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Our ref: HEM-13-71
Date: May 28, 2013

Subject: Westinghouse Hematite Decommissioning Project: Request for Third Alternate Disposal Approval and Exemptions for Waste Transferred to US Ecology Idaho (License No. SNM-00033, Docket No. 070-00036)

Dear Sirs:

Pursuant to 10 CFR 20.2002, Westinghouse Electric Company LLC (Westinghouse) requests that the U.S. Nuclear Regulatory Commission (NRC) approve alternate disposal of specified low-activity radioactive waste containing source material, byproduct material and Special Nuclear Material (SNM) from the Hematite Decommissioning Project (HDP), License No. SNM-0033. To accomplish the alternate disposal pursuant to 10 CFR 20.2002, Westinghouse also requests exemptions from 10 CFR 30.3 and §70.3 licensing requirements for byproduct material and SNM.

The receiving, treatment and disposal facility is the US Ecology Idaho, Inc. (USEI) RCRA Subtitle C treatment and disposal facility near Grand View, Idaho. Idaho is not an NRC Agreement State; however, Idaho regulations and the Grand View facility permit provide for the acceptance of this material based on the exemptions requested. This request is endorsed by USEI, which will issue separate correspondence to NRC to that effect.

Enclosure 1 is HDP-TBD-WM-909, "Safety Assessment for Third Alternative Disposal Request for Transfer of Hematite Project Waste to USEI". This document summarizes the characteristics of the candidate waste material; the proposed manner and conditions of disposal; the nature of the disposal environment; the nature and location of other potentially affected facilities; projected doses to members of the public during transport operations and to USEI workers during railcar receipt, unloading, transport, treatment, and disposal; and assessment of the potential post-closure doses, including cumulative impacts with 20.2002 requests approved as Amendments 58 and 60 to the HDP License, SNM-33.

This Safety Assessment demonstrates that this third 20.2002 request does not increase the cumulative dose estimate of 2.7 mrem from the two previous 20.2002 alternate disposal requests. This dose is consistent with NUREG-1757's Volume 1, Revision 2, "Consolidated Decommissioning Guidance - Decommissioning Guidance for Materials Licensees, Final Report," pp. 15-31 criteria of a "few millirem/yr" to a member of the public.

This third request is essentially an increase in volumes of waste for treatment and disposal using the same methods and data provided in support of Amendments 58 and 60. Enclosure 1 includes cumulative doses for all three requests.

Because this request is an extension of the 20.2002 requests approved in Amendments 58 and 60 to SNM-33, Westinghouse requests expedited review and approval of this request by August 2013.

Westinghouse suggests the following License Conditions be revised as indicated in the tracked changes:

15.d. Documents identified in the NRC's 10CFR20.2002 SERs associated with Amendment Nos. 58, ~~and~~ 60 and 63. (ADAMS Accession Nos. ML111441087, ~~and~~ ML12158A401 and ML##########)

17. Pursuant to 10 CFR 20.2002, the licensee may treat and dispose of solid materials (22,809 m^3 of soils and associated debris, ~~and~~ 23,000 m^3 of concrete/asphalt, piping, soil and miscellaneous equipment, and 22,000 m^3 of soil and soil-like waste) 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.

Please contact me at 314-810-3376, or Kevin Davis at 314-810-3348, should you have questions or need additional information.

Sincerely,

Dennis Richardson

Dennis C. Richardson
Deputy Director
Hematite Decommissioning Project

Enclosure: 1) HDP-TBD-WM-909, "Safety Assessment for Third Alternative Disposal Request for Transfer of Hematite Project Waste to USEI"

cc: J. J. Hayes, NRC/FSME/DWMEP/DURLD
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Enclosure 1

Safety Assessment for Additional Hematite Project Waste at USEI

HDP-TBD-WM-909

Westinghouse Electric Company LLC
US Ecology Idaho, Inc.

(25 pages)

Westinghouse Electric Company LLC, Hematite Decommissioning Project

Docket No. 070-00036

Hematite Decommissioning Project

Technical Basis Document

NUMBER: **HDP-TBD-WM-909**

TITLE: **Safety Assessment for Third Alternative Disposal Request for Transfer of Hematite Project Waste to USEI**

REVISION: **0**

EFFECTIVE DATE: **See Final Approved Date Below**

Approvals:

Author: Joseph S. Guido*

Owner / Manager: Joseph S. Guido*

**** Electronically approved records are authenticated in the electronic document management system. This record was final approved on May-28-2013. (This statement was added by the EDMS system to the quality record upon its validation.)***

REVISION LOG	
Revision No	**Change(s)**
0 See Cover Page	This document is needed to support a third 20.2002 alternate disposal request for transfer of HDP waste to USEI.

1.0 EXECUTIVE SUMMARY

This radiological safety assessment was developed in support of an alternate disposal request pursuant to 10 CFR 20.2002 for transfer of specified low-activity radioactive materials from the Hematite Decommissioning Project (HDP), License No. SNM-0033, to US Ecology Idaho, Inc. (USEI) Resource Conservation and Recovery Act (RCRA) treatment and disposal facility near Grand View, Idaho. This safety assessment identifies the candidate waste material; the proposed manner and conditions of treatment for RCRA hazardous waste (including volatile organics [VOCs]); the estimated doses to members of the public during transport operations and to USEI workers during receipt, unloading, transport and disposal; and provides an assessment of the potential post-closure doses.

This Safety Assessment incorporates by reference the large body of safety assessment documents performed for two previous 20.2002 alternate disposal requests for the transfer of HDP waste to this same USEI facility. References 1 and 2 document the approval of these two previous 20.2002 requests, and the associated Safety Evaluation Reports identify the documents that support those approvals. These documents are incorporated by reference into this third 20.2002 request.

