RH-TRU 72-B Packaging

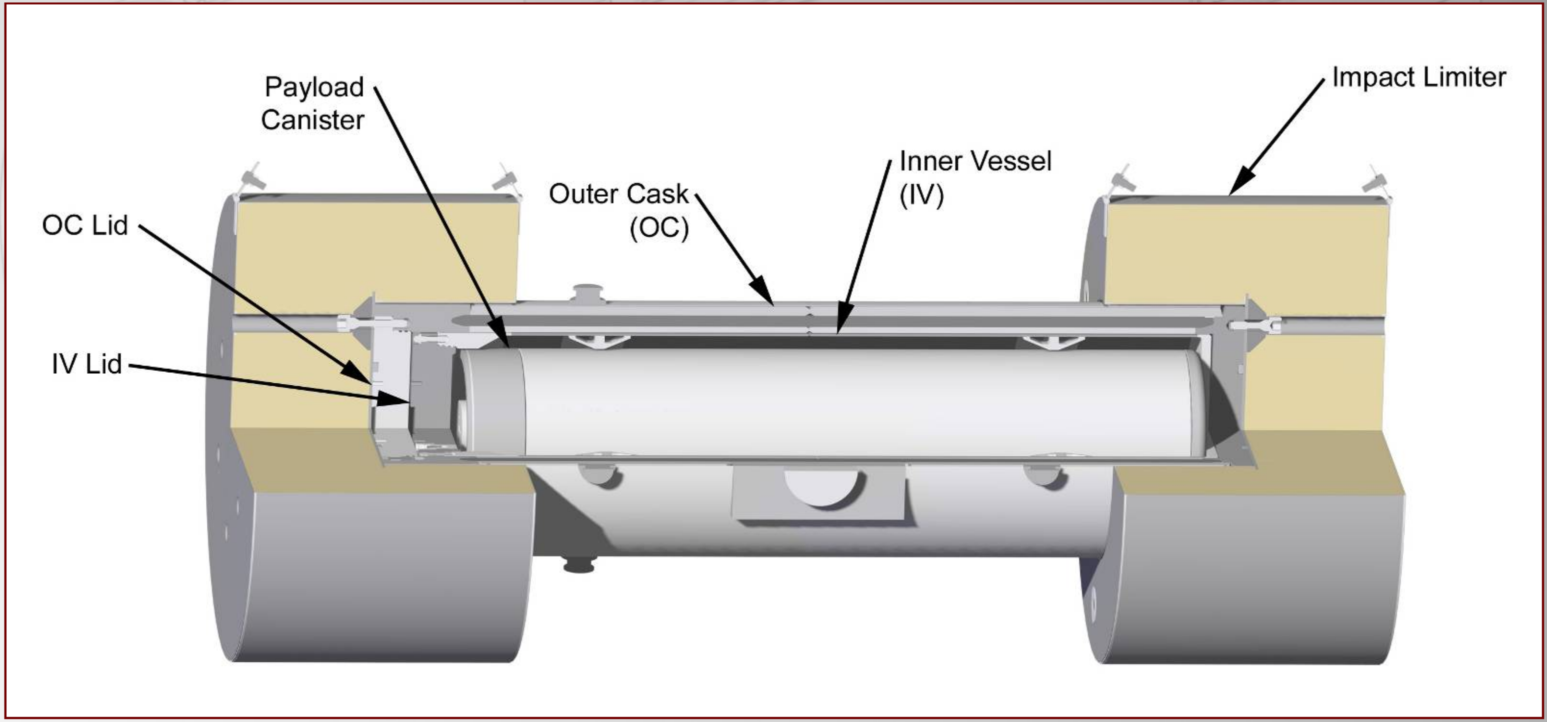
RH-TRU 72-B SAR Revision 5 Amendment

Request for Supplemental Information dated November 4, 2010

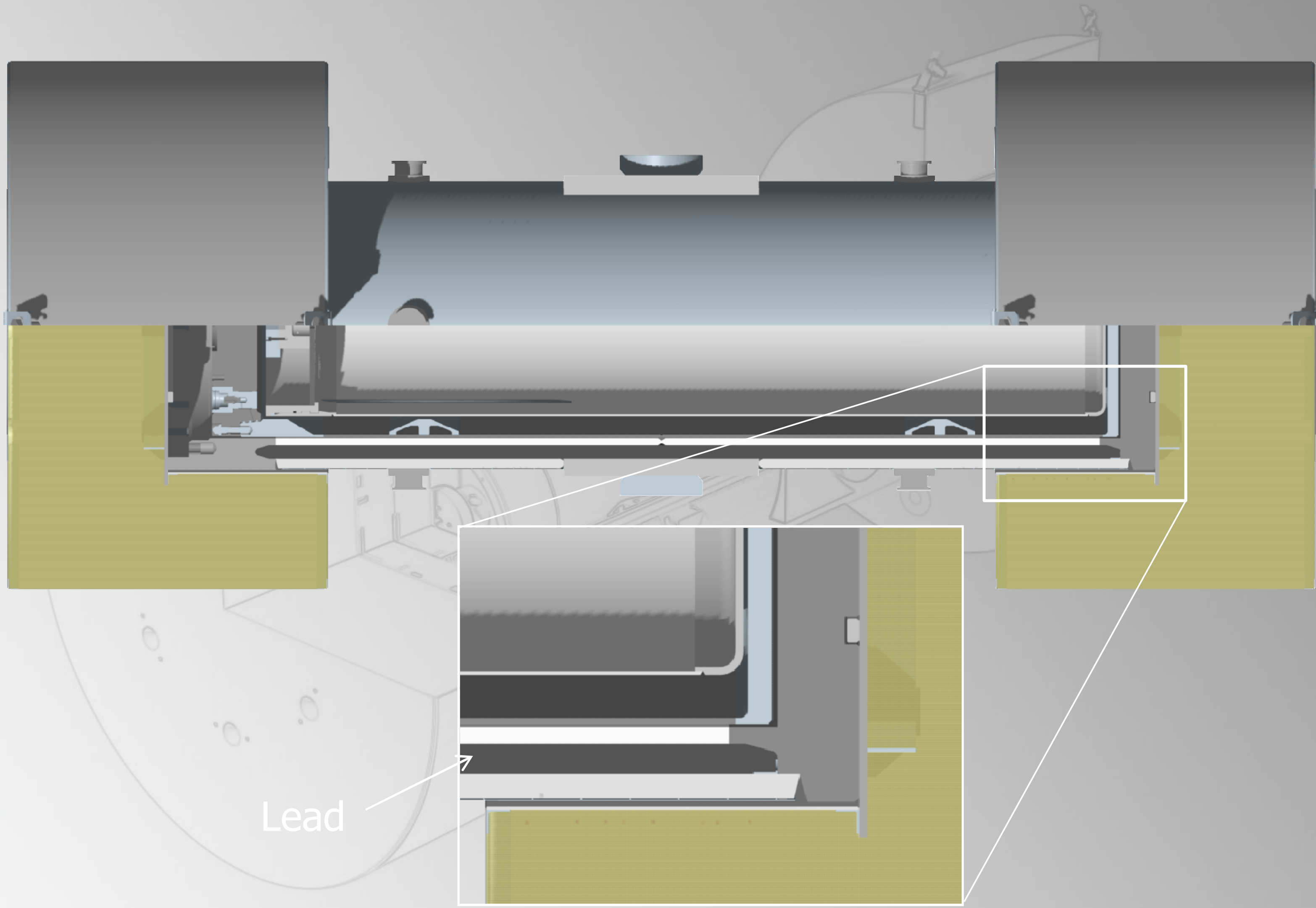
Presented to the US Nuclear Regulatory Commission
by Washington TRU Solutions on behalf of the US Department of Energy
December 9, 2010

RH-TRU 72-B

- Description
  - Overall Length = 187-%, Impact Limiter Diameter = 76”, Cask Length = 141-%”, Outer Cask OD = 41-.”, Inner Vessel Length = 130”, Inner Vessel OD = 32”, Cask Lead Thickness = 1-/”, Max. Package Weight = 45,000 lb, Max. Contents Weight = 8,000 lb

