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

SAFETY EVALUATION REPORT
Docket No. 71-9390
Model No. OPTIMUS® - L Package
Certificate of Compliance No. 9390
Revision No. 0

SUMMARY

By letter dated September 9, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20266G182), NAC International (NAC or the applicant) submitted an application for certificate of Compliance (CoC) No. 9390 for the Model No. OPTIMUS®-L package. The application was supplemented on February 23, 2021 (ADAMS Accession No. ML13107B541), June 15, 2021 (ADAMS Accession No. ML21179A640), and September 2, 2021 (ADAMS Accession No. ML21251A471). The applicant submitted a revised application “The OPTImal Modular Universal Shipping Cask Safety Analysis Report, Revision No. 0”, dated November 12, 2021 (ADAMS Accession No. ML21316A180) which supersedes all previous submittals.

The OPTImal Modular Universal Shipping Cask for Low activity contents (OPTIMUS®-L) package is transported by truck in a vertical orientation under exclusive use. The packaging consists of a Cask Containment Vessel (CCV), a CCV bottom support plate, and an Outer Packaging (OP) assembly. A Shield Insert Assembly (SIA) may be included inside the CCV for contents that require additional shielding.

The packaging contents include (i) byproducts, sources, special nuclear materials either as special form or non-special form, (ii) solid or solidified transuranic-containing wastes (TRU), fissile, non-fissile, or fissile-excepted; (iii) neutron activated metals or metal oxides in solid form; (iv) radioactive solid waste materials, including special form material; (v) irradiated fuel waste (IFW) consisting of low-enriched uranium (LEU) fuel and activated metal structural components. All radioactive contents are packaged in secondary containers which are used to prevent direct contact between the contents and the packaging in order to minimize the spread of contamination and to facilitate content loading and unloading operations. Secondary containers do not have a containment function.

The applicant proposed inerting the package with helium gas for TRU and IFW waste with a decay heat exceeding 50 watts. Staff reviewed the information provided by the applicant and determined that the applicant only shows a venting mechanism to allow gases to flow in or out of each confinement region of the contents but does not show its effectiveness in operation and does not demonstrate that the inert fill gas either “effectively” occupies the cavity of the cask containment vessel (CCV) or is in uniform concentration through the CCV. In order to allow inerting, the staff requires, at a minimum, a full demonstration that the inerting process will prevent the development of flammable gas mixtures in any confined area of the package throughout the entire shipment period, and a detailed evaluation to prove that there are no flammable gas mixtures (considering the worst-case concentrations) during shipment. In addition, the applicant needs to explain how the inerting gas is effectively introduced to all confined areas within the containment system of the package and, as such, is in uniform concentration throughout the CCV, Finally, the concentrations of combustible gases need to be able to be quantitatively analyzed. Therefore, the staff determined it could not reach a finding of reasonable assurance in order to accept inerting for the OPTIMUS®-L package. As a consequence, the CoC includes a condition preventing inerting and limiting waste, authorized for shipment, with a decay heat below 50 watts.

The staff notes that the application erroneously associates at times contents with A_2 , e.g., “The package which is designed to transport normal form content with a maximum activity greater than 3,000 A_2 and greater than 30,000 Ci”. Staff noted it is not appropriate to define contents in terms of A_2 value since A_2 is not an appropriate unit to define a source term and is intended solely for: (i) material selection purpose linked to the structural robustness of the package, and (ii) control and verification of leak rates only, as stated in 10 CFR 71.51(a)(1) and (a)(2). See also RIS 2013-04 “Content Specification and Shielding Evaluations for Type B Transportation Packages”.

Based on the statements and representations in the application, and the conditions listed in the CoC, the U.S. Nuclear Regulatory Commission staff (the staff) concludes that the package meets the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71.

EVALUATION

1.0 GENERAL INFORMATION

The OPTIMUS®-L packaging consists of a Cask Containment Vessel (CCV), a CCV bottom support plate, an Outer Packaging (OP) assembly, and Shield Insert Assemblies (SIAs). The CCV bottom support plate is a free-standing coated carbon steel plate positioned at the bottom end of the CCV cavity below the contents. The CCV fits within the cavity of the OP. The packaging may also be configured with a Shield Insert Assembly (SIA) within the cavity of the CCV. However, the CCV bottom support plate is not used with the 1-inch SIA.

The CCV is the packaging containment system. It is a stainless-steel cylindrical vessel that includes a body weldment, bolted lid, bolted port cover, and elastomeric O ring seals. The CCV has an outer diameter of 34.5 inches, which expands to 39.0 inches at the bolt flange and lid, and an overall height of approximately 51.4 inches. The internal cavity of the CCV has a 32.5 inches diameter and is 47.0 inches high. The CCV cylindrical shell and bottom plate have the same thickness. The CCV lid is fastened to the CCV body by 1-inch diameter socket head cap screws (e.g., CCV closure bolts).

The SIA is a coated carbon steel container that is placed inside the CCV cavity to provide additional gamma shielding. The SIA configurations used in the OPTIMUS®-L packaging are either 1-inch or 2¼-inch thick. The internal cavity of the SIA has a diameter of 24.0 inches and is 35.25 inches high. The 1-inch thick SIA is used inside the CCV cavity. The 2¼-inch thick SIA used in the CCV cavity requires an annular spacer plate placed underneath the bottom of the 2¼-inch thick SIA to position it near the top of the CCV cavity to facilitate loading operations.

The packaging is constructed and assembled in accordance with the following NAC Drawing Nos.:

70000.14-502, Rev. 1	Packaging Assembly – OPTIMUS-L
70000.14-510, Rev. 6	CCV Assembly - OPTIMUS
70000.14-511, Rev. 9	CCV Body Weldment - OPTIMUS
70000.14-512, Rev. 8	CCV Lid - OPTIMUS

70000.14-513, Rev. 3	Port Cover - OPTIMUS
70000.14-514, Rev. 2	CCV Bottom Support Plate – OPTIMUS-L
70000.14-540, Rev. 1	Outer Packaging Assembly – OPTIMUS-L
70000.14-541, Rev. 5	Outer Packaging Base – OPTIMUS-L
70000.14-542, Rev. 4	Outer Packaging Lid – OPTIMUS-L
70000.14-550, Rev. 4	1-inch Shield Insert Assembly (SIA) - OPTIMUS
70000.14-551, Rev. 5	2 ¼ -Inch Shield Insert Assembly (SIA) - OPTIMUS
70000.14-553, Rev. 2	2 ¼ -Inch SIA Annular Spacer Plate – OPTIMUS-L

The packaging contents include the following: (i) Byproduct, source, special nuclear material, non-fissile or fissile-excepted, as special form or non-special form in the form of process solids or resins; (ii) Dewatered, solid, or solidified transuranic-containing wastes (TRU), fissile, non-fissile, or fissile-excepted; (iii) Neutron activated metals or metal oxides in solid form, in secondary containers; (iv) Miscellaneous radioactive solid waste materials, including special form material; (v) Irradiated fuel waste (IFW) consisting of low-enriched uranium (LEU) fuel and activated metal structural components (e.g., cladding, liners, baskets, etc.). All radioactive contents shall be packaged in secondary container(s) (e.g., drums, liners, specialty bags, etc.).

Fissile contents must not exceed the fissile gram equivalents (FGE) in Table 1-1 below for the specified criticality configuration limits. Plutonium contents in quantities greater than 0.74 TBq (20 Ci) must be in solid form.

Table 1 - TRU Waste FGE Limits

Config. ID	FGE Criticality Configuration Description			FGE Limit (1) ^{239}Pu (^{235}U)
	Machine Compacted(2)	Weight % Special Reflector(3)	Minimum ^{240}Pu Credit	
FGE-1		≤ 1		340 (528) g
FGE-2a		≤ 1	≥ 5 g	350 (544) g
FGE-2b		≤ 1	≥ 15 g	375 (583) g
FGE-2c		≤ 1	≥ 25 g	395 (614) g
FGE-3		> 1		121 (188) g
FGE-5	x	≤ 1		250 (388) g

(1) FGE conversion based on a ratio of subcritical mass limits in ANSI/ANS-8.1, Section 5.2 of 0.7 kg (1.5 lb) for ^{235}U and 0.45 kg (1.0 lb) for ^{239}Pu. FGE equivalents determined using Table 7-1 of the application.

(2) For uncompacted or manually compacted TRU waste, materials with hydrogen density up to that of water (0.1117 g/cm^3) are unlimited, but materials with hydrogen density greater than water are limited to the hydrogen density of polyethylene (0.1336 g/cm^3) and may not exceed 15% of the total contents by volume. For machine compacted contents, hydrogenous materials in the contents are limited to the hydrogen density of polyethylene (0.1336 g/cm^3) in an unlimited quantity.

(3) Special reflector materials are defined as beryllium, beryllium oxide, carbon (graphite), heavy water, magnesium oxide, and depleted uranium. The weight% of the special reflector materials is calculated as the mass of all special reflector materials present divided by the total mass of all waste material contents inside the secondary container. For FGE-3, these materials are unlimited.

IFW contents shall not exceed the Fissile Equivalent Mass (FEM) limits from Table 2 for the specified criticality configuration limits.

Table 2 - IFW Waste FEM Limits

Config. ID(1)	LEU Waste Criticality Configuration Description		
	Weight % Special Reflector (2)	Enrichment Limit, (wt.% ^{235}U)	Uranium Mass Limit, lbs. (kg)
FEM-1	≤ 1	≤ 0.90 wt.%	2500 (1134)

(1) IFW contents must be non-machine compacted. Materials with hydrogen density up to that of water (0.1117 g/cm^3) are unlimited, but materials with hydrogen density greater than water are limited to the hydrogen density of polyethylene (0.1336 g/cm^3) and may not exceed 15% of the total contents by volume.
(2) Special reflector materials are defined as beryllium, beryllium oxide, carbon (graphite), heavy water, magnesium oxide, and depleted uranium. The weight% of the special reflector materials is calculated as the mass of all special reflector materials present divided by the total mass of all waste material contents inside the secondary container.

Contents shall not exceed the maximum activity limits in Tables 7.5-1 and 7.5-2 of the application using the procedure described in Attachment 7.5-1 of the application. Table 7.5-2 of the application applies to packages centered on the trailer and correspond to loadings with either no SIA, 1-inch SIA, or 2 ¼ inch SIA. Sample maximum loadings for key nuclides are in Table 3 for a package centered on the trailer.

Table 3 - TRU Waste and IFW Activity Limits for Key Isotopes

Isotope	Activity Limits (Ci) per package configuration		
	No SIA inside the CCV cavity	1-inch SIA inside the CCV cavity	2¼-inch SIA inside the CCV cavity
Co-60	8.197×10^{-2}	1.632×10^{-1}	4.284×10^{-1}
Cs-137	2.527×10^{2}	1.299×10^{3}	9.245×10^{3}
Ba-137m	3.846×10^{-1}	1.018	3.995
Cf-252	1.217×10^{-2}	1.289×10^{-2}	1.469×10^{-2}
Cm-244	3.819×10^{2}	4.074×10^{2}	4.654×10^{2}

The isotope inventory from TRU wastes that would be loaded in any particular package is variable. To comply with regulatory requirements on content specifications and dose rates, the applicant has loading requirements to determine the amount of each isotope and then sum the dose rate contribution from each nuclide. The sum cannot exceed 90% of the regulatory limit for every regulated location. This procedure is in Attachment 7.5-1 of the application and discussed in Section 5.4.2 of this SER.

The nominal weight of the empty packaging is approximately 6,050 pounds. The maximum weight of contents is approximately 3,500 pounds including radioactive waste, secondary containers, internal structures (e.g., CCV bottom support plate, SIA) and dunnage or shoring.

The maximum allowed decay heat is 50 watts. Shoring must be placed between loose fitting contents and the CCV cavity to prevent excessive movement during transport.

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

2.0 STRUCTURAL AND MATERIALS EVALUATION

2.`1 STRUCTURAL EVALUATION

2.1.1 Description of Structural Design

The OPTIMUS®-L is a Type B(U)F-96 transportation package per 10 CFR 71.4. The package consists of a Cask Containment Vessel (CCV), a CCV bottom support plate, and an Outer Packaging (OP) assembly.

The CCV is a stainless-steel vessel with a bolted closure designed to provide leak tight containment in accordance with the criterion of American National Standards Institute (ANSI) N14.5-2014.

The OP is made up of a base and lid bolted together to fully encase the CCV. The OP is designed to crush and absorb the impact energy when subjected to NCT (Normal Condition of Transport) free drop and HAC (Hypothetical Accident conditions) free drop tests, thereby limiting the loads imparted to the CCV. The OP also insulates the CCV from the direct effects of a fire during the HAC thermal test.

A Shield Insert Assembly (SIA) consisting of 1 inch and 2 1/4-inch thicknesses may be included inside the CCV for additional shielding.

The applicant provided licensing drawings with tolerances, dimensions, welding symbology, and definitions, material designation, and associated standards. Component descriptions and the arrangement of components relative to each other has been described and detailed by the applicant. In the application, the applicant describes the weight of the package with, and without its contents in Table 2.1-8 of the application. The overall physical dimensions of the package are shown in the listed drawings in the application in Appendix 1.3.3. The design bases maximum normal operating pressure of the package is 100 psi. The package is designed to be lifted vertically using a 3-legged bridle connected to the three lifting lugs located on the OP lid.

The staff has reviewed the package structural design description and concludes that the contents of the application satisfy the requirements of 10 CFR 71.31(a)(1)(c), 10 CFR 71.31(a)(2), 10 CFR 71.33(a), 10 CFR 71.35(a) and 10 CFR 71.33(b).

Identification of Codes and Standards for Package Design

The material standards used for the package comply with American Society for Testing and Materials (ASTM) and American Society of Mechanical Engineers (ASME) Section II, Part D, for the package. For simulation analyses, the applicant used LS-DYNA R5.1.1 (2012) and used ANSYS 19.0 to perform structural analyses. The fatigue analysis of the CCV and port cover closure bolts was conducted in accordance with ASME Section III, NB-3222.4, and NB-3232.3. NUREG/CR-6007 was used to analyze bolt stresses of the package under NCT and HAC. The applicant designed the lifting attachments of the OPTIMUS®-L package in accordance with the requirements of ANSI N14.6 for special lifting devices for critical lifts. The staff reviewed the proposed codes and standards and concluded that they are appropriate for the intended purpose and are properly applied.

Material Behavior

The package's containment boundary undergoes inelastic deformation when subjected to drop tests for both NCT and HAC. The applicant evaluated the stress in the CCV using a 3-D ANSYS finite element model which describes and characterizes the criteria used for elastic and inelastic stress in Section 2.6.7 of the application. The allowable elastic and inelastic buckling stresses for NCT and HAC are calculated in accordance with the formulas given in Section - 1713.1.1 and Section - 1713.2.1 of ASME Code Case N-284-1. The allowable buckling stresses include factors of safety of 2.0 for NCT and 1.34 for HAC in accordance with Section - 1400 of ASME Code Case N-284-1.

The staff has reviewed the structural codes and standards used in the package design and the post yield material behavior and finds that they meet the requirements of 10 CFR 71.31(c).

2.1.2 General Requirements

Minimum Package Size

The minimum package dimension is greater than 10 cm; thus, the staff finds that the package satisfies the requirements of 10 CFR 71.43(a) for minimum size.

Tamper-Indicating Feature

The closure of the package is facilitated by two one-piece wire cable tamper-indicating seals that are attached between the OP base and lid. Each seal is looped through holes in the alignment tabs located on the OP lid flange and under the tie-down arm located on the OP base flange. The location of the seal and its materials of construction minimize the potential for accidental damage during transport of the package. As a result, the staff reviewed the package tamper-indicating feature description and finds that the package satisfies the requirements of 10 CFR 71.43(b) for a tamper-indicating feature.

Positive Closure

Positive closure of the package is facilitated by bolts. Additionally, the CCV port cover is in installed with bolts. The applicant documented the fatigue analysis in the calculation CN-16007-213," OPTIMUS-L Bolt Fatigue Analysis," Rev. 0, and also described it in the SAR Section 2.4 for the positive closure of the package. The calculation documents the fatigue analysis of the CCV and port cover bolts in accordance with ASME Section III, NB-3222.4 and NB-3232.3 when transporting the OPTIMUS-L. Additionally, a stress analysis of the port cover bolts was documented in accordance with NUREG/CR-6007 to ensure the port cover bolts can withstand impact accelerations from NCT and HAC drop tests. The staff reviewed the package closure analysis and finds that the package satisfies the requirements of 10 CFR 71.43(c) for positive closure.

Package Valve

Other than the CCV lid closure and port cover closure, there are no penetrations to the containment system, and no valves, or pressure relief devices of any kind exist in the package. The staff reviewed the package closure description of the package and finds that it satisfies requirements of 10 CFR 71.43(e).

2.1.3 Lifting and Tie-Down Standards

Lifting Devices

The applicant describes lifting and handling of the package in calculation CN-16007-212," OPTIMUS-L Lifting and Tie-Down analysis," Rev. 1, and in SAR Section 2.5. The assembled package may be lifted using a forklift or using the lifting lugs installed on the top of the OP. Additionally, sub-assemblies are lifted using devices integrated in them. The lifting attachments of the package are designed in accordance with the requirements of ANSI N14.6. The ANSI N14.6 design criterion bounds the requirements of 10 CFR 71.45.

Based on the applicant's calculations, the minimum margin of safety is 0.02 with consideration of a limit of the lesser of Sy/6 and Su/10, where Sy is yield stress and Su is ultimate stress respectively. Further, the applicant evaluated the lifting attachment and summarized the calculation in Table 2.5-1. The minimum design margin is +0.02 for the maximum equivalent stress resulting from shear tear-out. The lifting attachments are also designed so that failure of any lifting device under excessive load would not impair the ability of the package to meet the other requirements of 10 CFR 71 Subpart E. Therefore, the OP lift lug satisfies the stress limits of ANSI N14.6 and 10 CFR 71.45(a).

The staff has reviewed the lifting for the package and concludes that it satisfies the requirements of 10 CFR 71.45(a) for lifting.

Tie-Down Devices

The package is tied down via the feet of the package which are bolted to a dedicated flat rack. The applicant provided calculations showing that the feet may carry five times the weight of the package in the lateral direction, ten times the weight of the package in the axial direction, and two times the weight of the package in the vertical direction. Bolts at the tie-down location are not considered to be part of the package, and failure of the tie-down system (bolts) will not impair the ability of the package to meet other 10 CFR 71 requirements.

In accordance with the requirements of 10 CFR 71.45(b)(2), any other structural part of the package that could be used to tie-down the package must be capable of being rendered inoperable for tying down the package during transport or must be designed with strength equivalent to that required for tie-down devices. The applicant stated that the only other structural part of the package that could be used for tie-downs are the lifting lugs located at the top of the OP lid. In order to prevent the lifting lugs from mistakenly being used for tie-downs, they are disabled during transport. The staff has reviewed the tie-down requirements for the package and concludes that they satisfy the standards of 10 CFR 71.45(b) for tie-down.

2.1.4 General Considerations for Structural Evaluation of Packaging

The applicant evaluated the package with a combination of analytical tools and physical drop testing comparison to determine the structural integrity of the package after being subjected to both NCT and HAC conditions. The applicant did an analysis using finite element analysis (FEA) tools to simulate drop testing to determine adequacy of the package design for the structural integrity. Specifically, the pre-analysis examined a sequence of drops with various package orientations as described in the calculation CN-16007-214, “OPTIMUS-L LS-DYNA Impact Analysis,” Rev.1, and in SAR Section 2.6, “Normal Condition of Transportation,” and Section 2.7,” Hypothetical Accident Conditions.” With respect to the drop tests cited for NCT and HAC, the applicant focused on assuring the cylinder was free from inelastic deformations as any damage to these components could cause a breach to the containment boundary. The staff reviewed the application and finds that the package satisfies the requirements of 10 CFR 71.41(a).

LS-DYNA model

The applicant used the LS-DYNA explicit dynamic program to simulate response of the OPTIMUS-L package to the NCT free drop, HAC free drop, and HAC puncture tests. A full-scale, half symmetry model was developed of the package using ANSYS parameter Design Language (APDL). The APDL was used for generating the LS-DYNA finite element model input file for the specific drop orientations and geometries of the package. Specifically, the bounding conditions in the APDL were applied to the LS-DYNA model for the NCT free drop, HAC free drop, and HAC puncture test. These conditions are discussed in Section 6.3, 6.4 and 6.5 of the applicant calculation CN-16007-214, “OPTIMUS-L LS-DYNA Impact Analysis,” Rev.1.

Material properties have been provided through catalog cuts and tabulated values in relevant codes and standards.

The applicant constructed a one-half symmetry (180°), three-dimensional model of the package including lids, bolts, CCV body, and flange using an ANSYS high order solid element to simulate the full package. The staff agrees that this model is representative of the performance of the package for the NCT free drop, HAC free drop, and HAC puncture test conditions.

Contents Modeling

The package 3-D model consists of three major structural components. These are the top impact limiter (OP lid), the bottom impact limiter (OP base) with attachments that comprise the impact limiter system (ISL), and the CCV assembly. The applicant used the ANSYS APDL customized script file for meshing and exporting the LS-DYNA geometry file. The finite element solid model is comprised of 3-D eight node brick elements represented by the LS-DYNA Solid Element Formulation Option 2. For the metal used to encapsulate the impact limiter foam the model used LS-DYNA 4-node shell Element Option 16. To simulate the polyurethane foam material, the LS-DYNA *MAT_CRUSHABLE_FOAM (Material Type 63) option was used. The polyurethane foam is not sensitive to differences in grain direction. The steel shells of the OP were modeled using LS-DYNA option *MAT_PIECEWISE_LINEAR_PLASTICITY (Material Type 24). This option allows for the input of the stress-strain curve and define failure based on the plastic strain. The applicant also modeled the OP and CCV bolt shaft using LS-DYNA option *MAT_ELASTIC (Material Type1). The applicant described various boundary conditions applied to the simulation models and documented them in CN-16007-214. The staff reviewed and accepted the applicant’s various modeling approaches for the simulation and the applied boundary conditions for the models.

