer scenario as well as the dose to the inadvertent human intruder who digs a well or constructs a house with a basement that intrudes into the disposal cell. WEC used the RESRAD code Version 6.4, applying site-specific parameters where appropriate, to calculate the long-term post-closure dose. To calculate dose to the intruder, WEC used the methods from NUREG/CR-4370, *Update of Part 61 Impacts Analysis Methodology*, January 1986.

The appropriateness of the RESRAD model for the Grand View site was reviewed by US Ecology health physics staff upon US Ecology purchasing the site from EnviroSAFE in 2001. The US Ecology staff concluded that the code was appropriate for the site conditions. In 2005, USEI

hired consultants to review the input values used for RESRAD, and determine site-specific inputs to be used with the code that more accurately reflect the site environmental conditions. Most of the site-specific parameters are explained in the 2005 report titled "Site-specific RESRAD Water Pathway Parameters for the Contaminated Soil, Vadose Zone, and Saturated Zone," provided in the RAI response dated December 29, 2009 (Attachment 5, ML100320540). For those parameters not described in the report, WEC provided a justification with its March 31, 2010 submittal (ML100950397).

The long-term stability of the site is important when considering long-term post-closure dose. Site-stability can be impacted by natural surface and subsurface processes, and is also impacted by the stability of the waste and engineered barriers of the disposal facility. In WEC's March 31, 2010 submittal, WEC provided a technical basis for the site stability of USEI stating that the facility was "constructed in compliance with the Resource Conservation and Recovery Act (RCRA) standards and the applicable Minimum Technology Requirements (MTRs). These requirements provide conservative criteria for cell construction to insure long-term stability and are consistent with the erosion design requirements in 10 CFR Part 61, and the joint NRC/EPA guidance document with guidelines on drainage and processes impacting stability." The staff finds this technical basis sufficient for demonstrating long term site-stability.

3.2. Dose Assessment Results

WEC supplied spreadsheet calculations, or results from RESRAD or Microshield, as appropriate, for each of the job functions and long-term scenarios associated with the waste disposal. The doses to workers are from the original submittal (ML091480071). The doses to members of the public are from the May 24, 2010 submittal (ML101450240) and the doses to intruders are from the March 31, 2010 submittal (ML1009503860).

WEC's scenarios included appropriate assumptions about working conditions and realistic exposure times. The staff finds the selection of scenarios and site-specific parameters to be acceptable considering the site environment and characteristics.

3.2.1. Operational External Dose

The external dose per worker at USEI is based on the external dose rate, the handling time indicated for each conveyance, the number of conveyances, and the number of workers sharing a job function. An external dose rate per conveyance in mR/hr was estimated by WEC using Microshield with the average concentrations and the data presented in Tables 3-1 and 3-2. This dose rate is multiplied by the amount of time that it takes to process a single conveyance (shown in the second column in Table 3-2) to obtain an external dose per conveyance. The dose to each worker is obtained by multiplying the dose per conveyance by the number of times that the worker will have to perform the task (number of conveyances) and then dividing by the number of workers. The inventory representing the present concentration of the parent radionuclides was decayed for 30 years to allow for in-growth of short-lived progeny. Since the parent radionuclides have long half lives relative to 30 years, this approximation in source term inventory for the Microshield analysis is reasonable. The staff finds the methods applied for estimating external worker doses to be acceptable.

3.2.2. Operational Internal Dose

The internal dose is dependent on the concentration of respirable dust in the air at the work locations (0.23 mg/m³), an inhalation rate (1.2 m³/hr), the radionuclide concentrations, the dose conversion factors from U.S. Environmental Protection Agency Federal Guidance Report (FGR) 11 (EPA 520/1-88-020 September 1988), and the handling times per conveyance. The concentration of respirable dust assumed in the dose assessment is based on a dust study that was performed at the US Ecology Idaho facility. In this study, measurements were made of the workers' exposure rates to total and respirable dust. The concentration of respirable dust assumed in the dose assessment is based on a dust study that was performed at the USEI facility in August 2008 (Attachment 2 of ML100320540). The dose was assessed for workers engaging in the same activities that they would perform when handling the Hematite waste. The types of workers evaluated included the Rail Transfer Facility (RTF) excavator operators, the workers responsible for sweeping and shoveling waste in the gondola, field technicians who perform radiological surveys, and process supervisors. The respirable dust concentration used in the internal dose calculations, 0.23 mg/m³, was based on the highest result measured in the USEI dust study. The staff finds that the calculations performed for the internal dose due to inhalation of dust from the Hematite waste were performed correctly and that the parameter values used in this calculation were appropriate.

For both internal and external dose, the dose per conveyance is multiplied by the total number of conveyances per year and divided equally among the number of workers in the job function. The dose results calculated by WEC are summarized in Table 3-3. The staff finds the methods applied for estimating internal worker doses to be acceptable.

Table 3-3. Annual Dose (mrem) per Person for Individual Job Function

Job Function	Internal (mrem/yr)	External (mrem/yr)	Total (mrem/yr)
Gondola Surveyor	9.0E-02	2.0E-02	1.1E-01
Excavator Operator	4.1E-01	6.3E-02	4.7E-01
Gondola Cleanout	4.5E-02	1.4E-02	5.9E-02
Truck Surveyor	6.8E-02	2.5E-02	9.3E-02
Truck Driver	3.5E-01	1.5E-01	4.9E-01
Stabilization Operator	1.4E-02	2.1E-03	1.6E-02
Cell Operator	2.7E-01	1.1E-01	3.8E-01

3.2.3. Post-Closure Dose

WEC estimates the post-closure long-term dose to be approximately 2 mrem. This dose is a slight decrease from the original estimate in the May 21, 2009 submittal. In response to the staff's RAI dated May 24, 2010 (ML101450240), WEC revised the post-closure dose using the RESRAD computer code. The dose decreased from 0.029 mSv to 0.019 mSv (2.9 mrem to 1.9 mrem) with the peak dose occurring around year 247 following disposal. This decrease resulted from changes to two input parameters: contaminated zone thickness and radionuclide source term concentrations. Specifically, the contaminated zone height was adjusted from 33.6 m to 14.93 m to reflect the height of the waste that would occupy the cell. The height was

determined from the volume of waste disposed, the density of the waste, and the area over which the waste was spread. The radionuclide concentrations were corrected to account for the difference in density of the waste as it is shipped (1.69 g/cm^3) to when it is emplaced with other waste (1.5 g/cm^3). The dose is delivered through the groundwater pathway, and Tc-99 is the primary contributing radionuclide.

Since Tc-99 is the primary contributing radionuclide, the total quantity of Tc-99 (as opposed to the concentration) will drive the dose consequences. RESRAD applies the concentration of Tc-99 and the volume of soil in the contaminated zone to determine the total activity quantity of Tc-99 that is available in uptake pathways. The value that WEC applies for the expected concentration of Tc-99 is 27 pCi/g . This concentration over $22,809 \text{ m}^3$ yields an expected total Tc-99 inventory of approximately 1 Ci.

The staff finds that if the total inventory of Tc-99 based on the average concentration and total volume shipped remains below 1 Ci, the proposed request will not yield a dose that is more than a few mrem/yr. WEC plans to sample the outgoing shipments to ensure that the inventory calculated from the mean activity concentrations, derived from the mass-weighted concentrations of each stockpile, remains below 1 Ci. In addition, the sampling plan will ensure that a 1.6 Ci limit for the 95th upper confidence limit will not be exceeded. WEC derived the 1.6 Ci upper confidence limit by assuming a standard deviation roughly equivalent to 1 mrem. The dose that WEC calculated resulting from a total inventory of 1.6 Ci is approximately 3 mrem. The staff finds this approach for determining the upper confidence limit to be acceptable because the dose consequences at the upper confidence limit remains a few mrem/yr. A detailed discussion of the review of the waste shipment characterization plan is contained in the Health Physics Evaluation of this report in Section 4.

3.2.4. Inadvertent Intruder Dose

WEC performed inadvertent intruder analyses (ML100950386) based on the intruder construction scenario and the intruder well drilling scenario described in Appendix G of NUREG-0782, "Draft Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste". The Pathway Dose Conversion Factors (PDCFs) applied are taken from NUREG/CR-4370, Volume 1.

In all intruder analyses performed by WEC, two different assumptions for the concentration shipped from Hematite to USEI were applied. The first assumption utilized Expected Average Concentration values from Table 1 of the May 21, 2009 submittal. The second assumed that the total sum of radionuclide material shipped is at the USEI Waste Acceptance Criteria (WAC). For both assumptions, two dilution scenarios were also applied for the waste that the intruder contacts. One scenario, the Average Cell Concentration scenario, assumes waste is uniformly mixed within the USEI cell. The second, the One-Ft Layer scenario, assumes the intruder contacts a one-foot layer of waste at its shipping concentration.

3.2.4.1. Intruder Well-Driller Scenario

Two intruder well-driller scenarios were considered by WEC. One was the acute well-driller. The other was the chronic well-driller.

The acute well-driller scenario assumes that the intruder digs a well by drilling through the waste disposal cell to reach the underlying aquifer at a depth of 93.1 m. The total period of exposure is 40 hours, 8 of which occur during the drilling through the contaminated layer. Therefore, for 8 hours, the driller is exposed to undiluted drill cuttings, and for the remaining 32 hours, the driller is exposed to the cuttings diluted by the ratio (0.31/93.1 or 3.3E-3) of the 1-ft contaminated layer (0.31 m) to the total well depth of 93.1 m. This dilution ratio is multiplied by the average cell concentration or the WAC concentrations. WEC calculated a dose to the acute well-driller of 2.9 mrem based upon the intruder drilling through a 1-ft layer at the WAC concentrations.

The chronic well-driller scenario assumes that the intruder spreads the exhumed drill cuttings around the residence and grows a garden in soil containing the drill cuttings. The concentration in the soil around the house is estimated to be 0.1 multiplied by the *expected average concentration* of the waste transported in Table 3-1. The staff finds this dilution factor reasonable for the Average Cell Concentration scenario since it results in less dilution of the material than what is assumed for the drill cuttings resulting from the drilling action in the acute well-driller scenario. The dose to the chronic well-driller calculated by WEC was 2 mrem/yr based upon the average concentrations (not the WAC concentrations).

The NRC staff finds the assumptions and pathways considered for the well-driller scenarios to be reasonable based on comparison to the guidance in Appendix G of NUREG-0782 and NUREG/CR-4370 Volume 1. The staff notes that WEC did not consider a scenario where the chronic well-driller encounters waste that is estimated to be 0.1 multiplied by the undiluted WAC values as opposed to average values (see Section 4.5 for the NRC's independent evaluation of this scenario).

3.2.4.2. Intruder Construction Scenario

In the construction scenario described by WEC, which is partly based on NUREG-0782, the inadvertent intruder is assumed to excavate or construct a building on a disposal site following a breakdown in institutional controls. The intruder is exposed to dust particles through the inhalation pathway, and may also be exposed to direct gamma radiation resulting from airborne particulates and by working directly in the waste-soil mixture.

For the Average Cell Concentration scenario, the waste is diluted by a factor of 0.0212 to account for mixing within the USEI cell with 2 million tons total waste. The 0.0212 factor is calculated by taking the ratio of Hematite waste to total waste received (42,425 tons / 2.0E+6 tons). For the One-Ft Layer scenario, the concentration is diluted by a factor of 0.31 (12 in/39 in) to account for USEI's practice of layering materials into pits in 1-ft layers and an assumption of 1 meter (39 in) of waste at the time of intrusion. The dose from the inhalation and from external gamma exposure is evaluated for a duration of 500 working hours, or a construction period of 3 months.

In both the Average Cell Concentration and the One-Ft Layer scenarios, WEC assumes that the shipped waste is further diluted by a factor of 0.5 due to particular disposal practices regarding waste emplacement (i.e., most of the waste is soil or soil like material and will be not be in containers). This assumption results in a total dilution factor of 0.5×0.31 , or 0.15, for the One-Ft Layer scenario, and a factor of 0.5×0.0212 , or 0.01, for the Average Cell Concentration scenario. WEC derives this 0.5 further dilution factor from the description of the site and the design operation factor associated with decontainerized waste noted in NUREG-0782. The design operation factor takes into account dilution due to particular disposal practices regarding waste emplacement.

The staff agrees that the 0.5 dilution factor for decontainerized waste (pg G-43 of NUREG-0782) is appropriate. However, the staff did not believe that it was appropriate to use this dilution factor in conjunction with additional dilution assumptions related to waste emplacement such as USEI's practice of layering materials into pits in 1-ft layers. Still, WEC did not assume any credit for the mixing of the waste with the cover material, which ranges from 0.76 m (2.5 ft) across the top to 6.10 m (20 ft) down the side slopes (see RAI Response to Performance Assessment RAI No. 9, ML100320540). Since USEI restricts the emplacement of any radioactive waste to within 3.6 meters of the surface of the finished cap of the cell, the construction scenario could be disregarded as not a feasible scenario. Furthermore, WEC does not take credit for decay up to the intrusion event, or for waste form or solidification. Therefore, the staff considers the total dilution factor of 0.15 acceptable for the One-Ft Layer scenario and the total dilution factor of 0.01 acceptable for the Average Cell Concentration scenario. The bounding dose for the construction intruder that WEC calculates is 10 mrem, and assumes waste shipped at the WAC values is encountered in a One-Ft Layer.

Based on the discussion above, the staff finds the assumptions and pathways considered in the intruder construction scenario to be reasonable. The staff finds this dose acceptable, given that WEC did not assume credit for the cover material. The staff notes that the time for this scenario was limited to 500 hours. The intruder construction scenario that WEC analyzed does not account for the chance that the intruder could subsequently live and grow food onsite. The reasons cited by WEC include the site's remote location and arid environmental conditions. The staff agrees with the technical basis for why intruder agricultural practices at the site are highly improbable.

3.3. NRC Staff Independent Calculations

As part of the review, the staff conducted independent calculations to verify the accuracy and appropriateness of all the calculations submitted by WEC. In addition, the staff performed independent calculations for some scenarios either not performed correctly by WEC or not considered by WEC. These additional calculations are summarized in the following sections.

3.3.1. Worker Dose

Because WEC did not calculate the potential USEI worker dose considering gondolas shipped at the WAC, the staff conducted an independent analysis to verify the doses to workers for shipments at the WAC. Assuming every package has concentrations of radionuclides that are at the WAC, the highest dose to USEI workers is to the truck driver and is found to be a few mrem/yr and acceptable.

3.3.2. Post-Closure Dose

Since the compliance limit for Tc-99 is a quantity of material as opposed to a concentration, the staff calculated the postclosure dose assuming that shipments were sent at a maximum possible Tc-99 concentration and maximum shipping rate (20 railcars/wk) until the 1 Ci limit was reached. The value of 599 pCi/g was chosen for this bounding scenario since this is the maximum value that WEC will permit for a single composite sample in a stockpile. This concentration would result in approximately 15 railcars (125 tons/railcar) sent in under one week (or about 0.7 wks). Assuming that 13,673 tons/wk, or about 100 railcars/wk, total waste (Hematite and non-Hematite) is shipped to USEI during the shipping campaign, the Tc-99 concentration would be diluted by a factor of 15/100 or 0.15. The staff found the dose for this scenario to be within a few millirem per year to individual members of the public and determined that the peak dose occurred within 250 years post-disposal.

The staff also considered the case of disposal of the Hematite waste with no additional dilution. This scenario assumes that the material contained in the 15 railcars with 599 pCi/g Tc-99 is spread out to a thickness of 0.15 m (6 in) in accordance with USEI practice of applying 6 – 12 in layers. The area for the contaminated zone is derived to be 7,413 m², and the concentration remains at 599 pCi/g Tc-99. This conservative analysis of post-closure dose, conducted by the staff, also yielded a dose less than the Part 20 dose limit of 1.0 mSv/yr (100 mrem/yr) to members of the public.

The staff concluded that the limits for all other radionuclides should be concentration-based in order to result in a dose of less than a few millirem/yr. For uranium, WEC would hypothetically be permitted to send up to the WAC (3,000 pCi/g), whereas Ra-226 and Th-232 are limited to 13 pCi/g and 16 pCi/g, respectively. The staff calculated the post-closure dose assuming that the total amount of material (22,809 m³) was shipped at the WAC value for each uranium isotope. The staff's calculations found the consequences to be a few mrem/yr. Similarly, if the total amount of material was shipped at the maximum Ra-226 and Th-232 concentrations, the post-closure dose would also be a few mrem.

While staff does not evaluate cumulative doses from multiple exemption requests from different licensees that could involve the same disposal facility, the staff recognizes that there might be cumulative effects. In this consideration, USEI has defined the WAC that determine specific concentrations of radioactive waste that can be accepted. The WAC are set to ensure the facility is able to meet its 15 mrem/yr limit. Each year, the facility produces an annual report that accounts for the total Ci inventory of radionuclides for the entire site.

3.3.3. Inadvertent Intruder Dose

Since WEC did not consider a scenario where the chronic well-driller encounters waste that is at the maximum as opposed to average values, the staff performed an independent calculation assuming that the uranium is sent at the WAC, Ra-226 and Th-232 were shipped at 13 and 16 pCi/g, and Tc-99 was shipped at 599 pCi/g. The well is assumed to be 93.1 m deep with an 0.28 m (8 in) diameter. Instead of assuming a 0.3048 m layer (1 ft) of waste, the staff assumed a 0.55 m layer (1.8 ft) of contamination within the well volume. The 0.55 m thick layer was derived from the total volume of 22,809 m³ spread over half the cell area, or 40,469 m². The staff's assumption of a thicker layer of Hematite waste in the cell was to account for the possibility that all the shipments could be sent at their maximum concentration values and placed in the cell without dilution from other non-Hematite waste. The total volume of material in the drill cuttings is 26 m³, and this is spread out over 500 m² to a depth of 0.052 m. With the mixing of the total drill cuttings, the concentrations of radionuclides in the contaminated layer are diluted by a factor of 0.006 (0.55 / 93.1). The staff found the all-pathways dose for this scenario, including radon, to be within a few mrem.

The peak dose for the intruder occurs in the first year when the intruder comes into contact with the waste when the concentration levels of the Ra-226 in the soil at the time of burial are highest. The staff also considered a conservative analysis of an inadvertent intruder who exumes contaminated material that is at the WAC values while constructing a home, and subsequently lives on the site. The scenario is modeled after the intruder construction scenario followed by the chronic intruder-agriculture scenario as depicted in NUREG-1757. The analysis conducted by NRC staff yielded doses less than the Part 20 dose limit of 1.0 mSv/yr (100 mrem/yr) to members of the public.

4. HEALTH PHYSICS EVALUATION

4.1. Need for Additional Radiological Characterization

The staff determined that additional information was needed on the characterization of waste materials. During the RAI process, the staff requested WEC to provide a description of the radiological sampling and survey measurement procedures and the quality control and assurance procedures that it would employ to ensure compliance with the USEI WAC. The staff also requested that WEC provide the methods and logistics to be employed to ensure radioactive waste homogeneity and the measures to be used to ensure that non-contaminated soil and materials are not blended or intentionally mixed with radioactive soil and debris to reduce the specific activity of the waste.

WEC provided a general overview of sampling and survey processes that would be used to ensure compliance with the USEI WAC in its December 29, 2009 response (ML100320540) to the RAI. The staff concluded that WEC's response was insufficient and that more specific details were needed on characterization activities and instrumentation before the staff could complete its radiological safety analysis of the proposed waste characterization and disposal activities. A teleconference between NRC and WEC staff was held on March 3, 2010, and WEC's RAI responses were discussed. An additional teleconference was held on April 19, 2010, in which WEC addressed the health physics points that required clarification. At that time, the staff requested that WEC provide a characterization plan demonstrating that WEC will be

able to adequately measure Tc-99 concentrations at the level (27 pCi/g) used for its dose analysis. Tc-99 is a hard-to-detect nuclide, and WEC had indicated that surrogate ratios of Tc-99:U-235 would be used to quantify Tc-99. The staff also stated that the plan should demonstrate how laboratory sampling would be performed throughout waste removal processes in order to re-establish and confirm the surrogate ratios.

4.2. WEC Initial Proposal for Additional Radiological Characterization

In a May 24, 2010, response (ML101450240), WEC provided information regarding the detection capabilities for radiological surveys and field measurements of soil during excavation and waste packaging. Inferred U-234 and Tc-99 values were presented based upon gamma instrumentation measuring U-235. Surrogate ratios for U-234 and Tc-99 were based on the WEC report titled "Derivation of Surrogates and Scaling Factors for Hard-To-Detect Radionuclides" (ML092870492). The U-234:U-235 ratio was based on observations of the enrichment in a large number of characterization samples, assumptions regarding the consistency of the enrichment shown by the characterization data, and on published values for the enrichment (based on isotopic ratios). Surrogate ratios for Tc-99:U-235 were developed for three specific areas: the Technetium Soil Area (TSA), the Burial Pit Area (BPA), and the Plant Soil Area (PSA). Within each area, additional subsets of ratios were developed for the following soil strata: Surface Soil (0 to 15 cm), Root Stratum Subsurface (15 cm to 1.5 m), and Deep Subsurface (> 1.5 m). WEC indicated in the report that the laboratory instrument's associated Minimum Detectable Concentrations (MDC) were substituted when Tc-99 or U-235 results were below the lower limit of detection. In order to confirm WEC's proposed correlation of U-235 to Tc-99, the staff reviewed the number of laboratory samples that were below the detection limit. The results show that MDC values were substituted with the following frequencies: 6.74% for the Tc-99 values and 41.35% for the U-235 values in the TSA; 43.82% for the Tc-99 values and 28.09% for the U-235 values in the BPA; and 35.16% for the Tc-99 values and 32.42% for the U-235 values in the PSA. The staff concluded that these results clearly indicated that U-235 and Tc-99 were not co-located at the site. Consequently, the staff requested WEC to provide a plan for sampling Tc-99 directly.

The staff informed WEC about its additional concerns regarding the proposed sampling plan during September and October 2010 conference calls. One concern in particular was a large variability (± 1447 pCi/g) noted within the Tc-99 characterization data. The staff wanted assurance that WEC would have in place continued quality assurance/quality control (QA/QC) checks to confirm assumptions about the distribution of the data (e.g., assumption of normality, assumed standard deviation).

4.3. WEC's Final Proposal for Additional Radiological Characterization

In Attachment 1 to its February 18, 2011 submittal (ML110530153), WEC provided a "Technical Basis for Characterization of Decommissioning Soils Waste to be Sent to U.S. Ecology Idaho, Inc." Several of the staff concerns were addressed in Section 3.0 of this attachment as follows:

- WEC will not assume normal distribution for Tc-99 data. Non-parametric statistics will be used for the final compliance calculation and the inventory check calculations during shipping.

- To address variability, the Visual Sampling Plan (VSP) was used to determine the number of samples required to ensure that the final inventory compliance calculation is made with sufficient confidence.
- WEC acknowledges that the surrogate ratios for Tc-99/U-235 vary substantially across the site. Consequently, only laboratory analyses for Tc-99 will be used to calculate ⁹⁹Tc inventories.
- To ensure control of rail shipments prior to dispatch, WEC will perform a number of in-process data checks to ensure all applicable inventory and disposal site WAC limits are met prior to shipment.
- A comprehensive QA/QC program will be in place.
- WEC will have contingency plans in place that are tied to specific action levels to ensure that unexpected conditions are identified.

4.4. Sampling Plan

In order to specifically address the staff's variability concerns, WEC committed in Section 5.0 of the Technical Basis document (Attachment 1 to its February 8, 2011 submittal) to independently characterize areas of elevated Tc-99 concentration located in the vicinity of the evaporation ponds. Material in the vicinity of the two most elevated Tc-99 results (EP-08-00-SL and EP-10-00-SL) will be held prior to shipment to USEI. WEC will perform additional characterization on this material prior to determining its ultimate disposition. However, the 1 Ci and 1.6 Ci Tc-99 limits must be met, regardless of whether this material is disposed of at USEI or at an NRC-licensed facility. To further address the staff's variability concerns and ensure compliance with these limits, WEC committed to using a statistically based sample plan to demonstrate that the weighted average mean concentration of Tc-99 in waste material disposed during the duration of the project is well known.

The proposed sampling approach, as described in Section 6.0 of the Technical Basis document, was initially based on the Visual Sample Plan (VSP) module for calculating a one-sided confidence interval for the population mean using simple random sampling. WEC generated a sampling requirement of 704 samples, based on a Tc-99 standard deviation of 225 pCi/g (the deviation associated with the characterization data excluding the two elevated evaporation pond results) and a confidence interval width of 14 pCi/g (equivalent to 1 mrem of increased dose). Since this is an a priori estimation of sample size, the standard deviation could fluctuate as excavation proceeds and could directly change the required number of samples. As a bounding measure, the total Tc-99 inventory will be monitored throughout excavation, and prior to each shipment the running mean and UCL₍₉₅₎ (Upper Confidence Level at 95%) of the inventory will be calculated (as Tc-99 analysis results are reported from an off-site laboratory). The shipment will not be made if either the 1 Ci or the 1.6 Ci limit is exceeded. In practice, WEC's plan for the number of samples is one per 15- 20 yd³ of waste material, whereas 704 samples would be approximately one per 42 yd³ (based on the projected volume to be shipped of 22,809 m³).

A material handling summary was provided in Section 7.0 of the Technical Basis document. The material handling and shipping limit decision points were provided in Figure 1 (Waste Handling Flowchart) of this document. Details of how materials from nuclear criticality safety (NCS) control areas would be handled were provided along with handling requirements for non-NCS control area materials. In either case, WEC indicated that material destined for USEI will

be weighed prior to transportation to a Waste Holding Area (WHA). At the WHA, one composite sample will be taken from each dumped truckload (approximately 17 m³) from four randomly selected points. Truckloads will be added to the WHA until enough material for 5 rail cars is accumulated. The composite samples will be analyzed for all radionuclides of concern and the data will be added to the overall data set. A new mean and UCL₍₉₅₎ will be calculated, and compared to the limits, before shipment. The UCL₍₉₅₎ will be calculated using the Chebyshev inequality based UCL using the sample mean and the standard deviation. This method does not assume normality, so this represents an acceptable non-parametric analysis.

4.4.1. Quality Assurance

Quality Assurance requirements are described in Section 8.0 of the Technical Basis document. WEC intends to implement field duplicate samples, field blanks, and laboratory control samples throughout the excavation process. Field duplicates will be collected at a frequency of 1 per 20 samples, and the results will be evaluated to determine the relative difference or relative percent difference between two data sets. Guidance from the Multi-Agency Radiological Laboratory Analytical Protocols Manual (MARLAP) will be used to compare these results to pre-determined warning and control limits. Field blanks will be collected at a frequency of 1 per 100 samples, and these results will be used to evaluate bias. Laboratory control samples, matrix spikes (if applicable), and replicate counts will be performed at a frequency of 1 per 20 samples in order to assess overall laboratory performance.

4.4.2. WEC Compliance Calculations and Contingency Plans

The compliance calculations were detailed in Section 9.0 of the Technical Basis document. In that document, WEC stated: "Compliance with the Tc-99 inventory limit (1 Ci) and UCL_(0.95) will be determined prior to each shipment of material and will comprise a 'running inventory.' As each stockpile of material is generated, analytical data from that stockpile will be pooled with the data from all previous stockpile samples to calculate a mean concentration and a 95% Chebyshev UCL on the mean. These two values will be multiplied by the sum of the total mass of material already shipped and the mass of the current stockpile. Once this is done, the two values (representing the mean Tc-99 inventory and UCL_(0.95) of the mean Tc-99 inventory) will be compared to the compliance limits (1 and 1.6 for the mean and UCL_(0.95) of the mean) to determine if the stockpile may be shipped."

Pre-shipment contingency plans were also provided in the Technical Basis document and are shown in Table 4-1 below:

Table 4-1. Pre-Shipment Contingency Plans Proposed by WEC

Parameter	Action Level	How Monitored	Actions
Total Quantity of Tc-99 shipped to USEI (mean)	>1 Ci	Running total activity (both shipped and pending shipment), based on laboratory sample results prior to shipment	<ul style="list-style-type: none"> • Reanalyze composite sample and/or analyze individual aliquots used to create the composite sample; • Resample stockpile and re-evaluate;^c • Ship material to alternate facility.
95% Upper Confidence Level of the mean Tc-99 shipped to USEI (UCL(0.95)).	>1.6 Ci	Running confidence interval (both shipped and pending shipment) based on laboratory sample data prior to shipment	<ul style="list-style-type: none"> • Reanalyze composite sample and/or analyze individual aliquots used to create the composite sample; • Resample stockpile and re-evaluate;^c • Ship material to alternate facility.
Total activity contribution from all radionuclides within individual railcar	>3000 pCi/g > 40 μ R/hr ^a	Laboratory sample results for stockpile evaluated at 95% UCL prior to shipment Gamma radiation levels on railcars prior to shipment.	<ul style="list-style-type: none"> • Analyze additional aliquot of composite sample; • Unload railcar (at HDP) and re-load with material containing lower concentration (either blended or alternate material from onsite waste stream);^c • Ship material to alternate facility.
Unexpected Tc-99 results for stockpile samples	>99 th percentile of the site wide dataset (599 pCi/g) ^b	Laboratory sample results for stockpile evaluated prior to shipment	<ul style="list-style-type: none"> • Analyze additional aliquot of composite sample; • Resample stockpile and re-evaluate;^c • Blend with less contaminated material, resample stockpile and re-evaluate; • Ship material to alternate facility.
Maximum average concentration of Ra-226 and Th-232 within individual railcar	Ra-226 >13 pCi/g Th-232 >16 pCi/g	Laboratory sample results for each railcar evaluated prior to shipment	<ul style="list-style-type: none"> • Analyze additional aliquot of composite sample; • Resample stockpile and re-evaluate;^c • Blend with less contaminated material, resample stockpile and re-evaluate; • Ship material to alternate facility.

