

**CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES**

1.	a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
	9365	3	71-9365	USA/9365/B(U)-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*)

Robatel Technologies, LLC
5115 Bernard Drive
Suite 304
Roanoke, VA 24018

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Robatel Technologies, LLC, application, Revision
No. 10, dated June 20, 2023.

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.: RT-100

(2) Description

A cylindrical stainless steel, lead shielded, packaging, with a 35 mm thick outer and 30 mm thick inner stainless-steel shell, designed for the transport of radioactive waste materials. The internal cavity of the packaging is 1,956 mm high, with a diameter of 1,730 mm. The annular space between the inner and outer shells is filled with a 90 mm thick lead for shielding. The cylindrical shell is attached to a circular forged bottom plate, with a full penetration weld, while the inner shell is attached to a circular forged flange, with a full penetration weld, at the top of the packaging. The base of the packaging consists of a 30 mm thick stainless steel outer bottom plate, a 75 mm thick gamma shield of poured lead, and a 50 mm thick stainless steel inner bottom forging.

The primary lid, consisting of a 210 mm thick stainless steel forging, is fastened to the packaging body with thirty-two M48 hex head bolts. The secondary lid, made of a 100 mm thick stainless steel upper plate, a 60 mm thick lead gamma shield and a 10 mm thick lower stainless steel plate, is attached to the primary lid with eighteen M36 hex head bolts.

Two tie-down arms, welded to the external shell, are considered as a structural part of the packaging. When not in use for package tie-down, shear pins prevent the tie-down arms from being used to lift the package. Removable lifting lugs are utilized for removal and handling of the primary and secondary lids, as well as of the impact limiters.

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5(a)(2) Packaging Description (Continued)

The stainless steel impact limiters have an outside diameter of 2,587 mm: the lower impact limiter extends 494 mm beyond the base of the packaging; the upper impact limiter extends 498 mm beyond the primary lid. The volume inside the impact limiter shells is filled with crushable shock-absorbing and thermal-insulating polyurethane foam.

The maximum gross weight of the package, including impact limiters, is 41,500 kg. The maximum (empty) weight of the packaging, including impact limiters, is 34,696 kg.

The containment boundary consists of the inner shell, the bottom forging, the top flange, the primary lid, the primary lid inner O-ring, the stainless steel vent port cover plate and its inner O-ring, the secondary lid and the secondary lid inner O-ring. A vent port penetrates the primary lid into the main cask cavity. The vent penetration contains a quick disconnect valve and is sealed with the vent port cover plate. The primary lid, secondary lid, and the cover plate are sealed with O-rings.

(3) Drawings

The packaging is constructed and assembled in accordance with Robatel Technologies, LLC, Drawing Nos:

RT100 PE 1001-1 Rev. H - RT-100 General Assembly Sheet 1/2

RT100 PE 1001-2 Rev. H - RT-100 General Assembly Sheet 2/2

RT100 PRS 1011 Rev. E - RT-100 Cask Sub Assembly Weld Map Cask Body

RT100 PRS 1013 Rev. C - RT-100 Cask Sub Assembly Weld Map Secondary Lid

RT100 PRS 1031 Rev. D - RT-100 Cask Sub Assembly Weld Map Lower Impact Limiter

RT100 PRS 1032 Rev. D - RT-100 Cask Sub Assembly Weld Map Upper Impact Limiter

102885 MD 1031-06 Rev. F - RT-100 Sub Assembly Fabrication Drawing Impact Limiter Foam

(b) Contents

- (1) Type and form of material: low density ($2 \text{ g/cm}^3 \leq \rho < 7.5 \text{ g/cm}^3$) activated hardware, high density ($\rho \geq 7.5 \text{ g/cm}^3$) activated hardware, dispersible solids, in the form of both dewatered and grossly dewatered resins, and filters contained within secondary containers.

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5(b) Packaging Contents (Continued)

(2) Maximum quantity of material per package

(i) Activity not to exceed 3,000 times a Type A quantity, along with the following limits:

- (1) As prescribed by the procedure in Section No. 7.6 of the application, for beta and gamma emitting radionuclides.
- (2) As prescribed by 10 CFR 71.15, for exempting materials from classification as fissile material.
- (3) A maximum total package neutron source of $3.5 \cdot 10^{-6}$ Ci/g for materials that produce neutrons (other than fissile materials) through any means, including spontaneous fission, alpha-neutron reactions, and gamma-neutron reactions.

(ii) Maximum decay heat: 200 watts.

(iii) Maximum weight of contents: 6,804 kg including shoring and secondary containers.

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

- (a) The package must be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application.
- (b) The packaging must be tested and maintained in accordance with the acceptance tests and maintenance program described in Chapter 8 of the application.

7. Except for close fitting contents, shoring must be placed between the secondary container and the package cavity's walls to prevent both radial and axial movements during transport.

8. Flammable gas (e.g., hydrogen) concentration is limited to less than 5% by volume.

9. A pre-shipment leakage rate test is required for all shipments.

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

11. Revision No. 2 of this certificate may be used until July 31, 2024.

12. Expiration date: July 31, 2029.

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REFERENCES

Robatel Technologies, LLC, application, Revision No. 10, June 20, 2023.

Supplements dated:

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Yaira K. Diaz-Sanabria

Signed by Diaz-Sanabria, Yaira
on 07/28/23

Yaira K. Diaz-Sanabria, Chief
Storage and Transportation Licensing Branch
Division of Fuel Management
Office of Nuclear Material Safety
and Safeguards

Date: July 28, 2023





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

SAFETY EVALUATION REPORT

Docket No. 71-9365
Model No. RT-100
Certificate of Compliance No. 9365
Revision No. 3

SUMMARY

By letter dated August 29, 2022, as supplemented November 22, 2022, December 14, 2022, May 19, 2023, and June 20, 2023 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML22262A264, ML22335A081, ML23005A121, ML23143A177 and ML23181A127 respectively), Robatel Technologies, LLC submitted an amendment request to revise the certificate of compliance (CoC) for the Model No. RT-100 package. The applicant proposed to add activated hardware as new contents and provide flexibility to ship filters of varying activities. In addition, on June 20, 2023, Robatel requested renewal of CoC No. 9365 for the Model No. RT-100 package. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the application using the guidance in NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material." Based on the statements and representations in the application, as supplemented, the staff agrees that these changes do not affect the ability of the package to meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71.

