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

July 28, 2006

Mr. Phillip. C. Gregory
Manager, Packaging Engineering
Washington TRU Solutions, LLC
P.O. Box 2078
Carlsbad, NM 88221-2078

SUBJECT: REVISION 4 OF THE RH-TRU 72-B SHIPPING PACKAGE (DOCKET 71-9212)

Dear Mr. Gregory:

As requested by your application dated October 14, 2005, as supplemented June 5, 2006, enclosed is Certificate of Compliance No. 9212, Revision No. 4 for the Model No. RH-TRU 72-B package. The Certificate of Compliance has been issued to the U.S. Department of Energy (DOE). The staff has documented its review of the changes associated with this revision in the enclosed Safety Evaluation Report.

The 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. 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. Registered Users may request by letter to remove their names from the Registered Users List.

If you have any questions concerning this matter, please contact me or Jill Caverly of my staff at (301) 415-6699.

Sincerely,

Stewart Brown for

Christopher M. Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9212
TAC No. L23913

Enclosure: 1. Certificate of Compliance No. 9212, Rev. No. 4
2. Safety Evaluation Report
3. Registered Users

cc: w/o encls: R. Boyle, Department of Transportation
J. Shuler, Department of Energy
RAMCERTS

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	4	71-9212	USA/9212/B(M)F-96	1	OF	4

2. PREAMBLE

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

Westinghouse TRU Solutions, LLC application dated November 27, 2002, 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.

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.

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

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 Packaging Technology Drawing No. X-106-500-SNP, Sheets 1-8, Rev. 4.

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.

(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 (Revision 0).

(2) Maximum quantity of material per package.

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

Fissile material not to exceed limits described in Section 3.1, "Nuclear Criticality" of RH-TRAMPAC (Revision 0). Pu-239 equivalent is determined in accordance with RH-TRAMPAC (Revision 0). Low enriched uranium is authorized for waste containers containing material that is primarily uranium (in terms of heavy metal component) and the waste matrix is 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 for organic wastes and 300 watts for inorganic waste, and not to exceed the limits in RH-TRAMPAC (Revision 0).

(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 (Revision 0).

7. Each waste canister must not exceed the decay heat limits determined as specified in RH-TRAMPAC (Revision 0), or must be tested for gas generation in accordance with RH-TRAMPAC (Revision 0), 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 RHTRAMPAC (Revision 0).

10. 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.

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

12 Packages may be marked with Package Identification Number USA/9212/B(M)F-85 until July 31, 2007 and must be marked with Package Identification Number USA/9212/B(M)F-96 after July 31, 2007.

13. Revision No. 3 of this certificate may be used until July 31, 2007.

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

15. Expiration date: February 28, 2010.

REFERENCES

Westinghouse TRU Solutions, LLC, application dated November 27, 2002.

Supplements dated: Washington TRU Solutions, LLC, November 1, 2004; October 14, 2005; and June 5, 2006.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Stuart W. Brown for

Christopher M. Regan, Acting Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: July 28, 2006

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

SAFETY EVALUATION REPORT

Docket No. 71-9212
Model No. RH-TRU 72-B Package
Certificate of Compliance No. 9212
Revision 4

1.0 SUMMARY

By application dated October 14, 2005, and supplemented June 5, 2006, Washington TRU Solutions, LLC, submitted to the U.S. Nuclear Regulatory Commission (NRC), Revision 4 to Certificate of Compliance No. 9212 for the Model No. RH-TRU 72-B transport package. The application requested approval of the following revisions to the Safety Analysis Report (SAR):

- Reformatting of the SAR,
- Consolidation of payload requirement into a separate document,
- Creation of appendices for document supporting information,
- Revision of criticality evaluation,
- Revision of leakage testing,
- Inclusion of revised methodology for site- and waste-specific descriptions of authorized contents,
- Inclusion of a maximum shipping period,
- Inclusion of a revised pressure analysis,
- Payload and packaging related changes, and
- Changes to the RH-TRU packaging and RH-TRU canister design drawings.

