

**CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES**

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGE
9392	0	71-9392	USA/9392/B(U)F-96	1	OF 8

2.. PREAMBLE

- a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."
- b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.
3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

- | | | |
|----|--|---|
| a. | ISSUED TO (<i>Name and Address</i>) | b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION |
| | NAC International, Inc.
2 Sun Court, Suite 220
Peachtree Corners, GA 30092 | OPTIMUS®-H OPTImal Modular Universal Shipping
Cask Safety Analysis Report – Revision No. 0, dated
April 2024. |

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

- (1) Model No.: OPTIMUS®-H
- (2) Description

The OPTIMUS®-H packaging consists of (i) the Cask Containment Vessel (CCV), (ii) the Outer Shield Vessel (OSV), and (iii) upper and lower impact limiters.

The CCV fits within the cavity of the OSV and 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 diameter of 32.5 inches by 47.0 inches high. The CCV cylindrical shell and bottom plate are both 1.0-inch thick. The CCV lid, which is 3.38 inches thick, is fastened to the CCV body by twelve (12) equally spaced 1-inch diameter socket head cap screws. The CCV includes an optional drain port that is used to drain water from the CCV cavity for wet-loading operations.

The OSV is a ductile cast iron vessel consisting of a body and lid. The OSV has an overall height of 61.5 inches and an outer diameter of 49.0 inches. The internal cavity of the OSV, with a diameter of 35.0 inches by 51.9 inches high, is large enough to accommodate the CCV with sufficient clearance to assure free differential thermal expansion. The OSV lid is fastened to the OSV body by twelve (12) 1.25-inch diameter heavy hex screws.

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

The cylindrical-shaped impact limiters fit over the upper and lower ends of the OSV assembly. The upper and lower impact limiters are identical, with a pocket that fits over the end of the OSV assembly and consisting of energy absorbing closed-cell polyurethane foam cores sealed inside the exterior stainless-steel inner and outer shells.

The packaging may also be configured with a Shield Insert Assembly (SIA), within the cavity of the CCV, to provide additional gamma shielding. The SIA is a coated carbon steel container, placed inside the CCV cavity, that is either 1-inch thick, 2¼-inch thick or 3¾-inch thick. The internal cavity of the 1-inch thick and 2¼-inch thick SIA has a diameter of 24.0 inches and a length of 35.3 inches, whereas the 3¾-inch thick SIA has a diameter of 24.0 inches and a length of 41.0 inches. The 2¼ inch and 3¾ inch SIAs both include lids providing supplemental top end gamma shielding.

The maximum weight of the contents (including the CCV, SIA, dunnage or shoring) is 7,300 lbs. and the maximum gross weight of the package is approximately 32,200 lbs.

(3) Drawings

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

70000.14-L501	1P	Packaging Assembly – OPTIMUS-H
70000.14-L510	0P	CCV Assembly – OPTIMUS
70000.14-L511	0P	CCV Body Weldment – OPTIMUS
70000.14-L512	0P	CCV Lid – OPTIMUS
70000.14-L513	0P	Port Cover – OPTIMUS
70000.14-L520	0P	OSV Assembly – OPTIMUS-H
70000.14-L521	2P	OSV Body – OPTIMUS-H
70000.14-L522	1P	OSV Lid – OPTIMUS-H
70000.14-L530	1P	Impact Limiter Assembly - OPTIMUS-H
70000.14-L550	1P	1-Inch Shield Insert Assembly (SIA) - OPTIMUS
70000.14-L551	1P	2¼-Inch Shield Insert Assembly (SIA) - OPTIMUS
70000.14-L552	1P	3¾-Inch Shield Insert Assembly (SIA) - OPTIMUS

(b) Contents

(1) Type and Form of Material

Radioactive contents of the package include two (2) classes of waste, transuranic containing waste (TRU) and irradiated fuel waste (IFW), as summarized in Table 1.

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5.(b)(1) Type and Form of Material (continued)

Table 1 - Content Designations

Contents			Content ID
Class	Type	Sub-Type	
TRU waste	Compliant ⁽¹⁾	---	1-1
	Non-Compliant ⁽¹⁾	Aerosol Cans, Type 1 ⁽²⁾	1-2A
		Aerosol Cans, Type 2 ⁽³⁾	1-2B
		Standard DOT 3E Lecture Bottle	1-2C
Irradiated Fuel Waste	LEU ⁽⁴⁾	---	2-1
	CANDU	---	2-2

Notes:

- (1) Refers to Waste Isolation Pilot Plant (WIPP) Waste Acceptance Criteria (WAC) for TRU waste.
(2) Aerosol cans with compressed gas propellant.
(3) Aerosol cans with liquified gas propellant or unknown propellant.
(4) Low enriched uranium.

- (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, either dewatered, solid, or solidified waste (Content 1-1).
- (ii) Dewatered, solid, or solidified transuranic-containing wastes (TRU), fissile, non-fissile, or fissile-excepted (Content 1-1).
- (iii) Neutron activated metals or metal oxides in solid form, including reactor components or segments of components of waste from a nuclear process or power plant (Content 1-1).
- (iv) TRU waste containing standard DOT 2P or 2Q, 1 liter aerosol cans containing compressed gas propellant (Content 1-2A).
- (v) TRU waste containing standard DOT 2P or 2Q, 1 liter aerosol cans with liquified gas propellant or unknown propellants (Content 1-2B).
- (vi) TRU waste containing standard DOT 3E lecture bottles with known contents of nonflammable gases or with unknown contents (Content 1-2C).
- (vii) Radioactive material in the form of LEU waste with or without activated metal structural components (e.g., cladding, liners, baskets, etc.) (Content 2-1).

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5.(b)(1) Type and Form of Material (continued)

(viii) Irradiated CANDU fuel waste contents and irradiated hardware contents. Irradiated CANDU fuel contents are restricted to natural UO_2 fuel of a typical CANDU fuel bundle design with cladding and bundle structure comprised exclusively of Zircaloy material (Content 2-2).

(ix) CANDU fuel baskets of irradiated hardware materials (Content 2-2).

5.(b)(2) Maximum quantity of material per package

- (i) Compliant TRU Waste – Content 1-1:
- a) Maximum weight of contents: 7,300 pounds including radioactive waste, secondary containers, SIA, and shoring.
 - b) Fissile contents must not exceed the fissile gram equivalents (FGE) in Table 2 for the specified criticality configuration limits. Plutonium contents in quantities greater than 0.74 TBq (20 Ci) must be in solid form.
 - c) TRU waste contents shall comply with regulatory dose rate limits, as determined in accordance with Chapter 7, Attachment 7.5-1.
 - d) Compliant TRU waste may contain only one (1) form of non-compliant TRU waste contents. TRU waste containing any approved non-compliant TRU waste items must also satisfy the maximum quantity requirements for the respective non-compliant TRU waste contents.
 - e) Except for approved non-compliant TRU waste contents, any other explosives, corrosives, non-radioactive pyrophorics, and sealed items containing compressed and/or flammable gas (e.g., aerosol cans, lecture bottles, etc.) are prohibited. Pyrophoric radionuclides may be present only in residual amounts less than 1 wt.%. All nonradioactive pyrophoric material shall be reacted (or oxidized) and/or otherwise rendered nonreactive prior to placement in a secondary container (e.g., drum).
 - f) Free liquids shall not exceed 1% of the CCV cavity volume.

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5.(b)(2) Maximum quantity of material per package (continued)

Table 2 - TRU Waste FGE Limits

	FGE Criticality Configuration Description				FGE Limit, (g ²³⁹ Pu) ⁽¹⁾
	Machine Compacted ⁽²⁾	Weight % Special Reflector ⁽³⁾	Minimum ²⁴⁰ Pu Credit	Chemically or Mechanically Bound	
FGE-1		≤ 1			335
FGE-2a		≤ 1	≥ 5 g		350
FGE-2b		≤ 1	≥ 15 g		370
FGE-2c		≤ 1	≥ 25 g		390
FGE-3		> 1			121
FGE-4		> 1		✓	320
FGE-5	✓	≤ 1			245

- (1) FGE equivalents determined as described in Section 6.3.4 of the Application. FGE conversion based on a ratio of subcritical mass limits in ANSI/ANS-8.1 [1-7], SAR Section 5.2 of 0.7 kg (1.5 lb) for ²³⁵U and 0.45 kg (1.0 lb) for ²³⁹Pu (See 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/ cm³) are unlimited, but materials with hydrogen density greater than water are limited to the hydrogen density of polyethylene (0.1336 g/cm³) 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/cm³) 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.

(ii) Non-Compliant TRU – Content 1-2A, Content 1-2B, and Content 1-2C

- a) Content 1-2A - A maximum of ninety-five (95) standard DOT 2P or 2Q 1 liter aerosol cans.
- b) Content 1-2B - A maximum of 4.4 liters of liquified gas propellant in any number of standard DOT 2P or 2Q 1 liter aerosol cans when the volume is known based on process knowledge x ray (or other) interrogation methods, or a maximum of four (4) full standard DOT 2P or 2Q 1 liter aerosol cans if the amount of liquified propellant, or type of propellant, in the aerosol cans is not known.
- c) Content 1-2C - A maximum of eight (8) full Standard DOT 3E Lecture bottles with known contents of non-flammable gases, or a maximum of one (1) Standard DOT 3E Lecture bottle with unknown contents.

(iii) Irradiated Fuel Waste

- a) Shall not exceed the Fissile Equivalent Mass (FEM) limits from Table 3 for the specified criticality configuration limits.

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5.(b)(2) Maximum quantity of material per package (continued)

- b) IFW contents shall comply with regulatory dose rates, as demonstrated in accordance with Chapter 7, Attachment 7.5-1 of the Application.
- c) Irradiated CANDU fuel waste contents shall have a maximum burnup of 5 GWd/MTU, a minimum cooling time of 40 years, and a maximum fuel mass of 1808.8 kg (3,988 lb) UO₂.
- d) Free liquids shall not exceed 1% of the CCV cavity volume.
- e) CANDU fuel baskets of irradiated hardware materials may not be shipped in the same package as other irradiated CANDU fuel waste contents.
- f) Maximum weight of contents, including IFW in a secondary container, internal structures (e.g., SIA, etc.) and dunnage or shoring shall not exceed 7,300 lbs (3,311 kg).
- g) CANDU fuel baskets of irradiated hardware materials are limited to 2.5 kg (5.5 lbs) of Inconel, 200 Kg (441 lbs) of stainless steel, and unlimited Zircaloy per package.

Table 3 - IFW FEM Limits

Config. ID ⁽¹⁾	LEU Waste Criticality Configuration Description			Uranium Mass Limit (lb)
	Weight % Special Reflector ⁽²⁾	Enrichment Limit (wt% U-235)	Particle Size Restriction (cm)	
FEM-1	≤ 1	≤ 0.96 wt%	≤ 0.1 and/or ≥ 8.0	5,000
FEM-2	≤ 1	≤ 0.80 wt%	N/A	

(1) IFW contents must be non-machine compacted. Materials with hydrogen density up to that of water (0.1117 g/cm³) are unlimited, but materials with hydrogen density greater than water are limited to the hydrogen density of polyethylene (0.1336 g/cm³) 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.

(iv) Plutonium contents in quantities greater than 0.74 TBq (20 Ci) must be in solid form.

5.(c) Criticality Safety Index (CSI) 0.0

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

- (a) The package must be loaded and prepared for shipment in accordance with the Package Operations in Section 7 of the application.

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- (b) The package must be tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Section 8 of the application.
7. 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.
8. All radioactive contents shall be packaged in secondary container(s) (e.g., drums, liners, specialty bags, etc.). TRU waste contents may be shipped on a nonexclusive use conveyance or under exclusive use controls. IFW shall be shipped under exclusive use.
9. Packaging preparations for shipment:
- (a) Each package prepared for shipment may only contain one form of non-compliant TRU waste contents, alone or in combination with compliant TRU waste (Content 1-1, Content 1-2A, Content 1-2B, Content 1-2C, Contents 1-1 and 1-2A, Contents 1-1 and 1-2B, and Contents 1-1 and 1-2C) and shall otherwise satisfy the Waste Isolation Pilot Plant (WIPP) Waste Acceptance Criteria (WAC).
- (b) The package shipment for TRU waste shall ensure that hydrogen and other flammable gases make up less than 5 percent by volume of the total gas inventory, or lower if warranted by the flammable gas, within any confined volume, for shipment within the allowable shipping time frame, as determined by Attachment 7.5-3 of the application.
- (c) Maximum decay heat for shipment of TRU waste is 200 watts. For TRU waste, with a total decay heat exceeding 50 watts, the CCV cavity shall be filled with helium gas per Table 3.1-1 of the application.
- (d) Maximum decay heat for shipment of IFW is 1,500 watts. For any heat load (0-1,500 watts), the CCV cavity shall be filled with helium gas per Table 3.1-1 of the application.
10. If not transported by private carriers, with individual monitoring of personnel in conformance with 10 CFR 20.1502, the minimum distance to occupied spaces shall be 20 ft from the centerline of the nearest package.
11. Transport by air is not authorized.
12. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.
13. Expiration date: August 31, 2029.

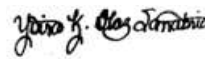
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REFERENCES

NAC International, Inc., Application – OPTIMUS®-H OPTImal Modular Universal Shipping Cask Safety Analysis Report – Revision No. 0, dated April 2024.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION



Signed by Diaz-Sanabria, Yaira
on 08/05/24

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

Date: August 5, 2024





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

**SAFETY EVALUATION REPORT
Docket No. 71-9392
Model No. OPTIMUS® - H Package
Certificate of Compliance No. 9392
Revision No. 0**

SUMMARY

By letter dated December 22, 2021 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML21361A089), NAC International (NAC) submitted an application for approval of the Model No. OPTIMUS®-H radioactive material transport package (Optimal Modular Universal Shipping cask for High activity contents).

On August 16, 2022, (ML22228A222), NAC submitted its responses to the staff's Request for Supplemental Information for the Review of the Model No. OPTIMUS®-H Package, dated April 28, 2022 (ML22115A197). On October 4, 2022, the application was accepted for a detailed technical review (ML22270A156).

On May 5, 2023, (ML23128A028, ML23128A029, and ML23128A030), NAC submitted its responses to the staff's Request for Additional Information (RAI) letter dated March 6, 2023 (ML23061A139). On August 14, 2023, NAC provided supplemental responses to the RAIs, pursuant to a July 12, 2023, clarification call held between NAC and the staff.

On September 25, 2023, staff issued a second request for additional information (ML23261C400) regarding ductile cast iron (DCI) components subjected to impact loads, propagation of fabrication flaws through the DCI components causing complete structural failure, and clarification of TRU waste content types, their corresponding heat load limits, for those deemed to require inerting. NAC submitted responses on December 13, 2023, and January 30, 2024, and revised and supplemented its responses on May 2, 2024, (ML24124A176) by including a fracture mechanics evaluation of DCI using fracture toughness limits, DCI testing requirements, DCI acceptance and updates to the lowest service temperature for the cask. The final safety analysis report dated April 30, 2024, is referenced in the Certificate of Compliance (CoC) (ML24124A177).

The packaging consists of a Cask Containment Vessel, an Outer Shield Vessel and impact limiters. A Shield Insert Assembly may be included inside the CCV for contents that require additional shielding. The internal cavity of the CCV is large enough to accommodate a 110-gallon drum. Radioactive contents of the package include two classes of waste, transuranic (TRU) containing waste and irradiated fuel waste. Acceptable TRU waste includes those which meet the waste acceptance criteria of the Waste Isolation Pilot Plant, aerosol cans with compressed gas propellant, aerosol cans with liquified gas or unknown propellant, or Department of Transportation 3E lecture bottles. Irradiated fuel waste includes low enrichment uranium fuel waste or CANDU fuel waste and their associated hardware.

Shielding for the OPTIMUS®-H contents is provided by the stainless steel and ductile cast iron structure of the packaging. Additional shielding for high activity payloads is provided by shielded insert assemblies made of carbon steel. Because the package contents can be highly variable the applicant developed inventory limits for each expected radionuclide. The Criticality Safety Index (CSI) for the package is 0.0.

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®-H packaging consists of (i) the Cask Containment Vessel (CCV), (ii) the Outer Shield Vessel (OSV), and (iii) upper and lower impact limiters.

The CCV fits within the cavity of the OSV and 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 diameter of 32.5 inches by 47.0 inches high. The CCV cylindrical shell and bottom plate are both 1.0-inch thick. The CCV lid, which is 3.38 inches thick, is fastened to the CCV body by twelve (12) equally spaced 1-inch diameter socket head cap screws. The CCV includes an optional drain port that is used to drain water from the CCV cavity for wet-loading operations.

The OSV is a ductile cast iron vessel consisting of a body and lid. The OSV has an overall height of 61.5 inches and an outer diameter of 49.0 inches. The internal cavity of the OSV, with a diameter of 35.0 inches by 51.9 inches high, is large enough to accommodate the CCV with sufficient clearance to assure free differential thermal expansion. The OSV lid is fastened to the OSV body by twelve (12) 1 ¼ inch diameter heavy hex screws.

The cylindrical-shaped impact limiters fit over the upper and lower ends of the OSV assembly. The upper and lower impact limiters are identical, with a pocket that fits over the end of the OSV assembly and consisting of energy-absorbing closed-cell polyurethane foam cores sealed inside the exterior stainless-steel inner and outer shells.

The packaging may also be configured with a Shield Insert Assembly (SIA), within the cavity of the CCV, to provide additional gamma shielding. The SIA is a coated carbon steel container, placed inside the CCV cavity, that is either 1-inch thick, 2¼-inch thick or 3¾-inch thick. The internal cavity of the 1-inch thick and 2¼-inch thick SIA has a diameter of 24.0 inches and a length of 35.3 inches, whereas the 3¾-inch thick SIA has a diameter of 24.0 inches and a length of 41.0 inches. The 2¼ inch and 3¾ inch SIAs both include lids providing supplemental top end gamma shielding.

The maximum weight of the contents (including the CCV, SIA, dunnage or shoring) is 7,300 lbs. and the maximum gross weight of the package is approximately 32,000 lbs.

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

70000.14-L501	1P	Packaging Assembly – OPTIMUS®-H
70000.14-L510	0P	CCV Assembly – OPTIMUS®
70000.14-L511	0P	CCV Body Weldment – OPTIMUS®
70000.14-L512	0P	CCV Lid – OPTIMUS®
70000.14-L513	0P	Port Cover – OPTIMUS®
70000.14-L520	0P	OSV Assembly – OPTIMUS®-H
70000.14-L521	2P	OSV Body – OPTIMUS®-H
70000.14-L522	1P	OSV Lid – OPTIMUS®-H
70000.14-L530	1P	Impact Limiter Assembly - OPTIMUS®-H
70000.14-L550	1P	1-Inch Shield Insert Assembly (SIA) - OPTIMUS®
70000.14-L551	1P	2¼-Inch Shield Insert Assembly (SIA) - OPTIMUS®
70000.14-L552	1P	3¾-Inch Shield Insert Assembly (SIA) - OPTIMUS®

Radioactive contents of the package include two classes of waste, transuranic containing waste (TRU) and irradiated fuel waste (IFW), as summarized in Table 1.

Table 1 - Content Designations

Contents			Content ID
Class	Type	Sub-Type	
TRU waste	Compliant	---	1-1
	Non-Compliant	Aerosol Cans, Type 1	1-2A
		Aerosol Cans, Type 2	1-2B
		Standard DOT 3E Lecture Bottle	1-2C
Irradiated Fuel Waste	LEU	---	2-1
	CANDU	---	2-2

Contents designated as Content 1-1 include (ii) 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, either dewatered, solid, or solidified waste, (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, including reactor components or segments of components of waste from a nuclear process or power plant.

Contents designated as Content 1-2A include (i) TRU waste containing standard DOT 2P or 2Q, 1 liter aerosol cans containing compressed gas propellant.

Contents designated as Content 1-2B include TRU waste containing standard DOT 2P or 2Q, 1 liter aerosol cans with liquified gas propellant or unknown propellants.

Contents designated as Content 1-2C include TRU waste containing standard DOT 3E lecture bottles with known contents of non-flammable gases or with unknown contents.

Content 2-1 is Radioactive material in the form of low-enriched uranium (LEU) waste with or without activated metal structural components (e.g., cladding, liners, baskets, etc.) (Content 2-1).

Content 2-2 is irradiated CANDU fuel waste contents and irradiated hardware contents. Irradiated CANDU fuel contents are restricted to natural UO_2 fuel of a typical CANDU fuel bundle design with cladding and bundle structure comprised exclusively of Zircaloy material, as well as CANDU fuel baskets of irradiated hardware materials.

The maximum quantity of material per package for Content 1-1 is 7,300 pounds including radioactive waste, secondary containers, SIA, and shoring. Fissile contents must not exceed the fissile gram equivalents (FGE) in Table 2 for the specified criticality configuration limits. Plutonium contents in quantities greater than 0.74 TBq (20 Ci) must be in solid form.

Table 2 - TRU Waste FGE Limits

	FGE Criticality Configuration Description				FGE Limit, (g ^{239}Pu)
	Machine Compacted	Weight % Special Reflector	Minimum ^{240}Pu Credit	Chemically or Mechanically Bound	
FGE-1		≤ 1			335
FGE-2a		≤ 1	≥ 5 g		350
FGE-2b		≤ 1	≥ 15 g		370
FGE-2c		≤ 1	≥ 25 g		390
FGE-3		> 1			121
FGE-4		> 1		✓	320
FGE-5	✓	≤ 1			245

The maximum quantity of material is 95 standard DOT 2P or 2Q 1 liter aerosol cans (content 1-2A), 4.4 liters of liquified gas propellant in any number of standard DOT 2P or 2Q 1 liter aerosol cans (Content 1-2B), 8 full Standard DOT 3E Lecture bottles with known contents of non-flammable gases (Content 1-2C)

Irradiated Fuel Waste shall not exceed the Fissile Equivalent Mass (FEM) limits from Table 3 for the specified criticality configuration limits and shall comply with the dose rate limits, as stated in Attachment 7.5-1 of the Application. Irradiated CANDU fuel waste contents shall have a maximum burnup of 5 GWd/MTU, a minimum cooling time of 40 years, and a maximum fuel mass of 1808.8 kg (3,988 lb) UO_2 . The maximum weight of contents, including IFW in a secondary container, internal structures (e.g., SIA, etc.) and dunnage or shoring shall not exceed 7,300 lbs (3,311 kg). CANDU fuel baskets of irradiated hardware materials are limited to 2.5 kg (5.5 lbs) of Inconel, Kg (441 lbs) of stainless steel, and unlimited Zircaloy per package.

Table 3 - IFW FEM Limits

Config. ID	LEU Waste Criticality Configuration Description			Uranium Mass Limit (lb)
	Weight % Special Reflector	Enrichment Limit (wt% U-235)	Particle Size Restriction (cm)	
FEM-1	≤ 1	≤ 0.96 wt%	≤ 0.1 and/or ≥ 8.0	5,000
FEM-2	≤ 1	≤ 0.80 wt%	N/A	

The Criticality Safety Index (CSI) of the package is 0.0.

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 EVALUATIONS

2.1 STRUCTURAL EVALUATION

The objective of the U.S. Nuclear Regulatory Commission's (NRC's) structural evaluation is to verify that the applicant has adequately analyzed the structural performance of the transportation package (packaging plus contents) so that it meets the regulations in Title 10 of the 10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

2.1.1 Description of Package Components

The Model No. OPTIMUS®-H is designated as Type B(U)F-96 per 10 CFR 71.4. The radioactive contents of the package include Type B quantities of normal form Transuranic (TRU) waste and Irradiated Fuel Waste (IFW) contents.

The OPTIMUS®-H packaging consists of a Cask Containment Vessel (CCV), an Outer Shield Vessel (OSV), and upper and lower impact limiters, together referred to as the impact limiter System (ILS). The CCV fits within the cavity of the OSV and the upper and lower impact limiters are attached to the respective ends of the OSV. The three main components of the packaging are identified as the CCV, OSV and ILS. The packaging may also be configured with a Shield Insert Assembly (SIA) within the cavity of the CCV to provide additional shielding when required to demonstrate compliance of the contents with regulatory dose rate limits.

The applicant provided licensing drawings with tolerances, dimensions, welding symbology, and definitions, material designation, and associated standards. Safety Analysis Report (SAR) Section 1.3.3 lists these drawings as Drawing No. 70000-14-501 through No.70000.14-552 and includes component descriptions and the arrangement of components relative to each other.

The CCV is the containment vessel for the contents of the package. The CCV shell is constructed from American Society for Testing and Materials (ASTM) Type 304 or 316 stainless steel plates welded to form a cylindrical shell. The shell is closed at the bottom by welding a 1.0-inch-thick plate of the same material as the shell. The top of the CCV body has a welded tapered stainless steel, ASTM SA-182, Type F304 or F316 or SA-240, Type 304 or 316, forged flange to accommodate a bolted lid. The smaller diameter of the forged flange is welded to the shell body and the larger diameter at the top accommodates the lid and lid bolts. The CCV lid includes a port used for inerting the CCV cavity and contents. The lid is equipped with bolted

port covers and O-ring seals, designed to make the containment leak-tight for the contents in accordance with the criterion of American National Standards Institute (ANSI) N14.5-2014. SAR Figure 1-2 shows an expanded view of the CCV assembly, and the materials of construction are listed in SAR Table 2.2.1. The internal cavity of the CCV is large enough to accommodate a 110-gallon drum. The CCV lid, which is approximately 3.38 inches thick, is fastened to the CCV body by twelve (12) closure lid bolts made of steel. The lid port is sealed with a Type 304 or 316 port cover and bolted. The CCV is designated as a Category I component based on the cask contents.

The OSV is a 7-inch thick-walled ductile cast iron (DCI) vessel with a three plus inch thick DCI bolted lid, which is closed by 12 closure bolts. The bolts are fitted in threaded anchor sleeves of the same material. The OSV is sized to contain the CCV and protect it from direct impact and fire in the event of a transportation accident. An expanded view of the OSV assembly is shown in Figure 1-3. The OSV is cast from DCI as a monolithic unit including the four (4) tiedown lugs, two (2) lifting trunnions, and twelve (12) impact limiter attachment brackets on the exterior of the sidewalls. The OSV has an overall height of around 61.5 inches with an approximate outer diameter of 49.0 inches, excluding protrusions such as the lifting trunnions, tiedown lugs and impact limiter attachment brackets. The internal cavity of the OSV is large enough to accommodate the CCV with sufficient clearance to assure free differential thermal expansion under Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC).

The ILS consists of two (2) identical foam-filled impact limiters designed to deform and absorb energy when subjected to NCT and HAC free drops, thereby limiting the load imparted to the OSV and CCV. The impact limiters are constructed from stainless steel sheets completely encasing energy-absorbing closed-cell polyurethane foam core, protecting the foam core from the external environment. The impact limiter inner stainless-steel weldment is a robust structure constructed from ½-inch thick Type 316 stainless steel plate with six (6) 1-inch-thick lugs. The outer shells of the impact limiter are constructed from 14-gauge Type 304 or 316 stainless steel sheets with rolled angles for corner supports. The cores of each impact limiter are comprised of two different densities of closed-cell polyurethane foam for optimal performance in the NCT and HAC free drop tests. The impact limiters are secured to the OSV body by six (6) turnbuckle jaw-end fittings, or swing bolts. The swing bolts are connected to attachment lugs on the OSV body. SAR Figure 1-4 shows such an attachment. Stainless steel rub strips are attached to the inside pocket of each impact limiter to minimize wear between the OSV and impact limiter. Each impact limiter includes three (3) drain tubes. The drain tubes prevent water from collecting inside the lower impact limiter.

The package can accommodate a painted carbon steel SIA of 1 inch and 2 1/4-inch to 3 ¾ inch thicknesses added inside the CCV for additional shielding, as required. These inserts are not considered as structural elements of the package and do not contribute to its structural integrity.

The OSV has two diametrically opposite trunnions for lifting and four (4) tiedown lugs. The trunnions can, in addition, be used as tiedown points during transportation. A general arrangement of the component parts is shown in Figure 1-3. The OSV trunnions and tiedown lugs are structural parts of the packaging.

In this section of the safety evaluation report (SER) the staff evaluates the SAR information to ensure that the applicant has satisfied the requirements of 10 CFR 71.31(a)(1) and 71.31(a)(2), 10 CFR 71.33(a) and (b), and 10 CFR 71.35(a). SAR Section 2.1 provides a description of the structural design of the package. SAR Sections 2.6 and 2.7 address the structural performance of the package under NCT and HAC, as required by 10 CFR 71.35(a) and 10 CFR 71.31(a)(2).

The applicant provided a description of the packaging in accordance with 10 CFR 71.33(a):

- (1) SAR Section 1.1 classifies the package as Type B(U)-F and identifies its contents characteristics in SAR Tables 1-1 through 1-4.
- (2) SAR Table 2.1-8 provides the gross weight of the package as 31,300 pounds.
- (3) SAR Section 1.1 identifies the NAC transport package as model number OPTIMUS®-H.
- (4) In SAR Section 1.1 states that the package has one containment provided by the CCV and its closure system, and an outer shell, OSV, for shielding and protection of the CCV.
- (5) The materials of construction and component dimensions are identified in the design drawings listed in SAR Section 1.3.3. SAR Section 2.1.3 provides the weights and the center of gravities of the different components of the package.

The applicant provided a description of the package contents in accordance with 10 CFR 71.33(b):

- (1) SAR Tables 1-1 through 1-4 establish the maximum radioactivity of the constituents, the maximum quantities of fissile material and additional details on the maximum normal operating pressure, maximum weight, and the maximum amount of decay heat in SAR Section 1.2.

The staff reviewed the package structural design description and concludes that the contents of the application include a description of the proposed package in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation of the package. The application satisfies the requirements of 10 CFR 71.31(a)(1) and 71.31(a)(2), 10 CFR 71.33(a) and (b), and 10 CFR 71.35(a).

2.1.2 Identification of Codes and Standards Used for Package Design

SAR Section 2.1.4 identifies the codes and standards used in design, fabrication, testing and maintenance of the package. The OPTIMUS®-H CCV is identified as a Category I container. The applicant used the guidance in RG 7.6 and NUREG/CR-3854 in selecting the codes and standards. The design code selected for the containment (CCV) is consistent with categorization as Category I, which is American Society of Mechanical Engineers (ASME) Code, Section III, Division 1, Subsection NB.

The non-containment structural portions of the packaging are designed to applicable requirements from ASME Code, Section III, Division 1, Subsection NF. Design criteria for behavior under specific types of loads are identified in SAR Chapter 2

The material standards used for the package comply with ASTM and ASME Section II, Part D, for the package. For simulation analyses, the applicant has used LS-DYNA R5.1.1 (2012) and used ANSYS 19.0 to perform the structural analyses. The fatigue analysis of the CCV and port cover closure bolts is conducted in accordance with ASME Section III, NB-3222.4 and NB-3232.3. The guidance in NUREG/CR-6007 is used to analyze bolt stresses of the package under NCT and HAC. The applicant designed the lifting attachments of the OPTIMUS®-H package in accordance with the requirements of ANSI N14.6 for special lifting devices for critical lifts.

The package's containment boundary undergoes elastic deformation when subjected to drop tests for both NCT and HAC. The damage is confined to the ILS and does not result in any inelastic deformation in the CCV and OSV. The applicant evaluated the stress in the CCV using a 3-D ANSYS finite element model, which is described and characterizes the criteria used for elastic analysis 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 a factor 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 notes that this version of the Code Case has not been approved by NRC, but subsequent versions are approved without restrictions. The staff expects that the version reference will be corrected in the next amendment. Since the equations used in the stress analysis are the same as in the approved version, the analysis results are acceptable.

The staff finds that the applicant has identified the appropriate codes and standards to be applied to the procurement, design, fabrication, and examination of the components of the OPTIMUS®-H transport package, which complies with the requirements for the application in 10 CFR 71.31(c).

2.1.3 General Requirements for OPTIMUS®-H Package

Minimum Package Size

The minimum package dimension is required to be greater than 4 inches. SAR Section 2.4.1 states that the OPTIMUS®-H package has an overall height of 74.2 inches. The staff finds that the package satisfies the requirements of 10 CFR 71.43(a) for minimum size.

Tamper Indicating Feature

Access to the cask lid is prevented by the ILS. Access to the closure bolts of the package is prevented by wire cable tamper-indicating seals that are attached to the upper and lower IL attachment lugs on the body of the OSV. The wire is fixed such that the IL cannot be removed without breaking the wire. SAR Section 2.5.2 presents more details on the tamper-indicating feature. The staff finds that the package satisfies the requirements of 10 CFR 71.43(b) for tamper indicating features.

Positive Closure

The ILS prevents any access to the OSV lid and the CCV is completely enclosed within the bolted OSV. The CCV is bolted and the access port for potential inerting the CCV is closed by a bolted cover plate. The staff review finds that the positive closure of the containment is assured by the multiple access prevention layers. Based on the review, the staff finds that the package satisfies the requirements of 10 CFR 71.43(c) for positive closure.

Chemical, Galvanic, or other Reactions among packaging components

SAR Section 2.2.2 describes the package components to be made from stainless steel, coated carbon steel, cast iron and polyurethane foam. The materials do not have the potential for galvanic, chemical, or other reaction when exposed to the operating environment. SAR Section 2.2.2.1 further discusses the potential of interaction between the contents of the CCV. SAR Section 2.2.2.2 presents information on the potential of reactions between the contents and the packaging and SAR Section 2.2.2.3 presents information on the interaction between packaging.

The staff's review of the information presented in the different sections of the SAR determined that there is no significant potential for chemical, galvanic or other reactions between the contents and components of OPTIMUS®-H in air, helium, and water environments. The staff finds that the package satisfies the requirements of 10 CFR 71.43(d).

