SAFETY EVALUATION REPORT

Docket No. 71-9365
Model No. RT-100
Certificate of Compliance No. 9365
Revision No. 1

SUMMARY

By letter dated January 30, 2015, as supplemented on March 5 and May 21, 2015, Robatel Technologies submitted an amendment request to revise the certificate of compliance (CoC) for the Model No. RT-100 package. The applicant revised shielding analyses, gas generation analyses as well as analyses associated with the lead shielding to address stresses associated with fabrication of the lead shielding and the performance of the lead shielding during hypothetical accident conditions (HAC), to allow the transport of material which has undergone "gross" dewatering, to allow additional leak testing methods, and to update operational procedures as appropriate. NRC staff reviewed the application using the guidance in NUREG-1609, "Standard Review Plan for Transportation Packages for Radioactive Material." Based on the statements and representations in the application, as supplemented, the staff agrees that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

1.1 Packaging Description

The applicant replaced statements in the safety analysis report (SAR) identifying how the packaging lifting pockets are rendered inoperable with a generic statement that the lifting pockets are rendered inoperable. Staff finds this change acceptable because the purpose of this SAR section is to provide the reader with an overall understanding of the package and not to provide specific detail on all of the operational characteristics of the package.

1.2 Content Description

The applicant revised the description of ion exchange resins in the SAR allowed for transport to include "grossly" dewatered resins, i.e., ion exchange resins which have essentially the same amount of water when presented for transport as when they were purchased from the manufacturer. Staff finds this change acceptable since it identifies the maximum amount of water associated with the contents presented for transport.

1.3 Findings

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

2.0 STRUCTURAL

2.1 Description of Structural Design

The major components of the RT-100 package are the 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. None of these components have changed as a result of this amendment. This portion of the safety evaluation report is concerned only with the structural integrity of the lead shielding as it pertains to this amendment.

2.2 Discussion

The applicant revised their original lead slump calculations and also calculated stresses induced during the manufacture of the lead shielding. No changes occurred to the lead shielding with regards to physical geometry or material property. The staff limited the scope of the structural review to the areas of the SAR pertaining to normal conditions of transport (NCT) and HAC and their relationship to stresses induced by lead pouring during the manufacturing process. Lead slump under HAC conditions was also examined since HAC is the bounding case.

2.3 Normal Conditions of Transport

The applicant calculated stresses due to lead shrinkage on the inner canister shell. The applicant determined that the pressure on the inner shell induced by the lead shrinkage as a result of cooling after lead pouring is 0.087 MPa. The resulting stresses were used as an initial condition for the NCT analyses and are small compared to the NCT allowable stresses. The applicant reported in Table 2.6.7-2 that the total stress as 10.5 MPa. This value is very small compared to the allowable stress of 138 MPa and produces a large margin of safety (12.1 minimum).

The staff determined that the applicant considered manufacturing stresses due to lead shrinkage as per Regulatory Guide 7.6. Given the large margin of safety, the staff concluded that fabrication stresses due to lead pouring under NCT will not produce a substantial reduction in the effectiveness of the lead shielding and satisfy the requirements of 10 CFR 71.71.

2.4 Hypothetical Accident Conditions

The applicant calculated stresses due to lead shrinkage on the inner canister shell. The applicant determined that the pressure on the inner shell induced by the lead shrinkage as a result of cooling after lead pouring is 0.087 MPa. The resulting stresses were used as an initial condition for the HAC analyses and are small compared to the HAC allowable stresses. The applicant reported in Table 2.7.1-2 the total stress as 38.4 MPa. This value is very small compared to the allowable stress of 331 MPa and produces an adequate margin of safety (6.7 minimum).

The staff determined that the applicant considered manufacturing stresses due to lead shrinkage as per Regulatory Guide 7.6. Given the adequate margin of safety, the staff concluded that fabrication stresses due to lead pouring under HAC of transport will not result in a substantial reduction in the effectiveness of the lead shielding and satisfy the requirements of 10 CFR 71.73.

2.5 Lead Slump Evaluation

In the original application, the applicant treated the lead as a solid and calculated the lead slump, i.e., deformation, due to HAC end drop to be 1.6 mm. Since the staff questioned the lead slump results in the original application, the applicant calculated lead slump under end drop conditions using a more conservative approach. The updated calculation assumed the lead behaved as a liquid and calculated the volumetric change of the lead at -40^{o}C. This more conservative approach calculated the lead slump to be 4.8 mm. This value is utilized in the current HAC shielding conditions.