Additionally, the applicant requested that the package be updated with the designation of "-96" signifying that it meets the requirements of updated 10 Code of Federal Regulations (CFR) Part 71 (effective October 2004).

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 and as approved in this Safety Evaluation Report (SER).

1.1 Package Description

The RH-TRU 72-B package was developed as a safe means of transporting, via ground transportation only, remote-handled transuranic wastes from various sites around the United States. The package is composed of an inner vessel (IV), that provides an inner containment boundary, an outer cask (OC) that provides an outer containment boundary and acts as an environmental barrier, and polyurethane foam filled energy absorbing impact limiters at each end of the outer cask. The package is designed for truck and rail transport. The maximum allowable total weight of the loaded RH-TRU 72-B package is 45,000 pounds. The empty package weighs approximately 37,000 pounds. The maximum allowable total weight of the loaded payload canister is 8,000 pounds. The payload of the RH-TRU 72-B package consists of one payload canister of waste.

The quantity and type of material permitted in this package is byproduct, source and special nuclear materials and waste, within a stainless or carbon steel waste canister. Specific requirements of the payload are described in the document titled, "The Remote-Handled Transuranic Waste Authorized Methods for Payload Control" (RH-TRAMPAC). This document serves as the governing document for shipment for the RH-TRU 72-B package by defining applicable requirements for a payload, describing acceptable methods that shall be used to prepare and characterize payload, and identifying the quality assurance program to be applied. The RH-TRAMPAC has been revised to be consistent with the payload characterization requirements of the previously approved RH-TRAMPAC for the TRUPACT-II and HalfPACT.

The package is constructed and assembled in accordance with Washington TRU Solutions, LLC, Drawings No. X-106-500-SNP, Sheets 1-8, Rev. 4; X-106-501-SNP, Rev. 4; and, X-106-502-SNP, Rev. 2.

1.2 Compliance With the Requirements of 10 CFR PART 71, FINAL RULE "-96" Update

10 CFR Part 71, Packaging and Transportation of Radioactive Material, was updated for compatibility with the International Atomic Energy Agency standards on October 1, 2004 (69 FR 3786). As a result of this update, transportation packages should be updated to meet the revised requirements. The new designation of "-96" has been requested for Model No. RH-TRU 72-B. The staff has reviewed the issues associated with the updated 10 CFR Part 71.

Of the 19 issues identified during the update, several issues pertain to this package. Below is the staff's analysis of the issues associated with the update of this package.

Issue 5, Criticality Safety Index (CSI) requirements - requires the use of the term CSI instead of TI. The CoC has been updated to include the CSI.

Issue 7, Deep Immersion Test - requires enhanced water immersion test. The SAR includes an analysis for deep water immersion and the staff agrees with the conclusion that the external water pressure is of negligible consequence for this package.

Issue 10, Crush test fissile material packages - requires crush test for fissile material. Model No. RH-TRU 72-B weight exceeds 1100 pounds and, therefore, no crush test is required.

Issue 11, Fissile material for Aircraft Transport - requirements for aircraft transport of plutonium. The package is not authorized for aircraft transport. The CoC has been updated to explicitly state that this type of transport is not allowed.

Issue 17, Double Containment of Plutonium - the new rule removes double containment requirement under certain circumstances. The revised design does not require double containment, but gives the user the option of testing if double containment is desired for a particular shipment. The staff has reviewed the requirements associated with leak testing of the containment systems and finds them acceptable.

The staff has determined that the package meets the requirements of the revised 10 CFR Part 71 and can be designated with a "-96" indicating compliance with the new regulations. The CoC has been updated with the new designation.

2.0 STRUCTURAL

The RH-TRU 72-B package analysis was revised to include additional material and conditions associated with the requriements of the revised 10 CFR Part 71 (effective October, 2004). The applicant revised the material used for the containment to include Type F304 and Type 304L stainless steels in addition to the original Type 304 stainless steel. Since the properties of these new added materials are similar to the original SS-304 in terms of mechanical, chemical and fracture properties, no brittle fracture and corrosive or galvanic reactions are expected.

