SAFETY EVALUATION REPORT

Docket No. 71-9356

Model No. MAGNATRAN Package

Certificate of Compliance No. 9356

Revision No. 0

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SUMMARY

By application dated November 26, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13010A094), as supplemented, February 15, 2013 (ADAMS Accession No. ML13107B451), March 29, 2013 (ADAMS Accession No. ML13093A233), December 1, 2014 (ADAMS Accession No. ML14356A385), January 13, 2015 (ADAMS Accession No. ML15023A393), January 21, 2015 (ADAMS Accession No. ML15023A050), October 15, 2015 (ADAMS Accession No. ML15296A084), May 9, 2016 (ADAMS Accession No. ML16137A061), August 11, 2016 (ADAMS Accession No. ML16229A205), January 11, 2017 (ADAMS Accession No. ML17017A184), January 13, 2017 (ADAMS Accession No. ML17031A401), February 2, 2017 (ADAMS Accession No. ML17034A378), October 6, 2017 (ADAMS Accession No. ML17291A395), April 9, 2018 (ADAMS Accession No. ML18102A852), August 30, 2018 (ADAMS Accession No. ML18262A298), October 30, 2018 (ADAMS Accession No. ML18311A259), and March 25, 2019 (ADAMS Accession No. ML19086A116), NAC International (NAC), submitted a request for a new certificate of compliance (CoC) for the Model No. MAGNATRAN as a Type B(U)F-96 package. NRC staff reviewed the application, as supplemented, using the guidance in NUREG-1617[1].

The package was evaluated against the regulatory standards in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71, including the general standards for all packages and the performance standards specific to fissile material packages under normal conditions of transport and hypothetical accident conditions. The analyses performed by the applicant demonstrate that the package provides adequate structural and thermal protection to meet the containment, shielding, and criticality requirements after being subject to the tests for normal conditions of transport and hypothetical accident conditions.

Based on the statements and representations in the application, and the conditions listed in the CoC, the staff concludes that the package meets the requirements of 10 CFR Part 71. Therefore, U.S. Nuclear Regulatory Commission (NRC) is issuing CoC No. 9356, for the MAGNATRAN package, for a 5-year term expiring on March 31, 2024.

References

1. NUREG-1617, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, U.S. Nuclear Regulatory Commission, March 2000.

1.0 GENERAL INFORMATION

1.1 Packaging

The MAGNATRAN package is canister-based and is part of a dual-purpose system for the storage and transportation of spent nuclear fuel for transporting the MAGNASTOR® transportable storage canister (TSC). The MAGNATRAN packaging includes the package body, upper and lower impact limiters, and TSC. The package body consists of the inner and outer shells, lead and upper forging, lid, bottom plate, bottom forging and solid neutron shield.

The packaging body is a cylinder with multiwall construction consisting of inner and outer stainless steel shells separated by a lead gamma radiation shielding. The inner and outer stainless steel shells are 1.75 and 2.25 inches thick, respectively. The lead gamma shield is 3.2 inches thick. Welded above the inner and outer steel shells is the upper forging. The upper forging is 7.2 inches thick where it attaches to the inner and outer shells.

The bottom of the package body consists of the bottom inner forging, the bottom outer forging and the bottom plate. The bottom inner forging is cup shaped and welded to the inner shell and the bottom forging. The ring-shaped bottom outer forging is welded to the outer shell and to the bottom plate. The bottom plate is welded onto the outer ring. The bottom inner forging is 5 inches thick and the bottom plate is 8.65 inches thick for a total of 13.65 inches of stainless steel shielding through the bottom.

The package lid is a 7.75-inch-thick stainless steel disk used to close the package. The lid is attached to the top forging by forty-eight, 2-8 UN-2A socket head cap screws. The socket head cap screws screw into the tapped holes in the upper forging. The package lid is sealed by two concentric O-rings, as is the coverplate for the lid port, using inner metallic and outer ethylene propylene diene monomer (EPDM) O-rings. The MAGNATRAN package contains a lid port that is closed by a bolted Type 304/304L stainless steel coverplate with dual O-rings. There are four stainless steel coverplate bolts. The lid port provides access to the port opening and the quick-disconnect fitting for backfilling and sampling the cavity gas during loading and unloading.

The neutron shield is comprised of NS-4-FR encased in stainless steel enclosures. The neutron shield material and its enclosure have two thicknesses, 5.8 inches and 6.4 inches, and is attached onto the outer shell along the length of the active fuel region around the circumference of the package cavity

Two diametrically opposite lifting trunnions are bolted to the outside of the top forging to lift the transport package. Prior to transport, the lifting trunnions are removed and replaced with trunnion plugs. Two rotation trunnions are located on the outer shell near the bottom of the package to permit rotation between the horizontal and vertical positions and to provide longitudinal tiedown restraint in the aft direction. The rotation trunnions are located approximately 5 inches off the cask centerline to ensure that the cask rotates in the proper direction.

A cavity spacer is used for the short TSCs to locate and support the canister and to minimize excessive longitudinal movement in the transport cask cavity, which is sized to accommodate the long TSC.

The MAGNATRAN package has cup-shaped impact limiters, consisting of a combination of redwood and balsa wood encased in a stainless-steel shell. The impact limiters are bolted over each end of the package to limit the g-loads acting on the package during a package drop event.

The impact limiters are attached to the lid and bottom plate via 16 tapped holes for retaining rods and nuts.

The TSC is constructed of a stainless steel cylindrical shell, bottom-end plate, closure lid, closure ring, and redundant port covers. The TSC confines the fuel basket structure and the spent fuel or the Greater-Than-Class C (GTCC) waste basket liner and GTCC waste. The TSC cylindrical shell is dual certified 304/304L stainless steel with a 72-inch diameter and is ½ inch thick and either 191.8 or 184.8 inches long, depending on the contents. The bottom end plate is welded onto the lower end of the TSC shell and is 2.75 inches thick. The closure lid is 9 inches thick and is either a solid stainless steel closure lid or stainless steel/carbon steel closure lid. The closure lid is welded onto the upper end of the TSC shell. The dual port covers provide a dual-welded closure system for the vent and drain ports. The GTCC TSC is similar in design and construction to the TSC's for spent fuel, but instead of a basket, it contains a GTCC waste liner.

Each TSC containing spent fuel includes a pressurized-water reactor (PWR), PWR damaged fuel, or boiling-water reactor (BWR) fuel basket that positions and supports the spent fuel. Consistent with the TSC design, there are two different length fuel baskets (the two lengths are the same for both the PWR and the BWR fuel baskets), with the PWR damaged fuel basket being shorter. The design of the basket is similar for the PWR and BWR basket configurations.

The PWR fuel basket design is an arrangement of 21 square, stainless steel fuel tubes held in a right-circular cylinder configuration by side and corner support weldments that are bolted to the outer fuel tubes. The 21 tubes develop 37 positions within the basket for the PWR spent fuel. Each PWR basket fuel tube has a nominal 8.86-inch square opening. Each developed cell fuel position has a nominal 8.76-inch square opening. The fuel tubes support an enclosed neutron absorber sheet on up to four interior sides of the fuel tube. Each neutron absorber sheet is covered by a thin stainless steel sheet to protect the neutron absorber during fuel loading and to keep it in position. The neutron absorber and stainless steel cover are secured to the fuel tube using weld posts distributed across the width and along the length of the fuel tube. The neutron absorber sheets may be replaced by commercial aluminum sheets on the two outside surfaces of the eight outermost fuel tubes of the PWR fuel basket (refer to sheet 3 of Drawing No. 71160-575).

The PWR damaged fuel basket is designed to store up to four damaged fuel cans in the damaged fuel basket assembly in the short TSC. The damaged fuel basket assembly has a capacity of up to 37 undamaged PWR fuel assemblies, which includes the four damaged fuel can locations. A damaged fuel can may be placed in each of the four damaged fuel can basket locations. The arrangement of tubes and fuel positions is the same as in the standard fuel basket, but the design of each of the four corner support weldments is modified with additional structural support to provide an enlarged position for a damaged fuel can at the outermost corners of the fuel basket. Each damaged fuel can location has a nominal 9.80-inch square opening. A damaged fuel can or an undamaged fuel assembly may be loaded in a damaged fuel can location.

Similar to the PWR basket, the BWR basket consists of 45 stainless steel fuel tubes that develop 87 basket locations for the BWR spent fuel. Each BWR basket fuel tube has a nominal 5.86-inch square opening. Each developed cell fuel position has a nominal 5.77-inch square opening. The BWR basket fuel tubes are held in a right-circular cylinder configuration by side and corner support weldments that are bolted to the outer fuel tubes. The fuel tubes support an enclosed neutron absorber sheet on up to four interior sides of the fuel tube for criticality control. Each neutron absorber sheet is covered by a sheet of stainless steel to protect the neutron absorber during fuel loading and to keep it in position. The neutron absorber and stainless steel cover are secured to the fuel tube using weld posts distributed across the width and along the length of the fuel tube.

The neutron absorber sheets may be replaced by commercial aluminum sheets on the three outer surfaces of the outermost fuel tubes of the BWR fuel basket (refer to sheet 3, Drawing 71160-599).

The damaged fuel can confines the fuel material within the can to minimize the potential for dispersal of the fuel material into the TSC cavity. The side plates that form the upper end of the damaged fuel can are 0.15-in thick and the tube body walls are 0.048-in thick (18-gage sheet). The damaged fuel can lid plate and bottom thicknesses total 11/16 inches and the lid overall height is 2.32 inches. The damaged fuel can bottom plate thickness is 5/8 (0.625) inch. The damaged fuel can is designed in two lengths: an overall length of 166.9 inches with a nominal cavity length of 164.0 inches; or an overall length of 171.8 inches with a nominal cavity length of 169.0 inches (shorter fuel assemblies may be accommodated with a fuel assembly spacer to limit axial movement). For the shorter damaged fuel can, a spacer is used in the damaged fuel basket assembly or alternatively fixed to the damaged fuel can bottom plate to provide an overall height of 171.5 inches. The DFC lid and bottom include screened drain holes.

The stainless steel GTCC waste basket liner is designed to hold GTCC waste and dimensionally fit in a TSC. The GTCC waste basket liner is 173 inches long with a 1-inch-thick bottom plate welded onto it. The GTCC stainless steel liner shell is2 inches thick for structural and gamma shield functions, and has lifting lugs welded on the inside diameter of the shell. The liner design also includes an outer ring and a middle support under the bottom plate and drain holes in the bottom plate to facilitate free flow drainage from the liner. The GTCC TSC includes a sump location in the bottom plate and the closure lid includes a drain tube assembly to enable draining and drying of the loaded TSC.

The MAGANTRAN containment system consists of the bottom inner forging, inner shell, top forging, package lid and lid bolts, metal inner lid O-ring, coverplate and bolts, and metal inner coverplate O-ring. The two concentric O-rings on the lid and coverplate forms the annulus to facilitate leakage testing after closure.

The package has approximate dimensions and weight as follows:

Cavity diameter	72 inches
Cavity length	193 inches
Package body outer diameter	87 inches
Impact limiter diameter	128 inches
Package length	
without impact limiters	214 inches
with impact limiters	322 inches

The maximum gross weight of the package is about 312,000 lbs.

