**SAFETY EVALUATION REPORT**
**Docket No. 71-9365**
**Model No. RT-100 Package**
**Certificate of Compliance No. 9365**
**Revision No. 0**

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# SAFETY EVALUATION REPORT
## Model No. RT-100 Package
## Certificate of Compliance No. 9365
## Revision No. 0


### SUMMARY

By application dated October 9, 2012, as supplemented September 18, 2013, January 14 and February 13, 2014, Robatel Technologies, LLC, requested a certificate of compliance (CoC) for the Model No. RT-100 package. Revision No. 4 of the package application, dated February 13, 2014, supersedes in its entirety the application dated October 9, 2012.

The Model No. RT-100 package consists of 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 annular space between the inner and outer shells is filled with a 90 mm thick lead for shielding. The primary lid, consisting of a 210 mm thick stainless steel forging, is fastened to the packaging body with 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 hex head bolts. Removable lifting lugs are utilized for removal and handling of the primary and secondary lids, as well as of the impact limiters.

The maximum gross weight of the package, including impact limiters, is 41,500 kg. Authorized contents include dispersible solids, in the form of dewatered resins and filters, contained within secondary containers. The maximum activity is not to exceed 3,000 times a Type A quantity.

The package was evaluated against the regulatory standards in 10 CFR Part 71. The analyses performed by the applicant demonstrate that the package provides adequate structural, thermal, containment, and shielding protection under normal and accident conditions.

NRC staff reviewed the application using the guidance in "Standard Review Plan for Transportation Packages for Radioactive Waste Materials," NUREG-1609. Based on the statements and representations in the application, and the conditions listed in the certificate of compliance, the staff concludes that the package meets the requirements of 10 CFR Part 71.


### References


Robatel Technologies, LLC, application “RT-100 Type B Cask Safety Analysis Report,” Revision No. 4, dated February 13, 2014.

## 1.0 GENERAL INFORMATION

The Model No. RT-100 package is a Type B (U)-96 package designed for the transport of resins and filters.

### 1.1 Packaging

The Model No. RT-100 packaging consists of a 35 mm thick outer stainless-steel shell, a 30 mm thick inner stainless-steel plate, 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 is a 210 mm thick stainless steel forging, fastened to the packaging body with thirty-two hex head bolts. The secondary lid, made of 100 mm thick stainless steel plate, a 60 mm thick lead gamma shield and a 10 mm thick stainless steel plate, is attached to the primary lid with eighteen hex head bolts. The annular space between the inner and outer shells is filled with a 90 mm thick lead. The internal cavity of the packaging is 1956 mm high with a diameter of 1,730 mm.

Impact limiters have an outside diameter of 2587 mm and are attached to the packaging through stainless-steel bolt ring flanges. The lower impact limiter extends 494 mm beyond the base of the packaging while the upper impact limiter extends 498 mm beyond the primary lid. The external shells of the impact limiters are stainless-steel to withstand large plastic deformations without fracturing. The volume inside the shells is filled with crushable shock-absorbing and thermal-insulating polyurethane foam, which is preformed and inserted into the shells to fill up all void spaces. The use of pre-formed foam ensures a homogeneous density.

The containment boundary of the Model No. RT-100 packaging 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. Pressure test ports are provided between the twin O-rings for the primary lid, between the O-rings for the secondary lid, and between the O-rings for the vent port cover plate. These ports are used for leak testing of the package in accordance with ANSI N14.5-1997. The vent port is provided for venting pressures within the containment cavity which may be generated during transport and prior to lid removal. Each port is sealed with an EPDM O-ring. Specification information for all O-rings is found in Chapter 4, Section 4.1.3, of the application.

The approximate dimensions and weights of the Model No. RT-100 packaging are the following:

| | |
|---|---|
| Inside Diameter of the Cavity: | 1,730 mm |
| Length of the Cavity: | 1,956 mm |
| Nominal Empty Packaging Weight: | 34,696 kg |
| Maximum Gross Weight of the Package: | 41,500 kg |

### 1.2 Contents

Authorized contents include dispersible solids, in the form of dewatered resins and filters, contained within secondary containers. The maximum activity of the contents shall not exceed 3,000 times a Type A quantity along with (i) the limit determined per the procedure in Section No. 7.6 of Chapter No. 7 of the application, for beta and gamma emitting radionuclides; and (ii) the mass limits for fissile materials as prescribed by 10 CFR 71.15 for exempting materials from classification as fissile material.

The maximum decay heat is 200 watts and the maximum weight of contents is 6,804 kg, including shoring and secondary containers.

Secondary containers, constructed of carbon steel, stainless steel, polyethylene or polypropylene, are required to be passively vented within the package cavity during shipment. The RT-100 stainless steel inner cavity does not interact with polyethylene or metal liners typically used in the nuclear industry for the shipment of resins and filters.

## 1.3 Materials

The inner and outer shells of the packaging, the primary and secondary lids, the outer bottom plate, and the inner bottom forging are made of ASTM A240, Type 304L stainless steel.

The primary lid is fastened to the packaging body with thirty-two (32) M48 hex head bolts (ASTM A354 Gr. BD or equivalent). The secondary lid is attached to the primary lid with eighteen (18) M36 hex head bolts (ASTM A354 Gr. BD or equivalent).

Lead shielding is provided between the inner and outer radial shells, as well as between the stainless steel bottom forging and bottom plate. The impact limiter external shells are stainless steel, and the volume inside the shell is filled with preformed polyurethane foam.

## 1.4 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

## 1.5 Evaluation Findings

A general description of the Model No. RT-100 package is presented in Chapter 1 of the package application, with special attention to design and operating characteristics and principal safety considerations. Drawings for structures, systems, and components important to safety are included in Section 1.3 of the application.

The package application identifies the Robatel Technologies, LLC, Quality Assurance Program for Packaging and Transportation of Radioactive Material, dated January 31, 2012.

The staff concludes that the information presented in this section of the application provides an adequate basis for the evaluation of the Model No. RT-100 package against 10 CFR Part 71 requirements, for each technical discipline.

## 2.0 STRUCTURAL REVIEW

The objective of the structural review is to verify that the structural performance of the package meets the requirements of 10 CFR Part 71, including performance under both normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

### 2.1 Structural Design

The major components of the packaging are the following: packaging body including the impact limiter attachment rings, bolting ring, primary and secondary lids, lifting pockets and tie-down arms, and two (upper and lower) impact limiters. These components are designed so that the structural responses of the package meet 10 CFR Part 71 requirements.

The general assembly drawings of the Model No. RT-100 package identify the major components of the package.

#### 2.1.1 Discussion

The cylindrical body of the package is comprised of a 35 mm (1.38 in) thick outer stainless steel shell and a 30 mm (1.18 in) thick inner stainless steel plate (ASTM A240, Type 304L). The annular space between the shells is filled with 90 mm (3.54 in) thick lead. The base of the packaging consists of a 30 mm (1.18 in) thick stainless steel outer bottom plate, a 75 mm (2.95 in) thick gamma shield of poured lead, and a 50 mm (1.97 in) thick stainless steel inner bottom forging (ASTM A240, Type 304L). The internal cavity dimensions are 1730 mm (68.11 in) in diameter and 1956 mm (77.0 in) high.

The primary lid, a 210 mm (8.27 in) thick stainless steel forging, is fastened to the packaging body with thirty-two (32) M48 hex head bolts (ASTM A354 Gr. BD or equivalent). The secondary lid, made of 100 mm (3.94 in) thick stainless steel plate, a 60 mm (2.36 in) thick lead gamma shield and a 10 mm (0.39 in) thick stainless steel plate, is attached to the primary lid with eighteen (18) M36 hex head bolts (ASTM A354 Gr. BD or equivalent).

Lead shielding is provided between the inner and outer radial shells, as well as between the stainless steel bottom forging and bottom plate. Shielding is provided by the following design features: (i) a 90 mm (3.54 in) of lead layer from the sidewall, (ii) a 75 mm (2.95 in) lead layer from the cask bottom, (iii) 210 mm (8.27 in) thick stainless steel of the primary lid, and (iv) 170 mm (6.69 in) thick stainless steel of the secondary lid with an embedded 60 mm (2.36 in) thick lead layer. Under NCT, the top and bottom impact limiters provide additional gamma shielding from the 10 mm (0.39 in) inner steel shell, polyurethane foam, and 4 mm (0.16 in) outer steel shell.

The two (upper and lower) impact limiters have an outside diameter of 2587 mm (101.85 in). The upper impact limiter extends 498 mm (19.61 in) beyond the primary lid, while the lower impact limiter extends 494 mm (19.45 in) beyond the base of the package. The impact limiter external shells are stainless steel, and the volume inside the shell is filled with preformed polyurethane foam.

The impact limiters are attached to the package via two stainless steel bolt ring flanges, welded along the circumference of the packaging, and considered a structural part of the package. Each impact limiter is equipped with twelve (12) M36 studs and attached to the bolt ring using twelve (12) M36 stainless steel hex head nuts.

The staff reviewed the drawings for completeness and accuracy, and found that the geometry, dimensions, material, components, notes, and fabrication details were adequately described throughout the application.

2.1.2 Design Criteria

The Model No. RT-100 package design complies with the "General standards for all packages," as specified in 10 CFR 71.43, and the "Lifting and tie-down standards" specified in 10 CFR 71.45. These criteria were discussed in Sections 2.4 and 2.5 of the application, and were used to evaluate the structural performance of the package.

The containment boundary is evaluated based on the American Society of Mechanical Engineers (ASME) Code requirements for Level A and D service, and is consistent with RG 7.6 "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," which provides design criteria based on the ASME B&PV Code, Section III. Therefore, the applicant used the allowable stress values for NCT Service Level A Limits and HAC Service Level D Limits. The tables in Section 2.1 of the application list the allowable stresses for various stress components under NCT and HAC loading conditions. The staff performed an independent verification of the calculated allowable levels, and confirmed that the values are accurate. The staff finds that the bolt allowable limits are in accordance with NUREG/CR-6007.

The load combinations used in performing the structural evaluations of the package are in accordance with RG 7.8, "Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Materials." Load combinations used for the analysis are summarized in Table 2.1.2-1 of the application.

Based on the review of the design criteria presented in Section 2.1.2 of the application, the staff finds that the structural design criteria for the package have adequate structural integrity to meet NCT and HAC requirements of 10 CFR Part 71.

2.1.3 Weights and Centers of Gravity

The nominal weights and centers of gravity are provided in Table 2.1.3-1 of the application. These weights are utilized in the structural evaluations to demonstrate compliance with NCT and HAC requirements.

2.1.4 Identification of Codes and Standards for Package Design

The package is designed as a Type-B, Category II, package in accordance with RG 7.11. The Codes and Standards used for the design of the package were selected based on the guidance provided in RG 7.6, which is consistent with ASME B&PV Section III, Subsection NB, and NUREG/CR-3854 for packages transporting Category II contents. As such, the package containment system is fabricated in accordance with the ASME Code, Section III, Subsection ND, while the tie-downs are fabricated in accordance with Subsection NF. The fabrication, examination, and inspection of the containment boundary components of a Category II package are evaluated per ASME B&PV Section III, Subsection ND.

In addition, RG 7.8 is used in identifying the load combinations to be used in the package design evaluation, and NUREG/CR-4554 is used in evaluating buckling of the containment vessel while NUREG/CR-6007 is followed for the bolt evaluations.

Based on the review of the Codes and Standards presented in Section 2.1.4 of the application, the staff finds that the Codes and Standards identified by the applicant are appropriate to evaluate the structural performance of the package under NCT and HAC conditions to meet the regulatory requirements of 10 CFR Part 71.

## 2.2 Materials

The following discussion focuses primarily on the structural components and/or items Important-to-Safety (ITS) for the package; however, a review of the materials listed in the Bill-of-Materials was also conducted.

