Westinghouse Non-Proprietary Class 3

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Subject: Response to Request for Additional Information Concerning Hematite Decommissioning Plan: Chapter 5, Dose Modeling (License No. SNM-00033, Docket No. 070-00036)

References:
1) NRC (J. J. Hayes) letter to Westinghouse (E. K. Hackmann), dated July 12, 2010 , "Westinghouse Hematite Decommission Plan Review Requests for Additional Information"
2) Westinghouse (E. K. Hackmann) letter to Document Control Desk (NRC), HEM-09-94, dated August 12, 2009, "Decommissioning Plan and Revision to License Application"

This letter provides the Westinghouse Electric Company LLC response to selected questions from NRC's Reference 1 request for additional information concerning Chapter 5, Dose Modeling, of the Decommissioning Plan of Reference 2. Westinghouse Electric Company LLC is continuing further technical evaluation of RAIs designated HDP-C5-Q2, Q3, Q7, Q9, Q10, Q11 and Q15. The response to these RAIs will be submitted upon completion of the technical evaluatons.

Attachment 1 provides responses to the request for additional information for the selected RAIs, and provides an explanation of associated changes to the Decommissioning Plan resulting from those responses. The actual changes to the Decommissioning Plan will be provided under separate cover.

NMSS01
FSME

Please contact Gerard Couture, Licensing Manager of my staff at 803-647-2045 should you have questions or need any additional information.

Sincerely,

E. Kurt Hackmann
Director, Hematite Decommissioning Project

Attachments:
1) Response to Request for Additional Information Hematite Decommissioning Plan: Chapter 5, Dose Modeling
2) Replacement Figure 5-3, response to RAI HDP-C5-Q1
3) Comparison of Radionuclide Mixture Using Mean and 95% UCL for Non-Uranium Isotopes, response to RAI HDP-C5-Q6
4) Summary of Modeled Geometries for External Dose from Buried and Excavated Piping, response to RAI HDP-C5-Q19

cc: J. J. Hayes, NRC/FSME/DWMEP/DURLD
J. W. Smetanka, Westinghouse, w/o attachments
J. E. Tapp, NRC Region III/DNMS/DB, w/o attachments

ATTACHMENT 1

Response to Request for Additional Information
Hematite Decommissioning Plan:
Chapter 5, Dose Modeling

Westinghouse Electric Company LLC, Hematite Decommissioning Project

Docket No. 070-00036

Decommissioning Plan Chapter 5, Dose Modeling

The following reiterates the NRC requests for additional information (RAI) of letter dated July 12, 2010, followed by the Westinghouse response for each RAI. Some of the responses will result in changes, as noted, to the Decommissioning Plan ((DP; Hematite Decommissioning Plan, DO-08-004, Revision 0.0). The changes to the DP will be provided under separate cover, denoted by vertical lines in the right margin of the document.

These RAI responses are organized in the same manner as the RAIs concerning DP Chapter 5. For each RAI, the NRC's Comment, Basis and Path Forward is reiterated, and followed by the Westinghouse Response.

1. (HDP-C5-Q1) Comment: The Hematite Decommissioning Plan (HDP), Section 5.2, states that Ra-226 and Th-232 are only found in certain locations of the site, but the location and extent of contamination is not thoroughly illustrated. Also, the Th-232 model assumes the entire surface area of the site is contaminated even though it is stated to be found in limited locations. Also, it is unclear if these radionuclides will be measured for across the entire site area during the Final Status Survey (FSS) to ensure they meet the derived concentration guideline levels (DCGLs).

   Basis: The area of the contaminated zone is a necessary RESRAD input parameter for determining the DCGLs. Section 5.2 of the DP discusses how Th-232 and Ra-226 are modeled separately from the other Radionuclides of Concern (ROCs). In HDP Figure 5-3, Ra-226 Impacted Areas, the legend uses a triangle to designate "Elevated Ra-226 Locations." However, the diagram does not have any triangles identifying the Ra-226 locations within the "Ra-226 Impacted Area." The DP states that Th-232 is only found in certain locations within the area of buried waste and that Ra-226 is only present at two locations within the buried waste. The Hematite Radiological Characterization Report (HRCR) describes Th-232 in isolated areas in the Burial Pits Soil, and Soil Southeast of the Process Buildings and Surrounding Areas. However, in Table 5-6 of the DP describing the RESRAD Input Parameters, the size of the Th-232 model contaminated zone is the entire site area of 153,375 m^2 while the Ra-226 model assumes a limited area of 1,292 m^2. It is unclear why the area is not reduced in the Th-232 model if the contaminant is only found in certain areas. If a larger area was intended to be used, then it is unclear why Th-232 was modeled separately from the other ROCs.

   Path Forward: Please provide an updated Figure 5-3 identifying the elevated Ra-226 locations. Also, provide further basis for the contaminated zone areas used in the Th-232 model. Also, clarify if Ra-226 and Th-232 will be measured for across the whole site area during the FSS and explain how the absence of Th-232 and Ra-226 in other areas of the site will be verified if those radionuclides will not be measured for in those areas in the FSS.

Westinghouse Response:

Figure 5-3 has been updated to identify the elevated Ra-226 locations and is included as Attachment 2. The confidence interval of 1.6 pCi/g provided in Hematite Decommissioning Plan (DP) Section 4.3.5 was used to define elevated Ra-226 concentrations.

The development of the Th-232+C $DCGL_W$ presented in DP Chapter 5 is conservative with regard to the contaminated zone area parameter. The justification for using the full impacted area (153,375 m^2) as the area of the contaminated zone is two-fold. First, Th-232 was determined to be distinguishable from background in isolated areas of the site (RCR Appendix A) as well as at isolated locations where sample results exceeded the calculated BTV of 1.7 pCi/g. As such, a smaller impacted area to be used as the area of the contaminated zone could not be justified. Second, DP Table 5-13 (Area Factors for Soil) shows that as the area of the contaminated zone increases, the area factor also increases. Because the calculation of the area factor is the DCGL value that corresponds to the elevated area divided by the $DCGL_W$ value, the DCGL value corresponding to the elevated area is therefore also increasing. As such, it is most conservative to assume a larger area of the contaminated zone in the development of the Th-232+C $DCGL_W$.

Modeling the Th-232+C $DCGL_W$ separately was done as a matter of convenience in the RESRAD software. The development of the Th-232+C $DCGL_W$ began with entering the following radionuclides all with an arbitrary initial concentration of 1 pCi/g: Th-232 and its long-live progeny Ra-228 and Th-228. Unlike, for example U-235+D, the $DCGL_W$ for Th-232+C cannot be obtained from the "Single Radionuclide Soil Guidelines" section of the RESRAD Summary Report. Instead, the $DCGL_W$ was calculated as follows. Since 1 pCi/g was the entered for the Th-232 initial soil concentration, the value RESRAD calculated as the "Maximum TDOSE(t)" provided in the "Contaminated Zone and Total Dose Summary" of the RESRAD Summary Report was considered to have units of [mrem/yr]/[pCi/g]. The $DCGL_W$ was calculated by dividing the TEDE (25 mrem/yr) by the "Maximum TDOSE(t)", which yields a $DCGL_W$ value with units of pCi/g that accounts for the dose from Th-232 and its long-lived progeny. Note that this discussion also applies to the separately modeled Ra-226+C DCGL.

Th-232 and Ra-226 will be included in the analysis of FSS samples site wide. Gamma spectroscopy will be performed on all samples and Ac-228 and Pb-214/Bi-214 will be used to quantify Th-232 and Ra-226 activity, respectively.

Thorium-232 will only be included for demonstrating compliance in areas distinguishable from background or when an individual result exceeds the BTV. DP Section 14.4.4.2.6 states: "Thorium-232 radioactivity concentration will be reported for use in areas distinguishable from background or for sample results greater than the BTV of 1.7 pCi/g (see Section 14.2)."

