

**CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES**

1. a. CERTIFICATION NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9383	0	71-9383	USA/9383/B(U)F-96	1	OF 5

2. PREAMBLE

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

a. ISSUED TO (Name and Address)

GNS Gesellschaft für Nuklear-Service mbH
Frohnhauser Straße 67
45127 Essen Germany

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

GNS Report 1014-SR-00001, Rev. 6
Safety Analyses Report Type B(U)F Transport
Package CASTOR® geo69, dated
October 28, 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: CASTOR® geo69
- (2) Description

The Model No. CASTOR® geo69 is a dual-purpose cask design intended for the transport, as a type B(U)F package, of up to 69 undamaged BWR fuel assemblies. The packaging consists of the cask with the cask body and the cask lid enclosing the canister, the fuel basket, and the shielding elements. The packaging includes two impact limiters fastened to the face ends of the cask in order to protect the cask and its content. The cask itself mainly consists of a monolithic cask body made of ductile cast iron that is sealed by the bolted cask lid. The cask body and the cask lid form the outer containment of the radioactive content. Neutron shielding is maintained by two circumferential rows of moderator columns inside the cask body wall and by moderator discs on the cask bottom and lid end. The package has trunnions at the top of the cask and tilting studs toward the bottom.

The canister, which contains the fuel basket and the fuel assemblies, forms the inner containment. The canister consists of a welded stainless steel shell (canister body), which is sealed by the stainless steel canister lid. The canister components (shell, lid, lid seal, bottom plate) and cask components (body, lid, lid seal) provide a redundant enclosure for the contained spent fuel. The fuel basket consists of a structural skeleton that maintains the subcritical arrangement of the fissile material contained in the fuel assemblies and holds additional components required for heat dissipation and shielding.

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5.(a)(2) Continued.

During transportation on public roads, the cask is protected by both lid-end and bottom-end impact limiters that cover the trunnions and tilting studs. The impact limiters consist of a steel casing filled with polyurethane foam. The selected materials and protection measures of the packaging are suitable for corrosion protection and decontamination under operating and transport conditions.

The weight of the fully loaded package, including the impact limiters is 148,880 kg (328,224 lbs.).

5.(a)(3) Drawings

The package shall be constructed and assembled in accordance with the following GNS Drawing numbers:

1014-DD-44719, Rev. 1, 1 Sheet CASTOR® geo69 Transport Configuration
1014-DD-37023, Rev. 0, 1 Sheet CASTOR® geo69 Transport & Storage Cask Assembly
1014-DD-36931, Rev. 0, 1 Sheet CASTOR® geo69 Internals Transport & Storage Cask
1014-DD-30934, Rev. 0, 8 Sheets CASTOR® geo69 Cask Transport & Storage Cask
1014-DD-36855, Rev. 0, 4 Sheets CASTOR® geo69 Canister Internals
1014-DD-30984, Rev. 1, 1 Sheet CASTOR® geo69 Basket Internals
1014-DD-33604, Rev. 2, 1 Sheet CASTOR® geo69 Shielding Elements Internals
1014-DD-38772, Rev. 1, 4 Sheets CASTOR® geo69 Impact Limiter

5.(b) Contents of Packaging

(1) Type and Form of Material

- (a) Undamaged spent BWR UO₂ fuel in a fuel basket.
- (b) Non-fuel components integral to a fuel assembly such as fuel channels, spacer grids, end fittings and/or plenum springs.
- (c) The fuel shall not be damaged. Any fuel rod or fuel assembly that cannot meet the pertinent fuel-specific or system-related regulations for the transportation package may not be loaded into the basket. Fuel rods may not contain pinholes, hairline cracks, or gross ruptures. Fuel rod assemblies may have assembly defects if able to meet the pertinent fuel-specific or system-related regulations.

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

- (a) 69 BWR UO₂ fuel assemblies (General Electric (GE) 8x8-1, GE 8x8-2, SPC 8x8-2, GE9B 8x8, GE12 LUA, and ATRIUM-10A) with characteristics described in Table 1.2-12 of the application. Fuel assemblies may contain unirradiated replacement dummy rods provided they displace equal or greater volume of water. The fuel assemblies are restricted to assembly burnup, enrichment, and cooling times as specified in Table 1.2-12 of the application.
- (b) The combined weight of all fuel assemblies, shall not exceed 22,000 kg (48,502 lbs.).
- (c) The total amount of initial heavy metal shall not exceed the limit listed in Table 1.2-11 of the application.
- (d) Fuel shall be loaded in accordance with one of the thermal requirements loading diagrams shown in Figure 1.2-9 of the application (with general loading schematic defined in Figure 1.2-8) and the maximum decay heat shall not be more than 18.5 kW.

5.(c) Criticality Safety Index: 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71, the CASTOR® geo69 package shall:

- (a) Be prepared for shipment and operate in accordance with the Package Operations in Chapter 7 of the application,
- (b) Meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application,
- (c) During transport, the fuel basket's aluminum-boron metal matrix composite (Al-B₄C-MMC) neutron absorber shall have no contact with the outer environment,
- (d) B₄C content in the aluminum-boron metal matrix composite (Al-B₄C-MMC) shall be in accordance with Section 3, Table 6 of Appendix 2-8, revision 6 of the application,
- (e) Mechanical properties of the aluminum-boron metal matrix composite (Al-B₄C-MMC) shall be in accordance with Section 3, Table 4 of Appendix 2-8, revision 6 of the application,
- (f) Acceptance criteria for uniform boron distribution of the aluminum-boron metal matrix composite (Al-B₄C-MMC) shall be in accordance with Section 2.5.2 of Appendix 2-8, revision 6 of the application,

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6. Continued.

- (g) Visual inspection criteria for of the aluminum-boron metal matrix composite (Al-B₄C-MMC) shall be in accordance with Section 2.6 of Appendix 2-8, revision 6 of the application,
- (h) Acceptance criteria for moderator material shall be in accordance with Section 8.1.6.1.2 of the application,
- (i) Acceptance criteria for moderator material mechanical properties shall be in accordance with Section 5.2, Table 5-1 of Appendix 2-7, revision 2 of the application,
- (j) Acceptance criteria for moderator material thermal properties shall be in accordance with Section 5.2, Table 5-2 of Appendix 2-7, revision 2 of the application,
- (k) Acceptance criteria for moderator material chemical composition shall be in accordance with Section 5.2, Table 5-3 of Appendix 2-7, revision 2 of the application, and
- (l) All test procedures used in demonstrating compliance with the above requirements shall conform to those in the application.

- 7. The package must be transported under exclusive use.
- 8. If a personnel barrier is placed over the package, an evaluation will be performed to ensure the barrier does not result in package temperatures above allowable values in the application due to impeded heat transfer.
- 9. Prior to transport by rail, the Association of American Railroads must have evaluated and approved the railcar and the system used to support and secure the package during transport.
- 10. Prior to marine or barge transport, the National Cargo Bureau, Inc., must have evaluated and approved the system used to support and secure the package to the barge or vessel, and must have certified that package stowage is in accordance with the regulations of the Commandant, United States Coast Guard.
- 11. Transport by air is not authorized.
- 12. The CASTOR® geo69 package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.
- 13. Expiration Date: November 30, 2029.

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REFERENCES

CASTOR® geo69 Transportation Package Safety Analysis Report, 1014-SR-00001, Revision No. 6, dated October 28, 2024.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION



Signed by Diaz-Sanabria, Yaira
on 11/15/24

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

Date: November 15, 2024





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

SAFETY EVALUATION REPORT

**Docket No. 71-9383
Model No. CASTOR® Geo69 Package
Certificate of Compliance No. 9383
Revision No. 0**

SUMMARY

By letter dated January 14, 2021 (Agencywide Documents Access and Management System Accession No. ML21033A353), as supplemented on July 2, 2021, August 25, 2021, March 2, 2022, August 30, 2023, June 28, 2024, September 26, 2024, October 28, 2024 (ML21196A374, ML21245A203, ML22073A009, ML23257A087, ML24192A184, ML24285A183, and ML24306A034 respectively), Gesellschaft für Nuklear-Service mbH (GNS) submitted an application for a certificate of compliance for the Model No. CASTOR® geo69 spent fuel transportation package (geo69 or the package).

By letter dated July 10, 2020 (ML20198M431), as supplemented on May 15, 2023, and June 27, 2024 (ML23257A089 and ML24192A182, respectively) GNS also requested approval of its quality assurance program.

EVALUATION

1.0 GENERAL INFORMATION

1.1 Packaging Description

The CASTOR® geo69 transportation package is shipped horizontally and is designed to transport 69 undamaged uranium BWR fuel assemblies (FA) within a basket fabricated from aluminum (with boron constituents) inner sheets and steel outer sheets. The packaging is comprised of a monolithic ductile cast iron cask body sealed with a bolted cask lid. Two circumferential rows of neutron shielding moderator columns are placed inside 56 holes bored into the cast iron cask body and moderator discs are placed on the cask bottom and lid. The inner and outer cask surfaces are coated to increase the emissivity to optimize heat transfer and the coatings are required to have stability over the entire intended service time (up to 60 years). Solid aluminum shielding segments, which also aid in conduction heat transfer, are located between the basket corners and the inner canister shell. Radial cooling fins are machined into the outer cask surface.

The cask includes trunnions and top and bottom impact limiters made of aluminum housing filled with polyurethane foam, which protects the cask during structural and thermal accident conditions. The cask interior and the canister are backfilled with pressurized helium. Metallic gaskets seal the cask and canister vessels.

The fuel basket axially and radially transfers the decay heat from the loaded fuel assemblies by the basket's center, bottom and top sheets via conduction and thermal radiation; in addition, the

heat is radially transferred to the round aluminum segments and the shielding elements to the canister body.

The canister, which holds the fuel assemblies within the basket structure, is made of stainless steel and includes a bolted lid arrangement. Metal gaskets at the lid are parts of the cask containment and pressure boundaries.

1.2 Drawings

The package shall be constructed and assembled in accordance with the following GNS Drawing numbers:

Drawing Number	Title
1014-DD-44719, Rev. 1, 1 Sheet	CASTOR® geo69 Transport Configuration
1014-DD-37023, Rev. 0, 1 Sheet	CASTOR® geo69 Transport & Storage Cask Assembly
1014-DD-36931, Rev. 0, 1 Sheet	CASTOR® geo69 Internals Transport & Storage Cask
1014-DD-30934, Rev. 0, 8 Sheets	CASTOR® geo69 Cask Transport & Storage Cask
1014-DD-36855, Rev. 0, 4 Sheets	CASTOR® geo69 Canister Internals
1014-DD-30984, Rev. 1, 1 Sheet	CASTOR® geo69 Basket Internals
1014-DD-33604, Rev. 2, 1 Sheet	CASTOR® geo69 Shielding Elements Internals
1014-DD-38772, Rev. 1, 4 Sheets	CASTOR® geo69 Impact Limiter

1.3 Contents

The geo69 is designed to hold 69 undamaged, spent BWR fuel assemblies restricted to the fuel assembly types with each having associated limits listed in tables 1.2-11 and 1.2-12 of the application. Dummy rods may replace individual fuel rods provided they have an equal or greater volume than an intact fuel rod for that assembly type.

1.4 Evaluation Findings

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 EVALUATION

The objective of the U.S. Nuclear Regulatory Commission's (NRC) structural evaluation is to verify that the applicant (GNS) has adequately analyzed the structural performance of the transportation package(cask, canister together with contents and impact limiters) so that it meets the regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71, "Packaging and Transportation of Radioactive Material." CASTOR® geo69 is a dual use system

and is separately licensed in the United States as a transportation package and as a dry cask storage system. The two major functions of a transportation package are to provide the cask with impact protection and maintain cask integrity under specified events both normal and accidental. The staff first evaluated the features of the impact limiters designed to absorb the impact energy and subsequently evaluated the stress analysis of the package components subject to inertial and other prescribed loads.

2.1 Description of Package Components

The CASTOR® geo69 is a dual-purpose cask for the storage and transport of boiling-water reactor (BWR) spent nuclear fuel (SNF). It is designated as a B(U)F package that contains more than $10^5 A_2$, and is designed to meet the requirements of 10 CFR 71.61.

The transportation package is an assembly of a closed bottom hollow cylindrical cask with a lid, enclosing a sealed canister containing a fuel basket holding up to 69 FA. The cask is protected at either end with impact limiters designed to allow energy dissipation in the event of a drop. To facilitate handling, the cask, is equipped with lifting trunnions and tilting studs attached to the outer side of the cask. The canister and the cask lid seal provide a double containment for the contained spent fuel, making the package amenable for use with high-burnup fuels. The safety analysis (SAR) figure 1.2-1 shows the main components of the package described below.

2.1.1 Cask Body

The cask body is a one-piece ductile cast iron (DCI) hollow cylinder closed at the bottom; the open top is closed by a bolted lid. The post casting the outer surface of the cask cylinder is machined to create radial cooling fins and deep holes are drilled from the bottom side into the thick cask wall. Each hole accommodates a moderator rod held in place by a short steel rod and spring resting on the stainless steel closure plate bolted to the bottom of the cask. A bottom moderator plate is screwed to the cask in the groove at the bottom of the cask. The closure plate is equipped with a test port for leak testing of the bottom closure system. For underwater operations the test port is made watertight by a sealing screw. The bottom moderator plate is held in place by the bottom stainless steel closure plate. The closure plate is equipped with threaded blind holes to attach the bottom impact limiter. The holes are sealed against the water intrusion during underwater operation and prior to impact limiter installation.

2.1.2 Cask Lifting and Tilting

The cask is equipped with a pair of side trunnions dedicated as points of attachment for lifting devices. The stainless steel trunnions are form fitted in recesses that are machined into the cask body and attached to the cask body using stainless steel screws which are prestressed using ultrasonic tightening devices. The trunnions allow the cask to be lifted vertically and supports tilting operations.

A pair of tilting studs at the bottom of the cask are milled out of the cask surface and serve only to support any cask tilting operation. It is not designed for any lifting. The wear surface of the tilting stud is shrink fitted with a stainless steel wear surface for protection against wear and corrosion. The tilting studs are provided with two screws to prevent twisting of the wear surface during operation.

During transportation, both the trunnions and the tilting studs are rendered inoperable as they are enclosed by the impact limiters preventing any outside access.

2.1.3 Cask Lid System

The cask lid system provides a hermetic seal to the cask. The lid is fixed to the cask body using hexagon headed screws which seals the cask by compressing the metal gasket fixed in a groove in the lid. Access to two service orifices on the lid, for vacuum drying and helium filling of the cask interior, is separately covered by blind flanges and protection caps secured by screws. A moderator plate is attached with screws to the underside of the lid. The lid is kept in place by a retention ring via threads in the inner side of the cask. The retention ring allows adjustment of the gap between the lid and the canister and prevents contact between the moderator plate and canister.

2.1.4 Canister

The canister is fabricated from stainless steel sheets welded together to form a cylinder. The cylinder is closed by welding a thick stainless steel bottom plate. To the top of the cylinder is welded a thick head ring made of stainless steel, to accommodate the canister lid and sealing system. The canister forms the first containment for the SNF. The canister is considered as hermetically sealed. A metal gasket, sealing ring and a washer ensures the leak tightness of the canister lid. Holes in the side of the head ring accommodated the clamping elements which are expanded using the form piece, locking and clamping the lid in place as the threaded lid bolts are tightened. There are two service orifices on the canister lid. The first for dewatering, vacuum drying and helium filling of the canister interior and the second for testing the leak tightness of the metal gasket. The access to these openings is separately covered by blind flanges and protection caps secured by screws.

2.1.5 Fuel Basket

The fuel basket is designed to ensure criticality safety and the removal of the decay heat for the BWR FA stored in the basket. The basket is made by stacking borated aluminum structural sheets into a grid. The stack is held together by the outer stainless steel sheets that support the entire height of the stack. The gap between the outer stainless steel sheets and the canister wall is fitted with stainless steel round shaped segments to provide support, radiation protection and heat dissipation. The gaps at the four corners between the canister wall and fuel basket skeleton is closed with shielding elements. At the bottom of each grid is a FA shoe that holds the FA in place within the grid enclosure.

2.1.6 Impact Limiters

Impact limiters (ILs) are added to the top and bottom of the cask to protect the cask and contents in the event of drops. The impact limiters consist of an aluminum housing with polyurethane (PU) foam filling to dissipate the impact energy. To provide protection against penetration, a penetration protection plate is installed on top of the load distribution plate in the lid impact limiter. The impact limiter is connected through the load distribution plate to the cask body. This prevents the impact limiter from separating from the cask body.

2.1.7 Additional package description

This section summarizes information about the package and its contents that the applicant has provided to satisfy the requirements of 10 CFR 71.33.

a) With respect to the packaging:

- (1) The applicant in SAR section 1.0.2 classifies the package as Type B(U)F and identifies its contents characteristics in SAR table 1.2-11 and states in SAR section 1.2.3 that the package contains plutonium in solid form.
- (2) The applicant in SAR table 1.2-4 provides the gross weight of the package as 148,880 kgs.
- (3) SAR section 1.0.2 identifies the GNS transport package as model number CASTOR® geo69.
- (4) In SAR section 1.2.1.8, the applicant states that the package has two containments. The inner containment is provided by the canister and its closure system, and the second outer containment is provided by the cask and its closure system. The components of both systems are listed in SAR tables 1.2-6 and 1.2-7 respectively.
- (5) The applicant has identified the materials of construction and component dimensions in the Parts lists and associated design drawings, provided in the SAR as appendix 1-3 through appendix 1-11. SAR table 1.2-3 lists the nominal weights of the package components, and the fabrication methods are identified in table 1.0-1. Materials specifically used as neutron absorbers or moderators are identified in SAR table 1.2-8 and table 1.2-9 for the cask and the canister, respectively. The packaging has no designed receptacle hence no internal or external support system is included in the design nor is any volume of the receptacle specified. The applicant has identified that heat transfer is primarily by conduction from the fuel basket via the round structural elements to the walls of the canister.

b) With respect to the contents of the package:

In SAR table 1.2-11, the applicant establishes the maximum radioactivity of the constituents, the maximum quantities of fissile material, the maximum normal operating pressure, maximum weight, and the maximum amount of decay heat. In addition, the applicant provides information about the characteristics of the FA that can be transported in this package in SAR table 1.2-12.

2.1.8 Staff Finding

The staff finds that GNS has provided a description of the package that is adequate to meet the requirements of 10 CFR 71.35 for the staff evaluation of the SAR of the package to the standards specified in 10 CFR Part 71 Subparts E and F. The staff finds the application in compliance with the contents of the application required under 10 CFR 71.31(a)(1) and 10 CFR 71.31(a)(2).

2.2 Codes and Standards

The applicant in SAR table 1.0-1 defines that governing Codes and Standards that is applied to the procurement, design, fabrication, and examination of the components of the CASTOR® geo69 transport package.

2.2.1 Staff Finding

The staff reviewed the information provided and determined that the applicant has used Codes and Standards that are consistent with the guidance in NUREG/CR-3854 for the design of components important to safety (ITS). More specifically, the identified American Society of

Mechanical Engineers (ASME) design code is dedicated to the design of transportation packages for SNF. For specialized components that are design specific, the applicant has provided in SAR chapter 2.0 the detailed design methodology used to achieve the desired component safety objectives. As necessary, the applicant has established the safety margins that exist within the component design. Whenever GNS has used German standards, a comparison to demonstrate its equivalence with U.S. codes of practice is provided.

The staff finds that the applicant has identified appropriate Codes and Standards for use in the procurement, design, fabrication, and examination of the components of the CASTOR® geo69 transport package in compliance with contents of the application required by 10 CFR 71.31(c),

2.3 General requirements for all packages

- (1) **Minimum package size:** SAR table 1.2-3 shows that the smallest overall dimension of the cask is significantly more than the 1000 mm required by 10 CFR 71.43(a).
- (2) **Tamper-indicating feature:** SAR section 1.2.1.2 specifies a protection seal attached to the cask lid system fixed with special screws to prevent unauthorized opening. This tamper-indicating feature is further described in SAR section 2.4.4. The staff finds this feature complies with the requirements of 10 CFR 71.43(b).
- (3) **Containment securing fastening device:** SAR section 1.2.1.2 presents the description of the closure of the cask with 72 hexagon headed screws which compresses a metal gasket. This provides for a positive fastening system that cannot be opened unintentionally or by pressure. The canister inside the cask serves as the primary containment, the lid of which is secured with another positive fastening system. The features of the positive closure of the containment are further described in SAR section 2.4.3. The staff finds these positive fastening systems for containment comply with the requirements of 10 CFR 71.43(c).
- (4) **Chemical, Galvanic, or other reactions:** SAR section 2.2.2 address the chemical, galvanic and other reactions between the materials of the package and the environment. The applicant states that no significant galvanic, chemical, or other reactions impact the safety-related components of the package. Protective coatings are applied to surfaces that can have galvanic reactions. The brief period of in-pool loading operation does have the potential for galvanic reaction, but the period is short to develop and adversely affect the components. For other portions of the transportation operation, the environment is dewatered and inerted with helium, eliminating the potential for galvanic reactions. The staff finds this to be compliant with the requirements of 10 CFR 71.43(d).
- (5) **Protection of valves against unauthorized operation:** The SAR states and the drawings depict that all the quick connect valves are blind plugged and covered with a protective cover screwed to the lid. The staff finds this design as complying with the requirements of 10 CFR 71.43(e).

2.3.1 Staff Finding

The staff, on reviewing the information provided in the SAR, find that the applicant has met the requirements of 10 CFR 71.43(a) through 10 CFR 71.43(e).

2.4 Evaluation of the Lifting Attachment and Tilting Studs

2.4.1 Lifting Attachment

The package is fitted with a pair of trunnions, that diametrically opposite each other, on the upper end of the cask. The trunnion locations are shown in SAR Drawing 1014-DD-37023 Rev. 0 and described in SAR section 1.2.1. The load carrying capacity of the trunnions is summarized in SAR section 2.5.1 from the analysis in appendix 2-3. The load factors used in the design are consistent with the criteria in NUREG-0612 and American National Standards Institute (ANSI) N14.6, as amplified by a dynamic factor of 1.15 in accordance with the Crane Manufactures of America publication #70. The design load factor of 6.9 is used in the design code criteria for yield stress and 11.5 for ultimate stress. The minimum temperature used in the design is -29 °C and the maximum temperature is 110 °C. The material properties for these temperatures are taken from ASME Boiler and Pressure Vessel Code (BPVC).II.D. The material of construction has a ratio of yield to ultimate of less than 0.8. The stress in the trunnion is computed following the stress intensities given in ASME section III, division 3, WB-3215 and evaluated to the Tresca criterion. The staff accepts the use of the Tresca criterion over the Von Mises criterion as it is slightly more conservative than the Von Mises criterion.

The trunnion is qualified with a minimum safety factor of 2.12 against yield and a minimum safety factor of 1.7 against ultimate. The trunnion bolts similarly are shown to have a minimum safety factor of 1.09 against yield and 1.40 against ultimate.

The engagement between the trunnion bolt threads in the cask body is verified using the German Standard KTA 3201.2. To demonstrate the equivalency of the German code of practice to that used in the US, the applicant has compared the engagement length computed using the KTA methodology and that using the "Machinery's Handbook." The more restrictive length of engagement was used in the design. Tables 2.1-24 through 2.1-35 provide the calculations undertaken to develop the comparison.

The staff find this approach to ensure that there is no stripping of the cask body threads and threads of the screw to be acceptable, in the absence of an equivalent U.S. code, as it is comparable to the U.S. reference of practice.

The acceptable preload forces in the trunnion bolts, considering the fabrication condition and the temperature conditions, are computed using coefficients from German industry standard VDI 2230-2. The applicant has made a comparison of the German code to the methodology used in the U.S. to compute these bolts stresses. The calculation shows that sometimes the German methodology leads to a higher computed safety factor than that of the U.S. methodology. The applicant adopted the lower safety factor in the design as both methodologies show a positive margin (both safety factors are greater than 1.0). The staff reviewed the adopted approach and finds that the engineering principles used in the determination of the bolt forces to be appropriate. The applicant has used the acceptance criteria in ANSI N14.6 and CMAA #70 in evaluating the bolt safety factors.

2.4.2 Tilting Studs

The design of the tilting studs is conducted using the criteria of ANSI N14.6 and CMMA #70. The design factors for the tilting studs are 3.45 for yield and 5.75 for ultimate. The design temperatures for the tilting studs are a minimum of -29 °C and maximum of 100 °C. The resulting minimum safety factor is 1.14 against yield and 1.17 against ultimate at maximum temperature.

2.4.3 Staff Finding

The staff has reviewed the lifting system for the package and has reasonable assurance that the lifting system is designed to meet the performance targets expected by the code stipulations. The staff finds that the design satisfies the requirements of 10 CFR 71.45(a) for lifting.

2.5 Drop Analysis

The applicant has summarized the structural design in the SAR and provided details of the structural analysis in the Appendices to the SAR. The staff therefore has, as needed, presented their review of an appendix prior to any discussion of the results presented in the SAR, in their evaluation.

2.5.1 Validation of the FEM for stress analysis

The applicant, in appendix 2-5, has presented the results of a scaled model drop test with round impact limiters. Numerical analysis is performed using a ANSYS finite element method (FEM) for validation and the results of the analysis compared to the recorded strain in the cask components during the drop test deceleration. The same finite elements are used for the CASTOR® geo69 model. For the static analysis in ANSYS, the drop deceleration was increased by an amplification factor to include dynamic effects, with a further increase using a coefficient to allow for any slap down effects. This is the same methodology used in the stress analysis of CASTOR® geo69. Even though wood was used in the impact limiters, it has no bearing on the validation of the FEM and the methodology. The applicant used their in-house software for the computation of rigid body deceleration and amplification factors in the design of transportation casks. The test program was used to extend the use of the software to round impact limiters with a scaled test cask from the applicant's design series.

Comparisons of the recorded strain with the computed strain, at different locations of interest, are presented in figure A 1 through figure A 76. Section 4 of the appendix provides a detailed discussion of the comparison of strain at different cutting planes. In general, there is reasonable agreement between the measured and computed strains. The deviations are explained and justified.

The applicant has applied the same methodology and presented the validation of their approach to the determination of the amplification factors for different components of the CASTOR® series fuel basket and cask under different drop orientations in appendix 2-5.

The staff finds that there is reasonable assurance that the methodology of using the rigid body deceleration, amplified by factors to include dynamic and slap down effects along with the FEM of the cask, can be used to perform a static stress analysis for inertial loads on the cask components. The staff concludes that the resulting stresses in the components would represent reasonable estimates of the demand in the cask components from drops. The cask design would be acceptable, if these stresses are shown to be meet the design stress limits of the

applicable code, when combined with other design stresses as identified by the load combinations.

2.5.2 Assessment of Impact Limiter Performance

In appendix 2-4, the applicant provides their assessment of the ILs ability to transfer the drop energy via contact and its dissipation through the deformation of the PU foam material. In addition, the applicant provides their estimate of the inertial loads developed in the IL components from the deceleration from the free fall on impact with the unyielding surface.

The rigid body decelerations of the CASTOR® geo69 transportation package for normal conditions of transport (NCT) and hypothetical accident conditions (HAC) drop requirements are assessed by the applicant in appendix 2-4. The applicant has used the FE software LS-DYNA to perform simulations of the required drop conditions. In the different drop simulations, the cask and the ILs are aligned relative to the unyielding drop surface by means of appropriate shifts and rotations of the FEM axis. The PU foam material information in the LS -DYNA FEM is derived from test data of the PU foam material, simulation behavior displayed during testing, and standard information provided by the manufacturer. The staff has separately evaluated the PU foam material testing program under the section titled “Characterization of the PU foam” in this safety evaluation report (SER).

The IL is evaluated in two stages. First, the rigid body decelerations of the package for a drop on an unyielding foundation is determined using a dynamic finite elements calculation in LS-DYNA for both minimum (-29 °C) and maximum (+80 °C) design temperatures.

