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

June 17, 2011

Mr. Todd Sellmer
Packaging Integration
Washington TRU Solutions LLC
P.O. Box 2078
Carlsbad, New Mexico 88221-2078

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

Dear Mr. Sellmer:

As requested by your application dated February 12, 2010, as supplemented April 19, 2010, August 30, 2010, February 16, 2011, and April 14, 2011, enclosed is the Certificate of Compliance No. 9212, Revision No. 6, for the Model No. RH-TRU 72-B package. The staff's Safety Evaluation Report is also enclosed.

Washington TRU Solutions LLC is registered as user of the package under the general license provisions of 10 CFR 71.17. Those on the attached list have been registered as users of the package under the general license provisions of 10 CFR 71.17 or 49 CFR 173.471. 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.

The staff notes that a shielding calculation basis is traditionally needed to demonstrate that normal condition of transport (NCT) dose rates can be achieved in accordance with 10 CFR 71.35(a) and 71.47. However as discussed in the SER for this approval, you provided a well-defined characterization of the contents, a rigorous pre-shipment dose measurement program, and data regarding shielding performance and dose rates from past shipments. Therefore, for this specific amendment request, the staff has reasonable assurance that the requirements of 71.35(a) are satisfied. Depending on the nature of the change, the staff notes however, that future certification requests may require an evaluation with an appropriate shielding calculation to demonstrate compliance with NCT requirements. Additional guidance to staff on this matter can be found in NUREG-1609, "Standard Review Plan for Transportation Packages of Radioactive Material."

If you have any questions regarding this certificate, please contact me or Jennie Rankin of my staff at (301) 492-3268.

Sincerely,

MD Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9212
TAC No. L24419

Enclosures:
1. Certificate of Compliance No. 9212, Rev. No. 6
2. Safety Evaluation Report
3. Registered Users

cc w/encls 1 & 2: R. Boyle, Department of Transportation
J. Shuler, Department of Energy
Registered Users

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9212	6	71-9212	USA/9212/B(M)F-96	1 OF	4

PREAMBLE

2.

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Department of Energy
Washington, DC 20585

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Washington TRU Solutions LLC consolidated application dated February 12, 2010, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No: RH-TRU 72-B

(2) Description

A stainless steel, lead-shielded cask designed to provide double containment for shipment of transuranic waste materials. The packaging consists of a cylindrical stainless steel and lead cask body, a separate inner stainless steel vessel, and foam-filled impact limiters at each end of the cask body.

The cask body (outer cask) consists of a 1 1/2-inch thick, 41 5/8-inch outer diameter stainless steel outer shell, and a 1-inch thick, 32 3/8-inch inside diameter stainless steel inner shell, with 1 7/8 inches of lead shielding between the two shells. The cask bottom is 5-inch thick stainless steel plate. The cask is closed by a 6-inch thick stainless steel lid, and 18, 1 1/4-inch diameter bolts. The main closure lid has a double bore-type O-ring seal. The containment seal is the inner butyl O-ring seal, which is leak testable. The cask lid has a single vent/sampling port that is sealed with leak testable butyl O-ring seals.

The separate inner vessel consists of a 3/8-inch thick, 32-inch outside diameter stainless steel shell, and a 1 1/2-inch thick stainless steel bottom plate. The inner vessel is closed by a 6 1/2-inch thick stainless steel lid, and eight, 7/8-inch diameter bolts. The inner vessel closure lid has three bore-type O-ring seals. The containment seal is the middle butyl O-ring seal, which is leak testable. The inner vessel lid has a helium backfill port and a combination vent/sampling port that are sealed with leak-testable butyl O-ring seals.

5.(a) (2) Description (Continued)

A polyurethane foam-filled stainless steel impact limiter is attached to each end of the cask body using six, 1 1/4-inch diameter bolts. The radioactive contents are packaged within a stainless or carbon steel waste canister that is placed in the inner vessel.

The approximate dimensions and weights of the package are as follows:

Overall package length	187 3/4 inches
Impact limiter diameter	76 inches
Cask length	141 3/4 inches
Cask outer diameter (OD)	41 5/8 inches
Inner vessel length	130 inches
Inner vessel OD	32 inches
Cask lead shield thickness	1 7/8 inches
Maximum package weight (including contents)	45,000 pounds
Maximum weight of contents (including waste canister)	8,000 pounds

(3) Drawings

The packaging is constructed and assembled in accordance with AREVA Federal Services LLC, Drawing No. X-106-500-SNP, sheets 1-8, Rev. 5.

The fixed lid waste canister is constructed and assembled in accordance with Packaging Technology Drawing No. X-106-501-SNP, Rev. 4. The removable lid waste canister is constructed and assembled in accordance with Packaging Technology Drawing No. X-106-502-SNP, Rev. 2. The neutron shielded waste canister is constructed and assembled in accordance with AREVA Federal Services LLC, Drawing No. X-106-503-SNP, Rev. 0.

(b) Contents

(1) Type and form of material

Byproduct, source, and special nuclear material in the form of dewatered, solid or solidified materials and waste, within the stainless or carbon steel waste canister described in Item 5(a)(3). Explosives, corrosives (pH less than 2 or greater than 12.5), and compressed gases are prohibited. Within a waste canister radioactive and non-radioactive pyrophorics must not exceed 1 weight percent. Flammable volatile organics are limited along with hydrogen to ensure the absence of flammable gas mixtures in RH-TRU waste payloads as described in RH-TRAMPAC (Rev. 1).

(2) Maximum quantity of material per package.

Not to exceed 8,000 pounds, including the weight of the waste canister.

5.(b) (2) Contents (Continued)

Fissile material not to exceed limits described in Section 3.1, "Nuclear Criticality" of RH-TRAMPAC (Rev. 1). Pu-239 equivalent is determined in accordance with RH-TRAMPAC (Rev. 1). Low enriched uranium is authorized for waste containers containing material that is primarily uranium (in terms of heavy metal component) and the waste matrix material must be distributed within the canister in such a manner that the maximum enrichment does not exceed 0.96% uranium (U-235) fissile equivalent mass in any location of the waste material.

Maximum decay heat per package not to exceed 50 watts per canister for all payloads in accordance with RH-TRAMPAC (Rev. 1).

(c) Criticality Safety Index: 0.0

6. Waste content codes and classification, physical form, chemical properties, chemical compatibility, gas generation, fissile content, decay heat, isotopic inventory, weight, and radiation dose rate must be determined and limited in accordance with RH-TRAMPAC (Rev. 1).

7. Each waste canister must not exceed the decay heat limits determined as specified in RH-TRAMPAC (Rev. 1), or must be tested for gas generation in accordance with RH-TRAMPAC (Rev. 1), Section 5.0, "Gas Generation Requirements."

8. A RH-TRU waste canister may be comprised of inner containers with different content codes provided that the hydrogen gas generation rate limit or decay heat limit for all of the inner containers within the payload is assumed to be the same as the content code with the lowest hydrogen gas generation rate limit or decay heat limit.

9. The waste canister and any sealed secondary containers greater than 4 liters in size overpacked in the waste canister must be vented in accordance with the minimum specifications in Section 2.4, Filter Vents, of RH-TRAMPAC (Rev. 1).

10. Shipments must not exceed the 60 or 10 day maximum shipping period requirement specified in RH-TRU Payload Appendices (Rev. 1), Sections 2.3 and 2.4, respectively.

11. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each package must be prepared for shipment and operated in accordance with the procedures described in Chapter 7.0, "Operating Procedures," of the application, as supplemented.

(b) Each packaging must be tested and maintained in accordance with the procedures described in Chapter 8.0, "Acceptance Tests and Maintenance Program," of the application, as supplemented.

(c) Each containment O-ring seal material formulation and each batch of containment O-ring seal material must be qualified and tested in accordance with the procedures described in Section 3.6.4 of RH-TRU 72-B (Rev. 5).

12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

13 Packages must be marked with Package Identification Number USA/9212/B(M)F-96.

14. Revision No. 5 of this certificate may be used until June 30, 2012.

15. This package may not be used for transport by aircraft.

16. This package is for exclusive use shipments.

17. Expiration date: February 28, 2015.

REFERENCES

Washington TRU Solutions LLC consolidated application dated February 12, 2010.

Washington TRU Solutions LLC supplements dated April 19, 2010; August 30, 2010; February 16, 2011; April 14, 2011.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

M D Waters

Michael D. Waters, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: June 17, 2011

SAFETY EVALUATION REPORT

Model No. RH-TRU 72-B Package
Docket No. 71-9212
Certificate of Compliance No. 9212
Revision No. 6

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

Safety Evaluation Report
Model No. RH-TRU 72-B Package
Docket No. 71-9212
Certificate of Compliance No. 9212
Revision No. 6

1.0 SUMMARY

By application dated February 12, 2010, as supplemented April 19, 2010, August 30, 2010, February 16, 2011, and April 14, 2011, Washington TRU Solutions LLC submitted to the U.S. Nuclear Regulatory Commission (NRC), an amendment request for the Model No. RH-TRU 72-B package. The application requested approval of the following revisions to the Safety Analysis Report (SAR):

- Addition of new authorized payload canisters.
- Changes to design drawings for a new payload assembly configuration.
- Creation of appendices for document supporting information.
- Revision to qualification testing for containment O-ring seals.
- Revisions to facilitate packaging fleet repair and maintenance activities, enhance operational effectiveness and provide for implementation of improvements.

Based on the statements and representation in the application, as supplemented, and the staff's review and interpretation of the applicable requirements, the staff concludes the revision to the Certificate of Compliance (CoC) meets the requirements of 10 CFR Part 71. Accordingly, the CoC has been updated to include the revisions requested in the application as approved in this Safety Evaluation Report (SER).

1.1 Packaging

RH-TRU 72-B is a shipping container for transporting remote-handled transuranic wastes by truck and rail transport. The packaging is composed of an inner vessel (IV) that optionally provides an inner containment boundary, an outer cask (OC) that provides an outer containment boundary and acts as an environmental barrier, and energy-absorbing impact limiters at each end of the OC. All RH-TRU waste will be loaded directly into the payload canister or into inner containers within the payload canister. The payload canister is basically a one- or two-piece construction unit (fixed or removable lid versions). The removable lid canister may be utilized as a neutron shielded canister in two configurations, NS15 and NS30, which incorporate neutron shielding components that provide neutron shielding for approximately 15 and 30 gallon inner containers, respectively. The RH-TRAMPAC and RH-TRU Payload Appendices have been revised to define the payload controls necessary for the Neutron Shielded Canister and summarize use of the Neutron Shielded Canister with the RH-TRU 72-B packaging.

Revisions have been made to provide an improved safety basis by rigorously defining O-ring seal material qualification and acceptance requirements. Revisions have also been made to allow lifting the OC and IV lids with a handling fixture in lieu of solely using eyebolts.

The package is constructed and assembled in accordance with AREVA Federal Services LLC, Drawing No. X-106-500-SNP, Rev. 5, sheets 1 through 8. The fixed lid waste canister is constructed and assembled in accordance with Packaging Technology Drawing No. X-106-501-SNP, Rev. 4. The removable lid waste canister is constructed and assembled in accordance with Packaging Technology Drawing No. X-106-502-SNP, Rev. 2. The neutron shielded waste canister is constructed and assembled in accordance with AREVA Federal Services LLC, Drawing No. X-106-503-SNP, Rev. 0.

2.0 STRUCTURAL AND MATERIALS EVALUATION

The application (SAR, Revision 5) seeks the addition of new authorized payload canisters (Neutron Shielded Canister, NS15 and NS30 designs) that incorporate internal neutron shielding, for shipment in the RH-TRU 72-B packaging. The RH-TRU 72-B SAR has been revised to incorporate NS15 and NS30 into the RH-TRU 72-B packaging design basis. It also adds a drawing into the SAR for the new payload assembly configuration. The application incorporates revisions to establish comprehensive qualification test requirements for containment O-ring seal materials. Although RH-TRU 72-B was primarily designed to ship waste with high gamma source terms, when NS15 and NS30 canisters are employed, the packaging system is designed to ship transuranic waste forms with high neutron source terms.