Because Ra-226 was identified only in a limited portion of the site, Ra-226 radioactivity concentration will only be included for demonstrating compliance in the Ra-226 Impacted Area (RIA, HDP Figure 5-3). DP Section 14.4.4.2.6 states: "Finally, gamma spectroscopy results for each sample will be reviewed for other gamma-emitting radionuclides present." This process allows Ra-226 contamination outside of the Ra-226 Impacted Area to be identified, however unlikely based on the Historical Site Assessment (HSA) and RCR findings. Should Ra-226 contamination exceeding the confidence interval of 1.6 pCi/g (provided in DP Section 4.3.5) be identified outside the RIA, the magnitude and spatial distribution of the contamination will be investigated to ensure compliance to the release criterion.

2. (HDP-C5-Q2) Westinghouse continues further technical evaluation of this RAI and will submit the response upon completion of the technical evaluation.

3. (HDP-C5-Q3) Westinghouse continues further technical evaluation of this RAI and will submit the response upon completion of the technical evaluation.

4. (HDP-C5-Q4) Comment: Section 5.3.4.2 discusses the value assumed for the evapotranspiration coefficient. This value is inconsistent with the assumptions regarding related parameters such as irrigation rate, precipitation rate, evapotranspiration rate, and runoff coefficient. It is not evident that the value chosen for the evapotranspiration coefficient is adequately conservative. Furthermore, the value chosen for the runoff coefficient does not match the site-specific data.

   Basis: The evapotranspiration coefficient is correlated with irrigation rate, which is correlated with precipitation rate, and well pumping rate. Precipitation rate is correlated with runoff coefficient. Table 5-6 of the DP states that the calculated value for evapotranspiration coefficient (based on the assumed values for evapotranspiration rate, precipitation rate, irrigation rate, and runoff coefficient) is greater than 1.0. (The value calculated is 1.02). Since the evapotranspiration coefficient cannot possibly be larger than 1.0, the DP states that the maximum of the PDF provided in NUREG-6697 was used instead. However, the maximum value in NUREG-6697 is 0.75, and the value assumed in the DP is 0.8.

   The timing of the Np-237 peak dose is sensitive to the assumed value for the evapotranspiration coefficient and related parameters, so the value for evapotranspiration coefficient should be sufficiently justified. Furthermore, the values chosen for the evapotranspiration coefficient could impact the sensitivity of results to the values chosen for the Kds of certain radionuclides such as Np-237.

   The runoff coefficient is estimated based on a method described in NUREG-6697, Attachment C, Table 4.2-1 assuming flat cultivated land with intermediate combination of clay and loam. This method results in a runoff coefficient of 0.8. However, in a document supporting the 2004 DP titled "Derivation of Site-Specific DCGLs for Westinghouse Electric Co. Hematite Facility", on pg A-3, of the value chosen for runoff coefficient is 0.305 based on 12" annual average runoff, and 38" average annual precipitation. In Section 3.3 of the 2009 DP, p. 3-5, it is restated that the area receives 12" of average annual runoff. It is not clear why the runoff coefficient assumptions changed from the 2004 DP if the site-specific data has not changed. It is also not clear why applying a generic empirical method in NUREG-6697 is the chosen approach over using the site-specific data.

Path Forward: Justify the values assumed for the evapotranspiration coefficient and runoff coefficient. Given the uncertainty in these parameters, explain how the values chosen are adequately conservative. Rectify the inconsistencies in the assumptions regarding correlated parameters such as irrigation rate, precipitation rate, evapotranspiration rate, and the runoff coefficient. Reevaluate the sensitivity of the Kds of relevant nuclides if a new value is assumed for the evapotranspiration coefficient.

Westinghouse Response:

Westinghouse Electric Corporation LLC (Westinghouse) agrees that the evapotranspiration coefficient is correlated with the runoff coefficient, irrigation rate, and precipitation rate. This correlation is the reason that the evapotranspiration coefficient was calculated using Equation 4.3-1 from NUREG/CR-6697, Att. C. as stated in Table 5-6.

Equation 4.3-1 requires data for the irrigation rate, precipitation rate, evapotranspiration rate, and runoff coefficient. The site-specific irrigation rate was selected from References 5-14, "Missouri 2004 irrigation Survey" and 5-15, "USGS Estimated Use of Water in the United States in 2000". The site-specific precipitation rate was selected from "Festus Weather Station Web Page". The evapotranspiration rate was selected from Reference 5-15 "USGS Estimated Use of Water in the United States in 2000". The runoff coefficient was calculated using site topography and soil type and the data provided in NUREG/CR-6697, Att. C, Table 4.2-1.

There is an inconsistency within the DP regarding precipitation rate. DP Section 3.3 states that the precipitation rate is 38 inches per year according to the "Missouri Water Atlas 1986". The precipitation rate of 39.9 inches provided in Table 5-6 was found on the "Festus Weather Station Web Page". These values are very close and actually serve to confirm the accuracy of both values since the difference is insignificant.

The comment states that a site-specific value for runoff coefficient may be more appropriate than using the site topography and site soil type to derive the runoff coefficient using NUREG/CR-6697 methods. Westinghouse has reviewed these two approaches and found very little difference. The derived runoff coefficient provided in Table 5-6 was 0.4 (unitless) and the calculated runoff coefficient assuming 12 inches average annual runoff and a precipitation rate of 38 inches was 0.32 (unitless). The site-specific evapotranspiration coefficient was calculated using the precipitation rate (38 inches) and runoff coefficient (0.32 = 12/38) from Section 3.3. The resulting evapotranspiration coefficient was 0.96 (unitless) which is very similar to the value of 1.02 (unitless) calculated in Table 5-6.

In conclusion, the site-specific calculation of evapotranspiration coefficient is the most applicable approach to use for parameter selection. It would be inappropriate to use a value based on a percentile of the NUREG/CR-6697 PDF because there are known discrepancies between the variety of conditions assumed in the development of the PDF and the known conditions at the Hematite Site. In addition, the change in evapotranspiration using the precipitation value and calculated runoff coefficient based on the data provided in Section 3.3 is insignificant and does not change the conclusion that the highest value of the evapotranspiration coefficient listed in NUREG/CR-6697 would be used.

The selected value of 0.8 is conservative since the calculation indicates the parameter value could actually be higher. The dose increases as the evapotranspiration coefficient decreases due to increased groundwater dose.

Finally, Westinghouse agrees that an incorrect value of 0.8, as opposed to 0.75 was selected as the maximum from the NUREG/CR-6697 PDF. The 0.75 value will replace the 0.8 value in the revised DCGL calculation.

5. (HDP-C5-Q5) Comment: Westinghouse Electric Company (Westinghouse) departs from their parameter selection approach for the plant transfer factor for Pa-231 and the milk transfer factor for Ra outlined in Section 5.3.4. The values assumed are neither the default RESRAD parameter values, nor are they the median value of the probability distribution function (PDFs) in NUREG-6697.

Basis: Pa-231 is a progeny of U-235, so the dose from this radionuclide impacts the DCGL for U-235. The milk transfer factor for Ra impacts the U-234 and the Ra DCGLs. The Pa-231 plant transfer factor and the Ra milk transfer factor are determined to be non-sensitive, so the median value of the PDF in NUREG-6697 should be applied if the value is changed from the RESRAD default. The Pa-231 plant transfer factor assumed (1e-3) does not match the median of the PDF in NUREG-6697, Table 6.2.1, pg. 6-5 (1e-2) or the RESRAD default (2e-2). Similarly, the milk transfer assumed for Ra (1e-4) does not match the median of the PDF or the RESRAD default (both are 1e-3).