^a Based on analysis previously transmitted in HEM-10-46 (May 24, 2010 Submittal - ML101450240)

^b Value shown is the 99th percentile of the pooled site wide Tc-99 dataset with EP-08-00-SL and EP-10-00-SL excluded using Microsoft[®] Excel[®] spreadsheet software.

^c Resampling of material will generally occur after down blending of stockpile material. When such sampling is performed, the new sample dataset will replace the initial data for the purpose of subsequent calculations. If re-sampling is performed without down blending (which would be the case if the material was sampled insitu railcars) then, the additional samples will be used to augment the initial dataset.

4.4.3. Measurement Instrumentation

The Waste Characterization Plan describing sampling implementation during excavation was provided in Appendix A to the Technical Basis document. Details on instrumentation to be used were also provided in this Appendix. For materials originating in areas that require NCS controls, the surface of the material will be scanned for gamma radiation (using a sodium iodide detector) within the excavation and prior to each successive lift. Waste trucks that originate

from the burial pit area will also pass through a box counter prior to proceeding to the WHA. Four sample aliquots will be randomly taken from each 15-20 yd³ of material as it is placed in the stockpile area. Samples collected for laboratory analysis will be analyzed via gamma spectroscopy for Ra-226, Th-232, U-235, and U-238; and Tc-99 samples will be sent for off-site analysis. The levels of U-234 will be inferred based upon enrichment calculations. For materials that originate in non-NCS controlled areas, sodium iodide scanning will take place once the materials are received at the WHA, and four aliquots per 15-20 yd³ will be taken for laboratory analysis. WEC provided, in Appendix A, the minimum detectable activities (MDA) for the sodium iodide, box counter, and gamma spectroscopy systems. The laboratory detection capabilities of the key nuclides were also provided as follows: Ra-226 (3 pCi/g via High-Purity Germanium Detector [HPGe]), Th-232 (3 pCi/g via HPGe), and Tc-99 (1 pCi/g sent for off-site analysis). The on-site detection capabilities are within a reasonable magnitude of the Ra-226 and Th-232 limits, which are 13 pCi/g and 16 pCi/g, respectively. For uranium, WEC would hypothetically be permitted to send up to the WAC (3,000 pCi/g), and the MDA values provided for uranium are within a reasonable fraction of the WAC. Tables 4-2 through 4-6 provide details regarding the sodium iodide, box counter, and gamma spectroscopy MDAs.

Table 4-2 Scan MDC for Total Uranium Based on Degree of Enrichment

Enrichment (wt% U-235)	Total U ^a (pCi/g)
3	65
20	77
50	95
75	109

^a MDC values assume a surveyor efficiency of 50

Table 4-3 Estimated Gamma Emitter MDA Values for Box Counter System

Count Time	Ra-226 MDA ^a	Th-232 MDA	U-235 MDA	U-238 MDA
(minutes)	(pCi/g)	(pCi/g)	(pCi/g)	(pCi/g)
10	10/1	1	1	10

^aMDA values shown for both direct analysis using 186 keV peak (higher value) and indirect analysis using daughters (i.e., Bi-214/Pb-214).

Table 4-4 Estimated Inferred U-234 MDA Values for Box Counter System

Count Time	Natural U	5 wt% U-235	20 wt% U-235	95 wt% U-235
(minutes)	(pCi/g)	(pCi/g)	(pCi/g)	(pCi/g)
10	21	18	20	32

Table 4-5 Estimated Gamma Emitter MDA Values for Gamma Spectroscopy System

Count Time	Ra-226 MDA	Th-232 MDA	U-235 MDA	U-238 MDA
(minutes)	(pCi/g)	(pCi/g)	(pCi/g)	(pCi/g)
Variable	3	3	5	20

Table 4-6 Estimated Inferred U-234 MDA Values for Gamma Spectroscopy System

Count Time	Natural U	5 wt% U-235	20 wt% U-235	95 wt% U-235
(minutes)	(pCi/g)	(pCi/g)	(pCi/g)	(pCi/g)
Variable	105	91	100	159

4.5. Health Physics Conclusions

The staff has determined that the proposed statistical evaluation, sampling plan, QA/QC program, and contingency plans are acceptable and allowed the licensee to demonstrate that its proposed disposal will result in a dose to individual members of the public that does not exceed a few millirem per year.

5. NUCLEAR CRITICALITY SAFETY

This section of the SER addresses the nuclear criticality safety aspects of WEC's §20.2002 alternate disposal request.

5.1. WEC Assessment

WEC performed a nuclear criticality safety assessment to demonstrate that the risk of criticality is not credible based on the process conditions at the Hematite site, very low concentrations of uranium in the waste, and disposal activities at the USEI site. The scope of the assessment was limited to the safe handling and disposal of the solid wastes at USEI based on the following low level waste streams being shipped from the Hematite site:

1. Exhumed burial waste from the Hematite burial pits and contaminated soils and backfill material associated with the Hematite burial pits and other remediation areas at the Hematite site; and
2. Solids recovered from the Water Treatment System (i.e., used filter media, IX beds, solids in the holding tanks, etc.)

WEC expects that under normal conditions, Hematite decommissioning wastes would contain trace quantities of radionuclides, or a very low presence of fissile nuclides. WEC indicated that potential items or regions containing fissile material would be identified through in-situ radiological survey and visual inspection of the area to be exhumed prior to removal of buried waste. The in-situ radiological surveys would identify any item or region of soil with a fissile concentration exceeding 1 gram U-235 in any contiguous 10 liter volume to assure that any items with non-trivial levels of U-235 contamination would be identified. Two independent

measurements would be performed. Items or regions containing fissile material, bulky objects, and items that resemble intact containers or metallic items would be removed and a more detailed characterization performed.

WEC also determined that ground water seepage and rainwater would be expected to intrude into the open excavations during the recovery of the contaminated solid wastes from the burial pits. The water would be removed and treated to remove entrained and soluble contaminants prior to release to the site water outfall. The treatment processes would result in the following solid wastes: sediments in the tanks, filter bags, filter media, and treatment media. Wet sediments from the bottom of the waste treatment system holding tanks would be removed using a drum vacuum. Each loaded 55-gallon drum would be transferred to a waste evaluation area and/or material assay area to determine the radiological content. The assay results must demonstrate that the drums do not contain a U-235 concentration greater than 1 gram U-235/10 liters in order to be shipped to USEI for burial. Two independent measurements would be performed. Filtration and treatment media would require removal and replacement, and would also be evaluated using the 1 gram U-235/10 liter criteria.

WEC postulated abnormal conditions concerning the potential for an increase in uranium mass and/or concentration levels on receipt, or following placement within the disposal system. WEC credits administrative controls to ensure that there is not a potential for the following: ship waste with an unanalyzed concentration to the USEI site, form a maximum safe mass at the USEI site, or ship high concentration wastes that may normally require spacing.

5.2. Staff Assessment

The staff has conducted considerable research on the technical basis for criticality safety at low-level waste facilities. (The current regulatory approach to ensure criticality safety at low-level waste facilities relies on limiting the average concentration of U-235 in the waste.) An NRC-sponsored study (NUREG/CR-6505, Vol.1, "The Potential for Criticality Following Disposal of Uranium at Low-Level Waste Facilities") evaluated the potential for uranium to be concentrated by hydrogeochemical processes to permit nuclear criticality. The NUREG identified a combination of variables that could lead to or support criticality in a waste matrix (or "soil"). The variables included the composition of the soil (i.e., concrete debris, iron scrap, etc.), the enrichments, the density of the soil, the degree of neutron moderation, and the degree of neutron reflection. The NUREG used silicon dioxide (SiO_2) and water to represent a waste matrix. Nuclear criticality evaluations were performed for finite-media and infinite media assuming various densities of U-235 and water for "SiO₂ soil" and "nominal soil." Silicon oxide (SiO) soil was used as the most conservative media because pure SiO is the least likely soil composition to absorb neutrons, thereby enhancing the potential for criticality. Using data from NUREG/CR-6505, the minimum critical infinite sea concentration for a fictitious bounding medium consisting of silicon dioxide and U-235 (highly enriched) was 1.4 grams U-235/ liters (39.6 grams U-235/ft³). WEC indicated that it has controls in place to ensure that waste shipped from the Hematite site to the USEI site will not exceed an average concentration of 1 gram U-235/10L, which is below the minimum critical infinite sea concentration.

5.3. Staff Conclusion:

Based on the information in the WEC submittal, the staff determined that a criticality is not credible due to the low concentrations of uranium in the waste while in the gondola railcar or at the USEI site. The staff further determined that the controls in place at the Hematite site provide reasonable assurance that an inadvertent criticality will not occur.

6. MATERIAL CONTROL AND ACCOUNTABILITY

This section of the SER addresses the material control and accountability (MC&A) aspects of WEC's §20.2002 alternate disposal request. The staff's assessment of these aspects involve three Hematite decommissioning actions: (1) shipment of the waste material via gondola cars to USEI; (2) unloading of the waste material from the gondola cars on to trucks for disposal at the USEI burial cell and (3) disposal of the waste material in the burial cells.

WEC Hematite maintains a Material Control and Accounting program in accordance with the NRC-approved Fundamental Nuclear Material Control Plan (FNMCP) per 10 CFR Part 74, Material Control and Accounting of Special Nuclear Material. The FNMCP contains the reporting requirements of 10 CFR §74.15 associated with DOE/NRC Form 741 for the WEC Hematite facility.

In Section 9 of WEC's May 21, 2009, Safety Assessment, WEC indicated that 10 CFR §70.42 (d)(2) requires a written certification by the transferee that the recipient is authorized by license or registration certificate to receive the type, form, and quantity of SNM to be transferred, specifying the license or registration certificate number, issuing agency, and expiration date. WEC stated further that since USEI would be exempted from the 10 CFR §70.3 requirement for a NRC licensee to possess SNM, the §70.42 requirement would not apply to it. WEC proposed that the permit issued to USEI by the State of Idaho serve as an alternative written certification. DOE/NRC Form 741, *Nuclear Material Transaction Report*, would be used by WEC, as it has in the past, to document all transfers of SNM to the disposal facility. A radioactive materials manifest would accompany each shipment, would be signed by USEI upon receipt, and would provide a further confirmation that proper accountability for the material was maintained. 10 CFR §70.42 (d)(2) identifies the following as an acceptable method for satisfying the verification requirement in §70.42(c): a written certification by the transferee that the recipient is authorized by license or registration certificate to receive the type, form, and quantity of SNM to be transferred, specifying the license or registration certificate number, issuing agency, and expiration date.

WEC's proposal raised an issue regarding the Nuclear Materials Management and Safeguards System (NMMSS) record keeping requirements associated with 10 CFR Part 74. This regulation requires that the SNM be accounted for at all times, so the material's whereabouts are always known. Consequently, with respect to WEC's Hematite's alternate disposal request, the recipient of the SNM is required to have a NMMSS account before any waste containing SNM may be received and disposed. Therefore, in order for USEI to accept shipment of the Hematite alternate disposal material, it must possess an NMMSS account. Therefore, a condition for approval of the §20.2002 request will be that USEI have an NMMSS account. A request for an NMMSS account may be made to Mr. Brian Horn (brian.horn@nrc.gov),

International Safeguard Analyst, U.S. Nuclear Regulatory Commission. Such a request should include the name of the entity requesting the account, the location of the entity, and the names, addresses, telephone numbers, and e-mail addresses for 1-2 points of contact for the NMMSS account.

With the approval of the §20.2002 alternate disposal request involving the USEI facility in Idaho, the NRC would approve of an NMMSS account being assigned to the USEI facility. Consequently, WEC would continue to use DOE/NRC Form 741, Nuclear Material Transaction Report, as it has in the past, to document all transfers of 1 gram or more of SNM, and USEI facility would report all SNM receipts, including SNM contained in waste, to NMMSS. Both facilities will report the SNM activity to NMMSS using the DOE/NRC Form-741 procedure. Once all of the WEC material is received and disposed of below ground at the USEI facility, the disposal facility may request that its NMMSS account be de-activated.

7. PHYSICAL SECURITY

This section of the SER addresses the physical security aspects of WEC's §20.2002 alternate disposal request. The staff's assessment of these aspects involve three Hematite decommissioning actions: (1) shipment of the waste material via gondola cars to USEI; (2) unloading of the waste material from the gondola cars on to trucks for disposal at the USEI burial cell and (3) disposal of the waste material in the burial cells.

7.1. Transportation Security

WEC will ship the waste to USEI in gondola railcars. The contents of each gondola railcar will be entirely enclosed in form-fitting, sift-proof, and closable wrappers meeting U.S. Department of Transportation (DOT) Industrial Type-I Package (IP-1) requirements. The IP-1 package precludes dispersal of waste to the air or loss of material during transport. WEC is responsible for the safe and secure transport of the material in accordance with the provisions of the Transportation, Physical Security and Fundamental Nuclear Material Control Plans. The custody of the SNM-bearing waste remains WEC's until the shipment arrives on-site in Idaho and USEI accepts custody of the waste.

In WEC's Safety Assessment in support of its May 21, 2009, submittal, it indicated that the expected concentration of U-235 was 5.5 pCi/g and the enrichment for the estimated 22,809 cubic meters of soil and debris was expected to be at enrichment levels averaging below 10%. In WEC's December 29, 2009, response (ML100320540) to the staff's RAI, WEC indicated that no HEU material would be shipped to USEI. However, material of intermediate enrichment (greater or equal to 10%, but less than 20%, enrichment) could be shipped to USEI.

The staff performed the following calculations to assess the need for transportation security. A single gondola railcar will contain approximately 127 tons of soil and debris. As noted in previous sections of this SER, USEI's WAC is 3000 pCi/g for all radionuclides. From Table 1 of Appendix A of WEC's March 31, 2010, submittal (Table 1), the average enrichment is 3.8% U-235. At this enrichment, and assuming that the SNM concentration in the waste is at the USEI WAC (no other radioactive elements are present), a single railcar could contain approximately 183 kg total U and 6.9 kg U-235. This value represents the maximum amount of U-235 (at the expected enrichment) that can be shipped to USEI in a single railcar. From Table

1, the expected average U-235 concentration is 32.2 pCi/g. At this concentration, a single railcar would contain approximately 1.7Kg U-235. If data from WEC's Safety Assessment in support of its May 21, 2009, submittal is used (5.5 pCi/g U-235), a single railcar would contain approximately 0.3 Kg U-235.

Any shipment containing 10 Kg or more of low enriched uranium (LEU) material would result in a situation where the SNM is considered to have low strategic significance (LSS). From the above calculations, WEC would have to ship multiple railcars, at one time, to meet the definition of LSS. As long as the amount of SNM being shipped is less than 10 Kg of LEU, no special security would be required. For this situation, WEC would be required to have a physical security plan but the plan would not have to be submitted to the NRC for approval. However, as stated in the Physical Security Plan (PSP), WEC has committed to implement the transportation security requirements in 10 CFR §73.67(g) for the transport of SNM of LSS.

If WEC shipped a railcar or multiple railcars containing 10 Kg or more of intermediate-enriched uranium, it would be required to have a Transportation Security Plan to address Category II SNM shipments of moderate strategic significance in accordance with 10 CFR §73.67(e). WEC's July 28, 2011, Physical Security Plan (ML11214A106) contains a Transportation Security Plan that addresses the transportation security of Category III SNM shipments of LSS in accordance with 10 CFR §73.67(g). It does not address railcar shipments containing 10 kg or more of intermediate-enriched uranium. A Transportation Security Plan for shipments of Category II SNM of moderate strategic significance is required by 10 CFR 73.67(c) and must be submitted to the NRC for review and approval. SNM enriched to 10% or more but less than 20% in the amount of more than 1Kg but less than 10Kg is considered to be of low strategic significance or Category III SNM. SNM enriched to 10% or more but less than 20% in amounts less than 1Kg are less than Category III and thus are not covered under 10 CFR §73.67(g). However, since WEC committed to protect all shipments in accordance the transportation security requirements in 10 CFR §73.67(g), there would be no security concerns associated with the shipment of this type and quantity of material.

The staff's review concluded that any gondola railcar(s) shipment involving 10 kg or more of LEU would be transported in accordance with the security requirements of 10 CFR §73.67(g) for the transport of SNM of LSS. If a shipment involved less than 1 kg of LEU, while WEC would be required to have a Physical Security Plan, no special transportation security would be required. WEC has committed to the transportation security requirements in 10 CFR §73.67(g) for the transport of SNM having LSS. Therefore, the staff has concluded that the appropriate security exists for the transportation of LEU material from Hematite.

The staff's review concluded that any railcar shipment involving intermediate-enriched uranium in an amount of 10 kg or greater would constitute a shipment of Category II SNM (of moderate strategic significance). A Transportation Security Plan would be required by 10 CFR 73.67(c) and submittal of the plan to the NRC for review and approval would be required. If the amount was less than 1 kg, there would be no physical security concerns associated with the shipment of this type and quantity of material since WEC committed to protect all shipments in accordance with the transportation security requirements in 10 CFR §73.67(g). Therefore, as noted above, the staff has concluded that the appropriate security exists for the transportation of less than 1 kg of intermediate-enriched uranium material from Hematite but for quantities 10 kg

or greater, a Category II Transportation Security Plan will be required and the plan will require NRC review and approval before implementation.

7.2. Security of SNM Prior to Waste Disposal

At the USEI site, SNM-bearing waste is stored in gondola railcars, unloaded from the railcars in a controlled environment to trucks for transport to the burial cell, possibly treated in a controlled environment for volatile organic compounds, and then disposed of in the USEI cell. Waste consignments are routinely emplaced for disposal within a few days of receipt of the waste.

Because of their robust design features and the use of tamper-indicating devices, railcars effectively represent individual SNM-containing structures (areas). A single railcar contains approximately 127 tons of soil and debris. Assuming that the waste contains SNM at the USEI WAC, a single railcar would contain approximately 6.9 kg U-235. However, as noted above, it is extremely unlikely that any gondola car would contain only uranium waste at the USEI WAC. At the average concentration of 32.2 pCi/g noted above or at the anticipated concentration of 5.5 pCi/g of U-235 as presented in Table 1 of WEC's Safety Assessment for the §20.2002 disposal request, a railcar would contain approximately 1.7 kg and 0.3 kg, respectively. As noted in the previous section, if the railcar contained 10 Kg or more of LEU, the material in the railcar would be considered as SNM of LLS.

Under the NRC's SNM categorization approach, the amount of recoverable SNM contained in a single railcar is no greater than a Category III SNM quantity (SNM of LSS). Considering credible SNM diversion scenarios, the storage and processing of SNM waste at the USEI site prior to disposal could be considered as no greater than Category III SNM activities. Due to the difficulty, time, and necessary equipment required to separate 10 kg of SNM from 127 tons of waste and due to the additional processing that would be required to make the SNM useful in either an improvised nuclear device (IND) or a radiological dispersal device (RDD), this material would have to be considered as highly unattractive to adversaries. Therefore, the staff has concluded that no additional security steps need to be taken at USEI during the period in which the waste is handled in preparation for burial and during burial.

7.3. SNM Security after Waste Disposal

The difficulty of recovering SNM from waste after disposal would increase considerably compared to the recovering the material prior to disposal. The difficulty would be precipitated by the following. It is anticipated that the Hematite waste would be buried over an area covering 30 acres. The Hematite waste will be intermixed with waste from other sources and those sources will not contain SNM. The cell in which the Hematite waste will be buried will have a soil cover which will vary in depth from 2 feet at the crown to 20 feet at the side slopes. The burial cell has a depth of approximately 49 feet, which would make it more troublesome. Potential adversaries would now have to excavate the waste, identify SNM-bearing materials, and separate these materials from soil and non-SNM-bearing debris. The additional processing that would be required to make the SNM useful in an IND or RDD would make this material highly unattractive to adversaries. The existing industrial security measures at the USEI site are adequate to address credible SNM diversion scenarios. Based upon the above discussion, staff found that there would not be a security issue with the material once it was buried at USEI.

7.4. Summary

The staff has assessed the physical security aspects associated with the shipment of waste material containing SNM in soil and debris. Time periods assessed were from shipment from the Hematite site to receipt at the USEI facility, from offloading of the material from the gondola cars until burial in the USEI cell, and after burial in the cell. With respect to transportation of the material from Hematite to the USEI site, the staff concluded that security aspects are appropriately covered in all cases except for the shipment of 1 kg or more of intermediate-enriched uranium. For this case, a revision to the Physical Security Plan would be required and so would review and approval by the NRC. With respect to the offloading of the material, its handling while at the USEI site, burial and after burial, the staff concluded that the material would have to be considered highly unattractive to adversaries due to the difficulty, time, and necessary equipment required to separate 10 Kg of SNM from the tons of waste and due to the additional processing that would be required to make the SNM useful in either an IND or RDD.

8. POTENTIAL FOR RECONCENTRATION

The staff assessed the potential for reconcentration in the leachate system at the USEI facility given the half lives of the SNM and the impact of leachate control system.

In 2008, USEI's permit was modified to include receipt of specified quantities of SNM that were exempt from the NRC regulations. The Idaho Department of Environmental Quality (IDEQ) granted this exemption after a detailed safety evaluation and criticality analysis was performed.

The potential for the generation of leachate is minimized by the site's waste acceptance requirement that the waste contain no free liquids. Further reducing the potential for leachate generation is the site's location in a desert environment that averages approximately 7.3 inches of precipitation per year with an evaporation rate of approximately 42 inches per year.

The potential to generate leachate is further reduced by the facility's design to completely encapsulate the waste in a low permeability (1×10^{-7} cm/sec) cover system. Requirements for the construction of a waste cell include a base layer of compacted clay three-feet thick overlain by a composite liner with a sump to collect any leachate that might be generated. The composite liner is overlain by a 30-inch soil layer as a protection barrier for the liner. Waste placed in the cell is compacted to minimize the potential for future subsidence and when the cell is full is overlain by a low permeability multi-layer cap 11.8 feet thick that includes nine feet of non-radiological material.

As a result of the above design features and the above noted site conditions, the staff has concluded that reconcentration in the leachate system should not be an issue with respect to the disposal of the SNM at USEI.

If the USEI Idaho site were compared to the NRC-licensed low-level radioactive waste disposal facility operated by Energy Solutions at Clive, Utah, one would find that the two facilities share similar site and design characteristics. The Clive facility is located in a desert environment similar to that of USEI's Idaho facility. Precipitation at the Clive facility averages approximately 8.6 inches of precipitation per year with an average evaporation rate of 59 inches per year. The Clive facility is allowed to accept SNM with concentrations up to 1,900 pCi/g of waste. As noted

previously, the USEI facility has an overall WAC of up to 3,000 pCi/g. There is no specified limit on SNM. However, as noted in Section 7 above, Table 1 of Appendix A to WEC's March 31, 2010, RAI response indicates a weighted mean value of U-235 in soil of 32.2 pCi/g. The maximum U-235 activity in any soil sample was shown in Table 3 of Appendix and found to be 1940 pCi/g.

The cell and cover design criteria of the Clive facility are comparable to the design criteria for the USEI facility. The Clive cell is underlain by a two-foot clay layer with a 1×10^{-6} cm/sec permeability with a leachate collection system. The cap consists of a 24-inch radon barrier, a six-inch filter zone to move water away from the buried waste material, a 12-inch silt loam sacrificial layer, and a second filter zone that is six inches thick. The final layer consists of an 18-inch thick layer of riprap rock. The NRC approved the SNM limits for Clive after performing a detailed safety evaluation and criticality analysis.

9. CONCLUSIONS

On May 21, 2009, WEC requested that the NRC approve alternate disposal, in accordance with 10 CFR §20.2002, of specified low-activity radioactive materials from the HDP. Granting this request would allow WEC to send up to approximately 22,809 m³ (or approximately 50,000 tons) of soil and debris with low concentrations of both SNM and byproduct material contaminants to USEI RCRA Subtitle C disposal facility near Grand View, Idaho.

Activities and potential doses associated with transportation, waste handling and disposal have been evaluated as a part of the review of this 10 CFR §20.2002 application. The staff has determined that WEC has provided an adequate description of the waste containing licensed material to be disposed of, including the physical and chemical properties important to risk evaluation, and the proposed manner and conditions of waste disposal.

The staff has determined that the proposed statistical evaluation, sampling plan, QA/QC program, and contingency plans are acceptable and allowed the licensee to demonstrate that its proposed disposal will not result in a dose to individual members of the public exceeding a few millirem per year.

Independent review of the post-closure and intruder scenarios using RESRAD estimated that the maximum projected dose per year over a period of 1,000 years is within "a few millirem". A conservative bounding analysis conducted by the staff yielded doses less than the Part 20 dose limit of 1.0 mSv/yr (100 mrem/yr) to members of the public. The projected doses to individual USEI workers have been conservatively estimated and demonstrate that the proposed disposal will not result in a dose to members of the public exceeding a few millirem per year.

In addition, because this 10 CFR §20.2002 application involves SNM, nuclear criticality safety, material control and accounting, and physical security assessments were performed. Only one issue was identified. If WEC wishes to ship waste to USEI containing 1 kg or more of intermediate-enriched uranium, then a revision to the Hematite Physical Security Plan will be required, which will require NRC review and approval.

In conclusion, there are no concerns that this request will greatly impact the annual cumulative dose from all exempted and naturally occurring radioactive material at the USEI disposal facility as long as actual source term concentrations reflect those applied in this assessment.

Further, in accordance with the provisions of 10 CFR §30.11 and 10 CFR §70.17, the NRC may, upon application by an interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in those parts of 10 CFR as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Based on the above analyses, the staff concludes that (1) this material authorized for disposal poses no danger to public health and safety; (2) the authorized disposal does not involve information or activities that could potentially impact the common defense and security of the United States; and (3) it is in the public interest to dispose of wastes in a controlled environment, such as that provided by the U.S. Ecology Idaho facility located in Grand View, ID. Therefore, to the extent that the material authorized for disposal in this §20.2002 authorization is otherwise licensable, the staff concludes that the site authorized for disposal is exempt from NRC licensing requirements in 10 CFR §30.3 and §70.3.

SAFETY EVALUATION REPORT

**REQUEST FOR ALTERNATE DISPOSAL APPROVAL AND
EXEMPTIONS FOR SPECIFIC HEMATITE DECOMMISSIONING
PROJECT WASTE AT
US ECOLOGY'S IDAHO FACILITY**

April 11, 2013

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1. INTRODUCTION

1.1. Westinghouse Request

By letter dated January 16, 2012 (ADAMS Accession Nos. ML12017A188, ML12017A189, and ML12017A190), Westinghouse Electric Company, LLC (WEC) requested that the U.S. Nuclear Regulatory Commission (NRC) approve an amendment to its Hematite license (SNM-33) to permit alternate disposal of licensed material in accordance with Title 10 of the Code of Federal Regulations (10 CFR) §20.2002. (January 16, 2012 request) The disposal would involve low-activity radioactive materials generated by the Hematite Decommissioning Project (HDP) containing source, byproduct, and special nuclear material (SNM). The January 16, 2012 request includes a request for an exemption from NRC licensing requirements in 10 CFR §30.3 and 10 CFR §70.3 for byproduct material and SNM, respectively. Granting these exemptions would allow these materials to be disposed of at the US Ecology Idaho, Inc. (USEI) facility, even though USEI is not an NRC licensee. On October 4, 2012, USEI requested that it be considered a party to WEC's January 16, 2012, alternate disposal request and exemption request (ADAMS Accession No. ML12313A014). WEC did not request, nor does it need, an exemption for its proposed disposal of source material because the quantities involved are "unimportant" and are exempt from licensing under 10 CFR §40.13(a). The 0.05 weight % referenced in 10 CFR §40.13(a) translates to approximately 339 pCi/g for natural uranium (including U-234, U-235, and U-238, but omitting consideration of decay products). Enclosure 1 to WEC's January 16, 2012 submittal shows in Section 5.1 that the average total activity concentration (sum of all nuclides and progeny and not just uranium) for this waste is approximately 110 pCi/g. Therefore, the 10 CFR §40.13(a) exemption is applicable here.

Granting the January 16, 2012 request would allow WEC to ship the HDP waste to USEI's Resource Conservation and Recovery Act (RCRA) Subtitle C disposal facility in Idaho.

The January 16, 2012 request follows a similar request submitted by WEC (HEM-09-52) on May 21, 2009 (ADAMS Accession No. ML091480071). That request was approved on October 27, 2011 as Hematite License Amendment No. 58 (ADAMS Accession Nos. ML111441087, ML112560105, and ML112560193).

Various types and quantities of SNM are discussed below. SER Section 5 (Criticality Safety) and Section 6 (Material Control and Accountability) discuss SNM in quantities of 1 g or more of U-235. SER Section 7 (Physical Security) pertains to SNM enriched in the U-235 isotope, in quantities of approximately 45 Kg of U-235.

1.2. USEI Facility

The USEI facility is a RCRA Subtitle C hazardous waste disposal facility permitted by the Idaho Department of Environmental Quality (IDEQ), and is not an NRC licensee. On October 4, 2012, USEI submitted a letter (ADAMS Accession No. ML12313A014) to the NRC stating that it had worked with WEC in the preparation and submittal of WEC's alternate disposal request and supporting documentation.

