ROBATEL technologies

August 25, 2014
Docket No. 71-9365
TAC No. L24876

Attn: Document Control Desk,
Director, Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety and Safeguards
US Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: **Response to Request for Additional Information letter dated August 8, 2014 for the Model No. RT -100 Package, Docket N° 71-9365.**

Reference: **Letter from P. Saverot (NRC) to C. Dane (Robatel Technologies), Request for additional information for the model No. RT-100 package, Docket No. 71-9365, August 8, 2014.**

Document Control Desk, dear Sir or Madam:

By the referenced letter, NRC requested that Robatel Technologies provide additional information for the NRC staff to complete their review of the request for an authorization to ship activated metals from the Rancho Seco decommissioning project using the Model No.RT-100 package. Please find in enclosure the Robatel Technologies, LLC response to NRC referenced Request for Additional Information letter. Should you or any member of the NRC staff have questions, please contact the undersigned at (540)-989-2878 or by email at cdane@robateltech.com.

Respectfully:

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cc: Pierre Saverot

NMSS24

Robatel Technologies Response to the Request for Additional Information for the Model N° RT-100 Package Docket N° 71-9365.

The NRC Request for Additional Information (RAI) identifies information needed by the staff with regard to its review of the Robatel Technologies request for an authorization to ship activated metals from the Rancho Seco decommissioning project using the Model No.RT-100 package. Each RAI question is followed by the Robatel Technologies response.

<u>RAI 1 Question:</u>

On page 6 of the report "RT-100 Transport Cask Contents Description, Rancho Seco Reactor Vessel Internals," the applicant states: "The dose rate obtained by modeling with Microshield® assumes a uniform distribution throughout the entire volume of the liner. This might explain the discrepancy between some of the measured and the estimated dose rate. This difference is larger for some activated metal and might be due to the geometrical form of the hardware and the way the waste was packed in the liner."

On page 5 of the report "Radiological Characterization of Rancho Seco Reactor Core Region Internals," the applicant states: "The Microshield calculation was based on a uniform source distribution. Results were normalized to the maximum measured dose rate at approximately the mid-point of the range."

From the data provided in Table 3, it appears that MicroShield may be inadequate for these calculations. Based on the information provided in the application, it appears that the homogenization of the source also might have introduced significant errors in the dose rate calculations because the activation of the various parts of the RVI is highly dependent on the neutron flux intensity of the locations. The side of a component facing the reactor will have significant more activation than the other side of the same component. This is true for all reactor internals components. Averaging the activity over a component might have severely underestimated the activation of the materials on the side facing the reactors and overestimated the activity on the opposite side. As such, if loaded in such a way that the highly activated side of a segment of the RVI was facing toward to outside of the liner, the dose rate could be substantially higher than what was calculated based on an average source.
The applicant needs to explain if any actions have been taken to correct this problem, when determining the source distribution in its shielding model for the RT-100 package.

The staff needs this information to determine compliance with the regulatory requirements of 10 CFR 71.47 and 71.51.

<u>RAI 1 response:</u>

Due to its inability to model non-uniform sources, the use of Microshield likely is insufficient for calculating dose rates for this specific application. The data in Table 3 was calculated by WCS using MicroShield for the purpose of estimating which liners would require a Type B cask based on the 3 m dose rate of the unshielded contents. This was necessary because the liners have been in shielded storage for years and are not readily available for inspection. The MicroShield analysis was used for this purpose only, and is not used at all in the basis for the requested authorization. Please note that the documentation provided to the NRC should be considered in the following order:

Robatel Technologies Response to the Request for Additional Information for the Model N° RT-100 Package Docket N° 71-9365.

