UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 19, 2016

Mr. T. E. Sellmer
Manager – Transportation Packaging
Nuclear Waste Partnership, LLC
P.O. Box 2078
Carlsbad, NM 88221

SUBJECT: REVISION NO. 9 OF CERTIFICATE OF COMPLIANCE NO. 9212 FOR THE MODEL NO. RH-TRU 72-B PACKAGE

Dear Mr. Sellmer:

As requested by your application dated November 10, 2014, as supplemented February 18, July 31, October 22, and December 17, 2015, enclosed is Certificate of Compliance No. 9212, Revision No. 9, for the Model No. RH-TRU 72-B package. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's safety evaluation report is also enclosed.

This approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR 173.471. Registered users of the package under the general license provisions of 10 CFR 71.17 or 49 CFR 173.471 have been provided a copy of this certificate.

If you have any questions regarding this certificate, you may contact me or Huda Akhavannik of my staff at 301-415-5253.

Sincerely,

/RA/

Steve Ruffin, Acting Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9212
TAC Nos. L24965 and L24975

Enclosures:
1. Certificate of Compliance No. 9212, Rev. No. 9
2. Safety Evaluation Report
3. Registered Users

Upon removal of Enclosure 3, this document is uncontrolled

cc w/encls 1 & 2: R. Boyle, Department of Transportation
J. Shuler, Department of Energy, c/o L. F. Gelder

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ADAMS P8 Package No.: ML16020A218 Letter No.: ML16020A227

OFC:	SFST	SFST	SFST	SFST	SFST	SFST
NAME:	HAkhavannik	DWalker	DForsyth	VWilson	JSolis	SEverard
DATE:	12/31/15	1/12/16	1/12/16	1/5/16	1/11/16	1/4/16
OFC:	SFST	SFST	SFST	SFST	SFST	SFST
NAME:	DTarantino	JChang	MRahimi	CAraguas	ACsontos	SRuffin
DATE:	1/6/16	1/5/16	1/15/16	1/15/16	1/15/16	1/19/16

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION REPORT
Docket No. 71-9212
Model No. RH-TRU 72-B
Certificate of Compliance No. 9212
Revision No. 9

SUMMARY

By application dated November 10, 2014, as supplemented February 18, July 31, October 22, 2015, and December 17, 2015, Nuclear Waste Partnership LLC (NWP), on behalf of the U.S. Department of Energy (the applicant) requested revisions to Certificate of Compliance (CoC) No. 9212 for the Model No. Remote-Handled Transuranic (RH-TRU) 72-B Package.

The applicant requested the following changes to the package:

- increasing the decay heat limit from 50 watts to a maximum of 90-270 watts for the Removable Lid Canister (RLC) and Fixed Lid Canister (FLC) payloads,
- a new quantitative activity based limits shielding methodology in place of qualitative preshipment dose rate surveys,
- a new “Case E” criticality analysis that addresses a non-homogenous low enriched uranium (LEU) payload configurations case,
- updated fatigue, lifting and tie down, canister buckling, corner drop, and side drop analyses to include new temperature limit,
- updated material properties for new temperature limits to evaluate canister stress, and,
- revised licensing drawings, operational procedures, acceptance and maintenance procedures, and quality assurance sections.

This application also incorporates revisions to the RH-TRAMPAC and RH-TRU Payload Appendices that are consistent with the above safety analysis report (SAR, or application) changes.

Additionally, the applicant requested renewal of the certificate. The certificate has been renewed for 5 years. Accordingly, CoC No. 9212 has been amended based on the statements and representations in the application, and staff agrees that the changes do not affect the ability of the packages to meet the requirements of 10 CFR Part 71.

EVALUATION

1 GENERAL INFORMATION

Staff evaluated the changes made to this section of the application. The applicant made administrative changes such as improving the figures describing the package, revising nomenclature of the O-ring compound, and updating the references. Additionally, the text was revised to reflect the new 270-watt decay heat limit and to make the design details and presence of the lid alignment pins optional during transportation. These pins are used to improve remote and non-remote lid installation operations by assisting in lid rotational alignment for bolt installation.

The licensing drawings were also updated. As part of the update, all the drawings were converted to be NWP controlled documents. The packaging and assembly drawings, "Nuclear Waste Partnership LLC, Drawing No. X-106-500-SNP, sheets 1-8, Rev. 6," contained updates to reflect the O-ring compound name change and added note 57 to identify those items that are optional for the operating procedures. The note was used for the lid alignment pins on the outer cask (OC) and inner vessel (IV).

The drawing for the construction and assembly of the fixed lid waste canister is now called, "Nuclear Waste Partnership LLC, Drawing No. X-106-501-SNP, Rev. 5." The drawing for the construction and assembly of the removable lid waste canister is now called, "Nuclear Waste Partnership LLC, Drawing No. No. X-106-502-SNP, Rev. 3." The drawing for the construction and assembly of the neutron shielded waste canister is now called, "Nuclear Waste Partnership LLC, Drawing No. X-106-503-SNP, Rev. 1."

2 STRUCTURAL EVALUATION

2.1 Description of Structural Design

The structural design of the Model No. RH-TRU 72-B waste shipping package in revision 7 of the SAR is identical to revision 6.

2.1.1 Discussion

In revision 7 of the structural evaluation, the applicant updated the structural analysis based on changes in temperatures and temperature gradients as a result of the thermal analysis revision. In addition, the applicant revised the approach used to address potential fatigue of bolted fasteners and provided a previously omitted Model No. RH-TRU 72-B waste canister buckling evaluation. The staff limited the scope of the structural review to the updated structural analysis performed by the applicant as a result of the thermal analysis, the fatigue evaluation of the closure bolts, and the buckling evaluation of the canister.

The staff has reviewed the package structural design description and concludes that the contents of the application meets the requirements of 10 CFR 71.31.

2.2 Materials Evaluation

The Model No. RH-TRU 72-B packaging is constructed of a stainless steel OC, which establishes the primary containment boundary, a stainless steel IV, intended and expected to provide a secondary containment boundary, stainless steel and foam impact limiters and the carbon or stainless steel payload canister. Section 1.2.1 of the application discusses the Model No. RH-TRU 72-B packaging materials.

Structural components are primarily fabricated from American Society of Testing and Materials (ASTM) A240, Type 304, stainless steel for the shells and ASTM A240, Type 304, or ASTM A182, Type F304, austenitic stainless steel for the lid and end closure. Non-structural materials include butyl rubber O-ring seals (Rainier Rubber, R0405-70, or equivalent), Nitronic 60 port closure bolts and ASTM A320, Grade L43, carbon steel closure bolts. The lead shielding is per Federal Specification QQ-L-171e, or ASTM B29 (copper-bearing material).

Impact limiter casings are fabricated from 300 series ASTM A276 and ASTM A269 stainless steels, filled with energy absorbing closed-cell polyurethane foam (nominal density of 11.5 lbs/ft^3) and attached using ASTM A320, Grade L43 bolts.

The Model No. RH-TRU 72-B packaging is designed to transport payloads loaded into the payload canister or into inner steel containers within the payload canister. The payload canister is a cylinder (fixed or removable lid versions) fabricated of carbon or stainless steel as the outer shell. The removable lid canister may be utilized as a neutron shielded canister with a high-density polyethylene thermoplastic neutron shield insert in two configurations, NS15 and NS30, that provide neutron shielding for approximately 15- and 30-gallon inner steel containers, respectively.

Specifications and temperature dependent mechanical properties evaluated at the new enveloping temperature of 210 ºF , including yield strength, tensile strength, allowable strength, modulus of elasticity, and coefficient of thermal expansion conform to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB are presented in summary Table 2.6-1 of the application. Steel density is 0.283 lb/in^3, and Poisson's ratio is 0.3. Lead density is 0.41 lb/in^3, Poisson's ratio is 0.45, and the melting point is 620 ºF. The staff reviewed the materials selected and found them to be acceptable based on review of above standard and various engineering literature. The staff notes the applicant to be aware of possible superseding specifications.

2.2.1 Chemical or Galvanic Reactions:

Section 2.4 of the application discusses reactions due to chemical, galvanic or other reactions. The applicant provided the following information that the materials used to fabricate the package components (i.e., stainless steel, carbon steel, lead, and polyurethane foam) will not cause significant chemical, galvanic, or other reactions in air or water environments. These materials have been previously used in radioactive material packages for transport of similar chemical material without incident. In addition, successful use of these materials has been demonstrated in similar industrial environments.

The Model No. RH-TRU 72-B package is primarily constructed of Type 304 stainless steel. This material is highly corrosion-resistant to most environments. The actual metallic structure of the Model No. RH-TRU 72-B package is composed entirely of stainless steels and appropriate weld material. The weld material and processes have been selected in accordance with the ASME Code to provide as good or better material properties than the base material, including corrosion resistance. The various steels used also have approximately the same electrochemical potential, minimizing any galvanic corrosion that could occur.