Path Forward: Justify the value used for the plant transfer factor for Pa-231 and for the milk transfer factor for Ra. If new values are applied, reevaluate the DCGLs for these radionuclides.

Westinghouse Response:

The values for protactinium plant and radium milk transfer factors provided in Hematite Decommissioning Plan (DP) Table 5-6 were not the correct values. The median values of the NUREG/CR-6697 PDF protactinium plant (1.00 E-02) and radium milk

(1.00 E-03 pCi/L per pCi/d) transfer factors cited by NRC in the Basis for HDP-C5-Q5 are correct and will be included in the revised HDP RESRAD dose factor library. A review of all other transfer and fish bioaccumulation factors listed in DP Table 5-6 was completed and identified the need to revise the following additional factors to be consistent with the NUREG/CR-6697 PDF median values: lead plant (4.00 E-03), lead milk (3.00 E-04 pCi/L per pCi/d), and thorium bioaccumulation for fish (100 pCi/kg per pCi/L). The revised RESRAD model will be prepared and provided to the NRC under a separate transmittal. When approved by the NRC, this revision will be incorporated into Chapter 5 of the DP.

6. (HDP-C5-Q6) Comment: The DP, Section 5.3.4.3, states that the sensitivity of the non-site specific parameters was determined using the probabilistic method in RESRAD for each CSM, assuming a different ratio of ROCs was used for the source term in each CSM. The ratios assumed require further justification. Also, it is not clear why the Alternative Excavation was not included as one of the CSMs in the sensitivity analysis.

Basis: NUREG-1757, Vol. 2, Appendix I states that the licensee may use expected concentrations or relative ratios, but also says they should evaluate the effect of uncertainty on the relative ratios. The ratios are based on the average concentration characterizing each soil layer, but the standard deviation of these data points is not provided. Since the Alternate Excavation CSM is used to determine the DCGLs that will be used in the DEEP zone (with the exception of Np-237), a sensitivity analysis should be performed for this CSM to determine any additional key parameters. It is unclear why the sensitivity of parameters was not tested for the Excavation CSM.

Path forward: Describe the data source for the averages used in Table 3-1 of reference 5-4. Evaluate the effect of uncertainty on the relative ratios of soil activity concentrations used. Perform a sensitivity analysis for the Alternative Excavation CSM, or explain why the important parameters for this model have already been identified using one of the other CSMs. Evaluate the sensitive parameters for the Alternate Excavation CSM.

Westinghouse Response:

The source of the radionuclide mixture data provided in Table 3-1 in Reference 5-4 (and duplicated in Table 5-4 in DP Chapter 5) was the characterization data from the surrogate the evaluation area (SEA) most applicable to the given CSM as follows:

- Surface CSM – Plant Soil SEA
- Root CSM – Tc-99 SEA
- Deep CSM – Burial Pit SEA
- Uniform CSM – Entire Impacted Area

The averages of the respective data sets were used to select the concentration values in Table 5-4. Data variability could possibly affect the uncertainty results for radionuclides that are present in low percentage but variability would not affect uncertainty results for radionuclides present in a high percentage, i.e., uranium. To evaluate the potential for data variability to influence the results of the uncertainty analysis the relative concentrations of the radionuclides other than uranium were set to the upper 95 percent confidence level of the mean 95% UCL. The mixture was then recalculated, in terms of relative concentrations, using the mean value for uranium and the 95% UCL for the remaining radionuclides. The results are provided in Attachment 3 to HEM-10-85.

From Attachment 3, it is seen that the relative fractions of the trace ROC's (Am-241, Np-237, Pu-239/240, Ra-226, and Th-232) using the conservative 95 % UCL values remain very low. It is unlikely that additional sensitivity analysis of trace radionuclides using the conservative radionuclide mixture would change the results of the sensitivity analysis. The radionuclide mixture remains dominated by U-234.

Tc-99 is present at an intermediate percentage that could possibly be affected by variability in the mixture. However, the data in Attachment 3 indicates minor changes in the Tc-99 relative fraction when applied at the 95 % UCL. Another consideration is that the Tc-99 fraction assumed in the Root CSM sensitivity analysis was fairly high at 20%. This was sufficient to cause the Tc-99 plant transfer factor as being identified as sensitive along with other plant uptake related parameters such as depth of soil mixing layer and root depth. As stated in DP Section 5.3.4.3, if a parameter was identified as sensitive for any of the CSMs (Surface, Root, or Deep, Uniform) the conservatively selected parameter was applied to all CSMs. Therefore, if low fractions of Tc-99 concentration in the Surface and Deep CSMs masked a Tc-99 sensitive parameter the parameter would still have been identified in the root zone sensitivity analysis and applied to the Uniform and Deep CSM. It is unlikely that additional sensitivity analysis of Tc-99 using the conservative radionuclide mixture would change the results of the sensitivity analysis.

The RAI also asked why a sensitivity analysis was not performed for the Excavation scenario. Sensitivity analysis of the Excavation scenario was considered unnecessary. As stated above, if a parameter was identified as sensitive for any of the CSMs (Surface, Root, or Deep) the conservatively selected parameter was applied to all CSMs. The Excavation scenario is essentially a combination of the Surface and Root CSMs because the assumed soil depth is 0.9 m as opposed to 0.15 m for surface and 1.5 m for root. Therefore, any parameter that would be identified as sensitive in the Excavation scenario would also be identified as sensitive in the surface and/or root sensitivity analyses and therefore already accounted for.

7. (HDP-C5-Q7) Westinghouse continues further technical evaluation of this RAI and will submit the response upon completion of the technical evaluation.

8. (HDP-C5-Q8) (HDP-C5-Q8) Comment: The value chosen for Root Depth in Section 5.3.4.4 of the DP requires further justification.

Basis: On p 5-15 of the DP, a site-specific value (0.6 m), estimated from Missouri data, is described for root depth. The DP also states that the generic weighted average root depth for fruits, vegetables and grains, and leafy vegetables is 1.1 m. The value of 0.6 m is averaged with 1.1 m to obtain the chosen value of 0.9 m. The basis for why these values are averaged is not provided. The value of 0.9 m is also coincidentally the RESRAD default value for Root Depth, but this is not the reason provided for using the value of 0.9 m. The range of values in NUREG-6697 is 0.3 m to 4.0 m.

Root depth is negatively correlated with dose for SURFACE and ROOT because the longer the root, the more clean soil the root is able to reach in proportion to its total length. It is positively correlated with dose for DEEP because the longer the root, the greater the proportion of it that lies in the DEEP section. Applying a value of 0.9 m instead of 0.6 m would lower the doses for SURFACE and ROOT, but increase the doses for DEEP. The value chosen for Root Depth should be adequately conservative for the SURFACE, ROOT, and Alternative Excavation Conceptual Site Models.

Path Forward: Explain the basis for why the site-specific value of 0.6 m was averaged with non-site specific data from NUREG-6697, or choose the site-specific value. Demonstrate that the value chosen for Root Depth is adequately conservative for the SURFACE, ROOT and Alternative Excavation CSMs through a sensitivity analysis

Westinghouse Response:

Hematite Decommissioning Plan (DP) Section 5.3.4.4 states two methods for determining the RESRAD root depth parameter value. First, the amount of acres harvested for crops planted in Jefferson County, MO for human consumption were used as the weighting factors for calculating a root depth of 0.6 m. Second, the values entered into the RESRAD model for plant consumption were used to weight average root depths provided in NUREG/CR-6697 for calculating a root depth of 1.1 m. As noted in the DP section and the NRC's RAI, choosing a larger value for the root depth is conservative for the Surface and Root models. As stated in the DP section, the value of 1.1 m does not fully consider regional agricultural crops and was considered overly conservative. In order to introduce a reasonable level of conservatism in the parameter selection, the two values were simply averaged and coincidentally the average was equal to the RESRAD default value of 0.9 m. The final root depth value of 0.9 m was bounded by the NUREG/CR-6697 statement that most root depths of plants harvested for nutrients usually extend less than 1 m below the surface. In conclusion, a root depth of 0.6 m is an acceptable site-specific value; however, the averaged value of 0.9 m is considered a reasonably conservative value for the Surface and Root models.