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 and finds that it satisfies the requirements of 10 CFR 71.43(e).

Absence of any continuous venting provision

The staff's review of the design drawings and information presented in the SAR did not reveal that the package has any provisions for continuous venting. The staff review finds that the package satisfies the requirements of 10 CFR 71.43(h).

The staff finds that the package satisfies the requirements of 10 CFR 71.43 as identified in the preceding SER Section 5.0.

2.1.4 Lifting and Tie-Down Points

Lifting Trunnions

The applicant described the qualification of the lifting trunnions for the OPTIMUS®-H package in SAR Section 2.5.1. The trunnion sleeve and base plate are fabricated from ASTM 240, Type 304 or 316 material. The trunnion is DCI and is integrated to the cask body during forging. The maximum lift weight per trunnion is 17,250 pounds. This load is evaluated to not exceed 1/3 the yield nor 1/5 the ultimate strength of DCI. In addition, the stress is further reduced by a factor of 0.5 as a single failure proof lifting device without redundancy. The dynamic load factor from crane operations further reduces the stress by another 15 percent (%). Thus, the resulting acceptable stress in the trunnion is $F_y/6 \times 1.15$ and $F_u/10 \times 1.15$. These stress limits are based on ANSI N14.6 and are lower than those required by 10 CFR 71.45(a).

The calculated bending stress and shear stress at the base of the trunnion is 1.8 ksi and 0.6 ksi, respectively. The design temperature based DCI yield stress is 25.4 ksi and the ultimate is 43.4 ksi, considering 60 % of allowable tensile as the shear allowable. The computed minimum margins for bending and shear are 1.33 and 3.17, respectively. The DCI cask wall is approximately 7 inches at the base of the trunnions. In the event there is a loss of the trunnions the wall has adequate thickness to maintain the other safety functions of the cask.

In addition to the trunnions the OSV has four (4) tie-down lugs which can be used to lift the package. Any two of these lugs can be used to lift the package. The F_y of DCI at 200°F is 25.4 ksi. The allowable lug loads for single plane fracture and double plane shear are computed using the methodology of ASME BTH-1-2005 and using the ANSI N14.6 stress limits.

The minimum design margins for the tie-down lugs for shear tear-out and bearing stress are 0.15 and 0.71, respectively. If under excessive loads the tie-down were to fail by shear tear-out, this would not impact the OSV in meeting the other requirements of 10 CFR 71.

The staff finds that the package complies with the requirements of 10 CFR 71.45(a) for lifting attachments.

Tie-Down Devices

In SAR Section 2.5.2, the applicant analyzes the tie-down attachments, which are four (4) lugs on the OSV body. The lugs are designed to withstand a static force applied to the center of gravity of the package defined by 2g vertical, 5g lateral and 10g along the direction of travel without exceeding the package material yield. SAR Figure 2.5-1 shows the tie-down arrangement using the tie-down lugs. Based on this configuration, the applicant finds the maximum tie-down tensile force is 131 kips. The minimum shear tear-out margin in the lug is +0.39, using the resulting shear tear-out stress and a shear allowable of 60% of base material tensile allowable.

The minimum margin for bearing stress was computed as +0.07, using the maximum bearing stress between the shackle pin and the lug with the DCI yield at 200°F. The computed combined tensile and shear stress (von Mises stress) at the base of the tie-down lug results in a design margin of +0.60 for combined tensile and shear.

The results show that under excessive load the failure of the tie-down would occur due to shear tear-out instead of failure at the base of the lug. Such a failure does not impair the ability of the package to perform the other requirements of 10 CFR 71 as the cask itself would not be compromised and the lug is therefore not required for any safety function.

Package tie-down configuration using the trunnions is shown in SAR Figure 2.5-2. The computed combined tensile and shear stress at the base of the trunnion results in a design margin for combined stress of +0.26.

Staff review finds that the package complies with the requirements of 10 CFR 71.45(b)(1) and (b)(3) for tie-down attachments.

2.1.5 Drop Evaluation Methodology

Package drop scenarios are evaluated under the NCT and HAC and follow the same analytical process for both conditions. In the analysis for the OPTIMUS®-H package, the applicant has considered the effect of deformation of the ILS in computing the acceleration and stresses in the OSV and CCV. The design considered that the kinetic energy of the drop is fully absorbed by the deformation of the impact limiter. The impact force on the cask is related to the yield strength of the impact limiter material and the contact area at the time of impact, which determines the maximum intensity and duration of impact on the cask.

A FE model using the LS-DYNA computer code captures the performance of the ILS, including deforming and absorbing the drop energy and for recording the rigid body acceleration time-history of the cask and contents. The maximum rigid body acceleration along with a dynamic load factor is used in the equivalent-static-linear-elastic analysis of the cask using the FE model in the ANSYS computer code. The LS-DYNA analysis demonstrates the structural adequacy of the ILS in the free drop tests and that the ILS does not bottom out. The staff reviewed the application to evaluate compliance with the requirements of 10 CFR 71.41(a).

LS-DYNA Model

The applicant used the LS-DYNA explicit dynamic finite element code to simulate the response of the package to the NCT free drop, HAC free drop, and HAC puncture tests. A full-scale, half-symmetry model of the package was developed using Autodesk Inventor and the ANSYS Workbench was used to create the FEM. This FEM was imported into LS-DYNA and the material models added for the dynamic analysis. SAR Section 2.6.7.1 discusses the essential parts of the FEM developed using the elements and material models in the LS-DYNA code. A 3-D, half symmetric model of the package was developed in LS-DYNA and is shown in SAR Figure 2.6-4. Individual finite element models of OSV and CCV are shown in SAR Figures 2.6-6 and 2.6-7 respectively. The OSV and CCV lid models both include features important for evaluation of closure.

The model of the impact limiter is shown in SAR Figure 2.6-5 which includes the end and corner/side foam cores and all steel components except the steel angle supports at the outer corners of the shell weldment. The mesh size is reduced in areas as needed to capture the areas of stress concentration and locations with the significant hourglass energy are modeled using fully integrated selective-reduced solid elements. Fully integrated shell elements are used to model the sheet metal of the IL. The non-linear contact between various components of the package modeled with surface-to-surface contacts. The foam is modeled using crushable foam material model. The dynamic compressive stress strain properties used for the densities of the foam are taken from SAR Figure 2.2-1 and Figure 2.2-1, respectively.

Material Model

The piecewise-linear plasticity material model is used for the components of the ILs, OSV and CCV for the input of stress strain and definition of failure based on the plastic strain. The true stress-true strain data is used for stainless steel and DCI. The bolt shafts are modeled using a plastic kinematic hardening material model using a bilinear stress strain curve. All steel properties are conservatively based on an upper-bound temperature of 300°F.

Drop Analysis Simulation Model

The details of the drop analysis are presented in Calculation: CN-16007-204 Rev. 2 referred to here as the calculation. Table 2.3-1 in the calculation provides a summary of the boundary conditions for the LS-DYNA analysis of OPTIMUS®-H. The summary includes the condition, drop orientation drop angle, temperature, and payload.

For the polyurethane foam used in the OPTIMUS®-H ILS, the calculation states that the material is treated as is treated as an isotropic material as the foam crush strength data shows little sensitivity to grain direction. A foam acceptance criterion is specified by an average static compressive strength at room temperature as a percentage of the nominal crushing value parallel and perpendicular to the direction of foam rise. The dynamic crush strength of the foam was computed in Calculation: CN -16007-214 and used in the OPTIMUS®-L certification. It was calculated using the static strength and the regression data provided by the vendor. Calculation: CN-16007-204 Rev. 2. Tables 4.2.3.1 and 4.2.3.2 provide the dynamic crush strength data used as input to the LS-DYNA analysis.

For each case the boundary condition is set by the initial velocity of the package just before contact with the rigid drop surface. This is computed using the drop height and conversion of the drop kinetic energy to potential energy. The maximum rigid body acceleration is extracted from

the LS-DYNA files. The Dynamic Load Factor (DLF) is computed using the guidance in NUREG/CR-3966. The highest DLF resulted from the shortest acceleration time-history pulse duration and the longest natural period of the package. The DLF is conservatively estimated as 1.13. The product of the maximum rigid body acceleration and the DLF is used in the stress analysis of package.

Evaluation of Simulation Model

The staff reviewed the acceleration time-history plots and the energy balance plots of different drop scenarios, which indicated that the kinetic energy of the drop was fully dissipated, and the acceleration had peaked. The internal energy plots showed how much of the kinetic energy was converted to strain via the elastic and inelastic deformation in the package. The plots in addition showed the trace of the hourglass energy—numerically produced proportional to the strain energy used to control the distortion in the solid finite elements.

The plot of the sliding energy provides information on how well the contact surfaces responded to the drop simulation. The simulation included a provision to adding external work which is needed to capture additional work done during a corner drop as the model rotates about the corner.

The review indicates that the LS-DYNA model of the OPTIMUS®-H package has the features to capture the energy balance of the package under different drop scenarios. The results show that the initial kinetic energy is converted into strain energy due to crushing of the impact limiter.

The hourglass energy plotted for several scenarios in 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 plots in the calculation, the staff finds that the LS-DYNA modeling approach of the drop cases is acceptable.

2.1.6 Benchmarking and Validation

The information on benchmarking of the OPTIMUS®-H package is presented in SAR Section 2.12.3. The applicant scaled the 30-foot one-quarter ($\frac{1}{4}$) scale side drop test results of NAC-UMS package for comparison to a one-half ($\frac{1}{2}$) scale NAC-UMS package, as a full-scale OPTIMUS®-H package is like a $\frac{1}{2}$ scale NAC-UMS as shown in SAR Table 2.12-1. The acceleration time-history curve from the $\frac{1}{4}$ -scale NAC-UMS drop test was adjusted using mass scaling laws (i.e., accelerations divided by two and time multiplied by 2) for comparison to the simulated results for a $\frac{1}{2}$ -scale NAC-UMS. Figure 2.12-1 shows the acceleration time-history curve for the OPTIMUS®-H package (labeled "SMP").

The applicant states that even though the SMP impact limiter utilizes foam, and the UMS uses redwood, the nominal stress strain curve of the material are similar. Therefore, adjusting the difference in the material stiffness would be appropriate for scaling the drop response. The benchmarking is to ensure that the FEM used in the simulation along with the material models of analysis can replicate the response of an actual drop test of a similar package.

The staff review of the benchmarking of the LS-DYNA acceleration T-H simulation for OPTIMUS®-H package finds that the model of the package can simulate the drop scenarios to be considered under the NCT and HAC. The benchmarking comparison shows that the simulated response of the OPTIMUS®-H package is the response of ½ scale UMS package scaled from the results of a ¼ scale drop test result of the UMS package. The staff finds this as an acceptable demonstration of the fidelity of the FEM in LS-DYNA for drop simulations.

ANSYS Model

The applicant used the ANSYS computer program to generate a three-dimensional model of the package and determine its response to NCT and HAC. The ANSYS code performed an equivalent static analysis with bounding g-loads calculated from 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 higher 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. Postprocessing 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 under combined load cases.

The applicant used the FEM within the LS-DYNA computer code to determine the acceleration response of the package to drop scenarios and the stress analysis of component parts using ANSYS to ensure that the closure design of the package will not fail under NCT and HAC load conditions.

The staff reviewed the approach of developing the ANSYS model and concluded that the model can perform the stress analysis of the package components under combined loads resulting from NCT and HAC. Based on this review the staff concluded that the ANSYS model for the package is acceptable.

2.1.7 Normal Conditions of Transport

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

Heat

The applicant stated that package ambient temperature conditions correspond to an ambient temperature of 100°F, with solar insolation. This matches the 38°C ambient temperature required by 10 CFR Part 71.71. NCT heat is a combination of ambient temperature, maximum decay heat, maximum insolation, maximum internal pressure, and fabrication stresses. The stresses in the CCV are computed using the ANSYS model. The results of the NCT heat stress analysis show that the maximum total ($P_b + P_m + Q$) stress intensity of 20.2 ksi is at the center of the bottom plate. The allowable $P_b + P_m + Q$ stress intensity is 60.0 ksi. The minimum design margin of the CCV due to the heat load is +1.97. The maximum separation between the CCV lid and bolting flange is less than 0.3 % of the O-rings nominal compression. The maximum compression set for the O-ring is 15.6 %. Therefore, the CCV containment seal will be

maintained under the NCT heat load without any increase in the external surface radiation levels.

The staff concluded that the ambient heat requirements for the package satisfy the standards of 10CFR 71.71(c)(1) and meets the requirements of 10 CFR 71.71(a). The package in addition complies with the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) when subject to the NCT heat test.

Differential Thermal Expansion

The applicant considered differential thermal expansion of the package as described in Section 2.6.1.2.1 of the application. For the thermal expansion between the CCV and the OSV, the applicant evaluated it conservatively using hand calculations, assuming an upper-bound temperature of 330°F, for the CCV, and a lower-bound temperature of 200°F, for the OSV. The calculation results show that differential thermal expansion between the CCV and OSV reduces the nominal axial and radial clearances to 0.30-inch and 0.22-inch., respectively. Therefore, the CCV will expand freely within the OSV 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 clearance to 0.28-inch and 0.17-inch, respectively. 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 in SAR Section 2.6.2 evaluated the effect of a steady -40°F ambient state in air and shade consistent with the requirements of 10 CFR 71.71(c)(2). The NCT cold condition is evaluated in combination with zero insolation, zero decay heat and zero internal pressure, resulting in a uniform -40°F throughout the package. Because the coefficient of expansion of the bolt is lower than that of the lid, the cold temperature will reduce the CCV bolt stress but not enough to overcome the bolt preload maintaining the closure seal.

The NCT cold structural evaluation shows that the allowable stress in the package from other NCT load combination to be within the package design allowable stress. The NCT cold does not result in any loss of containment or increase the external surface radiation level.

Thus, the staff finds that the ambient cold requirements for the package satisfy the standards of 10 CFR 71.71(c)(2) and meets the requirements of 10 CFR 71.71(a). The package in addition complies with the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) when subject to the NCT cold test.

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 psi. 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 psi. Hence, the greatest pressure difference between inside and outside of the containment system is applied for the design. SAR Section 2.6.3 presents the details of the applicant's analysis of the package under reduced external pressure conditions using the ANSYS model.

The results of the applicant's analysis for the NCT reduced external pressure is presented in SAR Table 2.6-1. The staff's review of the results show that the package containment system satisfies the ASME allowable stress design criteria with a minimum design margin of +0.69. The maximum stress ratio in the CCV closure bolts due to NCT reduced external pressure loading is summarized in SAR Table 2.12-6, which shows the maximum stress ratios as 0.98 from the average tensile stress. This results in a minimum margin of safety of +0.02. Reduced external pressure loading does not cause any permanent deformation of the package to reduce the effectiveness of the packaging, which would have resulted in the loss or dispersal of radioactive contents, thereby increasing the surface radiation levels.

The staff concluded that the reduced external pressure requirements for the package satisfy the standards of 10 CFR 71.71(c)(3) and meets the requirements of 10 CFR 71.71(a). The package in addition complies with the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) when subject to the NCT reduced pressure test.

Increased External Pressure

SAR Section 2.6.4 discusses the effect of increased external pressure on the package. 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. This results in a net design pressure differential of 5.3 psi. This increase is not evaluated for the SIA and the OSV as they are not pressure retaining boundaries. The magnitude of the external pressure load for deepwater immersion is 290 psi which 14.5 times greater than the 20-psi increased external pressure. The ratio of HAC-to-NCT allowable stress limits is slightly greater than 2, therefore the margin in this condition will be 25 times those reported for the deep water immersion.

Thus, the staff concludes that the increased external pressure requirements for the package satisfy the standards of 10 CFR 71.71(c)(4) and meets the requirements of 10 CFR 71.71(a). The package in addition complies with the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) when subject to the NCT increased pressure test.

Vibration

SAR Section 2.6.5 addresses assessment of the package 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 the bottom IL and tied down by using the four (4) tiedown lugs or the two (2) trunnions on the OSV body.

Based on testing performed in part by Sandia Laboratories, the peak vibration accelerations for transport are much lower than those resulting from the NCT free drop evaluated in SAR Section 2.6.7. SAR Table 2.12-6 summarizes the stress in the OSV and CCV closure bolts in combinations containing the NCT vibration loads. The minimum margins of safety in the CCV and OSV closure bolts for NCT vibration based on stress ratio limit are +0.04 and +3.35 respectively.

Thus, the staff concludes that the vibration requirements for the package satisfy the standards of 10 CFR 71.71(c)(5) and meets the requirements of 10 CFR 71.71(a). The package in addition complies with the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) when subject to the NCT vibration loading.

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 two in/h for at least 1 h. The applicant stated that the CCV assembly is isolated from the quenching effects of the water spray by the OSV 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).

NCT Free Drop

SAR Section 2.6.7 discusses the details of the NCT free drop analysis. The drop analysis is conducted using the FEM with the LS-DYNA computer code to compute the response of the ILS under the drop conditions and the development of acceleration time-history as the ILS deforms under the impact. The different drop orientations considered in the drop analysis under NCT is shown in SAR Figure 2.6-3. A summary of the free drop cases evaluated are listed in SAR Table 2.6-2.

The drop analysis is used to predict the acceleration loading on the CCV, OSV and contents for each NCT drop impact analyzed. The maximum rigid body acceleration amplified by the DLF is then used with the ANSYS FEM to evaluate the loads in the different components of the package. The maximum stress is then compared to the applicable allowable stress limits of the design criteria. The results of NCT free drop evaluation are summarized in SAR Table 2.6-3.

The table shows that the highest tensile force in the ILS attachment bolt results from the NCT side drop and the highest tensile force in the OSV and CCV closure bolts result from the NCT corner drop. These loads are combined with other loads and compared with the design criteria allowable. The comparison of the allowable to the demand shows that the impact limiter attachments satisfy the applicable design criteria for NCT drops.

The stress in the OSV and CCV are evaluated using the ANSYS model and the equivalent static g-values for each case as shown in SAR Table 2.6-3. The stress is evaluated at the locations shown in SAR Figure 2.1-1. The stress summary of the end drop and side drop are listed in SAR Table 2.6-4 and Table 2.6-5.

Stresses in CCV and OSV Components

The applicant has used the guidance in RG. 7.8 for establishing the load combinations of the drop scenarios with Minimum Normal Operating Pressure (MNOP), NCT heat and bolt preload stresses. SAR Tables 2.6-4 and 2.6-5 provide the stress summaries for the NCT top end drop and NCT side drop, respectively, at controlling locations. The controlling locations are shown in SAR Figure 2.1-1.

The minimum design margin is +0.05 for primary membrane plus bending ($P_m + P_b$) stress intensity in the CCV shell (at stress Section C5) for NCT side drop. The minimum margin for the OSV in primary bending plus membrane at (stress Section N9) of the lid is +0.51 due to the NCT end drop. Therefore, the packaging satisfies the applicable allowable stress design criteria for the NCT free drop.

The separation of the CCV lid from the bolting flange is 5.2% of the O-ring's nominal compression. For the CCV containment to be retained the maximum O-ring compression must not exceed 14.8%. The maximum O-ring compression set for combined temperature and radiation does not exceed this value. The containment function of the CCV is retained during the NCT conditions.

The stresses in the CCV and OSV closure bolts due to NCT free drop are determined using the methodology described in NUREG/CR-6007. SAR Section 2.12.5 provides a detailed discussion of the analysis including establishment of the thread engagement length. Two separate load combinations are evaluated for NCT free drop loading. The first (L.C. N5) combines NCT free drop with NCT heat temperature loading, MNOP, and maximum bolt preload and the second (L.C. N6) combines NCT free drop with NCT cold temperature loading, MNOP, and minimum bolt preload. The minimum margins of safety in the CCV and OSV closure bolts for the NCT free drop are shown in Table 2.12-6 as a stress ratio limit of +0.04 and +3.35, respectively.

The staff concluded that the free drop requirements for the package satisfy the standards of 10 CFR 71.71(c)(7) and meet the requirements of 10 CFR 71.71(a). The package in addition complies with the requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) when subject to the NCT free drop loading.

CCV Shell Buckling

The applicant in their SAR state that the buckling evaluations of the CCV shell are performed for the NCT free drop test in accordance with the requirements of ASME Code Case N-284-1. The maximum combined stresses used for the CCV shell buckling analysis are summarized in Table 2.6-6.

The stress is maximized to provide the most critical condition for buckling. The CCV shell stresses due to the NCT bottom end drop are combined with the stresses at the mid-length of the CCV shell (i.e., Section C7 in Figure 2.1-1) resulting from NCT increased external pressure and NCT heat. For increased external pressure loading the internal pressure is conservatively assumed to be zero, resulting in a net external pressure load of 5.3 psig. The maximum combined stresses used for the CCV shell buckling analysis are summarized in Table 2.6-6. The allowable stresses and interaction ratios for elastic and inelastic buckling are summarized in Table 2.6-7.

The staff's review of the tables in the SAR indicates that the highest buckling interaction ratio in the CCV shell for the NCT free drop, including both elastic and inelastic buckling, is 0.53 for inelastic buckling due to axial compression plus shear, which is less than the limit of 1.0. Therefore, the CCV shell satisfies the buckling design criteria of ASME Code Case N-284-1 for the NCT free drop.

Stresses in SIA

The stresses in the 1-inch, 2¹/₄-inch and 3³/₄-inch SIA are evaluated in SAR Section 2.6.7.2.2 for NCT free drop loading. The free drop loads in the inserts are combined with the other loads using the guidance in RG 7.8. The shield inserts are designed to ASME Subsection NF. Most of the loads are eliminated because of the function of the inserts. Consequently, no other loads are combined in the stress analysis. The applicant reviewed the different sizes and the boundary conditions of the SIA and determined the absolute minimum margin of safety for each drop condition.

For the Top End Drop: The corresponding minimum margin of safety in shear is +0.12, based on an allowable shear stress of 13.4 ksi for A516, Grade 70 carbon steel at 300°F. The corresponding minimum margin of safety for primary membrane stress is +1.46, based on an allowable primary membrane stress of 22.4 ksi for A516, Grade 70 carbon steel at 300°F.

For the Bottom End Drop: The allowable primary membrane and membrane plus bending stress intensities are 22.4 ksi and 33.6 ksi, respectively. The corresponding minimum margin of safety for primary membrane and membrane plus bending stress intensity is +0.19.

For the Side Drop: The corresponding minimum margin of safety for shear stress in the SIA shell-to-bottom plate is +0.51, based on an allowable average shear stress of 13.4 ksi for A516, Grade 70 carbon steel at 300°F.

2.1.8 Hypothetical Accident Conditions

HAC Free Drop

Like NCT conditions, the acceptance criteria used by the applicant for the HAC free drop was to demonstrate that the port cover and screws are undamaged during HAC, and that the CCV is not breached (containment boundary). In SAR Section 2.7 the applicant describes the response of the package when subject to the HAC per the requirement of 10 CFR 71.73. The structural evaluation for HAC is based on a sequential application of the HAC tests specified in 10 CFR 71.73(c) to determine the cumulative effect on the package as required by 10 CFR 71.73(a).

SAR Section 2.7 presents the assessment of the HAC free drop conditions. LS-DYNA results provide the rigid body accelerations for each HAC free drop and demonstrate the structural adequacy of the impact limiter assembly. The results of the analysis show that the tensile loads in the IL attachments are lower than the ultimate tensile capacity of the IL attachment, and the crush depth of the foam is lower than the allowable crush depth for each HAC drop.

The stress analysis of the CCV and OSV is performed using the acceleration input from the drop amplified by the DLF in the ANSYS model. SAR Table 2.7-1 summarizes the different free drop conditions considered for HAC and Figure 2.7-1 shows their orientations. The drop tests considered the 30-foot free drop, and the puncture test for cumulative damage with relevant package orientations.

The applicant also considered ambient temperatures ranging from -40°F (Cold) to +100°F (Hot). The HAC free drop was evaluated for the heaviest content weight of 7300 lbs. including the weight of the CCV bottom support plate. Upper-bound and lower-bound (500 lbs) analyses were performed for each HAC free drop impact orientation. The package is evaluated for a total of five (5) different HAC free drop orientations. They include upper-bound and lower-bound analyses for a top end drop, top corner drop, horizontal side drop, 5-degree bottom end oblique drop, and 10-degree bottom end oblique drop.

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, resulting in 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 and the tests were conducted as required in 10 CFR 71.73(a) under the test condition prescribed in 10 CFR 71.73(b) and in the sequence required by 10 CFR 71.73(c).

End Drop

SAR Table 2.7-2 summarizes the results of the HAC end drop IL analysis and list the crush depth as a % of the total foam thickness, considering the equivalent static acceleration experienced by the package and the maximum bolt tension in the ILS, OSV and CCV. The table

shows that the maximum crush depth occurs under the Hot/Heavy case at 43% of the nominal foam dimension at the top end and the maximum ILS, OSV and CCV bolt tension occurring in the Cold/Light case at the top end, as the equivalent static acceleration is also the highest in this case.

Side Drop

SAR Section 2.7.1.2 presents the information on the results of the package side drop. SAR Table 2.7-6 lists the results of impact limiter analysis for the side drop condition. The results show that the maximum crush depth occurs for the Hot/Heavy condition at around 55% of the nominal dimension, and the ILS bolt tension is also the highest in this case. The tension in the OSV and the CCV lid bolts are insignificant as there are no longitudinal forces acting on the lid or bolts.

Oblique Drops

SAR Section 2.7.1.4 presents the results of the oblique drop of the package. The maximum accelerations resulting from each HAC oblique drop test case evaluated are summarized in Table 2.7-9. The impact limiter deformations resulting from the 5° oblique drop (Case HO1) and 10° oblique drop (Case HO2) are shown in Figure 2.7-11 and Figure 2.7-13, respectively. The plots show that the highest rigid body accelerations occur during the primary impact and not during the secondary impact as the aspect ratio of the package is low. The closure bolt forces in the ILS attachments, OSV and CCV are low as there is only a small longitudinal component to the drop. The results of the oblique drop are bounded by the results of the corner and end drop.

Corner Drop

SAR Section 2.7.1.3 discusses the applicant's analysis of the corner drop. Table 2.7-8 presents the results of the HAC corner drop analysis. The results show that the maximum crush occurs during the Hot/Heavy drop condition for the corner drop, with 77% crush of the nominal foam dimension. The maximum tension in the CCV bolt occurs under the Cold/Light condition. The maximum tension in the ILS bolts is under the Cold/Heavy condition. Of all the drop scenarios the corner drop causes the maximum crush and the highest tension in the ILS bolts. The nominal foam thickness in this case is 15.1 inches and the foam does not bottom out under this loading. The highest tensile load on the impact limiter attachment is 38.8 kips. The tensile ultimate for the turnbuckle jaw-end is 50 kip. Based on this load the safety margin of the impact limiter attachment bracket and impact limiter lug is +0.38 and +0.91, respectively.

Stresses in CCV and OSV Components

The stresses in the CCV and OSV are computed and summarized in SAR Table 2.7-3 for end drop cases and Table 2.7-7, for side drop cases. The staff's review of these tables show that the OSV and CCV stresses are bounded by the results of the end drop and side drop. SAR Section 2.7.1.5 presents a summary of the results of the HAC free drops based on the review of the summary. Based on the tabularized results, the staff concluded that the HAC free drops do not cause any permanent deformation in the OSV and CCV. The drop impact energy is entirely absorbed by the ILS.

The staff finds that the induced stresses have minimum design margins greater than 1.0 and maximum stress intensities lower than the allowable stress intensities.

Shell Buckling

SAR Section 2.7.1.1.3 addresses CCV shell buckling under the governing HAC end drop conditions in accordance with ASME Code Case N-284-1. The allowable buckling stresses are shown in SAR Table 2.1-7. The maximum axial compressive stress at the mid-length of the CCV shell occurs from the HAC bottom end impact. The maximum combined stress in the CCV shell occurred under HAC end drop and was used in the buckling analysis is shown in SAR Table 2.7-4.

SAR Table 2.7-5 shows the results of the buckling analysis along with the interaction ratios, which are less than 1.0. The minimum margin of safety against buckling of the CCV shell for the HAC bottom end drop is +0.69, demonstrating that the shell meets the buckling criteria.

The staff finds that the CCV shell buckling meets the design requirements of ASME.

Shield Inserts

The minimum design margin in the 3 ¾-inch SIA for the HAC end drop is +1.09 for axial compressive stress at the top end of the outer shell due to a top end drop. The minimum design margin in the 3 ¼ -inch SIA for the HAC side drop is +0.41.

The staff finds that the SIA are adequate for use with the package under HAC loads.

Fatigue

SAR Section 2.1.2.4 addresses fatigue and established that the evaluation of cyclic loading is not required for the packaging components other than bolts. Analysis of the packaging structural components for cyclic service is not required because the conditions stipulated in NB-3222.4(d)(1) through (6) are met, as explained in SAR Section 2.1.2.4.1.

The CCV closure bolts are evaluated for fatigue failure due to cyclic loading using the methods of NB-3221.9(e) in accordance with the requirements of NB-3232(d)(2). As per the requirements of NB-3232(d)(2)(d), a fatigue strength reduction factor of 4.0 is used for the CCV closure bolts. The analysis is conservatively based on the assumption that the CCV closure bolts will be replaced after 5 years of service and the packaging will be used for one shipment per week, for a total of 260 shipments over the life of the CCV closure bolts. The CCV closure bolt usage factor for startup-shutdown cycles (U1) is 0.65 and the CCV closure bolt usage factor for normal operating thermal and pressure cycles (U2) is 0.09 (1,825/19,640). S_{alt3} is much lower than S_a at the endurance limit of 1E6 cycles, and the usage factor for NCT vibration is insignificant (i.e., U3 = 0.00). Since the cumulative usage factor is less than 1.0, the CCV closure bolts will not fail due to fatigue during their 5-year design life transport.

The OSV closure bolts are not subject to high cycle (> 20,000) fatigue loading and do not require evaluation for fatigue failure per NF-3331.1. Although the number of significant vibration cycles may exceed 20,000, vibration loading does not produce significant stress in the OSV closure bolts. Therefore, the OSV closure bolts will not fail due to fatigue during their 5-year design life.

The staff concluded that the applicant's assessment of the effect of fatigue failure is acceptable.

2.1.9 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 1,100 lbs.

2.1.10 Puncture

The applicant addressed the puncture drop test in SAR Section 2.7.3. The puncture drop test is performed in sequence after the HAC free drop test in accordance with 10 CFR 71.73(a). Therefore, the package damage resulting from the HAC free drop is considered in the HAC puncture drop evaluation. The portion of the OSV shell that is not protected by the upper and lower impact limiters is 7.0-inches thick, which is more than 8.4 times greater than the thickness required to prevent perforation (estimated using Nelm's equation). Therefore, the OSV will not be perforated by a side puncture impact.

The packaging is also evaluated for HAC puncture impact on the side and top end to determine the extent of cumulative damage. The HAC side and top end puncture evaluations are performed using the 3-D half-symmetry LS-DYNA explicit dynamic finite element model. The HAC side puncture impact results in a maximum deformation on the exterior surface of the OSV sidewall of less than 0.15 inches.

For the HAC Top End Center Puncture impact, the results show no damage to the OSV or CCV. The resulting cumulative deformation of the impact limiter is shown in Figure 2.7-16. Therefore, the extent of package damage resulting from the top end off-center impact is limited to local deformation (i.e., denting) of the impact limiter outer shell and end foam and localized tearing of the impact limiter outer shell at the point of impact with the puncture bar.

Based on its review of the results of the puncture test, the staff finds the package meets the requirements of 10 CFR 71.73(c)(3).

2.1.11 Thermal

In 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 and maintains the integrity of the containment. SAR Table 2.7-10 summarizes the maximum stress intensities in the CCV components and compares them to the corresponding allowable. The minimum design margin is +0.67 for primary membrane plus bending stress intensity at the center of the CCV bottom plate. The maximum separation between the CCV lid and the bolting flange at the O-ring is 10% of the O-ring nominal compression. The maximum compression to maintain containment is set at 14%. The combined effects of pressure and temperature and radiation will not exceed this value. The stresses in the CCV closure bolts due to HAC internal pressure loading are evaluated in combination with the maximum bolt preload and HAC temperature loading using the methodology of NUREG/CR-6007. The average tensile stress in the CCV closure bolts due to HAC thermal load combination (L.C. H3) is 86.3 ksi. The allowable average stress for the bolt at 350°F is 87.7 ksi, with a corresponding maximum stress ratio in the CCV closure bolt of 0.99.

As a result, the staff finds that the package under HAC thermal loads continue to maintain containment without the loss of any radioactive material and concludes that the package satisfies the requirements of 10 CFR 71.73(c)(4).

2.1.12 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.

2.1.13 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 or an equivalent external pressure load of 36.4 psi. The package design is bounded by the 290 psig for an external pressure as required by 10 CFR 71.61, which exceeds the external pressure load of 36.4 psi. The staff reviewed the package for immersion and concluded that it satisfies the standards of 10 CFR 71.73(c)(6).

2.1.14 Air Transport Accident Conditions for Fissile Material

The requirements of 10 CFR 71.55(f) do not apply since air transport is not authorized.

2.1.15 Immersion - Special Requirement for Type B Packages with more than 10^5 A₂

The requirements of 10 CFR 71.61 apply. The applicant considered the deepwater 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 results from the 3-D FEM used to determine the stresses in the CCV components for deepwater immersion are summarized in SAR Table 2.7-11.