The polyurethane foam used in the impact limiter of the Model No. RH-TRU 72-B package is the same as that which has been successfully used in a number of transportation packages. The polyurethane foam in the impact limiters is closed-cell foam that is very low in free halogens. The foam material is sealed inside a dry cavity in each impact limiter, to prevent exposure to the elements. If moisture were available for leaching trace chlorides from the foam, very little chloride would be available, since the material is closed cell foam and water does not penetrate the material to allow significant leaching.

The butyl rubber and other elastomers used in the O-ring seals contain no corrosives. These materials are organic in nature and non-corrosive to the stainless steel body of the Model No. RH-TRU 72-B package. In addition, the butyl rubber O-ring seal material was evaluated for chemical compatibility with identified payload chemicals. This evaluation determined that only a few payload constituents were potentially incompatible with butyl rubber under expected operating conditions. These compounds interact with butyl rubber by permeating and straining the polymer structure, causing the O-ring seal to swell. The combined effect of the very small quantities of incompatible solvents found in the payload, combined with the mechanism of attack on the butyl, indicate that the payload will not notably degrade the performance of the seals. Additionally, exposure to gamma radiation in the doses expected will not adversely affect the performance of the O-ring seals.

Corrosive materials are prohibited from the payloads. All payloads are contained directly in the payload canister or typically, confined within drums and liners placed in the canister. This configuration ensures that the chemistry of the payloads has very little interaction with the payload canister. However, the evaluation of compatibility is based on complete interaction of payload materials with the packaging. Since corrosives are prohibited from the payload, the only potential material that could be of concern is the release of free chlorides. Gaseous free chlorides could potentially be available only from the radiolysis of polyvinyl chloride and/or halogenated organic compounds within the payload. In both cases, the total quantity of gaseous free chlorides that would be available for diffusion into the IV is very small.

The staff concludes that, during normal conditions of transportation (NCT), the IV internals will not be subject to continuous or frequent exposure to moisture or that any moisture formation is not likely to occur in great quantities. The galvanic potential between the different metals used in fabrication is low. Therefore, the conditions required to create the possibility for galvanic corrosion is small.

In addition, visual inspections are to be performed on the accessible interior surfaces of the IV for indications of surface corrosion. Elevated non-destructive examination will be performed on IV interior surfaces including accessible shell, head, flange and weld surfaces in accordance with ASME Code should visual confirmation of corrosion exist. These visual inspections of the payload cavity at various timed intervals provide reasonable assurance against any significant corrosion occurring un-noticed.

Finally, non-reactive plastic threaded plugs are used to prevent moisture from entering the impact limiter cavities. And as noted in Section 8.2.4 of the application, O-ring seals and gaskets shall be replaced yearly or if they become impaired and new O-ring seals are to be leak rate tested per American National Standards Institute (ANSI) N14.5.

2.2.2 Brittle Fracture

Section 2.1.2.2 of the application discusses material brittle fracture concerns. The applicant provided that with the exception of the closure bolts and the canister, all containment and nearly all non-containment structural components, are fabricated of austenitic stainless steel. The closure bolts are fabricated from alloy steel in compliance with the applicable ASME Code brittle fracture requirements. Bolts are generally not considered as fracture critical components because multiple load paths exist and because bolted systems are designed to be redundant.

To comply with 10 CFR Part 71.71(b), the prototypical packaging was performance tested at minimum service temperature of -29ºC (-20ºF) to assure that no brittle fracture of structural containment components occurs. No observable differences in test results (e.g., drop test results) were noted for the low temperature test versus the normal ambient temperature test. Austenitic stainless steels (containment components) do not undergo a sharp ductile-to-brittle transition in the temperature range to -40 ºF. Therefore, the staff finds that low temperature has no reasonable detrimental influence on the Model No. RH-TRU 72-B packaging material performance.

2.2.3 Materials and Material Testing

Section 8.0 of the application discusses acceptance and maintenance testing. The metallic materials of construction (OC, IV, impact limiters and payload canister) are procured and fabricated to accepted industry standards. The various containment components are fabricated from ASTM standard materials. Welding is performed in accordance with ASME Code and inspected in accordance with the ASME Code and the American Welding Society D1.1 and D1.6 Welding Codes.

2.2.4 Conclusion

The staff finds that the Model No. RH-TRU 72-B transportation package meets the regulatory requirements for preventing or mitigating galvanic or chemical reactions, is unaffected by cold temperatures and is constructed with materials and processes in accordance with acceptable industry codes and standards.

2.3 Lifting and Tie-Down Standards for All Packages

The applicant revised the evaluation of the lifting and tie-down trunnions, the center-pivot trunnion, the OC and the attachment welds to account for the increase in temperature from 143 °F to 153 °F. The applicant obtained new stress allowable values for the increase in temperature through linear interpolation of the material properties between 100 °F and 200 °F.

Based on the applicant's evaluation, the staff concludes that the increase in the stress allowable values for temperatures from 143 °F to 153 °F has little effect on the performance of the outer shell, the lifting trunnions, the center pivot trunnion, the tie-down trunnions and the associated attachment welds, which all retained a positive margin of safety.

The staff reviewed the lifting and tie-down systems for the package and concludes that they meet the requirements of 10 CFR 71.45.

2.4 Normal Conditions of Transport (NCT)

2.4.1 Heat

The applicant's thermal analysis determined that the payload canister temperature reaches a maximum of 202 °F; therefore, the applicant changed the maximum temperature for analysis of the canister from 200 °F to 210 °F.

The applicant's evaluation determined that the maximum temperature difference between the OC end-closure plate and the OC inner and outer shells increased from 3 °F to 7 °F. The thermal gradient applicable for IV shell increased from 5 °F to 9 °F. The applicant determined the new secondary stresses associated with the revised temperature gradients, and combined the highest NCT secondary stress intensity with the highest NCT primary membrane-plus-bending stress intensity to determine the primary-plus-secondary stress intensity. The applicant assumed fully reversing stress states to determine the range of primary-plus-secondary stress intensity of 30,208 psi. When compared to the allowable ($3S_m$) of 60,000 psi, the margin of safety is 0.99 which is less than the previous value of 1.25. While the margin of safety has been reduced due to the updated thermal analysis, it is still positive; therefore, the staff finds that heat does not affect the effectiveness of the package.

2.4.2 Vibration and Fatigue

The applicant no longer determines the number of service cycles for the OC closure bolts and the IV closure bolts. The applicant stated that in the event of an unexpected failure of a bolt during installation, all bolts of the same type currently in use will be replaced prior to continued use of the packaging. Because of the redundant nature of the fasteners, and because all bolts of the same type will be changed following the failure of one, the staff concludes that this is a reasonable approach.

The applicant stated that the expected number of operating cycles (defined as the process of going from an empty package, to one with maximum heat load, at maximum normal operating temperature, and back again) for the Model No. RH-TRU 72-B package is below 10,000. The applicant's evaluation of fatigue at the normal operating temperature of 210 °F produces the same fatigue allowable alternating stress intensity amplitude as the 200 °F of Revision 6 of the SAR. The staff reviewed the applicant's fatigue calculations and concludes that the non-fatigue stress allowable criterion still governs and that fatigue does not reduce the effectiveness of the package.

2.4.3 Free Drop

The applicant revised the allowable stress intensity of the canister for the increase in temperature from 200 °F to 210 °F. The applicant's analysis determined that the reduced allowable stress had little effect on the margin of safety for the canister shell which remains positive.

The applicant performed a buckling evaluation using hand calculations in which they determined the critical buckling compressive stress to be 129,272 psi. Because this is well above the canister yield strength of 24,750 psi, the staff concludes that the canister is not susceptible to buckling.

The staff has reviewed the packaging structural performance under NCT and concludes that there will be no substantial reduction in the effectiveness of the packaging.

2.5 Hypothetical Accident Condition (HAC)

2.5.1 Thermal

The applicant increased the average through-wall temperature from 600 °F to 700 °F for their analysis of the pressure stress calculations as a result of the HAC fire evaluation. According to the applicant, this increased the total primary-plus-secondary stress intensity in the OC shell, which reduced the margin of safety for this accident condition from 1.31 to 0.70. Because the margins of safety remains positive, the staff finds the results of the temperature increase are acceptable.

The staff has reviewed the packaging structural performance under HAC and concludes the packaging has adequate structural integrity to satisfy the subcriticality, containment, shielding, and temperature requirements of 10 CFR Part 71.

2.6 Evaluation Findings

The staff reviewed the Model No. RH-TRU 72-B SAR as well as the RH-TRAMPAC document and RH-TRU Payload appendices. Based on the statements and representations contained in the application and the conditions given in the CoC, the staff concludes that the packages and payloads have been adequately described and evaluated to demonstrate their structural capabilities, including materials performance, to meet the 10 CFR Part 71 requirements.

