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An AECOM-led partnership with B&W and AREVA

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ATTN: Document Control Desk
Director, Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: SUPPLEMENTAL INFORMATION REGARDING REVISION 7 OF THE RH-TRU 72-B SHIPPING PACKAGE APPLICATION, DOCKET NO. 71-9212

References:
1. Letter from T. E. Sellmer to Document Control Desk, dated November 10, 2014, subject: Revision 7 of the RH-TRU 72-B Shipping Package Application, Docket No. 71-9212
2. Letter from Huda Akhavannik (NRC) to T. E. Sellmer, dated December 19, 2014, subject: Application for Revision No. 8 of Certificate of Compliance No. 9212 for the Model No. RH-TRU 72-B Package – Supplemental Information Needed, TAC No. L24965

Dear Sir or Madam:

Nuclear Waste Partnership LLC (NWP), on behalf of the U.S. Department of Energy, hereby submits supplemental information regarding the application for Revision 8 of the Certificate of Compliance for the RH-TRU 72-B Packaging, U.S. Nuclear Regulatory Commission (NRC) Docket No. 71-9212 (Reference 1). This information is provided in response to the NRC request for supplemental information (Reference 2).

The supplemental information provided consists of the following three attachments:

- Attachment A – Shielding Summary
- Attachment B – Responses to RSI
- Attachment C – Supplemental Reference

Attachment A provides a summary of key background information relative to current and historical RH-TRU 72-B shielding evaluations. Attachment B provides specific responses to the 3 shielding-related RSIs, 3 thermal observations and 10 shielding observations. Attachment C is provided in support of the response to RSI 5-2. These three attachments have been prepared to provide the necessary and sufficient supplemental information for the NRC to continue with its full review of the application. If you have any questions regarding this submittal, please contact Mr. R. A. Johnson of my staff at (360) 438-6145.

NMSS01

Sincerely,

Todd Sellmer

T. E. Sellmer, Manager
Packaging

TES:clm

cc: J. R. Stroble, CBFO
J. C. Rhoades, CBFO
H. Akhavannik, USNRC

Summary of RH-TRU 72-B Shielding Evaluations and Satisfaction of Dose Rate Limits

The currently approved version (Revision 6) of the RH-TRU 72-B SAR imposes radionuclide activity limits that ensure if all source material within a given payload canister were to coalesce at a single location immediately adjacent to the worst case packaging damage from HAC (i.e., that associated with side puncture), the Part 71 imposed HAC dose rate limit of 1,000 mrem/hr at 1 meter from the package surface would still be satisfied. No NCT based activity limits are currently imposed. Rather, pre-shipment dose rate surveys on the package surface and 2 meters from the vertical planes at the sides of the transport vehicle are relied upon to ensure satisfaction of corresponding, exclusive use, NCT limits (i.e., 200 mrem/hr and 10 mrem/hr, respectively) at the time of transport. It should also be noted for the over 700 shipments of RH-TRU waste that have been completed to date in the RH-TRU 72-B, en route dose rate surveys that have been performed at state borders, ports of entry, etc., have never identified a case where the NCT dose rate limits were exceeded during transit.

Consistent with guidance provided in NRC Regulatory Issue Summary (RIS) 2013-04[1], the well characterized nature of TRU waste coupled with other considerations such as the use of carefully developed, detailed documents and operating procedures[2] associated with 1) payload characterization, 2) payload and package loading activities, and 3) performance of pre-shipment dose rate surveys further establish that the above controls are reasonable. In fact, although the RH-TRU 72-B is not mentioned by name, the last paragraph of the portion of RIS 2013-04 under the heading "Package Evaluation" is specifically based on the RH-TRU 72-B and ends by stating, "Considering the above, the staff found that the evaluation demonstrated that the package meets the NCT dose rate requirements".

A more thorough discussion of the shielding basis for the RH-TRU 72-B can be found in the NRC's Safety Evaluation Report (SER) associated with the approval of Revision 5 of the SAR, which resulted in issue of Revision 6 of the NRC CoC for the package. Sections 5.1.3, 5.4.4 and 5.4.5 of the SER provide information directly relevant to the current shielding related RSIs and Observations that resulted from the NRC's acceptance review of the Revision 7 SAR (NRC letter of 12/19/14, TAC No. L24965). From a review of those SER sections, a few specific items are worth noting as follows:

> First, currently approved (Rev. 6 SAR) activity limits were found to fully satisfy the HAC dose rate limits of 10CFR71.51(a)(2). This conclusion was supported by NRC confirmatory analyses that conservatively treated potential lead slump and specifically looked at possible streaming effects in the event that point sources were located at the end of the canister. Given the fact that all new activity limits now being proposed in Revision 7 of the SAR continue to allow coalescing of all source material to a single, worst case location under HAC, any potential concerns with the satisfaction of HAC dose rate limits should continue to be addressed.

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[1] NRC Regulatory Issue Summary 2013-04, *Content Specification and Shielding Evaluations for Type B Transportation Packages*, April 23, 2013, http://pbadupws.nrc.gov/docs/ML1303/ML13036A135.pdf

[2] All documents are controlled consistent with the NRC approved RH-TRAMPAC and implemented via documents such as the RH-TRUCON and O&M Manuals.

However, as pointed out in the last paragraph of Section 5.4.5 of the SER, if a 72-B package was to be loaded with the maximum currently allowed (HAC based) activity, there is no analytic assurance that NCT dose rate limits would be satisfied. Again, this is because pre-shipment surveys are currently relied upon as the primary means of ensuring that NCT limits are satisfied at the time of transport.

Potential concerns stated in RIS 2013-04 included "relying completely on pre-shipment dose measurements to determine if a package meets its regulatory dose rate limits may not address the possibility of contents shifting or settling during transport." A potential concern stated in the last paragraph of Section 5.4.5 of the NRC SER was that "some, potentially significant amount of self-shielding is needed in any evaluation to demonstrate compliance with NCT dose rate limits." To address these potential concerns, Revision 7 of the SAR has now incorporated NCT based activity limits.

Given the new NCT based activity limits, if the confirmatory analyses discussed in the last paragraph of SER Section 5.4.5 (for both neutron and gamma sources) were repeated, the conclusion would now be that an evenly distributed source at its activity limit and without self-shielding would satisfy the NCT dose limits of 10CFR71.47(b). In fact, obtained dose rates would actually fall comfortably within allowable limits because of an inherent conservatism used in the new NCT based shielding analyses. Namely, as mentioned in the fourth paragraph of SAR Section 5.0, *Shielding Evaluation*, analytically calculated NCT dose rates are being held to the non-exclusive use shipment limit of 10 mrem/hr at 2 meters from the package surface instead of the exclusive use limit of 10 mrem/hr at 2 meters from a vertical plane at the edge of the transport vehicle. For the 72-B transportation configuration (see RH-TRAMPAC Figure 3.2-1, *Dose Rate Measurement Locations*), this corresponds to an additional 0.70 meters of distance attenuation that could have been used in the evaluation. Considering that the inside radius of the payload canister shell is only 0.31 meters, this conservatism alone largely addresses the RIS 2013-04 concern with the potential shifting or settling of contents during transport. The use of a 10% administrative margin, which has been added in the Revision 7 SAR when summing fractions, provides additional conservatism.

Lastly, the pre-shipment dose rate surveys will continue to play a role, albeit now a significantly lesser one, in that if a given payload is actually at or near its analytically established activity limit, the activity will need to be reasonably distributed within the payload container as assumed by the NCT analysis to meet the NCT dose rate limits. If not reasonably distributed, the measured dose rates may not satisfy NCT dose rate limits, which would result in the shipment being rejected. Modest concentrations of activity within a given payload container are readily accommodated by a combination of having conservatively used the non-exclusive use dose rate location when analytically establishing allowed activity limits and imposition of the 10% administrative margin. Significant concentrations of high activity materials in a given payload container are precluded in that the pre-shipment dose rate surveys would fail. By way of example, consider two extreme cases as follows. First, consider a payload that is right at its maximum allowed activity limit and that the activity is reasonably distributed throughout the payload container. In this case, as analytically demonstrated, actual pre-shipment dose rates would have to fall comfortably below their allowable limits. That margin is what accommodates the potential shifting or settling of payloads during transport. At the other extreme, assume a payload contains activity that is locally concentrated at or near the inner wall of the payload container. In this case, the pre-shipment surveys will only pass if the actual activity is well below the analytically established activity limit. Further, potential shifting of the payload to a worse location during transport is highly restricted by the fact that the activity started essentially at its worst case location.