The applicant revised the definition of normal operating cycles for the OC and IV bolts from "time-based" to "cycle-based" in the evaluation of fatigue resistance. Since fatigue is controlled by stress cycles rather than real time (such as 460 operating cycles for the OC bolts and 550 service cycles for the IV bolts) according to ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, it is acceptable to redefine it as cycle-based rather than time-based.

The applicant revised the stress analyses of the trunnions, OC outer shell, and trunnion attachment weld under normal conditions of transport (NCT), including lifting and tie-down, NCT side drops and end drops, differential thermal expansion, immersion and vibration, by using a more accurate ANSYS finite element model. The resulting margins of safety for each component are all positive, meaning the calculated stresses are all less than the allowable stresses.

The applicant also revised stress calculations under hypothetical accident conditions including side and end drops, corner and oblique drops, crush, and end and side punctures. In these cases, the revised stresses were determined to be less than the allowable stresses. For the end puncture analysis, maximum stresses occurred in the end plate due to a 29.5g end puncture. In its initial submittal, the applicant applied different boundary conditions to the top and bottom plates. In order to verify the applicants analysis, the staff performed an analysis that applied the same boundary conditions to both top and bottom plates. The results of the staff's calculation was a maximum stress of 88,799 psi which exceeds the allowable stress (67700 psi) by about 31%. Staff questioned this result in its Request for Additional Information. In response to the staff's questions, the applicant utilized a more precise approach of a finite element method (as given in Appendix 2.10.1.6 of the application) to calculate maximum stress. The results of this analysis determined the maximum stress is 66,760 psi and below the allowable stress. The staff reviewed the revised methodology and determined that the final stress calculation is based on a more precise methodology and that the result, although different from the original stress calculation, is acceptable.

Improved evaluations and design criteria of the buckling of the OC and IV shells are incorporated in Appendix 2.10.5 to ensure its accuracy, consistency, and completeness. A complete summary of stresses was included in Appendix 2.5.6. The detailed buckling evaluation was revised for the worst case for the OC inner shell using Load Combination 6 in Appendix 2.10.5.7. The staff reviewed the revisions and agrees that they are reasonable.

In Section 2.10.6.2, Analysis Methodology, a revision was made to include a possibility of using chrome plated bolts instead of using Cd plated bolts. It was determined that the torque coefficient K factor for the Cd plated bolts lies between 0.11-0.15, and this value will have a significant impact on the resulting pre-stress computed for the bolts. The K value used to calculate the pre-stress of the new chrome plated bolts is 0.2. The data or reference source on which this value is based is hereby provided, http://raskcycle.com/techtip/webdoc14.html.

The applicant has shown and the staff agrees that the Model No. RH-TRU 72-B, with the changes described above and in the SAR, Revision 4, continues to meet the structural requirements of 10 CFR Parts 71.31, 33, 35, 45, and 71.

3.0 THERMAL

The package is composed of three concentric cylindrical vessels: the OC, the IV, and the payload canister. The OC is a stainless steel cylinder that provides a containment boundary and also acts as an environmental barrier. The IV is a stainless steel cylinder placed inside the OC and can provide an optional containment barrier. The payload canister is made of carbon or stainless steel and is positioned inside the IV. All RH-TRU waste will be loaded directly into the payload canister or into inner containers within the payload canister. Up to three 50-gallon drums can be placed inside the payload canister. The package thermal protection is provided by a fire shield surrounding the outer cask exterior wall, and the polyurethane foam impact limiters covering each end of the OC. The fire shield consists of a 10-gauge stainless steel sheet offset from the OC body by a 12-gauge stainless steel wire wrap on a 3-inch pitch.

3.1 Shipping Period, Decay Heat, and Radioactive Contents

The applicant proposed two shipping procedures in order to conservatively establish the number of days from the closure of the package at the originating site until its venting at the receiving site (Waste Isolation Pilot Program, WIPP). Under NCT and monitoring, the applicant conservatively derives a maximum period of 60 days as necessary for the shipment to be completed. Shipments by rail should meet the same 60-day period limit requirement for truck shipment. Using special administrative controls, described in appendix 2.4 of the RH-TRU Payload Appendices, a 10-day shipment can be achieved. Included in these administrative controls are the requirements for a maximum of 24 hours for both loading (time spent between sealing of the IV and departure of the shipment) and unloading (time spent between arrival of the shipment and venting of the IV) of the waste. The 10-day controlled shipment is an option available to sites that elect to impose all the extra steps necessary to follow the specified conditions.