3.0 THERMAL EVALUATION

The staff reviewed the Model No. RH-TRU 72-B shipping package application to verify that the thermal performance of the package has been adequately evaluated for the tests specified under NCT and HAC and that the package design satisfies the thermal requirements of 10 CFR Part 71. Staff also reviewed the application to determine whether the package fulfills the acceptance criteria listed in Chapter 3 of NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," as well as associated Interim Staff Guidance (ISG) documents.

3.1 Description of Thermal Design

The SAR states that the Model No. RH-TRU 72-B package dissipates the thermal loads entirely by passive heat transfer mechanisms. The primary heat transfer mechanisms utilized in the Model No. RH-TRU 72-B thermal analyses are component-to-component from conduction and radiation within the package, and free convection and radiation from the exterior of the package to the ambient. Due to the relatively close coupling of the bodies within the packaging, any convective heat transfer in spaces within the package is negligible. The maximum decay heat load is 270 thermal watts per payload canister. Summary tables of temperatures and maximum pressures in the containment system are included in the SAR. The thermal evaluation provided in the SAR includes analysis for the following three different boundary conditions for NCT:

1) Steady-state conditions at an ambient temperature of 100ºF without insolation in still air, as defined in 10 CFR 71.43(g), for limiting the maximum accessible package surface temperature to not exceed 185ºF for exclusive use shipments.

2) Transient conditions at an ambient temperature of 100ºF with insolation in still air, as defined in 10 CFR 71.71(c)(1).

3) Steady-state conditions at an ambient temperature of -40ºF without insolation in still air, as described in 10 CFR 71.71(c)(2).

Also, the thermal evaluation provided in the SAR includes the following sequence of events during HAC:

1) Transient, pre-fire conditions at an ambient temperature of 100ºF in still air, as defined in 10 CFR 71.73(b), with insolation.

2) Transient conditions in sufficiently quiescent ambient conditions to provide an average emissivity coefficient of at least 0.9 at an ambient temperature of 1475ºF, without insolation, for 30 minutes and with a package surface emissivity of at least 0.8 and an external convection coefficient based on a fire environment, as defined in 10 CFR 71.73(c)(4).

3) Transient, post-fire conditions at an ambient temperature of 100ºF with insolation in still air, as described in 10 CFR 71.73(c)(4), until maximum temperatures for all package components have been achieved.

As described above, the staff reviewed the applicant's description of its thermal design, specified thermal loads, and summary tables of temperatures and pressures and confirmed that it is consistent with the guidance specified in Chapter 3 of NUREG-1609. Therefore, the staff finds the description of the thermal design meets the thermal requirements of 10 CFR Part 71.

3.2 Material Properties and Component Specifications

Material property tables for the Model No. RH-TRU 72-B components are included in SAR Section 3.2. The applicant states that the materials present in the package include Type 304 stainless steel, carbon steel, lead, high-density polyethylene, polyurethane foam, payload material, and air. The staff reviewed the thermal properties provided in the SAR which include thermal conductivity, density, heat capacity, emissivity, and absorptivity. SAR Section 3.3 summarizes limiting temperatures for the Model No. RH-TRU 72-B components. Separate limiting temperatures are provided for NCT and the HAC fire. The staff reviewed the references provided by the applicant, supporting the material properties and confirmed that the properties were either determined experimentally or by referencing approved codes and standards [(ASME Boiler and Pressure Vessel Code, Section II (Materials), Part D (Properties)]. The staff reviewed the component specifications and verified that the structural analysis performed in SAR Chapter 2 demonstrates materials stresses are within acceptable limits for the design limiting temperatures for the Model No. RH-TRU 72-B components.

The staff has reviewed the material properties and component specifications used in the thermal evaluation and concludes that they are sufficient to provide a basis for evaluation of the package against the thermal requirements of 10 CFR Part 71.

3.3 Description of Model No. RH-TRU 72-B Thermal Model

To perform the thermal evaluation of the Model No. RH-TRU 72-B package the applicant developed a one-half symmetry three-dimensional (3-D) thermal model using Thermal Desktop with RadCAD, Version 5.6, and SINDA/FLUINT general purpose thermal/fluid network analyzer, Version 5.6. RadCAD allows radiation heat transfer analysis using one of two methods: progressive radiosity or Monte Carlo ray-tracing. Monte Carlo ray-tracing was used for its superior accuracy due to the relative complexity of the packaging configuration in the regions where radiation exchange occurs. The SINDA/FLUINT computer program is a general purpose code that handles problems defined in finite difference (i.e., lumped parameter) and/or finite element terms and can be used to compute the steady-state and transient behavior of the modeled system.

The applicant's 3-D half symmetry thermal model is shown in SAR Figures 3.4-1 through 3.4-8. These figures show the entire package and its contents and also individual components. A RLC is used as the thermal model for the Model No. RH-TRU 72-B waste canister, the NS15 (neutron shielding for 15-gallon drum) and the NS30 (neutron shielding for 30-gallon drum inner containers). Both radiation and air conduction are used to transfer heat across the radial gap between the RLC cylindrical shell and the IV cylindrical shell, and across radial gap between the RLC cylindrical shell and the two IV spacer rings. Similarly, both radiation and air conduction are used to transfer heat across the axial gap between the RLC base and the inside surface of the IV base, across the axial gap between the RLC lid plate and the inside surface of the IV lid, across the axial gap between the RLC pintle cap and the inside surface of the IV lid, and across axial gap between the RLC flange top and the inside surface of the IV lid.

The IV is modeled entirely of finite difference (FD) surface and solid elements. The entire IV assembly is stainless steel. The IV base, upper forging, lid, and center region of the IV spacers use FD solid elements. The IV O-ring seal temperature is determined using nodes positioned near the geometric location of the actual O-ring sealing region. Both radiation and air conduction are used to transfer heat across the radial gap between the IV cylindrical shell and the inside surface of the OC inner cylindrical shell. Similarly, both radiation and air conduction are used to transfer heat across the axial gaps between the outside surface of the IV base and the inside surface of the OC base, and between the outside surface of the IV lid and the inside surface of the OC lid.

The OC is modeled primarily of FD solid elements, with finite elements (FE) only used to transition between the trunnions and the cylindrical OC outer shell. With the exception of the lead shell, the entire OC assembly is stainless steel. The lead shell is assumed to be in direct contact at its inner and outer surfaces, both radially and axially. The thermal shield is directly connected at each end of the OC body and around the periphery of each trunnion penetration. Both radiation and conduction heat transfer are used within the gap between the inside of the cylindrical thermal shield and the outside of the outer OC cylindrical shell. The effective conductivity of the thermal shield gap is a composite of air and stainless steel wire.

The impact limiters are modeled entirely of FE solids and surfaces. With the exception of polyurethane foam filling the entire cavity, the entire structure is stainless steel. Since the thermal gradient across the impact limiter shell structures is not important, these components are modeled using a feature of Thermal Desktop that allows the free faces of FE solids to be surface coated with planar elements. These planar elements share the same nodes as the underlying FE solids and are defined by material, thickness, and optical properties for radiation heat transfer, and may also include surface heat loads and convection heat transfer parameters.

The staff reviewed the applicant’s developed thermal models, thermal properties, assumptions, and boundary conditions applied to the model, and based on the heat transfer characteristics of the analyzed geometry, the staff determined the analysis adequately captures the physics of the heat transfer problem posed by the applicant’s design.

The staff has reviewed the methods used in thermal evaluation and concludes that they are described in sufficient detail, consistent with Chapter 3 of NUREG-1609, to permit an independent review of the thermal design.

3.4 Thermal Evaluation under Normal Conditions of Transport

3.4.1 Heat and Cold

The applicant used the thermal model described earlier to perform the package thermal evaluation under NCT and subject to solar insolation. Table 3.4-3 of the application summarizes the NCT results. The maximum predicted temperature for exposure to 100°F are all below the allowable limit with sufficient margin.

The applicant stated that the Model No. RH-TRU 72-B package is an exclusive use shipment, such that the 10 CFR 71.43(g) requirement of 185°F maximum temperature at the accessible surface of the package is applicable. The thermal analysis performed by applicant shows that the maximum temperature of the accessible surface of the package is less than 118°F which satisfies the regulatory requirement.

The applicant stated that the minimum temperature distribution for the RH-TRU package will occur with no decay heat load and an ambient air temperature of -40ºF per 10 CFR 71.71(c)(2)1. The applicant assumed that all packaging components would reach the -40ºF temperature under steady state conditions. For a package with the thermal capacitances of the RH-TRU package, prolonged exposure to low temperature environments is required to significantly depress package temperatures. The applicant stated that the -40ºF temperature is within the package's allowable temperature limits for the temperature sensitive components according to FSAR Section 3.3, Technical Specifications of Components.