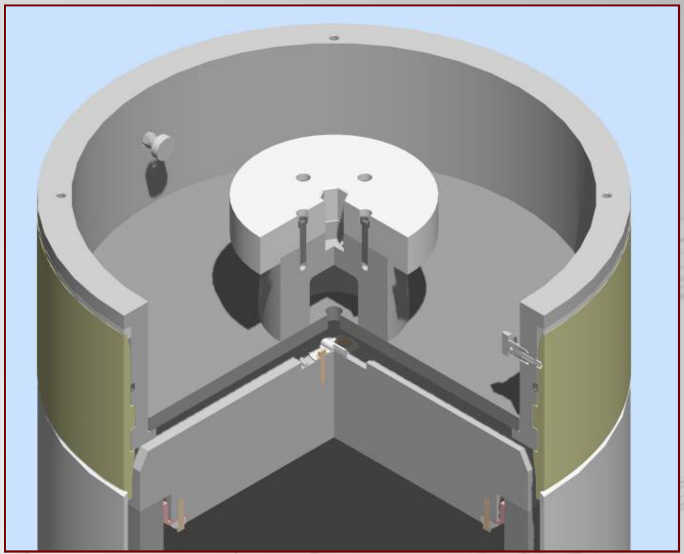


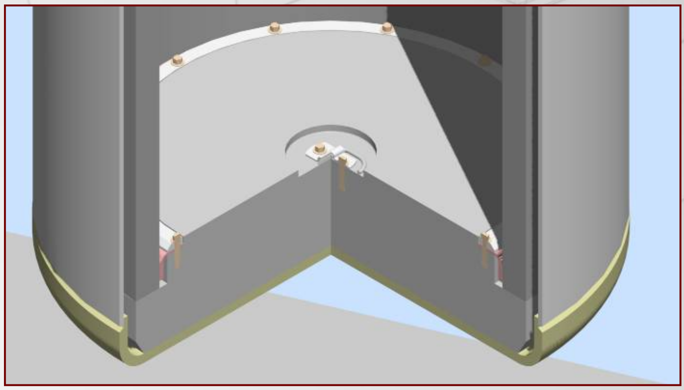
RH-TRU 72-B Gamma Shielding

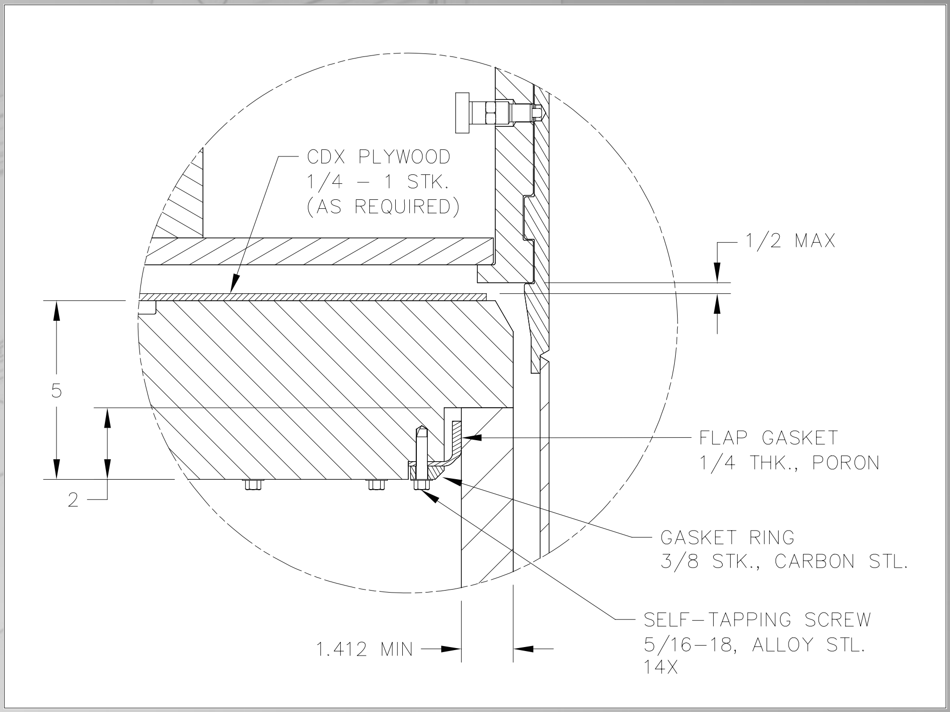


Neutron Shielded Canisters

- NS15 & NS30 (shown)







Thermal RSI-1

Update the NS15 and NS30 thermal analyses presented in the Safety Analysis Report (SAR) to reflect the recently provided NCT and HAC thermal analyses.

The response from the Request for Additional Information (RAI) teleconference (9/27/10) provided three shielded NS15/NS30 thermal analyses that do not use bulk spatial and temporal-averaged insolation boundary conditions. As a result of the analyses, the applicant mentions that the RH-TRU 72-B SAR design decay heat limit will be changed from 300 W to 50 W per canister. In addition, the new modeling methodology and the higher temperatures of the components found as a result of the updated NS15 and NS30 analyses should be incorporated in the appropriate sections of the SAR, such as Appendix 5.1 of the RH-TRU Payload Appendices.

This information is requested by staff to determine compliance with 10 CFR 71.71 and 71.73.

Thermal Sensitivity Analysis

- Performed 2-D transient thermal with 400 W/m^2 insolation to all exposed external surfaces (NS15 Sensitivity A) to compare with SAR 2-D steady-state with 127 W/m^2 (Current SAR)
  - all component temperatures remain well within the NCT allowable temperature limits
  - even with solar loads applied, the maximum external surface temperatures remain within the regulatory limits for exclusive-use shipments
  - the NS15 shield insert maximum temperature remains less than its 150 °F design basis
  - all sensitivity analysis temperatures are below the current SAR 300-watt metallic waste case temperatures that are utilized to establish the bounding values for all hot temperature structural component evaluations
  - hot foam structural evaluations are performed using a bulk average foam temperature of 140 °F, which remains above the bulk average foam temperature of 133 °F established by the sensitivity analysis

Thermal Sensitivity Analysis (cont.)

Location/Component	Maximum Temperature (ºF)					
	NS15 (50W)*		NS30 (50W)	50W	300W	Allowable
	Sensitivity A	Current SAR	Current SAR	Current SAR	Current SAR	
Waste Centerline	250	247	234	217	181	302
NS_ _ Shield Insert	149 (142 Avg.)	141	137	N/A	N/A	256
Canister Shell	142	133	132	132	167	2,600
IV Shell	140	128	128	127	150	800
IV Void Space Bulk Avg	140	127	127	-	-	N/A
OC Inner Shell	140	126	126	126	143	800
OC Lead Shield	140	126	126	126	143	620
OC Outer Shell	140	126	126	126	143	800
OC Thermal Shield	145	125	125	125	142	185
OC Upper Ring Forging	134	125	125	126	137	800
IV O-Ring Seal	134	125	126	126	140	225
OC O-Ring Seal	133	125	125	126	137	225
IV Lid	134	125	126	127	141	800
OC Lid	134	125	125	126	137	800
Impact Limiter Foam	150 (133 Avg.)	132 (127 Avg.)	132 (127 Avg.)	132	143	300
Impact Limiter Shell	158	133	133	133	142	185

*The NS15 case was chosen since it has the highest waste centerline, shield insert (as applicable), and canister shell temperatures for all 50-watt cases with all other packaging component temperatures within 2 °F of the current NS15 SAR analysis.

