ATTACHMENT 1

Responses to Requests for Additional Information on Decommissioning Plan Chapters 10 and 12

Westinghouse Electric Company LLC,
Hematite Decommissioning Project

Docket No. 070-00036

Responses to Requests for Additional Information on Decommissioning Plan Chapters 10 and 12

NRC issued requests for additional information (RAI) concerning the Hematite Decommissioning Plan (DP) in letter dated December 1, 2010. Westinghouse Electric Company LLC (Westinghouse) provides responses to those RAIs herein. Some of the responses will result in changes, as noted, to the DP. Those changes will be provided under separate cover.

These RAI responses are organized in the same manner as the RAIs of NRC letter dated December 1, 2010. For each RAI, the NRC's Comment, Basis and Path Forward is reiterated, followed by the Westinghouse Response.

Hematite Decommissioning Plan Chapter 10 - Health & Safety During Decommissioning

1. (HDP-10-Q1, Section 10.2.2.1)

   Comment: Section 10.2.2.1 of the Hematite Decommissioning Plan (DP) states that "When monitoring is required for the purpose of determining occupational exposure, sampling is accomplished through the use of a personal air sampler (lapel pump), or a portable low volume air sampler. The personal air sampler is the preferred method because the filter cartridge can be easily located within approximately 12 inches of the worker's head during sample collection, increasing the probability of being representative of the concentration in the worker's breathing zone." These statements do not make a clear commitment to providing an air sampling program representative of worker's breathing zones. There should be a commitment to using a lapel pump within the workers' breathing zones, or adequate justification should be provided to demonstrate that a portable low volume air sampler will be representative of the breathing zone. Simply calling one method "preferred" is not a commitment to using that method.

   Basis: Per guidance in NUREG-1757, Vol. 1, Rev. 2, Appendix D, Section X.a., licensees should provide "a description which demonstrates that the air sampling program is representative of the workers' breathing zones."

   Path Forward: Provide a description which demonstrates that the air sampling program is representative of the workers breathing zones. Clarify statements on preferred methods, so it is clear what methods will Westinghouse actually be committing to use for breathing zone air sampling.

Westinghouse Response

The air sampling program at HDP is consistent with Table 1 of USNRC Regulatory Guide 8.25, *Air Sampling in the Workplace,* and provides activity concentration data that is representative of the workers' breathing zones. For situations where an individual is unlikely to exceed 10 percent of the annual limit on exposure, and the average concentration in the workplace is not likely to exceed 10 percent of the derived air concentration, intermittent or grab samples are appropriate, and representative sampling of the breathing zone is not required in all cases.

Prospective estimates of air concentration are routinely performed during work planning and radiation work permit preparation based on the specific work and radiological conditions. For example, conservatively assuming a re-suspension factor of 1E-4 during soil excavation, an exposure period of 2,000 hours without credit for engineering controls or personnel protective equipment, and an average total uranium concentration of 140 pCi/g (the site average from characterization data) results in an estimate of 0.014 DAC, or an individual exposure of 70 millirem per year. In order to exceed either of the thresholds where representative sampling is required, the average concentration in soil handled during the course of one year would need to be approximately 1,000 pCi/g, which is a highly unlikely variance from the expected mean concentration.

In cases where the prospective estimates, or the results of intermittent or grab sampling indicate that the annual exposure or concentration thresholds listed above are likely to be exceeded, representative sampling for the purpose of determining occupational exposure will be conducted. The representative sampling will consist of the use of lapel samplers or other portable low volume air samplers located within about 1 foot of the worker's head to obtain the sample since this method is accepted as being representative without further demonstration.

10.2.2.1 Personal Air Sampling will be revised as follows:

> "10.2.2.1 Personal Air Sampling
> When monitoring is required for the purpose of determining occupational exposure, representative sampling is accomplished through the use of a personal air sampler (lapel pump), or a portable low volume air sampler. The air filter is located within approximately 12 inches of the worker's head during sample collection, increasing the probability of being representative of the concentration in the worker's breathing zone."

2. (HDP-10-Q2, Section 10.2.4)

   Comment: Section 10.2.4 of the DP indicates that Health Physicist (HP) staff will determine the "...type, frequency, and appropriate location of [air] sampling to be performed for the activity to be monitored..." based upon the Radiation Protection Plan (RPP) and associated implementing procedures. Westinghouse's statement in Section 10.2.4 does not provide sufficient information on the sensitivities of air sample measurements and the frequency of sample collection. It only refers to separate documents.

   Basis: Per guidance in NUREG-1757, Vol. 1, Rev. 2, Appendix D, Section X.a., licensees should provide "a description of the criteria which demonstrates that air samplers with appropriate sensitivities will be used, and that samples will be collected at appropriate frequencies."

   Path Forward: Provide a description of the criteria which demonstrates that air samplers with appropriate sensitivities will be used and that samples will be collected at appropriate frequencies.

Westinghouse Response

Summary

The criteria for ensuring the appropriate sensitivities are met for air samplers and for collecting air samples at the appropriate frequencies are discussed in other sections of the DP, namely 10.8.4 and 10.2, respectively. HDP has established processes for estimating the potential air concentration before work begins and validating those estimates through sampling during work activities.

Discussion

The detection sensitivity for air samples is addressed by Section 10.8.4 of the DP. This Section indicates that air samples will be of sufficient volume to achieve the sensitivity to meet the thresholds for sampling. These thresholds are described in Section 10.2 and 10.4.3 of the DP. In addition to being of sufficient volume, the samples will be analyzed using radiological instrumentation with a sensitivity that is sufficient to detect airborne radioactivity at the established thresholds. The equation used to calculate MDC is as follows:

$$\text{Air Sample MDC(uCi/ml)} = \frac{MDCR}{(\varepsilon_i)\,(\varepsilon_f)\,(V_s)\,(2.22\ \text{E09})}$$

where:

	MDCR	=	Minimum Detectable Count Rate (cpm),
	ε_i	=	instrument efficiency (c/d),
	ε_f	=	filter collection efficiency (default value is 0.99),
	V_s	=	sample volume (l),
	2.22 E09	=	conversion from dpm to μCi, and L to ml.