1.0 GENERAL INFORMATION

1.1 Packaging Description

The applicant did not propose any changes to the packaging or its components.

1.2 Drawings

The applicant revised the safety classification of a quick disconnect valve from safety class "A," i.e., critical to safety, to safety class lower "B," i.e., important to safety. The applicant explained that both the quick-disconnect cover plate and the inner seal are considered part of the containment boundary; therefore, the applicant assigned both components to safety class "A." The applicant asserted that the quick disconnect valve is not part of the containment boundary and does not need to be safety class "A" because a radioactive release can only occur if both the quick disconnect fails along with either quick-disconnect cover plate or the inner seal. The staff reviewed these changes for conformance to NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," and found them acceptable.

1.3 Content Description

The applicant proposed to add activated hardware as authorized contents. The applicant identified two categories of activated hardware: low density and high density. The applicant stated that activated hardware with densities greater than or equal 2 grams per cubic centimeter (g/cm^3) but less than 7.5 g/cm^3 were considered low density activated hardware, e.g., aluminum and zircaloy, while activated hardware with densities greater than or equal to 7.5 g/cm^3 and less than or equal to 9.0 g/cm^3 were considered high density activated hardware, e.g., steel and

Enclosure 1

Inconel. After reviewing these changes, the staff finds that the applicant adequately characterized the contents.

1.4 Findings

Based on a review of the statements and representations in the application, the staff concludes that the package has been adequately described to meet the requirements of 10 CFR Part 71.

2.0 STRUCTURAL

The objective of the structural evaluation is to verify that the applicant has adequately evaluated the structural performance of the proposed transport package and demonstrated that it satisfies the regulations in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." The staff reviewed and evaluated the proposed changes primarily in section 2.0 of the safety analysis report (SAR), revisions 8, 9 and 10, as provided by the applicant. This section of the safety evaluation documents the staff's review, evaluation, and conclusions with respect to the structural safety aspects of the proposed transport package.

2.1 Structural Design Description

The applicant proposed the following design changes that are evaluated in the structural review: the inclusion of activated hardware as package contents and the revision of the closure bolt evaluation in SAR section 2.13 due to a self-identified error in the load combination method employed. The applicant also cited a new reference, CN-21004-21, revision 1, "RT-100 Cask Bolting Load Combinations Verification," in SAR section 2.16.

2.1.1 Design Criteria

The applicant stated that the design criteria guidance of Regulatory Guide (RG) 7.6, "*Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels*", is followed for the normal conditions of transport (NCT) and hypothetical accident conditions (HAC) evaluations of the package. The applicant also stated that the load combination guidance provided in RG 7.8, "*Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material*", was followed in the structural evaluations. SAR table 2.1.2-1 summarized the load combinations for the RT-100 cask body analysis. The applicant cited NUREG/CR-6007, "*Stress Analysis of Closure Bolts for Shipping Casks*", in SAR section 2.1.2.2 for bolt design and allowable stresses, confirming in SAR sections 2.1.4 and 2.13 that NUREG/CR-6007 guidance is followed for the bolt evaluations.

2.1.2 Weights and Centers of Gravity

SAR section 1.2.2.3.3 described the activated hardware contents. In SAR section 1.2.2.7, the applicant stated that the maximum weight of the activated hardware is 5,896 kg. The applicant made an allowance of 900 kg for a container and shoring materials. SAR table 2.1.3-1 identified the maximum total payload weight as 6,805 kg and the maximum total package weight, with payload, as 40,845 kg. Sheet 1 of drawing RT100 PE 1001-1, revision H, restricted the maximum gross package weight to 41,500 kg; the applicant employed this value for most structural calculations. Sheet 1 of drawing RT100 PE 1001-1, revision H, the center-of-gravity of the cask with payload is unchanged with the proposed contents revision, at 1,648 mm above the base of the lower impact limiter. Per section 2.1.3 of the SAR, a ± 28 mm deviation from this location is permissible without changing the evaluation results. The staff finds that none of the values of the weight or the center-of-gravity input parameters for the cask structural

evaluation have been revised, therefore the addition of the activated hardware contents does not affect the structural evaluations presented in the SAR.

2.2 Normal Conditions of Transport

Staff compared the NCT load combinations in SAR sections 2.6.1.1 and 2.6.2 employed by the applicant to the cited design criteria. Staff noted that the cold conditions were determined including decay heat and a structural design internal pressure of 35 pounds per square inch gage (psig) which exceeded the design basis maximum normal operating pressure. Use of these inputs did not align with the recommendations of table 1 of RG 7.8. The applicant explained on June 20, 2023, (ADAMS Accession No. ML23181A127) that the package had been reanalyzed per the cold case load combination guidance of RG 7.8 using -40 °C and demonstrated that the resulting safety margin did not bound the values already presented in SAR tables 2.6.7-1 and 2.6.7-2. The applicant added a note to SAR section 2.6.7.2.2 stating that this confirmatory analysis was performed and demonstrated that the cold case at -40 °C does not govern the cask design. The staff reviewed the applicant's confirmatory analysis and concluded that the NCT load combinations employed as documented in the SAR yield the most unfavorable structural analysis results for both the cask and the bolt components for the conditions evaluated.

The staff reviewed the NCT internal pressure values cited in the SAR and noted that the maximum normal operating pressure and design pressures were often cited in both units of "psig" and pounds per square inch absolute, "psia." This created uncertainty as to what numerical values were employed in the structural analyses. In response to the request for additional information (RAI) and subsequent clarifications, the applicant confirmed that the internal design pressure of 35 psig was employed in the NCT analyses which is conservative versus the maximum normal operating pressure value of 11.8 psig calculated in SAR section 3.3.2.5. In the response, the applicant referenced section A.4 of calculation RTL-001-CALC-ST-0402, revision 4, which documents the use of 35 psig for the internal containment pressure employed in the structural finite element analysis. SAR section 2.6.1.1 clearly stated that 35 psig is employed in the NCT structural analysis, and the applicant revised other SAR sections, notably section 3.3.2.5 and tables 2.13.3-1, 2.13.2-3, to employ consistent units. Based on the applicant's clarifications and SAR revisions, the staff finds that the appropriate design internal pressure for NCT has been employed in the analyses and yield the most unfavorable results for both the cask and the bolt components for the conditions evaluated.