The minimum design margin in the CCV for the deepwater immersion test is +0.33 for membrane plus bending stress intensity at the center of the CCV bottom plate. Buckling evaluations of the CCV shell are performed for the deep immersion pressure test in accordance with the requirements of ASME Code Case N-284-1. HAC allowable buckling stresses shown in Table 2.1-7, which include a factor of safety of 1.34. The maximum interaction ratios do not exceed 1.0 as shown in SAR Table 2.7-12.

The results are acceptable to the staff for the CCV design to meet the special immersion requirements. The staff concluded that the package satisfies the immersion-special requirements of 10 CFR 71.61.

2.1.16 Air Transport of Plutonium

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

2.1.17 Summary of Findings

The staff finds that structural performance of the OPTIMUS®-H 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 tiedown 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 tiedown. The SAR described tiedown design, 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 290 psig 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.

2.1.18 Conclusion

The staff reviewed the information provided in the SAR and supporting calculations and based on the findings made because of the review, concluded that the applicant has demonstrated that the structural integrity of the OPTIMUS®-H spent fuel transportation package meets the regulations in Title 10 of the Code of Federal Regulations (10 CFR) Part 71, " Packaging and Transportation of Radioactive Material.

2.2 MATERIALS EVALUATION

General Considerations

The purpose of the staff's materials evaluation for the NAC International Type B(U)F-96 OPTIMUS®-H package is to determine whether the application adequately describes and evaluates the materials used in the package for ensuring that it meets the regulatory requirements of 10 CFR Part 71. The NRC staff performed its materials evaluation for the OPTIMUS®-H package by following the technical guidance in NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," August 2020 (ADAMS Accession No. ML20234A651).

The applicant provided information concerning the OPTIMUS®-H package materials in the NAC International OPTIMUS®-H Safety Analysis Report (SAR, hereafter referred to as the application). This information is included in Chapter 1, "General Information," Chapter 2, "Structural Evaluation," Chapter 3, "Thermal Evaluation," Chapter 4, "Containment," Chapter 5, "Shielding Evaluation," Chapter 6, "Criticality Evaluation," Chapter 7, "Package Operations," and Chapter 8, "Acceptance Tests and Maintenance Program" of the application. The staff identified that these chapters contain materials information and data that are used in the design and safety analyses of the package for demonstrating compliance with the applicable requirements of 10 CFR Part 71.

In letters dated May 5, 2023, December 13, 2023, May 2, 2024, the applicant provided responses to the staff's requests for additional information (RAIs) related to the OPTIMUS®-H packaging materials. As documented in the SER sections below, the RAI responses included updates to the application sections addressing material specifications and the evaluation of material properties and performance for the OPTIMUS®-H packaging components.

2.2.1 Drawings and Description of Packaging Components and Materials

Chapter 1 of the application provides the OPTIMUS®-H packaging drawings and a general description of the packaging components, materials, design functions, and package contents. The application states that the radioactive contents of the package include Type B quantities of normal form transuranic waste and low-enriched irradiated fuel waste. Chapter 2 of the application includes a description of the packaging component assemblies, subcomponents, individual parts, and their materials of construction. The application states that the four major packaging components include (1) the cask containment vessel (CCV) assembly, (2) the outer shield vessel (OSV) assembly, (3) the impact limiters, and (4) the shield insert assembly (SIA). The design functions and construction (specifically, materials and fabrication) of the four major packaging components, as described in the application, are summarized below.

- (1) The application states that the CCV assembly is an austenitic stainless steel cylindrical vessel that includes a cylindrical body weldment, a bolted closure lid, a bolted port cover, and O-ring seals. The CCV assembly is designed to be the pressure-retaining and leak-tight containment vessel for the radioactive contents. The stainless-steel cylindrical body weldment consists of a cylindrical shell, a bolting flange, and a bottom plate; all three items are joined by complete joint penetration welds. The bolting flange includes threaded holes for the CCV closure lid bolts. The application states that the stainless steel CCV closure lid is a circular plate with holes for the CCV closure lid bolts, leak test ports, a vent/fill port, and threaded, blind holes for securing lifting attachments. The vent/fill port, which is used for air evacuation and helium backfill, is closed and sealed by the CCV port cover. The CCV port cover is a stainless-steel circular plate with holes for the CCV port cover closure bolts and leak test ports. The application states that CCV containment closure devices include the bolted CCV closure lid and the bolted CCV port

cover. The CCV closure lid is secured to the CCV body by alloy steel CCV closure lid bolts that are intended for low-temperature surface. The CCV port cover is secured to the CCV closure lid by stainless steel port cover closure bolts. Elastomeric O-rings are used to establish the containment seals for the CCV closure lid and CCV port cover.

- (2) The application states that the OSV assembly is a thick-walled ductile cast iron (DCI) vessel with a bolted closure lid. The OSV assembly does not perform any containment function; specifically, it is not designed to retain internal pressure or to prevent or mitigate leakage of radionuclides to the outside environment. The OSV functions as a secondary enclosure for the CCV assembly to provide radiation shielding, structural support during lifting operations and for tie-down of the package, and to protect the CCV from the direct effects of NCT and HAC impact loading and HAC fire test conditions. The OSV assembly includes the cylindrical OSV body, OSV closure lid, and OSV closure lid bolts. The cylindrical OSV body is predominately a monolithic unit that is cast from ductile iron. The cylindrical OSV body casting includes an integral DCI bolting flange, integral DCI tie-down lugs, and integral DCI lifting trunnions. The OSV body's bolting flange includes alloy steel anchor bolts and sleeve nuts with threaded holes to accommodate the OSV closure lid bolts. The OSV body may also include either integral DCI impact limiter attachment lugs or non-integral stainless steel impact limiter attachment lugs that are bolted onto the OSV body. The DCI OSV closure lid is a circular plate with untapped holes for the OSV closure lid bolts and threaded holes for securing lifting attachments. The OSV closure lid is secured to the OSV body by alloy steel OSV closure lid bolts. The OSV closure lid does not include any ports or penetrations. No containment seal is used for the OSV closure lid. The OSV components do not have any weld joints.

The application states that the DCI OSV surfaces are covered with a specified type of high-temperature, radiation-resistant epoxy coating for corrosion protection. The application states that the epoxy coating compound is highly resistant to chemical reactions and has very good abrasion resistance.

- (3) The application states that two identical foam-filled impact limiters attach to and fit over the upper and lower ends of the OSV assembly. The impact limiters are designed to crush and absorb kinetic energy for the analyzed NCT and HAC impact events, including the NCT and HAC free drop tests and the HAC puncture drop test conditions, thereby limiting the impact loads transmitted to the OSV, CCV, and its contents. The impact limiters also function to thermally insulate the upper and lower ends of the OSV and CCV assemblies during the HAC fire test. The impact limiters consist of energy-absorbing closed-cell polyurethane foam cores that are sealed inside welded stainless-steel inner and outer shells. The cores of each impact limiter have two different densities of closed-cell polyurethane foam for optimal impact limiter performance during the NCT and HAC free drop tests. As shown in the application, the foam cores include corner/side region foam cores and end region foam cores. The welded stainless-steel inner and outer shells completely encase the energy-absorbing closed-cell polyurethane foam cores to protect the foam cores from the external environment.

- (4) The application states that the SIA is a coated carbon steel enclosure located inside the CCV that provides additional shielding for certain high activity radioactive contents, as specified in the application, to ensure compliance with package external radiation limits in 10 CFR Part 71. The SIA is placed inside the CCV and encloses the radioactive contents to provide additional gamma shielding. The application states that the SIA does not perform a containment or thermal function. The staff also noted that it does not perform any criticality safety function since the application states that the package does not rely on any internal support or positioning features to maintain an analyzed subcritical configuration of fissile contents. The application states that no credit is taken for the SIA in the structural evaluation of the other packaging components or in the package criticality analyses; however, the SIA is designed to withstand the HAC drop tests without structural failure such that its shielding integrity is maintained for conditions where the SIA is credited in the shielding evaluation.

The application states that the carbon steel SIA surfaces are covered with a specified type of high-temperature, radiation-resistant epoxy coating for corrosion protection. The application states that the epoxy coating compound is highly resistant to chemical reactions and has very good abrasion resistance.

The OPTIMUS®-H package application includes drawings showing the design and construction of the OPTIMUS®-H packaging components. The drawings include lists of parts and their material specifications for each of the four major packaging components. The drawings also include welding requirements, nondestructive examination (NDE) requirements, material testing requirements, and component dimensions.

The NRC staff reviewed the packaging drawings and the description of the packaging components. The staff verified that the application, as revised in accordance with the applicant's May 5, 2023, RAI response, includes drawings that adequately convey the geometry and dimensions of the packaging components, identification of subcomponents and parts, materials of construction, fabrication methods, welding qualification requirements, NDE requirements, and material testing requirements. Therefore, the staff finds that the packaging drawings and the description of the packaging components in the application are acceptable.

Based on the foregoing evaluation, the staff finds that the OPTIMUS®-H packaging drawings and the description of the packaging components are acceptable since they describe component assemblies, design functions, materials of construction, and fabrication methods in sufficient detail to provide an adequate basis for the detailed materials evaluation of the package. Accordingly, the staff finds that the packaging drawings and the description of packaging components and materials meets the requirements in 10 CFR 71.33(a).

2.2.2 Codes and Standards for Materials, Fabrication, and Nondestructive Examination

Chapters 1, 2, and 8 of the application provide information on the codes and standards that are used for materials, fabrication, and nondestructive examination (NDE) of the ITS packaging components. These chapters include information on standard material specifications for ITS packaging base metal components; standards for material selection and material qualification testing such as destructive tests on representative samples; standards for component fabrication including standard requirements for producing weld joints and qualification of welding processes and personnel; and NDE performance standards and NDE acceptance criteria for acceptance of production components and associated welds that are to be placed into service. In general, materials for ITS packaging components include both metallic and nonmetallic materials. Metallic materials are used for the structural components of the CCV, OSV, SIAs, and the deformable impact limiter shells. Non-metallic organic materials are used for several non-structural components including CCV O-ring seals, the deformable energy-absorbing impact limiter foam, and the protective coating used on the ductile cast iron (DCI) OSV and the carbon steel SIAs.

With certain exceptions that are described in the application and evaluated in subsequent sections of this SER, the application documents that metallic structural components of the packaging (CCV, OSV, and SIAs) are designed and constructed in accordance with the applicable requirements of the 2010 Edition with the 2011 Addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1 (hereafter ASME BPVC Section III). Specific exceptions to the use of the ASME BPVC Section III are discussed in the application, as revised in the applicant's May 5, 2023, December 13, 2023, May 2, 2024, RAI responses, and are applicable to the materials and fabrication of non-containment structural components, including the OSV and the SIA. The application requires that the metallic structural containment boundary (i.e., CCV) materials must comply with the applicable standard material specifications in the ASME BPVC Section II (2010 Edition with 2011 Addenda), whereas the application requires that the non-containment boundary structural materials must comply with the applicable standard material specifications. The application identifies the ASME BPVC Section II and standard material specifications that are used for the metallic structural components.

The staff's general evaluation of the applicant's use of codes and standards for selection, qualification testing, and NDE of metallic base materials used for fabrication of the CCV, OSV, and SIA structural components and the deformable impact limiter shells, including use of standard material specifications and exceptions to codes and standards for these materials, is documented in the SER subsections below. The staff's more specific evaluation of the applicant's use of codes and standards for the design, fabrication, and NDE of weld joints in the metallic CCV, SIA, and impact limiter shell components is documented in SER Section 7.3. The staff's specific evaluation of the applicant's use of codes and standards for determining the mechanical properties of the CCV, OSV, SIA, and impact limiter materials and for fracture toughness-related testing and evaluation of ferritic steel and DCI structural materials is documented in SER Section 7.4.

Cask Containment Vessel (CCV)

The application states that the CCV assembly is designed and fabricated in accordance with the applicable requirements of the 2010 Edition with 2011 Addenda of the ASME BPVC, Section III, Subsection NB. The staff confirmed that the application does not include any exceptions to the use of Subsection NB for materials, fabrication, and examination of the metallic CCV components. With the exception of the more rigorous fracture toughness criteria for ferritic steel

base material for transportation package containment vessels (addressed below in Section 7.4 of this SER), NUREG/CR-3854, "Fabrication Criteria for Shipping Containers," March 1985, and NUREG/CR-3019, "Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials," March 1984 state that Subsection NB requirements for materials, fabrication, and examination are acceptable for the construction of containment vessels for all categories of radioactive material contents, including spent nuclear fuel. NUREG-2216 identifies that NUREG/CR-3854 and NUREG/CR-3019 are acceptable for specifying the ASME BPVC, Section III requirements for the control of materials, fabrication processes, and examinations of containment vessels. Therefore, the staff determined that the applicant's use of Subsection NB for the materials, fabrication, and examination of the CCV is acceptable.

The staff reviewed the ASME BPVC Section II standard material specifications included in the application for the metallic CCV components, including the stainless steel CCV body, CCV closure lid, CCV port cover, alloy steel CCV closure lid bolts, and stainless steel CCV port cover closure bolts. The staff confirmed that the application, as revised in the May 5, 2023, RAI response, includes drawings that appropriately identify the standard material specifications for these components consistent with the description of the CCV materials in Section 2.2 of the application. The staff verified that the ASME BPVC Section II standard material specifications for the metallic CCV components, as specified in the application, are listed in the applicable material property tables of the 2010 Edition with 2011 Addenda of the ASME BPVC Section II, Part D, as required by Subsection NB, Paragraph NB-2121 for forging and plate materials and Paragraph NB-2128 for bolt materials. Therefore, the staff determined that the applicant's use of codes and standards for the materials, fabrication, and examination of the metallic CCV components is acceptable.

Non-Containment Structural Components

The non-containment structural components of the OPTIMUS-H packaging include the OSV and SIAs. The staff's evaluation of the applicant's use of codes and standards for materials, fabrication, and examination OSV and SIAs follows below.

Outer Shield Vessel (OSV)

As addressed above in this SER, the OSV is a secondary non-containment enclosure for the CCV assembly with safety functions that include radiation shielding, structural support during lifting operations and for tie-down of the package, and protection of the CCV from the direct effects of NCT and HAC. The OSV components include the DCI OSV body and closure lid, alloy steel OSV closure lid bolts, alloy steel OSV anchor bolts and sleeve nuts, and optional bolted impact limiter attachment hardware. The application states that these items are designed in accordance with the applicable requirements of the 2010 Edition with 2011 Addenda of the ASME BPVC, Section III, Subsection NF; however, the application indicates that the use of the identified standard material specification for the DCI OSV body and closure lid constitutes an exception to the Subsection NF requirements. NUREG/CR-3854 states that either the ASME BPVC, Section III, Subsection NF or the ASME BPVC, Section VIII, Division 1 requirements for materials, fabrication, and examination are acceptable for the construction of secondary non-containment enclosures that are used for radiation shielding and structural support for all categories of radioactive material contents. The staff noted that NUREG/CR-3019 is not applicable to the fabrication of the OSV components since there are no weld joints in any of the OSV components.

The staff reviewed the standard material specifications included in the application for the OSV components. The staff confirmed that the application, as revised in the May 5, 2023, RAI

response, includes drawings that appropriately identify the standard material specifications for these components consistent with the description of the OSV materials in Section 2.2 of the application. The staff noted that the standard material specifications for the alloy steel OSV closure lid bolts, alloy steel OSV anchor bolts and sleeve nuts, and optional bolted impact limiter attachment hardware have equivalent ASME BPVC Section II material specifications in the 2010 Edition with 2011 Addenda of the ASME BPVC; the staff verified that the equivalent ASME BPVC Section II material specifications are listed in the applicable material property tables of the 2010 Edition with 2011 Addenda of the ASME BPVC, Section II, Part D, as required by Subsection NF, Paragraph NF-2121 for permissible material specifications. Therefore, the staff determined that the standard material specifications for the metallic OSV bolting components and optional bolted impact limiter attachment components are acceptable.

Based on its review of the applicant's May 5, 2023, response to an RAI on impact test criteria for ensuring adequate fracture toughness of the alloy steel OSV closure lid bolts, anchor bolts, and sleeve nuts, the staff identified an additional exception to the Subsection NF requirements for these items. Specifically, the application, as revised per the RAI response, clarified that there is no requirement for any impact testing of the alloy steel OSV bolting materials. The staff determined that this constitutes an exception to the Subsection NF requirements since Subarticle NF-2300 of Subsection NF requires that, for bolt sizes greater than one inch, the bolt material must be Charpy impact tested, and the impact test results must meet the applicable acceptance criteria specified in this subarticle for ensuring adequate fracture toughness. As addressed in the staff's detailed evaluation of this issue, documented in this SER, the application was revised per the RAI response to include an acceptable demonstration that the closure function of the OSV lid will be maintained even if a majority of the alloy steel OSV closure lid bolts and associated anchor bolts and sleeve nuts were to fail by brittle fracture, thereby supporting the applicant's conclusion that no impact testing is needed to ensure adequate fracture toughness of these bolting materials. Based on the staff's evaluation and acceptance of this exception to the Subsection NF fracture toughness criteria for the alloy steel OSV bolting materials, as documented in this SER, the staff determined that the applicant's implementation of all other requirements of the ASME BPVC Section III Subsection NF requirements for the alloy steel OSV bolting components is acceptable.

Ductile Cast Iron

Importantly, the staff noted that the use of the subject DCI material is not endorsed by the NRC staff as a generic basis for using the subject DCI material in the construction of transportation package containment vessels (NRC Regulatory Guide 1.193, Revision 7, "ASME Code Cases Not Approved for Use," December 2021, ADAMS Accession No. ML21181A224).

The staff noted that the application does not propose to use this DCI material for any containment function. As addressed in this SER, the DCI OSV components are not designed to retain internal pressure or to prevent or mitigate leakage of radionuclides to the outside environment. The OSV functions as a secondary enclosure for the CCV assembly to provide radiation shielding, structural support during lifting operations and for tie-down of the package, and to protect the CCV from the direct effects of NCT and HAC. Based on the consideration of these safety functions, the staff identified that the most severe loads that could potentially result in a large through-wall fracture of the OSV that adversely affects the performance of OSV safety functions are the impact loads caused by the NCT and HAC regulatory free drop and puncture drop tests, in particular the HAC impact loads caused by the 30-ft free drop and puncture drop test conditions specified in 10 CFR 71.73.

Therefore, for the NCT and HAC regulatory drop tests, the staff determined that the DCI material must have sufficient fracture toughness at the lowest service temperature (LST) to ensure adequate performance of radiation shielding and CCV structural protection safety functions. Specifically, the staff determined that adequate DCI fracture toughness should be ensured through demonstration of adequate protection against OSV fracture that could result in any of the following events:

- (1) OSV fracture that results in any of the CCV components exceeding their allowable stress limits in Table 2.1-3 of the application for NCT and HAC free drop and puncture drop tests;
- (2) For the NCT 2-ft free drop test, OSV fracture that results in a significant increase in external surface radiation levels, such that the package is not in compliance with 10 CFR Sections 71.43(f) and 71.51(a)(1);
- (3) For the HAC 30-ft free drop test and puncture drop test, OSV fracture that results in an external radiation dose rate exceeding 10 mSv/hour (1 rem/hour) at 1 m (40 in) from the external surface of the package, such that the package is not in compliance with 10 CFR 71.51(a)(2).

In its May 2, 2024, response to an RAI addressing the need for DCI fracture toughness information to demonstrate adequate protection against OSV fracture that could result in any of the above three events, the applicant provided updates to the application sections addressing OSV DCI fracture resistance that include the following provisions:

- Section 8.1.5.2 of the application was updated to include new DCI fracture toughness qualification test requirements specifying that DCI material shall be subjected to fracture toughness testing at the LST, according to the specified test method, and the test data shall be evaluated against the fracture toughness acceptance criterion specified therein. These new DCI fracture toughness qualification test requirements replaced the prior requirements for qualification of DCI fracture toughness.
- Section 2.1.2.5.2 of the application was updated to include the results of a fracture mechanics calculation.
- Sections 2.1.2.5.2 and 8.1.5.2 of the updated application specify that the value for the fracture toughness, as determined by the test measurements and test data evaluation methods described in Section 8.1.5.2, shall be greater than or equal to the acceptance criterion specified therein. The application notes that the acceptance criterion is consistent with the acceptance criterion for load conditions.

The staff reviewed the updated information in the application regarding the methods and requirements for qualification of OSV DCI fracture toughness and found them to be acceptable. The staff's detailed evaluation of the updated application information provided in the RAI response is documented below in this SER. As explained in further detail in this SER, the staff's review of the applicant's May 2, 2024, RAI response determined that the fracture toughness test requirements, test data evaluation methods, and acceptance criterion ensures that the DCI used to construct the OSV components is adequately protected at the LST against fracture that could

result in any of the above three events. Therefore, staff determined that the applicant's use of specific code requirements for ensuring adequate LST fracture toughness for the OSV DCI material is acceptable.

Based on the foregoing review of the standard material specifications identified in the application and evaluation of their use considering the applicable provisions of the Code, with exceptions to Code requirements addressed and evaluated above, the staff determined that the applicant's use of codes and standards for the materials, fabrication, and examination of the OSV components is acceptable. Since the above noted exceptions to the requirements for the OSV materials are directly related to the brittle fracture resistance of the OSV components under the analyzed NCT and HAC impact loads, the staff's detailed review of these exceptions is covered in this SER.

Shield Insert Assembly (SIA)

As addressed above in this SER, the SIA is located inside the CCV and encloses the radioactive contents to provide additional gamma shielding. The application states that the SIA does not perform a containment, thermal, or criticality safety function. The application states that no credit is taken for the SIA in the structural evaluation of the other packaging components; however, the SIA is designed to withstand the NCT and HAC drop tests without structural failure such that its shielding integrity is maintained. The SIA is a welded carbon steel enclosure fabricated from carbon steel plate material conforming to the ASTM standard material specification identified in the application. The application states that the SIA is designed in accordance with the applicable requirements of the 2010 Edition with 2011 Addenda of the ASME BPVC, Section III, Subsection NF. The application did not identify any exceptions to the use of Subsection NF for the design and construction of the SIA. NUREG/CR-3854 and NUREG/CR-3019 state that either the ASME BPVC, Section III, Subsection NF or the ASME BPVC, Section VIII, Division 1 requirements for materials, fabrication, and examination are acceptable for the construction of secondary non-containment enclosures that are used for radiation shielding.

The staff reviewed the ASTM standard material specification included in the application for the carbon steel plate material used to fabricate SIA components. The staff confirmed that the application, as revised in the May 5, 2023, RAI response, includes drawings that appropriately identify the ASTM standard material specification for the SIA carbon steel consistent with the description of the SIA material in Section 2.2 of the application. The staff noted that the ASTM standard material specification for the carbon steel SIA components has an equivalent ASME BPVC Section II material specification in the 2010 Edition with 2011 Addenda of the ASME BPVC; the staff verified that the equivalent ASME BPVC Section II material specification is included in the applicable material property tables of the 2010 Edition with 2011 Addenda of the ASME BPVC, Section II, Part D. Therefore, the staff determined that the standard material specification for the carbon steel SIA components is acceptable.

The staff identified one exception to the Subsection NF requirements since the application does not specify any impact tests to qualify the fracture toughness for the welded carbon steel plate material used for the SIA. The staff determined that the lack of impact test requirements for the carbon steel plate material and associated welds constitutes an exception to the Subsection NF requirements since Subarticle NF-2300 of Subsection NF requires Charpy notched bar impact testing of these materials. Specifically, for welded carbon steel plate material with a nominal section thickness greater than five-eighths inch, the base material, weld procedure qualification test specimens, and weld metal must be Charpy impact tested, as specified in Paragraph NF-2331, and the impact test results must meet the applicable Charpy impact test acceptance

standards of NF-2331 for ensuring adequate fracture toughness. As an alternative to the Charpy impact test requirements and acceptance criteria of Subarticle NF-2300, the application specifies that, considering the SIA safety function, the SIA is designed in accordance with the Category III fracture toughness requirements of NUREG/CR-1815, which provide sufficient fracture toughness to prevent fracture initiation at minor defects typical of good fabrication practices.

The staff identified that Regulatory Guide (RG) 7.11 does not endorse the use of NUREG/CR-1815 Category III fracture toughness criteria for radiation shielding components. Rather, the ASME BPVC Section III, Subarticle NF-2300 fracture toughness requirements for ferritic steel materials are endorsed in NUREG/CR-3854. Therefore, the staff could not determine whether the subject SIA material would have sufficient fracture toughness at the LST to ensure adequate protection against SIA fracture that results in an unacceptable decrease in shielding performance.

Therefore, the staff issued an RAI requesting the applicant to state whether the welded carbon steel SIA material is required to be Charpy impact tested in accordance with the requirements of Subarticle NF-2300 of the ASME BPVC, Section III, Subsection NF. If not, the staff requested the applicant to provide information that demonstrates, for the NCT and HAC regulatory drop tests, adequate protection at the LST against SIA fracture that results in an unacceptable decrease in shielding performance, such that radiation levels exceed the regulatory limits of 10 CFR Sections 71.43(f), 71.51(a)(1), and 71.51(a)(2).

In its December 13, 2023, RAI response, the applicant stated that the SIA materials do not require Charpy impact testing in accordance with ASME BPVC Section III Subsection NF requirements since the SIA meets the Category III fracture toughness criteria of Table 6 of NUREG/CR-1815. The applicant's RAI response included information to justify the application of the NUREG/CR-1815, Category III fracture toughness criteria to ensure that the SIA has adequate protection at the LST against SIA fracture that results in an unacceptable decrease in shielding performance for the NCT and HAC drop tests. The applicant revised the text in the application to clarify that the use of the NUREG/CR-1815 Category III fracture toughness criteria for the SIA constitutes an exception to the requirements of ASME BPVC Section III Subsection NF.

The staff's detailed review of the RAI response information, including review of specific SIA material requirements, design configurations, and SIA stresses, is documented in this SER. Based on its review of case-specific SIA material and design requirements, the staff determined that the applicant's proposal to use the fracture toughness criteria for Category III containments from NUREG/CR-1815 and RG 7.11 for the SIA carbon steel ensures that the carbon steel is adequately protected at the LST against fracture that could result in an unacceptable decrease in shielding performance, where radiation levels exceed the applicable regulatory limits. Therefore, staff determined that the applicant's proposed alternative to use the NUREG/CR-1815 Category III fracture toughness criteria for the SIA in lieu of the fracture toughness requirements of the ASME BPVC Section III Subsection NF is acceptable.

Based on the foregoing review of the ASTM standard specification for the carbon steel SIA plate material and evaluation of its use considering the applicable provisions of the ASME BPVC, Section III, Subsection NF, with the exception and alternative to these code requirements addressed above, the staff determined that the applicant's use of codes and standards for the SIA components is acceptable. Since the above noted exception to the Subsection NF requirements for the carbon steel SIA material is directly related to the brittle fracture resistance

of the SIA components under the analyzed NCT and HAC impact loads, the staff's detailed review of this exception is covered in this SER.

Stainless Steel Impact Limiter Shells and Nonmetallic Items

The application states that the outer thin sections of the welded stainless steel impact limiter shells are designed to deform plastically and absorb kinetic energy along with the polyurethane foam cores when subjected to NCT free drop, HAC free drop, and HAC puncture drop load conditions to protect the OSV, CCV, SIAs, and contents during these impact events. Therefore, the application specifies the use of strain-based design criteria for the impact limiters. Accordingly, the staff identified that the ASME BPVC Section III structural design criteria (e.g., Article NB-3000 or Article NF-3000) are not applicable to the outer sections of the stainless-steel impact limiter shells or the foam cores since they are not a structural containment boundary, and they do not maintain structural rigidity during the analyzed NCT and HAC impact events. The welded stainless steel impact limiter shells also function to protect the polyurethane foam cores of the impact limiters from moisture-induced degradation during routine operations. Therefore, the impact limiter shells must be adequately designed and fabricated to prevent the intrusion of water and moisture into the foam cores during routine operation.

The staff reviewed ASTM standard material specification identified in the application for the stainless steel plate and sheet metal used to fabricate the impact limiter shells and determined that it is acceptable since this same stainless steel material specification is included in the ASME BPVC Section II material specifications and would be acceptable for use in the fabrication of containment components in accordance with the ASME BPVC Section III, Subsection NB, subject to additional requirements for containment components specified therein. The staff noted that the dimensional specifications for the outer thin sections of the stainless-steel shells is suitable for ensuring the needed deformation characteristics, as addressed in the application's structural analyses of impact limiter performance for the analyzed NCT and HAC impact events. With respect to its properties as a water and moisture barrier for the protection of the polyurethane foam cores from moisture-induced degradation, the staff identified that properly welded and properly maintained stainless steel shells are not prone to general corrosion or water penetration, although over extended periods, localized corrosion effects (e.g., pitting or crevice corrosion) that are known to occur in outdoor environments could lead to penetration of the outer thin sections. The details of the staff's review of the stainless-steel shell weld joint design, fabrication, and weld NDE for ensuring weld joint integrity to provide adequate protection of the polyurethane foam cores is addressed in Section 7.3 of this SER. The staff's review of the corrosion performance and associated maintenance inspections of the stainless-steel impact limiter shells is addressed in Section 7.7 of this SER. Considering these review findings, the staff determined that the applicant's use of codes and standards for the materials, fabrication, and examination of the welded stainless steel impact limiter shells is acceptable.

The nonmetallic components of the packaging are fabricated or synthesized from organic compounds and include the elastomeric O-rings for the CCV containment seals, the energy-absorbing polyurethane foam cores for the impact limiters, and the epoxy coating used to coat the untapped surfaces of the DCI OSV and carbon steel SIAs. There are no consensus codes or standards specified for these component materials and their associated production, fabrication, and examination methods since their functionality is ensured through package design and procurement controls in accordance with the NRC-approved QA program. Specific controls described in the application include the selection of suitable supplier materials, evaluation of material properties and specifications to ensure capability to perform intended design functions, qualification testing to ensure measured material properties meet their

functional design requirements for applicable service conditions, and nondestructive examinations to ensure production components are suitable for placement into service.

Summary of Review Findings Regarding Codes and Standards

Based on the foregoing evaluation, the staff finds that the codes and standards specified in the application for materials, fabrication, and examination of metallic packaging components are acceptable since the applicant's use of the applicable sections of the ASME BPVC and applicable ASTM standards for material selection and qualification ensures these components are adequately designed and constructed to meet their functional performance requirements. Further, the staff finds that the applicant has adequately described and justified exceptions to the use of specific provisions of the ASME BPVC that are applicable to specific OSV and SIA components. Accordingly, the staff finds that the applicant's use of codes and standards for materials, fabrication, and examination of metallic packaging components meets the requirements in 10 CFR 71.31(c).

2.2.3 Weld Design, Fabrication, and Examination

The application documents the use of weld joints in the design and fabrication of the CCV body, SIAs, and the stainless-steel impact limiter shells. The application states that the OSV does not have any weld joints. Section 2.3 of the application states that the CCV body weldment is fabricated in accordance with the applicable requirements of the ASME BPVC Section III, Subsection NB (2010 Edition with 2011 Addenda), and the welded SIA components are fabricated in accordance with the applicable requirements of the ASME BPVC Section III, Subsection NF (2010 Edition with 2011 Addenda). The staff confirmed that Subsection NB and Subsection NF include the needed controls on welding and related fabrication processes for ensuring the structural integrity of the weld joints in the CCV body and SIA, consistent with the recommendations in NUREG/CR-3854 and NUREG/CR-3019 considering the intended design functions of these components. The staff noted that the fabrication requirements of Articles NB-4000 (Subsection NB) and Article NF-4000 (Subsection NF) include requirements for the use of the ASME BPVC Section IX, "Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators," for qualification of welding procedures and personnel to ensure that welding processes produce weld joints that can sustain the required loads, in accordance with the applicable design criteria. The staff confirmed that the application, as revised in the May 5, 2023, RAI response, includes a requirement that all weld joints in the impact limiter shells are to be fabricated using welding procedures and personnel that are qualified in accordance with the ASME BPVC Section IX to ensure that the weld joints adequately function as a moisture barrier during normal package operation. The staff verified that the application, as revised in the May 5, 2023, RAI response, includes CCV, SIA, and impact limiter drawings that appropriately specify that all welding procedures and qualifications are to be in accordance with the ASME BPVC Section IX, consistent with the description of the welding procedure and personnel qualifications in Section 2.3 of the application. Therefore, the staff determined that the information in the application concerning weld fabrication methods and associated requirements for qualification of welding processes for the CCV body, SIA, and the stainless-steel impact limiter shells is acceptable.

With respect to weld design, the staff reviewed the CCV body, SIA, and impact limiter shell drawings provided in the application and confirmed that the design of the welds is adequately depicted in the drawings using weld symbols that are consistent with the standard nomenclature in American Welding Society (AWS) standard, AWS A2.4:2020, "Standard Symbols for Welding, Brazing, and Nondestructive Examination" January 2020. The staff verified that the design of the CCV body welds and SIA welds meets the applicable requirements of Article NB-3000 and

Article NF-3000, respectively, for permissible types of welded joints. Generally, the structural integrity of the weld joint design for the CCV body and SIAs for NCT and HAC loadings is based on acceptable implementation of the structural design requirements of Article NB-3000 for the CCV body and Article NF-3000 for the SIA, which is demonstrated by detailed structural analyses provided in Sections 2.6 and 2.7 of the application for NCT and HAC loadings, respectively. The staff's detailed review of the applicant's structural evaluation of the CCV and SIA is documented in this SER. Since the impact limiters are designed to undergo deformation and absorb kinetic energy for the NCT and HAC impact loadings, the staff noted that the allowable stress design criteria of Article NF-3000 are not applicable to the design of the impact limiter shell welds; however, the staff verified that the stainless steel impact limiter shell weld joints are adequately designed to protect the polyurethane foam cores against the intrusion of water and moisture during normal package operation. Therefore, the staff determined that the information in the application concerning the design of the weld joints for the CCV body, SIA, and the stainless-steel impact limiter shells is acceptable.