The following example illustrates the evaluation of an air sample obtained over a 2 hour interval (120 minutes), and at a flow rate of 50 liters per minute (6,000 liters). The counting parameters for analysis include a nominal counting interval of 10 minutes, a background count rate of 0.6 cpm, and counting efficiency of 0.26 c/d. This example calculation results in an MDA of 3.5E-13 uCi/ml. Compared to the DAC value for Uranium from 10 CFR 20 Appendix B Table 1 value of 2E-11 uCi/ml, the MDC for this air sample equates to 0.018 DAC, indicating the sensitivity for the air sample met the threshold.

$$\text{AirSampleMDC (uCi/ml)} = \frac{1.2}{(0.26)\,(0.99)\,(6000\ \text{liters})\,(2.22\ \text{E09}\ )} = 3.5\text{E-}13\ \text{uCi/ml}$$

Depending upon the period of use, lapel air samplers may not obtain a sufficient volume to meet sensitivity requirements using the approach outlined above. For those instances, the sample may be evaluated, and the internal exposure assigned based on a comparison of the amount of radioactivity on air filter to the annual limit on intake. This is discussed in section 10.4.1 of the DP. A target for the sensitivity using this approach has been established at 2 mrem. The basis for the selection of this target is as follows:

- 10 CFR Part 20.1502 requires monitoring of occupational exposure from airborne radioactivity when levels are likely to exceed 10-percent of the applicable regulatory limit. The regulatory limit for CEDE dose is 5,000 mrem, therefore monitoring is required if internal exposure is likely to exceed 500 mrem CEDE.

- The DAC values presented in 10 CFR 20, Appendix B are based upon 2,000 working hours in a year. Assuming a standard 8-hour working day, this equates to 250 total working days in a year. If an individual radiation worker were to be assigned a dose at the MDC goal (2 mrem) for each working day, the cumulative dose would be equal to 500 mrem.

The frequency of air sample collection is addressed by the following Sections of the DP.

- Section 10.2 of the DP requires that work place sampling will be performed when airborne radioactivity concentrations are likely to exceed 2 percent of the occupational DAC values in general areas. Sampling that is representative of the concentrations within the breathing zone will be performed when airborne radioactivity concentrations are likely to exceed 10 percent of the occupational DAC values in the breathing zone, or when airborne radioactivity concentrations are likely to exceed 2 percent of the occupational DAC values in the breathing zone of a declared pregnant female. Sampling will also be performed when respirators are worn for the purpose of protecting individuals from exposure to airborne radioactivity.
- Section 10.4.3 of the DP indicates that air samples will be collected at a minimum of a weekly frequency. However, operational grab or continuous air samples will be collected daily during any work activities for which the projected air concentrations are estimated to exceed 2 percent of the occupational DAC values. The response to RAI HDP-10-Q1 describes the HDP routine process for calculating prospective air concentration estimates during work planning and radiation work permit preparation based on the specific work and radiological conditions.

3. (HDP-10-Q3, Section 10.5)

   Comment: Section 10.5 of the HDP makes statements that "... although monitoring for external exposure is not required, the HDP has conservatively elected to implement a program that includes provisions for monitoring occupational exposure to beta, gamma and neutron radiation for those personnel who routinely handle radioactive materials," and "...the HDP may discontinue the external dosimetry program provided actual conditions support that determination." NRC expects to be provided with a detailed dose analysis and justification before any external dosimetry program is discontinued.

   Basis: The "...conditions requiring individual monitoring of external and internal occupational dose..." are provided in 10 CFR 20.1502.

   Path Forward: Section 10.5 should be modified to include a commitment from Westinghouse that, prior to discontinuing an external dosimetry program, a technical justification demonstrating that no potential for exposures above thresholds given in 10 CFR 20.1502 will be provided. The justification should contain a detailed dose analysis of current and past site conditions and historical trends in worker doses.

Westinghouse Response

Westinghouse will maintain an external exposure monitoring plan consistent with the requirements of 10 CFR 20.1502(a). Westinghouse has performed an evaluation of the potential exposures and the associated requirements for monitoring. The outcome of this

evaluation indicates that monitoring for external exposure is not required in accordance with 10 CFR 20.1502. The third paragraph in DP Section 10.5 will be deleted.

The following excerpt from the aforementioned evaluation provides summary statements regarding the limited potential for exposure anticipated for the balance of HDP decommissioning activities:

> *From 2001 through 2005, the highest annual individual external dose during this period was 123 mrem (received in 2002). This was the only annual individual external dose exceeding 100 mrem in the five-year period. The amount of U-235 being handled and shipped off-site in the 2002 timeframe (06/23/2001 through 07/22/2003) was approximately 50 times what is expected to be encountered during the remaining decommissioning work. External dose to individuals during the remaining decommissioning work is not expected to exceed that from past activities for the following reasons:*
>
> > *The estimated mass of U-235 in the burial pits is anticipated to be less than 2 percent of the mass removed from fuel processing buildings between 2002 through 2005.*
> >
> > *The schedule for the excavation and removal activity is planned to span just over eight (8) calendar quarters and will likely occur in three different calendar years.*
>
> *Based on the prior work experience (2001 through 2005), individual annual external exposure is expected to be less than 100 mrem.*

Notwithstanding the conclusion that monitoring for external exposure is not required, Westinghouse has elected to implement an external dosimetry program at this time. This decision is in response to a request from our insurer. Westinghouse will continue to evaluate actual and potential external exposures during the course of the project, and may elect to discontinue monitoring in the future should the evaluation continue to indicate that monitoring for external exposure is not required, and our insurer concurs with our evaluation. This evaluation will be available for inspection.

4. (HDP-10-Q4, Section 10.5)

   Comment: Section 10.5 of the HDP states that "...dosimetry will be utilized that is capable of detecting beta, gamma, x-ray and neutron radiation, such as the Global Dosimetry Model 760 TLD. The Model 760 has an energy response of 0.766 MeV to 5 MeV for beta, 5 keV to 6 MeV for photons, and thermal to 6 MeV for neutrons." There are some concerns that the stated energy response will not provide an accurate dose measurement for beta emitters. For example, the maximum energy of Tc-99 is 0.295 MeV (Radiological Health Handbook, 1970). Additional information is needed on the accuracy of dosimetry measurements for all expected radiation energies.