DP Table 5-5 documents that the root depth parameter is negatively correlated with dose for the Surface and Root models. Therefore, the selection of 0.9 m for the root depth is adequately conservative. Because the Excavation Scenario model is bounded by the Surface and Root models, the root depth parameter will also be negatively correlated and a sensitivity analysis is not required. The response to RAI HDP-C5-Q6 further discusses that a sensitivity analysis was not required for the Alternative Excavation Scenario model.

9. (HDP-C5-Q9) Westinghouse continues further technical evaluation of this RAI and will submit the response upon completion of the technical evaluation.

10. (HDP-C5-Q10) Westinghouse continues further technical evaluation of this RAI and will submit the response upon completion of the technical evaluation.

11. (HDP-C5-Q11) Westinghouse continues further technical evaluation of this RAI and will submit the response upon completion of the technical evaluation.

12. (HDP-C5-Q12) Comment: The value used for direct ingestion in RESRAD-BUILD as described in Section 4.2.3.1 of the DP seems to be based on information about indirect ingestion rates.

    Basis: The information in NUREG/CR-5512 that was used as the basis for the direct ingestion parameter is for "secondary ingestion" which is defined in NUREG/CR-5512 as the "Ingestion of removable surface contamination inside buildings that is transferred from contaminated surfaces via hands, food, and other items to the mouth". This differs from direct ingestion, which corresponds to the consumption of the contaminated material directly from the source. Therefore, it might not be appropriate to use the information on secondary ingestion in NUREG/CR-5512 as the basis for the direct ingestion parameter.

    Path Forward: Provide additional justification for the value selected for the direct ingestion parameter or provide a revised analysis using an appropriate value for this parameter.

    Westinghouse Response:

    Section J.3.6 of the *User's Manual for RESRAD-BUILD Version 3* (ANL/EAD/03-1, June 2003) provides a definition for the Indirect Ingestion Rate (INGE2) parameter.

    This parameter represents the ingestion rate of deposited material for a receptor at a specified location inside the building. This rate represents the transfer of deposited contamination from building surfaces to the mouth via contact with hands, food, or other objects. The indirect ingestion rate is expressed as the surface area contacted per unit time. This rate is different from the direct ingestion rate, since an individual does not need to be in the same room as the source to be exposed by this pathway.

    Section J.4.8 provides the following discussion for the Direct Ingestion Rate (INGE1) parameter.

This is the direct ingestion rate of the source by any receptor in the room. Each receptor will ingest the source at a rate determined by the product of the ingestion rate and the amount of contamination in the source at that time. Direct ingestion is possible only if the receptor and the source are in the same room.

Table 3.1 of the *User's Manual for RESRAD-BUILD Version 3* provides a value of 1.12E-04 m^2/h for Building Occupancy for the "Receptor indirect ingestion rate" (INGE2), which is consistent with the average ingestion rate of 1.1E-04 m^2/hr from NUREG/CR-5512. For the "Direct ingestion rate" (INGE1), Table 3.1 describes that the Building Occupancy value is calculated "from the default ingestion rate of 1.1E-04 m^2/hr in NUREG/CR-5512." For example, the Direct Ingestion Rate for the Small Office in DP Table 5-16 is calculated by dividing the ingestion rate of 1.1E-04 m^2/hr by the source area of 37.44 m^2.

The above discussions do not imply that the ingestion mechanisms are different. In both cases, the ingestion mechanisms are the same, e.g. hands to mouth. For indirect ingestion, the source is the deposited material. For the direction ingestion, the source is the existing material present on the surfaces.

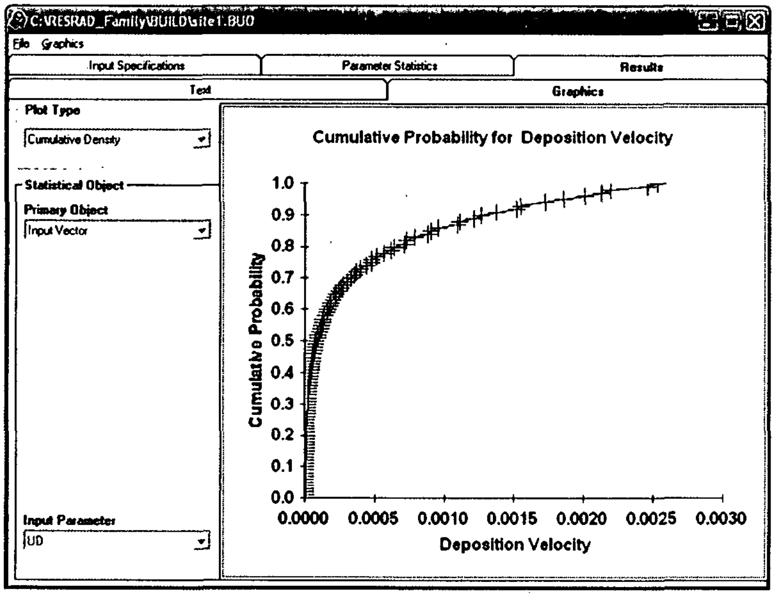
13. (HDP-C5-Q13) <u>Comment</u>: It is not clear how the median, 25th, and 75th percentile values used in the RESRAD-BUILD analyses were calculated from the distributions provided in NUREG-6697.

<u>Basis</u>: The values used in the RESRAD-BUILD analyses for parameters such as the deposition velocity, resuspension rate, building exchange rate, and the source lifetime were based on median, 25th percentile, and 75th percentile of distributions presented in NUREG-6697 as described in Section 5.4.3.1 of the DP. However, it is not clear how these values were calculated from these distributions.

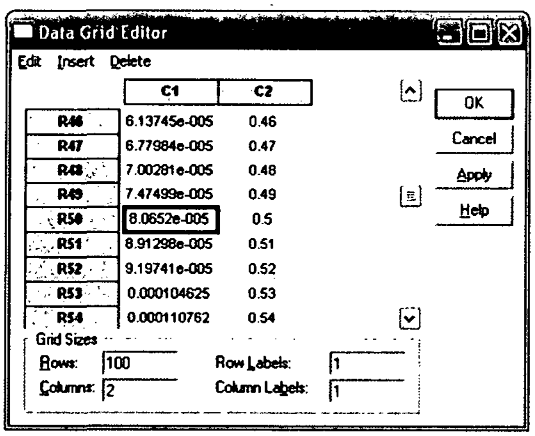
<u>Path Forward</u>: Provide a description of the methodology used to calculate the median, 25th percentile, and 75th percentile values from the distributions in NUREG-6697.

<u>Westinghouse Response:</u>

The median and 25th and 75th percentiles of NUREG/CR-6697 distributions were calculated using the following process. A RESRAD-BUILD default case was used with all default values except that the source type was set to area. All default "Sample specifications" under "Uncertainty Analysis Input Summary" were used except that the number of repetitions was changed from 3 to 1. For each parameter of interest, the default parameter distribution was set (by hitting F8 on the parameter of interest). After RESRAD-BUILD performed the calculations, the "Uncertainty Graphics" module was run. From the "Graphics" tab, the "Cumulative Density" "Plot Type" was viewed with the "Primary Object" set to "Input Vector." For Deposition Velocity, "UD" was selected as the "Input Parameter." See the following figure.