1.2 Contents

The MAGNATRAN package is designed to transport up to 37 undamaged PWR fuel assemblies in a 37 PWR basket assembly, up to 87 undamaged BWR fuel assemblies in an 89 BWR basket assembly, up to 37 undamaged PWR fuel assemblies or a combination of undamaged fuel assemblies and up to four damaged or high burnup fuel assemblies each in a damaged fuel can (or fuel material equivalent to a single fuel assembly) in the 37 PWR damaged fuel basket assembly, or a TSC containing up to 55,000 pounds of GTCC waste in a GTCC waste liner.

The PWR fuel assembly's characteristics are shown in the safety analysis report (SAR) Tables 1.3-6 and 1.3-7. The fuel assemblies shall be Zirconium-based alloy cladding with a maximum assembly average burnup of ≤ 60,000 MWd/MTU. High burnup fuel assemblies (fuel assemblies with maximum assembly average burnup > 45,000 MWd/MTU) shall be placed in a damaged fuel can and treated as damaged fuel. Undamaged PWR fuel assemblies may contain non-fuel hardware. Fuel assembly lattices not containing the nominal number of fuel rods must contain solid filler rods that displace a volume equal to, or greater than, that of the fuel rod that the filler rod replaces. Fuel assemblies may have stainless steel rods inserted to displace guide tube "dashpot" water. Fuel assemblies may contain any number of unirradiated (i.e., not inserted in-core) non-fuel solid filler fuel replacement rods. Activated stainless steel rods are limited to five per assembly, one assembly per basket. Fuel assemblies may contain a hafnium absorber assembly at a maximum burnup/exposure of 4.0 GWd/MTU.

The TSC will store up to 87 undamaged BWR fuel assemblies within the BWR fuel basket in accordance with the limiting values in SAR Tables 1.3-19 and 1.3-20. The fuel assemblies shall be Zirconium-based alloy cladding with average assembly burnup in accordance with SAR Tables 1.3-19 through 1.3-21. The BWR fuel assemblies may be unchanneled or channeled with zirconium-based alloy channels. BWR fuel assemblies with stainless steel channels are not authorized.

GTCC waste is defined in 10 CFR 61.55(a)(3) and (4) and consists of radiation activated and surface contaminated steel, and/or plasma cutting debris (dross). Stainless steel core baffle structure – baffle plates and angles, baffle formers, and lower core plates, located adjacent to the reactor vessel in a high neutron flux field, is the major component of GTCC waste. The specific Curie content source of the GTCC waste is limited to:

- a maximum of 2.7 Curies ^{60}Co/lb averaged over the GTCC contents,
- a localized peak of 16.1 Curies ^{60}Co/lb, and
- a total ^{60}Co activity of 85,760 Ci at the time of transport.

The maximum decay heat of the contents shall not exceed:

- 23 kW for PWR fuel loaded in a basket with neutron absorbers having Type 2 thermal conductivity,
- 22 kW for PWR fuel loaded in a basket with neutron absorbers having Type 1 thermal conductivity,
- 22 kW for BWR fuel, or
- 1.7 kW for GTCC waste content.

1.3 Criticality Safety Index

The criticality safety index (CSI) for undamaged PWR and BWR fuel contents described in 5(b)(1)(1)(i), 5(b)(1)(iii) and limited in 5(b)(1)(2)(i), and 5(b)(2)(iii) of the CoC is 0.0.

The CSI for damaged PWR fuel described in the CoC in 5(b)(1)(1)(ii) and limited in 5(b)(1)(2)(ii) is 100.

1.4 Drawings

The package is constructed and assembled in accordance with NAC drawings:

71160-500, Rev. 5P	Shipping Configuration, Transport Cask, MAGNATRAN
71160-501, Rev. 0	Assembly, Transport Cask, MAGNATRAN
71160-502, Rev. 6P	Transport Cask Body, MAGNATRAN

71160-504, Rev. 2	Misc. Details, Transport Cask, MAGNATRAN
71160-505, Rev. 6P	Lid Assembly, Transport Cask, MAGNATRAN
71160-506, Rev. 1	Cask Cavity Spacer, MAGNATRAN
71160-511, Rev. 1	Personnel Barrier, Shipping Configuration, Transport Cask, MAGNATRAN
71160-512, Rev. 1	Nameplate, MAGNATRAN
71160-530, Rev. 1	Misc. Details, Impact Limiter, MAGNATRAN
71160-531, Rev. 2P	Impact Limiter, Transport Cask, MAGNATRAN
71160-551, Rev. 10P	Fuel Tube Assembly, MAGNASTOR - 37 PWR
71160-559, Rev. 0	Lifting Trunnion, Transport Cask, MAGNATRAN
71160-571, Rev. 10P	Details, Neutron Absorber, Retainer, MAGNASTOR – 37 PWR
71160-572, Rev. 9P	Details, Neutron Absorber, Retainer, MAGNASTOR–87 BWR
71160-574, Rev. 6	Basket Support Weldments, MAGNASTOR - 37 PWR
71160-575, Rev. 11P	Basket Assembly, MAGNASTOR - 37 PWR
71160-581, Rev. 5	Shell Weldment, TSC, MAGNASTOR
71160-584, Rev. 8	Details, TSC, MAGNASTOR
71160-585, Rev. 12	TSC Assembly, MAGNASTOR
71160-591, Rev. 8P	Fuel Tube Assembly, MAGNASTOR - 87 BWR
71160-598, Rev. 7P	Basket Support Weldments, MAGNASTOR - 87 BWR
71160-599, Rev. 8P	Basket Assembly, MAGNASTOR - 87 BWR
71160-600, Rev. 5P	Basket Assembly, MAGNASTOR - 82 BWR
71160-601, Rev. 0	Damaged Fuel Can (DFC), Assembly, MAGNASTOR
71160-602, Rev. 1	Damaged Fuel Can (DFC), Details, MAGNASTOR
71160-620, Rev. 1P	Top Fuel Spacer, MAGNASTOR
71160-671, Rev. 2P	Details, Neutron Absorber, Retainer, For DF [Damaged Fuel] Corner Weldment, MAGNASTOR - 37 PWR
71160-673, Rev. 1	Damaged Fuel Can (DFC), Spacer, MAGNASTOR
71160-674, Rev. 4P	DF Corner Weldment, MAGNASTOR
71160-675, Rev. 3P	DF Basket Assembly, 37 Assembly PWR, MAGNASTOR
71160-681, Rev. 1	DF, Shell Weldment, TSC, MAGNASTOR
71160-684, Rev. 2	Details, DF Closure Lid, MAGNASTOR
71160-685, Rev. 6	DF, TSC Assembly, MAGNASTOR
71160-711, Rev. 1	GTCC Waste Basket Liner, MAGNASTOR
71160-781, Rev. 1	Shell Weldment, GTCC TSC, MAGNASTOR
71160-785, Rev. 3	GTCC TSC, Assembly, MAGNASTOR

1.5 Evaluation Findings

A general description of the Model No. MAGNATRAN package is presented in Section 1 of the application, with special attention to design and operating characteristics and principal safety considerations. Drawings for packaging components are included in the application. The package application identifies the NAC Quality Assurance Program and the applicable codes and standards for the design, fabrication, assembly, testing, operation and maintenance of the package.

2.0 STRUCTURAL REVIEW

The objective of this review is to verify that the structural performance of the package has been adequately evaluated for the tests and conditions specified for normal conditions of transport and hypothetical accident conditions and that the package design has adequate structural integrity to meet the requirements of 10 CFR Part 71.

2.1 Description of Structural Design

2.1.1 Descriptive Information

The MAGNATRAN packaging consists of five principal structural entities: (1) the package body, (2) the TSC, (3) the fuel basket, (4) the GTCC waste canister and waste basket liner and (5) the impact limiters. Considering specifically the content configuration, the packaging is designed to transport two categories of PWR fuel assemblies and two categories of BWR fuel assemblies within two lengths of TSCs (long and short). A cavity spacer is used to axially position the short TSC and limit its potential movement under normal conditions of transport and hypothetical accident conditions. The short TSC is also used to transport GTCC waste in a basket liner. The structural designs of these entities have been reviewed as discussed below.

2.1.1.1 Package Body

The principal structural features of the package body include a multi-walled cylinder including inner and outer stainless steel shells, separated by poured-in-place lead gamma shielding, a bottom forging and plate assembly, a package lid, two diametrically opposite lifting trunnions bolted to the outside of the top forging, and two removable rotation trunnion pins located at the outer shell and bottom outer forging near the bottom of the package.

A package cavity spacer approximately 7 inches in length is attached to the lid in the upper end of the package cavity to limit the axial movement of the short TSCs. As depicted in Figure 1.2-1 of the application, neutron shield material is held in stainless steel enclosures, which are, in turn, attached to the outer shell along the length and around the circumference of the package about the active fuel region.

NAC Drawing No. 71160-501, Rev. 0 shows the general arrangement of the package with a package cavity measuring 192.5 inches long by 72.25 inches in diameter. Table 1.3-1 of the application summarizes design characteristics of the package body and its components. The package is designed to transport a TSC with up to 37 undamaged PWR fuel assemblies or up to 87 undamaged BWR fuel assemblies. The package is also designed to transport a TSC containing up to four damaged PWR fuel assemblies in damaged fuel cans in the damaged fuel basket assembly or up to 55,000 pounds of GTCC waste using a GTCC waste basket liner.

Section 1.3.1 of the application provides a detailed description of the package. As noted in the Section 1.4.3 licensing drawings, the package containment system components include a bottom inner forging; an inner shell; a top forging; a package lid with forty-eight, 2-8 UN American Society of Mechanical Engineers (ASME) SB-637, Grade N07718 nickel alloy closure bolts; port cover plates; and inner O-rings on the lid and port cover plates. The bottom and top forgings, inner shell, and package lid are all constructed of Type 304 stainless steel.

2.1.1.2 Transportable Storage Canister

The TSC consists of an assembly of stainless steel cylindrical shell, bottom, closure lid, closure ring, and port covers. The 72-inch diameter TSC, which measures either 191.8 or 184.8 inches long, maintains a leaktight containment of the contents during transport in all the evaluated conditions. The 9-inch-thick closure lid, which is either an integral stainless steel or a composite stainless-plus-carbon steel construction, is designed with removable hoist rings so that the loaded TSC can be lifted. Section 1.3.1 of the application provides a detailed description of the TSC design parameters for the transport of different classes of PWR and BWR fuel and for GTCC waste. It also provides the geometry and materials of fabrication defined in the license drawings in Section 1.4.3.

2.1.1.3 Fuel Basket

The carbon steel fuel basket assembly inside the TSC has two different lengths. It is comprised of an array of square fuel tubes joined in the interior by the pin-slot connections. At the peripheral points, they are attached by bolting to an assembly of side and corner weldments that form a circular cross section for emplacement in the canister. The fuel tubes function as individual cells, as well as sidewalls for the developed cells for fuel assemblies. In conjunction with the side and corner weldments, which also partially serve as sidewalls, 21 square tubes provide 37 PWR fuel loading positions. Similarly, an array of 45 square tubes provides 87 loading positions for BWR fuel assemblies. Section 1.3.1 of the application provides a detailed description of the basket assemblies, including the PWR damaged fuel basket assembly for transporting up to four PWR fuel assemblies, each in a damaged fuel can. Section 1.3.1 also provides the geometry and materials of fabrication which are defined on the licensing drawings in Section 1.4.3.