3.4.2 Maximum Normal Operating Pressure

The applicant stated that the pressure within the sealed IV of the Model No. RH-TRU 72-B package may potentially be changed (increased or decreased) due to one or more of the following factors: radiolytic gas generation (or consumption), temperature-related pressure change, barometric pressure change, chemical reactions, biological gas generation, and/or thermal decomposition. The applicant stated that based on the thermal loads and materials for construction of the package, the major factors affecting the IV internal pressure are radiolytic gas generation, thermal expansion of gases, and the vapor pressure of water within the IV cavity. The applicant calculated each component separately and added them up to obtain the maximum normal operating pressure of the IV. The applicant calculated a maximum IV pressure at the end of a shipment for a bounding maximum thermal load of 270 watts of 8.995 psig. The SAR states that the maximum design pressure of the Model No. RH-TRU 72-B package is 138.8 psig (when the maximum barometric pressure change is included). The applicant's calculated pressure is well below the maximum design pressure. The applicant used standard accepted methods to calculate the maximum normal operating pressure and the staff finds the methods and calculations acceptable.

3.4.3 Maximum Thermal Stresses

The maximum thermal stresses during NCT event are discussed in Chapter 2 of the SAR. These maximum thermal stresses are evaluated in Chapter 2 of the safety evaluation report.

The staff has reviewed the package design, construction, and preparation for shipment and concludes that the package material and component temperatures will not exceed the specified allowable limits during NCT consistent with the tests specified in 10 CFR 71.71.

3.5 Thermal Evaluation under Hypothetical Accident Conditions

3.5.1 Initial Conditions

The applicant states that the initial temperature distribution within the package prior to the HAC fire is based on an ambient temperature of 100ºF in still air, as defined in 10 CFR 71.73(b), and includes the "12-hour on/12-hour off" insolation cycle that was used for NCT. The pre-fire transient analysis is continued for sufficient time until a pseudo-steady state condition is achieved (600 hours, i.e., 25 days). The staff confirmed that all temperatures used as initial conditions are at their maximum values, per NCT analysis. The thermal model used to evaluate the HAC fire event is essentially identical to the model developed for NCT. The notable exception is the application of 30-foot free drop and 1-meter puncture drop damage to the impact limiters. In addition, the HAC thermal model addresses the effect of foam thermal degradation and resulting external char layer. Damage from these two HAC drop scenarios and char reduction of the polyurethane foam are included in a single model, with an HAC analysis performed for each of the four payload configurations.

3.5.2 Fire Test Conditions

The applicant used the Model No. RH-TRU 72-B thermal model described earlier to determine the temperature of the package during the fire accident specified in 10 CFR Part 71. A 30 minute, 1475°F fully engulfing fire was simulated followed by a sufficient cooling period to ensure that maximum local temperatures are reached.

A transient calculation was performed with heating from a constant 1475°F fire for 30 minutes followed by cooling. The calculation started from the temperature profile obtained for NCT with insolation. No solar insolation was applied in the transient calculation during the heating phase of the fire test; however, insolation was applied during the cooling phase following the fire. Established correlations for natural convection were again used to derive the appropriate convection coefficient, as described in the SAR.

3.5.3 Maximum Temperatures and Pressures

The maximum temperatures experienced by the components of the Model No. RH-TRU 72-B package calculated by the applicant under a HAC fire event described earlier are provided in Table 3.5-2 (after end of fire) and Table 3.5-3 (post-fire) of the SAR. The staff confirmed that all the temperatures predicted during the HAC are below the component allowable limits. The applicant calculated a maximum pressure of 16.54 psig based on a calculated air averaged temperature of 334.25°F during HAC. This calculated pressure is well below the pressure of 300 psig (used in Chapter 2 of the SAR to calculate wall stresses), and is therefore acceptable.

3.5.4 Maximum Thermal Stresses

The maximum thermal stresses during HAC fire event are addressed in Chapter 2 of the SAR. These maximum thermal stresses are evaluated in Chapter 2 of the safety evaluation report.

The staff has reviewed the package design, construction, and preparation for shipment and concludes that the package material and component temperatures will not exceed the specified allowable short time limits during HAC consistent with the tests specified in 10 CFR 71.73.

3.6 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the Model No. RH-TRU 72-B transportation package thermal design has been adequately described and evaluated, and that the thermal performance of the package meets the thermal requirements of 10 CFR Part 71.

5.0 SHIELDING EVALUATION

The purpose of the shielding review is to verify that the Model No. RH-TRU 72-B package design meets the external radiation limits prescribed in 10 CFR Part 71 under NCT and HAC. The Model No. RH-TRU 72-B package is designed to transport RH-TRU materials for the DOE. The allowable payload materials are determined by the method set forth in the RH-TRAMPAC.

Specifically, the activity limits are determined using the method described in Section 5.5.4 of the SAR.

Since the Model No. RH-TRU 72-B is a waste package, its contents are varied and cannot be concisely defined. The applicant performed analyses for sources with the applicable range of energies for concentrated and distributed geometries and created a table of maximum activity for each energy. The procedure in Section 5.5.4 of the SAR instructs the user to determine all of the nuclides and associated gamma and neutron energies and sum the fractions at each energy bin to determine if the package is in compliance with regulatory dose rate limits.

The staff reviewed the shielding design and method using the guidance in Section 5 of NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material."

5.1 Description of Shielding Design

5.1.1 Design Features

The Model No. RH-TRU 72-B consists of an OC and a separate IV. The waste will be further placed within a payload canister. Users can load the radioactive contents directly into the RH-TRU waste canister, or place the contents into a smaller container and load up to three smaller containers (30 or 55 gallon drums) into the payload canister.

The payload canister has two different configurations. One is the fixed lid configuration (Drawing X-106-501-SNP, Rev. 5) and the other is a removable lid configuration (Drawing X-106-502-SNP, Rev. 3). The removable lid configuration includes the option to use two neutron shielded canisters: the NS15 and NS30 (Drawing X-106-503-SNP, Rev. 1). These canisters are nearly identical except the NS30 has a wider internal diameter and therefore less neutron shielding. The neutron shielded canisters (NS15 and NS30) must be loaded with 3 or fewer inner canisters (15 or 30 gallon drums). If fewer than three canisters are loaded into the RH-TRU waste canister or the neutron shielded canisters, the user must use dunnage to position the drums. This is discussed in Sections 2.8.1 and 2.8.2 of the RH-TRAMPAC document.

The package shielding consists of 1 7/8 inches of lead and 2 ½ inches of steel in the radial direction. The bottom of the package has 6 ½ inches of steel and a minimum of 12 ½ inches of steel at the top. The RH-TRU waste canister has an additional ¼ inches of carbon steel or stainless steel. The canisters equipped with neutron shields contain high density polyethylene with a minimum thickness in the radial direction of 3.288 inches for the NS15 and 1.412 inches for the NS30, and both canisters have a top and bottom end cap that is nominally 5 inches thick.

5.1.2 Summary Table of Maximum Radiation Levels

The applicant included a summary table of maximum radiation levels for a concentrated Co-60 source for gamma dose and a concentrated Cf-252 source for neutron dose for each of the 3 analyzed canister configurations (RH-TRU waste canister, NS15 and NS30 neutron shielded canisters) in Tables 5.1-1 through 5.1-3 of the SAR. The applicant evaluated the maximum amount of Co-60 and Cf-252 that would meet regulatory dose rate limits. Although the package is authorized to ship nuclides other than these, due to the varied nature of the waste content, the applicant chose representative nuclides for these summary tables of maximum radiation levels. The limiting dose rate is at 2 meters, which is very close to the regulatory limit of 10 mrem/hr with all other evaluated dose rates below their respective limits.

For all other energies and configurations (i.e. distributed source) the applicant evaluated the dose rate on a per particle basis and then scaled up the allowable contents so that the dose rate would meet the regulatory limit at 2 meters. The applicant modeled discrete gamma and neutron energies in the distributed and concentrated source configurations for each of the three canisters to come up with maximum activity limits for each of these configurations.

The applicant provided evaluations (ADAMS Accession No. ML15054A110, ML15054A111) demonstrating that the dose rate limit at 2 meters is the most challenging to meet and therefore by evaluating limits at this location for both gamma and neutrons at all energies, the surface and HAC dose rate limits will be met. Based on the applicant's evaluation the staff found this acceptable.

The results of the evaluations performed per particle at each energy are shown in Tables 5.4-4 through 5.4-6 and in Tables 5.5-2 through 5.5-5 of the SAR. The staff notes that these tables state the allowable particles/sec as "allowable activity." As activity for a specific nuclide is defined as "decays per time" (i.e. decays/second), activity in this terminology is meant to express particles per second as a nuclide typically has more than one particle per decay. In this SER as well as the applicant's SAR, expression of "activity" hereafter refers to particles/sec (either gamma/sec or neutrons/sec).