2.3 Hypothetical Accident Conditions

The staff reviewed the HAC load combinations employed by the applicant and noted that, per SAR section 2.7.1, the thermal effects from the hot ambient conditions were not included in the various 30-foot drop event analyses. Omitting this input did not align with the recommendations of table 1 of RG 7.8. In response to RAIs and subsequent clarifications, the applicant stated that the thermal stresses were omitted from the analysis and stress intensity determinations based on the guidance of RG 7.6 which defines thermally-induced stresses as secondary stresses. Under accident conditions, the guidance in Regulatory Position 6 of RG 7.6 omits secondary stresses in the determination of stress intensity. The applicant presented additional structural analysis results in their June 20, 2023, RAI clarification response letter (ADAMS Accession No. ML23181A127) indicating that the fabrication-induced thermal stresses from lead shielding installation bound those of the HAC fire-induced thermal stresses. The applicant also added an explanation for the thermal stress omission in SAR section 2.7.4.2. Therefore, staff finds that, based on the RG 7.6 guidance and the analytical results presented by the applicant, the omission of thermal stress in the HAC 30-foot drop event structural evaluation does not affect the results of the analysis presented in the SAR.

Staff reviewed the HAC internal pressure values cited in the SAR and noted that this pressure was often cited in both units of “psig” and “psia.” This created uncertainty as to what numerical values were employed in the structural analyses. In response to RAIs and subsequent clarifications, the applicant confirmed that the internal design pressure of 85.3 psig was employed in the HAC analyses. In their responses, the applicant referenced section 7.2.2 of calculation RTL-001-CALC-ST-0402, revision 4, which documents the use of 85.3 psig for the HAC internal containment pressure employed in the structural finite element analysis. The applicant also revised SAR section 2.7.1 to clearly state that 85.3 psig is employed in the HAC structural analysis. Based on the applicant’s clarifications and SAR revisions, staff finds that the appropriate design internal pressure for HAC has been employed in the analyses and yields the most unfavorable results for the conditions evaluated, for both cask and bolt components.

2.4 Closure Bolt Evaluation

In SAR section 2.13.1 the applicant stated that the closure lid bolt loadings are determined in accordance with NUREG/CR-6007. However, SAR section 2.1.4 stated that the load combinations identified in RG 7.8 are employed for the bolt evaluations. Staff reviewed these guidance documents and compared them to the load determinations and combinations documented in the SAR and found several deviations as described below.

SAR sections 2.13.2.2.2 and 2.13.2.2.4 presented HAC thermally-induced closure bolt load determination that were based on results from SAR table 3.1.3-2. The staff asked the applicant to verify that the temperatures in SAR table 3.1.3-3 are larger than those in table 3.1.3-2 and would subsequently produce higher bolt loads. The applicant’s RAI responses confirmed that the temperature values in SAR table 3.1.3-3 would produce the greater bolt loads, and the applicant subsequently revised the bolt load determinations in the SAR sections 2.13.2.2.2 and 2.13.2.2.4.

Because the SAR did not appear to address vibration-induced bolt loads, including prying effects, the staff requested supplemental information and issued an RAI. In response, the applicant included a calculation of vibration-induced loads in SAR section 2.13.2.8 as well as adding the results to SAR tables 2.13.3-1 and 2.13.3-2. The applicant stated that the magnitude of the vibration-induced bolt loads was negligible and did not include their load effects in the total bolt load determination.

The staff reviewed SAR tables 2.13.3-1 and 2.13.3-2 and found that not only were the vibration-induced loads not included, but also load effects were incorrectly labeled. Additionally, some values appeared where none would be expected, some values disagreed with the referenced calculation, and the source of some values was not noted. In response to RAIs, the applicant revised or added the following to the SAR: tables 2.13.3-1 and 2.13.3-2, sections 2.13.2.3.1 and 2.13.2.3.2.

After staff’s initial review, the SAR did not appear to include a low-cycle fatigue stress evaluation. Therefore, the staff requested that a low-cycle fatigue stress evaluation be performed for the lid closure bolts. In response to RAIs, the applicant included a bolt fatigue stress evaluation in SAR section 2.13.4 and added an operational control in SAR section 8.2.3.2 to limit the use of all lid bolts to 500 occurrences of preload.

The applicant stated in SAR section 2.13.2.1.1 submitted with the initial application that any shear load effects on the primary lid bolts are prevented due to existing gaps between the lid and the cask wall, as well as between the lid and the bolt. These existing gaps included the fabrication tolerances shown on the design drawings. The staff suggested that the applicant

perform a thermal analysis to confirm that thermal expansion or contraction would not result in shear load being transferred to a bolt. In responses and clarifications to a supplemental information request and an RAI, the applicant explained that alignment pins are employed during installation of the primary and secondary lids to facilitate lid placement and alignment. The applicant also performed the suggested lid thermal analysis based on the most critical as-built cask measurements (ADAMS Accession No. ML22356A050) and temperature values presented in the SAR. The applicant provided analyses on June 20, 2023 (ADAMS Accession No. ML23181A127), that confirmed adequate gaps exist between the lid and the cask wall as well as the lid and the bolt under thermal expansion and contraction conditions. The performance of this confirmatory analysis is noted in SAR sections 2.13.2.1.1 and 13.2.1.3.

Based on the information in the application, the staff finds that the applicant has evaluated the most critical loading conditions for the lid closure bolts and they maintain an adequate safety margin during NCT and HAC events.