   Basis: Per guidance in NUREG-1757, Vol. 1, Rev. 2, Appendix D, Section X.d., licensees should provide "...a description of the type, range, sensitivity, and accuracy of each individual-monitoring device..." and "...a description of the procedure to insure that surveys necessary to supplement personnel monitoring are performed."

Path Forward: Clarify how external dosimetry will provide an accurate measurement for all potential radiation energies at the site. Provide additional details on how Tc-99 will be measured. If dosimetry will not be capable of such measurements, provide a description of the procedure to insure that surveys necessary to supplement personnel monitoring are performed.

Westinghouse Response

The dosimetry system that Westinghouse is currently using for the purpose of confirmatory measurements has an adequate range to measure shallow external dose from uranium and uranium progeny, which are the predominant radionuclides that have the potential to contribute to external exposure at the HDP site. The concentration of Tc-99 at HDP is insufficient to result in shallow external exposure where monitoring for Tc-99 is required.

The determination for Tc-99 is based on the dose conversion factors contained in Federal Guidance Report 12 (1.06 E-4 mrem per pCi- y/cm^3); and the assumptions that an individual would need to be continuously present during the course of one year (2,000 hours) within an area having an average Tc-99 soil contamination level of 558,000 pCi/g in order to receive a shallow exposure of 100 mrem (based on a soil density of 1.69 g/cm^3). The highest Tc-99 soil contamination level measured at the HDP is 17,100 pCi/g, which is only 3 percent of this value. In other words, if an individual was continuously present at the location of the highest soil contamination measured on site, and assuming that the soil contamination was present in a sufficient quantity to approximate an infinite plane source, the maximum annual shallow exposure would be 3 mrem.

The evaluation of shallow external exposure due to skin contamination in HDP procedures includes dose conversion factors, including Tc-99, to be used when converting measured skin contamination to shallow dose.

5. (HDP-10-Q5, Section 10.5)

   Comment: Section 10.5 of the HDP (External Exposure Determination) does not provide a description of the use of extremity and whole body monitors when the external radiation field is non-uniform.

   Basis: Per guidance in NUREG-1757, Vol. 1, Rev. 2, Appendix 0 , Section X.d., licensees should provide "a description of the use of extremity and whole body monitors when the external radiation field is non-uniform."

   Path Forward: Provide a description of the use of extremity and whole body monitors when the external radiation field is non-uniform.

Westinghouse Response

The fifth paragraph of DP Section 10.5 will be revised as follows:

> Secondary dosimetry usage will be implemented in accordance with the RPP and associated implementing procedures. The requirements as presented in the implementing procedures which utilize the guidance of NRC Regulatory Guide 8.4, "Direct-Reading

and Indirect-Reading Pocket Dosimeters," (Reference 10-16), and NRC Regulatory Guide 8.28, "Audible-Alarm Dosimeters," (Reference 10-17).

- Monitored personnel shall wear dosimetry on the chest or other portion of the whole body as directed by the RWP or HP Staff so that the primary dosimetry device is located on the portion of the whole body likely to receive the highest exposure.
- If it is uncertain which part of the whole body may receive the highest exposure (due to worker movement or multiple radiation sources), monitored personnel may wear additional dosimetry devices on those portions of the whole body that could receive the highest exposure. The RSO will determine which dosimeter will be used to assign the worker's dose when multiple dosimeters are worn.
- For work situations in which extremity exposures are expected to be at least five times greater than the whole body exposures, or if extremity exposures are expected to exceed 1,250 millirem per calendar quarter, the RSO shall specify additional dosimetry devices for the extremities to measure and control extremity dose.

In addition, HDP procedures provide guidance on the performance of radiation surveys, and address the identification of non-uniform radiation fields with an emphasis on the potential need to relocate external dosimetry as follows:

> When performing a radiation survey in support of a specific job or activity that has the potential for changing the radiological conditions in an area, then attention should be given to the location of a workers' whole body in relation to the radiation source to evaluate the necessity for relocating whole body dosimetry or, for multiple dosimetry where non-uniform dose rates exist.

6. (HDP-10-Q6, Section 10.5)

   Comment: Section 10.5 of the HDP (External Exposure Determination) does not provide a description of action levels for worker's external exposure and actions to be taken if those levels are exceeded.

   Basis: Per guidance in NUREG-1757, Vol. 1, Rev. 2, Appendix 0 , Section X.d., licensees should provide "...a description of the action levels for worker's external exposure, and the technical bases and actions to be taken when they are exceeded."

   Path Forward: Provide a description of the action levels for worker's external exposure, and the technical bases and actions to be taken when they are exceeded.

Westinghouse Response

The fourth paragraph of Section 10.1 of the HDP (External Exposure Determination) will be revised to include the following:

> In accordance with the RPP, annual individual occupational doses for employees should not exceed the administrative occupational exposure limit of 2,000 mrem TEDE. The action level for investigation and possible work restrictions shall be 1,000 mrem for DDE. Approval by the Project Director and RSO is required for an employee to exceed the administrative limits.

Note that the HDP administrative controls for radiation exposure are defined in HDP-PO-HP-100, Radiation Protection Plan. Specifically, Section 9.2 states the following:

9.2.1 Annual individual occupational doses for employees should not exceed 2,000 mrem TEDE. The action level for investigation and possible work restrictions shall be 1,000 mrem for DDE.

9.2.2 Approval by the Project Director and RSO is required for an employee to exceed the administrative limits.

9.2.3 Persons under 18 years of age are not permitted access to Restricted Areas at HDP.

9.2.4 Planned Special Exposures as defined in 10 CFR 20.1206 are not authorized at HDP.

The administrative control that is defined in HDP-PO-HP-100 is conservatively chosen to allow ample margin to ensure that the exposure limits within 10 CFR 20.1502 are not exceeded.