The staff verified that the applicant correctly calculated the allowable activity by verifying that the calculated dose rate per particle plus the Monte Carlo N-Particle (MCNP) tally uncertainty multiplied by the allowable activity did not exceed 10 mrem/hr. All of the applicant's results were at 10 mrem/hr. Although there is no margin to the limit, the applicant does apply a 10% administrative margin in the loading procedures in Section 5.5.4 of the SAR.

5.2 Radiation Source

The radiation source is RH-TRU materials for the DOE. The RH-TRAMPAC document discusses the requirements and acceptable methods for preparing and characterizing a RH-TRU payload. From a radiation protection perspective, it is important to know the isotopic composition and quantity. This is discussed in Section 3.0 of the RH-TRAMPAC. Section 5.5.4 of the SAR has the procedure for establishing maximum allowable activity limits. These sections (Section 3.0 of the RH-TRAMPAC and Section 5.5.4 of the SAR) discuss fissile limits as well as gamma and neutron emission rate limits but do not mention alpha or beta radiation. The waste material will contain nuclides that have alpha and/or beta emissions, however these emissions are stopped by the Model No. RH-TRU 72-B packaging material and can therefore be transmitted in unlimited quantity as the Model No. RH-TRU 72-B has been demonstrated to meet Transportation Category I Fracture toughness requirements per Regulatory Guide 7.11. Realistically the maximum allowable alpha and beta emitting nuclides will be limited by the decay heat requirements as well as other limitations discussed within the RH-TRAMPAC document.

The RH-TRU SAR discusses limits for concentrated (very little credit for self-shielding) and distributed gamma sources with energies ranging up to 10 MeV and concentrated and distributed neutron sources with energies up to 15 MeV.

5.3 Shielding Model

The staff reviewed the structural and thermal sections of the SAR and found that conditions consistent with NCT and HAC were appropriately represented in the shielding model. The results of the NCT tests specified in 10 CFR Part 71.71 result in negligible damage to the package that is confined to the impact limiter and NCT shielding models are consistent with the SAR drawings. The HAC model is similar to the NCT model with the exception that the lead thickness is reduced by 0.656 inches to account for puncture damage. Results of the thermal evaluation to address the fully engulfing fire in 10 CFR 71.73(c)(4) show that the internal temperatures of the cask do not reach those that would cause the high density polyethylene to melt and therefore assuming the neutron shielding during HAC is appropriate. The applicant assumed no lead slump, and this was consistent with results from the 30-foot drop evaluations in the structural evaluation (Section 2.6.7.1(9) of the SAR).

<u>5.3.1 Source and Shielding Configuration</u>

The staff examined the dimensions of the RH-TRU 72-B package and three canisters in the sample MCNP input files in Section 5.7.1 of the SAR. The staff verified that the dimensions were consistent with the drawings presented in Section 1.3 of the SAR.

5.3.1.1 Source Geometry and Self-Shielding

The applicant has activity limits for two source configurations, a concentrated and distributed source. The concentrated sources are represented as a 1 inch diameter and 1 inch height cylinder using zirconium at a density of 1 g/cm^3. Since this is a small size compared to actual intended RH-TRU waste contents and 1 g/cm^3 is a fictitiously low density for zirconium, and zirconium was demonstrated to have the least amount of attenuation for typical RH-TRU contents with a constant density (see discussion in Section 5.3.2 of this SER) the staff finds that this is a reasonable representation of a concentrated source geometry for this package.

The applicant divided the source material into three equal sources and modeled it within three drums, such that each drum contained 1/3 of the total allowable activity. The concentrated sources are centered within each of three drums. For the distributed sources, the applicant homogeneously distributed the sources in a right circular cylinder. The drums are stacked vertically and placed within a canister centered within the payload cavity. The size of the source depends on the size of the available drums and the maximum allowable weight of the contents.

5.3.1.2 Source concentration and relocation

The applicant determined the package limits in Tables 5.5-2 through 5.5-7 of the SAR by assuming that the source was uniformly divided and centered into three drums. This is not a conservative assumption as the package canister could be directly loaded and source material concentrated in a single location, or drums could be loaded with unequal amounts of activity. In addition sources could shift or settle toward the side of the drum resulting in the possibility of increased dose rates. To determine the activity limit per drum (or per package for a direct loaded canister with a concentrated source), the applicant performed an evaluation of the increase in dose rates assuming all of the activity was concentrated into a single point source. The applicant discusses this evaluation in Section 5.3.1 of the SAR.

The applicant evaluated a point source at the center of a single drum containing the entire allowable activity for the package. The applicant determined that assuming all of the activity was concentrated into a single point produced similar dose rates at 2 meters but that the surface dose rate increased by almost a factor of two. Both surface and 2 meter dose rates exceeded the regulatory limits in this configuration. Therefore the applicant established a limit that no more than half of the allowable package activity is allowed in a single drum. Since packages are administratively limited to a loading fraction of 0.9 of the allowable limit, half of this is 0.45 of the allowable limit. This analysis is not comprehensive, given it is only for a single energy, and does not investigate all situations such as, for example, 2 drums, each loaded with half of the allowable activity. Recognizing this, the staff still finds that this limit is acceptable based on the following considerations. The staff considered the conservatisms within the analysis such as the 10% administrative margin on loading, evaluating dose rates at 2 meters from the package, rather than the vehicle surface, the inclusion of comprehensive measurement procedures and the review of submitted measurement data showing that the procedure for evaluating activity limits is conservative with respect to previous shipments (see Section 5.3.1.4 of this SER).

5.3.1.3 Concentrated versus Distributed Sources

In the loading procedure in Section 5.4.4 of the SAR, the applicant divides the loading into two "Cases." These two Cases are Case A, drums overpacked within a payload canister and Case B, waste direct-loaded into a payload canister. For both of these cases waste in this configuration can be treated as "distributed" (i.e. utilizing the "distributed" limits within Tables 5.5-2 through 5.5-7) if the contents meet the definition of "distributed throughout" from NUREG-1608. For Case A, this means all drums in the package must meet this definition. Case B only applies to the RH-TRU waste canister as the neutron shielded canisters (the NS15 and NS30) require the contents to be loaded into drums before being loaded into the waste canister.

5.3.1.3.1 Distributed Throughout

The applicant references the definition of "distributed throughout" from NUREG-1608, "Categorizing and Transporting Low Specific Activity Materials and Surface Contaminated· Objects." The definition from Section 4.2.3 in NUREG-1608 is as follows:

"For distributed throughout, the material can be divided into ten or more equal volumes. The volume of each portion should be no greater than 0.1 m³. The specific activity of each volume should then be assessed (through measurements, calculations, or process knowledge) and compared. Specific activity differences between any two volumes should not vary by more than a factor of 10."

NUREG-1608 applies to low specific activity (LSA) material. Therefore for higher specific activity waste, such as RH-TRU waste which will be shipped within the Model No. RH-TRU 72-B, the applicant provided no justification that the possibility of having a portion of the waste with a difference in specific activity as much as a factor of 10 would not cause external dose rates to increase beyond that of regulatory limits should it re-configure. In addition, as this definition was created with LSA material in mind, there was no need to consider the specific activity of the material and without qualifying either the specific activity or the volume of the material, this definition could easily be misinterpreted. For example, a highly concentrated 1 mm x 4 mm Co-60 source could meet the definition as "distributed throughout." Further NUREG-1608 states that "*For small (i.e., smaller than 0.2 m³ (7.5 ft³), or a 55 gal. drum) LSA material packages, the IAEA Safety Series No. 37 method described above should not be applied."* This last sentence implies that this definition would be inappropriate for the 55 gallon drums typically transported within the Model No. RH-TRU 72-B.

With these considerations the staff still accepts this definition for qualifying the Model No. RH-TRU 72-B contents as distributed for both drums (Case A) and direct load (Case B). For direct load into a canister (Case B), source geometry effects can be substantial and therefore the applicant limited the allowable source by the volume fraction that the waste occupies within the canister. For example a half full canister is allowed half of the package activity limit, and an 80% full canister is allowed 80% of the activity limit. This is consistent with the analyses performed in Chapter 5 of the SAR. For waste loaded into drums, the applicant does not have a requirement to reduce the allowable source to the volume fraction of the waste. The staff found this acceptable because increases in external dose rates due to source compression for waste essentially occupying a third of the package (as in one drum out of three) are far less than that of waste occupying the entire canister (as in the Case B of the direct load). In addition limiting allowable activity to 0.45 of the established limits already reduces the allowable activity per loaded volume to some extent.

The staff recognizes that the above paragraph does not specifically address all identified shortcomings of using the definition of “distributed throughout,” the staff factored other considerations into determining acceptability of this definition. The staff considered the conservatisms within the analysis such as the 10% administrative margin on loading, evaluating dose rates at 2 meters from the package rather than the vehicle surface, the inclusion of comprehensive measurement procedures and the review of submitted measurement data showing that the procedure for evaluating activity limits is conservative with respect to previous shipments (see Section 5.3.1.4 of this SER).