Hour Glassing

In the calculation CN-16007-214, the applicant documented the energy balance time-history results for the NCT and HAC drop conditions, which showed that all the initial kinetic energy is converted into strain energy due to crushing of the impact limiter. The hourglass energy plotted in several test figures of the calculation show essentially zero, indicating that the strain energy used to control the distortion of the model's brick elements is low. The sliding energy remains positive throughout the impact, which indicates proper behavior of the model contact interfaces. Based on the staff's review of the calculation and plots, the staff agrees that the hourglass energy to internal energy is low and that the LS-DYNA modeling approach of the drop cases is acceptable.

ANSYS model

The applicant used the ANSYS computer program to generate a three-dimensional model of the OPTIMUS-L package and determine its response to NCT and HAC. The ANSYS code performed an equivalent static analysis with bounding g-loads calculated using the LS-DYNA dynamic analysis. Specifically, the applicant constructed a one-half symmetry (180 degrees), three-dimensional model of package including lid, bolts, CCV body, and flange using ANSYS high order solid elements.

The simulation of the model included applied loads and boundary conditions. In the analysis, thermal stresses were calculated using input temperatures from the bounding NCT thermal analyses. Post-processing was accomplished by linearizing the stress across several locations where maximum stresses were calculated.

The calculated stress intensities were compared to appropriate ASME code allowable stresses and the margins of safety were calculated.

The applicant determined that the closure device will not fail under NCT and HAC conditions. The staff reviewed the approach of developing the ANSYS model and concluded that it is acceptable.

Benchmarking and validation

The applicant described the benchmarking evaluation that was used for validation of the LS-DYNA software and its use for analysis of material properties for rigid polyurethane form. The applicant described that the LS-DYNA model had previously been validated for the OPTIMUS-H design as well as for the 30-foot (9 m) side drop test of a ¼ -scale model of the NAC-UMS package.

The applicant documented the LS-DYNA benchmark analysis in calculation CN-16007-204, Revision 2," OPTIMUS-H LS-DYNA Impact Analysis." The OPTIMUS-L uses a material for the OP similar to that used in the OPTIMUS-H, so that use of a validated LS-DYNA model of the materials used in the OPTIMUS-H would be representative of and could be used for an evaluation of the OPTIMUS-L OP.

The applicant compared the OPTIMUS-H impact limiter acceleration time-history of HAC side drop analysis results to the results from the 30-foot (9 m) side drop test of a ¼ -scale model of the NAC-UMS package. The applicant selected the NAC-UMS drop test results for the bench analysis because of the impact limiter material similarity between the OPTIMUS-L OP and the

OPTIMUS-H. The total weight of ½-scale NAC-UMS and the maximum gross weight of the OPTIMUS-H are also similar (approximately 32 kips). The differences between the ½-scale NAC-UMS and full-scale OPTIMUS-H include the impact limiter material, overall dimensions as presented in the SAR TABLE 2.12-1, and the required foam density of OPTIMUS-H. These differences were adjusted for proper benchmarking. The applicant also used an appropriate spring stiffness in the model that was applied in the benchmark analysis. As stated in the SAR, the applicant adjusted the acceleration time-history curve from the ½-scale NAC-UMS drop test using the mass scaling law (i.e., accelerations divided by two and the time is multiplied by two) to provide the equivalent acceleration time-history curve for a 1/2-scale NAC-UMS and the OPTIMUS-H.

The results of the comparison are provided in the SAR Figure 2.12-1, which showed a close correlation of the LS-DYNA results for OPTIMUS-H and the ½-scale NAC-UMS (the applicant clarified in a response to an NRC request for additional information that the small modular package (SMP) shown in the SAR Figure 2.12-1 was the original name of the OPTIMUS-H).

As a result of the staff's review of the methodology and the close correlation of results, the staff concluded that the benchmark evaluation for the LS-DYNA is acceptable for use in the design and analysis of the OPTIMUS-L OP.

2.1.5 Normal Conditions of Transport

The acceptance criteria used by the applicant for NCT was to demonstrate that the lid and port cover closure remains secured and that the CCV is not breached during NCT.

Heat

The applicant stated that package ambient temperature conditions correspond to an ambient temperature of 38 °C (100°F) with solar insolation. This matches the 38°C ambient temperature required by 10 CFR Part 71. Thus, staff concludes that the ambient heat requirements for the package satisfy the standards of 10 CFR 71.71(c)(1).

Differential Thermal Expansion

The applicant considered differential thermal expansion of the package as described in Section 2.6.1.2 of the application. For the thermal expansion between the CCV and the OP, the applicant evaluated it conservatively using hand calculations, assuming an upper-bound temperature of 220°F, 112°F, and for a lower bound temperature of 100°F for the OP. The calculation results show that differential thermal expansion between the CCV and OP reduces the nominal axial and radial clearances. In comparison, the clearances are nominally 0.52-inch axially and 0.25-inch radially between the OP cavity and the outside surfaces of the CCV. Therefore, the CCV will expand freely within the OP cavity under NCT heat.

Similarly, the differential thermal expansion between the SIA and CCV was evaluated conservatively, assuming an upper-bound temperature of 700°F for the SIA, and 70°F for the CCV. The results show that differential thermal expansion of the between the SIA and CCV reduces the nominal axial and radial clearances. In comparison, the clearances are nominally 0.5-inch axially and 0.25-inch radially between the CCV cavity and the outside surfaces of the SIA. Therefore, the SIA will expand freely within the CCV cavity under NCT heat conditions. Based on the review of the results, this is acceptable to the staff.

Cold

The applicant used the temperature -40°F to perform a drop test evaluation and used material properties at this temperature. The staff reviewed the cold temperature requirements for the package and concludes that they satisfy the standards of 10 CFR 71.71(c)(2).

Reduced External Pressure

In accordance with 10 CFR 71.71(c)(3), the package is designed to withstand the effects of a reduced external pressure of 3.5 psia (18.2 psig). The CCV is designed to ASME Section III, Subsection NB, for a reduced external pressure of 3.5 psia and an internal pressure of 100 psig (85.3 psia). Hence, the greatest pressure difference between inside and outside of the containment system is applied for the design.

The results of the applicant's analysis for the NCT reduced external pressure demonstrated that the package containment system satisfies the ASME allowable stress design criteria. As a result, the staff concludes that the package satisfies the standards of 10 CFR 71.71(c)(3) for reduced external pressure.

Increased External Pressure

In accordance with 10 CFR 71.71(c)(4), the package is designed to withstand the effects of an increased external pressure of 20 psia. Since the OP is not a pressure retaining component, it is not affected by increased external pressure. The applicant designed the CCV for an increased external pressure of 5.3 psig (increased above 14.7 psia atmospheric pressure) and an internal pressure of 0 psig.

As stated below in this SER Section, the water immersion evaluation for an external pressure of 290 psi is much higher than the increased external pressure of 5.3 psig. Since the CCV is designed to a higher value of external pressure (290 psi), the staff concludes that the package satisfies the standards of 10 CFR 71.71 (c)(4).

Vibration and Fatigue

According to the requirements of 10 CFR 71.71(c)(5), the package is subjected to vibration normally incident to transport. The package is transported by truck in a vertical orientation. The package is supported by a foam filled OP and tied down by four tie-down arms.

The applicant stated that the effects of vibration during transportation of the package are considered negligible since the natural frequency of the CCV is calculated as 448 Hz. This is much greater than 33 Hertz, which is the minimum natural frequency of a nominal rigid body that is generally accepted in the industry. Therefore, the package is not affected by vibration normally incident to transport.

The applicant described in the SAR Section 2.1.2.4 that consideration of vibration and fatigue in the method of analysis of the packaging structural components (other than the bolts) for which cyclic service is not required if the conditions stipulated in ASME NB-3222.4(d)(1) through (6) are met. The analysis is conservatively based on the assumption that the packaging will be used for 20 years of service and be used for one shipment per week, for a total of 1,040 shipments. For the CCV closure bolts and port cover bolts, the applicant documented the results of a fatigue analysis in the SAR Section 2.1.2.4 and Calculation CN-16007-213, Revision 0. The results of the fatigue analysis show that the CCV bolts and port bolts satisfy the fatigue design criteria of ASME NB-3232.3 for NCT cyclic loading. The cyclic loading includes startup-shutdown cycles, normal operating cycles due to thermal and pressure fluctuation, and vibration cycles. The fatigue analysis assumes one shipment every week for a period of 5 years, after that period it assumed that the bolts will be replaced. The cumulative usage factors calculated for the CCV closure and port cover bolts are 0.56 and 0.03, which are less than 1. Hence the CCV bolts and port bolt satisfy the fatigue design criteria. The staff has reviewed the vibration and fatigue requirements for the package and bolts and concludes that they satisfy the standards of 10 CFR 71.71(c)(5).

Water Spray

In accordance with the requirements of 10 CFR 71.71(c)(6), the package must be subjected to a water spray that simulates exposure to rainfall of approximately 2 in/hour for at least 1 hour. The applicant stated that the CCV assembly is isolated from the quenching effects of the water spray by the OP assembly, which insulates the CCV from sudden environmental changes. As a result, the staff agrees that the water spray test will not impair the package and concludes that they satisfy the standards of 10 CFR 71.71(c)(6).

Free Drop

The applicant evaluated five different NCT free drop impact orientations. These include bottom end drop, top end drop, bottom corner drop, top corner drop, and side drop. The applicant performed them using the finite element code LS-DYNA to generate a three-dimensional model of the package and determine its response to NCT. The ANSYS code is used to perform an equivalent static analysis with bounding g-loads calculated using the LS-DYNA dynamic analysis. Specifically, a one-half symmetry (180°), three-dimensional model of package including lids, bolts, CCV body, and flange is constructed using ANSYS high order solid elements. The OP is simulated using elastic foundation elements.

The NCT free drop was evaluated for the heaviest content weight of 3500 lbs. including the weight of the CCV bottom support plate. The drop load analysis evaluated for the worst thermal cold condition of -40 °F. The hot thermal condition of 100 °F is not considered for the NCT because the cold condition accelerations are bounding. The applicant summarized the results in the calculation report CN-16007-215, “Cask Containment Vessel Stress Analysis,” Revision 1, for the different orientations of the CCV for an NCT free drop. The calculations used proper allowable stress intensities for the CCV material from the ASME Section III, Subsection NB.

The results show a minimum margin of 0.24 for all NCT free drop orientations, which occurred for the end drop. The CCV closure bolts were qualified in accordance with the requirements of NUREG/CR-6007. In the calculation, the bolt minimum margin of safety of 0.04 occurs for the top corner drop orientation. The staff concluded that sufficient margin exists for the CCV and the closure bolts stresses based on the ASME code and NUREG/CR-6007 methodologies. As a result, the staff concludes that the package satisfies the standards of 10 CFR 71.71 (c)(7).

Corner Drop

The corner drop is addressed in this SER. The staff reviewed the package for the corner drop and concluded that the results satisfy the standards of 10 CFR 71.71(c)(8).

Compression

In accordance with 10 CFR 71.71(c)(9), packages weighing up to 11,000 pounds must be subjected to a compressive load. The gross weight of the OPTIMUS-L package, including the maximum contents weight, is approximately 9,200 pounds. Although the package gross weight is less than 11,000 pounds, the lifting attachments located on the OP lid prevent stacking of packages. Therefore, the package is not evaluated for the compressive test. The staff agrees with the applicant that the package satisfies the standards of 10 CFR 71.71(c)(9).

Penetration

In accordance with 10 CFR 71.71(c)(10), the package must be subjected to an impact of the hemispherical end of a vertical steel cylinder of 1.25-inch diameter and weighing 13 pounds, that is dropped from a height of 40 inches onto the exposed surface of the package that is expected to be most vulnerable to puncture. The OPTIMUS-L package is large in size and does not have any vulnerable location on the package surface; therefore, the package need not to be evaluated for penetration. The staff agrees with the applicant and concluded that it satisfies the standards of 10 CFR 71.71(c)(10).

NCT Conclusion

The staff reviewed the structural performance of the packaging under the normal conditions of transport required by 10 CFR 71.71 and concludes that there will be no substantial reduction in the effectiveness of the packaging that would prevent it from satisfying the requirements of 10 CFR 71.51(a)(1) for a Type B package and 10 CFR 71.55(d)(2) for a fissile material package.

2.1.6 Hypothetical Accident Conditions

Like NCT conditions, the acceptance criteria used by the applicant was to demonstrate that the valve and port are undamaged during HAC, and that the CCV is not breached (containment boundary). The applicant describes the CCV packages' ability to withstand HAC conditions in the SAR Section 2.7, "Hypothetical Accident Conditions." The drop tests considered the 30 feet free drop, the 9 m drop, and the 40 in puncture test for cumulative damage with relevant package orientations as described in Section 2.7.1 of the SAR. The applicant also considered temperatures ranging from -40°F and +100°F. The applicant described that the higher g-loads will be experienced by the package at -40°F since the material of the package is stiffer along with expected smaller deformations, while the opposite is true at +100°F. The staff agrees that the applicant used the most damaging ordinations to challenge the package.

Free Drop

The applicant evaluated for five different HAC free drop impact orientations. These include bottom end, top end, bottom corner, top corner, side, 10° bottom oblique, and 10° top oblique drops. The applicant performed using the finite element code ANSYS to generate a three-dimensional model of the package and determine its response to HAC. The ANSYS code was used to perform an equivalent static analysis with bounding g-loads calculated using the LS-DYNA dynamic analysis. The LS-DYNA analysis methods used is the same for the NCT as described in this SER Section 2.4.5. Specifically, a one-half symmetry (180°), three-dimensional model of package including lids, bolts, CCV body, and flange constructed using ANSYS high order solid elements. The HAC free drop was evaluated for the heaviest content weight of 3500 lbs. including the weight of the CCV bottom support plate. Upper-bound and lower bound analyses were performed for each HAC free drop impact orientation. The upper-bound analyses were performed using the impact limiter material upper-bound strength properties for the cold thermal condition temperature of -40°F. The applicant summarized the results in the calculation report CN-16007-215, "Cask Containment Vessel Stress Analysis," Revision 1, for the different orientations of the CCV for an NCT free drop. The calculations used proper allowable stress intensities for the CCV material from ASME Section III, Subsection NB. The results show a minimum margin of 0.25 for all HAC free drop bounding orientations, which occurred for the side drop case. The CCV closure bolts were qualified in accordance with the requirements of NUREG/CR-6007. In the calculation, the bolt minimum margin of safety of 0.12 occurs for the side drop orientation. The staff concluded that sufficient margin exists for the CCV and the closure bolts stresses based on the ASME code and NUREG/CR-6007 methodologies.

Based on the applicant's modeling and analysis, the staff agrees that the OPTIMUS -L package meets the requirements for free drop and concludes that the standards of 10 CFR 71.73(c)(1) are satisfied.

Crush

The crush test of 10 CFR 71.73(c)(2) is required only when the specimen has a mass not greater than 1,100 pounds (500 kg). This test is not applicable since the package weighs more than 1100 lbs.

Puncture

The applicant considered the three most damaging orientations that could damage key components of the package. The applicant performed the puncture drop test sequence following the HAC free drop test in accordance with 10 CFR 71.73(a). The applicant evaluated the orientation where the HAC puncture test could damage to the OP lid and expose the CCV top end. In addition, potential plastic deformation of the CCV shell resulting from a side puncture impact was considered for evaluation. All HAC puncture impact cases were evaluated for the maximum allowable content weight of 3500 pounds. The puncture evaluation was performed by the applicant using the 3-D half symmetry LS-DYNA finite element model. The puncture cases were summarized in the SAR Table 2.7-11, "Summary of HAC Puncture Cases Evaluated," including the three HAC puncture orientations and conditions associated with the tests. In all three test cases, the extent of the damage to the package from the impact was limited to local deformation of the OP, foam, and minimum plastic deformation of the CCV shell. However, the CCV was not punctured. The staff reviewed the results in the SAR Section 2.7.3 and concluded that they satisfy the standards of 10 CFR 71.73(c)(3).

Thermal

In the SAR Section 2.7.4, the applicant described the structural evaluation for the HAC thermal test to demonstrate the packaging satisfies the ASME allowable stress design criteria. The applicant summarized the maximum stress intensity in the CCV components and the maximum average tensile stress in the CCV closure bolts as shown in the SAR Table 2.7-12. The table results show that the minimum design margin for the bottom plate for primary membrane plus bending stress intensity ($P_m + P_b$) is +0.65 and the CCV closure bolt margin is +0.60 for average tensile stress. The applicant in Section 2.7.4 of the SAR describes how the CCV will continue to have pressure values below the design pressure during the fire test. As a result, the staff has reviewed the package for thermal effects and concludes that the package satisfies the requirements of 10 CFR 71.73(c)(4).

Immersion - Fissile Material

The criticality evaluation presented in the SAR Chapter 6 considered the effect of water in leakage. Thus, the requirements of 10 CFR 71.73(c)(5) do not apply. As a result, the staff reviewed the package for free drop and concluded that it satisfies the standards of 10 CFR 71.73(c)(5).

Immersion - All Packages

In accordance with 10 CFR 71.73(c)(6), an undamaged package is subjected to a water pressure equivalent to immersion under a head of water of at least 50 feet (15 m) or an equivalent external pressure load of 36.4 psi (21.7 psi gage +14.7). The package design is bounded by the 290 psi for an external pressure as required by 10 CFR 71.61, which exceeds the external pressure load of 36.4 psi. As a result, the staff reviewed the package for immersion and concluded that it satisfies the standards of 10 CFR 71.73(c)(6).

Air Transport Accident Conditions for Fissile Material

Air transport of the package is not permitted and as a result, the requirements of 10 CFR 71.55(f) do not apply.

Immersion - Special Requirement for Type B Packages Containing More Than 10^5 A_2

The requirements of 10 CFR 71.51, 10 CFR 71.55(e) and 10 CFR 71.61 apply. The applicant considered the deep-water pressure of 290 psig per 10 CFR 71.61 on the CCV external surface and modeled it with a maximum bolt preload to evaluate the stresses in the CCV. The evaluation results show that the minimum margin of safety of +0.32 occurs at the center of the bottom plate of the CCV. The CCV closure bolt margin of safety calculated in the evaluation was +0.76. The results are acceptable to the staff for the CCV design to meet the special immersion requirements. As a result, the staff concluded that the package satisfies the immersion-special requirements of 10 CFR 71.51, 10 CFR 71.55(e) and 10 CFR 71.61.

Air Transport of Plutonium

The requirements of 10 CFR 71.74 do not apply since the package does not contain plutonium.

HAC Conclusion

The staff concludes that structural performance of the OPTIMUS®-L package meets the HAC requirements of 10 CFR 71.73, and has the structural integrity to satisfy the subcriticality, containment, and shielding requirements of 10 CFR 71.55(e) for a fissile material package.

- The staff has reviewed the package structural design description and concludes that the contents of the application satisfy the requirements of 10 CFR 71.31(a)(1) and (a)(2) as well as 10 CFR 71.33(a) and (b).

- The staff has reviewed the structural codes and standards used in package design and finds that they are acceptable and therefore satisfy the requirements of 10 CFR 71.31(c).

- The staff has reviewed the lifting and tie-down systems for the package and concludes that they satisfy the standards of 10 CFR 71.45(a) for lifting and 10 CFR 71.45(b) for tie-down. The SAR described design and tie-down requirements and operation, and the staff finds they satisfy the regulations.

- The staff has reviewed the package description and finds that the package satisfies the requirements of 10 CFR 71.43(a) for minimum size. The SAR Section 2.4.1 describes the height and package diameter that satisfy the regulation requirements.

- The staff reviewed the package closure description and finds that the package satisfies the requirements of 10 CFR 71.43(b) for a tamper-indicating feature. The package closure description in SAR Section 2.4.2 satisfies the requirements.

- The staff reviewed the package closure system and the applicant's analysis for normal and accident pressure conditions and concludes that the containment system is securely closed by a positive fastening device and cannot be opened unintentionally or by a pressure that may arise within the package and therefore satisfies the requirements of 10 CFR 71.43(c) for positive closure. The staff reviewed the positive closure requirements and it meet the regulatory requirements.

- The staff reviewed the package description and finds that the package valve, the failure of which would allow radioactive contents to escape, is protected against unauthorized operation, and provides an enclosure to retain any leakage and therefore satisfies the requirements of 10 CFR 71.43(e). The containment system does not include any covers, valves, or other access that could be inadvertently opened.

- The staff reviewed the structural performance of the packaging under the hypothetical accident conditions required by 10 CFR 71.73 and concludes that the packaging has adequate structural integrity to satisfy the subcriticality, containment, and shielding requirements of 10 CFR 71.51(a)(2) for a Type B package and 10 CFR 71.55(e) for a fissile material package.

- The staff reviewed the packaging structural performance under an external pressure of 2 MPa [290 psi] for a period of not less than 1 hour and finds that the package does not buckle, collapse, or allow the in leakage of water, and therefore satisfies the requirements of 10 CFR 71.61.

2.2 MATERIALS EVALUATION

The materials review was conducted using the guidance in NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," issued August 2020. The review was mainly focused on the primary containment and structural components; however, a comprehensive review was conducted of all packaging materials.

2.2.1 Package Description

Sections 1.2, 2.1, Table 2.2-1 and associated licensing drawings 502, 510 thru 514, 540 thru 542, 550, 551 and 553, of the application, described the design and material specifications of the OPTIMUS-L packaging. The packaging consists of a Cask Containment Vessel (CCV), a CCV bottom support plate, and an Outer Packaging (OP) assembly.