The RH-TRU 72-B package is designed to accommodate a maximum heat load of 300 watts. At this thermal loading condition and with the package immersed in 38°C (100°F) still air and in the shade, all its accessible surfaces are predicted at or below 50°C (122°F). Therefore, per 10 CFR 71.43(g), the RH-TRU 72-B package qualifies for a non-exclusive use shipment.

Other important factors, however, also contribute when establishing the allowable heat load for a given shipment. Due to radiolysis, some of the waste can generate flammable gases (e.g., hydrogen) within the leaktight transportation package. The internal pressure during shipment may also be affected by the generation of internal gases. For these reasons, gas concentration and pressures during transport are restricted as follows:

- For any package containing water and/or organic substances that could radiolytically generate combustible gases, determination must be made by tests and measurements or by analysis of a representative package such that the following criterion is met over a period of time that is twice the expected shipment time: The hydrogen generated must be limited to a molar quantity that would be no more than 5% by volume in the innermost layer of confinement (or equivalent limits for other inflammable gases) if present at standard temperature and pressure.

- The gases generated in the payload and released into the RH-TRU 72-B IV cavity shall be controlled to maintain the pressure within the IV cavity below the acceptable design pressure of 150 psig.

The applicant presented a systematic procedure for deriving the limiting flammable gas generation rate (FGGR), based on the characteristics of the waste, its constituent materials, as well as its packaging components. The same procedure must be applied for all wastes that may be shipped inside the RH-TRU 72-B. The derived FGGR can be "translated" into a decay heat limit, following a methodology proposed by the applicant. The resulting decay heat limit is, most often, much smaller than the 300 watts limit that guarantees the accessible surfaces remain below 50°C. The applicant also presented a limiting calculation addressing the build up of gases and the rising internal pressure. Assuming a bounding cellulose laden waste, the calculation indicates that, after the 60-day shipping period, the internal pressure would be just below the design limit of 150 psig if the payload heat load was 23.5 watts.

In order to establish the limiting FGGR for any given content code, the minimum release rate values for filters and liner bags that may exist within the waste, given in Table 2.5-1 of the RH-TRU Payload Appendices, must be applied. These values must be verified prior to the shipping qualification of any waste.

3.2 Thermal Evaluation Under Normal Conditions of Transport

The applicant considered a single package in a horizontal position to analyze the thermal performance of the RH-TRU 72-B package design under Normal Conditions of Transportation. The cask internal heat generation was assumed to be uniformly distributed along the internal walls of the containment vessel. Two payload models were considered: a 50-watt payload consisting primarily of paper products and a 300-watt payload consisting primarily of metallic items. Inside the cask, both conduction and radiation are allowable means of heat transfer. The cask walls exchange heat with the surrounding environment through convection and radiation. Solar heat fluxes are applied to the external surfaces.

Using the SINDA '85/FLUINT computer program, the applicant simulated the loaded cask during both shaded and sunny conditions. Parametric studies were performed in order to evaluate the package responses to smaller heat loads, besides the proposed 50 and 300 watts.

The applicant predicted all component temperatures well below any operational limits even when insolation is accounted for. The temperature-sensitive O-rings were predicted to be at or below 140°F (60°C) when the package was exposed to the sun, which is far from the butyl material's allowable limit of 250°F (120°C) for long term use. The applicant also demonstrated that, for a heat load limit of 300 watts, the accessible external surface temperature remained below the regulatory limit of 50°C (122°F) without insolation, required for packages under nonexclusive use.