The USEI RCRA facility is located near Grand View, Idaho in the Owyhee Desert. The HDP material would be disposed in Cell 15, which has an area of 88,220 m² (21.7 acres) and a depth of 33.6 m. The most important natural site features that limit the transport of radioactive material are the low precipitation rate (i.e., 18.4 cm/y (7.4 in. per year)) and the long vertical distance to groundwater (i.e., 61-meter (203-ft) thick on average unsaturated zone below the disposal zone).

As is usual with a RCRA Subtitle C site, a number of engineered features are present to enhance confinement of contaminants over the long term. These features include an engineered cover, liners, and leachate monitoring systems. Operations at the site include a number of systems that minimize the potential for exposure of workers to any waste handled by the facility. These systems include a closed facility with filtered ventilation exhaust for transfer of incoming waste material from the shipping conveyance to trucks for transport to the cell, mechanized equipment for disposition of waste material in the cell, and the application of an asphaltic spray at the end of each day's operations. The site is permitted to receive non Atomic Energy Act material or exempted radioactive material that meets site permit requirements.

1.3. Overview of NRC Review

The NRC reviews §20.2002 requests from the standpoint of the safety implications of disposing of licensed material at disposal facilities that are not licensed by the NRC or an NRC Agreement State.

The NRC's review of a 10 CFR §20.2002 request for disposal of low-activity waste at a RCRA facility covers protection of individuals, inadvertent intruders, and the public. The period of performance is 1,000 years after the expected date of license termination of the facility, consistent with 10 CFR 20.1401 (the License Termination Rule in Subpart E of 10 CFR Part 20). While the 10 CFR Part 20 dose limit for individual members of the public is 1 mSv/yr (100 mrem/yr) (10 CFR §20.1301), the NRC's practice is to approve §20.2002 requests if calculations demonstrate that disposal would not result in a dose exceeding more than a few millirem per year

Because this 10 CFR §20.2002 disposal request includes SNM, the NRC's review must -- in addition to a dose limit analysis -- evaluate nuclear criticality safety, material control and accounting, and physical security issues. .

The potential exists that the waste material approved for disposal by Amendment 58 and the material approved for disposal in this SER will be available for shipment to USEI at the same time. Therefore, this SER will discuss the cumulative impact of the alternative disposal of material from both requests.

1.4. Additional Westinghouse Supporting Information

The NRC's review of WEC's January 16, 2012 request resulted in a need for WEC to supplement its request. On May 1, 2012, the NRC made a request for additional information

(RAI) (ADAMS Accession No. ML120890557). WEC provided responses to that request in letters dated June 19, 2012 (ADAMS Accession Nos. ML12173A427, ML12173A428, ML12173A430 and ML12173A431), July 24, 2012 (ADAMS Accession Nos. ML12209A200 and ML12209A201), and October 17, 2012 (ADAMS Accession No. ML12293A029).

2. BACKGROUND

The Hematite site was used for the manufacture of low-enriched, intermediate-enriched, and high-enriched materials during the period of 1956 through 1974. In 1974, the production of intermediate- and high-enriched material was discontinued and all associated materials and equipment were removed from the facility. From 1974 to cessation of manufacturing operations in 2001, the Hematite facility produced nuclear fuel assemblies for commercial nuclear power plants. In 2001, fuel manufacturing operations terminated and the facility license was amended to authorize only decommissioning operations.

Activities at the Hematite site generated a large volume of process wastes contaminated with uranium of varying enrichment. Based on historic documentation, 40 unlined pits were excavated and used for the disposal of contaminated materials generated by fuel fabrication processes at Hematite between 1965 and 1970. The May 2009 alternate disposal request and License Amendment 58 approval covers the disposal of material from these burial pits, other undocumented burial pits, and other soil associated with the remediation of the Hematite site. This January 16, 2012 request involves the disposal of source, byproduct, and special nuclear materials contained in building slabs, asphalt, soils, buried piping and miscellaneous equipment associated with the HDP. While the primary waste types covered by the May 2009 alternate disposal request were expected to be solid materials in the form of soils and associated debris (i.e., trash, empty bottles, floor tile, rags, drums, bottles, glass wool, lab glassware, and filters), the primary waste types covered by the January 2012 request are expected to be concrete, asphalt, piping, soil and miscellaneous equipment.

WEC plans to ship the material associated with the January 16, 2012 request to the USEI facility by rail if the material meets criteria established by WEC and approved by the NRC for this §20.2002 disposal request. Discrete quantities of highly enriched uranium (HEU) will not be shipped to the USEI facility. However, the proposed rail shipments may contain diffuse quantities of HEU spread throughout the waste materials, as discussed further in Section 6 below.

3. DOSE EVALUATION

This SER section evaluates WEC's description of the types of material it plans to ship and its potential to generate radiological dose to various members of the public. WEC supplied information on the source material and a description of the job functions which permitted them to evaluate different possible exposures for various members of the public. These scenarios included the doses to the transportation workers and USEI workers and the post-closure dose to the general public, and to an intruder. For §20.2002 reviews, all the scenarios treat exposed individuals as members of the public because the material is proposed to be sent to a facility that is not licensed by the NRC or an NRC Agreement State. Therefore, the NRC's occupational dose criteria do not apply to workers at USEI.

3.1. Types and Quantities of Material

WEC estimates the volume of the waste that will be a candidate for disposal at USEI associated with this request to be approximately 23,000 m³ at a waste density of 1.5 g/cm³ (i.e., approximately 38,700 tons). Since the dose assessment calculations assume this amount as a limit, 23,000 m³ will be an upper bound on the amount of waste that WEC is permitted to send to USEI under this request. License Amendment 58 had approved for disposal approximately 23,000 m³ at a waste density of 1.69 g/cm³ (i.e., approximately 50,000 tons). Therefore, the combined waste amount for both requests is approximately 46,000 m³. The waste covered by the January 2012 request consists of concrete/asphalt, piping, soil and miscellaneous equipment, and contains low concentrations of source, SNM, and byproduct material contaminants. WEC determined the radionuclides of concern based on studies in the Hematite Historical Site Assessment (ADAMS Accession Nos. ML092870417 and ML092870418). This is summarized in Chapter 4 of the Hematite Decommissioning Plan (ADAMS Accession No. ML092330136).

In Table 4-1 of Attachment 1 (HDP-TBD-WM-906) of Enclosure 1 of the January 16, 2012 Westinghouse submittal (ADAMS Accession No. ML12017A189), WEC presented the expected curie quantities to be shipped to USEI in a volume of approximately 23,000 m³ of waste. That information is presented in this SER as Table 3-1. The technical basis for each estimate is described in the following sections.

Table 3-1: Source Term Radionuclides and Expected Total Curie Amounts⁽¹⁾

Material	Shipped Volume (m ³)	U-234 (Ci) ⁽²⁾	U-235 (Ci) ⁽²⁾	U-238 (Ci) ⁽²⁾	Tc-99 (Ci) ⁽²⁾	Wt% U-235
Concrete/Asphalt	8,249	1.4E+00	6.3E-02	2.9E-01	4.0E-02	3.3
Piping	348	1.1E-01	3.9E-03	1.2E-02	2.6E-03	5.0
Misc. Equipment	39	3.0E-03	1.7E-04	5.4E-04	3.8E-05	4.5
Soil	14,212	6.2E-01	3.2E-02	1.4E-01	2.1E-01	3.4
Total Weighted Average	22,848	2.2	0.1	0.4	0.3	3.4

⁽¹⁾ Values in the table reflect a multiplier of 1.5 to account for uncertainty

⁽²⁾ Multiply Ci by 3.7x10¹⁰ to obtain Bq

WEC based the average expected concentration on the totals in Table 3 1, while for the average cell concentration WEC assumed that the shipped materials will be evenly distributed over 725,000 tons of total waste anticipated to be sent to USEI from various waste generators. In addition, in response to RAI No. CH-22, on page 70 of 167 of HEM-12-67, (ADAMS Accession No. ML121740265) WEC assigns a bounding concentration for each radionuclide corresponding to the Waste Acceptance Criteria (WAC). The bounding concentration was based on 100 percent of the activity being from 3,000 pCi/g of total uranium, with the isotopic composition based on existing sample data. Tc-99 was not considered in the WAC concentration since WAC concentrations were used for the intruder scenarios and Tc-99 was not an important radionuclide for the intruder scenarios. When summed, the WAC concentrations (assuming the progeny radionuclides are at equilibrium) equal the overall WAC of 3,000 pCi/g. The source term estimations are reproduced as Table 3-2 in this SER.

Table 3-2: Assumed Concentrations of Radionuclides *

Radionuclide	Average (Expected) Concentration Shipped from Hematite (pCi/g)	Average Cell Concentration if Shipped at Expected Concentration (pCi/g)	USEI WAC Concentration in Rail Cars (pCi/g)
Tc-99	7.2	0.38	0
U-234	62	3.3	1815
U-235	2.8	0.15	81
U-238	13	0.68	341

* Multiply Ci by 3.7x10¹⁰ to obtain Bq

3.1.1. Concrete and Asphalt

WEC approximated the volumes for concrete and asphalt through visual inspection and physical measurements of the various structures and items. WEC approximated the concentration levels using the results of a total of 50 sample cores taken over two phases. The locations of the samples were selected on the basis of the results of the first of two gamma walkover surveys. In the first phase of core sampling, WEC collected 21 cores and subsampled in the top ¼ inch, next ½ inch, and remainder of these cores. In the second phase, WEC collected 29 additional

cores. These 29 cores were either: (i) analyzed as a whole core (20 cores); (ii) subsampled in the top three inches and the remainder (five cores); or (iii) subsampled in the top ¼ inch and the next ½ inch (four cores). The samples were analyzed for isotopic uranium, Tc-99, Am-241, Np-237, Pu-239, Ra-226, and Th-232. Of the 50 cores, 23 were analyzed for Am-241, Np-237 and Pu-239. Because no samples exceeded the minimum detectable concentration (MDC) for Am-241, and only three samples were slightly above the MDC for both Np-237 and Pu-239, WEC concluded that these three transuranics were present only at trace levels. WEC presented this information in Section 6.1 of Revision 1 of HDP-TBD-WM-906.

WEC performed a second gamma walkover after the buildings at its Hematite site had been demolished to more precisely delineate areas associated with elevated activity from those that are relatively uncontaminated. WEC identified six areas of elevated activity based on the gamma walkover and sample results. Due to high activity results in Area 5 and a portion of Area 1, WEC excluded these areas from the alternative disposal request. WEC calculated the average of the samples within each non-excluded elevated area. The averages were presented in Table 6-5 of Revision 1 of HDP-TBD-WM-906. WEC calculated a total curie amount for each elevated area and a weighted average for each of the elevated areas using the relative size of each. The radionuclide concentration in concrete outside the process building and the asphalt areas are based on the average concentrations for the non-elevated areas of the process buildings. Finally, WEC calculated an overall weighted average by weighting the included elevated (18%) and non-elevated areas (82%) by relative size.

3.1.2. Piping

WEC approximated the volume and weight of piping based on data obtained from engineering drawings. WEC approximated the concentration of the piping based on swipe and scale/sediment samples taken in 2010. Swipe samples were targeted at areas with high uranium concentrations. Piping was classified based upon system segments according to physical location or system function. A total curie amount for each system was calculated based on the assumed amount of debris within each pipe segment. WEC excluded the piping under Buildings 240 and 260 from this alternate disposal request. Since this piping contains 87 percent of the Tc-99, this piping is not a candidate for disposal at USEI.

3.1.3 Miscellaneous Equipment

WEC characterized equipment based on gamma radiation measurements taken in 2008. The gamma radiations levels were interpreted to a total U-235 enrichment and a U-235 amount using the Monte Carlo N-Particle (MCNP) Transport Code. WEC estimated the U-235 activity concentration (pCi/g) by dividing the total amount of U-235 activity by the mass of each miscellaneous equipment component (HDP-TBD-WM-906). Then WEC calculated concentrations of U-234, U-238, Tc-99, Th-230, Th-232, and Np-237 by applying scaling factors from HDP-TBD-WM-901 (ADAMS Accession No. ML12090A191). WEC stated that "the scaling factors are appropriate because they were based on samples obtained from surfaces that were exposed to the same radionuclide mixture [as the miscellaneous equipment]."

3.1.3. Sub-Slab Soils

WEC predicted the volume of soil that will require excavation based on soil sample results that exceed remediation goals or Derived Concentration Guideline Limits (DCGLs). A total of 94 samples were collected from the soil beneath the former process buildings down to a depth of 16.5 feet (5.03 m) (HDP-TBD-WM-906). The total curie amount contained in the soil to be excavated was estimated based on the concentration results within these areas and analytical calculations as described in Section 3.2.

WEC determined that Ra-226 was present only at background levels, and WEC analyzed for Ra 226 using gamma counts of radium progeny from the top ¼ inch and the remainder of the core of 23 sample cores at 21 locations. WEC calculated the lowest ratio of Ra-226 to U-234 ($1.8E-5$) among the samples taken from the top ¼ inch and multiplied this by each U-234 activity to find a lower bound for the Ra-226 attributable to U-234 contamination present in the top ¼ inch. This Ra-226 concentration was subtracted from the observed values and this adjusted set of observed values was then compared to the set of observed values from below ¼ inch which are representative of background. Because the adjusted concentration profile for the top ¼ inch was less than or equal to the background sample profile, WEC determined that Ra-226 was present only at background levels (HDP-TBD-WM-906). Therefore, Ra-226 was not included as a radionuclide of concern. Based on interviews with former employees, WEC believes that Ra-226 was introduced into the burial pits from the disposal of contaminated equipment or materials from the Mallinckrodt Site Uranium Division near St. Louis, MO. Radium-226 was not a licensed radionuclide for Hematite, and therefore would not have been expected to have been used in the processes at Hematite (HDP TBD WM 906).

WEC determined that Th-232 was present only at trace levels. WEC measured the Th-232 concentrations for 23 sample cores using alpha spectroscopy. The ratio of Th-232 to U-234 ranged from $4.1E-3$ to $3.7E-6$. Considering this low ratio range compared to the observed levels of U-234 contamination, WEC concluded that Th-232 is present only at trace levels. Therefore, Th-232 was not included as a radionuclide of concern.

3.2. NRC Evaluation of WEC's Material Characterization

Given that the source term values presented by WEC are an estimate, WEC committed to performing additional characterization of the concrete/asphalt, soils, and piping prior to shipment to verify amounts and to ensure adherence to the Tc-99 limits associated with License Condition 17 of the Hematite license. The adequacy of these future sampling plans is discussed in Section 4 of this SER. The following sections describe NRC's evaluation of the available characterization data, which was used to estimate the dose and help define the limits imposed under License Condition 17.

3.2.1. NRC Evaluation of Concrete/Asphalt Characterization

The NRC's review resulted in several RAIs pertaining to the existing characterization of the concrete/asphalt. These requests were mainly focused on the adequacy of the existing characterization for Tc-99 and on obtaining clarifying information regarding the data presented

in the Tables and Figures in the January 16, 2012 request and supporting characterization documents.

The NRC staff requested that WEC provide justification for its conclusion that no areas of Tc 99 have been overlooked given that sample locations were biased based on the gamma walkover survey results and given that a gamma walkover survey does not detect Tc-99, which is a beta emitter. WEC provided additional details regarding the current dataset, and also committed to perform additional sampling on a systematic grid for Tc-99. In Enclosure 2 (ADAMS Accession No. ML12209A201) of their July 24, 2012 RAI response WEC clarified that 33 of the 50 sampling stations were biased. Of these 33 stations, eight stations were in five areas defined as having historical operations involving materials contaminated with Tc-99 (locations 2 – 7; 20, and 21), and 12 stations served to bound the five areas with elevated Tc-99 activity. The other 17 of the 50 sampling stations were not biased. These 17 stations were selected as being representative of the non-elevated areas.

In RAI SA-3, the NRC staff requested clarification on the relationship between the data presented in Tables 6-2 thru 6-4 and Fig 1 of Appendix D of HDP TBD WM-906. In WEC's response to RAI SA-3, WEC clarified that the values for each station shown in Tables 6-2 and 6-3 are weighted average concentrations for all samples from each specific location. For example, if three samples were taken at a certain location (i.e., from top 1/4 inch, next 1/2 inch, remainder), then each sample result was weighted by the mass or thickness of concrete it represented to determine the average for that location. WEC's response resulted in revisions to Tables 6-2 thru 6-3 and Figure 1 of Appendix D of HDP TBD WM-906 which were presented in Westinghouse response associated with HEM-12-67.

In their response to RAI SA-5, WEC explained that Table 6-5 of HDP-TBD-WM-906, which shows the concentration for each elevated area, is an average of all the samples assumed to be in each of the elevated areas, excluding the bounding samples around the perimeter of the elevated areas which had lower concentrations. WEC provided additional details on how the calculations were performed for each of the elevated areas.

As noted in Section 3.1.1 of this SER, WEC is excluding Area 5 and a portion of Area 1 from the request for alternative disposal due to high Tc-99 activity results. The NRC staff asked WEC in RAI CH-10 how they would distinguish between these excluded and included areas during the review. In WEC's response to RAI CH-10 they provided Figure A which shows with different colored fixatives those areas that were included (green) and those that were excluded (blue). WEC stated that this methodology for the excluded portions of the concrete slabs is the same type of identification and control measures (e.g., separate staging areas and containers) as will be used to segregate burial pit soil/debris covered in Hematite Amendment 58 that does not meet USEI criteria (HEM-12-67).

3.2.1.1. NRC Findings

The NRC has concluded that WEC has presented a reasonable explanation of the existing characterization data for the concrete and asphalt and how this data was used to determine the

estimates reproduced in Table 3 1 of this SER. The NRC staff finds the averaging of the data to be appropriate, and thus the values presented in Table 3 1 and Table 3 2 to be reasonable for the purposes of dose estimation. However, due to the uncertainty in the data and the fact that some areas were not previously sampled for Tc-99, the NRC staff has requested WEC to perform additional characterization to verify that amounts sent to USEI do not exceed those assumed in the dose analysis. The adequacy of the future systematic grid sampling plans for concrete/asphalt is discussed in the Health Physics Evolution section of this report found in Section 4 of this SER.

3.2.2. NRC Evaluation of Piping Characterization

Since the samples collected from the pipes in 2010 were targeted at elevated gamma areas or from areas with debris buildup, the NRC staff has concluded that WEC's uranium results for piping are likely to be conservative. However, since Tc-99 contamination may not have been discovered using this approach, the NRC has requested WEC to perform additional surveys or inspections of the piping. During the review of WEC's response to RAI SA-7, NRC asked for additional graphics or tables to clearly segregate and identify the location of piping to which additional surveys and inspections would apply. WEC included this information in Appendix F of the revision of HDP-TBD-WM-906.

WEC is excluding piping from Building 240 and Building 260 from this alternate disposal request based on high Tc-99 activity results for these piping systems. WEC will not send piping from these buildings to USEI. WEC clarified in their June 19, 2012, response to RAI CH-10 that the same type of identification and control measures (e.g., separate staging areas and containers) used to segregate burial pit soil/debris that do not meet USEI criteria will be employed for the excluded portions of the piping and miscellaneous equipment (HEM-12-67).

3.2.2.1. NRC Findings

As detailed in Section 7.2 of HDP-TBD-WM-906, WEC will perform additional systematic characterization of the piping prior to shipment, and will also perform biased sampling based on uranium content as detailed in response to RAI CH-8 (HEM-12-67). The NRC evaluation of the additional sampling to be performed on piping is presented in Section 4 of this SER.

The NRC staff finds the existing characterization of piping to be adequate for the purposes of dose analysis given that WEC has excluded the known high Tc-99 areas from the request. Since the piping material makes up a small relative volume of the disposal material, it contributes a relatively small proportion of the dose. In addition, WEC will has committed to future systematic and biased sampling to ensure that any areas with Tc-99 contamination were not overlooked in the existing characterization.

3.2.3. NRC Evaluation of Miscellaneous Equipment Characterization

The NRC staff reviewed the technical basis for the surrogate factors provided by WEC (HDP TBD-WM-901). The scaling factors were based on smear samples obtained from the various process building areas. In October 2004, ten smears were collected from each building area

and were composited to a single sample for each area. In April 2010, nine biased concrete samples were obtained from the process building walls. While these samples were not of the equipment themselves, WEC stated that the scaling factors are appropriate since the samples were obtained from surfaces that were exposed to the same radionuclide mixture as the equipment. Specifically, the ventilation system would have drawn air from the same facility conditions that resulted in surface contamination identified in the swipe samples. WEC cited the common facility conditions as justification for the use of the same scaling factors. Since the scaling factors are based on average concentrations, WEC also pointed out that even if the maximum ratio of Tc-99 to U-235 were used, the total Tc-99 associated with the ventilation equipment would only change from 0.962 MBq (2.6×10^{-5} Ci) to 5.55 MBq (1.5×10^{-4} Ci), which is insignificant in relation to the total quantity of Tc-99 associated with the application of 11,840 MBq (0.32 Ci) (HEM-12-67).

3.2.3.1. NRC Findings

As indicated in Section 8.1 of HDP-TBD-WM-906 and the response to RAI SA-1, while WEC will not be re-evaluating the inventory of uranium for the equipment listed in Table 8.1 of HDP TBD WM-906, WEC will collect swipe samples of the miscellaneous equipment to verify the Tc-99 scaling factor prior to shipment to USEI (HEM-12-67). The NRC staff has concluded that this approach is acceptable given that WEC will verify Tc-99 scaling factors and will adjust the associated inventory accordingly.

3.2.4. NRC Evaluation of Sub-Slab Soil Characterization

In RAI CH-10, the NRC staff requested that WEC provide additional information regarding the calculations for determining the total curie amount in the sub-slab soils. WEC's response provided additional details on the methods of calculating soil volumes and curie amounts (HEM-12-67). WEC derived contours, which were presented in Figure H-1 in Appendix H of HDP-TBD-WM-906, using a Geographical Information System (GIS) program based upon the data from the 94 samples. These contours represented the volume of soil that is expected to be above the DCGLs. WEC calculated the in-situ volume based on a soil density of 1.69 g/cm^3 using the same GIS program. The volume of each depth layer was multiplied by the average concentration for that layer to calculate a curie amount. The in-situ volume was then multiplied by $1.69/1.44$ to obtain the post-excavation volume that would be shipped. (The density of the soil post-excavation is assumed to be 1.44 g/cm^3 .) The NRC staff concluded that the methods used to calculate the volumes and total curie amount for the sub-slab soil to be acceptable based on the information provided in the RAI responses.

Given the low activity levels in the characterization data provided, and the knowledge that the transuranic were not significant radionuclides for the Hematite DP, the NRC staff finds it reasonable to exclude Am-241, Np-237, and Pu-239 from the list of radionuclides of concern for this analysis.

3.2.4.1. NRC Findings

Based on the characterization data provided, and the historical knowledge of the facility, the NRC staff finds it acceptable to exclude Ra-226, Th-232, as well as the transuranic Am-241, Np-237, and Pu-239 from the list of radionuclides of concern for this analysis. NRC staff notes that even if these radionuclides were assumed to be present at their maximum concentration reported in the characterization data, the disposal of this material at USEI would contribute negligible dose to any member of the public either at the USEI facility or in transportation to the USEI facility.

3.3. WEC Assessment of Doses

3.3.1. Transportation and USEI Worker Doses

WEC analyzed the dose to USEI workers as well as the potential dose during transportation of the waste to USEI. The USEI workers included a gondola surveyor, an excavator operator, gondola cleanout worker, truck driver, stabilization operator, and cell operator. These dose assessments were similar to those provided by WEC in its 2009 alternate disposal request. WEC estimated that 352 gondola railcars will be used to transport the waste from the Hematite site to USEI. The contents of the gondola railcar will be enclosed in wrappers meeting the U.S. Department of Transportation (DOT) Industrial Type-1 Package (IP-1) requirements, which preclude dispersal of waste to the air or loss of material during transport. Once the waste is received at the USEI site, the gondola railcar will be surveyed and then off-loaded into trucks for transport to the USEI disposal cell. Once the waste is off-loaded, USEI personnel will remove any residual material in the railcar using shovels and brooms. The truck is surveyed prior to being driven to the USEI disposal cell, where the waste is spread and compacted in the cell. A fraction of the waste (less than 5%) is expected to contain hazardous constituents that require stabilization. This waste will be treated inside the USEI containment building prior to disposal.

Table 3 3 summarizes the job function scenario assumptions. The times assigned are the times for one person to perform each function once. In WEC's analysis, it is assumed that a specific number of workers per year will be available to carry out each of the job functions, and the total dose for the job function is divided equally among all workers within a job function group. Job functions are not shared among employees tasked as an excavator operator, truck driver, stabilization operator, or cell operator. These workers' responsibilities are not assumed to overlap. However, the groups performing tasks as gondola surveyors, gondola clean-out crews, and truck surveyors may involve the same individual employees.

Table 3-3: Job Function Scenario Assumptions

Job Function	Number of Workers in Group	Minutes to Perform Task	Type of Conveyance (count)
Gondola Surveyor	8	20	Gondola (352)
Excavator Operator	4	45	Gondola (352)
Gondola Cleanout	8	10	Gondola (352)
Truck Surveyor	8	5	Truck (1056)
Truck Driver	14	45	Truck (1056)
Stabilization Operator	6	45	Gondola (18)
Cell Operator	2	15	Gondola (352)

The MicroShield 7.02 code was used to calculate the external doses for the workers. The parameters used to estimate the external dose were identical to those used in the previous Hematite §20.2002 request except for the shielding thickness assumed in the calculation of potential dose to the gondola surveyor and the size and shape of the stabilization tank used in the calculation of dose for the stabilization worker. WEC stated that the changes in these assumptions were made in order to more accurately reflect the actual conditions for the gondola surveyor and stabilization operator. WEC also recalculated the dose to these workers for the prior request and found that these changes in assumptions only result in a slight increase to the calculated dose for these workers. The method and parameters used by WEC to calculate the internal dose for the excavator operator, gondola cleanout worker, stabilization operator, and cell operator are the same as those used in the previously approved §20.2002 request. The internal dose from the inhalation of contaminated dust was calculated based on an assumed concentration of dust in the building of 0.23 mg/m³, an assumed inhalation rate of 1.2 m³/hr, the concentrations of radioactivity in Table 3-2, and the FGR 11 Inhalation Dose Conversion Factors (DCFs). The assumed dust concentration was based on a study that found that the respirable dust concentrations at the USEI facility ranged from 0.17 to 0.23 mg/m³. WEC did not take credit for the respiratory protection program at USEI, so the actual inhalation dose would likely be smaller than what was calculated. Unlike in the previously approved §20.2002 request, an internal dose was not calculated for the gondola surveyor, truck surveyor, or the truck driver. WEC clarified that internal doses were not assigned to these workers because the truck bed and gondola railcar remains covered while they are being surveyed and the truck bed remains covered during the trip to the disposal cell, so these workers would not be expected to receive an internal dose.

Table 3-4: Annual Dose per Person for Individual Job Function*

<i>Job Function</i>	<i>Internal Dose (mrem/yr.)</i>	<i>External Dose (mrem/yr.)</i>	<i>Total Dose (mrem/yr.)</i>
Gondola Surveyor	NA	1.6×10^{-3}	1.6×10^{-3}
Excavator Operator	1.8×10^{-1}	2.7×10^{-3}	1.9×10^{-1}
Gondola Cleanout	2.0×10^{-2}	1.7×10^{-3}	2.2×10^{-2}
Truck Surveyor	NA	2.1×10^{-3}	2.1×10^{-3}
Truck Driver	NA	1.2×10^{-2}	1.2×10^{-2}
Stabilization Operator	6.1×10^{-3}	1.4×10^{-4}	6.3×10^{-3}
Cell Operator	1.2×10^{-1}	7.8×10^{-3}	1.3×10^{-1}

*multiply mrem/yr. by .01 to obtain mSv/y

To evaluate the potential dose to the public during transport of the waste by rail to USEI, the maximum external dose at 1 m and 1 ft from a loaded gondola railcar was calculated by WEC using Microshield. It was found that the maximum dose at 1 m is 0.18 μ R/hr and at 1 ft is 0.25 μ R/hr. WEC stated that based on these dose rates, an individual would have to spend 1,007 hours at 1 m from the gondola railcar or 793 hours at 1 ft from the railcar to receive a higher dose than a site worker. WEC stated that these exposure times are orders of magnitude higher than the expected worker exposure time of less than 20 hours.

3.3.2. Post-Closure Dose

The appropriateness of the RESRAD model for the USEI site was reviewed by USEI staff upon USEI purchasing the site from EnviroSAFE in 2001. The USEI staff concluded that the code was appropriate for the site conditions. In 2005, USEI hired consultants to review the input values used for RESRAD, and determine site-specific inputs that should be used with the code to more accurately reflect the site environmental conditions. Most of the site-specific parameters are explained in the 2005 report titled "Site-specific RESRAD Water Pathway Parameters for the Contaminated Soil, Vadose Zone, and Saturated Zone". This report was provided in WEC's December 29, 2009 RAI response noted as HEM-09-146 (ADAMS Accession No. ML100320540) to the May 2009 alternative disposal request. For those parameters not described in the report, WEC provided additional justification with its March 31, 2010 (HEM-10-38) submittal (ADAMS Accession No. ML100950397.)