Proposed Thermal RSI-1 Resolution

- Option 1 (preferred) – Rely on RSI response that documents the sensitivity analysis to demonstrate that the current SAR analyses are adequate to ensure safety because all limits are met even with the application of 400 W/m^2 insolation boundary conditions to all exposed external surfaces
  - All shipments to date << 50 watts (1 @ 13.83 W, 93% < 1 W)
  - Shipments under the 300-watt case to be disallowed in the RH-TRAMPAC to ensure sensitivity results are bounded by existing 300-watt analysis
  - SAR Chapter 3 revised under timely renewal process to formally incorporate currently accepted practice
- Option 2 – Revise SAR Chapter 3 to formalize a "currently accepted" application of insolation boundary conditions
  - Propose consistency with IAEA guidance to apply a more reasonable insolation boundary condition of 400 W/m^2 on upward facing curved surfaces and 200 W/m^2 on downward facing curved surfaces

Shielding RSI-1

Provide an analysis of the effect of lead slump on the HAC dose rates.

The applicant provided some discussion in response to the shielding RAI #1 on this subject, indicating that slumping will not occur. However, staff does not find the basis for this conclusion to be applicable. Thus, a shielding analysis should be provided for lead slump, as predicted using the method in the "Cask Designers Guide" document. The analysis should also account for any void between the top of the lead shielding and the outer cask top flange resulting from package fabrication. The analysis should account for the assumed 2% of the source escaping the canister's neutron shield insert and lodging as close to the slump area as allowed by the package HAC configuration. The remainder of the source should likewise be positioned as near as possible to the slump zone while remaining within the canister's neutron shield insert. Using analyses for a few radionuclide contents (e.g., Co-60), the applicant may demonstrate that the dose rates for the puncture HAC configuration bound those for the lead slump configuration.

This information is needed to confirm compliance with 10 CFR 71.51 and 71.73.

Lead Slump by Analysis

- Cask Designer's Guide Equation 2.16
  - $\Delta H = RWH/[\pi(R^2 - r^2)(t_S\sigma_S + R\sigma_{Pb})]$
  - Utilizes concept of flow stress
    - σ_{Pb}=5,000 to 10,000 psi
  - Equation is scalable
  - Specifically applies to bare, unbuffered cask geometries where the flow stress of lead is exceeded
  - Should not have been applied to 72-B which is buffered by relatively soft energy absorbing impact limiters where the flow stress is not exceeded

Lead Slump by Test

- Avoids undue conservatism of Cask Designer's Guide formula when addressing buffered casks such as 72-B and 125-B
- Scale testing considered to be an acceptable basis
  - Designer's Guide formula scales
  - Other references also defend scaling for lead response
- Valid testing does require use of prototypic manufacturing and lead installation techniques

72-B Lead Slump

- Results obtained from ¼-scale testing of highly similar 125-B are applicable and justifiable
  - ¼-Scale 125-B very closely simulates 3/8-scale 72-B

Physical Parameter of Importance to Lead Slump	Full-Size 72-B	38.5%-Scale 72-B	1/4-Scale 125-B	Δ%
Outer Cask Outer Shell Thickness, in	1.500	0.578	0.500	+15.6%
Outer Cask Inner Shell Thickness, in	1.000	0.385	0.250	+54.0%
Outer Cask Lead Thickness, in	1.875	0.722	0.969	-25.5%
Outer Cask Lead Column Height, in	124.250	47.836	44.750	+6.9%

  - 125-B and 72-B manufacturing and lead installation techniques and controls are the same
- Post-test radiographic inspection of the ¼-scale 125-B concluded no measurable lead slump

Lead Installation Process

- Lead installation set-up, preheat, pouring and cooldown procedures are carefully prescribed and controlled (SAR Section 8.3.1)
  - Package is inverted for lead fill with open end of outer cask facing down
  - Lead is introduced at lower end
  - Package is cooled from lower end up
  - Additional molten lead is added at upper end as lower end lead solidifies and shrinks
  - Result is complete fill of lead cavity and minimal axial gaps at either end of the lead column following cooldown

Gamma Scan Acceptance

- To confirm a void free lead fill and the absence of any significant gaps/shine paths at the ends of the lead column, gamma scan acceptance testing is required (SAR Section 8.1.5)
  - Starting at bottom inside of outer cask, an iridium-192 or cobalt-60 source is raised in increments, while measuring and recording dose rates on the entire exterior surface of the package
- All twelve 72-Bs fabricated to date and now in service have successfully passed required gamma scans

Lead Slump Position Summary

- The following are technically appropriate and utilized by multiple NRC-certified Type B packages, including the currently certified RH-TRU 72-B:
  - Dependence upon scale drop testing to ascertain lead slump
  - Controlled lead installation and post-pour gamma scan inspection to ensure shield integrity
- Fabrication and/or slump induced gaps at the ends of the 72-B lead column are negligible and HAC activity limits are adequately and properly established by considering only the side puncture location

Proposed Shielding RSI-1 Resolution

- Option 1 (preferred) – Rely on RSI response that provides additional detail regarding the basis and justification for use of the 125-B ¼-scale drop testing, lead installation procedure, and post-pour gamma scan to ensure that lead gaps are negligible
  - SAR Chapter 2 revised under timely renewal process to formally remove Cask Designer's Guide based lead slump calculations and incorporate detailed 125-B ¼-scale test data and correlations
- Option 2 – Revise SAR Chapter 2 to formalize the 125-B ¼-scale drop test data and correlations

Shielding RSI-2

Provide sufficient detail regarding the pre-shipment dose rate measurements and results of previous measurements to demonstrate the acceptability of this method, for the current amendment only, for use to meet the requirements of 10 CFR 71.35(a) and 71.47.