The additional volume of waste in this 20.2002 request is based on encountering more material during excavation that requires disposal than was accounted for by contingencies in the original estimates. The additional material is from areas that were previously characterized, but the volumetric extent of the contaminated material was underestimated. The additions of soil-like waste (e.g., dewatered sanitary sludge) and treatment of volatile organic (VOC) waste are due NRC letters dated February 26, 2013, and March 18, 2013 (References 3 and 4). In these letters NRC concluded that HDP is not authorized to transfer dewatered sanitary sludge or mixed VOC hazardous and radiological waste to USEI for treatment unless another 20.2002 request is submitted.

This Safety Assessment demonstrates that this third 20.2002 request does not increase the cumulative limiting dose estimate from the two previous 20.2002 alternate disposal requests of 2.7 mrem. This dose is consistent with NUREG 1757's Vol.1, Rev.2, Consolidated Decommissioning Guidance - Decommissioning Guidance for Materials Licensees, Final Report, p. 15-31 criteria of a "few millirem/yr" to a member of the public.

This information and radiological safety assessment was developed in coordination with USEI.

This Safety Assessment also supports a request for NRC to exempt USEI from byproduct material and Special Nuclear Material (SNM) license requirements of 10 CFR 30.3 and 10 CFR 70.3, as allowed in 10 CFR 30.11 and 10 CFR 70.17. Such exemptions from regulation under the Atomic Energy Act for transfer (treatment and disposal purposes are consistent with the diffuse, low concentrations of contaminants in the waste and with such exemptions issued by NRC for managing equivalent or higher concentration wastes. Each railcar of waste shipped to USEI will meet the 10 CFR 40.13 requirements for unimportant quantities of source material.

2.0 USEI FACILITY OPERATING HISTORY

USEI's operating history is described extensively in the information associated with Reference 1, with supplemental information provided in the information associated with Reference 2. There have been no significant changes to USEI's operating permit, radioactive materials acceptance limits, or regulatory status since the date of Reference 2.

Subsequent to the information associated with Reference 2, USEI has received approval to dispose of additional low-activity waste. This waste is from the Pacific Gas & Electric Humboldt Bay Nuclear Power Plant (HBNPP) near Eureka, California. USEI has not received any of this additional HBNPP waste to date. There are no pending 20.2002 requests for USEI at the time of this third HDP request. The additional HBNPP approvals (References 5 and 6) at USEI are:

- HBNPP Unit 3: 2,000,000 ft^3 of soil, debris, and miscellaneous waste from decommissioning, but does not include SNM.
- HBNPP Unit 3: 100,000 ft^3 of soil, debris, piping, and miscellaneous waste (or 1375 tons at 27.5 lb/ft^3), and 50,000 ft^3 of water (or 1,560 tons at 62.4 lb/ft^3 or 1 g/cc prior to solidification). This waste may include low levels of SNM at concentrations of 2 pCi/g for U-235, 4 pCi/g for Pu-239, and 150 pCi/g for Pu-241. SNM mass concentration of waste from HBNPP would be less than 1 g/m^3, or 0.001 g/L.

All of this other SNM material was below the average 0.1 $g^{235}U/L$ concentration limit for HDP waste consigned to USEI. The concentration is further reduced since these shipments were received between 2010 and 2012 and were combined with approximately 1.7 million tons of other non-SNM soils and debris in the USEI landfill. Therefore, no additional impact needs to be evaluated.

USEI remains in compliance with its operating permit and has maintained VPP "Star" status with the OSHA Voluntary Protection Program. Information on occupational illness and injuries for 2012 for both USEI and HDP are provided in Appendix A.

3.0 DISPOSAL FACILITY CHARACTERISTICS

A description of the USEI facility and waste placement practices is found in the information associated with Reference 1. The USEI disposal cell for this waste would continue to be Cell 15, consistent with the information associated with References 1 and 2. Cell 15 has additional capacity of approximately 700,000 tons, which represents about two years of burial space at current projections. Two years is beyond the scheduled duration of HDP disposal activities. As a contingency, USEI can establish controls to reserve space within Cell 15 while material from other sites is placed into Cell 16 once that cell is operational.

4.0 USEI WASTE ACCEPTANCE CRITERIA

USEI's waste acceptance criteria (WAC) associated with this 20.2002 request is included as Appendix B.

USEI's waste acceptance criteria are maintained on their website, nominally at:
usecology.com/idaho_documents_forms.htm

Idaho rules regulating radioactive materials (IDAPA 58.01.10) are maintained on their website, nominally at:

adminrules.idaho.gov/rules/current/58/index.html

In summary, USEI is authorized to accept low concentration SNM and byproduct material if:

- The NRC specifically exempts the material under 10 CFR 30.11 §70.17 and
- The sum of all activity within the material is less than 3,000 pCi/g, and
- IDEQ reviews and concurs with the NRC exemption and USEI Safety Assessment.

5.0 MATERIAL DESCRIPTION AND CHARACTERIZATION

5.1 Material Description

Westinghouse estimates the volume of the excavated waste that is a candidate for treatment and disposal at USEI to be up to 22,000 cubic meters at an average waste density of 1.69 g/cm^3 (e.g., approximately 41,000 tons). The waste consists of soil and soil-like material (e.g., dewatered sanitary sludge) with low levels of SNM, source material, and byproduct material meeting the definition of diffuse material, as defined in the Fundamental Nuclear Material Control Plan, dated February 18, 2011, and approved by Amendment 59 to License SNM-33.

The expected concentrations for the radionuclides are based on the information associated with Reference 1, with the following exceptions: 1) the total amount of Tc-99 will not exceed the numerical limits identified in Table 2, which are unchanged from Reference 2. The radionuclides concentrations from Reference 1 are appropriate for this request since those concentration were based on an assessment of soil from across the HDP Site, are higher and thus more conservative than the Reference 2 concentrations, and the Reference 1 waste matrix (soil with debris) is consistent with this third 20.2002 request for an additional volume of soil and soil-like material. Consistent with the information provided in References 1 and 2, the overall uranium enrichment for this 22,000 m^3 of waste averages less than 10 percent.