With respect to weld nondestructive examination (NDE), the staff reviewed the CCV body weldment, SIA, and impact limiter shell drawings provided in the application, as revised in the May 5, 2023, RAI response, and confirmed that the weld NDE methods are adequately depicted in the drawings using NDE symbols that are consistent with the standard nomenclature in AWS A2.4:2020. The staff confirmed that the drawings, as revised in the May 5, 2023, RAI response, specify weld NDE methods, including methods for visual examinations, surface examinations, and volumetric examinations, that meet the applicable requirements of the ASME BPVC, Section III, Subsection NB for CCV body welds and Subsection NF for SIA welds and impact limiter shell welds for detection of fabrication defects in the weld joints. The staff verified that the drawings, as revised in the May 5, 2023, RAI response, specify that NDE of weld joints in the CCV body, SIA, and impact limiter shells is to be performed in accordance with the requirements of the ASME BPVC Section V that are applicable to the specific NDE methods. The staff also verified that the drawings, as revised in the May 5, 2023, RAI response, specify that the acceptance standards of Subsection NB, Subarticle NB-5300 for CCV body welds and Subsection NF, Subarticle NF-5300 for SIA welds and impact limiter shell welds are to be used for evaluating relevant indications to determine whether such indications are unacceptable fabrication defects. Therefore, based on review of the information in the application concerning weld NDE criteria and review of associated ASME BPVC Section V and Section III requirements for performance of weld NDE and NDE acceptance standards, the staff determined that the CCV body, SIA, and impact limiter shell weld examinations are acceptable.

Based on the foregoing evaluation, the staff finds that the codes and standards specified in the application for the design, fabrication, and NDE of weld joints in the CCV body, SIAs, and impact limiter shells are acceptable since the use of these code requirements ensures that weld joints in these components are adequately designed and fabricated to perform applicable containment, shielding, structural support, and water barrier safety functions. Accordingly, the staff finds that the applicant's use of codes and standards for design, fabrication, and NDE of weld joints for the package meets the requirements in 10 CFR 71.31(c).

2.2.4 Mechanical Properties of Materials

Sections 2.1, 2.2, and 8.1 of the application address the mechanical properties of the ITS packaging materials that are relied on to ensure adequate structural performance of the package for NCT and HAC. Mechanical properties of metallic materials for structural components include tensile and elastic properties to protect against ductile failure and buckling; fatigue performance properties to protect against failure due to formation and growth of fatigue cracks during normal service due to cyclic loading; and fracture toughness properties to protect

against brittle fracture of the components at the lowest service temperature (LST). Mechanical properties for the energy-absorbing impact limiter polyurethane foam cores include dynamic stress strain curves that are used for the structural analysis of impact limiter performance for NCT and HAC impact events, which include the NCT and HAC free drop tests and HAC puncture drop test conditions.

Tensile and Elastic Properties

Sections 2.1.2.2, 2.1.2.3 of the application describe the structural design criteria and associated stress limits for evaluation of packaging structural components to ensure adequate protection against ductile failure and buckling. Section 2.2 of the application provides the tensile and elastic properties of metallic materials for the structural evaluation of packaging components, including the stainless steel CCV body, CCV closure lid, and CCV port cover; ductile cast iron (DCI) OSV body and closure lid; carbon steel SIA components; alloy steel CCV closure lid bolts; alloy steel OSV closure lid bolts; and alloy steel OSV anchor bolts and sleeve nuts. Section 2.2 also includes the tensile and elastic properties for the stainless-steel impact limiter shells and associated components for use in the structural evaluation of impact limiter performance for the analyzed NCT and HAC impact events.

For the evaluation of the CCV structural components to demonstrate adequate protection against ductile failure, the applicant used the tensile and elastic properties to demonstrate that stress conditions in the components are lower than the allowable stress limits for NCT and HAC, in accordance with the applicable requirements of the ASME BPVC Section III, Subsection NB and NUREG/CR-6007 for closure bolts. For the non-containment structural components, including the OSV and SIA, the applicant used the tensile and elastic properties to demonstrate that stress conditions in these components are lower than the allowable stress limits for NCT and HAC, in accordance with the applicable requirements of the ASME BPVC Section III, Subsection NF. For the buckling evaluation, the applicant used the tensile and elastic properties, along with the component geometric parameters, to demonstrate that stress conditions in the CCV shell and SIAs are lower than the allowable buckling stresses, in accordance with the applicable requirements of the ASME Code Case N-284, (which is evaluated by the staff in Table 4 in NRC Regulatory Guide (RG) 1.193, Revision 7), for the CCV shell and Subsection NF for the SIAs. The staff's evaluation of the applicant's detailed structural analyses for demonstrating that these code requirements are satisfied is covered in Section 2.1 of this SER.

Tables 2.2-2 thru 2.2-10 of the application, as revised in the May 5, 2023, RAI response, list the tensile and elastic properties of the metallic materials used in the structural evaluation. The staff confirmed that these tables include the properties that are needed to demonstrate that the stress conditions in the metallic structural components satisfy the allowable stress limits for protection against ductile failure and buckling, as per the applicable ASME BPVC Section III, NUREG/CR-6007, and ASME Code Case N-284 (which is evaluated by the staff in Table 4 in RG 1.193, Revision 7) requirements. The tensile and elastic properties used for these aspects of the structural evaluation include design stress intensity, yield strength, tensile strength, modulus of elasticity, mean coefficient of thermal expansion, Poisson's ratio, and material density.

The staff reviewed the tensile and elastic properties of the structural component materials, as reported in Tables 2.2-2 thru 2.2-10 of the May 5, 2023, revised application. The staff confirmed that all tensile and elastic properties for the stainless steel, alloy steel, and carbon steel components are consistent with those specified in the ASME BPVC, Section II, Part D for the applicable material specifications, as required by Subsections NB and NF of the ASME BPVC

Section III. The staff also confirmed that the application lists temperature-dependent properties, including design stress intensity, yield strength, tensile strength, modulus of elasticity, and mean coefficient of thermal expansion, for a range of temperatures that include the highest and lowest analyzed component temperatures that are applicable to the structural evaluations for the NCT and HAC tests. Therefore, the staff determined that the tensile and elastic properties for the stainless steel, alloy steel, and carbon steel components are acceptable for use in the structural evaluation to demonstrate adequate protection against ductile failure and buckling, in accordance with the applicable requirements of Subsections NB and NF of the ASME BPVC Section III, NUREG/CR-6007, and ASME Code Case N-284 (which is evaluated by the staff in Table 4 in RG 1.193, Revision 7).

Ductile Cast Iron

For the OSV DCI conforming to the applicable material specification, the staff confirmed that the application reports temperature-dependent tensile and elastic properties, including the design stress intensity, yield strength, tensile strength, elastic modulus and mean coefficient of thermal expansion. The staff verified that the highest values of the ASME BPVC-specified yield strength and tensile strength for the material at the lowest temperature range specified in the code are slightly lower than the corresponding strength requirements of the DCI material specification, based on tension testing at room temperature. Therefore, the staff determined that the DCI tensile and elastic properties are sufficiently conservative and acceptable for use in the structural evaluation to demonstrate adequate protection against ductile failure.

For the OSV DCI conforming to the applicable material specification, the staff confirmed that the application reports temperature-dependent tensile and elastic properties.

As addressed in this SER, there is no generic basis for using the subject DCI material specification for transportation package containment vessels. Importantly, the staff noted that, for the OPTIMUS®-H package, the DCI material specification is used for OSV outer enclosure components that are designed to provide radiation shielding and structural protection of the CCV containment boundary; the OSV does not itself perform any containment function. Therefore, based on the non-containment OSV design functions and the conservatism of the above tensile properties, the staff determined that the applicant's use of these tensile properties for the structural evaluation of the DCI OSV components is acceptable for the OPTIMUS®-H package. However, it should be noted that the staff's acceptance of the applicant's use of the DCI tensile properties in a specific Code for the OSV does not constitute an endorsement of that Code or the ASME BPVC Section III, Division 3 as a basis for using the subject DCI material specification for transportation package containment vessels.

The staff further noted that adequate specification of tensile properties for the DCI material to protect against ductile failure does not ensure adequate protection against brittle fracture at the LST. The staff's evaluation of the applicant's qualification of DCI material fracture toughness is documented below in this SER.

Fatigue Resistance Properties

Section 2.1.2.4 of the application discusses the fatigue evaluation of the packaging components including the stainless steel CCV body and lid, alloy steel CCV closure lid bolts, and alloy steel OSV closure lid bolts. The staff confirmed that application adequately describes the properties of these component materials that are used in the fatigue analyses. These properties include the design stress intensity, elastic modulus, mean coefficient of thermal expansion, and the design fatigue curves showing allowable amplitude of the alternating stress intensity as a

function of allowable stress cycles for the applicable material types (stainless steel base material and alloy steel bolts). The staff confirmed that the application cites design fatigue curves from Appendix I of the ASME BPVC Section III that are valid for the stainless-steel materials used for the CCV body and lid and the alloy steel materials used for the CCV and OSV closure bolts. Therefore, the staff determined that the material properties used for the fatigue evaluation of these CCV and OSV components are acceptable. The staff's evaluation of the applicant's detailed fatigue analysis for meeting the applicable ASME BPVC Section III design requirements for ensuring adequate protection against fatigue failure is covered in Section 2.1 of this SER.

Fracture Toughness

Section 2.1.2.5 of the application includes a fracture toughness evaluation to address the potential susceptibility of the package structural materials to brittle fracture at the LST. Fracture toughness evaluation for package structural components is performed to ensure that applicable ferritic steel materials, which may exhibit a ductile-to-brittle transition at low temperature, have adequate toughness to resist brittle fracture at the LST. For transportation package applications, fracture toughness evaluation for ferritic steel materials that have standard material specifications included in the ASME BPVC Section II, as authorized in the applicable subsections of the ASME BPVC Section III, is generally based on standard requirements and acceptance criteria for standard notched bar impact tests. Generic criteria for implementation of notched bar impact tests according to these standard test specifications are included in applicable subsections of the ASME BPVC Section III and, for ferritic steel base material used for transportation package containment vessels, NRC Regulatory Guides (RGs) 7.11 and 7.12.

The NRC staff's guidance for ensuring adequate fracture toughness of ferritic steel base materials for transportation package containment vessels is included in RGs 7.11 and 7.12. RG 7.11 applies to ferritic steels for containment vessels with a maximum wall thickness of four inches, and RG 7.12 applies to ferritic steels for containment vessels with a wall thickness greater than four inches but not exceeding 12 inches. Fracture toughness criteria for ferritic steel containment boundary welds should follow the applicable requirements of ASME BPVC Section III, Subsection NB. Fracture toughness criteria for ferritic steel containment closure bolting should follow the applicable requirements of Subsection NB and NUREG/CR-6007.

RGs 7.11 and 7.12 do not include guidance for ensuring adequate fracture toughness for non-containment boundary structural components and materials. However, NUREG/CR-3854 refers to the applicable subsections of the ASME BPVC, Section III that include material fracture toughness criteria for non-containment structural components. NUREG/CR-3854 refers to the ASME BPVC Section III, Subsection NG, Article NG-2000 requirements for materials (including fracture toughness criteria) for internal support structures located inside containment vessels that are relied upon to ensure an analyzed subcritical spatial arrangement and geometry of fissile contents. NUREG/CR-3854 refers to the ASME BPVC Section III, Subsection NF or ASME BPVC Section VIII, Division 1 requirements for materials (including fracture toughness criteria) for non-containment secondary enclosures, such as the OSV, that perform radiation shielding and structural support safety functions.

For non-containment structural components, the staff noted that the design of the OPTIMUS®-H package does not include any internal support structures or dedicated neutron absorbing materials for maintaining subcriticality of fissile contents; the criticality evaluation for the package does not include any such design features for demonstrating subcriticality. Therefore, the requirements of Subsection NG are not applicable to the design or construction of any OPTIMUS®-H packaging component. The application generally refers to Subsection NF, with

certain exceptions (as described and evaluated above in this SER), for design and construction of non-containment structural components; these include the OSV and the SIAs. The structural integrity of both the OSV and the SIAs are relied upon for radiation shielding. The OSV also provides structural support during lifting operations and is relied upon to protect the interior CCV from the direct effects of NCT and HAC impact loading during transport.

CCV Material Fracture Toughness

CCV Body and Lid Material Fracture Toughness

The staff confirmed that, with the exception of the alloy steel CCV closure lid bolts, all structural components of the CCV containment boundary, including the CCV body weldment, CCV closure lid, CCV port cover, and port cover bolts, are fabricated from wrought austenitic stainless steel. The staff identified that the wrought austenitic stainless-steel types specified for these CCV components does not undergo a significant ductile-to-brittle transition as a function of decreasing temperature down to the LST of the package. Notably, neither the ASME BPVC Section III nor RGs 7.11 or 7.12 require any specific material evaluation or material qualification test to ensure adequate fracture toughness for such wrought austenitic stainless-steel materials. Since the wrought austenitic stainless-steel types used in the fabrication of CCV are not susceptible to a significant ductile-to-brittle transition for temperatures down to the LST of the package, the staff determined that these materials are not susceptible to brittle fracture at the LST; therefore, they are acceptable, with respect to their fracture toughness.

CCV Bolting Material Fracture Toughness

The CCV closure lid bolts are fabricated from ferritic alloy steel and require a fracture toughness evaluation to ensure they will not be prone to brittle fracture at the LST. NUREG/CR-6007 recommends that fracture toughness criteria for containment boundary bolting meet the applicable fracture toughness requirements of Subarticle NB-2300 of the ASME BPVC, Section III, Subsection NB. Paragraph NB-2311 lists the types of materials for which notched bar impact testing is not required to meet Subsection NB requirements; this list includes ferritic alloy steel bolting items with a nominal size of one inch and less. Also, the requirements of Paragraph NB-2333 and Table NB-2333-1 specify that no impact test is required for bolts that have a nominal diameter of one inch and less. The staff noted that the Subarticle NB-2300 impact testing waiver for bolts with a nominal diameter of one inch and less, as well as non-bolting material with a nominal section thickness of five-eighths inch and less, is due to the fact that for ferritic carbon and alloy steels, thinner materials show significantly greater fracture toughness under axial loading (normal to a postulated crack face) in comparison to thicker materials due to better microstructural uniformity and plane stress conditions that favor greater crack tip plasticity. Therefore, since the CCV closure lid bolts meet the nominal size criterion of Subarticle NB-2300, they are not required to be impact tested in order to meet the requirements of Subsection NB.

Notwithstanding the impact testing waiver of Subarticle NB-2300, the application specifies that the CCV closure lid bolt material is Charpy impact tested in accordance with the requirements of the applicable ASME BPVC Section II bolting material specification to demonstrate that its Charpy impact test properties meet the acceptance criteria specified therein. The staff reviewed the applicable Charpy impact test requirements and acceptance criteria in the bolting material specification and confirmed that the CCV lid bolt material must be Charpy impact tested at a specified temperature that is significantly lower than the LST for the package, and the bolting material shall meet the specified minimum impact energy absorption requirements at the specified low-test temperature. Using an equation from Section 4.2 of NUREG/CR-1815 that

correlates dynamic fracture toughness with the measured absorbed impact energy from the Charpy impact test, the applicant performed an evaluation to estimate the dynamic fracture toughness of the bolt material based on the minimum impact energy absorption requirements and low-test temperature included in the material spec. Using this estimate of dynamic fracture toughness at the low Charpy impact test temperature from the material specification, the applicant estimated the nil-ductility transition (NDT) temperature for the bolting material based on the fracture toughness curve in Figure 2 of NUREG/CR-1815 showing the relationship between the dynamic fracture toughness for the material and the temperature of the material minus the NDT temperature. The applicant cited its estimate of the NDT temperature as a basis for determining that the bolting material would have adequate fracture toughness at the package LST since the estimate of the bolting material NDT temperature is significantly lower than the LST of the package.

The staff reviewed the applicant's supplemental evaluation for demonstrating adequate fracture toughness in the CCV closure lid bolts. The staff noted that the specific equation in Section 4.2 of NUREG/CR-1815 correlating dynamic fracture toughness with measured absorbed impact energy from the Charpy impact test, as cited in the application, is not endorsed by the staff for the purpose of determining dynamic fracture toughness based on specifications or measurements of absorbed Charpy impact energy since it has not been reviewed and approved by the staff for generic application to the CCV bolting material specification or any other material specification. However, considering the large value by which the LST of the package exceeds the applicant's estimate of the NDT temperature—where the NDT temperature was determined using an NRC-endorsed correlation between dynamic fracture toughness and material temperature minus NDT temperature—and the general positive correlation between ferritic material fracture toughness and absorbed Charpy impact energy in the ductile-to-brittle transition region of the Charpy impact test curve, the staff determined that the applicant's estimate of the dynamic fracture toughness and associated NDT temperature is credible for the case-specific application of qualitatively determining that the CCV bolting is highly unlikely to behave in a non-ductile manner at the LST. Accordingly, considering the ASME BPVC Section III, Subsection NB impact testing waiver in Subarticle NB-2300 for ferritic alloy steel bolts with a nominal diameter of one inch and less, the staff determined that the applicant's fracture toughness evaluation of the CCV closure lid bolts provides assurance that these containment bolts are not susceptible to brittle fracture at the LST. Therefore, the staff found that the applicant's fracture toughness evaluation of the alloy steel CCV closure lid bolts is acceptable.

OSV Material Fracture Toughness

The staff reviewed information in the application addressing the fracture toughness of the ferritic OSV materials. The staff's review of OSV material fracture toughness addressed the DCI OSV body and closure lid and the ferritic steel OSV closure devices, which include the alloy steel OSV closure lid bolts, bolt anchors, and sleeve nuts. Considering the lack of adequate OSV fracture toughness information in the initial application, the staff identified that several RAIs were needed to determine whether the OSV components have sufficient fracture toughness at the LST to ensure OSV integrity and safety function performance for NCT and HAC drop test conditions.

OSV Bolting Material Fracture Toughness

The staff identified that the initial application did not include any information on impact test criteria for ensuring adequate fracture toughness of the alloy steel OSV closure lid bolts, anchor bolts, and sleeve nuts. Therefore, the staff issued an RAI requesting the applicant to address

whether the impact testing criteria of ASME BPVC Section III, Subarticle NF-2300 for bolting material are required for the OSV closure lid bolting items at the LST.

In its May 5, 2023, RAI response, the applicant stated that impact testing of the OSV closure lid bolts is not required for the OPTIMUS®-H OSV since failure due to brittle fracture of multiple OSV lid bolts as a result of NCT and HAC impact loading at the LST will not lead to a loss of OSV lid closure function. The applicant stated that the structural evaluation of the OSV lid bolts presented in the application shows that the maximum stress ratios (computed load stress / design allowable stress) for the OSV lid bolts are very low for the most bounding NCT and HAC drop test conditions. The applicant described how the results of the OSV lid bolt stress analysis for the bounding NCT and HAC drop tests demonstrate that failure of a specified number OSV lid bolts will not lead to a loss of OSV lid closure function, considering the margin between maximum stresses in the remaining intact OSV lid bolts and the design allowable values. Based on these results, the applicant concluded that the OSV lid will adequately maintain its closure function during the NCT and HAC drop tests even if a specified number OSV lid bolts fail due to brittle fracture at the LST. The RAI response provided an update to the application that includes this additional information, describing how the results of the OSV lid bolt stress analysis for the bounding NCT and HAC drop tests support a determination that failure of a specified number OSV lid bolts will not lead to a loss of OSV lid closure function.

The staff reviewed the applicant's RAI response, including the update to the application, and confirmed that the margin between the maximum stresses in the OSV lid bolts and the design allowable values for the bounding NCT and HAC drop tests shows that the maximum stresses in the bolts are only a fraction of the yield stress. The staff determined that this provides reasonable assurance that the OSV lid bolts will not be susceptible to brittle fracture at the LST, considering that the bolting material is required to be free of significant fabrication defects that could initiate brittle fracture. Considering the information provided in the RAI response, the staff noted that even in the unlikely event that the most highly stressed OSV lid bolts were to fracture during the bounding HAC drop test, it is extremely unlikely that the drop test conditions would be capable of fracturing enough of the bolts to cause a loss of OSV lid closure function that would result in CCV damage or unacceptable radiation streaming from the CCV through the OSV opening at the location of the failed bolts. Therefore, the staff determined that the information provided in the applicant's RAI response is sufficient to support the determination that impact testing of the OSV lid bolting material is not needed to ensure adequate resistance to brittle fracture at the LST.

OSV Ductile Cast Iron Body and Lid Fracture Toughness

To address DCI fracture toughness at the LST for the DCI OSV body and closure lid, the initial application included case-specific requirements and associated acceptance criteria for performing tests and microstructure characterization tests on samples of DCI material used to construct the OSV body and closure lid. The application also included case-specific requirements for nondestructive examination (NDE) of the DCI OSV components and a requirement that the chemical composition and tensile properties of the DCI meet the requirements of the applicable standard material specification. The staff noted that, while the methods are specified in the application to be in accordance with the specified standard test methods, the use of these methods and the associated specific acceptance criteria for these tests are not included in the Code as an acceptable basis for qualifying fracture toughness properties of DCI at the LST, as addressed further below.

The requirements cited above are applicable to DCI used for transportation package containment vessels. The staff identified that the Code does not include less rigorous

qualification testing requirements to address fracture toughness of non-containment DCI structural components, such as those used for shielding and for structural supports. The structural integrity of the DCI OSV components is relied upon in the package safety analysis for radiation shielding, structural support for lifting operations, and for ensuring that the inner CCV is adequately protected from direct impact loads associated with NCT and HAC free drop and puncture drop tests. However, the DCI OSV is not a containment boundary, and therefore does not necessarily require the same rigor of fracture toughness qualification testing and acceptance criteria as that required for transportation containment vessels. However, the staff identified that the DCI OSV components have adequate resistance to brittle fracture at the LST.

Based on consideration of the OSV safety functions, the staff identified that the most severe loads that could potentially result in a large through-wall fracture of the OSV that adversely affects the performance of OSV safety functions are the impact loads caused by the NCT and HAC regulatory free drop and puncture drop tests, in particular the HAC impact loads caused by the 30-ft free drop and puncture drop test conditions specified in 10 CFR 71.73.

Therefore, for the NCT and HAC regulatory drop tests, the staff determined that the DCI material must have sufficient fracture toughness at the lowest service temperature (LST) to ensure adequate performance of radiation shielding and CCV structural protection safety functions. Specifically, the staff determined that adequate DCI fracture toughness should be ensured through demonstration of adequate protection against OSV fracture that could result in any of the following events:

- (1) OSV fracture that results in any of the CCV components exceeding their allowable stress limits in Table 2.1-3 of the application for NCT and HAC free drop and puncture drop tests;
- (2) For the NCT 2-ft free drop test, OSV fracture that results in a significant increase in external surface radiation levels, such that the package is not in compliance with 10 CFR Sections 71.43(f) and 71.51(a)(1);
- (3) For the HAC 30-ft free drop test and puncture drop test, OSV fracture that results in an external radiation dose rate exceeding 10 mSv/hour (1 rem/hour) at 1 m (40 in) from the external surface of the package, such that the package is not in compliance with 10 CFR 71.51(a)(2).

Therefore, the staff issued an RAI requesting the applicant to provide information that demonstrates adequate protection at the LST against OSV fracture that could result in any of the above three events. The staff's RAI stated that the information to demonstrate adequate protection against fracture should consider the maximum allowable fabrication flaw sizes based on the applicable NDE flaw acceptance criteria for the DCI OSV components.

In its May 2, 2024, RAI response, the applicant provided updates to the application sections addressing OSV fracture resistance. The staff reviewed the updated information in the application regarding the methods and requirements for qualification of OSV fracture toughness and found them to be acceptable based on the following considerations:

The updated application requires that DCI material be subjected to fracture toughness testing at the LST according to the specified standard test method. The staff noted that fracture toughness testing at the LST according to the specified standard test method is

consistent with the fracture toughness test method requirements for DCI material. Therefore, since the OSV DCI material is not used for containment (where it would need to be protected against essentially any significant fracture) and needs to be resistant only to large through-wall fractures that could result in any of the above three events, the staff determined that the use of test methods is also acceptable for the non-containment DCI OSV components.

The staff verified that the fracture toughness test data will be statistically meaningful and will conservatively account for measurement uncertainty. The staff reviewed these requirements and found them to be consistent with fracture toughness test data requirements and data evaluation methods for DCI material.

The staff confirmed the credible and conservative assumptions made by the applicant and determined that the applicant's calculation is sufficiently conservative and therefore acceptable.

In summary, the staff's review of applicant's May 2, 2024, RAI response determined that the fracture toughness test requirements, test data evaluation methods, and acceptance criterion ensures that the DCI used to construct the OSV components is adequately protected at the LST against fracture that could result in any of the above three events. Therefore, the staff determined that the applicant's fracture toughness evaluation, qualification test methods, and acceptance criterion for the OSV DCI material are acceptable.

SIA Material Fracture Toughness

The application states that no credit is taken for the carbon steel SIA in the structural evaluation of the other packaging components; however, the SIA is designed to withstand the most severe regulatory tests without structural failure, such that its shielding integrity is maintained for those conditions in which the SIAs are credited in the shielding evaluation. For the evaluation of brittle fracture, the application states that the carbon steel SIA provides radiation shielding, but has no structural support, thermal, or containment function. Considering its safety function, the application specifies that the SIA is designed in accordance with the Category III fracture toughness requirements of NUREG/CR-1815, which provide sufficient fracture toughness to prevent fracture initiation at minor defects typical of good fabrication practices.

The staff noted that the application does not include any requirements for performing Charpy impacts tests on samples of the SIA carbon steel material at the LST, although such impact tests are required by Subarticle NF-2300 of the ASME BPVC Section III, Subsection NF. Therefore, the staff issued an RAI requesting the applicant to state whether the welded carbon steel SIA material is required to be Charpy impact tested in accordance with the requirements of Subarticle NF-2300 of the ASME BPVC, Section III, Subsection NF.

In its RAI response, the applicant indicated that, considering the SIA shielding function, the carbon steel for the SIA does not require impact testing to ensure adequate fracture toughness since the carbon steel specification is for normalized steel made to fine grain practices. The applicant revised the text in the application to clarify that this is an exception to the fracture toughness criteria of ASME BPVC Section III, Subsection NF. The applicant explained that, although RG 7.11 does not specifically endorse the use of NUREG/CR-1815 Category III fracture toughness criteria for radiation shielding components, the application of these Category III criteria to shielding components, which have a lower safety significance than Category III containment components, is considered conservative and acceptable. The applicant identified that this approach is consistent with the graded quality approach of NUREG/CR-6407, which

classifies gamma shielding components of packaging for Type B (normal form) shipments as Category B items, recognizing that they have a lower safety significance than containment components that are classified as Category A. The applicant further explained that the smaller thickness SIA configurations are only credited for shielding under NCT conditions, not HAC conditions. Therefore, for these SIA designs, the potential for brittle fracture failure under HAC impact loading has no impact on the calculated dose rates under NCT. Only the largest thickness SIA configuration is credited for shielding under HAC conditions. The applicant stated that the largest thickness SIA radial wall is constructed from separate shells, such that even if brittle fracture failure were to occur in any one of the shells, the resulting crack would not result in radiation streaming that would significantly increase the dose rates outside the packaging.

The RAI response provided application updates stating that, based on application of the Category III fracture toughness criteria of NUREG/CR-1815, the SIA materials do not require Charpy impact testing to ensure adequate fracture toughness, and this constitutes an exception to the fracture toughness criteria of the ASME BPVC Section III Subsection NF.

The staff reviewed the applicant's RAI response, including the updates to the application, and noted that, for Type B packages, gamma shielding components, such as the SIAs, are generally categorized as having a lower safety significance than containment components. The staff also considered that, since the SIA carbon steel plate material is required to undergo a normalizing heat treatment and be manufactured to a fine grain practice, it meets the Category III fracture toughness criteria in NUREG/CR-1815 and RG 7.11. However, the staff identified that RG 7.11 specifies that Category III fracture toughness criteria are to be used for containment vessels requiring the lowest level of fracture protection based on the lowest radioactivity contents. As per RG 7.11, the Category III fracture toughness criteria for containment vessels carrying the lowest radioactivity contents are not generically endorsed or approved for radiation shield components. Therefore, the staff also considered additional case-specific factors covered in the applicant's RAI response, including the design of the largest thickness SIA configuration, and the fact that the lower thickness SIA configurations are only credited for shielding for the NCT loads and not for the HAC loads.

Based on review of the applicant's structural evaluation results for NCT drop tests, and considering the carbon steel fracture toughness characteristics discussed above, the staff determined that the maximum calculated stresses at the critical locations in the SIA during the NCT drop tests are sufficiently below the NCT design allowable values for carbon steel to provide confidence that a large fracture that results in an unacceptable decrease in SIA shielding performance is unlikely to occur for NCT drop test conditions. The staff also confirmed that the lower thickness SIA configurations are not considered in the structural evaluations of the HAC drop tests since they are not credited in the shielding evaluations for HAC. Therefore, since the NCT drop test conditions represent the peak loads and stresses that should be considered with respect to fracture tolerance for the lower thickness SIAs, the staff determined that meeting the fracture toughness criteria for Category III containments in NUREG/CR-1815 and RG 7.11 is sufficient for the lower thickness SIAs.

The structural evaluation of the largest thickness SIA for HAC drop test conditions shows maximum stresses that are below the HAC design allowable values for carbon steel, with tensile stresses significantly below the yield strength of the material. Further, the multi-shell design of the largest SIA provides some additional protection against fracture through the shell region since the shells and associated shell welds are not continuous material. Considering the unlikely case where the stress intensity factor for a postulated flaw at the maximum stress location in one of the shell welds reaches a critical value for crack propagation, it is even more unlikely that crack propagation would proceed into the adjacent shell sections. The staff determined that

these conditions provide adequate confidence that a large fracture that results in an unacceptable decrease in SIA shielding performance is unlikely to occur in the largest SIA for the HAC drop tests.

Based on review of the case-specific design configurations and the maximum stresses in the SIAs for the OPTIMUS®-H package, the staff determined that the applicant's proposal to use the fracture toughness criteria for Category III containments from NUREG/CR-1815 and RG 7.11 for the SIA carbon steel ensures that the carbon steel is adequately protected at the LST against fracture that could result in an unacceptable decrease in shielding performance, where radiation levels exceed the applicable regulatory limits of 10 CFR Sections 71.43(f), 71.51(a)(1), and 71.51(a)(2). Therefore, staff determined that the applicant's fracture toughness evaluation for the SIA carbon steel material is acceptable.

Impact Limiter Foam Mechanical Properties

The impact limiter system for the OPTIMUS®-H package consists of two identical foam-filled impact limiters. The impact limiters are attached to and fit over the upper and lower ends of the OSV assembly. As addressed in this SER, the impact limiters consist of energy-absorbing closed-cell polyurethane foam cores that are sealed inside welded stainless-steel shells. With respect to structural performance of the package, the impact limiters are designed to crush and absorb kinetic energy for the analyzed NCT and HAC impact events to limit the impact loads transmitted to the OSV, CCV, SIA, and contents. The staff's evaluation of the mechanical deformation properties of the impact limiter foam is provided below. Additional details concerning the staff's review of thermal performance and qualification tests for the impact limiter foam are provided in this SER.

Section 2.2.1.2 of the application provides a discussion of impact limiter foam deformation properties that are relied on for the structural evaluation of package performance for NCT and HAC regulatory drop tests. Section 8.1.5.3 of the application describes the acceptance tests that are performed to qualify new impact limiter foam material for service, specifically, to ensure that the foam has the deformation properties and performance characteristics needed to protect the structural integrity of packaging shielding and containment components (i.e., OSV, CCV, and SIAs) and package contents against the impact loads associated with NCT and HAC drop tests. The staff reviewed these sections to determine whether the description of the impact limiter foam deformation properties, and associated qualification tests to verify impact limiter foam properties and performance, are adequate to ensure that that foam can perform its requisite impact limiting safety functions for NCT and HAC.

Section 2.2.1.2 of the application reports the nominal foam densities and bounding dynamic stress-versus-strain data for the foam, which are used for the NCT and HAC drop test evaluations. This section states that upper-bound and lower-bound dynamic stress-versus-strain curves are developed considering foam crush direction, temperature effects, strain rate effects, and the effects of manufacturing tolerance on foam crush strength. The application reports the minimum and maximum foam temperatures considered in the development of the bounding dynamic stress-versus-strain curves and identifies that these temperatures represent the range of initial foam temperatures that are required to be established for the evaluation of the NCT and HAC drop tests. The application states that the nominal static crush strength of the foam at the minimum and maximum temperatures, both parallel and perpendicular to the direction of foam rise, are based on test data provided by the foam manufacturer. Section 8.1.5.3 of the application documents room temperature qualification testing of the foam to verify that measured foam density and static compressive strength at the specified strain intervals are within the specified bounding percentages of the nominal values provided by the foam

manufacturer, considering foam crush direction. The application states that the bounding dynamic crush strength of the foam is calculated from the static crush strength data and adjusted for temperature effects and fabrication tolerance using static-to-dynamic crush strength correlation coefficients provided by the foam manufacturer.