Since Tc-99 is the primary contributing radionuclide, the total quantity of Tc-99 (as opposed to the concentration) will drive the dose consequences. RESRAD applies the concentration of Tc 99 and the volume of soil in the contaminated zone to determine the total quantity of Tc 99 that is available in uptake pathways. The value that WEC applied for the expected concentration of Tc-99 in the waste shipped to USEI was 7.2 pCi/g (Table 3 2). This concentration spread over approximately 23,000 m³ yields an expected total Tc-99 inventory of approximately 0.2 Ci, to which WEC has multiplied an uncertainty factor of 1.5 to account for the potential to encounter more material than estimated based on existing data. This results in an approximate 0.3 Ci of Tc-99 as shown in Table 3 1.

WEC plans to treat the material identified in this request cumulatively with the material from the previous request. To ensure that the inventory calculated from the mean activity concentrations

(derived from the mass-weighted concentrations of each stockpile) remains below the cumulative limit, WEC plans to sample the outgoing shipments of material. The sampling plan and associated contingency limits, which are discussed in Section 4 of this SER, will ensure that the cumulative mean and 95th percentile upper confidence limit (UCL) of the mean will not be exceeded. WEC selected the UCL of the mean in order to maintain the dose at the UCL within the 'few mrem' criterion. Table 3-5 shows the Tc-99 mean and UCL inventory limits for the prior request, and the current request, as well as the cumulative limit.

Table 3-5: Cumulative Tc-99 Limits for §20.2002 Requests*

	Prior §20.2002 Request	This §20.2002 Request	Cumulative Action Threshold
Total Quantity of Tc-99 shipped to USEI (Mean)	1.0 Ci	0.3 Ci	1.3 Ci
Equivalent Dose for Mean	1.9 mrem/yr	0.8 mrem/yr	2.7 mrem/yr
95% UCL of the Mean of Tc-99 shipped to USEI	1.6 Ci	0.45 Ci	2.05 Ci
Equivalent Dose for the 95% UCL of the Mean	3 mrem/yr	1.2 mrem/yr	4.2 mrem/yr

*multiply mrem/yr by .01 to obtain mSv/y

WEC included a long-term post-closure analysis assuming a resident farmer scenario. WEC used the RESRAD code Version 6.4, applying site-specific parameters where appropriate, to calculate the long-term post-closure dose.

WEC estimated the post-closure long-term dose for the material associated with this request to be approximately 0.008 mSv (0.8 mrem). The dose is delivered through the groundwater pathway, and Tc-99 is the primary contributing radionuclide. WEC provided an estimate of the cumulative long term post closure dose, adding the long term dose of 0.019 mSv (1.9 mrem) associated with the previous request to the current predicted 0.008 mSv (0.8 mrem), or a total of 0.027 mSv (2.7 mrem).

WEC also performed a sensitivity analysis to evaluate the impact of a shorter project duration and therefore a decrease in the volume of non-Hematite waste that is available for mixing with Hematite waste. WEC analyzed a scenario in which the waste is sent over the shortest possible duration of 13 weeks, which resulted in a post-closure dose of approximately 0.016 mSv (1.6 mrem) as compared to 0.008 mSv (0.8 mrem).

3.3.3. Inadvertent Intruder Dose

To calculate dose to the intruder post-burial, WEC used the methods from NRC Guidance NUREG/CR 4370, Volume 2 (ADAMS Accession No. ML100250917). WEC performed inadvertent intruder analyses similar to those performed in their March 31, 2010 analysis performed in support of the May 2009 §20.2002 alternate disposal request (ADAMS Accession No. ML100950386). The analyses included variations on assumptions about the concentration of the material as it is shipped and the extent to which the shipping concentrations are diluted once it has been disposed of in the cell as detailed in Figure 3-1. WEC did not evaluate the Average Cell Concentration scenario for material shipped at the WAC for all radionuclides because the volume of material and concentration limits for Tc-99 are such that it would not be

possible for WEC to ship the total volume of waste under this request at the WAC. Instead, WEC did a sensitivity analysis assuming that the total volume was shipped at the WAC containing uranium at values listed in Table 3 2, but not containing Tc-99.

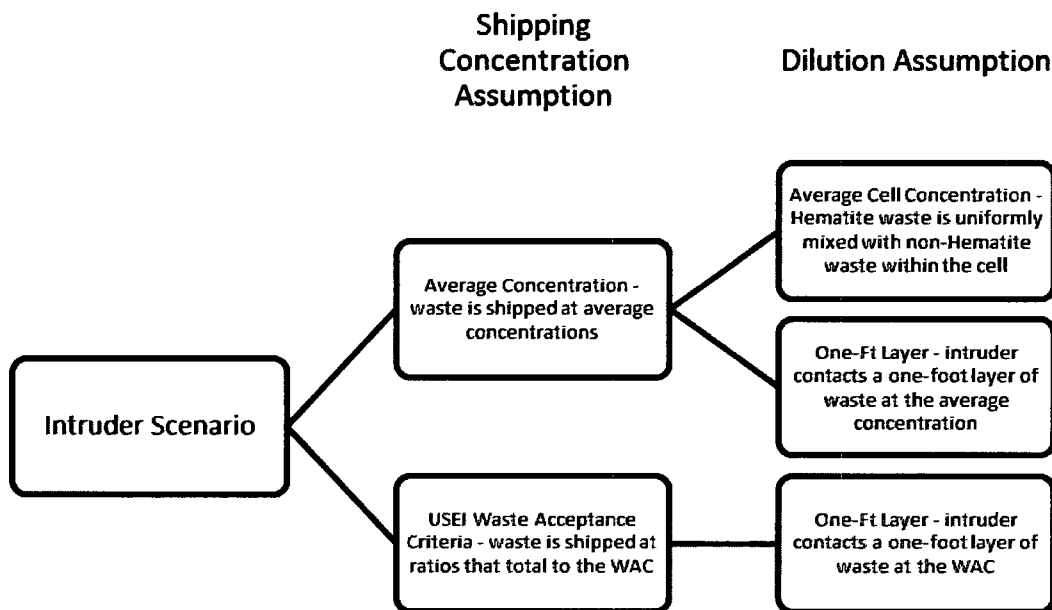


Figure 1: Intruder Scenario Waste Concentration Assumptions

3.3.4. Intruder Well-Driller Scenario

WEC evaluated two intruder well-driller scenarios (acute and chronic) as detailed below.

Acute Well-Driller	
Description	Intruder digs a well by drilling through the waste disposal cell to reach the underlying aquifer at a depth of 93.1 m. The total period of exposure is 40 hours, 8 of which occur during the drilling through the contaminated layer.
Concentration of Contaminated Layer	Concentration of the contaminated layer of Hematite waste, which is either the Average Cell Concentration, or the WAC Concentration as shown in Error! Reference source not found..
Additional Dilution of Contaminated Layer During Exhumation	Concentration of the contaminated layer multiplied by the ratio of 0.31/93.1 or 3.3×10^{-3} , which is the ratio of a 1-ft contaminated layer (0.31 m) to the total well depth (93.1 m).
Dose	0.029 mSv/yr (2.9 mrem/yr) based upon the intruder drilling through a 1-ft layer at the WAC.

Chronic Well-Driller	
Description	Intruder spreads the exhumed drill cuttings around the residence and grows a garden in soil containing the drill cuttings over the course of one year. His time for the year is spent either gardening (100 hours), outdoors (1,800 hours) or indoors (4,380 hours).
Concentration of the Waste	Maximum concentration resulting from the acute well-drilling (based on the soil disposed at the WAC in 1-ft layer).
Dose	3.0 mrem/yr based upon the intruder drilling through a 1-ft layer at the WAC.

3.3.5. Intruder Construction Scenario

WEC evaluated the intruder construction scenario as detailed below.

Construction Intruder	
Description	Intruder is assumed to excavate or construct a building on a disposal site following a breakdown in institutional controls. The intruder is exposed to dust particles through the inhalation pathway, and may also be exposed to direct gamma radiation resulting from airborne particulates and by working directly in the waste-soil mixture. The dose from the inhalation and from external gamma exposure is evaluated for duration of 500 working hours, or a construction period of 3 months.
Concentration of Waste to Which Intruder is Exposed	<ul style="list-style-type: none"> • Average Cell Concentration – Shipping concentration (either Average or WAC) multiplied by 0.053, which is calculated by taking the ratio of Hematite waste to total waste received (38,710 tons/ 725,000 tons). • 1-Ft Layer – Shipping Concentration (WAC) multiplied by a factor of 0.31 (12 in/39 in) to account for USEI's practice of layering waste into pits in 1-ft layers and an assumption that 1 meter (39 in) of waste is excavated.
Dose	Results range from 0.1 mrem - 16 mrem, with the highest value assuming the intruder encounters a 1 ft. layer at the WAC values.

3.4. NRC Assessment of Doses

3.4.1. Evaluation of Transportation and USEI Worker Dose

The NRC staff finds that the scenarios selected for the transportation and USEI worker dose assessment are consistent with the manner in which the waste will be transported to and handled at USEI. Additionally, the NRC staff finds that the parameter values selected appropriately represent the job functions and the site conditions at USEI. NRC staff performed independent calculations of the external doses using MicroShield and obtained similar results to those obtained by WEC. In addition, NRC staff performed independent calculations of the internal dose and obtained similar results to those obtained by WEC.

3.4.1.1. NRC Findings

Since the waste disposal covered by the approved 2009 §20.2002 request is still ongoing, there is some potential for the USEI workers to receive -- during the same year -- a dose both from that action and the current January 16, 2012 request.. However, as seen in Table 3 6, even if the workers were to receive the total expected annual dose from both sets of waste during the same year, the cumulative dose would still be less than one millirem. Therefore, the results of the dose assessment for the USEI workers indicate that the dose to these individuals will be within the "few millirem" criteria.

Table 3-6: Potential Cumulative Dose from Previous and Current §20.2002 Requests*

Job Function	§20.2002 Request Approved in Amendment 58 (mrem/yr)	Current §20.2002 Request (mrem/yr)	Total Dose (mrem/yr)
Gondola Surveyor	1.1×10^{-01}	1.6×10^{-03}	1.1×10^{-01}
Excavator Operator	4.7×10^{-01}	1.9×10^{-01}	6.6×10^{-01}
Gondola Cleanout	5.9×10^{-02}	2.2×10^{-02}	8.1×10^{-02}
Truck Surveyor	9.3×10^{-02}	2.1×10^{-03}	9.5×10^{-02}
Truck Driver	4.9×10^{-01}	1.2×10^{-02}	5.0×10^{-01}
Stabilization Operator	1.6×10^{-02}	6.3×10^{-03}	2.2×10^{-02}
Cell Operator	3.8×10^{-01}	1.3×10^{-01}	5.1×10^{-01}

*multiply mrem/yr by .01 to obtain mSv/y

3.4.2. NRC Evaluation of Post-Closure Dose

The staff finds that approval of the January 16, 2012 request will not yield a post closure long-term dose that is more than a few mrem/yr provided the total inventory of Tc-99 remains within the limits of 2.05 Ci. The staff finds this upper confidence limit to be acceptable because the dose resulting from the total inventory is also within a few mrem. A detailed discussion of the review of the sampling plan and contingency limits is contained in Section 4 of this SER.

Regarding cumulative post-closure doses, the staff agrees that it is acceptable in this case to treat the material cumulatively and to calculate a cumulative long term post-closure dose given that Tc-99 (through the groundwater pathway) is the primary contributor to dose. The staff finds the expected cumulative dose of 0.027 mSv (2.7 mrem) to be within the acceptable range of 'a few millirem'. The staff notes that while WEC separately analyzed impacts of shipping schedules on this and the prior request, WEC did not analyze the combined impacts of a faster shipping schedule for both requests. In absence of an assessment provided by WEC of a combined effect of a fast shipping schedule for both this request and the prior request (ADAMS Accession No. ML100950386), the NRC staff analyzed the cumulative impact of faster shipping schedules by adding the prior estimated 0.041 mSv (4.1 mrem) dose for the May 2009 request (ADAMS Accession No. ML110560334) assuming a 20 railcar/week shipping rate to the 0.016 mSv (1.6 mrem) estimate for the January 16, 2012 request. Because the cumulative 4.1 millirem dose in this scenario is still within a 'few millirem', the NRC staff finds the post-closure cumulative doses acceptable.

3.4.2.1. NRC Findings

NRC staff finds the parameter values and assumptions used in calculating the post-closure dose acceptable based on review of the USEI 2005 report and the RAI responses (HEM-09-146 and HEM-10-38). NRC staff performed independent assessments of WEC's calculations for post-closure dose and finds the post-closure doses submitted by WEC within the criteria of 'a few millirem'.

3.4.3. NRC Evaluation of Intruder Doses

The NRC staff considered the assumptions and pathways for the intruder scenarios to be reasonable based on comparison to the guidance in Appendix G of NUREG-0782 and NUREG/CR-4370 Volume 1.

Staff considers the dilution factor of 0.31 acceptable for the Construction One-Ft Layer scenario after reviewing the standard practices at USEI. They also considered the dilution factor of 0.53 acceptable for the Average Cell Concentration scenario after reviewing historical data for waste volumes sent to USEI. The staff notes the following conservatisms were presented in Section 7.2 of Enclosure 1 WEC's January 2012 submittal:

- No credit taken for the mixing of the waste with the cover material as noted in the RAI Response to Performance Assessment RAI No. 9, (ADAMS Accession No. ML100320540).
- USEI restriction of the emplacement of any radioactive waste to within 3.6 meters of the surface of the finished cap of the cell, which could rule out the construction scenario as not a feasible scenario.
- No credit taken credit for decay up to the intrusion event, for waste form, or solidification.

During the review, the NRC staff requested that WEC provide a discussion of the cumulative intruder doses for the prior §20.2002 request and this request. Table 3 7 shows the cumulative intruder doses, which are simply the sum of the doses assumed for the prior and current requests (HEM-12-67). The NRC staff notes that assuming an arithmetic sum for the cumulative intruder dose is conservative given that the intruder is not likely to encounter waste from both requests in the same location.

Table 3-7: Cumulative Intruder Doses

Scenario	Max Dose for Prior §20.2002 Request (mrem/yr)	Max Dose for this §20.2002 Request (mrem/yr)	Max Cumulative Dose (mrem/yr)
Intruder Construction	10	16	26
Intruder Acute Well Drilling	2.9	2.9	5.8
Intruder Chronic Well Drilling	2	3.0	5.0

*multiply by 0.01 to convert mrem/yr to mSv/yr

3.4.3.1. NRC Findings

The NRC staff finds the assumptions and pathways considered for the intruder scenarios to be reasonable based on comparison to the guidance in Appendix G of NUREG-0782 and NUREG/CR-4370 Volume 1. The NRC staff finds the intruder doses acceptable, given the conservative approach. The staff notes that the time for the intruder construction scenario was limited to 500 hours. The intruder construction scenario that WEC analyzed does not account for the chance that the intruder could subsequently live and grow food onsite due to the site's remote location and arid environmental conditions. The staff agrees with the technical basis for why intruder agricultural practices at the site are highly improbable. The NRC staff find the concentration assumptions for the WAC (that the 3,000 pCi/g is attributable fully to uranium and not Tc-99) in the sensitivity analyses performed by WEC acceptable because Tc-99 is not a significant radionuclide for the intruder scenarios and because uranium, through the air and direct gamma pathways, is the main contributor to dose for the intruder scenarios.

3.5. Stability of the Disposal Facility Following Closure

3.5.1. Westinghouse Assessment

Site-stability can be impacted by natural surface and subsurface processes, and is also impacted by the stability of the waste and engineered barriers of the disposal facility. In WEC's March 31, 2010 submittal associated with the prior alternative disposal request, WEC provided a technical basis for the stability of the USEI site stating that the facility was "constructed in compliance with the Resource Conservation and Recovery Act (RCRA) standards and the applicable Minimum Technology Requirements (MTRs). These requirements provide conservative criteria for cell construction to insure long-term stability and are consistent with the erosion design requirements in 10 CFR Part 61, and the joint NRC/EPA guidance document with guidelines on drainage and processes impacting stability."

3.5.2. NRC Evaluation and Findings

The NRC has noted that site-stability can be impacted by natural surface and subsurface processes and by the stability of the waste and engineered barriers of the disposal facility. The NRC staff has evaluated WEC's technical basis for the stability of the USEI site. The NRC staff has concluded that construction of the USEI facility to the Resource Conservation and Recovery Act (RCRA) standards and to the applicable Minimum Technology Requirements (MTRs) sufficient to provide long-term stability and to be consistent with the erosion design requirements in 10 CFR Part 61 and the joint NRC/EPA guidance document with guidelines on drainage and processes impacting stability.

4. HEALTH PHYSICS ASSESSMENT

4.1. WEC's Waste Material Characterization

WEC provided the characterization data for the waste to be shipped by rail to USEI in Attachment 1, Characterization Data Summary in Support of Additional USEI Alternate Disposal Request, HDP-TBD-WM-906, to Enclosure 1 of its January 16, 2012 request.

4.1.1. Soil Characterization

In Section 5.2.1 of Revision 2 of HDP-TBD-WM-908, WEC committed to following the same soil sampling plan described in the "Technical Basis for Characterization of Decommissioning Soils Waste That is Subject to the Alternate Disposal Request for US Ecology Idaho, Inc., Revision 1 (ADAMS Accession No. ML110530155)." This sampling plan was transmitted to the NRC in WEC's February 18, 2011 submittal and was previously approved by the NRC with the issuance of Amendment 58 to the Hematite license (ADAMS Accession No. ML112560105). Sampling protocols, detection capabilities, and activity limits for U-234, U-235, U-238, Th-232, Ra-226, and Tc-99 were provided by WEC in the aforementioned technical basis document and remain the same for the current request, with the exception of the Tc-99 limits. In order to reflect the lower quantity of Tc-99 in the current alternate disposal request, as compared to the quantity associated with the License Amendment 58 request, WEC adjusted the mean Tc-99 concentration to 13 pCi/g and standard deviation associated with soils to 36 pCi/g. Additionally, in Section 5.2.1 of HDP-TBD-WM-908, WEC indicated that a total TC-99 inventory will be maintained by combining the soil and debris concentrations from this request to the inventory approved with Amendment 58. Accordingly, Section 13.4 of the previously approved "Waste Characterization Plan" for soils (provided as Attachment A to the "Revised Technical Basis for Characterization of Decommissioning Soils Waste That is Subject to the Alternate Disposal Request for US Ecology Idaho, Inc.") was updated to indicate that, if it is determined that the mean Tc-99 activity of 0.30 Ci and 95% UCL of 0.45 Ci are within the established limits, the material will be authorized for rail shipment to USEI. An updated listing of action levels and associated contingencies was provided in Appendix R (Contingency Plan Table) of HDP-TBD-WM-906 and is provided as Table 4-1 of this SER.

4.1.2. Piping Characterization

WEC committed in Section 5.2.2 of HDP-TBD-WM-908 to perform additional characterization of piping prior to disposal at USEI. WEC intends to quantify uranium and gamma emitting radionuclides using High Resolution Gamma Spectroscopy (HRGS). Tc-99 concentrations will be determined through laboratory sampling. Further details were provided in Attachment 11 to HDP-TBD-WM-908 (Sampling Plan for Piping Destined for USEI). WEC considered two sampling approaches using the Visual Sampling Plan software package. The first approach was to compare a true average to a fixed threshold using data from the four nuclides: Tc-99, U-234, U-235, and U-238. The Tc-99 data required the most number of samples (at a rate of one sample per 7.1 m³ of material). The second approach determined the number of samples required to define the confidence interval on the mean activity, where the half-width of the confidence interval was set to half of the mean concentration. This approach resulted in a

sampling frequency of one sample per 12.1 m³ of piping. WEC decided to use the more conservative approach of one sample per 7.1 m³. Since prior sampling did not indicate a relationship between Tc-99 and uranium in piping, WEC will utilize random sampling for piping that is eligible for disposal at USEI. The exception will be for piping that is segregated for criticality safety evaluation at a Material Assay Area/Waste Evaluating Area. In the case of piping that segregated for criticality safety evaluation, one sample of such piping --consisting of 4 aliquots -- will be taken from each batch of segregated material. These samples will be biased since they represent a smaller batch which has been removed from a larger randomly sampled population. As noted in Attachment 11 to HDP-TBD-WM-908 (Sampling Plan for Piping Destined for USEI), this represents one sample for each container that was segregated for criticality safety analysis. This will still maintain a sampling frequency of at least one sample per 7.1 m³ of material.

4.1.3. Concrete/Asphalt Characterization

WEC committed in Section 6.6 of HDP-TBD-WM-906 to perform additional characterization of concrete and asphalt prior to disposal at USEI and provided a "Sampling Plan for Concrete and Asphalt" as Enclosure 3 in the July 24, 2012 final responses to the NRC's RAIs. WEC developed a sampling approach using the Visual Sampling Plan software to determine the confidence interval on a mean specific to the Hematite decommissioning project. The half-width of the confidence interval was set to half of the mean Tc-99 concentration outside the five elevated areas identified in HDP-TBD-WM-906, and the standard deviation of the same data set was used. The resultant sampling frequency was 20 samples per area, and buildings 240, 253, 254, 255, 256, 260, and 235/252 were each designated as 7 separate sampling areas. WEC has committed to taking concrete samples on a systematic grid, to depths of 0.75 inches and 1.5 inches, as shown in Appendix A of Enclosure 3 of the July 24, 2012 WEC RAI response. Samples from the 0.75 to 1.5 inch depth will be used to assess the contamination within the remaining thickness of the concrete slab since existing characterization data indicates that radioactivity of concern is located in the upper 0.75 inch layer of concrete. Asphalt will be sampled at a rate of 20 samples per area throughout five areas adjacent to the process building slab, as shown in Appendix B of Enclosure 3 of the July 24, 2012, WEC RAI response. A 100% beta contamination scan will be performed on the accessible designated asphalt sampling areas, and core samples will be biased toward elevated beta areas followed by random samples within each area in order to meet the 20 sample per area frequency. For both concrete and asphalt, uranium will be measured via gamma spectroscopy and Tc-99 will be measured via laboratory analysis.

4.2. NRC Assessment of WEC's Waste Material Characterization

In response to the staff's RAIs, WEC provided Revision 2 to HDP-TBD-WM-908, "Safety Assessment for Additional Hematite Project Waste at USEI," via an October 17, 2012, letter (ADAMS Accession No. ML12293A029). In Enclosure 1 to this letter, WEC stated that Section 5.2 of HDP-TBD-WM-908 would be modified to indicate that additional characterization of soils, piping, concrete, and asphalt would be completed prior to their shipment by rail to USEI. The associated characterization plans were reviewed by NRC staff, and the staff's assessment follows..

NRC staff performed a health physics review of WEC's January 16, 2012 request, and WEC's RAI responses. NRC staff determined that WEC's January 16, 2012 request did not provide a clearly developed characterization plan nor sufficient justification to demonstrate that the characterization performed to date was adequate to justify the disposal of wastes at a non NRC licensed facility. The staff recommended that Revision 0 of WEC document HDP TBD-WM-906, Characterization Data Summary in Support of Additional USEI Alternate Disposal Request, be revised to present a clear discussion of quantifiable characterization objectives followed by a description of how WEC would demonstrate if and how their characterization activities achieved those goals. The staff also noted that while historical data may be acceptable for use, there are numerous data gaps that require WEC to perform additional investigations and sampling. The staff's May 1, 2012, RAIs enumerated specific areas requiring additional characterization and recommended that WEC develop a formal characterization plan that includes additional systematic probabilistic sampling based on the Data Quality Objectives (DQO) process.

4.2.1.1. NRC Findings

The NRC staff has reviewed WEC's plans for additional soil, piping, concrete, and asphalt sampling and finds that WEC's plans represent acceptable sampling protocols and frequencies to adequately characterize materials prior to shipment to USEI.

4.3. Quality Assurance and Contingency Plans

4.3.1. WEC Quality Assurance and Contingency Plans

WEC developed several quality assurance and contingency plans in order to assess the additional soil, piping, concrete, and asphalt characterization results. Sampling data quality objectives were also provided as Appendix P in Revision 1 to HDP-TBD-WM-906. Associated with the May 2009 §20.2002 alternate disposal request, WEC had provided a detailed quality assurance plan for soils. This plan was described in the "Technical Basis for Characterization of Decommissioning Soils Waste That is Subject to the Alternate Disposal Request for US Ecology Idaho, Inc." (ADAMS Accession No. ML110530155), and was approved as part of the staff's review and approval associated with Hematite License Amendment 58. It was noted in the plan that WEC intends to implement field duplicate samples, field blanks, and laboratory control samples throughout the excavation process at its Hematite site. WEC will collect field duplicates at a frequency of 1 per 20 samples and the results will be evaluated to determine the relative difference or relative percent difference between two data sets. WEC intends to utilize guidance from the Multi Agency Radiological Laboratory Analytical Protocols Manual (MARLAP) to compare results to pre-determined warning and control limits. Field blanks will be collected at a frequency of 1 per 100 samples and these results will be used to evaluate bias. Laboratory control samples, matrix spikes (if applicable), and replicate counts will be performed at a frequency of 1 per 20 samples in order to assess overall laboratory performance.

WEC provided a contingency plan for piping in Section 7.2 of Revision 1 of HDP-TBD-WM-906. WEC indicated that post-collection data analysis will be performed to determine whether the results are adequate in both quality and quantity to support the primary sampling objectives.

Accordingly, WEC indicated that they would review the dataset to ensure that the requisite sampling frequency is met. WEC also committed to compare the Tc-99 results to the action levels provided in Appendix R of Revision 1 of HDP-TBD-WM-906. These action levels are presented below in Table 4-1..

Table 4-1: Pre-Shipment Contingency Plans Proposed by WEC

Parameter	Action Level	How Monitored	Actions
Total Quantity of Tc-99 shipped to USEI (mean)	>1.3 Ci	Running total activity (both shipped and pending shipment), based on laboratory sample results prior to shipment	<ul style="list-style-type: none"> • Reanalyze composite sample and/or analyze individual aliquots used to create the composite sample; • Resample stockpile and re-evaluate; and • Ship material to alternate facility.
95% Upper Confidence Level of the mean Tc-99 shipped to USEI [UCL(0.95)]	>2.05 Ci	Running confidence interval (both shipped and pending shipment) based on laboratory sample data prior to shipment	<ul style="list-style-type: none"> • Reanalyze composite sample and/or analyze individual aliquots used to create the composite sample; • Resample stockpile and re-evaluate; and • Ship material to alternate facility.
Total activity contribution from all radionuclides within individual railcar	>3000 pCi/g > 40 μ R/hr	Laboratory sample results for stockpile evaluated at 95% UCL prior to shipment Gamma radiation levels on railcars prior to shipment	<ul style="list-style-type: none"> • Analyze additional aliquot of composite sample; • Unload railcar (at HDP) and re-load with material containing lower concentration (either blended or alternate material from onsite waste stream); and • Ship material to alternate facility.
Unexpected Tc-99 results for stockpile samples (soil)	>99 th percentile of the site wide dataset (573 pCi/g)	Laboratory sample results for stockpile evaluated prior to shipment	<ul style="list-style-type: none"> • Analyze additional aliquot of composite sample; • Resample stockpile and re-evaluate; • Blend with less contaminated material, resample stockpile and re-evaluate; and • Ship material to alternate facility.
Unexpected Tc-99 results for stockpile samples (concrete)	>99 th percentile of the site wide dataset (1590 pCi/g)	Laboratory sample results for stockpile evaluated prior to shipment	<ul style="list-style-type: none"> • Analyze additional aliquot of composite sample; • Resample stockpile and re-evaluate; • Blend with less contaminated material, resample stockpile and re-evaluate; and • Ship material to alternate facility.

Parameter	Action Level	How Monitored	Actions
Unexpected Tc-99 results for stockpile samples (piping internal debris / residue)	>99 th percentile of the dataset (162 pCi/g)	Laboratory sample results for stockpile evaluated prior to shipment	<ul style="list-style-type: none"> • Analyze additional aliquot of composite sample; • Resample stockpile and re-evaluate; • Blend with less contaminated material, resample stockpile and re-evaluate; and • Ship material to alternate facility.
Unexpected Tc-99 results for stockpile samples (piping average concentration)	>99 th percentile of the dataset (125 pCi/g)	Laboratory sample results for stockpile evaluated prior to shipment	<ul style="list-style-type: none"> • Analyze additional aliquot of composite sample; • Resample stockpile and re-evaluate; • Blend with less contaminated material, resample stockpile and re-evaluate; and • Ship material to alternate facility.
Maximum average concentration of Ra-226 and Th-232 within individual railcar	Ra-226 >13 pCi/g Th-232 >16 pCi/g	Laboratory sample results for each railcar evaluated prior to shipment	<ul style="list-style-type: none"> • Analyze additional aliquot of composite sample; • Resample stockpile and re-evaluate; Blend with less contaminated material, resample stockpile and re-evaluate; and • Ship material to alternate facility.

Section 6.6 of Revision 1 of HDP-TBD-WM-906 describes a contingency plan for concrete and asphalt which includes a retrospective analysis of the data results to verify that a sufficient number of samples were collected to meet the data quality objectives. If an insufficient number of samples are collected, WEC will review the data to determine the cause of the insufficiency. WEC will review the data from each sampling area to determine if it is normally distributed. Data sets which are not normally distributed will be reviewed to identify areas of elevated results. If elevated areas are identified, additional samples will be collected as needed to bound the area, and the results will be compared to the action levels provided in Appendix R of Revision 1 of HDP-TBD-WM-906 and Table 4-1 of this SER.

4.3.2 NRC Assessment of WEC Quality Assurance and Contingency Plans

The staff has reviewed WEC's quality assurance/quality control programs, data quality objectives, and contingency plans. The staff has found them acceptable and their implementation should permit WEC to demonstrate that the NRC's alternate disposal dose requirement (of not more than "a few millirem per year" to any member of the public) can be met.