The CCV (i.e., containment system) is a stainless-steel cylindrical vessel that includes a body weldment, bolted lid, bolted port cover, and O-ring seals. The containment system is formed by CCV body (cylindrical shell/bottom base plate using American Society of Mechanical Engineers

(ASME) SA-240, Type 304 or 316 stainless steel, bolt flange (ASME SA-182, Type F304 or F316), and all associated welds), CCV lid and its closure bolts (ASME SA-320, Grade L43) and containment O-ring seal, and the port cover (ASME SA-240 or SA-479, Type 304 or 316) and its closure bolts (ASME SA-193, Grade B8, Class 1) and containment O-ring seal.

The internal cavity of the CCV is large enough to accommodate a 110-gallon carbon steel drum. The CCV lid is fastened to the CCV body. The CCV lid design includes a port used for inerting the CCV cavity and contents; however, inerting is not allowed for this package. A bolted port cover is used to seal the CCV port during transport.

The CCV bottom support plate is a free-standing coated carbon steel plate (ASTM A36 or A516. Gr. 70) that is positioned at the bottom end of the CCV cavity below the contents. The CCV bottom support plate is designed to spread the loading on the CCV bottom end plate from the contents under normal conditions of transport (NCT) and hypothetical accident conditions (HAC) bottom end drop conditions. The CCV bottom support plate is not required when using the 1-inch Shield Insert Assembly (SIA). Shoring must be placed between loose fitting contents and the CCV cavity to prevent excessive movement during transport. The shoring may be made from any material that does not react negatively with the packaging materials or contents. Shoring materials should also have a melting temperature above 300°F (149°C) to ensure shoring maintains its geometry under NCT.

The CCV is fully encased in the cavity of the cylindrical-shaped OP during transport. The OP has a cavity that is sized to accommodate a CCV with sufficient radial and axial clearances to permit free differential thermal expansion of the CCV during NCT and HAC. The OP base and lid consist of energy-absorbing closed-cell polyurethane foam cores sealed inside stainless steel inner and outer shells (ASTM A240, Type 304 or 316). The OP lid is secured to the overpack base.

The fully-assembled package is designed to be lifted by a forklift from a pallet on which the package is mounted or using a 3-legged sling attached to OP lid lifting lugs (ASTM A240, Type 304 or 316).

The SIA is a coated carbon steel container inside the CCV cavity to provide supplemental gamma shielding. The SIA configurations used in the OPTIMUS-L packaging include only an open-top body, is provided in two (2) thicknesses; 1-inch and 2¼-inch thick. The internal cavity of the SIA is large enough to accommodate a 55-gallon drum.

The 1-inch SIA is designed to fit inside the CCV cavity without the CCV bottom support plate. Instead, the bottom annular plate of the 1-inch SIA serves the same function as the CCV bottom support plate by spreading the load from the weight of CCV contents over the outer portion of the CCV bottom plate under NCT or HAC bottom end impact conditions. The 2¼-inch thick SIA does require the use of the CCV bottom support plate plus a spacer plate; the annular spacer plate is placed underneath the bottom of the 2¼-inch thick SIA to position it near the top of the CCV cavity to facilitate loading operations. The SIA is also not relied upon for thermal or containment functions. No structural credit is taken for the SIA in the structural evaluation however, the SIA is designed to withstand the most severe regulatory tests (e.g., free drop) without structural failure.

2.2.2 Drawings

The staff reviewed the licensing drawings 502, 510 thru 514, 540 thru 542, 550, 551 and 553. The staff verified that the drawings included design features considered in the package evaluation, including:

- the containment system
- closure device
- internal supporting or positioning structures
- gamma shielding
- outer packaging
- heat-transfer features
- energy-absorbing features
- lifting and tie-down devices

The staff verified that the drawings include the information described in NUREG-2216 on the (1) materials of construction, (2) dimensions and tolerances, (3) codes, standards or other specifications for materials, fabrication, examination and testing, and (4) welding specifications, including location and nondestructive examination (NDE). The staff determined that the drawings for the package provide the necessary information identified in the NRC guidance documents and the engineering drawings provided by the applicant are consistent with the design and description of the package, in accordance with 10 CFR 71.33, “Package Description.” Therefore, based on the above discussion, the staff finds that the drawings provided by the applicant are acceptable.

2.2.3 Design Criteria

Sections 1.2.1.10, 2.1.2 and 2.1.4 of the application described the OPTIMUS-L design criteria and codes and standards for the package. The applicant stated that the codes and standards selection is based on guidance provided in Regulatory Guide (RG) 7.6, “Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels,” and NUREG/CR-3854, “Fabrication Criteria for Shipping Containers.”

Codes and Standards

The applicant stated that ASME B&PV Code, Section III, Division 1, Subsection NB was used for the design and fabrication of the containment system of the packaging. The non-containment structural components of the packaging are designed and fabricated in accordance with the applicable requirements from the ASME B&PV Code Section III, Division 1, Subsection NF. The applicant stated that the polyurethane foam material (i.e., OP base and lid shells) is fabricated, installed, and tested in accordance with the foam vendors’ standard practices. The applicant stated that the buckling evaluation of the packaging cylindrical shells is performed in accordance with ASME Code Case N-284-1, “Metal Containment Shell Buckling Design Methods.”

The staff notes that containment structures systems and components SSCs should be designed and fabricated to ASME Code criteria. In addition, non-containment SSCs should be designed to ASME, ASTM, or ANSI material requirements. The staff finds that the identified codes and standards are consistent with the NRC guidance in NUREG-2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material,” except for the use of ASME Code Case N-284-1, “Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC.” The staff notes that ASME Code Case N-284-1 was not approved by the NRC and is included in Table 1 of RG 1.193 Revision 6 (ML19128A269); however, RG 1.193 states that the use of Code Case N-284-1 by licensees for storage canisters and transportation casks is permissible provided it has been reviewed and approved by the NRC. The staff's review of the applicant's buckling evaluation is included in SER Section 2. Therefore, the staff finds the codes and standards for the OPTIMUS-L package to be acceptable.

Weld Design and Inspection

Section 2.3 and associated licensing drawings of the application discussed fabrication and examination of the OPTIMUS-L package.

The applicant stated that welding is performed in accordance with a written welding procedure specification (WPS) that is qualified in accordance with the applicable requirements of the ASME Code Section IX. In addition, all personnel performing welding are qualified to use the welding procedure, and their qualifications are documented in accordance with the applicable requirements of Section IX of the ASME Code.

The applicant stated that examination and testing of the packaging is performed under an NRC approved Quality Assurance (QA) program. In addition, the components and assemblies of the packaging are inspected to assure that the packaging satisfies the dimensional requirements shown on the licensing drawings (i.e., Appendix 1.3.3) of the application.

The applicant stated that all welded joints receive a workmanship, visual (VT), liquid penetrant (PT), or magnetic particle (MT) examination to ensure that they do not include visible surface defects, such as lack of fusion, linear or crack like indications, or porosity. In addition, the full-penetration welds that form the CCV body weldment are examined using either radiography (RT) or ultrasonic testing (UT) methods to ensure that they do not include any indications of weld flaws.

The applicant stated that examinations of welded joints are performed in accordance with the applicable requirements of Section V and Section III, Subsection NB, of the ASME Code for the CCV assembly and Section III, Subsection NF, of the ASME Code for all other components. In addition, written reports of each weld examination are prepared and maintained with the final records package.

The staff verified that the weld design, fabrication, and inspections are consistent with the NRC guidance in NUREG-2216, which includes the use of ASME Code Section III, Subsection NB, for containment boundary welds, and Subsections NF for other code welds, as appropriate. In addition, non-code welds are examined in accordance with ASME Code Section V, with acceptance criteria per Subsection NF. The staff notes that, although ASME Code Section III, Subsection NB, does not require visual examination of welds, the applicant stated that welds will be visually examined to ensure conformance with the drawings (e.g., proper geometry, workmanship, etc.). The staff finds, based on the above discussion, that the weld design and inspections of the OPTIMUS-L packaging meet the requirements of the ASME Codes, as applicable.

2.2.4 Material Properties

Mechanical Properties

Section 2.2.1, Table 2.2-1 through Table 2.2-9, Figure 2.2-1 and 2.2-2 of the of the application discussed materials (i.e., mechanical) properties used in the OPTIMUS-L packaging structural analyses. The staff notes that the material properties were obtained from ASME Code, Section II, Part D. The staff independently verified the temperature-dependent values for the allowable stress, modulus of elasticity, Poisson's ratio, weight density, and coefficient of thermal expansion. The staff finds, based on the above discussion, that the mechanical properties of structural materials used by the applicant for the design of OPTIMUS-L packaging components are acceptable.

Brittle Fracture

Section 2.1.2.5 of the application discussed brittle fracture of the OPTIMUS-L packaging materials. The applicant stated that the CCV assembly (i.e., containment vessel) is designed in accordance with the fracture toughness requirements of RG 7.11, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches," and NUREG/CR-1815, "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up to Four Inches Thick" for Category I containers.

The applicant stated that the entire CCV body and closure lid are fabricated from austenitic stainless steels, which do not undergo a ductile-to-brittle transition down to −40°F (−40°C) and, thus, do not need to be evaluated for brittle fracture. In addition, RG 7.11 states that since austenitic stainless steels are not susceptible to brittle failure at temperatures encountered in transport, their use in containment vessels is acceptable to the staff and no tests are needed to demonstrate resistance to brittle fracture.

The applicant stated that the CCV bottom support plate, which is a coated carbon steel plate (i.e., ASTM A36 or A516 Gr. 70), is located inside the CCV cavity. The CCV bottom support plate is positioned at the bottom of the CCV cavity and rests on the interior surface of the stainless steel CCV base plate (i.e., ASME SA-240, Type 304 or 316). The CCV bottom support plate is not part of the containment boundary.

The applicant stated that the OP assembly is designed in accordance with the Category III fracture toughness requirements of NUREG/CR-1815. In addition, the OP shell assemblies are fabricated entirely from austenitic stainless steels, which does not undergo a ductile-to-brittle transition down to −40°F (−40°C) and, thus, do not need to be evaluated for brittle fracture.

The staff notes that both containment and non-containment boundary structural metallic components and associated welds are austenitic stainless-steel and exempt from brittle fracture testing in accordance with ASME Code Section III due to their lack of a ductile-to-brittle transition at low service temperatures (i.e., -40°F (-40°C)). Therefore, based on the above discussion, the staff finds the applicant's design against brittle fracture to be acceptable.

Thermal Properties

Section 3.2 of the application discussed material thermal properties and component specifications. The applicant stated that the OPTIMUS-L packaging is fabricated primarily from Type 304 or 316 stainless steel (e.g., CCV and OP) and polyurethane foam materials. In addition, Tables 3.2-1 thru 3.2-3 of the application provide thermal properties for stainless steel, polyurethane foam and an insulation (i.e., OP lid), respectively.

Table 3.2-6 of the application provides temperature limits of packaging components for NCT and HAC. The staff evaluated the applicant's thermal properties of the materials credited in the thermal analysis and determined that the thermal properties (e.g., thermal conductivity, thermal expansion, etc.) are consistent with those in the material specifications, technical literature, and the product manufacturer's specifications. Therefore, the staff finds the OPTIMUS-L packaging material thermal properties to be acceptable.

Shielding Materials

Section 1.2.1.4 of the application stated neutron absorbers for criticality control are not necessary for the specified radioactive material contents of the OPTIMUS-L package.

Sections 1.2.1.3, 1.2.1.13 and 2.1.1.3 of the application discussed gamma shielding features of the OPTIMUS-L packaging. The applicant stated that neutron shielding is not necessary for the specified radioactive material contents; however, gamma shielding is required. The packaging bottom end includes the carbon steel CCV bottom support plate, the CCV stainless steel bottom plate, the OP inner bottom plate, the OP bottom foam cover shell, and the OP outer bottom plate, for a combined steel thickness of 3.14 inches. In addition, the packaging radial gamma shielding includes the stainless steel CCV cylindrical shell, the stainless-steel OP inner and outer shell and the OP radial polyurethane foam (i.e., NCT only) for radial stainless-steel thickness of 1.27 inches. Finally, gamma shielding in the top end of the cask is provided primarily by the CCV lid, the OP lid inner and outer end plates and the OP lid end foam for a combined stainless-steel thickness of 3.72 inches.

The applicant stated that optionally, additional gamma shielding is provided on the package side and bottom by a coated 1-inch or 2.25-inch carbon steel SIA, taken only for NCT, conservatively assuming that the contents escape the secondary container cavity and the SIA following the HAC free drop.

The staff reviewed the material properties (e.g., density) used in the applicant's shielding analyses and verified that the material proprieties are consistent with the specifications/technical literature and with those used in previously approved transport packages. Therefore, the staff finds the shielding materials to be acceptable.

Corrosion Resistance

Section 2.2.2 of the application discussed chemical, galvanic, or other reactions of the OPTIMUS-L package. The applicant stated that the packaging's materials of construction, consisting primarily of stainless steel, coated carbon steel, and polyurethane foam, will not cause significant chemical, galvanic, or other reactions in the operating environment. In addition, no significant interactions are expected to occur between the contents of the package, which consist of fuel waste or transuranic (TRU) waste contained in drums or irradiated fuel waste and the packaging materials to which they are exposed. The applicant stated that the packaging materials have been used in other radioactive material (RAM) packaging for transport of similar contents without incident.

The applicant stated that exposed surfaces of the OP and CCV assemblies are all constructed of austenitic stainless steel, with high corrosion resistance in the operating environments of the packaging. In addition, the contents are packaged in secondary containers, such as drums or liners, which limits the chemical interaction between the payload and CCV. In addition, since corrosives are prohibited from the payload, there are no chemical, galvanic, or other reactions between the contents and the CCV.

The applicant stated that the CCV bottom support plate and SIA carbon steel surfaces are coated with epoxy coating, which is commonly used in the nuclear industry for similar applications; it is highly resistant to chemical reactions and has very good abrasion resistance. In addition, the coated surfaces of the CCV bottom support plate and SIA assembly contact the stainless-steel surfaces of the CCV and, therefore, no chemical, galvanic, or other reactions are expected between the coated surfaces of the CCV bottom support plate or SIA and stainless steel.

The applicant stated that the polyurethane foam material used for the cores of the OP base and lid has a long history of use in RAM packages without any adverse reactions. In addition, the foam material is very low in free halogen content and leachable chlorides. The applicant stated that the closed-cell polyurethane foam material is sealed inside the cavity of the impact limiter stainless steel shells and in a dry environment. In the unlikely event moisture was to enter the impact limiter cavity, it could not penetrate the closed-cell structure of the foam to cause leaching of chlorides. The applicant stated that, therefore, no chemical, galvanic, or other reactions are expected between the foam and stainless steel.

The applicant stated that the Fluorocarbon-Viton rubber O-ring material that contacts the stainless-steel base material of the CCV contains no corrosives to adversely affect the packaging, is organic in nature, and has not had any chemical, galvanic, or other reactions with stainless steel.

The staff reviewed the licensing drawings and applicable sections of the application to evaluate the effects, if any, of degradation of cask components due to exposure to the service environment and due to contact between various materials in the OPTIMUS-L package materials of construction during all phases of operation. The staff evaluated whether chemical or galvanic reactions could result in corrosion that could adversely affect safety. The staff notes that, due to the vacuum drying operations, and containment seals that prevent moisture ingress, the OPTIMUS-L internals will not be subject to sufficient moisture to promote corrosion or other adverse reactions.

Further, visual inspections are to be performed of the payload cavity prior to loading and following off-loading, which provide reasonable assurance against any considerable corrosion occurring unnoticed. The OPTIMUS-L package is constructed of materials (e.g., alloy steel, stainless-steel, coated carbon steel, and polyurethane foam) that are commercially available and have a long history of use in RAM packages without any adverse reactions. Therefore, the staff finds, based on the above discussion, that no credible corrosion or other adverse reactions of the package will occur during transport.

Content Reactions

Section 1.2.2 of the application described the acceptable contents and restrictions of the package, which includes transuranic (TRU) waste and irradiated fuel waste, consisting of low enriched uranium (LEU) fuel and metal structural components (e.g., cladding, liners, baskets, etc.). Section 1.2.2.1 of the application stated that radioactive contents are packaged in secondary containers (e.g., drums or liners).

Section 4.5.2 of the application discussed flammable gases produced by the OPTIMUS-L package contents. The applicant stated that the TRU waste contents present a potential risk for the introduction of flammable gases from hydrogen gas through radiolysis (i.e., all TRU waste contents). In addition, for all TRU waste contents, limits are set to ensure there is no risk of a flammable gas mixture in any confinement region in the TRU waste contents due to radiolysis or the release of aerosol propellant gases. The applicant stated that hydrogen gas generation from mechanisms other than radiolysis are insignificant and hydrogen gas from chemical reactions is prohibited.

Section 4.5.2 of the application states that the maximum bulk-average temperature of the TRU waste contents in the CCV during normal transport is 248°F (120°C), which is well below the 302°F (150°C) threshold temperature at which gas would be generated through thermal decomposition of plastics and other polymer waste materials in air. Table 3.3-2 of the application, Maximum package temperature for NCT heat, showed that the contents/CCV fill gas (average) temperature at 50 watts to be 248°F (120°C). The applicant stated that given the estimated transportation time, nature of the waste, and environment of the payload, biological mechanisms are considered insignificant.

Section 4.5.2.1 of the application discussed oxidant control and that a package with a total heat load exceeding 50 watts must be evacuated to an oxygen content of one volume percentage (vol%) or less and backfilled with helium gas prior to shipment to reduce the quantity of oxygen inside the CCV below the threshold at which a flammable gas mixture can develop in the CCV during the shipping period. Section 4.5.2.2 and Table 4.5-1 of the application discussed content limits and provides a summary of the flammability limits for TRU waste content based on the initial quantity of oxygen in the CCV at package closure and radiolysis of water. Section 4.5.3 of the application discussed the chemical compatibility of TRU waste contents and that each TRU waste stream is defined by a content code, with a chemical list for the contents, based on process knowledge and any chemical not included in the chemical list for the specific content code is limited to less than one wt.%, and the total quantity of trace materials is restricted to less than five wt.%. Section 4.5.4 of the application discussed hydrogen concentration calculations and stated all limits are equated to a limiting hydrogen gas concentration based on the radiolysis of water. The applicant stated that compliance with the hydrogen gas limits must be demonstrated for each shipment of the package with TRU waste contents. Section 4.5.4.1 and Table 4.5-2 of the application discussed G-value Data. The applicant stated that the G-values used for flammable gas generation calculations are specific to the contents, in a given payload, based on the chemical properties of the materials. Section 4.5.4.2 of the application discussed release rate data. The applicant stated that the release rates used for calculating the concentration of flammable gas are specific to the materials of the confinement layers in a given payload. Section 4.5.4.3 of the application discussed hydrogen gas accumulation calculations.

The staff reviewed the information provided by the applicant and determined that the applicant only shows a venting mechanism to allow gases to flow in or out of each confinement region of the contents but does not show its effectiveness in operation and does not demonstrate that the inert fill gas either “effectively” occupies the cavity of the cask containment vessel (CCV) or is in uniform concentration through the CCV. In order to allow inerting, the staff requires, at a minimum, a full demonstration that the inerting process will prevent the development of flammable gas mixtures in any confined area of the package throughout the entire shipment period and a detailed evaluation to prove that there are no flammable gas mixtures (considering the worst case concentrations) during shipment. In addition, the applicant needs to explain how the inerting gas is effectively introduced to all confined areas within the containment system of the package and, as such, is in uniform concentration throughout the CCV, Finally, the concentrations of combustible gases need to be able to be quantitatively analyzed. Therefore, the staff determined it could not reach a finding of reasonable assurance in order to accept inerting for the OPTIMUS-L package. As a consequence, the CoC includes a condition preventing inerting and limiting waste, authorized for shipment, with a decay heat below 50 watts.

Radiation Effects

Section 2.2.3 of the application discussed effects of radiation on materials. The applicant stated the packaging is designed using materials of construction such as austenitic stainless steel, carbon steel and ferritic bolting steel that are unaffected by the radiation levels in this package.

The staff notes that the gamma radiation associated with the decay of TRU waste and irradiated fuel waste, consisting of LEU fuel and metal structural components, is expected to have no detrimental effect on the austenitic stainless steel, carbon steel and ferritic bolting steel comprising the primary structural components of the OPTIMUS-L package during transportation. In addition, this SER addresses the radiation effects on the polyurethane foam and the fluorocarbon O-ring seals. The staff finds, based on the above discussion, there will be no deleterious radiation effects on the OPTIMUS-L packaging materials, and therefore they are acceptable.

2.2.5 Component-Specific Reviews

Protective Coatings

Section 2.2.2 of the application discussed chemical, galvanic or other reactions. The applicant stated that all exposed surfaces of the carbon steel CCV bottom support plate and SIA are coated with epoxy coating for corrosion protection, which is commonly used in the nuclear industry for similar applications. In addition, the epoxy is high-temperature, radiation-resistant, highly resistant to chemical reactions and has very good abrasion resistance. The CCV and SIA licensing drawings stated to remove oil and/or grease from all exposed surfaces, commercial blast clean and apply Carboline 890 epoxy coating to all exposed carbon steel surfaces.

The staff notes that the epoxy coating identified are commercially available with years of proven performance. The staff finds the protective coating (i.e., epoxy) to be acceptable based on the above discussion, independent review of various technical literature (e.g., data sheets, handbooks, etc.), and the coating’s ability to prevent oxidation, withstand radiation and the maximum service temperatures without undergoing adverse reactions that could impact package performance of the OPTIMUS-L during transport.