As illustrated in Figure 5.1-1 of the RH-TRU Payload Appendices, Rev. 1, the NS30 neutron shielded canister is a 26-inch diameter, 120-1/2" tall removable lid canister with a High-Density Polyethylene (HDPE) neutron shield insert, weighing approximately 1,660 pounds empty and having a maximum gross weight of 3,100 pounds. It is designed to carry three approximately 30-gallon steel payload drums with internal lever-lock closures, where each drum and its contents are limited to approximately 480 pounds. The NS30 neutron shielded canister incorporates the NS30 neutron shield insert which consists of HDPE top and bottom end caps (nominally 5-inch thick) with an integral open-cell urethane filter/gasket and a 24-inch outside diameter cylindrical pipe body (nominally 1.454 inches thick). The minimum thickness of HDPE neutron shielding provided by the body is 1.412 inches.

As illustrated in Figure 5.1-2 of the RH-TRU Payload Appendices, the NS15 neutron shielded canister is essentially identical to the NS30 with appropriate modification to accommodate a thicker 24-inch outside diameter cylindrical pipe body (nominally 3.387 inches thick) with 3.288 inches minimum thickness and an approximate tare weight of 2,070 pounds and maximum gross weight of 3,100 pounds. The NS15 is designed to carry three approximately 15-gallon steel payload drums with internal lever lock closures, where each drum and its contents are limited to approximately 337 pounds.

Both the NS30 and NS15 neutron shielded canisters incorporate HDPE end caps that are slip-fit into the HDPE pipe body. The end caps are engaged and remain captured within the pipe body through a 2-inch deep end cap step and the limited (1/2-inch maximum) axial geometric clearance of the shield assembly within the surrounding canister structure. An optional, 1/4-inch to 1-inch thick, 24-inch to 24-1/2-inch diameter disc of CDX grade plywood may be located above the top end cap or below the bottom end cap to ensure the maximum axial clearance is maintained. Both end caps incorporate lifting D-rings utilized to facilitate handling during remote operations. In addition to the respective payload drums, the neutron shielded canisters may optionally contain a mesh "bag" used to facilitate remote installation of each payload drum into the shield insert assembly. The neutron shielded canister must be installed with a filter vent; Section 2.4 of the RH-TRAMPAC (Rev. 1) provides the minimum specification for the neutron shielded canister filter vent.

The packaging is composed of an IV that optionally provides an inner containment boundary and an OC that provides an outer containment boundary and acts as an environmental barrier. The IV consists of a 1½-inch thick bottom plate, a 3/8-inch thick, 32-inch outside diameter shell, a lid end forging, and a 6½-inch thick lid. The OC consists of a 5-inch thick bottom plate, a 1-inch thick, $32^{3}/_{8}$-inch inside diameter inner shell, a lid-end forging, and a 6-inch thick lid. The RH-TRU 72-B package schematic is presented in Figure 1.1-1 of the SAR, Rev. 5.

The impact limiter skins are fabricated from 300 series stainless steel. Non-stainless members include the closed-cell polyurethane foam and the American Society of Testing and Materials (ASTM) A320, Grade L43, impact limiter attachment bolts. The polyurethane foam has a nominal density of 11½ pounds per cubic feet. Six (6), 1¼-7UNC bolts necked down to a 1-inch diameter secure each impact limiter to the OC. The outside diameter of each impact limiter is 76 inches and the length is 46 inches. The approximate weight of each impact limiter is 2,547 pounds.

All RH-TRU waste will be loaded directly into the payload canister or into inner containers within the payload canister. The payload canister uses a 26-inch outside diameter, $^{1}/_{4}$-inch thin-wall cylinder fabricated of carbon or stainless steel as the outer shell. Including a lift pintle at the top of the payload canister, the overall length is 120½ inches. The payload canister is basically a one- or two-piece construction unit (fixed or removable lid versions, respectively, either of which may be configured with a through-pintle fill port and plug) capable of transporting RH-TRU waste.

The removable lid canister may be utilized as a neutron shielded canister in two configurations, NS15 and NS30, as discussed above. Gross shipping weight of the RH-TRU 72-B package is 45,000 pounds, maximum.

Tie-down Devices

Of the eight trunnions located on the exterior of the package, four are intended for lifting (four trunnions near the lid-end) and six for tie-down (two pivot trunnions, two trunnions near the lid, and the two trunnions near the base). The two trunnions nearest the base of the package and two of the four near the lid of the package are also used to stabilize the package during loading and unloading. A more detailed discussion of the package lifting and tie-down features is provided in Section 2.5 of the SAR, Rev. 5, "Lifting and Tie-down Standards for All Packages." Both the OC and the IV are fitted with four-point handling fixture holes and a pintle receptacle where a pintle can be installed. The lids can be lifted with standard eyebolts and four-point rigging and/or handling fixture. The lid can also be lifted using the pintle grapple that is used for handling the payload canister.

A description of the neutron shielded canister structural evaluation is provided in Appendix 5.1 of the *RH-TRU Payload Appendices*.

FEA Models

The computer program CASKDROP was used to demonstrate compliance of the package with 10 CFR 71.71(c)(7) and 10 CFR 71.73(c)(1) for normal conditions of transport (NCT) and hypothetical accident conditions (HAC) free drop analyses, respectively. Appendix 2.10.2.1 of the SAR, Rev. 5, describes the CASKDROP analysis methodology, and provides an example problem with input and output.

The package is protected by polyurethane foam-filled energy absorbing end buffers, called impact limiters. For purposes of the regulatory free drop analyses using the CASKDROP computer program, the impact limiters are assumed to absorb, in plastic deformation of the polyurethane foam, all of the potential energy of the drop event. In other words, the drop analyses assume that none of the potential energy of the free drop event is transferred to kinetic or strain energy of the target (i.e., the "unyielding" surface assumption of 10 CFR Part 71), nor strain energy in the package body itself.

CASKDROP evaluates all angles of drop from 0° (horizontal) to 90° (vertical) by performing a quasi-static analysis that ignores rotational effects. At orientations where rotational effects are important, use of a dynamic analysis computer program such as SCANS or SLAPDOWN was required, utilizing the force-deflection data developed by CASKDROP. Three orientations where rotational motions (or pitch) play no role in the evaluation of the free drop analyses were:

- END DROP on the circular end surface of the impact limiter,
- SIDE DROP on the cylindrical side surfaces of the impact limiters, and
- CORNER DROP with the package center of gravity directly over the impact limiter corner.

For all orientations of impact, the prediction of impact limiter deformation behavior was approached from energy balance principles.

Impact limiter deflections and package accelerations were calculated using the Sandia National Laboratories-developed computer code SLAPDOWN. This program solves the rigid-body equations of motion for a transportation package, given parameters such as weight, rotational moment of inertia, geometric relationships, and impact limiter force-deflection curves. As further confirmation of the SLAPDOWN code analysis methodology, the sample problem was compared to output from the public domain program SCANS. The results compared well and were demonstrated in Table 2.10.2-6 of the SAR, Rev. 5. Input data for SLAPDOWN is provided in Table 2.10.2-2 of the SAR, Rev. 5. A 15° initial impact angle was used for direct comparison.

The RH-TRU 72-B impact limiters were tested in half-scale, 30-foot free drop tests. Two test articles were tested in end, side, and 55° from horizontal oblique orientations.

A NS30 neutron shielded canister was assembled into a neutron shielded canister test fixture, configured to simulate the RH-TRU 72-B packaging and a bounding deceleration environment, to facilitate hot and cold drops of the canister while the test fixture is at ambient conditions. The reusable test fixture and two NS30 neutron shielded canisters were each subjected to two 30 ft. free drops, one canister under hot and one canister under cold temperature conditions, onto a flat, essentially unyielding, horizontal surface. Only the NS30 neutron shielded canister was physically tested as the NS15 design is less limiting from a drop perspective.

2.1 Structural Design

2.1.1 Design Criteria

The structural design criteria are developed to assure that the transport system has adequate structural strength to meet the requirements of NCT as well as HAC. These criteria are designated as those that affect the containment boundary and those that affect other package structures which contribute to the overall structural performance. The containment boundary is

evaluated based on the ASME code requirements for level A and D service and is consistent with Regulatory Guide 7.6. The remaining design requirements address the shield cylinders, impact limiter deformation, as well as impact limiter attachment.

2.1.1.1 Loading and Load Combinations

Loads and Load Combinations are evaluated using Regulatory Guide 7.6, 10 CFR Part 71 for NCT and HAC loads, and ANSI N14.6 for handling loads.

2.1.1.2 Acceptance Criteria

The acceptance criteria established for the package are for the containment boundary, shielding components, and the impact limiters.

The containment boundary must meet the stress intensity limits of Subsection NB of the ASME Code for design pressure under service level A conditions. The containment boundary must meet sealing performance requirements under the Free Drop event as well as satisfy the ASME Code limits for Section III, service level A and D stress intensity limits for the respective drop heights. Under a penetration event, the containment boundary must not be breached, remain leak tight, and service level D stress intensity limits must be satisfied away from the point of impact. The containment boundary materials must not be susceptible to brittle fracture. Closure lid seals must remain functional under all events to maintain a sufficient seal such that established leak rates are not exceeded.

With damages sustained from NCT and HAC, the radiation doses must remain below the regulatory limit. Brittle fracture damage resulting in through-thickness cracks, causing a loss in shielding function, is not permitted.

The impact limiters must perform impact limiting functions such that ASME Section III, Subsection NB stress limits are satisfied for the applicable service condition. The impact limiters must remain permanently attached to the package during HAC sequence and have adequate crush characteristics to prevent bottoming out of the cask body. Decelerations under the 30-foot drop must be limited such that the peak flexural plastic strains do not exceed the failure strain and joints with gaskets in the containment boundary must remain fully functional.

2.1.2 Weights and Centers of Gravity

The SAR summarizes several content configurations with varying weights and center of gravity locations.

2.1.3 Codes and Standards

The Codes and Standards utilized for package design are identified in Section 2.1.2 of the SAR, Rev. 5. Regulatory Guide 7.6 was used for all package containment boundaries for both NCT and the HAC. Material data used in the evaluation correspond to the design stress values, Sm, yield strengths, Sy, and ultimate strengths, Su, given in the ASME Code. The allowable stresses used for containment structures and fasteners are consistent with Regulatory Guide 7.6 and Section NB- 3000 and Appendix F of the ASME Code, Section III.
Structural evaluations of non-containment boundaries, such as the OC outer shell, and tie-down and center-pivot trunnions, use allowable stresses for NCT and HAC as presented in Table 2.1-2 of the SAR, Rev. 5. The impact limiters are allowed to exceed yield for all conditions. The

acceptance criterion for all impact related loads within the impact limiters is that no package "hard points" directly come into contact with the impact surface. Adherence to this criterion is assured by designing the impact limiters so that no impact limiter deformation for any regulatory drop event ever exceeds the maximum strain level achieved in polyurethane foam crush tests. Foam samples have been tested to strains of up to 87%. Static testing of prototypic, half-scale RH-TRU 72-B package impact limiters imposed strains of up to 97% (for 35° from vertical orientation, see Section 2.10.7 (appendix) of SAR, Rev. 5, *Static and Dynamic Testing*). A strain limit of 85% is imposed for the polyurethane foam. Requirements for foam mechanical properties are presented in Section 2.3 of SAR, Rev. 5, *Mechanical Properties of Materials*. For lifting and handling loads, the "non-containment" NCT allowable of Table 2.1-2 of SAR, Rev. 5, are utilized in conjunction with a load factor of three (3), per 10 CFR 71.45.

2.2 Materials Evaluation

The 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 RH-TRU 72-B packaging materials.

Structural components are primarily fabricated from 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, RR0405-70, 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 1-1/4-7UNC, ASTM A320, Grade L43 bolts.

The 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 HDPE 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, 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, 1986 Edition are presented in Tables 2.3-1 through 2.3-3 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°F. The staff reviewed the materials selected and found them to be acceptable.

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 (RAM) packages for transport of similar chemical material without incident. In addition, successful use of these materials has been demonstrated in similar industrial environments.

The 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 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 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 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 NCT, the IV internals will not be subject to continuous or frequent exposure to moisture and 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. Visual inspections (prior to use and annually thereafter) of the payload cavity 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. As noted in Section 8.2.4 of the application, O-ring seals and gaskets shall be replaced yearly or if they become impaired. New O-ring seals are to be leak rate tested per 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 71.71(b), the prototypical packaging was performance tested at a minimum service temperature of -29°C (-20°F) 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°C (-40°F). Therefore, the staff finds that low temperature has no reasonable detrimental influence on the RH-TRU 72-B packaging material performance.