The staff reviewed the applicant's analysis of lead slump. This phenomena occurs during end drop conditions cited in 10 CFR 71.73. Staff determined that the applicant's lead slump analysis is conservative since a value larger than what was determined by analysis was cited.

2.6 Findings

Based on a review of the statements and representations in the application, the staff concludes that the lead shielding has adequate structural integrity to meet the requirements of 10 CFR Part 71.

3.0 CONTAINMENT EVALUATION

The objective of the review was to verify that the Model No. RT-100 package containment design was adequately described and evaluated under normal conditions of transport and hypothetical accident conditions, as required per 10 CFR Part 71. There were no changes to the containment boundary as part of this certificate of compliance revision. Rather, in regards to containment issues, the applicant's revised safety analysis report reflected the inclusion of grossly dewatered resins as content, provided clarifications associated with leakage rate testing, and specified that shipments are allowed only if conditions in Table 7.5.2-1 and Table 4.4.5-1 are met. Regulations applicable to the containment review include 10 CFR 71.31, 71.33, 71.35, 71.43, and 71.51.

3.1 General Considerations

3.1.1 Combustible Gas Generation

The applicant slightly changed the containment chapter that discussed the analyses necessary to limit the generation of flammable gas to less than 5% mole fraction. In the initial application, the applicant limited the dewatered resin's amount of free water after mechanical draining to approximately 1%. However, the applicant's previous analyses assumed that 25.75% of the waste volume was free water; this was a sizeable margin above the applicant's limit of 1%. In the current application, the grossly dewatered resin was limited to a free water amount that is 20% of the ionic resin volume. Staff determined that the applicant's previous flammable gas generation analysis remains acceptable because grossly dewatered resins limited to a 20% free water percentage are bounded by the 25.75% free water percentage in the applicant's previous analysis.

3.1.2 Leakage Rate Testing

The applicant updated the safety analysis report to clarify items associated with leakage testing, including the following: measuring base metal temperature as a parameter to determine the duration of the leakage test (either pressure rise or pressure drop), specifying the volume of the interspaces between the O-ring seals, and specifying the pre-shipment leakage test sensitivity. Staff reviewed these changes and determined that, for this package, they were consistent with the guidance presented in ANSI N14.5-1997.

3.2 Findings

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

4.0 SHIELDING EVALUATION

The staff reviewed the application using the criteria listed in NUREG-1609, “Standard Review Plan for Transportation Packages for Radioactive Material,” to verify that the changes made to the package by this revision provide adequate protection against radiation and meet the external radiation requirements of 10 CFR Part 71 under NCT and HAC. The applicant revised both the source configuration and shielding configuration descriptions in the RT-100 SAR and provided revised MCNP results.

4.1 Shielding Design Description

The staff reviewed Chapter 1 (general information), Chapter 5 (design information), and Chapter 7 (package operation) in the SAR, as well as both the CoC and the model drawings, provided with the application. The staff also considered relevant information in the application attachments. The RT-100 package is designed, and constructed, in accordance with 10 CFR 71.71 so that the maximum external dose rates do not exceed the exclusive use limits in 10 CFR 71.47. The RT-100 package is designed for a Type B quantity of radioactive material in the form of contaminated resins and filters from nuclear power reactors.

The package is designed for transport on an open (flat-bed) vehicle. The packaging is fabricated from concentric cylindrical stainless steel shells. The outer and inner shells are 35 mm and 30 mm thick respectively. A 90 cm thick layer of lead fills the annulus between the stainless steel shells. The base of the cask consists of a 30 mm thick stainless steel outer bottom plate, a 75 mm thick gamma shield of poured lead, and a 50 mm thick stainless steel inner bottom plate. The package has two lids. 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. Both the primary and secondary lids are bolted to the cask body. Since the package is not designed to ship contents with significant neutron source terms, neutron shielding materials are not utilized in the packaging design. A foam filled cylindrical impact limiter is installed on each end. The internal cavity dimensions are 1730 mm in diameter and 1956 mm in height. A secondary container is required for all shipments to shore the content in the cavity.