2.5 Evaluation Findings

The staff reviewed the drawings and amendment package for the proposed addition of activated hardware content and concludes that it satisfies the requirements of 10 CFR Parts 71.31(a)(1). The staff reviewed the structural performance of the package under the NCT required by 10 CFR 71.71 and the HAC required by 10 CFR Part 71.73 and concludes that it satisfies the requirements of 10 CFR 71.51(a)(1) and (2) for a Type B package. Based on a review of the statements and representations in the amendment request, the NRC staff finds that the RT-100 package has been adequately described and evaluated to demonstrate that it satisfies the structural integrity requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The applicant updated the thermal section by removing the following statement from SAR section 3.1.2: *“and is conservative for the contaminated resin and filter contents that are transported in the cask.”* The applicant also updated SAR section 3.2.3 to identify low density and high density activated hardware as authorized contents and updated the Quality Assurance Plan reference in SAR section 3.6. In addition, the applicant revised SAR section 3.3.2.5, which expressed the maximum normal operating pressure in both pounds per square inch absolute and pounds per square inch gage, by removing the maximum normal operating pressure psig expression. The staff determined that these changes are editorial in nature because they changed neither the maximum allowable decay heat for the cask nor the thermal analyses which evaluated the thermal performance of the cask; therefore, the staff finds them acceptable.

4.0 CONTAINMENT EVALUATION

4.1 Design Evaluation Changes

The objective of the review was to verify that the Model No. RT-100 package containment design is adequately described and evaluated under NCT and HAC as required per 10 CFR Part 71. There were no changes to the containment boundary or its testing as part of this CoC revision. Rather, the applicant's revised safety analysis report reflected the inclusion of activated metal hardware as new content (described in SAR section 1.2.2.3.3) which, according to SAR section 4.4, does not contribute to hydrogen or flammable gas generation. As noted below, SAR changes also included edits to the chapter 4 and chapter 7 hydrogen gas generation equations that accounted for activated hardware volume. Regulations applicable to the containment review included 10 CFR 71.31, 71.33, 71.35, 71.43, and 71.51.

4.2 Staff Evaluation

The applicant included slight edits to SAR chapter 4 “Containment Evaluation” and chapter 7 “Package Operations” that discussed analyses to limit the generation of flammable hydrogen gas to less than 5 percent mole fraction; NUREG/CR-6673 “Hydrogen Generation in TRU Waste Transportation Packages” formed the basis for these analyses. Since activated metal does not directly contribute to flammable gas generation, the changes addressed the impact of the activated metal hardware volume. As an example, SAR section 4.4.3 and section 4.4.5 subtracted the activated hardware content volume as a term in the free volume equations and hydrogen gas generation analytical equations, respectively. In addition, SAR table 4.4.4-1 indicated that activated hardware is grouped with resins and filters when the simplified loading model is used. Staff finds that these edits satisfactorily addressed the presence of the new content (i.e., activated hardware) in the hydrogen generation calculations.

4.3 Evaluation Findings

Based on a review of the containment-related sections of the application, the staff concludes that the containment design has been adequately described and evaluated and has reasonable assurance that the package meets the containment requirements of 10 CFR Part 71.

5.0 SHIELDING EVALUATION

The purpose of this evaluation is to verify the shielding design for the Model No. RT-100 Type B(U) Cask (RT-100) provides adequate protection for direct radiation from its packaged contents to meet the external dose rate limits that are specified in 10 CFR Part 71 under NCT and HAC. The RT-100 is designed for exclusive use shipments.

5.1 Shielding Design Description

The applicant designed the RT-100 package to transport radioactive materials including contaminated resin and filter media generated from nuclear power plant operation as well as both low and high-density hardware activated hardware. The RT-100 utilized a robust gamma shielding design comprised of a steel/lead/steel body with a steel primary lid and a steel/lead/steel secondary lid. Bolts secured the primary lid onto the body. Bolts also secured the secondary lid to the primary lid.

5.1.1 Design Features

The applicant made no modifications to the cask shielding design. The applicant only proposed to add activated hardware as well as mass limited filters and resins containing byproduct radioactive materials as authorized contents. The applicant stated that the contents do not contain fissile materials in quantities exceeding the fissile material exemption defined in 10 CFR 71.15. SAR chapter 7 specified the procedure for determining the maximum allowable content in the package, and the SAR shielding evaluation section specified the activity acceptance criteria of beta, gamma, and neutron emitting radionuclides for each specific nuclide.

5.1.2 Maximum External Radiation Levels Summary Tables

The applicant’s proposed content changes had no impact on either the NCT or HAC dose rates reported in SAR table 5.1.2-1. The applicant did revise the text in SAR chapter 5.1.2 and the title for SAR table 5.1.2-1. These changes clarified that the dose rates in SAR table 5.1.2-1 were not associated with the dose rates for either mass limited filters and resins or activated hardware. The applicant provided the dose rates for mass limited filters and resins in SAR

section 5.5 while SAR section 5.6 provided the dose rates for activated hardware. The staff determined that the change to SAR section 5.1.2, are editorial; therefore, the staff finds them acceptable.

5.2 Radiation Source

The proposed content change did not alter the previous staff evaluations of the RT-100 radiation source. Therefore, staff did not perform a new evaluation.

5.3 Shielding Model

The applicant used the Monte Carlo N-Particle® (MCNP), Version 6 (MCNP6) computer code with ENDF/B-VI Release 8 Photo-atomic Data gamma cross section library and MCPLIB84 for the shielding analyses. The applicant modeled the package under NCT and HAC conditions as specified in 10 CFR 71.71 and 71.73 respectively. For each model, the applicant based the geometry on the drawings provided in SAR Appendix 1.4. SAR sections 5.3.1.2 and 5.3.1.3 described the RT-100 cask MCNP NCT and HAC shielding models, respectively, and SAR figures 5.3.1-1 and figure 5.3.1-3 displayed the RT-100 cask MCNP NCT and HAC shielding models, respectively.