2.2.3 Fatigue

The fatigue evaluation is limited to fatigue on the bolts that is caused by applying the required pre-load and by the shock and vibration loading during transportation of the package. Pressurization and thermal loads do not significantly contribute to fatigue loading, because of their magnitude and long period. The allowable life of the package component is the number of cycles given on the design fatigue curve in Figure 1-9.2.2 (from Appendix 1 of the ASME Code), corresponding to the maximum corrected value of the fatigue stress (S_{alt}). The value of S_{alt} is corrected by the ratio of the modulus of elasticity given on the design fatigue curve to the modulus of elasticity of the component material used in the analysis. Operational procedures specify replacement of the OC closure bolts prior to the completion of 460 service cycles, and replacement of the IV closure bolts prior to 550 service cycles.

Normal operating cycles do not present a fatigue concern for the RH-TRU 72-B package components which have no stress concentrations. This is because the allowable stress for NCT (3.0Sm) will not exceed the allowable fatigue stress limit for the expected number of operating cycles. From Table I-1.2 in Appendix I of the ASME Code, the maximum value of the stress intensity (Sm) at the maximum normal operating temperature of 200°F, is 20,000 psi for the Type 304 stainless steel used in the RH-TRU 72-B package. The expected number of operating cycles (defined as the process of going from an empty package, to one with maximum heat load, at the maximum normal operating temperature, and back again) for the RH-TRU 72-B package is below 10,000.

2.2.4 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 Code 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.

2.2.5 Conclusion

The staff finds that the 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 Fabrication and Examination

2.3.1 Fabrication

The staff has evaluated the applicant's fabrication materials regarding the following seven criteria.

1) Tolerances are achievable.
2) Design is not overly reliant on stringent tolerances.
3) Weldability of dissimilar materials.
4) Specification of post heat treatment or other residual stress relief.
5) Manufacturing sequence must allow for unimpeded Non-Destructive Evaluation as well as remedial repairs.
6) Manufacturing sequence must allow for unimpeded access to relevant post weld machining of critical surfaces.
7) Manufacturing sequence does not engender unnecessary risk to worker safety.

The staff has reviewed DWG X-106-500-SNP, sheets 1-8, Rev. 5; DWG X-106-501-SNP, Rev. 4; DWG X-106-502-SNP, Rev. 2; and DWG X-106-503-SNP, Rev. 0; and determined that they adequately describe the package fabrication.

2.3.2 Examination

The staff has evaluated the applicant's examination materials regarding the following eleven criteria.

1) Materials of construction must be specified on the licensing drawings. Materials Important to Safety (ITS) shall be obtained with appropriate certification and documentation required by Sections II and III of the ASME Code where applicable. All materials and components will be inspected for visual and dimensional defects, adherence to specification requirements, and traceability markings where applicable.

2) Welders and weld procedures shall be qualified in accordance to Section IX and applicable Section III Subsections of the ASME Code.

3) Welds shall be examined utilizing Section V of the ASME Code with acceptance criteria in accordance with Section III of the ASME Code. The acceptance criteria for non-destructive examination shall be consistent with the code requirements for the component that was fabricated. Post weld inspections shall be identified in a weld inspection plan, which details the weld, examination requirements, the examination sequence and the acceptance criteria, and is subject to a mandatory review and approval in accordance to the applicant's QA program prior to its implementation. Non-destructive examination inspections shall be performed in accordance to written and approved procedures by qualified personnel.

4) The containment boundary shall be examined and tested via a helium leak test, pressure test, ultrasonic testing, magnetic particle testing, and/or liquid penetrant testing as applicable. Category A and B welds are subject to volumetric examinations based on Subsection NB of the ASME Code.

5) Grinding and machining operations shall be examined by ultrasonic testing to ensure that the metal wall thicknesses are not reduced beyond design limits.

6) Dimensional inspections shall occur to confirm compliance with design drawings and to verify fit-up of individual components.

7) Trunnions are designed, inspected, and tested in accordance to ANSI N14.6. A visual examination following a test at a maximum of 150% maximum design service loading applied for a minimum of 10 minutes shall be performed to verify that no gross deformation or cracking has occurred.

8) Upon completion of hydrostatic or pneumatic pressure tests the internal surfaces of the package shall be inspected for cracking or deformation. Subsequent discovery and repair for deformation or cracking shall require retesting of the package and the test results shall be documented and incorporated into the final quality documentation.

9) Each plate or forging used for the containment boundary shall be drop weight tested per Regulatory Guides 7.11 and 7.12, where applicable, and ASME Charpy V-notch testing shall be performed on these materials. Test results shall be recorded in the final quality document.

10) Leak tests shall be performed upon completion of the fabrication of the containment boundary per ANSI N14.5.

11) All required tests, inspections, and examinations shall be documented and included in the final quality documentation report(s).

The staff has determined that DWG X-106-500-SNP, sheets 1-8, Rev. 5; DWG X-106-501-SNP, Rev. 4; DWG X-106-502-SNP, Rev. 2; and DWG X-106-503-SNP, Rev. 0, and Section 8 of the SAR, Rev. 5, adequately describe the package examination.

2.4 General Standards for All Packages

2.4.1 Minimum Package Size

The smallest overall dimension exceeds the specified requirement of 4 inches, therefore, the package meets the requirements of 10 CFR 71.43(a) for minimum size.

2.4.2 Tamper-Proof Feature

A "lock-wire," or equivalent, is utilized between two tie points on the closure end of the package during a loaded shipment. Specifically, one tie point is affixed to the OC thermal shield, and one tie point is affixed to the closure end impact limiter. Failure of the said device will indicate purposeful tampering in accordance with 10 CFR 71.43(b). Thus, the requirements of 10 CFR 71.43(b) are satisfied.

2.4.3 Positive Closure

Inadvertent opening of the package closures cannot occur for the RH-TRU 72-B package. Following installation of the package payload, the IV lid is secured via eight (8), 7/8-9UNC bolts. The OC closure lid is then secured via eighteen (18), 1¼-7UNC bolts, thus eliminating access to the IV closure. The closure-end impact limiter is attached using six (6), 1¼-7UNC bolts. When installed, the impact limiter eliminates access to the OC closure. With this double containment closure and the presence of the impact limiter, inadvertent opening of the package cannot occur.

The package was adequately analyzed for maximum internal and external differential pressures as well as expected external and internal pressures during NCT and HAC. Thus, the requirements of 10 CFR 71.43(c) are satisfied.

2.4.4 Chemical and Galvanic Reactions

The materials from which this package is fabricated (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 (RAM) packages for transport of similar chemical material without incident. This RAM packaging history, combined with successful use of these materials in similar industrial environments, ensures that the integrity of the package will not be compromised by any chemical, galvanic, or other reactions. The materials of construction and the payload are evaluated in Section 2.4.4 of the SAR, Rev. 5, for potential reactions.

2.5 Lifting and Tie-Down Standards for all Packages

2.5.1 Lifting Devices

The applicant evaluated all devices or components related to a lifting operation which included the trunnions, the closure lid lifting holes and the containment baseplate. The requirements imposed by 10 CFR 71.45(a) require a minimum safety factor of three against yielding. The lifting attachments were evaluated for static lifting including three times inertial load multiplier and found acceptable based on the allowable stresses.

The analysis uses WRC Bulletin 107. Loads are applied first in the longitudinal direction to simulate vertical lift, and then in the circumferential direction to represent horizontal lift. Both the trunnions and lid lifting holes have adequate margin for lifting operations.

The applicant also evaluated the effects of the failure of the lifting devices permanently attached to the cask and determined that such a failure would occur away from the containment boundary such that the containment and shielding functions would not be compromised.

The staff reviewed the calculations and justifications presented by the applicant in Section 2.5 of the SAR, Rev. 5, and found them acceptable. Therefore, the requirements of 10 CFR 71.45(a)(1) for lifting devices are met.

2.5.2 Tie-Down Devices

The package does incorporate structural features that are used as tie-down devices. Trunnion loading in response to the regulatory requirements of 10*g* longitudinal, 5*g* transverse, and 2*g* vertical were evaluated.

The staff reviewed the analysis presented in Section 2.5.2 of the SAR, Rev. 5, and found the results acceptable. Thus the requirements of 10 CFR 71.45(b)(1) are met.

2.6 Normal Conditions of Transport

2.6.1 Heat

The thermal evaluation for the NCT heat condition was presented in Section 3.4 of the SAR, Rev. 5, *Thermal Evaluation for Normal Conditions of Transport*. The NCT heat condition consists of exposing the package to direct sunlight and 100°F still air per the requirements of 10 CFR 71.71(c)(1), with insolation per Regulatory Guide 7.8. A maximum internal heat load of 50 watts (combustible payload) and 300 watts (non-combustible payload) is used for the evaluation.

Differential Thermal Expansion

The conditions of normal heat result in rather modest temperatures throughout the package. Maximum temperatures for the various package components are presented in Table 2.6-2 of the SAR, Rev. 5. As indicated, all package structural component temperatures remain below 160°F. For many of the analyses, a 160°F maximum temperature for both the IV and OC is conservatively employed. The payload canister temperature reaches a maximum of 167°F. Therefore a 200°F maximum temperature for analysis of the canister was conservatively employed. The design was governed by primary, load-controlled stresses and the corresponding limits specified in Section C.2 of Regulatory Guide 7.6 rather than by primary-plus-secondary stresses and the 3.0Sm limit specified in Section C.4.

Initial pressure in both the IV and the OC is one (1) atmosphere (14.7 psia). Per Section 3.4.4, *Maximum Internal Pressure*, the design pressure is 150 psig for any package component.

The staff reviewed the applicant's analyses for pressures and temperatures, differential thermal expansion – radial expansions of shells relative to their end hardware, and stresses due to maximum pressures presented in Section 2.6.1 of the SAR, Rev. 5, and concurred with the results.

Lid Bolts and Seals

The applicant performed evaluations to determine the state of stress in the lid bolts as well as the degree of compression in the seals due to the NCT loadings. The applicant concluded that no bolt overstresses occurred nor did the closure plate seals unload.

The staff reviewed the statements and conclusions made by the applicant, reviewed calculations presented in supporting documents and determined that the structural performance of this package under the hot condition of NCT satisfied the requirements of 10 CFR 71.71(c)(1).

2.6.2 Cold

The shipping package must be able to withstand an ambient temperature of -40°C (-40°F) and -29°C (-20°F) in still air and in the shade per 10 CFR 71.71(c)(2).

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. Since this material does not undergo a ductile-to-brittle transition, per Regulatory Guide 7.11 and Regulatory Guide 7.12, the RH-TRU 72-B is not suspect to brittle fracture.

The only concern identified with the cold condition is with shrinkage of the lead onto the inner shell of the OC. The applicant has performed calculations in Section 2.10.8 (appendix) of the SAR, Rev 5, and determined that the stress levels from the lead pour and subsequent cool down would be relatively small, and it is concluded that buckling of the OC inner shell should not occur due to the shrinkage of lead.

The staff reviewed the calculations performed and conclusions made by the applicant and determined that the structural behavior of this package under the cold condition will satisfy the allowable stresses for the materials and components of construction.

The requirements of 10 CFR 71.71(c)(2) are satisfied.

2.6.3 Reduced External Pressure

The applicant has determined that the effect of a reduced external pressure of 3.5 psia, as required per 10 CFR 71.71(c)(3), is considered negligible. It was argued that the maximum NCT internal pressure of 150 psig bounds the reduced external pressure of 3.5 psia.

Under a reduced external pressure to 3.5 psi (25 kPa), the structural behavior is bound by the design internal pressure, therefore the requirements of 10 CFR 71.71(c)(3) are satisfied.

2.6.4 Increased External Pressure

The applicant has determined the effect of the increased external pressure of 20 psia (5.63 psig external pressure) to be negligible to the RH-TRU 72-B due to the thick OC outer shell and end closures.

The staff noted that this approach is consistent with the approach taken for the 290 psi external water pressure requirement of 10 CFR 71.61 discussed in Section 2.7.7, of the SAR, Rev. 5. Therefore, the staff has determined that the requirements of 10 CFR 71.71(c)(4) are satisfied.

2.6.5 Vibration

The applicant has demonstrated that the RH-TRU 72-B package is capable of withstanding normal vibration incident to transport. The analysis concludes that the package has adequate (approximately 6) margin of safety.

Based on the analyses presented in Section 2.6.5 of the SAR, Rev. 5, the staff has determined that the requirements of 10 CFR 71.71(c)(5) are satisfied.