Per 10 CFR 71.35(a), an application for a Part 71 Certificate of Compliance (CoC) must include a demonstration that the package containing the proposed contents at the proposed quantity limits satisfies, among other things, the requirements in 10 CFR 71.47. The current amendment application seeks to use pre-shipment dose rate measurements to meet this requirement. While pre-shipment measurements are normally not accepted as fulfilling this requirement, they may be found acceptable in the current case for only the current amendment only with certain additional conditions and the provision of further information to justify that the package's compliance with 10 CFR 71.47 will be ensured for the proposed contents at the proposed quantities. Package operations descriptions in Chapter 7, "Package Operations," of the SAR should also be modified to incorporate (by reference is acceptable) these conditions.

Shielding RSI-2 (cont.)

The applicant has provided some information regarding performance of pre-shipment measurements; however, this information does not completely satisfy the RAI. In addition to the measurement descriptions currently provided by the applicant, descriptions should be included that explicitly state that the neutron and gamma dose rate measurements are performed on the package surface and at 2 meters from the package surface. This ensures clarity as to which surfaces are being referenced. A statement should be added that clearly states that both gamma and neutron dose rate measurements are always performed and that they are done with appropriate instruments of appropriate/adequate dose rate ranges. The descriptions should also include that a grid is established for the entire package surface with squares no larger than a few inches (4 inches for example) on a side, with measurements taken at every grid location. Similarly, a description of how the 2-meter dose rate measurements are/will be comprehensive is also needed.

To justify the use of measurements in this case, the applicant should provide the results of representative cases from the measurements performed on previous shipments under the current CoC. The information should demonstrate the comprehensive nature of the measurements.

Shielding RSI-2 (cont.)

The applicant should provide the maximum measured surface and 2-meter dose rates for the package radial side and axial end surfaces, the contents descriptions (including Curie quantity(ies) and the form of the contents) for each result case included in the information. Results should be provided that cover the range of contents forms that are (to be) shipped in the RH-TRU 72-B package. Also, if the cases are not for the maximum allowed Curie quantity(ies) of the radionuclides present in the given cases, the applicant should also provide an evaluation of the dose rates for a package containing the maximum allowed quantities. A conservative approach to this task would be to take the highest dose rate contributor (both for gamma sources and neutron sources) present in the particular case and scale up its quantity to the maximum allowed by the CoC. Then, because the NS15 and NS30 differ from the current waste canisters, justification should be provided as to why the supplied results are sufficient to demonstrate that the higher quantities in the NS15 and NS30 canisters will meet 71.47 limits. The justification should be quantitative as well as qualitative, noting effects of geometry and shielding differences between the proposed canisters and the currently approved canisters.

The applicant should modify the application to include the requested information.

This information is needed to confirm compliance with 10 CFR 71.35(a) and 71.47.

RH-TRU Waste Dose Rates

- Dose rates from RH-TRU waste vary on a container by container basis as a function of:
  - Mixture and quantity of gamma and neutron emitting radionuclides
  - Physical and chemical composition of source matrix and associated self-shielding properties (internal material attenuation)
  - Distribution of radionuclides in the source matrix (distance attenuation)