For the NCT and HAC models, the applicant used the minimum values, both in dimensions and material densities, to identify the bounding package dose rates. SAR table 5.3-1 gave the nominal and minimum shield thicknesses. The shielding evaluations neglected any shielding provided by the high integrity container used to store and transfer resin into the RT-100 cavity. The applicant modeled the effects of resin and filter density changes, as well as redistribution of the content media due to NCT and HAC, by decreasing the volume occupied by the source term. After evaluating the applicant's assumptions, the staff finds them acceptable since using minimum tolerances and densities will produce conservative results.

5.3.1 Configuration of Source and Shielding

The RT-100 source configuration consisted of resins and filters within a secondary container placed in the package cavity. Shoring positioned the secondary container within the cavity. The radioactive source term volume within the RT-100, as modeled in the analyses, took no credit for the reduction in available volume associated with a secondary container or any shoring.

5.3.1.1 Source Term Configuration

The NCT and HAC shielding models uniformly distributed the photon source throughout the geometry cell representing the resin and filter media. This approach based the source strength density limit on the assumption that the maximum specific activity is evenly distributed throughout the entire cask cavity. In the actual cask operations, the contents will not be homogeneously distributed. However, because the contents of a secondary container liner are characterized by the shipper prior to cask loading, the staff finds this approach acceptable.

The applicant also developed a shielding model for the activated hardware in a similar manner to the filter and resin model. The applicant limited the activated hardware contents by mass while still assuming the maximum activity in the analysis. The applicant proposed mass limits of 1,000 lbs., 2,000 lbs., 8,000 lbs., and 13,000 lbs. Although the 13,000 lbs. limit is less than the maximum package capacity of 15,000 lbs., the applicant used it due to the need for a secondary container and potential shoring.

The primary nuclide responsible for radioactivity in the hardware proved to be Co-60, but the applicant explicitly analyzed the seven other nuclides that are specified in SAR sections 5.4 and 5.5. They considered radioactivity from all other nuclides to be negligible. For their analysis, the applicant modeled the hardware in two groups: high density (density between 7.5-9.0 g/cm³) and low density (density between 2.0 and less than 7.5 g/cm³). In determining the dose rate, the applicant modeled the contents using bounding cask loading conditions to determine the largest dose rate per Curie, i.e., mrem/hr/Ci, for the NCT and HAC models. When calculating the specific activity limits, the applicant only used the upper bound mass for the content group being evaluated. SAR table 5.6.7.1 showed the specific activity limit for all eight nuclides analyzed in both the high density and low density groups at each of the proposed mass limits.

The applicant accounted for uncertainties by requiring users to round up to the energy or the particles to the next energy line in the loading table. For example, if the particle energy is 1.61 MeV, the applicant required the package user to employ a particle energy of 1.7 MeV when determining the maximum allowable contents. This approach provided some safety margin for most radionuclides except for Co-60 because the sources are explicitly modeled in dose rate response calculations.

The model had the following key fundamental assumption: there is a fixed one-to-one relationship between dose rate and particle type, particle energy, and location regardless of the media through which the particle travels. For a package with a material composition similar to the model, this assumption would provide acceptable results.

The applicant estimated external package radiation levels using the methodology described above. The source term energy and attenuation from both the package contents and the packaging material impacted the estimates. The applicant calculated the relationship between the specified contents' variables and their effect on the radiation levels. The applicant also performed parametric studies to determine the dose rate response for different media composition and densities.

The applicant performed "forward dose rate" calculations of the package to confirm the evaluated contents are valid and the package satisfies the dose rate response calculation assumptions. SAR table 5.1.2-1 provided a dose rate summary of the maximum allowable quantities of these nuclides. The results of the applicant's shielding analysis showed that the package design meets the regulatory requirements with the maximum content. SAR table 5.4.4-1 showed more details of the nuclides used in the calculations and the corresponding shielding evaluation results.

The applicant modeled the resin and filter materials in the analysis as carbon with a density of 0.65 g/cm³. The applicant also modeled the material as polystyrene, nylon, and zeolite, and evaluated all four materials densities in the range of 0.65 g/cm³ to 1.0 g/cm³. The parametric study results found that increasing the media density decreased the allowable source strength density of the radionuclides. The study also found that carbon results in the most limiting case for source strength density; therefore, the applicant chose to calculate dose rate response with the filter media modeled as carbon at a density of 1.0 g/cm³.

5.3.1.2 NCT Model

For the NCT model, the applicant modeled the resin material in the RT-100 cavity as a void for the generic line energy dose rate response calculations. This assumption neglected all photon attenuation in the resin and filter media. Dose rate response calculations for the eight individual nuclides modeled the cavity with carbon at 1 g/cm³. This assumption took some credit for

photon attenuation in the resin material. Staff finds this approach acceptable because the calculation "Updated Resin/Filter Shielding Evaluation of the RT-100 Transport Cask" demonstrated that carbon at 1 g/cm³ produced the most restrictive Ci/g limits for all radionuclides.

5.3.1.3 HAC Model

The HAC assumed that the impact limiters were lost. The model also assumed that the nine meter drop and one meter puncture tests damaged the lead shield as follows. The pin puncture test created a 1 inch by 6 inch diameter indentation in the lead shield, and the nine meter drop test created a 5 millimeter annular void, which is based on calculations the applicant performed, due to lead slump at the top of the lead column. For the generic energies dose rate response calculations, the applicant did not take credit for the content. For the eight individual nuclides, the applicant modeled the resin that filled the cavity as carbon at a density of 1.0 g/cm³. Additionally, the applicant made two HAC models, one used for one meter dose rate calculations at the bottom of the cask, and the second for the one meter dose rate calculations at the top and the side of the cask. For each model, the applicant placed the content in the most restrictive location such that the calculated dose rates were bounding and conservative. Both HAC models incorporated the lead shield damage caused by the pin puncture test. SAR Figures 5.3.1-1 through 5.3.1-6 showed multiple two dimensional and one three dimensional visualizations of the NCT and HAC models. After evaluating the RT-100 transportation package HAC Model, the staff determined that the model adequately portrayed the potential deformation to the cask and lead shield, e.g., lead slump and lead shield indentation.