Polyurethane Foam Impact Limiter Material

Section 2.2.1.2 of the application discussed impact limiter energy-absorbing materials. The OP base and lid are filled with rigid, closed-cell polyurethane foam. The foam pieces are oriented with the direction-of-rise parallel to the longitudinal axis of the package. The dynamic stress versus strain data for the polyurethane foam materials are developed based on data provided by a foam manufacturer General Plastics Manufacturing Co., Last-A-Foam FR-3700. The applicant stated that the minimum and maximum foam temperatures considered for foam crush strength properties are −40°F (−40°C) and 180°F (82°C), respectively. In addition, these temperatures represent the range of temperatures the foam will experience under all initial conditions for the NCT and HAC free drop tests. The applicant stated that the resulting upper-bound and lower-bound dynamic crush strength-versus-strain curves foam densities are summarized in Figure 2.2-1 and 2.2-2, of the application, respectively.

Section 2.1.4 of the application stated that the polyurethane foam material is fabricated, installed, and tested in accordance with the foam vendors’ standard practices. In addition, the foam segments are manufactured with the foam rise parallel to the longitudinal axis of the package and encased in the stainless-steel shells. Section 8.1.5.2 of the application stated that each batch of closed-cell polyurethane foam used to construct the foam segments of the OP base and lid assemblies shall be tested for the following attributes and foam not meeting the acceptance criteria shall be rejected:

- Leachable Chlorides: Assure that it has no more than one part per million (ppm) of leachable chlorides.
- Average Density: The density of each pour from each batch of foam shall be tested at room temperature (i.e., 75 °F ± 10 °F) in accordance with ASTM D1622. The average apparent foam density from each pour, determined based on a minimum of three samples, shall be within ±20% of the nominal values
- Static Crush Strength: The static compressive strength of each pour from each batch of foam shall be tested in both the parallel-to-rise and perpendicular-to-rise directions at room temperature (i.e., 75 °F ± 10 °F) in accordance with ASTM D1621. A minimum of three samples from each pour from each batch shall be tested for each orientation to determine the compressive stress at strains of 20%, 40%, and 60%. The average foam compressive stress results of the foam in each foam core shall meet the acceptance criteria in Table 8.1-1 of the application.
- Flame Retardancy: Each batch of foam shall be tested to assess the relative burning characteristics of the foam material under controlled laboratory conditions in accordance with the foam manufacturer’s test procedures, which generally comply with the requirements for the Federal Aviation Regulation (FAR) 25.853 flame test. A minimum of three test samples from each batch of foam shall be tested.
- Intumescence: Each batch of foam shall be tested to determine its average intumescence in accordance with the foam manufacturer’s test procedures. A minimum of three test samples from each batch of foam shall be tested.

The applicant stated that the polyurethane foam material used for the OP base and lid cores is unaffected by gamma radiation exposure up to 2×10^{8} rad, equivalent to 1,000 rad/hour for a period of 20 years. In addition, at radiation exposure up to 2×10^{8} rad, testing shows no effect on density or crush strength and the resistance of the polyurethane foam material to water absorption is unaffected by radiation exposure up to 1×10^{7} rad.

The staff notes that the polyurethane foam material FR-3700 used for the cores of the OP base and lid is commercially available and has a long history of use in RAM packages without any adverse reactions. The staff reviewed the foam material properties (e.g., density, temperature range, etc.) used in the applicant’s structural and thermal analyses and verified that the material proprieties are consistent with the technical literature (e.g., data sheet). Therefore, the staff finds the polyurethane foam material to be acceptable for use in the OPTIMUS-L package.

Bolting

Section 2.1.2.5 of the application stated that the CCV assembly closure bolts are fabricated from ASME SA-320, Grade L43 stainless steel bolting material that is intended for low-temperature service. The applicant stated that the bolting material is required to have a minimum impact energy absorption of 20 ft-lbf (27 N-m) at a temperature of −101°C (−150°F). Tables 6-17 and 2.2-4 of the application shows structural (mechanical) properties of the CCV closure bolts.

Section 2.2.1 of the application stated that ASME SA-193, Grade B8, Class 1 stainless steel bolting is used for the port cover bolts and ASTM A574 socket-head cap screws are used for the OP closure bolts. Tables 6-16 and 2.2-5 of the application shows structural (mechanical) properties of the port cover bolts. Application Tables 6-20 and 2.2-8 alloy socket-head cap screws of the application shows structural (mechanical) properties of the OP closure bolts. For the ASTM A574 socket head cap screws, the applicant stated that in accordance with the guidance in NUREG/CR-1815, Category III components specified without fracture toughness testing should manufactured for normalized steel made to “fine grade practice” or better.

The staff notes that the bolting material and thermal properties were obtained from ASME Code, Section II, Part D, and the allowable stress limits for the bolts was determined using the methodology described in NUREG/CR-6007. The staff independently verified the temperature-dependent values for the allowable stress limits, modulus of elasticity, Poisson’s ratio, weight density, thermal conductivity, and coefficient of thermal expansion. The staff finds, based on the above discussion, that the mechanical properties of the bolt materials used by the applicant for the design of OPTIMUS-L packaging components are acceptable.

The staff notes that both containment and non-containment boundary bolting materials are manufactured using austenitic stainless-steel and exempt from brittle fracture testing in accordance with ASME Code Section III due to their lack of a ductile-to-brittle transition at low service temperatures (i.e., −40°F (−40°C)). The staff reviewed the ASTM A574 specification, which covers quenched and tempered steel socket head cap screws up to 4 inches in diameter. The staff verified that ASTM A574 specifies that the screws shall be fabricated from alloy steel made to a fine grain practice and that for the grade specified in the application, the fine grain quenched and tempered steel has acceptable fracture toughness at low temperatures and would not be susceptible to brittle fracture. In addition, the staff notes that all the bolting materials for closure applications are commercially available and have been successfully used in RAM packages. Therefore, based on the above discussion, the staff finds the applicant’s design against brittle fracture to be acceptable.

Seals

Section 2.2.3 of the application stated that the fluorocarbon polymer (FKM) O-ring material has good radiation-resistance properties and that radiation exposure below 10^6 rad, a level attained only after many years of operation, produces no change to the physical properties of the O-ring material. In addition, normal wear (as opposed to radiation exposure) is the main factor affecting their replacement frequency. The applicant stated that the O-rings are coated with a thin film of silicone-based lubricant to help protect the O-ring from damage by abrasion, pinching, or cutting. In addition, the lubricant also helps to seat the O-ring properly and protect the polymer from environmental damage. The applicant stated that because the O-ring lubricant is frequently cleaned and replaced, and because most of the lubricant’s benefit occurs during installation, radiation damage is not a concern.

Section 7.1.1 of the application discussed preparation for loading and provided instructions to visually inspect the O-ring seals for signs of damage or defects (e.g., cracks, tears, cuts, or discontinuities) that may prevent them from sealing properly when the package is assembled. In addition, any damaged or defective O-ring seals should be replaced with new O-ring seals in accordance with the requirements of the maintenance program described in Section 8.2.3.1 of the application.

Section 8.1.5.1 of the application stated that containment O-rings will be made from the Fluorocarbon-Viton compound specified on the licensing drawings that has been qualified based on testing to verify material composition, physical properties (hardness, tensile strength, elongation, and specific gravity), low-temperature properties, and compression set at high temperature. In addition, each O-ring will be subjected to dimensional acceptance testing.

Section 8.1.4 of the application discussed leakage rate tests. The applicant stated that the CCV assembly (i.e., the packaging containment boundary) shall be leakage rate tested in accordance with Section 8 of ANSI N14.5, “American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment,” to an acceptance criterion of 1×10^{-7} ref-cm^3/s. In addition, a CCV assembly that does not meet the acceptance criteria shall be reworked, replaced, or repaired, as required, and retested prior to acceptance.

The staff notes that the fluorocarbon (Viton) O-ring seal material used for the cores of the OP base and lid is commercially available and has a long history of use in RAM packages without any adverse reactions. The staff verified the O-ring material properties provided in technical literature (e.g., elastomer handbook) and the application. The staff concludes that, based on the discussion above, the containment seal material can perform in the thermal and radiation environments under NCT and HAC. Based on the above discussion and that seals are visually inspected, leak tested prior to shipment, and are replaced within a 1-year period prior to any shipment, the staff finds that the O-ring seals used in the OPTIMUS-L package are acceptable.

Insulation

Section 1.2.1.11 of the application stated that the packaging includes an insulation, bonded to the inner surface of the OP lid outer end plate, to minimize heating of the OP lid foam from insolation. The applicant stated that this reduces the volume-average temperature of the foam upon which the lower-bound foam stress-strain properties are based for the drop analyses. In addition, this feature would also minimize the heating of the CCV closure O-ring seals during the HAC fire. However, no credit is taken for the insulation during the HAC thermal test. Table 3.2-3 of the application provides the thermal properties of the insulation used in the OP lid.

The staff notes that the insulation material used to minimize heating of the OP lid foam is commercially available and has a long history of use in RAM packages without any adverse reactions. The staff verified that the insulation material would adequately perform at temperatures expected during NCT, based on the staff’s review of the service conditions, and the vendor’s technical data. Therefore, based on the discussion above, the staff finds that the insulation material is acceptable for use in the OPTIMUS-L package.

Evaluation Findings

The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.33. The applicant described the materials used in the transportation package in sufficient detail to support the staff’s evaluation.

The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.31(c). The applicant identified the applicable codes and standards for the design, fabrication, testing, and maintenance of the package and, in the absence of codes and standards, has adequately described controls for material qualification and fabrication.

The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a). The applicant demonstrated effective materials performance of packaging components under normal conditions of transport and hypothetical accident conditions.

The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.85(a). The applicant has determined that there are no cracks, pinholes, uncontrolled voids or other defects that could significantly reduce the effectiveness of the packaging.

The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.43(d), 10 CFR 71.85(a), and 10 CFR 71.87(b) and (g). The applicant has demonstrated that there will be no significant corrosion, chemical reactions or radiation effects that could impair the effectiveness of the packaging. In addition, the package will be inspected before each shipment to verify its condition.

The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a) for Type B packages. The applicant has demonstrated that the package will be designed and constructed such that the analyzed geometric form of its contents will not be substantially altered and there will be no loss or dispersal of the contents under the tests for normal conditions of transport.

Conclusion

The NRC staff concludes, based on review of the statements, and representations in the application, that the materials used in the OPTIMUS-L transportation package design have been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

3.0 THERMAL EVALUATION

The purpose of this evaluation is to verify that the OPTIMUS-L transportation package provides adequate protection against the thermal tests specified in 10 CFR Part 71, and meets the thermal performance requirements of 10 CFR Part 71 under Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC).

3.1 Description of Thermal Design

As stated in the application, the OPTIMUS-L is designed as Type B(U)F transportation package to ship contents of Type B quantities of normal form transuranic (TRU) waste and fuel waste material. The OPTIMUS-L package is considered as a Category I package and is designed to be transported by highway, in an open conveyance, under exclusive-use control.

The package consists of a cask containment vessel (CCV), a CCV bottom support plate, and an outer packaging (OP) assembly, as shown in Figure 1.1-1 of the application. All details and relevant dimensions of the packaging components are provided in the Licensing Drawings, Nos. 70000.14-502 thru 553 in Appendix 1.3.3, “Packaging General Arrangement Drawings.” The thermal design of the package is described in Section 3.1, “Description of Thermal Design,” and is summarized below:

> The CCV is the innermost vessel of the packaging and serves as the primary containment boundary of the package. The body of the CCV is comprised of stainless-steel shells and a deep machined or forged stainless-steel flange. The CCV lid is a stainless-steel plate, that protrudes into the CCV cavity, due to the stepped lid design. The O-rings used in the CCV lid and port cover are composed of fluorocarbon compound (Viton).

> The OP is comprised of a base and lid that form an internal cavity inside which the CCV is placed. The OP base and lid are both constructed of the stainless-steel shells that are filled with polyurethane foam to provide impact and thermal protection for the CCV. The polyurethane foam components are comprised of different densities in the different regions of the OP.

The applicant stated that the maximum total decay heat of the contents is 100 watts. However, since inerting is not authorized for this package, the total decay heat of the contents to be shipped will be limited to 50 watts and the CCV cavity be filled only with air. The staff reviewed the description of thermal design provided in Section 3.1 of the application and determined that the description of the thermal design is appropriate for a thermal evaluation: (a) the package is designed to safely dissipate heat under the passive conditions and (b) the packaging and contents' temperatures will remain within their respective allowable values or criteria for NCT and HAC, as required in 10 CFR Part 71.

3.2 Material Properties and Component Specifications

The applicant specified material properties and packaging components in Section 3.2, "Material Properties and Component Specifications," and provided material properties of the packaging components, including Shield Insert Assembly (SIA), in Tables 3.2-1/1A and Tables 3.2-2 through 3.2-5 used for the thermal model. The applicant provided the temperature limits of packaging components in Table 3.2-6 of the application and specified the minimum temperature limit of -40°C (-40°F) for all packaging components.

The applicant stated, in Section 3.2, that the minimum temperature limit for all components is -40°C (-40°F). The applicant provided General Plastics Data Sheets of LAST-A-FOAM for verification of the 250°F limit for polyurethane foam used in the package.

The staff reviewed the material properties provided in Tables 3.2-1/1A and Tables 3.2-2 through 3.2-5, and the component specifications provided in Section 3.2 and determined that they are appropriate to provide a basis for the thermal evaluation of the package to meet the requirements of 10 CFR Part 71.

3.3 General Considerations

3.3.1 Thermal Model

The applicant stated, in Section 3.3, "Thermal Evaluation under Normal Conditions of Transport," that the contents, CCV and OP are modeled for thermal evaluations and the design features of OP lifting lugs, OP tiedown arms, and CCV O-rings, are not explicitly simulated in the thermal model because these components do not significantly affect the thermal performance of the packaging.

The staff reviewed the package configuration and thermal design and agrees with the applicant's choice of packaging components included in the thermal model. The staff also reviewed the assumptions, methodology, and initial/boundary conditions used in the thermal model and concludes that those are acceptable for evaluation of the package thermal design.

3.3.2 Thermal Contact Resistance

The applicant stated, in Section 3.3, that the thermal contact between packaging components is modeled by specifying the thermal contact conductance (TCC) of the interface as a real constant. The TCC is defined as the reciprocal of the thermal contact resistance. As described in Section 3.3, the thermal contact resistance is due primarily to the surface roughness of the mating parts and is also a function of the mating materials, interstitial fluid/gas, and contact pressure, with TCC values of 1,000 Btu/hr-in^2-˚F for surfaces with low thermal contact resistance, 15 Btu/hr-in^2-˚F for surfaces with low/moderate contact resistance, 5 Btu/hr-in^2-˚F for surfaces with moderate contact resistance, 1 Btu/hr-in^2-˚F for surfaces with high/moderate contact resistance, and 0.5 Btu/hr-in^2-˚F for a surface/gap with high contact resistance.

The applicant stated, in Section 3.5.2.1, that sensitivity analyses were performed for NCT condition by changing all TCC values from the mixed values, discussed in Section 3.3 of the application, to 1,000 Btu/h-in^2-°F for low thermal contact resistance and 0.5 Btu/h-in^2-°F for high thermal contact resistance. The applicant presented the details of the sensitivity analyses in Appendix B of Calculation CN-16007-311 Rev. 4. As shown in Table B.2, the difference in the package temperatures is no more than 2°F between the analysis results from the model using mixed TCC values and the models with low or high TCC values.

The staff reviewed the sensitivity analyses in Section 3.5.2.1 and Calculation CN-16007-311 Rev. 4 of the application and determined that the TCC values assigned to the model adequately represent the thermal contact conditions between the various components of the OPTIMUS-L packaging configuration.

3.3.3 Shield Insert Assembly (SIA) Design Features

The applicant stated, in Section 3.1, that an optional Shield Insert Assembly (SIA) may be used inside the CCV to provide supplemental shielding for some contents. The SIA, a painted carbon steel shield sized to accommodate a 55-gallon drum, is placed inside the CCV cavity for contents that require additional shieling.

In response to a Request for Additional Information (RAI) issued by the staff, the applicant performed thermal analyses with the bounding thick SIA included in the thermal model for both NCT heat and HAC fire conditions. The analytical results in Appendix E, “OPTIMUS-L with SIA,” of Calc. CN-16007-311 Rev. 4, show that the SIA does not significantly affect the thermal performance of the package under NCT and HAC.

The staff reviewed Section 3.1 and Appendix E of Calc. CN-16007-311 Rev. 4 and accepts that the design feature of the SIA is described in sufficient detail for thermal evaluations under NCT and HAC. After comparing Table E.9. (with SIA) with SAR Table 3.3-2 (without SIA) for 55-gallon drum under NCT and Table E.10. (with SIA) with SAR Table 3.4-1 (without SIA) for 55-gallon drum under HAC, the staff agrees with the applicant’s conclusions that the SIA does not significantly affect the thermal performance of the package under NCT and HAC.

3.4 Thermal Evaluation under NCT

3.4.1 Heat and Cold

Heat

The applicant performed the NCT thermal analyses for the cases of (1) 50-watt volumetric heat load, air fill gas, and 55-gram/110-gram drums inside a thicker SIA in the CCV and (2) 100-watt volumetric heat load, helium fill gas, and 55-gram/110-gram drum in the CCV. The applicant evaluated the package under NCT with solar insolation, thermal contact resistance between packaging components, and natural convection and thermal radiation between package surface and the ambient, as described in SAR Section 3.3. The applicant stated in Section 3.3 that the results shown in Tables 3.1-1 and 3.3-2 indicate that (1) the CCV cavity gas does not significantly affect the temperatures of the packaging but does have a significant effect on the temperature of the contents and CCV fill gas, which affects the internal pressure of the CCV, and (2) all packaging component temperatures do not vary significantly with the content configuration (with/without SIA) and remain below their allowable temperature limits for NCT.

The staff reviewed the model description, thermal contact resistances, boundary conditions and the content configuration used for the NCT thermal evaluation. The staff confirmed that the temperature results shown in Tables 3.1-1 and 3.3-2 are acceptable and the maximum temperatures of the package components, including SIA, containment seals, and contents/fill-gas, are below their allowable limits for NCT.

<u>Cold</u>

The applicant stated, in Section 2.6.1.3, "Stress Calculations," that the package is designed to withstand the effects of a steady state ambient temperature of -40°F (-40°C) in still air and shade in accordance with 10 CFR 71.71(c)(2). Per Table 2.1-1, the NCT cold environment is evaluated in combination with zero insolation, zero decay heat, and zero internal pressure. Therefore, the NCT cold environment results in a uniform temperature of -40°F (-40°C) throughout the package.

The staff reviewed the service temperature ranges of the packaging components and verified that the minimum allowable service limit of all components is less than or equal to -40°C (-40 °F). The staff accepts that the package will sustain NCT cold conditions even at an ambient temperature of -40°C (-40°F).

The staff reviewed the package design and evaluation for shipment under both heat and cold conditions and concludes that the package material and component temperatures will not extend beyond the specified allowable limit during NCT consistent with the tests specified in 10 CFR 71.71.

3.4.2 Maximum Normal Operating Pressure (MNOP)

The applicant stated, in Section 3.3.2, "Maximum Normal Operating Pressure," that the MNOP is calculated by treating all gases in the CCV as ideal gases and determining the partial pressure contributions from temperature change, water vapor, and gas generation from radiolysis. The applicant presented the maximum pressures for NCT in Table 3.3-3.

The staff reviewed calculations of the MNOPs contributed by temperature change, water vaporization and radiolysis, and confirmed that the MNOPs, as shown in Table 3.3-3, are below the design pressure of 100 psig, as provided in Section 2.6.1.1, "Summary of Pressures and Temperatures."

3.4.3 Differential Thermal Expansion

The applicant stated, in Section 2.6.1.2, "Differential Thermal Expansion," that differential thermal expansion of the packaging components is evaluated considering possible interference resulting from a reduction in gap sizes. The differential thermal expansion evaluation includes radial and longitudinal differential thermal expansion between the CCV assembly and the OP cavity and between the SIA and the CCV cavity.

The applicant hand-calculated the nominal axial and radial clearances which are reduced under NCT between the CCV and the OP as well as between the SIA and the CCV. Compared to the nominal axial and radial clearances allowed in the design, the applicant stated, in Section

2.6.1.2, that the CCV expands freely within the OP cavity and the SIA expands freely within the CCV cavity under NCT thermal loading.

The staff reviewed Chapter 3 of the application to confirm (a) the upper-bound temperature for the CCV and the lower-bound temperature of the exterior surface for the OP and (b) the upper-bound temperature for the SIA and the lower-bound temperature for the CCV, used in the applicant's NCT thermal expansion calculations. The staff finds that the calculated nominal axial and radial gap sizes still allow for free expansion with negligible thermal stress for the CCV within the OP cavity and the SIA within the CCV cavity under NCT.

3.5 Thermal Evaluation under HAC

3.5.1 Original Damage Model and Modified Damage Model

As described in Section 3.5.2.3, "HAC Damage Model," the applicant performed a thermal analysis using the original damage model which assumes that the OP lid outer end plate and the insulation attached to the inside surface of the OP lid outer end plate are not present and the heat flux from the HAC fire is directly applied to the OP end foam steel liner.

The applicant performed an additional thermal analysis of the packaging using the modified damage model which includes the cumulative damage from the HAC side drop and HAC top end center puncture impact. The results in Table 3.5-2 show the peak temperatures from the modified damage model are within a few degrees of the results obtained using the original damage model and are well below the allowable temperature limits under HAC.

After reviewing descriptions of both the HAC original damage model and modified damage model and Table 3.5-2, the staff confirmed that the HAC thermal evaluation using the original damage model, described in Section 3.4.1, provides reasonable bounding results for the worst-case cumulative damage predicted for the HAC test sequence.