3.3 Thermal Evaluation Under Accident Conditions

The same SINDA/FLUINT model is used for the 30-minute hypothetical accident condition fire transient calculation with the exception that the impact limiter model is now conservatively modified so that drop damages can be accounted for. Once again, all components are predicted to remain below the allowable temperature limits. The highest lead temperature remains more than 60°F below the melting point of lead, while the highest O-ring seal temperature remains about 90°F below the recommended upper limit for the butyl O-ring seals of 250°F.

3.4 Internal Pressure

The Maximum Normal Operating Pressure (MNOP) was calculated at approximately 139 psig, after conservatively assuming a limited internal space, the release of gases due to radiolysis during the 60-day shipping period, and 100% humidity at loading time. The MNOP pressure was below the design limit of 150 psig. For the HAC conditions, the applicant arrives at a maximum internal pressure of approximately 179 psig, which is still below the transient design pressure value of 300 psig.

3.5 Conclusion

The staff independently verified some of the details of the proposed transportation package, confirming that the package design provides sufficient thermal safety margins for all its components. Based on the documentation and information supplied by the applicant, and the review and calculations performed by staff, there is reasonable assurance that the package design meets the thermal requirements of 10 CFR Part 71.

4.0 CONTAINMENT

The RH-TRU 72-B shipping package is used for ground (by truck or rail) transportation of remote-handled transuranic (RH-TRU) wastes from various sites within the United States. The package is basically composed of three concentric cylindrical vessels: the outer cask (OC), the inner vessel (IV) and the payload canister. The OC is a stainless steel cylinder that provides a containment boundary and also acts as an environmental barrier. The IV is a stainless steel cylinder placed inside the OC and can provide an optional containment barrier. The payload canister is made of carbon or stainless steel and is positioned inside the IV. All RH-TRU waste will be loaded directly into the payload canister or into inner containers within the payload canister. Up to three 50-gallon drums can be placed inside the payload canister.

The RH-TRU 72-B package is designed with two independent levels of containment (the IV and the OC vessels). In the current amendment, however, the applicant proposes to make the preshipment leakage test of the IV (inner vessel) optional. All the other leakage rate tests (fabrication, maintenance, and periodic) are unchanged: a leaktight (1×10^{-7} ref-cm^3/sec air) condition must be determined, based on procedures described in the ANSI N14.5-1997 standards. The optional approach for the pre-shipment leakage rate test of the IV is acceptable, based on the revised 10 CFR Part 71, which became effective on October 1, 2004. In the new rule, the double containment requirement for packages containing greater than 20 Curies of plutonium has been removed. Also proposed by the applicant is the option of performing a maintenance leakage rate test (1×10^{-7} ref-cm^3/sec air) instead of the usual preshipment leakage rate test (no detected leakage when tested to a sensitivity of 1×10^{-3} ref-cm^3/sec air). This proposal is also acceptable since it encompasses a more conservative approach.

The procedure for pre-shipment leakage rate testing has been modified, as described in Appendix 7.4.1 of the SAR. Based on a gas pressure rise process, the applicant established the criteria and procedures that will allow the successful determination that the package is ready for shipment.

In Chapter 8, Acceptance Tests and Maintenance Program, the applicant has clarified that the 150% MNOP hydrostatic tests of both the inner vessel and the outer cask are performed after the fabrication leakage rate tests. Section A.3.5 of the ANSI N14.5-1997 states that "for leaks smaller than 10^{-6} ref-cm^3/s, wetting of the test item before leakage rate test should be avoided," as some of the leak paths may become clogged by liquid if the hydrostatic test is conducted prior to the leakage rate test.

The staff reviewed the applicant's containment evaluation and agrees with the applicant's assumptions. The staff concludes that the leak test methods and acceptance, maintenance, periodic and preshipment criteria specified in the application are adequate, and that the package meets the containment requirements of 10 CFR 71.51.

5.0 SHIELDING

This section of the SAR was revised to reference RH-TRAMPAC, a document that contains previous appendices to the SAR. RH-TRAMPAC was called out from the main SAR to improve and simplify use.