The impact limiters are constructed from stainless steel (SS), American Society Mechanical Engineers (ASME) SA240, Type 304L. The volume inside the shell is filled with crushable shock-absorbing, thermal-insulating, preformed and inserted polyurethane foam, General Plastics (GP). The impact limiters are attached to the packaging via two stainless-steel bolt ring flanges welded along the circumference of the packaging. Each impact limiter is attached to the bolting ring with twelve M36 SS studs/nuts, SA479, Type 316L to ensure attachment for NCT and HAC conditions, as well as to facilitate removal during loading/unloading operations. Fusible plugs for over pressure relief/venting gases during HAC are fabricated of polyethylene with a melting point <160°C.

The package containment system is defined as the inner shell, together with the associated lid, O-ring seals, and lid closure bolts. The inner shell or containment vessel consists of a right circular cylinder, attached to a circular forged bottom/circular forged flange (top) with a full penetration weld. The primary lid is attached to the packaging body with thirty-two Carbon Steel (CS) M48 hex head bolts, American Standard Testing Materials (ASTM) A354, Grade BD with a secondary lid that covers an opening in the primary lid attached using eighteen CS ASTM S354, Grade BD M36 hex head bolts. The gamma shielding design is comprised of a SS/lead/SS body with a SS primary lid and a SS/lead/SS secondary lid. Lead is specified to be ASTM B29. 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 EPDM O-rings, ASTM D2000, M3DA 814 A26 B36 F19 retained in machined grooves. O-rings have a usable temperature range going from -50°C to 150°C which meets or exceeds both NCT and HAC requirements. The quick-disconnect valve is housed under a SS cover plate and is attached to the primary lid with six equally spaced SS, commercial, ISO 3506-1 M10 hex head bolts.

Material properties used in the package structural analyses are shown in Tables 2.2.1-1, 2.2.1-2, and 2.2.1-3. Material properties for the structural analyses of the polyurethane foam used in the impact limiter evaluations are provided in Appendix 2.12. Properties of both packaging materials and foam used in the thermal analyses are provided in Section 3.2.1.

Structural components of the packaging body are specified to be ASME SA240 Type 304/304L steel, with the exception of the tie-down straps, which are ASME SA240 UNS S31803, Type 318 SS. The primary and secondary lids are ASME SA240 Type 304/304L SS. Strength properties for these materials are presented in Table 2.2.1-1 while Table 2.2.1-2 provides

density and Poisson's ratio values. The lead properties are provided in NUREG/CR-0481 and are presented in Table 2.2.1-2.

The staff reviewed the materials selected and determined that they are acceptable and provide reasonable assurance for safety of the package. Specifications and temperature dependent mechanical properties, including yield strength, tensile strength, allowable strength, modulus of elasticity, and coefficient of thermal expansion conform to ASME Code, Section II, Part D.

### 2.2.1 Brittle Fracture

Section 2.2.1 of the application discusses material brittle fracture concerns. The applicant states that all the materials of structural components have sufficient fracture toughness to preclude brittle fracture under NCT and HAC. Regulatory Guides 7.11 and 7.12 are used to provide criteria for fracture toughness. The applicant shall procure all materials under the Quality Assurance Program with the specifications for each material. Regulatory Guides 7.11 and 7.12 do not apply to the use of SS ASTM A240, Type 304, Type 304L, and A240 UNS S31803 precludes brittle fracture under both NCT and HAC.

The staff finds that, by avoiding the use of ferritic steels, brittle fracture concerns are precluded; however, the use of ferritic steels in the Model No. RT-100 package is limited. Specifically, most primary structural packaging components are fabricated of Type 304/304L SS. Since this material does not undergo a ductile-to-brittle transition in the temperature range of interest (down to - 40ºC), it is safe from brittle fracture. The staff finds that, in austenitic SS metal, the force required to move dislocations is not strongly temperature dependent and the dislocation movement remains high (i.e., will deform more readily under load before breaking), even at low temperatures, and the material remains relatively ductile. Regulatory Guide 7.11 states that austenitic SS is not susceptible to brittle facture at temperatures encountered in transport. The HAC drop tests have been conducted at minus 40°C to determine if brittle fracture has any effect on the package, with compliance demonstrated if the containment vessel is undamaged and leaktight on completion of testing.

### 2.2.2 Chemical or Galvanic Reactions

Section 2.2.2 of the application discusses reactions due to chemical, galvanic, or other reactions. The applicant states that the materials used in the fabrication and operation of the Model No. RT-100 package, including coatings, lubricants, and cleaning agents, are evaluated to determine whether chemical, galvanic, or other reactions among the materials, contents, and environments can occur. All phases of operation, i.e., loading, unloading, handling, storage, and transportation, are considered for the environments that may be encountered under normal, off-normal, or accident conditions. Based on the evaluation, there are no potential reactions that could adversely affect the overall integrity of the packaging or the structural integrity and retrievability of the contents from the package. The evaluation conforms to NRC guidelines and demonstrates that the package meets the requirements of 10 CFR 71.43(d).

The staff concludes that, during normal conditions of transportation, the package internals will not be subject to continuous or frequent exposure to moisture or that any water intrusion is not likely to occur in great quantities. The number of any galvanic potential between the different metals used in fabrication is low. Therefore, the conditions required to create the possibility for galvanic corrosion is small. Further, visual inspections of the Model No. RT-100 package, of its ITS components and its payload cavity, are performed at various timed intervals and provide reasonable assurance against significant corrosion occurring unnoticed.

### 2.2.3 Effects of Radiation on Materials

Section 2.2.3 of the application discusses the effects of radiation. The applicant states that gamma radiation has no significant effect on metal and, therefore, the radiation produced by the contained radioactivity does not cause any measurable damage to the metallic components of the package. For seals, the absorbed dose in a year is expected to be below 350 rad which is below the polymer damage threshold of $10^5$ rad. Additional support information about the EPDM resistance to radiation up to 5 $10^8$ rads, while retaining reasonable flexibility and strength, and hardness and good compression resistance, is provided by an IEEE paper. For the ceramic thermal shield, the absorbed dose is expected to be below 350 rad. However, ceramic materials are insensitive to gamma radiation damage and, thus, the ceramic thermal shield is expected to be unaffected by radiation.

The staff finds that radiation effect on the elastomeric seals is not a concern as the seals are replaced on an annual basis. The contents of the package may emit alpha, beta, gamma, and neutron or a combination of these radiations. Austenitic SS, carbon steel and lead are durable materials that are able to withstand the damaging effects from the radiation. The EPDM O-ring seals fitted to the containment system are the only material on which the radiation may have an effect; however, it has been shown that, for the radioactive contents, the maximum dose to the containment seal is below 1 X $10^6$ rad whereas no change of physical properties of the EPDM containment seal is expected at radiation levels up to 1 X $10^6$ rad.

The staff finds that the Model No. RT-100 package meets the regulatory requirements for mitigating galvanic or chemical reactions, is unaffected by cold temperatures and is constructed with materials and processes in accordance with acceptable industry codes and standards.

## 2.3 Fabrication and Examination

### 2.3.1 Fabrication

As indicated in Section 2.1.4 above, all containment components are fabricated, examined, and inspected in accordance with ASME B&PV Section III, Subsection ND. In addition, all non-containment components are fabricated, examined, and inspected in accordance with ASME B&PV Section III, Subsection NF.

Reviewing NUREG/CR-3854, the staff concludes that the fabrication methods of the package use ASME B&PV Section III, Subsection ND, for all containment components and Subsection NF for all non-containment components.

### 2.3.2 Examination

All containment components are fabricated, examined, and inspected in accordance with ASME B&PV Section III, Subsection ND. All non-containment components are fabricated, examined, and inspected in accordance with ASME B&PV Section III, Subsection ND or NF.

The applicant provided acceptance tests and maintenance programs in Chapter 8.0 of the application that included requirement for visual examinations and measurements as well as weld examinations to be performed on the package.

The staff determines that the applicant provided reasonable details to describe the examination requirements.

## 2.4 General Requirements for all Packages

### 2.4.1 Minimum Packaging Size

The smallest overall dimension of the packaging body is 200 cm (78.7 in), which is larger than the requirement of 10 cm (4.0 in). The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.43(a).

### 2.4.2 Tamper-Indicating Features

The package incorporates a tamper-indicating seal that is installed on the aligning pin of the upper impact limiter to ensure that removal of the impact limiter by unauthorized individuals can be detected. These seals, when breached, will indicate that the package has been tampered with. The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.43(b).

### 2.4.3 Positive Closures

The lid and cover plate of the packaging are secured by multiple bolts. These bolts are tightened to a set torque value that cannot be inadvertently loosened. In addition, the applicant performed a stress analysis for the bolts to demonstrate that the bolts can maintain positive closure during operation. The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.43(c).

## 2.5 Lifting and Tie-Down Standards for All Packages

### 2.5.1 Lifting Devices

The packaging is designed to be lifted using a lifting yoke. The primary lifting device is a set of two lifting pockets that are welded to the outer shell of the packaging, where the lifting pockets are designed to allow the package to be lifted using a yoke after removal of the impact limiters. The primary and secondary lids and the upper and lower impact limiters are fitted with threaded bolt holes which provide for attachment of lifting rings that are used in lifting each component.

A dynamic factor of 1.35 was used for the lifting evaluations. Five lifting conditions were evaluated: (i) lifting pocket tear-out stresses, (ii) lifting pocket weld stresses for weld, (iii) lifting pocket weld stresses for cask, (iv) lifting pocket bearing stresses, and (v) lifting pocket average pure shear. The applicant provided the results of the evaluations that all of the lifting conditions meet the required factor of safety greater than 3.0 against yield, the factor of safety greater than 5.0 against ultimate stress for the tear out and weld stress, and the factor of safety greater than 1.0 for the bearing stresses and average pure shear. Based on the review of the evaluations, the staff finds that the results are acceptable. The applicant also provided the results of the evaluations for (i) primary lid lifting, (ii) secondary lid lifting, (iii) upper impact limiter lifting and, (iv) lower impact limiter lifting. The results showed that the calculated factors of safety are larger than the required factors of safety. The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.45(a).

### 2.5.2 Tie-Down Devices

The packaging uses two sets of tie-down arms, which are welded to two different tie-down plates that in turn are welded to the outer shell of the packaging. Each set of arms is designed

to securely position the packaging and absorb the latitudinal, longitudinal, and vertical forces required by 10 CFR 71.45.

The applicant evaluated the tie down arms and plates to demonstrate that these structural members of the packaging must withstand the required loads without impairing the safety of the packaging. The results of the evaluation showed that the tie-down arms and plates have acceptable margins of safety for design and can be operated safely.

The applicant also stated that, if the tie-down attachments were to fail due to excessive loading, the packaging is designed so that the failed provision would not impair the ability of the packaging to meet the other requirements of 10 CFR Part 71. The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.45(b).

## 2.6 Normal Conditions of Transport

The applicant analyzed the packaging using the ANSYS finite element analysis code, and performed scale drop testing to demonstrate compliance to 10 CFR 71.71.

The applicant developed a three-dimensional, 180° (half symmetry) finite element model, which included modeling of the major components, i.e., the inner and outer shells, flange, bottom plate, primary and secondary lids, and closure bolts. The body of the packaging was represented by using solid elements, contact elements, mass elements and spring/damper elements. The solid portion of the model was constructed using ANSYS solid (SOLID185) elements, while surface-to-surface contact elements were used to simulate the interaction between adjacent components. In addition, contact elements were also used to bond dissimilarly meshed components. To simulate the impact limiters, the interaction between the packaging body and the impact limiters was modeled using CONTAC52 gap elements, which act as compression only elements.

The applicant used temperature dependent material properties of the packaging components, and considered deadweight, internal pressure, and temperature for the NCT loading conditions. Additionally, the applicant performed finite element model verification and mesh density, as presented in Appendix A.4 of the calculation package RTL-001-CALC-ST-0402, Rev. 4. A hand calculation was also performed to verify that the stresses calculated by ANSYS are reasonable.