2.1.1.4 Greater-Than-Class C Waste Transportable Storage Canister and Waste Basket Liner

The GTCC TSC, which measures 184.8-inch in length, is the short version of the two TSC designs. The stainless steel GTCC waste basket liner is sized to dimensionally fit in a GTCC TSC. The liner's structural design includes a shell and welded bottom plate and the lifting lugs welded on the inside diameter of the shell so that the liner may be loaded with GTCC waste prior to being inserted into a TSC. Section 1.3.1 of the application provides a detailed description of the liner. Section 1.3.1 also provides the geometry and materials of fabrication defined in the license drawings in Section 1.4.3.

2.1.1.5 Impact Limiter

The packaging is equipped with two identical impact limiters each bolted to the package lid or the package bottom plate with 16 equally spaced retaining rod assemblies. The impact limiter, which measures {proprietary information removed} diameter, is fabricated with balsa wood and redwood wedge-shaped sections glued together to form a cylindrical shape. The wood is completely enclosed in a stainless steel shell. Section 1.3.1 of the application provides a detailed description of the impact limiters. Section 1.3.1 also provides the geometry and materials of fabrication defined in the license drawings in Section 1.4.3.

2.1.2 Design Criteria

Section 2.1.2 of the application summarizes the structural design criteria for the packaging, including applicable codes and standards, as well as load combinations. These design criteria are reviewed as follows.

2.1.2.1 Load Combinations

Section 2.1.2.1 of the application notes the loading conditions for evaluating the packaging structural performance for the normal conditions of transport and hypothetical accident conditions in 10 CFR Part 71. As delineated in Table 2.1.2-1 of the application, individual loading conditions are considered together with a set of initial conditions such as ambient temperature, solar insolation, decay heat, internal pressures, and fabrication stresses. The loading conditions with applicable initial conditions establish load combinations for which total stress effects on the structural performance of the package body components are evaluated. The staff reviewed the load combinations and finds them acceptable because they are consistent with those of the Regulatory Guide 7.8[1] guidelines.

2.1.2.2 Stress Limits and Failure Mode Criteria

Tables 2.1.2-2 and 2.1.2-3 of the application list stress allowables for containment and non-containment structures of the package, respectively. As noted in Section 2.1.2.2, the stress allowables for the package containment boundary and the TSC shell are designed per the ASME Boiler and Pressure Vessel Code[2] (the ASME B&PV Code), Section III, Division 1, Subsection NB, and Appendix F. The staff finds these stress allowables acceptable because they are essentially identical to those of the ASME B&PV Code, Section III, Division 3, Subsection WB guidance provided in NUREG-1617[3].

Section 2.1.2.3 of the application presents structural performance criteria for miscellaneous structural failure modes, including brittle fracture, fatigue, buckling, and impact limiter deformation limits. In Section 2.2 of the SER below the staff summarizes its review for the brittle fracture performance of the packaging structural components. The packaging structural components are fabricated primarily with the Type 304 stainless steel except for the package lid, lifting and rotation trunnions. The fuel basket tubes and associated support weldments are fabricated with carbon steel. In Section 2.1.2.3.2 of the application, NAC evaluates the package against the ASME B&PV Code, Section III, NB-3222.4 cyclic operation provisions and determined that the packaging structural components meet the ASME B&PV Code exemption criteria in NB-3222.4(b) and do not require a detailed fatigue analysis. Nevertheless, considering the normal operating cycle of spent fuel loading, transporting, and unloading, the applicant assumed 1,200 cycles of operation for evaluating the package closure lid bolts.

In Section 2.1.2.3.3 of the application, NAC notes that the package body inner shell is evaluated for structural stability in accordance with the interaction equations of the ASME B&PV Code Case N-284-1. The applicant evaluated buckling of basket tube walls using the criteria defined in NUREG/CR-6322[4].

There is currently no industry standard for the impact limiter structural performance; however, the applicant designed the impact limiter to absorb the impact energy by the crushing of the energy-absorbing wood materials during the package free drops. This protects the transportation package from a direct landing onto an essentially unyielding surface.

The stress allowables and structural failure modes, as reviewed above, meet the intent of NUREG-1617, and are, therefore, acceptable.

2.1.3 Identification of Codes and Standards for Packaging

Section 2.1.4 of the application summarizes the codes and standards used for the design, fabrication, assembly, and acceptance testing of the package containment and non-containment components. Table 2.1.4-1 lists the specific alternatives to the ASME B&PV Code requirements. The bases for justifying the alternatives are also provided. The staff reviewed the code and standard considerations and finds them acceptable because they are consistent with those identified in NUREG-1617.

2.1.4 Weights and Centers of Gravity

Table 2.1.3 of the application lists the calculated weights and centers of gravity of the major components of the package containing either PWR fuel, BWR fuel, or GTCC waste. The table also summarizes the weights and center of gravity locations of the package for several configurations, including the loaded TSC and the transport-ready package with the top and bottom impact limiters. The center-of-gravity locations, which are measured from the bottom outer surface of each configuration, are identified along the package vertical axis. For the transport ready configurations, the largest package weight is 312,000 lbs. and the corresponding center of gravity is located at {proprietary information removed} the exterior surface of the bottom impact limiter.

2.2 Material Properties

The Model MAGNATRAN is a Type B(U)F-96 package for the transport of spent fuel and GTCC waste. An overview of the transportation package is discussed in Section 1.2 of the application. The package description and materials of fabrication are provided in Section 1.3. Note that the MAGNASTOR TSC, which is transported in the MAGNATRAN package, has been previously approved by the staff for storage pursuant to the requirements in 10 CFR Part 72, and the SER will briefly discuss the major components and fuel cladding associated heat limits of the TSC. The MAGNATRAN materials review was comprehensive; however, the discussion primarily focuses on the non-proprietary structural components that are considered to be a part of the package.

2.2.1 MAGNATRAN Transportable Storage Canister Description:

Structural components of the MAGNASTOR TSC (Structural components of the MAGNATRAN TSC (shell, bottom plate, closure lid, and port covers) are fabricated from ASME SA-240, dual certified, Type 304/304L austenitic stainless steel. The applicant may use ASME SA-182 or SA-336, Type 304 stainless steel as a substitute material for the closure lid, provided the material has a yield strength and ultimate strength greater than, or equal to, those of the ASME SA-240 material. In addition, a shield plate, American Society of Testing and Materials (ASTM) A36 carbon steel is included as part of a carbon steel/stainless steel closure lid assembly. The MAGNATRAN TSC holds the PWR/BWR fuel basket or GTCC waste basket liner assembly and contains the associated used-fuel/non-fuel/waste contents. Two different lengths of the MAGNATRAN TSC accommodate the various PWR/BWR fuel assembly lengths, damaged fuel, and GTCC waste. Vent and drain ports are through-lid penetrations, which provide access for auxiliary systems to drain, dry and helium backfill the TSC. The MAGNATRAN TSC has four configurations, TSC1 through TSC4. TSC1 and TSC2 include a solid stainless steel closure lid assembly. TSC3 and TSC4 include a composite lid assembly consisting of a stainless steel closure lid and a carbon steel shield plate that is coated using an ASTM B733-97 electroless nickel-phosphorus plating. The shield plate is attached to the closure lid by 10 ASTM A193, Grade B6 bolts.

2.2.1.1 PWR/BWR Baskets

The MAGNATRAN TSC basket is a mechanical assembly fabricated from ASME SA-537, Class 1, carbon steel square fuel tubes and support weldments, designed to accommodate PWR/BWR fuel assemblies and PWR damaged fuel can. The assembled basket is coated with ASTM B733-97, electroless nickel-phosphorus plating, an immersion process utilized to reduce corrosion and prevent generation of combustible gases during fuel loading. Following plating of the carbon steel, the neutron absorber panels (borated metallic composite) and their stainless steel retainers are installed on the basket structure. ASME SA-240/312, Type 304 stainless steel fuel spacers may be used in the TSC to reduce axial gaps for the PWR spent fuel assemblies, non-fuel hardware or damaged fuel can. The fuel basket designs minimize horizontal surfaces that could allow water to accumulate. In addition, open paths for water flow to the drain tube and sump in the bottom of the TSC are provided. The fuel baskets are supported from the TSC bottom plate by standoffs at the corner of the fuel tubes enabling the TSC to fill and drain evenly.

The fuel tubes support an enclosed neutron absorber borated metallic composite sheet on up to four interior sides of the fuel tube for criticality control in the basket. Each neutron absorber sheet is enclosed by a thin ASME SA-240, Type 304 stainless steel sheet (the retainer) to protect the neutron absorber during fuel loading and to keep it in position. The neutron absorber and stainless steel retainer are secured to the fuel tube using ASME SA-479 stainless steel weld posts distributed across the width and along the length of the fuel tube. The neutron absorber sheets may be replaced by commercial aluminum sheets on the two outside surfaces of the eight outermost fuel tubes of the PWR fuel basket and on the three outer surfaces of the outermost fuel tubes of the BWR fuel basket.

2.2.1.2 Damaged Fuel Can

The MAGNATRAN PWR damaged fuel can is fabricated from ASME SA-240, Type 304 stainless steel. The DFC is designed in two lengths, the shorter fuel assemblies may be accommodated with a fuel assembly spacer to limit axial movement. For the shorter damaged fuel can, a damaged fuel can spacer is used in the damaged fuel basket assembly or alternatively fixed to the damaged fuel can bottom plate. The side plates that form the upper end of the damaged fuel can are ASME SA-240/479, Type 304 stainless steel and the tube body walls are 18-gage sheet, ASME SA-240, Type 304 stainless steel. The damaged fuel can lid plate and bottom are ASME SA-240, Type 304 stainless steel. The damaged fuel can lid and bottom include screened drain holes fabricated from commercial stainless steel. The damaged fuel can is sized to accommodate a damaged PWR fuel assembly or fuel debris equivalent to, or less than, one PWR fuel assembly. The damaged fuel can may also contain PWR fuel assemblies in an undamaged condition. Up to four damaged fuel cans may be loaded, one into each outer corner, in the damaged fuel Basket Assembly. The primary function of the damaged fuel can is to confine the fuel material within the can to minimize the potential for dispersal of the fuel material into the TSC cavity. ASME SA-537, Class 1, carbon steel corner support weldments in the damaged fuel basket assembly that are modified with additional structural support to provide an enlarged fuel tube opening for a damaged fuel can at the four outermost corners of the fuel basket. A damaged fuel can or an undamaged fuel assembly may be loaded in a damaged fuel can location.