6.0 CRITICALITY

The applicant revised the criticality analysis for the Model No. RH-TRU 72B for fissile material loadings based on the bounding moderating and reflecting properties of RH-TRU waste forms, and the amount of ^{240}Pu that can be verified to be present in the material. These analyses reflect the criticality analyses that have been previously approved for the TRUPACT-II and HalfPACT packages.

The applicant's revised analysis considered various configurations of fissile, moderating, and reflecting material, bounding the most reactive conditions which could occur in the packaging contents. These configurations are summarized in Section 6.0 of the SAR and include the following:

- Manually compacted (i.e., not machine compacted) waste with less than 1% special reflectors (Be, BeO, C, D2O, MgO, and depleted uranium (less than 0.72 wt% and greater than or equal to 0.3 wt% ^{235}U), which may contain up to 25 grams ^{240}Pu which can be credited in the criticality analysis,
- Manually compacted waste with greater than 1% special reflectors where the fissile material may or may not be chemically or mechanically bound to the special reflector,

- Machine compacted waste with less than 1% special reflectors, and
- Manually compacted waste with less than 0.96 weight percent ^{235}U fissile equivalent mass (FEM) distributed within non-fissile ^{238}U.

The low enriched uranium case is the same as was previously approved in Revision 3 of the RH-TRU 72-B SAR.

The resulting fissile material limits, in terms of either ^{239}Pu fissile gram equivalent (FGE) or ^{235}U FEM are summarized in Table 6.1-1 of the SAR, and in the following table:

Fissile Content Conditions	Fissile Material Limit Per Package			
	Case A	Case B	Case C	Case D
None	315 ^{239}Pu FGE	100 ^{239}Pu FGE	245 ^{239}Pu FGE	0.96 wt% ^{235}U FEM
>5g ^{240}Pu	325 ^{239}Pu FGE	-	-	-
>15g ^{240}Pu	350 ^{239}Pu FGE	-	-	-
>25g ^{240}Pu	370 ^{239}Pu FGE	-	-	-
Fissile Material Bound to Special Reflector	-	305 ^{239}Pu FGE	-	-

For all criticality calculations, the applicant used the CSAS25 module of SCALE 4.4a, with the KENO V.a three-dimensional Monte Carlo code and the associated 238 group ENDF/V cross-section library. The applicant's model for cases A through C consisted of a sphere of fissile and moderating material, either centered in the RH-TRU 72-B IV, or offset to one side of the package in an attempt to neutronically communicate with adjacent packages in an infinite array. The remaining volume of the IV was filled with varying concentrations of water, polyethylene, and special reflector, when applicable. The case A through C models are illustrated in Figures 6.3-2 through 6.3-4 of the SAR. The case D model consists of variously configured cylindrical solutions of fissile material, surrounded by reflecting mixtures of water, polyethylene, and special reflector, as illustrated in Figures 6.3-5 and 6.3-6. The resulting maximum k_{eff}s for each case are summarized in Section 6.4 of the SAR.

The applicant provided separate benchmarking analyses for the plutonium and low enriched uranium criticality analyses. For cases A through C, the applicant modeled 227 critical benchmark experiments from the OECD Nuclear Energy Agency's "International Handbook of Evaluated Criticality Safety Benchmark Experiments," 196 of which involved plutonium and the other 31 of which involved U-233 and beryllium. The same code and cross section set were used to model the critical benchmark experiments as were used for the RH-TRU 72B criticality analysis. The applicant used the results from the benchmarks, and the USLSTATS code, to develop an upper subcritical limit (USL) according to USL Method 1 described in NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages." The resulting USL was calculated to be 0.9382, which is above the highest k-eff calculated for the RH-TRU 72B package.

The applicant modeled 82 critical experiments involving low enriched uranium systems from the OECD Nuclear energy Agency's "International Handbook of Evaluated Criticality Safety Benchmark Experiments," including systems with uranium solutions and rods with reflectors of water or polyethylene. The criticality analysis enrichment of 0.96% was below the range of applicability of the benchmark experiments. However, due to the large margin in k_{eff} calculated by the applicant and the confirmatory analysis performed by the NRC staff, the calculated upper subcritical limit (USL) of 0.9257 is considered acceptable.