5. NUCLEAR CRITICALITY SAFETY

This section of the SER addresses the nuclear criticality safety aspects of WEC's January 16, 2012 request, which addresses the shipment of waste to USEI and its disposal there. Disposal at USEI must be done in a manner which ensures that any U-235 in the waste is not placed in a configuration which could result in a criticality safety event. In this regard, WEC has committed that each gondola car of shipped waste to USEI will be below an average concentration of 1 gram of U-235 per 10 liters of waste. WEC identifies this limit as its "NCS exempt material limit."

At this concentration limit, this permits the handling of fissile material without any additional NCS controls since the limit is conservatively set well below the NRC-endorsed minimum critical infinite sea concentration of 1.4 g U-235/liter. The latter value is based upon the data in NUREG/CR-6505, Vol. 1, "The Potential for Criticality Following Disposal of Uranium at Low Level Waste Facilities."

5.1. WEC Criticality Assessment

The decommissioning operations at the Hematite site include the excavation, recovery and collection of contaminated waste, waste characterization, waste treatment, and off-site shipping preparation. WEC performed an NCS assessment to demonstrate that the NCS exempt material limit will be met for waste disposal at USEI and therefore the risk of criticality is not credible (NCSA of the US Ecology Idaho (USEI) Site, NSA-TR-HDP-11-11, Rev. 0, dated December, 2011). WEC's assessment describes the process conditions used at the Hematite site and the characterization of the uranium concentration in the waste streams which are relied upon to ensure that the NCS exempt material limit is met.

5.1.1. Concrete/Asphalt Removal

In order to excavate the subterranean structures, the overlying concrete must be removed. Spills during past manufacturing operations at the Hematite site may have contaminated the overlying concrete, even though such spills were cleaned up (either scrubbed clean or scabbled and then re-surfaced). WEC performed an extensive radiological non-destructive surface assay during 2009 to quantify the residual mass of U-235 associated with the concrete surfaces. This survey was complemented by destructive analysis of cored concrete during 2010 and 2011. Based upon the sampling and assay of the concrete slabs, WEC determined that the total amount of U-235 present in the floor regions of all Hematite facility buildings is less than 4,565 g U-235. With the exception Building 252, the U-235 concentration that was confined in the upper 1/2" of the floor regions is well below the NCS Exempt Material limit of 0.1 g U 235/liters (or 1 g U-235/10 liters).

Once the concrete is removed, WEC will remove any soil and other overlying material (i.e., gravel and stones) that covers the subterranean structures. Since the soil/material of concern was covered by the concrete slabs, the only mechanisms for any non-trivial amount of contamination of the underlying soil are fissile solution spills that reached the soil via a seam or crack in the concrete. Operations that involved fissile solutions were confined to Buildings 240

and 260. Therefore these are the only areas where WEC will assay the underlying soil. Excavation of areas that are found to be below the NCS Exempt Material limit will be performed without any additional NCS controls. However, if an area of soil is found to exceed the NCS Exempt Material limit, then WEC will remove the material and package it in a field container that will be assayed to determine radiological content. Once the contaminated soil is exhumed, two independent surface assays will be performed over the uncovered soil regions. WEC will perform this sequence of operations until soil is determined to be below the NCS Exempt Material limit.

For the disposal of the concrete slab waste, the licensee performed in-situ assays (dual independent measurements) and took core samples that were destructively assayed to determine the U-235 mass present. Based upon these actions and the utilization of a scaling factor of 1.7 to account for the attenuation of gamma rays through the concrete substrate, WEC estimated that the total U-235 mass contained in all the slabs is approximately 4,600 grams. This results in an average concentration of 0.039 grams U-235/liter assuming a ½ inch cut depth which is conservative since the cut depth is typically greater than ½ inch (Table 1.6 of NSA-TR-HDP-11-11) [(ADAMS Accession No. ML12209A200)]. Since a small amount of underlying soil may also be inadvertently excavated with the concrete, WEC took core samples of the soil around seams to verify that the concentration in these areas will not contribute significantly to the amount of U-235 in the concrete slab debris. While two slabs were identified to have a slightly higher concentration (0.105 grams U-235/liter and 0.171 grams U-235/liter), these concentrations are still well below the minimum critical infinite sea concentration for a bounding soil/U-235 medium of 1.4 g U-235/liters. WEC has also implemented a requirement to inspect the concrete during excavation to ensure that any attached debris is characterized.

5.1.2. Subterranean Piping and Sewage Septic Treatment Tank and Drain Field and Drain Line Removal

In 2010, WEC conducted an in-pipe survey to quantify the residual mass of U-235 in subsurface piping that resides mainly beneath the former process buildings. Over one thousand feet of subsurface piping was surveyed. Because the assayed pipe length is a significantly large sample, and the assayed pipes represent pipes with drains that were in the vicinity of the fuel manufacturing operations, results of the in-pipe radiological surveys are assumed to be a bounding representation of all the subterranean piping.

WEC will perform a set of independent measurements on the subterranean piping to ensure the U-235 concentration does not exceed the NCS Exempt Material limit. If the independent assays confirm the pipe meets the NCS Exempt Material limit, the pipe may be transferred to a waste handling area for potential shipment to USEI. Subterranean piping that exceeds the NCS Exempt Material limit will be re-assayed using HRGS equipment to determine the precise fissile nuclide content. If the U-235 concentration exceeds the limit, WEC may commingle the material with a lesser contaminated waste so that it meets the NCS Exempt Material Limit. The resultant debris will be subject to two independent assays to ensure the resultant debris meets the NCS Exempt Material limit. Some of the piping system may be constructed of concrete or vitrified clay, which may be crushed during decommissioning operations. Prior to exhuming the debris

(i.e., mixture of pipe contents, piping material, and any soil/stones/gravel), a set of two independent surface assays will be performed on the debris. If the surface assays establish that the crushed debris meets the NCS Exempt Material limit, the material may be transferred to a waste handling area for potential shipment to USEI. However, if it exceeds the NCS Exempt Material limit, then the associated portion will be removed and packaged per NCS limits.

The Hematite site contains two sewage treatment systems and a concrete septic tank which were connected to the lavatories within the former process buildings. Only one sewage treatment system and the associated sanitation lines and drain lines remain in service. The older sewage treatment tank and concrete septic tank were previously abandoned in place. Prior to exhuming the contents of the current sewage treatment tank, sanitation lines leading to the treatment tank will be exhumed and disposed of following the process used for the subterranean piping as discussed above. If the sanitation lines leading to the current sewage treatment tank meet the NCS Exempt Material limit, and the U-235 activity linearly decreases as the sanitation lines approach the sewage treatment tank, then WEC assumes that the sewage treatment tank meets the NCS Exempt Material limit. WEC indicated in NSA-TR-09-08, Rev. 1, NCSA of the Sub-Surface Structure Decommissioning at the Hematite Site, (ADAMS Accession No. ML12293A029) that this assumption is supported by results of the in-pipe radiological surveys of the subterranean piping beneath the former process buildings. The results of the in-pipe radiological survey demonstrated that the highest observed dose rates were at the elbow section of the pipes. WEC found that as measurements were taken downstream from the elbow sections, the measured dose rates decreased. However, should WEC find sanitation lines which are demonstrated to contain material exceeding the NCS Exempt limit, or U-235 activity which does not decline as the sanitation lines approach the current sewage treatment tank, the treatment tank will then be assumed to contain fissile material. WEC is assuming that soil surrounding the current sewage treatment tank potentially contains U-235 concentrations above the NCS Exempt Material limit. If WEC determines that the soil does not exceed the NCS Exempt Material limit, the soil will be treated as waste and the sewage tank will be assumed to meet the NCS Exempt Material limit. If WEC finds that any of the soil exceeds the NCS Exempt Material limit, then the soil will be removed and packaged in a field container, and subjected to two independent assays. If the soil is found to be contaminated it is most likely due to a leak from the sewage tank. Therefore WEC will assume that the sewage tank also contains fissile material.

Since solids or solutions denser than water settle or layer in the bottom of a treatment tank, any uranium (solids or solutions) discarded into sanitation lines during fuel manufacturing operations could have settled to the tank bottom. Because of this, WEC will require two independent surface assay measurements of the current sewage treatment tank targeted for exhumation. If the content of the current sewage treatment tank is determined to meet the NCS Exempt Material limit, then WEC will assume that the associated drain line will also meet the NCS Exempt Material limit and the lines may be transferred to a waste handling area for potential shipment to USEI. If WEC determines that the current sewage treatment tank contents contain non-NCS Exempt Material then the associated drain line and the sewage treatment tank structure will be assumed to also contain non-NCS Exempt Material and the drain line will be

excavated in accordance with the soil exhumation and subterranean piping removal procedures described above. WEC will subject the resultant debris to two independent assays.

For the decommissioned sewage treatment tank or concrete septic tank, the material residing within the treatment tanks cannot be interpreted as representative of the material in the associated common drain field (i.e., filled with gravel). Thus, WEC will dispose of the common drain field in accordance with the soil exhumation and subterranean piping removal procedures.

5.1.3. Components Remaining as a Result of Building Demolition Operations

WEC performed a radiological survey in 2009 on the components that remained from the building demolition operations. WEC performed Monte Carlo N-Particle (MCNP) calculations to estimate the U-235 mass on components that may be disposed of at the USEI site. WEC performed decontamination and demolition (D&D) operations for the remaining equipment, piping, ventilation ducts, and miscellaneous items/components to prepare these items for removal and decontaminate select items to ensure they meet the limit for transportation and disposal at the USEI site. Following decontamination, WEC applied additional fixative to the contaminated surfaces of these items, as necessary, any material collected during these decontamination activities is not intended to be shipped to USEI. Based on the results of site characterization work, WEC determined that the remaining equipment, piping, ventilation ducts, and miscellaneous items/components have little to no loose UO₂ holdup.

5.1.4. Miscellaneous Equipment as a Result of Decontamination and Decommission

Decontamination and Decommissioning (D&D) efforts may result in contamination of equipment. However, due to the types of equipment used for D&D operations and the nature of the decommissioning waste materials, it is expected that only surface contamination of D&D equipment will occur. WEC will survey this equipment for potential UO₂ contamination.

5.1.5. Waste Generated as a Part of Demolition of Select Auxiliary Building Operations

The three auxiliary buildings remaining at the Hematite site are buildings 235, 115, and the Sanitary Waste Treatment Plant (SWTP) shed. Building 235 was used for storage of Special Nuclear Material (SNM) during plant operations, and is currently empty. Building 115, the Fire Pump House, had a generator and a fire pump. Building 115 has no history of radioactive material use. Buildings 115 and 235 may be used during future decommissioning operations. Any operations conducted in these buildings will only involve material contained within approved containers, and the operations will be conducted using controlled processes, therefore minimizing the potential for contamination. Prior to demolition, WEC will remove any contaminated materials from these buildings.

The SWTP shed received discharges from sinks, toilets, showers and drinking fountains. The SWTP was also used to receive laundry water (after the water was filtered and held for

sampling) and waste water from the former process water demineralizer system and laboratory sinks. The SWTP shed consists of a series of settling and aeration tanks and an adjacent building that contains data logging and electronic instrumentation, floor drains and an open work area. The portions that have been impacted by licensed activities are limited to the process components that came in contact with waste water, and that have the potential to collect solids that would have settled. Prior to demolition of the SWTP shed, WEC will remove the equipment described above and will separately disposition it.

The above noted buildings were surveyed as a part of the 2009 site radiological characterization program. The radiological survey results estimated that there was a combined total of 55 grams of U-235 on the surfaces of all three of these buildings.

5.2. NRC Staff's Criticality Assessment

The NRC staff's review focused on whether WEC had adequately evaluated NCS risks associated with the proposed waste streams for both normal and credible abnormal conditions. The staff relied upon information in NUREG/CR-6505, Vol. 1, "The Potential for Criticality Following Disposal of Uranium at Low Level Waste Facilities." In NUREG/CR-6505, Vol. 1 the potential for low levels of uranium to concentrate in soil by hydrogeochemical processes such that a criticality event could occur was evaluated. Based upon that evaluation the minimum critical infinite sea concentration for a bounding soil/U-235 medium is 1.4 g U-235/liter. The limit for disposal at USEI is 0.1 gram U-235/liter which is below the minimum critical concentration.

WEC's sample size of the piping surveyed is large. Even if the amount of material has been underestimated, WEC has committed to performing a set of independent measurements to determine the U-235 concentration prior to disposal. Because of the comprehensive sampling performed prior to removal of the piping, the independent sampling performed during the decommissioning operations, and the margin in the NCS limits for the material shipped to USEI, the NRC staff has reasonable assurance that a criticality is not credible from the disposal of the subterranean piping at USEI.

The other waste areas associated with WEC's request, namely, concrete/asphalt, soil underneath the slabs, components remaining after building demolition, miscellaneous equipment as a result of decontamination and decommissioning, and wastes generated as a part of demolition of selected auxiliary building operations generally involve very low contamination levels of fissile material, and thus are not a NCS concern. Therefore, the staff has concluded that a criticality event is not credible for these wastes.

5.1.6. NRC Findings

The NRC staff determined that a criticality event is not credible at the USEI disposal site for the WEC waste described above, because multiple controls related to identifying and segregating waste, as identified in Section 5.1 above, would have to fail before a criticality event could occur. In addition, the NRC staff determined that a criticality event is not credible during the

proposed rail shipments, due to the low concentrations of uranium in the waste to be shipped in the gondola railcars.

MATERIAL CONTROL AND ACCOUNTABILITY

5.2. Westinghouse Assessment

This section of the SER addresses the material control and accountability (MC&A) aspects of WEC's January 16, 2012 request. The staff conducts such a review due to the general reporting and record keeping requirements of subpart B of 10 CFR Part 74, which are applicable to those who possess SNM of 1 g or more of U-235.

WEC Hematite maintains a MC&A program in accordance with the NRC-approved Fundamental Nuclear Material Control Plan (FNMCP) per 10 CFR Part 74, Material Control and Accounting of Special Nuclear Material. The FNMCP contains the reporting requirements of 10 CFR §74.15 associated with DOE/NRC Form 741, Nuclear Material Transaction Report, for the WEC Hematite facility.

WEC's January 16, 2012 request is similar to its May 21, 2009, alternate disposal request. The differences between the two requests are twofold : (1) the type of material; and (2) the total quantity of radionuclides. License Amendment 58 was primarily for soil. The January 16, 2012 request involves concrete/asphalt, piping, miscellaneous equipment and soils. License Amendment 58 involved an average concentration of U-235 of 5.5 pCi/g of while the January 16, 2012 request involved an expected concentration of less than 2.8 pCi/g.

The staff reviewed WEC's January 16, 2012 request and determined that additional information was needed to complete the review, as documented in the staff's RAIs. WEC's RAI responses included its June 19 submittal, which along with the MC&A RAIs are not publicly available because of the sensitive nature of the information.

In its RAI response, WEC confirmed that the proposed waste to be disposed of at USEI is diffuse material as defined in Hematite's Fundamental Nuclear Material Control Plan, dated February 18, 2011. WEC's response also confirmed that it will continue to meet 10 CFR 74.15 requirements to document the transfers of 1 gram or more of SNM to the disposal facility through use of DOE/NRC Form 741, and that USEI will report SNM receipts using its existing account with the Nuclear Material Management & Safeguards System (NMMSS).

5.3. NRC Evaluation and Findings

As noted above WEC will continue to use DOE/NRC Form 741 to document all transfers of 1 gram or more of SNM to NMMSS and USEI will report all SNM receipts, including SNM contained in waste, to NMMSS. Once all of the WEC material is received and disposed of below ground at the USEI facility, USEI may request that its NMMSS account be de-activated, as previously approved. Based upon the above-noted WEC and USEI commitments, the staff has concluded that WEC's alternate disposal request is acceptable with regards to MC&A.

6. PHYSICAL SECURITY

6.1. Assessment

This section of the SER addresses the physical security aspects of WEC's January 16, 2012 request. Based upon the quantity of U-235 associated with this alternate disposal request, the transportation of the materials to USEI and its disposal at USEI has been assessed in accordance with the physical security requirements of 10 CFR Part 73. Section 5.1 of Enclosure 1 to WEC's January 16, 2012 request states that approximately 0.1 Ci of U-235 in total would be shipped to USEI for disposal. This curie amount equates to approximately 45 Kg of U-235.

The NRC staff finds that, from a physical security perspective, the physical security section (Chapter 7) of the SER associated with Hematite Amendment No. 58 presents a bounding analysis for the January 16, 2012 request. The elements of that conclusion are presented below as well as the relationship of the present request to the request associated with Amendment 58.

The physical security issues associated with Amendment 58 remain relevant, and regard: (1) rail shipment of waste that may contain SNM of average enrichment less than 10% U-235 to USEI; (2) transferring such SNM from the gondola cars to trucks for transport to the USEI burial cell; and (3) disposal of the SNM in the burial cells. From a physical security standpoint, any assessment needs to consider the concentration and the enrichment of the SNM being shipped to USEI and handled there, the attractiveness of the form of the SNM being disposed, and the ability of an adversary to efficiently and timely segregate such material after disposal.

In License Amendment 58, the average concentration of U-235 estimated to be shipped to USEI was 5.5 pCi/g. For the U-235 associated with the January 16, 2012 request, the average expected concentration is less than 2.8 pCi/g. The volume of waste associated with the disposal in Amendment 58 and this §20.2002 request is about the same, about 23,000 m³. Therefore, approximately half as much U-235 will be disposed at USEI in this §20.2002 request compared to Amendment 58.

While some of the SNM going to USEI will be HEU, WEC will not be shipping to USEI any HEU that is in a discrete form. Rather, the HEU will be dispersed throughout the waste material being shipped.

In terms of the attractiveness of the SNM for malicious use and its form, the SER for Hematite Amendment No. 58 bounds the analysis here. In neither case is the SNM in a useful form, because it is mixed with dirt found on concrete slabs and asphalt, or is on or in piping and miscellaneous equipment. Thus, the timely and efficient removal of the SNM by an adversary for unauthorized purposes is improbable. The combination of the existing physical security at the USEI site and the effort to identify SNM under such conditions would effectively prevent any opportunities for extracting SNM from its disposal cell.

6.2. NRC Findings

The NRC staff has reviewed the physical security aspects of the January 16, 2012 request. The staff has concluded that there are no physical security concerns associated with the disposal of the Hematite material at the USEI facility. The average U-235 activity levels are low. While SNM will be disposed at USEI, WEC has committed to removing discrete forms of HEU. The SNM will be dispersed throughout the waste material, thereby not lending itself for efficient and timely removal for unauthorized purposes.

7. POTENTIAL FOR RECONCENTRATION

7.1. Assessment

The staff assessed the potential for reconcentration of U-235 in the leachate system at the USEI facility given the half-lives of the SNM and the impact of leachate control system.

In 2008, USEI's permit was modified by the Idaho Department of Environmental Quality (IDEQ) to authorize receipt of specified quantities of SNM, provided that the SNM was made exempt from NRC regulations and licensing requirements. The potential for the generation of leachate is minimized by the site's acceptance requirement that any incoming waste contain no free liquids. Further reducing the potential for leachate generation is the site's location in a desert environment that averages approximately 7.3 inches of precipitation per year with an evaporation rate of approximately 42 inches per year.

The potential to generate leachate is further reduced by the USEI facility's design to completely encapsulate the waste in a low permeability (1×10^{-7} cm/sec) cover system. Requirements for the construction of a waste cell include a base layer of compacted clay three-feet thick overlain by a composite liner with a sump to collect any leachate that might be generated. The composite liner is overlain by a 30-inch soil layer as a protection barrier for the liner. Waste placed in the cell is compacted to minimize the potential for future subsidence and when the cell is full is overlain by a low permeability multi-layer cap 11.8 feet thick that includes nine feet of non-radiological material.

7.2. NRC Findings

As a result of design features such as a low permeability cover, the base layer of compacted clay with a composite liner as an overlay and the compaction of the waste upon burial, the staff has concluded that reconcentration in the leachate system should not be an issue with respect to the disposal of the SNM at USEI.

8. LICENSE CHANGES

Approval of WEC's January 16, 2012 request will be effectuated by issuing License Amendment No. 60 to the Hematite License including the following changes to Hematite License Conditions.

The first three changes are administrative in nature. The first administrative change arises from a previous numbering error (the present license goes from License Condition 10 to License Condition 12). Therefore, after License Condition 10, all License Conditions will be renumbered accordingly.

The second administrative change involves Item 9 of the Hematite License. Presently, the Authorized Uses involve Items A through E as described in the August 12, 2009 Decommissioning Plan and associated supporting documents noted in Hematite Decommissioning Plan SER (ADAMS Accession No. ML112101630) and July 5, 2011, License Application (ADAMS Accession No. ML111880290). When the Decommissioning Plan was approved in Amendment 57 to the Hematite License, Item 9 should have indicated that Authorized Use was for Items A through H. This license amendment corrects that omission.

The third administrative change more definitively defines the appropriate Westinghouse License Application and the July 5, 2011 Westinghouse letter by referring to the Westinghouse document number and providing the NRC's ADAMS numbers associated with the documents. Since both documents are part of the same submittal and have the same ADAMS number, they were listed as one reference.

The fourth change to the Hematite license revises License Condition 15 to list the documents referenced in this SER and the SER for License Amendment 58.

The fifth change is revises License Condition 17 to include the total volume of waste material that WEC is authorized to ship to USEI for disposal there and the total amount of Tc-99. This includes the 22,809 m³ of soils and associated debris covered by the approval of WEC's May 2009 alternate disposal request, and the 23,000 m³ of concrete/asphalt, piping, soil and miscellaneous equipment covered by the approval of WEC's January 16, 2012 request. Therefore, the revisions to Item 9 and to License Conditions 15 and 17 would be as follows:

9. Authorized Use: Items A through H. Uses as described in August 12, 2009 Decommissioning Plan and associated supporting documents noted in Hematite Decommissioning Plan SER (ADAMS Accession No. ML112101630) and July 5, 2011 License Application (ADAMS Accession No. ML111880290).

15. Except as specifically provided otherwise in this license, the licensee shall conduct its program in accordance with the statements, representations, and procedures contained in the documents, including any enclosures, listed below. The NRC's regulations shall govern unless the statements, representations, and procedures in the licensee's application and correspondence are more restrictive than the regulations.

- a. Westinghouse HEM-11-96, "Final Supplemental Response to NRC Request for Additional Information on the Hematite Decommissioning Plan and Related Revision to a Pending Licensing Action", July 5, 2011. (ADAMS Accession Nos. ML111880290 and ML111880292)
 - b. Documents identified in Chapter 1 of NRC Decommissioning Plan SER. (ADAMS Accession No. ML112101630)
 - c. Westinghouse HEM-11-56, "Evaluation of Technetium-99 Under the Process Buildings", May 5, 2011. (ADAMS Accession No. ML111260624)
 - d. Documents identified in the NRC's 10CFR20.2002 SERs associated with Amendment Nos. 58 and 60. (ADAMS Accession Nos. ML111441087 and ML12158A401)
17. Pursuant to 10 CFR 20.2002, the licensee may dispose of solid materials (22,809 m³ of soils and associated debris and 23,000 m³ of concrete/asphalt, piping, soil and miscellaneous equipment) provided the total inventory of Tc-99 based on the average concentration and total mass shipped remains below 1.3 Ci or 2.05 Ci based upon the 95th upper confidence limit as waste at the U.S. Ecology Idaho facility in Grand View, ID. Pursuant to 10 CFR 30.11 and 10 CFR 70.17, this material is exempt from the requirements in 10 CFR 30.3 and 10 CFR 70.3.

9. CONCLUSIONS

On January 16, 2012, WEC requested that the NRC approve alternate disposal, in accordance with 10 CFR §20.2002, of specified low-activity radioactive materials from the HDP. These waste materials total approximately 23,000 m³ of concrete/asphalt, piping, soil and miscellaneous equipment, and contain low concentrations of source, SNM and byproduct material contaminants. WEC plans to ship these materials by rail to USEI RCRA Subtitle C disposal facility near Grand View, Idaho.

Activities and potential doses associated with transportation, waste handling and disposal have been evaluated in reviewing this 10 CFR §20.2002 application. The staff has determined that WEC has provided an adequate description of the waste to be disposed of, including the physical and chemical properties important to risk evaluation, and the proposed manner and conditions of waste disposal.

The staff has determined that WEC's proposed statistical evaluation, sampling plan, QA/QC program, and contingency plans are acceptable, and demonstrate that its proposed disposal will not result in a dose to individual members of the public exceeding a few millirem per year.

Independent review of the post-closure and intruder scenarios using RESRAD estimated that the maximum projected dose per year over a period of 1,000 years is within "a few millirem". A conservative bounding analysis conducted by the staff yielded doses less than the Part 20 annual dose limit of 1.0 mSv/yr (100 mrem/yr) to members of the public. The projected doses to individual USEI workers have been conservatively estimated and demonstrate that the proposed disposal will not result in a dose to members of the public exceeding a few millirem per year.

In addition, because this 10 CFR §20.2002 application involves SNM, nuclear criticality safety, material control and accounting, and physical security assessments were performed.

The staff finds that this proposed action will not significantly impact the annual cumulative dose from all exempted and naturally occurring radioactive material at the USEI disposal facility. This finding is based upon the dose evaluations discussed in Section 3 above.

Further, in accordance with the provisions of 10 CFR §30.11 and 10 CFR §70.17, the NRC may, upon application by an interested person or upon its own initiative, grant such exemptions from the requirements of the regulations in those parts of Title 10, Chapter 1 of the Code of Federal Regulations as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Based on the above analyses, the staff concludes that: (1) this material authorized for disposal poses no danger to public health and safety; (2) the authorized disposal does not involve activities that could potentially impact the common defense and security of the United States; and (3) it is in the public interest to dispose of wastes in a controlled environment, such as that provided by the US Ecology Idaho facility located in Grand View, ID. Therefore, to the extent that the waste authorized for disposal contains byproduct material and SNM that would otherwise be

licensable, the staff concludes that the receipt and possession of this material by USEI is exempt from NRC licensing requirements in 10 CFR §30.3, and §70.3, respectively.

10. REFERENCES

For convenience, the references have been organized according to their WEC, LLC identification number.