5.3.2 Material Properties

Contents transported in the RT-100 included resins and filter media. The applicant considered the following four materials, which are typical of resins and filter media, as the package contents:

- Polystyrene based resins such as Duralite
- Activated Charcoal
- Nylon filter media
- Zeolite - hydrated aluminosilicates such as Faujasite

The composition of typical activated hardware components and activated metals included steels, Inconels, and zirconium alloys and possibly aluminum alloys. Because material density significantly affects the resulting permissible specific activity limits, the applicant divided the materials in this analysis into 'high-density hardware' and 'low-density hardware' groups.

The staff evaluated the material properties and finds them acceptable. The staff also confirmed that the applicant described and used appropriate material properties in the shielding models for all packaging components, package contents, and the conveyance.

5.4 Shielding Evaluation

SAR Sections 5.5 and 5.6 described the shielding evaluation for mass restricted filters and activated hardware which is an extension of the dose rate calculations outlined in SAR section 5.4. The applicant retained the packaging geometry, materials, and all assumptions for the effects of NCT and HAC, but the applicant used an alternative approach to modeling the contents. The supplemental evaluation calculated dose rates for content volumes equivalent to 500, 1,000, and 1,500 lbs. of radioactive filters. The applicant implemented a mass restriction

on the total quantity of radioactive contents that allowed increased radionuclide specific activity limits.

5.4.1 Methods

The applicant used the MCNP6 computer code to perform the RT-100 package shielding calculations. The staff finds the use of this code acceptable because MCNP6 is a Monte Carlo transport code that offers a full three-dimensional combinatorial geometry modeling capability. This means that MCNP6 required no gross approximations to represent the RT-100 package in the shielding analysis. In addition, the applicant used bounding shielding material thicknesses in the MCNP6 models which is conservative.

5.4.2 Dose Rate Calculations

SAR Section 5.4.1.2 described the methodology to calculate the package dose rates. For the generic energy line outputs, the applicant binned each dose rate response tally by the emission energies and subsequently reported a dose rate response for each generic energy line. For the eight radionuclides that are calculated individually, the applicant included all nuclide specific energy lines above a threshold energy in the MCNP6 model source term.

The MCNP6 model calculated the dose rate response for a particle at the specified energy, e.g., a 1.0 MeV gamma. Dividing the allowable dose rates by the calculated dose rate response determined the maximum number of particles at the specified energy. Dividing the maximum number of particles by the number of particles released by 1 Ci of the radionuclide calculated the maximum allowable content in terms of activity for each radionuclide. For contents that emit multiple particles at different energy levels with each decay, like Co-60, the allowable activity of each particle must be determined based on the energy distribution and branch fraction. The applicant presented the equations used to determine the loading table and dose rates of bounding conditions in SAR sections 5.4.4.4 and 5.4.4.5.

The applicant explicitly calculated a one-to-one dose rate per Curie, i.e., mrem/hr/Ci, using their MCNP model and provided them in SAR tables 5.5.6-1 through 5.5.6-8. SAR tables 5.5.6-1 through 5.5.6-8 also identified the overall activity limit for each radionuclide. Dividing the overall activity limits for the radionuclide by the respective filter content mass in grams generated specific activity limits for the mass restricted filters. SAR table 5.5.7-1 provided the specific activity limits for the eight radionuclides evaluated associated with mass restricted filters. The applicant used the same methodology to calculate specific activity limits for activated hardware. SAR tables 5.6.6-5 thru 5.6.6-12 identified the maximum one-to-one dose rate per Curie values, as well as the overall activity limit for each radionuclide, for activated hardware. SAR table 5.6.7-1 identified the overall activity limit for each radionuclide for activated hardware. In determining the maximum quantity of each radionuclide, the applicant specified that the dose rate response of the next higher energy of the same particle must also be used except for the two gammas emitted by Co-60 in its decay.

For Co-60, the applicant did not require using the dose rate response for the next higher energy because the two gammas emitted by Co-60 were explicitly modeled in the analyses. The RT-100 package had a total content mass limit of 15,000 lbs. Use of a secondary container, which is always required, and the possible need for shoring, prevented users from reaching this radioactive content mass limit. The applicant also developed instructions to assist the package user to use the loading tables in determining the maximum content quantity allowed. The staff reviewed the instructions in SAR section 7.1.1 of the application and found them acceptable. In the instructions, among others, the applicant directed that the RT-100 be surveyed for surface

contamination to ensure it is within allowable limits, and if the package exceeds the contamination limits, to decontaminate the RT-100 prior to performing the next step.

The applicant reduced the regulatory dose rate limit specified in 10 CFR 71 for a given location by 5 percent. To be more specific, the applicant reduced the regulatory NCT limit for the package surface from 200 mrem/hr to 190 mrem/hr, the 2-meter limit from the transport vehicle from 10 mrem/hr to 9.5 mrem/hr, and the cab limit from 2 mrem/hr to 1.9 mrem/hr. Similarly, the applicant reduced the HAC limit at 1 meter from 1000 mrem/hr to 950 mrem/hr. This reduction in the regulatory limits accounted for uncertainties in modeling the actual packaging and characterization of the contents.

The applicant also used the specific activity limits to demonstrate compliance with the regulatory dose rate limits by calculating the sum of the fractions based on the maximum specific activities of all filter contents. The applicant presented the following conservatisms associated with the calculations in SAR section 5.5.7: a 5 percent margin applied directly to the regulatory limits, no consideration of both the additional spacing and the shielding provided by the secondary container in the analysis, and modeling the filter contents solely as activated carbon to minimize photon attenuation of four materials as discussed in SAR section 5.4.4.2.

The staff finds the dose rates calculations acceptable since the shielding evaluation methods used are appropriate for evaluating the package radiation levels. The applicant effectively represented the shielding evaluation methods. The applicant also effectively evaluated the material properties, geometries and configurations of the packaging components, package contents, and the radiation source-term properties. The staff confirmed that the methods effectively represented the NCT evaluations and the HAC tests as well as evaluated the effects of the NCT evaluations and the HAC tests on the package.