1. **RT-100 Transport Cask Contents Description, Rancho Seco Reactor Vessel Internals**

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2. **Rancho Seco Class B/C Waste Segmentation-Source Term Definition (TR-13039-001)**

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3. **Rancho Seco Activated Metal Shielding Analysis (CN-13039-501)**

Each preceding document is an input to the next such that the WCS report "RT-100 Transport Cask Contents Description, Rancho Seco Reactor Vessel Internals" is a document whose purpose is to introduce the contents that are to be shipped in the RT-100 cask. More specifically this document gives a simple description of the activated reactor components and for each of the liners that are filled with the segmented components. The DAHER-TLI report TR-13039-001 draws information from the preceding WCS report and documentation from the Rancho Seco facility to outline a source term that will be used in the MCNP shielding analysis. The calculation report CN-13039-501 outlines the MCNP calculations for each individual liner in the RT-100 cask. It should be noted that the basis for shipping the liners filled with segmented reactor internals from the Rancho Seco facility to the WCS site is detailed in the report CN-13039-501. To provide more detail (inhomogeneity) of the source term, MCNP is used to ensure that the dose rates experienced during shipment of the package are less than the established regulatory dose rates.

In the MCNP shielding model, the distribution of activity between the components is considered as described in Section 2.3.1 (see figure 4) and Appendix B of CN-13039-501. For the MCNP models the components with the highest activities were placed closest to the side wall of the inner cavity of the RT-100 cask, this could be considered conservative since records of the actual loading configuration indicate that the more activated segments were placed in the center of the liner to take advantage of shielding from less activated segments. However, no extra consideration was given for the activity distribution within in each segment. It should be noted that the measured values provided in Table 3 are exposure rates recorded in 2005 with the components **only shielded by the other components and the liner itself**. With the RT-100 cask to provide additional shielding, and the more explicitly modeled MCNP source, the recorded dose rates will not be as sensitive to localized areas of increased activity as the case with little or no shielding provided and a simple homogenous source (as in Table 3). It can also be considered that the worst case calculated dose rates for these shipments were determined to be about 50% of the regulatory dose rate limits, leaving ample margin to cover the few simplifications made in the MCNP model.

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RAI 2 Question:

Explain how the corrected dose rates were determined and why the corrected dose rates for RVI-19 are smaller than that of RVI-008, RVI-016, RVI-017, and RVI-018 given the fact that the 2005 measured dose rates of RVI-19 were much greater than that of those components.

The applicant provides, in Table 3 of the report "RT-100 Transport Cask Contents Description, Rancho Seco Reactor Vessel Internals," corrected dose rates on contact and at 3 meters adjusted for decay from 2005 to February 2014. From this data, it is not clear how such adjustments were made. The applicant needs to explain whether these adjustments were made, i.e., based on measured or calculated dose rates, as reported in the table.

In addition, the applicant provides in Table 3 of the report "RT-100 Transport Cask Contents Description, Rancho Seco Reactor Vessel Internals" the estimated and measured contact dose rates in 2005. The applicant further provides corrected contact dose rates adjusted for decay from 2005 to February 2014. Given this fact that the dose rates measured in 2005 for RVI-19 were much larger than those of the other segments, it would be expected that the decay time corrected dose rate for RVI-19 is proportionally greater than the dose rates of the others. However, from these data, it appears that corrected dose rates for RVI-19 are smaller than that of RVI-008, RVI-016, RVI-017, and RVI-018.

The applicant needs to explain the inconsistency between these calculations and provide justification for the results. The applicant also needs to provide justification for the adequacy of its quality assurance program in determining the source terms which are critical to the assurance of compliance with the requirements of 10 CFR 71.47 and 71.51 on the package dose rates.

The staff needs this information to determine compliance with the regulatory requirements of 10 CFR 71.47 and 71.51.

RAI 2 response:

The results in Table 3 were used only as a predictive indicator of how many of the liners would require a Type B cask for shipment, since current dose rate measurements of the liners are not available. These results are not used to estimate dose rates for the purposes of the requested authorization.

However, to clarify the values presented in Table 3, all values are calculated except for the one column labeled "Measured Dose rate in 2005". Thus the corrected dose rates on contact (column 4) are determined with the activities used to calculate the contact dose rates in 2005 (column 2), decayed from 2005 to February 2014. This explains why the corrected dose rate (in Feb 2014) for RVI-019 is less than those for RVI-008, RVI-016, RVI-017, and RVI-018.