The above, and the responses to the individual RSIs and Observations provided in Attachment B, *Responses to RSI*, support the appropriateness of the proposed activity limits in the Rev. 7 SAR.

Responses to NRC Request for Supplemental Information (RSI) on Revision 7 of the RH-TRU 72-B Safety Analysis Report (SAR), Revision 3 of the Remote-Handled Transuranic Waste Authorized Methods for Payload Control (RH-TRAMPAC), and Revision 3 of the RH-TRU Payload Appendices

5.0 Shielding

5-1 Provide analyses, or justification, for the determination of the limiting dose rate location(s) for all allowable source geometries, energies, and canister configurations.

Section 5.5, "Activity Limits," states: "Since the results in Section 5.4, *Shielding Evaluation*, demonstrate that contents meeting the NCT limits at 2 meters will meet NCT surface and HAC at 1 meter limits, maximum activity was determined only for NCT."

Section 5.4 appears to only include analyses for concentrated Co-60 and Cf-252 whereas Section 5.5 includes activity limits for both distributed and concentrated sources of gammas from 0.5-10 MeV and neutrons 0.1-15 MeV. The staff requests that the applicant provide appropriate analyses to support the conclusion that the limiting conditions for all allowable contents is NCT at 2 meters (versus NCT surface dose rate limits or HAC dose rate limits for distributed sources). For example, in a similar package, the surface dose rate regulations were the most limiting for lower energy gammas (less than 1 MeV), and certain distributed source configurations were limited by HAC conditions. The applicant should provide analyses justifying the limiting dose rate location(s) that encompasses both source geometries for all canister configurations and the range of allowable gamma and neutron energies.

This information is needed for the staff's review to determine compliance with 10 CFR 71.47 and 10 CFR 71.51(a)(2).

Response:

Additional RH-TRU Waste Canister HAC shielding evaluations, like that presented in RH-TRU 72-B SAR Table 5.4-4 for Co-60 and Cf-252, were performed for each discrete gamma and neutron energy listed in SAR Table 5.5-1 in both concentrated and distributed source forms to determine the allowable activity associated with the 1 rem/hr at 1 meter HAC dose rate limit. The results were subsequently compared with the allowable activity to reach the NCT at 2 meter dose rate limit listed in SAR Table 5.5-2 in addition to the allowable activity to reach the NCT at surface dose rate limit. The comparison, details of which are provided below, demonstrate that the NCT at 2 meter dose rate is limiting such that NCT dose rate compliance ensures that HAC dose rate requirements are satisfied.

The HAC concentrated source evaluations implemented the extremely conservative geometry assumptions utilized for the Co-60 and Cf-252 evaluations depicted in SAR Figure 5.3-2. The HAC distributed source evaluations similarly implemented a conservative geometry assumption that concentrates the distributed sources down to a single, approximately 16-gal, volume in line with the puncture damage location that minimizes the shield material thickness and the distance to the 1-meter detector. The NCT and HAC geometry models implemented in the MCNP evaluations are shown in Figure RSI 5-1.1.

Tables RSI 5-1.1 through RSI 5-1.4 provide the resulting concentrated and distributed gamma and neutron source dose rates for NCT and HAC, based on the associated activity for each case that is most limiting; namely, the activity required to meet the NCT at 2 meter dose rate limit. In all cases, except for the lowest energy concentrated gamma source, the NCT at 2 meter dose rate is limiting by a reasonably large margin in comparison to the NCT at surface and/or HAC at 1 meter dose rates. The low-energy concentrated gamma HAC result is unrealistically high because the large quantity of low-energy gamma required to challenge the NCT at 2 meter dose rate would necessarily be somewhat distributed initially and could not subsequently coalesce into a single point, as assumed in the HAC analysis, and would therefore have a lower HAC dose rate more consistent with the distributed low-energy gamma source case. Therefore, the use of the NCT at 2 meter dose rate to ensure compliance with the HAC dose rate is justified over applicable source types, distributions, and energies.

Finally, the RH-TRU Waste Canister based results are conservatively bounding for the NS15 and NS30 neutron shielded canisters (NSC) because:

- NSCs provide a greater amount of material attenuation in comparison to the RH-TRU waste canister under NCT
- NSCs undergo a lesser amount of material and distance attenuation loss in comparison to the RH-TRU waste canister due to HAC
- The resulting ratio between the bounding NCT dose rate and HAC dose rate for a given activity is therefore increased for the NSCs in comparison to the RH-TRU Waste Canister.

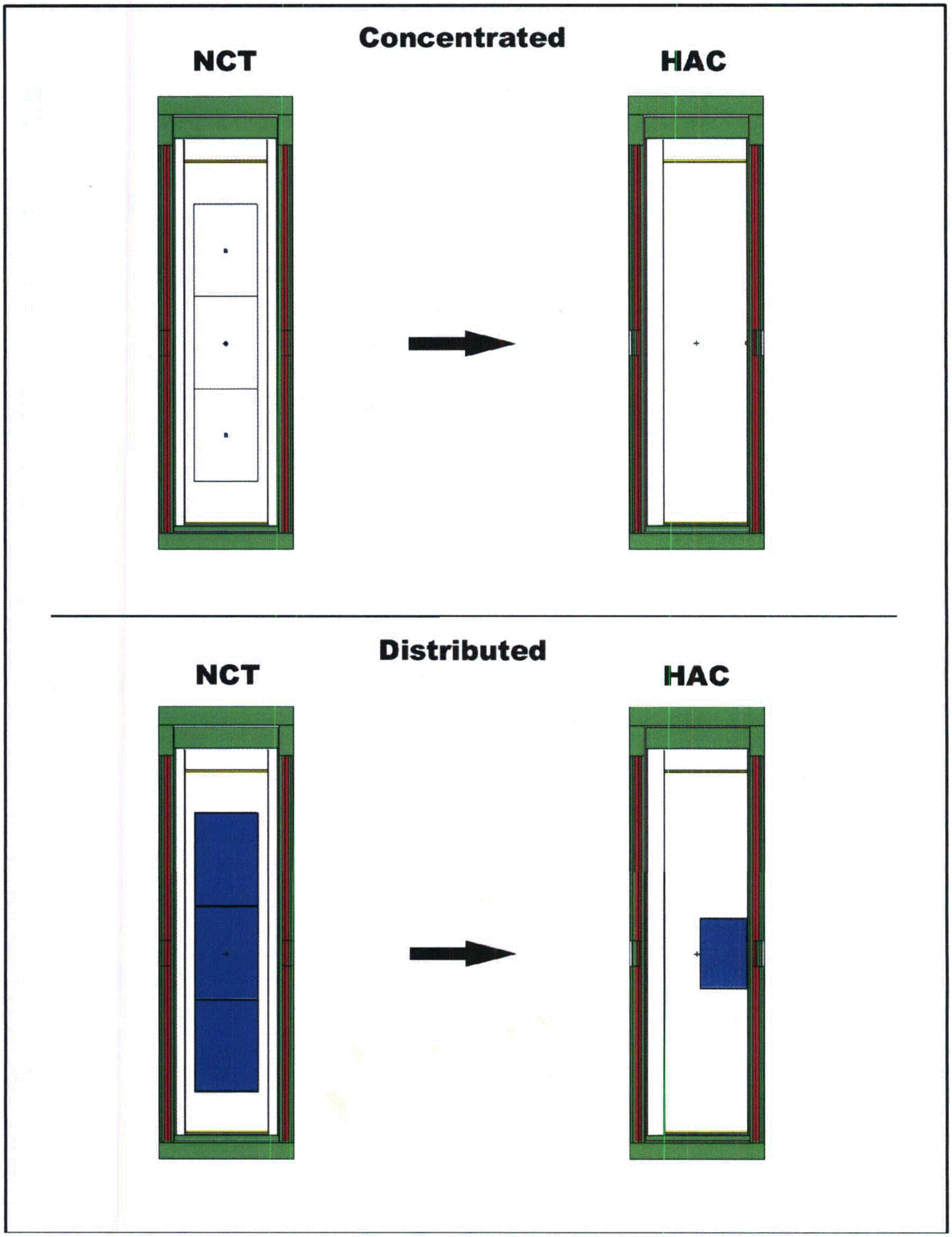


Figure RSI 5-1.1 – MCNP Geometries for NCT and HAC Analyses

Table RSI 5-1.1 – RH-TRU Waste Canister Concentrated Gamma Source Dose Rate Comparison