2.2.1.3 Greater-than-Class C Waste Basket Liner and Waste

The GTCC waste basket liner structural components are fabricated from ASTM A240, dual certified, Type 304/304L stainless steel and designed to hold GTCC waste and dimensionally fit inside the TSC. The waste basket liner design includes a shell for structural and gamma shield functions, a welded bottom plate, and lifting lugs welded on the inside diameter of the shell so that the liner may be loaded with GTCC waste prior to being inserted into a TSC. The liner design also includes an Inner/middle/outer ring(s) under the bottom plate and drain holes in the bottom plate to facilitate free flow drainage from the liner. The GTCC TSC includes a sump location in the bottom plate and the closure lid, ASTM A240/A182, includes a drain tube assembly to enable draining and drying of the loaded TSC.

The package is designed to lift, confine and provide additional gamma shielding to the GTCC waste placed within the basket liner. The GTCC waste basket liner can be separately loaded with GTCC waste and then placed into a GTCC waste TSC prior to final TSC closure operations. GTCC waste consists of radiation activated and surface contaminated steel, and/or plasma cutting debris (dross), including stainless steel core baffle structure baffle plates and angles, baffle formers, and lower core plates, located adjacent to the reactor vessel in a high neutron flux field, which is the major component of GTCC waste.

2.2.1.4 Fuel Spacers

Fuel cavity spacers may be used in the TSC to reduce axial gaps for the spent fuel assemblies, non-fuel hardware or damaged fuel can. BWR fuel assemblies do not contain non-fuel hardware and therefore are not subject to length variations occurring in PWR assemblies due to the various non-fuel hardware loading (e.g., “bare” fuel, burnable poison rod assemblies (BPRAs) plus fuel, reactor control cluster assemblies (RCCAs) plus fuel).

2.2.1.5 Spent Fuel Cladding Integrity

The potential degradation mechanisms identified for the spent nuclear fuel assemblies include oxidation, corrosion, cladding creep, cladding annealing, and hydride redistribution and reorientation within the cladding. Oxidation of the zirconium alloy cladding and the irradiated fuel pellets can occur if the fuel is exposed to air, causing the pellets to swell and potentially impact the fuel cladding. Excessive oxidation of the fuel cladding, combined with internal stress, could cause a fuel cladding breach.

The integrity of the TSC containment boundary maintains the inert environment and prevents oxidation of the fuel. During loading operations, the water level in the TSC will be lowered (blown down) by about 70 gallons to facilitate lid welding. The lowering of the water level will not expose the spent fuel rods to an air atmosphere, thereby assuring that there is no inadvertent oxidation of the fuel rods during this operation. The MAGNASTOR technical specifications (Docket No. 72-1031) also require that the licensee ensure that fuel cladding oxidation does not occur during this stage. Corrosion of the fuel assembly components can only occur if they are exposed to moisture; however, residual water within the TSC is limited to very low levels through the vacuum drying process, which requires that the TSCs are dried as described in the MAGNASTOR technical specifications. Therefore, no significant amount of moisture is within the TSC cavity during extended storage, and water ingress is considered not credible due to the performance of the MAGNASTOR TSC confinement boundary. Therefore, there are no credible corrosion mechanisms of the fuel assembly subcomponents. Hydride redistribution and reorientation can occur in cladding during high temperature, high hoop stress scenarios such as vacuum drying, which could adversely affect the structural properties of the cladding. Per Interim Staff Guidance – 11[5], (ISG-11) Rev. 3, significant hydride re-orientation is not expected to occur in low burnup fuel assemblies, i.e., burnups ≤45,000 MWd/MTU. High burnup fuel assemblies, i.e., burnups >45,000 MWd/MTU, will be placed in damaged fuel cans.

2.2.1.6 Neutron Absorber Materials

Criticality control is achieved through a combination of neutron absorber sheets on the interior faces of the fuel tubes and the use of burnup credit for PWR fuel assemblies. The neutron absorber plates provide criticality control and a heat conduction path from the fuel assemblies to the canister shell. Neutron poison plates are composed of: 1) a borated aluminum alloy, 2) a boron carbide aluminum metal matrix composite, or 3) Boral.

Following manufacture, each sheet of neutron absorber material will be visually and dimensionally inspected for damage, embedded foreign material, and dimensional compliance. The neutron absorber sheets are intended to be defect/damage free.

2.2.2 MAGNASTOR Transportable Storage Canister Evaluation

The following is a summary evaluation of the MAGNASTOR TSC that was previously reviewed and approved by the NRC staff for inclusion of the MAGNASTOR storage system in 10 CFR 72.214. The staff reviewed the materials selection (i.e., steel to be used and absorbers to be used in the cask); brittle fracture; applicable codes and standards; weld design and specifications; corrosion (i.e., environmental; chemical and galvanic; and uniform and localized corrosion); and cladding integrity. Additionally, staff verified that materials selections are appropriate for the environmental conditions to be encountered during loading, unloading, transfer and storage operations (i.e., vacuum drying of the canister). The staff reviewed the discussion on material temperature limits with respect to the following regulatory requirements: 10 CFR 72.122(h)(1) requires the spent fuel cladding to be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage. The staff verified the MAGNASTOR TSC materials of construction (e.g., stainless steel and carbon steel) are readily weldable using commonly available welding techniques. The TSC shell assembly is designed, fabricated, examined and tested in accordance with the requirements of Subsection NB of the ASME B&PV Code. The circumferential and longitudinal shell plate welds are multi-layer full penetration welds. The TSC welds were well-characterized on the license drawings, and standard welding symbols and notations in accordance with American Welding Society (AWS) Standard A2.4, “Standard Symbols for Welding, Brazing, and Nondestructive Examination” were used.

The GTCC waste basket liner and TSC are designed, fabricated and inspected in accordance with the ASME B&PV Code, Section III, Division 1, Subsection NF. The lifting features of the GTCC components are designed for non-critical lifting in accordance with NUREG-0612[6] and American National Standards Institute (ANSI) N14.6[7].

The staff concluded that the selection of neutron absorbers will ensure these materials will be sufficiently durable during the service life of the storage cask. These materials have been tested to perform as intended within design limits and have been evaluated over decades of use in reactor plants, and storage and transportation canister fabrication. The staff reviewed the design requirements, testing for durability (e.g., corrosion and thermal damage), and testing to demonstrate the ^{10}B uniformity. The staff found the qualification tests acceptable for this application. The staff verified that the cladding temperatures for each fuel type proposed for storage are below the temperature limits that would preclude cladding damage that could lead to gross rupture. In addition, the staff finds that the proposed method of drying acceptable based on elimination of water to prevent corrosion of internal components, and the prevention of degradation of the cladding due to cycling. The materials of the MAGNASTOR TSC have been previously approved by staff in its review of the MAGNASTOR spent fuel storage system (SER

ADAMS Accession No. ML090350589). The staff notes that materials used in design and fabrication standards, of the MAGNASTOR, are similar to those materials/standards used in design/fabrication of MAGNATRAN and various other transportation packages. The staff has reasonable assurance that no significant degradation of the MAGNASTOR canister or its contents would be expected during transportation based on the discussion above and the MAGNASTOR TSC will meet the requirements in 10 CFR 71.43(d).

2.2.3 MAGNATRAN Transport Packaging

MAGNATRAN package is a cylindrical, multi-walled structure for the transport of a TSC. The major structural components of the package body are the inner and outer shells, upper and lower forgings, lid, and the bottom plate. Poured-in place chemical-copper grade lead fills the annulus between the inner and outer shells and serves as the primary gamma radiation shield. The package lid is recessed and bolted in the top forging. The lid port is recessed into the package lid and is protected by a cover-plate. The neutron shield consists of NS-4-FR material that is positioned in stainless steel enclosures attached to the outer shell along the length and around the circumference of the package cavity about the active fuel region.

2.2.3.1 Package Body

The inner shell, top forging, bottom inner forging, bottom outer forging, bottom plate and the neutron shield enclosure shells of the MAGNATRAN package are ASME SA-240 or SA-336, Type 304, stainless steel. The outer shell is Type XM-19 (Nitronic 50/UNS S20910) austenitic stainless steel. The lower end of the package body is comprised of the bottom inner forging, the bottom outer forging and the bottom plate. The bottom inner forging is cup-shaped to form the bottom of the package cavity and is welded to the inner shell and to the bottom outer forging. The ring-shaped bottom outer forging is also welded to the outer shell and to the bottom plate.

The inner shell forms the side of the package cavity. The inner shell is welded to the top forging and to the bottom inner forging. Sixteen holes are tapped into the bottom plate for attachment of the lower impact limiter. The outer shell is welded to the top forging and to the bottom outer forging. Primary radial gamma radiation shielding is provided by the ASTM B29 chemical-copper grade lead that is poured into the annulus between the inner and outer shells. The upper ends of the inner shell and the outer shell are welded to the ring-shaped top forging. The inner diameter of the top forging includes a recess to accommodate the package lid.

NS-4-FR is a solid synthetic polymer and provides radial neutron shielding for the package. The NS-4-FR is enclosed in stainless steel assemblies that are attached to the outside diameter of the outer shell along its length. The MAGNATRAN package incorporates ASTM B152 copper (cooling fin-A), and ASTM B209 copper/1100 Aluminum alloy (cooling fin-B), evenly distributed around the package, which are retained by the bolted neutron shield assemblies against the outer shell to enhance heat transfer from the package body.

All of the ASME SA-240/SA-479, Type 304, stainless steel port covers are recessed into the package lid and none protrude above the package surface. The ASME SA-193, Type 410, Grade B6 stainless steel package lid bolts are recessed in the package lid and do not project above the lid surface. The heat dissipation features of the package are entirely passive. No active or support cooling mechanisms are required during transport. No coolants are used within the MAGNATRAN package. An inert helium gas atmosphere is used to backfill the MAGNATRAN package's cavity prior to transport.

2.2.3.2 Package Lid/Bolts

The lid is fabricated from ASME SA-693/SA-705, Type 630, 17-4 PH, Condition H1025, stainless steel. Forty-eight lid bolts, 2-8 UN-2A socket head cap screws fabricated from SB 637, Grade N07718 nickel alloy steel, secure the lid to the top forging. Sixteen holes, equally spaced are tapped into the top of the lid for attachment of the upper impact limiter. The lid, when bolted to the top forging, provides a sealed closure of the MAGNATRAN package containment vessel. The outer periphery of the lid is stepped down so that the bolt heads are below the top surface of the lid. The bottom surface of the lid is sealed to the top forging by two concentric O-rings, an inner, Garlock Helicoflex, HN 200, metal O-ring and an outer EPDM O-ring. An inter-seal test port is provided between the O-rings to facilitate leak testing of the O-rings. After the leak test is completed, the inter-seal test port is closed by a stainless steel threaded plug fitted with a boss seal. The top surface of the lid has threaded holes for the attachment of a lid-lifting device. The lid is lifted using four of the sixteen holes provided for attachment of the upper impact limiter. The lid provides attenuation, on top of the package, for radiation from the fuel assemblies, fuel assembly hardware, or GTCC waste contained within the MAGNATRAN package cavity.