5.3.1.4 Measurement Data

The staff notes that it does not accept pre-shipment measurement as a means for determining compliance with NCT dose rate limits in 10 CFR 71.47, however the staff recognizes that they can assist in quantifying uncertainties. As part of CoC Revision 6, the RH-TRU 72-B has incorporated comprehensive measurement procedures in Section 3.2 of the RH-TRAMPAC, Revision 3. The package operations incorporate this procedure by reference in Section 7.1.2.22. Since Chapter 7 of the SAR is incorporated by reference into the CoC along with the RH-TRAMPAC revision number, the comprehensive measurement procedure is a condition of the certificate. Also in support of CoC Revision 6, the applicant submitted data from past shipments.

To address many of the uncertainties discussed throughout this SER, such as the uncertainty related to activity distribution and content movement during transport, the applicant submitted measurement data from past shipments. The applicant submitted updated data similar to that submitted during the review of Revision 6 of the CoC (ML15212A770.)

Although the applicant stated that there were 230 past shipments, they provided details for 14 of the shipments that had the highest 2-meter dose rate fraction (greater than or equal to 0.25). When using measurement alone to determine if packages meet regulatory dose rate limits, the applicant stated that none of the shipments resulted in a reportable condition of exceeding regulatory dose rates before, during or after shipment. If following the new procedure (in the current application) for determining maximum allowable activity, 8 shipments would have been precluded from shipment. The data submitted by the applicant shows that using the method of determining activity limits within the SAR, packages with activity fractions near or exceeding the new limits still had a measured dose rate that was within the regulatory limits, and often it was well below the limit.

The data indicates that there can be substantial differences between pre- and post-shipment measurements indicating possible content shift during transport. The applicant states that *“on average, post-shipment surveys showed a 36% increase over pre-shipment surveys”* in discussing the measurement data from previous Model No. RH-TRU 72-B shipments. They further stated that *"the significance of these changes can be attributed to a combination of items such as measurement uncertainties, potential shifting of source material during transport, and variations in the implementation of pre- and post-shipment surveys."* The staff notes that although there was a 36% increase on average, the highest surface dose increase was almost 3 times the pre-shipment surface dose rate. And although the 2 meter dose rate is limiting for this package, the applicant did not provide post shipment 2 meter dose rate, and the staff could not verify if there was an increase. Therefore, not knowing the exact cause of the increase in dose rates, the staff finds that the reconfiguration potential could be great for this package. The staff finds that implementing the concentrated versus distributed source limits will provide reasonable assurance to prevent external dose rate increase beyond regulatory limits due to relocation.

Overall the applicant's data gives the staff confidence that the method within the SAR for determining activity limits is conservative when compared to using pre-shipment measurement alone.

5.3.1.5 Hypothetical Accident Conditions (HAC)

Under HAC the applicant models the source as a single point source on the side of the package. Consistent with the information in Appendix 5.1 of the RH-TRU payload appendices, the applicant assumes that 98% of the source stays inside the package and 2 percent leaves the package which is an order of magnitude more than what was observed in the testing. The HAC assumptions are consistent which those reviewed by the staff during Revision 6 of the CoC and the staff finds them acceptable.

5.3.1 6 Streaming

The applicant did not specifically address streaming however the staff does not find streaming to be an issue for this package. This is due to the cask having a double lid as well as the waste being placed within a canister.

5.3.2 Material Properties

Materials used in the cask shielding evaluation are presented in Section 5.3.2 of the SAR. The staff verified that these materials were appropriately represented in the MCNP input files and that these properties are consistent or bounded by those in the package drawings in Section 1.3.1 of the SAR.

The waste medium for evaluating self-shielding is highly varied so it was not practical for the applicant to evaluate all possible materials. Therefore the applicant analyzed a conservative medium at a unit density and created density correction factors when shipping materials of different density.

Both the concentrated and distributed sources are modeled using zirconium. The applicant submitted the results of an evaluation (ML15054A110 and ML15054A111) showing that Zirconium had representative gamma attenuation characteristics of materials that are to be shipped in the RH-TRU. In response to staff observation 5-9 (ML15054A110 and ML15054A111), the applicant shows that the choice of zirconium has no measurable effect on neutron attenuation and therefore the staff finds that this is an acceptable choice to represent the TRU waste contents for both gamma and neutron sources.

The applicant modeled the density of the self-shielding material (Zirconium) in a range from 0.1 to 8 g/cc. Although these are mostly fictitious densities for Zirconium, the applicant demonstrated that the attenuation properties of Zirconium at different densities is equivalent to that of other Model No. RH-TRU 72-B materials at these densities.

The applicant used this density range to calculate a density correction factor (DCF) that they apply to "distributed" source loadings. The allowable activity from SAR Tables 5.5-2 through 5.5-6 is multiplied by the DCF based on the density of the contents. The DCF is the ratio of the maximum activity at a density other than 1 g/cm^3 to the maximum activity for the source with a density of 1 g/cm^3. Information on how the DCFs were created was supplied in response to staff RSI 5-3 (ML15054A110 and ML15054A111.) The staff reviewed this information and found that the DCF was evaluated in a conservative manor and finds its use acceptable for accounting for the differing payload densities for distributed contents.

5.4 Shielding Evaluation

5.4.1 Methods

The applicant performed shielding calculations with MCNP5, version 1.60. MCNP is a three dimensional Monte Carlo transport code developed and maintained by Los Alamos National Laboratory. The code's capabilities include modeling of and determining dose rates from package design features where radiation streaming may be a concern. This code is used extensively for shielding calculations by industry. Given this and the code's capabilities and its extensive application in industry (ensuring the code is well-vetted), the staff found the code acceptable for use in the present application.

5.4.2 Input and Output Data

The applicant provided input files for the MCNP calculations used to determine the maximum radioactivity of the contents in Section 5.7.1 of the SAR. Section 5.7.2 of the SAR contains representative output files for these input files. Staff reviewed the sample input files and found that the information regarding material properties and dimensions used in the calculations is consistent with descriptions of the calculations given in the application. The staff reviewed the output files and ensured that the calculations had achieved proper convergence and that the calculated radiation levels agree with those reported in the application.

5.4.3 Flux-to-Dose-Rate Conversion

The applicant used conversion factors that were derived from the ANSI/ANS 6.1.1-1977 standard for both gammas and neutrons. These are shown in Tables 5.4.2 and 5.4.3 of the SAR. This is the standard that staff finds acceptable (as discussed in NUREG-1609) and therefore the staff finds them acceptable for use in this application. The conversion factors used in the input files are consistent with those described in the application.

5.4.4 External Radiation Levels

The applicant evaluated the dose rates at the radial side of the package. The applicant did not evaluate dose rates at the top and bottom and instead states in response to staff Observation 5-7 (ML15054A110 and ML15054A111) that under NCT this location was chosen due to the extra distance attenuation from the impact limiters. The staff found the maximum impact limiter crush under NCT in Section 2.6.7.2 of the SAR where it says the maximum crush to the impact limiter corner is 9.68 inches. The staff performed an independent scoping calculation to see if this was enough distance attenuation for dose rates to be lower than on the side of the package. The staff used Microshield 9.06 and compared the dose rate at the side surface of the package to that of the top surface plus the distance occupied by the remaining impact limiter. Based on the dimensions of the impact limiter from drawing X-106-500-SNP Rev. 6, the staff conservatively assumed that the entire 9.68 inches was crushed in the vertical direction leaving approximately 13 inches of distance attenuation left to credit. The results of the staff's evaluation showed that the additional 13 inches is still enough for dose rates to be higher on the side surface and therefore found that this is an appropriate location for performing dose rate evaluations under NCT.

For HAC, the applicant used similar logic – utilizing the distance attenuation provided by the impact limiters -- to justify the location of the dose rate tallies. Under HAC the staff finds it inappropriate to credit the distance occupied by the impact limiter. Section 2.7.1.2 of the SAR states that the maximum crush to the impact limiter due to HAC corner/oblique drops is 23.7 inches. In addition, Section 3.5.2 of the SAR estimated a char depth of 3.3 inches to the impact limiter due to the HAC fire event. Given these two effects, the staff does not have confidence that there will be enough distance remaining from the damaged impact limiter to provide any significant distance attenuation. However for HAC, the applicant does reduce the thickness of the lead shield by 0.656 inches in the radial direction to account for the puncture damage (Section 5.3.1.1 of the SAR). The staff's evaluations show that when accounting for this reduction in lead thickness that the dose rates at the side are higher (i.e. more limiting) under HAC when compared to the top of the package with no impact limiter and therefore finds evaluating HAC dose rates at the side acceptable.