2.6.6 Water Spray

The containment capabilities of the RH-TRU 72-B Transport Packaging System are not compromised by water spray, because all external surfaces are composed of stainless steel, and the closure seal is impervious to water. Therefore, the staff has determined that the requirements of 10 CFR 71.71(c)(6) are satisfied.

2.6.7 Free Drop

The structural evaluation of a 1 foot (0.3 meter) free drop was reviewed and evaluated in Section 2.6.7 of the SAR, Rev. 5. Based on the staff's review of the information presented by the applicant, the requirements of CFR 71.71(c)(7) are satisfied.

2.6.8 Corner Drop

The corner drop test does not apply since the gross weight of the package exceeds 100 kg (220 lb) – for cylindrical fissile packages, in accordance with 10 CFR 71.71(c)(8).

2.6.9 Compression

The compression drop test does not apply since the gross weight of the package is 20,455 kg (45,000 lb) and exceeds 5,000 kg (11,000 lb), in accordance with 10 CFR 71.71(c)(9).

2.6.10 Penetration

Due to the lack of sensitive external protuberances, and based on the guidance presented in Regulatory Guide 7.8, it is determined that the NCT penetration test is not significant to the structural design of the RH-TRU 72-B packaging system due to its large overall size (45,000 lb, 141 in length). Based on the staff's review of the information presented by the applicant, the requirements of 10 CFR 71.71(c)(10) are satisfied.

Structural Evaluation (Design condition, Normal Operating Condition, and 0.3 meter Free Drop)

Maximum impact limiter deflections and residual clearances, and impact limiter separation moments are given in Figure 2.10.3-1 through Figure 2.10.3-8 (appendix) of the SAR, Rev. 5, and are presented as a function of the initial package impact angle, measured from horizontal.

A summary of the impact limiter deformations and package accelerations for the end and side impact orientations is represented in Table 2.10.3-10 (appendix) of the SAR, Rev. 5.

A summary of the maximum package responses as a function of initial impact angle is presented in Table 2.10.3-11 and Table 2.10.3-12 (appendix) of the SAR, Rev. 5.

A summary of the maximum impact limiter separation moments as a function of initial impact angle is represented in Table 2.10.3-13 (appendix) of the SAR, Rev. 5. The maximum impact limiter separation moments are similar for both normal and accident conditions because the maximum moment occurs at a relatively small crush depth. In other words, the moment arm decreases with additional crush depth at a higher rate than the crush force increases.

2.7 Hypothetical Accident Conditions

The RH-TRU 72-B was licensed by analysis with supplemental static and dynamic testing of ½ scale impact limiters, provided in Section 2.10.7 of the SAR, Rev. 5, to validate and define dynamic load factors. Although no stress results from the impact limiter tests were compared with CASKDROP, the deformations were compared and found acceptable. The staff concurs with the applicant's approach and finds it reasonable.

2.7.1 30-foot Free Drop

The staff observed no significant discrepancies between the peak decelerations obtained from the CASKDROP solutions and the actual tests conducted for the RH-TRU 72-B. The CASKDROP results were predicting peak g loads that were higher than the results obtained during the tests, which is conservative.

Table 1

Source: SAR Table - 2.10.3-10 Summary of End and Side Drop Impact Accelerations And Impact Limiter Deflections				
Drop Height (ft)	**Drop Orientation**	**Impact Limiter Temperature (°F)**	**Impact Acceleration (g)**	**Impact Limiter Deflection (in)**
1 [NCT]	End	-20	42.5	0.59
		140	28.6	0.86
	Side	-20	19.3	1.44
		140	12.4	2.12
30 [HAC]	End	-20	89.7	4.65
		140	51.1	8.67
	Side	-20	81.2	7.79
		140	68.9	11.52

Section 2.10.4 (appendix) of the SAR, Rev. 5, presents an evaluation of the RH-TRU 72-B package for potential secondary impacts, or slapdowns, following the primary impacts associated with 30-foot corner or oblique drop events. The conclusion is that stresses associated with slapdown are within allowable HAC stresses. Additionally, impact limiter deformations resulting from slapdown are acceptable as they will not allow "bottoming out" on any package external surface (i.e., trunnions and the package body itself).

In Section 5.1.3 of Appendix 5.1 of the RH-TRU Payload Appendices (RPA), the staff noted that the 30-foot drop testing of the neutron shielded canister was simply to ascertain the response of the canister and internal shielding to HAC. Because the gross weight of the neutron shielded canister is less than the fixed or removable lid canister previously analyzed, and due to the fact that interface boundary conditions between the packaging and payload are unchanged, the structural design basis for the RH-TRU 72-B packaging is not affected by the addition of the neutron shielded canister as an authorized payload container.

Part (9) of Section 2.7.1.1 of SAR, Rev. 5, shows that the RH-TRU 72-B lead shielding does not generate enough flow stress to undergo deformations that would create a lead slump due to the HAC end drop. The staff noted that the analysis was performed by using Equation 2-16 from OR-NSIC-68 (Oak Ridge Cask Designer's Guide), to determine the required flow stress (temperature dependent) of the lead to undergo slumping. The calculated margins of safety are shown in Table 2.7-12 of the SAR, Rev. 5.

Appendix 5.1 of the RPA references Petersen Engineering Report 7953-R-027, Revision 0, November 2009, for details regarding the 30-foot drop testing of the neutron shielded canister in a test fixture that simulated the RH-TRU 72-B interface boundary conditions and g-loads. The staff has reviewed this report, and in general concurs with its conclusions.

The staff reviewed the modeling methodologies, calculation packages, results from the dynamic and quasi-static evaluations, and comparisons with allowable stresses. The staff also verified values obtained from the analytical models by recreating reported data and associated pictorial plots showing deceleration time histories, stress contours, and deformed and un-deformed configurations of the package. Finally, the staff verified that energy balances for each of the drop scenarios were consistent with good practice in analysis.

The analytical modeling, in aggregate, satisfied the requirements of 10 CFR 71.73(c)(1).

Buckling

There are two shells within the RH-TRU 72-B package where buckling prevention criteria are applicable: the inner shell of the OC and the IV shell. For additional conservatism, the OC outer shell was also evaluated. For reference purposes, the principal geometric features of these shells are listed in Table 2.1-3 of the SAR, Rev. 5. To demonstrate that buckling will not occur for these three shell geometries, the methodology of ASME Code Case N 284 is directly applied.

Consistent with the Regulatory Guide 7.6 philosophy, factors of safety corresponding to ASME Code Level A and D Service conditions were employed respectively for NCT and HAC loadings, with applicable factors of safety of 2.0 for NCT and 1.34 for HAC, as specified in ASME Code Case N 284 therefore, the buckling of the RH-TRU 72-B package is not a mode of failure due to NCT or HAC.

2.7.2 Crush

The regulatory requirement for Crush is not applicable as all models have an individual weight greater than 1,100 lb. (500 kg). Therefore, the requirements of 10 CFR 71.73(c) are met.

2.7.3 Puncture

The RH-TRU 72-B Transport Packaging System cask component is evaluated for accidental drops as evidenced in the SAR. The RH-TRU 72-B transport package is analyzed for 40-inch drop onto a 6-inch diameter steel bar. The applicant has evaluated the RH-TRU 72-B for Side Puncture, End Puncture, Puncture on the Trunnions, Puncture on Containment Penetrations, Puncture on the OC Closure Lid, and Puncture on the OC Gas Sampling Port. Localized deformation is expected, resulting in thinning of the lead shielding. This is noted in more detail in the shielding evaluation.

The staff reviewed the analyses presented by the applicant and found that the results presented support the conclusions made by the applicant relative to the puncture drop event. The requirements of 10 CFR 71.73(c)(3) are met.

2.7.4 Thermal

The applicant utilized temperature information from the evaluated fire event to determine the effects on the structural integrity of the package. The main conclusions drawn by the applicant are that the metal temperature averaged across any section of the containment boundary remains below the maximum allowable temperature for level A conditions for ASME, Section III, Subsection NB components, the outer surface of the package directly exposed to the fire does not slump, internal part interferences do not occur due to differential thermal expansion both

during and after the fire event, and the cask closure lid bolts do not unload, leading to a reduction of compressive load on the gasketed surfaces. The staff reviewed the evaluation presented by the applicant and found their reasoning and conclusions credible. Therefore the requirements of 10 CFR 71.73 (c)(4) are met.

2.7.5 Immersion - Fissile

10 CFR 71.73(c)(5) requires that all fissile packages that have not assumed inleakage of water, immersion under a head of water of at least 3 feet must be considered.

The applicant has considered water inleakage as required per 10 CFR 71.55 in the RH-TRU 72-B criticality analysis, and therefore 10 CFR 71.73(c)(5) is not applicable.

2.7.6 Immersion - All Packages

This condition is less demanding to the package than the Deep Water Immersion condition, and is therefore covered by the Deep Water condition of 10 CFR 71.61. Therefore, the requirements of 10 CFR 71.73(c)(6) are met.

2.7.7 Deep Water Immersion

The RH-TRU 72-B contains more than 10^5 A_2. Therefore, the package must be analyzed against a pressure load of 2MPa (290 psia). The applicant subjected the RH-TRU 72-B to 300 psig internal pressure, which bounds the 290 psia scenario. The buckling analysis was evaluated using ASME Boiler and Pressure Vessel Code Case N-284. The staff agreed that the evaluation performed by the applicant was adequate and the requirements of 10 CFR 71.61 are satisfied with respect to stress limits and stability requirements.

2.8 Fuel Rods

This section does not apply to the RH-TRU 72-B Transport Packaging System.

2.9 Evaluation Findings

The structural review of RH-TRU 72-B packaging resulted in the following findings.

2.9.1 Description of Structural Design

The staff reviewed the structural design description of the package and concluded that the contents of the application met the requirements of 10 CFR 71.31. The staff reviewed the codes and standards used for package design and determined that they are acceptable.

2.9.2 Material Properties

To the maximum credible extent, there are no significant chemical, galvanic or other reactions among the packaging components, among package contents, or between the packaging components and the contents in dry or wet environment conditions. The effects of radiation on materials are considered and package containment is constructed from materials that met the requirements of RGs 7.11 and 7.12.

2.9.3 Lifting and Tie-down Standards for All Packages

The staff reviewed the package lifting and tie-down standards and concluded that they met the requirements of 10 CFR 71.45.

2.9.4 General Considerations for Structural Evaluation of Packaging

The staff reviewed the structural evaluation for the packaging and concluded that the application met the requirements of 10 CFR 71.35.

2.9.5 Normal Conditions of Transport

The staff reviewed the structural performance of the packaging under NCT and concluded that there will be no substantial reduction in the effectiveness of the packaging, and it met the requirements of 10 CFR 71.71.

2.9.6 Hypothetical Accident Conditions

The staff has reviewed the structural performance of the packaging under HAC and concluded the packaging satisfied the sub-criticality, containment, shielding, and temperature requirements of 10 CFR 71.73.

2.9.7 Special Requirement for Irradiated Nuclear Fuel Shipments

N/A

2.9.8 Internal Pressure Test

The staff reviewed the containment structure and DWG X-106-500-SNP, Rev. 5, and concludes that the package met the 10 CFR 71.85(b) requirements for pressure test without yielding.

2.9.9 Evaluation Summary

Based on review of the statements and representations in the application, the staff concludes that the structural design was adequately described and evaluated and that the package has adequate structural integrity and meets the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

3.1 General Considerations

The RH-TRU 72-B package is a Type B package designed to transport remote-handled transuranic waste. Paper waste and metallic waste can be shipped in the shielded and non-shielded packaging designs. The amount of waste that can be shipped is limited by the following conditions: 1) decay heat, 2) flammable gas generation rate, or 3) pressure within the container generated by radiolysis. Although all canisters are limited to 50 W of decay heat, the applicant's calculations assumed metallic waste and paper waste decay heats of 300 W (non-shielded canister) and 50 W, respectively. However, the amount of metallic or paper waste allowed within a package must be determined by detailed procedures described in the "Remote-Handled Transuranic Waste Authorized Methods for Payload Control" (RH-TRAMPAC), Rev. 1 (April 2011) manual based on the decay heat and the radiolysis activity. This evaluation

focused on the NS15 and NS30 shielded canisters; these shielded canisters formed the basis of this amendment application.