Table 1. Expected Radionuclide Concentrations in Westinghouse Hematite Waste

Shipped Volume (m^3)	U-234 (pCi/g)	U-235 (pCi/g)	U-238 (pCi/g)	Tc-99 (pCi/g)	Ra-226 (pCi/g)	Th-232 (pCi/g)
22,000	113	5.5	18	27*	1	1.2

* The total amount of Tc-99 will not exceed the limits identified in Table 2.

The waste subject to Reference 1, Reference 2, and this request will be cumulatively counted against the limits in Table 2. Cumulatively, the total Tc-99 inventory for all three requests will be maintained at or below 1.3 Ci, and the amount of Tc-99 at the 95 percent confidence interval on the mean will be maintained below 2.05 Ci. This will ensure that the cumulative dose to the public from all three requests is maintained below 2.7 mrem based on the mean inventory and below 4.2 mrem at the 95 percent upper confidence interval on the mean. Both of these values are within the 'few millirem' criterion in NUREG 1757 for 20.2002 requests.

Consistent with Reference 1, the average total activity concentration (sum of all nuclides and progeny) for this waste is approximately 226 pCi/g or approximately 8 percent of USEI's 3,000 pCi/g total activity concentration limit.

Up to 22,000 m^3 of this waste may contain concentrations of hazardous constituents, such as metals and volatile organics, that exceed the levels identified in 40 CFR 261 for classification as hazardous waste and require treatment prior to disposal. This waste will require treatment at USEI.

Since the waste proposed for transfer to USEI will contain more than 1 gram of SNM, HDP 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).

Table 2, Contingency Plan Table

Prior to shipment, the following conditions will be evaluated, combining results for all three 20.2002 requests for HDP-USEI:

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	• Reanalyze composite sample and/or analyze individual aliquots used to create the composite sample; • Resample stockpile and re-evaluate[a]; • 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	• Reanalyze composite sample and/or analyze individual aliquots used to create the composite sample; • Resample stockpile and re-evaluate[a]; • Ship material to alternate facility.
Total activity contribution from all radionuclides within individual railcar	>3000 pCi/g > 40 μR/hrb	Laboratory sample results for stockpile evaluated at 95% UCL prior to shipment Gamma radiation levels on railcars prior to shipment.	• 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)a; • Ship material to alternate facility.
Unexpected Tc-99 results for stockpile samples (soil)	>99th percentile of the site wide dataset (573 pCi/g)c	Laboratory sample results for stockpile evaluated prior to shipment	• Analyze additional aliquot of composite sample; • Resample stockpile and re-evaluate[a]; • Blend with less contaminated material, resample stockpile and re-evaluate; • Ship material to alternate facility.
Unexpected Tc-99 results for concrete slab samples	>99th percentile of the dataset (1590 pCi/g)	Laboratory sample results for concrete slabs evaluated prior to shipment	• Analyze additional aliquot of sample; • Resample and re-evaluate[a]; • Blend with less contaminated material, resample and re-evaluate; • Ship material to alternate facility.
Unexpected Tc-99 results for stockpile samples (piping internal debris / residue)	>99th percentile of the dataset (162 pCi/g)	Laboratory sample results for stockpile evaluated prior to shipment	• Analyze additional aliquot of composite sample; • Resample stockpile and re-evaluate[a]; • Blend with less contaminated material, resample stockpile and re-evaluate; • Ship material to alternate facility.

Table 2, Contingency Plan Table

Prior to shipment, the following conditions will be evaluated, combining results for all three 20.2002 requests for HDP-USEI:

Parameter	Action Level	How Monitored	Actions
Unexpected Tc-99 results for stockpile samples (piping average concentration)	>99th percentile of the dataset (125 pCi/g)	Laboratory sample results for stockpile evaluated prior to shipment	• Analyze additional aliquot of composite sample; • Resample stockpile and re-evaluate[a]; • 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	• Analyze additional aliquot of composite sample; • Resample stockpile and re-evaluate[a]; Blend with less contaminated material, resample stockpile and re-evaluate; • Ship material to alternate facility.

[a] Re-sampling 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 in-situ railcars) then, the additional samples will be used to augment the initial dataset.

[b] Based on analysis previously transmitted in HEM-10-46, 5/24/10.

[c] 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 spreadsheet software.

5.2 Waste Characterization Plan

Detailed characterization data and accompanying analyses for soil are contained in Reference 1. This characterization information continues to apply to the waste subject to this request. The increase in volume is due to an underestimate in the volume of soil that require disposal due to contamination, but the nature of the contamination is consistent with the Reference 1 characterization.

Westinghouse will subject soils, which may include spent limestone used as backfill, associated with this request to the same sampling plan that was detailed in Reference 1, the sample results will be compared to the combined contingency plan limits in Table 2, and will use the same radiological controls and programmatic elements detailed in Reference 1.

The characterization of soil-like material, e.g., dewatered sanitary system sludge, for shipment will follow the same requirements associated with Reference 1 soils prior to their shipment for treatment and disposal. Available characterization data of sanitary system sludge is provided in Table 3. HDP will take additional samples of the sanitary system sludge for laboratory analysis for Tc-99, isotopic uranium, and gamma spectroscopy. These sample results will be submitted to NRC to augment this 20.2002 request.

The approach used to sample soil and soil-like material associated with this request will be identical to that approved by Reference 1. The sample results will be compared to the contingency plan limits shown in Table 2, above.

Sample analysis for Tc-99 will follow the methodology currently employed, which is inductively coupled plasma mass spectrometry. Since this analysis is destructive, the sample aliquot is not returned; however, the excess portion of the sample is returned.