4.2 Radiation Source Specification

Authorized contents of the RT-100 are contaminated resins and filters from nuclear power plants. The radionuclides in the contaminated resins and filters emit primarily gamma and beta radiation. Since the packaging shield material is steel and lead, betas do not contribute to the external dose, but secondary gamma radiation from Bremsstrahlung reactions of high energy beta particles from Y-90, Sb-124, La-140, and Ce-144 with the steel and lead contribute to the packaging external dose rates. Neither radionuclides which undergo either spontaneous fission or alpha-n reactions nor fissile materials are authorized for transport in the RT-100 except in small quantities consistent with contaminated resins and filters. Although beta radiation from the contents does not contribute to external package dose rates, beta emitting radionuclides can contaminate external packaging surfaces. The activity of beta, gamma, and neutron emitting radionuclides shall not exceed the limits established in the shielding evaluation for each nuclide.

4.3 Shielding Model Specification

The applicant used MCNP6, a Monte Carlo transport code that offers a full three-dimensional combinatorial geometry modeling capability, to perform shielding analyses of the RT-100 package. MCNP6 allows the user to track gamma radiation emitted by the contents. Bremsstrahlung radiation generated by high energy beta particles from Y-90, Sb-124, La-140, and Ce-144 interacting with the steel and lead are evaluated by explicit electron-photon transport calculation techniques within MCNP6. The applicant modeled the package under NCT and HAC scenarios, as prescribed in 10 CFR 71.71 and 71.73 respectively. The applicant employed the mesh based weight windows approach variance reduction technique. The weight windows cards used in the MCNP6 input file includes the WWG and MESH for generating mesh based weight window files, and WWP for calling in the generated weight window mesh file. The weight windows estimated by the MCNP6 WWG card are subject to statistical fluctuations; thus, some manual refining of the generated weight window mesh may be necessary. A cylindrical grid of cells independent of the MCNP6 geometry that extends slightly beyond the boundaries of the model was used to define this mesh. This cylindrical spatial mesh is superimposed over the geometry with the MESH card. 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.

External radiation levels depend on the source term energies and attenuation provided by the contents and packaging materials. Since the contents can be a wide range of radionuclides, the applicant evaluated the packaging shielding on a per curie basis for each radionuclide with a maximum source specific activity of 1Ci/g. Although resins and filters typically have a density of 0.65 g/cm^3, the applicant used a density of 1.0 g/cm^3. Calculations provided by the applicant showed 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. Thus, 1.0 g/cm^3 was used as the design basis upper limit because it bounds all lower density contents. Since the contents will be non-uniformly distributed after loading, the applicant uniformly distributed the maximum specific activity source throughout the cask cavity for both the NCT and HAC shielding models.

For NCT and HAC shielding evaluations of the package, thicknesses for the steel shells, steel lids, and lead shielding are reduced by subtracting the manufacturing tolerance from the nominal dimensions. Using reduced thicknesses in the model bounds the shielding material variations caused by the fabrication processes that may impact the external dose rates. Neutron shielding features are not considered in the MCNP modeling because neither

radionuclides with significant neutron sources nor fissile materials are authorized for transport in the RT-100 except in small quantities consistent with contaminated resins and filters.

In its previous review, staff noted that the calculation results presented for the MCNP models had not converged and suggested the applicant perform the analyses again using either variance reduction techniques, longer run times or both to ensure the analytical results pass the MCNP statistical diagnostic checks. The applicant followed this suggestion and used a variance reduction technique in their MCNP modeling. In Revision 5 of the SAR, the staff discovered that 17 of 72 MCNP analysis outputs had not converged. The MCNP tallies are binned by energy, and each energy bin is divided up into multiple segment bins. For each energy bin of a given tally, only the one segment at which the maximum dose rate is calculated is used for the specific activity limit calculation. After analyzing all segment bins to make sure that the maximum segment bin is selected for the respective energy bin, and that the shape of the flux along the tally surface seems appropriate, the segment bins that aren't used are essentially discarded. The applicant indicated that tally segment bins with a fractional standard deviation greater than 0.1 are far from the maximum segment, with a calculated dose rate that is orders of magnitude less than the value reported at the maximum segment. All 17 of the tally segment bins with a fractional standard deviation greater than the MCNP criterion of 0.1 were discarded due to not being the maximum-recorded tally segment bin.