On the plot, the cumulative probability curve was right-clicked with the mouse and the option "Edit Chart Data..." was selected. For Deposition Velocity, the median value was determined by first finding the value of "0.5" in column "C2." Then, the corresponding "C1" value was selected as the median of the NUREG/CR-6697 distribution for Deposition Velocity. See the figure below which shows the source of the 8.07 x 10^{-05} m/sec provided in Section 5.4.3.2 of the DP.



To determine the 25th percentile, the above process was used except that a value of 0.25 in column "C2" was used. Similarly, a value of 0.75 in column "C2" was used to determine the 75th percentile.

14. (HDP-C5-Q14) Comment: The value used for the source lifetime parameter in RESRAD-BUILD may not sufficiently bound the calculated dose.

Basis: The source lifetime parameter was identified in Section 5.4.3.2 as being a sensitive parameter with a negative correlation to dose, and the 25th percentile of the distribution in NUREG-6697, or 17,918 days, was selected for this parameter value. However, the source lifetime distribution in NUREG-6697 is a triangular distribution with a most likely value of 10,000 days. Because the calculated dose is sensitive to the value of this parameter, it may not be appropriate to select a value for this parameter that results in a lower dose than the "most likely" value.

Path Forward: Provide additional information supporting the parameter value used for the source lifetime in RESRAD-BUILD, or provide a revised analysis using an alternate value for this parameter.

Westinghouse Response:

The 17,918 day source lifetime parameter is the 25th percentile and was chosen in accordance with the methodology described in Chapter 5 and is consistent with NRC guidance. The 17,918 day value has been accepted by the NRC in DCGL calculations by other licensees (NASA Plum Brook and Connecticut Yankee). The median value of the distribution is 33,254 days. The 10,000 days value is the 9th percentile. The 25th percentile value is considered sufficiently conservative.

15. (HDP-C5-Q15) Westinghouse continues further technical evaluation of this RAI and will submit the response upon completion of the technical evaluation.

16. (HDP-C5-Q16) Comment: The range of ratios observed for the radionuclides in the samples taken from the buildings was not considered in the RESRAD-BUILD sensitivity analysis.

Basis: The sensitivity analysis that was performed for the RESRAD-BUILD calculations described in Section 5.4.3.2 was based on the relative ratios of radionuclides that are presented in Table 4-1 of the DP. These values seem to be based on the total amount of a radionuclide in the samples divided by the total activity of all radionuclides in the samples. This corresponds to a mass weighted average of the ratios over all of the building samples. However, based on the data presented in Table 4-1, the relative ratios of the radionuclides seem to vary widely from sample to sample. Because the dose from different radionuclides can be sensitive to different parameters, it is important to consider uncertainty in the relative ratios of the radionuclides in the sensitivity analysis.

Path Forward: Provide an evaluation of the effect of uncertainty in the relative ratios of the radionuclides on the outcome of the sensitivity analysis.

Westinghouse Response:

The distribution of Radionuclides of Concern (ROC) in structures that is presented in Table 4-1 of the Decommissioning Plan is based upon two material samples taken in Building 230 during characterization, and additional samples obtained from the drain systems in Buildings 230 and 110. The small population of samples obtained from surfaces outside of the drain system was caused by the absence of any significant amount of contamination on surfaces. Given this limitation on the number of samples, and the fact that assuming the trace amount of transuranic radionuclides found in the additional samples taken from the drain systems in Building 110 and Building 230 would result in a conservative estimate of the potential dose, the two groups of samples were combined.

Two additional structures, Building 115 and the Sanitary Wastewater Treatment Plant (SWTP) Shed, may be demolished prior to license termination or, if deemed economically advantageous, may remain intact after license termination. If the decision is made that these additional structures are to remain and be subjected to FSS, the same radionuclide mixture as identified in Building 230 will be assumed given the absence of a sufficient amount of contamination in these two additional buildings to provide a sample for laboratory analysis.

The distribution presented by Table 4-1 shows a mixture dominated by the uranium isotopes at 99.12% of the distribution with the other five ROC (Am-241, Np-237, Pu-239/240, Tc-99 and Th-232) representing only 0.88% of the distribution. It should also be noted that in the fifteen samples that represent this population, Am-241 was detected at a concentration greater that Minimum Detectable Concentration in only one sample; Np-237 was positively detected in three samples; and Tc-99 was detected in a concentration greater than MDC in four of the fifteen samples. Using the dose to source ratios presented in Table 5-19 of the DP, the relative dose contribution from the uranium isotopes account for 23.20 mrem/yr and the other five radionuclides contribute 1.64 mrem/yr.

To assess the variability and to conservatively estimate the potential contribution to dose from the five non-uranium ROCs, the upper confidence limit of the mean ($UCL_{0.95}$) for the sum of the activities contributed by Am-241, Np-237, Pu-239/240, Tc-99 and Th-232 was compared to the average concentration of the uranium isotopes. Under these assumptions, the uranium isotopes account for 99.08% of the activity and the non-uranium isotopes account for 0.92% of the activity. Again, using the dose to source ratios presented in Table 5-19 of the DP, the relative dose contribution from the uranium isotopes account for 23.4 mrem/yr and the other five radionuclides contribute 1.70 mrem/yr. This illustrates that the variability and contribution to dose from the non-uranium radionuclides in the radionuclide distribution used for the structures at HDP is not sensitive.

17. (HDP-C5-Q17) <u>Comment</u>: It is not clear how the volumetric contamination in the buildings will be evaluated if it is found.

<u>Basis</u>: The DCGL values described in Section 5.4.4 derived for the buildings that will remain on site are areal concentrations for surface contamination. It is not clear what criteria will be used if areas of volumetric contamination are found in the buildings. In addition, Chapter 4 of the DP states that isolated spots of elevated activity were identified within the seams and joints on the concrete floor, which implies that

volumetric contamination may exist in these areas.

Path Forward: Provide information about how volumetric contamination will be evaluated if any is found within the buildings. If volumetrically contaminated building material will remain on site, provide volumetric DCGL values for the buildings.

Westinghouse Response:

Westinghouse does not believe that volumetric contamination is present within buildings expected to remain at the time of license termination. Based on operational history and process knowledge, the pathways for volumetric contamination such as spills of radioactive liquids and neutron activation were not identified in the buildings to remain. HDP does recognize that a viable avenue for sub-surface concrete contamination to occur in remaining buildings exists via cracks, seams etc. Section 8.3.3 of the DP describes the focus on such crack/seam areas, and the use of Remedial Action Support Surveys and remediation techniques to identify potential sub-surface contamination and remove it. Further focus is given to such areas during FSS.

Building 110 is used for administrative functions and as a security point ingress/egress for the site. Building 115 housed pumps for fire protection, and has no history of the use of radioactive materials. Building 231 is located southwest of Building 230 and is used for covered storage of non-radioactive supplies and equipment. Building 230 was constructed in 1992 and was designed to support rod loading and fuel assembly operations. Fabricated fuel pellets were taken to Building 230 where they were loaded into empty fuel rods, plugged, and seal-welded. Sanitary Wastewater Treatment Plant Shed and System processed waste water containing low levels of radioactivity. The steel piping, and aeration and digestion tanks are not subject to volumetric contamination to any significant degree. The operational history for these buildings did not include the use of process liquids beyond laboratory quantity, therefore migration into the concrete matrix and volumetric contamination is considered very unlikely.

In response to NRC's question regarding the method to evaluate volumetric contamination , if volumetric contamination is potentially present, Section 14.4.4.1.5.1 of the DP describes the use of drilling and coring to obtain samples of the concrete for laboratory analysis and subsequent verification that appropriate DCGLs have been met. Further emphasis on biased sampling during FSS for areas of stress cracks, floor and wall interfaces, penetrations etc. is included in Section 14.4.4.1.5.1. If volumetric contamination were present, core samples of the concrete would be obtained for laboratory analysis.