2.2.3.3 Port Cover Plate

The MAGNATRAN package has a lid port that is closed by a bolted ASME SA-240/SA-479, Type 304/304L, stainless steel cover-plate with dual O-rings. The four cover-plate socket head cap screws are ASME SA-193, Type 410, Grade B6 stainless steel. The cap screws are countersunk flush with the top of the cover-plate. The basic configuration of the lid port and cover-plate includes an opening to recess the cover-plate and for access to the port opening and the quick-disconnect installed there. Two concentric O-rings are located on the bottom face of the cover-plate. An inner Garlock Helicoflex, HN 200, metal O-ring and an outer EPDM O-ring. The inner O-ring provides the containment boundary seal for the lid port. The outer O-ring and a test port located between the two O-rings provide the means to leak test the containment boundary seal. After the leak test is completed, the seal test port is closed by a stainless steel threaded plug fitted with a metal boss seal.

2.2.3.4 Lifting Trunnions/Rotation Trunnions

The MAGNATRAN package utilizes two ASME SA-564/SA-693, Type 630, 17-4 PH stainless steel lifting trunnions that are bolted, using nine ASME SB-637, Grade N07718, nickel alloy steel bolts, to the top forging at diametrically opposite locations around the package circumference. A retainer, or flange, on the outer end of each lifting trunnion acts as a safety stop to ensure that proper engagement with the lift yoke is maintained. This permits critical package lifting and handling using a single pair of lifting trunnions, load tested, in accordance with ANSI N14.6 and that meets the requirements in 10 CFR 71.45(a). The MAGNATRAN package trunnions are replaced for transport operations with ASTM A240, Type 304, stainless steel trunnion plugs using Type 304, stainless steel bolts. Two ASME SA-240/SA-276, Type XM-19, stainless steel rotation trunnions, located approximately 180° apart, are welded to the exterior of the outer shell and bottom outer forging near the bottom of the neutron shield and are off-set from package centerline to ensure proper rotation. The neutron shield assemblies are shaped to accommodate the location and operation of the rotation trunnions. The rotation trunnion pins are 17-4 PH stainless steel. The rotation trunnion pins mate with the rear package support to permit rotation of the package between the horizontal and vertical orientations. The rotation trunnions also serve as the package tie-down restraint in the aft longitudinal direction.

2.2.3.5 Cavity Spacer

An ASME SA-240, Type 304, stainless steel cavity spacer is used in the upper end of the MAGNATRAN package cavity to limit the axial movement of the short TSC. The cavity spacer consists of six concentric, ASME SA-240, stainless steel rings welded to a flat, ASME SA-240, stainless steel plate. The depth of the rings/length of the spacer, is approximately 7 inches representing the difference in length between the short and long TSCs. The spacer is bolted utilizing 1-8 UNC-2B stainless steel Hex Nuts through the flat plate to the underside of the package lid.

2.2.3.6 Impact Limiters

The MAGNATRAN package is protected by two energy-absorbing impact limiters that fit over each end of the package and dissipate kinetic energy by crushing redwood and balsa wood. The MAGNATRAN package is equipped with removable impact limiters that are bolted over each end of the package to ensure that the design impact loads for the package are not exceeded for normal conditions of transport or hypothetical accident conditions. The upper and lower impact limiters are dimensionally identical. The impact limiters consist of redwood and balsa wood wedge-shaped sections glued together to form a cylindrical shape that is completely enclosed in an ASTM A240, Type 304, stainless steel shell. The force required to crush the impact limiter is determined by the amount, location and grain direction of the redwood and balsa wood. The lower impact limiter is bolted to the ASTM A240, Type 304, SS, MAGNATRAN package bottom plate by 16 equally spaced ASME SA-193, Grade B8S, stainless steel retaining rods and nuts. The upper impact limiter is similarly bolted to the MAGNATRAN package lid. ASTM A269, Type 304, stainless steel screw tubes serve as the penetrations for the impact limiter retaining rods.

2.2.4 MAGNATRAN Package Materials Evaluation

2.2.4.1 Material Properties

Section 2.2.1 of the SAR provides material specifications, mechanical and physical property data for the MAGNATRAN package structural materials and other major components. The following material properties evaluation centers on all major structural and containment metallic components as described above. In Section 2.2.4.3, "Other Materials," and throughout the remaining SER other packaging materials (for example: criticality, shielding, seals, foam, impact limiter wood, etc.) will be evaluated. The staff notes that the coefficient of thermal expansion for the bolt material is very similar to that of the lid and the ASME SA-336, Type 304, stainless steel top forging materials, minimizing differential thermal expansion effects. The staff notes that most of the values in the SAR tables were obtained from the ASME B&PV Code, Section II, Part D; however, some of the values were obtained from other acceptable technical references. The staff notes that structural components of the MAGNATRAN package are selected based on their strength, ductility, resistance to corrosion, and brittle fracture characteristics. The staff independently verified the temperature-dependent values for the allowable stress, modulus of elasticity, Poisson's ratio, weight density, thermal conductivity and coefficient of thermal expansion. The staff concludes that the material mechanical properties are acceptable and appropriate for the expected load, temperature and environmental conditions during the transportation period for the MAGNATRAN package. In addition, materials used in fabrication of the MAGNATRAN packaging components are identical to those previously reviewed and approved. The staff reviewed the application to confirm that the applicant has identified materials and package components used for heat transfer in accordance with 10 CFR 71.33(a)(5) and (6).

2.2.4.2 Drawings

The staff reviewed the license assembly drawings and finds the drawings contain a bill of materials, including appropriate consensus code information, that include AWS, ASME, ASTM specification number(s) for the material(s) used in fabrication.

2.2.4.3 Other Materials

The staff notes that the MAGNATRAN packaging utilizes other materials as part of the transportation package evaluated as follows.

Foam:

The applicant states that the package utilizes layers of expansion foam and strips of insulation in the solid neutron shield regions of the MAGNATRAN package. The expansion foam permits thermal expansion of the solid neutron shield material during normal conditions of transport, and the insulation protects the expansion foam during final closure welding of the neutron shield shell to the end plate. The foam and the insulation are non-flammable, non-toxic and non-corrosive silicone. The applicant states that expansion foam within each neutron shield assembly also provides for pressure reduction in case of a vapor pressure increase resulting from the neutron shield material being exposed to a transport thermal accident condition.

The staff notes that Silicone Foam HT-800 is a highly versatile, medium firmness silicone that offers the lightness of a foam along with the enhanced sealing capabilities of a traditional sponge rubber protecting in part against wind-driven rain or fire. The material is also used to reduce shock or isolate vibration. Auburn silicone foams are available in various thicknesses and manufactured in roll form which can be easily converted into tapes, sheets or custom die cut gaskets. Environmental seals to protect against penetration of dust, moisture, air, or light into outdoor enclosures.

The staff reviewed the vendor's HT-800 technical specification for various material properties as follows: environmental properties such as ultraviolet light to Society of Automotive Engineers (SAE) J-1960, and corrosion resistance to Aerospace Material Specification (AMS)-3568; physical/mechanical properties such as density/compression set to ASTM D1056, and tensile strength/elongation to ASTM D412; thermal properties such as conductivity to ASTM C518, and temperature range (-67 to 392°F) to SAE J-2236. In addition, flame resistance was tested per ANSI approved test method UL 94, "Standard for Safety of Flammability of Plastic Materials for Parts in Devices and Appliances." The staff verified the foam material would adequately perform at the exposure temperatures the foam adjacent to the neutron shielding would experience. Therefore, the staff finds there are no credible potential reactions associated with the silicone expansion foam or insulation based on the material technical specification and discussion above.

Lubricants:

The applicant states that dry film lubricants used in the package meet the performance and general compositional requirements of the nuclear power industry. The staff notes the following. Neolube is used primarily on threaded/mechanical connection surfaces. Dow Corning High Vacuum Grease is used as an adherent/lubricant to lubricate and retain the O-ring seals in their grooves. NEVER-SEEZ is a high-temperature/pressure anti-seize lubricant containing particles of pure nickel, graphite with other additives and used on the trunnion surfaces of the package.

Neolube is graphite particles in isopropanol, noncorrosive, and resistant to cold/radiation. No coatings are applied to the stainless steel or nickel alloy steels.

The staff evaluated the hypothetical accident condition temperature limits stated in the SAR and the associated non-metallic material decomposition references and notes that there is sufficient temperature margin to ensure that gas generation will not affect the containment boundary. Therefore, the staff finds that there is reasonable assurance that the transportation package can be operated without undue risk to the health and safety of the public. In addition, the quantity of these non-metallic materials is considered insufficient to over-pressurize the package during a hypothetical accident conditions fire event. In addition, these lubricants have been previously evaluated, accepted and used successfully in radioactive material packages for transport of similar contents.

Seals:

The applicant states that O-ring seals are formed from stainless steel, silicone rubber, EPDM, and Viton compounds. Viton is a silicon elastomer. EPDM or elastomer O-rings are used for MAGNATRAN package applications because of their excellent short-term sealing capabilities and ease of handling. Stainless steel was previously discussed. The components of all of the seal and gasket materials are stable and nonreactive. No potential reactions are associated with the packaging seal materials.

The applicant identified the dimensions, tolerances, and materials for the inner lid containment boundary seals and port cover plate boundary seals in the corresponding drawings. The applicant specified seal performance requirements in the application, including (1) ensuring testing of the containment boundary is performed to ANSI N14.5-1997, "American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment," leaktight criteria, (2) adequate performance of the stainless steel metallic seals in the temperature range of -40°F to 500°F (which bounds the maximum temperature during the tests for normal conditions of transport and hypothetical accident conditions in 10 CFR 71.71 and 10 CFR 71.73, respectively), (3) pressure rating of 300 psig (which bounds the maximum internal pressure of the package and (4) maximum external pressure due to package immersion), and minimal compression forces for the inner lid and port coverplate bolts (consistent with bolt pre-load and torque specifications in the respective drawings). The staff reviewed the seal specifications in the design drawings and concludes that the seals are capable of adequately meeting these performance requirements.

Wood:

The applicant states that the impact limiters for the MAGNATRAN package are fabricated from balsa wood and redwood encased in a stainless steel shell. The impact limiters absorb the kinetic energy of the loaded package in a drop impact by crushing the redwood and balsa wood. The static and dynamic crush properties have been determined by performing tests on redwood and balsa wood of the types used in the impact limiters. The results of these tests are discussed in Section 2.6.7.5.1 of the SAR.

The applicant stated that no potential reactions occur between the wood and the stainless steel shells. The wood is coated with a preservative prior to installation in the impact limiter shell. These are standard applications of preservatives and adhesives, and no post application reaction occurs. No potential reactions are associated with the energy-absorbing or insulating material. The redwood and balsa wood provide high capacity, predictable energy absorption; both static and dynamic crush properties are provided. The staff reviewed the wood material properties provided against various technical publications, handbooks, and scientific articles and determined that the application used the appropriate wood properties in the structural calculations.

Resin:

The applicant states that NS-4-FR is a solid synthetic polymer—a borated hydrogenous material with neutron absorption capabilities similar to those of borated water. The NS-4-FR neutron shield material may be poured into, or may be pre-cast prior to placement into, a series of stainless steel enclosures. Each of these neutron shield assemblies are then welded closed and attached to the outer shell of the MAGNATRAN package. . The applicant states that each mixed batch of NS-4-FR shall be tested to verify that the material composition (aluminum and hydrogen), boron concentration and neutron shield density meet the requirements specified on the licensing drawings in the SAR. Testing shall be performed by qualified laboratories in accordance with written and approved procedures. Material composition, boron concentration and density data for each lot of neutron shield material shall become part of the quality record documentation package. Samples of each lot of neutron shield material shall be maintained as part of the quality record documentation package.