Although they are close, the decayed time corrected dose rates for the liners don't decrease proportionally to one another because the Microshield calculations used to generate these results considered ALL isotopes. These dose rates are based on an inventory of nuclides from the DW James reports where some segments contained different distributions of nuclides. It is not a simple decay of Co-60, but rather a collection of nuclides of which Co-60 and Fe-55 were the main contributors to gamma dose. So for a liner that has more of

a given shorter-lived isotope, the corrected dose rate will decrease at a greater rate than a liner with longer-lived isotopes.

For example, if the calculated dose rate for RVI-19 results from having a greater Fe-55 content than other liners, then the dose rate would decrease faster than those liners over time due to Fe-55's shorter half-life (Co-60 half-life=5.27 years, Fe-55 half-life=2.73 years).

The source term for each of the liners were calculated under contract by DW James & Associates as outlined in TR-13039-001. DW James & Associates operates under an independent NQA-1 quality assurance program, and calculates activity estimates using their SNAP program, which is benchmarked with ORIGEN 2.1.

RAI 3 Question:

Provide a summary table to demonstrate that the calculated dose rates meet the requirements of 10 CFR 71.47 and 71.51.

The applicant provides a calculation package and input files for the dose rates at NCT and HAC. However, there is no summary of dose rates or reference to other document in the main application document to demonstrate compliance with the regulations. The applicant needs to a summary table to demonstrate that the calculated dose rates meet the requirements of 10 CFR 71.47 and 71.51.

The staff needs this information to determine compliance with the regulatory requirements of 10 CFR 71.47 and 71.51.

RAI 3 response:

The summary table per RG 7.9 format and content is in TLI Calculation Note No. CN-13039-501 titled "Rancho Seco Activated Metal Shielding Analysis," in Tables 5 and 6 as shown below. These tables summarize the range of dose rates calculated for all of the included liners for NCT and HAC. The calculated dose rates of each individual liner are shown in Appendix F of CN-13039-501.

Table 5 Highest Calculated NCT Dose Rates

Normal Conditions of Transport	Package Surface mSv/hr (mrem/hr)			2 Meter from Edge of Vehicle mSv/hr (mrem/hr)	Vehicle Occupied Position mSv/hr (mrem/hr)
Radiation	Top	Side	Bottom	Side	Side
Gamma (Co-60)	0.0215 - 0.2767 (2.15 – 27.67)	0.0912 - 0.7468 (9.12 – 74.68)	0.0347 - 1.129 (3.47 – 112.9)	0.0111 - 0.054 (1.11– 5.40)	0.0027 – 0.0125 (0.27– 1.25)
Neutron					
10 CFR 71.47 (b)(2)	2 (200)	2 (200)	2 (200)	0.1 (10)	0.02 (2)

Table 6 Highest Calculated HAC Dose Rates

Hypothetical Accident Conditions	1-meter from Package Surface mSv/hr (mrem/hr)		
Radiation	Top	Side	Bottom
Gamma (Co-60)	0.2303 – 2.6391 (23.03 – 263.91)	0.0551 – 1.4746 (5.51 – 147.46)	0.2402 – 2.7563 (24.02 – 275.63)
Neutron			
10 CFR 71.51 (a)(2) Limit	10 (1000)	10 (1000)	10 (1000)

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RAI 4 Question:

Explain how the source term used in the shielding analysis model was determined.

From the TLI Calculation Note No. CN-13039-501 titled "Rancho Seco Activated Metal Shielding Analysis," it appears that a uniformly distributed source was used in the MCNP models for both NCT and HAC analyses and that the source was defined with explicit geometries of the content blocks in the NCT model. However, it is not clear how the source distribution for each segment was determined from the information provided in application and the calculation note. The applicant needs to explain how the source term used in the shielding analysis model was determined.

The staff needs this information to determine compliance with the regulatory requirements of 10 CFR 71.47 and 71.51.

RAI 4 Response:

The method for determining the source distribution for each of the components is shown in the DW James reports DWJ-RS-001 through DWJ-RS-004, which are summarized in the DAHER-TLI report TR-13039-001. This document summarizes the work that was done by DW James and Associates to characterize the activity built up in each of the reactor components. For this process, the first DW James calculation report is DWJ-RS-001, which is used to calculate the flux profiles used for the activation analysis. Below is the analysis description for this document.