Furthermore, the applicant compared its finite element model analysis with the model analyses previously generated for other package designs. The results of the analysis are consistent with those from previous designs, where peak stresses were expected. In addition, the applicant performed confirmatory scale model testing of the package to demonstrate that the finite element methods used to calculate the accelerations and impact limiter deformation are consistent with the drop test results.

The staff finds that the finite element model analyses with a 1/3 scale model testing are adequate to determine realistic results for the NCT load conditions.

### 2.6.1 Heat

The applicant performed a thermal analysis for the packaging body and closure lids to demonstrate the structural adequacy of the design for the temperatures specified in 10 CFR 71.71(c)(1). Detailed thermal analysis and its results are presented in Chapter 3 of the application. The applicant provided a brief summary of the thermal analysis in the tables of

Section 2.6 of the application. The staff reviewed the results, and finds that the stress intensity values at critical components are within the allowable limits of the material.

The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.71(c)(1).

2.6.1.1 Summary of Pressure and Temperatures

Chapter 3 of the application presents the maximum normal operating pressure evaluation as well as the maximum component temperature evaluation for NCT. The pressure and temperatures were utilized to determine the stress allowables used in the structural evaluation for NCT.

2.6.1.2 Differential Thermal Expansion

The package used temperature dependent material properties of the components in the finite element analysis. Therefore, the differential thermal expansion was implicitly included in the stress calculations.

2.6.1.3 Stress Calculations

Using an ANSYS finite element model, the applicant analyzed the package with the range of primary plus secondary stresses for the combined normal events (including heat, cold, normal operating pressure, 0.3-m end drop, and 0.3-m side drop conditions) to satisfy the requirements of RG 7.6. Table Nos. 2.6.7-1 and 2.6.7-2 of the application tabulated the calculated stress intensities for NCT and HAC conditions against the material allowable limits.

2.6.1.4 Comparison with Allowable Stresses

The applicant compared the calculated stress intensities with the material allowable limits for NCT and HAC conditions in Table Nos. 2.6.7-1 and 2.6.7-2 of the application. The comparison shows that the margins of safety are all positive; therefore, the package satisfies the regulatory requirements of 10 CFR 71.71(c)(1).

2.6.2 Cold

The regulatory requirements of 10 CFR 71.71(c)(2) require that the package be subjected to a temperature of -40°C (-40°F) in still air and shade. The applicant performed a thermal evaluation for the Model No. RT-100 package in accordance with the temperatures specified in 10 CFR 71.71(c)(2) to demonstrate the structural adequacy of the package. Chapter 3 of the application provides a thermal evaluation for cold conditions using the methodology discussed in Section 2.6.1 of the application. The combined stress results are presented in Tables 2.6.7-1 and 2.6.7-2 of the application. The tables show that the calculated margins of safety are all positive. The staff reviewed the results, and finds that the packaging satisfies the regulatory requirements of 10 CFR 71.71(c)(2).

2.6.3 Reduced External Pressure

The regulatory requirements of 10 CFR 71.71(c)(3) require that the package be subjected to a reduced external pressure of 25 kPa (3.5 lbf/in$^2$) absolute. The applicant analyzed the package with an internal pressure of 241 kPa (35 psig), and provided the results in Tables 2.6.7-1 and

2.6.7-2 of the application. The staff reviewed the results presented in the tables of Chapter 2 of the application, and finds that the stress intensity values at critical components are within the allowable limits of the material.
The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.71(c)(3).

2.6.4 Increased External Pressure

The regulatory requirements of 10 CFR 71.71(c)(4) require that the package be subjected to an external pressure of 140 kPa (20 $lbf/in^2$) absolute. The applicant stated that an increased external pressure of 20 psia (5.3 psig external pressure), as specified in 10 CFR 71.71(c)(4), has a negligible effect on the package because of the thick outer shell and end closures of the package, and the conservatisms added in the analysis presented in Section 2.6.7 of the application for different loading cases which exceed the prescribed external pressure requirements. The staff reviewed the results of the analysis presented in Section 2.6.7 of the application, and finds that the stress intensity values at critical components are within the allowable limits of the material. The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.71(c)(4).

2.6.5 Vibration

The applicant stated that the Model No. RT-100 package consists of thick section of materials that should not be affected by vibration normally incident to transport. However, the package may be subjected to a cycle range typically associated with high-cycle fatigue ($>10^8$ cycles) during transportation. Therefore, the applicant performed vibration evaluations for both primary and secondary lid bolts to determine an endurance limit of the material for high cycle fatigue. The results of the evaluation show that the stress in the bolts is well below the endurance limit of the material; therefore, the primary and secondary lid bolts are not subjected to transport-related fatigue damage during their service life.

The applicant also evaluated both the primary and secondary lid bolts under an impact load because the lid closure bolts may be subjected to shock loading during transportation. The results of the evaluation provided in Sections and 2.12.4.1 and 2.13.3.3 of the application show that the primary and secondary lid bolts withstand a 125 g impact load, which is much larger than the shock loading during transportation. The applicant concluded that the primary and secondary lid closure bolts are acceptable for shock loadings.

The staff reviewed the calculation performed by the applicant, and finds that the stress intensity values are within the allowable limits of the materials. The staff determines that the requirements of 10 CFR 71.71(c)(5) for normal vibration incidents and shock loading conditions during transport are met.

2.6.6 Water Spray

The regulatory requirements of 10 CFR 71.71(c)(6) require that the package must be subjected to a water spray test that simulates exposure to rainfall of approximately 5 cm/h (2 in/h) for at least 1 hour. The water spray test is primarily intended for packaging relying on material that absorb water and/or are soften by water material bounded by water soluble glue. The packaging outer layer is designed to be fabricated entirely of metal. Thus, the water spray test is not applicable and the packaging satisfies the regulatory requirements of 10 CFR 71.71(c)(6).

2.6.7 NCT Free Drop

The applicant evaluated the package for the free drop requirements of 10 CFR 71.71 by a combination of classic calculations, finite element analysis, and scale model drop testing. The evaluations included the qualification of the cover bolt design for the combined effects of free drop impact force, internal pressures, thermal stress, O-ring compression force, and bolt preload using the methodology provided in NUREG/CR-6007. Analyses were performed in three orientations – end, side, and center of gravity over corner.

The package was evaluated for the NCT 0.3 m (1 ft) side-drop onto a flat, unyielding, horizontal surface. Table 2.6.7-1 of the application presents the stress results for the side drop combined loading conditions. As shown in Table 2.6.7-1, the margins of safety are positive when compared with the stress intensity for each category. The most critically stressed component is the inner lid. The locations of the critical sections correspond to the maximum stress location shown in Figures 2.6.7.3-1 through 2.6.7.3-11 of the application. The staff reviewed the analysis, and verified that the most critically stressed component is the inner lid.

The results for the top and bottom-end drop combined loading conditions are documented in Table 2.6.7-2 of the application, which shows the primary membrane (Pm), primary membrane plus primary bending (Pm+Pb), and primary membrane plus primary bending plus secondary peak stress (Pm+Pb+Q) in accordance with the criteria presented in RG 7.6. The margins of safety for the primary stress intensity category are positive for all of the top-end drop conditions.

The most critically stressed component in the system is the packaging flange region from the bending of the flange due to the inertial load imposed by the lids. The locations of the critical sections correspond to the maximum stress location and are shown in Figure 2.6.7-12 through Figure 2.6.7-17 of the application. The staff determines that the package meets the regulatory requirements of 10 CFR 71.71(c)(7).

2.6.8 Corner Drop

The corner drop test is not applicable because (i) the Model No. RT-100 package is composed of materials other than fiberboard or wood, and (ii) its weight exceeds 100 kg. As such, the packaging satisfies the regulatory requirements of 10 CFR 71.71(c)(8).

2.6.9 Compression

The compression test is not applicable because the packaging weight is greater than 5,000 kg (11,000 lb). As such, the packaging complies with the regulatory requirements of 10 CFR 71.71(c)(9).

2.6.10 Penetration

The Model No. RT-100 package was not tested for penetration based on the statement provided in RG 7.8, i.e., “the penetration test of 10 CFR 71.71 is not considered by the NRC staff to have structural significance for large shipping casks (except for unprotected valves and rupture disks) and is not considered as a general requirement.”

The package has no unprotected valves or rupture disks that could be affected by normal conditions of transport. Thus, the packaging is in compliance with the regulatory requirements of 10 CFR 71.71(c)(10).

## 2.7 Hypothetical Accident Conditions

The evaluation of the package for HAC conditions was done by analytical methods. The applicant described the details of its 3-D finite element model in Section 2.6 of the application, and used the same 180°ANSYS finite element model than the one used for the NCT loading conditions, as discussed in Section 2.6 above.

### 2.7.1 Free Drop

In order to determine the orientation that produces the maximum damage, the applicant evaluated the package for impact orientations in which the package strikes the impact surface on its bottom end and side. The evaluation of each drop orientation was performed by using finite element analysis techniques. A complete description of the 3-D model used to analyze the package body is presented in Section 2.6.7.2 of the application. The package was analyzed under a 9 m (30 ft) end drop, side drop, corner drop (38° from vertical), and oblique drop conditions onto an unyielding surface. The impact limiters were evaluated in Appendix 2.12 for all loading conditions and orientations. From the staff's review, the staff finds that the calculations for the impact-limiter reaction forces are reasonable.

#### 2.7.1.1 End Drop

The stress intensity levels for components at the worst locations were tabulated in Table 2.7.1-2 of the application. The table documented the primary membrane (Pm), primary membrane plus primary bending (Pm+Pb) stresses in accordance with the criteria presented in RG 7.6.

The applicant calculated the potential lead slump under an end drop impact. Based on the lead material properties and the quality controls in place during the lead pouring procedure, the maximum gap, that may develop due to lead shrink, is 1 mm between the inner and outer shells housing the lead layer. The applicant determined that, with this available gap, the potential for lead slump under HAC is a 2.478 cm void gap at the end.

The staff reviewed the analysis results, and verified that, when compared with the allowable stresses in RG 7.6, a minimum safety factor of 1.6 was reached at the flange for the stress category of primary membrane and primary bending (Pm+Pb), which was most critically stressed due to bending, as a result of the inertial loads on the cask lids. The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.73(c)(1).

#### 2.7.1.2 Side Drop

The stress intensity levels for components at the worst locations were tabulated in Table 2.7.1-3 of the application. The table documented the primary membrane (Pm), primary membrane plus primary bending (Pm+Pb) stresses in accordance with the criteria presented in RG 7.6.

The applicant calculated the potential lead slump under a side drop impact. Based on the available space developed due to lead shrinkage, the applicant determined that the lead slump will form a void gap, in the shape of a crescent, with a 2.3216 cm thickness at the widest point of the crescent.

The staff reviewed the analysis results, and verified that, when compared with the allowable stresses in RG 7.6, all calculated margins of safety were positive. The most critically stressed component in the system was the outer shell due to the ovalization of the packaging body and

the inertial load of the lead shield. The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.73(c)(1).

2.7.1.3 Corner Drop

The stress intensity levels for components at the worst locations were tabulated in Table 2.7.1-5 of the application. The table documented the primary membrane (Pm), primary membrane plus primary bending (Pm+Pb) stresses in accordance with the criteria presented in RG 7.6. The staff reviewed the analysis results, and verified that, when compared with the allowable stresses in RG 7.6, a minimum safety factor of 1.0 was reached at the inner lid for the stress category of primary membrane and primary bending (Pm+Pb). The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.73(c)(1).

2.7.1.4 Oblique Drops

The packaging was analyzed for an oblique drop case (also referred to as shallow angle drop condition), using the equations presented in Section 2.7.1.4 of the application. While package designs with a length-to-diameter ratio greater than 1.37 may result in oblique impact velocities greater than the side drop, the Model No. RT-100 package has a length-to-diameter ratio of 1.28; therefore, the impact velocities are lower than for the side drop and stresses in the package during the oblique drop event are lower than those experienced during the side drop.