7. (HDP-10-Q7, Section 10.8)

Comment: A general description of instruments to be used during decommissioning is provided in Section 10.8, Table 10-5, and Table 10-6 of the HDP. There also seems to be a heavy reliance on the RPP, as it is stated that "...the RPP provides guidance on the use, calibration and maintenance of radiological instrumentation and the guidance is implemented through approved site procedures." The RPP should be provided for NRC review as a supplement to the DP.

Path Forward: Provide the RPP or provide a description of the guidance contained in the RPP for NRC review.

Westinghouse Response

HDP-PO-HP-100, Radiation Protection Plan, is provided with this response.

The RPP, Section 12.0, Instrumentation Program provides a list of radiological instrumentation and detectors currently in use at HDP, types of radiation detected, method of detection, detection ranges, nominal efficiencies and target MDA values, if applicable. Section 12.1 of the RPP establishes the calibration frequency for radiological instrumentation and section 12.2 describes the functional QC checks required to be performed prior to instrument use to ensure proper function. Westinghouse procedures provide the detailed requirements for instrumentation setup, functional checks and proper documentation.

8. (HDP-10-Q8, Section 10.8.1)

Comment: A description of instrument calibration and quality assurance procedures was not provided for instrumentation that will be calibrated on site.

Basis: NRC guidance in NUREG-1757, Vol. 1, Rev. 2, Appendix D, Section X.g. states that licensees should provide "...a description of instrumentation calibration and quality assurance procedures."

Path Forward: Provide a description of instrumentation calibration and quality assurance procedures.

Westinghouse Response

Summary

The descriptions of instrumentation calibration and quality assurance are discussed in other sections of the DP, namely 14.4.4.2.2 and 14.4.4.2.3, respectively.

Discussion and DP Revision

Radiological instrumentation, detectors, and air sample flow meters are sent to a qualified vendor for calibration, with the exception of a few instruments. (The vendor calibration and quality assurance programs are reviewed and approved in accordance with Westinghouse vendor qualification programs; see the response to HDP-C13-Q1; Westinghouse letter to NRC dated January 12, 2011, HEM-11-2, Response to Request for Additional Information on Decommissioning Plan Chapter 13 (License No. SNM-00033, Docket No. 070 00036)). These exceptions include two stationary gas-proportional counting systems and two portal monitors which are calibrated on-site. These instruments are calibrated by HDP using specific procedures that address the actual calibration protocol, the use of NIST traceable sources, quality assurance measures, and proper documentation of the calibration. Westinghouse performs routine operability and quality checks of instrumentation, Procedure implementation effectiveness is routinely assessed as required by Westinghouse quality assurance programs.

DP Section 14.4.4.2.2 provides additional detail on the calibration and maintenance of radiological survey instrumentation. However, the text does not address that instrumentation, including some used for final status survey, is sent off-site for calibration by an approved vendor. DP Section 14.4.4.2.2 will be revised as follows:

> Instruments and detectors will be calibrated for the radiation types and energies of interest or to a conservative energy source. Calibration will be performed on-site using HDP procedures or off-site by an approved vendor. Instrument calibrations will be documented with calibration certificates and/or forms and maintained with the instrumentation and project records. Calibration labels will also be attached to all portable survey instruments. Prior to using any survey instrument, the current calibration will be verified and all operational checks will be performed. Radioactive sources used for calibration will be traceable to the National Institute of Standards and Technology (NIST) and have been obtained in standard geometries to match the type of samples being counted. When a characterized high-purity germanium (HPGe) detector is used, suitable NIST-traceable sources will be used for calibration, and the software set up appropriately for the desired geometry.

Existing DP Section 14.4.4.2.3 provides additional detail on instrument quality assurance procedures (response checks):

> Prior to use on-site, all project instrument calibrations will be verified and initial response data collected. These initial measurements will be used to establish

performance standards (response ranges) in which the instruments will be tested against on a daily basis when in use. An acceptable response for field instrumentation is an instrument reading within ±20 percent of the established check source value. Laboratory instrumentation standards will be within ±3-sigma as documented on a control chart.

The DQO process determines the frequency of response checks, typically before issue and after an instrument has been used (typically at the end of the work day but in some cases this may be performed during an established break in activity, e.g., lunch). This additional check will expedite the identification of a potential problem before continued use in the field. Instrumentation will be response checked in accordance with HDP Site procedures. If the instrument response does not fall within the established range, the instrument will be removed from use until the reason for the deviation can be resolved and acceptable response again demonstrated. If the instrument fails a post-survey source check, all data collected during that time period with the instrument will be carefully reviewed and possibly adjusted or discarded, depending on the cause of the failure. In the event that FSS data are discarded, replacement data will be collected at the original locations.

9. (HDP-10-Q9, Section 10.8.1)

Comment: A description of the methods used to estimate uncertainty bounds for each type of instrumental measurement was not provided.

Basis: NRC guidance in NUREG-1757, Vol. 1, Rev. 2, Appendix 0 , Section X.g. states that licensees should provide "...a description of the methods used to estimate uncertainty bounds for each type of instrumental measurement."

Path Forward: Provide a description of the methods used to estimate uncertainty bounds for each type of instrumental measurement.

Westinghouse Response

DP Section 10.8.1.1 will be revised to add:

Uncertainty bounds are established at ± 3 standard deviations of the mean source count for bench counters and ± 20% of the expected source response for friskers, portable scalars, and dose rate meters.

The error and uncertainty introduced by the instrumentation used to collect the data are assumed to be controlled by the performance of instrument calibration and response checks.

10. (HDP-10-Q10, Section 10.8.4)

Comment: Section 10.8.4 of the HDP provides minimum detectable concentrations (MDC) and MDC calculations only for air samples. Additional information is needed on MDC and minimum detectable activity (MDA) calculations for all radiological instrumentation that will be used during decommissioning.

Basis: NRC guidance in NUREG-1757, Vol. 1, Rev. 2, Appendix D, Section X.g. states that licensees should provide "...a description of the method used to estimate the MDC or MDA (at the 95 percent confidence level) for each type of radiation to be detected."