Source Type	Source Density (g/cc)	Energy (MeV)	NCT @surf (mrem/hr)	NCT @2m (mrem/hr)	HAC @1m (mrem/hr)
Concentrated	1	0.500	158	10	1000
Concentrated	1	0.600	147	10	537
Concentrated	1	0.800	134	10	282
Concentrated	1	1.000	127	10	209
Concentrated	1	1.250	119	10	169
Concentrated	1	1.500	116	10	147
Concentrated	1	1.750	113	10	136
Concentrated	1	2.000	111	10	132
Concentrated	1	2.500	109	10	119
Concentrated	1	4.000	106	10	108
Concentrated	1	6.000	104	10	118
Concentrated	1	10.000	105	10	125

Table RSI 5-1.2 – RH-TRU Waste Canister Concentrated Neutron Source Dose Rate Comparison

Source Type	Source Density (g/cc)	Energy (MeV)	NCT @surf (mrem/hr)	NCT @2m (mrem/hr)	HAC @1m (mrem/hr)
Concentrated	1	0.100	129	10	56
Concentrated	1	0.200	131	10	60
Concentrated	1	0.300	129	10	59
Concentrated	1	0.500	131	10	58
Concentrated	1	0.750	129	10	55
Concentrated	1	1.000	130	10	57
Concentrated	1	1.250	130	10	62
Concentrated	1	1.500	129	10	65
Concentrated	1	1.750	129	10	58
Concentrated	1	2.000	130	10	62
Concentrated	1	2.500	131	10	59
Concentrated	1	5.000	130	10	63
Concentrated	1	10.000	129	10	63
Concentrated	1	15.000	129	10	66

Table RSI 5-1.3 – RH-TRU Waste Canister Distributed Gamma Source Dose Rate Comparison

Source Type	Source Density (g/cc)	Energy (MeV)	NCT @surf (mrem/hr)	NCT @2m (mrem/hr)	HAC @1m (mrem/hr)
Distributed	0.1	0.500	82	10	280
Distributed	0.1	0.800	83	10	101
Distributed	0.1	2.000	84	10	63
Distributed	0.1	10.000	81	10	63
Distributed	1	0.500	80	10	328
Distributed	1	0.600	81	10	195
Distributed	1	0.800	81	10	121
Distributed	1	1.000	81	10	97
Distributed	1	1.250	83	10	83
Distributed	1	1.500	81	10	76
Distributed	1	1.750	83	10	73
Distributed	1	2.000	81	10	70
Distributed	1	2.500	84	10	67
Distributed	1	4.000	82	10	72
Distributed	1	6.000	83	10	68
Distributed	1	10.000	81	10	81
Distributed	2	0.500	80	10	340
Distributed	2	0.800	81	10	130
Distributed	2	2.000	81	10	78
Distributed	2	10.000	78	10	81
Distributed	4	0.500	79	10	337
Distributed	4	0.800	80	10	133
Distributed	4	2.000	80	10	84
Distributed	4	10.000	78	10	90
Distributed	6	0.500	77	10	333
Distributed	6	0.800	80	10	135
Distributed	6	2.000	78	10	86
Distributed	6	10.000	75	10	76
Distributed	8	0.500	77	10	314
Distributed	8	0.800	80	10	129
Distributed	8	2.000	79	10	83
Distributed	8	10.000	76	10	73

Table RSI 5-1.4 – RH-TRU Waste Canister Distributed Neutron Source Dose Rate Comparison

Source Type	Source Density (g/cc)	Energy (MeV)	NCT @surf (mrem/hr)	NCT @2m (mrem/hr)	HAC @1m (mrem/hr)
Distributed	1	0.100	125	10	43
Distributed	1	0.200	124	10	45
Distributed	1	0.300	124	10	47
Distributed	1	0.500	128	10	44
Distributed	1	0.750	128	10	42
Distributed	1	1.000	127	10	44
Distributed	1	1.250	126	10	43
Distributed	1	1.500	125	10	44
Distributed	1	1.750	126	10	42
Distributed	1	2.000	126	10	44
Distributed	1	2.500	127	10	43
Distributed	1	5.000	126	10	43
Distributed	1	10.000	125	10	44
Distributed	1	15.000	125	10	45

5-2 Provide the analyses that justify the use of Zr-40 as the source material to address source self-shielding.

Section 5.3.2, "Material Properties," states: "Zirconium is modeled in the source region at densities ranging from 0.1 to 8 g/cc to represent a conservative basis for self-shielding in distributed sources." Section 5.5, "Activity Limits," further states: "The material for the source region was selected as Zr (Z=40); multiple calculations with various materials showed a material selection of Zr was reasonably representative of TRU waste for gamma calculations and inconsequential to neutron calculations." Provide the analyses demonstrating that Zr-40 is representative of TRU waste and discuss how it specifically represents the materials that can be shipped in the RH-TRU 72-B.

This information is needed for the staff's review to determine compliance with 10 CFR 71.47 and 10 CFR 71.51(a)(2).

Response:

The justification for use of Zr-40 as the source material to appropriately address source self-shielding was evaluated extensively for TRU waste with the TRUPACT-II Rev. 23 and HalfPACT Rev. 6 SAR application, which obtained an NRC CoC in June 2013.

A calculation package, 72B-CAL-0003, *Self-shielding Surrogate Material Evaluation for RH-TRU 72-B*, Rev. 0, is provided in Attachment C to further justify the use of Zr as the distributed source material in the shielding analysis. The calculation package presents a comparison between Zr and common TRU waste material constituents with the following primary conclusion, "Zirconium is a reasonable surrogate material to represent the dose rate attenuation due to self-shielding in remote-handled transuranic waste."

5-3 Provide clear and detailed information regarding how the density correction factors (DCFs) were determined and how they are to be used.

The applicant performed an analysis from which it developed a method for package users to determine the acceptability of materials for shipment in the RH-TRU 72-B. The analysis and acceptability determination method make use of DCFs. The description of the analysis and use of the DCFs is unclear, making it difficult to understand and evaluate the analysis and the part of the acceptability determination method that uses the DCFs.

For example, terminology referring to source activities appears to not be used in a consistent manner. This makes it difficult to understand what activity is being referred to, whether it is the activity loaded into the package or it is the applicable activity limit from Tables 5.5-2, 5.5-4, and 5.5-6. Accordingly, it is difficult to follow the description of how the DCFs were developed and how they are to be applied by a package user and whether the results are conservative or appropriate.

In addition, the information provided is limited in detail that is needed to understand the DCFs' development and use. Thus, details should be provided that illustrate, for at least one payload configuration, how, in a step-by-step manner, the DCFs were calculated and the DCF curves were developed. The information should include calculation results that were used in each applicable development step. Section 5.5.4 of the application should also include, for at least one of the examples, more details that illustrate step-by-step the process of calculating the DCFs from the DCF curves and how they are applied. This information could also be provided in a separate calculation package.

This information is needed for the staff's review to determine compliance with 10 CFR 71.47 and 10 CFR 71.51(a)(2).

Response:

The fifth paragraph in the RH-TRU 72-B SAR Section 5.5, page 5.5-1, states that, "For distributed gamma sources, the 1 g/cc (unit density) results are utilized along with a density correction factor (DCF) to apply the distributed gamma unit-density source results to any source density in the range. A 3rd order polynomial curve-fit of the minimum DCF, determined as the smallest DCF for all evaluated energies at each source density, was produced to facilitate calculation of a DCF for any density of contents." The first full paragraph on page 5.5-2 further states that, "The gamma DCF is the ratio of the maximum activity at a density other than 1 to the maximum activity for the unit density source."

To clarify for the RH-TRU Waste Canister, results similar to that presented in SAR Table 5.5-2 for densities other than 1 g/cc were determined for a range of discrete source densities (0.1, 2, 4, 6, and 8 g/cc) over a range of discrete gamma energies (0.5, 0.8, 2, and 10 MeV) and were reported as data points in Figure 5.5-2. The data points in Figure 5.5-2 were obtained by dividing the maximum allowable activity to satisfy the NCT at 2 meter dose rate limit at each source density evaluated by the corresponding maximum allowable activity associated with the 1 g/cc source (for each energy). For each density, the minimum ratio was determined and a curve fit was obtained to provide a polynomial equation for the DCF that provided a density correction factor that was independent of energy and a function of density.