3.2 Thermal Design Features

The RH-TRU 72-B packaging includes a radioactive source payload canister that is placed within a 304 stainless steel IV. The IV is placed within a 304 stainless steel OC vessel. A cylindrical annular cavity within the OC vessel wall contains a lead shield. A stainless steel thermal shield placed around the OC offers protection from the thermal input by fire. Impact limiters, placed on each end of the packaging, limit the damage that can result during the HAC tests described in 10 CFR 71.73. The open cell polyurethane (LAST-A-FOAM FR-3700) filling of the impact limiters also acts as a thermal barrier to fire. The package is passively cooled.

3.3 Decay Heat

Section 5.0 of the RH-TRAMPAC manual states that the design decay heat limit for the RH-TRU 72-B is 50 W for all payload types. The amount of metallic or paper waste allowed within a package must be determined by detailed procedures described in the RH-TRAMPAC manual. Following the RH-TRAMPAC procedure generally results in packages with decay heats less than the 50 W limit, due to restrictions associated with radiolysis pressure generation and flammable gas generation.

3.4 Summary of Temperatures

Table 3.1-1 and Table 3.1-2 of the February 2011 SAR provide a summary of component temperatures for the NCT and HAC discussed in 10 CFR 71.71 and 10 CFR 71.73, respectively. Component temperatures associated with packaging based on the NS15 and NS30 shielded canisters are provided in Tables 5.1-1 to 5.1-6 of the "RH-TRU Payload Appendices" (February 2011). These tables showed that the maximum calculated component temperatures were within the allowable limits.

3.5 Thermal Evaluation for Normal Conditions of Transport

The applicant considered an isolated horizontal package in order to analyze the thermal performance of the RH-TRU 72-B package design under NCT. A two-dimensional axisymmetric lumped model was modeled using the Thermal Desktop and SINDA/FLUINT computer programs. The applicant modeled the NS15 and NS30 canisters assuming a 50 W paper waste content. Thermal transport mechanisms included conduction and radiation heat transfer between the package's gaps and convection and radiation heat transfer to the surroundings.

The solar insolation applied to the package was modeled by the applicant using two methodologies in order to understand the sensitivity to the calculation's package component temperatures. One method used projected areas and averaged the insolation values delineated in 10 CFR 71.71(c)(1) over 12 hours as a steady state model. The second method applied insolation values over the full geometric areas in a "12 hour on / 12 hour off" transient model. Analyses for maximum temperature assumed a 100°F ambient temperature.

The applicant's analyses indicate that the transient methodology resulted in package component temperatures higher than those for the steady methodology by approximately 10°F for the O-ring seals and 20°F for the outer thermal shield. For both methodologies, the surface

temperatures of the packaging components were below the 185°F 10 CFR 71.43(g) regulatory limit for exclusive use shipping.

Under NCT, the minimum temperature of the components occurs for an ambient temperature of -40°F, which is in agreement with 10 CFR 71.71(c)(2).

The SAR analyses indicated that the maximum normal operating pressure (MNOP) occurs for a paper waste content with a 21.7 W decay heat. Due to radiolysis of the paper's organic matter, the 21.7 W decay heat content resulted in a NCT package pressure of 138.8 psig, which is within the 150 psig design pressure. The package pressure is based on a 60-day shipping period. The staff evaluated the thermal performance of the package during NCT and found it acceptable.

3.6 Thermal Evaluation for Hypothetical Accident Conditions

The applicant considered an isolated horizontal package in order to analyze the thermal performance of the RH-TRU 72-B package design under HAC. The analytical model for HAC was similar to the NCT model. However, in order to accommodate the damage to the model associated with the HAC tests, the applicant changed the geometry of the model by reducing the polyurethane foam thickness at the impact locations during the crush test. The initial conditions prior to the HAC fire included a 100°F ambient temperature without insolation. This analysis assumed that the model was exposed to a 1475°F radiation source for 30 minutes with a fire emissivity of 0.9, which is in agreement with the regulatory requirements in 10 CFR 71.73(c)(4).

Under HAC, the analyses indicate that the package components were at temperatures below the allowable values. The highest O-ring seal temperature was approximately 210°F below the short term limit and 75°F below the long term operating limit. The lead shield was approximately 75°F below the lead melting point. For the shielded canister, the extra-high molecular weight, HDPE was 50°F below the allowable limit.

The maximum pressure during HAC was calculated by the applicant to be 217.9 psig, which is below the 300 psig internal pressure value assumed for the HAC stress analysis. The staff evaluated the thermal performance of the package during HAC and found it acceptable. The staff agrees that the temperatures stated by the applicant are within the regulatory limits.

3.7 Conclusion

Based on a review of the thermal sections of this amendment application, the staff has reasonable assurance that the package meets the thermal requirements of 10 CFR Part 71.

4.0 CONTAINMENT EVALUATION

4.1 General Considerations

RH-TRU 72-B is a Type B(M)F-96 package designed to transport, by ground, remote-handled paper and metallic transuranic waste. The authorized contents are defined in the following DOE governing documents:

a) "Remote-Handled Transuranic Waste Authorized Methods for Payload Control" (RH-TRAMPAC), Rev. 1, April 2011.
b) "RH-TRU Payload Appendices," Rev. 1, February 2011.

4.2 Description of the Containment System

The RH-TRU waste is loaded into a payload canister or into inner containers placed within the canister. The canister consists of a 26-inch outside diameter ¼-inch thick cylindrical vessel with an overall length of 120-1/2 inches. The canister design is either a one or two piece construction (i.e., fixed or removable lid versions). The removable lid version can be configured into either a NS15 or NS30 design, which includes HDPE neutron shielding surrounding either approximately 15 or 30 gallon inner containers.

The payload canister is placed within an IV consisting of a 3/8-inch thick, 32-inch outside diameter Type 304 stainless steel shell that is 130 inches long. The vessel includes three butyl O-ring seals. The middle O-ring seal acts as a secondary containment barrier, if an optional preshipment leakage rate test is successfully performed. The IV includes a Seal Test Port, Backfill Port, and Gas Sampling Port that are used to test for leaktightness.

The IV is placed within an OC consisting of a 41 5/8-inch outside diameter Type 304 stainless steel shell that is 141-3/4 inch long. The OC consists of 1.75-inch thick outer shell, 1 7/8-inch thick lead shield, and 1-inch thick inner shell. The vessel includes two Rainier Rubber butyl O-ring seals; the lower O-ring seal acts as the primary containment barrier. The OC includes a Seal Test Port and a Gas Sampling Port that are used to test for leaktightness.

4.3 Containment and Leakage Rate Tests

A number of features exist to prevent the inadvertent opening of the package. The IV lid is secured with eight 7/8-9UNC bolts. The OC lid is secured with eighteen 1 ¼-7UNC bolts. Access to the ends of the OC is prevented by the attachment of the closure-end impact limiters via six 1 ¼-7UNC bolts.

The O-ring material (Rainier Rubber RR0405-70, or equivalent) must meet the test requirements and specifications described in Section 3.0, "Thermal Evaluation" (including Section 3.6.4) of the RH-TRU 72-B SAR, Rev. 5, February 2011. Specifically, each containment O-ring seal material formulation shall be initially qualified for use in the RH-TRU 72-B packaging through the application of performance tests that demonstrate the material's ability to achieve and maintain a "leaktight" seal at or beyond hot and cold temperature extremes, duration, minimum seal compression, and maximum seal compression changes.

The IV and OC shall be leak tested as described in the February 2011 SAR Section 4.1.3.1.1, "Fabrication Leakage Rate Test," Section 4.1.3.1.2, "Maintenance/Periodic Leakage Rate Test," and Section 4.1.3.1.3, "Pre-shipment Leakage Rate Test," and conform to the requirements of ANSI N14.5. Information on the leakage rate tests are found in the SAR, Chapter 7.0, "Operating Procedures," and Chapter 8.0, "Acceptance Tests and Maintenance Program."

Section 4.2 and 4.3 of the SAR states that the results of the structural and thermal analyses verify that there will be no release of radioactive materials per the "leaktight" criterion of ANSI N14.5 under NCT and HAC.

4.4 Conclusion

Based on a review of the containment sections of the application, the staff has reasonable assurance that the package meets the containment requirements of 10 CFR Part 71.

5.0 Shielding Review

Washington TRU Solutions LLC submitted an application, on behalf of the U.S. Department of Energy (DOE), for an amendment to the RH-TRU 72-B CoC to incorporate NS15 and NS30 neutron shielded canisters as approved canisters for use in the RH-TRU 72-B transportation package. The applicant also requested an increase in the allowable activities of radionuclides that can be transported in the package due to the additional shielding provided by the NS15 and NS30 canisters.

The RH-TRU 72-B transportation package has previously been approved to transport transuranic waste. The staff's previous shielding review was documented in a staff SER dated December 27, 2002 (Reference 5.4).

The staff reviewed the addition of the new contents using the guidance in Section 5 of NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," May 1999. The staff's evaluation of the applicant's changes to the shielding evaluation follows. This evaluation is based on SAR Revision 5 (Reference 5-1), RH-TRAMPAC Revision 1 (Reference 5-2), and RH-TRU Payload Appendices Revision 1 (Reference 5-3). These documents were initially submitted as Revision 1 dated February 2010; however, the applicant modified the documents in response to the staff's review and resubmitted them in their entirety with their same revision numbers but dated August 2010 and then again as the same revision numbers but dated February and April 2011. Hence, the references are for the latest date of the revision.

5.1 Description of Shielding Design

5.1.1 Design Features

The staff reviewed the general information chapter in the SAR as well as additional information on the shielding design in Chapter 5.0 of the SAR, Reference 5-2, Section 3.2, and Reference 5-3, Appendix 5.1. The staff determined that the figures, drawings, tables, and discussion describing the shielding features are sufficiently detailed to support an in-depth evaluation. The shielding design features of the NS15 and NS30 canisters include layered shielding materials that completely surround the waste. The shielding materials are stainless steel 304, lead, and HDPE for neutron shielding. The NS15 Neutron Shielded canister uses a HDPE pipe with a nominal thickness of 3.387 inches (min 3.288 inches) and HDPE end caps with a nominal thickness of 5 inches. The NS30 Neutron Shielded canister uses a HDPE pipe with a nominal thickness of 1.454 inches (min 1.412 inches) and HDPE end caps with a nominal thickness of 5 inches.

5.1.2 Codes and Standards

The applicant identified the appropriate regulations in 10 CFR Part 71 throughout Section 5 in the SAR, and Section 3.2 in Reference 5-2. The staff also verified that the applicant appropriately identified ANSI/ANS-6.1.1 1977 for their flux to dose rate conversion factors; and ANSI/ANS-6.4.3 1991 for their mass attenuation and build up factors.

5.1.3 Maximum Radiation Levels

The staff reviewed the information provided in SAR Section 5 to ensure that the applicant meets the requirements in 10 CFR 71.47 and 10 CFR 71.51. Since the applicant states that the RH-TRU 72-B will be operated under "exclusive use," the radiation levels must not exceed those specified in 10 CFR 71.47(b). The applicant chooses to evaluate the dose rate at the surface of the transportation cask.

The staff verified that the applicant's methodology is to ensure that the external radiation dose rate during HAC does not exceed 1 rem/hr at 1 meter from the external surface of the package. The applicant used the maximum allowable dose rate under HAC to calculate the curie limits for each radionuclide inventory. The staff finds that this meets, in part, a requirement of 10 CFR 71.51(a)(2).

The applicant relies upon the pre-shipment dose rate measurements to ensure compliance with 10 CFR 71.47 limits for NCT and does not provide an evaluation in the application for showing the proposed contents limits meet the NCT dose rate limits. The pre-shipment dose rate measurements fulfill the requirement given in 10 CFR 71.87(j). These measurements only indicate that a particular shipment can be transported and meet the NCT dose rate limits; they do not necessarily indicate that the maximum contents proposed in the CoC meet those limits as a given shipment may not be loaded with the maximum contents quantities specified in the CoC. Per 10 CFR 71.31(a)(2) and 71.35(a), the application should include a demonstration that the package satisfies the NCT dose rate limits. The demonstration for the package includes the packaging and the maximum contents limits as proposed in the CoC. As indicated previously, a commitment to pre-shipment measurements alone may not meet this requirement.

