SAFETY EVALUATION REPORT

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

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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 m^2 (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/cm^3 (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 to1.5 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 m^3) than the projected amount to be sent to USEI (22,809 m^3) 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 m^3 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 m^3 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 m^3 (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 m^3. 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 711,00 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 μR/1.2μR/hr@1m), or 326 hr at 1 foot away (490 μR/ 1.5μR/ 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/m^3), an inhalation rate (1.2 m^3/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/m^3, 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 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 underone 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 occured 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 m^2, 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 m^3) 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 m^3 spread over half the cell area, or 40,469 m^2. 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 m^3, and this is spread out over 500 m^2 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 ^{99}Tc 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 yd^3 of waste material, whereas 704 samples would be approximately one per 42 yd^3 (based on the projected volume to be shipped of 22,809 m^3).

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 m^3) 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	• 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	• 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 μR/hr[a]	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);[c] • Ship material to alternate facility.
Unexpected Tc-99 results for stockpile samples	>99th percentile of the site wide dataset (599 pCi/g)[b]	Laboratory sample results for stockpile evaluated prior to shipment	• 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	• 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 yd^3 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 yd^3 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 (minutes)	Ra-226 MDA[a] (pCi/g)	Th-232 MDA (pCi/g)	U-235 MDA (pCi/g)	U-238 MDA (pCi/g)
10	10/1	1	1	10

[a]MDA values shown for both direct analysis using 186 keV peak (higher value) and indirect analysis using daughters (i.e., Bi-241/Pb-214).

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

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

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

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

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

Count Time (minutes)	Natural U (pCi/g)	5 wt% U-235 (pCi/g)	20 wt% U-235 (pCi/g)	95 wt% U-235 (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-galllon 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 “SiO_2 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/ft^3). 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 vertification 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 RECONCENTRATON

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 x 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 m^3 (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 milirem". 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.