Table 3. Sample Results from the Sanitary Sludge

	Potassium-40 (pCi/g)			Radium-226 (pCi/g)			Thorium-232 (pCi/g)			Uranium-235/236 (pCi/g)			Uranium-238 (pCi/g)		
Sample ID	**Result**	**Error**	**MDC**	**Result**	**Error**	**MDC**	**Result**	**Error**	**MDC**	**Result**	**Error**	**MDC**	**Result**	**Error**	**MDC**
4111-WM-130110-01-01	14.11	2.08	1.31	1.37	0.18	0.32	ND		0.71	1.55	0.25	0.34	3.92	3.62	5.89
4111-WM-130110-01-02	12.10	1.79	1.12	1.09	0.14	0.25	ND		0.61	3.33	0.31	0.29	10.80	12.32	5.97
4111-WM-130110-01-03	13.90	2.08	1.38	1.32	0.16	0.12	ND		0.76	1.11	0.21	0.30	ND		7.28
4111-WM-130110-01-04	13.46	2.00	1.47	1.66	0.17	0.11	ND		0.81	44.36	1.90	0.29	183.47	19.78	9.25
4111-WM-130110-01-05	11.34	1.73	1.33	1.65	0.16	0.25	ND		0.66	48.16	1.99	0.45	184.70	18.72	8.72
4111-WM-130110-01-06	14.53	2.10	1.29	1.14	0.15	0.26	ND		0.75	3.07	0.30	0.34	9.95	3.66	6.28
4111-WM-130110-01-07	14.98	2.22	1.55	1.40	0.18	0.13	0.58	0.20	0.34	17.84	0.98	0.41	69.02	13.23	7.45
4111-WM-130110-01-08	14.10	1.98	0.81	1.02	0.15	0.22	ND		0.74	0.27	0.14	0.22	ND		6.61
4111-WM-130110-01-09	10.81	1.75	1.21	1.11	0.15	0.29	ND		0.71	0.23	0.15	0.24	ND		6.40
4111-WM-130110-01-11	14.85	2.18	1.33	1.14	0.15	0.27	ND		0.78	1.87	0.27	0.35	5.36	3.69	5.88
4111-WM-130110-01-12	10.02	1.64	1.23	1.05	0.14	0.24	0.47	0.19	0.31	2.29	0.25	0.28	16.37	17.23	5.33
4111-WM-131110-01-10	14.52	2.04	1.01	1.16	0.14	0.25	ND		0.76	3.07	0.29	0.27	10.13	3.46	6.41

ND – not detected.

6.0 TRANSPORT AND DISPOSAL DOSE ASSESSMENTS

The waste in this third 20.2002 request will be shipped by rail using the same methods and requirements as described in the information associated with Reference 2. The dose assessment methodology for rail transport and USEI treatment and disposal activities associated with Reference 2 is unchanged in this request. The information for the Stabilization Operator is updated from Reference 2 to provide additional information on the treatment activities. This information on treatment activities was initially provided to NRC in Westinghouse letter HEM-12-77 (Reference 7). The dose estimates associated with transport and USEI workers are evaluated for impacts by the increase in the volume of the waste and the treatment of the waste by USEI.

For the public along the transport route, the dose would be less than the more restrictive dose calculated below for the USEI work, as discussed in the information supporting Reference 2. In order for the dose to a bystander during rail transport to exceed that of the site worker and therefore be bounding, the individual would have to spend more than 1000 hrs at 1 meter from the gondola railcar or 790 hrs at 1 foot, which are not credible external exposure scenarios during transportation. Since the waste within the gondola railcars is contained during transport, no internal dose would be assigned to a by-stander.

The dose estimate for USEI workers is based on the dose estimates associated with References 1 and 2. These estimates were then modified for the volume and density of waste being transported and treated in this 20.2002 request. More specifically, the dose estimates for USEI workers in Reference 1 are carried forward in the development of this assessment, with the exception of the gondola surveyor and stabilization operator. Reference 2 identified revised dose estimates for gondola surveyor and stabilization operator from Reference 1, so those Reference 2 estimates are carried forward in the development of this assessment. Appendix C contains the Microshield®[1] calculations for the stabilization operator. From those baseline estimates, the number of railcars is adjusted from 400 in Reference 1 to 372 (372 is based on 110 tons per railcar). The number of cubes for the Stabilization Operator is increased to 372 to reflect up to 100 percent of the waste in this 20.2002 request being treated, as compared to 5 percent in Reference 1. Section 6.1, provides a detailed summary of the waste stabilization process. Table 4 contains the exposure estimates for the USEI workers.

The potential cumulative USEI worker doses are estimated by the following equation, and the results are in the right column of Table 4. This equation is based on the 58 percent of Reference 1 waste that was not shipped as of December 31, 2012, 100 percent of the waste associated with Reference 2, 100 percent of the waste associated this 20.2002 request are conservatively assumed to be disposed at USEI in 2013.

$$\text{Cum. Dose } 2013 = 0.58(\text{Dose Reference 1}) + (\text{Dose Reference 2}) + (\text{Dose Reference 3})$$

The highest cumulative dose is 0.86 mrem/year, which is less than the bounding post-closure dose of 2.7 mrem/year.

[1] MicroShield is a trademark of Grove Software, Inc., registered in the U.S. and other countries.

6.1 Stabilization Operator

Up to 22,000 m^3 of the HDP waste may contain chemical constituents at concentrations identified in 40 CFR 261 to be hazardous waste. Such constituents are Tetrachloroethylene (D039), Trichloroethylene (D040), Vinyl Chloride (D043), Arsenic (D004), Mercury (D009), and Lead (D008). USEI's RCRA Part B permit allows the facility to accept this waste, treat it to meet USEPA Land Disposal Restrictions and dispose it in the facility's disposal cell.

All treatment activities will be conducted inside a containment building, with 50,000 cfm of negative airflow. The waste will be mixed by a stabilization operator. Six stabilization operators share this task. It normally takes the stabilization operator up to 45 minutes to perform this operation. The operator occupies a position approximately 2.8 meters from the material. The estimated radiation field, internal and external dose rates per gondola railcar and per operator are provided in Table 4 (based on up to 372 gondola cars containing waste with constituents requiring stabilization).