Path Forward: For all radiological instrumentation to be used during decommissioning provide a description of the method used to estimate the MDC or MDA at the 95 percent confidence level for each type of radiation to be detected.

Westinghouse Response

For operational radiological instrumentation used during decommissioning, the following methods for estimating the MDC at the 95 percent confidence interval will be added to Section 10.8.4:

For operational surveys, the MDC formulations equivalent to those in Section 14.4.4.2.5 are used with the exception that the efficiency term represents the instrument efficiency (ε_i) as opposed to a total weighted efficiency (ε_t).

The protocol for establishing the MDC for bench counter and scalar instrumentation for operational (non-FSS) surveys includes the following formulas for MDC calculation:

MDC for a Portable Counter (timed count)

$$\text{MDC (dpm/100cm2)} = \frac{3 + 3.29\sqrt{(R_b)(T_g)\left(1 + \frac{T_g}{T_b}\right)}}{\frac{DA}{100}(\varepsilon_i)(T_g)}$$

Where;

DA	=	detector area (cm^2)
ε_i	=	instrument efficiency (c/d)
R_b	=	background count rate (cpm)
T_b	=	background count time (minutes)
T_g	=	gross count time (minutes)

Bench Counter Smear MDC

$$\text{Smear MDC (dpm/100cm2)} = \frac{3 + 3.29\sqrt{(R_b)(T_g)\left(1 + \frac{T_g}{T_b}\right)}}{(\varepsilon_i)(T_g)}$$

Where;

ε_i	=	instrument efficiency (c/d)
R_b	=	background count rate (cpm)
T_b	=	background count time (minutes)
T_g	=	gross count time (minutes)

In addition, the equation in DP Section 10.8.4 for the Air Sample MDC will be revised as follows:

$$\text{Air Sample MDC (uCi/ml)} = \frac{3 + 3.29\sqrt{(R_b)(T_g)\left(1 + \frac{T_g}{T_b}\right)}}{(\varepsilon_i)(\varepsilon_c)(T_g)(V_s)(2.22E^9)}$$

Where;

V_s	=	sample volume (liters)
ε_i	=	instrument efficiency-intra-
ε_c	=	collection efficiency (default 0.99)
R_b	=	background count rate (cpm)
T_b	=	background count time (minutes)
T_g	=	gross count time (minutes)
$2.22E^9$	=	conversion factor (dpm to uCi and liters to ml)

For the performance of Final Status Survey, the calculation of MDC is addressed in detail in Chapter 14 of the DP.

Section 14.4.4.2.4 addresses the total weighted efficiency, and Section 14.4.4.2.5 addresses Static MDC for FSS of Building and Structural Surfaces. In this case, MDC is calculated as follows:

$$\text{MDC (dpm/100cm2)} = \frac{3 + 3.29\sqrt{(R_b)(T_s)\left(1 + \frac{T_s}{T_b}\right)}}{\frac{A}{100cm^2}(\varepsilon_t)(T_s)}$$

Where;

A	=	probe area (cm^2)
ε_t	=	total weighted efficiency (c/d; 4π)
R_b	=	background count rate (cpm)
T_b	=	background count time (minutes)
T_s	=	sample or measurement count time (minutes)

Section 14.4.4.2.6 addresses the calculation of MDC when performing HPGe Spectrometer Analysis for FSS. In this case, the MDC as calculated by the software is similar to the following equation:

$$\text{MDC (pCi/g)} = \frac{3 + 4.65\sqrt{B}}{(K)(W)(t)}$$

Where;

B	=	Number of background counts during the count interval t
K	=	Proportionality constant that relates the detector response to the radioactivity level in a sample for a given set of measurement conditions

W = Sample weight (dry grams)

t = Count time (minutes)

Section 14.4.4.2.7 addresses the calculation of Scan MDC when performing FSS building and structural surfaces as well as gamma scan of open land areas. For the scanning of building and structural surfaces during FSS, the scan MDC is calculated as follows:

$$\text{Scan MDC} = \frac{MDCR}{\left(\sqrt{p}\right)\left(\varepsilon_t\right)\left(\frac{A}{100cm^2}\right)}$$

Where; $MDCR$ = minimum detectable count rate (cpm)

ε_t = total efficiency (c/d)

p = surveyor efficiency (unitless – typically assumed to be 0.5)

A = detector area (cm^2)

For the calculation of scan MDC for the scanning of open land areas, section 14.4.4.2.9 addresses the process used in detail, including the use of Microshield modeling to establish minimum detectable exposure rates (MDER) for the primary radionuclides of interest. In addition, Table 14-14 of the DP lists the typical MDC at a 95% confidence level for the instruments that will be used at HDP.

11. (HDP-10-Q11, Section 10.9.1.3.1)

Comment: Section 10.9.1.3.1 of the HDP states that nuclear criticality safety assessments (NCSAs) are subject to independent documented review by a qualified "NCS specialist" reviewer, however this term is not defined in the HDP. Additionally, the qualification process of key NCS personnel (i.e. , NCS Engineers), is not specified or described.

Basis: NUREG 1757, Vol 1, states that a description of the technical qualifications of safety personnel should be provided.

Path Forward: Please define "NCS specialist." Also, describe how key NCS personnel are qualified outside of meeting the minimum requirements as stated in Section 10.9.1.1.2.

Westinghouse Response

The term "NCS Engineer" will be replaced throughout the DP with "NCS Specialist." DP Section 10.9.1.1.2 specifies the minimum qualifications for this position. The minimum qualifications are stated to be "a Bachelor's degree in science or engineering, or equivalent, with at least three years of nuclear industry experience in criticality safety." Since these qualifications are stated to be the minimum qualifications for an NCS Specialist, there is no provision for key NCS personnel to be qualified outside of meeting the minimum requirements as stated in Section 10.9.1.1.2.

12. (HDP-10-Q12, Section 10.9.1.3.2)

Comment: Section 10.9.1.3.2 of the HDP provides the hierarchy of the NCS control philosophy and indications "alternative processes" to be the preferred control. It is unclear what is meant by "alternative processes" in the context of NCS controls.