An example of the determination of the minimum DCF associated with a specific source density is provided below for the RH-TRU Waste Canister, where $A_{m,n}$ is the maximum allowable activity to satisfy the NCT at 2 meter dose rate limit for density "m" and energy "n". The minimum DCF for a specific source density is determined using the following equation:

$$DCF_m = \min\left[\frac{A_{m,n}}{A_{1,n}}\right]$$

where

$m = 0.1, 2, 4, 6, \text{ and } 8$ g/cc

and

$n = 0.5, 0.8, 2, \text{ and } 10$ MeV

The resulting DCF_m values are then used to fit a 3rd order polynomial equation to provide a DCF for the RH-TRU Waste Canister as a function of density. Table RSI 5-3.1 lists the maximum allowable activity values utilized to determine the final DCF equation presented in SAR Section 5.5.1:

$$DCF = (0.00002)\,\rho^3 - (0.0048)\,\rho^2 + (0.6322)\,\rho + 0.3065$$

The DCF equation, as described in SAR Section 5.5.4 for distributed payloads that meet the Case A or Case B qualification requirements, is utilized to calculate the allowable activity for any distributed source density as a multiplier for the unit-density activity allowable given in SAR Table 5.5-2. For example, the DCF and associated allowable activity for a 2 MeV gamma with a source density of 4 g/cc is as follows:

$$DCF_4 = (0.00002)\times 4^3 - (0.0048)\times 4^2 + (0.6322)\times 4 + 0.3065 = 2.76$$

and

$$A_{4,2} = A_{1,2}\times DCF_4 = 3.59E{+}11\times 2.76 = 9.91E{+}11 \text{ gamma/sec}$$

An example implementation of the above calculations, per the SAR Section 5.5.4 procedure, for a payload that consists of three drums having dose rates that vary by less than a factor of 10 from the average, with waste densities of 8, 5, and 4 g/cc (use 4 g/cc as minimum), and a single 2.0 MeV gamma total source strength for the canister (as obtained from the payload assay results for each drum summed plus the addition of assay error) of 5.00E+11 gamma/sec would result in a sum of fractions of

$$SF = 5.00E{+}11/9.91E{+}11 = 0.51 < 0.9$$

that meets the requirements for shipment in the RH-TRU Waste Canister.

The overall process for determining the compliance with SAR Section 5.5.4 allowable fraction is practically implemented within a controlled compliance software application that simplifies the dose rate compliance determinations. Due to the nature of TRU waste with variable waste densities, gamma and neutron energies, and total source terms, the generalized energy-based method is necessary to allow the evaluation of all waste forms anticipated for shipment in the RH-TRU 72-B. The methodology described above is currently implemented for the TRUPACT-II and HalfPACT packagings utilized at WIPP.

Table RSI 5-3.1 – RH-TRU Waste Canister Allowable Activity for Distributed Source DCF Calculation

Distributed Source Density (g/cc)	Energy (MeV)	Allowable Activity (par/sec)	DCF Ratio	Min DCF Ratio
0.1	0.500	3.29E+15	0.34	0.34
	0.800	1.57E+13	0.42	
	2.000	2.06E+11	0.57	
	10.000	3.94E+10	0.61	
1	0.500	9.72E+15	1.00	1.00
	0.800	3.76E+13	1.00	
	2.000	3.59E+11	1.00	
	10.000	6.47E+10	1.00	
2	0.500	1.87E+16	1.93	1.53
	0.800	6.83E+13	1.82	
	2.000	5.79E+11	1.61	
	10.000	9.88E+10	1.53	
4	0.500	3.69E+16	3.80	2.73
	0.800	1.33E+14	3.54	
	2.000	1.08E+12	3.01	
	10.000	1.77E+11	2.73	
6	0.500	5.41E+16	5.56	3.96
	0.800	1.98E+14	5.28	
	2.000	1.60E+12	4.47	
	10.000	2.56E+11	3.96	
8	0.500	6.88E+16	7.08	5.06
	0.800	2.49E+14	6.64	
	2.000	2.05E+12	5.71	
	10.000	3.27E+11	5.06	

Observations

Thermal

3-1 Provide adequate explanation and justification to neglect internal convection in the thermal evaluation of the RH-TRU 72-B waste shipping package during HAC (fire).

Page 3.4-1 of the SAR states that "Consistent with NRC-accepted thermal analysis methodology, internal convection between bodies is conservatively ignored."

Neglecting internal convection during HAC may underestimate package maximum temperatures and pressures because during the fire event internal convection could increase heat transfer from the heat source (fire) to the package internals.

The information is needed to demonstrate compliance with 10 CFR 71.73.

Response:

The RH-TRU 72-B waste shipping package is transported horizontally. Natural convection heat transfer across a gap between two long, horizontal concentric cylinders may be evaluated using Equations 3.101 through 3.104 from Eric C. Guyer, *Handbook of Applied Thermal Design*, CRC Press, 1999. When sufficient radial gap is present between the two concentric cylindrical surfaces, and when buoyancy forces are able to overcome viscous forces, natural convection heat transfer will establish a convective flow pattern, such as what is shown in Figure RSI 3-1.1.

Figure RSI 3-1.1 illustrates the condition where the inside cylinder is hotter than the outside cylinder. If the reverse is true, then the convective flow pattern would be reversed from what is shown.

From Equation 3.101 of Guyer (1999), the natural convection heat transfer, q', across the gap between two long, horizontal concentric cylinders, where the annulus contains air and the inside cylinder is hotter than the outside cylinder, is:

$$q' = \frac{2\pi k_{air} Nu}{\ln(D_o/D_i)}(T_i - T_o) \qquad (3.101)$$

where k_{air} is the thermal conductivity of air, Nu is the Nusselt number, D_o is the diameter of the outer cylinder's inside surface, D_i is the diameter of the inner cylinder's outside surface, and $T_i - T_o$ is the temperature differential between the inner and outer cylindrical surfaces, respectively.

Substituting k_{eff} for $k_{air}Nu$ in Equation 3.101 reduces the equation to simple thermal conduction between two concentric cylinders:

$$q' = \frac{2\pi k_{eff}}{\ln(D_o/D_i)}(T_i - T_o)$$

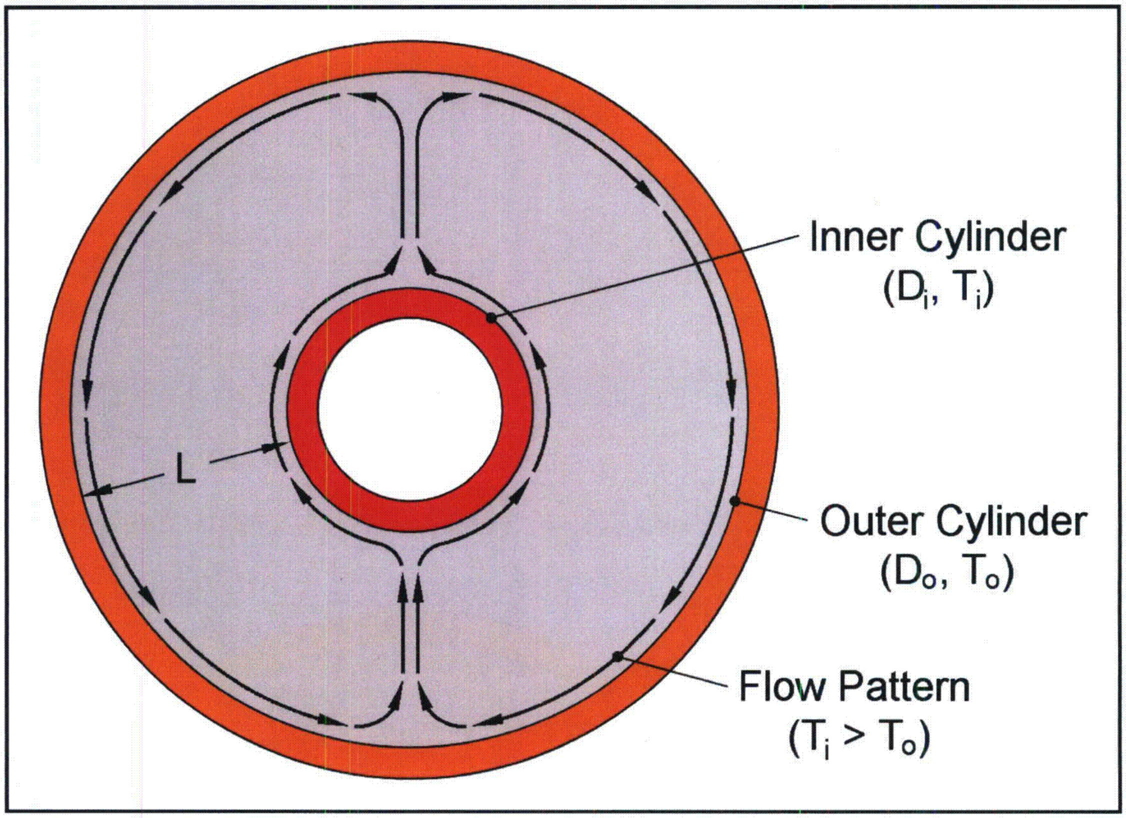


Figure RSI 3-1.1 – Typical Convective Flow Pattern between Concentric Cylinders

Thus, the Nusselt number, Nu = k_{eff}/k_{air}, and k_{eff} becomes the effective thermal conductivity of air for heat transfer across the radial cylindrical gap. Equation 3.102 from Guyer (1999), defines the Nusselt number, Nu, as:

$$\mathrm{Nu} = 0.386\left(\frac{\mathrm{Pr}}{0.861+\mathrm{Pr}}\right)^{1/4}\left(\mathrm{Ra}_c\right)^{1/4} \qquad (3.102)$$

where the Prandtl number for air, Pr = 0.7, and Ra_c is the critical Rayleigh number. Equation 3.102 is valid for Pr ≥ 0.7 and 100 < Ra_c < 10^8, and establishes the following limits:

$$\mathrm{Ra}_c < 100 \quad \Rightarrow \quad \mathrm{Nu} = \frac{k_{eff}}{k_{air}} = 1$$

$$100 < \mathrm{Ra}_c < 10^8 \quad \Rightarrow \quad \mathrm{Nu} = 0.386\left(\frac{\mathrm{Pr}}{0.861+\mathrm{Pr}}\right)^{1/4}\left(\mathrm{Ra}_c\right)^{1/4}$$

As shown above for Ra_c < 100, heat transfer across a radial gap becomes purely air conduction with no contribution due to natural convection.

From Equation 3.103, the critical Rayleigh number, Ra_c, is

$$Ra_c = \frac{[\ln(D_o/D_i)]^4}{L^3\left(D_i^{-3/5} + D_o^{-3/5}\right)^5} Ra \qquad (3.103)$$

where the characteristic length, $L = (D_o - D_i)/2$, and Ra is the Rayleigh number. From Equation 3.104, the Rayleigh number, Ra, is:

$$Ra = \frac{g\beta L^3}{\nu\alpha}(T_i - T_o) \qquad (3.104)$$

where the gravitational constant, $g = 386.0886$ in/s^2, β is the thermal expansion coefficient for air, ν is the kinematic viscosity of air, α is the thermal diffusivity of air, and $\Delta T = T_i - T_o$ is the maximum average temperature differential between the two cylindrical surfaces. The coefficients β, ν, and α are determined using the average temperature of the two surfaces at the time when the maximum average temperature differential occurs, with values linearly interpolated from Table A.4 of Incropera, DeWitt, Bergman, and Lavine, *Fundamentals of Heat and Mass Transfer*, 6th Edition.

Determining the convective heat transfer coefficients is an iterative process. The temperatures reported in Table RSI 3-1.1 are taken from the final iteration of the thermal analyses that use the calculated Nusselt numbers as coefficients for the effective radial thermal conductivity of air in the annulus between the outer cask and inner vessel (OC/IV), in the annulus between the inner vessel and removable lid canister (IV/RLC), and in the annulus between removable lid canister and payload drums (RLC/Drum). As summarized in Table RSI 3-1.1, the coefficients in the OC/IV annulus are unity because the radial gap is insufficient to support a convective flow pattern, and the same is true for the gap between the OC thermal shield and the OC outer shell since the radial gap there is even less.

Tables RSI 3-1.2 and RSI 3-1.3 summarize temperatures from the final iteration of the HAC pre-fire and HAC fire/post-fire analyses that used the coefficient discussed above, and compare these results to the original results without internal convection. With a few negligible exceptions, temperatures are lower when internal convection is included in the evaluations. These temperature reductions occur because under normal conditions, any convective modes of heat transfer that exist internal to the package will reduce initial condition, pre-fire temperatures, and because the two regions that are most affected by the HAC fire event, the OC thermal shield air gap and OC/IV air gap, are too closely coupled to support an internal convection flow pattern. Therefore, the original analyses with all package components centered, but without internal convection, are reasonable for determining worst-case package temperatures and pressures.

Table RSI 3-1.1 – HAC Pre-Fire and HAC Fire/Post-Fire Internal Convection Coefficient Calculations

HAC Pre-Fire Internal Convection Coefficients			
Variable	**OC/IV**	**IV/RLC**	**RLC/Drum**
D_o (in)	32.375	31.250	25.500
D_i (in)	32.000	26.000	18.250
L (in)	0.1875	2.6250	3.6250
ΔT (°F)	10.6	28.1	34.3
T_{avg} (°F)	139.2	158.3	189.3
β (°F^{-1})	$1.6719(10)^{-3}$	$1.6196(10)^{-3}$	$1.5425(10)^{-3}$
ν (in^2/s)	$2.9662(10)^{-2}$	$3.1354(10)^{-2}$	$3.4178(10)^{-2}$
α (in^2/s)	$4.2245(10)^{-2}$	$4.4759(10)^{-2}$	$4.9024(10)^{-2}$
Ra	$3.5994(10)^{1}$	$2.2647(10)^{5}$	$5.8073(10)^{5}$
Ra_c	$1.0483(10)^{-1}$	$1.0291(10)^{4}$	$4.6703(10)^{4}$
Nu	1.00	3.18	4.64
HAC Fire/Post-Fire Internal Convection Coefficients			
Variable	**OC/IV**	**IV/RLC**	**RLC/Drum**
D_o (in)	32.375	31.250	25.500
D_i (in)	32.000	26.000	18.250
L (in)	0.1875	2.6250	3.6250
ΔT (°F)	145.1	45.2	33.3
T_{avg} (°F)	261.8	226.8	190.0
β (°F^{-1})	$1.3872(10)^{-3}$	$1.4587(10)^{-3}$	$1.5408(10)^{-3}$
ν (in^2/s)	$4.1104(10)^{-2}$	$3.7709(10)^{-2}$	$3.4242(10)^{-2}$
α (in^2/s)	$5.9599(10)^{-2}$	$5.4422(10)^{-2}$	$4.9120(10)^{-2}$
Ra	$2.0911(10)^{2}$	$2.2437(10)^{5}$	$5.6103(10)^{5}$
Ra_c	$6.0904(10)^{-1}$	$1.0195(10)^{4}$	$4.5117(10)^{4}$
Nu	1.00	3.17	4.60

Table RSI 3-1.2 – HAC Pre-Fire Steady State Temperatures with Insolation (°F) for an RLC with 30-Gallon Drums Containing a 100 lb/ft³ Payload with 270 Watts Decay Heat; without and with Internal Convection

Location	No Internal Convection	Internal Convection	ΔT	Allowable Limit
Lower Drum Contents Peak	311.8	295.5	-16.3	402
Lower Drum Contents Average	252.7	234.1	-18.6	302
Lower Drum Wall Peak	251.7	237.8	-13.9	—
Middle Drum Contents Peak	326.1	306.7	-19.4	402
Middle Drum Contents Average	270.9	252.0	-18.9	302
Middle Drum Wall Peak	252.2	234.5	-17.7	—
Upper Drum Contents Peak	310.5	294.2	-16.3	402
Upper Drum Contents Average	250.6	235.9	-14.7	302
Upper Drum Wall Peak	251.2	233.6	-17.6	—
HDPE Shielding Wall Peak	—	—	—	170
HDPE Shielding Wall Average	—	—	—	150
RLC Shell Peak	201.9	189.2	-12.7	210
RLC Lid Peak	161.3	155.8	-5.5	212
IV Shell Peak	158.0	158.5	+0.5	160
IV Base Peak	151.5	149.6	-1.9	160
IV Upper Forging Peak	146.8	144.6	-2.2	160
IV Lid Peak	145.4	143.0	-2.4	160
IV O-ring Seal Peak	145.4	143.0	-2.4	225
OC Inner Shell Peak	150.2	150.3	+0.1	160
OC Lead Shield Peak	150.1	150.3	+0.2	160
OC Outer Shell Peak	150.2	150.3	+0.1	160
OC Base Peak	144.9	144.5	-0.4	160
OC Upper Forging Peak	143.6	143.0	-0.6	160
OC Lid Peak	143.6	143.0	-0.6	160
OC O-ring Seal Peak	141.3	140.7	-0.6	225
OC Thermal Shield Peak	151.6	151.8	+0.2	185
OC Main Trunnions Peak	149.5	149.6	+0.1	153
OC Small Trunnions Peak	149.4	149.4	0.0	153
Impact Limiter Foam Peak	156.2	156.2	0.0	260
Impact Limiter Foam Average	138.0	137.8	-0.2	140
Impact Limiter Shell Peak	157.2	157.2	0.0	185