Radionuclide Mixture

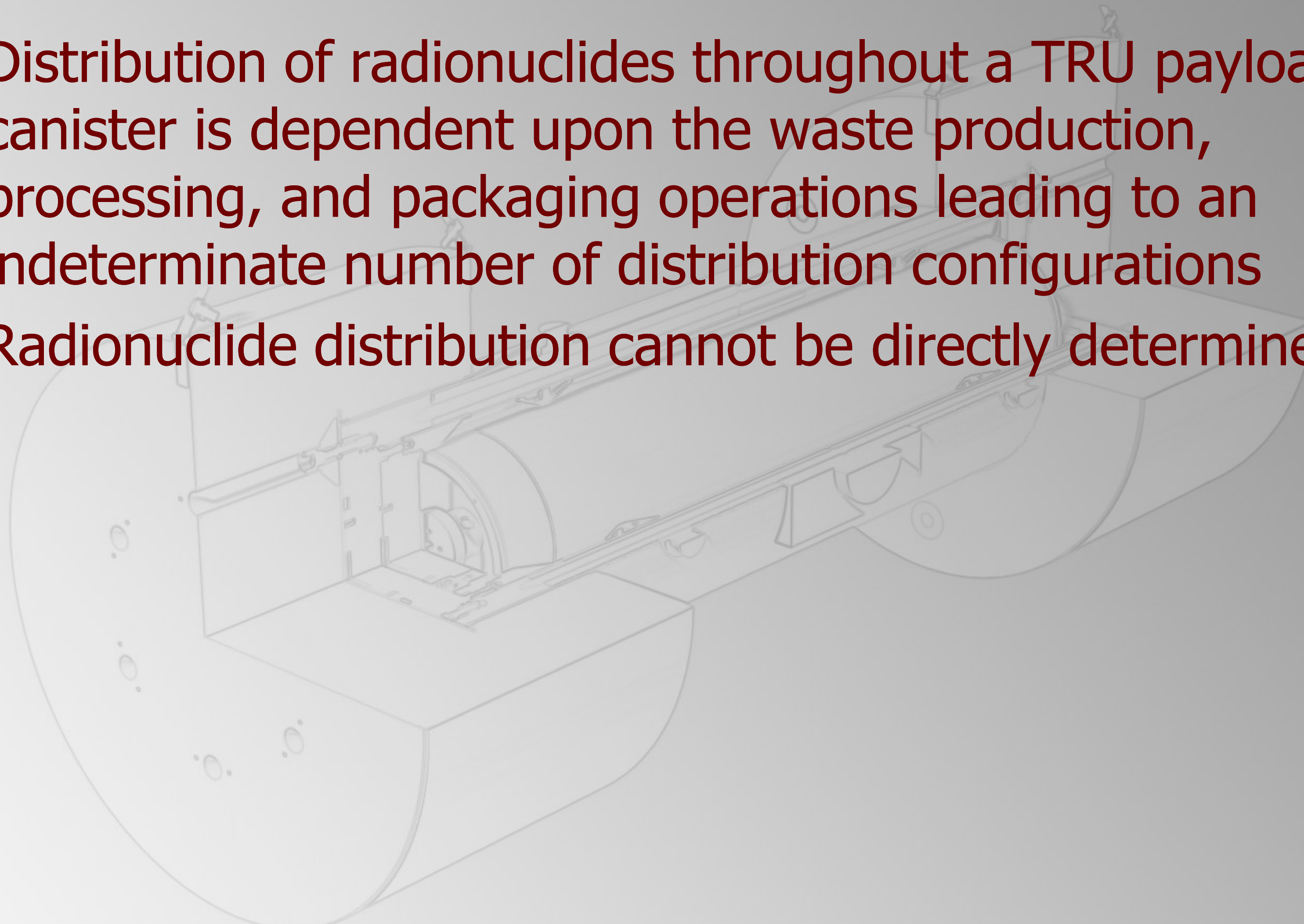
- RH-TRU waste inventory identifies the presence of possible isotopes
  - 169 gamma
  - 37 neutron
  - 31 of the above are combined gamma/neutron
- Radionuclide content is determined via measurement or records

Chemical Composition

- 39 different basic material and chemical constituents in quantities >1% by weight are defined in the chemical lists for 88 TRUCON codes generated to date
- Chemical and material content is determined via records, procurement controls, or sampling

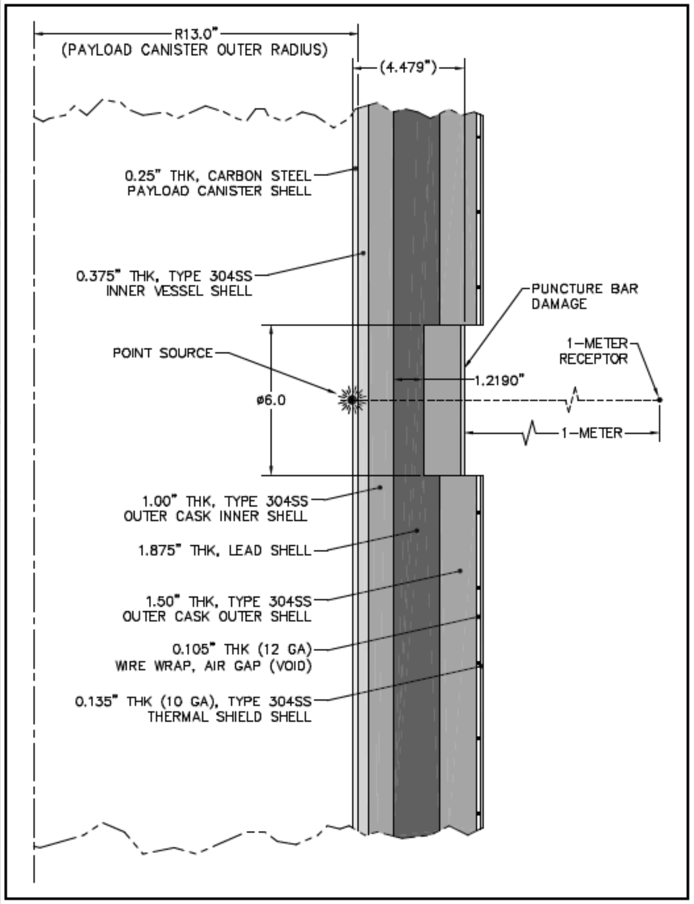
Radionuclide Distribution

- Distribution of radionuclides throughout a TRU payload canister is dependent upon the waste production, processing, and packaging operations leading to an indeterminate number of distribution configurations
- Radionuclide distribution cannot be directly determined