DWJ-RS-001:

"ANISN radial and axial case studies are performed based on Rancho Seco specific dimensional parameters summarized in Reference 5. The ANISN models were developed using BUGLE-96, 47 scattering group cross section libraries with a Legendre P3 order of scattering. Reference 3 developed an axial ANISN model for the analysis of fuel assembly hardware specific to a B&W reactor core featuring 165 inch fuel assemblies equivalent to those used at Rancho Seco. This model was extended to include the upper and lower internal assemblies for Rancho Seco. The calculation was performed in two parts. The first extended upward from the core center line to the top of the control rod guide tubes, and the second extended downward to the inner surface of the reactor vessel bottom head. The radial model for Ranch Seco was developed using averaged core parameters from the Reference 3 axial model. It extended from the center of the core to 30 cm inside the reactor biological shield."

Reference 5: Rancho Seco Reactor Vessel Internals Packaging Plan, Physical Characterization Supplement, September 2004

Reference 3: PNL-6906, Vol.2, "Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal, Luksic, et. al, June 1989

The following DW James calculation reports: DWJ-RS-002, DWJ-RS-003, and DWJ-RS-004 were used to estimate the activity content of the Rancho Seco upper internals, core region internals, and lower internals respectively. For each analysis the flux profile for the respective core region, taken from the first DW James report (DWJ-RS-001), was used to calculate the activation in this area. Section 1.3 in the DAHER-TLI report TR-13039-001 provides plots showing the flux profiles calculated for these regions. With each flux profile,

activation calculations were performed using the DW James & Associates code "SNAP", and were compared with calculations using ORIGEN 2.1. Each activation analysis accounted for 13 years of decay (reactor shutdown in 1989 to analysis date in 2003). Radiation dose rate surveys of the major components were performed to provide a point of reference for the activity estimates and the calculated radionuclide spectra were adjusted to better match the results of the surveys.

The report DWJ-RS-002 also outlines the procedure used for calculating the contamination activity. This process is summarized in the provided report TR-13039-001. The contamination activity is included in the total activity summary used for the source term.

It was determined that Co-60 accounts for nearly 100% of the gamma source for both the activated components and the additional activity in the contamination. Appendix II of the report TR-13039-001 includes the material packaging summary for each liner, where the activity (decayed to the loading date in 2005) of the segmented components included in each liner is summarized. These are the values that are used for the activity distribution in the RT-100 shielding calculation in CN-13039-501. For example the information used for liner RVI-016 is shown below:

In Appendix II of TR-13039-001, one can find the activity distribution between the CB pieces in the Package Contents Summary Table, and the total Co-60 Activity in the Package Activity Summary Table:

RVI-017 DW JAMES REPORT

Package Contents Summary

Component	Weight (lbs)	Activity (Ci)
ACB1-5	1530	418
ACB1-6	1530	418
ACB2-4	1240	341
ACB2-5	1240	341
ACB2-6	1240	341
ACB3-3	960	263
ACB3-4	960	263
ACB3-5	960	263
TOTAL	9660	2648

Package Activity Summary

Nuclide	Activity (Ci)
Co-60	1.39E+03
Ni-63	1.03E+03
Fe-55	2.17E+02
Am-241	1.68E-03
C-14	1.52E+00
Pu-238	3.19E-04
Pu-239	2.78E-04
Pu-241	8.85E-03
Cm-244	2.07E-04
Nb-94	2.95E-02
Sr-90	1.26E-03
Mn-54	2.87E-03
Cs-137	3.19E-04
Cm-242	1.99E-07
Tc-99	5.75E-06
Zn-65	1.72E-06
Ni-59	1.06E+01
TOTAL	2649

In Appendix B of CN-13039-501, one can find the activity distribution summary used for the shielding

<u>analysis, which matches the values shown in TR-13039-001:</u>

RVI-017 Component Model		
Item	Activity (Ci)	Activity (%)
CB1-5	418	15.79
CB1-6	418	15.79
CB2-4	341	12.88
CB2-5	341	12.88
CB2-6	341	12.88
CB3-3	263	9.93
CB3-4	263	9.93
CB3-5	263	9.93
TOTAL	2648	-