To create the maximum allowable activity tables (Tables 5.5-2 through 5.5-7 of the SAR), the applicant calculated the maximum activity that would not exceed the 2 meter NCT dose rate limit. The applicant did not evaluate dose rates for the other regulated locations (surface or HAC). In response to Staff RSI 5-1 (ML15054A110 and ML15054A111) and Staff RAI 5-1 (ML15212A770), the applicant submitted results of the evaluations performed that determined that the 2 meter NCT dose rate limit was the most limiting (i.e. highest) for gammas and neutrons in the 3 canisters for the entire energy range over the other regulatory locations.
The staff verified the tally surfaces from the MCNP input files. The applicant evaluates the dose rate using the surface average tally (F2 tally) in MCNP. The input file shows that the tally surface is divided into sections including a 2.54 cm radius cylinder's intersection with the 1 meter Z-axis cylinder at the height of the source for HAC and for NCT they subdivided the cylindrical surface by planes at the central source location with a height of 5 cm. It is the staff's judgment that the size of these cells are sufficiently small to capture the highest dose rate.

5.5 Evaluation Findings

Based on its review of the statements and representations in the application and independent evaluations, the staff has reasonable assurance that the shielding design has been adequately described and evaluated and that the package meets the external radiation requirements of 10 CFR Part 71.

6.0 Criticality Evaluation

DOE submitted an SAR documenting analysis and testing to support the authorized use of the RH-TRU 72-B waste shipping package to include heterogeneous LEU payloads. The applicant intends to show in the SAR that the package continues to meet the safety requirements with the additional authorized contents, which the applicant calls “Class E.” The objective of this review is to verify the applicant’s findings that the RH-TRU 72-B meets the requirements of 10 CFR 71.

6.1 Areas of Review

The applicable regulations considered in the review of the criticality safety portion of this application include the fissile material requirements in 10 CFR Part 71, specifically the general requirements for fissile material packages in 10 CFR 71.55 and the standards for arrays of fissile material packages in 10 CFR 71.59. The staff also used the review guidance contained in NUREG-1609, “Standard Review Plan for Transportation Packages for Radioactive Material.”

Staff reviewed the information in the SAR and verified that the information is consistent, as well as that all descriptions, drawings, figures and tables are sufficiently detailed to support an in-depth staff evaluation. The staff evaluated the package design to determine if the package meets the criticality safety requirements of 10 CFR 71.

6.1.1 Packaging and Design Features

Packaging design features are identical to the previous revision.

6.1.2 Summary Table of Criticality Evaluations

Case	Homogeneous LEU 0.96 wt% max ^{235}U FEM	Heterogeneous LEU 0.84 wt% max ^{235}U FEM
Normal Conditions of Transport (NCT)		
Single Unit Maximum k_s	Bound by HAC single unit k_s	
Infinite Array Maximum k_s	Bound by HAC infinite array k_s	
Hypothetical Accident Conditions (HAC)		
Single Unit Maximum k_s	0.9180	0.9295
Infinite Array Maximum k_s	0.9224	0.9316
Upper Subcritical Limit (USL)	0.9257	0.9323

6.1.3 Criticality Safety Index (CSI)

The applicant demonstrated that an infinite array of RH-TRU 72-B packages with the most reactive contents in both NCT and HAC remains adequately subcritical. Therefore the CSI is 0.0 in accordance with 10 CFR 71.59(b).

6.2 Fissile Material Contents

The applicant is requesting one additional form of waste material to be authorized for shipment with the RH-TRU 72-B package, “Class E” heterogeneous LEU. The LEU contents are restricted to material that is primarily uranium, in terms of the heavy metal component. The applicant provided conversion factors for calculating the contribution of other fissile isotopes to the % ^{235}U fissile equivalent mass (FEM) in the RH-TRAMPAC.

The material requirements of “Class D” LEU homogeneous waste remain unchanged.

“Class E” LEU heterogeneous contents do not have the particle/lump size limitations of “Class D” LEU homogeneous waste. However, the following criteria must be met:

- The waste matrix material is distributed within the canister such that the maximum enrichment does not exceed 0.84 wt% ^{235}U FEM in any location.
- There are no variations in the fissile concentration that exceed 0.84 wt% ^{235}U FEM in any location.
- Total contents net weight shall not exceed 3,100 lbs.

6.3 General Considerations

6.3.1 Model Configuration

The applicant split the RH-TRU 72-B model into two evaluations, contents and packaging. The packaging uses dimensions that have previously been reviewed and considered to be conservative. The applicant models the package dimensions as they occur under HAC, particularly the impact limiter. The applicant states that, since the inner and outer vessel and contents sustain little damage, and the expected minor variations in these components has little effect on system reactivity, the evaluation under HAC accounts for both HAC and NCT.

The applicant’s evaluation varies the composition, shape and position of the contents in the computational model to verify maximum system reactivity.

6.3.2 Material Properties

The applicant modeled the fissile material as either cylinders (rods) or spheres of 0.84 wt% ^{235}U. The remaining void space within the IV is filled with a moderating mixture of 25% polyethylene, 74% water and 1% beryllium. The applicant selected this moderator composition as a result of prior scoping calculations for evaluation of “Class D” homogeneous LEU waste.

The applicant makes an additional calculation with a reflector material that contains beryllium, up to 75% by volume. The applicant shows that the most reactive axial reflector composition for an array is 50% beryllium, 25% water and 25% polyethylene.

The applicant ignores the mass of the moderating and reflecting material when modeling the contents at maximum net weight, with the exception of additional axial beryllium reflector, which is included with fissile material in the total net weight of the contents.

6.3.3 Computer Codes and Cross-Section Libraries

The applicant used the KENO-V.a module in the SCALE 5.1 code suite driven by the CSAS25 utility for criticality analysis. KENO-V.a is a three-dimensional Monte Carlo transport code. The cross section library used by the applicant was the 238-group library based on the ENDF/B-V nuclear data. Both this software and cross-section library have been previously evaluated and found to be appropriate for use in this application.

6.3.4 Demonstration of Maximum Reactivity

In single package evaluations, the applicant used the optimum diameter of 1.25 cm for arrays of both spheres and cylinders. The applicant varied the pitch and demonstrated that the optimum spherical element pitch is 1.569 cm, which results in an $H/^{235}U$ ratio of 469, and the optimum cylindrical element pitch is 2.0287 cm with a corresponding H/^{235}U ratio of 397. The IV and interstitial array space is flooded with the 74/25/1% mixture of water, polyethylene and beryllium. The applicant models the package as fully reflected by water. The most reactive configuration resulted when the applicant modeled the array at the center of the IV.

The applicant's analysis included several scenarios which varied the location of the fissile array within the IV. The applicant modeled an infinite, square array of packages with radially and axially reflective boundary conditions. A series of calculations were done with both spheres and cylinders. The applicant showed their most reactive case resulted when the array was placed in the top of the IV. The applicant's most reactive infinite array model is one where the system is reflected and moderated by the 50/25/25 beryllium, water and polyethylene mixture with a 16 cm axial reflector, the interstitial moderator is minimized (0.01% by volume), and optimum pitch is 2.7273 cm.

Tables 6.4-15 through 6.4-20 in the SAR show the results of the "Case E" calculations and sufficiently demonstrate the configuration of maximum reactivity.

6.3.5 Confirmatory Analysis

Staff, using the same assumptions as the applicant, analyzed the RH-TRU 72-B package using KENO-VI in the SCALE 6.1 suite with a continuous energy ENDF/B-7 cross-section library. The geometric configuration assumed by the applicant was retained, and all other dimensions used in the analysis were obtained from the SAR.

Staff varied the same model parameters as the applicant to determine maximum reactivity: pitch; reflection, including special reflector materials; interstitial moderator density; and array location. Staff results were in close agreement with those of the applicant for both a single package and arrays under HAC. The conditions that maximized reactivity in the staff's analysis matched those of the applicant.

6.4 Single Package Evaluation

6.4.1 Configuration

The applicant considered only HAC for the package configuration, which is conservative as the applicant evaluates water in-leakage. In all cases, the package is fully reflected with a close-fitting full density water reflector that is 12 inches (30.5 cm) thick. The single package model does not include the impact limiter, and instead considers the space to be occupied by full density water.

The applicant modeled the interstitial space between the IV wall and the fissile array as a mix of beryllium, polyethylene and water, in addition to varying the axial reflector thickness. While the applicant ignores the mass of the radial beryllium reflector, additional axial beryllium replaces fissile mass in the modeled system.

The applicant varies the location of the fissile contents to determine optimum k_{eff} for the system.

6.4.2 Results

The most reactive configuration models the optimum spherical fissile element diameter and pitch in a cylindrical array. The most reactive configuration results in a system k_{eff} of 0.9218 after consideration of bias and uncertainty.

The applicant demonstrates that a fully flooded and reflected package will remain subcritical.