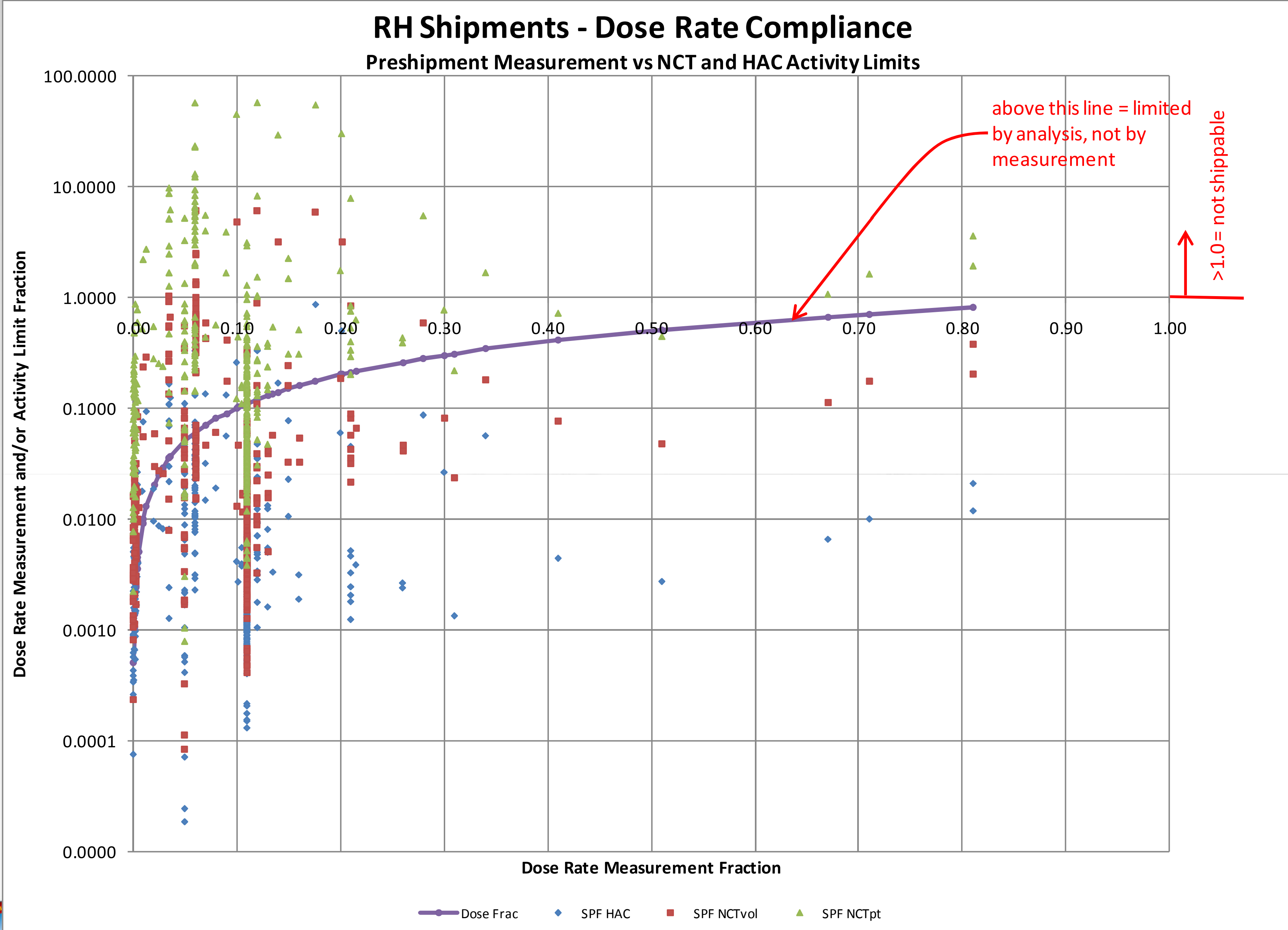
HAC Shielding Analysis

- Bounding Configuration
  - Each radionuclide individually analyzed under conditions of:
    - maximum shield damage (puncture)
    - minimum distribution/distance (point source aligned with puncture at inner surface of payload canister)
    - no credit for self-shielding
    - assume iron build-up factor for composite lead-steel shield
  - Sum of partial fractions utilized to apply individual analyses to each unique mixture of radionuclides



NCT Shielding Analysis

- Bounding Configuration
  - Would unnecessarily reduce shipping efficiency by disallowing shipments (67 or 15%) that have been historically shown (435 evaluated) to satisfy NCT dose rate requirements via a comprehensive preshipment survey, surveys at points of entry, and surveys during receipt at WIPP
    - Shipments would fail due to the bounding analyses not crediting the actual waste matrix properties (distribution, self-shielding); properties are real and present, yet vary on a canister by canister basis
  - Could significantly reduce future shipments as 64% of historical shipments would have been more limited by NCT activity than NCT dose rate measurement
- "Reasonable" Configuration
  - Would require monte-carlo evaluation of all radionuclides to credit radionuclide distribution, waste matrix self-shielding, and accurately calculate build-up in the composite shield
  - A <u>meaningful</u> reasonable configuration cannot be defined due to the variations in chemical/materials, distribution, self-shielding, etc.
  - Purpose is unclear from a regulatory compliance perspective


RH Shipments - Dose Rate Compliance
Preshipment Measurement vs NCT and HAC Activity Limits
Dose Rate Measurement and/or Activity Limit Fraction
100.0000
10.0000
1.0000
0.1000
0.0100
0.0010
0.0001
0.0000
0.00
0.10
0.20
0.30
0.40
0.50
0.60
0.70
0.80
0.90
1.00
Dose Rate Measurement Fraction
above this line = limited by analysis, not by measurement
>1.0= not shippable
Dose Frac
SPF HAC
SPF NCTvol
SPF NCTpt