18. (HDP-C5-Q18) Comment: DCGL values discussed in Section 5.4.4 were not provided for the ventilation ducts in the buildings that will remain on site and the dose to the building occupant from contaminated ducts was not considered.

Basis: It is not clear how the residual contamination in the interior of ventilation ducts will be evaluated because DCGL values were not provided for the ducts. In addition, the potential for the residual contamination in the ducts to cause a dose to the building occupants was not evaluated. Because the building surface DCGL values correspond to a dose of 0.25 mSv/yr, an individual who receives a dose from the building surfaces as well as the interior of the ducts could receive a dose that is above 0.25 mSv/yr.

Path Forward: Provide information about the DCGL values that will be used for the interior of ducts. Also provide information about the potential dose from the residual contamination in the ducts to a building occupant. If the potential dose from the residual contamination in the ducts is not insignificant, revise the building surface DCGL values to account for the fact that the occupant could receive a dose both from residual contamination on the building surfaces and the residual contamination in the ducts.

Westinghouse Response:

DCGL have not been established for the interior surfaces of ventilation components that are independent of the DCGL that apply to the building surfaces and the exterior surfaces of ventilation components. As an alternative to the modeling that would be required to develop DCGL that are specific to the interior surfaces of ventilation components, the levels of surface contamination within ventilation components to remain at the time of license termination will be compared to the limits for surface contamination measurements specified for U-nat, U-235, U-238 and associated decay products in Table 1, Acceptable Surface Contamination Levels "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," dated April 1993.

Although the basis for this approach differs from the approach used for building surfaces, it is believed to be sufficiently conservative since these alternate criteria are only a small fraction of the values that have been proposed as acceptable for building surfaces; and these values are accepted as an industry standard for release of materials and equipment from similar facilities.

In addition to the measurements of surface contamination, air sampling will be performed at outlets of ventilation ducting remaining on site to directly assess the dose contribution from ventilation ducting. Air sampling locations will be rotated to various ventilation ducting openings of the ventilation systems that will remain. The average of the calculated dose contributions from the air samples associated with the remaining ventilation systems will be added to the dose associated with the surface contamination measurements within each surface and structure survey unit as a final compliance measure to ensure the 25 mrem/yr criterion is met.

To illustrate, using the average of two air sample results, assuming the activity is all attributable to Uranium, applying a 2,000 hour/year occupancy and a breathing rate of 1.4 cubic meters/hour resulted in a potential intake of 1E-5 uCi of Uranium. Comparing this value to the ALI for Uranium in 10 CFR 20, Table 1, Column 2 would result in 0.8 mrem/yr. If these two samples were the entire population of air samples, an increment of

0.8 mrem/yr would be assigned to each of the survey units (effectively reducing the acceptable amount of residual surface contamination) to demonstrate compliance.

The following revisions will be made to DO-008-04, Hematite Decommissioning Plan:

The first bullet in paragraph 3 of Section 8.4.1 will be reworded to remove reference to ventilation ducts, as ventilation ducts are discussed in Section 8.4.2. The revision to the wording will also address the proposed change to address HDP-C5-Q19 as follows:

> *RASS will be performed on the interior surfaces of drain systems to determine if remediation will be required. Contaminated drain systems will be remediated to levels that do not exceed the DCGLs that are approved for building surfaces; or will be physically removed and packaged for disposal at an off-site facility; or will be remediated to levels that do not exceed the DCGLs that are approved for buried piping and filled with grout.*

Section 14.4.4.1.5.4 will be revised to include a discussion on obtaining measurements of surface contamination followed by a comparison to the alternate criteria and collection of air samples at ventilation openings. The text will be revised to read as follows:

> *14.4.4.1.5.4 Ventilation Ducts – Interiors*
> *Measurements of total and removable surface contamination will be obtained at access points, and at locations where radioactivity is most likely to have accumulated. (e.g., bends, transitions, filter housings) The measurements of surface contamination will be compared to the limits for surface contamination measurements specified in "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," dated April 1993. Air sampling will be performed at outlets of ventilation ducting remaining on site to directly assess the dose contribution from ventilation ducting. Air sampling locations will be rotated to various ventilation ducting openings of the ventilation systems that will remain. The average of the calculated dose contributions from the air samples associated with the remaining ventilation systems will be added to the dose associated with the surface contamination measurements within each surface and structure survey unit as a final compliance measure to ensure the 25 mrem/yr criterion is met.*
>
> *Measurements of surface contamination obtained from the exterior surfaces of ventilation components will be compared to the DCGL that apply to the building surfaces.*

19. (HDP-C5-Q19) Comment: The conceptual model used for calculating the DCGL values for the pipes described in Section 5.5 does not seem consistent with the expected configuration of the residual contamination following decommissioning activities.

Basis: There are several aspects of the conceptual model assumed in calculating the DCGL values for pipes that need additional clarification. In the calculation of these DCGL values, it is assumed that the pipe is filled with soil that has an activity of radionuclides equal to the DCGL values calculated for the root depth soil layer (i.e., 15 cm to 1.5 m). These volumetric DCGL values were then converted to an areal value on the interior surface of the pipe, which is effectively equivalent to assuming a dilution of the material on the surface of the pipe over the whole volume of the pipe. This resulted in the proposed areal DCGL values to be larger for large pipes than for small pipes because in the large pipes the material was averaged over a larger volume. The pipes that remain on site at the time of license termination may remain in service and may not be filled with soil, so it is not clear that this conceptual model applies. In addition, the dose to an individual may be different from a thin skin of contamination on the surface of the pipe than from a volumetric source containing the same amount of activity. Furthermore, the contribution of the pipes to the dose for the resident and building occupant scenarios is not considered and the potential dose from the pipes being excavated and reused is not considered. Information on the depth of the pipes below the ground surface was also not provided. The depth of the pipes can have a large impact on the dose because of the shielding from the soil or pavement above the pipe.

Path Forward: Provide additional justification for the assumed conceptual model used in the development of the DCGL values for pipes, including the basis for assuming that the residual contamination in the pipes is averaged over the volume of the pipe. Alternatively, a new calculation of DCGL values for the pipes can be performed and provided along with justification for the conceptual model assumed in the new calculation. The potential dose from the pipes to the building occupant and resident receptors should be considered, and the potential dose from a pipe that is dug up and reused should also be considered. Information on the depth of the pipes below the ground surface should also be provided.

Westinghouse Response:

The conceptual model for buried piping DCGL calculations assumes that the pipes remain *insitu*, and was primarily intended to address smaller diameter pipes with potential for contamination from operations. The dose in an occupied building due to external exposure from buried piping underneath the building foundations and containing residual radioactivity equivalent to the proposed buried piping DCGL is very low as shown in Attachment 4, Figures 1 and 2. The dose to an excavation worker from excavated piping containing residual radioactivity equivalent to the proposed buried piping DCGL is also expected to be low due to limited contact time with the excavated pipe as shown Attachment 4, Figures 3 and 4.

HDP expects the contamination levels in the majority of the piping to be very low relative to the proposed DCGL for buried piping. To ensure conservatism, HDP will utilize the Building DCGLs (Small Office) from Chapter 5 Table 5-19 as release criteria for buried piping to remain in place. HDP will provide NRC with details of the buried piping FSS survey methodology as committed to in HDP's response to NRC RAI HDP-C14-Q6. An updated list of piping to remain (Table 5-21) is included with this correspondence as part of the HDP response to HDP-C5-CC4.

In the unlikely event buried piping to remain exceeds Building DCGLs and cannot be practically decontaminated or removed, HDP will verify the piping meets Buried Pipe DCGLs (Table 5-22) and grout the piping in place. The grout will eliminate the potential for re-use of the pipe, ensure remaining contamination is fixed in place, and add a volume of clean material to the pipe interior. HDP will evaluate the specific dose from piping to be left as in Attachment 4, Figures 3 and 4 examples, and account for the dose in the affected survey unit. Note that this approach has been accepted by NRC at several sites including NASA Plum Brook and Connecticut Yankee.