HDP-TBD-WM-906. "Characterization Data Summary in Support of Additional USEI Alternate Disposal Request, Revision 0, Westinghouse Electric Company LLC, Hematite Decommissioning Project. January 19, 2012. (ADAMS Accession No. ML12017A188)

HDP-TBD-WM-906. "Characterization Data Summary in Support of Additional USEI Alternate Disposal Request, Revision 1, Westinghouse Electric Company LLC, Hematite Decommissioning Project. July 24, 2012. (ADAMS Accession No. ML12209A201)

HDP-TBD-WM-908. "Safety Assessment for Additional Hematite Project Waste at USEI, Revision 2," Westinghouse Electric Company LLC, Hematite Decommissioning Project October 17, 2012. (ADAMS Accession No. ML12293A029)

HEM-09-94. "Decommissioning Plan and Revision to License Application," Westinghouse Electric Company LLC, Hematite Decommissioning Project. August 12, 2009. (ADAMS Accession No. ML092330136)

HEM-09-52. "Westinghouse Electric Company, LLC - Request for Alternate Disposal Approval and Exemptions for Specific Hematite Decommissioning Project Waste," Westinghouse Electric Company LLC, Hematite Decommissioning Project, May 21, 2009. (ADAMS Accession No. ML091480071)

HEM-09-94. Hematite Decommissioning Project Report DO-08-005, Rev. 0, "Historical Site Assessment," Westinghouse Electric Company LLC, Hematite Decommissioning Project. July 2009. (ADAMS Accession Nos. ML092870417 and ML092870418)

HEM-12-2. "Request for Additional Alternate Disposal Approval and Exemptions for Specific Hematite Decommissioning Project Waste at US Ecology Idaho," Westinghouse Electric Company LLC, Hematite Decommissioning Project, January 16, 2012. (ADAMS Accession Nos. ML12017A188, ML12017A189, ML12017A190)

HEM-12-67. "Partial Response to NRC Requests for Additional Information Dated May 1, 2012 on the January 16, 2012 Hematite 20.2002 Alternate Disposal Request. Westinghouse Electric Company LLC, Hematite Decommissioning Project. June 19, 2012. (ADAMS Accession No. ML121740265)

HEM-12-88. "Final Responses to NRC Requests for Additional Information Dated May 1, 2012 on the January 16, 2012 Hematite 20.2002 Alternate Disposal Request." Westinghouse Electric Company LLC, Hematite Decommissioning Project. July 24, 2012. (ADAMS Accession Nos. ML12209A200 and ML12209A201)

HEM-12-121. "Further Information for the Final Response Dated July 24, 2012 to NRC Requests for Additional Information Dated May 1, 2012 on the January 16, 2012, Hematite Alternate Disposal Request," October 17, 2012. (ADAMS Accession No. ML12293A029)

HDP-TBD-WM-901. "Scaling Factors for Radioactive Waste Associated with the Above Slab Portion of the Process Buildings. Westinghouse Electric Company LLC Hematite Decommissioning Project. March 28, 2012. (ADAMS Accession No. ML12090A191)

HEM-09-146. "WEC Response to Request for Additional Information - Alternate Waste Disposal," Westinghouse Electric Company LLC, Hematite Decommissioning Project, December 29, 2009. (ADAMS Accession No. ML100320540)

HEM-I0-38. "Additional Information for Alternate Waste Disposal Authorization and Exemption," Westinghouse Electric Company LLC, Hematite Decommissioning Project, March 31, 2010 (ADAMS Accession No. ML100950397)

Nuclear Criticality Safety Associates, NSA-TR-09-08, Rev. 1, "NCSA of the Sub-Surface Structure Decommissioning at the Hematite Site", November 2011. (ADAMS Accession No. ML12293A029)

Nuclear Criticality Safety Associates, NSA-TR-HDP-11-11, Rev. 1, "Nuclear Criticality Safety Assessment of the US Ecology Idaho (USEI) Site for the Land Fill Disposal of Additional Decommissioning Waste from the Hematite Site", July 2012. (ADAMS Accession No. ML12209A200)

U. S. Environmental Protection Agency, Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," September 1988.

U.S. Nuclear Regulatory Commission, NUREG/CR-6505, Vol. 1, "The Potential for Criticality Following Disposal of Uranium at Low Level Waste Facilities." June 1997. U.S. Nuclear Regulatory Commission, NUREG/CR-0782, Volume 4, "Draft Environmental Impact Statement on 10 CFR Part 61 Licensing Requirements for Land Disposal of Radioactive Waste: Appendices G-Q," September 1981. (ADAMS Accession No. ML052590354)

U.S. Nuclear Regulatory Commission, NUREG/CR-4370, "Update of Part 61 Impacts Analysis Methodology, Vol. 2". January 1986. (ADAMS Accession No. ML100250917)

U.S. Nuclear Regulatory Commission, Hematite Amendment No. 58. "US NRC Safety Evaluation Report Request for Alternate Disposal Approval and Exemptions for Specific Hematite Decommissioning Project Waste at US Ecology's Idaho Facility," October 27, 2011. (ADAMS Accession No. ML111441087)

U.S. Nuclear Regulatory Commission, "Hematite Amendment No. 58 Transmittal Letter and License," October 27, 2011. (ADAMS Accession Nos. ML112560105 and ML112560193)

U.S. Nuclear Regulatory Commission, SECY-07-0600, "Basis and Justification for Approval Process for 10 CFR 20.2002 Authorizations and Options for Change," March 27, 2007. (ADAMS Accession No. ML070220045)

U.S. Nuclear Regulatory Commission, NUREG-1757, Vol. 1, Rev. 2 "Consolidated Decommissioning Guidance: Decommissioning Process for Material Licensees," September 2006. (ADAMS Accession No. ML070390074)

U.S. Nuclear Regulatory Commission, NUREG-1757, Vol. 2, Rev. 1 "Consolidated Decommissioning Guidance: Characterization, Survey, and Determination of Radiological Criteria," September 2006

U.S. Nuclear Regulatory Commission, "R. Copp Letter Requesting Additional Information on Hematite 20.2002 Alternate Disposal Request," May 1, 2012

U.S. Nuclear Regulatory Commission, "NRC Request for Additional Information during 6/21/10 Teleconference between NRC and WEC Regarding Westinghouse Electric Corporation Hematite Project 20.2002 Soil Exemption Request," Westinghouse Electric Company LLC, Hematite Decommissioning Project, Jun 25, 2010. (ADAMS Accession No. ML110560334)

U.S. Ecology Idaho, "Request for Exemptions under 10CFR30.11 and 10CFR70.17 for Alternate Disposal of Wastes from Hematite Decommissioning Project under 10CFR20.2002," October 4, 2012. (ADAMS Accession No. ML12313A014)

ATTACHMENT 2

USEI Part B Permit EPA ID IDD073114654

Latest Revision Date: February 26, 2013 - Part C.3

Waste Acceptance Criteria (5 pages)

C.3 WASTE ACCEPTANCE CRITERIA

C.3.1 Pre-acceptance Review

The preacceptance protocol has been designed to ensure that only hazardous and radioactive material that can be properly and safely stored, treated and/or disposed of by USEI are approved for receipt at the facility. A two-step approach is taken by USEI. The first step is the chemical and/or radiological and physical characterization of the candidate waste stream by the generator. The second step is the preacceptance evaluation performed by USEI to determine the acceptability of the waste for receipt at the facility. Figure C-2 presents a logic diagram of the preacceptance protocol that is utilized at the facility.

C.3.2 Radioactive Material Waste Acceptance Criteria

The following waste acceptance criteria are established for accepting radiological contaminated waste material that is generally or specifically exempted from regulation by the Nuclear Regulatory Commission (NRC) or an Agreement State under the Atomic Energy Act of 1954 ("AEA"), as amended. Material may also be accepted if it is not regulated or licensed by the NRC 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 based on statute or regulation or specifically approved by the NRC or an Agreement State 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 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 Table C.1 through C.2 radioactive material waste receipts showing volumes and radionuclide concentrations disposed at the USEI site on a quarterly basis. USEI will also submit a Table C.3 through C.4b annual report of exempted products devices, materials or items within 60 (sixty) days of year end (December 31st). The annual report will provide total volumes or mass of isotopes and total activity by isotope listing the activity of each radionuclide disposed during the preceding year, and the cumulative total of activity for each radionuclide disposed at the facility. The report will include an updated analysis of the impact on the facility performance assessment.

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.4b 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 ²³⁸U, ²³⁵U, ²³⁴U; ^{234m}Pa, ²³¹Th

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 ²²⁶Ra 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.

¹ 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).

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

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 <i>in vivo</i> diagnosis of humans	¹⁴ C, one µCi per capsule
40.13(a)	Unimportant quantity of source material: see Table C.1	≤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.

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.

ATTACHMENT 3

SPFM Annualized Dose Assessment Data Summaries for Truck and Rail Shipment Scenarios

USEI T&D Dose Calculation Worksheet
(for US NRC 20.2002 Alternate Disposal Requests)

Customer: Studsvik SPFM - Average Annual Inventory

Project: Alternate Disposal Request and License Amendment

Scenario: IMC Shipments by Rail

I. Waste Stream Information

Volume of waste (CF):	489,000
Waste Density (lb/CF):	27.5
Waste Density (g/cc):	0.44
Waste Mass (lbs):	1.34E+07
Waste Mass (g):	6.10E+09

II. Radionuclides of Concern - Provided by Customer

Isotope	Waste Stream Concentration, mass (pCi/g)	Waste Stream Concentration, volume (pCi/cm ³)	uCi/cm ³ for Microshield ^{1,2}	Baseline RESRAD Input ² (Conc. over volume of "CZ" (pCi/g)) ³	Conc. RESRAD Scenario ³ (pCi/g)	Ci/m ³ (for Intruder Scenario)	pCi/g (for Conc. Intruder Scenarios)	Ci/m ³ (for Conc. Intruder Scenarios)
Ag-108m	5.00E+00	2.20E+00	2.20E-06	6.85E-03	1.82E-01	2.20E-06	1.50E+01	6.60E-06
Ag-110m	2.50E+01	1.10E+01	1.10E-05	3.43E-02	9.10E-01	1.10E-05	7.50E+01	3.30E-05
Am-241	5.00E-01	2.20E-01	2.20E-07	6.85E-04	1.82E-02	2.20E-07	1.50E+00	6.60E-07
Au-195	3.50E+00	1.54E+00	1.54E-06	4.80E-03	1.27E-01	1.54E-06	1.05E+01	4.62E-06
Ba-133	3.00E-01	1.32E-01	1.32E-07	4.11E-04	1.09E-02	1.32E-07	9.00E-01	3.96E-07
Be-7	2.00E+01	8.80E+00	8.80E-06	2.74E-02	7.28E-01	8.80E-06	6.00E+01	2.64E-05
C-14	2.50E+01	1.10E+01	1.10E-05	3.43E-02	9.10E-01	1.10E-05	7.50E+01	3.30E-05
Ce-139	1.00E+00	4.40E-01	4.40E-07	1.37E-03	3.64E-02	4.40E-07	3.00E+00	1.32E-06
Ce-141	1.15E+01	5.06E+00	5.06E-06	1.58E-02	4.19E-01	5.06E-06	3.45E+01	1.52E-05
Ce-144	1.80E+02	7.92E+01	7.92E-05	2.47E-01	6.55E+00	7.92E-05	5.40E+02	2.38E-04
Cm-242	1.00E-01	4.40E-02	4.40E-08	1.37E-04	3.64E-03	4.40E-08	3.00E-01	1.32E-07
Cm-243	1.30E+00	5.72E-01	5.72E-07	1.78E-03	4.73E-02	5.72E-07	3.90E+00	1.72E-06
Cm-244	5.00E-01	2.20E-01	2.20E-07	6.85E-04	1.82E-02	2.20E-07	1.50E+00	6.60E-07
Cm-245	5.00E+00	2.20E+00	2.20E-06	6.85E-03	1.82E-01	2.20E-06	1.50E+01	6.60E-06
Co-57	1.20E+01	5.28E+00	5.28E-06	1.64E-02	4.37E-01	5.28E-06	3.60E+01	1.58E-05
Co-58	2.00E+02	8.80E+01	8.80E-05	2.74E-01	7.28E+00	8.80E-05	6.00E+02	2.64E-04
Co-60	6.50E+02	2.86E+02	2.86E-04	8.91E-01	2.37E+01	2.86E-04	1.95E+03	8.58E-04
Cr-51	5.50E+01	2.42E+01	2.42E-05	7.54E-02	2.00E+00	2.42E-05	1.65E+02	7.26E-05
Cs-134	1.75E+02	7.70E+01	7.70E-05	2.40E-01	6.37E+00	7.70E-05	5.25E+02	2.31E-04
Cs-137	5.00E+02	2.20E+02	2.20E-04	6.85E-01	1.82E+01	2.20E-04	1.50E+03	6.60E-04
Eu-152	7.00E+00	3.08E+00	3.08E-06	9.60E-03	2.55E-01	3.08E-06	2.10E+01	9.24E-06
Eu-154	2.50E+00	1.10E+00	1.10E-06	3.43E-03	9.10E-02	1.10E-06	7.50E+00	3.30E-06
Eu-155	4.00E+00	1.76E+00	1.76E-06	5.48E-03	1.46E-01	1.76E-06	1.20E+01	5.28E-06
Fe-55	1.00E+03	4.40E+02	4.40E-04	1.37E+00	3.64E+01	4.40E-04	3.00E+03	1.32E-03
Fe-59	8.00E+00	3.52E+00	3.52E-06	1.10E-02	2.91E-01	3.52E-06	2.40E+01	1.06E-05
H-3	3.25E+02	1.43E+02	1.43E-04	4.45E-01	1.18E+01	1.43E-04	9.75E+02	4.29E-04
I-125	1.00E-01	4.40E-02	4.40E-08	1.37E-04	3.64E-03	4.40E-08	3.00E-01	1.32E-07
I-129	3.00E-01	1.32E-01	1.32E-07	4.11E-04	1.09E-02	1.32E-07	9.00E-01	3.96E-07
I-131	1.80E+01	7.92E+00	7.92E-06	2.47E-02	6.55E-01	7.92E-06	5.40E+01	2.38E-05
Mn-54	8.00E+01	3.52E+01	3.52E-05	1.10E-01	2.91E+00	3.52E-05	2.40E+02	1.06E-04
Na-22	2.00E-01	8.80E-02	8.80E-08	2.74E-04	7.28E-03	8.80E-08	6.00E-01	2.64E-07
Nb-94	4.00E+00	1.76E+00	1.76E-06	5.48E-03	1.46E-01	1.76E-06	1.20E+01	5.28E-06
Nb-95	2.50E+01	1.10E+01	1.10E-05	3.43E-02	9.10E-01	1.10E-05	7.50E+01	3.30E-05
Ni-59	1.00E+02	4.40E+01	4.40E-05	1.37E-01	3.64E+00	4.40E-05	3.00E+02	1.32E-04
Ni-63	9.25E+02	4.07E+02	4.07E-04	1.27E+00	3.37E+01	4.07E-04	2.78E+03	1.22E-03
Pu-238	5.00E+00	2.20E+00	2.20E-06	6.85E-03	1.82E-01	2.20E-06	1.50E+01	6.60E-06
Pu-239	3.00E-01	1.32E-01	1.32E-07	4.11E-04	1.09E-02	1.32E-07	9.00E-01	3.96E-07
Pu-240	3.00E-01	1.32E-01	1.32E-07	4.11E-04	1.09E-02	1.32E-07	9.00E-01	3.96E-07
Pu-241	1.50E+01	6.60E+00	6.60E-06	2.06E-02	5.46E-01	6.60E-06	4.50E+01	1.98E-05
Pu-242	2.00E-01	8.80E-02	8.80E-08	2.74E-04	7.28E-03	8.80E-08	6.00E-01	2.64E-07
Ra-226 ⁴	1.00E+01	4.40E+00	4.40E-06	1.37E-02	3.64E-01	4.40E-06	3.00E+01	1.32E-05
Ru-103	1.50E+00	6.60E-01	6.60E-07	2.06E-03	5.46E-02	6.60E-07	4.50E+00	1.98E-06
Ru-106	1.00E+01	4.40E+00	4.40E-06	1.37E-02	3.64E-01	4.40E-06	3.00E+01	1.32E-05
Sb-124	8.00E+00	3.52E+00	3.52E-06	1.10E-02	2.91E-01	3.52E-06	2.40E+01	1.06E-05
Sb-125	1.00E+02	4.40E+01	4.40E-05	1.37E-01	3.64E+00	4.40E-05	3.00E+02	1.32E-04
Sn-113	2.00E+00	8.80E-01	8.80E-07	2.74E-03	7.28E-02	8.80E-07	6.00E+00	2.64E-06
Sr-89	1.70E+01	7.48E+00	7.48E-06	2.33E-02	6.19E-01	7.48E-06	5.10E+01	2.24E-05
Sr-90	1.40E+01	6.16E+00	6.16E-06	1.92E-02	5.09E-01	6.16E-06	4.20E+01	1.85E-05
Tc-99	4.00E+00	1.76E+00	1.76E-06	5.48E-03	1.46E-01	1.76E-06	1.20E+01	5.28E-06
Te-123	4.00E+00	1.76E+00	1.76E-06	5.48E-03	1.46E-01	1.76E-06	1.20E+01	5.28E-06
Th-228	1.00E+00	4.40E-01	4.40E-07	1.37E-03	3.64E-02	4.40E-07	3.00E+00	1.32E-06
Th-232 ⁴	4.00E+00	1.76E+00	1.76E-06	5.48E-03	1.46E-01	1.76E-06	1.20E+01	5.28E-06

Attachment 3
Exhibit A

U-233 ⁵	8.00E+00	3.52E+00	3.52E-06	1.10E-02	2.91E-01	3.52E-06	2.40E+01	1.06E-05
U-234 ⁵	1.90E+02	8.36E+01	8.36E-05	2.60E-01	6.91E+00	8.36E-05	5.70E+02	2.51E-04
U-235 ⁵	1.00E+01	4.40E+00	4.40E-06	1.37E-02	3.64E-01	4.40E-06	3.00E+01	1.32E-05
U-238 ⁵	1.90E+02	8.36E+01	8.36E-05	2.60E-01	6.91E+00	8.36E-05	5.70E+02	2.51E-04
Natural Uranium ⁶	2.25E+02	9.90E+01	9.90E-05	3.08E-01	8.19E+00	9.90E-05	6.75E+02	2.97E-04
Zn-65	1.15E+02	5.06E+01	5.06E-05	1.58E-01	4.19E+00	5.06E-05	3.45E+02	1.52E-04
Zr-95	3.00E+01	1.32E+01	1.32E-05	4.11E-02	1.09E+00	1.32E-05	9.00E+01	3.96E-05

III. Summary of Potentially Exposed Workers

a) Total Project Dose

Job Function	No. Workers	Waste Contact Time (hr)	External Exposure Rate ⁷ (mR/hr)	Internal Dose Rate ⁹ (mrem/hr)	Distance (m)	No. Required Trips or Reps ¹⁰	Total External Dose (mrem)	Total Internal Dose (mrem)	Total Project Dose (mrem)
FED Truck Drivers ⁸	6	0.09	8.86E-02	0.00E+00	4.0	561	7.53E-01	0.00E+00	7.53E-01
Long-Haul Truck Drivers ⁸	6	32.73	8.86E-02	0.00E+00	4.0	0	0.00E+00	0.00E+00	0.00E+00
RTF Equipment Operator	4	0.25	9.51E-02	0.00E+00	4.9	561	3.33E+00	0.00E+00	3.33E+00
Truck/IMC Surveyors	8	0.08	7.71E-01	0.00E+00	1.0	561	4.33E+00	0.00E+00	4.33E+00
BED Truck Drivers	10	0.75	8.86E-02	0.00E+00	4.0	561	3.73E+00	0.00E+00	3.73E+00
Landfill Cell Operators	4	0.25	1.16E-01	2.40E-02	2.0	135	9.77E-01	2.02E-01	1.18E+00

IV. RESRAD Modelling Results

A. Baseline Case 2.42E-01 mrem (@ t= 326.1 yrs) Baseline DF = 1.37E-03
Assumption(s): Project waste is distributed evenly over entire contents of USEI waste cells (88,221 m² area, 33.6 m depth)

B. Concentrated Case 6.42E+00 mrem (@ t= 326.1 yrs) Conc. DF = 3.64E-02
Assumption(s): Project waste is concentrated in smaller portion of landfill. Assumes entire annual SPFM activity inventory arrives at USEI in 4-months (instead of 12 months).

V. Notes & Assumptions

- 1 Microshield model inputs for transportation and handling tasks are adjusted to units of uCi/cc using as-shipped density
- 2 Microshield model inputs for landfill workers is adjusted to units of uCi/cc using in situ compacted landfill density of 1.5 g/cc.
- 3 The waste stream concentrations are multiplied by the Dilution Factors in Section IV for each scenario (Baseline, Concentrated)
- 4 Th-232 and Ra-226 assumed to be in complete equilibrium with their entire progeny chains
- 5 Individual uranium isotopes are analyzed as they are reported with no additional decay.
- 6 Natural uranium is assumed to be in complete equilibrium with all decay progeny for Microshield runs.
Natural uranium isotopic concentrations added to other U-234, U-235, and U-238 concentrations for RESRAD runs.
- 7 The External Exposure Rates are the sum of two separate Microshield runs. One for natural uranium in equilibrium and another for all other Radionuclides of Concern shown in Section II. U-Nat was run independently to ensure equilibrium with all progeny.
- 8 Dose to SPFM Long-Haul Truck Drivers is not evaluated since these personnel are trained radiation workers monitored under a licensed RPP.

USEI T&D Dose Calculation Worksheet
(for US NRC 20.2002 Alternate Disposal Requests)

Customer: Studsvik SPFM - Average Annual Inventory

Project: Alternate Disposal Request and License Amendment

Shipment Scenario: IMC Shipments by Truck

I. Waste Stream Information

Volume of waste (CF):	489,000
Waste Density (lb/CF):	27.5
Waste Density (g/cc):	0.44
Waste Mass (lbs):	1.34E+07
Waste Mass (g):	6.10E+09

II. Radionuclides of Concern - Provided by Customer

Isotope	Waste Stream Concentration, mass (pCi/g)	Waste Stream Concentration, volume (pCi/cm ³)	uCi/cm ³ for Microshield ^{1,2}	Baseline RESRAD Input ² (Conc. over volume of "CZ" (pCi/g)) ³	Conc. RESRAD Scenario ³ (pCi/g)	Ci/m ³ (for Baseline Intruder Scenarios)	pCi/g (for Conc. Intruder Scenarios)	Ci/m ³ (for Conc. Intruder Scenarios)
Ag-108m	5.00E+00	2.20E+00	2.20E-06	6.85E-03	1.82E-01	2.20E-06	1.50E+01	6.60E-06
Ag-110m	2.50E+01	1.10E+01	1.10E-05	3.43E-02	9.10E-01	1.10E-05	7.50E+01	3.30E-05
Am-241	5.00E-01	2.20E-01	2.20E-07	6.85E-04	1.82E-02	2.20E-07	1.50E+00	6.60E-07
Au-195	3.50E+00	1.54E+00	1.54E-06	4.80E-03	1.27E-01	1.54E-06	1.05E+01	4.62E-06
Ba-133	3.00E-01	1.32E-01	1.32E-07	4.11E-04	1.09E-02	1.32E-07	9.00E-01	3.96E-07
Be-7	2.00E+01	8.80E+00	8.80E-06	2.74E-02	7.28E-01	8.80E-06	6.00E+01	2.64E-05
C-14	2.50E+01	1.10E+01	1.10E-05	3.43E-02	9.10E-01	1.10E-05	7.50E+01	3.30E-05
Ce-139	1.00E+00	4.40E-01	4.40E-07	1.37E-03	3.64E-02	4.40E-07	3.00E+00	1.32E-06
Ce-141	1.15E+01	5.06E+00	5.06E-06	1.58E-02	4.19E-01	5.06E-06	3.45E+01	1.52E-05
Ce-144	1.80E+02	7.92E+01	7.92E-05	2.47E-01	6.55E+00	7.92E-05	5.40E+02	2.38E-04
Cm-242	1.00E-01	4.40E-02	4.40E-08	1.37E-04	3.64E-03	4.40E-08	3.00E-01	1.32E-07
Cm-243	1.30E+00	5.72E-01	5.72E-07	1.78E-03	4.73E-02	5.72E-07	3.90E+00	1.72E-06
Cm-244	5.00E-01	2.20E-01	2.20E-07	6.85E-04	1.82E-02	2.20E-07	1.50E+00	6.60E-07
Cm-245	5.00E+00	2.20E+00	2.20E-06	6.85E-03	1.82E-01	2.20E-06	1.50E+01	6.60E-06
Co-57	1.20E+01	5.28E+00	5.28E-06	1.64E-02	4.37E-01	5.28E-06	3.60E+01	1.58E-05
Co-58	2.00E+02	8.80E+01	8.80E-05	2.74E-01	7.28E+00	8.80E-05	6.00E+02	2.64E-04
Co-60	6.50E+02	2.86E+02	2.86E-04	8.91E-01	2.37E+01	2.86E-04	1.95E+03	8.58E-04
Cr-51	5.50E+01	2.42E+01	2.42E-05	7.54E-02	2.00E+00	2.42E-05	1.65E+02	7.26E-05
Cs-134	1.75E+02	7.70E+01	7.70E-05	2.40E-01	6.37E+00	7.70E-05	5.25E+02	2.31E-04
Cs-137	5.00E+02	2.20E+02	2.20E-04	6.85E-01	1.82E+01	2.20E-04	1.50E+03	6.60E-04
Eu-152	7.00E+00	3.08E+00	3.08E-06	9.60E-03	2.55E-01	3.08E-06	2.10E+01	9.24E-06
Eu-154	2.50E+00	1.10E+00	1.10E-06	3.43E-03	9.10E-02	1.10E-06	7.50E+00	3.30E-06
Eu-155	4.00E+00	1.76E+00	1.76E-06	5.48E-03	1.46E-01	1.76E-06	1.20E+01	5.28E-06
Fe-55	1.00E+03	4.40E+02	4.40E-04	1.37E+00	3.64E+01	4.40E-04	3.00E+03	1.32E-03
Fe-59	8.00E+00	3.52E+00	3.52E-06	1.10E-02	2.91E-01	3.52E-06	2.40E+01	1.06E-05
H-3	3.25E+02	1.43E+02	1.43E-04	4.45E-01	1.18E+01	1.43E-04	9.75E+02	4.29E-04
I-125	1.00E-01	4.40E-02	4.40E-08	1.37E-04	3.64E-03	4.40E-08	3.00E-01	1.32E-07
I-129	3.00E-01	1.32E-01	1.32E-07	4.11E-04	1.09E-02	1.32E-07	9.00E-01	3.96E-07
I-131	1.80E+01	7.92E+00	7.92E-06	2.47E-02	6.55E-01	7.92E-06	5.40E+01	2.38E-05
Mn-54	8.00E+01	3.52E+01	3.52E-05	1.10E-01	2.91E+00	3.52E-05	2.40E+02	1.06E-04
Na-22	2.00E-01	8.80E-02	8.80E-08	2.74E-04	7.28E-03	8.80E-08	6.00E-01	2.64E-07
Nb-94	4.00E+00	1.76E+00	1.76E-06	5.48E-03	1.46E-01	1.76E-06	1.20E+01	5.28E-06
Nb-95	2.50E+01	1.10E+01	1.10E-05	3.43E-02	9.10E-01	1.10E-05	7.50E+01	3.30E-05
Ni-59	1.00E+02	4.40E+01	4.40E-05	1.37E-01	3.64E+00	4.40E-05	3.00E+02	1.32E-04
Ni-63	9.25E+02	4.07E+02	4.07E-04	1.27E+00	3.37E+01	4.07E-04	2.78E+03	1.22E-03
Pu-238	5.00E+00	2.20E+00	2.20E-06	6.85E-03	1.82E-01	2.20E-06	1.50E+01	6.60E-06
Pu-239	3.00E-01	1.32E-01	1.32E-07	4.11E-04	1.09E-02	1.32E-07	9.00E-01	3.96E-07
Pu-240	3.00E-01	1.32E-01	1.32E-07	4.11E-04	1.09E-02	1.32E-07	9.00E-01	3.96E-07
Pu-241	1.50E+01	6.60E+00	6.60E-06	2.06E-02	5.46E-01	6.60E-06	4.50E+01	1.98E-05
Pu-242	2.00E-01	8.80E-02	8.80E-08	2.74E-04	7.28E-03	8.80E-08	6.00E-01	2.64E-07
Ra-226 ⁴	1.00E+01	4.40E+00	4.40E-06	1.37E-02	3.64E-01	4.40E-06	3.00E+01	1.32E-05
Ru-103	1.50E+00	6.60E-01	6.60E-07	2.06E-03	5.46E-02	6.60E-07	4.50E+00	1.98E-06
Ru-106	1.00E+01	4.40E+00	4.40E-06	1.37E-02	3.64E-01	4.40E-06	3.00E+01	1.32E-05
Sb-124	8.00E+00	3.52E+00	3.52E-06	1.10E-02	2.91E-01	3.52E-06	2.40E+01	1.06E-05
Sb-125	1.00E+02	4.40E+01	4.40E-05	1.37E-01	3.64E+00	4.40E-05	3.00E+02	1.32E-04
Sn-113	2.00E+00	8.80E-01	8.80E-07	2.74E-03	7.28E-02	8.80E-07	6.00E+00	2.64E-06
Sr-89	1.70E+01	7.48E+00	7.48E-06	2.33E-02	6.19E-01	7.48E-06	5.10E+01	2.24E-05
Sr-90	1.40E+01	6.16E+00	6.16E-06	1.92E-02	5.09E-01	6.16E-06	4.20E+01	1.85E-05
Tc-99	4.00E+00	1.76E+00	1.76E-06	5.48E-03	1.46E-01	1.76E-06	1.20E+01	5.28E-06
Te-123	4.00E+00	1.76E+00	1.76E-06	5.48E-03	1.46E-01	1.76E-06	1.20E+01	5.28E-06
Th-228	1.00E+00	4.40E-01	4.40E-07	1.37E-03	3.64E-02	4.40E-07	3.00E+00	1.32E-06
Th-232 ⁴	4.00E+00	1.76E+00	1.76E-06	5.48E-03	1.46E-01	1.76E-06	1.20E+01	5.28E-06

Attachment 3
Exhibit B

U-233 ⁵	8.00E+00	3.52E+00	3.52E-06	1.10E-02	2.91E-01	3.52E-06	2.40E+01	1.06E-05
U-234 ⁵	1.90E+02	8.36E+01	8.36E-05	2.60E-01	6.91E+00	8.36E-05	5.70E+02	2.51E-04
U-235 ⁵	1.00E+01	4.40E+00	4.40E-06	1.37E-02	3.64E-01	4.40E-06	3.00E+01	1.32E-05
U-238 ⁵	1.90E+02	8.36E+01	8.36E-05	2.60E-01	6.91E+00	8.36E-05	5.70E+02	2.51E-04
Natural Uranium ⁶	2.25E+02	9.90E+01	9.90E-05	3.08E-01	8.19E+00	9.90E-05	6.75E+02	2.97E-04
Zn-65	1.15E+02	5.06E+01	5.06E-05	1.58E-01	4.19E+00	5.06E-05	3.45E+02	1.52E-04
Zr-95	3.00E+01	1.32E+01	1.32E-05	4.11E-02	1.09E+00	1.32E-05	9.00E+01	3.96E-05

III. Summary of Potentially Exposed Workers

a) Total Project Dose

Job Function	No. Workers	Waste Contact Time (hr)	External Exposure Rate ⁷ (mR/hr)	Internal Dose Rate ⁹ (mrem/hr)	Distance (m)	No. Required Trips or Reps ¹⁰	Total External Dose (mrem)	Total Internal Dose (mrem)	Total Project Dose (mrem)
Long-Haul Truck Drivers ⁸	6	32.73	0.00E+00	0.00E+00	4.0	561	0.00E+00	0.00E+00	0.00E+00
RTF Equipment Operator	4	0.25	9.51E-02	0.00E+00	4.9	0	0.00E+00	0.00E+00	0.00E+00
Truck/IMC Surveyors	8	0.08	7.71E-01	0.00E+00	1.0	561	4.33E+00	0.00E+00	4.33E+00
BED Truck Drivers	10	0.75	8.86E-02	0.00E+00	4.0	0	0.00E+00	0.00E+00	0.00E+00
Landfill Cell Operators	4	0.25	1.16E-01	2.40E-02	2.0	135	9.77E-01	2.02E-01	1.18E+00

IV. RESRAD Modelling Results

A. Baseline Case 2.42E-01 mrem (@ t= 326.1 yrs) Baseline DF = 1.37E-03
Assumption(s): Project waste is distributed evenly over entire contents of USEI waste cells (88,221 m² area, 33.6 m depth)

B. Concentrated Case 6.42E+00 mrem (@ t= 326.1 yrs) Conc. DF = 3.64E-02
Assumption(s): Project waste is concentrated in smaller portion of landfill. Assumes entire annual SPFM activity inventory arrives at USEI in 4-months (instead of 12 months).