Stabilization treatments applicable to waste shipped from the Hematite site to the USEI site are as follows:

1. Stabilization is a proven treatment process that irreversibly bonds target molecules into an environmentally inert material that reduces the leachability of the contaminants of concern. Stabilization uses limebearing material such as Portland cement, kiln dust, or other lime sources. Stabilization results from the chemical reaction of the lime, waste and water (supplied or in the waste). All of these reactions contribute to the reduced leachability of the constituents of concern.
2. Chemical Reduction involves reagents which, when used in conjunction with stabilization, reduce the leachability of inorganic and organic constituents. Reducing reactions, oxidation reactions, and competing reactions may all occur during the use of these reagents. These processes allow the formation of compounds, which are insoluble or have significantly reduced solubility.
3. Macro-encapsulation is a confining or immobilization process used to treat all types of hazardous debris independent of the hazardous constituents involved. The macro-encapsulation process encases the debris to provide a physical barrier that prevents/minimizes potential leaching of hazardous constituents from the debris.
4. Micro-encapsulation is a confining or immobilization process that requires the stabilization of the debris with the following types of reagents (or waste reagents) such that the leachability of the hazardous contaminants is reduced: (1) Portland cement; or (2) lime/pozzolans (e.g., fly ash and cement kiln dust). Additional reagents (e.g., iron salts, silicates, carbon, polymers or clays) may be used as appropriate.

Table 4. Summary of Doses to USEI Workers during Transport, Treatment and Disposal of Westinghouse Hematite Waste

	Input Parameters				External dose rate (mrem/hr)	Dose (mrem/conveyance)		Individual Dose (mrem/worker)			Cumulative Dose Estimate for 2013 (mrem/ worker)
	No. of Workers	Minutes to perform task[1]	Distance from object (meters)	Type of conveyance modeled (count)[2]		Internal[3]	External[4]	Internal[5]	External[6]	Total[7]	
Gondola surveyor	8	20	1	Gondola (372)	2.5E-03	1.8E-03	8.5E-04	NA[8]	4.0E-02	4.0E-02	1.2E-01
Excavator operator	4	45	2	Gondola (372)	4.2E-04	4.1E-03	3.2E-04	3.8E-01	3.0E-02	4.1E-01	8.6E-01
Gondola Cleanout	8	10	1	Gondola (372)	1.6E-03	9.0E-04	2.7E-04	4.2E-02	1.3E-02	5.4E-02	1.1E-01
Truck Surveyor	8	5	1	Truck (1117)	2.0E-03	4.5E-04	1.7E-04	NA[8]	2.4E-02	2.4E-02	8.0E-02
Truck Driver	14	45	0.6	Truck (1117)	2.3E-03	4.1E-03	1.7E-03	NA[8]	1.4E-01	1.4E-01	4.3E-01
Stabilization Operator	6	45	2.8	Cube (372)	6.9E-04	4.1E-03	5.2E-04	2.5E-01	3.9E-02	2.9E-01	3.1E-01
Cell Operator	2	15	1	Gondola (372)	2.3E-03	1.4E-03	5.7E-04	2.6E-01	1.1E-01	3.7E-01	7.2E-01

1 The minutes assigned for each job function listed in Table 4 are the times estimated by knowledgeable and experienced site personnel for one person to perform each function one time.

2 Calculations based on volume of material in a gondola railcar.

3. Internal dose per conveyance calculated based on the product of the intake quantity of 0.23 mg/m3 of respirable dust, 1.2 m3/hr inhalation rate, individual radionuclide concentrations based on Table 1, dose conversion factors from FGR 11, and the handling times shown.

4 External dose per conveyance calculated based on the product of the external dose rate and handling time indicated.

5 Internal dose per individual worker calculated based on the internal dose per conveyance times the number of conveyances per year and divided by the total number of workers.

6 External dose per individual worker calculated based on the external dose per conveyance times the number of conveyances per year and divided by the total number of workers.

7 Total dose per individual worker is the sum of the internal dose per individual and external dose per individual.

8 Internal dose is not applicable for this job function because waste remains in the conveyance and is covered.

7.0 POST-CLOSURE AND INTRUDER DOSE ASSESSMENTS

The dose assessment methodology for post-closure (1000 years) and intruder dose associated with Reference 2 is unchanged in this request. The results for post-closure and intruder dose are evaluated for impacts by the increase in the volume of the waste and the additional amount of radioactivity, except for Tc-99, which is capped at the Reference 2 limits in Table 2 of this request.

7.1 Post-Closure Analysis

As a conservative parameter, the waste material referenced in this submittal is estimated to be removed and disposed of at USEI within calendar year 2013. If the work extends into 2014, that will reduce the estimated post-closure exposure because the additional time will allow more material from other waste streams to comingle and dilute this HDP waste.

The post-closure analysis associated with References 1 and 2 determined that Tc-99 was the only nuclide contributing to the post-closure dose. Since the cumulative amount of Tc-99 is not being increased with this third 20.2002 request, there will be no increase in the post-closure dose from Tc-99 in this waste. In essence, the additional volume of waste associated with this third 20.2002 request amounts to diluting HDP waste with respect to Tc-99.

The dose contributions from the other radionuclides are on the order of 10^{-4} to 10^{-30} mrem/yr from the information associated with References 1 and 2. The volume of HDP waste is being increased by 50 percent over the combined volume associated with References 1 and 2. A 50 percent increase in a dose on the order of 10^{-4} mrem/yr will be on the order of 10^{-3} mrem/yr. The effect of less non-HDP waste for commingling and dilution should also be considered. Assuming the amount of non-HDP waste is reduced by 50 percent, the resulting dose would double, but still be less than 10^{-2} mrem/yr from radionuclides other than Tc-99. The post closure dose contribution from Tc-99 remains the controlling (highest) dose.

The maximum shipping schedule was established in References 1 and 2 at 20 railcars per week. The combined effect of a shipping schedule of 20 railcars per week for all three requests is calculated as follows, considering the 14,210 m^3 (62 percent) of waste shipped through March 31, 2013, associated with Reference 1. The dose contribution from Reference 1 waste is weighted according to the shipping schedule. From Reference 1 the dose associated with the nominal shipping rate is 1.9 mrem and the dose associated with the maximum shipping rate is 4.1 mrem; these doses are from Tc-99. From Reference 2, the dose from the maximum shipping rate is 1.6 mrem (this dose is from Tc-99). Since this 20.2002 request does not increase the amount of Tc-99 from the combined amounts from References 1 and 2, there is no contribution in the following equation.

$$\text{Cumulative Dose} = 0.62(1.9\ \text{mrem}) + 0.38(4.1\ \text{mrem}) + (1.6\ \text{mrem})$$

The cumulative dose from the maximum shipping rate is 4.3 mrem.