The applicant has shown and the staff agrees that the Model No. RH-TRU 72-B package with revised fissile material contents meets the criticality safety requirements for fissile material packages in 10 CFR 71.55 and the requirements for fissile material package arrays in 10 CFR 71.59.

7.0 OPERATING PROCEDURES

Chapter 7 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 outer cask, and optionally, the inner vessel seals, and 4) securing the external impact limiters to the RH-TRU 72-B package. Loading the payload canister will be done in accordance with the appendix to the SAR titled, *Remote-Handled Transuranic Waste Authorized Methods for Payload Control.*

Chapter 7.0 of the SAR has been updated to include a revised leakage rate testing procedure for the Pre-shipment Leakage Rate test and Maintenance Leakage Rate Test. Specifically, the procedures have been revised to provide a pre-shipment leakage rate test procedure that is consistent with the same procedures in the TRUPACT-II and HalfPACT SARs. The loading procedures have been revised to specify that the pre-shipment leakage rate testing of the IV is optional.

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

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Chapter 8 of the SAR describes the acceptance test and maintenance program. Section 8.1 of this chapter has been revised to include: Lifting Device Load Testing - revised to limit load testing for a center-pivot trunnions to the total gross weight of the package instead of twice the gross weight; Fabrication Leakage Rate Tests - revised to clarify correct package fabrication leakage rate testing; Polyurethane Foam - revised to be consistent with chemical composition and test procedure requirements previously approved in the TRUPACT-II and HalfPACT SARs; and Tests for Shielding Integrity - revised for consistent working for quality assurance disposition of nonconforming issues.

Section 8.2 of this chapter has been revised to include: Maintenance/Periodic Leakage Rate Tests - revised to clarify and correct package maintenance/periodic leakage rate testing; Fasteners - revised to clarify the number of "round trips" between fastener replacements; and Sealing Area Routine Inspection - revised to clarity the seal area inspection process.

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 where made to the Certificate of Compliance:

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

Condition No. 5(b)(1) was revised to remove the reference to Appendix 1.3.7 of the June 2002 application and reference the new document, RH-TRAMPAC submitted with Revision 4 of the SAR.

Condition No. 5(b)(2) was revised to remove the reference to Appendix 1.3.7 of the June 2002 application and reference the new document, RH-TRAMPAC submitted with Revision 4 of the SAR.

Condition No. 6 was revised to remove the reference to Appendix 1.3.7 of the June 2002 application and reference the new document, RH-TRAMPAC submitted with Revision 4 of the SAR.

Condition No. 7 was revised to remove the reference to Appendix 1.3.7 of the June 2002 application and reference the new document, RH-TRAMPAC submitted with Revision 4 of the SAR.

Condition No. 9 was revised to include a reference to the new document, RH-TRAMPAC submitted with Revision 4 of the SAR.

Condition No. 12 of the certificate was added and allows a package to be marked with the previous package identification number, USA/9212/B(M)F-85, until July 31, 2007. This is to allow time to replace the packaging nameplate that shows the revised package identification number, USA/9212/B(M)F-96.

Condition No. 13 of the certificate was added and authorizes use of the previous revision of the certificate for a period of approximately one year.

Condition No. 14 of the certificate was added that identifies the limitation that the package is not approved for air transport.

The October 14, 2005, and June 5, 2006, submittals were included in the References section.

CONCLUSION

Based on the statements and representation in the application, the staff concludes that the Model No. RH-TRU 72-B meets the requirements of the revised 10 CFR Part 71 and can be designated with "-96" in the identification number and that the revisions to the SAR are acceptable.

Issued with Certificate of Compliance No. 9212, Revision No. 4
on July 28, 2006.

REGISTERED USERS

U. S. Department of Energy
Attn: James M. Shuler
EM-24/CLV-1081
1000 Independence Ave., S.W.
Washington, DC 20585-2040

Westinghouse Government Environmental
Attn: Mr. Tom Lewis
Services Company, LLC
P.O. Box 2138
Carlsbad, NM 88221-2138