4.4 Dose Rate Response Calculation

The applicant estimated dose rate responses for most radionuclides of interest using a generic energy line method. The generic energy line dose rate responses were calculated for photons ranging in energy from 0.5 MeV to 8 MeV by multiplying the dose rate per curie per radioactive decay calculated by MCNP6 for a particle with energy E, the total number of particles emitted at energy E and the associated branching ratios (i.e., percentage of particles emitted at energy E). The resulting products were summed to calculate a dose response function for each specific nuclide. In using the generic energy line method, the applicant used specific particle energies. If a radionuclide emitted particles which did not match these specific energies, the particles emitted by the radionuclide were rounded up to the next highest energy. For example, if the energy line closest to particles of 1.15 MeV and1.21 MeV is 1.3 MeV, it is assumed that the energy of the particles emitted by the radionuclide are 1.3 MeV. This rounding up may provide some safety margin by accounting for uncertainties associated with assumptions and approximations included in the dose rate calculations.

The fundamental assumption of this method is that a fixed one-to-one relationship exists between the dose rate and the particle type, particle energy, and location regardless of the medium the particle traverses. This approach provides acceptable results only if the material composition in the package is sufficiently similar to that used in the model for the dose rate response calculations; otherwise, the model is no longer valid. Dose rate responses were also generated for eight individual nuclides of interest in a manner similar to that described above. However, the safety margin associated with the generic energy line method is not available for eight radionuclides which are explicitly modeled in the dose-rate response calculations. The applicant employed the dose rate response at the various energy levels to update the loading table that is to be used by the package user to determine the maximum allowable content that meets the dose rate limits prescribed in 10 CFR 71.47 and 71.51.

4.5 Dose Rate Results

The total dose rate is evaluated based upon the total activity in curies of all radionuclides in the resin or filter media loaded into the package. Table 5.1.2-1 summarizes the dose rates for NCT and HAC. Table 5.4.4-5 and -6 of the application shows all the nuclides used in the calculations and the corresponding shielding evaluation results for NCT and HAC. The data shows the package meets the regulatory requirements with the maximum content.

Using the dose rate response functions discussed above, the applicant employed an inverse approach to determine the content limits. This approach divided the regulatory limit dose rate at a specific location outside the package by the dose rate response described previously. The final result is a loading table that provides the maximum content in curies for each nuclide. 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.

The applicant specified that the quantity of radioactive materials is not to exceed 3,000 A_2. The package user must ensure that the maximum content does not exceed 3,000 A_2. The shipper shall use the procedures presented in Chapter 7 of the application to determine the maximum allowable content. The contents of a liner are shipper responsibility prior to cask loading. The shipper must determine the maximum specific activity for any nuclides used in the loading table, and therefore considering it as the source strength density throughout the entire contents. Decontamination procedures should be employed to reduce contamination levels to acceptable limits.

4.6 Findings

The staff reviewed the description of the package design features related to shielding, the source terms, the analytical methods, and instructions for determining the contents presented for transport. The staff also reviewed the shielding analyses, the assumptions, and approximations used in the analyses as well as the maximum dose rates for NCT and HAC presented in the application. Based on its review of the statements and results provided in the application, the staff determined 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 conditions stated in the CoC.

5.0 PACKAGE OPERATIONS

5.1 Evaluation

The applicant proposes to revise the operating instructions for the following reasons:

- to allow the use of both pressure rise and pressure drop leak tests,
- to revise lifting equipment terminology,
- to update the maximum allowable activity concentration based upon the revised shielding evaluation, and
- to add information, in the form of notes and tables, which will aid users in loading the package.

The staff reviewed the proposed changes and determined that they ensure the package will satisfy the requirements in 10 CFR Part 71 during operation.

5.2 Findings

Based on a review of the statements and representations in the application, the staff concludes that the operating procedures meet the requirements of 10 CFR Part 71 and that these procedures are adequate to assure the package will be operated in a manner consistent with its evaluation for approval.

6.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM REVIEW

6.1 Evaluation

In addition, the applicant proposes to modify the pre-shipment leak test instructions as follows:

- to clarify testing organization references,
- to add both notes and steps to the pre-shipment leak test procedure,
- to update the minimum test duration time period,
- to update leak test acceptance criteria, and
- to allow the use of the gas pressure drop method.

The staff reviewed the proposed changes and determined that they will ensure the package is maintained in a manner which will allow the package to satisfy the requirements of 10 CFR Part 71.

6.2 Findings

Based on review of the statements and representations in the application, the staff concludes that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71, and that the maintenance program is adequate to assure packaging performance during its service life.

CONDITIONS

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

Condition 5(b)(1) was revised to incorporate grossly dewatered resins as allowable contents.

The references section has been updated to include this request.

Minor editorial corrections were made.

CONCLUSIONS

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

Issued with Certificate of Compliance No. 9365, Revision No. 1
on July 27, 2015.