Basis: NUREG 1757, Vol 1, states that a description of NCS requirements should be provided.

Path Forward: Please clarify what is meant by "alternative processes."

Westinghouse Response

Summary

The term "alternative processes" is based on the first bullet point of DP Section 10.9.1.3.2, which represents the first preferred approach to mitigating criticality hazards by selecting an alternate process to remove or design out the criticality hazard.

Discussion

Eliminating a criticality hazard by altering the process is always preferred over establishing criticality safety controls. This is because designing out the criticality hazard reduces the probability of the criticality event to zero, whereas mitigating the criticality hazard with the provision of criticality safety controls will reduce the criticality hazard to an acceptably low, but non-zero frequency. As an example, a decommissioning process/operation that involves the excavation of bulk quantities of material potentially contaminated with U-235 may result in the identification of numerous criticality hazards. Following the hierarchy of NCS control, the first preference would be to adjust/refine the decommissioning process/operation so that the quantity and risk of the criticality hazards presented by the process/operation is reduced to a minimum practicable level. In this example, this could be achieved by modifying the process/operation such that the materials are characterized by in-situ radiation measurements prior to excavation, with any identified quantities of U-235 of concern being separately extracted and segregated.

13. (HDP-10-Q13, Section 10.9.1.3.3)

Comment: Section 10.9.1.3.3 of the HDP provides the NCS control parameters analyzed for the HDP. In the description of geometry control, it is not clear if the facility configuration management program is used to maintain the dimensions of geometry controlled features. In the description of mass control, is it not clear if the instrumentation for mass measurements is subject to facility management measures.

Basis: NUREG 1757, Vol 1, states that a description of NCS requirements should be provided.

Path Forward: Please clarify whether the configuration management program is used to maintain the dimensions of geometry controlled features. Please clarify whether the instrumentation for mass measurements is subject to management measures.

Westinghouse Response

While HDP uses geometry controls and has controls for mass measurements, these controls are not subject to a 10 CFR 70.72 configuration management program or 10 CFR 70.62 management measures. As stated in 10 CFR 70.60, the requirements of 10 CFR 70.61 - §70.76 do not apply to decommissioning activities. HDP declared its decommissioning status on September 11, 2001, and NRC license amendment number 42, dated April 11, 2002, recognized HDP's decommissioning status pursuant to 10 CFR 70.38.

When NCS controls require procurement of items to maintain geometry control, those dimensional requirements are quality assurance requirements, verified by the quality assurance receipt inspection program. In addition, instrumentation for mass measurements when performed for NCS control purposes are procured under, and calibration is assured by, the quality assurance program. Also, NCS controls established by nuclear criticality safety assessments (NCSA), such as key dimensions for geometry control, are confirmed when the NCSA requirements are implemented.

14. (HDP-10-Q14)

Comment: Table 10-8 through Table 10-13 of the HDP provide a summary of credible NCS hazards established in the HDP NCSAs. In Section 10.9.1.3.2, the NCS control hierarchy is described and indicates that enhanced administrative and simple administrative controls are the least preferred, respectively. However, overall, there appears to be an abundance of administrative controls and very few passive or active engineered controls.

Basis: NUREG 1757, Vol 1, states that a summary of the review of NCSAs should be provided.

Path Forward: Please explain, where applicable, the perceived reliance on administrative controls.

Westinghouse Response

Hematite is no longer an active facility with static processing operations. The nature and variability of decommissioning activities (e.g., excavation slope, uneven terrain, potential presence of physically large burial items, etc.) necessitates reliance on administrative actions to fulfill the criticality safety requirements established, Moreover, the requirement to identify, segregate, package, and evaluate items/materials of concern (e.g., due to fissile nuclide mass content) necessitates reliance on human agency for most criticality safety functions.

However, where viable, administrative controls have been augmented/enhanced by passive design features (e.g., container geometry, spacing devices, etc.) and physical devices (e.g., radiation detectors). Since the approach used is consistent with the NCS control hierarchy in Section 10.9.1.3.2 (because preference has been placed on designing out criticality hazards or employing passive engineered, active engineered, and enhanced administrative controls over simple administrative controls) no change is considered necessary to DP Section 10.9.

15. (HDP-10-Q15)

Comment: Table 10-8 through Table 10-13 of the HDP provide a summary of credible NCS hazards established in the HDP NCSAs. Throughout the tables, the terms "at least unlikely" and "unlikely" are used.

Basis: NUREG 1757, Vol. 1, states that a summary of the review of NCSAs should be provided.

Path Forward: Please clarify what is meant by "at least unlikely" and how this differs from "unlikely."

Westinghouse Response

The term "unlikely" used throughout DP Section 10.9 refers to the frequency of occurrence prescribed to contingencies (or adverse changes to process conditions) in the Double Contingency Principle (DCP). The term "at least unlikely" (also used throughout DP Section 10.9) is attributed to events or conditions that have a frequency of occurrence that is smaller (i.e., less frequent) than "unlikely". Thus, a criticality hazard that is mitigated by controls that are "at least unlikely" to fail would present less risk than a criticality hazard that is mitigated by controls that are just "unlikely" to fail. This is because a control that is "at least unlikely" to fail is more robust that a control that is "unlikely" to fail. Therefore, use of the term "at least unlikely" is intended to portray that the described event is in fact less probable than "unlikely". Based on this clarification, no change is considered necessary to DP Section 10.9.

Hematite Decommissioning Plan Chapter 12 - Radioactive Waste Management Program

1. (HDP-12-Q1)

   Comment: In Section 12.2.2.5 of the DP it is stated that Chapters 8 and 10 of the DP contain a discussion of the actions to be taken in the event of elevated radioactivity measurements. Review of these two Chapters provided no descriptions of such actions for this situation.

   Path Forward: Provide in Chapters 8 and 10, the description of the actions that will occur in the event of elevated radioactivity measurements while performing a radiological survey prior to excavation in the burial pit area.