Consequently, the response to the inertial loads is calculated for NCT and HAC using a static FEM of the IL for the penetration protection plate and load distribution plate for the lid IL along with the spacer and bolted joints (cap screws and washers), using the recorded deceleration time-history.

A schematic of the lid IL components is shown in figure 1. The design of the IL includes PU foam fillings, which are completely encapsulated by a thin aluminum sheet shell. A penetration protection system is located between the lid IL and the cask lid system. The penetration protection system consists of a load distribution plate, a penetration protection plate, a spacer and a PU foam plate. The lid IL is connected through the load distribution plate and spacer to the cask body. The load distribution plate is connected to the cask. The load distribution plate is also screwed to the spacer. The penetration protection is bolted to the load distribution plate. 8 pipes are arranged between the penetration protection plate and the load distribution plate to absorb shear forces. The spacer is connected to the ring by 16 screws. The details of the screw connections are shown in figure A 1. There is a nominal vertical gap between the load distribution plate and the cask lid to prevent any load transfer via interaction.

A schematic of the bottom IL components is shown in figure 2. Like the lid IL, the construction includes PU foam fillings, which are completely encapsulated by thin aluminum sheets. The aluminum sheets are welded to each other and the spacer. The bottom IL is connected through a spacer to the cask body with screws. The screws are mounted through the closure plate. Details of the screw connections are shown figure A 2.

The numerical model of the package includes all essential components such as the cask body, the penetration protection system, the top and bottom impact limiters along with the aluminum and PU foam. The model is based on the nominal dimensions of the cask body and ILs. SAR figure 5 shows the ANSYS FEM while figure 6 shows the details of LS-DYNA FEM for the lid IL

used in the dynamic analysis. The lid and bottom ILs are modeled with three-dimensional elements. Contact elements are defined at those locations where components come in contact or are expected to meet each other. The screws, which connect the ILs to the cask body and clamp the load distribution plates and the spacer against the cask body, are explicitly modeled. The contacts between threads and thread holes are modeled as bonded. The cross section of the screws in the threaded area is the stress cross section. Figure 8 shows the local coordinate system used for recording the deceleration time-history.

The material behaviors are assigned to the different elements of the numerical model in LS-DYNA. For the cask, a rigid material behavior is used. The load distribution plate and spacers are considered as elastic material, the aluminum sheets and penetration protection plates are considered as elastic-plastic, and the PU foam is modeled as crushable foam material.

The IL PU foam is further characterized in appendix 2-5 to establish the properties and behavior of the PU foam that can be input into the material definition cards for crushable foam so that the physical response to the drop is reasonably represented in the simulation.

The material properties assigned to FEM models for the drop analyses are listed in table 3. For the IL aluminum shell, stress-strain curves are presented for the maximum and minimum design temperatures in figure 4. The higher strength curve is used for the minimum design temperature. For the PU foam, the stress- strain curves in figure 3 are used along with the dynamic characteristic curves associated with each design temperature.

Package performance in drops is simulated using dynamic finite element analysis (FEA) in LS-DYNA to assess the deceleration time-history and the impact forces. The staff reviewed the energy time-history plots presented in appendix 2-4, figure A 3 through figure A 6. The staff observed that as the kinetic energy is being converted to internal energy the kinetic energy reaches its minimum as the internal energy reaches its maximum. The hourglass and sliding energies are small. The sliding energy remains positive and there is no energy lost in damping. The staff concludes that there is reasonable assurance that the IL material has performed its intended function in converting the drop kinetic energy into internal deformational energy.

The staff reviewed the contact energy at surfaces that play a significant role in load transfer. The applicant provided, in figure A 23 and figure A 24, plots of contact energies at the cask body to spacer, spacer to aluminum housing and the housing to the PU foam depicted in figure A 25. For the bottom IL, the contact surfaces of interest are the cask body to spacer and spacer to PU foam as shown in figure A 26. In addition, the applicant, in figure A 25 and figure A 26, provided deformation plots at various time points from which the interaction of the involved parts can be seen (figure A 27 shows the locations of the sections of the deformation plots). Additionally, in figure A 28 through figure A 32, time-history plots of selected points in the contact areas show that the penetrations are negligibly small, with values of 0 μm –34 μm . The staff, based on these plots, concludes that the contacts work as expected and there is no reduction in impact force due to energy transfer via the defined contacts.

The staff reviewed the drop displacement time histories and the timed development of the contact surface. The applicant provided this information in figure A 7 through figure A 10 for the displacement time histories of points on the IL closest to the rigid surface and in figure A 11 through figure A 14 for the cask center of gravity; figure A 15 through figure A 22 shows the deformation of the contact area. The staff concludes that the displacement of any point on the IL does not meet the drop surface under any of the prescribed drop conditions, hence there is

adequate crushable material in the IL to convert the drop energy into internal energy without the IL bottoming out.

2.5.3 Characterization of the PU foam material

The manufacturer, in their data sheets, provides information on their products using American Society for Testing and Materials testing protocol to establish the characteristics of the PU foam. The applicant performed additional testing to address modeling parameters that would be required to appropriately capture the performance of the PU foam in the LS-DYNA simulation.

Calculation 1014-TR-00071 Rev.1, appendix 2-5 of the SAR presents the applicant's characterization of the PU foam for use in the LS-DYNA FEA. The applicant has performed tests of the PU foam under the different conditions to which the foam will be exposed when used in the IL of the CASTOR® geo69 transport package. The IL uses two types of foam from the LAST-A-FOAM® FR-3700 series; each of these three types are characterized by additional tests performed by the applicant.

Appendix 2-5 provides a compilation of the mechanical characterization of a PU foam material LAST-A-FOAM® Rigid Polyurethane Foam. The compilation includes experimental testing, the theoretical derivation of the parameters for the material model and the material cards used in the analysis.

The special material models available in programs like LS-DYNA are extensively dependent on the density and type of the PU foam material in terms of the desired damping characteristics. Damping is dependent on the speed of load application and the range of temperatures to which the foam is exposed. The material is not tested for the complete set of parameters needed in the material model. However, for appropriate verification of the analysis results, testing was carried out for different densities, load application speeds and temperatures to determine the necessary material parameters. Following testing, theoretical characterization of the foam material was carried out by means of subsequent simulation of selected tests to assess reproducibility and appropriateness of the selected material model.

Five sets of PU foam tests were conducted and are identified as AP1 through AP5:

- AP1 - Uniaxial pressure tests at various compression rates + cyclical quasi-static pressure tests.
- AP2 - Uniaxial hot/cold pressure tests.
- AP3 - Hydrostatic pressure tests and confined compression tests.
- AP4 - Tensile tests.
- AP5 - Ball indentation tests.

The compression rate and speed of load application are provided in the Fraunhofer EMI report on page 3 for each of the foam types along with other specifics of the testing program

In the AP1 through AP5 tests, the testing direction (the direction of the test force) was parallel to the growth direction of the PU foam ('parallel to rise') and thus in the thickness direction of the supplied sheet material.

The vendor manufactures PU foam sheets of varying thicknesses. The manufacturer (General Plastics) data sheets are provided in appendix 1 of 1014-TR-00071, appendix 11 (1014-TR-00071 is SAR appendix 2-5) of the calculation referenced above. To establish the consistency of the foam, 25 mm³ test samples were taken across the width of the test sheet

from the upper and lower half. The test results from a hardness test showed no change in the compression stress curve with an almost constant value across the height for all densities of the foam material. Hence the staff concludes that the location on the sheet from which the test samples are taken has no influence on the testing program.

For all test involving a measurement of the change in height, the height at each load increment or load cycle is measured using the data from a Linear Variable Differential Transformer and an optical tracking method thus, providing an independent check on any compliance in the hydraulics of the test equipment.

Table 2-1 of appendix 2-5 presents the test plan identifying the test type with its associated material temperature, and rate of load application for the different tests.

The staff on reviewing the testing data presented by the applicant in appendix 2-5 finds that:

1. Even though the stress and strain behavior are the same between foam with different densities, density has a pronounced influence on the stress and strain at failure.
2. The state of stress in the specimen influences the failure behavior of the foam with little impact on strain hardening.
3. In the cyclic mode, the deformation is only reversible to a very minor extent.
4. Test speed has a significant effect on the stress level as the strain rates change and thus influences the failure time.
5. Test temperature has a significant influence on the stress level and has a minor influence on the failure time.

The applicant, based on the behavior of the different density of foams identified by the test program, established requirements for the selection of the material model cards in LS-DYNA.

For each density of foam, different material cards are used for behaviors that are different between the densities, while for properties that are similar, the same card is used for all densities.

For each foam density, the material model is expected to capture:

- Strain hardening behavior for compression corresponding to the characteristic stress-strain curve (upslope–plateau – upslope).
- Asymmetrical tension/compression failure behavior where the yield locus is represented at different stress levels for different stress states.
- Permanent deformation of the foam. This deformation is proportional to the elastic rebound when the overall deformation is considerably lower or negligible.
- Strain rate dependency of strain hardening (different strain rates can occur within a simulation).
- Temperature dependency - the planned simulations should be carried out in isothermal form. The temperature dependency is therefore not taken into consideration in the material model; instead, separate material cards are created for the different temperatures that are analyzed.

The applicant, using the above set of observations, selected the most suitable material model in LS-DYNA that can best represent these material characteristics. The Bilkhu/Dubois foam model was selected for use in the LS-DYNA analysis.

Section 4 of appendix 2 presents the basis for the selection of the material properties that are used as inputs in the material cards for each of the foam densities used in the IL. A separate set of material cards were created to capture the temperature dependency of the material properties.

To check how well these models simulate the material behavior, the applicant, in section 5 of appendix 2, simulated the real-life tests with strain rate-independent material cards. Because of the limitations in the use of LCARTE strain rate dependent strain hardening, the simulations are carried out in strain rate-independent form but with material cards scaled for the respective strain rate range. As a result of this, strain distribution inaccuracies are possible, because zones with a momentarily high strain rate are not stabilized by the influence of the strain rate. However, it is expected that the force or stress level of the real-life test is reproduced in an initial approximation.

A comparison of simulated results of unconfined compression at different strain rates was provided by performing a confined compression test, a tension test, and a ball test while keeping the temperature at +23 °C. The details of the observations and the results comparison is presented in section 5 of appendix 2.

Based on these comparisons, the staff concludes that the simulation provides reasonable assurance that the material model provides a good approximation of the real-life test results. This simulation can reproduce characteristic effects such as the dependency on the foam density, the strain rate and temperature dependency of the stress/strain behavior, and the dependency of the material behavior on the stress state.

However, the staff notes that there are areas of discrepancy that could not be addressed due to the limitations of the material model such as the absence of strain rate dependency close to element collapse and the monotonic strain hardening curves used in the simulation.

2.5.4 Staff Finding

The staff, after reviewing of all the information presented by the applicant, finds that there is reasonable assurance that the foam in the impact limiter can be represented in the LS-DYNA FEM for each of the drop analysis thus demonstrating compliance with 10 CFR 71.71 and 71.73.

2.6 Deceleration and Impact Force

The rigid body decelerations under NCT for a drop height of 0.3 m are presented in figure A 34 through figure A 37. Table 6 summarizes the maximum rigid body decelerations. Table 7 summarizes the top and bottom contact forces between the IL and an unyielding foundation. The maximum IL deformations are enveloped by the deformations under HAC drop.

The rigid body decelerations for load cases under HAC are shown in figure A 38 through figure A 51. Table 8 through 10 summarizes the maximum rigid body decelerations. Table 11 through table 13 summarizes the top and bottom contact forces between the IL and unyielding foundation. The cask does not come into direct contact with the unyielding foundation in any of the case scenarios considered.

Since higher loads occur at slap down compared to horizontal load drops, the applicant presents their assessment of the slap down condition and the determination of the enhancement factor C_{slap} in section 7. The maximum contact force between the IL and an

unyielding foundation for a side drop with an inclination angle of 0° is shown in figure A 60. The effect of slap down is further simulated for various inclination angles. The enhancement factor is the ratio of the maximum value of the contact force between the IL and the unyielding surface for slap down to the maximum value for side drop (inclination angle 0°). The results for the contact forces are shown in table 14. The greatest enhancement factor for slap down for the minimum design temperature is $C_{slap, T_{min}}=1.01$ for an inclination angle of 10°. Figure A 61 shows the contact force. For the maximum design temperature, the enhancement factor is $C_{slap, T_{max}}=1.09$ for an inclination angle of 5°.

2.6.1 Staff Finding

The staff finds that the applicant has accounted for the different dynamic load effects by the use of coefficients in their pseudo-static stress analysis of the cask components in ANSYS. They have justified the development of these coefficients using the dynamic characteristics of the transportation cask components and these results can be used in the subsequent stress analysis for component design of the CASTOR® geo69 system.

2.7 Analysis of Inertial Loads on the IL components

In addition to the energy absorbed by the IL PU foam, the bolts and screws, the load distribution plate, and the penetration protection plate also experience the inertial forces from the different drop conditions.

The applicant established the load bearing capacity of these components of the IL using the limit state analysis criteria of ASME Code section III division 1, subsection NF (NF-3221.4). To meet the criteria for Service Level D loadings, the HAC loads are increased by a 10/9 factor in the analysis.

For Service Level D, the lower bound service load is determined using a yield stress that is the greater of 1.2 Sy and 1.5 Sm, but not greater than 0.7 Su. For non-ASME materials the smaller value of 1.2 Sy and 0.7 Su are used. For bolt materials, the yield strength Sy is used.

The highest loads on the penetration protection system that are used in the limit analysis occur at the minimum design temperature. Conservatively, no credit is taken for the higher material strengths at lower temperatures.

In section 6.2.2.1, the applicant evaluates the lid impact limiter for the 9.0 m flat drop. The staff finds that the maximum impact force occurs at -29 °C. The staff notes from figure A 55 and figure A 56 that the penetration plate remains unloaded due to the design configuration and the resultant reaction from the drop only loads the distribution plate. Figure A 56 shows the increase in the reaction-force over time. The staff concludes that the distribution plate has adequate capacity because its ultimate load carrying capacity is greater than the resulting vertical force from the analysis and thus meets the code requirements. The staff notes that there is additional capacity given the conservatism used. In addition, from the review of figure A 57, the staff concludes that there is no contact between the penetration protect system and the cask lid as the deflection in the limit analysis is less than the construction gap between the distribution plate and the cask lid.

A similar approach is taken by the applicant for the 9.0 m side drop. The staff concludes that the applicant has shown that the ultimate lateral load carrying capacity of the penetration system is higher than the factored demand on impact limiter.

Performing a similar analysis for the 9.0 m edge drop at 600 ° inclination, the demand in the top impact system is bounded by the limit analysis in the vertical and side drop analysis.

For the puncture test, a vertical, cylindrical, mild steel bar 150 mm in diameter is placed in the longitudinal center of gravity of the cask. The impact force exerted by the bar on the cask surface is calculated assuming the bar behaves as an elastic, perfectly plastic material with a yield strength of 280 MPa, which is a typical yield strength of mild steel. The impact load is calculated from the properties of the puncture bar.

For all load cases of the cask and canister assembly, the design temperature used in the stress analysis is shown in table 2.7.29 and all the applied internal and external pressures are shown in table 2.7-30

2.8 Evaluation of Cask and Canister Components

The structural safety analysis considers the mechanical loads and load combinations to confirm that the load conditions required according to 10 CFR Part 71 (division 3, WA-3351.2 i) are met. The structural safety analysis includes the consideration of loads, materials and other specified design information. The analysis demonstrates the integrity of the containment function of the cask and canister including their seals and fastenings under NCT and HAC test conditions for a Type B(U) packaging

2.8.1 Modeling

The integrity of the components that support the containment function of the package are analyzed using two finite elements models. The first simulation model is the cask model. This model details the features of the cask incorporating a simplified model of the canister. This allows the inclusion of the inertial properties of the canister in the cask analysis. The second simulation model describes a detailed canister model inside a simplified cask.

Figure A 1 through figure A 5 of appendix 2-1 shows the FEM used in the cask stress analysis and contains the cask body, the cask lid with its respective sealing and bolting, the retention ring, and the closure plate with its respective sealing and bolting. The lid and bottom ILs, as well as the simplified canister, are considered for load application only.

Each run is based on the basic cask model and with the same meshing where the general loads like bolt preloads, pressures, temperature loads and accelerations are applied. Boundary conditions and contacts are varied based on the needs of specific conditions. All parts are meshed with solid elements (SOLID185). Parts are connected via frictional contacts and the threads of the respective bolting have bonded contacts with their plug holes.

Figure A 46 through figure A 49 shows the FEM with the mesh details used in the stress analysis of the canister, canister lid and respective bolting, with a simplified representation of the cask. All the elements are solid tetrahedron or pyramids. The contact definitions and the coefficient of friction is shown in table A 174.

2.8.2 Analysis condition under NCT and HAC

Minimum and maximum bolt pretensions of different bolting at assigned temperatures are shown in table A 180. The pretension loads are applied in a first load step via pretension elements (PRETS179). Therefore, in this step the gasket of the cask lid is compressed.

The gasket is modeled with gasket elements (INTER195). At the first step, the bolt preloading the gasket is compressed to ensure full compression prior to application of external loads. Table A 173 shows the metal gasket sealing values used in the analysis.

The temperature loading is applied as stationary load when bolt prestress is considered for maximum, minimum or room temperature. The applied temperature values are shown in table A 2. The temperature definitions are shown in figure A 10 through figure A 12.

The fire load and the hot and cold environmental case starts at room temperature, while the temperature load is applied incrementally in the second load step.

External and internal pressures are applied via surface elements (SURF154) which share their nodes with the underlying solid elements. The extent of the surface to which these loads are applied is shown in figure 8A for internal pressure and figure 9A for external pressure. The values of the pressures are shown in table A 181.

The decelerations from the drop test cases and the puncture test forces, as defined in table A 183 and table A 184 respectively, are applied in the last load step. The post drop contact area and acting pressure distributions resulting from the decelerations are derived from the dynamic simulations in LS-DYNA.

Component weights of the 3-dimensional (3-D) model are adjusted to the weights in table A 1 in section 1.3, by modifying their densities.

The welds are treated as base metal with the properties and allowable given in ASME BPVC section III division 1, subsection NF.

The average stress in tension, average stress in shear, stress ratio, and the maximum stress (stress intensity S) are adopted from NUREG/CR-6007.

To ensure leak tightness at the critical section, the pure shear stress criterion in ASME, division 3, WB3227.2 is used for NCT and division 3, WB-3224.1(d) for HAC.

The admissible elastic stress analysis stress intensity limits for the cask and canister are defined by the rules of division 3 (WB-3222) for NCT and division 3, (WB-3224) for HAC.

The admissible stresses under NCT and HAC conditions are shown in SAR table 2.1-2 and the admissible yield strength of the cask and canister bolts is shown in table 2.1-3.

2.8.3 Analysis conditions under test load

The admissible stress criteria under testing is shown in table 2.1-6. The maximum shear stress intensity criterion in division 3 (WB-3235) is applied to cask and canister threaded bolts.

2.8.4 Analysis conditions for Fatigue

The fatigue assessment of the lid bolting is performed using the procedure described in division 3 (WB-3232.4) to determine the design life for a given stress intensity amplitude. For each of the load cases, the total usage factor is determined by summing the usage factors.

2.9 Results of stress analysis

SAR section 2.6 presents the results of the analyses conducted using the requirements of the normal conditions of transport.

2.9.1 Normal conditions of Transport

Hot Environment - For the hot environment load case, the applicant summarized the minimum factors of safety of the cask and canister components in table 2.6-6 and table 2.6-7, respectively. The results used the local stress intensities to report the component factor of safety. The details of the stress assessment are presented in appendix 2-1, table A 95, table A 96, and figure A 80 through figure A 85. All the factors of safety are above 1.0, hence in compliance with the acceptance criteria. The minimum cask lid and canister lid bolt factors of safety are 2.6 and 2.1 respectively, under maximum bolt prestress at maximum design temperature. The minimum factors of safety under min and max preloads are shown in table 2.6-8 and table 2.6-9 for the cask and canister bolts respectively, under the hot environment.

Cold Environment - For the cold environment load case, the applicant summarized the minimum factors of safety of the cask and canister components in table 2.6-10 and table 2.6-11. The results used the local stress intensities to report the component factors of safety. The details of the stress assessment are presented in appendix 2-1, table A 99, table A 100, and figure A 86 through figure A 91. All the factors of safety are above 1.0, hence in compliance with the acceptance criteria. The minimum cask lid and canister lid bolt factors of safety are 2.4 and 2.0 respectively, under maximum bolt prestress at maximum design temperature. The minimum factors of safety under min and max preloads are shown in table 2.6-12 and table 2.6-13 for the cask and canister bolts respectively, under the cold environment.

Vibration and Shock – For transport shock loadings, the applicant used a combination of the maximum inertia loads for transport via truck and ship applied in two of the three x-y-z coordinate directions. Table 2.6-14 shows the g-values adapted from NUREG 766510 and SSG-26. The assessment is performed for the maximum and minimum design temperatures with the bolt preloads shown in table 2.6-3, internal and external pressures shown in table 2.6-5, and the shock loads shown in table 2.6-14. The simulation is performed in appendix 2-1.

Table 2.6-15 presents the minimum factors of safety of the cask parts as 1.44 under maximum design temperature. The details of the stress assessment are presented in appendix 2-1. Table 2.6-16 shows the assessed minimum factors of safety of the canister parts as 1.56. The details of the stress assessment are shown in appendix 2-1. The minimum factor of safety in the cask lid bolting is 2.9 at maximum bolt preload under maximum design temperature. The minimum factor of safety in the canister lid bolting is 1.8 at maximum bolt preload under maximum design temperature. Therefore, the seal tightness of the cask and canister lids is ensured. The evaluated bolt stresses are summarized in appendix 2-1.

Reduced External Pressure – enveloped under the condition of Vibration and Shock loading

Increased External Pressure - enveloped under the condition of Vibration and Shock loading

0.3m side drop – The package is only able to drop onto the side due to the given transport orientation. Table 2.6-19 presents the assessed minimum factors of safety of the critical components of the cask parts. The minimum factor of safety is 1.43. Table 2.6-20 presents the assessed minimum factors of safety of the critical components of the canister parts. The minimum factor of safety is 1.47. The minimum factor of safety in the cask lid bolting is 1.7 at

maximum bolt preload under maximum design temperature. The minimum factor of safety in the canister lid bolting is 1.7 at maximum bolt preload under maximum design temperature. The seal tightness of the cask and canister lids is assured as the bolt maintains a positive factor safety. The details of the stress assessment are presented in appendix 2-1.

Corner Drop – Not required for weight more than 100 kg.

Compression – Not required for weight more than 5000 kg.

Penetration – Enveloped by the Puncture load case under HAC

The summary of all evaluated minimum factors of safety during normal conditions is shown in table 2.6-23 for the cask parts and in table 2.6-24 for the canister parts. Factors of safety are greater than 1.0. The staff therefore conclude that the requirements under NCT are met by the transportation cask and canister components in compliance with 10 CFR 71.71.

2.9.2 Hypothetical Accident Conditions

The initial conditions for each load case are per table 1 of the NRC Regulatory Guide 7.8. The inertia loads under HAC conditions are applied to the CASTOR® geo69 components as determined by simulations. A summary of the evaluated inertia loadings is presented in appendix 2-1, table A 184. The stress assessments for components other than lid bolts are performed under minimum and maximum temperature and maximum bolt preload since the bolt preloads do not impact the stress assessment results. In the static stress analysis, a dynamic amplification factor is used to cover dynamic effects for the loads on cask, canister and respective cask lid and canister lid evaluation. The dynamic load factors are shown in appendix 2-1, table A 184.

9m Flat Drop on to Lid Side - The results of the dynamic impact load used as inertia loading for the 9 m flat drop on lid side is 63.6 g under minimum design temperature as shown in appendix 2-1, table A 184.

SAR table 2.7-3 presents the computed minimum factors of safety for the critical components of the cask. The minimum factor of safety is 3.06 at the center of the cask lid under maximum design temperature conditions. The assessment uses the local stress intensity as a criterion. The details of the stress calculations are documented in appendix 2-1, subsection 12.1.6, table A 113 and figure A 116 through figure A 121.

SAR table 2.7-4 presents the assessed minimum factors of safety of the critical components of the canister parts. The form pieces and washers are not considered in the stress assessment, since they are not critical to the structural integrity of the containment. The minimum factor of safety is 1.24 under minimum design temperature. The assessment is performed with the local stress intensity value. The details of the stress assessment are presented in appendix 2-1 subsection 12.1.6, table A 115 and figure A 122 to figure A 127.

The minimum factor of safety in the cask lid bolting is 1.7 at maximum bolt preload under maximum design temperature as shown in table 2.7-5. The minimum factor of safety in the canister lid bolting is 2.4 at maximum bolt preload under maximum design temperature, shown in table 2.7-6. The seal tightness of the cask and canister lids is ensured as the bolt factors of safety are all greater than 1.0.

Evaluation of the retention ring - Under static inertia loading it is shown that the thread on the cask side has a factor of safety of 2.58 against stripping. For shearing of the thread in the cast ring from inertial loading, the factor of safety is 2.01, hence the retention ring stays in place under this drop condition. The detailed computation is shown in appendix 2-1, subsection 12.1.6, table A 114.

Evaluation of the canister buckling - The maximum allowable compressive stress is calculated according to the rules in division 3, WB 3133.6(b) for Level A and WB 3224.1(e) for Level D as shown in table 2.7-7. The nominal stresses in the canister shell cross section due to Level D loads from the flat drop load cases are less than the allowable axial compressive stress for Level D loads according to table 2.7- 8. The axial compression criterion is satisfied with a factor of safety of 1.48. The flat drop bottom side direction is evaluated, since the mass of the canister lid acts additionally onto the canister cross section. In the lid side orientation, the canister lid rests on the retention ring, which reduces the load onto the canister shell cross section.

End Drop - 9m flat drop on to bottom side – From the dynamic impact analysis, the inertia loading for the 9 m flat drop onto the bottom side is 61 g under minimum design temperature for the cask and canister. Appendix 2-1, table A 184 provides the details of the simulation. To account for dynamic effects on the lid bolting in the static analysis, a dynamic load factor computed in appendix 2-1, table A 184, is used. Table 2.7-9 shows the computed minimum factors of safety for the critical components of the cask. The minimum factor of safety is 1.8 in the cask lid under minimum design temperature. The calculation used the local stress intensity values. The details of the stress assessment are presented in appendix 2-1, subsection 12.1.9, table A 134 and figure A 156 through figure A 161. Table 2.7-10 shows the computed minimum factors of safety for the critical components of the canister. The minimum factor of safety is 2.20 in the canister lid under maximum design temperature. The factor of safety is calculated using the local stress intensity values. The details of the stress assessment are presented in appendix 2-1, subsection 12.1.9, table A 135 and figure A 162 through figure A 165. The minimum factor of safety in the cask lid bolting is 3.6 at maximum bolt preload under maximum design temperature shown in table 2.7-11. The minimum factor of safety in the canister lid bolting is 2.1 at maximum bolt preload under maximum design temperature and is shown in table 2.7-12. The computed bolt stresses are shown in appendix 2-1, subsection 12.1.9. The seal tightness of the cask and canister lids are ensured as the bolt factors of safety are greater than 1.0

9m Side Drop - The impact analysis of the 9 m side drops results in an inertial loading under minimum design temperature of 86 g for the cask and for 98 g for the canister as shown in appendix 2-1, table A 184. The inertia loadings include dynamic load factors of 1.30 (cask) and 1.48 (canister). The details of the performed calculations are described in appendix 2-1, chapters 10 and 11, subsections 10.8.7 and 11.8.7. Table 2.7-13 shows the computed minimum factors of safety of the critical components of the cask parts. The minimum factor of safety is 1.84 in the cask lid under minimum design temperature. The details of the stress computation are shown in appendix 2-1, subsection 12.1.8, table A 128 and figure A 144 through figure A 149. Table 2.7-14 shows the minimum factors of safety of the critical canister components. The minimum factor of safety is 2.72 for the pure shear criterion and occurs in the head ring under maximum design temperature. The details of the stress calculation are presented in appendix 2-1, subsection 12.1.8, table A 129 and figure A 150 through figure A 155. The minimum factor of safety of the cask lid bolting is 1.5 at maximum bolt preload under maximum design temperature as shown in table 2.7-15. The minimum factor of safety for the canister lid bolting is 1.5 at maximum bolt preload under maximum design temperature as

shown in table 2.7-16. The computed bolt stresses are shown in appendix 2-1, subsection 12.1.8. The seal tightness of the cask and canister lids is ensured as the bolt minimum factors of safety are greater than 1.0.