7.2 Intruder Scenario

The intruder construction scenario performed for this request is based on the same methodology associated with Reference 1. The lack of an agricultural pathway for the intruder scenario precludes pathways for the Tc-99 to significantly contribute to the dose

when compared to uranium. Accordingly, only uranium is considered for the intruder scenario. Since the volume of waste for this third 20.2002 request is slightly less than the volume of waste associated with Reference 1, the doses from the three intruder scenarios in Reference 1 are bounding for this third request.

The highest concentration of the isotopic mixture considered in Reference 1 is the disposal of waste at the USEI WAC within a 1 ft layer. This concentration is Ra-226 at 1.3E+01, Pb-210 at 1.3E+01, Tc-99 at 3.6E+02, Th-232 at 1.6E+01, Ra-228 at 1.6E+01, Th-228 at 1.6E+01, U-234 at 1.5E+03, U-235 at 7.3E+01, and U-238 at 2.4E+02.

Intruder Scenario	**Description**	**Estimated Dose** (mrem/yr)
Acute Well-Driller	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.	2.9
Chronic Well-Driller	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).	2
Construction Intruder	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.	10

Scenario	**Maximum Dose Reference 1** (mrem/yr)	**Maximum Dose Reference 2** (mrem/yr)	**Maximum Dose this Request** (mrem/yr)	**Cumulative Maximum Dose** (mrem/yr)
Intruder Construction	10	16	10	36
Intruder Acute Well Drilling	2.9	2.9	2.9	8.7
Intruder Chronic Well Drilling	2	3	2	7

This conservative analysis of cumulative intruder dose yields a dose less than the Part 20 dose limit of 100 mrem/yr to members of the public. This cumulative dose is unrealistic since it assumes the radioactivity from all three maximum scenarios are lined up in proximity together at the time of intrusion.

7.3 Sensitivity Analysis

The sensitivity analyses associated with References 1 and 2 demonstrated that the dose estimates are not highly sensitive to small changes in parameters. Since the methods of

calculating dose are the same as Reference 1 and 2, these sensitivity analyses remain valid for this third 20.2002 request.

8.0 NUCLEAR CRITICALITY SAFETY

Nuclear criticality safety assessments, NSA-TR-09-14 and NSA-TR-HDP-11-11 (References 8 and 9), demonstrate that the treatment and disposal of Hematite decommissioning waste at the USEI site can be safely performed. The assessments have determined that there are very large margins of safety under normal (i.e., expected) conditions and that there is considerable tolerance to abnormal conditions. Under all foreseen abnormal conditions a criticality event is considered either not credible or is precluded by controls in place at the Hematite site.

NSA-TR-HDP-11-11 includes assessment of the sanitary sludge being considered under this request as part of soil-like material.

This analysis applies to disposal of Hematite decommissioning wastes with a maximum average fissile nuclide concentration of 0.1g ^{235}U/L at the USEI site. This average fissile nuclide concentration is assessed on a per railcar basis. The conservative average of 0.1g^{235}U/L is derived based on the following analysis.

- NUREG/CR-6505 Vol. 1 derives a minimum critical infinite sea concentration of 1.4 g^{235}U/L (39.6 g^{235}U/ft^3) for a fictitious bounding medium consisting of only SiO_2 and ^{235}U.
- Section 10 of NUREG/CR-6505 (Ref. 15, pg. 45) concludes that a concentration factor of greater than ten is not considered credible for migration of 235U based on the hydrogeochemical modeling.

The treatment methods will not result in the waste exceeding an average of 0.1 g^{235}U/L, as described and analyzed in Nuclear Criticality Safety Assessments, NSA-TR-09-14 or NSA-TR-HDP-11-11. While these two documents discuss five percent of the waste requiring treatment, the amount being treated is not a limitation to ensuring nuclear criticality safety, which is achieved on a concentration basis. The estimate of waste, approximately 22,000 m^3, requiring stabilization prior to disposal, will exceed the five percent estimate mentioned in the NCSAs.

As discussed in Section 1 above, non-HDP SNM material shipped to USEI has been below the average 0.1 g^{235}U/L concentration limit for HDP waste consigned to USEI. USEI disposal cell 15 includes not only HDP and non-HDP SNM waste, but approximately 1.7 million tons of other non-SNM soils and debris disposed between 2010 and 2012. Therefore, no additional impact needs to be evaluated.

9.0 RECORDS OF TRANSFER

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. Since USEI would be exempted from the 10 CFR 70.3 requirement of an NRC licensee to possess SNM, the §70.42 requirement would not apply. However, Westinghouse will maintain as an alternative written registration certificate a copy of the permit issued to USEI by the State of Idaho and NRC approval of this alternate

disposal request for specific HDP wastes. DOE/NRC Form 741, *Nuclear Material Transaction Report*, would be used by Westinghouse to document transfers of SNM to the disposal facility. USEI will report SNM receipts using its existing account with the Nuclear Materials Management & Safeguards System.

10.0 REFERENCES

Note: The "ML#" is the accession number for the NRC ADAMS document database that is accessible via the internet.