The applicant states that the installation of the NS-4-FR shall be performed in accordance with written, approved and qualified procedures. The installation procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pouring techniques are controlled in order to prevent gaps, or unacceptable voids, from occurring in the NS-4-FR. Installation procedures shall be qualified by the use of mockups.

The staff notes that the original material formulation, Dow and Bisco Products, has published material specification technical data information stating that NS-4-FR retains long-term functional stability at temperatures from -40°F to 300°F. In addition to this specific data, Bisco Products has performed thermal tests showing stability of the material through temperatures as high as 338°F. Several organizations associated with the current owner of the NS-4-FR technology have performed independent investigations of off gassing and material loss when NS-4-FR is confined in different configurations and exposed to temperature both less than and greater than 300°F. The staff notes that reports published on long-term thermal degradation and the thermal stability of resin as neutron shielding material for spent fuel transport package document material stability under a number of different conditions and temperatures ranging from 257°F through 392°F. The tests demonstrate that NS-4-FR is a stable material without any observed failures in the form of cracks at prolonged exposure to temperatures above the published specification limit of 300°F. The staff finds NS-4-FR acceptable for transportation based on the above discussion, the test results demonstrating NS-4-FR material stability at temperatures above the identified design limits and that NS-4-FR has been previously accepted by staff.

2.2.4.4 Brittle Fracture

The applicant states that the brittle fracture evaluation for structural integrity of the MAGNATRAN package for normal conditions of transport requires that the MAGNATRAN package be evaluated for normal conditions of transport at the most unfavorable ambient temperature in the range from -20°F to +100°F. The evaluations are conservatively made over the ambient temperature range of -40°F to +100°F. As discussed above, the MAGNATRAN package containment boundary is comprised of the top forging, the bottom inner forging, both welded to the inner shell, all Type 304 stainless steel. The package lid, 17-4 PH, Type 630, Condition H1025 stainless steel, the package lid inner metal O-ring and lid bolts, SB-637, Grade N07718 nickel alloy steel. Finally, the lid port coverplate, Type 304 stainless steel, the coverplate inner metal O-ring and coverplate bolts, ASME SA-193, Type 410, Grade B6 stainless steel.

After reviewing the applicant's evaluation, the staff concludes that, with the exception of the closure lid/coverplate bolts, all structural and containment components of the MAGNATRAN packaging are primarily fabricated of austenitic stainless steel. Austenitic stainless steel does not undergo a ductile-to-brittle transition, but a progressive reduction in Charpy impact value in the temperature range of interest (i.e., down to -40°F) and therefore, does not require brittle fracture evaluation. The bolt material has been evaluated to -40°F and lid bolt Charpy impact values verified. Therefore, brittle fracture is not a failure mode of concern. The staff has evaluated the MAGNATRAN packaging materials susceptibility to brittle fracture and finds it to be acceptable based on the above discussion. In addition, other package designs similar to the MAGNATRAN, constructed of similar materials, have been previously approved and successfully transported.

2.2.4.5 Creep Considerations

The applicant states that the TSC is the package structural component exposed to the highest temperatures, which remain below 752°F. The staff notes the criteria in order to assure integrity of the cladding material, for all fuel burnups (low and high), the maximum calculated fuel cladding temperature should not exceed 752°F for short-term loading operations (e.g., drying, backfilling with inert gas, and transfer of the cask to the storage pad). Specifically, creep in fuel cladding as a function of the cladding temperature and hoop stress, with creep exceeding 1.0% strain potentially causes gross rupture; however, as discussed in ISG-11 creep will not cause gross rupture if the cladding temperatures do not exceed 752°F (400°C) during loading or storage. The staff notes that the calculated maximum (peak) cladding temperature for the spent fuel during normal conditions of transport and short-term loading operations (i.e. loading, drying, backfilling with inert gas) does not exceed 570°C (1,058°F) for low burnup fuel, or 400°C (752°F) for high burnup fuel. High burnup fuel will be treated as damaged fuel and placed in damaged fuel cans.

2.2.4.6 Chemical, Galvanic or Other Reactions

The applicant states that no potential chemical, galvanic or other reactions have been identified for the MAGNATRAN package and no corrosion of mechanical surfaces is anticipated. The MAGNATRAN package is loaded with a TSC and handled dry. The operating environment of the MAGNATRAN package cavity contains helium, but MAGNATRAN package external surroundings could be air, rain, a marine environment, snow and/or ice. The exposed surfaces of the MAGNATRAN package and TSC are all stainless steel, except for the leak test annulus (outer) O-ring seals of the package lid. The applicant further states that shielding materials in the MAGNATRAN package body (lead and NS-4-FR) and the energy-absorbing materials in the impact limiters (wood) that are exposed only to the temperature effects of the operating environment. The sealed shielding material regions are typically evacuated and backfilled with helium, and the impact limiter shells are leak tested following fabrication. The metals oxidize any oxygen trapped in the sealed region until thermodynamic equilibrium is reached, i.e., a thin oxide layer develops on the lead. Similarly, the hydrogen in the NS-4-FR material captures any oxygen present until thermodynamic equilibrium is reached. Because the quantity of oxygen present, if any, is very small, equilibrium is reached very quickly and active corrosion in sealed regions does not occur.

In addition, no potential reactions are associated with the lubricants or grease or potential reactions associated with the silicone expansion foam. The staff reviewed the design drawings and applicable sections of the SAR to evaluate the effects, if any, of intimate contact between various materials in the MAGNATRAN package materials of construction during all phases of operation. In particular, the staff reviewed whether these contacts could initiate a chemical or galvanic reaction that could result in corrosion or combustible gas generation that could adversely affect safety. A review of the MAGNATRAN package system, its contents, and its operating environments has been performed to confirm that no operation will produce adverse chemical or galvanic reactions.

The staff finds that the MAGNATRAN packaging components are fabricated from materials designated by appropriate commercial industry specifications. The materials are not susceptible to chemical or galvanic reactions. In addition, most components are fabricated from corrosion-resistant materials. The staff finds no significant material interactions or galvanic reactions are to be expected, based on the above discussion. The staff finds the applicant performed extensive materials data characterization, including temperature dependence of mechanical properties. Packages constructed of similar materials to those used in the MAGNAGTRAN design have been previously approved and successfully transported. Further, visual inspections are to be performed of the payload cavity prior to loading and following off-loading and provide reasonable assurance against any considerable corrosion occurring unnoticed.

2.2.4.7 Effects of Radiation on Materials

The applicant states that materials selected for the MAGNATRAN packaging components have a long, proven history of use in the nuclear industry and are not affected by the radiation levels produced by the spent nuclear fuel. Significant neutron radiation damage does not occur for neutron fluences below the threshold of 10^{19} n/cm^2. This value is much greater than the neutron fluence exposure that is experienced by the MAGNATRAN packaging components. Significant gamma radiation damage to metals only occurs for doses of 10^{18} rads, or more. This value is much higher than the gamma dose produced by spent nuclear fuel in the MAGNATRAN package, approximately 10^{10} rads over a 10-year period. The staff finds significant neutron or gamma radiation damage of stainless steel components is not expected for neutron fluences below 10^{19} n/cm^2 or gamma doses below 10^{18} rads. These threshold values are higher than those experienced by the transport package assembly components, approximately 10^{10} rads for gamma and 10^{13} n/cm^2 for neutrons over a 10-year period. The staff notes that the MAGNATRAN packaging lid and lid port coverplate metallic O-rings are replaced and leak tested prior to each shipment of spent fuel. The outer EPDM O-rings are replaced annually, or as required based on visual inspections performed during package operations. Therefore, based on these considerations, the staff concludes that radiation deterioration of the seals is not credible during transportation. The staff concludes that the application provides reasonable assurance that the requirement of 10 CFR 71.43(d) is met.

The staff notes that the ASTM B29 lead and ASME SA-240 stainless steel shells of the MAGNATRAN packaging provide shielding between the TSC and the exterior surface of the MAGNATRAN package for the attenuation of gamma radiation. The radiation associated with the decay of spent fuel will have no effect on the austenitic stainless steel comprising the structural components of the MAGNATRAN package. The containment seal receives a negligible exposure. For these reasons, the staff finds there will be no deleterious radiation effects on the packaging, and the requirements of 10 CFR 71.43(d) are met.

2.3 Codes, Standards, Fabrication and Examination

The applicant states that the MAGNATRAN containment boundary components are designed, fabricated, tested and inspected to the requirements of the ASME B&PV Code, Section III, Division 1, Subsection NB, to the extent practical. The MAGNATRAN structures (other than lifting trunnions) assembled to, or surrounding, the containment boundary components are designed, fabricated and inspected to the requirements of the ASME B&PV Code, Section III, Division 1, Subsection NF, to the extent practical.

The applicant states that visual examinations of the finished surfaces of all welds of the MAGNATRAN containment boundary components shall be performed in accordance with ASME B&PV Code, Section V, Articles 1 and 9, with acceptance per Section III, Subsection NF-5360. The final surface of the welds in the MAGNATRAN containment boundary components shall be dye penetrant examined (PT) in accordance with ASME B&PV Code, Section V, Articles 1 and 6, with acceptance per Section III, Subsection NB-5350. The longitudinal and the circumferential welds of the MAGNATRAN containment boundary components shall also be radiographic examined in accordance with ASME B&PV Code, Section V, Articles 1 and 2, with acceptance per Section III, Subsection NB-5320. Repairs to the MAGNATRAN containment boundary component welds shall be performed in accordance with ASME B&PV Code, Section III, Subsection NB-4450, and the welds shall be re-inspected per the original acceptance criteria applicable to the examination method.

The applicant stated that the non-containment components and structures of the MAGNATRAN packaging shall be visual examined in accordance with ASME B&PV Code, Section V, Articles 1 and 9 with acceptance per Section III, Subsection NF-5360. The final surfaces of the welds of the MAGNATRAN package non-containment structures shall be liquid penetrant examined in accordance with ASME B&PV Code, Section V, Articles 1 and 6, with acceptance per Section III, Subsection NF-5350. Repairs to the nonstructural MAGNATRAN packaging components shall be performed in accordance with ASME B&PV Code, Section III, Subsection NF-4450, and the welds shall be re-inspected per the original acceptance criteria.

The staff verified that the weld design and inspections are in accordance with the recommendations in both NUREG/CR-3019[8] and NUREG/CR-3854[9], which includes the use of the ASME B&PV Code Section III, Subsection NB for containment boundary welds, and Subsections NF for other code welds, as appropriate. Non-code welds are examined in accordance with the ASME B&PV Code Section V, with acceptance criteria per Subsection NF. The staff concludes that the welded joints of the MAGNATRAN packaging meet the requirements of the ASME B&PV Code and the AWS Code, as applicable.

2.4 General Standards for All Packages (10 CFR 71.43)

2.4.1 Minimum Package Size

The transverse dimension of the package is approximately 87 inches and the longitudinal dimension is approximately 214 inches. Both dimensions are greater than 4 inches. The staff reviewed the applicant's evaluation and finds the package meets the requirements of 10 CFR 71.43(a) for the minimum size.