The staff finds that the applicant's approach is acceptable based on the staff's review of the study presented in Sandia Report SAND90-2187, titled "An Analysis of Parameters Affecting Slapdown of Transportation Packages." The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.73(c)(1).

2.7.2 Crush

The dynamic crush is only required when the specimen has a mass not greater than 500 kg (1100 lb), an overall density not greater than 1000 kg/m$^3$ (62.43lb/ft$^3$) based on external dimension, and radioactive contents greater than 1000 $A_2$ not as special form radioactive material. As such, the Model No. RT-100 package, with a weight of more than 500 kg and an overall density greater than 1000 kg/m3, satisfies the regulatory requirements of 10 CFR 71.73(c)(2).

2.7.3 Puncture

The applicant evaluated the package for puncture drop by using the classical elastic analysis and finite element analysis methods. The calculation package, RTL-001-CALC-ST-0403, Rev. 4, details the evaluation of the regulatory puncture drop and the hypothetical drop test. The applicant considered two scenarios: (i) a puncture drop on the lid, and (ii) a puncture drop on the sidewall. For the analysis of the puncture drop on the lid, the applicant used the ANSYS finite element code. The results of the analysis were provided in Table 2.7.3-1 of the application which documents the primary membrane (Pm), primary membrane plus primary bending (Pm+Pb) stresses in accordance with the criteria presented in RG 7.6, and shows that the margins of safety are positive when compared to the stress intensity for each category.

For the analysis of the side puncture, the applicant calculated 29 mm (1.16 in) as the minimum required outer shell thickness using Nelm's Equation per ORNL-NSIC-68, titled "Cask Designers

Guide", dated February 1970. The applicant concluded that the outer shell is sufficient to resist the puncture since the package has a thickness of 35 mm (1.38 in).

The applicant determined that the maximum deformation occurs during a postulated puncture event when the packaging strikes the puncture probe approximately mid-span on the packaging outer shell. Figure 2.7.3-3 of the application shows the side puncture details. In these analyses, the applicant considered the package to be a closed cylinder subjected to a concentrated load at mid-span. The deformation is obtained from Table 31, Case 9 of the "Roark's Formulas for Stress and Strain, 6$^{th}$ Edition". Based on such analyses, the applicant determined that the maximum lead deformation is 1.7 cm (17 mm).

Based on the review of the analysis results, the staff determines that the packaging meets the regulatory requirements of 10 CFR 71.73(c)(3).

2.7.4 Thermal

The regulatory requirements of 10 CFR Part 71.73(c)(4) require exposure of the package with an average flame temperature of at least 800°C (1475°F) for a period of 30 minutes. The applicant performed an evaluation for the thermal expansion of the bolts and evaluation for the packaging body under pressures associated with the fire accident that produces a surrounding environment of 800°C (1475°F) for a period of 30 minutes. The HAC thermal evaluations are presented in Chapter 3 of the application and the staff's detailed thermal evaluation is provided in Chapter 3 below.

2.7.4.1 Summary of Pressure and Temperatures

The applicant used the ANSYS finite element computer code to analyze the package components temperatures under varying conditions. The cavity pressure was estimated based on the surface averaged temperature of the inner shell at the cavity side.

Table 3.1.3-2 of Chapter 3 of the application presented the maximum HAC temperatures along with the maximum surface averaged temperature of inner shell surface at the cavity side. Table 3.1.4-1 summarizes the maximum HAC pressures. A detailed safety evaluation regarding the issues related to pressure and temperatures is provided in Chapter 3 below.

2.7.4.2 Differential Thermal Expansion

The applicant performed an evaluation on the closure bolts for thermal expansion under the fire accident condition. Appendix 2.13 of the application presents the bolting evaluation and the effects of thermal expansion on the closure bolts. A detailed safety evaluation regarding the issues related to the thermal expansion is provided in Chapter 3 below.

2.7.4.3 Stress Calculation

The bolt stress evaluation was presented in Appendix 2.13 of the application. The applicant compared the calculated bolt stresses with the allowable stresses from Section 2.1.2.2 of the application, based on the recommendations of NUREG/CR-6007. The comparison shows that the calculated stresses are less than the allowables; therefore, it was concluded that the bolts continue to provide a tight seal, and that containment is maintained. Based on the verifications and review of the calculations, the staff finds that the applicant's calculations are acceptable.

2.7.4.4 Comparison with Allowable Stresses

The applicant presented the accident pressure stresses in Table 2.7.4-1 of the application. The table documented the primary membrane (Pm), primary membrane and plus primary bending (Pm+Pb) stresses in accordance with the criteria presented in RG 7.6. The margins of safety are positive when compared to the stress intensity for each category. The minimum margin of safety was found to be +6.4 for the primary membrane plus bending stress intensity (Pm+Pb). The staff verified the applicant's calculations and the results, and finds that the calculated margins of safety are all positive. The staff determines that the packaging meets the regulatory requirements of 10 CFR 71.73(c)(4).

2.7.5 Immersion – Fissile Material

The HAC immersion test for fissile materials is not applicable, as the package does not have any fissile material subject to the requirements of 10 CFR 71.55.

2.7.6 Immersion – All Packages

The regulatory requirements of 10 CFR 71.73(c)(6) are required for all packages. Based on the review of the results of the evaluations provided in Chapter 2 of the application, the staff confirms that the stress intensity values at critical components are within the allowable limits of the material. As such, the package meets the regulatory requirements of 10 CFR 71.73(c)(6).

2.7.7 Deep Water Immersion Test

The Model No. RT-100 package does not contain irradiated fuel or contents containing more than 3,000 $A_2$. Thus, the deep water immersion test is not applicable.

2.7.8 Summary of Damage

The applicant demonstrated in Section 2.7.8 of the application that the results of the analyses reported in Section 2.7.1 through 2.7.7 indicated that the damage incurred by the package during HAC is minimal, and that such damage does not diminish the ability of the package to maintain the containment boundary. A 9-meter drop or a 1-meter pin puncture accident may damage the outer shell, and result in a localized reduction in shielding ability.

The applicant calculated the potential lead slump under an end drop, a side drop, and a puncture event, and found that the maximum gap formed by lead shrinkage, in the space between the inner and outer shells housing the lead layer, is 1 mm. The applicant determined that the potential for lead to slump under HAC is a 2.478 cm void gap at the end due to the lead slump when the package is subjected to an end drop and a crescent shaped void on the side of the package when the package is subjected to a side drop. The crescent lead slump void gap is 2.3216 cm thick at the widest point of the crescent.

The applicant also performed structural analyses on the end and side puncture impacts. The applicant determined that the maximum deformation occurs during a postulated puncture event when the package strikes the puncture probe approximately mid-span on the package outer shell. Figure 2.7.3-3 of the application shows the side puncture details. In these analyses, the applicant considered the package to be a closed cylinder subjected to a concentrated load at the mid-span. The deformation was obtained from Table 31, Case 9 of “Roark’s Formulas for

Stress and Strain, 6th Edition". Based on these analyses, the applicant determined that the maximum lead deformation is 1.7 cm.

Based on its review, the staff determined that the assessed damages are conservative and acceptable, and that the package can safely withstand the HAC free drop, puncture, and fire test performed in sequence. The staff determines that the package meets the regulatory requirements of 10 CFR 71.73.

2.8 Accident Conditions for Air Transport of Plutonium, for Fissile Material Packages for Air Transport, Special Form, Fuel Rods

These sections are not applicable.

2.12 Evaluation Findings

On the basis of the review of the applicant's responses and the statements and representations in the application, the staff concludes that the Model No. RT-100 package is adequately described and evaluated to demonstrate that its structural capabilities meet the regulatory requirements of 10 CFR Part 71.

## 3.0 THERMAL REVIEW

The objective of the review is to verify that the thermal performance of the Model No. RT-100 package has been adequately evaluated for the tests specified under both NCT and HAC conditions and that the package design satisfies the thermal requirements of 10 CFR Part 71.

3.1 Description of Thermal Design

The package is constructed of 304/304L stainless steel inner and outer shells, with lead shielding between these radial shells as well as in between the 304/304L stainless steel bottom forging and the bottom plate. The impact limiters cover the top and bottom ends of the package and protect both the lead in the bottom of the package and the O-ring seals in the primary and secondary lids and the vent port cover plate. The polyurethane material of the impact limiters insulates the lead and the O-rings from the high temperatures under HAC. The thermal shield covering the radial surface of the package is made of a ceramic fiber material for preventing heat input during the HAC fire.

The Model No. RT-100 package is designed for a maximum decay heat of 200 watts, a value selected as the design basis and conservative for the contaminated resin and filter contents. The thermal criteria of the Model No. RT-100 package are to maintain (i) the lead shielding in the package body and the secondary lid below the melting point, and (ii) the maximum temperature of the O-ring seals below the maximum operating temperature.

3.2 Material Properties and Component Specifications

The applicant presented the material properties and specifications of the Model No. RT-100 package in Chapter 3.2 of the application. The applicant listed the temperature-independent properties in Table 3.2.1-1 and the temperature-dependent properties in Tables 3.2.1-2 to 3.2.1-4 for stainless steel 304, lead, ceramic paper, polyurethane, and the EPDM seal. The component specifications of 304/304L stainless steel, lead, polyurethane foam, and the seal (EPDM) are listed in Table 3.2.2-1.

The staff reviewed the material properties of the contents and of the packaging components and compared them with the materials source data. The staff confirmed that the material properties are appropriate and correctly incorporated into the thermal evaluation.

### 3.3 General Considerations

The applicant stated in Section 1.2.2.6 of the application that neither a determination of hydrogen gas generation nor a restriction of shipping time is required for the secondary package containing materials with a radioactive concentration less than LSA, if shipped within 10 days of preparation or within 10 days of venting the container.

The shipper should ensure that the hydrogen generation in the cavity will be below 5% by volume, which represents the lower flammability limit for hydrogen. The staff reviewed the application and confirms that there is no significant hydrogen generation issue with the Model No. RT-100 package.

### 3.4 Thermal Evaluation under Normal Conditions of Transport

#### 3.4.1 Heat and Cold

The applicant evaluated the thermal performance of the package using the 3-D ANSYS code. Under NCT, the heat is transferred by radiation with an emissivity of 0.8, natural convection with a heat transfer coefficient of 5 $W/m^2$-$^oC$, heat flux from solar insolation, and uniform internal heat flux of 13.04 $W/m^2$ from the internal decay heat of 200 watts. The constant insolation of 388 $W/m^2$ for the curved surface and 776 $W/m^2$ for the flat surface was used in the “Hot Case 1” (with insolation).

The applicant presented the predicted NCT component temperatures of the package surface, impact limiter surface, outer shell, inner shell, lids, base plate, lead, and O-ring seals in Table 3.1.3-1 of the application. Each of the components is below the corresponding maximum allowable temperature limit as described in the application.

The maximum temperature calculated for the package outer surface under NCT is 41.3°C (with no insolation), which is less than 50°C for non-exclusive use shipment or 85$^o$C for exclusive use shipment when the package is subject to the requirements of 10 CFR 71.43(g).

The applicant calculated an average cavity gas temperature of 70.0$^o$C, a maximum inner shell temperature of 73$^o$C, a maximum lead temperature of 73.1$^o$C which is below the allowable limit of 328$^o$C, and a maximum O-ring seal temperature of 72.5°C (among the primary seal, secondary seal, and cover plate seal) which is far below the allowable limit of 150°C under NCT.

The applicant stated that the O-ring seals (EPDM) have a minimum allowable service temperature of -45$^o$C, as shown in Table 3.2.2-1 of the application, which is less than -40$^o$C, required by 10 CFR 71.71. The staff reviewed the attached Appendices 3.6-1 and 3.6-2 and concludes that the EPDM seals have an operating range between -45°C and 150°C.

The staff reviewed the assumptions, boundary conditions, and parameters used in the computer model, and the temperature profiles output from the thermal evaluation. Based on the staff’s review, the staff finds that the applicant’s thermal evaluation under NCT is acceptable and meets the requirements of 10 CFR 71.71.