The staff does note, however, that given various circumstances surrounding this application (e.g., the pre-shipment measurement was accepted in previous amendments for the purposes of meeting 10 CFR 71.35(a)), that pre-shipment measurements may be relied upon in only this instance given that the applicant has described a rigorous pre-shipment survey, the survey method is a condition of the CoC, the applicant provided sufficient justification that this measurement is sufficient, and provided operational performance data on previous shipments. The pre-shipment measurements are performed in a rigorous and comprehensive manner as described in Section 3.2 of the RH-TRAMPAC, Revision 1. The package operations, described in Chapter 7.0 of the SAR, incorporate this measurement procedure by reference. With Chapter 7.0 of the SAR incorporated by reference in the proposed CoC (and the RH-TRAMPAC revision also captured in the CoC), the comprehensive measurement procedure is a condition of the certificate.

The applicant also provided information regarding the past shipments already conducted under the approved CoC. This information includes the dose rate measurements and the fraction of CoC curie quantity limits that were loaded in each shipment. The measurements for these shipments were performed using the comprehensive procedure described in the RH-TRAMPAC for the current application. While some shipments showed the same dose rates at the package surface and at 2 meters from the vehicle edge, this was because the surface dose rates were so low that they were at or below the lower limit of detection for the particular instrument used by the package user; so, the values of the lower limit of detection (summed for gamma and neutron dose rates) were reported for the package surface and measurements at 2 meters from the vehicle edge. The staff notes, that these reported values were always below the limit for dose rates at 2 meters from the vehicle edge. The applicant also indicated that the contents to be shipped under the proposed CoC are similar in nature (e.g., physical properties) to those the

same user has shipped under the current CoC, only with less decay time since generation (hence the higher neutron sources and the need to use the neutron shielded canister).

The staff reviewed this information and also considered the proposed changes in curie quantity limits in the CoC and the changes to the shielding and configuration resulting from the addition of the HDPE pipe. Based upon this consideration, the staff finds the proposed curie limit changes to be reasonable. The staff also noted a well defined characterization of the content, which is normally part of transuranic waste characterization for disposal purposes. Based upon a review of these descriptions and considerations, the staff finds that there is sufficient reasonable assurance that, for this application/CoC amendment only, the pre-shipment measurements may be relied upon in the place of a NCT evaluation to demonstrate compliance with 10 CFR 71.35(a) and 71.47. Depending on the nature of the request, any subsequent amendments to this package should provide an evaluation in the application to demonstrate compliance with 10 CFR 71.35(a) and 71.47,. This evaluation may be by analysis or by measurement; however, the measurement results should be provided in the application and demonstrated to encompass the scope of the contents proposed to be transported in the package, as defined in the CoC.

5.2 Source Specification

The source specification of the contents is listed in Reference 5-2, Tables 3.2-1 and 3.2-2. These show the amount of radionuclides that, if placed as a point source into the HAC configuration, results in a dose rate of 1 rem/hr at 1 meter from the surface of the package. The RH-TRU 72-B is designed to transport remote handled transuranic waste containing both gamma and neutron emitting radionuclides.

5.2.1 Gamma Source

The applicant modeled gamma sources using the equation for dose rate from an isotropic point source, which was implemented in MathCAD Version 13.1. The applicant did not credit the HDPE with gamma attenuation, but did take geometry credit. The applicant also used iron buildup factors in place of lead buildup factors for conservatism. The applicant used buildup and attenuation factors from ANSI/ANS-6.4.3 1991. The applicant requested slightly higher limits than the currently approved activity limits for gamma emitters for the case of the NS15 and NS30 neutron shielded canister; due to geometry credit for the HDPE pipe.

5.2.2 Neutron Source

The applicant used Monte Carlo N Particle (MCNP) 5 v1.40 to calculate the dose rates from neutron-emitting isotopes. The neutron source spectra were provided in Reference 5-9. The total source strength was set to 1 n/s, so once the dose rate is known the activity could be easily scaled up to produce the HAC limit of 1 rem/hr at 1 meter. This calculation is performed for the neutron emitting isotopes in Tables 7 and 8 of Reference 5-9.

5.3 Shielding Model

The staff reviewed Sections 2 (structural evaluation) and 3 (thermal evaluation) of the SAR to determine the effects of the NCT and HAC on the packaging and its contents. The staff reviewed the information in Section 2.6 of the applicant's SAR, and the applicant concludes that there will be no damage to the package as a result of NCT. This information was previously reviewed and approved by the staff and documented in Reference 5.4.

The staff also reviewed the information in Section 2.7 of the applicant's SAR to determine the effects of HAC on the package. From the tests required in 10 CFR 71.73, any damage is limited to the impact limiter from the drop tests. This is summarized in Sections 2.7.8 and 3.5.6 of the SAR. As stated in Section 5.1 on page 5.1-2, the applicant does not include the impact limiter in the shielding analyses and assumes nominal design dimensions. Localized deformation is expected from the puncture test; this deformation results in thinning of the lead shielding at the area of impact.

The applicant has accounted for the tolerances of the cask dimensions in their shielding analysis, as well as reducing the thickness of the HDPE neutron shield by 0.25 inch. The reduction in HDPE thickness accounts for the amount of material which may be removed as a result of HAC.

In its structural evaluation, the applicant considered the potential for the occurrence of lead slump and concluded that lead slump will not occur. The applicant also determined that, based upon the procedures for fabrication of the lead shielding, any axial gaps at the end of the lead column (as a result of the fabrication process) will be negligible. Based upon its review of the fabrication procedures and the evaluation related to lead slump (see Section 2 of this SER), the staff finds that lead slump will not occur and the HAC shielding model does not need to account for slump or any axial gap at the ends of the lead shielding column.

To model the effects of impacting a puncture bar, the applicant reduced the thickness of the lead shielding from 1.875 inches to 1.219 inches over a six inch diameter area.

The staff finds that these models are consistent with the effects from the accident analyses described in Sections 2 and 3 of the SAR.

5.3.1 Configuration of Source and Shielding

The staff examined Figures 5.1-1 and 5.5.5-1 in the SAR, and Figures 2.8-2 and 2.8-3 in RH-TRAMPAC (Reference 5-2), the description of the modeling, and the MCNP input decks to determine how the shielding is modeled. The staff verified that the cask dimensions are consistent with the package drawings provided in SAR Section 1.3.1. The reference of the drawing viewed by the staff is: U.S. Department of Energy, RH-TRU 72-B Packaging SAR Drawing, Drawing Number X-106-500-SNP, Revision 5.

The neutron shielded canisters are modeled as HDPE tubes with HDPE end caps. The transportation cask is modeled as concentric cylinders with an inner steel shield, surrounded in the radial direction by a lead shield and then surrounded by an outer steel shield, both end caps are manufactured out of steel. The applicant models the source as a point source.

For HAC, 98% of the source is placed at the inner surface of the canister next to the area where the shielding is thinnest due to puncture bar damage. The applicant assumes that the other 2% of the source gets out of the HDPE shield and reconfigures as a point source in line with the main source next to the thin area, a 6 inch diameter dent due to impact with a puncture bar.

The staff verified that the applicant has a dose point 1 meter from the surface of the package for HAC. The dose point was located in line with the dent and the point sources, which would result in the highest results. Since the applicant modeled HAC, the dose at the surface and 2 meters

from the package, or the vehicle edge, was not analyzed in the models, as would be done for NCT analyses.

5.3.2 Material Properties

The staff verified that the applicant identified the materials and mass densities of the shielding materials. The applicant identified the inner and outer shield as Type 304 Stainless Steel in Table 6 of Reference 5-8 with a density of 8.0128 g/cm^3, and the lead shield as elemental lead using the default isotopic concentration and a density of 11.35 g/cm^3 which were used for gamma and neutron dose rate calculations for the NS15 and NS30 canisters. In the MCNP input decks provided in Reference 5-8, and in Section 3.2 of that document, the applicant modeled HDPE as CH_2 (polyethylene) with a density of 0.94 g/cm^3. The staff finds that the applicant correctly specified the MCNP card POLY.60T to account for the hydrogen-carbon bonds in the polyethylene. The staff finds that the material specifications of the shield are acceptable.

The staff does not find that any of the shielding materials would degrade during the service life of the packaging and that there are no temperature sensitive materials present.

5.4 Shielding Evaluation

5.4.1 Methods

For the neutron shielding evaluation the applicant uses the MCNP5 computer code with cross sections from the ENDF/B-VI library. The staff has previously accepted the use of MCNP for similar shielding evaluations and finds that its use is acceptable.

For the gamma shielding evaluation the applicant uses the point kernel approximation to calculate gamma dose rates from basic physics. These calculations are performed in MathCAD. This approach has been previously accepted by the staff in Reference 5-4.

5.4.2 Input and Output Data

The staff performed a review of MCNP input decks and ensured that the geometry and materials were appropriately specified. In addition the staff reviewed the options selected and detector specification to confirm that they were used as stated in the SAR.

5.4.3 Flux-to-Dose-Rate Conversion

The staff confirmed that the applicant used the ANSI/ANS 6.1.1-1977 (Reference 5.6) standard and finds this acceptable. The applicant performed the neutron flux-to-dose-rate conversion using the MCNP code and Table 1 from the standard. The applicant performed the gamma flux-to-dose-rate conversion using MathCAD Version 13.1 and Table 3 from the standard. This is documented in Reference 5-8.

5.4.4 Radiation Levels

The staff confirmed that the applicant's calculated radiation levels under HAC satisfy the limits in 10 CFR 71.51(a)(2). The staff verified that the analysis showed that the selected locations are those where maximum radiation levels occur and included the thinning of the lead shielding due to an impact with a puncture bar.

For NCT the applicant relies on pre-shipment surveys to ensure that the waste is not loaded in a configuration that exceeds the allowable dose rates in 10 CFR 71.47. As discussed in Section 5.1.3 of this SER, this approach is found to be acceptable only for this application under the conditions described in that section of the SER. However, any subsequent amendment application should provide an evaluation to demonstrate, as required by 71.35(a), that the package with the proposed contents meets the NCT dose rate limits; the pre-shipment surveys will only be relied on for compliance with 10 CFR 71.87(j).

5.4.5 Confirmatory Analysis

The staff performed an independent analysis of the RH-TRU 72-B with NS30 to confirm the applicant's results. The staff used the SCALE6 code with the v7-27n19g cross section library to model neutron radiation and with the v7-200n47g cross section library to model gamma radiation. The staff modeled NCT and HAC.

To model the effect of an impact with a puncture bar, the staff modeled the transportation cask with a 6 inch diameter area where the thickness of the lead shielding was reduced in the radial direction. The thickness of the HDPE, lead, and steel used in the staff's model are the same as those in Figure 2 of Reference 5-8. The staff modeled a single point source at the HAC activity limit specified in Table 7 of Reference 5-8. The point source was located inside the HDPE shield opposite the center of the area of reduced shielding. The staff modeled both Pu-239 and Cm-244 neutron sources. The staff did not model the secondary point source shown in Figure 2 of Reference 5-8. The staff's calculation results confirmed that the HAC dose limits are not exceeded. This provides the staff reasonable assurance that the dose limits in 10 CFR 71.51(a)(2) will not be exceeded during an accident condition.

Additionally, the application initially included a discussion of some lead slumping, to the amount of 0.513 inches. Thus, the staff considered lead slumping by replacing the top 0.513 inches of lead near the top of the cask with void. The staff modeled a point source located in the top right corner of a NS30 canister that had been shifted to the top right of the RH-TRU cask. The staff modeled both neutron and gamma sources. The dose results for a neutron source (Pu-239) at 1 meter was similar to the dose results for the puncture bar case because the amount of lead doesn't have a large effect on the neutrons. The dose results for a gamma source (Co-60) were less than the puncture bar case because the package configuration does not allow the point source to reach a position of less shielding versus the shielding near the puncture impact, thus mitigating any streaming effect the lead slump region (of only 0.513 inches) may yield. However, as previously stated (See Sections 2 and 5.3 of this SER), the applicant modified its evaluations and demonstrated that lead slump will not occur.

To confirm that a canister loaded to the HAC activity limits specified in Table 7 of Reference 5-8 would meet the NCT dose limits in 10 CFR 71.47(b), the staff modeled a distributed source inside the NS30 canister. The staff's calculations show that an evenly distributed Pu-239 neutron source without self-shielding exceeds the NCT dose limits. Thus, some, potentially significant amount of self-shielding is needed in any evaluation to demonstrate compliance with the NCT dose rate limits. As noted previously (Section 5.1.3 of this SER), the applicant did not provide an evaluation in the application to demonstrate that the proposed contents limits meet the NCT dose rate limits.