9m Edge Drop onto Lid Side - In this drop the center of gravity is moved over the corner to avoid rotation of the package. A FEM is used in the simulation model, the details of which are presented in appendix 2-1, chapters 10 and 11, subsections 10.8.6 and 11.8.6. The edge drop onto lid side occurs at a drop angle of 66°, in which the longitudinal axis of the package is at an angle of 66° from the impact surface. The dynamic analysis of the 66° drop orientation calculated a maximum inertia loading of 50g for cask and canister under minimum design temperature in vertical drop direction appendix 2-1, table A 184.

Table 2.7-17 shows the minimum factors of safety of the critical components of the cask. The minimum factor of safety is 1.70 in the cask lid under minimum design temperature. The assessment is performed with the local stress intensity value. The details of the stress assessment are presented in appendix 2-1, subsection 12.1.7, table A 122 and fig. A 128 through fig. A 133. Table 2.7-18 presents the minimum factors of safety of the critical canister components. The minimum factor of safety is 2.05 in the canister lid for maximum design temperature. The details of the stress assessment are presented in appendix 2-1, subsection 12.1.7, table A 123 and figure A 134 through figure A 143. The minimum factor of safety in the cask lid bolting is 1.5 at minimum bolt preload under maximum design temperature shown in table 2.7-19. The minimum factor of safety in the canister lid bolting is 2.3 at maximum bolt preload under maximum design temperature shown in table 2.7-20. The evaluated bolt stresses are tabulated in appendix 2-1, subsection 12.1.7. The tightness of the cask and canister lids seals are ensured as the minimum factors of safety of the bolts are greater than 1.0.

9m Edge Drop onto bottom side - In this load case, the drop is a corner drop, where the center of gravity is moved over the corner due to rotation of the package. A FEM simulation is shown in appendix 2-1, subsection 11.8.9. The edge drop onto the bottom side occurs at a drop angle of 66° in which the longitudinal axis of the canister is at an angle of 66° with the impact surface. A 50g inertial force is computed for the canister under minimum design temperature in the vertical drop direction as shown in appendix 2-1, table A 184. The integrity of the cask is enveloped by the 9m edge drop onto lid side drop of the cask.

Table 2.7-21 shows the evaluated minimum factors of safety of the critical components of the canister. The minimum factor of safety is 1.07 at the bottom of the canister body under minimum design temperature. The details of the stress assessment are shown in appendix 2-1, subsection 12.1.10, table A 138 and figure A 166 through figure A 175.

The minimum factor of safety in the canister lid bolting is 1.7 at maximum bolt preload under maximum design temperature as shown in table 2.7-22. The details of the evaluated bolt stresses are shown in appendix 2-1, subsection 12.1.10. The seal tightness of the cask and canister lids are assured as the minimum factors of safety for the bolts are greater than 1.0.

Oblique drop - The simulations show that the resulting accelerations from the oblique drop are enveloped by those for the side drop simulation.

Crush – The mass of the package is greater than 500 kg, so the crush test is not applicable.

1m Puncture Laterally into Lid Side – The puncture on the unprotected center of cask body is more critical than this puncture test, hence no further evaluation is required. This area is protected by the impact limiter.

1m Puncture Laterally into Bottom side - The puncture on the unprotected center of cask body is more critical than this puncture test, hence no further evaluation is required. This area is protected by the impact limiter.

1 m Puncture Vertically onto Lid Side - The puncture on the unprotected center of cask body is more critical than this puncture test, hence no further evaluation is required. This area is protected by the impact limiter.

1m Puncture Vertically into Bottom of Cask - In this load case, the bar will need to puncture the impact limiter and closure plate before hitting the cask body. Table 2.7-24 shows the minimum factors of safety computed from the simulation. The minimum factor of safety is 1.15 in the center of the cask body bottom under maximum design temperature. The details of the stress assessment are presented in appendix 2-1, subsection 12.1.12, table A 141, figure A 178 and figure A 179.

1 m Puncture Vertically into center of cask body - When dropped laterally onto the center of the cask, the cask is not protected by the impact limiters in the central region and the deadweight of the whole package acts onto the puncture bar. This direction covers the 1 m lid side and bottom drop onto the puncture bar. Table 2.7-23 shows the minimum factors of safety of the critical components of the cask. The minimum factor of safety is 1.10 in the cask body center under maximum design temperature. The details of the calculation are shown in appendix 2-1, subsection 12.1.11, table A 140, figure A 176 and figure A 177.

Fire Accident - The stress analysis of the cask and canister components due to fire accident is performed as part of the HAC load combination. The details of the performed simulations are described in appendix 2-1, chapters 10 and 11, subsections 10.8.11 and 11.8.10. Bolt preload effects according to appendix 2-1, table A 180 and internal pressure as shown in appendix 2-1, table A 181) are considered. Table 2.7-25 shows the minimum factors of safety for the critical components of the cask. The minimum factor of safety is 1.82 in the cask webs between the moderator drill holes. The details of the stress computations are presented in appendix 2-1, subsection 12.1.13, table A 142 and figure A 180 through figure A 184. Table 2.7-26 show the computed minimum factors of safety for the critical components of the canister. The minimum factor of safety is 1.59 in the canister lid. The details of the stress computation are presented in appendix 2-1, subsection 12.1.13, table A 143, figure A 185 through figure A 191. The minimum factor of safety in the cask lid bolting is 1.1 with maximum bolt preload from table 2.7-27. The minimum factor of safety in the canister lid bolting is 2.0 at maximum bolt preload as shown in table 2.7-28. The evaluated bolt stresses are summarized in appendix 2-1, subsection 12.1.13. The tightness of the cask and canister lids are ensured as the minimum factors of safety of the bolts are greater than 1.0.

Differential Thermal Expansions - The differential thermal stress is not specifically computed as there exist sufficiently large gaps in the axial and radial direction for free thermal expansion, eliminating contact and restraint.

2.10 Water Immersion Test

10 CFR 71.61 requires that the package be subjected to an external water pressure of 2 MPa for a period of not less than one hour without collapse, buckling, or in leakage of water. Since

the canister is inside the cask, a demonstration of the integrity of the cask under this condition is acceptable as meeting the requirement. The analysis is conducted for the cask model. A linear-elastic material model is used. Since the cask has significant wall thickness, if the stresses are shown to be within the limits it can be accepted that there will be no buckling. A lack of a separate buckling analysis is acceptable.

The criticality analysis is conducted with water flooding and has no bearing on this requirement.

Table 2.7-31 shows the minimum factors of safety for the critical cask components. The minimum factor of safety amount is 1.02 in the fillet at the cask lid. The details of the stress calculation are presented in appendix 2-1, subsection 12.1.14, table A 148, figure A 192 through figure A 194. The minimum factor of safety in the cask lid bolting is 2.4 with a minimum bolt preload as shown in table 2.7-32. The evaluated bolt stresses are summarized in appendix 2-1, subsection 12.1.14. The tightness of the seal is ensured as the minimum factors of safety of the bolts are greater than 1.0.

2.11 Recheck of factors of safety due to new fission gas release rates

The internal and external pressures were recalculated due to increased fission gas release rates for Low Burn-up Fuel (LBF) as shown in SAR chapter 4. The factors of safety for all load cases where it is affected by the change in pressure is proportioned using a reduction factor. A check was conducted to verify if the pressures used in the structural stress evaluation under NCT and HAC are valid. Table A 181 shows that the pressures used envelopes the LBF condition except for the max. internal pressures under NCT and HAC (without fire) for the canister. Table A 67 and table A 68 shows the lowest factors of safety from the verifications of NCT, HAC, and increased pressure. Since the lowest factor of safety is above 1, as shown in table A 67 and table A 68, no new calculations are needed, and all verifications are still valid considering the increased pressures due to fission gas release rates.

2.12 Fatigue Assessment of Components

Cycles of loading and unloading occur on components of the package which are used in operations involving handling, transport and storage. These components are evaluated by the applicant for fatigue loading in accordance with ASME BPVC, Sec III, Div.3, 2017 Edition, WB 3232.4. The process for the computation and determining the parameters that influence the assessment is shown in section 2.2.4. Assembly state, test pressure, shock, hot environment and side drop are the load cases considered in the fatigue assessment. Appendix 2-1, table A 64 summarizes the fatigue assessment for the lid bolts. The results of the fatigue analysis for the containment parts are shown in table A 66. For the lid bolts and the containment parts the cumulative damage factors are less than 1.0 indicating that there is no fatigue related failure that can be expected during the service life of the system.

2.13 Comparison of German and US standards

The applicant has used German standards in lieu of U.S. standards for the design of some of the package components. To demonstrate that the German standards produce the same level of design safety as the U.S. codes the applicant has evaluated the same components using the US codes. This approach was considered acceptable as the components are readily fabricated using the German codes in Germany, the country of origin of the cask design.

The German Code VDI 2230-1 was used to compute the prestress in the lid bolts and other attachment bolts. NUREG / CR-6007 provides a methodology for bolt prestress computation.

Tables 2.1-15 through table 1.2-22 provide a comparison of the preload computation using NUREG / CR-6007 for different bolts applications which are prestressed. The results show that the preloads computed using the German codes results in comparable bolt preloads.

The applicant, in section 2.1.4.1.2, provides the factor of safety in the bolt stress computed using the methodology in Machine Design. The results are shown in table 2.1-23. The calculation is performed for a stress design factor of 6 and 10 (left and right side in table 2.1-23). The results show that the methodology in Machine Design results in computed factors of safety greater than 1.0, the same as that results from using the German code VDI 2230-2.

An evaluation of the minimum engagement length of bolts or screws is required to ensure that the threads do not strip under load leading to disengagement of the threaded component. The German Code KTA 3201.2 is used by the applicant for this assessment. Using the methodology in Machinery's Handbook, the applicant recomputed this parameter for all the screws and bolts. The results of this computation are shown in tables 2.1-24 through table 2.1-35. The use of KTA 3201.2 and "Machinery's Handbook" leads to required lengths of engagement which meet the design criteria.

The applicant has in addition provided a comparison of KTA 3201.2 with ASME BPVC.III.3 2017 and ASME BPVC.III.A 2017, as it is used in the design of the fuel basket load bearing components. The detailed analysis for the fuel baskets is presented in appendix 2-2 (Fuel basket analyses). KTA 3201.2 is used for the classification of load cases and the evaluation of stresses and strains. Many elements in KTA 3201.2 can also be found in ASME BPVC.III.3-2017 WD and ASME BPVC.III.A-2017, e. g. the allowable stresses for level A events and the analysis and acceptance criterion for load bearing capacity analysis in level D events.

The comparability of both standards for stress limits (according to KTA 3201.2 and ASME BPVC.III.3-2017 / ASME BPVC.III.A-2017) are given in table 2.1-37 and table 2.1-38 for level A and level D events for an elastic analysis.

The criteria for level A are the same in KTA 3201.2 and ASME BVC.III.3-2017. The criteria for level D differ between KTA 3201.2 and ASME BVC.III.A-2017. Since a load bearing capacity analysis with plastic calculations is performed in appendix 2-2, the difference does not matter.

If the elastic analysis for level D loadings is not successful, like in appendix 2-2 (fuel basket analyses), both standards allow for an alternative load bearing capacity analysis with the help of plastic calculations. These calculations assume material behavior that is ideally plastic with a fictitious yield stress to prove sufficient safety margins against the collapse load.

The criterion and yield stress considered in the load bearing capacity analyses are the same in KTA 3201.2 and ASME BVC.III.A 2017 as shown in table 2.1-39.

Therefore, both standards give identical criteria and acceptance criteria. The staff finds that the methodology used in appendix 2-2 is applicable for the mechanical design of fuel baskets.

2.13.1 Staff Finding

The staff finds that the comparison between the German codes of practice and the US codes and industry practices yield comparable levels of design safety and finds reasonable assurance in their use to meet the regulatory requirements of a transportation cask design. The staff finds that the few instances of the use of non-US codes and standards, to support compliance with

the regulatory safety under 10 CFR 71 requirements for this specific transportation package, to be reasonable given the information in the SAR and supporting documents.

2.14 Evaluation of Fuel Basket and Shielding Elements

Appendix 2-2 presents the details of the analysis of the fuel basket and associated shielding elements. The analysis establishes the adequacy of the basket and the shielding elements regarding their capability to support the loads resulting from the different conditions to which the package is subjected.

The cells holding the FA are made of boron alloyed aluminum (Al-B4C-MMC) stacked and interlocked among each other and are plugged in the continuous outer sheets, forming the inner frame structure of the fuel basket. The outer sheets are made of austenitic stainless steel and are supported by the round aluminum segments. The space between the fuel basket and the canister is filled by four aluminum shielding elements. The fuel basket holds 69 FA's. When the cask stands in the upright position, the FA shoes rest on the canister bottom. All components of the fuel basket are load-bearing components. Figure 1 shows the fuel basket in the horizontal position.

The analyses are performed for different drop positions and load orientations. A distinction is made between side drop positions with various orientation angles and flat drop positions with the loads acting in the axial direction of the cask. The side drop is indicated also as side load and the flat drop as vertical load.

The analysis of the fuel basket is performed with the assumption that the decelerations acting on the package are transferred completely to the fuel basket and the contents. The mechanical loads are modeled as static loads, and the dynamic effects are included using coefficients. The thermal loads are considered as steady-state temperature fields.

When determining the mechanical properties of the materials, maximum design temperature is taken as a basis to conservatively address their temperature dependence.

The verification of the load-bearing capacity of the component is accomplished by a linear-elastic stress analysis as described in section 3.3.1. If this does not result in acceptable outcomes, a nonlinear stress-strain analysis according to section 3.3.2 is performed. The assessment of residual deformations is performed following the methodology in section 3.3.3. The stability of the fuel basket is performed as described in section 3.3.4.

The stresses and strains are calculated using standard analytical and numerical methods. The numerical algorithm selected is the finite element method (FE-method). The calculations are carried out using the explicit finite element code LS-DYNA®.

The design of the fuel basket uses the German code KTA 3201.2, and the analysis of the fuel basket follows the sequence: linear elastic stress analysis, followed by load-bearing capacity analysis (if necessary) with ideally elastic-plastic material behavior, evaluation of the deformations of the fuel basket, and then verification of stability.

For all the fuel basket components, the allowable stresses and fictitious yield stresses are valid only for metallic materials, which possess enough deformation capacity (ductility) for the whole range of operating temperatures.

2.15 Mechanical Loads

Vertical Loads - Because of the gap between the canister lid and the fuel basket, thermal restraint in the vertical direction is neglected and the only strain in the vertical direction is caused by the deceleration loads. For the NCT, an inertial load of 30 g is considered. For HAC, an enveloping inertial load of 102 g is considered along with other loads. A linear-elastic analysis is performed. The resulting stresses are shown in table A 1 for the cell sheets, table A 2 for the outer sheets, table A 3 for the shielding elements, and table A 4 for the round segments. The factors of safety are defined in accordance with table 2 using the membrane stresses define in table 1.

Side Loads – For side drop loads, an explicit finite model is used as described in section 6, appendix 2-2. The associated material models and contacts are listed in subsection 6.2 and subsection 6.3, respectively. Different drop orientations are considered, along with the pressure load from the FA. Thermal load is applied during the analysis following the temperature field vs time curve as shown in figure 11.

For the side load calculations under NCT and HAC, a linear-elastic analysis did not yield acceptable results. A subsequent analysis using the load-bearing capacity methodology is adopted as described in section 3.3.2. A safety factor of 1.5 is used for NCT and 1.1 for HAC to demonstrate that a deceleration of 45 g (NCT) and 112.2 g (HAC), respectively, will not cause global structural failure. For HAC limit load analyses, a deceleration up to 140 g is performed to establish safety margins.

Figure A 5 through figure A 13 show the time-histories of the global reaction forces, the global energies for NCT with and without additional thermal load, and for HAC without thermal load, in all analyzed drop orientations. Stable equilibrium states are reached for all required decelerations. For all drop orientations the fuel basket remains stable even under a load of 140g, thus the resulting safety margin is at least 24.8 percent (%). Figure A 14 through figure A 22 shows the deformed structure at maximum deceleration.

Deformation – To show that under maximum loads the sheets will not bear on the FAs, the applicant performed a deformation analysis in which the residual deformations and plastic strains from the cumulative loads is computed under HAC with and without additional thermal load. Figure A 23 through figure A 28 show the time-histories of the global reaction forces and the global energies. In figure A 29 through figure A 34, the deformations of the structure sheets in the drop direction after the intermediate and after the final unloading are depicted for each load case. figure A 35 through figure A 40 show the exaggerated deformation (displacement factor 10 – 20) of the sheets at the point of maximum load and the variation of the internal diameter of the most compressed receptacle over time. A comparison of the minimum value of all receptacles shows that the gap between FAs and surrounding receptacle will not close.

Stability – The applicant, for all drop angles with and without additional thermal load, analyzed the stability of the fuel basket, considering initial imperfections as shown in figure 13, with a required safety factor of 1.1. In the analysis for HAC, the 102g acceleration was increased to 115g. Figure A 41 through figure A 46 show the time-histories of the global reaction forces and the global energies and figure A 47 through figure A 52 show the deformed structure under maximum load. Under all conditions the fuel basket remains stable.

2.16 Evaluation Findings

The structural analysis presented confirms the integrity of the principal structural members of the packaging ITS under NCT and HAC by showing that the performance criteria specified in 10 CFR Part 71 (§§ 71.71 and 71.73) are met and the resulting stresses are within the adopted design code strength limits. For NCT, stress summaries are provided in table 2.6-23 (for the cask), table 2.6-24 (for the canister), table 2.6-25 (for the cask lid bolts) and table 2.6-26 (for the canister lid bolts). For the HAC, stress summaries are provided in table 2.7-33 (for the cask), table 2.7-34 (for the canister), table 2.7-35 (for the cask lid bolts) and table 2.7-36 (for the canister lid bolts). The summary tables document that the factors of safety verify the integrity of the cask and canister along with their ability to maintain leak tightness for the loading requirements specified by the regulations of 10 CFR 71.71 and 10 CFR 71.73.

- F2.1** The staff has reviewed the package structural design description and concludes that the contents of the application satisfies the requirements of 10 CFR 71.31(a)(1) and (a)(2) as well as 10 CFR 71.33(a) and (b).
- F2.2** 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).
- F2.3** The staff has reviewed the lifting and tie-down systems for the package and concludes that they satisfy the standards of 10 CFR 71.45(a) for lifting and 10 CFR 71.45(b) for tie-down.
- F2.4** The staff has reviewed the package description and finds that the package satisfies the requirements of 10 CFR 71.43(a) for minimum size.
- F2.5** 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.
- F2.6** 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.
- F2.7** 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).
- F2.8** The staff reviewed the application and finds that the package was evaluated by subjecting a specimen or scale model to the specific tests, or by another method of demonstration acceptable to the Commission, and therefore satisfies the requirements of 10 CFR 71.41(a).
- F2.9** The staff reviewed the structural performance of the packaging under the normal conditions of transport required by 10 CFR 71.71 and concludes that there will be no substantial reduction in the effectiveness of the packaging that would prevent it from satisfying the requirements of 10 CFR 71.51(a)(1) and 10 CFR 71.55(d)(2).

- F2.10** 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) and 10 CFR 71.55(e).
- F2.11** The staff reviewed the packaging structural performance under an external pressure of 2 MPa [290 psi] for a period of not less than 1 hour and finds that the package does not buckle, collapse, or allow the inleakage of water and therefore satisfies the requirements of 10 CFR 71.61.

Based on its review and findings, the staff concludes that GNS's design for CASTOR® geo69 transportation package has met the regulatory requirements under 10 CFR Part 71.

3.0 THERMAL EVALUATION

3.1 Review Scope and Objective

The objective of this review is to verify that the CASTOR® geo69 package design meets the thermal requirements under normal conditions of transport and hypothetical accident conditions. The staff reviewed the thermal aspects of the CASTOR® geo69 transport cask and the package performance under normal conditions of transport and hypothetical accident conditions and evaluated the package design.

3.2 Description of the Thermal Design

3.2.1 Packaging Design Features

According to SAR section 1.2.2 and section 3.1.1, the CASTOR® geo69 transport cask, which is shipped horizontally, is designed to transport 69 undamaged uranium BWR FA within a basket fabricated from aluminum (with boron constituents) inner sheets and steel outer sheets. The packaging is comprised of the following major components:

- A monolithic ductile cast iron cask body sealed with a bolted cask lid. According to SAR section 1.2.1.1, 2 circumferential rows of neutron shielding moderator columns are placed inside 56 holes bored into the cast iron cask body and moderator discs are placed on the cask bottom and lid. SAR section 1.2.1.12.3 noted that the inner and outer cask surfaces are coated to increase the emissivity to optimize heat transfer and, according to SAR section 2.2.7, the coatings are required to have stability over the entire intended service time (up to 60 years).
- According to SAR section 1.2.1.5, solid aluminum shielding segments, which also aid in conduction heat transfer, are located between the basket corners and the inner canister shell.
- SAR section 1.2.1.1 indicated that radial cooling fins are machined into the outer cask surface (i.e., thus, no weld stresses if fins were welded to the surface). Details of the fins were provided in Drawing 1014-DD-30934 2/8 rev 0.
- SAR section 1.2.1.1 and section 1.2.1.6 indicated that the cask includes trunnions and top and bottom impact limiters made of aluminum housing filled with polyurethane foam, which protects the cask during structural and thermal accident conditions.

- SAR table 7.1-2 and appendix 4.2 provided the cask interior and canister interior backfilled helium pressure (approximately less than 40 kPa at 281 K). SAR section 2.2.5.1 indicated that metallic gaskets seal the cask and canister vessels. Staff notes the backfilled helium is an inert gas that mitigates spent fuel reactions (e.g., oxidation) and has relatively high thermal conductivity that aids in heat transfer.
- SAR section 1.2.1.12.1 noted that the fuel basket axially and radially transfers the decay heat from the loaded FA by the basket's center, bottom and top sheets via conduction and thermal radiation; in addition, the heat is radially transferred to the round aluminum segments and the shielding elements to the canister body.
- SAR section 1.2.1.3 indicated that the canister, which holds the FA within the basket structure, is made of stainless steel and includes a bolted lid arrangement.
- According to SAR section 4.1.1, metal gaskets at the lid are parts of the cask containment boundary. In addition, metal gaskets at the canister lid are parts of the canister pressure boundary.
- Details of the cask, canister, and basket were found in Drawings 1014-DD-30934, Rev 0, 1014-DD-36855, Rev. 0, and 1014-DD-30984, Rev. 1.

SAR section 3.6 and chapter 7 described thermal-related evaluations and operations associated with the package, including the following: loading of spent FA in the canister within the transfer cask located within the spent fuel pool, allowable time between lid placement on the transfer cask and completion of dewatering, vacuum drying operation and its time limit, helium backfilling, and transfer of the transfer cask within the reactor building. These analyses indicated, for example, an approximately 50-hour time limit period between lid placement and completion of dewatering and a completed vacuum drying time limit period of no more than approximately 100 hours. Conditions of these operations included a spent fuel pool water temperature of no more than 52 °C and an ambient temperature during dewatering and helium backfilling of no more than 52 °C.

3.2.2 Codes and Standards

SAR table 1.0-1 and SAR section 2.1.4 described and listed the code and standards used for the design, fabrication, assembly, inspection, and examination of the CASTOR® geo69 package and referenced the ASME BPVC for structural material properties. In addition, SAR section 7 referenced ANSI N14.5 for helium leak testing of the containment boundaries. Standards associated with the package components are discussed in SER section 3.3.2.

3.2.3 Content Heat Load Specification

SAR table 1.2-11, table 1.2-12, and table 3.1-1 provided details of the allowable fuel types in the CASTOR® geo69 cask. SAR figure 1.2-8, figure 1.2-9, and figure 3.1-1 provided the three allowable individual FA loading diagrams and the maximum allowable individual FA decay heat thermal requirements; similar loading diagrams are referenced in certificate of compliance (CoC) Condition 5(b)(1)(e), which make reference to figure 1.2-9 of the application. For example, the thermal requirements TR-1 uniform loading diagram limits the individual BWR FA to decay heat that is slightly higher than 250 W. The loading diagrams indicated that total package decay heats associated with the TR-1, TR-2, and TR-3 thermal requirements varied between approximately 18 kW to less than approximately 18.5 kW. According to SAR section 3.3.2.1, the decay heat in the finite element thermal model is conservatively

overestimated by a few percent relative to the thermal requirements (TR) provided in section 3.1.2. The use of the 18.9 kW decay heat in the thermal models (e.g., TR-2 and TR-3 mentioned in the NCT, HAC, and short-term SAR sections), is more than the 18.5 kW decay heat value mentioned in the SAR.

SAR section 3.3.1.5.4 indicated that the fuel assembly modeled in the thermal analyses considered bounding conditions and conservative assumptions, including low thermal conductivity, short active length, and high peaking factor.

3.2.4 Summary Tables of Temperatures

SAR table 3.3-7 provided the summary of package component temperatures for hot normal conditions of transport. These temperatures were comparable to the structural design temperatures reported in SAR table 2.6-4 and less than the NCT and HAC component allowable temperature limits provided in SAR table 3.2-10. SAR table 3.3-7 showed that the differences in component temperatures were minimal (approximately 2 °C) when comparing results associated with Thermal Requirement FA loading 1, 2, and 3.

In addition, SAR table 3.4-2 provided a summary of the maximum HAC package component temperatures and corresponding time to reach those temperatures after fire initiation; these temperatures were below the allowable values reported in SAR table 3.2-10. SAR figures 3.4-4 through 3.4-8 provided temperature profiles of various package components during the thermal HAC transient. Similarly, SAR table 3.6-6 and table 3.6-7 provided the summary of component temperatures for the vacuum drying and helium backfilling operations. SAR section 3.6.4.4 and section 3.6.5.4 indicated that the vacuum drying and helium backfilling temperatures of the fuel rod, transfer cask lead shield, transfer cask water chamber, basket sheet, and gasket were below their respective NCT allowable temperatures.

SAR section 2.7.4.1 stated that the HAC transient thermal analyses in SAR chapter 3 were used as input to the structural analyses, which indicated the cask and canister could withstand accident loadings.

SAR section 3.3.2.3 and table 3.3-8 indicated a maximum cask surface temperature of 67 °C when exposed to a 38 °C ambient temperature condition without insolation, thus meeting the 10 CFR 71.43 temperature limit of 85 °C (185 °F) for exclusive use shipment.

SAR section 3.3.2.2 stated that, for cold normal conditions, the minimum temperature of the package would be -40 °C (-40 °F) and the SAR indicated a low safe operating temperature of -40 °C (-40 °F) for cask components, as discussed in SER section 3.3.2 below.

3.2.5 Summary Tables of Pressures in the Containment System

SAR sections 4.2.1 and 4.3.1 presented the applicant's analyses for determining the maximum internal pressure within the CASTOR® geo69 cask and canister during normal conditions of transport and hypothetical accident conditions, respectively. SAR section 4.3.1 indicated there are no flammable-related gases or vapors associated with the CASTOR® system. In addition, SAR chapter 2 indicated that the integrity of the canister and cask would be maintained after the NCT tests. According to SAR table 4.2-1, the NCT calculations for pressure within the cask were based on the initial helium backfill gas pressure (per SAR table 7.1-2) and temperature; similarly, the canister pressure during NCT was based on the canister's initial helium backfill gas pressure (per SAR table 7.1-2) and temperature (281 K), fission gas release fraction (up to 30%) and 100% fuel rod (bounding) helium filling gas from 0.03 fuel rod failure fraction, and the

minimum free gas volume within the canister. SAR section 4.2.1 indicated the maximum absolute pressure within the canister was calculated to be less than approximately 75 kPa for high burnup fuel and low burnup fuel. The maximum absolute pressure within the cask based on the initially backfilled helium heating to NCT temperature was calculated as 54 kPa.