Westinghouse Response

Additional discussion regarding actions to be taken in the event elevated radioactivity measurements that exceed the NCS Exempt Material Limit are identified prior to excavation will be added to DP Chapters 8 and 12 as follows:

DP Section 8.5.1 provides a general sequence of soil and object evaluation and removal. The first bullet in Section 8.5.1 will be revised as follows:

> Soil will be evaluated using in-situ GWS, VOC monitoring (Photo-Ionization detector) and visual inspection of the exposed surface, repeated for each newly exposed surface. If elevated radioactivity measurements in excess of the NCS Exempt Material Limit are encountered prior to or during excavation, the detector response will be evaluated and the appropriate excavation depth determined.

Section 12.2.2.5, "Nuclear Criticality Safety" will be revised as follows:

> Prior to excavation of an area, a radiological and visual survey will be performed. Actions specified in section 8.5.1 will be implemented upon identification of an object/intact container or an elevated radioactivity measurement in excess of the NCS Exempt Material Limit.

The NCS Exempt Material Limit screening method and criteria can be found in DP 8.5.2.1 and DP 10.9.2.1.1.

2. (HDP-12-Q2)

   Comment: Westinghouse indicated in Section 12.1 of the DP that radioactive waste management activities will be conducted in accordance with the Hematite Waste Management and Transportation Plan (WMTP). Additional details are needed with respect to the manner in which site radioactive waste management activities will be conducted. Specifically, what details does the Plan have that are not presented in Chapter 12 of the DP.

   Basis: Per guidance in NUREG-1757, Vol. 1, Rev. 2, Appendix 0 , Section XII , licensees should provide a description of the waste management program for solid, liquid and mixed wastes.

Path Forward: Provide the WMTP or enhance the discussion in Section 12.1 with a description of the manner in which radioactive waste management activities will be conducted in accordance with the WMTP.

Westinghouse Response

HDP-PO-WM-900, Waste Management and Transportation Plan (WMTP) is attached.

The WMTP identifies regulatory responsibilities and other requirements for the characterization, packaging, transportation, security and disposal of the various types of radioactive, non-radioactive waste materials expected to be encountered. The types of waste anticipated for the HDP include: Asbestos Waste (radioactive and non-radioactive), Commercial Solid Waste (non-radioactive), Construction and Demolition Waste (radioactive and non radioactive), Electronic Waste (non-radioactive), Hazardous Waste (various types, non-radioactive), Infectious Waste, (radioactive and non-radioactive), Low Level Radioactive Waste (solid or liquid), Mixed Waste, (Radioactive and Hazardous), PCB Waste or PCB Bulk Product Waste, Universal Waste, and Used Oil.

3. (HDP-12-Q3)

Comment: Under Section 12.4.3, Waste Treatment, of the DP, it is stated that some materials may be treated onsite, e.g., toxic-dense non-aqueous phase liquid (12.4.3.3), toxic volatile organic compounds (12.4.3.4) and reactive uranium metal fines (12.4.3.5). However, the methods for treatment of such wastes are not described. The manner for treatment of wastes may result in additional types of wastes and effluents being generated. Has Westinghouse accounted for such and where is the information provided in the Decommissioning Plan?

Basis: Per guidance in NUREG-1757, Vol. 1, Rev. 2, Appendix 0 , Section XII , licensees should provide a description of how wastes are to be treated and the resultant volume, types, activity levels of waste and their classification.

Path Forward: Provide a description of the on-site treatment of the wastes described in Section 12.4.3.3 - 12.4.3.5 and the amounts, types, activity level and waste classification resulting from such treatment. Also provide the effluents which may/will be generate as a result of such treatment.

Westinghouse Response

Sections 12.4.1 and 12.4.3 are revised to describe how wastes are to be treated, how this treatment affects the volume of the final waste form, and any effluents generated by the treatment. The types, activity level, and waste classification for treated wastes are included in the DP Chapter 12 tables. As stated in DP section 12.4.1, mixed wastes have not been identified as being present at the HDP site. However, for the purpose of effective scheduling and planning and as a contingency, 5700 cubic feet of solid mixed waste (primarily VOC contaminated) and 1300 liters of liquid mixed waste will be assumed to be generated during the decommissioning.

Westinghouse has selected treatment methods that are unlikely to generate new waste types, and are expected to minimize the volume of any additional waste. The onsite treatment methods do

not result in concentrating radioactivity, and therefore activity levels that are in any secondary waste are expected to be consistent with the bulk of the decommissioning waste.

DP Changes

Note: The last paragraph of DP Section 12.4.1 will be removed from that section and combined into DP Section 12.4.3.3. DP Sections 12.4.3.1 -12.4.3.5 are revised as follows:

12.4.3.1 Corrosives-Acids And Bases

Due to the corrosive nature of this type of waste, intact metal containers are not expected to exist in a burial environment that would contain any significant quantity of an acid or base. Due to the burial environment and the variable ground water level at the Hematite Site, corrosives in soil should be attenuated and no longer present.. Upon identification, intact plastic or glass containers containing fluids will be evaluated for corrosive material. If acidic or basic liquids are identified, the waste will be neutralized with the appropriate reagent to achieve an acceptable pH in accordance with waste management procedures. The treatment may yield a final waste form that is up to twice the amount of the original waste form. No additional effluents are generated by neutralization.

12.4.3.2 Toxic-Heavy Metal Contamination

Metal contaminated soil could exist in the Burial Pit area, and any required remediation areas are identified in the ROD. Though not specifically identified during the HSA process and also not detected during soil sampling, it is recognized that lead was a commonly used metal for shielding, plumbing, paint and roofing. Mercury was commonly used in various thermometers, gauges and switches for process equipment. Arsenic levels in soil above site background were identified in approximately 10 percent of soil samples from areas co-located with radiological contaminants slated for excavation.

Soils contaminated with metals, at concentrations failing the TCLP test, will be mixed with an appropriate stabilization agent based upon the material of concern. This will bind the hazardous constituents in order to meet TCLP testing requirements and LDRs. The treatment may yield a final waste form that is up to five times the volume of the original waste form. No additional effluents will be generated by stabilization process. Lead forms will be recycled. If the lead cannot be recycled it will be shipped off-site for macro-encapsulation prior to disposal.