3.5.2 CCV Axial Position in the OP Cavity

As described in Section 3.5.2.2, "Axial position of the CCV within the OP Cavity," the applicant also performed separate HAC thermal analyses with the CCV and contents positioned at center, bottom and top ends of the OP cavity during the fire, in order to determine the maximum temperatures of the various packaging components. The maximum HAC package temperatures vs. CCV position in the OP cavity are presented in Table 3.5-1.

After reviewing Section 3.5.2.2 and Table 3.5-1, the staff finds that the maximum package temperatures result from the CCV positioned at either top or bottom end and the maximum package temperatures from the CCV at all axial positions within the OP cavity (center, top and bottom) are below the allowable limits and therefore, are acceptable for the HAC thermal evaluations.

3.5.3 Maximum Temperatures and Pressure

The applicant stated, in Section 3.4.3, "Maximum Temperatures and Pressure," that the package is evaluated for a 30-minute HAC fire to meet requirements of 10 CFR 71.73(c)(4). The package is evaluated for the HAC fire using a content heat load of 50 watts with air filling the CCV cavity. To bound the maximum package temperatures for the wide range of possible contents and configurations, the applicant performed HAC thermal analyses for two different content configurations: (1) uniformly distributed volumetric heat source from waste filling a 110-gallon drum in the CCV cavity, and (2) uniformly distributed volumetric heat source from waste filling a 55-gallon drum inside the cavity of a thick SIA in the CCV cavity. The applicant provided the maximum packaging component temperatures for HAC in Table 3.4-1, and Figures 3.4-1 thru 2.4-8/8A and the summary of HAC pressures in Table 3.4-2.

After reviewing Section 3.4.3, Table 3.4-1 of the application, the staff concludes that the package material and component temperatures will not extend beyond the corresponding allowable limits during HAC, consistent with the tests specified in 10 CFR 71.73(c)(4), and the maximum HAC pressure for the 50-watt heat load/air filling CCV, as shown in Table 3.4-2, is below the design pressure of 100 psig, which is the limit described in Section 2.6.1.1.

3.5.4 Maximum Thermal Expansion

The applicant stated, in Section 2.7.4.2, “Differential Thermal Expansion Stress,” that differential thermal expansion in the packaging components due to the HAC thermal loading causes the clearances between the packaging components to increase, because the temperature of the OP shells is higher than that of the CCV shell based on HAC thermal evaluation. Following the HAC fire, the temperature gradients between CCV and OP remain bounded by those resulting from NCT heat.

Therefore, the differential thermal expansion between CCV and OP during HAC fire will be bounded by the results for NCT heat (see Section 2.6.1.2 of the application). The applicant summarized the maximum temperatures of key packaging components shown in Table 3.4-1 along with their corresponding allowable temperatures. The OP shells/plates are only required to maintain confinement of the polyurethane foam; therefore, they are only required to remain below their respective melting temperatures during the HAC fire.

The staff reviewed Section 2.7.4.2 and Table 3.4-1 and confirmed that the maximum temperatures in all packaging components remain well below their allowable temperatures for the HAC fire and do not vary significantly with the assumed payload configurations. The staff also accepts that the maximum temperatures for the cases with the 55-gallon drum in the SIA are lower than the other cases due to the added thermal mass of the SIA.

3.6 Evaluation Findings

Staff has reviewed the package description and evaluation and concludes that they satisfy the thermal requirements of 10 CFR Part 71. Staff has reviewed the material properties and component specifications used in the thermal evaluation and concludes that they are sufficient to provide a basis for evaluation of the package against the thermal requirements of 10 CFR Part 71. Staff has reviewed the methods used in the thermal evaluation and concludes that they are described in sufficient detail to permit a thermal review of the package thermal design.

Staff has reviewed the accessible surface temperatures of the package as it will be prepared for shipment and concludes that they satisfy 10 CFR 71.43(g) for packages transported by exclusive-use vehicle. Staff has reviewed the package design, construction, and preparations for shipment and concludes that the package material and component temperatures will not extend beyond the specified allowable limits during normal conditions of transport consistent with the tests specified in 10 CFR 71.71. Staff has reviewed the package design, construction, and preparations for shipment and concludes that the package material and component temperatures will not exceed the specified allowable short time limits during hypothetical accident conditions consistent with the tests specified in 10 CFR 71.73(c)(4).

Based on review of the statements and representations in the application, the staff concludes that the impact of the proposed changes on the thermal design has been adequately described and evaluated, and that the thermal performance of the OPTIMUS-L package meets the thermal requirements of 10 CFR Part 71.

4.0 CONTAINMENT EVALUATION

The focus of this review is to ensure that the containment system for the OPTIMUS-L meets the regulatory requirements for containment performance found in 10 CFR Part 71 and complies with the standards in ANSI N14.5, 2014 as far as the applicant has committed to implement those standards.

4.1 Description of the Containment System

As described in Section 4.1 of the application, the OPTIMUS-L containment system is provided by the CCV, which is comprised of the body weldment, bolted closure lid, bolted port cover, lid port, and the associated lid and port cover containment O-ring seals. The containment boundary and containment system components are depicted in the drawings in Figure 4.1-1 of the application. The CCV's body is solid austenitic stainless steel (Type 304 or 316) with a 1" thick cylindrical shell, a 1" thick bottom plate, and a bolt flange.

The application states that, outside of the CCV lid closure and port cover closure, there are no penetrations to the containment system, and no valves or pressure relief devices of any kind. According to Section 4.1 of the application, the packaging does not rely on any filter or mechanical cooling system to meet containment requirements, nor does it include any vents or valves that allow for continuous venting, which is required by 10 CFR 71.51(c).

The discussion in Section 4.1 states that the CCV system was designed, fabricated, examined, tested, and inspected in accordance with the requirements of the ASME Code, Subsection NB, with certain exceptions provided in Chapter 2: the containment system materials of construction are discussed and evaluated in Section 2.2.1 of the application while these materials were selected to avoid chemical, galvanic, or reactions, as discussed in Section 2.2.2.

Based on staff's review of the design of the containment vessel, the design meets the requirements of 10 CFR Part 71.33 and 10 CFR 71.31(c) given that inerting will not be allowed and that the lid port cover will be closed.

4.2 Containment under Normal Conditions of Transport

The OPTIMUS-L has been designed to meet the criteria of containment given in 10 CFR 71.71 for normal conditions of transport (NCT). The analysis presented in Section 2.6 of the application demonstrates that all inner containment system components are maintained within their code-allowable stress limits during NCT tests. The analysis presented in Section 3.1 demonstrates that all inner containment system components are within their respective temperature limits for NCT.

The applicant determined the maximum pressure of 15.8 psig for NCT based on the maximum heat load of 50 W for air. The staff reviewed Sections 2.7.4 and 3.1.4 of the application and verified that the maximum pressure of 15.8 psig is bounded by the test pressure of 100 psig considered for the structural evaluation under NCT.

In Section 4.2.2 of the application, the applicant stated that this package is leaktight per the definition provided in ANSI N14.5. Regarding the containment criteria, the allowable leakage rate criteria (as defined by ANSI N14.5) and types of tests specified are provided in Tables 8.1.1 and 8.1.2 of the application. The applicant stated, in Section 4.3.3, that the thermal evaluation in Section 3.3.1 demonstrated that the seals, bolts, and containment system materials of construction do not exceed their temperature limits when subjected to the conditions of 10 CFR 71.71.

Based on review of the description for containment under NCT and reviewing the appropriate calculations provided by the applicant, the staff finds that the design of the containment system of the OPTIMUS-L meets the containment requirements in 10 CFR 71.71 for NCT. In addition, staff determined that inerting should be removed and that the maximum decay heat limit should be 50 W.

4.3 Containment under Hypothetical Accident Conditions of Transport

The OPTIMUS-L has been designed to meet the containment criteria provided in 10 CFR 71.73 for hypothetical accident conditions (HAC) of transport. The analysis presented in Section 2.7 of the application demonstrates that all inner containment system components are maintained within their code-allowable stress limits during HAC tests. The analysis presented in Section 3.1 demonstrates that all inner containment system components are maintained within their respective temperature limits for HAC.

The applicant determined the maximum pressure of 36.8 psig for HAC based on the maximum heat load of 50 W for air. The staff reviewed Sections 2.7.4 and 3.1.4 of the application and verified that the maximum pressure of 36.8 psig is bounded by the test pressure of 225 psig considered for the structural evaluation under HAC.

In Section 4.3.2, the applicant stated that this package is leaktight per the definition provided in ANSI N14.5. Regarding the containment criteria, the allowable leakage rate criteria (as defined by ANSI N14.5) and types of tests specified are provided in Tables 8.1.1 and 8.1.2 of the application. The applicant states, in Section 4.3.3, that the thermal evaluation in Section 3.4.3 demonstrates that the seals, bolts, and containment system materials of construction do not exceed their temperature limits when subjected to the conditions of 10 CFR 71.73.

Based on the staff's review of the description for containment under HAC and having reviewed the appropriate calculations, the staff finds the design of the containment system of the OPTIMUS-L meets the containment requirements in 10 CFR 71.73 for HAC. In addition, staff determined that inerting should be removed and that the maximum decay heat limit should be 50 W.

4.4 Leakage Rate Tests for Type B Packages

In Section 4.4 of the application, the applicant stated that all leakage rate testing of the OPTIMUS-L cask containment system shall be performed in accordance with the guidance in ANSI N14.5. Section 8 of the application provides the containment system components to be tested and prescribes the types of leakage tests to be performed for post-fabrication, pre-shipment, periodic, and maintenance of the package.

In Section 4.4.1, the applicant discusses the fabrication leakage rate test to ensure that the containment system, as fabricated, provides the required level of containment. The fabrication leakage test is to demonstrate that the leakage rate of the containment system, as fabricated, does not exceed 10^{-7} ref-cm^3/s. The fabrication leakage rate test requirements are described in Section 8.1.4 of the application.

The requirements for the pre-shipment leakage rate test are specified in Section 4.4.2: the CCV assembly is tested after maintenance of the CCV assembly, to confirm the leakage rate of the containment system after maintenance, repair, or replacement of components does not exceed 10^{-7} ref-cm^3/s. The maintenance leakage rate testing and the replacement or repair activities requiring a maintenance leak rate test are further described in Section 8.2.2.2 of the application.

The requirements for the periodic leakage rate test are specified in Section 4.4.3: the periodic leakage rate test is done on the CCV assembly. This assembly is tested within 12 months prior to each shipment to confirm the leakage rate of the containment system does not exceed 10^{-7} ref-cm^3/s. The periodic leakage rate test remains valid for 1 year. The periodic leakage rate test requirements are further described in Section 8.2.2.2 of the application.

The requirements for the pre-shipment leakage rate test are specified in Section 4.4.4: each package is tested prior to shipment to confirm the containment system is properly assembled for shipment. The pre-shipment leakage rate test is performed using the gas pressure rise method in ANSI N14.5, Section A.5.2, following the steps outlined in Section 7.1.3 of the application. The pre-shipment leakage rate test requirements are further described in Section 8.2.2.3 of the application.

Based on the staff's review of the description for the leakage rate tests for Type B packages, the proposed leakage tests for the OPTIMUS-L transportation system meet the test standards found in ANSI N14.5, which the applicant has committed to follow; therefore, the staff finds the description acceptable.

4.5 Appendix Review

4.5.1 Flammable Gas Calculations / Requirements

In Section 4.5.2 of the application, the applicant addresses the fact that this package is a transuranic (TRU) waste package, and its contents present a potential risk for the introduction of flammable gases from hydrogen gas through radiolysis. The applicant also stated that limits are set to ensure there's no risk of a flammable gas mixture within the confinement region in TRU waste contents because of radiolysis or release of aerosol propellant gases. The applicant mentions that hydrogen gas generation from mechanisms other than radiolysis are insignificant and that hydrogen gas from chemical reactions is prohibited.

The applicant states that, per the operating procedure described in Section 7.1.2, a package with a total heat load exceeding 50 W must be evacuated to an oxygen content of 1% (by volume or 1 vol%) or less and backfilled with helium gas prior to shipment. This reduces the quantity of oxygen inside the CCV below the threshold at which a flammable gas mixture can develop in the CCV during the shipping period.

The applicant also states that after evacuation of the gas from the CCV cavity, the CCV cavity is monitored to assure that any oxygen potentially trapped within the contents has been evacuated below the limiting oxidant concentration (LOC). The inerting process may be skipped if the decay heat of the contents is less than 50 W; however, if the inerting process is skipped, the hydrogen concentration limit is set at a lower point. With the requirement for flammability control based on oxidant removal to below the LOC, the primary concern for potential flammability during transport of the package is from oxidant reintroduction from radiolysis of water generating oxygen.

All oxidant control requirements for the package are equated to hydrogen concentrations (based on the radiolysis of water), as the limits for any TRU waste content are based on a hydrogen concentration. This is discussed further in Section 4.5.4 in the application.

Table 4.5-1 provides a summary of the flammability limits for TRU waste content based on the initial quantity of oxygen in the CCV at package closure and radiolysis of water. For contents that are inerted at closure, the only gaseous fuel introduced is hydrogen from radiolysis. As such, the oxidant concentration is limited to 5 vol% oxygen in any confinement region in accordance with Information Notice (IN) 84-72, “Clarification of Conditions for Waste Shipments Subject to Hydrogen Gas Generation, Section (1)(b).

The applicant states that, since the initial quantity of oxygen in the system is limited to 1 vol% oxygen for greater than 50 W contents, to limit the oxygen concentration to below the 5 vol%, the quantity of oxygen added from radiolysis is limited to 4 vol%. With the added oxygen concentration limited to 4 vol%, the corresponding hydrogen concentration limit, based on the radiolysis of water, is 8 vol%, resulting in total radiolysis gases concentration limit of 12.0 vol%.

When the inerting process is skipped for TRU waste contents having a heat load of 50 W or less, the hydrogen fuel from radiolysis is limited to 5 vol% in accordance with NUREG-1609, “Standard Review Plan for Transportation Packages for Radioactive Material," Section 4.5.2.3, “Combustible Gas Generation.” Based on the radiolysis of water, the maximum oxygen introduced to the system from radiolysis is 2.5 vol%, resulting in total radiolysis gases of 7.5 vol%.

Based on a review of the description in this section and review of the calculations within Section 4.5.4.4 of the application, staff determined that inerting should be removed and that the maximum decay heat limit should be 50 W.

4.5.2 Hydrogen Concentration Calculations

The applicant stated, in Section 4.5.4, that all limits (whether the flammability control limit is based on fuel or oxidant restrictions or not) are equated to a limiting hydrogen gas concentration based on the radiolysis of water (see Table 4.5-1). It is necessary for all TRU waste contents to calculate the concentrations of hydrogen gas in the confinement layers of the TRU waste containers over time, to demonstrate compliance with the imposed limits. The hydrogen gas calculations are not restricted to the radiolysis of water only, but all hydrogenous materials that could generate hydrogen gas through radiolysis, must be considered. The applicant also stated that Section 4.5.4 of the application demonstrates their compliance in calculating hydrogen concentrations using the guidance in NUREG/CR-6673,” Hydrogen Generation in TRU Waste Transportation Packages.”

The applicant explained that G-values are specific to the contents in a given payload, based on the chemical properties of the materials (alpha, beta, gamma energy, etc.) for flammable gas generation calculations. The bounding G-values shall be used for the calculations of a given content based on the materials present, and must be selected from acceptable industry standard references, such as Appendix D of NUREG/CR-6673.

The applicant described the following types of variables/equations that were used in the calculations:

1) Radiation Based G Values - The G-value for a given material, can vary depending on the fraction of the alpha, beta, and gamma energy, when dose dependent G-values are used.

2) Temperature Adjustment - The sources for G-values typically provide the data at 70°F. Thus, these values must be adjusted to account for the temperature effect on radiolytic gas generation. This temperature effect can be found in Equation 2.2 in NUREG/CR-6673.

3) Flammable Gas Generation Rate – This variable is calculated using the effective G-value and decay heat of the contents.

In Section 4.5.4.2 of the application, the applicant introduces release rate data. The release rates used for calculating the concentration of flammable gasses are specific to the materials of the confinement layers in a given payload and for a given content are based on the confinement layers present in the TRU waste container.

Once the release rate was calculated, the effective release rate or the effective resistance can be determined. The applicant used this information to calculate hydrogen gas accumulation in Section 4.5.4.3 of the application. There are two methods mentioned in this section: the method given in NUREG/CR-6673 and what is called the simplified calculation.

The first method follows the method from NUREG/CR-6673, Section 4.2, which outlines the hydrogen mole balance equations for multiple configurations of TRU waste containers (i.e., nested leaking enclosures in a non-leaking enclosure). More specifically, Sections 4.2.2.1 and 4.2.2.5 of NUREG/CR-6673 provide the approaches for calculating the hydrogen concentration in nested leaking enclosures, with a nonleaking outer enclosure at a given time. With the equations outlined in these sections, and the content specific hydrogen limits listed in Table 4.5-1, the time to reach the hydrogen concentration limit can be calculated based on the TRU waste container confinement volumes and release rates.

The second method is known as the simplified calculation method. This simplified calculation can be used in lieu of the NUREG/CR-6673 calculation for an unquantified innermost confinement region volume. For the initial conditions of the simplified approach, it is assumed that the concentration and flow of hydrogen through the confinement layers of the TRU waste contents has reached steady state, prior to CCV closure. At steady state condition, the flow of hydrogen across all confinement layers is equal to the hydrogen generation rate. This assumption neglects the removal of nearly all hydrogen from the system during the evacuation process, prior to shipment.

The applicant performed calculations using NUREG/CR-6673 in Section 4.5.4.4 of the application, because there was a hydrogen/flammability potential. The applicant's results yielded a hydrogen concentration limit greater than the prescribed 5%. As a result, the applicant introduces package inerting prior to shipment in order to prevent combustion.

It is common practice to use a test/experiment to demonstrate the effectiveness of inerting, and this cannot be accomplished through an analysis or set of calculations, as presented by the applicant. Furthermore, the test apparatus from Savannah River National Laboratory used by the applicant does not match with the OPTIMUS-L package; thus, staff is not able to accept a test in this case. In addition, the applicant used 8% hydrogen to calculate “the safe shipping period,” which is greater than the 5% prescribed in NUREG-1609.

In order to allow inerting, the staff requires, at a minimum, a full demonstration that the inerting process will prevent the development of flammable gas mixtures in any confined area of the package throughout the entire shipment period and a detailed evaluation to prove that there are no flammable gas mixtures (considering the worst-case concentrations) during shipment. In addition, the applicant needs to explain how the inerting gas is effectively introduced to all confined areas within the containment system of the package and, as such, is in uniform concentration throughout the CCV. Finally, the concentrations of combustible gases need to be able to be quantitatively analyzed. Therefore, the staff determined it could not reach a finding of reasonable assurance to accept inerting for the OPTIMUS-L package with the limited information presented by the applicant. As a consequence, the CoC includes a condition preventing inerting and limiting waste, authorized for shipment, with a decay heat below 50 watts.

4.6 Evaluation Findings

The staff has reviewed the applicant’s evaluation of the containment system of the OPTIMUS-L under NCT and concludes that the package is designed, constructed, and prepared for shipment such that under the tests specified in 10 CFR 71.71, “Normal Conditions of Transport,” the package satisfies the containment requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) for normal conditions of transport with no dependence on filters or a mechanical cooling system. The staff has reviewed the applicant’s description and evaluated the containment system based on the necessary codes and standards. Staff finds that the description satisfies the containment requirements for codes and standards found in 10 CFR 71.31(c).

The staff has reviewed the applicant’s description and evaluated the description regarding the containment system being securely closed. Staff finds that the description satisfies the requirements for a containment being securely closed found in 10 CFR 71.43(c).

The staff has reviewed the applicant’s evaluation of the containment system under HAC and concludes that the package satisfies the containment requirements of 10 CFR 71.51(a)(2) for HAC, with no dependence on filters or a mechanical cooling system.

Based on staff review of the statements and representations in the application, the staff concludes that the information presented regarding hydrogen generation meets 10 CFR 71.43(d) given that the maximum decay heat is 50 W and inerting is prohibited.

5.0 SHIELDING EVALUATION

The purpose of this evaluation is to verify that the shielding design of the OPTIMUS-L transportation package meets the dose rate limits set forth in 10 CFR Part 71.47(b) and 71.51(a)(2) under NCT and HAC under exclusive use. This SER documents the staff’s review of the shielding analysis for the OPTIMUS-L included in Chapters 1, 5, 7, and 8 of the application.

5.1 Description of the Shielding Design

5.1.1 Packaging Design Features

The OPTIMUS-L is made of two primary components, the cask containment vessel (CCV) and the protective outer packaging (OP). The CCV is made of stainless-steel shells and has a bolted closure to provide leak tight containment. The CCV has a bottom plate for support. The OP consists of a base and a lid bolted to enclose the CCV. The function of the OP is to limit the impact of crush and force to the CCV during NCT and HAC and isolate the CCV during fire. It is filled with foam which is used as the impact limiting material. For loads that require additional shielding, the Shield Insert Assembly (SIA) with 1-inch or 2.25 inches thickness can be inserted into the CCV. Figure 1.1-1 of the application depicts the components of the OPTIMUS-L packaging.

5.1.2 Summary Table of Maximum Radiation Levels

The applicant demonstrated that the package design meets the regulatory dose rate requirements of 10 CFR 71.47 and 71.51(a)(2) by performing a package evaluation which satisfied the requirements of 10 CFR 71.35(a).

The package will only be transported by exclusive use, so the applicant did not evaluate the transportation index.