SAR section 4.3.1 and SAR appendix 4-1 indicated the thermal fire HAC maximum absolute pressure inside the canister was slightly above 300 kPa based on the input parameters in SAR table 4.3-1; it was reported that this pressure would increase to approximately 375 kPa with an assumed fission gas fraction of 0.3. It was also reported that the cask pressure was approximately 60 kPa, based on no communication between the canister and cask. These pressures are based on the calculation that is a function of backfill helium pressure, backfill helium temperature, gas volume, and gas volume temperature during the fire HAC. In addition, the canister pressure also is a function of 15% (high burnup fuel) and 30% (low burnup fuel) fission gas released and 100% fuel rod (bounding) helium filling gas from 100% fuel rod failure. Staff notes that the structural analyses presented in SAR section 2.7 indicated that the canister and cask had factors of safety greater than one such that cask and canister integrity would be maintained during the NCT and HAC tests.

For the impact HAC scenario without communication between the canister and cask, SAR section 4.3.1 indicated the canister pressure can range from approximately 415 kPa (Impact II with fuel reconfiguration) to approximately 480 kPa (Impact I with no fuel reconfiguration) and the cask pressure would be less than approximately 60 kPa for Impact I and Impact II conditions.

SAR table 2.7-30 indicated that pressures applied for the structural calculations at NCT and HAC were generally consistent with those reported above. It is noted that SAR table 8.1-3 indicated the actual acceptance test pressures for the canister and cask were greater than the calculated minimum hydrostatic test pressures mentioned in SAR section 8.1.4.1.

3.3 Material Properties and Component Specifications

3.3.1 Material Properties

SAR section 3.2.1 provided the thermal conductivity, density, specific heat capacity, emissivity, and absorptivity thermal properties of package component materials. Likewise, SAR appendix 2-6 to document 1014-SR-00001 and SAR section 2.2 provided the thermal properties and thermo-mechanical properties, including thermal conductivity, Poisson's ratio and coefficient of thermal expansion of package component materials (e.g., polyurethane hard foam PUR 320, ultra-high molecular weight low pressure polyethylene (UHMW-PE) neutron moderator). The references associated with the properties listed in the tables were provided in SAR section 2.2 and section 3.2. Similarly, with regards to radiation heat transfer surface properties, SAR section 2.2.4, table 2-22, and table 3.2-8 listed the emissivity values associated with the package's various components. In addition, SAR table 3.2-9 indicated that the solar absorption coefficient of the cask's outer surfaces was chosen as 1. Appendix 2-11 (document 1014-TR-00083 Rev. 0) included the material qualification report for the cask's surface coating properties (i.e., which influence emissivity and thermal performance), including data sheets and testing. SAR section 2.2.7.1 indicated that the coating specification included stability up to 60 years. In addition, SAR section 8.2.5 indicated that each component of the CASTOR® geo69 system undergoes periodic inspection to ensure that component's thermal behavior and the heat dissipation of the package are consistent with the thermal analysis. Likewise, SAR table 7.1-3 stated that fin geometry and cleanliness are inspected and that fins are cleaned as part of package preparations.

3.3.2 Technical Specifications of Components

SAR section 1.0.2 and section 2.2.8 indicated there are no significant chemical or galvanic effects associated with the CASTOR® geo69 package. It mentioned that the canister and package are backfilled with helium, thereby producing an inert environment with a relatively high thermal conductivity.

SAR section 1.2.1 provided details of various components, including metal gaskets and seals, ultra-high molecular weight low pressure polyethylene (UHMW-PE) neutron moderator, and basket neutron absorber material (aluminum and boron carbide metal matrix composite). In addition, section 3.1.2 of appendix 2-9 to 1014-SR-00001 provided metal gasket properties and tolerances.

Details and specifications (e.g., test data of properties) for the above-mentioned important-to-safety components were provided throughout the SAR. These included the following documents: appendix 2-5 of document 1014-SR-00001 provided the property data of the polyurethane foams, appendix 2-7 of 1014-SR-00001 provided the material qualification of the tempered UHMW-PE (neutron moderator material), appendix 2-8 to 1014-SR-00001 provided the material qualification of the basket neutron absorber metal matrix composite material, appendix 2-9 of 1014-SR-00001 provided the material qualification of the metal gaskets, and appendix 2-10 of 1014-SR-00001 provided the material qualification of the General Plastics LAST-A-FOAM FR-3700 series (a rigid, closed-cell, flame retardant foam used as the impact limiter material).

3.3.3 Thermal Design Limits of Package Materials and Components

Temperature limits of package components at NCT and HAC were provided in SAR table 3.2-10. In addition, SAR section 2.6.1.1 and table 2.6-4 indicated a cold environment temperature of -40 °C and a minimum design temperature of -29 °C for metal components of the cask and canister. SAR section 2.6.2 discussed the structural aspects of the package components in the -40 °C environment and showed that factors of safety were greater than one.

Section 3.3 of SAR appendix 2-10 to 1014-SR-00001 (Material Qualification Polyurethan Foam) indicated that the General Plastics Manufacturing Company's LAST-A-FOAM FR-3700 series, which is a rigid, closed-cell, flame retardant foam used as the impact limiter material, has an operating temperature range of -40 °C to 100 °C. Section 6 of appendix 2-10 also addressed the thermal conductivity of the LAST-A-FOAM and its flame retardant properties.

Section 5.1 of SAR appendix 2-7 to document 1014-SR-00001 indicated that moderator rod and plates composed of ultra-high-molecular-weight polyethylene (UHMW-PE) had a service temperature range from -40 °C to slightly greater than 150 °C. SAR section 3.1.1.3 stated that the shielding HAC analysis assumed the moderator material was completely lost. This was a conservative assumption because the moderator material is not destroyed during HAC, as noted in section 4.3 of SAR appendix 2-7, which indicated that thermal degradation of the polyethylene begins at temperatures above 340 °C.

The round segment basket component shielding elements are listed as SB-221 UNS A95454 aluminum, which according to SAR section 3.2.2 have an NCT allowable temperature of approximately 600 °C.

Section 3.1 and table 4 of SAR appendix 2-8 to document 1014-SR-00001 provided thermal property values (e.g., thermal conductivity, density, specific heat) and indicated that the fuel

basket's aluminum-boron metal matrix composite neutron absorber had a service temperature range from -40 °C to less than approximately 305 °C.

SAR section 2.2.5.1 and section 2 of appendix 2-9 to document 1014-SR-00001 (Material Qualification Metal Gaskets) provided the permissible operating temperature range of the important-to-safety containment boundary metal gaskets have a permissible operating temperature range from -40 °C to less than approximately 255 °C. In addition, appendix 2-9 section 4.3.1 and SAR section 2.2.5.1 stated that the tightening plug, protection cover, and blind flange gaskets (small torus diameter) and the cask and canister lid gaskets (large torus diameter) have short-term temperature limits above 350 °C. SAR section 2.2.5.2 noted that the VMQ and FKM elastomeric seals, which are not important-to-safety but are used during package operations (e.g., leakage rate testing), have an operating temperature range from -40 °C to 200 °C.

3.4 Thermal Evaluation Methods

3.4.1 Evaluation by Analyses

The applicant used the ANSYS FEA code (release 2020 R2 and release 17.2) to perform thermal modeling of the CASTOR® geo69 transportation package during NCT and HAC. As noted in SAR section 3.3.1.1, the ANSYS code and its use would fall under the GNS 10 CFR Part 71 quality assurance program, which is applied to the design and analysis of the packaging and its fabrication, assembly, testing, maintenance, and repair.

According to SAR section 3.3.1.2, the NCT finite-element thermal model was 3-D with half symmetry. As noted in SAR figure 3.3-1 and figure 3.3-2, the thermal model included homogenized FA zones, basket sheets, outer sheets, and shielding elements, FA channels, aluminum round segments, canister, cask body, moderator rods and plates, and impact limiter components (e.g., polyurethane foam). It was stated that component dimensions were nominal values from design drawings and that temperature-dependent material properties from SAR section 3.2.1 were applied.

SAR section 3.3.1.3 stated the model included a small gap (less than 1 mm) between the bottom of the canister and bottom of the cask and a small gap (less than 1 mm) between the bottom of the canister and basket. In addition, SAR figure 3.3-3, figure 3.3-4, and figure 3.3-5 provided gaps between basket components. It was noted that the basket sheet effective conductivity considered the effects of the gaps and that convection and radiation heat transfer in the axial gaps were conservatively neglected. The thermal model assumed a radial gap (approximately 5 mm) between the cask and canister and a radial gap (approximately 15 mm) between the canister and basket. SAR section 3.3.1.3 stated that the 6 mm and 15.5 mm gaps were based on ambient temperature conditions and that the gaps would be smaller during NCT because of higher thermal expansions of the inner components compared to the outer components; the applicant's calculations showed that the two radial gaps were slightly reduced from their nominal dimensions at hot NCT. The geometric effect of the smaller gaps at NCT was taken into account by increasing the helium gas thermal conductivity within these gaps by factors corresponding to the reduced gap. The SAR noted that these gaps assumed a concentric arrangement while being transported. SAR section 3.3.3.1, section 3.3.3.2, and section 3.3.3.3 described a sensitivity analysis of the middle portion of the 3-D model (adiabatic boundary conditions were applied to the top and bottom surfaces) to compare a concentric arrangement and a non-concentric arrangement. Results in SAR section 3.3.3.2 and table 3.3-10 showed that some component temperatures were higher (e.g., up to approximately 14 K) for the non-concentric arrangement, including the cask surface, canister surface, round aluminum

segments, and moderator rods. The increase in component temperature was then applied to the concentric result from the 3-D model described above; component temperatures in the 3-D model were not decreased if the middle portion of the 3-D model results showed lower temperatures.

The SAR stated that the dry air between the cask and impact limiter, dry air around the bottom of the moderator plate, and helium within the canister were stagnant and, therefore, heat transfer was via heat conduction and thermal radiation; emissivity values of surfaces were listed in SAR table 3.2-8. Convection heat transfer was ignored between the gaps and within the helium atmosphere inside the canister and cask.

The outer boundary conditions included insolation, buoyant free convection (i.e., no wind), and radiation exchange between the outer cask and ambient. According to SAR section 3.3.1.7.2, the insolation applied to the non-horizontal flat surfaces, curved surfaces, and other surfaces was 200 W/m², 400 W/m², and 800 W/m², respectively; no insolation was applied to the base of the package. The SAR analysis assumed that these were constant values (i.e., applied over 24 hours), which is conservative (a factor of 2) relative to the conditions described in 10 CFR 71.71. The convection heat transfer was modeled by using a Nusselt correlation for free convection of a vertical surface (i.e., diameter of the cask was large relative to the boundary layer thickness), which was a function of the Grashof number and Prandtl number; further details were described in SAR section 3.3.1.7.1. SAR section 3.3.1.7.1 and section 3.3.1.2 stated that thermal radiation energy heat transfer was modeled using a Stefan-Boltzmann relation and the ANSYS radiation matrix method.

SAR section 3.3.1.4 stated the radial fins machined into the outer surface of the CASTOR® geo69 transportation package were not explicitly modeled in the 3-D thermal model. Rather, a surface enlargement factor, which was a function of fin area and fin efficiency, was calculated; details were derived in SAR section 3.3.1.4. The convection heat transfer coefficient based on the convection heat transfer coefficient correlation for an unfinned surface was multiplied by the surface enlargement factor. The SAR noted that the surface enlargement factor was conservatively not applied when calculating radiant heat transfer from the package to the ambient.

Likewise, although SAR section 3.3.1.4 indicated the insolation thermal input effect of the increased area due to the fins was not modeled, it was stated this was partly compensated by the conservative assumption of applying the 10 CFR 71.71 insolation values as a steady-state condition (i.e., 24-hour period versus the 12-hour period) and the assumption of an outer surface absorptivity value of one.

The bounding axial heat load profile imposed along the length of the FA was chosen after calculating the heat load distribution for all types of FA and for various decay times. As noted in SAR section 3.3.1.5.3, the most unfavorable assumptions from the different FA were combined as a bounding modeled FA (e.g., highest power density, lowest radial thermal conductivity, highest peaking factor).

SAR section 3.3.1.5 described the methodology for calculating effective thermal properties (thermal conductivity, density, and specific heat) of the FA in the radial and axial directions. The effective axial thermal conductivity and effective axial density properties were proportional to the summed product of area of the component in a radial cross-section and thermal property of each component divided by the summed area; the effective axial specific heat was proportional to the summed product of area of the component in a radial cross-section, specific heat, and

density of each component divided by summing the product of each component's density and area.

SAR section 3.3.1.5.2 described the method for determining the effective thermal conductivity in the radial direction. This property was determined by constructing both a detailed two-dimensional (2-D) finite element (FE) model of a single FA cross-section and a simplified homogeneous cross-section 2-D FE model, as noted in SAR figure 3.3-7. The detailed model included the pellets, cladding, water rods, fuel channels, surrounding basket sheets, and gas atmosphere. The simplified model consisted of homogenized material that modeled the effective thermal conductivity associated with the fuel, water rods, and gas atmosphere between the fuel rods. The models considered the heat production inside the pellets as well as the heat conduction inside the pellets, cladding, water rods, and gas atmosphere inside the FA cavity. In addition, radiant heat transfer was modeled between the cladding surface and water rods, inner and outer surfaces of the fuel channels, and the inner surface of the basket sheets; radiation heat transfer inside the water rods and between the fuel pellets and the inside cladding surface of the detailed model were conservatively neglected.

The next step in modeling the radial effective thermal conductivity was to set a particular temperature (e.g., 150 °C) to the outer boundary of the 2-D detailed model. The decay heat of the fuel rods was set at a small value in order to achieve a relatively small peak fuel temperature in the middle of the FA (e.g., 1 °C). The same decay heat and outer boundary temperature was imposed for the simplified 2-D FA model and the effective thermal conductivity was iteratively adjusted until the simplified 2-D model temperature profile matched the detailed 2-D model temperature profile. This process was repeated for a temperature range from 50 °C to 400 °C to determine the effective thermal conductivity as a function of temperature. Staff finds that the methodology was sufficient with the current decay heats considering the approximately 185 °C margin of cladding with its allowable value.

3.4.1.1 NCT sensitivity analyses

SAR section 3.3.3 presented a number of sensitivity analyses associated with the NCT model and boundary conditions. According to SAR section 3.3.3.1, these sensitivity analyses considered the baseline condition and comparative sensitivity conditions using an axial section of the NCT 3-D model (e.g., the axial section at the package half-length comprised of two layers of basket sheets); the difference in component temperatures between the two models was then added to the full 3-D NCT model baseline temperature. The baseline model assumed a concentric alignment of the basket, canister, and cask with gaps between the above-mentioned components based on nominal dimensions. Recognizing that the CASTOR® geo69 package is transported horizontally, which would result in an eccentric arrangement of the basket, canister, and cask with larger gaps located on the top, the comparative sensitivity model was based on small helium gaps (less than 1 mm) between the basket, canister, and cask at the eccentric bottom. Differences in component temperatures between the two axial section models provided in SAR table 3.3-10 showed that the cask surfaces, moderator rod components, and the basket's aluminum round shielding segments had greater maximum temperatures for the eccentric gap model whereas canister walls, fuel channels, and basket sheets had lower maximum temperatures. SAR section 3.3.3.3 further analyzed the impact of contacts between additional components, such as between round segments and between basket sheets. The results of SAR table 3.3-12 showed even higher maximum temperatures of moderator rods, accessible surface temperature, cask surfaces, basket's aluminum round shielding segments and outer sheets from contact due to radial displacement contact points. These higher sensitivity temperatures continued to be below allowable temperatures.

SAR section 3.3.3.4 presented the NCT sensitivity analysis for different axial gaps between basket sheets of the NCT 3-D FE model described in SAR section 3.3.1, which showed that corresponding effective axial thermal conductivity values varied greatly (often by a factor of 6 or greater). SAR table 3.3-14 showed that these large variations in effective axial thermal conductivity resulted in temperature changes that varied up to approximately 15 °C; this confirmed that the radial direction is the dominant heat transfer path from the spent fuel to the outer cask surface. Although the fuel rods and basket sheets showed higher temperatures by approximately 15 °C, both components continued to have large margin with allowable values at NCT.

SAR section 3.3.3.5 presented a sensitivity analysis for the gaps between the basket and canister and between the canister and cask of the NCT 3-D FE model. SAR table 3.3-15 showed maximum package component temperatures between the baseline model (nominal gaps) and maximum radial gaps that account for the most unfavorable manufacturing tolerances. Although most components showed increases of approximately 4 °C, the fuel rods, basket sheets, round segments, and shielding elements increased by approximately 10 °C. However, these higher component temperatures continued to have relatively large margin (e.g., approximately 175 °C for fuel rod) with their allowable temperatures.

SAR section 3.3.3.6 presented a mesh sensitivity analysis at NCT using the axial section model described in SAR section 3.3.3.1. Four mesh sizes were analyzed such that the number of elements increased by a factor of 2.2, 7.1, and 14.5 from the baseline mesh. Most of the components resulted in less than 1 °C temperature increases, although the fuel rods, fuel channels, and basket sheets increased by approximately 3 °C. However, these increased temperatures continued to be well below the allowable temperatures (e.g., fuel rod with approximately 180 °C margin). With regards to numerical heat flow residual convergence, SAR section 3.3.2.4 presented results that showed the thermal steady-state NCT analysis would converge to a near zero residual after a few iterations.

3.4.1.2 HAC sensitivity analyses

A number of sensitivity analyses were performed by the applicant for thermal-related HAC analyses. For example, according to SAR section 3.4.2.2, the basis of the fire HAC thermal analysis was a transient numerical scheme. SAR figure 3.4-9 through figure 3.4-11 showed that each time-step calculation satisfied the ANSYS code's convergence criterion.

Another convergence sensitivity analysis was described in SAR section 3.4.3.2, which discussed the effect of different time stepping schemes and step sizes for the transient fire HAC analyses. Results showed that a user-defined manual time stepping scheme with smaller timesteps (which tend to improve numerical accuracy) used for HAC fire calculations resulted in slightly higher temperatures compared to the ANSYS code's automatic time stepping scheme.

SAR section 3.4.3.1.1 discussed the thermal impact from compressed impact limiters due to the HAC drop tests and noted that, although the geometric finite element model was not changed, the effect of the compressed impact limiter was modeled by increasing the impact limiter polyurethane foam's thermal conductivity by a large factor. This larger value accounted for the shorter distance between heat source and ambient and the higher density of the compressed foam. The plots of temperature as a function of time for many package components (e.g., gaskets) indicated that, for many of the components, the improved heat transfer outward from the package after the fire ended, slightly outweighed the improved heat transfer into the package during the fire, as indicated by their slightly lower temperatures during most of the transient compared to an undeformed package.

SAR section 3.4.4 discussed a 0-dimensional transient energy balance calculation for approximating the surface temperature of the cask during the fire phase of the HAC transient, purportedly as a check of the finite element model. The 0-dimensional calculation necessitated simplifying assumptions to perform the analysis. Results showed a mean cask shell temperature at the end of the 30-minute transient that was nearly 20% greater than the finite element result. Staff notes that the calculation did not consider the effect of the decay heat's thermal input. In addition, although the calculation considered a comparison of the package surface temperature with the two models, it does not necessarily demonstrate the appropriateness of the entire finite element model.

Although the thermal analyses considered the increased convection heat transfer effects due to the presence of the fins during NCT and the convective effect of the HAC fire, SAR section 3.4.3.3 noted that the effect of the increased package surface area due to the fins on the radiant energy directed towards the package did not consider an effective surface amplification factor because of similar temperatures between opposing fins and that the radiant energy from the fire temperature was directed toward the package outer surface, thereby conservatively not considering the thermal resistance of the fins which would tend to reduce the temperature at the cask surface. Section 3.4.3.3 then discussed a sensitivity analysis of increasing the effective emissivity by approximately 15% (i.e., a factor of 1.15). SAR table 3.4-7 indicated that fuel rods and gaskets increased in temperature by approximately 2 °C, although some components towards the package surface (e.g., outer row of moderator rods) increased in temperature by over 20 °C. Staff notes that the impact of including the effect of increased surface area of the fins (an approximate surface area factor of 2.5) would result in an increase in PCT and O-ring temperatures in amounts greater than 2 °C. This is because the radiative thermal input towards the package is proportional to the exposed surface area (e.g., Stefan-Boltzmann component of an energy balance equation). In addition, the fire detailed in 10 CFR 71.73(c) is for a fully engulfing hydrocarbon fuel/air fire, which generally includes soot and is opaque such that the fins and package surface would be exposed to the fire's high temperature (rather than an opposing fin). Therefore, with regards to potential increases in component temperatures due to increased radiant energy towards the package, staff notes that SAR table 3.3-7 and table 3.4-2 showed that fuel rod temperatures increased from 215 °C to 237 °C between NCT and fire HAC; a difference of 22 °C with a 333 °C margin between the HAC temperature and its allowable temperature. Similarly, the O-ring temperature approximately increased from 93 °C to 127 °C between NCT and fire HAC; a difference of 34 °C with a 243 °C margin between HAC temperature and its allowable temperature. Increasing the difference in NCT and fire HAC component temperatures of a fuel rod or an O-ring by a 2.5 factor would still result in large margins between HAC temperatures and allowable temperatures. Staff notes that this simplified assumption is considered because of the above-mentioned large margins for this package. In addition, although the SAR's discussion indicated that the effect of not modeling the increased radiant energy to the fins was negligible, this may not be appropriate in future submittals, especially if there are content and package design changes that reduce temperature margins, such that a more accurate calculation for determining the effect of the increased fin area on radiant energy input to the package may be necessary.

3.4.1.3 Thermal evaluation of fuel rod failure for NCT, HAC fire, HAC impact

SAR section 3.5 presented the thermal analyses for NCT and accident conditions considering the failure of 3% and 100% of fuel rods, respectively. SAR section 3.5.1 discussed the calculations and values (reported in SAR table 3.5-1) for determining the thermal conductivity of the canister's gas mixture consisting of helium from the backfilled canister, fuel rod helium backfill, and the ruptured fuel rods' xenon fission gas (the fuel rod's krypton fission gas

component was conservatively associated with xenon, which has a lower thermal conductivity). These gas thermal conductivity values were then used to calculate the homogenized basket and canister region's effective axial and radial thermal conductivity, density, and specific heat, as reported in SAR table 3.5-2.

SAR section 3.5.2.1 stated that the NCT fuel rod failure analysis was based on the previously discussed NCT thermal FE model using the above-mentioned thermal properties. Results showed that fuel rod, moderator components, gasket, and basket maximum temperatures were only slightly greater (e.g., fuel rods were 4 °C higher) than the NCT results that did not consider fuel rod failure; all components continued to be well below allowable temperatures. For example, the maximum fuel rod temperature was reported as 219 °C, which is 181 °C less than the 400 °C allowable temperature.

SAR section 3.5.6 discussed another NCT thermal analysis for fuel rod failure but also included the effect of some fuel reconfiguration. Unlike the assumption for the second HAC impact scenario in which the fuel particle packed bed was adjacent to the lid (i.e., to maximize temperatures at the gaskets) as discussed below, this model assumed reconfigured fuel particles were located in the gaps from the canister's inner lateral surface to the upper edge of the aluminum shielding round segment (i.e., the particles accumulated by gravity into the lower section of the canister's NCT horizontal orientation, as shown in SAR figure 3.5-8). The effective thermal conductivity associated with the reconfigured fuel was lower than the intact FA effective thermal conductivity. SAR table 3.5-14 compared the calculated package component temperatures with the baseline hot NCT package component temperatures and showed there was little difference in temperature values. For example, the maximum temperature difference was observed for the canister lid gasket, which showed a 2.2 °C increase in temperature.

SAR section 3.5.3 discussed the analysis for the HAC fire condition with 100% fuel rod failure assuming 15% fission gas release, resulting in a canister gas atmosphere of approximately 56.5% helium and 43.5% xenon. The transient numerical model was the same as the HAC fire condition except for using the modified canister gas properties and modified effective homogenized FA thermal properties. Results were presented in SAR figure 3.5-2 and showed that the hottest fuel rod was 286 °C after 22 hours. Likewise, SAR section 3.5.3.2 noted the highest gasket temperature of 132 °C was over 100 °C below the allowable normal condition temperature and was over 200 °C below the short-term allowable accident temperature.

SAR section 3.5.5 discussed two analyses associated with a HAC impact condition. The first HAC impact scenario used the geometric thermal model discussed in SAR section 3.4 for the HAC fire but was based on a 100% fuel rod rupture assumption (without fuel reconfiguration) with approximately 37% helium and 63% xenon gas mixture in the canister, as presented in SAR table 3.5-1 and table 3.5-2, discussed above. The results presented in SAR section 3.5.5.2, figure 3.5-6, and table 3.5-12 showed that the maximum fuel rod temperature was 293 °C and gasket temperatures were over 100 °C below the allowable normal condition temperature and over 200 °C below the short-term allowable accident temperature.

For the second HAC impact scenario, the thermal model was modified by assuming fuel reconfiguration. As presented in SAR figure 3.5-5, the top portion of the canister (i.e., near the lid gaskets) was filled with fuel particles with the remaining canister length without fuel particles. This concentrated the decay heat into a smaller volume resulting in a locally high-power density. SAR section 3.5.5.1 presented the assumptions associated with the analyses, including volume fraction of fuel pellets and power density in the active FA zone, volume fraction of fuel packing and power density in the inactive fuel assembly zone, the fraction of helium and fuel pellets in

the homogenized active FA zone, and the percentage of fuel particles and gas mixture in the lid gap area. The assumptions conservatively assumed a model with a decay heat greater than 19 kW. SAR table 3.5-9 and table 3.5-10 presented the above-mentioned conditions that were used to calculate an effective thermal conductivity of the gaps and packed bed arrangement of the active FA zone and inactive FA zone. The effective thermal conductivity values presented in SAR table 3.5-7 were less than the intact FA effective thermal conductivity values presented in the SAR table 3.3-5. SAR figure 3.5-7 and table 3.5-13 provided the resulting temperatures for the second HAC impact scenario which showed the maximum gasket temperature of 175 °C was approximately 75 °C below the allowable normal condition temperature and approximately 200 °C below the short-term allowable accident temperature. Although the above-mentioned assumptions (and similar assumptions for reconfiguration analyses) had uncertainties associated with them and, therefore, staff cannot make definitive findings on the assumptions, the large margin between gasket temperature and allowable accident temperature (e.g., approximately 200 °C) indicates there is reasonable assurance that gaskets would continue to function under reconfiguration conditions.

3.4.2 Thermal HAC Model and Analysis

SAR section 3.4 discussed the thermal analyses associated with the HAC fire test associated with 10 CFR 71.73; staff notes that SAR section 2.8 and CoC Condition 10 stated that air transport is excluded and not authorized for the CASTOR® geo69 package. The thermal analyses utilized the 3-D FE NCT thermal model described above as the initial basis for the thermal HAC analysis. SAR section 3.4.1.1 mentioned that the thermal heat capacity of the fins (which are not explicitly modeled) was indirectly modeled by locally increasing the density of the transportation cask's outer periphery to account for the fin mass, which has an impact on the package's thermal mass during transient analyses. Staff finds that locally increasing the density to account for the fin mass is a reasonable approximation considering the large temperature margins for the package.