V. Notes & Assumptions

- 1 Microshield model inputs for transportation and handling tasks are adjusted to units of uCi/cc using as-shipped density
- 2 Microshield model inputs for landfill workers is adjusted to units of uCi/cc using in situ compacted landfill density of 1.5 g/cc.
- 3 The waste stream concentrations are multiplied by the Dilution Factors in Section IV for each scenario (Baseline, Concentrated)
- 4 Th-232 and Ra-226 assumed to be in complete equilibrium with their entire progeny chains
- 5 Individual uranium isotopes are analyzed as they are reported with no additional decay.
- 6 Natural uranium is assumed to be in complete equilibrium with all decay progeny for Microshield runs.
Natural uranium isotopic concentrations added to other U-234, U-235, and U-238 concentrations for RESRAD runs.
- 7 The External Exposure Rates are the sum of two separate Microshield runs. One for natural uranium in equilibrium and another for all other Radionuclides of Concern shown in Section II. U-Nat was run independently to ensure equilibrium with all progeny.

ATTACHMENT 4

Microshield Reports for Annualized SPFM Dose Assessment

MicroShield 7.02
US Ecology, Inc (08-MSD-7.02-1419)

Date	By	Checked

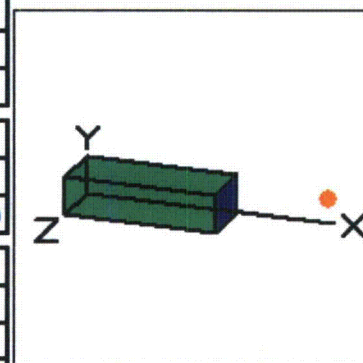
Filename	Run Date	Run Time	Duration
Truck-IMC Driver Chassis Trailer SPFM.ms7	August 15, 2012	1:37:24 PM	00:00:01

Project Info	
Case Title	SPFM IMC Driver
Description	IMC on Chassis. D=4m (13.1'), rho=0.44 g/cc, 1 pCi/g debris
Geometry	13 - Rectangular Volume

Source Dimensions	
Length	617.2 cm (20 ft 3.0 in)
Width	243.8 cm (7 ft 12.0 in)
Height	152.4 cm (5 ft 0.0 in)

Dose Points			
A	X	Y	Z
#1	1.0e+3 cm (33 ft 4.7 in)	121.9 cm (3 ft 12.0 in)	60.9 cm (1 ft 12.0 in)

Shields			
Shield N	Dimension	Material	Density
Source	2.29e+07 cm ³	Concrete	0.44
Shield 1	.5 cm	Aluminum	2.7
Air Gap		Air	0.00122



Source Input: Grouping Method - Standard Indices
Number of Groups: 25
Lower Energy Cutoff: 0.015
Photons < 0.015: Excluded
Library: Grove

Nuclide	Ci	Bq	µCi/cm ³	Bq/cm ³
Ac-228	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Ag-108m	5.0451e-005	1.8667e+006	2.2000e-006	8.1400e-002
Ag-110m	2.5225e-004	9.3334e+006	1.1000e-005	4.0700e-001
Am-241	5.0451e-006	1.8667e+005	2.2000e-007	8.1400e-003
Au-195	3.5315e-005	1.3067e+006	1.5400e-006	5.6980e-002
Ba-133	3.0270e-006	1.1200e+005	1.3200e-007	4.8840e-003
Ba-137m	4.7726e-003	1.7659e+008	2.0812e-004	7.7004e+000
Be-7	2.0180e-004	7.4667e+006	8.8000e-006	3.2560e-001
Bi-210	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
Bi-212	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Bi-214	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
C-14	2.5225e-004	9.3334e+006	1.1000e-005	4.0700e-001
Ce-139	1.0090e-005	3.7334e+005	4.4000e-007	1.6280e-002
Ce-141	1.1604e-004	4.2934e+006	5.0600e-006	1.8722e-001

Ce-144	1.8162e-003	6.7200e+007	7.9200e-005	2.9304e+000
Cm-242	1.0090e-006	3.7334e+004	4.4000e-008	1.6280e-003
Cm-243	1.3117e-005	4.8534e+005	5.7200e-007	2.1164e-002
Cm-244	5.0451e-006	1.8667e+005	2.2000e-007	8.1400e-003
Cm-245	5.0451e-005	1.8667e+006	2.2000e-006	8.1400e-002
Co-57	1.2108e-004	4.4800e+006	5.2800e-006	1.9536e-001
Co-58	2.0180e-003	7.4667e+007	8.8000e-005	3.2560e+000
Co-60	6.5586e-003	2.4267e+008	2.8600e-004	1.0582e+001
Cr-51	5.5496e-004	2.0533e+007	2.4200e-005	8.9540e-001
Cs-134	1.7658e-003	6.5334e+007	7.7000e-005	2.8490e+000
Cs-137	5.0451e-003	1.8667e+008	2.2000e-004	8.1400e+000
Eu-152	7.0631e-005	2.6133e+006	3.0800e-006	1.1396e-001
Eu-154	2.5225e-005	9.3334e+005	1.1000e-006	4.0700e-002
Eu-155	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Fe-55	1.0090e-002	3.7334e+008	4.4000e-004	1.6280e+001
Fe-59	8.0721e-005	2.9867e+006	3.5200e-006	1.3024e-001
H-3	3.2793e-003	1.2133e+008	1.4300e-004	5.2910e+000
I-125	1.0090e-006	3.7334e+004	4.4000e-008	1.6280e-003
I-129	3.0270e-006	1.1200e+005	1.3200e-007	4.8840e-003
I-131	1.8162e-004	6.7200e+006	7.9200e-006	2.9304e-001
In-113m	2.0180e-005	7.4667e+005	8.8000e-007	3.2560e-002
Mn-54	8.0721e-004	2.9867e+007	3.5200e-005	1.3024e+000
Na-22	2.0180e-006	7.4667e+004	8.8000e-008	3.2560e-003
Nb-94	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Nb-95	2.5225e-004	9.3334e+006	1.1000e-005	4.0700e-001
Ni-59	1.0090e-003	3.7334e+007	4.4000e-005	1.6280e+000
Ni-63	9.3334e-003	3.4534e+008	4.0700e-004	1.5059e+001
Pb-210	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
Pb-212	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Pb-214	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
Po-210	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
Po-212	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Po-214	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
Po-216	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Po-218	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
Pr-144	1.7903e-003	6.6239e+007	7.8067e-005	2.8885e+000
Pu-238	5.0451e-005	1.8667e+006	2.2000e-006	8.1400e-002
Pu-239	3.0270e-006	1.1200e+005	1.3200e-007	4.8840e-003
Pu-240	3.0270e-006	1.1200e+005	1.3200e-007	4.8840e-003
Pu-241	1.5135e-004	5.6000e+006	6.6000e-006	2.4420e-001
Pu-242	2.0180e-006	7.4667e+004	8.8000e-008	3.2560e-003
Ra-224	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Ra-226	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
Ra-228	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Rh-103m	1.5095e-005	5.5853e+005	6.5826e-007	2.4356e-002

	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
Rn-220	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Rn-222	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
Ru-103	1.5135e-005	5.6000e+005	6.6000e-007	2.4420e-002
Ru-106	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
Sb-124	8.0721e-005	2.9867e+006	3.5200e-006	1.3024e-001
Sb-125	1.0090e-003	3.7334e+007	4.4000e-005	1.6280e+000
Sn-113	2.0180e-005	7.4667e+005	8.8000e-007	3.2560e-002
Sr-89	1.7153e-004	6.3467e+006	7.4800e-006	2.7676e-001
Sr-90	1.4126e-004	5.2267e+006	6.1600e-006	2.2792e-001
Tc-99	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Te-123	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Th-228	1.0090e-005	3.7334e+005	4.4000e-007	1.6280e-002
Th-232	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
Tl-208	4.0361e-005	1.4933e+006	1.7600e-006	6.5120e-002
U-233	8.0721e-005	2.9867e+006	3.5200e-006	1.3024e-001
U-234	1.9171e-003	7.0934e+007	8.3600e-005	3.0932e+000
U-235	1.0090e-004	3.7334e+006	4.4000e-006	1.6280e-001
U-238	1.9171e-003	7.0934e+007	8.3600e-005	3.0932e+000
Y-90	1.4126e-004	5.2267e+006	6.1600e-006	2.2792e-001
Zn-65	1.1604e-003	4.2934e+007	5.0600e-005	1.8722e+000
Zr-95	3.0270e-004	1.1200e+007	1.3200e-005	4.8840e-001

**Buildup: The material reference is Source
Integration Parameters**

X Direction		20
Y Direction		20
Z Direction		20

Results

Energy (MeV)	Activity (Photons/sec)	Fluence Rate	Fluence Rate	Exposure Rate	Exposure Rate
		MeV/cm ² /sec No Buildup	MeV/cm ² /sec With Buildup	mR/hr No Buildup	mR/hr With Buildup
0.015	9.176e+03	2.518e-15	2.766e-15	2.159e-16	2.372e-16
0.02	1.947e+06	3.829e-08	4.488e-08	1.326e-09	1.555e-09
0.03	2.946e+07	2.375e-04	3.220e-04	2.354e-06	3.191e-06
0.04	1.295e+07	6.339e-04	1.039e-03	2.803e-06	4.596e-06
0.05	8.037e+05	9.991e-05	1.977e-04	2.662e-07	5.266e-07
0.06	1.284e+06	2.852e-04	6.879e-04	5.665e-07	1.366e-06
0.08	3.833e+06	1.702e-03	5.077e-03	2.694e-06	8.034e-06
0.1	7.040e+06	4.757e-03	1.564e-02	7.278e-06	2.393e-05
0.15	1.112e+07	1.424e-02	4.882e-02	2.344e-05	8.039e-05
0.2	6.791e+06	1.320e-02	4.375e-02	2.330e-05	7.721e-05
0.3	4.698e+06	1.629e-02	4.894e-02	3.090e-05	9.283e-05
0.4	2.138e+07	1.120e-01	3.085e-01	2.182e-04	6.011e-04
0.5	3.104e+07	2.245e-01	5.770e-01	4.407e-04	1.133e-03

0.6	2.724e+08	2.569e+00	6.216e+00	5.014e-03	1.213e-02
0.8	2.053e+08	2.953e+00	6.534e+00	5.618e-03	1.243e-02
1.0	2.749e+08	5.505e+00	1.141e+01	1.015e-02	2.103e-02
1.5	2.542e+08	9.328e+00	1.729e+01	1.569e-02	2.909e-02
2.0	1.687e+06	9.487e-02	1.648e-01	1.467e-04	2.549e-04
3.0	1.490e+06	1.511e-01	2.413e-01	2.050e-04	3.274e-04
Totals	1.142e+09	2.099e+01	4.290e+01	3.758e-02	7.729e-02

MicroShield 7.02
US Ecology, Inc (08-MSD-7.02-1419)

Date	By	Checked

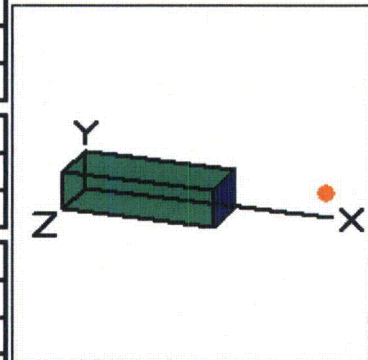
Filename	Run Date	Run Time	Duration
Truck-IMC Driver Chassis Trailer SPFM Unat.ms7	August 15, 2012	1:39:32 PM	00:00:01

Project Info	
Case Title	SPFM IMC Driver
Description	IMC on Chassis. D=4m, rho=0.44 g/cc, 1 pCi/g debris, U-nat
Geometry	13 - Rectangular Volume

Source Dimensions	
Length	617.2 cm (20 ft 3.0 in)
Width	243.8 cm (7 ft 12.0 in)
Height	152.4 cm (5 ft 0.0 in)

Dose Points			
A	X	Y	Z
#1	1.0e+3 cm (33 ft 4.7 in)	121.9 cm (3 ft 12.0 in)	60.9 cm (1 ft 12.0 in)

Shields			
Shield N	Dimension	Material	Density
Source	2.29e+07 cm ³	Concrete	0.44
Shield 1	.5 cm	Aluminum	2.7
Air Gap		Air	0.00122



Source Input: Grouping Method - Standard Indices				
Number of Groups: 25				
Lower Energy Cutoff: 0.015				
Photons < 0.015: Excluded				
Library: Grove				
Nuclide	Ci	Bq	µCi/cm ³	Bq/cm ³
Ac-227	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001
Bi-210	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Bi-211	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001
Bi-214	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Fr-223	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001
Pa-231	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001
Pa-234	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Pa-234m	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Pb-210	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Pb-211	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001
Pb-214	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Po-210	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Po-214	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Po-215	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001

Po-218	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Ra-223	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001
Ra-226	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Rn-219	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001
Rn-222	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Th-227	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001
Th-230	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Th-231	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001
Th-234	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
Tl-207	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001
U-234	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000
U-235	6.3568e-005	2.3520e+006	2.7720e-006	1.0256e-001
U-238	1.1030e-003	4.0812e+007	4.8100e-005	1.7797e+000

**Buildup: The material reference is Source
Integration Parameters**

X Direction	20
Y Direction	20
Z Direction	20

Results

Energy (MeV)	Activity (Photons/sec)	Fluence Rate	Fluence Rate	Exposure Rate	Exposure Rate
		MeV/cm ² /sec No Buildup	MeV/cm ² /sec With Buildup	mR/hr No Buildup	mR/hr With Buildup
0.015	5.276e+03	1.448e-15	1.590e-15	1.242e-16	1.364e-16
0.02	3.083e+04	6.063e-10	7.106e-10	2.100e-11	2.462e-11
0.03	5.657e+05	4.561e-06	6.183e-06	4.520e-08	6.128e-08
0.04	5.887e+04	2.883e-06	4.726e-06	1.275e-08	2.090e-08
0.05	3.123e+06	3.882e-04	7.682e-04	1.034e-06	2.046e-06
0.06	3.201e+06	7.108e-04	1.714e-03	1.412e-06	3.405e-06
0.08	1.126e+07	5.001e-03	1.492e-02	7.914e-06	2.360e-05
0.1	2.692e+07	1.819e-02	5.981e-02	2.784e-05	9.151e-05
0.15	1.354e+07	1.734e-02	5.944e-02	2.855e-05	9.789e-05
0.2	1.492e+07	2.901e-02	9.614e-02	5.120e-05	1.697e-04
0.3	1.282e+07	4.446e-02	1.336e-01	8.434e-05	2.533e-04
0.4	1.873e+07	9.814e-02	2.703e-01	1.912e-04	5.267e-04
0.5	4.491e+06	3.248e-02	8.349e-02	6.376e-05	1.639e-04
0.6	3.488e+07	3.290e-01	7.961e-01	6.422e-04	1.554e-03
0.8	3.601e+07	5.181e-01	1.146e+00	9.855e-04	2.180e-03
1.0	3.827e+07	7.665e-01	1.588e+00	1.413e-03	2.928e-03
1.5	1.348e+07	4.947e-01	9.168e-01	8.323e-04	1.543e-03
2.0	1.166e+07	6.555e-01	1.139e+00	1.014e-03	1.761e-03
Totals	2.440e+08	3.010e+00	6.307e+00	5.344e-03	1.130e-02

MicroShield 7.02
US Ecology, Inc (08-MSD-7.02-1419)

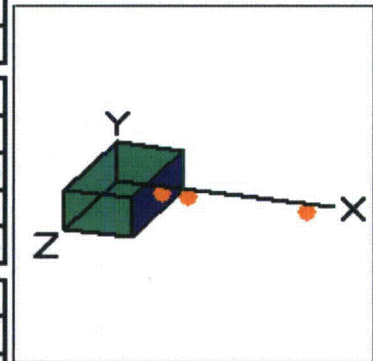
Date	By	Checked

Filename	Run Date	Run Time	Duration
Truck-IMC Surveyor RTF Taylor Operator SPFM.ms7	August 15, 2012	1:41:19 PM	00:00:02

Project Info	
Case Title	SPFM IMC Surv&Taylor
Description	1 pCi/g Debris, rho=0.44 g/cc, D=1m & 4.9m
Geometry	13 - Rectangular Volume

Source Dimensions	
Length	218.44 cm (7 ft 2.0 in)
Width	462.28 cm (15 ft 2.0 in)
Height	127.0 cm (4 ft 2.0 in)

Dose Points			
A	X	Y	Z
#1	234.315 cm (7 ft 8.3 in)	63.5 cm (2 ft 1.0 in)	231.14 cm (7 ft 7.0 in)
#2	319.075 cm (10 ft 5.6 in)	63.5 cm (2 ft 1.0 in)	231.14 cm (7 ft 7.0 in)
#3	706.755 cm (23 ft 2.2 in)	63.5 cm (2 ft 1.0 in)	231.14 cm (7 ft 7.0 in)



Shields			
Shield N	Dimension	Material	Density
Source	1.28e+07 cm ³	Concrete	0.44
Shield 1	.635 cm	Aluminum	2.7
Air Gap		Air	0.00122

Source Input: Grouping Method - Standard Indices
Number of Groups: 25
Lower Energy Cutoff: 0.015
Photons < 0.015: Excluded
Library: Grove

Nuclide	Ci	Bq	µCi/cm ³	Bq/cm ³
Ac-228	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Ag-108m	2.8214e-005	1.0439e+006	2.2000e-006	8.1400e-002
Ag-110m	1.4107e-004	5.2196e+006	1.1000e-005	4.0700e-001
Am-241	2.8214e-006	1.0439e+005	2.2000e-007	8.1400e-003
Au-195	1.9750e-005	7.3074e+005	1.5400e-006	5.6980e-002
Ba-133	1.6928e-006	6.2635e+004	1.3200e-007	4.8840e-003
Ba-137m	2.6690e-003	9.8754e+007	2.0812e-004	7.7004e+000
Be-7	1.1286e-004	4.1757e+006	8.8000e-006	3.2560e-001
Bi-210	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001
Bi-212	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Bi-214	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001
C-14	1.4107e-004	5.2196e+006	1.1000e-005	4.0700e-001

Ce-139	5.6428e-006	2.0878e+005	4.4000e-007	1.6280e-002
Ce-141	6.4892e-005	2.4010e+006	5.0600e-006	1.8722e-001
Ce-144	1.0157e-003	3.7581e+007	7.9200e-005	2.9304e+000
Cm-242	5.6428e-007	2.0878e+004	4.4000e-008	1.6280e-003
Cm-243	7.3356e-006	2.7142e+005	5.7200e-007	2.1164e-002
Cm-244	2.8214e-006	1.0439e+005	2.2000e-007	8.1400e-003
Cm-245	2.8214e-005	1.0439e+006	2.2000e-006	8.1400e-002
Co-57	6.7713e-005	2.5054e+006	5.2800e-006	1.9536e-001
Co-58	1.1286e-003	4.1757e+007	8.8000e-005	3.2560e+000
Co-60	3.6678e-003	1.3571e+008	2.8600e-004	1.0582e+001
Cr-51	3.1035e-004	1.1483e+007	2.4200e-005	8.9540e-001
Cs-134	9.8749e-004	3.6537e+007	7.7000e-005	2.8490e+000
Cs-137	2.8214e-003	1.0439e+008	2.2000e-004	8.1400e+000
Eu-152	3.9500e-005	1.4615e+006	3.0800e-006	1.1396e-001
Eu-154	1.4107e-005	5.2196e+005	1.1000e-006	4.0700e-002
Eu-155	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Fe-55	5.6428e-003	2.0878e+008	4.4000e-004	1.6280e+001
Fe-59	4.5142e-005	1.6703e+006	3.5200e-006	1.3024e-001
H-3	1.8339e-003	6.7855e+007	1.4300e-004	5.2910e+000
I-125	5.6428e-007	2.0878e+004	4.4000e-008	1.6280e-003
I-129	1.6928e-006	6.2635e+004	1.3200e-007	4.8840e-003
I-131	1.0157e-004	3.7581e+006	7.9200e-006	2.9304e-001
In-113m	1.1286e-005	4.1757e+005	8.8000e-007	3.2560e-002
Mn-54	4.5142e-004	1.6703e+007	3.5200e-005	1.3024e+000
Na-22	1.1286e-006	4.1757e+004	8.8000e-008	3.2560e-003
Nb-94	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Nb-95	1.4107e-004	5.2196e+006	1.1000e-005	4.0700e-001
Ni-59	5.6428e-004	2.0878e+007	4.4000e-005	1.6280e+000
Ni-63	5.2196e-003	1.9312e+008	4.0700e-004	1.5059e+001
Pb-210	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001
Pb-212	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Pb-214	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001
Po-210	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001
Po-212	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Po-214	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001
Po-216	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Po-218	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001
Pr-144	1.0012e-003	3.7044e+007	7.8067e-005	2.8885e+000
Pu-238	2.8214e-005	1.0439e+006	2.2000e-006	8.1400e-002
Pu-239	1.6928e-006	6.2635e+004	1.3200e-007	4.8840e-003
Pu-240	1.6928e-006	6.2635e+004	1.3200e-007	4.8840e-003
Pu-241	8.4642e-005	3.1317e+006	6.6000e-006	2.4420e-001
Pu-242	1.1286e-006	4.1757e+004	8.8000e-008	3.2560e-003
Ra-224	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Ra-226	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001

	8.4419e-006	3.1235e+005	6.5826e-007	2.4356e-002
Rh-106	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001
Rn-220	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Rn-222	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001
Ru-103	8.4642e-006	3.1317e+005	6.6000e-007	2.4420e-002
Ru-106	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001
Sb-124	4.5142e-005	1.6703e+006	3.5200e-006	1.3024e-001
Sb-125	5.6428e-004	2.0878e+007	4.4000e-005	1.6280e+000
Sn-113	1.1286e-005	4.1757e+005	8.8000e-007	3.2560e-002
Sr-89	9.5927e-005	3.5493e+006	7.4800e-006	2.7676e-001
Sr-90	7.8999e-005	2.9230e+006	6.1600e-006	2.2792e-001
Tc-99	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Te-123	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Th-228	5.6428e-006	2.0878e+005	4.4000e-007	1.6280e-002
Th-232	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
Tl-208	2.2571e-005	8.3513e+005	1.7600e-006	6.5120e-002
U-233	4.5142e-005	1.6703e+006	3.5200e-006	1.3024e-001
U-234	1.0721e-003	3.9669e+007	8.3600e-005	3.0932e+000
U-235	5.6428e-005	2.0878e+006	4.4000e-006	1.6280e-001
U-238	1.0721e-003	3.9669e+007	8.3600e-005	3.0932e+000
Y-90	7.8999e-005	2.9230e+006	6.1600e-006	2.2792e-001
Zn-65	6.4892e-004	2.4010e+007	5.0600e-005	1.8722e+000
Zr-95	1.6928e-004	6.2635e+006	1.3200e-005	4.8840e-001

Buildup: The material reference is Source	
Integration Parameters	
X Direction	20
Y Direction	20
Z Direction	20

Results - Dose Point # 1 - (234.315,63.5,231.14) cm					
Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm ² /sec No Buildup	Fluence Rate MeV/cm ² /sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	5.131e+03	4.912e-14	5.369e-14	4.213e-15	4.605e-15
0.02	1.089e+06	2.397e-07	2.793e-07	8.303e-09	9.675e-09
0.03	1.648e+07	1.862e-03	2.582e-03	1.845e-05	2.559e-05
0.04	7.239e+06	7.826e-03	1.358e-02	3.461e-05	6.006e-05
0.05	4.494e+05	1.546e-03	3.350e-03	4.117e-06	8.924e-06
0.06	7.183e+05	4.946e-03	1.315e-02	9.823e-06	2.612e-05
0.08	2.143e+06	3.264e-02	1.084e-01	5.165e-05	1.716e-04
0.1	3.937e+06	9.506e-02	3.501e-01	1.454e-04	5.357e-04
0.15	6.219e+06	2.965e-01	1.133e+00	4.882e-04	1.866e-03
0.2	3.798e+06	2.801e-01	1.022e+00	4.944e-04	1.804e-03
0.3	2.628e+06	3.533e-01	1.144e+00	6.701e-04	2.169e-03
0.4	1.196e+07	2.461e+00	7.187e+00	4.795e-03	1.400e-02

0.5	1.736e+07	4.977e+00	1.339e+01	9.770e-03	2.629e-02
0.6	1.523e+08	5.733e+01	1.437e+02	1.119e-01	2.806e-01
0.8	1.148e+08	6.646e+01	1.501e+02	1.264e-01	2.854e-01
1.0	1.537e+08	1.244e+02	2.605e+02	2.294e-01	4.801e-01
1.5	1.422e+08	2.115e+02	3.890e+02	3.559e-01	6.545e-01
2.0	9.436e+05	2.146e+00	3.658e+00	3.318e-03	5.657e-03
3.0	8.335e+05	3.383e+00	5.234e+00	4.590e-03	7.101e-03
Totals	6.388e+08	4.738e+02	9.766e+02	8.480e-01	1.760e+00

Results - Dose Point # 2 - (319.075,63.5,231.14) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate	Fluence Rate	Exposure Rate	Exposure Rate
		MeV/cm ² /sec No Buildup	MeV/cm ² /sec With Buildup	mR/hr No Buildup	mR/hr With Buildup
0.015	5.131e+03	3.588e-14	3.924e-14	3.077e-15	3.366e-15
0.02	1.089e+06	1.988e-07	2.317e-07	6.885e-09	8.027e-09
0.03	1.648e+07	1.255e-03	1.725e-03	1.244e-05	1.710e-05
0.04	7.239e+06	4.443e-03	7.530e-03	1.965e-05	3.330e-05
0.05	4.494e+05	7.909e-04	1.640e-03	2.107e-06	4.370e-06
0.06	7.183e+05	2.382e-03	5.997e-03	4.731e-06	1.191e-05
0.08	2.143e+06	1.481e-02	4.562e-02	2.344e-05	7.220e-05
0.1	3.937e+06	4.198e-02	1.420e-01	6.423e-05	2.172e-04
0.15	6.219e+06	1.271e-01	4.444e-01	2.093e-04	7.319e-04
0.2	3.798e+06	1.183e-01	3.965e-01	2.088e-04	6.998e-04
0.3	2.628e+06	1.464e-01	4.400e-01	2.778e-04	8.347e-04
0.4	1.196e+07	1.008e+00	2.757e+00	1.963e-03	5.371e-03
0.5	1.736e+07	2.019e+00	5.130e+00	3.963e-03	1.007e-02
0.6	1.523e+08	2.309e+01	5.502e+01	4.507e-02	1.074e-01
0.8	1.148e+08	2.649e+01	5.738e+01	5.038e-02	1.091e-01
1.0	1.537e+08	4.922e+01	9.953e+01	9.074e-02	1.835e-01
1.5	1.422e+08	8.270e+01	1.485e+02	1.391e-01	2.498e-01
2.0	9.436e+05	8.336e-01	1.396e+00	1.289e-03	2.159e-03
3.0	8.335e+05	1.306e+00	1.996e+00	1.772e-03	2.708e-03
Totals	6.388e+08	1.871e+02	3.732e+02	3.351e-01	6.727e-01

Results - Dose Point # 3 - (706.755,63.5,231.14) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate	Fluence Rate	Exposure Rate	Exposure Rate
		MeV/cm ² /sec No Buildup	MeV/cm ² /sec With Buildup	mR/hr No Buildup	mR/hr With Buildup
0.015	5.131e+03	6.629e-15	7.254e-15	5.686e-16	6.222e-16
0.02	1.089e+06	3.492e-08	4.067e-08	1.209e-09	1.409e-09
0.03	1.648e+07	1.716e-04	2.347e-04	1.701e-06	2.326e-06
0.04	7.239e+06	5.410e-04	9.100e-04	2.393e-06	4.025e-06
0.05	4.494e+05	9.186e-05	1.893e-04	2.447e-07	5.043e-07
0.06	7.183e+05	2.713e-04	6.823e-04	5.388e-07	1.355e-06
0.08	2.143e+06	1.666e-03	5.188e-03	2.637e-06	8.210e-06
0.1	3.937e+06	4.716e-03	1.625e-02	7.215e-06	2.486e-05
0.15	6.219e+06	1.432e-02	5.162e-02	2.357e-05	8.500e-05

0.2	3.798e+06	1.338e-02	4.653e-02	2.361e-05	8.212e-05
0.3	2.628e+06	1.667e-02	5.228e-02	3.163e-05	9.917e-05
0.4	1.196e+07	1.154e-01	3.304e-01	2.249e-04	6.437e-04
0.5	1.736e+07	2.326e-01	6.188e-01	4.565e-04	1.215e-03
0.6	1.523e+08	2.672e+00	6.674e+00	5.216e-03	1.303e-02
0.8	1.148e+08	3.092e+00	7.023e+00	5.882e-03	1.336e-02
1.0	1.537e+08	5.792e+00	1.227e+01	1.068e-02	2.262e-02
1.5	1.422e+08	9.886e+00	1.856e+01	1.663e-02	3.123e-02
2.0	9.436e+05	1.009e-01	1.762e-01	1.560e-04	2.724e-04
3.0	8.335e+05	1.608e-01	2.550e-01	2.181e-04	3.459e-04
Totals	6.388e+08	2.210e+01	4.608e+01	3.956e-02	8.301e-02