The staff reviewed the description of the methods used for developing the dynamic stress-versus-strain data for the foam and found them acceptable since they adequately ensure that the bounding crush strength properties of the foam are considered based on foam crush direction, strain rate effects, manufacturing tolerance, and the minimum and maximum initial foam temperatures required for evaluation of the regulatory drop tests. The staff determined that the application includes suitable acceptance tests for room temperature qualification of the static crush strength of the foam to ensure that the measured static crush strength of the foam is within the bounding percentages of the nominal values of static crush strength provided by the foam manufacturer. The staff verified that the range of allowable static crush strength values that are established based on qualification testing provide for the development of suitable upper-bound and lower-bound dynamic stress-versus strain data for evaluation of impact limiter performance for NCT and HAC drop tests. Therefore, the staff determined that the applicant's evaluation methods and qualification testing requirements for ensuring the required dynamic foam crush strength are acceptable for analyzing impact limiter performance for NCT and HAC drop tests.

Summary of Review Findings Regarding Mechanical Properties of Materials

Based on the foregoing evaluation, the staff finds that the information in the application regarding the mechanical properties of packaging materials is acceptable since the application specifies tensile and elastic properties, fatigue resistance properties, fracture toughness properties, impact limiter foam deformation properties, and associated qualification tests that ensure that the CCV, OSV, and SIAs are adequately protected against the analyzed modes of structural failure over the applicable range of service temperatures, as required for ensuring the structural integrity of package containment, shielding, and structural supports for NCT and HAC loadings. Accordingly, the staff finds that the applicant's evaluation of mechanical properties for OPTIMUS-H packaging materials meets the requirements in 10 CFR Sections 71.33(a), 71.35(a), 71.51(a), and 71.55(d) and (e).

2.2.5 Thermal Properties of Packaging Materials and Qualification of Organic Compounds

Chapter 3 of the application provides the thermal evaluation of OPTIMUS®-H package for NCT and HAC. The applicant's thermal evaluation was developed to show that the packaging component materials will remain within their applicable temperature limits, and the structural, containment, and shielding safety functions of the package are not adversely affected for simulated regulatory thermal tests associated with NCT and HAC. This SER addresses the staff's review of the thermal properties of metallic packaging materials. The staff's detailed review of the acceptability of the metallic packaging components for operation at the LST, based on fracture toughness considerations, is documented in this SER since fracture toughness is generally the limiting material property for low-temperature service of metallic structural components. This SER addresses the staff's review of the thermal properties and related qualification testing of the organic packaging materials, including the polyurethane foam cores for the impact limiters, the elastomeric O-rings for the containment seals, and the epoxy coating used for corrosion protection of ductile cast iron (DCI) OSV and carbon steel SIA components.

Thermal Properties of Metallic Packaging Materials

Sections 3.1 and 3.2 of the application provide thermal properties of the metallic packaging materials used in the package thermal analyses, the maximum allowable temperature limits for metallic packaging materials, and a single value for the minimum allowable temperature limit. The staff reviewed the maximum allowable temperature limits reported in the application for the metallic structural materials and confirmed that they are consistent with maximum values specified in the ASME BPVC Section II, Part D. The application indicates that the minimum allowable temperature limit is set equal to the lowest service temperature (LST) of the package for all components.

The thermal properties reported in the application for the metallic structural materials include density, thermal conductivity, specific heat capacity, and emissivity. The staff reviewed the values of the density, and thermal conductivity reported for the metallic packaging materials over the applicable service temperature range and verified that they are consistent with the applicable reference values specified in the ASME BPVC Section II, Part D. The staff noted that the ASME BPVC Section II, Part D does not include reference values for specific heat capacity and emissivity. Therefore, the staff confirmed that the room temperature values reported in the application for specific heat capacity and emissivity are generally consistent with those established in credible public sources and technical literature. Based on comparison of the thermal properties reported in the application with the applicable reference values provided in ASME BPVC Section II, Part D and other credible reference sources, the staff determined that the thermal properties reported in the application for the metallic structural materials are acceptable.

Thermal Properties and Qualification of Organic Materials

Chapter 3 of the application provides a description of the thermal properties and performance of the organic packaging materials, including the elastomer O-rings for the containment seals and the polyurethane foam for the impact limiters. Section 8.1.5 of the application describes acceptance tests for qualifying thermal properties of these materials to ensure the O-rings and impact limiters will perform as required to support package containment boundary integrity and impact limiter performance for NCT and HAC.

The staff verified that the application specifies maximum and minimum service temperature limits for the elastomer O-rings seals and the polyurethane foam and specifies that these limits are based on supplier data. The minimum service temperature limit for the O-rings and the foam is equal to the lowest service temperature (LST) for the package. The application shows that the calculated values of the maximum temperatures of these items for NCT, based on the thermal analyses, are less than the maximum service temperature limits of these materials. For the O-ring seals, the staff also confirmed that the HAC thermal analysis for the regulatory fire test demonstrates that the maximum temperature of the seals remains less than the maximum service temperature limit for the seals based on supplier data. The staff noted that the application does not include a maximum temperature limit for the polyurethane foam for the HAC fire test. However, the staff identified that, as per the regulatory test sequence requirements of 10 CFR 71.73, the impact limiters are not required to perform any impact energy-absorbing function during or after the HAC fire test since the HAC free drop and puncture drop tests of 10 CFR 71.73 precede the fire test. The evaluation of the HAC fire test shows no unacceptable damage to the impact limiter foam, which remains encased inside the stainless-steel shells during the fire, although the impact limiter design incorporates thermal relief plugs to allow gases generated by the foam material to escape during the fire. The evaluation of the HAC fire test also shows that the containment function of the CCV is maintained considering the maximum calculated temperature for the containment seals and the maximum calculated pressure inside the CCV for the bounding radioactive content heat loads.

The staff also reviewed the application description of the acceptance tests for qualifying the thermal properties and behavior of elastomer O-rings and polyurethane foam. The application states that the O-rings are to be made of the elastomer compound specified in the packaging drawings, and the elastomer shall be qualified based on testing to verify material composition; physical properties, including hardness, tensile strength, elongation, and specific gravity; low temperature properties; compression set at high-temperature; and dimensional acceptance testing. The staff confirmed that the application description of these qualification tests is adequate for ensuring the elastomer O-rings are capable of performing their requisite containment seal function over the full range of their service temperatures, as specified in the package thermal evaluation, which includes the evaluation of the HAC fire test. The staff also reviewed requirements for thermal testing of each batch of polyurethane foam used to construct the foam cores of the impact limiters. The foam thermal qualification tests include flame retardancy and intumescence based on exposure of the foam to the specified thermal conditions. The staff determined that the application description of these tests is adequate to support the thermal evaluation of the foam for HAC fire test conditions since the evaluation considers the damaged foam properties for heat transfer through the impact limiters during the fire test. Therefore, the staff determined that the application description of the thermal qualification tests of the elastomer O-rings and polyurethane foam is acceptable for ensuring that these materials are capable of performing in a manner that is consistent with their analyzed conditions for the NCT and HAC thermal evaluation.

For the epoxy coated surfaces of the DCI OSV and carbon steel SIA components, the staff determined that additional information was needed regarding the thermal properties and performance of the coating and the evaluation of its effect on the package thermal analyses. Therefore, the staff issued an RAI requesting the applicant to provide information regarding the safety classification, thermal properties and performance, and qualification of the coating for ensuring that the effects of the coating on the thermal performance of the package are adequately evaluated for NCT and HAC.

In its May 5, 2023, RAI response, the applicant identified that the coating used on the OSV and SIA components is categorized as not important to safety (NITS) because it is not credited in any of the safety analyses, including the thermal analyses, and its only function is corrosion protection. The applicant also stated that coated steel has a higher emissivity than uncoated steel; therefore, a lower-bound emissivity value for oxidized cast iron is used for the OSV for the NCT thermal evaluation. For HAC, the regulations specify the value of surface emissivity to be applied. Therefore, the applicant determined that the results of the thermal analyses are bounding even if the coating fails. The applicant further explained that since the coating is not credited for its higher emissivity in the NCT thermal analysis and HAC thermal pre-fire analysis, using a higher emissivity for coated OSV surfaces would result in lower packaging temperatures. Consequently, the applicant identified that it is conservative to not credit the higher emissivity of the OSV coating in the thermal analysis.

The applicant's RAI response also discussed the maximum service temperature limit of the epoxy coating for normal service conditions. The RAI response explained that since the maximum calculated temperatures of the OSV and SIAs for NCT are significantly less than the maximum service temperature limit of the coating, there is no thermal degradation of the coating for NCT. The applicant also stated that the manufacturer's specifications for the coating used on OSV and SIA are not verified through qualification testing because the coating is not credited in the thermal analysis, and it is not important to safety.

The staff reviewed the RAI response and determined that the applicant adequately explained the function, safety classification, and thermal properties of the coating. The staff confirmed that

the coating is appropriately categorized as not important to safety since it is not credited in any of the package safety analyses. The potential effect of the coating on the thermal performance of the package is adequately evaluated in the RAI response. Specifically, since the coating is used exclusively for protecting the OSV and SIA components against corrosion, and it is categorized as not important to safety, the higher emissivity of the coated DCI OSV components is appropriately not used in any of the package thermal analyses, which conservatively use the lower emissivity of the uncoated oxidized DCI surface. The staff also noted that the SIA is not evaluated in any of the package thermal analyses since the SIA is just used for gamma shielding and does not perform any heat transfer function. Therefore, the staff determined that the effect of the coating on potential thermal behavior of the SIA is moot and irrelevant to the evaluation of package thermal performance. The staff also located and reviewed the coating supplier product sheet specifications including the maximum temperature limit for continuous normal service. Based on review of the maximum service temperature limit, the staff verified that the coating will not thermally deteriorate during normal service conditions, so that the OSV and SIA components will be adequately protected against general corrosion during normal service. Therefore, the staff determined that the applicant's evaluation of the thermal properties and performance of the coating, and its effect on the package thermal analyses, is acceptable.

Summary of Review Findings Regarding Thermal Properties of Materials

Based on the foregoing evaluation, the staff finds that the information in the application addressing the thermal properties of packaging materials and associated qualification tests for organic materials is acceptable for the analyses of package thermal and containment performance for NCT and HAC. Therefore, the staff finds that the applicant's evaluation of thermal properties and associated qualification tests for packaging materials meets the requirements of 10 CFR Sections 71.33(a), 71.35(a), 71.51(a), and 71.55(d) and (e).

2.2.6 Materials for Shielding and Criticality Control

Chapter 5 of the application provides the shielding evaluation of OPTIMUS®-H package. The OPTIMUS®-H package relies exclusively on metallic structural components and the stainless-steel inner shells of the impact limiters for neutron and gamma shielding. The metallic structural components included in the shielding evaluation are the stainless steel CCV, ductile cast iron (DCI) OSV, and the carbon steel SIAs. The impact limiter foam and the stainless steel outer thin shells of the impact limiters are assumed to be void in the shielding analysis. The staff verified that the description of the package shielding model includes sufficient information on the component geometries and conservatively accounts for dimensional tolerances associated with component fabrication. The staff also verified that the shielding model description includes material elemental compositions and densities for all components credited in the shielding analyses, as needed to ensure that the calculated radiation dose rates are within the regulatory limits specified in 10 CFR Sections 71.47 and 71.51(a) for NCT and HAC. The staff determined that the analyzed structural and thermal performance of the metallic CCV, OSV, SIA, and impact limiter inner shell components for NCT and HAC, as demonstrated in the structural and thermal safety analyses of the package, are sufficient to ensure that the shielding performance of these component materials will be adequately maintained for NCT and HAC conditions. The staff identified that there are no organic materials or other special or unique materials credited for shielding. Therefore, there are no additional special controls on operating temperatures or other service conditions (beyond those already evaluated for the structural and thermal safety analyses) to ensure adequate shielding material performance. Based on the foregoing considerations, the staff determined that the applicant's description of the shielding characteristics of the packaging materials are acceptable for use in the shielding evaluation to

demonstrate compliance with the regulatory standards for radiation levels and dose rates specified in 10 CFR Sections 71.47 and 71.51(a) for NCT and HAC.

Chapter 6 of the application provides the criticality evaluation of the OPTIMUS®-H package. With respect to criticality control, the OPTIMUS®-H package design does not use any neutron absorbing materials for ensuring subcriticality, and it does not incorporate any internal support structure to ensure a subcritical geometry or spatial configuration of fissile contents. As addressed in the criticality safety analyses, the subcriticality safety margin of the package is ensured by controls on the quantities of fissile radioisotopes and hydrogenous materials present in the radioactive contents. The subcriticality of the package is also ensured based on the analyzed structural and thermal performance of the packaging enclosures (CCV and OSV) and the impact limiters for NCT and HAC; SIAs are not credited for ensuring subcriticality. The nuclear properties of packaging component materials and package contents that are used in the package criticality analyses are reviewed as part of the staff's criticality evaluation, provided in Section 6 of this SER. Therefore, the staff's materials evaluation does not address any specific aspect of criticality properties and performance for the OPTIMUS®-H package.

2.2.7 Corrosion, Coatings, Maintenance, Chemical Reactions, and Radiation Effects

10 CFR 71.43(d) specifies that a package must be made of materials and construction that assure that there will be no significant chemical, galvanic, or other reaction among the packaging components, among package contents, or between the packaging components and the package contents, including possible reaction resulting from inleakage of water, to the maximum credible extent. This regulation also states that account must be taken of the behavior of materials under irradiation. Section 2.2.2 of the application addresses the potential for corrosion, chemical, galvanic, or other reactions among the packaging components, among package contents, or between the packaging components and the package contents. Section 2.2.3 of the application addresses the behavior of packaging materials under irradiation. Section 8.2 of the application describes the OPTIMUS®-H package maintenance criteria, which includes periodic inspections, tests, and repair and replacement criteria to ensure continued acceptable performance of packaging components. The NRC staff's review of the application to ensure compliance with 10 CFR 71.43(d) requirements for packaging components is documented in the subsections below.

Section 4.5 of the application provides a detailed evaluation of the chemical compatibility and potential content reactions for the TRU waste contents, including detailed analysis of the potential for flammable gas generation and analysis of potential combustion reactions. The staff's evaluation of potential content reactions, and their potential effects on package safety, is included in the staff evaluation of the package thermal and containment analyses. Therefore, the staff's review of application sections addressing content reactions is documented in this SER.

Corrosion, Coatings, and Packaging Maintenance Criteria

Section 2.2.2 of the application addresses the potential for chemical, galvanic, or other reactions to cause corrosion of metallic packaging components and deterioration of organic materials. This section also describes the use of a coating to protect against corrosion of the OSV and SIA components. This section of the application determines that the packaging component materials, consisting primarily of stainless steel, coated carbon steel, coated cast iron, and polyurethane foam, will not undergo significant chemical, galvanic, or other reactions in the operating environment.

With respect to the potential for chemical and galvanic reactions among the metallic packaging components to cause corrosion of these components, the application states that, except for the threaded interior surfaces of bolt holes, the ductile cast iron (DCI) OSV surfaces and carbon steel SIA surfaces are coated with the specified type of high-temperature, radiation-resistant epoxy coating for corrosion protection. The application also states that this type of epoxy coating is commonly used in the nuclear industry for similar applications and is highly resistant to chemical reactions and has very good abrasion resistance. The application notes that the coated surfaces of the DCI OSV assembly contact the uncoated outer surfaces of the stainless steel CCV assembly and the uncoated surfaces of the stainless-steel impact limiter shells. The application also notes that the coated outer surfaces of the carbon steel SIA assembly contact the uncoated inner surfaces of the stainless steel CCV assembly. Based on these material contacts, the application identifies that no significant chemical, galvanic, or other reactions are expected among the coated surfaces of the DCI OSV assembly, the coated surfaces of the carbon steel SIA assembly, and the uncoated surfaces of the stainless steel CCV assembly and stainless-steel impact limiter shells.

With respect to potential reactions between the packaging components and the package contents, the application states that the radioactive contents are packaged in secondary containers, such as drums or liners, which limit chemical interaction between the payload and the CCV. The application identifies that TRU waste may contain very small amounts of halides, including hydrogen chloride (HCl) gas, that could originate from radiolysis of polyvinyl chloride or halogenated organics within the TRU waste, with a potential to cause stress corrosion cracking (SCC) of the CCV stainless steel. However, the application states that the nature of the TRU waste is expected to preclude the possibility of producing any significant quantities of free HCl gas inside the secondary containers, and any small quantities of HCl gas are likely to be absorbed by the moisture content of the TRU waste and retained inside the secondary containers. The application references gas sampling programs on drums of newly generated TRU waste at Idaho National Engineering Laboratory and Rocky Flats Plant that did not detect any HCl gas in the drum headspace or any layers of confinement within the waste. Therefore, the applicant determined that, since corrosives are prohibited from the payload, there are no significant chemical, galvanic, or other adverse reactions between the contents and the CCV.

With respect to potential reactions that may cause deterioration of components made of organic compounds, the application states that the polyurethane foam material used for the cores of the impact limiters has a long history of use in radioactive material transport packages without any adverse reactions. The application states that the polyurethane foam is very low in free halogen content and leachable chlorides, and it is sealed inside stainless steel shells in a dry environment. The application states that in the unlikely event that moisture enters the impact limiter cavity, it could not penetrate the closed-cell structure of the polyurethane foam to cause leaching of chlorides. The application states that the elastomer O-ring material that contacts the CCV stainless steel sealing surfaces contains no corrosives to adversely affect the packaging; the elastomer O-ring material does not have any chemical, galvanic, or other reactions with stainless steel.

The staff's review of the applicant's evaluation of corrosion and galvanic reactions, including its coating application and maintenance criteria for packaging components, follows below.

Ductile Cast Iron, Carbon Steel, and Alloy Steel Components

Since the body and lid components of the OSV and SIAs are fabricated from DCI and carbon steel, respectively, the staff identified that they are susceptible to general corrosion when directly exposed to sheltered air or outdoor air environments. Further, considering the material

contacts and service environments for the DCI OSV and carbon steel SIA, the staff identified that they are also susceptible to galvanic corrosion where their metal surfaces are in direct contact with the stainless steel CCV assembly. The staff noted that the application identifies that the OSV and SIA components have the specified type of epoxy coating applied to all exposed surfaces, except for OSV threaded bolt holes, for corrosion protection. The staff identified that the application of a suitable coating compound is used to protect the underlying DCI and carbon steel materials against the conduction of small galvanic electric currents across the dissimilar metal contacts (i.e., from the active DCI and carbon steel materials to the passive stainless steel) that would lead to galvanic corrosion. Irrespective of galvanic effects, the coating also protects against corrosive chemical attack of the DCI and carbon steel surfaces by air, water, and other chemical species present in the operating environment, especially DCI surfaces exposed to outdoor air with precipitation and dissolved compounds from salts, dirt, and road chemicals.

The staff noted that the application included only limited information on the coating compound and its use for protecting the OSV and SIA components against corrosion. To ensure that the OSV and SIA components are capable of performing their structural, thermal, and radiation shielding safety functions during their service lives, the staff determined that additional information was needed regarding the properties and application of the coating compound for protection of OSV and SIA surfaces against loss of material due to general and galvanic corrosion. Therefore, the staff issued an RAI requesting the applicant to provide information regarding the properties, qualification, acceptance testing, and maintenance inspections of the coating for ensuring adequate corrosion protection of OSV and SIA surfaces, such that the components are capable of performing their requisite safety functions for NCT and HAC.

In its May 5, 2023, RAI response, the applicant identified that the coating used on the OSV and SIA components is categorized as not important to safety (NITS) because it is not credited in the safety analyses, and its only function is corrosion protection. The applicant also identified that the coating would not undergo thermal deterioration during normal service since the maximum analyzed temperatures of OSV and SIA components for NCT are lower than the maximum continuous service temperature limit specified for the coating. Therefore, the applicant determined that the corrosion protection function of the coating would not be affected for NCT service temperatures, and for the highest HAC temperatures associated with the HAC fire test, the coating is not relied upon to perform any safety function. The applicant stated that the coating has no known adverse chemical reactions with steel or cast iron substrates, and it does not have any potential for flammable gas reactions with the transuranic waste contents.

The applicant also stated that, while there are no specific qualification tests performed on the coating compound to verify manufacturer product specifications, the coating supplier product data sheet includes performance data based on the identified standard abrasion tests and adhesion tests. With respect to the application of the coating to the OSV and SIA component surfaces, the applicant indicated that the minimum required coating thickness is verified, and the adhesion of the cured coating is also tested in accordance with the identified standard test method. The applicant also indicated that the coating is visually inspected for unacceptable defects, such as peeling, blistering, or damage caused by package handling or service conditions and repaired to meet specification requirements. For these inspections, the maintenance criteria specify that superficial defects, such as minor surface corrosion, scratches, blemishes, and adhered material/particles, may be removed by polishing the packaging surfaces with fine abrasives. Significant damage shall be repaired to restore the components to the applicable requirements of the packaging drawings or the damaged components may be replaced. Replacement components shall satisfy the applicable requirements of the packaging drawings and additional requirements specified therein.

The staff reviewed the RAI response and confirmed that the applicant provided an acceptable description of the product specifications and methods used to ensure that the coating will adequately protect the OSV and SIA components against the deteriorating effects of general and galvanic corrosion during the service lives of the components. The staff located and reviewed the coating supplier product sheet specifications including the maximum temperature for continuous normal service and the standard tests for coating adhesion and abrasion resistance. Based on review of the coating product specifications and the applicant's description of visual inspection criteria and tests for coated components, the staff verified that the applicant's coating selection, application, acceptance tests and inspections, and visual inspection criteria for maintenance of coated components are sufficient to ensure that the coating will provide adequate corrosion protection for the OSV and SIA components during normal service conditions.

The staff confirmed that the coating is not credited in any of the safety analyses for ensuring OSV and SIA structural, thermal, or shielding performance for NCT and HAC. Therefore, the staff determined that, for normal service conditions, the coating supplier product specifications, acceptance tests for verification of coating thickness and adhesion, and periodic visual inspections are sufficient to ensure that the OSV and SIA components are adequately protected against significant loss of material due to corrosion that could degrade these safety functions. Consequently, the staff determined that the applicant's RAI response is acceptable for demonstrating that the coating adequately protects the OSV and SIA against significant corrosive chemical and galvanic reactions during normal service, such that the base metals of the OSV and SIA are capable of performing their requisite safety functions for NCT and HAC.

Since the coating is not applied to any metal surfaces of the alloy steel CCV and OSV closure bolts or the threaded bolt holes in the OSV flange, the staff identified that galvanic corrosion is a credible corrosion mechanism for the alloy steel CCV closure bolts (since they are in contact with stainless steel CCV bolt holes), and general corrosion is a credible corrosion mechanism for the CCV and OSV closure bolts and the OSV threaded bolt holes. However, the staff confirmed that the application maintenance criteria include requirements for visual inspections to look for excessive wear, corrosion, or other damage on the bolts. The maintenance criteria require visual inspections prior to each shipment of all packaging threaded fasteners that are removed during package loading or unloading operations. The maintenance criteria also require visual inspections of all packaging threaded fasteners, including those not removed during package loading or unloading operations, within the 12 month period prior to any shipment. For visual inspection acceptance criteria and corrective actions, the maintenance criteria for threaded fasteners state that fasteners that have minor damage or wear may be refurbished by chasing the threads, and minor surface corrosion may be removed by polishing with fine abrasives. The maintenance criteria specify that fasteners that show visible signs of excessive wear or significant corrosion or damage shall be replaced with new fasteners that meet the requirements of the packaging drawings.

The application maintenance criteria state that tapped holes for threaded fasteners do not require visual inspections; however, tapped holes that do not fit-up properly with the mating fastener may be refurbished by chasing the threads or repaired as necessary using threaded inserts in accordance with the packaging drawings. The maintenance criteria require that all fastener holes with threaded inserts shall be visually inspected within the 12-month period prior to any shipment to verify that the threaded inserts are not displaced or damaged.

The staff verified that the visual inspection criteria for the CCV and OSV closure bolts are adequate for ensuring that unacceptable corrosion degradation is detected and corrected prior to a loss of bolt material integrity that could impair the structural integrity and proper function of

the CCV and OSV closure devices. For the threaded bolt holes in the OSV flange, the staff confirmed that visual inspections are not needed for bolt holes without threaded inserts since adequate material condition of the inner surfaces of the tapped holes is reasonably ensured by proper engagement with the mating bolts, and there are suitable corrective actions for cases where there is not adequate engagement between the bolts and the tapped holes. Therefore, considering potential corrosion mechanisms, the staff determined that the application maintenance criteria are adequate for ensuring that the alloy steel CCV and OSV bolts and the OSV threaded bolt holes are maintained in an adequate condition for proper function of the CCV and OSV closure devices. Accordingly, the staff determined that the application maintenance criteria are acceptable for ensuring that general and galvanic corrosion reactions are not significant such that they may cause unacceptable degradation of the alloy steel CCV and OSV closure bolts or the OSV threaded bolt holes.

Austenitic Stainless-Steel Components

The staff identified that the austenitic stainless steel alloy types used for fabrication of the CCV components and impact limiter shells are resistant to general corrosion, even if exposed to outdoor air and precipitation environments. The general corrosion resistance of the CCV and impact limiter shells is due to stainless steel passivity in sheltered air and outdoor air with precipitation environments. For such environments the chromium in the stainless-steel alloy reacts with the oxygen in the air to spontaneously form a resilient protective passive oxide layer that is highly resistant to general chemical attack by ambient air and water, thereby protecting the underlying metal from general corrosion. The staff also verified that stainless steel CCV components and impact limiter shells are also generally resistant to galvanic corrosion in the packaging operating environments since passive stainless steel in contact with either exposed or coated DCI or carbon steel would not draw significant galvanic electric current away from the passive stainless-steel surface.

While the stainless steel CCV assembly and impact limiter shells are generally resistant to general and galvanic corrosion, these components may be susceptible to localized corrosion effects, including pitting and crevice corrosion, during prolonged exposure to outdoor air and precipitation environments. Such localized corrosion effects are due to the presence of dissolved chlorides or other halide species in the outdoor air and precipitation environments. Over extended service periods, these dissolved anion species, which are commonly present in aqueous outdoor air environments, especially on road surfaces, can cause localized penetration of the protective passive oxide layer on stainless steel surfaces, resulting in the formation of pits and crevice corrosion. Further, stainless steel components under high tensile stress (such as weld residual stress) exposed to aqueous outdoor air environments are also susceptible to the formation of cracks due to chloride-induced stress corrosion cracking (SCC). Therefore, the staff reviewed the service environments, operating requirements, and maintenance criteria for the stainless steel CCV assembly and impact limiter shells to determine whether these components are adequately protected from significant localized corrosion effects that could potentially degrade component performance over long service periods.

The staff noted that during transport, the CCV assembly is housed inside the OSV assembly and is generally protected from direct exposure to outdoor precipitation, whereas the thin stainless steel impact limiter shells may be directly exposed to outdoor precipitation and road spray. Therefore, the staff considered that over long service periods, the impact limiter shells should be considered susceptible to localized corrosion effects. The exterior surfaces of the stainless CCV assembly components may also be susceptible to localized corrosion effects if these surfaces are allowed to accumulate atmospheric deposits that form aqueous electrolytes on the exterior surfaces. The interior of the stainless steel CCV is generally not as susceptible to

localized corrosion since, for cases where air environments are allowed during transport, it is required to be dried, closed, and sealed, and the CCV must remain closed when not in use. Therefore, the staff determined that the CCV interior is adequately protected against significant localized corrosion effects during road transport.

The staff reviewed the application maintenance criteria to determine whether they include adequate requirements for visual inspections and corrective actions to detect and repair significant degradation due to localized corrosion that could impair the safety function performance of stainless-steel packaging components. The application maintenance criteria state that, prior to each shipment, the exterior of the packaging, including the impact limiters and OSV assembly, shall be visually inspected to verify that the physical condition is unimpaired. Superficial defects on the exterior of the packaging, such as marks, scratches, or dents do not require repair. However, any significant damage to the packaging exterior, such as holes in the outer thin sections of the stainless-steel impact limiter shells, shall be repaired prior to shipment. Further, the application maintenance criteria require that all exposed interior and exterior surfaces of the impact limiters, OSV body and lid, OSV drain port plug, CCV body and lid, CCV vent and test port plugs, and SIA body and lid 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 staff verified that the visual inspection criteria for the exposed surfaces of the stainless steel CCV are adequate for ensuring that unacceptable degradation due to localized corrosion effects is detected and corrected prior to a loss material integrity that could impair the structural integrity and containment function of the CCV. For the pre-shipment visual inspection of the exposed surfaces of the impact limiter stainless steel shells, the staff determined that the criterion specifying that holes in the outer thin shell sections shall be repaired prior to shipment is sufficient for ensuring that the impact limiter foam is adequately protected against significant water intrusion during shipment that could degrade the impact energy-absorbing functionality of the polyurethane foam cores. To support its determination regarding adequate protection of the impact limiter foam, the staff also considers the application description of the foam properties and qualification tests (addressed below in SER Section 7.7.3) for ensuring that the foam has adequate resistance to water penetration in the unlikely event that moisture were to enter the impact limiter cavity. Therefore, considering potential localized corrosion mechanisms, the staff determined that the application maintenance criteria are adequate for ensuring that the exposed surfaces of the stainless steel CCV and impact limiter shells are maintained in an adequate condition to ensure the structural integrity and containment function of the CCV and the impact energy-absorbing function of impact limiters. Accordingly, the staff determined that the application maintenance criteria are acceptable for ensuring that potential localized corrosion reactions on the exposed surfaces of stainless-steel components are not significant such that they may cause unacceptable degradation of the stainless steel CCV and impact limiter shells.

Potential Water Inleakage

With respect to the potential for inleakage of water, Section 2.7.7 of the application provides a structural evaluation of the package for deep water immersion to demonstrate that the undamaged package containment system can withstand an external water pressure of 290 psi for a period of not less than one hour without collapse, buckling, or inleakage of water, in accordance with 10 CFR 71.61. This evaluation demonstrates that there is no significant inleakage of water into the containment system as a result the deep water immersion. The staff's in-depth review of the applicant's detailed structural analyses, including the deep water immersion condition, is documented in this SER. Based on this evaluation, the staff verified that there is no credible scenario where there would be inleakage of water into the undamaged

containment system. On this basis, the staff determined that the potential for inleakage of water does not need to be included in the evaluation to demonstrate, pursuant to 10 CFR 71.43(d), that there will be no significant chemical, galvanic, or other reaction among the packaging components, among package contents, or between the packaging components and the package contents.

Physical-Chemical Reactions for Non-Metallic Packaging Components

To ensure that the impact limiters perform as required for NCT and HAC drop test conditions, the staff reviewed application information regarding potential reactions that could degrade the impact energy-absorbing functionality of the polyurethane foam cores. The application states that the polyurethane foam has a long history of use in radioactive material transportation packages without any adverse reactions. The application states that the polyurethane foam is sealed inside the cavity of the impact limiter stainless steel shells in a dry environment. The application also states that the impact limiter foam is very low in free halogen content and leachable chlorides. The application explains that in the unlikely event that moisture enters the impact limiter cavity, it could not penetrate the closed-cell structure of the foam to cause leaching of chlorides. The application includes a specific test of leachable chlorides as part of its acceptance tests for qualifying each batch polyurethane foam. Therefore, the application determines that no adverse reactions are expected for the impact limiter polyurethane foam.

The staff reviewed this information and determined that it is adequate to demonstrate that there are no significant reactions that result in unacceptable degradation of the impact energy-absorbing functionality of the polyurethane foam. The staff confirmed that the application includes a specific acceptance test of leachable chlorides as part of the acceptance test program for qualifying each batch of polyurethane foam. Further, since the polyurethane foam is sealed inside the cavity of the impact limiter stainless steel shells, it is generally protected against significant water intrusion during normal service conditions, provided that the stainless-steel shells are adequately inspected for ruptures or other damage or deterioration that could result in water intrusion. As addressed above in this SER, the staff confirmed that the application maintenance program includes adequate visual inspection criteria and corrective actions to detect and repair holes, damage, or degradation that could result in significant water intrusion with the potential to impair the impact energy-absorbing function of the impact limiters. Therefore, the staff determined that the application information is acceptable for demonstrating that the impact limiter foam is adequately protected against significant physicochemical reactions that could cause an unacceptable decrease in the energy-absorbing performance of the impact limiters.

Section 4.5 of the application provides a detailed evaluation of the chemical compatibility and potential content reactions for the TRU waste contents, including detailed analysis of the potential for flammable gas generation and analysis of potential combustion reactions. The staff's evaluation of potential content reactions, and their potential effects on package safety, is included in the staff evaluation of the package thermal and containment analyses. Therefore, the staff's review of application sections addressing content reactions is documented in this SER.

Behavior or Materials Under Irradiation

Section 2.2.3 of the application discusses the evaluation of the effects of radiation on packaging materials. The application states that the packaging is designed to withstand the damaging effects from radiation through the use of durable materials of construction, such as austenitic stainless steel, carbon steel, alloy steel bolts, and ductile cast iron, all of which are unaffected

by the radiation levels of the radioactive contents contained in the package. The application also states that the polyurethane foam material used for the impact limiter cores is unaffected by gamma radiation exposure up to an absorbed radiation dose of 2×10^8 rad, equivalent to 1000 rads per hour over a continuous exposure period of over 20 years. The application references tests showing that this radiation exposure has no effect on foam density or crush strength. The application states that the elastomer material for the containment seal O-rings shows no change in physical properties at radiation exposures below 10^6 rad. The application identified that this exposure level for the containment seal O-rings is attained only after many years of operation. Therefore, the application determined that normal wear, as opposed to radiation exposure, is the main factor affecting the O-ring replacement frequency.