SAR section 3.4.1.2 and section 3.4.1.3 presented the initial and boundary conditions of the transient HAC thermal analysis. The initial conditions were the steady-state results of the hot NCT analysis with insolation and a 38 °C ambient temperature, as discussed earlier. Boundary conditions during the 30-minute fire included a fully engulfing pool fire condition at 800 °C with a fire emissivity and surface emissivity of 0.9. The fuel source extended horizontally at least 1 m but not more than 3 m from the transportation cask surface. Convection heat transfer and radiation heat transfer from the fire was applied on transportation cask outer surfaces. The convection heat transfer coefficient was based on a Nusselt convection correlation, which was a function of the surrounding air Prandtl number, Reynolds number, thermal conductivity, and cask diameter. The forced convection heat transfer coefficient, which was based on a 5 m/s gas velocity, was calculated as less than 10 W/m²-K and conservatively increased by over 50 percent to account for uncertainties of combustion gas composition. Within the outer finned zone area (modeled with a higher density, as discussed above), the effective surface enlargement factor of 2.2 was applied to the heat transfer coefficient. Radiation heat transfer to the package was applied using an effective radiant heat transfer coefficient based on the Stefan-Boltzmann law. In addition, insolation was applied during the 30 minute fire. SAR section 3.4.1.4 indicated that the thermal analysis continued nearly 40 hours after the 30 minute fire in order for package temperatures to reach their peak values. The 38 °C ambient temperature and insolation values applied during the hot NCT conditions, as well as the natural convection and radiation heat transfer correlations, were also applied during the cool-down period. The emissivity of the package surface was increased to 0.8 after the fire in order to account for soot.

In addition to the above-mentioned HAC thermal analysis based on an undeformed NCT thermal model, SAR section 3.4.3.1.1 discussed the thermal impact from compressed impact limiter which would occur after the HAC drop tests. For example, although the geometric FEA model was not changed, the effect of the compressed impact limiter was modeled by increasing the impact limiter polyurethane foam's thermal conductivity by a large factor. This larger value accounted for the shorter distance between heat source and the surface and the higher density of the compressed foam. The plots of temperature as a function of time for many package components (e.g., gaskets) indicated that the improved heat transfer outward from the package after the fire ended, slightly outweighed the improved heat transfer into the package during the fire, as indicated by the slightly lower temperatures of components during most of the transient compared to an undeformed package.

3.4.3 Short-Term Operations Analyses

SAR section 3.6 described the thermal models and analysis results of short-term operations inside the reactor facility, including fuel loading of the canister inside the transfer cask under water in the spent fuel pool, the dewatering, vacuum drying, and helium backfilling sequence of the canister, and the transfer of the loaded transfer cask inside the reactor building. The following discussion describes the methods of these analyses and their results.

The thermal properties of the transfer cask components and the conditions of the short-term operations were provided in SAR section 3.6.1. SAR table 3.6-1, table 3.6-2, and table 3.6-3 included the thermal conductivity, density, specific heat capacity, and emissivity of the relevant components and fluid. The effective thermal conductivity of the water filled volumes in the transfer cask's inner and outer water chambers was calculated based on the radial gap widths and free convective heat transfer coefficients associated with the inner and outer water chambers. SAR section 3.6.4.2 noted that a gap (less than 1.5 mm) was modeled on either side of the lead shield of the transfer cask to model thermal contact resistances that could occur during fabrication. The SAR presented the results of a sensitivity analysis of their assumed water free convection coefficient that indicated the variation of the inner and outer water chamber's convection heat transfer coefficients by 30% resulted in 1 K temperature differences across the water gaps. Therefore, the uncertainty associated with the water's heat transfer coefficient would not introduce significant changes in the temperature results associated with the water-filled transfer cask operations described below.

SAR section 3.6.2 described the underwater fuel loading operation, in which the transfer cask and canister loaded with FA are filled with demineralized pool water (per SAR section 1.2.4.3). Since the transfer cask lid is not installed, there is free exchange between the water inside the canister and transfer cask and the pool water at 52 °C. The thermal analysis to estimate cladding temperature was an energy balance that related the decay heat of a FA (i.e., maximum fuel assembly decay heat associated with thermal requirement 2) and the previously assumed free convection heat transfer coefficient in water; the calculation found the FA temperature was slightly higher than the pool water temperature.

SAR section 3.6.3 described the short-term operation of having a water-filled cask, loaded with the canister filled with water and FA (total decay heat associated with thermal requirement 2), when the cask lid is mounted; this situation occurs prior to the dewatering process. An adiabatic boundary condition calculation was performed to determine the time period for the transfer cask to increase in temperature from the 52 °C initial temperature to 100 °C, which is the boiling temperature of water at 1 bar. SAR table 3.6-4 provided the mass, specific heat, and heat capacity of the components associated with the filled transfer cask and content. The calculation

result indicated that temperatures would increase to 100 °C within approximately 50 hours. The applicant indicated this time period was conservative because of the adiabatic assumption, recognizing that there actually would be heat transfer from the transfer cask while it was immersed in water.

SAR section 3.6.4 described the vacuum drying thermal analysis, in which it was assumed water was removed from the canister and the annular region between the canister and transfer cask and replaced with water vapor. As noted in SAR table 3.6-3 and the RAI response submittal (dated 28/06/2024, Enclosure 2 to letter T1213-CO-00021), the vacuum drying condition assumed water vapor at 0.01 bar with SAR table 3.6-5 presenting the material properties (e.g., radial heat conductivity, axial heat conductivity, density, specific heat capacity) of the homogenized FA zones based on pure water vapor as the filling gas. SAR section 3.6.4.2 noted that the finite element model for the canister, fuel basket, and FA was the same as in SAR section 3.3.1, although it included transfer cask details rather than the storage cask, including a gap on either side of the lead shield to account for gaps and thermal contact resistances due to fabrication. Specifically, the three-dimensional ANSYS finite element thermal circumferential half-symmetry model included transfer cask components, including a mounted transfer cask lid, lead shield, and water chambers; figure 3.6-1, figure 3.6-2, and figure 3.6-3 listed the modeled components. SAR section 3.6.4.3 discussed the boundary conditions during vacuum drying, including a 52 °C temperature within the reactor facility. The free convection heat transfer correlation (a function of Grashof number and Prandtl number) for the vertical cylindrical surface of the transfer cask was listed and it was stated the resulting heat transfer coefficient was reduced by 10% to account for shielding or scaffolding; staff notes that although no basis for the 10% value was given, the cladding temperature had nearly 80 °C margin with its allowable temperature of 400 °C such that uncertainties in the reduction value would not have significant effects. In addition, a free convection correlation was considered to model the heat transfer from the transfer cask's horizontal lid. The Stefan Boltzmann radiation heat transfer law was applied to the external transfer cask and the emissivity values of the external surfaces were provided in SAR table 3.6-2. The bottom side of the cask included an adiabatic boundary condition.

Results of the vacuum drying thermal analyses showed that the maximum temperature of the fuel rods was 323 °C, which is less than the 400 °C allowable. The maximum lead shield temperature was 117 °C, which is 210 °C less than the 327 °C allowable temperature. The highest gasket temperature was 128 °C, which is approximately 120 °C less than the maximum continuous allowable temperature. The basket sheet did not exceed the 300 °C allowable temperature (reported in SAR table 3.2-10) up to the vacuum drying time limit (approximately 100 hours).

SAR section 3.6.5 described the analysis using the numerical model described in SAR section 3.6.4.2 in which the canister is backfilled with helium and the space between the canister and transfer cask is filled with dry air. The boundary conditions are similar to those described in SAR section 3.6.4.3, although the reduction in the heat transfer coefficient factor is removed because scaffolding is removed. SAR table 3.6-7 showed that temperatures were below allowable values; for example, fuel rod temperatures of 274 °C were below the 400 °C allowable temperature.

SAR section 3.6.6 and section 3.6.7 stated that component temperatures during transfer of the contents inside the reactor building and loading of the canister into the transport cask are bounded by the results provided in SAR section 3.6.5 for helium backfilling near the loading pool because the conditions away from the pool are more favorable for heat removal. In addition,

SAR section 3.6.7 stated that polyethylene shielding blocks, which are present during canister loading into the transport cask, are below allowable temperatures.

3.4.4 Evaluation of Accessible Surface Temperature

SAR section 3.3.2.3 indicated that the accessible surface was the outer package surface. Using the NCT thermal model with thermal requirement 2 loading pattern, discussed earlier, the surface temperature at NCT with no insolation was calculated to be 67 °C, which is less than the 85 °C limit described in 10 CFR 71.43(g) for exclusive use transport.

CoC condition 7 notes that an evaluation would be performed to ensure the barrier does not result in package temperatures above allowable values due to impeded heat transfer if a personnel barrier is placed over the package to meet dose requirements.

3.5 Thermal Evaluation under Normal Conditions of Transport

3.5.1 Heat and Cold

SAR section 3.3.2 presented the results of the steady-state NCT calculations using the NCT thermal model described earlier. The hot NCT results, presented in SAR table 3.3-7, showed component temperatures below allowable values. For example, maximum fuel temperature was reported as 215 °C, which indicates a 185 °C margin with the 400 °C limit. The highest temperature of the moderator components was 120 °C, which is below the 160 °C allowable limit. Likewise, the highest gasket temperature was 107 °C, which is approximately 140 °C below the maximum allowable temperature.

SAR section 3.5.2 presented the NCT results assuming the presence of SAR table 3.5-1 gas constituents in the canister from released gas of failed fuel rods (e.g., xenon). This model was the same as the previously discussed NCT model with intact fuel rods except for the different thermal properties of the canister filling gas. The impact from the filling gas properties were small. For example, it was reported that the fuel rod temperature was increased to 219 °C compared to the NCT baseline value of 215 °C.

SAR section 3.5.5 presented the results from the NCT 3% fuel rod failure analysis with fuel release. As noted earlier, this model assumed the canister-filled cask was in the horizontal position during transport. As a result, the released fuel particles accumulated in the direction of gravity along the canister's axial length, as shown in SAR figure 3.5-6. The packed fuel particle zone with an assumed porosity was modeled with the effective thermal conductivity reported in SAR table 3.5-6. SAR section 3.5.5.2 and table 3.5-10 presented the temperatures of canister and cask components, which showed negligible differences compared to the above-mentioned temperatures assuming no fuel rod failure.

SAR section 3.3.2.2 indicated that the cold condition assumed all components were at -40 °C. As noted in SER section 3.3.3 above, the various sections of the SAR indicated that package components have a -40 °C allowable temperature, including containment boundary gaskets, moderator components, and UHMW PE neutron moderator.

3.5.2 Maximum Normal Operating Pressure

SAR section 4.2.1 presented the inputs and analyses for determining maximum internal pressures within the CASTOR® geo69 cask body and canister. The pressure calculation for the canister and cask considered backfill gas pressures and, for canisters, 3% fuel rod rupture with

15% and 30% fission gas released into the canister using the conditions presented in SAR table 4.2-1. SAR section 4.2.1 also noted that radiolysis would not contribute to the results. SAR chapter 2 indicated that the integrity of the canister and cask would be maintained after the NCT tests. Based on the above, results showed that the maximum absolute pressure within the transport cask was 54 kPa and that the maximum absolute pressure within the inner canister containment was less than approximately 75 kPa for high burnup fuel and low burnup fuel). These reported pressures are below the design pressures reported in SAR table 2.7-30 and are below atmospheric pressure.

3.5.3 Maximum Thermal Stresses

SAR section 3.3.5 provided the nominal NCT radial gap width between the canister and cask is and the radial gap width between the canister and basket sheets; these gaps were slightly reduced for hot NCT such that there is no contact between the components in the radial direction. In addition, it was noted that the nominal axial gap width between the canister and basket as well as the nominal axial gap width between the canister and cask lid were slightly reduced for hot NCT, thus indicating there are no axial restraint forces during NCT. SAR section 3.3.5 noted that additional discussion of thermal stresses was presented in the Structural chapter of the SAR.

3.6 Thermal Evaluation under Hypothetical Accident Conditions

3.6.1 Conditions for the steady-state and transient analyses

SAR section 3.4 discussed the transient fire thermal analysis for evaluating the package with a decay heat (thermal requirement 2 per SAR section 3.3.2.1) less than 19 kW under the fire hypothetical accident condition. The initial conditions were based on the package at hot NCT conditions (e.g., 100°F ambient with insolation). Regarding boundary conditions, both convection and radiation heat transfer from the package to the ambient were modeled, as discussed in SAR section 3.4.2, above. Similarly, SAR section 3.5.3 and section 3.5.4 discussed the conditions and thermal analysis of a fire HAC thermal analysis that assumed 100% fuel rod failure with fuel rod fill gas release and 0.15 and 0.3 fuel rod fission gas release fraction, respectively, into the canister. The same decay heat and transient boundary conditions discussed in SAR section 3.4 also were applied to the fire HAC that assumed failed fuel rods. SAR section 3.5.5 discussed the thermal analysis and results of HAC impact conditions that assumed 100% fuel rod failure with fuel rod fill gas release and 0.35 fuel rod fission gas release fraction into the canister; the Impact I condition assumed the failed fuel rods maintained their form whereas Impact II condition assumed fuel reconfiguration with active and inactive FA zones. Based on the thermal contours of SAR figure 3.5-1 (NCT with fuel rod failure) and SAR figure 3.5-6 (HAC with fuel rod failure due to impact), the impact accident condition analyses assumed boundary conditions (e.g., ambient temperature) similar to NCT conditions.

3.6.2 Maximum Temperatures and Pressure

SAR section 3.4.2, table 3.4-2, and figure 3.4-2 through figure 3.4-8 presented the results of the thermal fire HAC condition. Specifically, SAR figures 3.4-4 through 3.4-8 showed that the transient analyses were performed beyond the peak temperatures when temperatures were decreasing over time. Results showed that the maximum temperature of the cladding of 237 °C was reached 17 hours into the HAC transient; this temperature is below the 570 °C allowable temperature with a 333 °C margin. In addition, the 127 °C maximum gasket temperature is less than the maximum allowable value with approximately a 240 °C margin. Similarly, SAR table 3.5-4 showed that fuel rod temperatures of 286 °C (reached 22 hours into the HAC

transient) and maximum gasket temperatures of 132 °C for the HAC fire condition that assumed 100% fuel rod failure (without fuel reconfiguration) were below allowable values. SAR table 3.5-8 indicated that with a 0.35 fission gas release fraction, fuel rod temperatures increased to 303 °C and maximum gasket temperatures increased to 135 °C.

For the HAC impact scenario I (without fuel reconfiguration) and HAC impact scenario II (with fuel reconfiguration), SAR table 3.5-12 and table 3.5-13 showed that canister and cask lid gaskets ranged from 94 °C and 175 °C, which were below the allowable temperature for NCT by at least 75 °C and the short-term HAC allowable temperatures by at least approximately 200 °C.

Regarding pressure, SAR section 3.5 stated the mechanical analyses in SAR chapter 2 showed the canister's integrity remained for NCT, HAC fire, and HAC impact. Specifically, SAR section 2.7 reported that factors of safety for the canister and cask were greater than one for NCT and HAC tests; therefore, there would be no fission gas released from the canister interior into the cask cavity. The pressures in the canister and cask for the fire and impact HACs were presented in SER section 3.2.5. SAR table 2.7-30 indicated that pressures applied for the structural calculations were generally consistent with the values presented in SAR section 4.2.1 and section 4.3.1, as noted in SER section 3.2.5. SAR section 8.1.4.1 provided the minimum hydrostatic test pressures based on the canister and cask pressures reported in SAR section 4.2; SAR table 8.1-3 show that actual acceptance test pressures are higher than the minimum calculated values. Further structural discussion on canister and cask pressures is provided in the SER Structural Evaluation.

3.6.3 Maximum Thermal Stresses

SAR section 3.4.5 and figure 3.4-24 described the changes in axial and radial gaps associated with the fire HAC and found that axial gaps between the canister and basket sheets and between the cask and canister would be slightly reduced from NCT gap dimensions; likewise, radial gap dimensions would be slightly reduced from NCT gap dimensions and, therefore, the gap widths would be sufficient to prevent restraint forces. The SAR noted that discussion of thermal stresses due to temperature gradients was provided in the SAR Structural chapter.

3.7 Computer Program Description

SAR section 3.3.1.1 and the response to RAI 3-9 (Enclosure 2 to letter T213-CO-00019, dated 25/08/2023) indicated that the thermal analysis was performed using the ANSYS Mechanical finite element analysis code (Release 2020 R2 and Release 17.2), which was verified and validated by ANSYS for quality assurance. Similarly, SAR section 3.7 included information on the validation of the ANSYS versions 17.2 and 2020 for thermal calculations. In addition, SAR section 1.0 stated that the thermal analysis (i.e., a design activity) of the CASTOR® geo69 structures, systems and components (SSCs) are performed in accordance with the U.S. NRC approved Quality Assurance Program.

3.8 Evaluation Findings

- F3.1** The staff has reviewed the package description and evaluation and has reasonable assurance that the information provided with associated CoC conditions satisfies the thermal requirements of 10 CFR Part 71.
- F3.2** The staff has reviewed the material properties and component specifications used in the thermal evaluation and has reasonable assurance that the information provides sufficient basis for evaluation of the package against the thermal requirements of 10 CFR Part 71.

- F3.3** The staff has reviewed the methods used in the thermal evaluation and has reasonable assurance that the models are described in sufficient detail to permit an independent review, with confirmatory calculations, of the package thermal design.
- F3.4** The staff has reviewed the accessible surface temperatures of the package, as it will be prepared for shipment, and has reasonable assurance that the requirements of 10 CFR 71.43(g) for packages transported by exclusive-use vehicle have been satisfied.
- F3.5** The staff has reviewed the package design, construction, and preparations for shipment and has reasonable assurance that the package material and component temperatures will not extend beyond the specified allowable limits during normal conditions of transport consistent with the tests specified in 10 CFR 71.71.
- F3.6** The staff has reviewed the package design, construction, and preparations for shipment and has reasonable assurance that the package material and component temperatures will not exceed the specified allowable short-term limits during hypothetical accident conditions consistent with the tests specified in 10 CFR 71.73.

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

4.0 CONTAINMENT EVALUATION

The objective of this containment evaluation review is to verify that the CASTOR® geo69 SNF Transportation Package design satisfies the containment requirements of 10 CFR Part 71, under NCT and HAC for transportation under the proposed CoC 9383, Revision 0.

The applicant, via a letter dated January 14, 2021, has provided their containment analysis for the CASTOR® geo69 package in the document 1014-SR-00001, Rev. 0, "*Safety Analyses Report Type B(U)F Transport Package CASTOR® geo69*" and, subsequently, in response to a request for additional information from the NRC staff (ML24038A289), provided the Rev. 4 of the document 1014-SR-00001 (ML24192A184) The staff's technical review of the applicant's analysis is described below.

4.1 Description of the Containment Design

In SAR section 1.2.1.8, "Components of the Containment System," the applicant describes the components of the containment vessel of the CASTOR® geo69 in the following manner:

The containment system consists of all the packaging components that prevent the release of radioactive material. The CASTOR® geo69 features a double-containment system with the outer containment formed by the DCI [ductile cast iron] cask and the inner containment formed by the canister containing the SNF.

Tables 1.2-5 and 1.2-6 of the SAR list the components of the "inner" and "outer" containment systems, respectively, of the CASTOR® geo69 transportation system.

4.1.1 Containment Vessel

As described by the applicant in section 4.1.1 of the SAR, the containment system for the CASTOR® geo69 consists of an inner containment (canister) and an outer containment (cask).

The containment boundary has been defined by the applicant as the “inner” containment system (canister) and is defined in figure 4.1-3 of the SAR. The applicant defines the components of both the inner and outer containment systems as listed below:

- “Inner” Containment (canister)
 - canister body (including associated welds)
 - canister lid (incl. clamping elements, thread bolts, and metal gasket (seal))
 - tightening plug and metal gasket (seal)
- “Outer” containment (cask)
 - Cask body
 - Cask lid (incl. hexagonal screws, hexagonal head screws for sealing, and metal gasket)
 - protection cap (in the cask lid), cap screws and metal gasket
 - blind flange (in the cask lid), cap screws and metal gasket

Although the applicant defines both “inner” and “outer” containment systems CASTOR® geo69 transportation system, for the purposes of this application no credit is taken for the “outer” containment (cask). Therefore, the “inner” containment (canister) is the only one considered in the containment analysis presented by the applicant.

The standard cited by the applicant for leakage testing for the CASTOR® geo69 transportation system inner and outer containment boundaries (i.e., the “canister” and the “cask, respectively”) is ANSI N14.5 – 2014, *American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment*. The ANSI N14.5 - 2014 standard defines the term “leaktight” as:

The degree of package containment that, in a practical sense, precludes any significant release of radioactive materials. This degree of containment is achieved by demonstration of a leakage rate less than or equal to 1×10^{-7} ref · cm³/s of air at an upstream pressure of 1 atmosphere (atm) absolute (abs), and a downstream pressure of 0.01 atm abs or less.

The ANSI N14.5 - 2014 standard provides the only definition of the term “leaktight”, as it relates to radioactive material package leakage testing, that is recognized by the NRC. The applicant was asked by the staff to clarify the use of the terms “leak-tight” and “leak-tightness” in their SAR, as it related to containment performance of the canister and cask containment boundaries of the CASTOR® geo69 transportation system and the definition of “leaktight” provided in the ANSI standard. The applicant did so in subsequent revisions to the CASTOR® geo69 transportation system SAR, to the satisfaction of the staff.

The applicant will verify that the monolithic cask bodies and closure lids are “leak-tight” by leakage testing of the entire containment boundary, per ANSI N14.5, for each unit of the CASTOR® geo69 transportation system fabricated and, therefore, only the gasket sealing system will need to be assessed for leakage prior to transport.

4.1.2 Sealing Gasket Parameters

The applicant takes the position that there is no credible leakage from the cask and canister bodies, which are monolithic ductile cast iron and stainless steel, respectively, and, as a result, the applicant states, in section 4.1.1 of the SAR, that the “containment analysis can be reduced to the gasket sealing system.”

The gasket used by the applicant to seal the containment boundary of the geo69 cask consists of an inner jacket of stainless steel surrounded by a helical spring made of a nickel-based, high-temperature, low creep superalloy clad with an outer jacket of silver.

Plastic deformation of the outer silver jacket of the seal, which results from the pretension force induced by the installation and tightening of the cask lid, ensures that the gasket will adapt to any sealing surface imperfections. The inner jacket of the gasket acts to uniformly distribute the force of the pressure generated by compression of the helical spring over the outer jacket.

For metal gaskets, capillary leakage is the only relevant potential leakage mechanism and continuous venting is precluded.

The applicant states that for both the “inner” and “outer” containment, a maximum reference air leakage rate of 1×10^{-7} ref cm³/s (leak test criterion) is demonstrated by a leakage test after loading (see SAR chapter 7, table 7.1-2). A pre-shipment leakage rate test is also conducted for the outer containment to demonstrate this leakage rate prior to transport (As indicated in table 7.1-3).

4.2 Containment under NCT

As described by the applicant in section 4.2.1 of the SAR, the “inner” containment (canister) of the CASTOR® geo69 cask, the interior of which is drained, dried, evacuated and backfilled with helium gas prior to final closure, contains the SNF to be transported. The dry annular space formed between the cask interior wall and the loaded canister’s outer wall is evacuated and backfilled with helium before final closure of the cask. The applicant therefore concludes that there are no gasses or vapors present within the annulus, or in the canister containing the fuel, that could potentially cause a gaseous reaction or conflagration.

4.2.1 Pressurization of the Containment System for NCT

The applicant described pressurization of the CASTOR® geo69 transportation package for NCT with both low burnup fuel (LBF) and high burnup fuel (HBF) contents in section 4.2.1 of the SAR. The applicant provided additional details for the pressure calculation in appendix 4-2 to the SAR.

Using the boundary conditions in SAR table 4.2-1 for the canister (inner containment), which includes the maximum helium backfill pressure for both the canister and the cask and gas temperatures enhanced by a multiplication factor for conservatism, the applicant calculated the maximum absolute pressure for the canister under NCT.

The results obtained above combine the areas of the canister backfilled with helium added to helium gas filling the fuel rods, plus fission gases that have been mobilized from the fuel rods, as described by the applicant in appendix 4-2.

The applicant reports that the Maximum Normal Operating Pressure (MNOP) is the value of the maximum absolute pressure less the atmospheric pressure (101.3 kPa or 14.7 psi) which results in a negative value for the MNOP.

For the NCT “Cold” condition, the applicant reported a minimum pressure for both the canister (inner) and the cask (outer) containment boundary based on a helium fill pressure (specified in appendix 4-2) and an environmental temperature of -40 °C.

The applicant concludes that the structural integrity and, therefore, the containment performance of the package, are not compromised under the NCT evaluations. The staff accepts the applicant’s conclusions.

4.2.2 Containment Criteria for NCT

In section 4.2.2 of the SAR, the applicant notes the containment performance criteria for NCT found in 10 CFR 71 § 71.51 which are summarized as the following:

- loss or dispersal of radioactive contents is restricted to a maximum of 10^{-6} A₂ per hour, for any given mixture of radionuclides
- no significant increase in external surface radiation levels may result, and
- no substantial reduction of the effectiveness of the CASTOR® geo69 packaging may result

While the applicant maintains that the containment boundary of the CASTOR® geo69 remains “leaktight” for the NCT tests, they provide a calculation of the A₂ value of mixtures of radionuclides in accordance with appendix A, paragraph IV, of 10 CFR Part 71. The results of these calculations are presented in three separate “mobilized activity and activity concentration” tables in the SAR: table 4.2-2 for gases and volatiles, table 4.2-3 for fines and crud, and table 4.2-4 for the total nuclide mixture.

The applicant, using the parameters provided in SAR table 4.2-4 and the containment criterion listed above, calculated an allowable leakage rate using the leakage rate calculation method described in appendix B of the ANSI N14.5.

The applicant, using the method described in SAR appendix 4-2 and the parameters provided in table 4.2-5, also calculated the maximum allowable leakage hole diameter (L_N) as reported in SAR table 4.2-5. Using the calculated leakage hole diameter, in addition to other parameters also in SAR table 4.2-6, the applicant calculated a reference air leakage rate (L_R) reported in table 4.2-6.

Following the procedure for leakage determinations in ANSI N14.5, the reference air leakage rate, L_R, calculated by the applicant would then become the acceptance criterion for leakage testing before first use of the package, periodic testing, and testing following maintenance or repair. The applicant has committed, however, to test to the “leaktight” criteria of 1×10^{-7} ref · cm³/s found in ANSI N14.5.

The staff reviewed the applicant’s leakage calculations, as presented in the SAR, and found them to be acceptable; however, the staff notes that, in accordance with NUREG 2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material”, a

release calculation is not required if a transportation package is demonstrated to meet the “leaktight” criterion in ANSI N14.5.

4.3 Containment under Hypothetical Accident Conditions (HAC)

As described by the applicant in section 4.3.1 of the SAR, the CASTOR® geo69 canister, in which SNF is loaded, is drained, dried, evacuated and backfilled (with helium gas) prior to final closure. The annular space between the cask interior wall and the loaded canister’s outer wall is evacuated and backfilled with helium before final closure of the cask. As a result of the helium backfill process described, the applicant concludes that there are no gasses or vapors present, either within the annulus or in the canister, that could cause a gaseous reaction or conflagration.

4.3.1 Pressurization of the Containment System for HAC

The applicant described pressurization of the CASTOR® geo69 transportation package for HAC with LBF and HBF contents in section 4.3.1 of the SAR as well as in appendix 4-2 to the SAR chapter 4. Additional details of the applicant’s pressure calculation are provided in appendix 4-2.

Using the boundary conditions in SAR table 4.3-1 for canister (inner containment), which includes the maximum helium backfill for both the canister and the cask, the applicant explores HAC scenarios for fire (HAC-fire) and impact (HAC-impact).

For the HAC-fire condition, the applicant calculates a maximum absolute pressure for the canister loaded with HBF using the boundary conditions reported in table 4.3-1. The pressure value for LBF is also reported in section 4.3.1 of the SAR. Although the value of maximum absolute pressure calculated by the applicant for LBF contents was higher for than for HBF contents, the applicant explains that this is an anomaly in the calculation and maintains that, due to the increase in fission gas releases as burn-up increases, lower pressures are realistically expected for LBF contents when compared to HBF contents.

The staff has reasonable assurance that the pressures reported do not adversely affect the ability of the containment boundary to perform its function regardless of the contents of the package (LBF or HBF) and therefore is not concerned with the calculational anomaly reported by the applicant in the SAR.