5.5 Evaluation Findings

Based on the staff's review of the statements and representations in the shielding sections of the application, the staff finds reasonable assurance that the package meets the shielding and the external radiation requirements of 10 CFR Part 71.

5.6 References

5-1 RH-TRU 72-B SAR Rev. 5, Docket No 71-9212, February 2011.

5-2 U.S. Department of Energy (DOE), *Remote-Handled Transuranic Waste Authorized Methods for Payload Control* (RH-TRAMPAC), U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico, Rev. 1, April 2011.

5-3 U.S. Department of Energy (DOE), *RH-TRU Payload Appendices*, U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico, Rev. 1, February 2011.

5-4 U.S. Nuclear Regulatory Commission, Safety Evaluation Report, Docket No. 71-9212, Model No. RH-TRU 72-B Package, Certificate of Compliance No. 9212, Revision 2, December 27, 2002 (ADAMS Accession No. ML023640220).

5-5 Title 10, Code of Federal Regulations, Part 71 (10 CFR Part 71), *Packaging and Transportation of Radioactive Material*, 01-01-10 Edition.

5-6 American Nuclear Society, ANSI/ANS 6.1.1 1977, *Neutron and Gamma-Ray Flux-to-Dose-Rate Factors*, La Grange Park, Illinois.

5-7 American Nuclear Society, ANSI/ANS 6.4.3 1991, *Gamma-Ray Attenuation Coefficients & Buildup Factors for Engineering Materials*, La Grange Park, Illinois.

5-8 R. J. Migliore and B. A. Day, *72-B with NS15 and NS30 Shielded Canister Shielding Analysis*, 01937.01.M005-03, Rev. 0, AREVA Federal Services LLC, Tacoma, WA, January 2010.

5-9 PacTec, *Neutron Source Rates for TRU Waste*, ED-042 Rev. 1, October 31, 2000, (ADAMS Accession No. ML003775749).

5-10 Washington TRU Solutions LLC. *Supplemental Information Regarding Application for Revision 5 of the RH-TRU 72-B Shipping Package*, April 19, 2010, (ADAMS Accession No. ML101121102).

6.0 CRITICALITY EVALUATION

Washington TRU Solutions LLC (the applicant) submitted an application, on behalf of the U.S. Department of Energy (DOE), for an amendment to the RH-TRU 72-B CoC to incorporate NS15 and NS30 neutron shielded canisters as approved canisters for use in the RH-TRU 72-B transportation package. The applicant also requested an increase in the allowable activities of radionuclides that can be transported in the package due to the additional shielding provided by the NS15 and NS30 canisters.

The RH-TRU 72-B transportation package has previously been approved to transport transuranic waste. The reactivity effect of the changes in the current amendment is bounded by Case C from the previous amendment. The staff's previous criticality review was documented in a staff SER dated July 28, 2006 (Reference 6-1).

The staff reviewed the addition of the new contents using the guidance in Section 6 of NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material," May 1999. The staff's evaluation of the applicant's changes to the criticality evaluation follows. This evaluation is based on SAR Revision 5 dated February 2011, RH-TRAMPAC Revision 1 dated April 2011, and RH-TRU Payload Appendices Revision 1 dated February 2011.

6.1 Description of Criticality Design

The applicant provided a criticality safety evaluation of the RH-TRU 72-B transportation package. This evaluation defines generic bounding contents to cover all possible constituents. The neutron multiplication factor (k-effective, or k_{eff}) will be less than 0.9382 during all normal and accident conditions.

6.1.1 Design Features

The applicant provided drawings of the package in Section 1.3.1 of Reference 5-1, and Section 2.8 of Reference 5-2. The staff reviewed these drawings and find that they sufficiently describe the locations, dimensions, and tolerances of the containment system including the NS15 and NS30 canisters. Therefore the staff finds that the applicant meets the requirements of 10 CFR 71.31(a)(1) and 10 CFR 71.33(a)(5) with respect to the criticality evaluation.

For criticality control the applicant relies on controlling the quantity of fissile material. Although the NS15 and NS30 canister will hold multiple drums of TRU waste, the applicant does not take credit for the presence or geometry of the drums. In addition there are no fixed neutron absorbers being credited in the analysis. However, there is a large amount of HDPE that is used as a neutron shield material and acts as an excellent moderator.

6.1.2 Codes and Standards

The applicant identified the appropriate regulations in 10 CFR Part 71 throughout SAR Section 6. The applicant used ANSI/ANS-8.1, -8.12, and -8.15 (References 6-4, 6-5, and 6-6) in the criticality design and control. These standards have been endorsed by the NRC in Reference 6-7, so the staff finds that following the guidance in the standards is acceptable. The applicant used the standards to create the ^{239}Pu Fissile Grams Equivalent (FGE) and ^{235}U Fissile Equivalent Mass (FEM) conversion factors listed in Reference 5-2 Table 5.1-1; that was previously approved in Reference 6-3. The staff finds that the applicant has met 10 CFR 71.31(c) by providing the codes and standards used in the criticality evaluation.

6.1.3 Summary Table of Criticality Evaluations

The staff reviewed the summary table of criticality evaluations (Reference 5-1, Table 6.1-2). The applicant divided the analyses into four cases depending on the RH-TRU 72-B contents descriptions (denoted to as Cases A through D). TRU waste in neutron shielded canisters falls under Case C, which assumes the RH-TRU 72-B's inner volume is filled with polyethylene, due to the large amount of HDPE in the neutron shielded canisters. The staff finds that the applicant

provided the k_{eff} of the configuration that had the highest reactivity. For Case C, the applicant determined that the highest reactivity resulted from an infinite array of HAC canisters with full density water in the spaces between the canisters. The canister spacing was based on half the thickness of the impact limiters.

In the summary table, the applicant shows that the maximum k_{eff} for the limiting configuration is less than the Case C upper subcritical limit (USL) of 0.9382. The staff finds that this meets the requirements of 10 CFR 71.55(b), (d) and (e).

6.1.4 Criticality Safety Index

In SAR Section 6.1, the applicant states that the Criticality Safety Index (CSI) is 0. This is based on analyses performed using a value of N to be infinity (i.e. an infinite array of casks). The staff finds that the CSI was appropriately determined per 10 CFR 71.59(b).

6.2 Fissile Material Contents

The applicant allows up to 245 ^{239}Pu Fissile Grams Equivalent (FGE) to be loaded into a neutron shielded canister. The applicant does not allow credit for ^{240}Pu to increase the fissile material limit for NS15 and NS30 canisters. Nor does the applicant allow the use of ^{235}U Fissile Equivalent Mass (FEM) limits. The applicant's use of FGE and FEM was previously accepted by the staff in Reference 6-1 Section 6. The staff finds that limiting the loading of fissile material to 245 FGE meets 10 CFR 71.33(b)(2). The content limits applicable to the NS15 and NS30 waste containers are listed in Table 3.1-1 of Reference 5-2, which is incorporated into the CoC by reference.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The staff reviewed Sections 2 (structural evaluation) and 3 (thermal evaluation) of the SAR to determine the effects of the NCT and HAC on the packaging and its contents. The staff reviewed the information in Section 2.6 of the applicant's SAR and the applicant concludes that there will be no damage to the package as a result of NCT, which includes the 1-foot NCT drop test. This information was previously reviewed and approved by the staff and documented in Reference 6-3. Single package NCT and HAC models are the same and neglect the impact limiter. The NCT package array is bounded by the HAC array; therefore, only the HAC array model was developed. This array model assumes the impact limiters are present but with only half their thickness for all contents except Case D contents, for which the impact limiters are neglected. Only the spacing afforded by the impact limiter is credited in the array model. The staff notes that this amount of spacing appears to be inconsistent and non-conservative with respect to the impact limiter deflection reported in Chapter 2.0 of the SAR (see Table 2.10.3-10 of Reference 5-1). In that chapter the deflection is about 11.5 inches, which would leave less than half the thickness of the impact limiter around the OC.

The staff examined the sketches of the models used for the criticality calculations and verified that the dimensions and materials are consistent with those in the drawings of the actual package. The staff verified that the SAR considers deviations from nominal design configurations by analyzing conservative conditions. The staff finds this acceptable and finds that the differences will not impact the criticality safety evaluation. The staff considered the

difference in the as-modeled HAC impact limiter configuration from that used in the applicant's criticality analysis as part of its confirmatory analyses.

6.3.2 Material Properties

The staff verified that appropriate mass fractions and densities are provided for materials used in the models of the packaging and contents. The applicant provided this information in Table 6.3-7 in the SAR. The HDPE neutron shield has a density of 0.94 g/cc, as shown in Section 3.2 of Reference 5-8. The applicant bounds the HDPE, as discussed in Section 6.3.4 of this SER, instead of explicitly modeling it. There are no materials used in the model that change form in HACs, such as a neutron shield or neutron absorbers that could melt, that are important to maintaining criticality calculations. For a discussion of the moderators modeled by the applicant see Section 6.3.4 of this SER. See Section 3 of this SER for the Thermal analysis.

The applicant does not use any materials as neutron absorbers and the applicant does not request credit for burnable poisons in the fuel. The staff did not identify any criticality properties that could degrade during the service life of the package. The applicant does assume that the waste reconfigures into an optimally moderated sphere; this is assumed for both HAC and NCT.

6.3.3 Computer Codes and Cross-Section Libraries

The applicant performed the criticality evaluations with the three-dimensional Monte Carlo computer code KENO-V.a of the SCALE v4.4a software package. This code is widely used in industry application for criticality calculations and has therefore been extensively benchmarked against critical experiments. The use of this code was previously accepted by the staff in Reference 6-1 Section 6.

The cross-section data used was the 238 energy-group (238GROUPNDF5) cross-section library based on the ENDF/B-V data. This library was used in both the criticality evaluations and in the benchmark calculations. The 238GROUPNDF5 cross-section library comes with the SCALE v4.4a software package and is commonly used by industry. The use of this cross-section library was previously accepted by the staff in Reference 6-1 Section 6.

The applicant stated that the criticality calculations were performed with enough neutron histories to reduce the computed uncertainty down to about 0.1%. The applicant provided a sample output file for Case C contents at various H-to-Pu-239 ratios, including the most reactive case.

6.3.4 Demonstration of Maximum Reactivity

The applicant models the fissile contents as 245 FGE of ^{239}Pu as this is bounding or equivalent for the other allowable fissile contents. The applicant uses polyethylene (CH_2) as a bounding hydrogenous moderating material. The package may include "special reflecting" materials; so, the applicant uses beryllium (Be) to bound these special reflecting materials. Transuranic wastes with more than 1% special reflectors, by weight, are not authorized to be shipped in the RH-TRU 72-B with NS15 or NS30 canisters. The staff finds these bounding assumptions to be acceptable, as they were approved in the previous criticality review of the RH-TRU 72-B (Reference 6-1 Section 6) and in Reference 6-3 Section 6.

The applicant's model assumes a homogeneous sphere of ^{239}Pu, polyethylene and beryllium. The applicant performed analyses to determine the optimal H/Pu ratio by varying the

polyethylene mass in the fissile sphere. The beryllium mass was adjusted to be 1% of the polyethylene mass.

In the applicant's previous analysis, for Case C, a polyethylene packing fraction of 100% was used (Reference 5-1 Section 6.2.3) and approved by the staff (Reference 6-1). In the analysis, an optimally moderated sphere of polyethylene and 245 FGE ^{239}Pu was placed in a RH-TRU 72-B that was modeled as filled with 99% polyethylene and 1% beryllium (by volume). Since the HDPE in the NS15 and NS30 canisters is equivalent to polyethylene with a 100% packing factor, and since the HDPE in the canisters does not fill the inner volume of the RH-TRU 72-B, the NS15 and NS30 canisters are bounded by Case C of the current analysis. For this reason the applicant did not perform new criticality models for the amendment. The staff finds modeling 1% beryllium (by volume) to be acceptable since beryllium is more dense than polyethylene; so, 1% by volume is more than 1% by mass resulting in a higher reactivity. The staff finds that the applicant has met 10 CFR 71.83, by assuming bounding moderation.

The applicant's model assumed that canisters are arranged in an infinite square pitched planar array. The spacing between canisters in the array is provided by half-thickness impact limiters.