3.4.2 Maximum Normal Operating Pressure

The contents of the Model No. RT-100 package are dewatered resins and filters and the quantity of water is limited. The gases within the package, i.e., a mixture of air, water, oxygen, and hydrogen generated from radiolysis of the residual water, behave as an ideal gas. The ideal gas law is used to calculate the cavity pressure.

The applicant used an upper bound temperature of 80°C and calculated a maximum total pressure of 26.5 psia by summing the pressures due to the gases initially present in the package, due to the water vapor from the contents or packaging, and due to the generation of gas by radiolysis under NCT (see RTL-001-CALC-TH-0102). The calculated pressure of 26.5 psia is below the MNOP of 49.7 psia (or 35.0 psig).

The staff reviewed the pressure calculation and finds acceptable the calculated pressure of 26.5 psia, which is below the MNOP of 49.7 psia under NCT.

3.4.2 Maximum Thermal Stress

The applicant showed the temperatures of the components of the package in Table 3.1.3-1 of the application, and stated that the thermal expansion is very limited because the temperatures of the components of the package differ by only a few degrees under NCT.

After reviewing the component temperatures, as presented in Figures 3.3.1-3 through 3.3.1-12, and the component characteristics, the staff concludes that the thermal stress is negligible under NCT. Based on the staff's review, the staff finds acceptable the applicant's evaluation of the thermal stresses under NCT.

3.5 Thermal Evaluation under Hypothetical Accident Conditions

The applicant evaluated two limiting events under HAC: (1) for the lid impact, the limiting configuration considered a pin puncture through the top impact limiter directly into the secondary lid at the location of the O-ring seal, and directly exposed the secondary lid to the hypothetical accident fire and (2) for the package sidewall impact, the limiting configuration considered the pin puncturing the thermal shield directly below the lifting block and increased the area of the outer shell of the package that is not protected by the thermal shield and maximize the heat input into the lead.

The applicant modeled the contact between the lead and the inner and outer shells as bonded surfaces for thermal analyses of HAC in order to maximize heat input to the lead. The contacts: (a) between the upper flange and the primary lid, (b) between the primary lid and the secondary lid, (c) between the impact limiters and the cask body, (d) the bolts with the primary lid, and (e) the bolts with the secondary lid were also modeled as thermal contacts to increase the heat input to the package. The applicant simulated the top impact limiter without the stainless steel plate covering the central hollow portion of the limiter, and exposed the concave area of the top impact limiter to solar insolation and/or fire. The staff agrees that such approaches lead to higher temperatures of the package components under the HAC fire.

Instead of natural convection in NCT, the transfer of heat from the fire source to the package takes by a combination of radiation with absorptivity of 0.9 and forced convection with heat transfer coefficient of 10 $W/m^2$-$^oC$ during the 30-minute HAC fire with no insolation. During the post-fire cooldown, the transfer of heat from the package to the ambient combines radiation with

an emissivity of 0.8, natural convection with heat transfer coefficient of 5 $W/m^2$-°C, and solar insolation.

The staff reviewed the assumptions, boundary conditions, and parameters used for the HAC thermal analysis. Based on the staff's review, the staff finds acceptable the applicant's HAC thermal evaluation, thus meeting the thermal requirements of 10 CFR 71.73.

### 3.5.1 Maximum Temperatures

The applicant presented the maximum calculated component temperatures of the inner shell average, lead shield, closure bolt, base packaging body, and O-ring seals under HAC with pin puncture damage at the top impact limiter (Table 3.1.3-2) and at the side of the packaging body (Table 3.1.3-3). Each component is below the allowable temperature limit, as described in the application.

For the top lid impact fire transient, the applicant calculated an average cavity gas temperature of 150°C, an average inner shell temperature of 136.3°C, a maximum lead temperature of 304.8°C which is below the allowable limit of 328°C for the lead, and a maximum O-ring seal temperature of 133.1°C (among the primary, secondary seal and cover plate seals) which is far below the allowable limit of 150°C under a top impact HAC fire.

For the package sidewall impact, the applicant calculated an average cavity gas temperature of 150°C, an average inner shell temperature of 137.0°C, a maximum lead temperature of 304.7°C which is below the allowable limit of 328°C for the lead, and a maximum O-ring seal temperature of 110.3°C which is far below the allowable limit of 150°C under the side impact HAC fire.

The staff reviewed the temperature profiles resulting from the thermal evaluations of the HAC fire and its post-fire cool-down. The staff confirmed that the component temperatures calculated from the HAC thermal analyses are below the corresponding allowable maximum temperature limits. Thus, the staff finds them acceptable.

### 3.5.2 Maximum Pressure

The applicant used an upper bound temperature of 150°C and calculated a maximum total pressure of 97.47 psia by summing the pressures from the initial gases, the water vapor, and the gas by the radiolysis under HAC (see RTL-001-CALC-TH-0202, Rev. 6) with no combustion. Even with an improbable significant combustion within the Model No. RT-100 package, the applicant also assumed a complete combustion of the wood as the bounding case and calculated a maximum pressure of 99.3 psia by summing the pressures from the water vapor, the gas by radiolysis and the phase transformation (including the pressure rise due to temperature rise and gas generation from combustion). The staff reviewed the calculation packages of RTL-001-CALC-TH-0202 and RTL-001-CALC-TH-0301 and concludes that the maximum pressures of 97.47 psia (with no combustion) and 99.3 psia (with combustion) are still below the maximum pressure of 100 psia, used for the package structural analysis under HAC. Therefore, the staff finds acceptable the applicant's evaluation of the maximum pressures.

### 3.5.3 Combustion and Over-pressurization

As noted in Section 3.4.3.2.6 of the application, package contents include filters that may be constructed from paper. The paper has an auto-ignition point of 232°C and may ignite when the package inner shell surface temperature is up to 275°C under an HAC fire with pin damage on

top impact limiter. However, the applicant delineated in Section 3.4.3.2.6 that the significant combustion of paper is improbable due to (1) a limited amount of paper in the package, (2) a limited amount of air in the package (3) an inert atmosphere, (4) a sealed container to prevent inleakage of air, and (5) desorption of water during the HAC fire. The applicant performed the pressure analysis to assure there is no over-pressurization to impact the containment integrity. By assuming the complete combustion of wood which has the highest heat of combustion, the applicant calculated a combustion pressure of 99.3 psia which is below 100 psia as used in the structural evaluation of the package.

The staff reviewed the calculations provided by the applicant and agrees that the auto-ignition of paper should not cause a significant combustion issue and an over-pressurization for both top-lid impact and sidewall impact HAC fires. This is because the limited paper mass is consumed with the limited air supply, the water desorption, and the inert atmosphere within the Model No. RT-100 package, and the pressure rise due to ignition and gas generation is still limited below the 100-psia limit.

### 3.5.4 Maximum Thermal Stresses

The applicant evaluated the thermal stress produced by the temperature gradients in the package under the HAC fire, and presented the details in Section 2.7.4 of the application.

With the closure bolts being the only components of concern during the HAC fire for ensuring the integrity of the containment boundary, the staff concludes the thermal stress is not significant on the closure bolts since the thermal gradients across clamped components and the primary/secondary closure lid bolts are almost zero.

The staff finds acceptable the evaluation of the maximum thermal stresses under the HAC fire.

### 3.6 Evaluation Findings

The staff reviewed the package description, the material properties, component specifications and the methods used in the thermal evaluation, and has reasonable assurance that they are sufficient to provide a basis for evaluation of the package against the thermal requirements of 10 CFR Part 71. The staff reviewed the accessible surface temperatures of the package as it will be prepared for shipment and has reasonable assurance that the temperatures satisfy 10 CFR 71.43(g) for packages transported by exclusive-use vehicle. The staff reviewed the package preparations for shipment and has reasonable assurance that the package material and component temperatures will not extend beyond the specified allowable limits during NCT, consistent with the tests specified in 10 CFR 71.71. The staff also has reasonable assurance that the package material and component temperatures will not exceed the specified allowable short-time limits during HAC, consistent with the tests specified in 10 CFR Part 71.

Based on review of the statements and representations in the application, the staff concludes that the thermal design has been adequately described and evaluated, and that the thermal performance of the Model No. RT-100 package meets the thermal requirements of 10 CFR Part 71.

## 4.0 CONTAINMENT REVIEW

The objective of the review is 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. The package is designed for ground transport of contents consisting of dewatered resins, filters, and mixtures of resins and filters that contain byproduct radioactive nuclear material; plutonium and fissile material are not included. The content is not in powdered form and the activity is not

greater than 3,000 A2. The resins and filters are loaded into secondary containers (liners), which are constructed of carbon steel, stainless steel, or thermoplastics, such as polyethylene or polypropylene. These secondary containers are placed within the package cavity. Regulations applicable to the containment review include 10 CFR 71.31, 71.33, 71.35, 71.43, and 71.51.

4.5.1 Description of the Containment System

The containment boundary consists of a stainless steel inner shell, bottom plate, and top flange, stainless steel primary lid and EPDM inner O-ring, stainless steel secondary lid and EPDM inner O-ring, and stainless steel vent port cover plate and EPDM inner O-ring. The vent port includes a quick disconnect valve that penetrates the primary lid near the lid's bolt circle. There are full penetration welds on the inner shell, between the inner shell and bottom plate, and between the inner shell and upper flange. The containment system is fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection ND.

Two concentric O-rings are used to seal and test each primary lid, secondary lid, and cover plate; the inner O-ring acts as a containment boundary and the outer O-ring bounds a test volume for leakage testing. The O-rings are supplied by Parker Hannifin Corporation or Trelleborg Sealing Solutions to standard specifications, as listed in Section 8.1.5.2 of the application. Seal groove dimensions are based on vendor data and the applicant's experience with the O-ring/gland design. The O-rings have an allowable temperature range between -45°C and 150°C.

A number of features exist to prevent the inadvertent opening of the package. The primary lid is bolted to the shell upper flange with thirty-two M48 hex head bolts. The secondary lid is attached to the primary lid with eighteen M36 hex head bolts. A cover plate, which fits over a recessed quick-disconnect valve, is attached to the primary lid with six M10 hex head bolts. The required bolt torque values for the primary lid, secondary lid, and cover plate are defined in Chapter 7 of the application. Access to the ends of the package is prevented by the attachment of upper and lower impact limiters via twelve M36 stud and bolt ring nuts. The upper impact limiter design includes polyurethane foam to protect the lids from structural impacts and thermal conditions. The secondary lid and center portion of the primary lid are covered by 130 mm of foam and the vent port cover plate and peripheral region of the primary lid are covered by 480 mm of foam.

The staff concludes that the applicant adequately described the package's containment features, as discussed above.

4.5.2 General Considerations

The application includes an analysis, based on NUREG/CR-6673 and EPRI NP-5977 reports, to determine the mole fraction of flammable gas generated within the package due to radiolysis of ionic resin beads, water, and a polyethylene container. Although contents within the package may include resin beads, filters, plastic liners, wood shoring, and 1% free water, flammable gas generation analyses treat all content as resin beads, polyethylene, or water in order to use the most conservative gas generation G-values. For example, stainless steel filters are treated as resin beads in the calculations whereas wood shoring, polyethylene filters, and polypropylene filters are treated as polyethylene. The calculations assume that all of the decay energy is absorbed by the contents. The calculations account for an increase in published G-values because the temperature inside the package is greater than ambient. The analysis models the

ionic resin as spheres with a random packing factor of 0.64, such that the remaining 0.36 fraction of free space consists of 0.25 fraction of water and 0.11 fraction of air. The analysis includes equations that determine the limited shipping time, decay heat, and waste volume to limit the generation of flammable gas to less than 5% mole fraction. To aid package users, the application includes Table 7.5.1-1 and Figure 7.5-1 that visually shows acceptable values of decay heat and waste volume that would limit the generation of flammable gas to less than 5% mole fraction within a 10 day shipment period. The shipment period is one half the period of time to reach a 5% mole fraction, thus providing margin to prevent concentrations from reaching the lower flammability limit. If the package content is such that decay heat and waste volume fall beyond the range of the curve in Figure 7.5-1 or the conditions in Table 7.5.1-1 are not met, users of the package must rely on the equations in Chapter 7 of the application to determine the acceptable content decay heat and waste volume.