1. NRC (McConnell) letter to Westinghouse (Hackmann), dated October 27, 2011, Issuance of Hematite Amendment No. 58 Approving Westinghouse Hematite Request for Alternate Disposal of Soil and Debris and Granting Exemptions to 10 CFR 30.3 and 10 CFR 70.3, (ML112560105). This reference includes the referenced documents in the Safety Evaluation Report enclosed in this letter.
2. NRC (Camper) letter to Westinghouse (Richardson), dated April 11, 2013, Issuance of Hematite Amendment No. 60 Approving Westinghouse Hematite Request for Alternate Disposal of Specified Low-Activity Radioactive Material and Granting Exemptions to 10 CFR 30.3 and 10 CFR 70.3 (ML12158A372). This reference includes the referenced documents in the Safety Evaluation Report enclosed in this letter.
3. NRC (Hayes) letter to Westinghouse (Copp), dated February 26, 2013, "Disposal of Sanitary Waste Solids and Water Treatment Materials at U.S. Ecology Idaho" (ML13036A331)
4. NRC (Hayes) letter to Westinghouse (Copp), dated March 18, 2013, "The U.S. Nuclear Regulatory Commission's Assessment of Westinghouse Hematite Unreviewed Safety Question Involving the Shipment of Radiologically Contaminated Waste Containing Volatile Organic Compounds for the Treatment of the Organics at U.S. Ecology" (ML13038A141)
5. NRC (McConnell) letter to Humboldt Bay (Conway) dated April 25, 2012, "Humboldt Bay Power Plant Unit 3 – Request for 10 CFR 20.2002 Alternate Disposal Approval and 10 CFR 30.11 Exemption for Plant Waste Disposal at US Ecology Idaho" (ML120620454)
6. NRC (Camper) letter to USEI (Weismann) dated December 19, 2012, "Request for 10 CFR 20.2002 Alternate Disposal Approval and Exemptions from 10 CFR 30.11 and 10 CFR 70.17 for Humboldt Bay Plant Waste Disposal at US Ecology Idaho" (ML12299A056)
7. Westinghouse (Copp) letter to NRC (Document Control Desk), HEM-12-77, dated July 30, 2012, "Additional Information on the Hematite Nuclear Criticality Safety Assessment of the US Ecology Idaho (USEI) Site for the Land Fill Disposal of Decommissioning Waste from the Hematite Site, Revision 6, July 2012" (ML12215A351)
8. NSA-TR-09-14, "Nuclear Criticality Safety Assessment of the US Ecology Idaho (USEI) Site for the Land Fill Disposal of Decommissioning Waste from the Hematite Site"
9. NSA-TR-HDP-11-11, "Nuclear Criticality Safety Assessment of the US Ecology Idaho (USEI) Site for the Land Fill Disposal of Additional Decommissioning Waste from the Hematite Site

Appendix A
HDP and USEI Occupational Injury and Illness Data

Work-related injuries at the HDP

Year	Work Hours	Injuries	OSHA Recordable Injury/Illness	Fatalities	Injuries per 10,000 hours
2001	438,404	67	50	0	1.5
2002	115,832	11	5	0	1.0
2003	86,736	1	0	0	0.1
2004	52,208	0	0	0	0
2005	169,739	18	3	0	1.1
2006	144,480	26	1	0	1.8
2007	57,760	0	0	0	0
2008	114,000	0	0	0	0
2009	120,623	0	0	0	0
2010	111,015	1	1	0	0.2
2011	146,727	5	0	0	0.3
2012	161,813	15	6	0	0.9
TOTAL	**1,719,337**	**144**	**66**	**0**	**N/A**

Work-related injuries at the USEI

Year	Work Hours	Injuries	OSHA Recordable Injury/Illness	Fatalities	Injuries per 10,000 hours
2001	87,362	9	5	0	1.0
2002	81,707	8	3	0	1.0
2003	93,490	18	2	0	1.9
2004	94,872	16	3	0	1.7
2005	121,048	20	4	0	1.6
2006	158,800	22	5	0	1.4
2007	180,683	40	7	0	2.2
2008	179,072	30	3	0	1.7
2009	149,929	16	3	0	1.1
2010	117,151	14	2	0	1.2
2011	133,366	5	2	0	0.4
2012	120,251	12	3	0	1.0
TOTAL	**1,517,731**	**210**	**42**	**0**	**N/A**

Appendix B
USEI Waste Acceptance Criteria

USEI Part B Permit
EPA ID. No.: IDD073114654
Revision Date: September 25, 2008

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.

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

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

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

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

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

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

Table C.3: Particle Accelerator Produced Radioactive Material

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

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

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

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

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

Exemption 10 CFR Part*	Product, Device or Item	Isotope, Activity or Concentration
30.15	As listed in the regulation	Various isotopes and activities as set forth in 30.15
30.14, 30.18	Other materials, products or devices specifically exempted from regulation by rule, order, license, license condition, concurrence, or letter of interpretation	Radionuclides in concentrations consistent with the exemption
30.19	Self-luminous products containing tritium, ^{85}Kr, ^{3}H or ^{147}Pm	Activity by Manufacturing license
30.20	Gas and aerosol detectors for protection of life and property from fire	Isotope and activity by Manufacturing license
30.21	Capsules containing ^{14}C urea for *in vivo* diagnosis of humans	^{14}C, one µCi per capsule
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.

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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.

Appendix C
Microshield Calculation for Stabilization

Case Summary of USEI Stab. Worker — Page 1 of 3

MicroShield 7.02
Westinghouse Electric Company (08-MSD-7.02-1424)

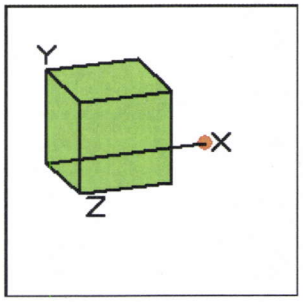
Date	By	Checked

Filename	Run Date	Run Time	Duration
HDP Stabilization Operator.ms7	May 21, 2013	6:15:57 PM	00:00:01

Project Info	
Case Title	USEI Stab. Worker
Description	USEI SSPA, Density=1.69 g/cc
Geometry	13 - Rectangular Volume

Source Dimensions	
Length	385.572 cm (12 ft 7.8 in)
Width	385.572 cm (12 ft 7.8 in)
Height	385.572 cm (12 ft 7.8 in)

Dose Points			
A	X	Y	Z
#1	665.988 cm (21 ft 10.2 in)	0.0 cm (0.0 in)	0.0 cm (0.0 in)