CC1. (HDP-C5-CC1) Comment: There appears to be a mistake in the units for Table 5-15.

Basis: Table 5-15 provides approximate building dimensions in meters. However, in Table 5-16, the same values for the building dimensions are reported as being in feet.

Path Forward: Provide the correct units for the dimensions provided in Table 5-15.

Westinghouse Response:

Westinghouse will revise the heading of Table 5-15 to denote the correct units of "feet".

"Approximate Dimensions (Feet)"

The values and units reported in Table 5-16 are correct.

CC2. (HDP-C5-CC2) Comment: There appears to be a typo in the number of years required for 0.6 m of erosion to occur.

Basis: On page 5-7, it is stated that "potential erosion over 100 years is estimated to be 0.6 m". However, the erosion rate used in RESRAD corresponds to 0.6 m of erosion in 1000 years.

Path Forward: Provide the correct number of years required for 0.6 m of erosion to occur.

Westinghouse Response:

The correct timeframe for erosion to occur is 1,000 years versus 100 years. Westinghouse will revise section 5.3.3.1 of DP Chapter 5 as follows to indicate that the potential erosion over 1,000 years is estimated to be 0.6 m.

"The Root stratum represents soil in the root zone (15 cm to 1.5 meters) and accounts for the potential removal of soil due to erosion over the 1,000-year modeling period. The root depth is assumed to be 0.9 m and potential erosion over 1,000 years is estimated to be 0.6 m (see Section 5.3.4.2 for parameter justification). Using the combined 1.5 m depth ensures that the thickness of the root stratum will equal or exceed the 0.9 m root depth for the entire 1,000-year period."

CC3. (HDP-C5-CC3) Comment: Section 5.4.3.1 seems to cite the wrong table.

Basis: On page 5-21, it is stated "The activity fractions of the ROCs in site buildings are provided in Chapter 4, Table 4-2". However, this information is not presented in Table 4-2.

Path Forward: Provide the correct reference for the location of this information.

Westinghouse Response:

The correct reference for this information should have been Table 4-1 versus Table 4-2. Westinghouse will revise section 5.4.3.1 of DP Chapter 5 as follows to indicate that the activity fractions of the ROCs in site buildings are provided in Chapter 4, Table 4-1.

"In accordance with NUREG-1757, Section I.7.5, the activity fractions (or mixture) of the ROCs were used in the sensitivity analysis. The activity fractions of the ROCs in site buildings are provided in Chapter 4, Table 4-1, which is reproduced in Table 5-17."

CC4. (HDP-C5-CC4) Comment: Some possible discrepancies have been identified in the information presented in Table 5-21 of the Decommissioning Plan about the buried piping expected to remain at license termination.

Basis: The piping listed in Table 5-21 for buried piping expected to remain at license termination includes piping for Building 240. However, Chapter 4 of the DP indicates that this building will be demolished prior to license termination. Additionally, this table does not include piping for Building 231, which is expected to remain at the time of license termination. Chapter 4 of the DP also states that Building 115 (Fire Pump House), the Sanitary Wastewater Treatment Plant (SWTP) Shed and Building 235 may also remain on site at the time of decommissioning, but piping associated with these buildings is not included in this table.

Path Forward: Clarify if piping associated with Building 240 will remain at the time of license termination and clarify if any piping associated with Building 231 will remain on site. Provide information on piping associated with Building 115, Building 235, and the SWTP that may remain on site at the time of license termination.

Westinghouse Response:

The Storm Drain and Sanitary/Grey Drains piping outside the footprint of Building 240 may remain at the time of license termination depending upon the cost associated with decontamination and/or survey versus the cost of removal and disposal. Piping within the footprint of Building 240 is expected to be removed following removal of the concrete floor slab. Although Table 5-21 lists the underground piping as being associated with Building 240, this is only to provide a general location of the piping. Figure 4-1 of the DP clearly shows the piping is outside the Building 240 footprint. The piping and area drains will be protected during demolition and slab removal of Building 240.

Although Building 231 is expected to remain at license termination, there are no systems which reside in or traverse the building with the exception of a sprinkler supply line which was roughed in (capped at the floor) but never activated. The sprinkler line is part of the Public Water System. Figure 4-1 and Table 5-21 will be revised to reflect this piping.

Building 235 will not remain at license termination and there are no systems which reside in or traverse the building.

Building 115 may remain at the time of license termination and the only system associated with this structure is the Public Water inlet and outlet that have been removed from within the structure. Figure 4-1 and Table 5-21 will be revised to reflect the remainder of the piping.

The Sanitary Wastewater Treatment Plant shed does contain 2"-4" drain lines which interface with the Sanitary System beneath the structure. Figure 4-1 and Table 5-21 will be revised to reflect this piping.

The Public Water System provides drinking water and potable water to Building 110 and Building 230 and is also used for fire suppression. It will remain at the time of license termination. Although contamination is extremely unlikely, this system will be added to Table 5-21, Table 14-16 (survey unit list), and Figure 4-1.

CC5. (HDP-C5-CC5) Comment: A DCGL value is not listed in Table 5-12 for Np-237.

Basis: The DP states, "Np-237 is an exception and the DEEP DCGL (of 0.3 pCi/g) will be used", but it is not clear if 0.3pCi/g will be used for SURFACE, ROOT, and UNIFORM layers as well as the DEEP layer.

Path Forward: Please clarify if the DEEP value of 0.3 pCi/g should appear in the 5-12 Table. Please also clarify the values intended to be used for SURFACE, ROOT and UNIFORM.

Westinghouse Response:

The heading to Table 5-11 which contains the values for Np-237 for the Excavation and Deep Scenarios will be revised as follows;

Radionuclide	DSR (mrem/yr per pCi/g)	Excavation Scenario Concentrations Corresponding to 25 mrem/yr (pCi/g)	Deep Scenario Concentrations Corresponding to 25 mrem/yr (pCi/g)

In addition, Table 5-11 will be revised to include the following footnote;

"The Excavation Scenario DCGLs listed in Table 5-12 were derived by multiplying the "Excavation Scenario Concentrations Corresponding to 25 mrem/yr" in Table 5-11 by a factor of two to account for the mixing with the assumed 1.5 m clean cover soil during excavation."

Table 5-12 will be revised to include the following footnote;

"The Deep DCGL of 0.3 pCi/g for Np-237 will be used in lieu of the derived Excavation Scenario DCGL of 11.2 pCi/g when located in the deep strata as it is more limiting."