The applicant determined the maximum dose rate under NCT at the trailer surface, 2-meters from the trailer surface and occupied cab position dose rates, and 1 meter from the package surface under HAC. Packages are transported in a vertical orientation, and the applicant evaluated the side dose rates under NCT. Under HAC the applicant also evaluated the dose rate at the side because this is the limiting surface.

As the OPTIMUS-L is designed to ship waste material, the isotope inventory from TRU wastes that would be loaded in any particular package is variable. To comply with regulatory requirements on content specifications and dose rate, the applicant has loading requirements to determine the amount of each isotope and then sum the dose rate contribution from each nuclide. The sum cannot exceed 90% of the regulatory limit for every regulated location. This procedure is in Attachment 7.5-1 of the application and discussed in Section 5.4.2 of this SER.

Because of the nature of the contents and how they are specified, it is less meaningful for the applicant to have a summary table of maximum dose rates, as the procedure for specifying contents is such that no dose rate will exceed 90% of the regulatory limit. However, as an example, the applicant evaluated the maximum dose rates at the various locations for isotopes Co-60 and Cf-252 when loaded to the limit using the procedure outlined in Attachment 7.5-1 of the application and presented the maximum radiation levels in Tables 5.1-2 and Table 5.1-4 of the application.

These values are in compliance with the regulatory dose rate limits in 10 CFR Part 71.47(b) and 71.51(a)(2) under NCT and HAC, respectively, and are consistent with the maximum activity values for these nuclides in Table 1.2-3 of the application.

5.2 Radioactive Materials and Source Terms

5.2.1 Source Term Calculation Methods

The applicant used the ORIGEN module of the SCALE code package, version 6.2, to determine the source spectra of each evaluated nuclide. The secondary particles (Bremsstrahlung and α, n) are considered by the ORIGEN code in determining the source spectra. The staff found the use of the ORIGEN code acceptable for this evaluation as ORIGEN is considered the industry standard and is recommended for use by NUREG/CR-6802, “Recommendations for Shielding Evaluations for Transport and Storage Packages,” March 2003.

The ORIGEN code has the ability to group the neutron and gamma spectra from the individual isotopes into a pre-defined energy group structure which the applicant has defined in Table 5.2-1 for gammas and 5.2-2 for neutrons. The applicant used the upper bound energy from the group structure to perform dose rate evaluations to represent gammas and neutrons that fall into each group which is conservative as most all gammas and neutrons will be represented by a slightly higher energy and acceptable to the staff.

5.2.2 Source Geometry

The applicant modeled the source as a point source at the center of the CCV cavity under NCT. There are no required shoring mechanisms that would maintain the source at the center of the cavity required during shipment. However, since the point source assumption is very conservative as it does not take credit for self-shielding or spatial distribution of the source, the staff took this into consideration (as well as other conservative assumptions such as assuming all source term energies are at the maximum energy for each energy bin) when making the determination that assuming the point source is at the center of the cavity is acceptable under NCT.

Under HAC the applicant modeled the point source next to the CCV wall closest to the detector. Having the source next to the CCV wall is very conservative as it minimizes distance to the detector. Given the additional movement of contents during HAC, the staff found it appropriate to relocate the source closest to the wall of the CCV cavity. The staff found the assumptions for the source geometry to be conservative and acceptable.

5.3 Shielding Model and Model Specifications

The applicant states in Section 5.3.1 of the application that it used nominal packaging dimensions within the shielding models. This is a non-conservative assumption because actual thicknesses of shielding components may be less than this when accounting for tolerances. However, given the conservative nature of the analysis, e.g., point source approximation and using maximum energy from the grouped energy bins to represent the source, the staff found the use of nominal dimensions acceptable for this package.

The MCNP models of the OPTIMUS-L under NCT and HAC are shown in Figure 5.3-1 of the application which shows the dimensions of all significant axial and radial shield thicknesses. The staff reviewed these dimensions and found that they are consistent with the package drawings.

The applicant did not model some additional features and attachments such as tie-downs, or support plates. The applicant modeled all bolts and inserts in the CCV as the same material as CCV components.

These modeling assumptions and simplifications were found acceptable by the staff based on other conservative assumptions such as assuming a point source and using maximum energy of each energy group. The minimum foam thickness is modeled for NCT only based on minimum polyurethane foam of the OP lid and polyurethane material composition provided in Table 5.3-4 of the application. This is conservative and acceptable to the staff.
The secondary packaging (such as drums or liners) prevents contamination between the contents and packaging and is used to prevent rapid movement of content during shipment.

Although this is credited to some degree in the dose rate analysis by keeping the contents centered within the cavity during NCT, it may also affect the external dose rate of the package as the material may serve as additional shielding but is not considered within the shielding model. Although these components will likely be present, they are not required as part of the packaging, and neglecting them is conservative, and appropriate and is therefore acceptable to the staff.

The applicant modeled the OP stainless steel shell as a single stainless-steel shell with minimum combined stainless steel in all directions. The applicant removed the foam for the HAC scenario to bound HAC effects on the package (such as loss of material during HAC fire), otherwise the HAC model is identical to the NCT model. Both NCT and HAC models include a crush depth for OP outer radial layer. Chapter 2 of the application shows that maximum crush depth off or the side drop, and for the end drop, during the HAC tests (Table 2.7-6 and 2.7-2 of the application). The staff found that this is conservative and acceptable.

The applicant calculated all dose rates from the side of the package because the top and bottom of the package have thicker shielding. The staff found this acceptable.

5.3.1 Material Properties

The only materials used in the MCNP model is stainless steel Type 304 and polyurethane foam. The compositions and densities of these two materials assumed by the applicant are in Table 5.3-4 of the application. The staff reviewed this information and found it consistent with the open literature, PNNL-15870 Rev. 1, “Compendium of Material Composition Data for Radiation Transport Modeling,” March 4, 2011, and acceptable for use within the OPTIMUS-L shielding model.

5.4 Shielding Evaluation

5.4.1 Methods

The applicant used the MCNP-6 code with ENDF/B-VII.1 nuclear data library to calculate dose rate for each individual energy group at the desired locations. Per the guidance in NUREG-2216 Section 5.4.4.1, the staff found that the MCNP code, with the latest nuclear data, is acceptable for the shielding evaluations of the OPTIMUS-L package with the requested contents.

5.4.1 Fluence-Rate-to-Radiation-Level Conversion Factors

The MCNP code calculates a fluence per emitted particle. Then this fluence is converted into a dose rate by using fluence-to-dose rate conversion factors to arrive at the dose rate per emitted particle. The applicant used the fluence-to-dose-rate conversion factors recommended by NUREG-2216, the 1977 ANS/ANSI-6.1.1 standard and are therefore acceptable to the staff. The applicant added an additional two sigma to the fluence calculated by MCNP to account for the statistical uncertainty of the monte Carlo code. The staff found it to be a conservative and acceptable way to account for this uncertainty.

5.4.1 Dose Rate Results

5.4.2 External Radiation Levels

The MCNP code uses “tallies” when determining particle flux at a location of interest. The tally cell represents the volume in space that the particles are collected. Tally cells need to be small enough to reasonably represent a maximum dose (versus an average). The cell tallies specified by the applicant for the OPTIMUS-L are approximately 100 cm^3 (10 cm wide x 10 cm tall x 1 cm thick) formed at the side of package with a 10-degree inner angle arc.

The staff used its judgment and consideration for the conservatism within the source term modeling (e.g., point source and energy bins represented as upper values) and found that the size of the tally for the dose rate calculations is acceptable with these considerations. The location of the tally cells are based on the locations specified in 10 CFR Part 71, e.g., surface, 2 meters, and 1 meter under HAC.

The applicant placed the surface tally at the surface of the deformed OP. For the 2-meter tally, it is placed 2 meters from the trailer surface assuming a 100-inch-wide trailer this is acceptable because the regulation in 10 CFR Part 71.47(b) has dose rate limits defined from the vertical planes projected by the outer edges of the vehicle for a flat-bed trailer. The applicant assumed a trailer width of 102 inches.

The staff found that the size of the trailer assumed by the applicant is a reasonable width based on the standard width of a trailer and standard width of U.S. roads. When the package is placed near the edge of the trailer, there’s less distance than when it is in the center which is why the applicant has different loading tables for these two configurations. For HAC the applicant located the tallies at 1 meter from the package surface. This is appropriate and acceptable to the staff as dose rate limits under HAC in 10 CFR Part 71.51(a)(2) are defined at 1 meter from the package.

The results of the applicant’s calculations for dose rate per particle emitted at the various regulatory locations are summarized in Tables 5.4-2 and 5.4-3 of the application for gammas and neutrons, respectively, for a package at the edge of the trailer. For a centered package with or without an SIA these results are in Tables 5.5-3 through 5.5-8 of the application. For the centered cask evaluation, the applicant has credited the additional distance to the detector. The evaluations crediting this additional distance to the detector and additional shielding provided by either of the two SIAs allow for higher loading of radioactive material.

The applicant determines the dose rate per curie of each isotope by summing the dose rate contribution of all of the gammas and neutrons within each energy bin as defined in Tables 5.4-2, 5.4-3 and 5.5-3 through 5.5-8 of the application. This is shown in Equation 3 in Section 5.4.1.3 of the application. The results of the applicant’s evaluations of the dose rate per curie of each individual isotope are in presented in the application for a cask with no SIA at the edge of the trailer, and for a centered cask with or without an SIA.

The loading procedure described in Section 7.5-1 of the application (and referenced in the CoC) requires the package user to determine the amount of each isotope in the package, then use Tables 7.5-1 or 7.5-2 of the application to add up the dose rate contribution from each isotope. If the resultant dose is less than 90% of the regulatory dose rate limit for all locations (surface, 2 meter and HAC at 1 meter) then the content is acceptable for loading. The staff found that this is conservative and is an acceptable demonstration that the OPTIMUS-L meets regulatory dose rate limits in 10 CFR 71.47(b) and 10 CFR 71.51(a)(2).

As required by the loading procedure in Section 7.5-1 of the application, the minimum distance from the driver cab to the centered package is 15 feet to ensure that driver cab location does not exceed regulatory limits. If dose rates exceed 50 percent of regulatory limits, the distance from the driver cab to the centered package is required to increase to at least 20 feet. The staff found that this demonstrates that the OPTIMUS-L meets the dose rate requirement in 10 CFR 71.47(b)(4) pertaining to the normally occupied space should the OPTIMUS-L not be transported by a private carrier with exposed personnel wearing radiation dosimetry devices in conformance with 10 CFR 20.1502.

5.5 Evaluation Findings

The staff concludes that the shielding design of the OPTIMUS-L when used as described in the application is in compliance with 10 CFR Part 71 and that the applicable design and acceptance criteria have been satisfied. The staff has reasonable assurance that the OPTIMUS-L design will provide safe transportation of TRU waste. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes, and standards, the applicant's analysis, and responses to requests for additional information, and acceptable engineering practices. Based on its review of the statements and representations provided in the application, the staff has reasonable assurance that the shielding evaluation is consistent with the appropriate codes and standards for shielding analyses and NRC guidance. Therefore, the staff finds that the package design and contents satisfy the dose rate limits in 10 CFR Part 71.

6.0 CRITICALITY EVALUATION

The OPTImal Modular Universal Shipping cask for Low activity contents (OPTIMUS-L) is a Type B(U)F radioactive material transportation package design. NAC International (NAC, the applicant) submitted an application for a Certificate of Compliance under NRC regulations 10 CFR Part 71. NAC provided the criticality safety evaluation in Section 6 of its application. The staff performed a review of the application following the guidance provided in NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material" to prepare its Safety Evaluation Report (SER). The staff's evaluation of the applicant's criticality safety evaluation follows.

6.1 Description of Criticality Design

The OPTIMUS-L package is designed to transport Type B quantities of transuranic (TRU) waste and irradiated fuel waste material. The applicant provided a criticality evaluation of the OPTIMUS-L package. The package achieves the criticality safety goal by limiting the amount of fissile, moderator, and reflector material that will be contained within the package. The neutron multiplication factor (k-effective, or k-eff) will be less than 0.95 during all normal and accident conditions. Details of the staff's evaluation of the package design features and summary follows.

6.1.1 Packaging Design Features

The OPTIMUS-L consists of a Cask Containment Vessel (CCV), a CCV bottom support plate, and an Outer Packaging (OP) assembly which is a foam impact limiter shell that fully houses the CCV. The OPTIMUS-L has an optional configuration that includes a Shield Insert Assembly (SIA). The SIA was not considered within the criticality evaluations because the applicant stated that it would restrict the volume of the CCV and would reduce reactivity, therefore neglecting it is conservative. The staff found this assertion acceptable because this modeling approach increases the amount of moderator and reflector in the package.

The staff reviewed the general information provided in Section 1 of the application. The applicant provided drawings of the package. The staff reviewed the drawings to determine that they are sufficiently detailed to perform a criticality safety review. The staff reviewed the following drawings:

- Packaging Assembly OPTIMUS-L, Drawing 502
- CCV Body Weldment OPTIMUS, Drawing 511
- Outer Packaging Assembly, Drawing 540
- Outer Packaging Base, Drawing 541
- Outer Packaging Lid, Drawing 542

With respect to the criticality evaluation, the staff found that the description of the packaging is described in sufficient detail to provide adequate information for its evaluation and that the description includes types and dimensions of materials of construction. Therefore, the staff found that the application meets the requirements of 10 CFR 71.31(a)(1) and 10 CFR 71.33(a)(5) as it pertains to criticality safety.

The applicant identified codes, standards, and regulations applicable to the criticality design as required in 10 CFR 71.31(c) in Section 1.2.1.10 of the application. These pertain to the structural design for the package. Additional codes and standards referenced related to the criticality safety design of the package also include ANSI/ANS 8-1 and 8-15 and ASME SA-20/SA-20M. Using the minimum critical masses given in ANSI/ANS 8-1 and 8-15 for various fissile/fissionable materials, the applicant developed fissile gram equivalent (FGE) limits for Pu-239 as specified in Table 1.2-1. The ASME SA-20/SA-20M standards are used for determining the permissible variation in specified plate material.

6.1.2 Summary Table of Criticality Evaluations

The applicant provided summary tables of the criticality evaluations in Tables 6.1-1 and 6.1-2 of the application. Table 6.1-1 of the application summarizes the results of the criticality evaluations for a single package and an array of packages under normal conditions of transport (NCT) and hypothetical accident conditions (HAC).

The quantities of the fissile materials in the TRU waste are given as FGE contents in Table 1.2-1 of the application, irradiated fuel waste (called fissile equivalent mass (FEM) contents by the applicant) is in Table 1.2-2 of the application. Table 6.1-2 of the application summarizes the results of the criticality evaluations performed by the applicant for a single package and an array of packages under NCT and HAC.

FGE represents the amount of Pu-239 or U-235 that would produce the equivalent k-eff as that determined for the fissile material in the container (assuming all containers are in an optimally moderated infinite array) as noted in Table 1.2-1 of the application. The procedure for determining the FGE for each nuclide is discussed in Section 6.2.1 of this SER. The procedures are consistent with the technical report WIPP/DOE – 069 Revision 3, “TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant.” The FEM is a single parameter limit for criticality safety control of system and used by the applicant for the irradiated fuel waste content and represents uranium with a maximum enrichment of 0.9%.

The staff verified that the tables include the maximum value of k-eff. These values include two standard deviations. The data presented in these table show that the maximum k-eff values are 0.93911, 0.93779, 0.93921, 0.93924, 0.93516, 0.93763 and 0.94002 for the FGE-1, 2a, 2b, 2c, 3, 5, and FEM contents respectively. The characteristics of the FGE contents are summarized in Table 6-1 of this SER. The summary table shows that all configurations have a k-eff below the upper subcriticality limit (USL) of 0.93930 for the contents FGE-1, 2a, 2b, 2c, 5, and below the USL of 0.9368 for content FGE-3 and below the USL of 0.94140 for the FEM contents.

For each content class the applicant performed a criticality safety evaluation for a single flooded package, an array of 50 dry packages under NCT, and an array of 20 flooded packages under HAC. The applicant performed analyses to identify the optimally moderated condition, which includes consideration of special reflectors, as discussed in Section 6.3.4 of this SER and included close full reflection of the single package and arrays of 50 cm full reflection of water.

The amount of water assumed in the reflector is more conservative compared to the 20 cm water reflector recommended by NUREG/CR-5661, “Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages,” April 1997, for full reflection. The staff found that the results of the summary table of criticality evaluations show that the package meets the requirements of 10 CFR 71.55(b), (d) and €, 10 CFR 71.59(a)(1) and 10 CFR 71.50(a)(2).

6.1.3 Criticality Safety Index (CSI)

Section 6.1.3 of the application specifies that the CSI of the package is 5.0. This is based on an array of 50 packages under NCT is subcritical, while an array of 20 packages under HAC is subcritical. The staff found that the CSI was appropriately determined per 10 CFR 71.59(b) and by specifying the CSI in accordance with 10 CFR 71.59 the applicant meets the requirement of 10 CFR 71.35(b). The staff found that the applicant meets 10 CFR 71.59(a)(3) because the value of N is not less than 0.5.

6.2 Fissile Material Contents

The OPTIMUS-L is designed to transport two different kinds of fissile material: (1) transuranic (TRU) waste and (2) irradiated fuel waste.

6.2.1 Transuranic Waste

The applicant defines TRU waste in Section 1.2.2.1 of the application as “intermediate-level radioactive waste exposed to alpha radiation or containing long-lived radionuclides in concentrations requiring isolation and containment for periods beyond several hundred years.” With respect to the fissile content of the allowable TRU waste, the applicant limits the quantities of fissile materials in terms of fissile gram equivalent (FGE).

The applicant discusses the procedure for determining FGE in Section 6.3-4 of the application where it determines conversion factors using a ratio of the single parameter critical mass limits as provided in ANSI/ANS standards 8.1 and 8.15 (2014) to that of Pu-239. The list of allowable fissile nuclides and the associated conversion factors are listed in Note (a) to Table 6.3-4 of the application. This table and the procedure for using the conversion factors to determine total FGE of the contents is acceptable and is included in Section 7, “Package Operations,” of the application.

The staff reviewed this information and has determined that it is consistent with the common practices used in the industry for treating TRU wastes that are typically a mixture of various fissile and fissionable materials. The transportation systems for TRU waste approved by the

NRC in Docket no. 71-9212, 71-9218, 72-9279 are evidence of applications of this approach. However, the conversion factors for the OPTIMUS-L were derived using values provided in the 2014 edition of the standard which is an updated version of the standard. On this basis, the staff found this approach to be acceptable for limiting contents for other fissile nuclides listed in Note (a) to Table 6.3.4a of the application.

These conversion factors are based on the minimum critical mass of the various fissile and fissionable nuclides as compared to that of Pu-239 in various mixture forms and with optimal moderations. For the main fissile nuclides, Pu-239, U-235 and U-233 the applicant used the minimum fissile mass from ANSI/ANS-8.1-2014 from Section 5.2 for an aqueous mixture. For the other nuclides, the applicant used the minimum subcritical mass from ANS/ANSI-8.15-2014 for a metal-water mixture and where data was not available for this configuration the applicant used subcritical mass limits for the water reflected spherical metal system.

Although the conditions needed to maximize reactivity are not necessarily the same for all nuclides as they are for Pu-239, the staff found that the use of the conversion factor method was conservative as it is unlikely to have critical mass quantities for these other nuclides because Pu-239 has the lowest critical mass in comparison with all other fissile and fissionable materials as presented in the TRU wastes to be shipped by this packaging system.

The FGE limits are stated in Table 1.2-1 of the application and is repeated in Table 6-1 below.

Table 6-1: TRU Waste FGE Limits

Configuration	**FGE Criticality Configuration Description**			**FGE limit Pu-239(g) (U-235 (g))**
	Machine Compacted	**Weight % Special Reflector**	**Minimum Pu-240 Credit**	
FGE-1		≤ 1		340 (538)
FGE-2a		≤ 1	≥ 5 g	350 (544)
FGE-2b		≤ 1	≥ 15 g	375 (583)
FGE-2c		≤ 1	≥ 25 g	395 (614)
FGE-3		> 1		121 (188)
FGE-5	X	≤ 1		250 (388)

For contents FGE-2a, FGE-2b, and FGE-2c, Pu-240 is used as a neutron absorber to reduce reactivity, and therefore a minimum mass is required. Sections 1.2.2.1 and 1.2.2.2 of the application state that free liquids shall not exceed 1% of the cavity volume.

However, per Note 2 to Table 1.2-1 of the application, for the TRU waste contents FGE-1, 2 and 3, materials with a hydrogen density up to that of water (0.1117 g/cm^3) are allowed in any amount and materials with a hydrogen density up to that of polyethylene (0.1336 g/cm^3) may not exceed 15% of the total contents by volume. Per Note 3 to Table 1.2-1 of the application, special reflector material is defined as beryllium, beryllium oxide, carbon (graphite), heavy water, magnesium oxide, and depleted uranium.

6.2.2 Irradiated Fuel Waste

The applicant defines irradiated fuel waste in Section 1.2.2.2 of the application as low enriched uranium (LEU) fuel and metal structural components. With respect to the fissile content of the irradiated fuel waste, the applicant has used the terminology fissile equivalent mass (FEM) to represent this content within the application.

The amount of FEM allowed for this content is summarized in Table 1.2-2 of the application to include ≤ 1% weight percent special reflectors, ≤ 0.9% U-235 enrichment, with a uranium mass limit of 2500 lbs (1134 kg). Similar to the TRU wastes, special reflectors are defined as beryllium, beryllium oxide, carbon (graphite), heavy water, magnesium oxide, and depleted uranium.

6.2.3 Fissile Material Content Conclusions

The staff reviewed the application and found that the applicant has defined adequately the type, maximum quantity, and chemical and physical forms of the fissile materials. On this basis, the staff found that this meets the requirements of 10 CFR 71.31(a)(1), 10 CFR 71.33(b)(1), 10 CFR 71.33(b)(2) and 10 CFR 71.33(b)(3).