2.4.2 Tamper-Indicating Feature

Section 2.4.2 of the application notes that crimped wire seals are used on the package as tamper indicators. This involves a numbered metal cup seal looped through a hole in the end of an upper impact limiter retaining rod to preclude removal of the hex nuts. Also, the hex nuts on the retaining rod must be removed to remove the upper impact limiter to gain access to the package closure assembly for which a severed seal will indicate purposeful tampering. The staff reviewed the applicant's evaluation and finds the package meets the requirements of 10 CFR 71.43(b) for the tamper-indication.

2.4.3 Positive Closure

The large pre-load applied to the lid bolts prevents inadvertent opening of the package closure lid from internal or external loads or loads induced by shock, vibration, or thermal expansion during normal handling and transportation of the package. The staff reviewed the applicant's evaluation and finds the package meets the requirements of 10 CFR 71.43(c) for the positive closure.

2.5 Lifting and Tie-Down Standards for All Packages (10 CFR 71.45)

2.5.1 Lifting Devices

The applicant evaluated two types of lifting systems: (1) lifting trunnions bolted at 180° intervals to the top forging of the package body and (2) four lifting bolts threaded into the holes located on the top surface of the package lid assembly. The applicant noted that the provision for the material's factor of safety under 10 CFR 71.45(a) is being enveloped by those associated with NUREG-0612 and ANSI N14.6. As such, the lifting devices were evaluated in a non-redundant configuration to a load factor of 6 on material yield strength and a load factor of 10 on ultimate strength for the lifting components.

Section 2.5.1.1 of the application describes the 3-dimensional (3-D) finite element analysis (FEA) model used for trunnion components stress evaluation. Section 2.5.1.2 of the application notes the use of linearized stress values at section cuts for evaluating membrane and membrane-plus-bending stresses for the top forging. Stress allowables are also identified for other components, including those for the thread stress of the trunnion attachment bolt assembly.

As design load input, the applicant used a package weight of 299,000 pounds, which bounds the maximum loaded package configuration without impact limiters. A dynamic load factor of 1.10 is also applied in the FEA model of the trunnion assembly. Sections 2.5.1.1.1 through 2.5.1.1.8 present detailed stress evaluations of the trunnion components, including the top forging and the attachment bolt assembly. The staff reviewed the calculation assumptions and the resulting stress factors of safety.

The staff finds the trunnion design adequate because, by virtue of satisfying the enveloping NUREG-0612 and ANSI-N14.6 design criteria reviewed in the paragraph above, it meets the requirements of 10 CFR 71.45(a) for lifting devices. Additionally, in evaluating the lifting trunnion for overload, the applicant calculated the ultimate shear strengths, in Sections 2.5.1.2.1 and 2.5.1.2.2, of the lifting trunnion shank and the top forging, respectively. The staff reviewed the calculation assumptions and the results and found them acceptable. Since the ultimate strength of the top forging is demonstrated to exceed that of the lifting trunnion, the staff concludes the lifting trunnion will fail before the forging. For this reason, the staff finds the trunnion design also meets the excessive load protection requirements of 10 CFR 71.45(a) for the lifting devices.

The package lid, which weighs 10,500 lbs., is lifted with four lifting bolts threaded to the equally spaced 1-¼ -7 UNC-2B × 2.5-inch deep threaded holes, which are located on a 70-inch bolt circle. Considering a lid bolt engagement length of 1.5 inches and the at-temperature shear allowables for the lid constructed of ASME SA-693/SA-705, 17-4PH stainless steel, the applicant, in Section 2.1.5.3 of the application, calculated large factors of safety against lid material yield and ultimate strengths. The staff reviewed the calculated factors of safety, which are greater than 3 against the material yield strength. Since failure of threads of a bolt hole is local, the staff finds the package lid lifting design adequate to meet the requirements of 10 CFR 71.45(a) for the lifting devices, including the provision for excessive load protection.

2.5.2 Tie-Down Devices

The MAGNATRAN package is supported horizontally when transported on a railcar. The applicant evaluated the package tie-down to resist the static force components at 2, 10, and 5 times the package weight along the vertical, horizontal axial, and horizontal transverse directions, respectively, in accordance with 10 CFR 71.45(b)(1). Since the package front is held down by straps, the tie-down devices that are a structural part of the package include only the two rear rotation trunnions and the front shear ring on the package body.

Section 2.5.2.1.3 of the application considered a bounding package weight of 315,000 lbs and three values of the axial package center-of-gravity locations at 102, 105, and 108 inches, to calculate the reaction forces exerted on the tie-down components. This includes both the bearing and shear forces as applied to the shear ring, the package body, and the rotation trunnion pin and its support weld.

The applicant computed stresses in the components by hand calculations. Table 2.5.2-7 summarizes minimum factors of safety for the tie-down components, including the minimums of 1.01, and 1.13 for the average shear stress in the rotation trunnion pin and the shear ring, respectively. The rotation trunnion support weld stress margin is calculated to be 1.16. The staff reviewed the calculation assumptions and noted the greater than unity factors of safety. Because of this, the staff concludes that these component stresses are adequate in meeting the requirements of 10 CFR 71.45(b)(1) for the tie-down devices.

Section 2.5.2.3 of the application compared shear capacities of the rotation trunnion support weld and shear ring weld to those of the package body to demonstrate that the welds would fail before the package body. The staff reviewed the calculation assumptions and the results and determined them acceptable. Since the shear capacities of the package body is greater than those of the welds associated with the rotation trunnion support and the shear ring, the staff finds the tie-down devices meet the requirements of 10 CFR 71.45(b)(3) for the excessive load protection.

2.6 General Considerations for Structural Evaluation of Package

The applicant evaluated the structural performance of the package primarily by analysis. For the evaluation by analysis, both the FEA and hand calculations were used. For the evaluation by testing approach, wood material property tests and scale model drop tests performed and approved previously for packages with similar impact limiter designs were used to benchmark the impact limiter FEA methodology for calculating package drop deceleration g-loads. In the following sections the staff reviewed, in general, the applicant's structural evaluation of the package with respect to evaluation by analysis and by testing.

2.6.1 Evaluation by Analysis

2.6.1.1 Finite Element Analysis Codes

The applicant used two general-purpose, commercially available FEA codes, ANSYS® and LS-DYNA, to perform structural analysis of the package. The ANSYS® code was used to perform quasi-static analysis of the structural components, including those associated with the package body, TSC, fuel basket, GTCC waste basket liner and impact limiter. As loading or response conditions dictate, the explicit dynamic analysis code, LS-DYNA, was used to calculate transient impact responses, including rigid body decelerations for the package subject to free drop tests.

2.6.1.2 Finite Element Analysis Models

The applicant developed three-dimensional (3-D) FEA models to represent structures with individual components each discretized as an assembly of finite elements associated with specific sets of material properties. The governing modeling attributes regarding selecting element types to simulate structural performance of key packaging components are summarily reviewed below.

Section 2.12.2.6 of the application provides detailed description of the package body FEA model. The ANSYS® model includes the eight-node solid, SOLID45 elements to represent the package body components, such as package shell, end forgings, lid, bottom end plate, lead gamma shielding, and neutron shielding. The stiffness of the closure lid bolts and their engagement with the package body and bolt heads are simulated with the two-node bar, BEAM4 element in conjunction with the gap, CONTAC52 element.

The model also involves gap elements for the interfaces between components which may not be mechanically connected. This includes varying the stiffness of gap elements of contacts and physical separations, such as the 0.015-inch gap modeled between the lead and package body contact shells. Section 2.6.7.7.4 of the application provides details of the ANSYS® model using SOLID45, SHELL63, and CONTAC52 elements for attaching the neutron shield assembly of cooling fins, NS-4-FR filled shielding tube, and the retention stud assembly to the package body outer shell.

Section 2.6.12.2.1 of the application presents details for the TSC ANSYS® model constructed with the SOLID45 elements. The CONTAC52 elements are used to simulate the interaction of the TSC with the package body, bottom forging, and lid for package free drop tests. For the TSC composite lid construction, the interaction between the 4-inch-thick stainless steel lid and the 5-inch-thick carbon steel shield plate is modeled with CONTAC52 gap elements. The LINK10 elements are used to model the bolts attaching the shield plate to the lid.

Section 2.6.13.2 of the application describes details of the two 10-inch segment periodic FEA models used to evaluate the structural integrity of the PWR basket for side drop conditions. The ANSYS® models are constructed with the commonly used SOLID45, BEAM4, CONTAC52, and LINK10 elements. Additionally, for the mechanically interlocked pin-fuel tube slot connections between fuel tubes, the applicant used CONTA173 elements around the periphery of the pins and TARGE170 elements at the inner surface of the fuel tube slots for the interface simulation.

As noted in Section 2.6.14.1 of the application, the model attributes for the PWR damaged fuel basket are identical to those for the PWR fuel basket. The applicant described model details for the BWR fuel basket FEA, which uses essentially identical element types to those for the PWR basket. Also, as noted in Section 2.6.15.2 of the application, ANSYS® SHELL43 elements are introduced to model the tie plates of the basket corner weldment; however, only CONTAC52, in lieu of the CONTA173 and TARGE170, elements are used to simulate the force and {proprietary information removed} .

Section 2.6.16.2 of the application provides FEA model details for the GTCC waste basket liner which uses SOLID45 elements for the basket liner shell, bottom plate and spacer rings. The model includes TARGE170 and CONTA174 surface-to-surface contact elements to simulate the interface between the outer diameter of the liner shell and the inner diameter of the canister shell. These contact pairs are also used between the top of the canister bottom plate and the bottom of the liner spacer rings, as well as between the top of the liner shell and the bottom of the TSC shield plate.

Section 2.6.7.5 of the application provides FEA model details for the impact limiter attached to the package body for the side drop, end drop, and corner drop analyses using the LS-DYNA code.

The redwood and the balsa wood are modeled as {proprietary information removed}. The stainless steel shells enclosing the redwood and balsa wood are modeled as {proprietary information removed}. A series of beam elements with stainless steel elastic-plastic properties, which allow each threaded rod to independently experience collapse, are used to simulate impact limiter attachment to the package body. Also, to simulate structural component interface, the LS-DYNA "Automatic Single Surface" contact algorithm is employed between each part in the impact limiter, the package body, and the impact limiter shells.

In reviewing the FEA models used by the applicant to perform the structural analysis of the package, the staff considered the modeling bases and assumptions to ensure that the analysis methods described by the applicant could be implemented appropriately and results interpreted properly. This would include considerations of temperature-dependent material properties, force and displacement boundary conditions, elastic-plastic material model, and load combinations associated with appropriate temperature, pressure, and drop orientation considerations.

Except for the impact limiter FEA model benchmarking, which is to be reviewed with respect to the evaluation by testing approach in SER Section 2.6.2.2 below, the staff noted that the applicant had followed commonly accepted FEA practices in selecting element types and in using contact elements to simulate interfaces among structural components. The staff also noted that specific model attributes, including force and displacement boundary conditions and elastic-plastic material properties unique to package components, would further be reviewed in other SER sections. For this reason, the staff has reasonable assurance that the FEA models submitted by the applicant in support of this package design are acceptable for evaluating the structural performance of the packaging by analysis.