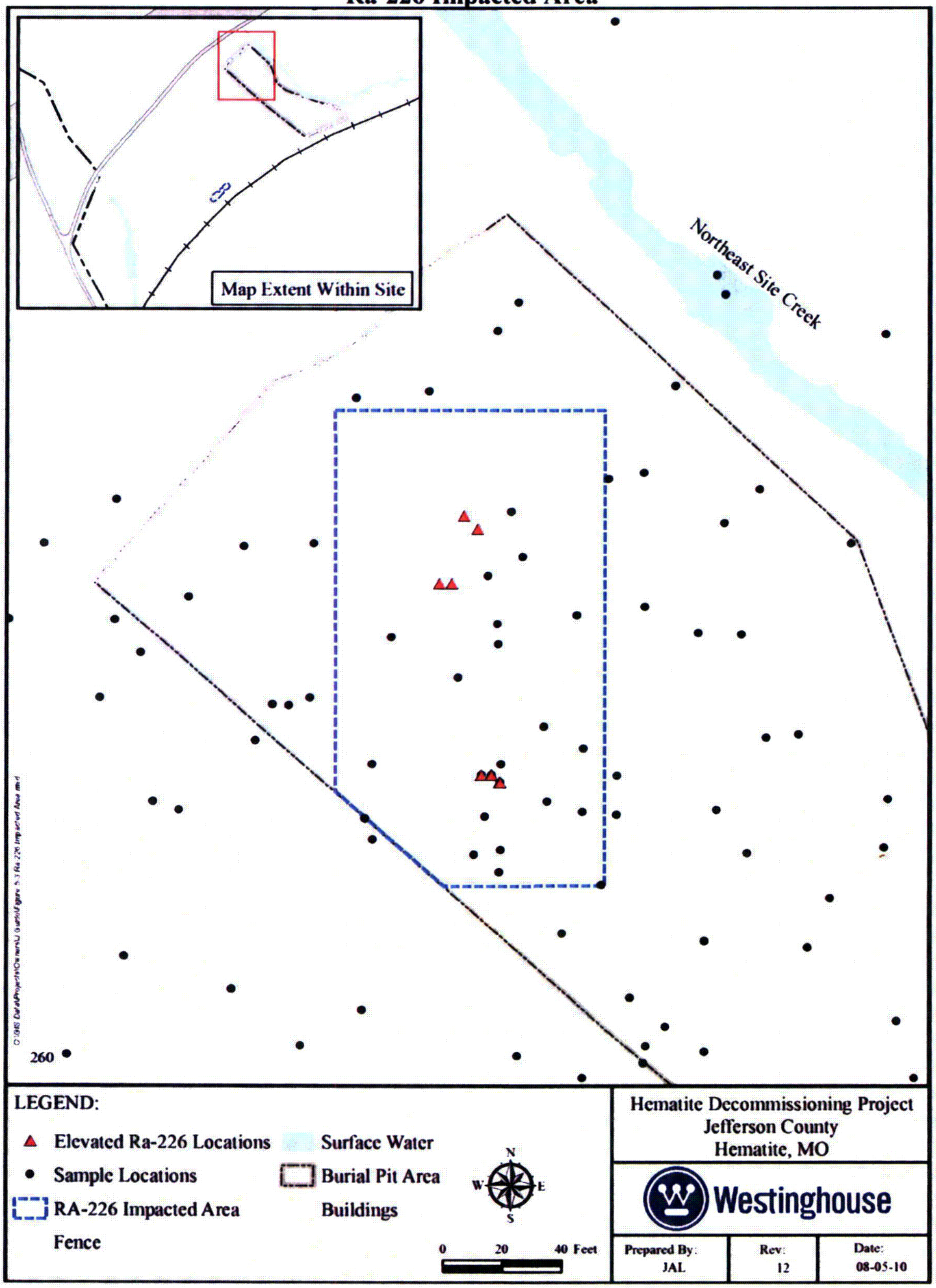
The Np-237 values that will be used in the surface, root and uniform CSM strata are as stated in Table 5-7, *"Soil DSRs and DCGLs – Surface"* (Np-237 – 17.3 pCi/g), Table 5-8, *"Soil DSRs and DCGLs – Root"* (Np-237 – 5.0 pCi/g) and Table 5-10, *"Soil DSRs and DCGLs – Uniform"* (Np-237 – 0.3 pCi/g).

ATTACHMENT 2

REPLACEMENT FIGURE 5-3
Ra-226 Impacted Area

(referred to in response to RAI HDP-C5-Q1)

REPLACEMENT FIGURE 5-3
Ra-226 Impacted Area



ATTACHMENT 3

Comparison of Radionuclide Mixture Using Mean and 95% UCL for Non-Uranium Isotopes

(Referred to in response to RAI HDP-C5-Q6)

Comparison of Radionuclide Mixture Using Mean and 95% UCL for Non-Uranium Isotopes

Radionuclide	Surface		Root		Deep		Uniform	
	Table 5-4 Values [a] (Relative Fractions)	95% UCL[a,b]	Table 5-4 Values [a] (Relative Fractions)	95% UCL[a,b]	Table 5-4 Values [a] (Relative Fractions)	95% UCL[a,b]	Table 5-4 Values [a] (Relative Fractions)	95% UCL[a,b]
Am-241	N/A	0.05 %	0.02 %	0.05 %	0.001 %	0.01 %	0.001 %	0.004 %
Np-237	0.02 %	0.02 %	0.1 %	0.1 %	0.1 %	0.1 %	0.004 %	0.005 %
Pu-239/Pu-240	0.001 %	0.01 %	0.002 %	0.01 %	0.002 %	0.005 %	0.0003 %	0.0006 %
Ra-226	N/A	N/A	0.05 %	0.1 %	3.5 %	7.2 %	0.1 %	0.2 %
Tc-99	11.0 %	16.9 %	19.6 %	25.3 %	0.5 %	0.9 %	4.6 %	8.4 %
Th-232	N/A	N/A	N/A	0.04 %	0.1 %	0.3 %	1.1 %	2.5 %
U-234	68.7 %	64.1 %	62.7 %	58.1 %	60.4 %	57.8 %	90.7 %	85.5 %
U-235	2.8 %	2.6 %	7.1 %	6.6 %	4.7 %	4.5 %	1.0 %	0.9 %
U-238	17.5 %	16.3 %	10.4 %	9.7 %	30.7 %	29.3 %	2.5 %	2.4 %

[a] Negative average or average plus 1.645 □ SDOM (standard deviation of the mean) activity concentrations were not included.
[b] Average used for U-234, U-235, & U-238. Average plus 1.645 □ SDOM used for all other radionuclides.

ATTACHMENT 4

Summary of Modeled Geometries for External Dose from Buried and Excavated Piping Figure 1 through 4

(Referred to in response to RAI HDP-C5-Q19)

Summary of Modeled Geometries for External Dose from Buried and Excavated Piping

Scenario	Geometry	Result
Figure 1 External Dose to Resident / Future Occupant Due to Abandoned Pipe. Exposure to exposed end, in situ or excavated pipe	25 foot cylinder, 24 inch radius. Source term equal to the applicable surface contamination at gross activity DCGL from Table 5-22, assigned to individual radionuclides using activity fractions from Table 5-17. Cylinder is filled with grout with density of 1.6 g/cm3. Dose measured at 1 foot from end of pipe.	2.0 E -5 mR/hr
Figure 2 External Dose to Resident / Future Occupant Due to Abandoned Pipe. Exposure at surface due to subsurface pipe.	25 foot cylinder, 24 inch radius. Source term equal to the applicable surface contamination at gross activity DCGL from Table 5-22, assigned to individual radionuclides using activity fractions from Table 5-17. Cylinder is filled with grout with density of 1.6 g/cm3. 3 foot shield composed of soil with density of 1.6 g/cm3 Dose measured at 1 foot from shield of pipe.	3.2 E -6 mR/hr
Figure 3 External Dose to Resident / Future Occupant Due to Abandoned Pipe. Excavated Pipe – Single Pipe	25 foot cylinder, 24 inch radius. Source term equal to the applicable surface contamination at gross activity DCGL from Table 5-22, assigned to individual radionuclides using activity fractions from Table 5-17. Cylinder is filled with grout with density of 1.6 g/cm3. Dose measured at 1 foot from pipe.	5.8 E -3 mR/hr
Figure 4 External Dose to Resident / Future Occupant Due Excavated Pipe – Multiple Pipes	Infinite plane source. Source term equal to the applicable surface contamination at gross activity DCGL from Table 5-22, assigned to individual radionuclides using activity fractions from Table 5-17. Cylinder is filled with grout with density of 1.6 g/cm3. Dose measured at 1 foot from plane. A reasonable example of 40 hours handling time for disposal of non-reusable pipe would result in 0.81 mR of exposure.	5.2 E -2 mR/hr

Summary of Modeled Geometries for External Dose from Buried and Excavated Piping
(continued)

Figure 1	Figure 2	Figure 3	Figure 4