The pressure inside the cask (outer containment) is also calculated by the applicant in SAR section 4.3.1, which takes into account the fill gas temperature, enhanced by a multiplication factor for conservatism, as prescribed in SAR table 3.4-2.

For the HAC-impact condition, two variations are considered: “Impact I” which is without a release from the fuel and no reconfiguration of the fuel contents, and “Impact II” which accounts for a release from the fuel as well as a reconfiguration of the fuel.

Again, using the boundary conditions in SAR table 4.3-2 for HBF contents, the applicant calculated the maximum absolute pressure values for the canister under “Impact I” and “Impact II” conditions and presented those in SAR section 4.3.1.

The pressure inside the cask (outer containment) is calculated by the applicant for HAC-Impact I and HAC-Impact II, which are based on the cask fill gas temperatures, enhanced by a multiplication factor for conservatism, found in SAR tables 3.5-12 and 3.5-13.

The applicant concludes that the structural integrity and, therefore, the containment performance, of the package, are not compromised under the evaluated HAC-fire and HAC-impact conditions. The staff accepts the applicant's conclusions.

4.3.2 Containment Criteria for HAC

In section 4.3.2 of the SAR, the applicant notes the containment performance criteria for HAC found in 10 CFR § 71.51 which are summarized as the following:

- no release of ^{85}Kr exceeding 10 A_2 per week and
- no release of other radioactive material exceeding a total amount an A_2 per week.

While the applicant maintains that the containment boundary of the CASTOR® geo69 remains "leaktight" following the HAC tests, they provide a calculation of the A_2 value of mixtures of radionuclides in accordance with appendix A, paragraph IV, of 10 CFR Part 71. The results of these calculations are presented in tables for "mobilized activity and activity concentration" in the SAR, for the HAC-fire condition: table 4.3-3 for gases and volatiles, table 4.3-4 for fines and crud, and table 4.3-5 for the total nuclide mixture and then for the HAC-Impact condition: table 4.3-6 for gases and volatiles, table 4.3-7 for fines and crud, and table 4.3-8 for the total nuclide mixture.

The applicant, using the parameters provided in SAR table 4.3-9 and the containment criterion listed above, applied the method described in ANSI N14.5 to calculate allowable leakage rates for the HAC-fire and HAC-impact conditions and reported the values in section 4.3.3 of the SAR.

Using the method described in appendix 4-2 and the parameters provided in table 4.3-9, the applicant calculated the maximum allowable leakage hole diameters for HAC-fire, HAC-impact I, and HAC-impact II as. This leakage hole diameter is then used, along with the parameters in SAR table 4.3-10, to calculate reference air leakage rates for HAC-fire, HAC-impact I, and HAC-impact II, with those values reported in table 4.3-10 of the SAR.

The staff reviewed the applicant's leakage and release calculations, as presented in the SAR, and found them to be acceptable; however, the staff notes that, in accordance with NUREG 2216, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material", a release calculation is not required if a transportation package is demonstrated to meet the "leaktight" criterion in ANSI N14.5.

4.4 Staff Evaluation

The applicant, in section 4.4 of the SAR, has concluded the following concerning the containment performance of the CASTOR® geo69 transportation package:

The leak-tightness of the containments (container and cask) is ensured with leak-tightness testing compliant with ANSI N14.5 [1]. The relevant procedures for fabrication leakage rate testing, pre-shipment leakage rate testing, periodic leakage rate testing and maintenance leakage rate testing are defined in chapter 7 resp. chapter 8. The leakage tests and the structural analysis in chapter 2 demonstrate that the requirements associated with the reference air leakage rates for NCT (cf. table 4.2-6) and HAC (cf. table 4.3-10) are met.

In section 4.1.1 of the SAR, the applicant makes the general assertion that the CASTOR® geo69 transportation system design, as fabricated, would not exhibit any credible leakage

based on the monolithic nature of the cast-iron cask body and the redundant closure system that provides multiple barriers to any leakage of radioactive material from the contents to the atmosphere. To support their argument related to leakage from the CASTOR® geo69 transportation system, the applicant has demonstrated that the closure system for the containment boundary may be demonstrated (through a leakage test) that it is indeed “leaktight” per the definition in ANSI N14.5.

In response to the RAI provided by the staff regarding the “leaktightness” of the CASTOR® geo69 transportation package (ML24038A289), the applicant provided a detailed overview of their revised leakage calculation, which resulted in the values provided above for allowable leakage rate, leakage hole diameter, and reference air leakage rate based on the acceptance criteria for releases provided in 10 CFR Part 71.

Based on the applicant’s demonstration that calculated leakage from the CASTOR® geo69 package would not result in a release of radioactive material greater than the limits prescribed in 10 CFR § 71.51 for both NCT and HAC of transport, the staff has reasonable assurance that the CASTOR® geo69 will meet the containment requirements found in 10 CFR Part 71.

4.5 Evaluation Findings

Based on a review of the statements and representations in the application, the NRC staff concludes that the CASTOR® geo69 package has been adequately described and evaluated to demonstrate that it satisfies the containment requirements of 10 CFR Part 71.

From NUREG 2216, “Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material”:

- F4.1** The staff has reviewed the applicant’s description and evaluation of the containment system and concludes that:
- the application identifies established codes and standards for the containment system
 - the package includes a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package
- F4.2** The staff has reviewed the applicant’s evaluation of the containment system under normal conditions of transport and concludes that the package is designed, constructed, and prepared for shipment so that under the tests specified in 10 CFR 71.71, “Normal Conditions of Transport,” the package satisfies the containment requirements of 10 CFR 71.43(f) and 10 CFR 71.51(a)(1) for normal conditions of transport with no dependence on filters or a mechanical cooling system.
- F4.3** The staff has reviewed the applicant’s evaluation of the containment system under hypothetical accident conditions and concludes that the package satisfies the containment requirements of 10 CFR 71.51(a)(2) for hypothetical accident conditions, with no dependence on filters or a mechanical cooling system.

5.0 SHIELDING EVALUATION

5.1 Review Objective

The objective of this NRC shielding evaluation is to verify that the Type B(U)F CASTOR® geo69 transportation package design meets the dose rate limits set forth in 10 CFR Part 71, “Packaging and Transportation of Radioactive Material.” Specifically, the requirements in 10 CFR 71.47(b) under NCT, 71.51(a)(2) under HAC, and during loading and unloading conditions. This SER documents the staff’s review of the shielding analysis for the CASTOR® geo69 included in chapters 1, 5, 7, and 8 of the application.

The CASTOR® geo69 package is a type B(U)F transportation package, intended for the transportation of 69 BWR high burn-up FA. There are six types of BWR spent FA (GE 8x8-1, GE 8x8-2, SPC 8x8-2, GE 9B 8x8, GE 12 LUA, and ATRIUM-10A) for transport with CASTOR® geo69 as indicated in section 1.2.2 of the SAR. There are three bounding loading patterns as described in section 1.2.2 of the SAR. The contents may include hardware such as end fittings, plenum springs, and grid spacers.

5.2 Description of the Shielding Design

The following evaluation follows the guidance for shielding evaluation as documented in NUREG 2216.

5.2.1 Packaging Design Features

The CASTOR® geo69 packaging is divided into three groups: the main components cask, internals, and impact limiters.

The cask consists of a monolithic cask body made of DCI that is sealed by a bolted cask lid. The cask body and lid form the outer containment. Neutron shielding consists of two circumferential rows of ultra-high molecular weight polyethylene (UHMW-PE) moderator columns inside the cask body walls and by UHMW-PE moderator discs on the cask bottom and top.

The internals are comprised of the canister, the fuel basket, and the shielding elements. The canister contains the fuel basket and the FA and acts as the inner containment. The canister is a welded stainless-steel shell, sealed with a stainless-steel canister lid. Total height of canister and total outer diameter is given in table 1.2-2 of the SAR. This double containment system qualifies the packaging for the transportation of high burnup fuel.

The fuel basket is the structure that physically holds the FA in place. The fuel basket consists of a structural skeleton that holds the FA in a subcritical arrangement. There is a grid of stacked borated aluminum plates in which the FA are inserted. The whole grids are kept in place by four stainless steel sheets on the outside of the grid. The stainless sheets are secured in place by circular ‘round segments’ that support the whole structure against the inner wall of the cylindrical canister and act as shielding and heat transmission elements as shown in figure 1.2-5 and drawing 1014- DPL-30984 of the application. The gaps between the four round segments are filled with four aluminum shielding elements. Total height of fuel basket and total outer diameter are depicted in the table 1.2.-2 of the SAR.

5.3 Summary Tables of Maximum External radiation levels

The package will be transported by exclusive use, so the applicant did not evaluate the transportation index. The applicant determined the maximum dose rate under NCT at the trailer surface, 2-meters from the trailer surface and occupied cab position dose rates, and 1 meter from the package surface under HAC. Packages are transported in a horizontal orientation. In case that package transported under non-exclusive use, the NCT dose limit is set 0.1 mSv/h at 1 meter from the surface of the package.

Tables 5.1-1 and 5.1-2 provide the maximum dose rates to the surface of package during NCT and HAC for fuel configuration TR1 through TR3. Since the surface dose exceeds 0.1 mSv/h at 1 meter, the package should be transport under exclusive use.

5.4 Source Specification

The applicant determined the bounding source term for each loading pattern. The radiation sources consist of gamma and neutron radiation from active zone of SNF and gamma from activated hardware. The applicant used the ORIGAMI sequence of SCALE 6.2 to evaluate the source term for both the in-core region and activated hardware. The applicant compares the decay heat, gamma, and neutron source term to find the maximum values of source term. The source terms evaluated for all SNF geometry using ORIGAMI, TRITON used to generate cross-section libraries for all FA types, with Evaluated Nuclear Data File (ENDF)/B-VII-252 group library and enrichment from 1.75 to 4.25 weight percent (wt%). The applicant checked the validity of output against several TRITON calculations using ORIGAMI. The burnup profiles are matched with the burnup profiles of NUREG/CR-7224. The minimum cooling times are shown in the table 5.2-2 of the application.

5.4.1 Gamma Source

As shown in table 5.2-3 of the SAR The gamma-radiation of the spent fuel is analyzed in seven energy lower group energy and upper energy group and average group. The applicant applied scaling factors to account for the burnup profile discussed above. In calculating the results from the ORIGAMI code, the applicant used these scaling factors to estimate plenum, top nozzle, and bottom nozzle gamma contributions. Tables 5.2-4 through 5.2-10 of the application show the gamma source for each type of SNF. Tables 5.2.11 through 5.2-14 of the application show the hardware activation gamma sources for TR1 through TR3 for different decay heat for all allowable fuels.

5.4.2 Neutron Source

Neutron source analyzed as function of energy the same as gamma sources. The applicant evaluated the neutron source from all contributing nuclides. The applicant also included spontaneous fission and (α , n) reactions. The applicant considered subcritical neutron multiplication in the MCNP model. The spontaneous fission source is shown in table 5.2-19 and the (α , n) reactions are shown in table 5.2-20 of the application. The applicant selected bounding neutron sources for each loading pattern for bounding decay heat as shown in the tables 5.2-21 through 5.2-23. The total neutron energy for spontaneous fission and (α , n) reactions are depicted in the tables 5.2-24 and 5.2-25 of the application.

5.5 Shielding Models

The applicant used MCNP 6.2 with continuous-energy cross-sections based on ENDF/B-VII Nuclear data. The applicant ran the MCNP in coupled mode (both neutron and gamma) to account for n-gamma reactions. Subcritical multiplication is handled automatically by the code. MCNP 6.2 is a 3-D, Monte Carlo particle transport code that can model complex geometry, such as those found with complex systems. The code and cross-section library have a long history of use in spent-fuel storage and transportation applications, and staff finds their use here acceptable. The applicant developed seven 3-D MCNP 6.2 models for CASTOR® geo69 package to test under different transport conditions and used these models to evaluate dose rates for both NCT and HAC. For each model, the applicant evaluated the contribution from neutron, secondary gamma from neutron interactions, and primary gamma contributions. Staff finds this acceptable since the applicant considered all possible contribution from secondary reaction beside primary gamma and neutron sources.

The applicant chose the Atrium-10A as the bounding fuel design. Atrium-10A has a large water channel, and partial length rods that minimize self-shielding. The Atrium-10A also has the thinnest fuel cladding and fuel channel with respect to the other authorized fuel types in the geo69 package. The applicant modeled four axial zones for the fuel, on each for the bottom end fitting, in-core region, plenum, and top end fitting. The models are shown in figure 5.3-4 of the SAR which includes the diameter of the fuel rod, cladding out diameter, and thickness of the cladding that includes tolerance and the fuel channel. No dummy fuel is assumed in the MCNP model. The heaviest metal in the spent fuel is modeled. The applicant modeled the basket as shown in figure 5.1-1. No mounting elements were modeled and left as void. The canister is modeled as steel and the diameter, height, and thickness are specified in the SAR. The canister lid is modeled as steel. The cask body and lid are modeled with nominal dimensions. The applicant modeled the total height of the cask as depicted in the SAR table 1.2-2. The applicant modeled the trunnions as a solid piece of steel with minimum dimensions. The minimum thickness used in the MCNP model is discussed above. The applicant modeled the cask lid with lid moderator installed on its lower surface. The applicant modeled the closure plate on the bottom side of the cask as steel. The applicant modeled cooling fins explicitly using base thickness and fin fine tip dimension. The applicant modeled the moderator rods as polyethylene with steel supporting bars. The applicant modeled the impact limiters for NCT with nominal dimensions. The applicant included foam filling, spacers, a load distribution plate (aluminum), and a penetration plate (steel) in its model. The applicant did not consider the interior sheets and plates in its shielding model. Staff finds this acceptable since conservative MCNP modeling was used with minimum thickness and no credit is taken for the sheets and plates which could reduce the dose rates.

The impact limiters protect the package during transportation and are made of steel casing filled with polyurethane foam. The impact limiters have aluminum housing and inner layer structure sheets.

All other cask body dimensions, including the height of the cask and the outer diameter (over cask fins), are given in appendices 1 through 4 of the application.

The package shown in the figure 5.1-2, and the top cross section view of the package is depicted in the figure 5.1-1 of the application. The minimum thickness of materials is used in the shielding calculation.

No credit is taken for the transport vehicle for CASTOR® geo69 which is transported with its impact limiters. The tip of the cooling fin is assumed to be located at the lateral surfaces of the vehicle which is a conservative assumption.

For CASTOR® geo69 the moderator rods and plates are made of UHMW-PE and serve as a neutron moderator. The applicant considers two cases in modeling: cold and hot. In the cold case, the applicant assumed the cask was recently loaded and thermal equilibrium is not reached. The UHMW-PE rods are still at room temperature with no expansion. In the hot case, the applicant assumed the cask has reached thermal equilibrium where the UHMW-PE rods have expanded and reduced density.

In case transportation takes place in the winter, the moderator rod temperature compensates the low ambient temperature, no shrinkage of polyethylene is considered in the model. The shrinkage will only be observed when the fuel cools down, and cannot compensate for ambient cool weather, but in this case the source term is weaker and therefore, will not generate a higher external dose rate.

The model that generates higher external dose is representative of the shielding model. Figures 5.1-4 and 5.1-5 of the application show two shielding models. Since CASTOR® geo69 is authorized to transport high burnup fuel, the applicant considers the impact of 3% fuel failure in NCT modeling (see NUREG-2224, Dry Storage and Transportation of High Burnup Spent Fuel).

The applicant also performed calculations for HAC. The external dose rate is limited to a maximum of 10 mSv/h at 1 meter, as required by 10 CFR 71.51. The two consequences the applicant considered in the HAC model are partial or loss of the neutron moderator due to fire, and damage to impact limiters as result of a drop. The applicant assumes total loss of moderator materials and impact limiters, as chapter 2 of the SAR shows no damage to the canister, basket, and cask during HAC. Staff finds this acceptable because it is a conservative model since there is no damage to impact limiters due to accidents. The applicant also evaluated the 100% fuel failure under HAC and relocation of the fuel. Staff finds this acceptable because it is a conservative assumption that considered 100% fuel failure.

The applicant homogenized the fuel, for the models with fuel reconfiguration, over the cross section of the fuel in basket and used the appropriate length and axial peaking factors for the core region. Four separate models for NCT and three HAC were used to find contributions from gamma, neutron, and secondary gamma. The source was modeled uniformly in radial position and varied axially to apply peaking factors.

The MCNP model was based on the drawing in Appendices 1 through 10 of the application. The basket was modeled as stainless-steel boxes encircled by aluminum plates. The shielding model under NCT condition evaluated for shortly after loading (moderator materials (polyethylene) at room temperature), under operating condition (hot shielding model, polyethylene with reduced density), 3% failure of fuel toward the bottom of canister, and 3% failure toward the top of canister. Under the HAC conditions, total loss of moderator and impact limiters, 100% fuel failure toward the bottom of canister, and 100% fuel failure toward the top of canister.

5.6 Detectors

The applicant modeled the package surrounded by 10 m of air to account for scattering effect. The dose rates obtained by detectors located around the cask are shown in figure 5.3-8 of the application. The applicant modeled a cylindrical mesh tally with variable cell sizes. For the

lateral region of the cask, the applicant used an angular cell mesh. The applicant set the smaller thickness of the cells at the surface of the cask, and a larger thickness at 1 m and at 2 m. Dose rates at the top and bottom sides of the package were obtained by subdividing the mesh tally radially from the axial of symmetry of the package as shown in figure 5.3-8 of the SAR. The mesh tally thickness increased the same as the side from smaller to larger.

5.7 Material properties

The composition and densities of the material used in MCNP model shown in table 5.3-1 of the application. The applicant modeled two different densities for moderator rods, nominal after loading into cask and reduced after equilibrium reached. The applicant assumed all steel components consist of SA-240M 316 in the model. The applicant reduced the boron carbide content of the aluminum-boron metal matrix composite (Al-B₄C-MMC) from nominal value. Since this also reduces the amount of B-10, a strong neutron absorber, staff finds this acceptable. The design basis properties for all materials are shown in the table 2.2-1 of the application.

5.8 Shielding Evaluation

The applicant states in section 5.4.1 of the application that it used reduced nominal packaging dimensions within the shielding models. This is a conservative assumption because actual thicknesses of shielding components greater than this when accounting for actual thickness as discussed above.

The MCNP 6.2 with ENDF/B-VII models of the package under NCT and HAC for six position groups shown in figure 5.0-1 of the application. The MCNP model benchmarked against experimental model data for gamma and neutron problems. There was a good agreement between the calculated and measured values. The loading patterns of TR1 through TR3 are evaluated for NCT and HAC.

5.9 Input and Output Data

The applicant provided sample MCNP input and output files used for gamma and neutron dose calculations under NCT and HAC in Appendices 5.4 and 5.5 of the application, respectively.

5.10 Flux to Dose Rate Conversion:

The MCNP code calculates a fluence per emitted particle. This fluence is converted into a dose rate by using fluence-to-dose rate conversion factors to arrive at the dose rate per emitted particle. The applicant used the fluence-to-dose-rate conversion factors recommended by NUREG-2216, and the 1997 version of ANS/ANSI-6.1.1 standard and are therefore acceptable to the staff. The applicant added an additional 2 σ to the fluence calculated by MCNP to account for the statistical uncertainty of the Monte Carlo code. The staff found it to be a conservative and acceptable way to account for this uncertainty.

5.11 Dose Rate Results

5.11.1 External Radiation Levels

The MCNP code uses “tallies” when determining particle flux at a location of interest. The tally cell represents the volume in space that the particles are counted. Tally cells need to be small enough to reasonably represent a maximum dose, and not so large as to represent an average

dose. The cell tallies specified by the applicant for the CASTOR® geo69 can be found in the SAR. The radial thickness varies from a shorter distance relative to the surface of the package, to a larger distance at a distance of 1 m, and at 2 m from the surface of the package. The staff used its judgment and consideration for the conservatism within the source term modeling (e.g., energy bins represented as upper values) and found that the size of the tally for the dose rate calculations is acceptable with these considerations. The locations of the tally cells are based on the locations specified in 10 CFR Part 71 (e.g., surface, 2 meter, and 1 meter under HAC).

Tables 5.4-3 through 5.4-5 of the application depict the maximum external dose rates for three patterns shortly after loading. For these cases, the applicant assumed the moderator material is at room temperature. The dose rates at surface of the package exceed 0.1 mSv/h, so the package only can be transported by exclusive use. TR1 dose rates bounded all other patterns. Figure 5.4-1 of the application shows the dose rate distribution on the surface of the package.

Tables 5.4-6 through 5.4-8 of the application show the maximum dose rates for the package under NCT at thermal equilibrium. The moderator temperature is at maximum temperature for these cases, so the applicant reduced the density analyzing these cases. The TR1 pattern generates the higher dose rates than other patterns. The dose rate at 1 m exceed 0.1 mSv/h in these cases so the package can be transported only by exclusive use.

The applicant also analyzed the fuel reconfiguration under NCT for TR1, since this pattern has the highest dose rates. The applicant considered two cases, one with 3% failed fuel at the bottom of canister and one with 3% failed fuel to the top of canister close to lid. Tables 5.4-9 and 5.4-10 of the application show the 3% fuel failure for TR1 pattern at the bottom side and lid side respectively. The calculated dose rates at 1 m exceed 0.1 mSv/h in these cases so the package can be transported only by exclusive use.

The dose rates for package under HAC for all 3 patterns is shown in table 5.4-11 of the application. The table also shows dose rates for pattern TR1 when the 100% of the fuel fails and concentrates at the canister bottom and canister lid.

5.12 Staff Evaluation

The applicant demonstrated that the package design meets the regulatory dose rate requirements of 10 CFR 71.47 and 71.51(a)(2) by performing a package evaluation which satisfied the requirements of 10 CFR 71.35(a). The staff concludes that the shielding design of the CASTOR® geo69, when used as described in the application, is in compliance with 10 CFR Part 71 and that the applicable design and acceptance criteria have been satisfied. The staff has reasonable assurance that the CASTOR® geo69 design will provide safe transportation of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes, and standards, the applicant's analysis, and responses to requests for additional information, and acceptable engineering practices. The staff also performed confirmatory analyses of the dose rates for transport package CASTOR® geo69. The staff finds the applicant dose rates are similar or very close to the staff's evaluation using SCALE 6.3 computer code for source term calculation and MCNP 6.2 in shielding analysis. The results were in agreement with the applicant evaluations. Based on its review of the statements and representations provided in the application and staff evaluation, the staff has reasonable assurance that the shielding evaluation is consistent with the appropriate codes and standards for shielding analyses and the NRC guidance. Therefore, the staff finds that the package design and contents satisfy the dose rate limits in 10 CFR Part 71.

5.13 Evaluation Findings

- F5.1** 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.
- F5.2** 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.
- F5.3** 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.
- F5.4** 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).
- F5.5** 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).
- F5.6** The staff has reviewed the application and finds that it includes operations descriptions, acceptance tests, and maintenance programs that will ensure that the package is fabricated, operated, and maintained in a manner consistent with the applicable shielding requirements of 10 CFR Part 71.

6.0 CRITICALITY EVALUATION

The objective of this review is to verify that the contents of the CASTOR® geo69 (geo69) 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 geo69 consists of an outer cask with the cask body, cask lid, and further attachments enclosing the canister and the fuel basket for accommodation of up to 69 BWR fuel assemblies. The cask body is a hollow cylinder with a closed bottom end which is made of ductile cast iron. The cask body is cast in one monolithic piece, which includes radial cooling fins that are machined into the outer surface of the cask body after the initial casting. Additionally, the neutron shielding material is housed in boreholes that are machined into the cask walls after

casting. A closure plate covers both the bottom moderator plate and the boreholes. The cask lid consists of austenitic stainless steel with a disk of neutron shield material on the underside. The lid is bolted to the cask body. The cask lid and body comprise the outer containment boundary.

The canister consists of stainless steel and is comprised of the canister body and the canister lid system. The canister body consists of a bottom, a liner, and a head ring. The liner is welded together from a series of plates, and the bottom and head ring are both welded to the liner. The canister lid system consists of a stainless-steel canister lid that is secured by a series of clamping elements. The canister lid system and body comprise the inner containment boundary.

The fuel basket consists of borated aluminum metal matrix composite (MMC) and is comprised of a series of vertically oriented sheets, stacked alternately on end to create a basket mesh. Outer sheets, consisting of stainless steel, maintain the MMC sheet placement.

6.1.2 Codes and Standards

The applicant based its criticality safety design philosophy on ANSI/ANS-8.1-2014. The applicant followed the guidance of the SRP for moderator exclusion under HAC to show subcriticality under 10 CFR 71.55(e). The NRC has endorsed ANSI/ANS-8.1-2014 for uranium systems as described in Regulatory Guide 3.71.

6.1.3 Summary Table of Criticality Evaluations

The applicant presented the most limiting results of its criticality analyses in table 6.1-1 of the application. The applicant applied its biases and uncertainties to each result rather than reduce the upper subcritical limit (USL) from the administrative limit of 0.95. The staff reviewed the results and noted all are less than the administrative k_{eff} limit with all biases and uncertainties applied.

6.1.4 Criticality Safety Index (CSI)

The applicant demonstrated that an infinite array of densely packed packages, both undamaged and damaged, will remain subcritical. As a result, the CASTOR® geo69 package has a CSI of 0.0 for the most limiting configuration according to 10 CFR 71.59(b).

6.2 Contents

6.2.1 Fuel Type

The geo69 is designed to hold 69 undamaged, spent BWR fuel assemblies restricted to the fuel assembly types with each having associated limits listed in tables 1.2-11 and 1.2-12 of the application. Dummy rods may replace individual fuel rods provided they have an equal or greater volume than an intact fuel rod for that assembly type.

Maximum initial heavy metal mass of a single assembly is 0.185 MTU. This yields a maximum initial heavy metal loading for a full geo69 package of 12.765 MTU.

6.3 General Considerations for Criticality Safety

The geo69 transportation package relies on fixed geometry, neutron absorbers in the basket material, and a limit to the amount of fissile material allowed.

6.3.1 Model Configuration

The applicant modeled the cask body, bottom and lid system with some of the associated structures. For the cask body and bottom, the applicant modeled it as a monolithic cylinder comprised entirely of cast iron. The applicant did not model the cooling fins machined into the outer surface of the cask body. Instead, the applicant modeled a reduced diameter representing the minimum diameter of the cooling fin valleys. This effectively reduces the spacing between the contents and potential external moderator, or results in a smaller pitched array. Both effects will increase calculated reactivity and the staff finds this simplification acceptable. The applicant omitted the moderator rods in the cask wall. The applicant also omitted the closure plate, void space and moderator disk from the bottom of the cask. While these features either include moderator, or may allow moderation via flooding, the location is separated from the contents and will have negligible effect on calculated k_{eff} . Staff calculations confirmed this conclusion (see section 6.8 below). As a result, the staff finds the applicant's modeling of the cask body and bottom acceptable.

The applicant modeled the lid system as a single, stainless-steel disk. The applicant omitted closure components (e.g., bolt holes), the top moderator disk, and the spacing between the lid system and the top of the canister. Since the features of the closure system are small, they are unlikely to have a significant impact on reactivity. Therefore, the staff finds their omission acceptable. Like the moderator and void space in the cask body and bottom, the moderating material and potential flooding of the void space between the cask lid and the canister will have a small impact on reactivity due to the separation from the contents. Staff calculations confirmed this conclusion (see section 6.8 below). As a result, the staff finds the applicant's modeling of the lid system acceptable.

The applicant did not credit any array spacing or moderator displacement that the impact limiters would have provided in its evaluations. Omitting the impact limiter from the models will conservatively increase calculated reactivity, and the staff finds it acceptable.

6.3.2 Material Properties

The applicant assumed 75% of the minimum boron content in the basket plates for its criticality analyses. The applicant detailed its MMC material qualification in appendix 2-8 of section 2.12 of the application. The staff noted the applicant conducts neutron transmission testing to verify ^{10}B areal density with each extrusion profile. This follows SRP guidance for 75% boron credit in fixed neutron absorbers, and the staff finds the applicant's minimum boron content assumption acceptable.