The application also discusses the potential effects of radiation on the lubricants used for the containment seal O-rings and the threaded fasteners. The O-ring lubricant is used to help protect the O-rings from damage by abrasion, pinching, or cutting, and to help seat the O-ring properly and protect the polymer from environmental damage. The application stated that since the O-ring lubricant is frequently cleaned and replaced, and because most of the lubricant's benefit occurs during installation, radiation damage of the lubricant is not a concern. The application identified that the thread lubricant used for the CCV and OSV bolts is commonly used for nuclear applications and is suitable for use in radiation environments. The application determined that since none of the threaded fasteners are in high exposure areas, and the lubricant is frequently cleaned and replaced, the thread lubricant is not prone to radiation damage.

The staff reviewed the application information on the evaluation of the effects of radiation on packaging materials. The staff confirmed that radiation levels of the radioactive contents contained in the package will have no effect on any of the properties or performance characteristics of any of the metallic materials used in the construction of packaging components. The staff also determined that the application includes credible information showing that the density and crush strength of the impact limiter foam are not affected by the radiation exposure levels associated with continuous use of a loaded package for over 20 years. The staff identified that the packaging is not going to be continuously loaded with radioactive contents during its service life. Therefore, there is essentially no concern with radiation-induced deterioration in the mechanical properties for the impact limiter foam for packaging service lives of 20 years. The staff noted that the physical properties of the containment seal O-rings could potentially be affected by radiation exposure after many years of operation. However, the staff reviewed the application maintenance criteria for the O-rings and confirmed that the visual inspection and replacement criteria for O-rings adequately ensures that the O-rings are unlikely to be affected by significant radiation-induced degradation prior to requiring replacement based on normal wear. The staff also identified that the maintenance criteria for the epoxy coating applied to the DCI OSV and carbon steel SIA components and the lubricants used on the O-rings and threaded fasteners adequately ensures the component-protection functions of these substances is not adversely affected by irradiation. Based on these considerations, the staff determined that the application adequately accounts for the behavior of packaging materials under irradiation. Therefore, the staff finds that the information in the application on the evaluation of the effects of radiation on the packaging materials is acceptable.

Summary of Review Findings Regarding Corrosion, Coatings, Maintenance, Chemical Reactions, and Radiation Effects

Based on the foregoing evaluation, the staff finds that the information in the application addressing the potential for chemical, galvanic, or other reactions among packaging components, including use of coatings and maintenance criteria for protection against such

reactions, is acceptable since the application adequately demonstrates that there are no such significant reactions that would degrade the safety function performance of the packaging components. Further, the staff finds that the application adequately accounts for the behavior of packaging materials under irradiation. Therefore, the staff finds that the application evaluation of corrosion, coatings, maintenance criteria, chemical reactions, and radiation effects for packaging components meets the requirements of 10 CFR 71.43(d).

2.2.8 Materials 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), 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 and 10 CFR 71.55(d)(2) for fissile 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.

Based on review of the statements and representations in the application, the NRC staff concludes that the materials used in the OPTIMUS-H 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®-H package: (1) provides adequate protection against the thermal tests specified in 10 CFR Part 71, and (2) meets the thermal performance requirements of 10 CFR Part 71 under normal conditions of transport (NCT) and

hypothetical accident conditions (HAC) to transport compliant Transuranic (TRU) waste (Content 1-1), non-compliant TRU waste (Contents 1-2A, 1-2B and 1-2C), and irradiated fuel waste (IFW) (Contents 2-1 and 2-2).

3.1 Description of Thermal Design

As stated in SAR Section 1.1, *Introduction*, the package is designed as Type B(U)F-96 transportation package to ship contents of Type B quantities of normal form TRU waste and IFW, as summarized in SAR Table 1-1. SAR Section 1.2.2, *Radioactive Contents*, notes that TRU waste is divided into two types: compliant TRU waste (Content 1-1) and non-compliant TRU waste (Content 1-2). Acceptable non-compliant TRU waste is subdivided into three sub-types of TRU waste drums: aerosol cans with compressed gas propellant (Content 1-2A), aerosol cans with liquified gas propellant or unknown propellants (Content 1-2B), and standard DOT 3E lecture bottles (Content 1-2C).

The OPTIMUS®-H package with TRU waste is shipped on a non-exclusive use shipment or under exclusive use control. The OPTIMUS®-H package with IFW is shipped under exclusive use control. The package accessible surface temperatures are less than 122°F for non-exclusive use shipment and 185°F for exclusive-use shipment under 100°F ambient temperature in shade, as specified in 10 CFR 71.43(g).

The package consists of a Cask Containment Vessel (CCV), an Outer Shield Vessel (OSV), and an Impact Limiter System (ILS), as shown in Figure 1-1 of the SAR. The package can be configured with an optional Shield Insert Assembly (SIA) inside the CCV to provide additional shielding for some contents. All details and relevant dimensions of the packaging components are provided in the Licensing Drawings, Nos. 70000.14-501 through 552 in SAR Appendix 1.3.3, *Packaging General Arrangement Drawings*. The applicant described the thermal design of the package in SAR Section 3.1, *Description of Thermal Design*, as below:

- The CCV is the innermost vessel of the packaging, which serves as the primary containment boundary of the package. The CCV is a stainless-steel vessel with a bolted lid closure designed to the leak-tight containment in accordance with the criterion of ANSI N14.5. The CCV lid includes operating and containment features, such as a port for evacuation and backfill of the cavity with the inert gas.
- The OSV, which consists of a body and lid, serves as the primary shielding component of the package, and provides impact and thermal protection to the CCV.
- The ILS consists of two impact limiters, with a thick stainless-steel inner shell and a stainless-steel outer skin filled with polyurethane foam, to provide impact and thermal protection for the package and protect the OSV lid. The impact limiter includes a skid plate that restricts the contact area between the ILS inner end and the OSV top end and restrict the heat flow into the OSV from the ILS during a fire accident. The applicant modeled an air gap between the ILS inner shell and the OSV lid to account for this feature in thermal analysis.

In SAR Section 3.1.2, *Content's Decay Heat*, the applicant outlined the requirements for TRU waste limited to heat loads of 200 W with helium CCV fill gas and 50 W with air CCV fill gas, and for IFW limited to heat load of 1,500 W with helium CCV fill gas. TRU waste is for open transport and closed transport, while IFW is for open transport only, as shown in SAR Table 3.1-1.

The staff reviewed the description of thermal design provided in SAR Section 3.1 and determined that description of the thermal design is appropriate for 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 SAR Section 3.2, "Material Properties and Component Specifications," and provided material properties of the packaging components, including polyurethane foam, in SAR Tables 3.2-1 thru 3.2-5 used for the thermal evaluation.

The applicant stated, in SAR Section 3.2, that the minimum temperature limit for all components is -40°C (-40°F) and the O-ring seal material, used as the containment boundary O-ring seal for the CCV lid, vent port cover and drain port cover (optional), has a continuous operating temperature range of -40°F (-40°C) to 400°F (204°C) as recommended by the Parker O-Ring Handbook.

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

3.3 General Considerations

3.3.1 Thermal Model

The applicant stated, in SAR Section 3.3, *Thermal Evaluation under Normal Conditions of Transport*, that the contents, CCV, OSV, and ILS are modeled for thermal evaluations and the design features of OSV lifting trunnions and tiedown lugs, OSV closure bolts, impact limiter mounting lugs are not explicitly simulated in the thermal model because these components do not significantly affect the thermal performance of the packaging. The applicant described in SAR Section 3.3 that the TRU waste and the IFW is modeled as a homogenous cylinder of helium which has a lower thermal conductivity than the package contents to result in higher fill gas temperatures. The applicant modeled the volumetric heat generation of the loaded contents as uniformly distributed within the helium inside the CCV cavity and specified the solar heat flux per 12-hour period in SAR Table 3.3-1 and described the boundary conditions in SAR Section 3.3.

The staff reviewed the package configuration and thermal design and accepts the packaging components simulated in the thermal model. The staff also reviewed the assumptions, methodology, thermal features, and initial/boundary conditions used in the thermal analyses and concludes that the NCT thermal model is acceptable for evaluation of the package thermal design.

3.3.2 Thermal Contact Resistance

The applicant stated, in SAR 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 in ANSYS computer code. The TCC is defined as the reciprocal of the thermal contact resistance. As described in SAR 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-in²-°F for surfaces with low thermal contact resistance, 15 Btu/hr-in²-°F for surfaces with low/moderate contact resistance, 5 Btu/hr-in²-°F for surfaces with moderate contact resistance, 1 Btu/hr-in²-°F for surfaces with high/moderate contact resistance, and 0.5 Btu/hr-in²-°F for a surface/gap with high contact resistance, as shown in SAR Table 3.3-4.

The applicant stated, in SAR section 3.5.2.1, that sensitivity analyses were performed for the NCT condition by changing all TCC values from the mixed values discussed in SAR Section 3.3 to 1,000 Btu/h-in²-°F for low thermal contact resistance and 0.5 Btu/h-in²-°F for high thermal contact resistance. The applicant presented the results of the sensitivity analyses in SAR Figures 3.5-1 through 3.5-6 which show the difference in the package temperatures is less than 1°F between the models with two levels of TCC values.

The staff reviewed the NCT sensitivity analysis for OPTIMUS®-H package and referred to the HAC sensitivity analysis for OPTIMUS®-L package (i.e., perfect contact during the 30-minute fire and high contact resistance during the post-fire cooldown) and accepts that the HAC sensitivity analysis for OPTIMUS®-L packaging can be applied to OPTIMUS®-H packaging because OPTIMUS®-L packaging includes the same CCV design and similar contents to OPTIMUS®-H packaging.

3.3.3 Shield Insert Assembly (SIA) Design Features

The applicant stated, in SAR Section 3.3, that thermal analyses of the packaging configurations that include a Shield Insert Assembly (SIA) are not performed because the package temperatures for these conditions (with SIA) will be bounded by those (without SIA) for the configuration evaluated. This conclusion is based on the results of the thermal evaluation of the OPTIMUS®-L packaging with TRU waste in a 110-gallon drum and a 55-gallon drum in a SIA, both with the same total decay heat load. The results demonstrate that the presence of the SIA has minor impact on the maximum packaging temperatures, and results in significantly lower temperatures for the CCV cavity fill gas and contents.

The staff reviewed both NCT and HAC thermal analyses with the SIA included in the thermal analyses of the OPTIMUS®-L Package and accepts that the demonstration of the OPTIMUS®-L package can be applied to the OPTIMUS®-H package. The staff determined that the SIA will not significantly affect the thermal performance of the OPTIMUS®-H 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:

- (1) TRU waste with 200 W and helium-filled gas (single package for open transport and two packages in International Standards Organization (ISO) Container for closed transport),
- (2) TRU waste with 50 W and air-filled gas (single package for open transport and two packages in ISO Container for closed transport), and

- (3) IFW with 1,500 W and helium-filled gas (single package for open transport). The applicant evaluated the package under NCT with two conditions of solar insolation and shade.

The applicant described the initial conditions and boundary conditions in SAR Section 3.3.

The applicant presented the results in SAR Tables 3.3-8 (TRU waste in a single package for open transport and TRU waste in two packages in an ISO Container for closed transport) and 3.3-9 (IFW in a single package for open transport). The results indicate that:

- 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,
- all packaging component temperatures remain below their allowable temperature limits for NCT. The applicant summarized the maximum packaging components temperatures from the NCT thermal analyses (1) ~ (3) mentioned above and the corresponding allowable temperature limits for NCT in SAR Table 3.1-3.

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 SAR Tables 3.3-8 and 3.3-9 are acceptable and the maximum temperatures of the package components, containment seals and contents/fill-gas, are below their allowable limits, as shown in SAR Table 3.1-3 for NCT.

The staff finds that in still air and shade, the maximum accessible surface temperature of 114°F (45°C) for TRU waste is below the limit of 122°F (50°C) for non-exclusive use shipment and the maximum accessible surface temperature of 177°F (80°C) for IFW is below the limit of 185°F (85°C) for exclusive use shipment.

Cold

The applicant stated, in SAR Section 2.6.2, *Cold*, 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 SAR 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, including containment O-rings 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 NCT heat and cold conditions and concludes that the package material and component temperatures will not extend beyond the specified allowable limits during NCT consistent with the tests specified in 10 CFR 71.71.

3.4.2 Maximum Normal Operating Pressure (MNOP)

The applicant stated, in SAR Section 3.3.2, *Maximum Normal Operating Pressure*, that the MNOP is calculated by treating all gases in the CCV as ideal gas and determining the partial pressure contributions from temperature change (both TRU waste and IFW), water vapor pressure (TRU waste), and gas generation from radiolysis (TRU waste). The applicant presented the MNOPs of IFW and TRU waste in SAR Tables 3.1-5 and 3.3-10.

The staff reviewed calculations of the MNOPs of the IFW (1,500 W with helium-fill-gas) and compliant and non-compliant TRU wastes (50 W with air-fill-gas and 200 W with helium-fill gas) and confirmed that the MNOPs, as shown in SAR Table 3.3-10, are below the design pressure of 100 psig for NCT, as provided in SAR Section 2.6.1.1, *Summary of Pressures and Temperatures*.

3.4.3 Differential Thermal Expansion

The applicant stated, in SAR Section 2.6.1.2, *Differential Thermal Expansion*, that differential thermal expansion of the packaging components is evaluated considering 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 OSV cavity and between the SIA and the CCV cavity.

The applicant calculated the nominal axial and radial clearances between CCV and OSV as well as the nominal axial and radial clearances between SIA and CCV. Compared to the nominal axial and radial clearances allowed in the design, the applicant stated, in SAR Section 2.6.1.2, that the CCV expands freely within the OSV cavity, and the SIA expands freely within the CCV cavity under NCT thermal loading.

The staff reviewed SAR Chapter 3 for (a) the upper-bound temperature for the CCV and the lower-bound temperature of the exterior surface for the OSV, 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 OSV cavity and the SIA within the CCV cavity under NCT.

3.4.4 International Standards Organization (ISO) Container

The applicant performed the NCT thermal evaluation for two packages inside an ISO Container, containing TRU waste. In thermal evaluation, the applicant assumed a low thermal conductivity of 0.001 Btu/hr-in-°F at bottom surface of the ISO Container, neglected the convection internal to the ISO Container, and only simulated the radiation heat transfer between the package and the ISO Container. The applicant presented the analytical results in SAR Table 3.3-8 and Figure 3.3-14 for two packages in an ISO.

The staff finds the thermal analysis for two packages inside an ISO Container (containing TRU waste), as described in SAR Section 3.3, is acceptable and the calculated maximum packaging component temperatures, shown in SAR Table 3.3-8 and Figure 3.3-14, are below the allowable limits for the NCT.

3.5 Thermal Evaluation under HAC

As described in SAR Section 3.4, *Thermal Evaluation under Hypothetical Accident Conditions*, the applicant performed thermal analyses for the package with damage consistent with a 30-ft drop and 40-inch drop on a 6-inch pin with TRU waste (200 W) and IFW (1,500 W).

3.5.1 Axial Position of the CCV within the OSV Cavity

As described in SAR Section 3.5.2.3, *Axial position of the CCV within the OSV Cavity*, the applicant also performed separate HAC thermal analyses with the CCV and contents positioned at bottom and top ends of the OSV cavity during the fire, to determine the maximum temperatures of the various packaging components. The maximum HAC package temperatures vs. CCV position in the OSV cavity are presented in SAR Table 3.5-2.

The staff finds that the maximum temperatures of the packaging components, including the containment CCV O-ring, from the CCV positioned at either top end or bottom end of the OSV cavity are below the allowable limits and therefore, are acceptable for the HAC thermal evaluations.

3.5.2 Package Orientation during HAC

The applicant described, in SAR Section 3.5.2.4, *Package Orientation during HAC*, that the package was modeled and compared for both horizontal orientation and top-down (inverted) orientation during the 30-minute fire and the post-fire cooldown. As presented in SAR Section 3.5.2.4 and Figures 3.5-8 and 3.5-9, the temperatures of the packaging components, including containment CCV O-ring, are below the allowable limits.

The staff finds that the HAC fire analysis assuming the package in horizontal orientation, as described in SAR Section 3.4, is appropriate with slightly higher CCV O-ring temperature than in inverted orientation. The staff confirmed that the maximum O-ring temperatures have been below the allowable limit of 400°F in both orientations of the CCV during the HAC fire.

3.5.3 Peak Temperatures for Different Cumulative Damaged Models

The applicant described, in SAR Section 3.5.2.7, *Peak Temperatures for Different Cumulative Damaged Models*, the HAC thermal evaluation for different cumulative damaged models on the sensitivity of the packaging temperatures to the extent of cumulative damage resulting from the HAC free drop and puncture tests. The applicant considered four damage scenarios of maximum center puncture damage (base case), maximum off-center puncture damage, bounding center puncture damage and extreme top end damage. Compared to the base case of the maximum center puncture damage (see SAR Table 3.5-3 for maximum HAC package temperatures for bounding damage cases), the applicant noted, in SAR Section 3.5.2.7, that (a) the highest peak temperatures result from the extreme top end damage HAC thermal evaluation and (b) the CCV O-ring reaches a peak temperature for this extreme worst-case scenario but remains below the allowable limit of 400°F and the service limit of 470°F for a 5-hour exposure time.

The staff reviewed SAR Table 3.5-3 and finds the packaging component temperatures in all four damage scenarios are below the allowable limits. The staff accepts that the peak CCV O-ring temperatures in all four damage scenarios remain below the temperature limit of 400 °F for continuous service and the service limit of 470°F for a 5-hour exposure time, as shown in the Parker O-ring Handbook.

3.5.4 Simulation of Polyurethane Foam

The applicant stated, in SAR Section 3.4.3.1, that the polyurethane foam will generate expanding char layer, exhibit superficial blackening when reaching temperatures of 500°F and is covered with continuous char foam at 600°F. Therefore, the applicant included the charring of the polyurethane foam in the package impact limiters for HAC simulations by assigning the properties of dry air to the polyurethane foam regions that have any of their nodes reaching a temperature of 500°F or greater. Since dry air has a thermal conductivity that is less than the polyurethane foam, the heat that has penetrated the impact limiter during the 30-minute fire will be impeded as it moves out to the impact limiter surfaces to be rejected to the 100°F ambient during the post-fire cooldown.

The staff referred to references and confirmed that thermal degradation of the polyurethane foam would be anticipated when temperature more than 250°C (482°F) and therefore accepts the simulation of the polyurethane foam by assigning the properties of dry air to the polyurethane foam regions reaching a temperature of 500°F or greater.

3.5.5 Maximum HAC Temperatures and Pressures

Temperature

The applicant presented the packaging component temperatures in SAR Tables 3.1-4 and 3.4-3 and the containment O-ring seals in SAR Figure 3.4-8 and Figure 3.4-9. The HAC results show that the packaging component temperatures remain below their allowable temperature limits and the O-ring seals will maintain containment under the HAC fire when loading the 200-W TRU waste and the 1500-W IFW, respectively.

The staff reviewed the HAC results and confirmed that the HAC thermal evaluations on TRU waste (50 W/air-filled and 200 W/helium-filled) and IFW (1,500 W, helium-filled) provides results acceptable for HAC test sequence and all packaging components, including the containment O-rings, will not extend beyond the corresponding allowable limits under HAC, consistent with the tests specified in 10 CFR 71.73(c)(4).

Pressure

The applicant stated, in SAR Section 3.4.3.2, *Maximum HAC Pressure Results*, that the maximum HAC pressures are calculated using the same equations as for the MNOP calculations but including the pressure increases from CCV gas temperature increase from a fire, water vapor, radiolysis gases, and release of gases in pressurized containers of non-compliant contents. The applicant described the HAC pressure calculations of compliant TRU waste (Content 1-1) and IFW (Contents 2-1 and 2-2) in SAR Section 3.4.3.2 and summarized the maximum HAC pressures of the compliant TRU waste (Content 1-1) and the IFW (Contents 2-1 and 2-2) in SAR Tables 3.1-5 and 3.4-4.

The applicant provided the HAC pressure calculations of non-compliant TRU waste: Aerosol Cans, Type 1 (Content 1-2A), Aerosol Cans, Type 2 (Content 1-2B), and Standard DOT 3E Lecture Bottles (Content 1-2C) in SAR Section 4.5.5.1, *Pressure Calculations for Non-Compliant TRU Waste*. SAR Table 4.5-5 summarized the parameters used for NCT and HAC pressure calculations and the resulting NCT and HAC pressures for non-compliant TRU waste (Contents 1-2A, 1-2B, and 1-2C).

The staff reviewed SAR Sections 3.4.3.2 and 4.5.5.1 and Tables 3.4-4 and 4.5-5, the staff confirmed that the maximum HAC pressures for non-compliant TRU waste (Content 1-1, 200 W), compliant TRU waste (Contents 1-2A, 1-2B and 1-2C, 200 W), and IFW (Contents 2-1 and 2-2, 1,500 W), as shown in SAR Tables 3.1-5 and 3.4-4, are below the design pressure of 225 psig (SAR Section 2.6.1.1).

3.5.6 Maximum Thermal Expansion

The applicant stated, in SAR 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 OSV shells is higher than that of the CCV shell based on HAC thermal evaluation. Following the HAC fire, the temperature gradients between CCV and OSV remain bounded by those resulting from NCT heat. Therefore, the differential thermal expansion between CCV and OSV during HAC fire will be bounded by the results for NCT heat (see SAR Section 2.6.1.2).

The staff reviewed SAR Section 2.7.4.2 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, and therefore there is no significant issue on thermal stress caused by thermal expansion under HAC.

3.6 EVALUATION FINDINGS

Staff has reviewed the package description and evaluations and concludes that they satisfy the thermal requirements of 10 CFR Part 71. It is noted that TRU waste is transported within a single package for open transport or with no more than two packages in a 20-foot enclosure (e.g., ISO container), and the IFW is only transported within a single package for open transport.

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 NCT and HAC thermal evaluations 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 the package satisfies 10 CFR 71.43(g) for exclusive shipment of the IFW and for non-exclusive shipment or exclusive use control of TRU waste.

Staff has reviewed the package design, construction, and preparations for shipment and concludes that the package material and component temperatures and the package internal pressure 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 and the package internal pressure will not exceed the specified allowable limits during hypothetical accident conditions consistent with the tests specified in 10 CFR 71.73(c)(4).

The staff agrees with the decay heat limit of 200 W for TRU waste if the package still meets the 5 vol% hydrogen and other flammable gases concentration of the total gas inventory within any confined volume for shipment within the allowable shipping time frame.

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

4.0 CONTAINMENT EVALUATION

The objective of the containment review is to verify that the OPTIMUS®-H package containment design is adequately described and satisfies the containment requirements of 10 CFR Part 71 under NCT and HAC to transport compliant Transuranic (TRU) waste (Content 1-1), non-compliant TRU waste (Contents 1-2A, 1-2B and 1-2C), and irradiated fuel waste, IFW (Contents 2-1 and 2-2).

4.1 Containment System

The OPTIMUS®-H package is designed and prepared for shipment to ensure no loss or dispersal of contents as demonstrated to a sensitivity of 10^{-6} A₂ per hour under NCT in compliance with 10 CFR 71.51(a)(1), and no escape of Krypton-85 exceeding 10 A₂ in one week and no escape of radioactive material exceeding a total of A₂ in one week under HAC in compliance with 10 CFR 71.51(a)(2). Each package prepared for shipment may only contain one form of non-compliant TRU waste, alone or in combination with compliant TRU waste.

Containment Boundary

The containment system of the package, as shown in SAR Figure 4.1-1, consists of the following components: cask containment vessel (CCV) body, CCV lid, CCV lid inner O-ring, CCV port cover and CCV port cover inner O-ring, and all containment welds. Other than the CCV lid closure and port cover closure, there are no penetrations of the containment system. The package has no valves or pressure relief devices for continuous venting and does not rely on a filter or mechanical cooling system to meet containment requirements. Both CCV lid inner O-ring and port cover inner O-ring are O-rings with a continuous operating temperature range of -40°F (-40°C) to 400°F (204°C). The containment system material of construction is evaluated and selected to avoid chemical, galvanic, or other reactions as discussed in SAR sections 2.2.1 and 2.2.2.

Welds and Closure

As described in SAR section 4.1, the CCV is a stainless-steel weldment including a CCV shell, a base plate, and a bolt flange. The CCV shell is formed into a cylinder by a full penetration longitudinal seam weld (containment weld). The bottom plate and bolt flange are connected to the CCV shell by full penetration circumferential welds (containment welds). All containment welds are examined using dye-penetrant testing (PT) and radiographic testing (RT) to verify that there are no unacceptable indications of weld flaws.

4.2 Containment under Normal Conditions of Transport (NCT)

The applicant stated, in SAR section 4.2, that:

- (1) the package is designed to a leak-tight containment criterion per ANSI N14.5, with a leakage rate less than or equal to 1×10^{-7} ref-cm³/sec under NCT,
- (2) the package maximum normal operating pressure (MNOP) is 100 psig,
- (3) the structural evaluation in SAR section 2.6 shows there would be no loss or dispersal of radioactive contents, and that the containment boundary, seal region, and closure bolts do not undergo any inelastic deformation when subjected to the conditions of 10 CFR 71.71, and
- (4) the thermal evaluation in SAR section 3.3.1 shows 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.

The staff reviewed the thermal evaluation described in SAR section 3.3.1 and confirmed that the maximum temperatures of the package containment components, including the containment O-ring seals, are below their allowable limits under NCT for both TRU waste and IFW, as shown in SAR Tables 3.1-3, 3.3-8 and 3.3-9.

The staff also confirmed that the calculated maximum pressures, as shown in SAR section 3.3.2 and tables 3.1-5 and 3.3-10, are below the MNOP/design pressure of 100 psig for both TRU waste (Contents 1-1, 1-2A, 1-2B, and 1-2C) and IFW (Contents 2-1 and 2-2) under NCT.

4.3 Containment under Hypothetical Accident Conditions (HAC)

The applicant stated, in SAR section 4.3, that:

- (1) the package is designed to a leak-tight containment criterion per ANSI N14.5 with a leakage rate less than or equal to 1×10^{-7} ref-cm³/sec under HAC,
- (2) the package maximum HAC pressures, shown in SAR Table 3.1-5, is below the bounding pressure of 225 psig,
- (3) the structural evaluation in SAR section 2.7 shows there would be no loss or dispersal of radioactive contents, and that the containment boundary, seal region, and closure bolts do not undergo any inelastic deformation when subjected to the conditions of 10 CFR 71.73, and
- (4) the thermal evaluation in SAR section 3.4.3 shows 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.

The staff reviewed the thermal evaluation described in SAR section 3.4.3 and confirmed that the maximum temperatures of the package containment boundary components, including the containment O-ring seals, are below their allowable limits under HAC for both TRU waste and IFW, as shown in SAR Tables 3.1-4 and 3.4-3.

The staff also confirmed that the calculated maximum pressures, as shown in SAR section 3.4.3 and Table 3.4-4 for both TRU waste (Contents 1-1, 1-2A, 1-2B, and 1-2C) and IFW (Contents 2-1 and 2-2), are below the design pressure of 225 psig under HAC.

4.4 Leakage Rate Tests

The applicant stated, in SAR section 4.4, that the package is tested to a sensitivity of at least 1.0×10^{-3} ref-cm³/sec for the pre-shipment leakage rate test and a leak-tight criterion of 1×10^{-7} ref-cm³/sec for fabrication, maintenance, and periodic leakage rate tests, in accordance with ANSI N14.5.

The applicant stated, in SAR Chapter 8, that the test procedures for fabrication, maintenance, and periodic leakage rate tests shall be reviewed and approved by personnel whose qualifications and certifications in the nondestructive method of leak testing include certification by a nationally recognized society at a level appropriate to the writing and/or review of leakage rate testing procedures (e.g., an American Society of Non-destructive Testing (ASNT) Level III in leak testing) as noted in Section 8.8, "Quality Assurance," of ANSI N14.5.

The staff reviewed SAR section 4.4 and Chapter 8 and accepts the leakage rate test descriptions and criteria, described in the application, for pre-shipment, fabrication, maintenance, and periodic leakage rate tests, including the fabrication leakage rate testing to be performed on the entire containment boundary of the package and the leakage rate testing procedures to be approved by personnel with an ASNT Level III certification.

4.5 Inerting Procedure

The applicant referred to Report No. 70000.14-R-07 Rev. 0, "OPTIMUS Proof Test Report for Inerting Nested Contents," provided as Attachment 1 to their RAI Response (ADAMS No. ML23128A030) and use this report for demonstrating the inerting process. Staff notes that SAR Attachment 7.5-2 does not refer to Report No. 70000.14-R-07 Rev. 0 but rather refers to a 2016 Savannah River National Laboratory report, "Proof of Principle Testing for Inerting a 9978 Containment Vessel," (SRNL-STI-2016-00674), that discussed the inerting of a small vessel; this report was not demonstrated to be relevant to the large containments associated with the OPTIMUS®-H package. Report No. 70000.14-R-07 Rev. 0 stated that the inerting test assembly used in the proof test is a full-scale licensed OPTIMUS® cask containment vessel (CCV) test assembly loaded with simulated waste contents (filtered bags and 55-gallon drum). The inerting was applied to three test configurations in which each configuration was tested with various non-radioactive contents:

- Configuration 1 Test Setup – 55-gal drum and loose waste within the CCV
- Configuration 2 Test Setup – three nested bags of waste within the CCV
- Configuration 4 Test Setup – three waste bags nested in a 55-gal drum within the CCV.

The applicant noted in Report No. 70000.14-R-07 Rev. 0:

- All types of drums, bags, cans, boxes, and bottles used to confine TRU waste (placed within the CCV) should have venting mechanisms with a minimum hydrogen diffusivity rate of $\geq 1.48 \times 10^{-5}$ mole/sec/mole (for drums) and $\geq 1.08 \times 10^{-5}$ mole/sec/mole (for bags) or meet the diffusivity rates provided in SAR Table 4.5-4.
- The operating range, operating temperature, accuracy, total device error, and calibration of the instruments used in the test should be specified in the test procedure.

Staff notes that SAR Attachment 7.5-2 did not associate instrument specifications (e.g., sensitivities) to the process details (e.g., number of helium backfills, system time constants) of the full-scale proof test described in Report No. 70000.14-R-07 Rev. 0.

The proof test showed the measured oxygen levels in various locations to be reduced to less than 1 vol% following the final helium backfill cycle during the inerting process for the loading configurations used in the test.

The staff reviewed the CCV license drawings in the application and recognized that the full-scale test CCV is consistent with the CCV described in the license drawings. However, the staff finds that:

- (1) the content loading configurations used in the inerting demonstration are not able to represent or bound the real-life conditions, and
- (2) the proof test has not fully demonstrated that the inerting gas (e.g., helium) could be introduced effectively to the innermost packaging to prevent the development of the flammable gas mixtures in any confined region within the containment boundary of the package throughout the entire transport operation.

Therefore, the inerting effectiveness has not been sufficiently proven.

4.6 Flammable Gas Calculations/Requirements for Compliant TRU Waste

The applicant stated, in SAR section 1.2.2.1, that each package prepared for shipment of TRU waste may only contain one form of non-compliant TRU waste contents, alone or in combination with compliant TRU waste.

The applicant stated, in SAR section 1.2.2.1.1, that

- (1) compliant TRU waste (Content 1-1) with a total decay heat exceeding 50 watts shall be inerted with helium gas per SAR Attachment 7.5-2,
- (2) compliant TRU waste contents that are not inerted with helium in accordance with Attachment 7.5-2, shall be limited to the lower flammable limit (LFL) of 5 vol% hydrogen of the total gas inventory within any confined volume,
- (3) compliant TRU waste contents that are inerted with helium gas in accordance with Attachment 7.5-2, shall be limited to the lower oxidant concentration (LOC) limit of 5 vol% oxygen of the total gas inventory within any confinement volume, and
- (4) the flammable gas concentration limits for TRU waste contents that include any approved non-compliant TRU waste items shall be limited in accordance with the requirements for the respective non-compliant TRU waste content, as described in the sections 1.2.2.1.2 through 1.2.2.1.4. Specifically, it states that TRU waste contents containing Content 1-2A, 1-2B, or 1-2C shall be inerted with helium per Attachment 7.5-2, regardless of heat load and shall be demonstrated to comply with the flammable gas concentration, using the procedure described in Attachment 7.5-3, using the LOC limit of 5 vol% oxygen of the total gas inventory within any confinement volume.

The staff makes no determination on the use of inerting, and the staff continues to find acceptable the use of the LFL of 5 vol% hydrogen (or lower LFL, if warranted by the flammable gas), instead of limiting the LOC less than 5 vol% oxygen, as the criteria to meet the flammable gas requirement, in accordance with NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material (Final Report)."

Based on the staff analysis of the application, the staff notes that

- Content loading configurations used in the inerting proof test do not demonstrate how these would represent or bound real-life conditions at sites. Thus, inerting effectiveness is not sufficiently proven by this proof test.
- The application does not demonstrate that the gas constituents, including hydrogen, oxygen, and helium, within the innermost packaging of any confined region, would be at non-flammable concentrations for all loading configurations during transport operations. It is noted there are numerous variables, including for physical testing, to ensure an inherently safe transport operation if there is a combustible amount of flammable gas within the package.
- The application does not demonstrate that the inerting process will prevent the development of flammable gas mixtures in any confined region of the package throughout its operation. Specifically, the application does not demonstrate how the uncertainties introduced by the complexity of the package inner configurations affect the effectiveness of an inerting procedure.
- There is a lack of sufficient basis and evidence to adopt LOC (≤ 5 vol%) as a criterion to prevent ignition and combustion-related events for package. It is noted that (1) National Fire Protection Association (NFPA) Code 69, "Standard on Explosion Prevention System," states: (1) where the LOC is greater than or equal to 5 vol%, a safety margin of at least 2 vol% below the worst credible case LOC shall be maintained, and (2) the research literature points that the LOC could be lower than the applicant's proposed 5 vol% limit when the gas in the package cavity has a higher temperature under NCT and HAC. As noted above, there also should be some margin to account for any uncertainty.
- Limiting the LOC to less than 5 vol% oxygen could allow the flammable gases (mainly hydrogen gas) to be generated in the CCV during transport, up to 8 vol% according to SAR section 4.5.2, which is a 60% increase in flammable gas content, and which is over the limit of 5 vol% hydrogen concentration stated in NUREG-2216.
- A combustion-related event could occur when the package is opened with a hydrogen concentration (e.g., 8 vol%) above the LFL of 5 vol% hydrogen, even with a LOC below 5 vol%.