The staff reviewed and accepted the applicant's discussion on limiting flammable gas generation.

4.5.3 Containment under Normal Conditions of Transport

Detailed containment release calculations that include release fractions, etc., were not necessary because the containment boundary is tested to the ANSI N14.5 "leaktight" criterion. The applicant performed relevant analyses to show the integrity of the containment boundary during NCT conditions, as described in 10 CFR 71.71. NCT thermal analyses indicated that the primary lid, secondary lid, and cover plate O-ring temperatures were below the allowable limits. The staff reviewed and accepted the applicant's description of the containment system's response under NCT.

4.5.4 Containment under Hypothetical Accident Conditions

As stated above, detailed containment release calculations that include release fractions, etc., were not necessary because the containment boundary is tested to the ANSI N14.5 "leaktight" criterion. Using the ANSYS computational modeling software, the applicant analyzed the structural response of the package to simulated 10 CFR 71.73 HAC events, including puncture and numerous drop orientations, to evaluate the effects on the package. The end drop and puncture at the lid center showed that, although there was deformation of the 130 mm thick polyurethane foam, complete exposure of the lid did not occur and the structural integrity of containment was not affected. The analyses applied stress concentration factors to the seal groove areas of the lids to show that there was no inelastic deformation during the HAC events and that seals remained compressed. HAC thermal analyses conservatively assumed exposure of the secondary lid near the O-rings due to puncture and showed that seal temperatures would remain below their allowable value. The staff reviewed and accepted the applicant's description of the containment system's response under HAC.

4.5.5 Leakage Rate Testing

Prior to leakage testing, the containment boundary is hydrostatically tested at an internal pressure that is at least 300 kPa, which is over 50% greater than the package's MNOP of 182.7 kPa. A fabrication leakage rate test of the entire containment boundary, including cask body, primary lid, secondary lid, and cover plate, is performed to a "leaktight" criterion [1 $10^{-7}$ ref-cm$^3$/sec (air)] using ANSI N14.5 Table A1, test A.5.3. Likewise, the primary lid, secondary lid, and cover plate inner O-rings are tested prior to initial use, periodically every 12 months, and after maintenance to a "leaktight" criterion using ANSI N14.5 Table A1, test A.5.3. The quick

disconnect valve is helium leakage tested prior to initial use, periodically every 12 months, and after maintenance using ANSI N14.5 Table A1, test A.5.3 such that there is no leakage detected when tested to a sensitivity of $10^{-3}$ ref-cm$^3$/sec. The applicant provided allowable leakage rates for different concentrations of helium tracer gas in Table 8.3-2 of the application.

A pressure-rise pre-shipment leakage test is performed per ANSI N14.5 Table A1, test A.5.2 such that there is no leakage detected when tested to a sensitivity of $10^{-3}$ ref-cm$^3$/sec. The components of the pre-shipment leakage test include the primary lid, secondary lid, and cover plate O-rings. All leakage test procedures are approved by ASNT NDT or COFREND Level III personnel certified in leakage testing. Likewise, testing is performed by ASNT NDT or COFREND Level II personnel certified in leakage testing. Leakage testing follows the ANSI N14.5-1997 standard.

The staff reviewed the leakage rate test descriptions and concludes that the leakage rate tests are to be performed in accordance with ANSI N14.5-1997.

### 4.6 Evaluation Findings

Based on a review of the containment 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 REVIEW

The purpose of the shielding evaluation is to verify that the Model No. RT-100 package shielding design provides adequate protection against direct radiation from its contents to ensure that the package design meets the external dose rate limit requirements of 10 CFR Part 71 under NCT and HAC.

### 5.1 Shielding Design Description

The authorized contents, contaminated resins and filters from nuclear power reactors, are mainly gamma emitters with negligible neutron radiations. The package is designed for exclusive use.

The packaging is comprised of concentric cylinders of stainless steel and lead. The outer and inner shells are 35 cm and 30 cm thick respectively, and the lead layer is 90 cm thick. The base of the cask consists of a 30 mm thick stainless steel bottom plate, a 75 mm thick gamma shield of poured lead, and a 50 mm thick stainless steel bottom forging. The primary lid is a 210 mm thick stainless steel forging. The secondary lid consists of a 100 mm thick stainless steel plate, a 60 mm thick lead gamma shield, and a 10 mm thick stainless steel plate. The secondary lid is attached to the primary lid with eighteen (18) hex head bolts. A cylindrical foam-filled impact limiter is installed on each end. The internal cavity dimensions are 1730 mm in diameter and 1956 mm in height. A secondary liner is required at all times for shipment to shore the content in the cavity. There is no drain port on the cask. Licensing drawings RT100 PE 1001-1, RT100 PE 1001-2, RT100 PRS 1011, and RT100 PRS 1031 provide details of the design, including the bill of materials and the dimensions of the packaging.

The contents are limited to contaminated resins and filters containing byproducts or otherwise radioactive nuclear materials, and do not include fissile materials in excess of quantities exempted from classification as fissile material per 10 CFR 71.15. The user shall use the procedure presented in Chapter 7 of the application to determine the maximum allowable

content. The activity of beta, gamma, and neutron emitting radionuclides shall not exceed the limits established in the shielding evaluation for each nuclide.

The applicant demonstrated through structural analyses that the package meets the regulatory requirements under the tests specified in 10 CFR 71.71 and 71.73 for packages under NCT and HAC. The applicant recognized the potential for lead to slump under HAC, and determined that there will be a 2.478 cm void gap at the end due to lead slump when the package is subjected to end drop and a crescent void on the side of the package when the package is subjected to side drop test. The crescent lead slump void gap is 2.3216 cm thick at the widest point of the crescent. Figure 5.3.1-5 in the application illustrates the assumed side lead slump of the package under HAC. The impact limiters are assumed to be completely lost under HAC conditions.

## 5.2 Radiation Source Specification

The radioactive nuclides in the authorized contents, i.e., resins and filters, produce primarily gamma radiations and, to a much lesser extent, neutrons from trace actinide spontaneous fission and alpha-n reactions in the media. Contents with significant neutron sources are not authorized.

The radiation sources from the authorized contents of the package are mainly gamma and beta radiations. Since the beta particles will be shielded by the steel and lead shields, no beta radiation is expected to contribute directly to the dose rates at the surfaces, and at two meters from the surface of the package under NCT except surface contamination. The secondary gammas from Bremsstrahlung reactions are considered in the shielding analyses by explicit MCNP calculations. The surface contaminations will be removed by decontamination procedures to acceptable limits as required by the operating procedures.

The applicant considered the secondary gamma radiations from the Bremsstrahlung reaction of high energy beta particles. Bremsstrahlung gamma radiation is evaluated by explicit mode e-p (electron-photon) transport calculation using MCNP5. Bremsstrahlung gammas from Y-90, Sb-124, La-140, and Ce-144 are evaluated in the calculations. Because of its typical high activity in resins and filters, Cs-137 is evaluated as well.

The Model No. RT-100 package is not designed to carry fissile material nor neutron sources (except typical small quantities consistent with contaminated resins and filters) and thus, it does not include features for shielding neutrons. Although the application contains some discussion on neutron shielding, the package is not designed for shipping contents with significant neutron emitters.

The form of the contents can be a combination of a wide variation of quantities of nuclides. To simplify the safety analyses and make the application flexible, the applicant chose to evaluate the shielding performance of the package, i.e., the dose rate, under a per Curie basis for each radionuclide with a maximum source concentration of 1 Curie per gram (Ci/g). The total dose rate is based on the total loaded activity, in Ci, of all radionuclides in the resin or filter media in the package.

## 5.3 Shielding Model Specification

The applicant performed shielding analyses using the MCNP computer code. The applicant modeled the package under NCT and HAC conditions, as prescribed in 10 CFR 71.71 and 71.73 respectively.

For the package under NCT, the package includes a shield thickness of 90 mm of lead and 70 mm of stainless steel including the thermal shield plate of 5 mm thickness and a shield thickness of 75 mm of lead and 80 mm of stainless steel at the bottom of the cask. The top end

includes a shield thickness of 210 mm of stainless steel for the primary lid and a shield thickness of 60 mm of lead and 110 mm of stainless steel for the secondary lid. Typical material densities are used in the shielding model for lead and steel.

A one millimeter (0.1 cm) gap is assumed between the lead layer and the stainless shells that hold the lead shell to account for the lead shrinkage during the lead solidification after the liquid lead is poured into the shell. The nominal thickness of the lead layer is 90 cm. The model assumed 85 cm of lead to account for the manufacturing tolerances. The acceptance criterion requires the space between the inner and outer shells that holds the lead layer to be measured thoroughly before the lead is poured to ensure the minimal thickness of the lead shell.

For the package under NCT, the impact limiters are modeled with two foam regions with densities bounding the lowest acceptable densities for the foam materials.

The design basis density of the content is 1.0 g/cm$^3$ with a uniform distribution, although the typical densities of the intended contents are around 0.65 g//cm$^3$. A sensitivity study shows that, in the range of 0.65 g/cm$^3$ to 1.0 g/cm$^3$, the dose rates decrease as the content density decreases. The 1.0 g/cm$^3$ is the design basis upper limit and bounds all contents with lower density. The effects of resin and filter density changes and redistribution of the content under NCT are studied by decreasing the volume occupied by the source term by 10% and a corresponding increase in source concentration. The results show no significant changes in the dose rates of interest. This is because the model increased the attenuation of the contents while increasing the source concentration when the content is compacted.

The applicant also developed an MCNP model for the package under HAC. The model assumes the loss of impact limiter together with a one inch puncture depth in the lead and lead slump to the bottom or to the side. Two shielding models are developed to address lead slump and the pin puncture. The first model includes an annular void at the top from the postulated end drop and also includes the 1 inch x 6 inch diameter indentation from the postulated pin puncture. The second model includes a crescent shaped void in the lead along the side of the outer shell from the postulated side drop.

5.4 Determination of the maximum allowable sources per package

A maximum quantity allowed to be transported in the package for each radionuclide is specified in the form of activity densities (Ci/g). The method used for calculating the maximum allowable activity density first calculates the average dose rates at the surface of the package and two meters from the projected surface of the vehicle under NCT for a pseudo particle with specific energy being emitted from the source uniformly distributed in the content, i.e., a dose rate per particle with a specific energy. The maximum allowable content for any source with this specific energy is determined by dividing the allowable dose rates by this dose rate per particle/energy ratio. This method made use of the intrinsic feature of the MCNP code for shielding calculations, i.e., the output of the calculations is the per particle dose rate at the selected detector location.

Dose rates for package containing single particle of energy of 0.3, 0.4, 0.5, 0.6, 0.7, 0.8, 0.9, 1.0, 1.1, 1.2, 1.3, 1.4, 1.5, 1.6, 1.7, 1.8, 1.9, 2.0, 2.5, 3.0, 3.5, 4.0, 4.5, 5.0, 5.5, and 6.0 MeV are calculated. The output of the calculations is the so-called “dose rate response” for a given particle at a given energy, e.g., a gamma at 1.0 MeV. The maximum number of particles at each energy level is determined by dividing the allowable dose rates by the dose rate response. A table is developed for the maximum allowable number of particles at each energy level for each case. The maximum allowable content in terms of curies for each radioactive isotope is determined by dividing the maximum allowable particles by the number of particles emitted by a Curie of activity of the said isotope. If a specific content contains radionuclides that emit

multiple particles per decay at different energies, such as Co-60, a fraction of the allowable activities for each particle must be determined based on the energy distribution and the branch fraction. Section 5.4.4.4 and Section 5.4.4.5 of the application presents the equations the applicant used to determine loading tables and the dose rates for bounding payloads.