<u>In the MCNP input RVI-016-017.inp, the cell restriction that matches the activity distribution in Appendix B of the report is found in the source definition:</u>

```
c Co-60 Source
sdef  x=d1 y=d5 axs=0 0 1 z=d6 erg=d4 cel=d3
si3 l 111 112 113 114 115 116 117 118
sp3   0.099320242 0.099320242 0.099320242 0.128776435 0.128776435 0.128776435
      0.157854985 0.157854985
si4  l 8.6830E-04  7.4609E-03  7.4781E-03  8.2647E-03  3.4693E-01  8.2628E-01
      1.1732e+00  1.3325E+00  2.1588E+00  2.5050E+00
sp4   1.4900E-06  3.2500E-05  6.4000E-05  1.3000E-05  7.6000E-05  7.6000E-05
      9.9900e-01  9.9982E-01  1.1100E-05  2.0000E-08
si1 -16 54
sp1 0 1
si5 -63 63
sp5 0 1
si6 0.63 177.8
sp6 0 1
```

The source term defined above is representative of a cobalt-60 source where the probability of a source point being generated in a given cell corresponds to the values on the sp3 card. Note that cells 111 through 118 are the cells that represent the core barrel pieces CB1-5 through CB3-5, with cell 118 representing the cell closest to the edge of the cask and cell 111 closest to the center, forcing the most active components to the edge of the liner cavity.

If needed for further clarification, the DW James reports have been provided as references 3, 4, 5, and 9 in the authorization request.

Amendment Request Reference No.	Electronic Reference Package File Name	DW James Report No.	Duratek Engineering Report No.	Document Title
9	[9]ER-05-004 RVI Flux Profiles.pdf	DWJ-RS-001	ER-05-004	Rancho Seco Reactor Vessel Internals Flux Profiles
3	[3]ER-05-005 Rad Char of Upper Internals.pdf	DWJ-RS-002	ER-05-005	Radiological Characterization of Rancho Seco Upper Internals
4	[4]ER-05-006 Rad Char of Core Region Internals.pdf	DWJ-RS-003	ER-05-006	Radiological Characterization of Rancho Seco Core Region Internals
5	[5]ER-05-007 Rad Char of Lower Internals.pdf	DWJ-RS-004	ER-05-007	Radiological Characterization of Rancho Seco Lower Internals Assembly

RAI 5 Question:

Justify the use of a disc source to represent the contents in the HAC model.

In the TLI Calculation Note No. CN-13039-501 titled "Rancho Seco Activated Metal Shielding Analysis," the applicant states "For the Side 1-meter dose calculations the source is restricted to the 1-inch thick disc representing the components or the chips at the lead slump or pin puncture." However, it was not clear how the sources distributed in the content were converted into a disc source. The applicant needs to explain how this conversion was done with a justification if it is conservative to do so.

The staff needs this information to determine compliance with the regulatory requirements of 10 CFR 71.47 and 71.51.

RAI 5 response:

The disc source was used to determine the side dose rates for the HAC case in an attempt to bound any possible realistic scenario. For the chips models and the components models the assumption is made that all of the chips or components are compressed into a one-inch thick disc located at the exact axial height of the lead slump or the pin puncture where the highest dose rates are calculated. For this case, the highest does rate was calculated at the axial location of the lead slump.

For the components HAC model, the density of the disc is equal to the full theoretical density of steel. The chips HAC model disc was modeled as a 10% theoretical steel density disc, assuming that all of the chips are compressed into a single disc aligned axially with the lead slump. No data was made available on the density of the chips free sitting, however it was determined that for the chips condensed into a single disc of 10% theoretical steel density is adequate. It should also be noted that the highest calculated dose rates for the HAC cases are about 25% of the regulatory dose rate limit.

Modeling the chips and components as a disc at the lead slump location during HAC is bounding because realistically, while the components and chips may rearrange themselves inside the cavity, they will still maintain their geometry and remain distributed throughout the cavity of the cask.