Table RSI 3-1.3 – HAC Fire Transient Temperatures (T) and Time (t) after End-of-Fire for an RLC with 30-Gallon Drums Containing a 100 lb/ft³ Payload with 270 Watts Decay Heat; without and with Internal Convection

Location	No Internal Convection		Internal Convection		ΔT	Allowable Limit
	T (°F)	*t (hr)*	T (°F)	*t (hr)*	(°F)	(°F)
Lower Drum Contents Peak	331.5	*59.0*	317.9	*56.5*	-13.6	402
Lower Drum Contents Average	275.6	*22.5*	264.1	*20.5*	-11.5	302
Lower Drum Wall Peak	280.9	*18.0*	269.4	*17.0*	-11.5	—
Middle Drum Contents Peak	345.3	*55.5*	329.2	*53.0*	-16.1	402
Middle Drum Contents Average	293.7	*19.0*	279.7	*17.5*	-14.0	302
Middle Drum Wall Peak	283.0	*17.0*	272.0	*16.0*	-11.0	—
Upper Drum Contents Peak	330.5	*56.5*	317.3	*54.0*	-13.2	402
Upper Drum Contents Average	274.1	*20.0*	263.7	*18.0*	-10.4	302
Upper Drum Wall Peak	282.5	*17.0*	271.6	*16.0*	-10.9	—
HDPE Shielding Wall Peak	—	—	—	—	—	256
HDPE Shielding Wall Average	—	—	—	—	—	—
RLC Shell Peak	284.2	*3.85*	282.6	*3.50*	-1.6	650/256
RLC Minimum	163.2	*3.85*	158.1	*3.50*	-5.1	—
RLC Lid Peak	200.4	*19.5*	200.0	*16.0*	-0.4	650/256
IV Shell Peak	384.4	*1.4*	377.7	*1.4*	-6.7	390
IV Base Peak	204.7	*17.5*	203.2	*17.0*	-1.5	300
IV Upper Forging Peak	190.4	*12.5*	193.3	*12.0*	+2.9	300
IV Lid Peak	186.7	*22.0*	187.7	*17.5*	+1.0	300
IV O-ring Seal Peak	186.4	*20.0*	187.7	*17.5*	+1.3	360
OC Inner Shell Peak	546.8	*0.4*	546.8	*0.4*	0.0	550
OC Lead Shield Peak	570.4	*0.1*	570.5	*0.1*	+0.1	620
OC Outer Shell Peak	701.5	*0*	701.0	*0*	-0.5	705
OC Base Peak	202.4	*10.0*	201.7	*10.0*	-0.7	500
OC Upper Forging Peak	242.2	*1.2*	235.2	*1.2*	-7.0	500
OC Lid Peak	197.5	*9.15*	195.7	*10.0*	-1.8	500
OC O-ring Seal Peak	193.9	*12.0*	192.3	*12.0*	-1.6	360
OC Thermal Shield Peak	1,323	*0*	1,323	*0*	0.0	2,600
OC Main Trunnions Peak	1,070	*0*	1,070	*0*	0.0	2,600
OC Small Trunnions Peak	1,096	*0*	1,096	*0*	0.0	2,600
Impact Limiter Foam Peak	1,473	*0*	1,473	*0*	0.0	—
Impact Limiter Foam Average	—	—	—	—	—	—
Impact Limiter Shell Peak	1,474	*0*	1,474	*0*	0.0	2,600

3-2 Explain why the thermal evaluation of the RH-TRU 72-B package does not consider a realistic configuration to determine package maximum temperatures and pressures.

Page 3.6.2-2 of the SAR states that "The use of a uniform radial gap is appropriate for NCT and HAC evaluations even though the RH-TRU 72-B package is transported horizontally since the decrease in the radial gap on one side of a component will be offset by a corresponding increase in gap on the opposite side."

However, the sensitivity study performed in the application (Section 3.6.2.3, "Radial Component Offsets") states that temperature increases as large as 47°F were calculated for some components. This sensitivity of the results to the package configuration combined with other modeling assumptions (see Observation 3-1) could result in larger package temperatures and pressures, especially during HAC (fire).

The information is needed to demonstrate compliance with 10 CFR 71.71 and 71.73.

Response:

The evaluation presented in the response to RSI Observation 3-1 determined that including the effects of internal convection results in a net decrease of package temperatures for the RLC and all components/materials internal to the RLC. Temperature changes for the IV and all components external to the IV were shown to be negligible. Therefore, the sensitivity study presented in SAR Appendix 3.6.2.3, with its conclusion that the original evaluations with all package components centered is reasonable for determining maximum package temperatures, is still valid and appropriately addresses and dismisses any adverse impact of adjacent components being in contact with one another.

As discussed in SAR Appendix 3.6.2.3, the only effect of any significance due to shifting components radially into contact with one another was to obtain a few localized hot spots on the IV where it was assumed to be in direct contact with the hot OC inner shell during the HAC fire. However, as indicated in Appendix 3.6.2.3, the increased temperatures remain well within the capability of the IV to perform its function (i.e., material strengths and large structural margins of safety are only slightly degraded by the increased temperatures). Further, if internal convection was considered in addition to the radial shifting, peak temperatures at the local hot spots would necessarily reduce given the added ability to transfer heat between adjacent components away from their points of contact.

3-3 Explain how the total decay heat load of the RH-TRU 72-B package is verified to be at or below allowable limits before transport.

Page 3.6.1-6 of the SAR states that "For potential use with RH-TRU waste canister (fixed or removable lid canister, FLC or RLC, respectively) shipments, maximum decay heat loads were developed for a variety of 30-gallon drum payload densities from 12 to 100 lb/ft3 using the thermal properties of polyurethane foam as a surrogate payload material."

The above statement implies that a surrogate material was used to determine acceptable temperatures. However, the application does not explain how to total decay heat load is verified and validated to be at or below allowable limits.

The information is needed to demonstrate compliance with 10 CFR 71.71 and 71.73.

Response:

The total decay heat load that actually exists for a given shipment is determined in accordance with the RH-TRAMPAC, Section 5.1.4, *Decay Heat Limits*. The procedure for determining compliance with decay heat limits is unchanged from the requirements currently approved for the RH-TRU 72-B package.

The basis for using polyurethane foam as a surrogate material is provided with note 1 of SAR Table 3.2-8, *Thermophysical Properties of the Payload Material*. As stated therein, this surrogate provides lower bound conductivities as compared to actual TRU waste. The surrogate material was not actually used to determine acceptable temperatures, but rather, by virtue of its conductivity being conservatively low compared to actual TRU waste, was used to analytically establish upper bound temperatures that could exist for the actual TRU waste. The governing allowable temperatures for the waste, as presented in SAR Section 3.3.3, *Payload Material*, are based on properties of the waste itself, not the surrogate material.

Shielding

5-1 Justify the procedure for qualifying contents as a distributed source.

Section 5.5.4, "Determination of Acceptable Activity," discusses the procedure for qualifying contents as a distributed source. Case A states: "For drums overpacked in a payload canister, the canister contents may be considered distributed if the measured surface dose rate of each drum in the canister varies by less than a factor of 10 from the average surface dose rate of all drums in the canister." Justify why this qualifies as "distributed" and discuss how it is ensured for these payloads that there is no reconfiguration during NCT or HAC that would cause the package to exceed regulatory external dose rate limits in 10 CFR 71.47 or 10 CFR 71.51(a)(2).

This information is needed for the staff to determine compliance with 10 CFR 71.47 and 10 CFR 71.51(a)(2).

Response:

Similar to the qualification under Case B in SAR Section 5.5.4, Case A is generally based on the NUREG-1608 summary of IAEA advisory material that classifies the degree of non-uniformity of material as "distributed throughout" when the specific activity of sample volumes vary by less than a factor of 10. From the perspective of the RH-TRU 72-B limiting dose rate at 2 meters from the package surface under NCT, the package contents are considered distributed when there is a reasonable distribution of dose-emitting activity throughout the payload canister.

Reconfiguration under NCT/HAC is addressed by the administrative margin for sum of fractions presented in SAR Section 5.4.4 and further discussed and supported by the Attachment A, *Shielding Summary*, and RSI 5-1 response contained herein.

5-2 Justify how concentrated sources will be shored within the center of the internal payload container.

Section 5.3.1, "Configuration of Source and Shielding," states: "Under NCT, the sources are evenly distributed to, and centralized within each of the internal payload containers inside the payload canister." Discuss how a concentrated source (i.e., a source not distributed within the canister) will be shored within the center of the internal payload container. Discuss how these shoring mechanisms survive NCT specified in 10 CFR 71.71.