The applicant performed a series of sensitivity studies with TSUNAMI on the densities of each mixture in its model. The results are shown in table 6.3-1 of the application. Only the fuel, basket sheet, and moderator density had significant impact on reactivity. These materials provide neutron multiplication, neutron absorption, and neutron moderation, respectively, and this result is within expectation. The applicant used the most reactive manufacturing tolerance limit for those materials, which maximizes calculated reactivity, and staff finds this acceptable. The applicant used nominal densities for the other mixtures. The staff finds this acceptable since calculated k_{eff} is not significantly impacted by changes in density for those materials.

The applicant also performed a series of sensitivity studies with TSUNAMI on the isotopic composition of the packaging materials. The results are shown in table 6.3-2 of the application. Only the boron content of the MMC basket sheets had significant impact on reactivity. Since the isotopic composition of the fuel is fixed in the applicant's analyses and the other manufactured

materials with composition tolerances have little effect on reactivity, this result is within expectation. The applicant used the most reactive manufacturing tolerance for the boron content of the MMC basket sheets, which maximizes calculated reactivity. Therefore, the staff finds the applicant's isotopic compositions acceptable.

The applicant considered the heavy metal content to consist entirely of ^{235}U and ^{238}U (i.e., fresh-fuel assumption). This ignores the presence of ^{236}U and ^{234}U , which are neutron absorbers, thus increasing the calculated reactivity. As a result, the staff finds the applicant's uranium composition acceptable.

To achieve more uniform radial power distribution, fuel rods may have different enrichments within the fuel lattice. The applicant evaluated the possible effect on reactivity that planar enrichment gradients could have. The applicant's planar enrichment study assumes full 10x10 lattice of full-length fuel pins (i.e., no water rods/channel and no part-length rods) for simplicity. The pin layout and enrichments are shown in figure 6.3-12 and table 6.3-6 of the application, respectively. The applicant's results are shown in table 6.3-7 of the application. The applicant's results show a negative reactivity trend as the difference between the maximum and minimum pin enrichment increases. Staff analysis confirms the same negative reactivity trend with increasing radial enrichment eccentricities, and the staff finds the use of planar-averaged uniform enrichment acceptable.

6.3.3 Analysis Methods and Nuclear Data

The applicant used the KENO-VI code in the SCALE 6.2 code suite with the 252-group cross-section library based on ENDF/B-VII.1 nuclear data. For its sensitivity studies, the applicant used TSUNAMI with TSUNAMI-IP which are also in the SCALE 6.2 code suite. TSUNAMI is a SCALE control module that relies on KENO-VI to determine the sensitivity of a computed response (e.g., k_{eff}) to each of the components or isotopes used in the calculation. KENO-VI is a 3-D Monte Carlo transport code that can model complex geometry, has a long history of use in spent fuel criticality applications, and is well vetted. TSUNAMI also has a long history of use for criticality safety analysis. For these reasons, the staff finds the applicant's criticality and sensitivity codes and cross-section libraries appropriate.

6.3.4 Demonstration of Maximum Reactivity

The applicant determined the Atrium-10 was the most reactive fuel assembly and varied several parameters to determine most reactive possible configuration. Those parameters were: length of the active fuel zone; fuel cladding inner and outer diameter (i.e., cladding thickness); the thickness of the water channel; and the thickness of the fuel channel. Of these parameters, only some significantly increased calculated reactivity. Two fuel assembly parameters, the thickness of the water channel and fuel cladding, represent the displacement of moderator by zircaloy material which is largely transparent to neutrons. Thus, it is not surprising that the nominal values for these parameters are not bounding. The applicant presented the results of its sensitivity studies on the water channel thickness and fuel cladding thickness in table 6.3-3 and figure 6.3-9 of the application. The staff reviewed the applicant's perturbations and finds the range is varied sufficiently to cover expected variations due to manufacturing tolerances. The staff also finds the applicant's use of the bounding parameters conservatively maximizes calculated reactivity. The applicant's investigation of the other fuel assembly parameters showed no statistically significant impact on calculated k_{eff} , as shown in table 6.3-3 and figure 6.3-9 of the application. As a result, the staff finds the applicant's use of nominal values for those parameters in its model acceptable.

The applicant also varied the position of the FA within the basket and shifting of part-length rods. The applicant evaluated the following parameters: axial shifting of the FA; radial shifting of the FA; clustering of FA; axial shifting of part-length rods in a fuel assembly; and since the Atrium-10 fuel pin arrangement is asymmetric, fuel assembly orientation. The applicant's results are shown in table 6.3-3 and figure 6.3-9 of the application. For the axial shifting of FA and orientation of the FA, the applicant's results showed no statistically significant impact on calculated reactivity. For the axial shifting of part-length fuel rods, the applicant's results show the most reactive configuration is with the baseline modeling of these rods at the bottom of the active fuel zone. Therefore, the staff finds the applicant's use of its baseline NCT configuration without rotation in its criticality analyses acceptable. For clustering of FA as shown in figure 6.3-7 of the application, the applicant's results show a decrease in reactivity. With all FA shifted toward the center of the geo69 package, the applicant's results show an increase in reactivity. As a result, the staff finds the applicant's use of inward shifting as its bounding condition acceptable.

Using the packaging assumptions described in section 6.3.1 above, the applicant varied several parameters related to the packaging. Those parameters were: partial and full flooding; radial shifting of the canister; the canister inner and outer diameters (i.e., canister wall thickness); thickness of the outer stainless-steel basket sheets; thickness of the basket sheets; the width of the basket cell. The results of the applicant's evaluations are shown in table 6.3-3 and figure 6.3-9 of the application. The thickness of the outer, stainless-steel basket sheets did not have any significant effect on reactivity. Since these sheets do not contain any neutron absorber and are located on the periphery of the basket, this result is expected. The applicant assumed minimum tolerances in the thickness of the internal basket sheets. This effectively minimizes the width of the basket locations and reduces the spacing of fissile material and maximizes the amount of moderator present. The applicant evaluated reactivity over the range of manufacturing tolerances and the results confirm the selection of minimum dimensions for modeling its basket components. As a result, the staff finds the applicant appropriately determined the most reactive configuration of the basket sheets. Offsetting the canister within the cask body showed no significant change in calculated reactivity. Therefore, the staff finds the applicant's use of the baseline, centered location acceptable. The applicant's results for both minimum and maximum canister wall thickness at the maximum outer diameter (OD) do not differ significantly. Considering the magnitude of other conservative assumptions and the applicant's margin in k_{eff} between the most reactive configuration and the geo69 USL, the staff finds the applicant's use of the baseline configuration acceptable. In evaluating partial flooding, the applicant conducted two assessments in a vertical and horizontal orientation. The applicant's results with its horizontal flooding evaluation showed the most reactive state is fully flooded. In the vertical orientation, the applicant's results showed calculated reactivity will increase as water level increases until a maximum k_{eff} value prior to a fully flooded condition. The applicant's results show k_{eff} values both slightly above and below that of the fully flooded condition in a flat distribution as water height is further increased. This indicates that this is likely an artifact of computational statistics and any real effect in this water height range would be insignificant. As a result, the staff finds the applicant's use of the fully flooded condition as its most reactive configuration acceptable.

The applicant varied the temperature of the materials in its models to determine the effect doppler broadening would have on calculated reactivity. The applicant's results show little effect between 0° and 20° C where reactivity might increase. The applicant modeled water at a density of 1.0000 g/cm³ in its baseline model, which is slightly greater than the maximum density of water around 3° C. Since the moderator density effects dominate the temperature effects, and the moderator temperature coefficient is driven largely by the associated change in density, the

applicant's baseline model already accounted for the largest contributor to any reactivity increase below 20° C. As a result, the staff finds the applicant's use of its baseline temperature acceptable.

The applicant varied the effective boundary conditions for both single packages and an array of packages. This is most important to determine the effect external moderation (i.e., flooding) will have on a single package, and whether any increased moderation would counteract increased package interaction in package arrays. For a single package, the applicant explicitly modeled 20 cm of water around the package and compared it to the voided case. For arrays, the applicant assumed the packages were close packed in a square-pitched array with the interstitial space being either water or void. The applicant's results in table 6.3-3 of the application show little to no statistical change in calculated k_{eff} due to external flooding. This is expected since internal flooding provides significant moderation, and the thick packaging walls separate external moderating material from the contents. Staff analyses confirm the applicant's results, and the staff finds the applicant's use of external void space in its most reactive configuration acceptable.

The applicant evaluated the effect of a single missing assembly in eight different locations along horizontal and diagonal basket radii. The empty locations modeled are shown in figure 6.3-10 of the application. The staff reviewed the locations and, considering the basket symmetry, finds they cover a sufficient range of positions to determine the maximum possible effect on reactivity. The applicant's results are shown in table 6.3-5 and figure 6.3-11 of the application. The staff reviewed the results and noted that the reactivity impact becomes more negative toward the central basket location. As a result, the staff finds reasonable assurance that one or more empty locations will result in a decrease in reactivity regardless of the position in the geo69.

6.4 Single Package Evaluation

6.4.1 Configuration

The applicant used the most reactive configuration reviewed by the staff in section 6.3.4 above with the addition of 20 cm of water reflection. The staff finds this configuration meets the requirements of 10 CFR 71.55(b). As part of its determination of the most reactive configuration, the applicant already evaluated the effect of external moderation on a single package. Those results showed that a flooded package would remain subcritical and external moderation had a minimal effect on calculated reactivity. Therefore, the staff finds reasonable assurance that a single flooded and water reflected geo69 package in the most reactive configuration will remain subcritical. The results of the applicant's single package evaluations are shown in table 6.4-1 of the application.

6.5 Evaluation of Package Arrays

6.5.1 Package Arrays under NCT

The applicant used the most reactive configuration reviewed by the staff in section 6.3.4 above except with no internal flooding. The staff reviewed the geo69 packaging design features and procedures in sections 2 and 7 in this SER and found they ensure no single packaging error would permit water leakage. Per 10 CFR 71.55(d), the staff finds the applicant's NCT array without water moderator acceptable. The applicant modeled an infinite array of dry geo69 packages in a square lattice with no separation between packages. As expected, the calculated k_{eff} of LEU systems without moderator is very low and is bounded by the single NCT package evaluation. Therefore, the staff finds reasonable assurance that an infinite array of dry geo69

packages will remain subcritical. The applicant's results are shown in table 6.5-1 of the application.

6.5.2 Package Arrays under HAC

The staff review of the geo69 structural design features in section 2 of this SER found the applicant showed no credible change to geometry as a result of the tests and conditions specified in 10 CFR 71.73; therefore, the staff finds applicant's use of its NCT model acceptable for package arrays under HAC. The applicant's NCT results from table 6.5-1 remain applicable and are repeated in table 6.6-1.

6.5.3 Package Arrays and CSI

The applicant showed that an infinite array of packages will remain subcritical; therefore, the CASTOR® geo69 package has a CSI of 0.0 for the most limiting configuration according to 10 CFR 71.59(b).

6.6 Benchmark Evaluations

The applicant performed its criticality and sensitivity analyses with SCALE 6.2 and the 252-group cross-section library based on ENDF/B-VII.1 nuclear data. These are the same code and cross-section libraries the applicant used in its criticality safety analysis. This follows SRP guidance and staff finds it appropriate. The applicant determined the correlation coefficients using the TSUNAMI-IP code with the 56-group covariance library that is distributed with SCALE 6.2. This covariance library has a history of use with criticality safety applications and is well vetted. As a result, the staff finds its use here appropriate.

6.6.1 Experiments and Applicability

The applicant selected 93 critical experiments from six series of benchmark experiments from reference 6-2 of this SER. The applicant listed the criticality results of the selected experiments in table 6.8-1 of the application. In the most reactive configuration, the BWR FA permitted in the geo69 are essentially water-moderated arrays of UO₂ rods. One critical experiment series the applicant selected consists of arrays of LEU metal tubes which are water moderated. Even though the fissile material is metal and not oxide, it is still a thermal LEU system that mimics the heterogeneity of a fuel assembly. The rest of the benchmark cases consist of water-moderated arrays of UO₂ rods. In the benchmark cases where there are neutron absorbers, the absorbing material is boron; this matches the neutron absorbing material in the geo69. For these reasons, the staff finds the applicant's selection of benchmark cases appropriate.

The applicant also performed a comparative sensitivity analysis with TSUNAMI-IP. TSUNAMI-IP calculates the sensitivity of the computed responses (e.g., k_{eff} and ratios of reaction rates) on changes in nuclide densities for a given system. Using these calculated sensitivities and the cross-section covariance data, the similarity of two systems can be quantified. Two systems that exhibit the same sensitivities to the same perturbation can be considered to have a high degree of similarity, which the code quantifies in a single correlation coefficient, c_k . A c_k value higher than 0.8 indicates marginal similarity, and a value greater than 0.9 indicates similarity. All but six of the critical experiments in the series the applicant selected had a c_k value greater than 0.8. As a result, the staff finds reasonable assurance that the critical benchmark series selected by the applicant are appropriate for the geo69 transportation package.

The applicant performed a trending analysis on the correlation coefficient, c_k . The applicant included the lower-correlated experiments (i.e., those with c_k values less than 0.8) from its selected benchmark series in its bias and bias uncertainty determination. The bias trend as a function of c_k is shown in figure 1 of reference 8, and the bias trends smaller with a small slope (< 0.0015) as the correlation value increases. This is expected since the similarity to the system being evaluated also increases. By including these less correlated benchmarks, the applicant increased its effective bias and bias uncertainty.

The applicant performed an area of applicability study on three parameters, enrichment, pitch, and cladding thickness. The results are shown in tables 1 and 2 of reference 8. Enrichment typically has a strong correlation with bias and bias uncertainty; pitch and cladding thickness contribute to the fuel-moderator ratio, and both also typically show strong correlation. As a result, the staff finds the selection of these parameters acceptable to show area of applicability. The staff noted the enrichment of the most reactive configuration of the geo69 package extends above the range covered by the benchmark data. The applicant calculated its bias uncertainties with a weighted mean approach with generalized linear least-squares method. The applicant also selected c_k as its trending parameter. This method has been shown to calculate higher bias uncertainties than other methods with c_k trending (7). The applicant showed the inclusion of less correlated experiments increased the magnitude of the bias and bias uncertainty. When applied to the calculated k_{eff} values in the criticality analyses, this results in a more conservative result. Given the relatively flat trend, the small extrapolation beyond the area of applicability, and conservative methodology used to determine bias and bias uncertainty, the staff finds reasonable assurance that the benchmark area of applicability sufficiently covers the applicant's evaluations.

6.6.2 Bias and Uncertainty Determination

The applicant describes its bias determination methodology in section 6.8.2 of the application. The applicant calculated the bias with a lower one-sided tolerance limit using equation 8.9 in section 6.8.2 of the application. The applicant relied on some approximations that require normality in the data distribution without apparent trends in the criticality results (5). The frequency distribution of the biases of the selected benchmark experiments is shown in figure 6.8-3 of the application. The distribution is nearly normal but skews non-conservatively toward a smaller bias. Therefore, the applicant's assumption of normality increases the bias correction. This conservatively increases the calculated k_{eff} values when applied to the applicant's analyses. As a result, staff finds the applicant's assumption of normality acceptable.

The applicant defined its total uncertainty as the sum of the associated uncertainties due to the following: nuclide inventory; fuel burnup; calculational tool; Monte Carlo method; and tolerances. The Monte Carlo uncertainty is already included in the upper one-sided 95/95 tolerance limit for each calculated k value, so there is no need to add it again to the total uncertainty. Since the applicant assumed bounding values for geometry and material compositions, and no burnup credit is applied, the uncertainties due to tolerances, nuclide inventory, and burnup are equal to zero. As a result, the applicant's uncertainty is simply equal to that of the calculational tool.

Rather than apply the total uncertainty as a reduction to the administrative USL k_{eff} of 0.95, the applicant applied the total uncertainty by increasing each calculated k_{eff} value, which is effectively the same.

This method follows many of the recommendations of references 6-4 and 6-5 of this SER. The staff reviewed the applicant's methodology and finds it applies appropriate statistical rigor, and the staff finds reasonable assurance that the applicant determined code bias and bias

uncertainty such that any additional calculation has a 95% chance to fall below the expected maximum value with a 95% confidence level.

6.7 Burnup Credit for Commercial LWR SNF

The CASTOR® geo69 package does not rely on burnup credit for criticality safety.

6.8 Staff Confirmatory Analysis

The staff performed confirmatory calculations using the KENO-VI code in SCALE 6.3.1. The staff used a continuous energy cross-section library based on ENDF/B-VII nuclear data. The staff's geo69 model included the bottom moderator plate, bottom closure plate, and associated void spaces at the bottom of the cask. The staff also included axial spacing at the top of the cask, along with the top moderator plate. Except as noted, the staff modeled the side wall of the geo69 as a monolithic cylinder of cast iron.

For the geo69 canister, the staff modeled the canister lid and body as a single, stainless-steel cylinder with a cavity void inside. The staff modeled the whole basket at the same length as the active fuel zone, which omits the top and bottom of the basket. Aside from the channel, the staff omitted non-fuel components from its fuel assembly models. The staff maintained the elevation of the active fuel zone that would occur due to the presence of lower hardware (e.g., nozzles). The staff included the pellet-clad gap in its fuel assembly models. This space was filled with either un-borated water or void depending on the evaluation. The staff modeled the internal water channel for the Atrium-10 fuel at the minimum thickness. Any basket or fuel assembly component omitted from the model was replaced with either void or water, depending on the evaluation.

The staff compared the reactivity of an array of ATRIUM-10A and GE12 LUA assemblies with the most reactive fuel parameters. The 8x8 assemblies have slightly higher heavy metal content but with significantly reduced maximum enrichment than the ATRIUM-10A and GE12 LUA. The staff results provided additional assurance that the ATRIUM-10A is the most reactive fuel type.

The staff evaluated the effect on reactivity that the thicknesses of the water and fuel channels had with a flooded geo69 package. The staff evaluations confirm that the minimum water channel thickness results in the highest calculated k_{eff} , and that thinner or absent water channels will result in higher reactivity. The staff evaluations showed little to no reactivity effect with varying thickness of the fuel channel.

To determine any effect the presence of the moderator rods might have on reactivity, the staff modeled the inner and outer moderator rods as homogenized annular cylinders matching the volume fraction of polyethylene and cast iron content in the moderator rod regions. The staff results showed no significant effects on calculated k_{eff} .

To determine if the presence of moderator in the bottom closure has any effect, the staff varied the materials in this region of the model. For full moderation, staff modeled the moderator plate as polyethylene and the void space flooded with full-density water. The staff repeated this evaluation with the polyethylene and empty space modeled as void. Staff results showed no significant effects on calculated k_{eff} .

6.9 Evaluation Findings

- F6.1** The staff has reviewed the CASTOR® geo69 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).
- F6.2** The staff has reviewed the CASTOR® geo69 package and concludes that the application identifies the codes and standards used in the package's criticality safety design in compliance with 10 CFR 71.31(c).
- F6.3** The staff has reviewed the CASTOR® geo69 package and concludes that the application specifies the number of packages that may be transported in the same vehicle through provision of an appropriate CSI in compliance with 10 CFR 71.35(b).
- F6.4** The staff has reviewed the CASTOR® geo69 package and concludes that the applicant used packaging features and package contents configurations and materials properties in the criticality safety analyses that are consistent with and bounding for the package's design basis, including the effects of the normal conditions of transport and the relevant accident conditions in 10 CFR 71.73. The applicant has adequately identified the CASTOR® geo69 package configurations and material properties that result in the maximum reactivity for the single package and package array analyses.
- F6.5** The staff has reviewed the CASTOR® geo69 package and concludes that the criticality evaluations in the application of a single package demonstrate that it is subcritical under the most reactive credible conditions, in compliance with 10 CFR 71.55(b), 71.55(d), and 71.55(e). The 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).
- F6.6** The staff has reviewed the CASTOR® geo69 package and concludes that the criticality evaluation in the application of the most reactive array of 5N undamaged packages demonstrates that the array of 5N packages is subcritical under normal conditions of transport to meet the requirements in 10 CFR 71.59(a)(1).
- F6.7** The staff has reviewed the CASTOR® geo69 package and concludes that the criticality evaluation in the application of the most reactive array of 2N packages subjected to the tests in 10 CFR 71.73 demonstrates that the array of 2N packages is subcritical under hypothetical accident conditions in 10 CFR 71.73 to meet the requirements in 10 CFR 71.59(a)(2).
- F6.8** The staff has reviewed the CASTOR® geo69 package and concludes that the applicant's evaluations include an adequate benchmark evaluation of the calculations. The applicant identified and evaluated experiments that are relevant and appropriate for the package analyses and performed appropriate trending analyses of the benchmark calculation results. The applicant has determined an appropriate bias and bias uncertainties for the criticality evaluation of the package.

F6.9 The staff has reviewed the CASTOR® geo69 package and concludes that the application identifies the necessary special controls and precautions for transport, loading, unloading, and handling and, in case of accidents, compliance with 10 CFR 71.35(c).

F6.10 The staff has reviewed the CASTOR® geo69 package and concludes that the evaluations in the application assume unknown properties of the fissile contents are at credible values that maximize neutron multiplication consistent with 10 CFR 71.83.

Based on review of the statements and representations in the application, the staff has reasonable assurance that the proposed CASTOR® geo69 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

- 6-1 U.S. Nuclear Regulatory Commission, "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," NUREG-2216, August 2020.
- 6-2 Organization for Economic Cooperation and Development – Nuclear Energy Agency, "International Handbook of Evaluated Criticality Safety Benchmark Experiments," NEA/NSC/DOC(95)03, 2014.
- 6-3 Oak Ridge National Laboratory, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages," NUREG/CR-6361, March 1997.
- 6-4 Idaho National Laboratory, "ICSBEP Guide to the Expression of Uncertainties," November 2007.
- 6-5 Science Applications International Corporation, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," NUREG/CR-6698, January 2001.
- 6-6 Oak Ridge National Laboratory, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," NUREG/CR-5661, April 1997.
- 6-7 A. Hoefer et al., "Bias and Correlated Data, Comparison of Methods," ICNC 2023 - The 12th International Conference on Nuclear Criticality Safety, October 2023.
- 6-8 Letter from GNS Gesellschaft für Nuklear-Service mbH to US Nuclear Regulatory Commission, V1130091-FSc, October 2, 2024.

7.0 MATERIALS EVALUATION

The staff's objective for the materials evaluation is to ensure that the SSCs ITS comply with the regulatory requirements in 10 CFR Part 71. The staff reviewed and evaluated the information provided by the applicant using the guidance in chapter 7 of NUREG-2216, (Published August 2020), "Standard Review Plan for Transportation Packages for Spent Fuel and Radioactive Material," to reach a conclusion of adequate materials performance under NCT and HAC. The areas of review covered in this SER section included system design, engineering drawings, material selection and material properties, environmental conditions and material compatibility.

7.1 Drawings

The applicant provided the package drawings in SAR appendices 1-3 through 1-10. The drawings include parts lists that identify which components are ITS and provide the material specifications and dimensions of each component. The drawings also detail the weld fabrication and examination requirements, including references to the applicable provisions of the ASME BPVC, as documented below in the staff's review of welding criteria. The staff reviewed the drawing content with respect to the guidance in NUREG/CR-5502, "Engineering Drawings for 10 CFR Part 71 Package Approvals," and confirmed that the drawings provide an adequate description of the materials and fabrication requirements, and, therefore, the staff finds them to be acceptable.

7.2 Materials Codes and Standards

As described in SAR section 2.1.4, "Identification of Codes and Standards for Package Design," the governing code for the package containment design, assembly, inspection, and examination is ASME BPVC section III, division 3, "Containment Systems for Transportation and Storage of Spent Nuclear Fuel and High-Level Radioactive Material," subsection WB, "Class TC Transportation Containments." The use of this code is consistent with the guidance in NUREG-2216, and, therefore, the staff finds it to be acceptable. The staff notes that subsection WB references ASME BPVC section II, "Materials" for materials procurement and properties. The staff's review of those aspects of the application follows.

The drawings and SAR section 2.2.1, "Material Properties and Specifications," describe the materials codes and standards for the procurement of the package materials. The staff reviewed the materials codes and standards to verify that they are consistent with design codes, when applicable, or are otherwise appropriate for the component.

The ITS SSC are primarily constructed of materials procured in accordance with ASME BPVC, section II, "Materials," with some exceptions described below. The outer cask body is constructed of ASME SA-874M DCI with a Type 304 or 316 austenitic stainless steel lid that conforms to either SA-182M, SA-240M, or SA-965M. The inner canister consists of an SA-240M Grade 316L stainless steel body and SA-965M Grade FXM-19 (nitride-strengthened austenitic stainless steel) lid. The cap screws that secure both the outer cask lid and engage the inner canister lid clamping elements are constructed of SA-540M Grade B22 chromium-molybdenum alloy steel. The cask lid protection system consists of a European Standard (EN) 10025-6 high strength steel penetration protection plate, EN AW-7075 high strength aluminum load distribution plate, and an EN AW-5383 aluminum spacer. The cap screws that secure the impact limiters, protection plate system, and trunnions to the cask are constructed of SA-193 Grade B6 high strength ferritic stainless steel. Various additional ITS SSCs that are not specifically detailed above (e.g., fuel basket peripherals, minor cask and impact limiter subcomponents) also are constructed of aluminum and steels that conform to ASME or EN standards.

The package also includes several proprietary materials that do not conform to a consensus code or standard, including the borated aluminum metal matrix composite fuel basket sheets, containment gaskets (seals), the polymeric shielding material, and the foam impact limiter. The staff's reviews of each of these materials are separately documented in their respective SER sections below. In addition, material of particular importance, namely DCI, have additional specifications, in-process examination, and tests detailed in SAR section 8.1.1 and appendix 2-12.

Based on the staff's verification that all non-proprietary materials of construction appropriately reference ASME BPVCs or European standards, the staff finds the applicant's design and procurement of the package materials to be acceptable.

7.3 Weld Design and Inspection

The applicant described the canister welding requirements in SAR section 2.3.1, "Fabrication," section 8.1.2, "Weld Examinations," and the package drawings. All welds of the containment (canister) are performed in accordance with ASME BPVC section IX, "Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operations." The package drawings include weld symbols that describe the weld location, geometry, and examination requirements.

The canister body welds are full penetration groove welds designed in accordance with ASME BPVC section III division 3, article WB, "Class TC Transportation Containments." Examinations of the canister welds are performed in accordance with ASME BPVC section III division 3 and include visual, radiographic, and liquid penetrant methods. Welds are also present in the aluminum casing of the impact limiter; however, the casing plates are classified as not important to safety.

The staff reviewed the welding criteria and verified that the ITS welds are designed, fabricated, and examined in a manner that is consistent the ASME BPVC, and therefore the staff finds them to be acceptable.

7.4 Mechanical Properties

7.4.1 Tensile Properties of Metallic Materials

The applicant described the mechanical properties of the structural materials in SAR section 2.2.1, "Materials Properties and Specifications," and proprietary appendix 2-6, "Material Data – Transport Package, CASTOR® geo69." The applicant stated that, unless otherwise noted, the property requirements used in the structural analyses are in accordance with the ASME BPVC, section II and section III, division 3.

For the containment and cask load attachment SSCs, the applicant provided tables of allowable stress criteria, thermal expansion coefficients, and elastic moduli in SAR section 2, "Structural Evaluation." The staff reviewed the tensile properties of the steel containment SSCs and verified that they are consistent with the values in the ASME BPVC, section II, Part D, tables 2A, 3, and TM-1.

For the aluminum alloys in the fuel basket and impact limiter, SAR section 2.2.1 includes tables of minimum allowable strengths and elongations. The staff reviewed the tensile properties of the aluminum SSCs and verified that they are consistent with the ASME BPVC section II or available values from the applicable European Standards. The staff's review of the mechanical properties of the aluminum metal matrix composite basket plates are documented in SER section 7.6, "Criticality Control Materials."