MicroShield 7.02
US Ecology, Inc (08-MSD-7.02-1419)

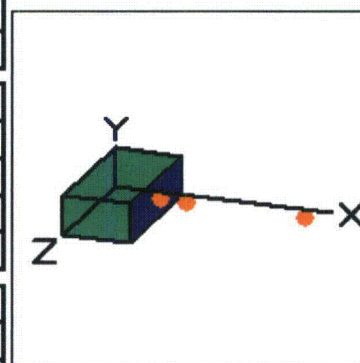
Date	By	Checked

Filename	Run Date	Run Time	Duration
Truck-IMC Surveyor RTF Taylor Operator SPFM Unat.ms7	August 15, 2012	1:42:48 PM	00:00:02

Project Info	
Case Title	SPFM IMC Surv&Taylor
Description	1 pCi/g debris, rho=0.44 g/cc, D=1m & 4.9 m, Unat only
Geometry	13 - Rectangular Volume

Source Dimensions	
Length	218.44 cm (7 ft 2.0 in)
Width	462.28 cm (15 ft 2.0 in)
Height	127.0 cm (4 ft 2.0 in)

Dose Points			
A	X	Y	Z
#1	234.315 cm (7 ft 8.3 in)	63.5 cm (2 ft 1.0 in)	231.14 cm (7 ft 7.0 in)
#2	319.075 cm (10 ft 5.6 in)	63.5 cm (2 ft 1.0 in)	231.14 cm (7 ft 7.0 in)
#3	706.755 cm (23 ft 2.2 in)	63.5 cm (2 ft 1.0 in)	231.14 cm (7 ft 7.0 in)



Shields			
Shield N	Dimension	Material	Density
Source	1.28e+07 cm ³	Concrete	0.44
Shield 1	.635 cm	Aluminum	2.7
Air Gap		Air	0.00122

Source Input: Grouping Method - Standard Indices
Number of Groups: 25
Lower Energy Cutoff: 0.015
Photons < 0.015: Excluded
Library: Grove

Nuclide	Ci	Bq	µCi/cm ³	Bq/cm ³
Ac-227	3.5550e-005	1.3153e+006	2.7720e-006	1.0256e-001
Bi-210	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Bi-211	3.5550e-005	1.3153e+006	2.7720e-006	1.0256e-001
Bi-214	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Pa-231	3.5550e-005	1.3153e+006	2.7720e-006	1.0256e-001
Pa-234	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Pa-234m	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Pb-210	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Pb-211	3.5550e-005	1.3153e+006	2.7720e-006	1.0256e-001
Pb-214	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Po-210	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Po-214	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000

Po-215	3.5550e-005	1.3153e+006	2.7720e-006	1.0256e-001
Po-218	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Ra-223	3.5550e-005	1.3153e+006	2.7720e-006	1.0256e-001
Ra-226	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Rn-219	3.5550e-005	1.3153e+006	2.7720e-006	1.0256e-001
Rn-222	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Th-228	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Th-230	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
Th-231	3.5550e-005	1.3153e+006	2.7720e-006	1.0256e-001
Tl-207	3.5550e-005	1.3153e+006	2.7720e-006	1.0256e-001
U-234	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000
U-235	3.5550e-005	1.3153e+006	2.7720e-006	1.0256e-001
U-238	6.1686e-004	2.2824e+007	4.8100e-005	1.7797e+000

**Buildup: The material reference is Source
Integration Parameters**

X Direction	20
Y Direction	20
Z Direction	20

Results - Dose Point # 1 - (234.315,63.5,231.14) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate	Fluence Rate	Exposure Rate	Exposure Rate
		MeV/cm ² /sec No Buildup	MeV/cm ² /sec With Buildup	mR/hr No Buildup	mR/hr With Buildup
0.015	2.951e+03	2.824e-14	3.087e-14	2.422e-15	2.648e-15
0.02	4.604e+03	1.013e-09	1.181e-09	3.511e-11	4.091e-11
0.03	3.150e+05	3.560e-05	4.938e-05	3.528e-07	4.894e-07
0.04	2.990e+04	3.232e-05	5.608e-05	1.429e-07	2.480e-07
0.05	1.206e+06	4.148e-03	8.992e-03	1.105e-05	2.395e-05
0.06	9.120e+05	6.279e-03	1.670e-02	1.247e-05	3.317e-05
0.08	6.318e+06	9.620e-02	3.196e-01	1.522e-04	5.058e-04
0.1	1.370e+07	3.307e-01	1.218e+00	5.059e-04	1.863e-03
0.15	7.617e+06	3.631e-01	1.388e+00	5.979e-04	2.285e-03
0.2	8.141e+06	6.005e-01	2.191e+00	1.060e-03	3.866e-03
0.3	6.907e+06	9.286e-01	3.006e+00	1.762e-03	5.703e-03
0.4	1.047e+07	2.156e+00	6.295e+00	4.200e-03	1.227e-02
0.5	2.495e+06	7.155e-01	1.925e+00	1.405e-03	3.779e-03
0.6	1.951e+07	7.342e+00	1.841e+01	1.433e-02	3.593e-02
0.8	2.013e+07	1.166e+01	2.632e+01	2.217e-02	5.006e-02
1.0	2.140e+07	1.733e+01	3.627e+01	3.194e-02	6.686e-02
1.5	7.539e+06	1.122e+01	2.063e+01	1.887e-02	3.471e-02
2.0	6.520e+06	1.483e+01	2.528e+01	2.293e-02	3.909e-02
Totals	1.332e+08	6.757e+01	1.433e+02	1.199e-01	2.570e-01

Results - Dose Point # 2 - (319.075,63.5,231.14) cm

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm ² /sec	Fluence Rate MeV/cm ² /sec	Exposure Rate mR/hr	Exposure Rate mR/hr
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		No Buildup	With Buildup	No Buildup	With Buildup
0.015	2.951e+03	2.063e-14	2.256e-14	1.770e-15	1.935e-15
0.02	4.604e+03	8.405e-10	9.798e-10	2.911e-11	3.394e-11
0.03	3.150e+05	2.400e-05	3.299e-05	2.379e-07	3.270e-07
0.04	2.990e+04	1.835e-05	3.109e-05	8.114e-08	1.375e-07
0.05	1.206e+06	2.123e-03	4.403e-03	5.655e-06	1.173e-05
0.06	9.120e+05	3.025e-03	7.615e-03	6.008e-06	1.512e-05
0.08	6.318e+06	4.365e-02	1.345e-01	6.908e-05	2.128e-04
0.1	1.370e+07	1.461e-01	4.939e-01	2.234e-04	7.557e-04
0.15	7.617e+06	1.557e-01	5.443e-01	2.564e-04	8.964e-04
0.2	8.141e+06	2.536e-01	8.499e-01	4.476e-04	1.500e-03
0.3	6.907e+06	3.849e-01	1.157e+00	7.302e-04	2.194e-03
0.4	1.047e+07	8.826e-01	2.415e+00	1.720e-03	4.705e-03
0.5	2.495e+06	2.903e-01	7.375e-01	5.698e-04	1.448e-03
0.6	1.951e+07	2.957e+00	7.046e+00	5.772e-03	1.375e-02
0.8	2.013e+07	4.645e+00	1.006e+01	8.836e-03	1.914e-02
1.0	2.140e+07	6.854e+00	1.386e+01	1.263e-02	2.554e-02
1.5	7.539e+06	4.386e+00	7.875e+00	7.379e-03	1.325e-02
2.0	6.520e+06	5.760e+00	9.645e+00	8.907e-03	1.492e-02
Totals	1.332e+08	2.676e+01	5.483e+01	4.756e-02	9.834e-02

Results - Dose Point # 3 - (706.755,63.5,231.14) cm					
Energy (MeV)	Activity (Photons/sec)	Fluence Rate	Fluence Rate	Exposure Rate	Exposure Rate
		MeV/cm ² /sec No Buildup	MeV/cm ² /sec With Buildup	mR/hr No Buildup	mR/hr With Buildup
0.015	2.951e+03	3.812e-15	4.171e-15	3.269e-16	3.578e-16
0.02	4.604e+03	1.476e-10	1.720e-10	5.114e-12	5.957e-12
0.03	3.150e+05	3.281e-06	4.487e-06	3.252e-08	4.447e-08
0.04	2.990e+04	2.234e-06	3.758e-06	9.881e-09	1.662e-08
0.05	1.206e+06	2.466e-04	5.081e-04	6.568e-07	1.354e-06
0.06	9.120e+05	3.444e-04	8.663e-04	6.841e-07	1.721e-06
0.08	6.318e+06	4.912e-03	1.529e-02	7.773e-06	2.420e-05
0.1	1.370e+07	1.641e-02	5.652e-02	2.510e-05	8.648e-05
0.15	7.617e+06	1.753e-02	6.322e-02	2.887e-05	1.041e-04
0.2	8.141e+06	2.867e-02	9.974e-02	5.060e-05	1.760e-04
0.3	6.907e+06	4.383e-02	1.374e-01	8.314e-05	2.607e-04
0.4	1.047e+07	1.011e-01	2.894e-01	1.970e-04	5.638e-04
0.5	2.495e+06	3.343e-02	8.896e-02	6.562e-05	1.746e-04
0.6	1.951e+07	3.423e-01	8.547e-01	6.680e-04	1.668e-03
0.8	2.013e+07	5.424e-01	1.232e+00	1.032e-03	2.343e-03
1.0	2.140e+07	8.064e-01	1.708e+00	1.486e-03	3.149e-03
1.5	7.539e+06	5.243e-01	9.843e-01	8.821e-04	1.656e-03
2.0	6.520e+06	6.970e-01	1.217e+00	1.078e-03	1.882e-03
Totals	1.332e+08	3.159e+00	6.748e+00	5.605e-03	1.209e-02

MicroShield 7.02
US Ecology, Inc (08-MSD-7.02-1419)

Date	By	Checked

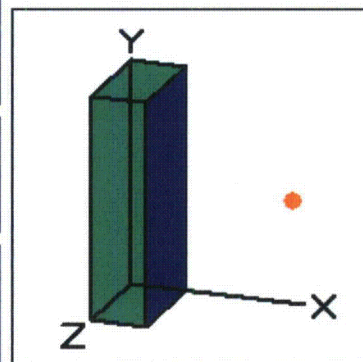
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Landfill Cell Operator_SPFM.ms7	August 15, 2012	1:45:15 PM	00:00:01

Project Info	
Case Title	USEI Cell Operator
Description	SPFM, rho=1.5 g/cc, D=2m, 0.5" steel shield
Geometry	13 - Rectangular Volume

Source Dimensions	
Length	91.44 cm (3 ft)
Width	182.88 cm (6 ft)
Height	381.0 cm (12 ft 6.0 in)

Dose Points			
A	X	Y	Z
#1	292.6 cm (9 ft 7.2 in)	190.5 cm (6 ft 3.0 in)	45.72 cm (1 ft 6.0 in)

Shields			
Shield N	Dimension	Material	Density
Source	6.37e+06 cm ³	Concrete	1.5
Shield 1	1.27 cm	Iron	7.86
Air Gap		Air	0.00122



Source Input: Grouping Method - Standard Indices
Number of Groups: 25
Lower Energy Cutoff: 0.015
Photons < 0.015: Excluded
Library: Grove

Nuclide	Ci	Bq	µCi/cm ³	Bq/cm ³
Ac-228	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Ag-108m	1.4017e-005	5.1862e+005	2.2000e-006	8.1400e-002
Ag-110m	7.0084e-005	2.5931e+006	1.1000e-005	4.0700e-001
Am-241	1.4017e-006	5.1862e+004	2.2000e-007	8.1400e-003
Au-195	9.8118e-006	3.6304e+005	1.5400e-006	5.6980e-002
Ba-133	8.4101e-007	3.1117e+004	1.3200e-007	4.8840e-003
Ba-137m	1.3260e-003	4.9062e+007	2.0812e-004	7.7004e+000
Be-7	5.6067e-005	2.0745e+006	8.8000e-006	3.2560e-001
Bi-210	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
Bi-212	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Bi-214	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
C-14	7.0084e-005	2.5931e+006	1.1000e-005	4.0700e-001
Ce-139	2.8034e-006	1.0372e+005	4.4000e-007	1.6280e-002
Ce-141	3.2239e-005	1.1928e+006	5.0600e-006	1.8722e-001

Ce-144	5.0461e-004	1.8670e+007	7.9200e-005	2.9304e+000
Cm-242	2.8034e-007	1.0372e+004	4.4000e-008	1.6280e-003
Cm-243	3.6444e-006	1.3484e+005	5.7200e-007	2.1164e-002
Cm-244	1.4017e-006	5.1862e+004	2.2000e-007	8.1400e-003
Cm-245	1.4017e-005	5.1862e+005	2.2000e-006	8.1400e-002
Co-57	3.3640e-005	1.2447e+006	5.2800e-006	1.9536e-001
Co-58	5.6067e-004	2.0745e+007	8.8000e-005	3.2560e+000
Co-60	1.8222e-003	6.7421e+007	2.8600e-004	1.0582e+001
Cr-51	1.5419e-004	5.7049e+006	2.4200e-005	8.9540e-001
Cs-134	4.9059e-004	1.8152e+007	7.7000e-005	2.8490e+000
Cs-137	1.4017e-003	5.1862e+007	2.2000e-004	8.1400e+000
Eu-152	1.9624e-005	7.2607e+005	3.0800e-006	1.1396e-001
Eu-154	7.0084e-006	2.5931e+005	1.1000e-006	4.0700e-002
Eu-155	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Fe-55	2.8034e-003	1.0372e+008	4.4000e-004	1.6280e+001
Fe-59	2.2427e-005	8.2980e+005	3.5200e-006	1.3024e-001
H-3	9.1109e-004	3.3710e+007	1.4300e-004	5.2910e+000
I-125	2.8034e-007	1.0372e+004	4.4000e-008	1.6280e-003
I-129	8.4101e-007	3.1117e+004	1.3200e-007	4.8840e-003
I-131	5.0461e-005	1.8670e+006	7.9200e-006	2.9304e-001
In-113m	5.6067e-006	2.0745e+005	8.8000e-007	3.2560e-002
Mn-54	2.2427e-004	8.2980e+006	3.5200e-005	1.3024e+000
Na-22	5.6067e-007	2.0745e+004	8.8000e-008	3.2560e-003
Nb-94	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Nb-95	7.0084e-005	2.5931e+006	1.1000e-005	4.0700e-001
Ni-59	2.8034e-004	1.0372e+007	4.4000e-005	1.6280e+000
Ni-63	2.5931e-003	9.5945e+007	4.0700e-004	1.5059e+001
Pb-210	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
Pb-212	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Pb-214	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
Po-210	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
Po-212	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Po-214	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
Po-216	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Po-218	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
Pr-144	4.9739e-004	1.8403e+007	7.8067e-005	2.8885e+000
Pu-238	1.4017e-005	5.1862e+005	2.2000e-006	8.1400e-002
Pu-239	8.4101e-007	3.1117e+004	1.3200e-007	4.8840e-003
Pu-240	8.4101e-007	3.1117e+004	1.3200e-007	4.8840e-003
Pu-241	4.2051e-005	1.5559e+006	6.6000e-006	2.4420e-001
Pu-242	5.6067e-007	2.0745e+004	8.8000e-008	3.2560e-003
Ra-224	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Ra-226	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
Ra-228	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Rh-103m	4.1940e-006	1.5518e+005	6.5826e-007	2.4356e-002

	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
Rn-220	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Rn-222	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
Ru-103	4.2051e-006	1.5559e+005	6.6000e-007	2.4420e-002
Ru-106	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
Sb-124	2.2427e-005	8.2980e+005	3.5200e-006	1.3024e-001
Sb-125	2.8034e-004	1.0372e+007	4.4000e-005	1.6280e+000
Sn-113	5.6067e-006	2.0745e+005	8.8000e-007	3.2560e-002
Sr-89	4.7657e-005	1.7633e+006	7.4800e-006	2.7676e-001
Sr-90	3.9247e-005	1.4521e+006	6.1600e-006	2.2792e-001
Tc-99	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Te-123	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Th-228	2.8034e-006	1.0372e+005	4.4000e-007	1.6280e-002
Th-232	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
Tl-208	1.1213e-005	4.1490e+005	1.7600e-006	6.5120e-002
U-233	2.2427e-005	8.2980e+005	3.5200e-006	1.3024e-001
U-234	5.3264e-004	1.9708e+007	8.3600e-005	3.0932e+000
U-235	2.8034e-005	1.0372e+006	4.4000e-006	1.6280e-001
U-238	5.3264e-004	1.9708e+007	8.3600e-005	3.0932e+000
Y-90	3.9247e-005	1.4521e+006	6.1600e-006	2.2792e-001
Zn-65	3.2239e-004	1.1928e+007	5.0600e-005	1.8722e+000
Zr-95	8.4101e-005	3.1117e+006	1.3200e-005	4.8840e-001

**Buildup: The material reference is Source
Integration Parameters**

X Direction		20
Y Direction		20
Z Direction		20

Results

Energy (MeV)	Activity (Photons/sec)	Fluence Rate	Fluence Rate	Exposure Rate	Exposure Rate
		MeV/cm ² /sec No Buildup	MeV/cm ² /sec With Buildup	mR/hr No Buildup	mR/hr With Buildup
0.015	2.549e+03	1.841e-255	3.971e-31	1.579e-256	3.406e-32
0.02	5.409e+05	2.001e-116	1.326e-28	6.931e-118	4.592e-30
0.03	8.185e+06	5.240e-39	4.436e-27	5.193e-41	4.396e-29
0.04	3.597e+06	2.600e-19	1.027e-18	1.150e-21	4.544e-21
0.05	2.233e+05	5.750e-13	3.780e-12	1.532e-15	1.007e-14
0.06	3.569e+05	2.872e-09	2.627e-08	5.704e-12	5.218e-11
0.08	1.065e+06	7.911e-06	8.765e-05	1.252e-08	1.387e-07
0.1	1.956e+06	2.109e-04	2.272e-03	3.226e-07	3.476e-06
0.15	3.090e+06	3.608e-03	3.023e-02	5.941e-06	4.979e-05
0.2	1.887e+06	5.429e-03	3.787e-02	9.582e-06	6.684e-05
0.3	1.305e+06	9.415e-03	5.167e-02	1.786e-05	9.802e-05
0.4	5.941e+06	7.528e-02	3.507e-01	1.467e-04	6.834e-04
0.5	8.623e+06	1.661e-01	6.829e-01	3.260e-04	1.341e-03

0.6	7.568e+07	2.039e+00	7.574e+00	3.980e-03	1.478e-02
0.8	5.703e+07	2.597e+00	8.297e+00	4.939e-03	1.578e-02
1.0	7.637e+07	5.213e+00	1.494e+01	9.609e-03	2.754e-02
1.5	7.063e+07	9.972e+00	2.390e+01	1.678e-02	4.021e-02
2.0	4.688e+05	1.088e-01	2.352e-01	1.683e-04	3.638e-04
3.0	4.141e+05	1.873e-01	3.564e-01	2.541e-04	4.835e-04
Totals	3.174e+08	2.038e+01	5.646e+01	3.623e-02	1.014e-01

MicroShield 7.02
US Ecology, Inc (08-MSD-7.02-1419)

Date	By	Checked

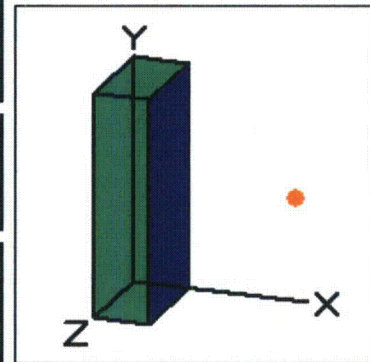
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Project Info	
Case Title	USEI Cell Operator
Description	SPFM - rho=1.5 g/cc, D=2m, 0.5" steel shield, Unat only
Geometry	13 - Rectangular Volume

Source Dimensions	
Length	91.44 cm (3 ft)
Width	182.88 cm (6 ft)
Height	381.0 cm (12 ft 6.0 in)

Dose Points			
A	X	Y	Z
#1	292.6 cm (9 ft 7.2 in)	190.5 cm (6 ft 3.0 in)	45.72 cm (1 ft 6.0 in)

Shields			
Shield N	Dimension	Material	Density
Source	6.37e+06 cm ³	Concrete	1.5
Shield 1	1.27 cm	Iron	7.86
Air Gap		Air	0.00122



Source Input: Grouping Method - Standard Indices				
Number of Groups: 25				
Lower Energy Cutoff: 0.015				
Photons < 0.015: Excluded				
Library: Grove				
Nuclide	Ci	Bq	μCi/cm ³	Bq/cm ³
Ac-227	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
Bi-210	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Bi-211	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
Bi-214	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Fr-223	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
Pa-231	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
Pa-234	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Pa-234m	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Pb-210	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Pb-211	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
Pb-214	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Po-210	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
Po-214	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Po-215	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001

Po-218	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Ra-223	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
Ra-226	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Rn-219	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
Rn-222	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Th-227	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
Th-230	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Th-231	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
Th-234	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
Tl-207	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
U-234	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000
U-235	1.7661e-005	6.5347e+005	2.7720e-006	1.0256e-001
U-238	3.0646e-004	1.1339e+007	4.8100e-005	1.7797e+000

**Buildup: The material reference is Source
Integration Parameters**

X Direction	20
Y Direction	20
Z Direction	20

Results

Energy (MeV)	Activity (Photons/sec)	Fluence Rate	Fluence Rate	Exposure Rate	Exposure Rate
		MeV/cm ² /sec No Buildup	MeV/cm ² /sec With Buildup	mR/hr No Buildup	mR/hr With Buildup
0.015	1.466e+03	1.058e-255	2.283e-31	9.079e-257	1.958e-32
0.02	8.565e+03	3.168e-118	2.099e-30	1.098e-119	7.272e-32
0.03	1.572e+05	1.006e-40	8.518e-29	9.972e-43	8.442e-31
0.04	1.636e+04	1.182e-21	4.672e-21	5.229e-24	2.066e-23
0.05	8.676e+05	2.234e-12	1.469e-11	5.952e-15	3.913e-14
0.06	8.894e+05	7.157e-09	6.547e-08	1.422e-11	1.300e-10
0.08	3.129e+06	2.324e-05	2.575e-04	3.678e-08	4.075e-07
0.1	7.481e+06	8.065e-04	8.689e-03	1.234e-06	1.329e-05
0.15	3.762e+06	4.393e-03	3.681e-02	7.234e-06	6.062e-05
0.2	4.147e+06	1.193e-02	8.323e-02	2.106e-05	1.469e-04
0.3	3.562e+06	2.569e-02	1.410e-01	4.874e-05	2.675e-04
0.4	5.205e+06	6.595e-02	3.073e-01	1.285e-04	5.988e-04
0.5	1.248e+06	2.403e-02	9.881e-02	4.716e-05	1.940e-04
0.6	9.692e+06	2.611e-01	9.699e-01	5.096e-04	1.893e-03
0.8	1.001e+07	4.556e-01	1.456e+00	8.665e-04	2.769e-03
1.0	1.063e+07	7.258e-01	2.080e+00	1.338e-03	3.834e-03
1.5	3.746e+06	5.288e-01	1.267e+00	8.897e-04	2.132e-03
2.0	3.239e+06	7.521e-01	1.625e+00	1.163e-03	2.513e-03
Totals	6.779e+07	2.856e+00	8.075e+00	5.021e-03	1.442e-02

ATTACHMENT 5

Internal Dose Calculations for Annualized SPFM Dose Assessment

USEI Internal Dose Rate Calculation Worksheet
 Studsvik SPPM - Average Annual Inventory

Rev. 0

I. Constants

Respirable dust loading (g/m^3): 2.30E-04
 Breathing rate (m^3/hr): 1.20E+00

Total Internal Dose Rate (mrem/hr)
2.395E-02

II. Nuclide-Specific Dose per Hour Calculations

Nuclide	Customer Waste Profile Concentration (pCi/g)	Assumed Transport Class	Dose Conversion Factor ¹ (Sv/Bq)	Dose Conversion Factor (mrem/pCi)	Dose Rate per Nuclide (mrem/hr)
Ag-108m	5.0	Y	7.66E-08	2.84E-04	3.92E-07
Ag-110m	25.0	Y	2.17E-08	8.04E-05	5.55E-07
Am-241	0.5	W	1.20E-04	4.44E-01	6.13E-05
Au-195	3.5	Y	3.50E-09	1.30E-05	1.25E-08
Ba-133	0.3	D	2.11E-09	7.81E-06	6.47E-10
Be-7	20.0	Y	8.67E-11	3.21E-07	1.77E-09
C-14	25.0	N/A	5.64E-10	2.09E-06	1.44E-08
Ce-139	1.0	Y	2.45E-09	9.07E-06	2.50E-09
Ce-141	11.5	Y	2.42E-11	8.96E-08	2.84E-10
Ce-144	180.0	Y	1.01E-07	3.74E-04	1.86E-05
Cm-242	0.1	W	4.67E-06	1.73E-02	4.77E-07
Cm-243	1.3	W	8.30E-05	3.07E-01	1.10E-04
Cm-244	0.5	W	6.70E-05	2.48E-01	3.42E-05
Cm-245	5.0	W	1.23E-04	4.56E-01	6.29E-04
Co-57	12.0	Y	2.45E-09	9.07E-06	3.01E-08
Co-58	200.0	Y	2.94E-09	1.09E-05	6.01E-07
Co-60	650.0	Y	5.91E-08	2.19E-04	3.93E-05
Cr-51	55.0	Y	9.03E-11	3.34E-07	5.08E-09
Cs-134	175.0	D	1.25E-08	4.63E-05	2.24E-06
Cs-137	500.0	D	8.63E-09	3.20E-05	4.41E-06
Eu-152	7.0	W	5.97E-08	2.21E-04	4.27E-07
Eu-154	2.5	W	7.73E-08	2.86E-04	1.98E-07
Eu-155	4.0	W	1.12E-08	4.15E-05	4.58E-08
Fe-55	1000.0	W	7.26E-10	2.69E-06	7.42E-07
Fe-59	8.0	W	3.30E-09	1.22E-05	2.70E-08
H-3	325.0	V	1.73E-11	6.41E-08	5.75E-09
I-125	0.1	D	6.53E-09	2.42E-05	6.68E-10
I-129	0.3	D	4.69E-08	1.74E-04	1.44E-08
I-131	18.0	D	8.89E-10	3.29E-06	1.64E-08
Mn-54	80.0	W	1.81E-09	6.70E-06	1.48E-07
Na-22	0.2	D	2.07E-09	7.67E-06	4.23E-10
Nb-94	4.0	Y	1.12E-07	4.15E-04	4.58E-07
Nb-95	25.0	Y	1.57E-09	5.81E-06	4.01E-08
Ni-59	100.0	W	2.48E-10	9.19E-07	2.54E-08
Ni-63	925.0	W	6.22E-10	2.30E-06	5.88E-07
Pu-238	5.0	Y	7.79E-05	2.89E-01	3.98E-04
Pu-239	0.3	Y	8.33E-05	3.09E-01	2.55E-05
Pu-240	0.3	Y	8.33E-05	3.09E-01	2.55E-05
Pu-241	15.0	Y	1.34E-06	4.96E-03	2.05E-05
Pu-242	0.2	Y	7.92E-05	2.93E-01	1.62E-05

II. Nuclide-Specific Dose per Hour Calculations

Nuclide	Customer Waste Profile Concentration (pCi/g)	Assumed Transport Class	Dose Conversion Factor ¹ (Sv/Bq)	Dose Conversion Factor (mrem/pCi)	Dose Rate per Nuclide (mrem/hr)
Ra-226	10.0	W	2.32E-06	8.59E-03	2.37E-05
Ru-103	1.5	Y	2.42E-09	8.96E-06	3.71E-09
Ru-106	10.0	Y	1.29E-07	4.78E-04	1.32E-06
Sb-124	8.0	W	6.80E-09	2.52E-05	5.56E-08
Sb-125	100.0	W	3.30E-09	1.22E-05	3.37E-07
Sn-113	2.0	W	2.83E-09	1.05E-05	5.79E-09
Sr-89	17.0	Y	1.12E-08	4.15E-05	1.95E-07
Sr-90	14.0	Y	3.51E-07	1.30E-03	5.02E-06
Tc-99	4.0	W	2.25E-09	8.33E-06	9.20E-09
Te-123	4.0	D	2.85E-09	1.06E-05	1.17E-08
Th-228	1.0	Y	9.23E-05	3.42E-01	9.44E-05
Th-232	4.0	Y	3.11E-04	1.15E+00	1.27E-03
U-233	8.0	Y	3.66E-05	1.36E-01	2.99E-04
U-234	190.0	Y	3.58E-05	1.33E-01	6.95E-03
U-235	10.0	Y	3.32E-05	1.23E-01	3.39E-04
U-238	190.0	Y	3.20E-05	1.19E-01	6.22E-03
Natural Uranium ²	225.0	Y	3.20E-05	1.19E-01	7.36E-03
Zn-65	115.0	Y	5.51E-09	2.04E-05	6.48E-07
Zr-95	30.0	Y	6.31E-09	2.34E-05	1.94E-07

Notes:

- 1 Dose Conversion Factors taken from FGR-11, Table 2.1 - "Inhalation Doses (Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation. Slowest transport class used.
- 2 Natural uranium modeled as U-238