12.4.3.3 Toxic-Dense Non-Aqueous Phase Liquid (DNAPL)

DNAPL is a general term for a class of chemicals that do not readily mix with water. Specific examples of DNAPL include chlorinated solvents, coal tar, creosote, polychlorinated biphenyls, mercury and extra heavy crude oil. If present, the material would originate from intact plastic or glass containers containing fluids or from pockets of liquid under ruptured containers. The liquids will be collected in a compatible container and will be assayed to determine the exact nature of the liquid. Any form of DNAPL that can be incinerated or processed through solvent recovery will be packaged for off-site treatment and disposal at an approved off-site facility.

If the form of the DNAPL is mercury, then the mercury will be treated on-site (e.g. by an amalgamation processes and stabilized to pass TCLP testing) or sent offsite for disposal. The treatment will yield a final waste form that is up to five times the volume of the original waste form. No additional effluents are generated by stabilization.

If other forms of DNAPL waste are encountered that are not amenable to incineration, then absorption or solidification on-site may be required. The treatment may yield a final waste form that is five times the amount of the original waste form. No additional effluents are generated by absorption or solidification.

12.4.3.4 Toxic-Volatile Organic Compounds (VOCs)

Low levels of VOC contamination, below Toxicity Characteristic Leaching Procedure (TCLP) values but above RG values, have been detected in samples from the Burial Pit area and underneath process building slabs. Materials with VOCs that are below the TCLP criteria will be segregated from materials with VOCs that exceed the TCLP criteria. Materials that have VOC contamination in excess of TCLP values will be treated as mixed waste.

VOC treatment will be conducted in treatment tanks by *ex-situ* soil vapor extraction (SVE). SVE uses a mechanical blower to induce a vacuum, which causes the VOCs to be stripped and volatilized into the air stream. The exhaust air is then treated to remove particulates and VOCs before it is emitted to the atmosphere. The system effluent is unlikely to have concentrations of radioactivity or chemicals that are detectable, however effluents will be monitored per the requirements of NESHAPs and 10 CFR 20. Radiological and chemical deposits on carbon bed filters may result in a limited volume of mixed waste (approximately 350 cubic feet).

Tanks utilized for VOC treatment will be specially constructed cells that meet the tank and tank system definition in Title 40 Code of Federal Regulation Part 265, Interim Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities (Reference 12-8). Waste treatment tanks will be lined and covered with a durable, flexible membrane liner, and designed, constructed and operated in accordance with United States Environmental Protection Agency (EPA) Management of Remediation Waste Under RCRA EPA530-F-98-026, October 1998 (Reference 12-9), interim status standards for tank systems. Solid structures will serve as the base/secondary liner (e.g., process building slabs, asphalt pavement) and structural sidewalls (e.g. Jersey barriers). The waste will be accumulated in the treatment tanks for sampling, and until sufficient material is available for effective treatment. Soil that initially was below the TCLP criteria and below reuse DCGLs is expected to be reused as fill material following on-site treatment; non-soil materials that initially were below the TCLP criteria are expected to be shipped for off-site disposal. Materials that exceed the TCLP criteria will be shipped for disposal at an off-site permitted facility following on-site treatment. Treated materials will be tested to ensure the disposal facility WAC is met. SVE treatment does not change the volume of the waste form.

12.4.3.5 Reactive or Ignitable - Uranium Metal Fines

Due to the oxidizing nature of the burial environment, reactive pyrophoric-metals (e.g., uranium metal shavings, fines or chips) are not expected to be present; however, if they do exist, they would be expected to be found inside intact containers. The treatment process for reactive pyrophoric-metals involves solidification of the uranium metal fines in a silica concrete mixture. The solidification process will occur in a metal box with the approximate dimensions of 4 ft X 6ft X 2ft. The solidification mixtures will be controlled to limit the package uranium content to less than 180 grams of metallic U-235 per 360 kilograms of concrete if the uranium is special nuclear material, or 15,000 pCi/g U-238 if the metallic uranium is depleted uranium. The treatment may yield a final waste form that is 20 times the amount of the original waste form, but no more than 90 cubic feet is projected to be generated. No additional effluents are generated by solidification. The treatment process will comply with the disposal site's WAC and will allow shipment for disposal.

4. (HDP-12-Q4)

Comment: Under Section 12.4.5, Permitting, of the DP, it is indicated that Westinghouse will be applying for a conditional exemption for low-level mixed waste storage and treatment as allowed under 40 Part 266 Subpart N, Conditional Exemption for Low-Level Mixed Waste Storage, Treatment, Transportation and Disposal. Has Westinghouse applied for this exemption and has it been granted?

Path Forward: Provide the status of the application for the conditional exemption for low-level mixed waste storage and treatment as allowed under 40 Part 266 Subpart N and whether it was or was not granted when the regulatory decision is reached.

Westinghouse Response

The terminology "apply for a conditional exemption for low-level mixed waste storage and treatment" used in the last paragraph of Section 12.4.5 is inaccurate. The correct description of this process is to "use a conditional exemption for low-level mixed waste storage and treatment".

The State of Missouri has adopted the Low Level Mixed Waste (LLMW) exemption provided in 40 CFR 266. In accordance with 40 CFR 266.230 (10 CSR 25-7.266), Westinghouse will notify MDNR in writing of the use of the exemption within 90 days after the storage unit has been placed into service for the storage of conditionally exempt LLMW; pre-notification of EPA is not required.

The 3rd paragraph of 12.4.5, Permitting, will be changed to the following:

Generators are not required to obtain a permit prior to storing or treating mixed waste (other than thermal treatment) under Title 40 Code of Federal Regulations 266, Subpart N, Conditional Exemption for Low Level Mixed Waste Storage, Treatment, Transportation and Disposal (Reference 12-20). Westinghouse intends to use this conditional exemption and notify MDNR of its use within 90 days after the storage unit for low-level mixed waste storage and treatment has been placed into service, as allowed in Reference 12-20.