Shields			
Shield N	Dimension	Material	Density
Source	5.73e+07 cm³	Concrete	1.69
Air Gap		Air	0.00122

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

Nuclide	Ci	Bq	µCi/cm³	Bq/cm³
Ac-227	1.9593e-007	7.2496e+003	3.4182e-009	1.2647e-004
Ac-228	1.1372e-004	4.2076e+006	1.9839e-006	7.3404e-002
Bi-210	6.8637e-005	2.5396e+006	1.1974e-006	4.4304e-002
Bi-211	1.9498e-007	7.2141e+003	3.4015e-009	1.2585e-004
Bi-212	1.1326e-004	4.1905e+006	1.9758e-006	7.3105e-002
Bi-214	9.5787e-005	3.5441e+006	1.6711e-006	6.1829e-002
Fr-223	2.7039e-009	1.0004e+002	4.7171e-011	1.7453e-006
Pa-231	4.5093e-007	1.6685e+004	7.8667e-009	2.9107e-004
Pa-234	2.7514e-006	1.0180e+005	4.8000e-008	1.7760e-003
Pa-234m	1.7196e-003	6.3627e+007	3.0000e-005	1.1100e+000
Pb-210	6.8654e-005	2.5402e+006	1.1977e-006	4.4315e-002
Pb-211	1.9498e-007	7.2141e+003	3.4015e-009	1.2585e-004
Pb-212	1.1326e-004	4.1905e+006	1.9758e-006	7.3105e-002
Pb-214	9.5787e-005	3.5441e+006	1.6711e-006	6.1829e-002
Po-210	6.8167e-005	2.5222e+006	1.1892e-006	4.4001e-002

Po-211	5.3228e-010	1.9695e+001	9.2860e-012	3.4358e-007
Po-212	7.2563e-005	2.6848e+006	1.2659e-006	4.6839e-002
Po-214	9.5767e-005	3.5434e+006	1.6707e-006	6.1816e-002
Po-215	1.9498e-007	7.2142e+003	3.4015e-009	1.2585e-004
Po-216	1.1326e-004	4.1905e+006	1.9758e-006	7.3105e-002
Po-218	9.5807e-005	3.5448e+006	1.6714e-006	6.1842e-002
Ra-223	1.9498e-007	7.2142e+003	3.4015e-009	1.2585e-004
Ra-224	1.1326e-004	4.1905e+006	1.9758e-006	7.3105e-002
Ra-226	9.5806e-005	3.5448e+006	1.6714e-006	6.1841e-002
Ra-228	1.1372e-004	4.2076e+006	1.9839e-006	7.3404e-002
Rn-219	1.9498e-007	7.2142e+003	3.4015e-009	1.2585e-004
Rn-220	1.1326e-004	4.1905e+006	1.9758e-006	7.3105e-002
Rn-222	9.5807e-005	3.5448e+006	1.6714e-006	6.1842e-002
Tc-99	2.6364e-003	9.7548e+007	4.5994e-005	1.7018e+000
Th-227	1.9264e-007	7.1277e+003	3.3607e-009	1.2435e-004
Th-228	1.1326e-004	4.1906e+006	1.9759e-006	7.3107e-002
Th-230	3.9207e-006	1.4507e+005	6.8399e-008	2.5308e-003
Th-231	5.3309e-004	1.9724e+007	9.3000e-006	3.4410e-001
Th-232	1.1464e-004	4.2418e+006	2.0000e-006	7.4000e-002
Th-234	1.7196e-003	6.3627e+007	3.0000e-005	1.1100e+000
Tl-207	1.9444e-007	7.1944e+003	3.3922e-009	1.2551e-004
Tl-208	4.0693e-005	1.5056e+006	7.0991e-007	2.6267e-002
U-234	1.0890e-002	4.0293e+008	1.8998e-004	7.0293e+000
U-235	5.3309e-004	1.9724e+007	9.3000e-006	3.4410e-001
U-238	1.7196e-003	6.3627e+007	3.0000e-005	1.1100e+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 MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	7.916e+07	3.760e-12	4.132e-12	3.225e-13	3.544e-13
0.02	7.341e+01	3.641e-13	4.281e-13	1.261e-14	1.483e-14
0.03	2.892e+06	1.194e-05	1.657e-05	1.184e-07	1.642e-07
0.04	4.301e+04	1.263e-06	2.076e-06	5.587e-09	9.181e-09
0.05	6.182e+05	4.306e-05	8.113e-05	1.147e-07	2.161e-07
0.06	2.614e+06	3.032e-04	6.735e-04	6.022e-07	1.338e-06
0.08	5.034e+06	1.088e-03	2.886e-03	1.721e-06	4.566e-06
0.1	6.433e+06	2.053e-03	6.049e-03	3.141e-06	9.254e-06
0.15	3.282e+06	1.927e-03	6.083e-03	3.174e-06	1.002e-05
0.2	1.485e+07	1.307e-02	4.067e-02	2.307e-05	7.178e-05

0.3	1.840e+06	2.846e-03	8.234e-03	5.399e-06	1.562e-05
0.4	1.464e+06	3.395e-03	9.155e-03	6.616e-06	1.784e-05
0.5	6.548e+05	2.088e-03	5.313e-03	4.099e-06	1.043e-05
0.6	3.079e+06	1.277e-02	3.085e-02	2.493e-05	6.022e-05
0.8	1.884e+06	1.190e-02	2.660e-02	2.263e-05	5.060e-05
1.0	4.235e+06	3.726e-02	7.872e-02	6.868e-05	1.451e-04
1.5	1.253e+06	2.033e-02	3.895e-02	3.421e-05	6.554e-05
2.0	9.630e+05	2.418e-02	4.373e-02	3.739e-05	6.763e-05
3.0	1.503e+06	6.933e-02	1.160e-01	9.406e-05	1.574e-04
Totals	**1.318e+08**	**2.026e-01**	**4.141e-01**	**3.300e-04**	**6.877e-04**