Based on the staff's verification that the properties used in the structural analysis are consistent with the applicable ASME BPVCs or European standards, the staff finds the tensile properties to be acceptable.

7.4.2 Fracture Resistance of Ferrous Materials

SAR tables 8.1-6 and 8.1-7 specify the toughness testing and acceptance criteria to verify the packaging materials have adequate low-temperature brittle fracture performance. The components requiring testing include the DCI cask body and the ferritic steel trunnion, trunnion attachment cap screws, cask lid cap screws, and the pressure nut associated with the canister top plug. The staff notes that the ASME BPVC section III does not have toughness testing requirements to verify low-temperature fracture performance for the metallic SSCs of the transportation package that are constructed of austenitic stainless steels and aluminum alloys, given the inherent toughness of these materials. Staff notes that SAR section 8.1.1 “Materials Testing” states that any nonconformities repairs of the casted cask body are not permitted.

SAR appendix 2-12, “Material Qualification Ductile Cast Iron” section 5.2 “Fracture Toughness” states that cast iron cask body is characterized with rapid-load fracture toughness testing at -40 degrees C with acceptance criteria consistent with ASME BPVC, section III division 3, subsection WB-2331.4, “Acceptance Standards for Ductile Cast Iron for Containments”. The trunnions are toughness tested in accordance with Subsection WB-2331.1, and the acceptance criteria are based on demonstrating the nil ductility transition temperature is sufficiently below the lowest service temperature per table WB-2331.2-1, “Required LST-RT_{NDT} Values for Ferritic Steel Material for Containment Material”. The remaining fastener components are tested to verify a minimum Charpy lateral expansion value that is consistent with ASME BPVC table WB-2331-1, “Required C_v Values for Bolting Material”.

Based on the staff’s verification that the applicant will conduct testing in a manner consistent with the ASME BPVC, the staff finds that the applicant has controls in place to ensure adequate low-temperature fracture performance of the DCI and alloy steel SSCs.

7.4.3 Mechanical Performance of Impact Limiter Foam

The applicant provided the mechanical properties of the polyurethane foams used in the impact limiters in SAR sections 2.2.3.10 and 2.2.3.11, and acceptance criteria for polyurethane foam as a shock absorber in section 8.1.6.1.5, “Polyurethane foam materials tests” and table 8.1-12. The impact limiters use General Plastics LAST-A-FOAM® 3700 series, which are described as chlorofluorocarbon-free, rigid, closed-cell, and flame-retardant. The foam designations differ with respect to their density, which is established by incorporating different gas contents during manufacture.

The bases for the foam properties used in the structural are described in SAR appendix 2-5 (sub-appendix 11), “Characterization of a Polyurethane foam material for different densities, strain rates and temperatures for application in LS-DYNA as a shock absorber material,” and appendix 2-10, “Material Qualification, Polyurethane Foam.” The applicant relied on both the manufacturer data (General Plastics) and the applicant’s own mechanical testing to support the properties used in the safety analyses. The applicant also cited the manufacturer’s test data to demonstrate that exposure to radiation and water will not significantly affect the mechanical performance of the foams. The staff reviewed the applicant’s test data and the manufacturer’s data and verified that it adequately supports the deformation behavior assumed in the structural analyses. The staff notes that the data demonstrates that neither radiation nor water exposure significantly affects the foam performance. Therefore, the staff finds the mechanical properties of the impact limiter foam used in the accident analyses to be acceptable.

7.4.4 Elevated Temperature Performance of Aluminum Basket Components

For the aluminum basket periphery components exposed to elevated service temperatures (i.e., solid aluminum shielding elements and round segments), the applicant cited mechanical properties of soft annealed, temper round segment EN AW-5454 aluminum. The staff notes that such a temper designation is consistent with a wrought aluminum product that is exposed to the elevated temperatures within the fuel basket. Therefore, the staff finds the properties of the aluminum basket periphery components to be acceptable.

The staff's review of the high temperature performance of the aluminum metal matrix composite fuel basket plates is documented in SER section 7.6, "Criticality Control Materials".

7.5 Thermal Properties of Materials

SAR appendix 2-6, "Material Data, Transport Package, CASTOR® geo69," and SAR tables 3.2-1 through 3.2-4 describe the material properties used in the thermal analyses, including density, thermal expansion, thermal conductivity, and specific heat capacity. For the metallic SSCs, the applicant generally based the properties on the applicable ASME BPVC or European standards. For the proprietary materials, the properties are based on the applicant's qualification data and manufacturer technical data sheets that are described in proprietary SAR appendices 2-7 (moderator shielding), 2-8 (fuel basket plates), and 2-10 (impact limiter foam).

The staff reviewed the applicant's thermal property values against those described in ASME BPVC section II, "Materials," Part D, "Properties," European steel data, technical handbooks, material qualification data, and manufacturer technical data sheets.

The staff verified that the material properties used in the thermal analyses are appropriate and consistent with the applicable standards or the manufacturer test data. Therefore, the staff finds the thermal properties of the components for the CASTOR® geo69 transportation package to be acceptable.

In addition, SAR table 3.2-9, "Temperature Limits of Components," identifies the service temperature limits for the fuel rods, seals, polyethylene neutron shielding, fuel basket sheets, and impact limiter foam. For the shielding analysis of the hypothetical fire accident, the applicant conservatively assumed that the neutron shielding material and the impact limiter foam are completely lost.

The staff reviewed the maximum allowable temperatures and verified that they are consistent with values specified in NUREG-2216 (fuel cladding temperatures) and limits identified by the manufacturer for the proprietary materials. For the ultrahigh molecular weight polyethylene neutron shielding material, the applicant's testing described in the proprietary report in SAR appendix 2-7, subjected the material to furnace treatments to demonstrate that the material is unaffected at temperatures greater than the maximum allowable NCT temperature described in SAR table 3.2-1 "Material data of cask and canister components". The staff notes that the technical literature also supports the stability of the ultrahigh molecular weight polyethylene at the NCT service temperatures (Bala, 2016). Therefore, the staff finds the applicant's analyses for the CASTOR® geo69 transportation package component temperature limits to be acceptable.

7.6 Radiation Shielding Materials

7.6.1 Neutron shielding

Neutron shielding is primarily provided by ultrahigh molecular weight polyethylene in the outer cask, although additional SSCs also contribute to neutron shielding performance of the package. SAR table 5.3-1 “Material properties in the shielding model” provides the material densities and compositions of the SSCs considered in the shielding analysis.

The staff reviewed the material property data used in the shielding analysis and verified that they are consistent with values available in technical handbooks and the polyethylene manufacturer’s technical data (Celanese). The staff also verified that the shielding performance of the SSCs are not diminished under NCT conditions and that the allowable service temperatures of the materials are not exceeded. In the HAC analysis, the polyethylene and the impact limiter foam are not relied on for shielding, as they are assumed to be lost.

In addition, the acceptance testing requirements in SAR section 8.1.6.1.2, “Moderator Material Tests,” include verification that each plate of the polyethylene has the minimum required density and is within the allowable thermal expansion.

Therefore, the staff finds the material properties used in the neutron shielding analysis to be acceptable and that the applicant includes sufficient controls to ensure the manufactured plates meet the property requirements.

7.6.2 Gamma shielding

Gamma shielding is primarily provided by the stainless steel and DCI SSCs of the inner canister and outer cask, although additional SSCs also contribute to gamma shielding performance of the package. The staff reviewed the gamma shielding material densities and compositions in SAR table 5.3-1 “Material properties in the shielding model” and verified that they are consistent or conservative with respect to the values available in technical handbooks and ASME BPVC, section III, Part D, table PRD, “Poisson’s Ratio and Density of Materials”. Therefore, the staff finds the material properties used in the gamma shielding analysis to be acceptable.

7.7 Criticality Control Materials

The fuel basket plates are borated aluminum metal matrix composite (Al-MMC) plates that provide both structural support and criticality control. The plates consist of a type 3004 aluminum alloy matrix with a uniform distribution of boron carbides (B_4C). Proprietary SAR appendix 2-8, provides details of the qualification and acceptance testing for Al-MMC material.

The manufacturer’s qualification data is provided in proprietary SAR Attachment I. The report describes qualification test results related to chemical composition, carbide particle size, product dimension variability, density, mechanical properties, thermal properties, corrosion, and behavior under irradiation and elevated temperatures.

Although nonferrous materials are generally excluded from fracture acceptance testing in consensus standards and the proprietary metal matrix composite is a non-code material that contains boron carbide ceramic particles that effect the fracture performance relative to the conventional aluminum material that are considered in ASME BPVC. To address this, the applicant provided scanning electron microscope observations of the fracture surfaces of test coupons of static tensile test and dynamically cracked (Charpy V-notch) test data at low-

temperature that's demonstrate the ductile nature of Al-MMC. Staff notes that mechanical properties provided in SAR appendix 2-8 are acceptable and of comparable range for similar aluminum-metal matrix composite materials. The applicant showed that the use of these properties for structural analysis provides adequate margin of safety that does not exceed material yield strength under NCT and HAC.

SAR section 8.1.6.1.1, "Neutron absorber material tests," describes the proposed acceptance testing to ensure each batch of material has adequate structural integrity, neutron absorption, and heat dissipation characteristics. The acceptance testing description references the qualification report in SAR appendix 2-8. Section 2, "Proposed Acceptance Criteria to be tested during Manufacturing." That appendix includes requirements for chemical composition, tensile properties, boron distribution (via neutron attenuation or combustion analysis), general surface condition, and dimensions. SAR appendix 2-8 section 2.6 "Visual Inspection of Final Profile or Plates" states all extruded plates are visually inspected before and after anodization for surface defects like scratches, notches, grooves, chipping, and cleanliness along with evaluation criteria.

The staff reviewed the qualification and acceptance testing of the Al-MMC fuel basket plates and verified that the material properties relied on in the structural and criticality evaluations are adequately supported by the manufacturer's data. In SAR section 6.3.2 "Material Properties" the applicant stated that the criticality calculation only takes credit for 75% of minimum specified boron content which is conservative with respect to the guidance in NUREG-2216. Additionally, the applicant's proposed acceptance testing for each batch of material is consistent with the guidance in NUREG-2216 and is considered to be capable of ensuring that those properties will be met in the manufacture of the CASTOR® casks. Therefore, the staff finds that the applicant has demonstrated that the material will be capable of performing its intended functions.

7.8 Corrosion Resistance

In SAR section 1.2.1.13, "Corrosion Protection," the applicant stated that all packaging surfaces that are exposed to the environment during transport, or come into contact with water during loading, are either inherently corrosion resistant or are protected by measures such as a coating. SAR section 2.2.8 "Chemical, Galvanic, or other Reactions" notes that the principal component exposures are (1) basket and canister exposure to pool water during fuel loading, after which they are in an inert helium environment, and (2) cask and impact limiter exterior surface exposure to the ambient environment. SAR section 2.2.8 also contains a table that identifies the service exposures of each package component, with a justification that chemical and galvanic reactions will not occur to an extent that they would detrimentally affect component safety functions, operations, or package performance.

Regarding the components exposed to pool water during fuel loading, the applicant stated that corrosive reactions are minimized due to the inherent corrosion resistance of the stainless steels, aluminum alloys, and gasket materials (nickel, silver), as well as the additional mitigation measure of anodizing the aluminum (including the borated aluminum metal matrix composites). The applicant also stated that galvanic couples can form between the stainless steel and aluminum; however, the brief water exposure and the anodization of the aluminum will minimize corrosion reactions. The staff notes that, as described in NUREG-2216, section 7.4.10.2, the NRC has previously reviewed a number of hardware components and materials to ensure that there are no significant chemical, galvanic, or other reactions as a result of exposure of these various contents to wet loading. In those evaluations, the staff concluded that the use of the stainless steel and aluminum materials to be acceptable because the materials have sufficient corrosion resistance to preclude corrosive reactions during the short-term fuel loading

operations. Also, the staff notes that the subject components are exposed to an inert helium environment after fuel loading, such that no additional corrosive reactions are expected during transport.

Regarding those components exposed to the ambient air environment during transport, the applicant stated that the materials generally either have a native corrosion resistance (stainless steels, aluminum) or are otherwise coated (e.g., epoxy-coated cask iron cask body, zinc-coated alloy steel cap screws). One exception is the steel penetration plate, which is covered by the impact limiter. The staff reviewed the material exposures and find that the material selections and coating mitigation measures provide sufficient corrosion resistance to preclude corrosive reactions during the transportation in ambient environmental exposures. The applicant provided an overview of all components with periodic testing and replacement schedule in SAR table 8.2-1.

Based on the discussions above, the staff finds that the applicant's package design to preclude significant corrosion to be acceptable.

7.9 Coatings

Steel and cast iron cask components are coated for corrosion prevention, ease of decontamination, and optimization of heat transfer. As described in SAR section 1.2.1.13.3, section 1.2.1.14, and table 3.2-8 "Surface emissivities", the outer and inner surfaces of the cast iron cask are coated with a multi-layer epoxy coating. In addition, as described in SAR sections 1.2.1.13.2 and 1.2.1.13.3, the alloy steel fasteners associated with the canister and cask lids are coated with zinc for corrosion protection. Coating requirements are provided in SAR section 2.2.7. The staff notes that although repairs of the coating are permitted, it's limited to local defects like scratches or chipping that may occur during handling operations and large-scale recoating is not permitted. Appendix 2-11 "Materials Qualification Report Coatings by Fa. Hempel" provided detailed requirements, descriptions, and properties for the coatings. The coating material has a qualified range of emissivity values that exceed the assumed values used in the SAR for thermal analysis. Staff notes that the thermal evaluation does not take credit for coating on the inner cask surface.

SAR chapter 7, "Package Operations," and chapter 8, "Acceptance Tests and Maintenance Program," describe the inspection of cask coatings and coated fasteners in preparation for loading, in preparation for transport, and per a periodic maintenance schedule.

The staff reviewed the applicant's use of coatings and the associated inspection activities and finds them to be sufficient to ensure that the cast iron and steel surfaces will be adequately protected against corrosion and will maintain the emissivity (heat transfer) characteristics credited in the thermal analysis. The inspections prior to fuel loading, prior to transport, and during periodic maintenance are considered to provide for timely identification of any degradation of the coating or underlying metal substrate, such that any adverse conditions can be promptly addressed.

7.10 Content Reactions

As described in SAR table 1.2-11, "Characteristics of the fuel assemblies," the fuel assembly cladding, water rods, and fuel channels are made of zirconium alloys, while the tie plates are stainless steel. As the FA were designed for, and in prior service in, a comparatively severe reactor water environment, their exposure to fuel pool water during fuel loading into the canister is not considered to introduce corrosion reactions that would significantly affect the assembly

performance. Also, as the cask cavity is subsequently dried and inerted with helium, no further content reactions are considered to be credible during transport.

SAR section 1.2.2 “Contents” states that only undamaged FA will be allowed for loading and do not contain any moisture after the cask drying procedure. The applicants stated in SAR section 7.1.2 “Loading of Contents” that due to the continuous vacuum drying, the amount of hydrogen produced (due to minimized galvanic reaction of aluminum and stainless steel and due to radiolysis of water as stated in SAR section 2.2.8 “Chemical, Galvanic, or other Reactions”) is tolerable during loading operation of the canister (before drying and refilling with helium). The steps for dewatering, drying, final inertization with helium and subsequent closure are carried out sequentially and without interruption and the canister is never left with an uncontrolled or undefined interior atmosphere, hence the FA are not exposed to any non-inert environment. In addition to the corrosion resistance of the packaging materials and contents as discussed above, the staff notes that the package does not include zinc or other coatings on the components that may undergo adverse (hydrogen-producing) reactions when submersed in the fuel pool. Finally, the lid closures of the canister and cask are bolted, thus precluding closure welding-related ignition of any flammable gases. Therefore, the staff finds that the package’s design to preclude adverse content reactions to be acceptable.

7.11 Radiation Effects

The applicant evaluated the effects of radiation on materials performance in SAR section 2.2.3, “Effects of Radiation on Materials.” SAR appendix 5-3, table 1, “Energy doses and neutron fluxes for different packaging components,” provides the calculated accumulated neutron and gamma radiation dose for the packaging components.

The applicant stated that metallic components are not impaired by either gamma or neutron radiation. In support of that conclusion, the applicant cited the inherent resistance to gamma radiation effects and relatively low neutron exposure (approximately 10^{15} n/cm² for 60 years) compared to the levels needed to damage the metals used in the package (greater than 10^{18} n/cm²). The staff noted that the applicant’s conclusion is consistent with the staff conclusions in NUREG-2216 and NUREG-2214, in which steels, stainless steels, and aluminum are not considered to be susceptible to changes in properties when exposed to levels of radiation expected for spent fuel storage systems over extended exposure (up to 60 years). Therefore, the staff finds that the metallic package components will adequately maintain their structural properties in the radiation service environment.

The staff also evaluated potential radiation effects on the polymeric package materials (foam impact limiter and polyethylene shielding). Regarding the impact limiter foam, the staff notes that the foam is external to the principal cask shielding components, and thus it is not exposed to significant levels of radiation. Also, as described in the manufacturer’s LAST-A-FOAM® design guide (General Plastics, 2015), laboratory exposures of 2×10^8 rads (gamma) did not affect the mechanical performance of the foam. This laboratory exposure is similar to the level of radiation the applicant stated is experienced by the interior cask shielding components, as stated in SAR section 2.2.3. Therefore, the staff finds that the impact limiter foam will adequately maintain its structural performance in the radiation service environment.

Regarding the ultrahigh molecular weight polyethylene shielding, the applicant stated that the approximately 10^6 Gy ($\sim 10^8$ rad) gamma exposure over 60 years of service is insignificant to shielding performance.

Based on the information provided, the radiation effects are minimal, and staff finds the package's design to be acceptable.

7.12 Spent Nuclear Fuel

7.12.1 Fuel Classification

SAR tables 1.2-12 "Characteristics of the fuel assemblies" describe the allowable spent fuel contents. The fuel rod cladding is constructed of Zircaloy-2, and the internal water rods and channels are constructed of Zircaloy-2 and Zircaloy-4, respectively. The maximum allowable assembly-averaged burnup is 58 GWd/MTHM.

SAR section 1.2.2 "Contents" states that only undamaged fuel assemblies may be loaded into the cask. Fuel assemblies which have been deformed or damaged during reactor operation or which are otherwise defective in their structural integrity and do not meet the fuel-specific or system-related regulations are not to be loaded. SAR section 0.5, "Glossary," defines undamaged fuel according to NUREG-2216 section 7.4.14.1 (fuel that can meet all fuel-specific and system-related functions).

The applicant's safety analyses assume reconfiguration of the fuel rods, and thus the mechanical properties of the cladding are not relied on in the structural analysis of the package.

7.12.2 Fuel Drying and Peak Cladding Temperature

SAR section 7.1.2 "Loading of Contents" describes the cask drying process following loading of the fuel contents. Both the cask and canister cavities are vacuum dried and inerted with helium. There is an additional requirement to verify that, after vacuum drying, the cavity pressure does not rise by more than 0.1 kPa per 15 minutes over the subsequent 60-minute period.

SAR sections 3.2.2 "Component Specifications" and 3.5.5.2 "Maximum Temperatures" define the maximum allowable cladding temperature and provide a safety analysis that shows that the temperature of 400°C for NCT and short-term loading operations and 570°C for HAC is not exceeded. The staff notes that these criteria are consistent with the guidance in NUREG-2216 section 7.4.14.2 Uncanned spent fuel.

7.13 Bolting Material

As discussed above in SER section 2.2, the principal bolting components in the package include (1) SA-540M Grade B22 chromium-molybdenum alloy steel cap screws that secure the outer cask lid, (2) SA-193 Grade B6 high strength ferritic stainless steel cap screws that secure the cask trunnion and secure the impact limiters and protection plate system to the cask, and (3) SA-540M Grade B22 thread bolts that engage the inner canister lid clamping elements.

The staff's evaluations of the brittle fracture performance and the corrosion resistance of the bolting materials are documented above in SER sections 7.4.2 and 7.8, respectively.

7.14 Seals

As described in SAR section 1.2.1.8.3 "Metal Gaskets and Seals" and the package drawings, the lid containment seals of the outer cask and inner canister are metallic helical spring gaskets that consist of an SA-240M Type 304L stainless steel inner liner, Nimonic® 90 nickel-chromium-cobalt alloy spring, and pure silver outer jacket. Proprietary SAR appendix 2-9, "Material

Qualification, Metal Gaskets,” describes the gaskets in greater detail, including their performance at elevated temperatures and radiation levels over extended service times.

The staff reviewed the applicant’s proprietary report on the metal containment seals and verified that the applicant appropriately evaluated for potential changes in mechanical properties and sealing performance at the temperatures and radiation levels present in the package. In addition, the staff notes that the applicant’s evaluation for corrosion resistance is consistent with the staff’s independent evaluation in SER section 2.8, above, in which the silver, nickel, and stainless steel components of the seals are considered to have sufficient corrosion resistance to preclude degradation that could affect sealing performance. Finally, in addition to the seal leakage testing during cask loading, SAR section 8.2.5 includes requirements for periodic visual examination and leakage testing of the seals at intervals of 15 transports or 3 years (when the package is empty).

Therefore, based on the applicant’s qualification data and periodic testing of seal performance, the staff finds that there is reasonable assurance that the metallic seals are capable of performing their containment function.

The staff notes that the package contains several elastomeric seals (e.g., test seals that support the package leakage testing operations); however, as stated in SAR section 1.2.1.8.3, those seals do not perform a safety function.

7.15 Material Evaluation Findings

- F7.1** The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.33. The applicant described the materials used in the transportation package in sufficient detail to support the staff’s evaluation.
- F7.2** The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.31(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.
- F7.3** 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.
- F7.4** 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.
- F7.5** 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.
- F7.6** The staff has reviewed the package and concludes that the applicant has met the requirements of 10 CFR 71.55(a) 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 transportation package design have been adequately described and evaluated and that the package meets the requirements of 10 CFR Part 71.

7.16 References

- 7-1 Bala, A.S. et al., "Processability and Thermal Properties of Ultra-high Molecular Weight Polyethylene/Polypropylene Blends," Journal of Engineering and Applied Sciences, Asian Research Publishing Network, Vol. 11, No. 8, April 2016.
- 7-2 General Plastics. "Design Guide: LAST-A-FOAM® FR-3700 Crash and Fire Protection of Radioactive Material Shipping Containers" (2015); and data sheets: "FR-3700-PDS" and "FR-3700-Extended-TDS_METRIC."

8.0 PACKAGE OPERATIONS

The staff reviewed the operating procedures for the transport package CASTOR® geo69 in chapter 7 of the application to ensure that the procedures reflect the acceptable operating sequences, guidance, and generic procedures for key operations as represented in the shielding analysis and meet the requirements of 10 CFR Part 71. The staff reviewed the loading procedures and finds that the applicant considered ALARA principles and surface contamination survey and has steps associated with the operations for preparation for loading, loading of contents and preparation for transport loading when needed.

8.1 Compliance with Dose Rate Limits

Table 7.1.3 of the application contains the loading procedure for determining if the contents are acceptable for loading. The staff reviewed this procedure and found that it is consistent with the analysis in the shielding evaluation in chapter 5 of the application as discussed in section 5.8 of this SER and that the procedure will ensure that the package as loaded will be below regulatory dose rate limits.

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The staff reviewed the acceptance tests and maintenance programs that are important to ensuring the shielding in the as-fabricated package meets the design specified in the technical drawings, as evaluated in the shielding analysis at the time of fabrication and use and will continue to do so over the course of its service life. The staff evaluated the information in section 8.1.6 of the application related to "Shielding Tests." The applicant states that no acceptance testing is required because shielding components are made from ductile cast iron. The staff accepts inspections of the package dimensions as a sufficient acceptance test for ensuring shielding performance of the steel components. These components are manufactured to industry standard specifications, and they are not subject to the material irregularities faced from non-standard materials or a cast poured but it is machined afterwards. Section 8.1.5 of the application requires that the dimensions and tolerances be verified by measurement on each package. The staff found this acceptable to ensure that the shielding is manufactured in accordance with the drawings in lieu of acceptance tests.

The applicant did not identify any maintenance tests that will need to be performed on the transport package CASTOR® geo69 in relation to the shielding performance. The staff has not identified any degradation mechanisms that would affect the shielding performance during the service lifetime of the package and found this acceptable.

10.0 QUALITY ASSURANCE

The staff reviewed the Quality Assurance Program Description (QAPD) as a part of a new application submittal for a certificate of compliance for the Model No. CASTOR® geo69 spent fuel transportation package. The applicant submitted the QAPD for review on October 7, 2020 (ML20198M431), May 15, 2023 (ML23257A089), and June 27, 2024 (ML24192A182).

The staff finds that the QAPD for the transportation packaging meets the requirements in subpart H to 10 CFR Part 71 and address all 18 criteria as applicable. The staff finds that the QAPD encompasses quality assurance program controls, as appropriate, to ensure that the package will allow safe transport of the radioactive material authorized in an NRC Quality Assurance Program approval under docket No. 071-00967. The staff reached this finding based on a review that considered applicable NRC regulations and regulatory guides and the statements and representations contained in the application.

CONDITIONS

In addition to the package description, drawings and contents, the following conditions were included in the CoC:

Condition 6 states that in addition to the requirements of Subpart G of 10 CFR Part 71, the CASTOR® geo69 package shall:

- (a) Be prepared for shipment and operate in accordance with the Package Operations in chapter 7 of the application,
- (b) Meet the Acceptance Tests and Maintenance Program of chapter 8 of the application,
- (c) During transport, the fuel basket's aluminum-boron metal matrix composite (Al-B₄C-MMC) neutron absorber shall have no contact with the outer environment,
- (d) B₄C content in the aluminum-boron metal matrix composite (Al-B₄C-MMC) shall be in accordance with section 3, table 6 of appendix 2-8 revision 6 of the application,
- (e) Mechanical properties of the aluminum-boron metal matrix composite (Al-B₄C-MMC) shall be in accordance with section 3, table 4 of appendix 2-8 revision 6 of the application,
- (f) Acceptance criteria for uniform boron distribution of the aluminum-boron metal matrix composite (Al-B₄C-MMC) shall be in accordance with section 2.5.2 of appendix 2-8 revision 6 of the application,
- (g) Visual inspection criteria for of the aluminum-boron metal matrix composite (Al-B₄C-MMC) shall be in accordance with section 2.6 of appendix 2-8, revision 6 of the application,
- (h) Acceptance criteria for moderator material shall be in accordance with section 8.1.6.1.2 of the application,

- (i) Acceptance criteria for moderator material mechanical properties shall be in accordance with section 5.2, table 5-1 of appendix 2-7, revision 2 of the application,
- (j) Acceptance criteria for moderator material thermal properties shall be in accordance with section 5.2, table 5-2 of appendix 2-7, revision 2 of the application,
- (k) Acceptance criteria for moderator material chemical composition shall be in accordance with section 5.2, table 5-3 of appendix 2-7, revision 2 of the application, and
- (l) All test procedures used in demonstrating compliance with the above requirements shall conform to those in the application.

Condition 7 states that the package must be transported under exclusive use.

Condition 8 states that if a personnel barrier is placed over the package, an evaluation will be performed to ensure the barrier does not result in package temperatures above allowable values in the application due to impeded heat transfer.

Condition 9 states that prior to transport by rail, the Association of American Railroads must have evaluated and approved the railcar, and the system used to support and secure the package during transport.

Condition 10 states that prior to marine or barge transport, the National Cargo Bureau, Inc., must have evaluated and approved the system used to support and secure the package to the barge or vessel, and must have certified that package stowage is in accordance with the regulations of the Commandant, United States Coast Guard.

Condition 11 states that transport by air is not authorized.

Condition 12 states that the CASTOR® geo69 package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

Condition 13 states that the packages expiration date is November 30, 2029.

CONCLUSION

Based on the statements and representations in the application, as supplemented, and the conditions listed above, the staff concludes that the Model No. CASTOR® geo69 package design has been adequately described and evaluated, and that these changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with CoC No. 9383, Revision No. 0.