Based on the conditions and observations above, the staff determined:

- The application does not fully demonstrate that there will not be an ignition/combustion-related event with the proposed contents with the proposed LOC 5 vol% of oxygen. Further, no conservatism have been introduced to cover the uncertainties and complexities discussed.
- Risk consequence would be high with more hydrogen (and other flammable gases) in a package. The proposed inerting to limit oxygen less than 5 vol% is not as inherently safe as the current guidance of limiting hydrogen and other flammable gases less than 5 vol%. To date, ignition/combustion events related to flammable gases of NRC certified packages, approved per NUREG-2216 guidance, have not been reported.

Therefore, the staff does not accept the use of LOC (≤ 5 vol%) together with an inerting method to replace the limiting hydrogen/flammable gas concentration (≤ 5 vol%) in order to prevent ignition and combustion-related events for the package.

4.7 Evaluation Findings

Based on the review of the statements and representations in this application, the staff concludes with the following findings:

- 1) The containment design has been adequately described and evaluated and the package design meets the containment requirement of 10 CFR Part 71.
- 2) The package is leak-tight and meets the requirements of 10 CFR 71.51(a)(1) under NCT and 10 CFR 71.51(a)(2) under HAC. The staff makes no determination on inerting procedure itself.
- 3) The application did not demonstrate the effectiveness and safety of inerting with helium as the sole means of safely transporting flammable content.
- 4) Each package, containing one form of non-compliant TRU waste contents, alone or in combination with compliant TRU, shall be limited to the LFL of 5 vol% hydrogen (or lower LFL if warranted by the flammable gas) of the total gas inventory within any confinement volume and shall satisfy the shipping requirements, including but not limited to, the contents, the heat load limit, the allowable pressure limits, the allowable package component temperature limits, and the maximum quantity requirements per thermal perspective.
- 5) As discussed in Chapter 3 of this SER, the staff agrees with the decay heat limit of 200 W for TRU waste, this requires that the package still meets the LFL of 5 vol% hydrogen (or lower LFL if warranted by the flammable gas) of the total gas inventory within any confined volume for shipment within the allowable shipping time frame.

5.0 SHIELDING EVALUATION

The staff reviewed the applicant's Safety Analysis Report (SAR), calculations, and computer inputs and outputs to ensure that there is reasonable assurance of adequate protection to the public and occupational workers. The review was performed to verify that the package design meets the external radiation requirements of 10 CFR 71.47 and 10 CFR 71.51 for normal conditions of transport (NCT) and hypothetical accident conditions (HAC) under exclusive and non-exclusive use. NUREG-2216, *Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material*, was used to guide the staff's review.

5.1 Shielding Design Features

Shielding for the OPTIMUS®-H contents is provided by the stainless steel and ductile cast iron structure of the packaging. Additional shielding for high activity payloads is provided by shielded insert assemblies made of carbon steel.

The geometric configuration of the contents during NCT will be fixed by the dunnage, shielded inserts, and Cask Containment Vessel (CCV). The applicant analyzed HAC with the source redistributed to the worst-case location inside the CCV for each detector location. Based on the review given in section 2 of this safety evaluation report the staff finds this approach to analyzing contents configuration during NCT and HAC acceptable.

5.2 Summary of Maximum External Radiation Levels

Because the package contents can be highly variable the applicant developed inventory limits for each expected radionuclide. The content limits were set so all expected contents would stay below the regulatory dose rate limits. The applicant provided two examples in the SAR using

these limits to calculate dose rates for non-exclusive use. Results for example one using ^{60}Co , a strong gamma source, were given in tables 5.1-2 and 5.1-3. A second example using ^{252}Cf , a strong neutron source, were given in tables 5.1-4 and 5.1-5. The staff finds that the maximum dose rates as summarized in table 1 are below regulatory limits.

Table 1: Summary of applicant calculated maximum dose rates for non-exclusive use

Nuclide	Condition of Transport	Detector Location	Dose Rate (mSv/hr)	Limit (mSv/hr)
^{60}Co	NCT	Package surface, side	0.73	2
^{60}Co	NCT	1-meter, side	0.09	0.1
^{60}Co	HAC	1-meter, side	0.22	10
^{252}Cf	NCT	Package surface, side	0.69	2
^{252}Cf	NCT	1-meter, side	0.09	0.1
^{252}Cf	HAC	1-meter, side	0.18	10

5.3 Radioactive Materials

Materials to be shipped inside the OPTIMUS®-H include transuranic (TRU) waste and irradiated fuel waste. Acceptable TRU waste includes that which meets the waste acceptance criteria of the Waste Isolation Pilot Plant, aerosol cans with compressed gas propellant, aerosol cans with liquified gas or unknown propellant, or Department of Transportation 3E lecture bottles. Irradiated fuel waste includes low enrichment uranium fuel waste or CANDU fuel waste and their associated hardware.

5.4 Source Term Calculation

The TRU waste contents identified by the applicant were variable so the applicant used ORIGEN in SCALE 6.2.2 to find the activity for 1 TBq of each expected radionuclide. Secondary particles were generated using a UO_2 matrix and the photon and neutron energies were grouped to ensure the inclusion of important emissions.

For CANDU fuel waste and its associated hardware the applicant used ORIGAMI in SCALE 6.2.2 to calculate the source term. The same energy grouping as TRU waste was used. The applicant used the CANDU19 library to calculate radionuclide inventory and the staff finds the use of this library acceptable because the effect on dose rate as shown in appendix A of calculation package CN-16007-503 is small compared to conservatism built into the shielding model. Parameters the applicant used for the fuel were 0.021 MTU/bundle, burnup of 5 GWd/MTU, and cooling time of 40 years.

The applicant set the mass of non-fuel hardware as 0.048 kgHW/kgU with cobalt impurities taken from DOE/RW-0184 Rev. 1. The staff finds these parameters are acceptable based on a review of the included reactor operating records.

Analysis performed by the applicant found that including subcritical multiplication in the source term resulted in a decrease in dose rate. For the reasons stated above, staff finds the method used to calculate the source term to be acceptable.

5.4.1 Isotope Activity Limits

The applicant set individual isotope activity limits which include collapsed parent and short-lived daughter decay chains.

The applicant used the dose rate contribution per nuclide activity that the applicant had previously calculated in "OPTIMUS®-H Waste Content Shielding Analysis – Single Package," 70000.14-5101 Rev. 2 and OPTIMUS®-H Waste Content Shielding Analysis – Nonexclusive Use," 70000.14-5102 Rev. 1. From this, the applicant determined activity limits for the OPTIMUS®-H using 10 CFR 71 Appendix A to collapse parent and short-lived daughter decay chains.

Staff previously evaluated the applicant's documents and found the results acceptable. Footnote a of 10 CFR 71 Table A-1 lists the short-term daughter nuclides that are included in the A_1 and/or A_2 values for the parent isotope. If an isotope is present in Table A-1, then that isotope has not been collapsed with the parent isotope in the listed A_1 and/or A_2 value.

The staff reviewed the parent and short-lived daughter decay chains from Table A-1 of 10 CFR 71 which are summarized in Table 4-1 of Supplement 3. For the activity of Cs-137, the staff noted that Table A-1 of 10 CFR 71 includes the Ba-137m daughter activity because Ba-137m has a half-life that is five orders of magnitude shorter (2.55 minutes versus 30 years) and quickly decays down to its stable ground state.

The staff also reviewed the branching fractions taken from ORNL/TM-13624 and listed in Table 4-2 of the application and noted they match.

Tables 6-1 through 6-8 of Supplement 3 show the maximum activity limits for each major isotope in a package. The applicant analyzed cases for each listed isotope individually by multiplying its branching fraction by the dose rate contribution of the parent plus the summation of the dose rate contribution of each i_{th} daughter.

This method was previously approved by staff in Revision 0 (ML20266G182) of the application, and each case is for one package containing the maximum activity of the respective isotope. As discussed above, the applicant used previously approved methods and data from regulation; therefore, the staff finds the applicant's individual isotope activity limit determination acceptable.

5.5 Shielding Model and Evaluation

The applicant evaluated the dose for each energy group of photons and neutrons using MCNP 6. The cross-section libraries used in MCNP were MCPLIB84 for photons and ENDF71x for neutrons. Staff finds these cross-section libraries acceptable since they are based on ENDF/B-7 nuclear data. The staff reviewed the material compositions and densities and found them to be acceptable.

The applicant took dimensions for the model from the engineering drawings using the tolerances that resulted in the thinnest material thicknesses. The staff finds that dimensions and tolerances from the engineering drawings were properly transferred to the MCNP model. The staff finds that an under thickness of 1/16-inch for the Outer Shield Vessel (OSV) wall and 1/4-inch for the OSV lid and bottom is acceptable. The basis for the staff's conclusion is two-fold:

- Actual contents will be distributed throughout the waste which will reduce the dose of low energy photons as shown in table B.3-3 of calculation package CN-16007-502.
- Confirmatory measurements will be performed as explained in Chapter 7 of the SAR.

The applicant did not model the attachments to the OSV (i.e., trunnions, tie-downs, turnbuckles, etc.). The applicant also did not model the foam or outer shells of the impact limiters. The staff concludes this is an acceptable simplification because the removal of materials adds additional conservatism to the model. Bolt holes and ports were not modeled individually. The staff accepts this simplification because those areas will be filled with materials with equal or better shielding performance during use.

The applicant also analyzed the design for possible streaming paths from the O-ring grooves. The applicant showed that the location of the O-ring grooves is along the radiation's path through a similar thickness of shielding material so no streaming paths are present. The staff concludes that the applicant's analysis appropriately addresses the potential for streaming paths.

The applicant also analyzed the use of shielded insert assemblies made of carbon steel. Analysis for NCT used 1-inch, 2 ¼-inch, and 3 ¾-inch inserts. Analysis during HAC included the 3 ¾-inch insert. The staff finds the analysis using the inserts during these conditions to be appropriate. The model for HAC includes a 6-inch diameter by ¼-inch deep depression on the side wall facing the tally. As discussed in section 2, the staff finds this modeling assumption acceptably represents expected damage due to HAC.

The location of the tallies used to calculate dose rate are provided in table 5.3-2 of the SAR. For models using the 3 ¾-inch shielded insert an additional surface tally was added for the flared-out region. The staff finds that these tally locations meet regulatory requirements.

The applicant modeled the radioactive source as a point with different locations based on transport conditions. During NCT the point source was centered in the CCV cavity. For HAC the source was placed inside the CCV cavity as near to the applicable detector as possible. The staff finds this method acceptable.

The applicant used the energy dependent dose responses from MCNP with the source term calculations to calculate the maximum activity for each isotope under each type of shipment. The staff finds that the method used to calculate maximum activity for each potential content is acceptable.

The applicant included uncertainty in the calculations by setting the dose rates to 90% of the regulatory limits. Uncertainty in MCNP dose rates from the shielding evaluation were covered by including two times the uncertainty in the response functions. The source was modeled as a point which is another conservatism because attenuation from the waste matrix was not included in the analysis. The staff finds that the methods used to include uncertainty are acceptable.

Flux was converted to dose using the recommended ANSI/ANS-6.1.1-1977, *Neutron and Gamma-Ray Flux-to-Dose-Rate Factors*.

5.6 Confirmatory Analyses

The staff performed confirmatory calculations using the MAVRIC module of the SCALE 6.2.4 code suite.

5.6.1 ⁶⁰Co Dose Rate Under Non-Exclusive Use

The staff created a model using a 0.188 TBq ^{60}Co source centered in the CCV cavity. The energy spectrum for the emitted photons is listed in table 2. Material compositions for the stainless steel and cast iron are the same as those listed in table 5.3-4 of the safety analysis report.

The geometry in the confirmatory model was the same as that used by the applicant except staff modeled the CCV flange as a simple cone. This simplification results in negligible change in material thickness and would result in a minor increase in dose rate. Staff used the v7.1-28n19g cross-section library and the results demonstrate that the applicant's overall method of analysis is acceptable.

Table 2: Gamma energy spectrum for 0.188 TBq ^{60}Co used in staff analysis

Upper Energy Boundary (MeV)	Source Strength (γ/s)
20	0.0000
10	0.0000
8	0.0000
6.5	0.0000
5	0.0000
4	0.0000
3	3.7600×10^3
2.5	2.2560×10^6
2.0	0.0000
1.66	1.8787×10^{11}
1.33	1.8772×10^{11}
1.0	1.4422×10^7
0.8	4.7747×10^5
0.6	1.3503×10^6
0.4	1.73117×10^7
0.3	1.9705×10^7
0.2	3.4087×10^8
0.1	1.4818×10^9
0.045	6.9242×10^9

5.7 Staff Findings

The staff has reviewed the application and finds that it adequately describes the package contents and the package design features that affect shielding in compliance with 10 CFR 71.31(a)(1), 71.33(a), and 71.33(b), and provides an evaluation of the package's shielding performance in compliance with 10 CFR 71.31(a)(2), 71.31(b), 71.35(a), and 71.41(a). The descriptions of the packaging and the contents are adequate to allow for evaluation of the package's shielding performance. The evaluation is appropriate and bounding for the packaging and the package contents as described in the application.

The staff has reviewed the application and finds that it demonstrates the package has been designed so that under the evaluations specified in 10 CFR 71.71 (normal conditions of transport), and in compliance with 10 CFR 71.43(f) and 10 CFR 71.51(a)(1), the external radiation levels do not significantly increase.

The staff has reviewed the application and finds that it demonstrates that under the evaluations specified in 10 CFR 71.71 (normal conditions of transport), external radiation levels do not

exceed the limits in 10 CFR 71.47(a) for nonexclusive use shipments or 10 CFR 71.47(b) for exclusive use shipments, as applicable.

The staff has reviewed the application and finds that it demonstrates that under the tests specified in 10 CFR 71.73 (hypothetical accident conditions), external radiation levels do not exceed the limits in 10 CFR 71.51(a)(2).

The staff has reviewed the application and finds that it identifies codes and standards used in the package's shielding design and in the shielding analyses, in compliance with 10 CFR 71.31(c).

6.0 CRITICALITY EVALUATION

The objective of this evaluation is to verify that the transportation package design meets the criticality safety requirements of 10 CFR 71 under the conditions described in 10 CFR 71.71 and 71.73. The NRC staff's evaluation follows the guidance of NUREG-2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material" (SRP).

6.1 Description of Criticality Design

6.1.1 Packaging Design Features

The OPTIMUS®-H consists of a cask containment vessel (CCV) within a ductile cast iron outer shield vessel (OSV). Two impact limiters (ILs) are affixed to the top and bottom of the OSV. The CCV body consists of a stainless-steel base plate, shell, and flange which are welded to form the side wall and bottom of the vessel. A stainless-steel lid is bolted to the flange with an O-ring seal. The CCV with the O-ring seal comprise the containment boundary of the package. The OSV consists of a monolithic, ductile cast iron body that comprises the bottom and sides of the cylindrical vessel. A ductile cast iron lid is bolted to the top of the body. The applicant designed the package to not rely on neutron absorbing materials in the packaging nor any geometry controls to ensure subcriticality. Rather, the contents are limited to remain subcritical in the most adverse configuration. The applicant does rely on minimum spacing provided by the impact limiters for criticality safety.

6.1.2 Codes and Standards

The applicant limited the contents on two bases: fissile gram equivalent (FGE) of ^{239}Pu ; and ^{235}U fissile equivalent mass (FEM). The applicant used ANSI/ANS 8.1, 8.12, and 8.15 for its FGE conversion factors. The applicant's FEM enrichment limits are based on ANSI/ANS 8.1. ANSI/ANS 8.12 and 8.15 are endorsed by NRC per RG 3.71. NRC endorsed ANSI/ANS 8.1 with the $\text{Pu}(\text{NO}_3)_4$ exception described in RG 3.71.

6.1.3 Summary Table of Criticality Evaluations

The applicant presented the most limiting results of its criticality analyses in Tables 6.1-1 and 6.1-2 of the application. The staff reviewed the results and noted all are less than the upper subcritical limit (USL) with all biases and uncertainties applied.

6.1.4 Criticality Safety Index (CSI)

The applicant demonstrated that an infinite array of optimally moderated packages, both undamaged and damaged, will remain subcritical. As a result, the package has a CSI of 0.0 for the most limiting configuration according to 10 CFR 71.59(b).

6.2 Nuclear Contents

Authorized contents of the package include up to 500 lbs. of contact handled (CH) and remote handled (RH) TRU waste with up to 1000 lbs. of steel shoring. In addition, a maximum of 7300 lbs. of irradiated CANDU fuel assemblies in 2 CANDU fuel baskets may be shipped in the OPTIMUS®-H. FEM contents have a maximum enrichment of 0.96 wt. % ^{235}U with particle size limits described in Table 6.2-2 of the application. FEM contents have a reduced maximum enrichment of 0.8 wt. % ^{235}U with unrestricted particle size. The applicant showed the FGE mass limits in Table 6.2-1 of the application.

Moderating and reflecting materials, which includes special reflector material (i.e., beryllium) are also limited. Specific moderator and reflector material limits for each of the waste types are given in Tables 6.2-3 and 6.2-4 of the application for FGE and FEM limited contents, respectively.

Only contents meeting the criteria in Table 6.2-3 for FGE-5-type waste may be machine compacted. All other contents must be manually compacted.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The applicant listed the dimensions used in its model in Table 6.3-1 of the application with an accompanying elevation view of the package in Figure 6.3-1 of the application. The applicant did not model tolerances for the CCV body and lid. Since the standards for ASME plate and forging components have tight, standardized tolerances (0.01 inches), staff finds that this will not significantly impact the criticality safety of the package. The applicant did not model the bolts as separate entities and omitted them from the CCV lid and body. Staff finds this acceptable since bolts largely fill the volume of the bolt hole, and any neutron moderation due to flooding will be negligible due to the small volume involved.

The applicant modeled the OSV as a solid piece of ductile cast iron. The applicant did not model external features and attachments (e.g., trunnions, brackets, etc.). Since these features are external to the package, they will likely have negligible impact on the calculated system k_{eff} . As with the CCV, the applicant did not model the bolts and holes in the OSV model either. Staff finds these omissions acceptable since they have negligible impact on criticality safety. The applicant assumed a 1/8-inch tolerance with the inner and outer dimensions of the OSV sides, lid, and base thickness. Staff finds this acceptable since it minimizes the displacement of moderating material which conservatively increases calculated reactivity.

The applicant modeled the part of the 0.5-inch stainless-steel shell of the IL that interfaces with the OSV. The applicant omitted the rest of the IL material. Staff finds the IL material omission acceptable since this will increase the neutron reflection and increase calculated reactivity. While the applicant omitted the IL material, the spacing provided by the IL was included in determining the pitch of an array of packages. For an array of packages under HAC, the applicant assumed the IL provided 35 % and 29 % of the original spacing at the top and sides of the package, respectively.

The applicant provided an analysis of the maximum displacement in Chapter 2 of the application, which staff found acceptable in Section 2 of this SER. The applicant conservatively assumed a greater maximum displacement than the analysis showed, which reduces spacing between packages and increases calculated reactivity. As a result, staff finds the applicant's modeling of the ILs acceptable.

6.3.2 Material Properties

The packaging consists largely of two materials, ductile cast iron and stainless steel. Specifically, the OSV body and lid are comprised of ductile cast iron. The standard provides a maximum value for some elements and a range for others. The applicant assumed the maximum value except for carbon and silicon. There is an additional requirement the carbon and silicon contents must satisfy the equation, $C + 1/3 Si \leq 4.5 \%$. The applicant satisfied this criterion by assuming the carbon and silicon components to be in the middle of their allowed ranges. This method was recommended to provide a degree of standardization in nuclear analyses (2) and has been shown to yield reliable, repeatable results. The isotopic composition of stainless steel and ductile cast iron used in the criticality evaluation is shown in Tables 6.3-2 and 6.3-3 of the application, respectively. Staff reviewed the composition the applicant used for ductile cast iron and stainless steel and noted they follow those recommendations in Reference 2; therefore, staff finds them acceptable.

For the contents, the applicant evaluated the presence of light water, polyethylene, beryllium, plutonium, and uranium. The applicant varied the composition of moderating and reflecting materials to establish FGE or FEM limits for each case. Prior evaluations have shown polyethylene to be the bounding hydrogenous moderating material that could credibly moderate transuranic waste in pure form (3). A 25/75 % polyethylene/water mixture has previously been found as the basis for determining "special reflector" material (4). Considering physical testing has shown the manually compacted polyethylene packing fraction to be 13.36 % (5), staff finds the applicant's assumption of 15 % polyethylene appropriate moderator and reflector composition for manually compacted contents. With the same considerations, staff also finds the applicant's assumption of 100 % polyethylene for bounding machine compacted moderator and reflector material appropriate. Since beryllium has been determined to be the bounding special reflector material (3), staff finds the applicant's modeling assumption that any special reflector material consists entirely of beryllium appropriate. The isotopic compositions of these materials are presented in Table 6.3-4 of the application.

6.3.3 Analysis Methods and Nuclear Data

For all of its criticality analyses, the applicant used MCNP6. The applicant used continuous energy cross-section libraries based on ENDF/B-VII nuclear data. The applicant's thermal scattering kernels (i.e., $S(\alpha, \beta)$) are based on data from the ENDF/B-VII.0 library. MCNP is a three-dimensional, Monte Carlo particle transport code that is capable of modeling the OPTIMUS-H transportation package. MCNP, and the cross-section and thermal scattering libraries used by the applicant are all well vetted with a long history of use in criticality safety applications. For these reasons, staff finds the applicant's choice of software and nuclear data appropriate.

The applicant included $S(\alpha, \beta)$ for light water, polyethylene, and beryllium. These materials comprise the moderator and reflector material that may be present inside the CCV cavity. Since these materials have such a large effect on reactivity, staff finds the inclusion of these scattering kernels will appropriately model thermal scattering in moderator and reflector materials. The

applicant also included $S(\alpha, \beta)$ for ^{56}Fe . The vast majority (>90 %) of naturally occurring iron is ^{56}Fe , and iron comprises more than 70 % and 96 % of SS-304 and ductile cast iron, respectively. As a result, staff finds the applicant's inclusion of this scattering kernel will appropriately model thermal scattering in the structural components.

6.3.4 Demonstration of Maximum Reactivity

The applicant determined the bounding condition to be an infinite array of packages under HAC. As a result, this Section discusses the methodology the applicant used to determine the most reactive configuration for an array of packages under HAC.

6.3.4.1 FGE Contents

For FGE cases, the applicant initially determined the optimum $\text{H}/^{239}\text{Pu}$ ratio with the maximum mass for each FGE contents type that still results in a calculated reactivity plus two standard deviations (i.e., $k_{\text{eff}} + 2\sigma$) that remains below the USL. The applicant varied the $\text{H}/^{239}\text{Pu}$ ratio by changing the fissile volume and holding the ^{239}Pu mass constant. The applicant then used the fissile volume with the maximum k_{eff} in subsequent evaluations. For all FGE cases, the applicant modeled the fissile mass as a homogenized sphere of plutonium and moderator, with the remainder of the CCV filled with a reflector material. Realistically, the fissile and moderator material will be more generally interspersed throughout the CCV cavity and staff finds the uniformity of the moderator and reflector compositions appropriate.

6.3.4.1.1 FGE-1

The applicant's process to determine the most reactive configuration of FGE-1 contents in the package is given in Section 6.5.2.5 of Reference 6. The applicant selected several different FGE mass and varied the fissile sphere volume to generate a series of results to determine the optimum $\text{H}/^{239}\text{Pu}$ ratio that yielded the highest k_{eff} . The applicant then determined the maximum FGE mass at optimum moderation that stayed below the USL. The applicant carried this limiting mass and fissile volume forward through additional sensitivity studies of the FGE-1 contents. Staff reviewed the results of these evaluations and finds the applicant considered a mass and volume range appropriately to determine a maximally reactive base configuration.

The applicant evaluated the effect of shifting the position of the fissile sphere. The applicant evaluated the following positions: the center of the CCV cavity; the top center of the CCV cavity; and the top corner (i.e., adjacent to both the CCV lid and side) of the CCV cavity. The applicant assumed the proximity of denser structural components would enhance neutron reflection and considered the top corner configuration as the baseline configuration. Staff finds the applicant's results confirm that the enhanced reflection of the stainless steel and ductile cast iron components bounds the centrally located proximity to polyethylene/water/beryllium reflector material.

The applicant evaluated the effect of flooding in the void space within the OSV cavity and external to the OPTIMUS-H package. The applicant evaluated the following scenarios: flooding only within the OSV cavity; flooding only external to the package; flooding in both the OSV cavity and external to the package. In addition, the applicant varied the volume fraction of the water in each of the scenarios from 0.0001 to 1. The applicant's results showed that maximum reactivity occurs with only OSV flooding at a water volume fraction of 0.001. Staff finds the applicant considered a wide enough range of scenarios and volume fractions appropriately to determine the most reactive flooded configuration.

Since the inclusion of water within the OSV could alter the optimum $H/^{239}\text{Pu}$ ratio, the applicant repeated the baseline configuration with water in the OSV void space at a 0.001 volume fraction. The applicant's results showed no statistically significant change in optimum moderator/fuel ratio from the existing baseline configuration, and that the baseline k_{eff} was bounding.

To account for special reflector material within the contents, the applicant evaluated the effect the presence of beryllium could have on reactivity. The applicant evaluated the following scenarios: 1 % beryllium in both the moderator and reflector; 1 % beryllium in the reflector only; and no beryllium present. The applicant's results showed that the presence of beryllium in both the moderator and reflector yields the maximum calculated k_{eff} . In order to comply with FGE-1 waste limits, beryllium content may not exceed 1 wt. %. Therefore, staff finds the applicant appropriately determined the bounding configuration with maximum special reflector material allowed for FGE-1 contents.

As discussed above, the applicant determined the optimum moderation for the maximum mass allowed in the most reactive configuration with bounding moderator, reflector, and special reflector material compositions. As a result, staff finds reasonable assurance that the applicant determined the most reactive configuration for FGE-1 contents.

6.3.4.1.2 FGE-2a, FGE-2b, and FGE-2c

The applicant's process to determine the most reactive configuration of FGE-2 contents in the OPTIMUS-H package is given in Sections 6.5.2.6 through 6.5.2.8 of Reference 6. For FGE-2 contents, the applicant credits the presence of ^{240}Pu in the contents. The applicant assumed 5, 10, and 15 grams of ^{240}Pu present in FGE-2a, FGE-2b, and FGE-2c waste types, respectively. Except for the addition of ^{240}Pu , the applicant's baseline configuration and material compositions are identical to that for FGE-1 contents. The applicant selected several different FGE mass and varied the fissile sphere volume to generate a series of results to determine the optimum $H/^{239}\text{Pu}$ ratio that yielded the highest k_{eff} . The applicant then determined the maximum FGE mass at optimum moderation that stayed below the USL. The applicant carried this limiting mass and fissile volume forward through additional sensitivity studies of the FGE-2 contents. Staff reviewed the results of these evaluations and finds the applicant considered a mass and volume range appropriately to determine a maximally reactive base configuration.

The applicant evaluated the effect of flooding in the void space within the OSV cavity and external to the OPTIMUS-H package. The applicant evaluated the following scenarios: flooding only within the OSV cavity; flooding only external to the package; flooding in both the OSV cavity and external to the package. In addition, the applicant varied the volume fraction of the water in each of the scenarios from 0.0001 to 1. For FGE-2a contents, the applicant's results showed that maximum reactivity occurs with no OSV flooding. For FGE-2b contents, the applicant's results showed that maximum reactivity occurs with only OSV flooding at a water volume fraction of 0.0001. For FGE-2c contents, the applicant's results showed that maximum reactivity occurs with only OSV flooding at a water volume fraction of 0.001. Staff finds the applicant considered a wide enough range of scenarios and volume fractions appropriately to determine the most reactive flooded configuration.

Since the inclusion of water within the OSV could alter the optimum $H/^{239}\text{Pu}$ ratio for FGE-2b and FGE-2c waste types, the applicant repeated the baseline configuration with water in the OSV void space at the most reactive volume fraction. The applicant's results showed no

statistically significant change in optimum moderator/fuel ratio from the existing baseline configuration for both FGE-2b and FGE-2c, and the baseline k_{eff} was bounding. Since the material properties of FGE-2 contents are nearly identical to those of FGE-1, the results of the beryllium evaluations for FGE-1 will also apply. Therefore, staff finds it acceptable that the applicant did not repeat the beryllium study for FGE-2 contents. As discussed above, the applicant determined the optimum moderation for the maximum mass allowed in the most reactive configuration with bounding moderator, reflector, and special reflector material compositions. As a result, staff finds reasonable assurance that the applicant determined the most reactive configuration for FGE-2a, FGE-2b, and FGE-2c contents.

6.3.4.1.3 FGE-3

The applicant's process to determine the most reactive configuration of FGE-3 contents in the OPTIMUS-H package is given in Sections 6.5.2.10 of Reference 6. For FGE-3 contents, the applicant evaluates the presence of special reflector material that is neither chemically nor mechanically bound to the waste. In a similar case that modeled the moderator and reflector material with 25 % polyethylene, the applicant varied the beryllium volume fraction from 1 % to 40 %. The remaining polyethylene/water volume fraction was kept at the same 15:84 ratio used for FGE-1 and FGE-2 contents. The applicant's results showed maximum reactivity occurs at with a 1 % beryllium volume fraction. Staff reviewed the applicants results and finds the applicant varied the beryllium fraction sufficiently to determine the maximum reactivity. The material composition of the FGE-3 contents is not significantly different from the other scenario and staff finds the results of the beryllium volume fraction study applicable. As a result, staff finds the applicant's determination of 1 % beryllium as the baseline configuration acceptable.

Except for the beryllium content, the applicant's baseline configuration and material compositions are identical to that for FGE-1 contents. Using its baseline FGE-3 moderator composition, the applicant selected several different FGE mass and varied the fissile sphere volume to generate a series of results to determine the optimum H/ ^{239}Pu ratio that yielded the highest k_{eff} . The applicant then determined the maximum FGE mass at optimum moderation that stayed below the USL. The applicant carried this limiting mass and fissile volume forward through additional sensitivity studies of the FGE-3 contents. Staff reviewed the results of these evaluations and finds the applicant considered a mass and volume range appropriately to determine a maximally reactive baseline configuration.

The applicant evaluated the effect of shifting the position of the fissile sphere. The applicant evaluated the following positions: the center of the CCV cavity; the top center of the CCV cavity; and the top corner (i.e., adjacent to both the CCV lid and side) of the CCV cavity. The applicant's results showed the center of the CCV cavity to be the most reactive location. Staff finds the applicant's results confirm that beryllium is a more effective reflector than the structural components of the packaging.

The applicant evaluated the effect of flooding in the void space within the OSV cavity and external to the package. The applicant evaluated the following scenarios: flooding only within the OSV cavity; flooding only external to the package; flooding in both the OSV cavity and external to the package. In addition, the applicant varied the volume fraction of the water in each of the scenarios from 0.0001 to 1. The applicant's results showed that maximum reactivity occurs with only OSV flooding at a water volume fraction of 0.01. Staff finds the applicant considered a wide enough range of scenarios and volume fractions to determine the most reactive flooded configuration. Since the inclusion of water within the OSV could alter the

optimum $H/^{239}\text{Pu}$ ratio, the applicant repeated the baseline configuration with water in the OSV void space at a 0.01 volume fraction.

The applicant's results showed a statistically significant change in maximum k_{eff} at the moderator/fuel ratio from the existing baseline configuration. As a result, the applicant carried this flooded configuration forward as the new baseline.

The applicant also evaluated the effect density of the beryllium reflector could have on reactivity. The applicant varied the volume fraction of the reflector from 0.1 to 1. The applicant's results showed that a full-density beryllium reflector yields the maximum calculated k_{eff} . Considering the beryllium content is likely to be less than modeled, staff finds the applicant appropriately determined the bounding configuration with maximum special reflector material allowed for FGE-3 contents.

As discussed above, the applicant determined the optimum moderation for the maximum mass allowed in the most reactive configuration with bounding moderator, reflector, and special reflector material compositions. As a result, staff finds reasonable assurance that the applicant determined the most reactive configuration for FGE-3 contents.

6.3.4.1.4 FGE-4

The applicant's process to determine the most reactive configuration of FGE-4 contents in the OPTIMUS-H package is given in Sections 6.5.2.11 of Reference 6. For FGE-4 contents, the applicant evaluates the presence of special reflector material that is chemically or mechanically bound to the waste. The applicant assumed the reflector is the same 15/84/1 polyethylene/water/beryllium composition as that for FGE-1. Since the special reflector material is chemically or mechanically bound to the fissile material, it is not likely to be present outside the fissile mass. Therefore, staff finds this assumption acceptable. The applicant varied the beryllium volume fraction in the moderator from 1 % to 80 %. The remaining polyethylene/water volume fraction was kept at the same 15:84 ratio used for FGE-1 and FGE-2 contents.