5.4.3 Code input and output data

The staff reviewed all relevant inputs and outputs for the gamma shielding analysis provided with proprietary calculation package CN-13039-502. The applicant performed post processing of the energy dependent responses into detailed dose rate responses (mrem/hr/Ci) for all radionuclides and provided these in SAR tables 5.7.2-1 and 5.7.2-2. Using these responses and the content activity loading, the applicant computed the total dose rate in mrem/hr for NCT and HAC conditions.

5.4.4 Flux-to-Dose Rate Conversion

The applicant used the ANSI/ANS 6.1.1-1977 - Gamma Flux-to-Dose Conversion Factors in the calculation of a photon flux (particles/s-cm²) at a particular tally or detector location given the source magnitude. MCNP6 converted these values into dose using the gamma flux-to-dose response functions in SAR table 5.4.3-1. The staff finds that use of the ANSI/ANS 6.1.1-1977 - Gamma Flux-to-Dose Conversion Factors is consistent with previous approvals by the NRC.

5.4.5 External Radiation Levels

The applicant determined the maximum external radiation levels by the quantity of each radionuclide in the contents to be shipped. The staff confirmed that the limiting quantity of each radionuclide was determined by the source strength density limit for each respective radionuclide. For the radionuclides considered, either the NCT 2 meter or the HAC side 1-meter regulatory requirements always limited the source strength density. The maximum dose rate that can be measured at any regulatory location can only be equal to the regulatory limit at the NCT 2 meter or the HAC side 1-meter locations.

SAR table 5.5.1-1 identified the maximum dose rates under NCT and HAC for mass restricted filters up to 1500 lbs. For filter contents exceeding 1,500 lbs., SAR Table 5.1.2-1 identified the maximum NCT and HAC dose rates. SAR table 5.6.1-1 identified the maximum dose rates under NCT and HAC for activated hardware.

The staff evaluated the input and output files and found them acceptable. All the radiation levels met the limits of 10 CFR 71.47(a) or 10 CFR 71.47(b), as appropriate, and 10 CFR 71.51(a)(2). The staff also verified that all radiation level point locations shown in the shielding analyses include all locations prescribed in 10 CFR 71.47(a) or 71.47(b) and in 71.51 (a)(2).

5.5 Staff Calculations

The applicant determined the maximum external radiation levels by the quantity of each radionuclide in the resin and filter media that is to be shipped. The staff confirmed that the limiting quantity for each radionuclide was determined by the respective source strength density limit of each radionuclide. For the radionuclides considered, either the NCT 2-meter or the HAC side 1-meter regulatory limits always limited the source strength density. The maximum dose rate that can be measured at any regulatory location can only be equal to the regulatory limit at the NCT 2-meter or the HAC side 1-meter locations.

SAR tables 5.7.2-1 and 5.7.2-2 listed the gamma radionuclide responses. SAR tables 5.4.4-5 and 5.4.4-6 identified the maximum dose rates under NCT and HAC for each individual radionuclide. SAR table 5.4.4-1 summarized the maximum calculated dose rate at each regulatory location, and the radionuclide responsible for each maximum dose rate.

The staff evaluated the input and output files and found them acceptable. All the radiation levels meet the limits of 10 CFR 71.47(a) or 10 CFR 71.47(b), as appropriate, and 10 CFR 71.51(a)(2). The staff also confirmed that all radiation level locations prescribed in 10 CFR 71.47(a) or 71.47(b) and in 71.51 (a)(2) are included in the shielding analyses.

The applicant stated that the RT-100 package is limited to a quantity of radioactive materials such that the package does not exceed 3,000 A₂. The staff considered this a problematic way of defining content limits due to different radionuclides having different A₂ values. Additionally, A₂ values did not consider the impact of different particles energies which are a key factor in shielding. However, staff finds it acceptable for this application because the package will not exceed the 3,000 A₂ based on the shielding and source terms calculations.

The applicant did not include any analysis on neutron shielding due to the contents being restricted to trace amounts of neutron emitters only. Neutron sources in the package are limited to 3.5×10^{-6} Ci/g source strength density based on Class C material burial limits.

5.6 Evaluation Findings

Based on the review of the information provided in the application and the independent confirmatory analyses that the staff performed, the staff has determined the proposed package design and contents satisfy the shielding requirements and radiation level limits specified in 10 CFR Part 71 with reasonable assurance. In addition, the staff also considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices in coming to this position.

F5-1 The staff has reviewed the application and have determined that it appropriately describes the package contents and the design features that affect compliance with shielding

regulations specified in 10 CFR 71.31(a)(1), 71.33(a), and 71.33(b). The application also shows that the shielding is compliant with 10 CFR 71.31(a)(2), 71.31(b), 71.35(a), and 71.41(a). Based on the description of the package in the application, the staff feels confident in the ability to adequately model the shielding performance. The evaluation of the shielding performance is also adequate as the applicant used the appropriate bounding tolerances and package contents as described in the application.

F5-2 The staff has reviewed the application and have determined that the package has been designed such that it can withstand the conditions described in 10 CFR 71.71 (NCT), as well as with the regulations specified in 10 CFR 71.43(f) and 10 CFR 71.51(a)(1), without resulting in a significant increase in external radiation levels.

F5-3 The staff has reviewed the application and have determined that under the evaluations specified in 10 CFR 71.71 (NCT), the package external radiation levels do not exceed the limits prescribed in 10 CFR 71.47(b) for exclusive-use shipments. The package however fails to comply with the limits prescribed in 10 CFR 71.47(a) for nonexclusive-use shipments and as such is only suitable for exclusive-use shipments as described in the application.

F5-4 The staff has reviewed the application and have determined that under the test procedure specified in 10 CFR 71.73 (HAC), the package external radiation levels do not exceed the limits prescribed in 10 CFR 71.51(a).

F5-5 The staff has reviewed the application and determined that the applicant has identified codes that are well benchmarked and appropriately applied the codes in the shielding analyses and design in compliance with the regulations specified in 10 CFR 71.31(c). Examples of these codes and standards include the usage of MCNP in the shielding model and flux-to-dose conversion factors in the calculations.