6.3.5 Confirmatory Analyses

The staff performed a number of independent calculations to confirm the applicant's results and that the most reactive conditions had been correctly identified. In particular, the staff evaluated whether the reactivity of fissile material in NS15 and NS30 neutron shielded canisters is bounded by the Case C model. The staff constructed its models using information from the SAR and RH-TRAMPAC. The staff used the KENO-VI code, of the SCALE6 code system, with the 238GROUPNDF5 cross section library derived from ENDF-V data. The staff assumed the composition of HDPE is 99% polyethylene and 1% beryllium (by volume) with densities of 0.923 and 1.85 g/cc respectively, which are the default densities in SCALE. The staff confirmatory analyses also included investigations of the effects of the thinner impact limiters (per the structural evaluation vs. the thickness assumed in the applicant's criticality analyses), variations in density of water between packages and changes to the array (i.e., hexagonal vs. square-pitch array).

The staff used the equations in SAR Section 6.3.5 to calculate the composition and radius of the moderated Plutonium sphere for a given H/Pu ratio. The staff also used those equations to check the compositions in SAR Table 6.3-4. The staff's calculations confirmed that the most reactive sphere composition was around 900 H/Pu as shown in SAR Table 6.4-9.

The staff's calculations indicate that there is not significant variation in k_{eff} for the different arrays that were considered. There was some indication that different water densities may result in a more reactive system; however, the difference was small. The difference in the thickness of impact limiters also had little impact on k_{eff}. The staff's calculations yielded maximum k_{eff} values slightly higher (<0.15%) than the applicant's USL. The applicant's most reactive case is only slightly lower than its USL (~0.1%). Thus, based upon the conservatisms in the contents model and risk-informed considerations, the staff finds there is reasonable assurance that an array of packages will remain subcritical.

6.4 Package Evaluation

6.4.1 Configuration

The staff verified that the applicant's evaluation demonstrates that a single package is subcritical under both NCT and HAC. The NCT and HAC models are the same for the applicant's criticality evaluation. As stated in Sections 6.3.1 of this SER, damage to the cask is limited to the impact limiter; so, the applicant modeled the impact limiter at half its nominal thickness in all calculations (See Reference 5-1 Figure 6.3-1). The applicant floods the empty spaces inside the cask with water when performing calculations for the damaged condition. However, for Case C there was no room for flooding because all available space was filled with the 99% polyethylene and 1% beryllium reflector. Due to the thickness and density of this reflector, the contribution to reactivity from the infinite array of canisters is negligible. Therefore, the infinite array analysis was used for the single package evaluation as well as the array evaluation (Reference 5-1 Section 6.4.3.1.3).

The applicant's calculations using an infinite array of packages were subcritical; therefore, the applicant specified a CSI of 0.0. The applicant modeled the most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents. This is discussed in Sections 6.1.1, 6.2, and 6.3.4 of this SER. The applicant modeled full density water reflection on all sides of the packages in the infinite array; the spacing of the array was half the thickness of the impact limiters. The applicant's Case C model has not changed since it was reviewed and accepted by the staff in Reference 6-1.

6.4.2 Results

The staff confirmed that the results of the applicant's criticality calculations are consistent with the information presented in the summary table discussed in Section 6.1.3 of this SER. Since values of k_{eff} are less than the USL of 0.9382 under the tests specified in 10 CFR 71.71 and 71.73, the staff verified that this meets the requirements of 10 CFR 71.55(d)(1) and 71.55(e) which requires that the contents be subcritical.

Since the applicant performed the criticality evaluations using reasonably bounding geometry, the staff verified that the geometry could not be altered so as to affect the conclusions of the criticality safety analysis. The staff finds that the applicant meets 10 CFR 71.55(d)(2). The applicant modeled the fissile material rearranging into the most reactive credible configuration. The staff finds that this meets 10 CFR 71.55(e)(1).

The applicant performed calculations where moderation is present to the maximum credible extent. The staff finds that this meets 10 CFR 71.55(d)(3) and 55(e)(2). The applicant performed calculations with the canister fully reflected by water on all sides. The staff finds that this meets 10 CFR 71.55(e)(3).

For the NCT tests specified in 10 CFR 71.71, the staff verified that there will be no substantial reduction in the effectiveness of the packaging for criticality prevention, including (1) the total effective volume of the packaging on which nuclear safety is assessed will not be reduced by more than 5%, (2) the effective spacing between the fissile contents and the outer surface of the package is not reduced by more than 5%, and (3) there is no occurrence of an aperture in the

outer surface of the packaging large enough to permit the entry of a 10 cm cube. The staff finds that this condition meets the requirements in 10 CFR 71.55(d)(4).

6.5 Benchmark Evaluations

The benchmarks the applicant uses have not changed since they were approved by NRC staff in the previous criticality review. The HDPE of the NS15 and NS30 Neutron Shielded canisters is within the area of applicability of the applicant's benchmarks; so, the staff finds the applicant's use of benchmarks to be acceptable. For the staff's review of the benchmarks see Reference 6.1 Section 6.0.

6.6 Evaluation Findings

Based on review of the statements and representations in the application and independent confirmatory calculations, the staff finds reasonable assurance that the nuclear criticality safety design has been adequately described and evaluated and that the package meets the subcriticality requirements of 10 CFR Part 71.

6.7 References

6-1 Letter from C.M. Reagan (U.S. NRC) to P.C. Gregory (Washington TRU Solutions), Safety Evaluation Report, *Revision 4 of the RH-TRU 72B Shipping Package (Docket No. 95-19-9518)*, July 28, 2006 (ADAMS Accession No. ML062130059).

6-2 NUREG-1609, *Standard Review Plan for Transportation Packages for Radioactive Material*, May 1999.

6-3 U.S. Nuclear Regulatory Commission, Safety Evaluation Report, Docket No. 71-9218, Model No. TRUPACT-II Package, Certificate of Compliance No. 9218, Revision 17, August 23, 2004 (ADAMS Accession No. ML042370439).

6-4 American Nuclear Society, ANSI/ANS-8.1-1998, *Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors,* La Grange Park, Illinois.

6-5 American Nuclear Society, ANSI/ANS-8.12-1987, *Nuclear Criticality Control and Safety of Plutonium-Uranium Fuel Mixtures Outside Reactors*, La Grange Park, Illinois.

6-6 American Nuclear Society, ANSI/ANS-8.15-1981, *Nuclear Criticality Control of Special Actinide Elements,* La Grange Park, Illinois.

6-7 U.S. Nuclear Regulatory Commission, Regulatory Guide 3.71 Revision 1, *Nuclear Criticality Safety Standards for Fuels and Material Facilities*, October 2005.

6-8 Oak Ridge National Laboratory (ORNL), *SCALE 4.4a: Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstation and Personal Computers*, NUREG/CR-0200, ORNL/NURE/CSD-2/R6, March 2000.

6-9 U.S. Nuclear Regulatory Commission, Safety Evaluation Report, Docket No. 71-9212, Model No. RH-TRU 72-B Package, Certificate of Compliance No. 9212, Revision 4, July 28, 2006 (ADAMS Accession No. ML062130059).

7.0 PACKAGE OPERATIONS

Chapter 7.0 of the SAR describes by what means the package is loaded, closed, and prepared for transport. Loading the RH-TRU 72-B package for transport involves: 1) prior loading and measuring the payload canisters, 2) dry-loading the prepared payload canister into the RH-TRU 72-B package, 3) leakage rate testing the RH-TRU 72-B package OC and, optionally, the IV seals following the guidelines of ANSI N14.5, and (4) securing the external impact limiters to the RH-TRU 72-B package.

The staff reviewed the package operations with respect to shielding and radiation protection. The operations include a description of the dose rate measurements that are made prior to shipment for a loaded package. These measurements are made to fulfill the requirement in 10 CFR 71.87(j). The operations in Chapter 7.0 of the SAR incorporate by reference Section 3.2 of RH-TRAMPAC, Revision 1, dated April 2011 to provide greater detail on the method of performing the measurements. The staff finds that the package operations meet the requirements of 10 CFR 71.87(j).

Additionally, as discussed in Section 5.0 of this SER, the applicant commits to comprehensive pre-shipment measurements to demonstrate package compliance with 10 CFR 71.35(a) and 71.47. The staff agrees the pre-shipment measurements are comprehensive, and cover the entire package surface and areas at 2 meters from the vehicle edge. An evaluation of the pre-shipment measurements for this purpose is described in Section 5.0 of this SER.

The staff also finds the operations for package unloading, transport of an empty package and shipment of the package as Low Specific Activity material are acceptable and meet the appropriate regulatory requirements.

Chapter 7.0 of the SAR has been updated to include a revised description regarding the use of IV holes with a handling fixture in lieu of only using eyebolts and clarified the option to remove/replace the package to/from the Center-Pivot Trailer. In addition, an option has been added to install the OC lid closure bolts and/or impact limiter bolts with 100-200 lb-ft torque for empty shipments. This reduces undue wear on the threads while maintaining sufficient torque to preclude loosening during transport.

The staff has reviewed the changes to the Operating Procedures and concludes that the applicable requirements of 10 CFR Part 71 have been met.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1 Acceptance Tests

Chapter 8.0 of the SAR describes the acceptance test and the maintenance program for the RH-TRU 72-B Package. Section 8.1.3.1.1 details testing the IV structure integrity using the evacuated envelope method and has been revised to evacuate the IV cavity at an ambient temperature of 40°F, or greater. Ambient temperature of 40°F, or greater is sufficient to ensure proper operation of leakage rate test equipment and consistent with that required for other WIPP Type B packages.

The staff reviewed the gamma shielding acceptance test (note there are 2 techniques for the test) and found that, accounting for the information regarding the grid size used in the test, that there is reasonable assurance that the shielding acceptance test meets the requirements of 10 CFR Part 71. The staff does note that the test in this instance is acceptable; however, the grid size should typically be specified in the acceptance test description in Chapter 8.0 of the application. Depending on the nature of the request, the next submittal for a licensing action for this certificate should include the grid size in the application's Chapter 8.0 description of the shielding acceptance test. The staff also notes that the drawings do not have a tolerance on the lead thickness. However, the drawings do have a note (see note 5 of Drawing No. X-106-500-SNP, Rev. 5, sheet 1 of 8), which ties the lead shielding explicitly to the acceptance test. Thus, the shielding acceptance test acceptance criterion acts to set a tolerance on the lead thickness. Therefore, the staff finds the shielding integrity test for lead shielding to be acceptable.

The staff also considered the need for an acceptance test for the neutron shield pipe in the NS15 and NS30 canisters. The pipe material is a known, non-proprietary material (high density polyethylene) with specifications governed by the standards listed on the licensing drawings. Thus, the staff finds that the acceptance test described in Section 8.1.1 to verify compliance with the licensing drawing requirements provide adequate assurance of neutron shielding integrity and that additional testing for neutron shielding integrity is not necessary.

The staff has reviewed the changes to the Acceptance Tests and Maintenance Program and concludes that the requirements of 10 CFR Part 71 have been met.

CHANGES TO CERTIFICATE CONDITIONS

The following changes were made to the Certificate of Compliance:

The title and identification of report or application 3(b) changed to reflect the consolidated application submitted.

Condition No. 5(a)(3) was updated to reference the revised and new drawings.

Condition No. 5(b)(1) was updated to reference the new revision to RH-TRAMPAC.

Condition No. 5(b)(2) was updated to reference the new revision to RH-TRAMPAC and the new decay heat limit. The low enriched uranium authorization was also clarified.

Condition No. 6 was updated to reference the new revision to RH-TRAMPAC.

Condition No. 7 was updated to reference the new revision to RH-TRAMPAC.

Condition No. 9 was updated to reference the new revision to RH-TRAMPAC.

New Condition No. 10 was added to note the maximum shipping periods specified in the RH-TRU Payload Appendices. This affected the numbering of Condition Nos. 11-17.

New Condition No. 11(c) was added to specify containment O-ring seal material qualification and tests as specified in the RH-TRU 72-B SAR.

Condition No. 13 was updated to remove package marking allowance prior to July 31, 2007.

Condition No. 14 was revised to allow the use of the previous revision of the certificate for a period of approximately one year.

New Condition No. 16. was added to specify exclusive use transport. This affected the numbering of Condition No. 17.

The February 12, 2010, consolidated submittal and supplements dated April 19, 2010, August 30, 2010, February 16, 2011, and April 14, 2011, were included in the Reference Section.

CONCLUSION

Based on the statements and representations contained in the application, the staff concludes that the Model No. RH-TRU 72-B package meets the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9212, Revision No. 6,
on June 17, 2011.