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

To account for the damage to the package under HAC, the applicant modeled 35% of the top, bottom and side thicknesses in the array spacing. The staff reviewed Section 2 (structural evaluation) and 3 (thermal evaluation) of the application to determine the effects of the normal conditions of transport and hypothetical accident conditions on the packaging and its contents and determined that nominal dimensions for NCT is acceptable and the crush bounds the stated effects on the packaging under HAC.

The staff reviewed the data presented in Tables 6.3-1, 6.3-2, 6.3-3, 6.3-3A, and 6.3-3B of the application for the dimensions of the packaging that are used in the criticality safety analyses. The staff found that they are consistent with Licensing Drawings 502, 511, 540, 541, and 542. Therefore, the staff found that the geometric data of the package used in the criticality safety analyses is acceptable.

6.3.2 Material Properties

The staff verified that the applicant provided the appropriate composition for all packaging materials used in the criticality safety models of the packaging and contents. The applicant provided this information in Tables 6.3.4 and 6.3.5 of the application. The applicant states that the material properties were from Pacific Northwest National Laboratory, “Compendium of Material Composition Data for Radiation Transport Modeling,” PNNL-15870 Rev. 1, 2011. The staff verified that these compositions are consistent with this reference and found them acceptable.

The applicant did not include the foam within the impact limiters in the model, therefore there are no materials in the packaging that need to be adjusted to be consistent with the package under HAC because there are no materials other than the foam used in the packaging that will change their form. The assumption used in the HAC model for the package is consistent with the damaged conditions and meets the regulatory requirements of 10 CFR 71.55(e).

6.3.2.1 Moderator Materials

6.3.2.1.1 TRU Waste

The applicant chose polyethylene as the bounding moderator material based on the studies performed in SAIC, “Reactivity Effects of Moderator and Reflector Materials on a Finite Plutonium System,” SAIC-1322-001 Rev. 1, 2004. This report contains a study that determined that this is the most effective moderator for TRU waste material. The staff reviewed this report and determined that this assumption is acceptable.

In addition, the applicant chose a polyethylene packing fraction of 15% for non-machine compacted waste (contents FGE-1, 2a, 2b, 2c, and 3 and FEM) based on a study performed in Washington TRU Solutions LLC, “Test Plan to determine the TRU Waste Polyethylene Packing Fraction,” WP 08-PT.09, Rev. 0, 2003. The staff reviewed this report and found that it was adequate for justifying that 15% packing fraction for polyethylene was acceptable. For machine compacted waste, the applicant used 100% packing fraction. The staff found this acceptable.

The OPTIMUS-L is allowed hydrogenous material up to the hydrogen density of water (0.1117 g/cm^3) for contents FGE-1, 2a, 2b, 2c and 3. For baseline studies, and due to the restriction that free-standing liquid cannot exceed 1%, the applicant included 1% water within the criticality safety evaluation, equal to 6372 g, as part of the moderator for contents FGE-1, 2a, 2b, 2c and 5 and FEM. The supporting analyses in the application are all based on this assumption.

To justify hydrogenous material up to the same hydrogen density as water, the applicant performed a sensitivity study where it used the flooded conditions (15% polyethylene, 84% water and 1% beryllium) in the NCT array. This differs from the NCT array with 1% water in that the applicant modeled the nominal OP dimensions for array spacing versus the more conservative HAC array spacing. The staff found this to be acceptable and appropriate for justifying the inclusion of hydrogenous material up to the hydrogen density of water.

For FGE-3, the fissile mass is smaller and therefore the volume of the fissile/moderator shape is smaller and optimum moderation is achieved before the entire 6372 g of water is added, therefore the applicant has modeled this content with the amount of water moderation that achieves optimal moderation. The applicant does not include water in the moderation of FGE-5 as it uses 100% polyethylene. As stated previously, the staff determined that this material provides moderation better than water does for TRU waste.

Contents FGE-1, 2a, 2b, 2c, and 5 are allowed up to 1% special reflector material and therefore the applicant has included 1% beryllium in the moderator material for FGE-1, 2a, 2b, and 2c. For content FGE-3, the allowable amount of special reflector material is more than 1%. The applicant also represented the special reflector material as 1% beryllium for these cases.

To justify this, it performed a sensitivity study, discussed in Section 6.6.2.9 of the application, varying the amount of beryllium. This study shows that at low quantities of beryllium (1-5 vol%) the effect on k-eff from varying beryllium in the moderator is insignificant, with a slight decrease. As the beryllium content is increased further, k-eff decreases significantly. The staff found that this demonstrates that the assumption of 1% beryllium for this content is appropriate. The applicant did not model beryllium present in the moderator material for content FGE-5. Also discussed in Section 6.6.2.9 of the application, the applicant performed a sensitivity study including beryllium and showed that the results are essentially the same justifying neglecting it in the criticality safety evaluations. The staff found this acceptable.

For single package and array conditions where the package is flooded the applicant flooded the remaining space with water meaning that contents FGE-1, 2a, 2b, 2c and 3 are modeled with 15% polyethylene (CH_2), 84% water (H_2O), at 1% beryllium by volume.

The volume percentages stated above for the FGE limits do not account for the volume of the Pu-239, however the volume for the largest amount of Pu-239 as compared to the volume of the cavity is very small, and based on the calculations being very conservative (e.g., neglecting secondary containers, also the likelihood of the waste forming the most reactive content and geometry is very small) the staff found this to be an acceptable simplification.

6.3.2.1.2 Irradiated Fuel Waste

For the irradiated fuel waste content, the applicant modeled a moderator of 15% polyethylene, 1% beryllium, and 1% H_2O. The applicant represented this material as both a homogenous mixture and a heterogeneous array of spherical and cylindrical pellets. For the homogenous model, because the FEM takes up a larger volume of the cavity as compared to the FGE content, the applicant accounted for this volume by subtracting it from the allowable H_2O volume. The heterogeneous configuration is the bounding condition and the therefore the 15% polyethylene, 1% beryllium, and 1% H_2O volume fractions are preserved. The staff found the approximation acceptable for the homogenous content as the heterogenous content is the most limiting.

6.3.2.2 Reflector Materials

The applicant modeled the remaining cavity volume that is not taken up by the fissile/moderator volume with material to act as a reflector.

6.3.2.2.1 TRU Waste

For the TRU waste content, the reflector material is made up of the same material as the moderator minus the fissile material, i.e., polyethylene at 15% packing fraction, 1% beryllium, and the remaining volume fraction composing of water for single package and HAC array conditions. For content FGE-3, this content is allowed > 1% special reflectors therefore the applicant modeled the reflector area around the fissile material as 100% Be.

As discussed in Section 6.3.5 of this SER the staff performed independent calculations that show that assuming 100% Be in the reflector region is conservative versus various mixtures with water and polyethylene.

For content FGE-5 the applicant modeled 99% polyethylene and 1% Be by volume. Based on the allowable contents and the information in SAIC-1322-001 Rev. 1, 2004, the staff found this acceptable because these materials were determined to be most effective in reflecting neutrons to increase the k-eff of the package.

6.3.2.2.2 Irradiated Fuel Waste

For the irradiated fuel waste content, the applicant assumed the reflector material is made up of the same material as the moderator minus the fissile material, i.e., polyethylene at 15% packing fraction, 1% beryllium, and the remaining volume fraction composing of water for single package and HAC array conditions. The staff also found this acceptable for the same reasons as discussed above for other contents.

6.3.3 Computer Codes and Cross Section Libraries

The applicant used the Monte Carlo N-Particle Code, Version 6 (MCNP6). The applicant uses the ENDF/B-VII.1 and ENDF/B-VII.0 continuous energy nuclear data set. Per the guidance in NUREG-2216 Section 6.4.3.3, the staff found that the MCNP code, with the latest nuclear data, is appropriate for the criticality evaluations of the OPTIMUS-L package with the requested contents.

The applicant provided representative input and output files. The staff reviewed a sample of these files and verified that the neutron multiplication factors, k-eff, from the output files agree with those reported in the evaluation and that the calculations have properly converged. For the FEM files, as the staff did not perform a confirmatory evaluation, the staff also verified that the information regarding the amount of fissile material, density, and enrichment, were properly represented in the input files.

6.3.4 Demonstration of Maximum Reactivity

To determine the maximum amount of contents that can be transported for the various content types, the applicant determined the USL and then varied the allowable amount of material such that the resultant k-eff including the standard deviation is below the USL. For the TRU waste (FGE content) the applicant determines the allowable mass and for the irradiated fuel waste (FEM content), the applicant determines the maximum allowable enrichment.

In all cases for the TRU waste (FGE content) the applicant represents the fissile/moderator mixture as a sphere. This is the most reactive geometry as it minimizes neutron leakage. The applicant determines the most reactive H/Pu-239 ratio by varying the size of the sphere while holding the fissile mass constant. The applicant also varies the positioning of this sphere within the CCV volume to determine the position of maximum reactivity.

For the irradiated fuel waste (FEM content), the applicant modeled three content geometries: (1) homogenous fissile sphere/cylinder, (2) heterogeneous cylindrical particle lattice in a cylindrical arrangement, and (3) a heterogeneous spherical particle lattice in a cylindrical arrangement. The staff found that this is a conservative representation of this material as these types of geometries increase reactivity and it is unlikely for the waste to achieve these geometries.

For a single package and an array of packages under HAC, the applicant modeled the package as flooded with varying the density of the water in the flooded region to determine the maximum reactivity.

For the OP region for the single package analyses the applicant performed sensitivity studies on the amount of flooding in this region and found that there was a slight difference for the FGE-1 and FGE-2b contents and reported the maximum value from this study in the single package results.

For package under HAC, the applicant performed a sensitivity study with flooding the foam region of the OP and interspersed between packages. The applicant neglected the foam within the OP impact limiter. This material is made of polyurethane and could act as a reflector or moderator, however the applicant found that flooding conditions external to the package (including OP) and between packages in an array calculation have very little effect on reactivity and therefore the staff found that neglecting the foam is acceptable as the various calculations examining the flooding of the OP are reasonably representative of including the foam within the evaluations.

Based on the above statements, the staff found that the applicant's analysis demonstrated that it has identified the maximum reactivity in accordance with the regulatory requirements of 10 CFR 71.55(b).

6.3.5 Confirmatory Analysis

The staff performed independent calculations to confirm some of the applicant's results. The staff preformed calculations with the CSAS6 criticality sequence of the SCALE 6.2.3 code package. SCALE 6.2.3 was developed by Oak Ridge National Laboratory for use in criticality and shielding analyses. The CSAS6 sequence is a criticality sequence that uses KENO-VI geometry and multi-group cross sections. The staff used the 252-group cross section library derived from ENDF-VII data.

The staff's calculations verified that the single flooded package was subcritical for the FGE-1, 2a, 2b, 2c, 3 and 5 configurations. The staff also performed sensitivity studies varying the amount of beryllium and confirmed the applicant's results that 1% is an appropriate assumption that produces maximum reactivity for the moderator material. The staff performed additional sensitivity studies on the FGE-3 configuration replacing the reflector material with various mixtures of polyethylene and water which confirmed that modeling 100% beryllium in the reflector produces the highest reactivity and is conservative.

As stated in Section 6.3.2.1.1 of this SER, hydrogenous material up to the hydrogen density of water is allowed in for FGE-1, 2a, 2b, 2c and 3 and the applicant justified this by performing a sensitivity study where the flooded condition was modeled in an array of packages under NCT to determine this was subcritical. There is some neutron absorption in water. This may contribute to a reduction in k-eff in comparison to another material with the same hydrogen density that does not experience absorption.

Because these allowable materials do not contain significant amount of water (as only 1% of free-standing liquids are allowed) they would consist of some other hydrogenous material that may not have the same parasitic neutron absorption as water. The staff performed a sensitivity study replacing the water with polyethylene, but reduced the density of the polyethylene, so that it would match the hydrogen density of water and found that although k-eff did increase that it was very insignificant, less than 0.2%, and therefore staff found that this provides additional assurance that the applicant's assumptions are appropriate for justifying the allowable hydrogenous material. Based on the results of its confirmatory calculations, the staff found that the applicant's criticality safety analyses are conservative and hence acceptable.

The staff did not perform an independent evaluation of the FEM material due to the additional complexity associated with that model. However, as stated in Section 6.3.3 of this SER, the staff did perform a more detailed review of the applicant's MCNP input files provided for this content.

6.4 Single Package Evaluation

The staff verified that the applicant's criticality safety evaluation demonstrated that a single package is subcritical under both NCT and HAC. The applicant models the inside of the cask flooded with water when performing calculations for the single package. The applicant performed a sensitivity study of the flooding in the OP region to determine the optimal flooding and did not find that there was a significant effect but reported the highest value of k-eff from this study regardless.

The applicant surrounded the single package with at least 20 inches (50.8 cm) of close water reflection. The staff found that this assumption is more conservative than the recommendation of NUREG/CR-5661, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages" and therefore acceptable.

The applicant modeled the most reactive configuration consistent with the condition of the package and the chemical and physical form of the contents as discussed in Section 6.3 of this SER.

6.4.1 TRU Waste

As stated in Section 6.3.4 of this SER, the applicant varied the size of the moderator and Pu-239 sphere to determine the optimal H/Pu-239 ratio and did this for various masses and chose the maximum mass that was under the targeted USL of 0.9393 or 0.9368 (FGE-3). The limiting configuration for all these contents was the HAC array so the mass was selected to meet USL under these conditions. The results for the single package analyses are summarized in the following Table:

Results of Single Package Criticality Evaluations

Content	H/Pu-239	Pu-239 mass (g)	k-eff + 2σ
FGE-1	900	340	0.93258
FGE-2a	950	350	0.93030
FGE-2b	900	375	0.93317
FGE-2c	900	395	0.93337
FGE-3	800	121	0.93505
FGE-5	900	250	0.93318

6.4.2 Irradiated Fuel Waste

As stated in Section 6.3.4 of this SER, the applicant used three different geometries for representing this content and determined the optimal H/U-235 or varied pitch and particle size combination. The applicant also performed sensitivity studies on flooding the package with various densities of water. The applicant determined that the spherical particle lattice is the geometry that produces the higher reactivity. The maximum k-eff + 2σ for 2500lb (1134 kg) of uranium at 0.9% enrichment of U-235 is 0.93941 which is below the USL of 0.94140.

6.4.3 Single Package Evaluation Conclusion

Since the k-eff is less than the USL under the tests specified in 10 CFR 71.71 for both the TRU waste and the irradiated fuel waste contents, the staff determined that this meets the requirements of 10 CFR 71.55(d)(1) which requires that the contents be subcritical under NCT tests as prescribed in 10 CFR 71.71.

The staff verified that the geometric form of the package contents would not be substantially altered as specified in 10 CFR 71.55(d)(2) based on the conclusion of the structural review of the package. On this basis, the staff determined that the package design meets the regulatory requirements of 10 CFR 71.55(d)(2).

Based on the conclusion of the structural review, the staff verified that there will be no substantial reduction in the effectiveness of the packaging for criticality prevention. The staff verified that there is no reduction in the effectiveness of the packaging including (1) the total volume of the packaging will not be reduced on which the criticality safety is assessed, (2) the effective spacing between the fissile contents and the outer surface of the packaging is not reduced by more than 5%, and (3) there is no occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10cm cube. The staff found that the package design meets the requirements in 10 CFR 71.55(d)(4).

6.5 Evaluation of Package Arrays under Normal Conditions of Transport

The applicant analyzed the most reactive credible configuration consistent with the condition of the package and the chemical and physical form of the contents under NCT. This is discussed in Section 6.3.4 of this SER.

The applicant demonstrated that 50 packages in a hexagonal array under NCT is subcritical. In accordance with 10 CFR 71.59(a), with “5N” being the size of an array of packages under NCT, the number “N” is 10.

6.5.1 TRU Waste

The results of the criticality evaluations for the array under NCT for the TRU waste material is summarized in the following table:

Results of NCT Package Array Criticality Evaluations

Content	H/Pu-239	Pu-239 mass (g)	k-eff + 2σ
FGE-1	900	340	0.93279
FGE-2a	950	350	0.93145
FGE-2b	950	375	0.93366
FGE-2c	950	395	0.93396
FGE-3	800	121	0.93470
FGE-5	900	250	0.93730

6.5.2 Irradiated Fuel Waste

The applicant found the spherical particle configuration to have the highest k-eff of the three different geometries modeled. The maximum k-eff + 2σ for 2500lb (1134 kg) of uranium at 0.9% enrichment is 0.71919 which is below the USL of 0.94140.

6.6 Evaluation of Package Arrays under Hypothetical Accident Conditions

The applicant specified a CSI of 5.0. The applicant assumed 20 packages in a hexagonal array the HAC array analyses. This represents an array size of “2N” (with N being derived consistent with 71.59(b)) which is consistent with the requirement in 71.59(a)(1).

6.6.1 TRU Waste

The applicant performed criticality safety analyses for an array of packages containing the TRU wastes under HAC. The results of the criticality evaluations for the array of TRU waste container packages under HAC are summarized in the following table:

Results of HAC Package Array Criticality Evaluations

Content	H/Pu-239	Pu-239 mass (g)	k-eff + 2σ
FGE-1	900	340	0.93911
FGE-2a	950	350	0.93779
FGE-2b	900	375	0.93921
FGE-2c	900	395	0.93924
FGE-3	800	121	0.93516
FGE-5	900	250	0.93763

6.6.2 Irradiated Fuel Waste

The applicant performed criticality safety analyses for array of packages containing the irradiated fuel wastes under HAC. The applicant found the spherical particle configuration to have the highest k-eff of the three different geometries modeled. The maximum k-eff + 2σ for 2500lb (1134 kg) of uranium at 0.9% enrichment is 0.94002 which is below the USL of 0.94140.

6.7 Computer code and Code Benchmarking

The applicant uses the Monte Carlo N-Particle Code, Version 6 (MCNP6). The applicant uses the ENDF/B-VII.1 and ENDF/B-VII.0 continuous energy nuclear data set for its criticality safety analyses. Because MCNP is one of the computer codes recommended in NUREG-2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material,” for criticality safety analyses, the staff found it to be acceptable for this application.

6.7.1 Experiments and Applicability

The applicant performed benchmark calculations with the same computer codes and cross section data that were used in the criticality safety calculations for the OPTIMUS-L package containing the various authorized contents.

The applicant selected experiments from the International Handbook of Evaluated Criticality Safety Benchmark Experiments. These are acceptable to use as benchmark experiments per the guidance provided in NUREG/CR-6698, “Guide for Criticality Validation of Nuclear Criticality Safety Calculational Methodology.”

6.7.1.1 Plutonium Experiments

The applicant provided the selected experiments containing plutonium (Pu) in Table 6.8-4 of the application including the area of applicability in Table 6.8-6 of the application. For the Pu systems, the applicant has included experiments without beryllium, even though all of the allowable Pu contents have beryllium, also there are some extra Pu isotopes (e.g. Pu-238, Pu-241, Pu-242) within the critical experiments that are not part of the modeled contents. For some of the allowable contents, the applicant included Pu-240 (FGE-2a, 2b, 2c) in its OPTIMUS-L criticality safety evaluations, however all benchmark experiments include Pu-240.

The staff reviewed the critical experiments selected by the applicant for code benchmarking analyses. The staff found that all of the OPTIMUS-L criticality evaluations with Pu contain polyethylene and moderator and/or reflector and none of the cited benchmark experiments have this content. However, the staff found that the degree of moderation is similar enough to the benchmark evaluations by comparing the H/X (hydrogen-to-fissile) ratio of the experiments to that of the OPTIMUS-L calculations.

The staff also compared the energy of the average lethargy causing fission (EALF) which gives an indication of the fission reaction distribution in the system as a function of energy. From the information in Tables 6.8-4 and 6.8-6 of the application, the staff determined that the experiments used by the applicant have a similar enough EALF to give reasonable assurance that the experiments selected are comparable to the Pu contents in the OPTIMUS-L package.

The benchmark experiments are for solution systems and therefore can be considered to represent the homogenous calculations performed for the OPTIMUS-L plutonium contents. The applicant grouped the experiments into those with and those without beryllium and calculated a different USL for FGE-3 which contains a significant amount of beryllium. The experiments the applicant used to calculate the USL with beryllium contain all of the experiments without beryllium but include additional experiments with beryllium.

Ideally for this content, the applicant would only use experiments with a beryllium reflector. The staff does not have enough information to determine if the cases without beryllium are similar enough to be appropriate to include within the benchmark evaluation for FGE-3. The staff found that although it is possible that using more appropriate experiments may increase the code bias or bias uncertainty, the staff found the applicant's analysis acceptable for this package based on the very low Pu-239 mass, i.e., 121 grams, allowed for this content. For comparison, under the conditions studied in the standard ANSI/ANS 8.1-2014, "nuclear criticality safety in operations with fissionable materials outside reactors," the minimum critical mass is 450 grams with stated conditions that are "unlikely to be approached in practice." Therefore, the staff found that even if there were increases to the code bias and bias uncertainty, there is enough safety margin to subcriticality based on the conservative assumptions within the analysis. The mass of Pu in the content is not sufficient to make the system critical under optimal moderation even with beryllium, which is a special moderator.

Despite the stated differences, the staff found that the benchmark experiments include significant fissile, moderator and reflector materials to perform the bias and bias uncertainty determination for OPTIMUS-L. Section 5.1 of NUREG/CR-6361 states that these should be as similar as possible. In making its determination that these experiments are acceptable for the benchmarking evaluations, the staff also took into consideration the conservative nature of the calculations, such as it being unlikely for a waste package to assume the most reactive geometry, moderation and reflection conditions and assuming there are no secondary containers.