F5-6 The staff has reviewed the application and determined that it includes operating descriptions, tests, and maintenance programs that ensure the package is fabricated, operated, and maintained such that it remains compliant with the shielding requirements specified in 10 CFR Part 71.

6.0 CRITICALITY EVALUATION

Contents authorized for transport in the RT-100s contained only trace quantities of fissile radionuclides. Consequently, the contents met the fissile exemption requirements of 10 CFR 71.15. Therefore, staff did not perform a criticality review.

7.0 MATERIALS EVALUATION

7.1 Introduction

The applicant's main purpose of this revision is to amend the Model No. RT-100 type B(U) CoC to allow for the transport of (1) activated hardware or activated metal contents packaged in a secondary container and (2) contaminated spent resins and filter media with activated hardware of varying activities by limiting the waste's mass.

The type and form of material to be transported will be contained within a secondary container. The chemical forms of the contents are resins and filter media containing radioactive materials and metallic activated hardware segments in the form of dispersible solids. The applicant states that no contents are in powdered form.

The staff reviewed and evaluated changes in revision 8 of the SAR as provided by the applicant. This section contains the materials review, evaluation, and conclusions for the RT-100 transportation package.

7.2 Material Properties and Specifications

The applicant added activated hardware or activated metal (terms used interchangeably) packaged in a secondary container to the package contents in addition to the contaminated spent resins and filters currently specified in the CoC. The applicant described this activated hardware content as low-density hardware, such as aluminum and zircoloy, as well as high-density hardware, such as steel and Inconel. The SAR limited low-density hardware to a density greater than or equal to 2 g/cm³ and less than 7.5 g/cm³. The SAR limited high density hardware to a density greater than or equal to 7.5 g/cm³ and less than or equal to 9.0 g/cm³. Common examples of activated hardware that could be shipped in the RT-100 cask included, but not limited to, fuel channels, velocity limiters, and reactor vessel internals from Pressurized and Boiling Water Reactors.

The applicant used secondary containers to package contaminated spent resins and filter media, activated low-density hardware, activated high-density hardware, or a mixture of spent resins and filter media with activated hardware generated by nuclear power plants. The applicant stated that contents packaged in the secondary container are manufactured using corrosion resistant, non-reactive materials. After reviewing the information provided by the applicant, the staff concludes the addition of activated hardware will not cause any significant chemical or galvanic reaction. Therefore, the staff finds that the RT-100 cask meets the requirements of 10 CFR 71.43(d).

7.3 Flammable and Explosive Reactions

In SAR section 4.4.5, the applicant described their use of guidance provided in NUREG/CR-6673 to determine the time to reach a hydrogen concentration of 5 percent. The applicant supplemented this guidance with the information in EPRI NP-5799 which provides parameters for a wide range of ion exchange resins. The applicant defined the shipping time as one-half the time required to reach a 5 percent hydrogen concentration per the guidance in NUREG/CR-6673. The applicant explained that activated hardware materials neither generated hydrogen nor retained water that cannot be evacuated like filters and resin wastes. The applicant stated that, if the contents of the package only contain activated hardware and no hydrogenous materials like filters and resins waste are included, the hydrogen gas build up is not a concern.

The staff reviewed the applicant's approach to determine the shipping time for contents that may generate hydrogen. The staff noted that the guidance in NUREG/CR-6673 is applicable to transuranic waste that usually consists of transuranic nuclides mixed with plastics, metal, glass, paper, salts, absorbents, oxides, filters, filter media, cloth, concrete, and other waste materials. The staff further noted that the guidance in NUREG/CR-6673 is limited to the evaluation of hydrogen generation from radiolysis. The applicant provided supplemental information to address hydrogen generation from chemical reactions, thermal degradation, or biological activity. The applicant referenced United States Environmental Protection Agency's report EPA-600/2-80-076 that describes a method for determining the compatibility of combinations of hazardous wastes categorized in specific groups. The staff reviewed the details provided and determined the information provided satisfactorily establishes that these waste groups are compatible. The staff reviewed the description of the package contents provided by the applicant and determined that the guidance in NUREG/CR-6673, supplemented with the

information in EPRI NP-5799, is acceptable for evaluating hydrogen generation from the dewatered or grossly dewatered spent resins and filter media contents.

7.4 Conclusion

The staff finds that the Robatel Model RT-100 transportation package meets the regulatory requirements for package contents and content reactions. The staff also finds that the Robatel RT-100 transportation package is constructed with materials and processes that are in accordance with acceptable industry codes and standards.

7.5 Findings

- F7.1 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.33. The applicant described the materials used in the transportation package in sufficient detail to support the staff's evaluation.
- F7.2 The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.43(d). The applicant has demonstrated that there will be no significant corrosion, chemical reactions, or radiation effects that could impair the effectiveness of the packaging.

8.0 PACKAGE OPERATIONS

The applicant revised SAR section 7.5 to address the impact of activated hardware on hydrogen gas calculations. The staff evaluated these changes in SER section 4.2 and found them acceptable. The applicant also revised SAR section 7.5 to address the impact of activated hardware on package dose rates. The staff evaluated these changes in SER section 5.4 and found them acceptable. Based on a review of the statements and representations in the application, the staff concludes that the operating procedures meet the requirements of 10 CFR Part 71 and that these procedures are adequate to assure the package will be operated in a manner consistent with its evaluation for approval.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW

The applicant updated the references to identify the most recent version of the Quality Assurance Program. The staff determined this change to be editorial in nature; therefore, staff finds it acceptable. Based on a review of the statements and representations in the application, the staff concludes that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71, and that the maintenance program is adequate to assure packaging performance during its service life.

CONDITIONS

The CoC includes the following condition(s) of approval:

Condition 5(b)(2) was revised to identify both low density and high density activated hardware as contents authorized for transport.

Condition 11 was revised to identify the length of time that Revision 2 of the certificate may be used.

The references section has been updated to include this request.

Minor editorial corrections were made.

CONCLUSIONS

Based on the statements and representations contained in the application, as supplemented, and the conditions listed above, the staff concludes that the design has been adequately described and evaluated, and the Model No. RT-100 package meets the requirements of 10 CFR Part 71.

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