Dose rate responses for package under both NCT and HAC have been calculated. Two loading tables have been developed and presented in Chapter 7, “Operating Procedures,” of the application, one for NCT and one for HAC. The user/shipper of the package can use this approach and these two loading tables to determine the maximum allowable contents for package under NCT and HAC conditions. The user/shipper must choose the lesser quantity of the content from these two determinations.

In addition, in determination of the maximum quantity of each nuclide, the dose rate response of the next higher energy of the same particle must be used for all the particles except the two gamma particles that are emitted by Co-60 decay. The applicant explicitly calculated the one-to-one dose rate per particle in the shielding calculations using the MCNP code. Instructions have also been developed to allow the user/shipper to use this table to determine the maximum allowable contents.

## 5.5 Shielding Evaluation

The applicant used the MCNP5 code version 4c for the shielding analysis of the Model No. RT-100 package. The Model No. RT-100 package is modeled with full three-dimensional details. The content region is modeled as a homogenized mass. Surface flux type tallies and point detector tallies are performed at the package surface, 1 m from the surface and at 2 m from the projected planes from the vehicle edge of the transportation trailer. The special feature of the MCNP5 mesh tally was used to facilitate the tallying of the neutron and gamma fluxes at the surfaces, 1 m and 2 m from the surface of the package. Dose rates are computed from surface tallies, multiplied by the ANSI/ANS-6.1.1-1977 flux-to-dose factors. The maximum dose rates are determined at key locations along the side of the package.

The applicant used an inverse approach to qualify the content’s limits. In essence, this approach determines the dose rate at a specific location outside the package from the radiation of a single particle emitted from the content using the MCNP code. The maximum allowable content for a specific nuclide is then calculated by dividing the total allowable dose rate by the dose rate from the single particle. The number of particles and their energy distributions together with the branching ratios for each specific nuclide are further formulated into the equation to develop a dose rates per Curie per cubic centimeter of each nuclide. The final result is a loading table that provides the maximum content in Curie for each nuclide. The user can then use the final results to determine the maximum allowable contents. For cases in which more than one radionuclide is in the content, the user must first determine the fractional allowable quantity for each nuclide and ensure that the sum of the dose rate fractions does not exceed 1.0.

To account for uncertainties other than assumptions and approximations included in the dose rate calculations, the applicant requires the users to round up the energies of the particles from the contents to the nearest energy line in the loading table. For example, if the energy of the particle is 1.21 MeV and the next evaluated energy line is 1.3 MeV, the user shall assume that the energy of the particles from the content is 1.3 MeV and determine the allowable content accordingly. Depending on the individual isotopes, this rounding up process may provide some additional safety margin for the shielding calculations, but this safety margin is not available for Co-60 sources which are explicitly modeled in the dose-rate response calculations.

The fundamental assumption of this method is that there exists a fixed one-to-one relationship between the dose rate and the particle type, energy, and location regardless of the medium the particle transverses. The staff noted that this approach may provide acceptable results if the material composition in the package is sufficiently similar to that used in the model for the dose rate response calculations. However, this one-to-one relationship is no longer valid if there is a significant change in material composition of the medium through which the particle transverses or the distribution of the sources. This is the fundamental flaw of this approach from both the fundamental particle transport theory and a practical application point of view.

External radiation levels estimated by the analysis depend on the source term energies and attenuation provided by the contents and packaging materials. Calculations were performed to show the relationship of the specified contents variables (content material composition and density) and their effect on radiation levels, but parametric analyses for the entire ranges of specified contents variables were not performed to identify a maximum radiation level.

In order to confirm the shielding calculations using the said method, the applicant performed “forward dose rate” calculations of the package to confirm the content determined in such a way is valid and the package satisfies the assumptions when dose rate response were calculated. The applicant performed shielding analyses for the potential content nuclides. Table 5.1.2-1of the application provides a summary for the dose rates of the maximum allowable quantities of these nuclides, respectively. The data presented in the table show that the results of the shielding analyses demonstrate that the design of the package with the maximum content meets the regulatory requirements. Table 5.4.4-1 of the application shows more details of the nuclides used in the calculations and the corresponding shielding evaluation results.

The composition and density of the resin and filter materials were represented in the analysis as the polystyrene at a density of 0.65 g/cm$^3$. The applicant studied the impact of material density with three other material compositions (nylon, Zeolite, charcoal) at 0.65 g/cm$^3$ to determine the effect of composition on the external radiation levels, and a range of polystyrene densities (0.35 to 1.00 g/cm$^3$) to determine the effect of material density on external radiation levels. The maximum radiation levels occurred for a carbon material at 0.65 g/cm$^3$ and polystyrene density of 1.00 g/cm$^3$. The maximum effect for the set of radionuclides in the evaluation was used to define a scaling factor that was applied to the determination of the limiting distributed source strength density for each of the radionuclides. The design of this parametric analysis is sufficient at this time to conclude with reasonable assurance that the maximum external radiation levels would occur for a charcoal media at 1.00 g/cm$^3$. However, calculation of radiation levels for the generic energies used to set the design limits for the content material type and form should have been performed at the optimum material composition and density.

The validity of the MCNP tallies is evaluated with respect to the fractional standard deviation (fsd) of the individual tally for each energy group in the source definition and ten statistical diagnostics provided in the output file. The MCNP code evaluates for each tally whether the ten statistical diagnostics pass the acceptance criteria. Following the guidance provided in the MCNP user manual, the applicant evaluated the statistical diagnostics that did not pass using the values provided in the tally function chart (TFC). The results suggest that the tallies at the bounding external radiation locations did pass the statistical diagnostic criteria and the fsd were also acceptable for the tallies at the defined source energies. Tallies for source energies below 0.5 MeV were not used to evaluate the external dose rates. All source energies below 0.5 MeV for the specified radionuclides were processed as if these were 0.5 MeV.

Although the evaluation of tallies and process used to calculate the external radiation levels is considered acceptable for such particular types of contents, the staff believes that the applicant

should have used variance reduction techniques and/or longer run times to ensure the results pass MCNP statistical diagnostics.

During its review, the staff performed confirmatory analyses. The staff discovered that the MCNP models and the calculation results presented in the safety analysis report did not converge properly for low- to mid-energy gamma shielding calculations. In general, this would be grounds for rejection of the application. However, the applicant took prompt corrective actions, recalculated the dose rate response per particle at the various energy levels, and updated the loading table that is to be used by the user/shipper to determine the maximum allowable content that meets the dose rate limits prescribed in 10 CFR 71.47 and 71.51.

The applicant states that the quantity of radioactive materials is not to exceed 3,000 $A_2$. As $A_2$ is typically used as a term for determining the structural category of the packaging and containment designs, a bounding content should be selected during the design of the package. The user of the package must ensure that the maximum content does not exceed the design assumption.

5.6 Evaluation Findings and Conclusion

The staff reviewed the description of the package design features related to shielding, the source terms, and the method and instructions for determining the contents. The staff reviewed also the shielding analyses, the assumptions and approximations used in the analyses as presented in the shielding safety analysis, and the results of the analysis, presented in the application, and the maximum dose rates for NCT and HAC to determine that the reported values were below the regulatory limit in 10 CFR 71.47 and 71.51 for an exclusive use package.

Based on its review of the statements and representations provided in the application, the staff determined with reasonable assurance that the shielding evaluation is consistent with the appropriate codes and standards for shielding analyses and NRC guidance, and that the shielding design of the Model No. RT-100 package, with the content's limits as determined from the instructions for determining allowable content and the loading table in Chapter 7 of the application, meets the regulatory requirements of 10 CFR Part 71 with the following conditions: (1) the maximum content density is 1.0 g/cm$^3$, (2) there shall be no neutron emitting nuclides, except for trace amounts; (3) the weight of water must be excluded when determining the Ci/g of content limits; and (4) the source must be uniformly distributed in the content with no concentration or shift during NCT.

References: X-5 Monte Carlo Team 2003, *MCNP—A General Monte Carlo N-Particle Transport Code, Version 5*, LA-UR-03-1987, Los Alamos National Laboratory, Los Alamos, N.M.

## 6.0 CRITICALITY REVIEW

The RT-100 package is not designed to carry fissile materials in excess of the exemption limits set forth in 10 CFR 71.15. No criticality safety evaluation is necessary.

## 7.0 PACKAGE OPERATIONS

Chapter 7.0 of the application provides a summary description of package operations, including package loading and unloading operations, to ensure that the package is operated in a safe and reliable manner under NCT and HAC. The preparation of an empty package for shipment is also described.

## 7.1 Package Loading

Section 7.1 of the application describes loading-related preparations, package loading sequences and inspections of the package, such as those performed to ensure the package is not damaged and that radiation and surface contamination levels are within allowable regulatory limits.

The applicant developed operating procedures for loading, transport, and unloading the RT-100 package. The most important step of the operating procedures is the determination of the maximum allowable quantity for the known content. Prior to loading, the shipper shall ensure that the contents meet the requirements of the loading table shown in Section 7.6.1.1 of the application and, in particular, a 1.0 g/cm$^3$ maximum density of the contents with the density ensured at any point of the contents. Dividing the total activity by the total weight to derive an average activity is not acceptable. The weight of water must also be excluded when determining the limit in Ci/g of the contents, and the shipper must first analyze the radioactive nuclides of the contents on a per-gram basis before determining the allowable content based on the loading tables. The shipper must ensure that the activity per gram, at any point of the content, does not exceed the limit that is determined according to the loading tables.

The loading table, completed by the shipper, includes several entries as follows: (i) actual content nuclides; (ii) maximum allowable activity concentration for the isotope being entered, in Ci/g, based on the methodology described in Section 5.4.4 and presented in Table 7.6.1-6 of the application; (iii) actual content activity, i.e., the maximum activity concentration of the isotope in Ci/g. This value should be the maximum for any waste stream; (iv) the percentage of the activity concentration for the isotope in question versus the maximum allowable activity concentration established by the methodology described in Section 5.4.4 of the application, (v) the total activity equivalent to 1 $A_2$ in curies for the isotope; (vi) the total $A_2$ quantity for the isotope; (vii) the amount of heat energy released per unit activity, in watts/Curie, for the isotope; and (viii) the total heat load contribution, in watts, for the activity concentration and mass entered by the user for the isotope.

In addition, the operating procedures instruct the shipper to perform a comprehensive measurement of the dose rates over the entire surface of the package to establish a profile prior to shipment and a comprehensive dose rate measurement at the arrival of the package. The shipper must stop shipments if the measured dose rates differ significantly from the pre-shipment measurement values because such evidence is an indication of source relocation during NCT, thus invalidating the assumption of the package shielding design.

Compliance with dose rate requirements is ensured by (i) determining the percent of each radionuclide activity relative to its maximum allowable value, and (ii) summing up this dose rate percentage for all radionuclides and assuring that the sum does not exceed 100%. In addition, the activity for the package shall be less than 3,000 $A_2$, and the total decay heat shall be below 200 watts. The shipper must calculate the $A_2$ value once the allowable contents are determined.

Users are required to ensure that flammable gas concentration within the package remains below 5% mole fraction during the shipping period. The rate of flammable gas generation is considered when evaluating the heat load, and the method for calculating the flammable gas generation is described in Section 4.4 of the application: a simplified model, described in Section 7.5.1 of the application, is used for most shipments, while an analytical model, described in Section 7.5.2, is used in more complex cases. Using the equations derived in Chapter 4, Section 4.4, the decay heat limit versus waste volume can be determined for a limit

of 5% of flammable gas in the cavity free volume. Figure 7.5-1 provides a curve illustrating the waste volume to decay heat value that would result in the generation of a flammable gas mixture within 10 days of shipment, assuming that all decay heat is absorbed by the waste material and the polyethylene container. The calculation assumes that the flammable gas generation occurs over a period of time that is twice the allowable shipping time. For most shipments, this simplified graphical model, i.e., loading curve, can be used to determine the maximum heat load.