NCT Dose Rate Survey

- Same procedures employed by shipping sites for all radioactive material shipments (e.g., Type A, Type B, …) to ensure that maximum measured dose rates are in compliance with regulatory requirements (e.g., 49CFR§173.441, 10CFR§71.47, …)
- Procedures and equipment are qualified under a DOE approved QA program to standards (e.g., ANSI-N323A)
- Dose rate emanating from a Type A package does not fundamentally differ from a Type B package
  - The limits are the same, the potential safety consequences are the same, so why not the compliance method?
- Measurement is the best and most direct method to ensure compliance with NCT dose rate limits for TRU waste
- Measurement is the compliance method for the vast majority of nuclear material shipments

SFST-ISG-20

- *Transportation Package Design Changes Authorized Under 10 CFR Part 71 Without Prior NRC Approval*
  - *Regulatory Basis*
    - *The application must include a description of the contents in sufficient detail to provide an adequate basis for evaluation of the packaging design. [10CFR§71.31(a)(1) and §71.33(b)]*
    - *Before each shipment, the licensee must ensure that the package meets the routine determination requirements of 10 CFR Part 71. [10CFR§71.87]*
  - *Section 2*
    - *… specificity of the contents description may be different for different package types and the safety significance of the contents.*
  - *Appendix B*
    - *… typical examples of what a reviewer should look for in a package design and operations in a well-prepared package application from the applicant. Examples are provided for the major package types, and represent actual experience with package approvals.*

SFST-ISG-20 – Example #1

Package Type - Feature	Examples of How Flexibility Can be Provided
Radiography Package - supplemental gamma shielding in drawings	Drawings should show a general arrangement for using supplemental shielding, if needed to meet normal condition dose rate limits. The materials of construction, maximum weight and thickness, and method of attachment should be shown. The specific details are not needed, because the supplemental shielding is intended for the maximum strength source to meet the normal conditions dose rate limit. If the radiation survey does not confirm that the shielding is adequate, the source may not be shipped. The shielding evaluation should show that the package can meet the accident conditions dose rate limit without the supplemental shielding, so its attachment is not critical to the safe performance of the package.

Interpretation: supplemental shielding is equivalent to waste self-shielding such that a preshipment survey satisfies NCT and an analysis not crediting self-shielding satisfies HAC = Current 72-B SAR Methodology

SFST-ISG-20 – Example #2

Package Type - Feature	Examples of How Flexibility Can be Provided
Type B Waste Package - contents specification	The exact isotopic distribution of the contents is often not known for these packages. The contents may be identified in terms of the number of A_2 values for containment considerations. For shielding considerations, a representative loading, along with normal conditions dose rates could be used. The radiation analysis should show that a reasonable rearrangement of contents under accident conditions would result in a dose rate less than the accident condition limit.

Interpretation: 10^5 A_2, HAC activity limits, FGE limits, …, and "leaktight" containment criteria provide sufficient definition of contents; full reconfiguration of activity under HAC = Current 72-B SAR Methodology

SFST ISG-20 Applicability to 72-B

- Contents limits are defined utilizing multiple methodologies per the RH-TRAMPAC to ensure compliance with the regulations and <u>provide a sufficient basis for evaluation of the package</u>.
- Dose Rate Compliance
  - NCT activity limits are not specified directly, but rather limited via preshipment radiological surveys to provide a <u>sufficient basis for evaluation</u>
    - Insignificant damage to the shield under NCT, blocking and bracing of contents, no credit for supplemental ALARA shielding, significant handling of the contents in multiple orientations prior to, during, and after loading in the package along with historical verification of no significant reconfiguration of the contents under NCT ensure measured dose rates are maintained and do not exceed 200 mrem/hr @ surface and 10 mrem/hr @ 2 meters
  - HAC activity limits are determined via analysis and enforced via use of sum of partial fractions to provide a <u>sufficient basis for evaluation</u>
    - The most extreme and conservative HAC analysis conditions ensure a large margin against the dose rate exceeding 1 rem/hr @ 1 meter from the package surface

NCT Dose Rate Position Summary

- Current measurement approach has a proven track record and is the currently approved compliance method for all NRC-certified Type B packages utilized by WIPP (TRUPACT-II, HalfPACT, TRUPACT-III, RH-TRU 72-B, CNS 10-160B)
- Analysis-based NCT activity limits for TRU waste either:
  - unnecessarily restrict safe shipments (including those previously made under an NRC CofC) when using a "bounding" analytical approach, or
  - don't enhance the safety basis of the package (due to not providing activity limits that are bounding for all TRU waste) when using a "reasonable" analytical approach
- A survey method that continuously scans the package surface and at 2 meters from the package surface to ascertain the maximum gamma and neutron dose rates effectively ensures compliance with 10CFR§71.47
  - Use of discrete grid spacing and recording of non-bounding values is onerous

Proposed Shielding RSI-2 Resolution

- Augment RH-TRAMPAC Section 3.2.2 to add the following:

  "Radiation dose rates shall be obtained through the implementation of site-specific procedures that direct the measurement of <u>both</u> gamma and neutron dose rates for the package at the surface and at 2 meters from the surface. Contact dose rate surveys shall be performed on all exposed external surfaces (full-length circumferential and ends) of the package and the highest value recorded. Two-meter dose rate surveys from the outer lateral surfaces of the package (excluding the top and underside of the package) shall be performed and the highest value recorded. Radiation monitoring instruments shall be maintained and calibrated at least annually in accordance with national standards (e.g., American National Standards Institute [ANSI]-N323A)."

Comments