This information is needed for the staff to determine compliance with 10 CFR 71.47.

Response:

As further clarified and discussed in SAR Section 5.3.1, the sources are not explicitly assumed to be shored within the center of the internal payload container(s) or assumed to absolutely remain in that location under NCT. The sources are initially assumed to be centralized in order to determine the maximum activity that meets the NCT at 2 meter dose rate limit, where that activity is subsequently shown to satisfy the HAC at 1 meter dose rate limit when fully reconfigured and coalesced to a single worst-case location in line with the puncture bar damage with minimal distance to the 1 meter detector. The distribution and location of the NCT sources is addressed with a system of controls as discussed in Section 5.3.1 as follows: "Any significant deviation from the assumed NCT starting configuration is precluded through the use of preshipment radiological surveys of the package to validate the source(s) are either well below their maximum allowed activity limits or are reasonably distributed within the package if they are at or near their maximum allowed activity limits. Further, implementation of a 10% administrative margin in the compliance evaluation is used to effectively accommodate the potential for source movement during NCT."

5-3 Clarify statements in the SAR that state that the distance from the source to the detector under NCT is maximized.

Section 5.3.1, "Configuration of Source and Shielding," states: "NCT assumptions maximize the distance from the source(s) to the nearest detector whereas HAC assumptions minimize the distance from the source(s) to the nearest detector." The staff does not understand this assumption and the discussion following. Clarify this statement. This modeling strategy has been used in package analyses that relied on pre-shipment measurement to meet regulatory dose rate limits to create a bounding source for HAC analyses. However in evaluations used to establish activity limits for allowable contents, this would produce a non-conservative limit as any reconfiguration could cause dose rates to increase. The staff believes that the distance to the detector should be minimized to establish the maximum amount of contents that would meet the regulatory dose rate limits. This statement appears to be inconsistent with the first statement in this section that states: "Under NCT, the sources are evenly distributed to, and centralized within, each of the internal payload containers inside the payload canister."

This information is needed for the staff to determine compliance with 10 CFR 71.47.

Response:

See Observation 5-2 response, above.

5-4 Justify the source locations assumed under HAC.

Section 5.3.1, "Configuration of Source and Shielding," states: "Under HAC, these concentrated sources are coalesced to a single location just inside the confinement boundary for the RH-TRU waste canister and coalesced to two locations (98% of the coalesced sources is located just inside the shield insert and 2% is located just outside of the shield insert next to the confinement boundary for the canister) for the NS15 and NS30 neutron shielded canisters to conservatively bound NSC test results." Discuss more specifically the source locations used in the HAC analyses and justify that these locations produce the most conservative dose rate.

This information is needed for the staff to determine compliance with 10 CFR 71.51(a)(2).

Response:

The HAC dose rate for the RH-TRU Waste Canister payload and the NS15 and NS30 neutron shielded canisters in the RH-TRU 72-B is maximized when the maximum allowed NCT activity (as determined by meeting the NCT dose rate limit at 2 meters for centered sources) is reconfigured to a location that minimizes the distance to the 1 meter detector and exploits damage to the packaging and canister shield components. For the RH-TRU Waste Canister, the canister is shifted in the radial direction to the maximum possible extent and the sources are coalesced into a single location up against the inner surface of the canister wall and in line with the HAC puncture damage in the RH-TRU 72-B composite lead/stainless steel shield structure. Due to the presence of the impact limiters and the additional radial and axial distance attenuation provided near the ends of the package, the evaluation of a maximally reconfigured coalesced source in line with the puncture damage at the midline of the package is bounding. The NS15 and NS30 evaluation is consistent with that for the RH-TRU Waste Canister with the exception of the presence of the HDPE shield insert inside of the canister shell and the presence of two maximally located and coalesced sources. The two sources contribute 98% of the activity within the shield insert and 2% of the activity outside of the shield insert and inside the canister shell. The distribution of activity in the two sources is conservatively based on HAC drop testing of the NS30 design, as discussed in Appendix 5.1 of the RH-TRU Payload Appendices, which accounts for an order of magnitude more release of material from within the shield insert than observed in the testing.

5-5 Justify the 10% margin used to account for source movement during NCT.

Section 5.3.1, "Configuration of Source and Shielding," states: "a 10% administrative margin in the compliance evaluation is used to effectively accommodate the potential for source movement during NCT." Justify that this 10% margin is enough to account for source movement during NCT.

This information is needed for the staff to determine compliance with 10 CFR 71.47.

Response:

SAR Section 5.5.4, Footnote 5, provides the basis and justification for the 10% administrative margin that results in a sum of fractions limit of 0.9 as follows: "Although under NCT conditions, source material movement would be expected to be primarily in a gravity driven, vertically downward direction with very little or no impact on the dose rate at 2 meters from a vertical plane at the side of the conveyance, there is a potential for some, random vibration induced, net lateral (radial) movement. Use of the 10% administrative margin, which allows for a net lateral (radial) relocation of all source material of up to 40% of the radius of the drum, will ensure that regulatory dose rate requirements are satisfied during transport."

The 40% relocation value referenced above is a reasonable upper bound on the amount of source movement and is approximately associated with a radial source location at which the package 2 meter dose rate from Cs-137 is increased by 10% over the baseline. The amount of radial displacement to have a similar increase in the 2 meter dose rate for Co-60 is over 60% of the drum radius. The offset values were determined by modifying the NCT shielding model for the RH-TRU Waste Canister and progressively moving the sources from the centered position radially toward the detector and determining the point at which the activity required to reach the limiting 2 meter dose rate limit was reduced by 10%. The Attachment A, *Shielding Summary*, provides additional discussion as to why source movement under NCT is not significant for TRU waste such that the administrative margin is sufficient to ensure NCT dose rate compliance even if movement were to occur.

5-6 Provide information on the effects of NCT on the canisters.

Section 5.3.1.1, "RH-TRU 72-B," includes a discussion on the effects of NCT on the package. Discuss the effects of NCT on the required canisters: RHTRU Waste Canister, NS15 Neutron Shielded Canister, and NS30 Neutron Shielded Canister.

This information is needed for the staff to determine compliance with 10 CFR 71.47.

Response:

SAR Sections 5.3.1.2, 5.3.1.3, and 5.3.1.4 summarize and clarify that the damage to the canister under NCT is negligible. These conclusions and the baseline NCT geometry models utilized in the shielding evaluation are justified when considered against the minimal elastic stresses experienced by the canister under NCT drop conditions as discussed in SAR Section 2.6 for the canister end drop (pp. 2.6-23), oblique drop (pp. 2.6-28), and side drop (pp. 2.6-36). Additionally, RH-TRU Payload Appendix 5.1 describes the results of an extensive HAC drop test evaluation of the NS30 under HAC drop test conditions that were significantly more severe than what would be experienced by the canister inside the RH-TRU 72-B. With g-loads under NCT drop conditions an order of magnitude less than that evaluated for the NS30 and the lack of significant damage to the canister and HDPE shield inserts under those extreme conditions, it is concluded that NCT damage to canisters and any impact of that damage on the NCT shielding evaluation is negligible.

5-7 Provide more information on the locations of the dose rate tallies.

Section 5.4.4, "External Radiation Levels," provides some information on the locations of the tally surfaces used. Provide additional information describing the locations of the dose rate tallies and demonstrate that these are the most limiting dose rate locations.

This information is needed for the staff to determine compliance with 10 CFR 71.47 and 10 CFR 71.51(a)(2).

Response:

For HAC, when taking into consideration the lack of presence of lead slump at the ends of the RH-TRU 72-B lead column, the possible locations of displaced sources at the extremes of the canister cavity, and the addition of significant amounts of additional distance attenuation ensured by the presence of the impact limiters that remain attached to the ends of the package, it becomes evident that the worst case tally location is 1 meter radially out from the side of the package and directly in line with both the assumed coalesced source and the puncture damage. Use of this configuration ensures that the HAC analysis model described in SAR Section 5.3 is appropriate and bounding. As briefly discussed in the Attachment A, *Shielding Summary*, prior NRC confirmatory analyses reached the same conclusion (see NRC SER supporting Rev. 6 of the RH-TRU 72-B CoC).

NCT evaluations similarly employ dose rate tallies at the side surface and 2 meters from the side surface of the package. This minimizes distance from the source to the detector such that the dose rate tallies discussed in SAR Section 5.4.4 are the most limiting. Under NCT, dose rate compliance is further ensured by the utilization of an extensive preshipment dose rate survey. This survey ensures that the maximum package dose rates are recorded and the regulatory dose rate requirements are satisfied.