The applicant's results showed maximum reactivity occurs at with a 40 % beryllium volume fraction. FGE-4 waste does not have an upper beryllium limit. Staff reviewed the applicants results and finds range of the beryllium content evaluated was sufficiently large to determine the bounding beryllium content that yields maximum reactivity. As a result, staff finds the applicant's determination of 40 % beryllium as the baseline configuration acceptable. Using its baseline FGE-4 moderator composition, the applicant selected several different FGE mass and varied the fissile sphere volume to generate a series of results to determine the optimum $H/^{239}\text{Pu}$ ratio that yielded the highest k_{eff} . The applicant then determined the maximum FGE mass at optimum moderation that stayed below the USL. The applicant carried this limiting mass and fissile volume forward through additional sensitivity studies of the FGE-4 contents. Staff reviewed the results of these evaluations and finds the applicant considered a mass and volume range appropriately to determine a maximally reactive baseline configuration.

The applicant evaluated the effect of shifting the position of the fissile sphere. The applicant evaluated the following positions: the center of the CCV cavity; the top center of the CCV cavity; and the top corner (i.e., adjacent to both the CCV lid and side) of the CCV cavity. The applicant assumed the proximity of denser structural components would enhance neutron reflection and considered the top corner configuration as the baseline configuration. Staff finds the applicant's results confirm that the enhanced reflection of the stainless steel and ductile cast iron

components bounds the centrally located proximity to polyethylene/water/beryllium reflector material.

The applicant evaluated the effect of flooding in the void space within the OSV cavity and external to the OPTIMUS®-H package. The applicant evaluated the following scenarios: flooding only within the OSV cavity; flooding only external to the package; flooding in both the OSV cavity and external to the package. In addition, the applicant varied the volume fraction of the water in each of the scenarios from 0.0001 to 1. The applicant's results showed that maximum reactivity occurs with only OSV flooding at a water volume fraction of 0.001. Staff finds the applicant considered a wide enough range of scenarios and volume fractions to appropriately determine the most reactive flooded configuration.

Since the inclusion of water within the OSV could alter the optimum $H/^{239}\text{Pu}$ ratio, the applicant repeated the baseline configuration with water in the OSV void space at a 0.001 volume fraction. The applicant's results showed a statistically significant change in maximum k_{eff} at the moderator/fuel ratio from the existing baseline configuration. As a result, the applicant carried this flooded configuration forward as the new baseline.

As discussed above, the applicant determined the optimum moderation for the maximum mass allowed in the most reactive configuration with bounding moderator, reflector, and special reflector material compositions. As a result, staff finds reasonable assurance that the applicant determined the most reactive configuration for FGE-4 contents.

6.3.4.1.5 FGE-5

The applicant's process to determine the most reactive configuration of FGE-5 contents in the OPTIMUS®-H package is given in Section 6.5.2.13 of Reference 6. The applicant's FGE-5 model is designed to approximate machine compacted waste. The applicant modeled the moderator as 100 % polyethylene and the reflector as 99 % polyethylene with 1 % beryllium. Polyethylene has been determined to be the bounding hydrogenous moderating material that could credibly moderate transuranic waste in pure form (3), and to comply with FGE-5 waste limits, the beryllium content may not exceed 1 wt. %. Therefore, staff finds the applicant's moderator and reflector composition acceptable. The applicant selected several different FGE masses and varied the fissile sphere volume to generate a series of results to determine the optimum $H/^{239}\text{Pu}$ ratio that yielded the highest k_{eff} . The applicant then determined the maximum FGE mass at optimum moderation that stayed below the USL. The applicant carried this limiting mass and fissile volume forward through additional sensitivity studies of the FGE-5 contents.

Staff reviewed the results of these evaluations and finds the applicant considered a mass and volume range appropriately to determine a maximally reactive base configuration. The applicant evaluated the effect of shifting the position of the fissile sphere. The applicant evaluated the following positions: the center of the CCV cavity; the top center of the CCV cavity; and the top corner (i.e., adjacent to both the CCV lid and side) of the CCV cavity. The applicant assumed the proximity of denser structural components would enhance neutron reflection and considered the top corner configuration as the baseline configuration. Staff finds the applicant's results confirm that the enhanced reflection of the stainless steel and ductile cast iron components bounds the centrally located proximity to polyethylene/beryllium reflector material.

The applicant evaluated the effect of flooding in the void space within the OSV cavity and external to the package. The applicant evaluated the following scenarios: flooding only within the OSV cavity; flooding only external to the package; flooding in both the OSV cavity and

external to the package. In addition, the applicant varied the volume fraction of the water in each of the scenarios from 0.0001 to 1. The applicant's initial flooding evaluations showed an increase in reactivity above the USL, so the applicant repeated the flooded evaluations with a reduced plutonium mass. The applicant's results showed that maximum reactivity occurs with flooding both internally within the OSV cavity and externally at a water volume fraction of 0.01. Staff finds the applicant considered a wide enough range of scenarios and volume fractions to determine the most reactive flooded configuration.

Since the inclusion of water within the OSV could alter the optimum H/²³⁹Pu ratio or yield a higher k_{eff} , the applicant repeated the baseline configuration with water in the OSV void space at a 0.001 volume fraction. The applicant's results showed no statistically significant change in optimum moderator/fuel ratio from the existing baseline configuration and the reactivity of the existing baseline configuration remained bounding.

The applicant varied the reflector volume fraction for FGE-5 to determine the bounding reflector density. The applicant varied the reflector volume fraction from 0.1 to 1. The applicant's results show that a full-density polyethylene reflector is the bounding, most reactive configuration. As discussed above, the applicant determined the optimum moderation for the maximum mass allowed in the most reactive configuration with bounding moderator, reflector, and special reflector material compositions. As a result, staff finds reasonable assurance that the applicant determined the most reactive configuration for FGE-5 contents.

6.3.4.2 FEM Contents

The applicant's process to determine the most reactive configuration of FEM contents in the package is given in Section 6.5.4.2 of Reference 6. The applicant evaluated heterogeneous spheres and cylinders of uranium metal with the moderator and reflector comprised of 15/84/1 % polyethylene/water/beryllium. As discussed above, staff finds this moderator and reflector composition acceptable. The applicant evaluated two ²³⁵U enrichments, 0.8 wt.% and 0.96 wt.%, in both a square and hexagonal array of spheres and cylinders of varying size and pitch. The applicant varied the sphere and cylinder outer radius from 0.05 cm to 7.00 cm. The applicant varied the half-pitch of the arrays from a minimum value equal to the particle outer radius (i.e., close packed) to 8.5 cm. The applicant modeled the fissile array of cylinders at the length of the CCV cavity and varied the radius to change the pitch. The applicant modeled the array of spheres at the CCV cavity radius and varied the cylinder height to change the pitch. The applicant effectively varied the H/²³⁵U ratio by changing the pitch of the fissile array. The applicant selected the parameters that yielded the maximum calculated k_{eff} to be its baseline configuration.

The applicant also performed an additional homogeneous evaluation on FEM contents in the package. The applicant varied the uranium mass at 0.96 wt.% enrichment to determine the most reactive mass and H/²³⁵U ratio. The results of this evaluation are shown in Table 6.4-11 of the application. The optimum H/²³⁵U ratio and maximum reactivity occurs at a loading with less than the maximum allowed uranium content. Staff finds the applicant varied the mass over a large enough range to determine maximum k_{eff} .

6.3.4.2.1 FEM-1

Summaries of the maximum for each sphere and cylinder particle result for 0.96 wt.% enriched uranium (i.e., FEM-1 contents) are shown in Tables 6.5.4-12 and 6.5.4-13 of Reference 6, respectively. The results are presented graphically in Figures 6.5.4-4 and 6.5.4-5 of Reference 6

for spheres and cylinders, respectively. Staff reviewed the applicant's results and finds the applicant selected a range of particle outer radii sufficient to determine a maximum reactivity based on particle size. Staff also finds the applicant selected a range of half-pitches sufficient to determine the most reactive pitch at a given particle size. The applicant's results showed that only arrays of spheres and cylinders with outer radii of 0.1, 0.25, 0.5, 1.0, 2.0, and 3.0 cm had calculated maximum k_{eff} values that exceeded the USL for the OPTIMUS-H package. As a result, staff finds the size limits of ≤ 0.05 cm and ≥ 4.0 cm appropriate for FEM-1 contents.

The applicant evaluated the effect of flooding on the bounding, most reactive case that remained below the USL. The applicant evaluated the following scenarios: flooding only within the OSV cavity; flooding only external to the package; flooding in both the OSV cavity and external to the package. In addition, the applicant varied the volume fraction of the water in each of the scenarios from 0.0001 to 1. Staff finds the applicant considered a wide enough range of scenarios and volume fractions to determine the most reactive flooded configuration. The applicant's results showed a slight increase in k_{eff} with flooding both inside and external to the OSV at a volume fraction of 0.0001. However, this result was within 2σ of the baseline configuration does not differ in a statistically significant way. As a result, staff finds the applicant's conclusion that flooding has a negligible effect on maximum reactivity acceptable.

As discussed above, the applicant determined the optimum moderation limited to the maximum mass allowed in the most reactive configuration. As a result, staff finds reasonable assurance that the applicant determined the most reactive configuration within the FEM-1 contents size limits.

6.3.4.2.2 FEM-2

Summaries of the maximum for each sphere and cylinder particle result for 0.8 wt.% enriched uranium (i.e., FEM-2 contents) are shown in Tables 6.5.4-15 and 6.5.4-16 of Reference 6, respectively. The results are presented graphically in Figures 6.5.4-6 of Reference 6 for cylinders. Staff reviewed the applicant's results and finds the applicant selected a range of particle outer radii sufficient to determine a maximum reactivity based on particle size. Staff also finds the applicant selected a range of half-pitches sufficient to determine the most reactive pitch at a given particle size. The applicant's results showed the calculated maximum k_{eff} values remained below the USL for the package at all half-pitches evaluated. As a result, staff finds the unrestricted size limit condition for FEM-2 contents appropriate.

The applicant evaluated the effect of flooding on the bounding, most reactive case that remained below the USL. The applicant evaluated the following scenarios: flooding only within the OSV cavity; flooding only external to the package; flooding in both the OSV cavity and external to the package. In addition, the applicant varied the volume fraction of the water in each of the scenarios from 0.0001 to 1. Staff finds the applicant considered a wide enough range of scenarios and volume fractions to determine the most reactive flooded configuration. The applicant's results showed the unflooded scenario to be the most reactive. As a result, staff finds the applicant's use of the baseline configuration as bounding acceptable. As discussed above, the applicant determined the optimum moderation limited to the maximum mass allowed in the most reactive configuration. As a result, staff finds reasonable assurance that the applicant determined the most reactive configuration with FEM-2 contents.

For FEM cases, the uranium mass is limited to 5000 lbs. The enrichment limits depend on whether debris size limits apply. In cases with unlimited debris size, the maximum ^{235}U enrichment is 0.80 wt.%. When size limits are met, the maximum ^{235}U enrichment is 0.96 wt.%.

6.3.5 Moderator Exclusion Under HAC

The applicant assumed the CCV volume was filled with various moderating and reflecting material, including water, and is not relying on moderator exclusion to maintain subcriticality.

6.4 Single Package Evaluations

6.4.1 Configuration

The applicant modeled the OSV and CCV identically under NCT and HAC. In Section 2 of this SER, staff found reasonable assurance that the applicant's analysis showed that no damage will occur to the OSV and CCV due to the conditions described in 10 CFR 71.71 and 71.73; therefore, staff finds the applicant's use of a single model to cover all conditions acceptable. The applicant modeled a single package, described in Section 6.3.1 above, surrounded by 20 inches of water. Prior staff review has found 20 inches of water satisfies the full reflection requirements of 71.55(b)(3) and 71.55(e)(3). The applicant omitted the IL from the single package model. Staff finds this acceptable since this will increase calculated reactivity due to the decreased distance between the contents and the water reflector external to the package.

6.4.2 Results

The applicant followed the same method described in Section 6.3.4 of this SER. The results of the single package evaluations are given in Tables 6.4-1 through 6.4-11 of the application. Since the single package evaluation does not involve potential neutron interactions with fissile masses in adjacent packages, some of the most reactive flooded conditions changed. This is due to the increased moderation provided by close flooding that does not also reduce neutron interaction between packages.

Except for FGE-2c contents, package flooding did not significantly change the optimum moderator/fuel ratios for FGE contents. Even with a changed $H/^{239}\text{Pu}$ ratio for FGE-2c, the calculated k_{eff} for none of the maximum single package evaluations exceeded that of the HAC array with identical FGE contents. As a result, the HAC evaluations are bounding for FGE contents.

For FEM contents, the most reactive single package did not change the optimum $H/^{235}\text{U}$ ratio. However, for FEM-1 contents, the calculated k_{eff} for a single package with spherical particles was higher than the most reactive FEM-1 HAC array. This is due to a more reactive smaller particle size which yielded a k_{eff} below the USL, and the applicant considered it the bounding case.

In all cases, the maximum k_{eff} values remained below the USL. Considering the applicant used the same iterative process as with HAC array, staff finds reasonable assurance that the applicant determined the most reactive single package configuration.

6.5 Evaluations of Package Arrays

6.5.1 Package Arrays under NCT

The applicant determined that no credible deformation of the IL would occur under NCT. The IL deformation modeled under HAC would allow closer package spacing and would therefore be bounding (i.e., more reactive) of an array of packages under NCT. As a result, staff finds it

acceptable that the applicant did not perform an additional evaluation of package arrays under NCT.

6.5.2 Package Arrays under HAC

Each package is modeled within a hexagonal prism with reflective boundaries on all eight faces. This effectively creates an infinite, triangular-pitch array in all directions. As discussed above, the applicant credits 29 % and 35 % for side and top spacing provided by ILs, respectively. For FGE cases where the top corner location was most reactive, the applicant moved the fissile mass to the corner that effectively placed it closest to the adjacent packages. Since the applicant used a mirror boundary condition on the axial faces, this effectively modeled alternate planes of the infinite array as inverted. This means the applicant essentially modeled the array with packages lid-to-lid, which conservatively placed the fissile masses in each layer closer together. Since OPTIMU-H packages are unlikely to be inverted on top of other packages, staff finds this approximation conservatively increased calculated k_{eff} .

Spherical particles and homogenous FEM cases don't have radial displacement in the CCV cavity due to the assumed mass filling the entire cross-sectional area of the CCV cavity. As discussed in Section 6.3.4 of this SER, the applicant varied the volume fraction of water between the packages to determine the most reactive array configuration. Staff finds this acceptable since this will maximize calculated k_{eff} and bound any other condition, flooded or otherwise.

6.5.3 Package Arrays Results and CSI

The results of the applicant's evaluations are presented in Tables 6.6-1 through 6.6-16 of the application. Staff finds the applicant provided reasonable assurance that an infinite number of packages under HAC will remain subcritical; therefore, the CSI of the Optimus-H package is 0.0 per 10 CFR 71.59(b).

6.6 Benchmark Evaluations

The applicant selected all of its benchmark experiments to determine bias and USL of FGE and FEM evaluations from Reference 7. This follows guidance in the SRP and staff finds it appropriate.

6.6.1 Experiments and Applicability

6.6.1.1 Plutonium Experiments

The benchmarks the applicant selected are shown in Table 6.8-3 of the application. The FGE waste types permitted in the OPTIMUS®-H effectively consist of plutonium metal in solution; the benchmark cases are all plutonium compounds in solution. The chemical form of the fissile nuclides does not change the fission cross-section, and staff finds the form of the plutonium in the benchmarks applicable. All of the selected benchmarks are thermal systems moderated by water. With two exceptions, the sets of benchmarks are also water reflected. One set of benchmarks includes a graphite (carbon) and beryllium oxide, and another has no reflector material (i.e., "bare"). Since the applicant evaluated the presence of carbon-containing material (polyethylene) and beryllium, and the CoC allows for the inclusion of such materials, staff finds the moderating and reflecting materials of the applicants selected benchmarks appropriate for its evaluations of the OPTIMUS®-H. For the fissile nuclides, the applicant assumed the package contents consisted of 95-100 wt. % ^{239}Pu , with ^{240}Pu comprising the remainder.

These benchmarks all include ^{240}Pu . While some of the applicant's evaluations included ^{240}Pu , all of the OPTIMUS[®]-H FGE contents will likely contain some amount of ^{240}Pu regardless. Since ^{240}Pu is a neutron absorber, this will provide some degree of conservative margin to the applicant's calculations. Staff noted that some of the benchmarks also include ^{238}Pu , ^{241}Pu , and ^{242}Pu . These isotopes occur in trace amounts and staff finds they are likely to affect the bias and USL insignificantly.

6.6.1.2 Low-Enriched Uranium Experiments

The benchmarks the applicant selected are shown in Table 6.8-4 of the application. All of the low-enriched uranium benchmarks contain small amounts of ^{234}U and ^{236}U , while the applicant's evaluations assumed only ^{235}U and ^{238}U were present. In actuality, the OPTIMUS[®]-H contents will include small amounts of ^{234}U and ^{236}U . Considering the low maximum enrichment (<1 wt.% ^{235}U), these two isotopes are unlikely to occur in greater than trace amounts. As a result, their presence would have negligible effects on bias and USL calculations, and staff finds the applicant's evaluations acceptable with only ^{235}U and ^{238}U . Most of the cases the applicant selected are water-moderated uranium solution experiments that are either bare or water reflected. The chemical form of uranium has no effect on its nuclear properties.

The applicant modeled FEM moderator and reflector materials as largely hydrogenous material (i.e., water) which also comprised the moderator and reflector in the benchmark experiment. As a result, staff finds the inclusion of those experimental benchmarks appropriate. Another set of benchmarks used uranium metal tubes with water moderation and reflection. While the size of the tubes is larger than the debris the applicant evaluated, the benchmark experiment captures the effects of heterogeneity on the system. As a result, staff finds these included benchmarks acceptable. One more set of benchmarks consisted of UF_4 and paraffin mixtures in single, large cubes, with either polyethylene, paraffin, or Plexiglas reflector, or bare. The benchmark report for this set concluded there was no significant change in bias between modeling these benchmarks as homogenous rather than realistically as small UF_4 particles suspended in paraffin. In terms of reflector and moderator material, this set of benchmark cases is most similar to the authorized FEM contents of the package. For these reasons, staff finds the inclusion of this benchmark set appropriate.

6.6.2 USL and Bias Determination

The applicant determined the USL of the package first by applying a statistical calculation of the bias and its uncertainty, plus an administrative margin, to a linear fit of critical experiment benchmark data. This is known as Method 1: Confidence Band with Administrative Margin. The applicant verified the USL with a second method that utilized statistical techniques with a rigorous basis applied to determine a combined lower confidence band plus subcritical margin. This is known as Method 2: Single-Sided Uniform Width Closed Interval Approach.

For both methods, the applicant used USLSTATS to calculate USL correlations using typical problem-specific parameters described in Appendix C of Reference 8. Staff finds this method acceptable since it follows established recommendations. The trending parameters that the applicant evaluated are: the energy corresponding to the average lethargy of neutrons causing fission (EALF); moderator/fuel (H/X) ratio; and fissile weight percent (i.e., enrichment). The applicant used the trending parameter with the largest correlation coefficient to determine the USL. This follows the recommendation of Reference 8 and staff finds it appropriate.

6.6.2.1 FGE Contents Without Beryllium Reflector

The area of applicability (AOA) for plutonium experiments is shown in Table 6.8-5 of the application. The contents of the OPTIMUS®-H evaluated by the applicant extend beyond the AOA for some of the trending parameters. For FGE contents without beryllium (i.e., FGE-1, FGE-2abc, and FGE-5), the applicant showed ^{239}Pu enrichment has a small correlation coefficient. This indicates that extrapolation beyond the experimental AOA for this parameter will likely have little effect on bias and calculated USL. The range of H/X ratios for the OPTIMUS®-H contents evaluated also extends above the AOA of the plutonium experiments. Staff reviewed the results of the H/X ratios and noted the limiting cases for FGE contents without beryllium lie within the AOA of the experiments. In addition, the value of k_{eff} decreases as H/X ratio increases in the region beyond the AOA.

Any additional bias is unlikely to change the USL to a point where the limiting contents k_{eff} would exceed it. The range of EALF for the OPTIMUS®-H FGE contents falls within the AOA of the benchmark experiments. For these reasons, staff finds the applicant's justification for evaluation of contents outside the AOA acceptable.

The applicant's trending analysis showed the H/X ratio had the largest correlation coefficient for FGE contents without beryllium reflector, and the applicant used this parameter to determine USL.

6.6.2.2 FGE Contents with Beryllium Reflector

The contents of the OPTIMUS®-H evaluated by the applicant extend beyond the AOA for some of the trending parameters. For FGE contents with beryllium (i.e., FGE-3 and FGE-4), the ^{239}Pu enrichment correlation coefficient is much larger. However, the applicant showed the trend is increasing above the AOA for enrichment, and therefore a small extrapolation above the upper limit would not result in greater bias penalty to the USL. The range of H/X ratios for the OPTIMUS®-H contents evaluated also extends above the AOA of the plutonium experiments. Staff reviewed the results of the H/X ratios and noted the limiting cases for FGE contents with beryllium lie within the AOA of the experiments. In addition, the bias is decreasing in the range outside of the AOA.

Any additional bias for FGE contents with beryllium would result in a higher USL than the value used by the applicant. The range of EALF for the FGE contents falls within the AOA of the benchmark experiments. For these reasons, staff finds the applicant's justification for evaluation of contents outside the AOA acceptable.

The applicant's trending analysis showed the ^{239}Pu enrichment had the largest correlation coefficient for FGE contents with beryllium reflector, and the applicant used this parameter to determine USL.

6.6.2.3 FEM Contents

The AOA for uranium experiments is shown in Table 6.8-6 of the application. The FEM contents of the OPTIMUS®-H evaluated by the applicant extend beyond the AOA for some of the trending parameters. All of the LEU benchmarks have higher enrichment than the FEM contents of the OPTIMUS®-H. The applicant showed that the trend line is relatively flat, and the bias introduced to the criticality evaluation based on ^{235}U enrichment is not significant. For EALF, the range of the OPTIMUS-H FEM contents goes above the range of the LEU benchmarks. For FEM-2 contents, the applicant determined the bounding configuration with smaller particle sizes had an EALF that fell within the AOA of the benchmark experiments. For FEM-1 contents, the larger

particle size yielded a limiting configuration with an EALF higher than the AOA of the benchmark experiments.

The applicant extrapolated the potential bias penalty with the USL equation for EALF shown in Table 6.8-10 of the application. Its results were less than 25 % of the margin between the limiting FEM-2 case and the USL for FEM contents. The applicant did not apply this additional bias penalty to the USL, but given the large margin to the USL, staff finds it appropriate. The range of H/²³⁵U ratios for OPTIMUS®-H FEM contents fell within the AOA of the benchmark experiments. For these reasons, staff finds the applicant's justification for evaluation of contents outside the AOA acceptable.

The applicant's trending analysis showed the H/X ratio had the largest correlation coefficient for FEM contents, and the applicant used this parameter to determine USL.

6.7 Staff Confirmatory Analyses

The staff conducted its own confirmatory analyses using the SCALE 6.3 code suite. Specifically, the staff used KENO-VI with continuous energy cross-section libraries based on ENDF/B-VII nuclear data. Staff omitted the bolts and holes, and lid/body interface of CCV and OSV and modeled both as monolithic solid shapes. Staff included the gap space between the CCV and OSV and the inner shells of the ILs. For HAC array spacing, staff used the same spacing as the applicant.

Rather than re-create the entire suite of iterative studies conducted by the applicant, staff chose to limit the scope of its confirmatory analyses to a handful of limiting cases to confirm some of the most reactive configurations. Staff perturbed the variable the applicant was evaluating and compared the difference in staff's results to verify the condition of peak reactivity.

For FGE contents, staff repeated the fissile mass location evaluations. Staff results confirmed the top corner is the most reactive location except for FGE-3 contents. Staff also repeated the H/X evaluation for FGE-1 and FGE-2c contents. Staff selected the FGE-2c due its different optimum H/X. Staff results confirmed the applicant's determination of optimum H/X for both arrays of packages under HAC and single packages. Staff also confirmed the optimum H/X for a single package with FGE-2c contents differs from the other FGE contents. Staff results provide additional assurance in the accuracy and thoroughness of the applicant's FGE analyses.

For FEM contents, staff repeated the pitch evaluation at the upper cylinder size limits for FEM-1 contents. Staff selected the upper limiting size (i.e., 4 cm) and cases on either side of that size limit. Staff also selected pitch cases adjacent to the most reactive pitch in the applicant's analyses. Staff also repeated the most reactive cylinder size evaluations for FEM-2 contents with additional pitch evaluations. Staff results confirm the applicant's particle upper size restriction for FEM-1 rods contents and confirm that the applicant determined the most reactive pitch.

For FEM-2 contents, staff results also confirm that the most reactive unlimited rod size determined by the applicant will remain subcritical at optimum pitch. Staff results provide additional assurance in the accuracy and thoroughness of the applicant's FEM analyses. Considering the applicant's conservative and bounding assumptions and accounting for two standard deviations, the most reactive configuration calculated by the applicant for each contents type remains below the package USL.

The staff results provide additional assurance that the applicant has demonstrated the OPTIMUS®-H package will remain subcritical.

6.9 Findings

The staff has reviewed the OPTIMUS®-H 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 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 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 staff 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 accident conditions in 10 CFR 71.73. 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 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 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, for Type B fissile packages, 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 concludes that the criticality evaluation in the application of the most reactive array of damaged 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 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 [or 10 CFR 71.74 for plutonium packages transported by air, per 10 CFR 71.64(a)(1)(iii)] demonstrates that the array of 2N packages is subcritical under hypothetical accident conditions in 10 CFR 71.73 [or under the accident conditions in 10 CFR 71.74] to meet the requirements in 10 CFR 71.59(a)(2).

The staff 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 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 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 and contents satisfy the nuclear criticality safety requirements in 10 CFR Part 71. 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

6.10 References

1. U.S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," NUREG-2216, August 2020.
2. Pacific Northwest National Laboratory, "Compendium of Material Composition Data for Radiation Transport Modeling," PNNL-15870, April 2006.
3. Science Applications International Corporation, "Reactivity Effects of Moderator and Reflector Materials on a Finite Plutonium System," SAIC-1322-001 Rev. 1, May 2004.
4. TRUPact II SAR 71-9218
5. Washington TRU Solutions LLC, "Test Plan to Determine the TRU Waste Polyethylene Packing Fraction," WP 08-PT.09, June 2003.
6. Daher-TLI, "NAC-SMP Criticality Safety Analysis" CN-16007-602 Rev. 1, May 2018.
7. Organization for Economic Cooperation and Development – Nuclear Energy Agency, "International Handbook of Evaluated Criticality Safety Benchmark Experiments," NEA/NSC/DOC(95)03, 2014.
8. Oak Ridge National Laboratory, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," NUREG/CR-6361, March 1997.

7.0 OPERATING PROCEDURES

Package Loading

The applicant stated, in SAR section 7.1.1, that loading operations shall be performed in a precipitation-free environment, e.g., under a protective cover. If standing water collects inside the CCV cavity and/or SIA cavity (if used), absorbent materials or another suitable method, such as a vacuum system, shall be used to remove the free-standing water from the CCV cavity and/or SIA cavity (if used), which may require the contents to be unloaded. The staff agrees with the objectives, as stated in the application, to prevent any water accumulation within the package in order to prevent hydrogen and other flammable gas generations from radiolysis within the package.

Preparation for Transport

The applicant stated in SAR 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 staff confirmed that this temperature survey on the exterior surface of the package will be performed by the users before shipment.

Determination of Flammable Gas Concentration and Shipping Time

The applicant stated, in SAR Attachment 7.5-3, "Procedure for Determination of Flammable Gas Concentration and Shipping Time," that the flammable gas concentration calculations may be performed using the CCV free volume for the specific packaging configuration and the bounding NCT temperature based on the packaging decay heat of the contents. The applicant presented the CCV free volume and fill gas temperatures of four assembly configurations (bare basket, 1-inch thick, 2 ¼ inch thick and 3 ¾ inch thick SIA) in SAR Table 7.5-9 and the average temperature of the contents and fill gas inside the CCV as a function of the total heat load of the contents and the type of fill gas used, as shown in SAR figure 7.5-2.

The staff reviewed SAR Attachment 7.5-3, table 7.5-9 and figure 7.5-2 and accepts the procedure presented by the applicant for determination of flammable gas concentration and shipping time; however, the application needs to meet the criteria that hydrogen and other flammable gases make up less than 5 vol% or lower by volume of the total gas inventory within any confined volume in the package for TRU waste, in accordance with NUREG/CR-6673. Staff also notes that the methods used for determining waste forms and quantities are to have the needed sensitivity and accuracy to result in calculated flammable gas concentrations that meet the LFL criterion.

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

Thermal Acceptance Tests

The applicant stated, in SAR Section 8.1.7, *Thermal Tests*, that thermal acceptance testing of the packaging is not required for OPTIMUS-H because the packaging does not include any specific features that require thermal acceptance testing and the material properties used for the thermal evaluation of the package are sufficiently conservative. As instructed in Section 9.4.1.8, *Thermal Tests*, of NUREG 2216, *Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material*, a thermal test is used to verify that the heat transfer performance is achieved in the fabrication process. The staff referred to Section 9.4.1.8 of NUREG-2216 and accepts that the thermal test is not required for OPTIMUS®-H package, based on thermal design features, acceptable temperature margins below the allowable limits, and NCT and HAC thermal evaluations presented in Chapter 3 of the application.

Pressure Testing of the Containment Boundary

The applicant stated, in SAR section 8.1.3, that each CCV assembly shall be pressure tested to 150% of the packaging design pressure of 100 psig to verify the capability of the containment system to maintain its structural integrity, in compliance with 10 CFR 71.85(b) and the test pressure will be maintained for a minimum of 10 minutes prior to initiation of the examination for leakage, in accordance with the requirements of NB-6223 of ASME B&PV Code, section III.

Fabrication Leakage Rate Tests

The applicant stated, in SAR section 8.1.4, that the CCV assembly (e.g., the packaging containment boundary) shall be leakage rate tested in accordance with ANSI N14.5 to an acceptance criterion of 1×10^{-7} ref-cm³/sec. 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 SAR section 8.2.2.1 prior to their initial use. The staff reviewed the OPTIMUS®-H package to ensure that the package will be helium leakage rate tested for fabrication, in accordance with ANSI N14.5 and the fabrication leakage rate test procedure shall be approved by personnel with an ASNT Level III certification in leak testing.

Periodic and Maintenance Leakage Rate Testing

The applicant described, in SAR section 8.2.2.1, "Periodic and Maintenance Leakage Rate Testing," that OPTIMUS®-H package is tested to a leak-tight criteria (1×10^{-7} ref-cm³/sec) for maintenance and periodic leakage rate tests, in accordance with ANSI N14.5.

The staff reviewed the OPTIMUS®-H package to ensure that the package will be helium leakage rate tested for maintenance in accordance with ANSI N14.5 and the periodic and maintenance leakage rate test procedures shall be approved by personnel with an ASNT Level III certification in leak testing.

Pre-shipment Leakage Rate Testing

The applicant described, in SAR section 8.2.2.2, "Pre-shipment Leakage Rate Testing," that the pre-shipment leakage rate testing of the CCV lid containment seal and CCV port cover containment seal of the loaded packaging is required before each shipment of a loaded package to verify that the containment system is properly assembled for shipment.

The staff accepts the applicant's statement and confirmed that the pre-shipment leakage rate test shall be performed using the methods described in section A.5.1 and A.5.2 of ANSI N14.5 and the pre-shipment leakage rate test procedures shall be approved by personnel with an ASNT Level III certification in leak testing.

Periodic and Maintenance Thermal Tests

The applicant stated, in SAR Section 8.2.4, *Thermal Test*, that the periodic or routine thermal tests are not required on the package.

The staff accepts this statement based on thermal design features, acceptable temperature margins below the allowable limits, and NCT and HAC thermal evaluations presented in Chapter 3 of the application, in accordance with Section 9.4.2.6, *Thermal Tests*, of NUREG 2216.

CONDITIONS

The following Conditions are included in the certificate:

The package must be loaded and prepared for shipment in accordance with the Package Operations in Section 7 of the application.

The package must be tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Section 8 of the application.

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.). TRU waste contents may be shipped on a nonexclusive use conveyance or under exclusive use controls. IFW shall be shipped under exclusive use.

Each package prepared for shipment may only contain one form of non-compliant TRU waste contents, alone or in combination with compliant TRU waste (Content 1-1, Content 1-2A, Content 1-2B, Content 1-2C, Contents 1-1 and 1-2A, Contents 1-1 and 1-2B, and Contents 1-1 and 1-2C) and shall otherwise satisfy the Waste Isolation Pilot Plant (WIPP) Waste Acceptance Criteria (WAC).

The package shipment for TRU waste shall ensure that hydrogen and other flammable gases make up less than 5 percent by volume of the total gas inventory, or lower if warranted by the flammable gas, within any confined volume, for shipment within the allowable shipping time frame, as determined by Attachment 7.5-3 of the application.

Maximum decay heat for shipment of TRU waste is 200 watts. For TRU waste, with a total decay heat exceeding 50 watts, the CCV cavity shall be filled with helium gas per Table 3.1-1 of the application.

Maximum decay heat for shipment of IFW is 1,500 watts. For any heat load (0-1,500 watts), the CCV cavity shall be filled with helium gas per Table 3.1-1 of the application.

If not transported by private carriers, with individual monitoring of personnel in conformance with 10 CFR 20.1502, the minimum distance to occupied spaces shall be 20 ft from the centerline of the nearest package.

Transport by air is not authorized.

CONCLUSION

Based on the statements and representations in the application, the staff finds that the package meets the requirements of 10 CFR Part 71.

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