6.5 Evaluation of Package Arrays under NCT

The applicant considered the package array configuration under HAC to account for package arrays under NCT. Staff finds this acceptable since the applicant's assumptions under HAC are more conservative.

6.6 Evaluation of Package Arrays under HAC

6.6.1 Configuration

All of the applicant's array analyses are performed with a reflective (i.e. mirror) boundary condition, effectively modeling an infinite array of packages in a close-packed, square-pitch configuration. The applicant reduced the impact limiter thickness by half to simulate damage under HAC. This reduced spacing bounds the analysis under NCT. Additionally, the applicant varies the interspersed moderator density between packages to further maximize system reactivity.

The applicant models the contents as arrays of uranium metal (0.84 wt% ^{235}U) both as a cylindrical array of cylinders (i.e. rods) and a cylindrical array of spheres. Staff finds that square pitched arrays are sufficiently representative of credible configurations. The only physical limitation placed on the system is the total mass of fissile and moderating mixture does not exceed a net weight of 3,100 lbs.

The applicant modifies the position of the contents sphere or cylinder to maximize interaction between packages.

6.6.2 Results

The applicant demonstrates that, except at trace quantities, the addition of hydrogenous material between the drums reduces reactivity. The applicant demonstrated that the most reactive scenario is where the fissile material is modeled as a reflected array cylinders, with the cavity flooded with a 75/25/1% mixture of water, polyethylene and beryllium, respectively. The most reactive reflector material is a 25/25/50% mixture of water, polyethylene and beryllium, but the applicant's calculated reflected system k_{eff} is lower than that of the unreflected system. Both mixtures have a hydrogen density above that of water. In the most reactive case, the maximum system k_{eff} considering bias and uncertainty is 0.9316. The applicant shows that the system will remain acceptably subcritical under HAC.

6.7 Benchmark Evaluations

The applicant used the USLSTATS code from Oak Ridge National Laboratory. This software has been developed for use with the SCALE suite and ENDF/B cross-section libraries. USLSTATS accounts for the combined effects of code bias, uncertainty in the benchmark experiments, uncertainty in the computational evaluation of the benchmarks, and an administrative margin of subcriticality. A package is determined to be adequately subcritical if the computed value of k_{eff} plus two standard deviations is below the upper subcritical limit (USL).

6.7.1 Experiments and Applicability

The applicant evaluated 82 benchmarks that included low enriched uranium experiments that included both uranium solutions and arrays of moderated uranium oxide rods. The moderators evaluated in the benchmarks included both water and polyethylene. The applicant revised the SCALE 4.4 version of these benchmarks to run with SCALE 5.1 and the ENDF/B-V cross-section library. The enrichment of the system being evaluated falls outside of that of the LEU benchmark cases. The applicant extrapolated the USL to the respective LEU enrichments for comparison with other USLs based on trending parameters.

The applicant identified two trending parameters for determination of the bias for LEU evaluations with SCALE 5.1, $H/^{235}U$ ratio and ^{235}U enrichment. The applicant trended the data against two parameters to determine the resulting bias, and found that the trend for the enrichment was more limiting. The applicant conservatively used the more limiting USL.

In the “Case E” analysis with unlimited special reflector material (i.e. beryllium), the applicant used energy of average lethargy causing fission (EALF) as an independent parameter to determine similarity to the plutonium benchmarks that contain beryllium. The applicant also used EALF for the single LEU beryllium case since the systems lacked similarity.

The benchmark experiments are listed in Table 6.5-6 of the SAR with the applicant’s observed trending parameters and computed results.

6.7.2 Bias Determination

USLSTATS fits an empirical trend line to the data based on the parameter.

The applicant’s resulting USL for trending against the $H/^{235}U$ ratio was higher than the USLSTATS result for enrichment and was conservatively discarded.

The results of the USLSTATS evaluations are presented in Table 6.5-7 of the SAR.

6.8 Evaluation Findings

Based on review of the statements and representations in the application, the staff concludes that the nuclear criticality safety design of the RH-TRU 72-B package has been adequately described and evaluated and that the package meets the criticality safety requirements of 10 CFR Part 71.

7.0 OPERATING PROCEDURES

The applicant revised the operating procedures to allow for the option of lid alignment pins to not be present during transport. Additionally, the wording for rendering impact limiter lift devices inoperable was revised to be consistent with note 40 of the licensing drawings. Additionally, the applicant made several changes to the pre-shipment leakage rate test section.

7.1 Pre-shipment Leakage rate test

DOE mandated specific leak test system

The applicant described the pre-shipment leakage rate test in SAR section 7.4.1 and stated in their request for additional information (RAI) response that the use of the specific leak test system for the pre-shipment leakage rate test is mandated in the DOE RH-Packaging Operations Manual (DOE/WIPP 02-3284).

The applicant noted that: (1) the specific leak test system uses a semi-automatic leak test equipment and performs a “self-check” of the system prior to each pre-shipment leak test to confirm that the system, including valves, is operating properly, and (2) implementation of the DOE mandated specific leak test system ensures that closure of the vacuum pump isolation valve effectively isolates the system such that the additional steps outlined in RAI 7-1(a) are not required to ensure an accurate leak test.

The staff reviewed the DOE mandated specific leak test system described in their response to NRC’s request for additional information (RAI) and accepts that: (1) the DOE mandated specific leak test system can properly implement a self-check of the system, and (2) the vacuum pump valve effectively closes and isolates the system for an accurate leak testing. The staff accepted that the steps of the pre-shipment leakage rate test, listed in SAR section 7.4.1, are acceptable for a pre-shipment leak test.

Test Time Required for Pre-shipment Leakage Rate Test

The applicant described the pre-shipment leakage rate test in SAR section 7.4.1 and stated in RAI response that actual test conditions that are different from the baseline pressure assumptions require an adjustment to the test time based on the standard conditions (1 atm and 25°C).

The applicant presented the adjusted equation, for determining the test period required for the pre-shipment leak testing, in step 9 of SAR 7.4.1.2 and provided derivation of the equation in the RAI response. The applicant used the formula in ANSI N14.5-2104 and derived the equation by assuming that (1) the test/gas temperature remains unchanged because the test duration is very short and the test volume is very small; and (2) the molecular flow has a relatively small effect on the leakage rate when compared to the continuum flow.

The staff reviewed the derivation provided in the RAI response, ensured that the derivation is consistent with the steps given in ANSI N14.5-2014, and accepted the assumptions used in the derivation. The staff accepted the equation shown in step 9 of SAR section 7.4.1.2 for a pre-shipment leak test with a relatively small test volume and a very short test duration.

Evaluation Findings

After review, the staff determined that the changes in SAR section 7.4.1 satisfy the thermal and containment requirements, in compliance with 10 CFR Part 71. The staff has reviewed the changes to the operating procedures and concludes that they do not affect the ability of the package to meet 10 CFR Part 71.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The applicant updated the acceptance tests to the 2014 version of ANSI N14.5 standard. Additionally, the component names for the IV surface were revised to be consistent. Previously, the component names were an inadvertent carryover from other package terminology. Additionally, a requirement that non-conformances in the structural and pressure tests be recorded and be dealt with in accordance to the quality assurance program has been added.

The maintenance program has been updated to require replacement of all fasteners of a given type in the event that any bolt of that type was to fail during installation. Previously, the replacement requirements were cyclic.

Staff reviewed these changes to the acceptance tests and maintenance program and concludes that they do not affect the ability of the package to meet 10 CFR Part 71.

9.0 QUALITY ASSURANCE

The quality assurance program has been revised to reference the currently applicable DOE Order 460.1C instead of the previous 460.1B. Additionally, a procurement in the list of activities performed under a quality assurance program verified to satisfy 10 CFR Part 71, Subpart H, has been included for completeness.

CONDITIONS

The following updates have been made to the CoC:

Condition No. 3.b., “Title and Identification of Report or Application,” and the “References” section have been updated to reflect the new consolidated SAR.

Condition No. 5.(a)(2), “Description,” has been updated to describe the potential for two levels of containment of the package.

Condition No. 5.(a)(3), “Drawings,” has been updated to reflect NWP as the owner and the latest revision numbers of the drawings.

Condition No. 5.(b)(2), “Contents,” has been updated to include discussion on the properties of the low enriched uranium. It has also been updated to specify that radioactive material containing gamma or neutron emissions must also meet the activity limits (in particles/sec) determined by the procedure in Section 5.5.4 of the application. Additionally, it has been updated to include the new decay heat limits.

Condition No. 13 has been updated to allow for use of the previous revision of the CoC for up to one year.

Condition No. 15 has been updated to reflect the new expiration date.

Revision numbers for the RH-TRAMPAC and RH-TRU Payload Appendices documents have been revised to reflect the latest revision throughout the CoC.

The references section has been updated to include this request.

CONCLUSION

Based on the statements contained in the application, and the conditions listed above, the staff concludes that the changes indicated do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

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