Pre-shipment leakage rate tests are performed on the primary lid, secondary lid, and cover plate O-rings. The pre-shipment leakage rate tests follow ANSI N14.5-1997 and have a leakage rate criterion such that there is no leakage detected when tested to a sensitivity of 1E-3 ref $cm^3$/sec.

The package is only lifted in the vertical position and not via the lifting rings located on the upper impact limiter. Loading of the package can take place either on or off the trailer, and with or without the lower impact limiter attached. Figure 7.1.2-1 of the application describes the process flow for loading operations.

Prior to installation of the primary lid, the two seal surfaces (on the body of the package and on the primary lid) are inspected and cleaned and O-rings are checked to ensure they are not damaged. The lid is positioned by an aligning pin to ensure its proper placement and the bolts are tightened, using the “star pattern” method to ensure an evenly distributed pressure on the lid and package body, with an initial torque of 400 N-m and a final torque of 850 N-m. A pre-shipment leak test of the primary lid O-ring is then performed as described in Section 8.2.2.2 of the application. The same methodology is followed to install the secondary lid, except that the initial and final torques are different, i.e., 150 N-m and 350 N-m, respectively.

**Lid Bolt Tightening Torques**

| Description | Dimensions | Qty. | Tightening Torque [N-m] | Tolerance |
|---|---|---|---|---|
| Primary lid bolts | HHCS M48x170 | 32 | 850 | ± 10% |
| Secondary lid bolts | HHCS M36x120 | 18 | 350 | ± 10% |
| Quick disconnect valve disconnect valve cover plate bolts | SHCS M10x30 | 6 | 27 | ± 10% |

Prior to transport, a contamination survey is completed on the external surfaces of the package to confirm that non-fixed (removable) radioactive contamination is as low as reasonably achievable, and is within the limits specified in 49 CFR 173.443, as required by 10 CFR 71.87. If contamination is within limits, preparation for transport may be conducted.

Figure 7.1.3-1 describes the process flow for transport preparation, including installation of the upper impact limiter, trailer placarding and package labeling, installation of security seals, inspection of the exterior of the package for any damage prior to shipment, and verification that the package is correctly tied-down to the trailer.

## 7.2 Package Unloading

Unloading procedures include receipt of the package and removal of the contents. Upon receipt from the carrier, the package is visually inspected to verify there are no indications of impaired physical conditions. The tamper indicating seal is also inspected at the upper impact

limiter aligning pin and the shipment may be rejected by the consignee if the tamper seal has been removed or tampered with in any way. Comprehensive contamination and dose rate measurements are performed: any significant differences from the pre-shipment measurement values would be an indication of source relocation during NCT and would invalidate the assumptions for the shielding licensing of the packaging.

Package unloading is conducted in a vertical position. The upper impact limiter is first removed, prior to the removal of the quick-disconnect valve cover plate, and the interior and exterior pressures are balanced. Both the primary and secondary lids are removed in accordance with Sections 7.1.1.4 and 7.1.1.5 respectively. During removal of the contents, care should be taken not to damage the cavity of the package or the sealing surfaces.

### 7.3 Preparation of Empty Package for Transport

Section 7.3 of the application describes the operations used to certify that the empty package is safe for transport, i.e., verification that the cavity has been emptied of its contents; survey, followed by potential decontamination, of the lid, quick connect cover and interior surface of the packaging; installation of the primary, secondary lids and of the quick-disconnect valve cover; decontamination of the exterior surfaces of the packaging, as necessary; installation of the lower impact limiter in accordance with Section 7.4.3; and positioning of the packaging on the RT-100 trailer and installation of the upper impact limiter in accordance with Section 7.1.3.1 of the application.

### 7.4 Evaluation Findings

The staff reviewed the operating procedures in Chapter 7 of the application to verify that the package will be operated in a manner that is consistent with its design evaluation provided that: (i) the maximum density of the contents is 1.0 g/cm$^3$ ensured at any point of the content (determining an average density by dividing the total activity by the total weight is not acceptable), (ii) there are no neutron emitting nuclides except trace amounts, (iii) the allowable contents are determined based on dry resins or filters, i.e., the weight of the water must be excluded when determining the Ci/g limit of the contents, and the nuclear physical characteristics, i.e., the gamma attenuation coefficient of the content, must not be smaller than that of the resin, and (iv) a comprehensive dose rate measurement is performed before and after transport. On the basis of its evaluation, the staff concludes that the combination of the engineered safety features and the operating procedures provide adequate measures and reasonable assurance for safe operation of the package in accordance with 10 CFR Part 71.

## 8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Chapter 8 of the application identifies the inspections, acceptance tests and maintenance programs to be conducted on the Model No. RT-100 package and verifies their compliance with the requirements of 10 CFR Part 71.

### 8.1 Acceptance Tests

The weld maps of the packaging provide the examination criteria for each weld. Radiographic testing, dye penetrant testing, and/or visual testing are performed in accordance with applicable ASME standards.

Since the maximum normal operating pressure (MNOP) of the packaging is 182.7 kPa, i.e., above 35 kPa, the packaging is tested, prior to first use, at an internal pressure at least 150% of the MNOP to verify its ability to maintain its structural integrity at that pressure, in accordance with 10 CFR 71.85(b). For conservatism, the minimum test pressure is set to 300 kPa. The hydrostatic test pressure is held for a minimum of 10 minutes. Afterwards, the primary lid and secondary lid closures are examined for leakage. After depressurization and draining, the cask cavity and seal areas are visually inspected for cracks and deformation. Any cracks or deformation are remedied, and the test and inspection repeated.

Section 8.1.4 of the application describes the leakage tests to be performed on the Model No. RT-100 packaging, prior to its initial use. A fabrication leakage rate test of the entire containment boundary, including packaging body, primary lid, secondary lid, and cover plate is performed to a “leaktight” criterion using the technique described in ANSI N14.5 Table A1, test A.5.3. Likewise, the primary lid, secondary lid, and cover plate inner O-rings are tested prior to initial use, periodically every 12 months, and after maintenance to a “leaktight” criterion using ANSI N14.5 Table A1, test A.5.3. The quick disconnect valve is helium leakage tested prior to initial use, periodically every 12 months, and after maintenance using ANSI N14.5 Table A1, test A.5.3 such that there is no leakage detected when tested to a sensitivity of $10^{-3}$ ref-$cm^3$/sec.

Containment boundary welds meet the acceptance requirements of ASME Code, Section III, Division I, Subsection ND, Article ND-5000. The containment boundary welds are also inspected by radiographic examination. Non-destructive examinations are performed by ASNT or COFREND certified inspectors, based on the note 102885 EQN 001, Rev. C.

## 8.2 Maintenance

No routine or periodic structural testing is performed on the Model No. RT-100 packaging, except for the lifting fixtures, tested annually in accordance with ANSI N14.6 requirements to verify continuing compliance.

Wear and tear from normal use will not impact the safety of the package. However, the Model No. RT-100 packaging is subject to routine inspection and periodic maintenance to ensure its performance, as intended, throughout its service life. Visual inspections and measurements ensure that the packaging conforms to the dimensions and tolerances specified on the licensing and fabrication drawings and that its effectiveness is not significantly reduced.
In particular, the exterior surfaces of the packaging, the upper and lower impact limiters, the condition of the fusible plugs in the impact limiters, and the proper marking and labeling of the packaging are inspected.

All major components of the packaging, e.g., primary and secondary lids; quick disconnect valve cover plate, quick disconnect valve, leak test port plugs, primary lid, secondary lid, and vent port cover plate O-rings, and the visible exterior surface welds and interior cavity welds are visually inspected for defects. The primary lid, secondary lid and quick-disconnect valve cover plate sealing surfaces are cleaned during maintenance. New inner and outer containment boundary O-rings are installed every 12 months.

Every four years, at the minimum, testing is performed on the cask body lifting elements, i.e., lifting pockets. At the same time, examination of the inner shell visible welds parts on the packaging body is performed in addition to the periodic maintenance every twelve months.

Threaded inserts may be used to repair threaded bolts holes. At a minimum, each repaired bolt hole will be tested for proper installation by assembling the joint components where the insert is used and ensuring the bolt can be tightened to the required torque.

Leak testing of the Model No. RT-100 package is performed after completion of the annual inspection and after maintenance or repair. These tests are identical to those performed on the packaging prior to its initial use, i.e., primary and secondary lid containment O-rings helium leak testing, quick disconnect valve helium leak testing, quick disconnect valve cover plate containment O-rings helium leak testing. This test is conducted using helium leak detection in accordance with ANSI N14.5-1997, Table A1 test A.5.3, to demonstrate compliance with the ANSI N14.5 leaktight criteria. Calibration of the helium detector is performed using an appropriate standard, in accordance with Section 10 of ASTM E-499 or equivalent.

Cleanliness of sealing surfaces is of the highest priority during package disassembly for maintenance and assembly. This requirement particularly applies to O-rings and seal surfaces. O-rings are replaced within a 12 month period of use in accordance with Regulatory Guide 7.9 and leak-tested.

Section 8 of the application discusses acceptance tests and maintenance program. The various structural components are fabricated from ASTM standard materials in accordance with ASME Sections III, V, and IX. A summary of maintenance requirements is discussed in Section 8.1 of the application. The maintenance program includes periodic testing, inspection, and replacement schedules.

The staff finds that visual inspections at various timed intervals provide additional reasonable assurance against corrosion occurring unnoticed.

8.3 Evaluation findings

The applicant described its acceptance tests and the acceptance criteria for the Model No. RT-100 package. The applicant also described the required maintenance program of the package. The staff reviewed these descriptions and determined that two acceptance criteria were considered as important during the review of this package: (i) prior to lead pouring, the gap between the shells holding the lead must be measured to ensure a minimal clearance of 8.6 cm (the minimum thickness of the lead shield is 8.5 cm for a nominal thickness of 9.0 cm) because of the gap formed by lead shrinkage; (ii) the maximum gap between the lead and the stainless shells is not to exceed 0.1 cm (1 mm) because the safety of the package is directly impacted by the size of this gap.

Based on the statements and representations in the application, the staff has reasonable assurance that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71. Further, the certificate of compliance is conditioned to specify that each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application.

**CONDITIONS**

The following conditions are included in the Certificate of Compliance:

(a) A maximum total package neutron source of 3.5 $10^{-7}$ Ci/g for materials that produce neutrons, other than fissile materials, through any means. This condition was included because, in the absence of neutron shielding, this effectively reduces the neutron

contribution to external doses and loading table limits by a factor of 1,000 to 10,000. Thus, the user will load only gamma emitting sources with only trace amounts of neutron emitters typically found in resins and filters, which are the authorized contents of the package.

(b) The package shall be prepared for shipment and operated in accordance with Chapter 7 of the application.

(c) The package must be tested and maintained in accordance with Chapter 8 of the application.

(d) Shoring must be placed between the secondary container and the package cavity's walls to prevent both radial and axial movements during transport.

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

(f) A pre-shipment leak test is required for all shipments, i.e., not only shipments of Type B quantities. This condition is motivated by a potential false interpretation of regulations for contents such as Low Specific Activity (LSA) or Surface Contaminated Object (SCO) materials as being exempted from a pre-shipment leak test. If a user is shipping less than a Type B quantity and does not want to perform a leak test, this user has the option of using a different shipping name for that arrangement, of covering the Type B markings and making it as an IP-3 shipment. Therefore, the Type B certificate of compliance would not be used for that shipment. In reality, such users always want to call the Model No. RT-100 a Type B package, thus a leak test has to be required. Furthermore, there are cases of materials meeting either the LSA or SCO definition but with dose rates preventing such materials to be put in an IP package: one could consider assigning an LSA or SCO UN number for such materials and putting them in the Type B package. In that case, the markings should not be covered and the Type B certificate applies, thus requiring a leak test.

**CONCLUSION**

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

Issued with Certificate of Compliance No. 9365, Revision No. 0,
on March 18, 2014.