6.7.1.2 Uranium Experiments

The applicant provided the selected experiments for a uranium-based system in Table 6.8-5 of the application including the area of applicability in Table 6.8-7 of the application. The benchmark experiments include additional uranium isotopes not included in the OPTIMUS-L calculations, however this a small amount.

The staff reviewed the critical experiments selected by the applicant for code benchmarking analyses. The staff found that the enrichment for the benchmark evaluations are slightly out of range for that of the OPTIMUS-L. However, based on the trendline the applicant found for enrichment, the applicant determined that there is not a strong correlation for enrichment and the bias in k-eff.

Although there are no benchmark experiments with the exact geometry used in the OPTIMUS-L package containing uranium contents, the applicant selected experiments with heterogeneous and homogeneous mixtures to account for the geometries studied within the OPTIMUS-L criticality safety calculations.

The moderator for the uranium cases is mostly water with a few using paraffin. Although all of the OPTIMUS-L cases contain polyethylene and beryllium, the staff found that the degree of moderation is similar enough to the benchmark evaluations by comparing the H/X (hydrogen-to-fissile) ratio of the experiments to that of the OPTIMUS-L calculations.

The staff also compared the energy of the EALF which gives an indication of the fission reaction distribution in the system as a function of energy. From the information in Tables 6.8-5 and 6.8-7 of the application, the EALF is just outside of the range of the benchmark comparisons, however none of these are for the limiting cases, therefore additional bias would not change the results of the criticality safety evaluation. The staff determined that the experiments used by the applicant have a similar enough H/X and EALF to give reasonable assurance that the experiments are comparable to the OPTIMUS-L uranium contents.

The applicant did not include critical benchmark experiments with U-235 and beryllium. The applicant did not reduce the USL to account for a potential increase to the uncertainty from beryllium. Although this is potentially non-conservative, it was the staff’s judgment that the benchmarking experiments were acceptable based on the relatively small amount of beryllium (1%) and the conservative nature of the analysis, e.g. assuming a configuration that would achieve maximum reactivity.

Section 5.1 of NUREG/CR-6361 states that benchmark experiments should be as similar as possible. In making its determination that these experiments are acceptable for the benchmarking evaluations for the OPTIMUS-L, the staff also took into consideration the conservative nature of the calculations, such as it being unlikely for a waste package to assume the most reactive geometry, moderation and reflection conditions and assuming there are no secondary containers.

6.7.2 Bias Determination

Section 4.1 of NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," discusses two methods for calculating the bias and bias uncertainty. Method 1, discussed in Section 4.1.1 of NUREG/CR-6361, uses a statistical calculation to a linear fit of critical benchmark data to determine the bias and bias uncertainty plus an administrative margin. Method 2, in Section 4.1.2 of NUREG/CR-6361 outlines a statistical method for determining a combined lower confidence band plus subcritical margin.

The applicant has applied Method 1 and used Method 2 as a verification of Method 1. This is consistent with NUREG/CR-6361 which also recommends Method 2 to be applied with Method 1 to verify that the administrative margin is conservative, therefore the staff found the use of these methods to determine the bias and bias uncertainty acceptable. Section 4.1 of NUREG/CR-6361 states that these methods are based on the assumption that the data is normally distributed.

The applicant used the USLSTATS code (discussed in Section C.3 of NUREG/CR-6361) to determine the USL which is based on these methods. Data needs to be normally distributed for these methods to be applicable. The applicant stated that the data without the beryllium reflector is normally distributed, as reported by USLSTATS, however the beryllium data is an outlier and causes the data to be non-normally distributed. Since its contribution does decrease the USL, which is conservative, and based on the conservative nature of the analysis, the staff found its inclusion acceptable for the USL determination for FGE-3.

The staff reviewed the USLSTATS code input in Tables 6.8-10A and 6.8-10B and found that they are appropriately considering experimental and calculational uncertainties.

The applicant determined the parameters responsible for variations in k-eff are EALF, H/X ratio and fissile weight percent. For the plutonium calculations the applicant determined that the parameter with the largest correlation coefficient was the fissile weight percent and used this parameter to determine the USL. For the uranium calculations the applicant determined that the parameter with the largest correlation coefficient was the H/U-235 ratio and used this parameter to determine the USL. This is consistent with the recommendations in NUREG/CR-6361 and therefore the staff found it acceptable.

The applicant determined a USL of 0.9393 for the plutonium calculations without beryllium that apply to contents FGE-1, 2a, 2b, 2c, and 5; 0.9368 for the plutonium calculations with beryllium that apply to contents FGE-3, and 0.9414 for the uranium calculations (FEM contents).

The staff reviewed the applicant's determination of bias and bias uncertainty associated with the computer code and found that it was determined appropriately by the applicant and found it acceptable.

6.8 Evaluation Findings

The staff has reviewed the package and concludes that the application adequately describes the package contents and the package design features that affect nuclear criticality safety in compliance with 10 CFR 71.31(a)(1), 71.33(a), and 71.33(b) and provides an appropriate and bounding evaluation of the package's criticality safety performance in compliance with 10 CFR 71.31(a)(2), 71.31(b), 71.35(a), and 71.41(a). The staff has reviewed the package and concludes that the application identifies the codes and standards used in the package's criticality safety design in compliance with 10 CFR 71.31(c).

The staff has reviewed the package and concludes that the application specifies the number of packages that may be transported in the same vehicle through provision of an appropriate CSI in compliance with 10 CFR 71.35(b). The applicant specifies an appropriate CSI for each type of fissile content. The staff has reviewed the package and concludes that the applicant used packaging features and package contents configurations and materials properties in the criticality safety analyses that are consistent with and bounding for the package's design basis, including the effects of the normal conditions of transport and the relevant accident conditions in 10 CFR 71.55(f), 71.73, or 71.74. The applicant has adequately identified the package configurations and material properties that result in the maximum reactivity for the single package and package array analyses.

The staff has reviewed the package and concludes that the criticality evaluations in the application of a single package demonstrate that it is subcritical under the most reactive credible conditions, in compliance with 10 CFR 71.55(b), 71.55(d), and 71.55(e) (the package is not authorized for air transport so 71.55(f), or 71.64(a)(1)(iii) do not apply). The evaluations in the application also demonstrate that the effects of the normal conditions of transport tests do not result in a significant reduction in the packaging's effectiveness in terms of criticality safety, in compliance with 10 CFR 71.43(f) and 10 CFR 71.55(d)(4) and, 10 CFR 71.51(a)(1). The evaluations in the application also demonstrate that the geometric form of the contents is not substantially altered under the normal conditions of transport tests, in compliance with 10 CFR 71.55(d)(2).

The staff has reviewed the package and concludes that the criticality evaluation in the application of the most reactive array of 5N undamaged packages demonstrates that the array of 5N packages is subcritical under normal conditions of transport to meet the requirements in 10 CFR 71.59(a)(1). The staff has reviewed the package and concludes that the criticality evaluation in the application of the most reactive array of 2N packages subjected to the tests in 10 CFR 71.73 demonstrates that the array of 2N packages is subcritical under hypothetical accident conditions in 10 CFR 71.73 to meet the requirements in 10 CFR 71.59(a)(2). The staff has reviewed the package and concludes that the applicant's evaluations include an adequate benchmark evaluation of the calculations. The applicant identified and evaluated experiments that are relevant and appropriate for the package analyses and performed appropriate trending analyses of the benchmark calculation results. The applicant has determined an appropriate bias and bias uncertainties for the criticality evaluation of the package.

The staff has reviewed the package and concludes that the application identifies the necessary special controls and precautions for transport, loading, unloading, and handling and, in case of accidents, compliance with 10 CFR 71.35(c). The staff has reviewed the package and concludes that the evaluations in the application assume unknown properties of the fissile contents are at credible values that maximize neutron multiplication consistent with 10 CFR 71.83.

Based on review of the statements and representations in the application, the staff has reasonable assurance that the proposed package design for the contents satisfy the nuclear criticality safety requirements prescribed in 10 CFR Part 71.55. In making this finding, the staff considered the regulation itself, appropriate regulatory guides, applicable codes and standards, accepted engineering practices, and the staff's own independent confirmatory calculations.

7.0 OPERATING PROCEDURES

The applicant detailed the loading-related preparations, tests, and inspections for the package. These include the inspections made before loading the package to determine that it is not damaged, and that radiation and surface contamination levels are within the regulatory limits. The package, transported by truck in a vertical orientation, may either be tied down to a custom-designed pallet that is secured to a trailer deck or tied down directly to the trailer deck.

The general procedure for preparing the package for loading is as follows: (i) perform radiation and removable contamination surveys of the package, clean or decontaminate the package as necessary, (ii) visually inspect the exterior surfaces of the package for any signs of damage, (iii) remove the tamper-indicating seal from the OP and remove the OP lid bolts, (iv) lift the OP lid, loosen the CCV closure bolts, attach swivel hoist rings to the CCV lid and torque them, verify that the captured lid bolts are completely disengaged from the threaded holes in the CCV bolt flange and lift the CCV lid vertically, (v) if required, remove all CCV port cover bolts, remove the CCV port cover, remove the plugs from the CCV lid and port cover test ports. A maintenance leakage rate test is required for any replaced CCV lid or vent port containment O-ring, per Section 8.2.2.1 of the application.

The general procedure for loading the contents into the package and closing the package is as follows: (i) confirm that the contents to be loaded meet the requirements of the CoC, (ii) verify that the packaging internals (e.g., cribbing/dunnage and SIA components) required for the shipment are properly configured in the CCV cavity, (iii) place the contents into the CCV cavity or SIA cavity (if used), clean and visually inspect the sealing surface for the CCV lid and the CCV port cover, lower the CCV lid onto the CCV body, tighten the CCV lid bolts to a torque of 300 ± 15 ft-lbs, (iv) install the CCV port cover and torque the port cover bolts to 15± 1 in-lbs. Backfilling with helium gas the CCV cavity is not authorized. The staff reviewed the operating procedures and determined that it includes reference to Table 6.3.6A of the application which includes the conversion factors for determining fissile gram equivalent for contents FGE-1, 2a, 2b, 2c, 3 and 5 described in Table 1.2-1 of the application. The staff found that this appropriately describes the package loading operations necessary to determine the amount of fissile material that is consistent with the criticality safety analyses. Pre-shipment inerting of the CCV is not allowed.

Preparation of the package for transport includes (i) a pre-shipment leakage rate test of the CCV lid and port cover O-ring seals prior to every shipment, even if the port cover is not removed for loading operations; pre-shipment leakage rate tests shall be performed using the Gas Pressure Drop or Gas Pressure Rise methods described in Sections A.5.1 and A.5.2 of ANSI N14.5 (ii) installation of the plugs in the leak test ports of the CCV lid and CCV port cover, (iii) decontamination of the exterior top surface of the CCV, (iv) placing the OP lid onto the OP base and tightening each of the OP lid bolts to a torque of 50 ± 5 ft-lbs, (v) install the tiedown and tamper-indicating devices, (vi) verification of the external radiation levels and of the levels of non-fixed contamination on the package. The applicant stated in Section 7.1.3, “Preparation for Transport,” that the users need to verify that the exterior surface of the package does not exceed 85°C (185°F) in accordance with the requirement of 10 CFR 71.43(g).

The general procedure for opening each loaded package and removing its contents is as follows: (i) remove all OP lid bolts, lift the OP lid, remove all CCV closure bolts, lift the CCV lid vertically and place it to a temporary storage location, (ii) remove the contents from the packaging.

The general procedure for preparing each empty package for transport is as follows: (i) visually inspect the CCV cavity or SIA cavity to confirm that it has been emptied of its contents, (ii) survey the interior of the internal surfaces of the package (i.e., CCV bottom support plate, CCV cavity, CCV flange, and underside of the CCV lid) and any empty payload internals (e.g., SIA body and dunnage, if used) to be shipped to verify that the interior contamination limits are satisfied. If the non-fixed surface contamination exceeds the limits for empty package shipment, then decontaminate the interior surfaces, as necessary, (iii) visually inspect the readily accessible surfaces of the packaging components for any signs of damage, (iv) install the CCV closure lid and tighten each lid bolt, in the sequence shown on the CCV lid, to a torque of 300 ± 15 ft-lbs, (v) install the CCV port cover and torque each of the port cover bolts to 15 ± 1 in-lbs, place the OP lid onto the OP base and torque the lid bolts to a torque of 50 ± 5 ft-lbs., (vi) verify that the package tiedowns are installed and the package is secured to the trailer, (vii) perform a radiation survey and a contamination survey prior to releasing the package for the empty packaging return shipment.

The staff reviewed the operating procedures for the OPTIMUS-L in Chapter 7 of the application to ensure that the procedures reflect the acceptable operating sequences, guidance, and generic procedures for key operations as represented in the shielding analysis and meet the requirements of 10 CFR Part 71. The staff reviewed the loading procedures and finds that the applicant considered ALARA principles and contamination and has steps associated with loading the SIA when needed. Appendix 7.5-1 of the application contains the loading procedure for determining if the contents are acceptable for loading. The staff reviewed this procedure and found that it is consistent with the analysis in the shielding evaluation in Chapter 5 of the application as discussed in Section 5.4.2 of this SER and that the procedure will ensure that the package as loaded will be below regulatory dose rate limits.

In Appendix 7.5, the applicant provided the inerting procedure (Attachment 7.5-2 of the SAR, “Example CCV Pre-Shipment Inerting Procedure”) and Attachment 7.5-3 to determine the maximum shipping time limit required to assure a flammable gas concentration would not accumulate in any confinement volume of the contents during the shipment. Staff reviewed the revision of Attachment 7.5-2 from the applicant’s 2nd round RAI response with the additional steps and clarifying notes. Upon review of the attachment, as stated in a previous portion of this RAI, the attachment was used for the Model No. 9978 package that is smaller and less complex than that of the OPTIMUS-L package, thus not providing a reasonable set of guidelines that may be applied for this package. In addition, the applicant proposed using the inerting procedure from SRNL for the OPTIMUS-L package. The staff reviewed the proposed procedure and finds that it does not provide the necessary information on how the SRNL procedure, if followed, would prevent the development of flammable gas mixtures in any confined area within the OPTIMUS-L package. In addition, the procedure provided by SRNL itself states, in its abstract: “Test results demonstrated only following a set of procedure steps would not ensure the inerting will be acceptable.” Therefore, staff determined that the applicant’s application of the SRNL to the OPTIMUS-L package is not appropriate due to the variance in the package size, port configuration, and the variance in the internal complexities between the package used by SRNL and the OPTIMUS-L package. NAC provides a discussion pertaining to the secondary containers being within the containment system of the OPTIMUS-L package. However, secondary containers do not have a containment function and are required to include a venting mechanism to allow gases to flow in or out of the secondary container or in or out of each confinement region of the contents. Staff determined that NAC only shows a venting mechanism to allow gases to flow in or out of each confinement region of the contents but does not show its effectiveness in operation and does not demonstrate that the inert fill gas either “effectively” occupies the cavity of the cask containment vessel (CCV) or is in uniform concentration through the CCV. Additionally, NAC did not demonstrate or discuss an injection path or port orientation of how helium can be introduced effectively within the containment system of the OPTIMUS-L package. The staff determined that it was not able to reach a reasonable assurance finding in order to accept inerting for the OPTIMUS-L package.

The applicant described loading and unloading procedures and the preparation of the empty package for transport. This information is contained in the SAR Section 7, “Package Operation.” The package is intended to be transported by truck in a vertical orientation. The applicant described the details of the package operation in the SAR Section 7.1.1, “Preparation for Loading,” Section 7.1.2, “Loading Contents,” and Section 7.1.3,” Preparation for Transport.” These SAR sections describe the preparations, inspections, and cautions that should be taken when handling the CCV, and how to install it within the OP. These instructions are provided in a written step by step format.

Specifically, when the package content is lifted using forklift, or using three lifting lugs installed on the OP, the applicant describes the angle of the slings as having to be approximately 70° from horizontal. The applicant also states that the package is designed to be lifted vertically using a three-legged bridle connected to the three lifting lugs located on the OP lid. The bounding vertical design lift load is 11.5 kip, or 3.83 kip per lifting lug.

The applicant describes the procedures used to tie the package to the conveyance (flat track). The package is secured to the transport conveyance by four (4) tie-down arms attached to the OP base bolt flange. The package is secured using tension on the tie-downs with turnbuckles.

The NRC staff has reviewed the description of the operating procedures and finds that the package will be prepared, loaded, transported, received, and unloaded in a manner consistent with its design. The NRC staff has reviewed the description of the special instructions to inspect, handle, and to safely open a package and concludes that the procedures for providing the special instructions to the consignee are in accordance with the requirements of 10 CFR 71.89.

8.0 ACCEPTANCE TESTS AND MAINTENANCE

The acceptance tests include the following : (i) each CCV assembly shall be pressure tested to 150% of the packaging MNOP (100 psi) to verify the capability of the containment system to maintain its structural integrity at the test pressure, (ii) the CCV assembly (i.e., the packaging containment boundary) shall be leakage rate tested to 1×10^{-7} ref-cm^3/s. Leakage rate testing shall be performed using the Evacuated Envelope-Gas Detector method of ANSI N14.5, Section A.5.4, with helium as the tracer gas and a suitable helium leak detector with a sensitivity of at least 5×10^{-8} ref-cm^3/s. All containment O-rings that are not used for the acceptance leakage rate test shall be subjected to the maintenance leakage rate testing described in Section 8.2.2.1 of the application prior to their initial use.

The packaging does not require shielding acceptance testing because the shielding component are made from solid steel. The packaging does not in include any special shielding features, such as a poured lead gamma shield, and the material properties used for the shielding evaluation of the package are sufficiently conservative. Thermal acceptance testing of the packaging is not required because the packaging does not include any special thermal features that require thermal acceptance testing and the material properties used for the thermal evaluation of the package are conservative. Although NUREG 1609, “Standard Review Plan for Transportation Packages for Radioactive Material,” states that a thermal test is used to verify that the heat transfer performance is achieved in the fabrication process, the staff agrees with the applicant that the thermal test is not required for OPTIMUS-L package, based on a small decay heat of 50 watts, with acceptable temperature margins below the allowable limits, and conservative thermal evaluations under NCT and HAC presented in Chapter 3 of the application.

Table 8.1-1 of the application summarizes the foam static crush strength acceptance criteria.

The staff reviewed the acceptance tests and maintenance programs that are important to ensuring the shielding in the as-fabricated package meets the design specified in the technical drawings, as evaluated in the shielding analysis at the time of fabrication and use and will continue to do so over the course of its service life. The staff evaluated the information in Section 8.1.6 of the application related to "Shielding Tests." The applicant states that no acceptance testing is required because shielding components are made from solid steel. The staff accepts inspections of the package dimensions as a sufficient acceptance test for ensuring shielding performance of the steel components. These components are manufactured to industry standard specifications and they are not subject to the material irregularities faced from non-standard materials or a poured lead shield. Section 8.1.1 of the application requires that the dimensions and tolerances be verified by measurement on each package. The staff found this acceptable to ensure that the shielding is manufactured in accordance with the drawings in lieu of acceptance tests.

The applicant did not identify any maintenance tests that will need to be performed on the OPTIMUS-L in relation to the shielding performance. The staff has not identified any degradation mechanisms that would affect the shielding performance during the service lifetime of the package and found this acceptable.

The maintenance program includes periodic inspections, tests, and maintenance activities designed to ensure continued performance of the packaging. This section describes the periodic testing, inspection, and replacement schedules, as well as the criteria for replacement and repair of components and subsystems on an as-needed basis. A periodic leakage rate test is required to be performed on every containment seal of the packaging within the 12-month period prior to every shipment but need not be performed for packages that are out-of-service (e.g., placed into temporary storage). As discussed in Section 8.2.3.1, all packaging O-rings and fastener seals are required to be replaced within the 12-month period prior to any shipment and, therefore, the maintenance leakage rate testing of the replaced containment seals also satisfies the requirement for periodic leakage rate testing.

Maintenance leakage rate testing of all packaging containment seals is performed in accordance with Section 7.4 of ANSI N14.5 prior to returning the package to service following maintenance, repair, or replacement of any components of the containment system to confirm that the CCV assembly is not degraded.

Maintenance leakage rate testing need only be performed on the affected seal or sealing surface of the containment system. Leak-tight acceptance criteria of 1×10^{-7} ref-cm^3/s shall be used for the periodic and maintenance leakage rate tests. All exposed interior and exterior surfaces of the OP base and lid assemblies, CCV body and lid assemblies, CCV vent and test port plugs, and SIA body assembly shall be visually inspected within the 12-month period prior to any shipment for damage or degradation that could impair the physical condition of the packaging.

The packaging maintenance requirements for the CCV containment O-rings (lead and port), the CCV leak test O-rings (lid & port), the CCV containment O-ring sealing surfaces, the CCV leak test O-ring sealing surfaces, CCV lid bolts and port cover, the OP lid bolts, threaded inserts, exposed packaging interior and exterior surfaces are summarized in Table 8.2-1 of the application.

CONDITIONS

The following Conditions are included in the certificate:

The maximum decay heat is 50 Watts.

The package must be transported under exclusive use.

Shoring must be placed between loose fitting contents and the CCV cavity to prevent excessive movement during transport. The shoring material shall not react negatively with the packaging materials or contents and should have a melting temperature above 300°F to ensure shoring maintains its geometry under routine and normal conditions of transport.

All radioactive contents shall be packaged in secondary container(s) (e.g., drums, liners, specialty bags, etc.).

Hydrogen must be limited to a molar quantity that would be no more than 5% by the volume of the innermost layer of confinement during transport. The port of the CCV lid shall be plugged. Compliance with the hydrogen and other flammable gas limit must be demonstrated for each shipment.

Transport by air is not authorized.

CONCLUSION

Based on the statements and representations in the application, the staff finds that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

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