5-8 Provide additional information on the distribution of the source contents within the canisters.

Per Figures 5.5-1, 5.5-3, and 5.5-5, it appears as though the canisters under NCT are broken into three regions. Discuss this modeling and state if the source is also broken into three regions. Justify this assumption.

This information is needed for the staff to determine compliance with 10 CFR 71.47.

Response:

SAR Section 5.3.1 clarifies that under NCT the sources are evenly distributed to, and centralized within, each of the internal payload containers inside the payload canister. As such, the sources under NCT are evenly distributed into three regions that represent the 3 payload drums within the canister.

The Attachment A, *Shielding Summary*, provides additional discussion and justification, along with the RSI Observation 5-2 response, for why the NCT source assumption is appropriate.

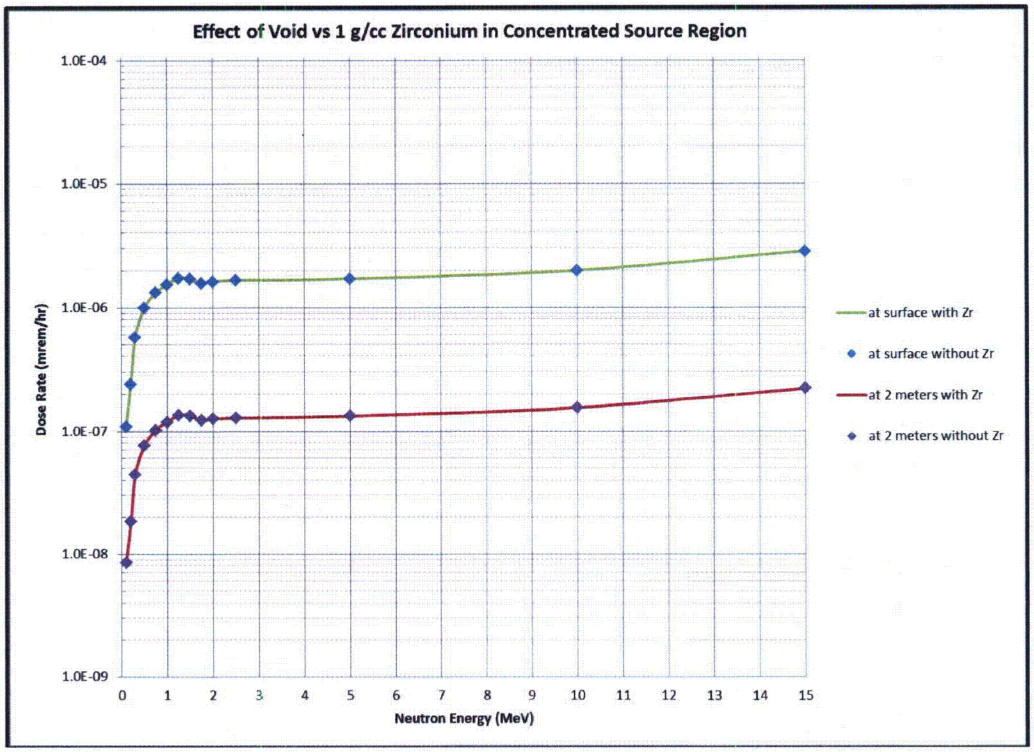
5-9 Provide additional information on the assumptions used with respect to self-shielding in the neutron source analyses.

Section 5.5 states: "For distributed neutron sources, the source was evaluated at a single density, 1 g/cc, which applies to any density in the source range without the use of a DCF since the Zr material does not attenuate neutrons. Therefore, the distributed neutron source has an activity allowable that differs from the concentrated neutron source due to distance attenuation effects alone whereas the distributed gamma source has an activity allowable that differs from the concentrated gamma source due to distance and material attenuation effects." Justify these statements related to the neutron attenuation. Given the fact that Zr does not attenuate neutrons, discuss why this was chosen for the modeling medium. Describe whether any calculations were performed for a distributed source without any type of material assumed for the source to show that it is equivalent to a distributed source composed of Zr.

This information is needed for the staff to determine compliance with 10 CFR 71.47 and 10 CFR 71.51(a)(2).

Response:

Zirconium is used extensively in nuclear reactor designs because it is understood to be largely "invisible" to and minimally interact with neutrons. Due to this fact and the desire to minimize the potential for input/modeling errors by utilizing a common and automated Excel template for the creation of gamma and neutron MCNP input decks, the template source region material was retained as Zr for all analyses. Scoping calculations were performed to ensure that the modeling methodology was appropriate and that significant neutron interaction with the Zr source region material did not occur. Figure RSIO 5-9.1 illustrates that the effect of Zr vs void in the source region for both concentrated and distributed source assumptions (utilizing a density of 1 g/cc) is negligible on the dose rate over the analyzed neutron energy spectrum.



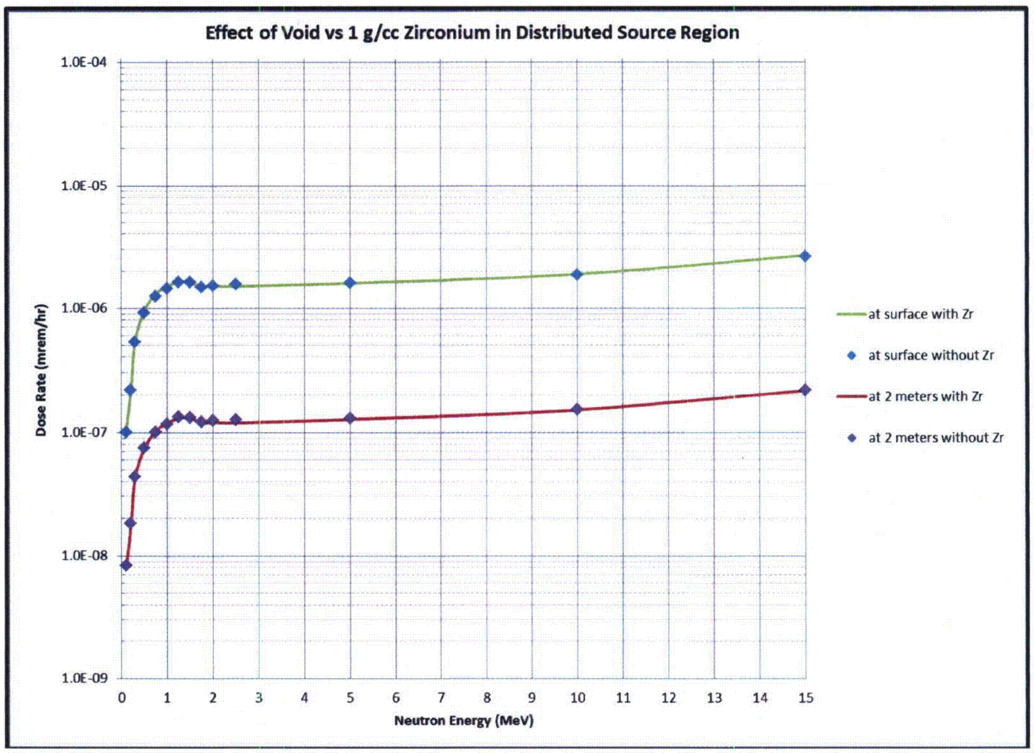


Figure RSIO 5-9.1 – Effect of Zirconium vs Void in Source Region for Neutrons

5-10 Update the SAR revision number in the footnote of the RH-TRAMPAC document.

Activity limits referenced in the certificate of compliance refer to RH-TRAMPAC document which then refers to RH-TRU SAR Section 5.5.4. The footnote to this reference states "current revision" of the SAR. This should be updated to reflect the actual revision number (for example Revision 7) of the SAR.

This information is needed for the staff to determine compliance with 10 CFR 71.47 and 10 CFR 71.51(a)(2).

Response:

The NRC Certificate of Compliance (CoC) has historically been utilized to ensure that the proper revision level of each document that comprises an application is invoked and utilized. Inter-document references within the RH-TRU 72-B SAR, RH-TRAMPAC, and RH-TRU Payload Appendices have not historically incorporated a specific revision level due to the reliance upon the CoC to define the specific application and supplements that form the basis of the issued certificate. Incorporation of a specific revision level to the inter-document references would require that any modification or change to one document in support of an amendment would unnecessarily require modification of all other documents to update the revision level of the modified reference. To avoid this inefficiency, it is therefore requested that the CoC continue to be utilized and relied upon to identify the dated application and supplements that ultimately define the applicable revision level of all SAR documents associated with the CoC.