2.6.2 Evaluation by Testing

The applicant used the 30-ft drop tests of a ¼-scale model, which had been performed for the Model No. NAC-STC package (Docket No. 71-9235) Connecticut Yankee configuration for the predominately balsa wood impact limiters, to evaluate the MAGNATRAN impact limiter FEA models. The evaluation by testing amounts to a validation of the FEA methodology for calculating conservatively the rigid-body decelerations of the package body and crush depths of the impact limiters. As reviewed in the SER sections below, it also serves to validate the structural integrity calculation to demonstrate that the impact limiters remain attached to the package body and in position after the package 30-ft free drops.

2.6.2.1 Quarter-Scale Package Model Drop Tests

Section 2.12.2.4.3 of the application provides a detailed description of the confirmatory testing of the ¼-scale NAC-STC, Connecticut Yankee configuration balsa impact limiter model. The testing for the 30-ft top-end drop, top-corner drop, and side drop were performed for another docket at the Sandia National Laboratories. Included also in the description are the instrumentation and data reduction details for recording package's impact response time histories, measuring impact limiter crush depths, and low-pass filtering raw data to obtain rigid-body decelerations of the package body. In Docket No. 71-9235, the staff reviewed the testing program implementation and determined that test results, such as peak rigid-body decelerations and corresponding response pulse durations and shapes, had been adequately reduced. The tests confirmed the capabilities of the balsa impact limiters for the NAC-STC package, Connecticut Yankee configuration, in addition to providing the needed experimental data for validating impact limiter FEA modeling for calculating rigid-body decelerations of the package body.

2.6.2.2 Validation of LS-DYNA Impact Limiter Analysis Methodology

Section 2.12.2.3 of the application presents details of the ¼-scale NAC-STC-CY finite element models used to evaluate the correlation between the calculated impact limiter performance and the observed test results. In this evaluation, the LS-DYNA FEA models were constructed primarily with the 8-node bricks and 4-node shells elements. Considering the primary objective of calculating rigid-body response of the package body, the applicant used a single-shell representation, but with adjusted elastic modulus, to simulate the cross-sectional properties of the package body for the steel-lead-steel multi-wall construction.

The dynamic crush tests of wood specimens, performed previously by the applicant for other approved packages, were used for deriving stress-strain curves for the woods at varied strain rates. Specifically, the material properties used in modeling the impact limiter include the {proprietary information removed}

to simulate large deformations of the steel shell and gussets of the impact limiter. Other modeling attributes considered include the automatic single surface contact for all parts in the interface treatment and the Rigidwall_Geometric_Flat option to represent the unyielding impact surface. An initial velocity of 527.4 in/sec was applied to the entire model to represent the terminal velocity upon package landing on the unyielding surface of the 30-ft drop accident.

Figures 2.12.2-18 and 2.12.2-19 of the application compare the calculated responses to the tested rigid-body deceleration time histories measured at the top- and bottom-end accelerometer locations, respectively. The calculated peak decelerations and response pulse durations and shapes are seen to correlate well with those obtained by the tests. Similar deceleration time history plots with calculated results are reported in Figure 2.12.2-23 for the top-corner drop and Figure 2.12.2-28 for the top-end drop. Since the calculated peak decelerations all bound the corresponding test results and the calculated and measured impact limiter crush depths correlate reasonably well, as shown in Tables 2.1 and 2.2 below, the staff concludes that the LS-DYNA FEA methodology was adequately validated for application to the impact limiters for the NAC-STC, Connecticut Yankee configuration.

Table 2.1 Calculated and Tested Peak Decelerations - ¼-Scale Model, 30-Ft Drop

Drop Orientation	Drop Test (g)		Calculated (g)	
	Top End	Bottom End	Top End	Bottom End
Top Corner	126	N/A	129	N/A
Top End	122	N/A	130	N/A
Side	150	164	199	176

Table 2.2 Calculated and Tested Peak Crush Depths - ¼-Scale Model, 30-Ft Drop

Drop Orientation/Location	Measured Crush (inches)	Calculated Crush (inches)
Top Corner	5.50	5.52
Top End	4.40	4.64
Side - Near the Trunnion	2.88	2.73
Side - Lower Impact Limiter	2.88	2.77

2.6.2.3 NAC-STC Impact Limiter Analysis Methodology for MAGNATRAN Application

Both the MAGNATRAN and NAC-STC, Connecticut Yankee configuration packages are equipped with the balsa-redwood impact limiters. Section 2.12.2.3.2 of the application compares the impact limiter design features of the two packages to establish that the impact limiter analysis methodology benchmarked for NAC-STC, Connecticut Yankee configuration is also applicable to MAGNATRAN. Table 2.12.2-1 lists the package and impact limiter dimensions and weights of the two package designs, which shows a slightly longer MAGNATRAN package body than NAC-STC, Connecticut Yankee configuration with a weight increase from 260,000 lbs. to 312,000 lbs. Except for the core balsa wood dimensions sized for the end drop package protection, which measure
{proprietary information removed}
all other impact limiter features are identical for the two designs. These include the outer diameter, stainless steel shell casing, number of redwood segments and orientation, and radial and axial thicknesses of the redwood.

By virtue of the same construction with stainless steel shell casing and gussets, the applicant also noted the same manner in which the redwood segments are to maintain in position during the package side drop. For the end drop, the applicant noted the same key design features for the two packages, including the grain orientation in parallel to the direction of crush, ensure that the same wood crush properties are called into action in the FEA in developing crushing forces to decelerate the package in the same manner. Since the balsa wood is enclosed by stainless steel shells for both designs, the applicant also noted that the casing and gussets restraint to maintain the balsa wood in position is the same for both designs.

In reviewing the bases for using the validated NAC-STC Connecticut Yankee configuration impact limiter analysis methodology for the MAGNATRAN package, the staff noted a close similarity, in all key design features, of the two impact limiter designs as reviewed in the preceding paragraph. This provides assurance that both impact limiter designs will undergo similar crushing of wood and develop similar forces to decelerate the package during the drop. As to the package body rigid-body response time history acceptance criteria, the staff verified that the calculated peak deceleration and response pulse duration and shape amounted to a conservative estimate of input force for the package design evaluation by analysis. For those reasons, the staff finds the LS-DYNA analysis methodology validated for the NAC-STC Connecticut Yankee configuration impact limiter design acceptable for analyzing the MAGNATRAN impact limiter performance.

2.6.3 LS-DYNA Analysis of the MAGNATRAN Impact Limiter

Following the analytical approach validated above, the applicant constructed the LS-DYNA FEA models of the MAGNATRAN package weighing 312,000 lbs. to calculate impact limiter crush depths and package rigid-body decelerations. Section 2.6.7.5 of the application presents the model attributes, including using Modified_Crushable_Foam material type to allow implementation of the strain rate dependent wood properties. To account for wood crush strength and fabrication tolerance, the stress-strain curves for the hot, 100°F, and cold, -40°F, conditions were further adjusted by the factors of 0.90 and 1.10, respectively.

In Section 2.12.2.3.16 of the application, the applicant performed a parametric study, using the NAC-STC-CY FEA model, for the package drop angles of 15°, 10° and 5° with respect to the horizontal surface, to evaluate the package shallow-angle drop effect. The study established that the side drop results are also bounding for the MAGNATRAN by noting the nearly identical slenderness ratio, L/r, where "L" is the length and "r" the radius of gyration of the package. The staff verified the calculated slenderness ratios of 1.82 and the 1.85 for the NAC-STC Connecticut Yankee configuration and the MAGNATRAN packages, respectively. These values are also smaller than 2, the threshold for which a shallow-angle drop will result in a higher head-end than the tail-end side impact of the package. As a result, the staff concludes that, for the MAGNATRAN, the shallow-angle drop is bounded by the side drop and no further evaluation is needed.

Figures 2.6.7-7 through 2.6.7-12 of the application present the calculated responses, at the temperature conditions of hot and cold, for the 1-ft free drop tests for the corner, end, and side drop orientations. The peak package rigid-body deceleration {proprietary information removed} .

Figures 2.6.7-13 through 2.6.7-18 of the application present the calculated decelerations for the 30-ft drop tests. The peak package rigid-body deceleration of {proprietary information removed} . Table 2.6.7-37 summarizes the calculated impact limiter crush depths and package rigid-body decelerations for the hot and cold temperature conditions. As summarized in Table 2.3 below, the calculated rigid body decelerations are all below the corresponding design basis deceleration of 60 g considered in Section 2.7.1 of the application for structural evaluation of the packaging components for the 30-ft free drops.

Table 2.3 Comparison of Peak Package Rigid Body Decelerations

Drop Orientation	Deceleration (g) Cold (-40°F), Calculated	Deceleration (g) Hot (+200°F), Calculated	Design Basis (g)
30-Ft Side Drop			60
30-Ft End Drop			60
30-Ft Corner Drop			60

{proprietary information removed from Table}

On the basis of the evaluation above, the staff finds it conservative, thus, acceptable for the applicant to consider the calculated peak decelerations to establish the design basis decelerations for evaluating the packaging normal conditions of transport and hypothetical accident conditions free drop tests.

2.7 Normal Conditions of Transport

The application presents a variety of analyses to demonstrate that, under the tests and conditions specified in 10 CFR 71.71 for normal conditions of transport, there would be no substantial reduction in the effectiveness of the packaging for meeting its safety functions.

2.7.1 Heat

In Section 2.6.1 of the application, NAC analyzes the MAGNATRAN package for the thermal heat condition. The applicant used the maximum package component temperatures listed in Table 3.4-1 of the application under the normal conditions of transport, the maximum decay heat and maximum ambient temperature of 100°F to perform a differential thermal expansion analysis of the package and the TSC. This resulted in a minimum calculated diametric clearance {proprietary information removed} between the canister and the package, which ensures that the TSC can be removed freely without binding.

For the ANSYS® stress analysis, an internal package cavity pressure of 135 psig was used, which also envelops the maximum normal operating pressure (MNOP) of 23 psig for the package. The applicant post-processed stresses for the stress categories, per the ASME B&PV Code, Subsection NB, which include primary membrane (P_m), primary membrane-plus-bending ($P_m + P_b$), and primary membrane-plus-bending-plus secondary ($P_m + P_b + Q$) stresses. They are evaluated against the at-temperature allowables for the individual loading cases, such as the 135 psig internal pressure and lid closure bolt preload. As summarized in Tables 2.6.1-8 and 2.6.1-9 of the application for the combined internal pressure, thermal heat, and 1-g gravity, there are large stress factors of safety for all critical section cuts of the package body.

The staff reviewed the analyses performed by the applicant, as described above and determined that the analysis methods had been implemented appropriately and results interpreted properly. For this reason, the staff finds the results acceptable to demonstrate that the package meets the requirements of 10 CFR 71.71(c)(1) for the heat condition.

2.7.2 Cold

Section 2.6.2 of the application presents an ANSYS® stress analysis of the MAGNATRAN by considering an ambient temperature of -40° F, an internal pressure of 135 psig, no decay heat load, no solar insolation, and in still air and shade. The applicant post-processed the stresses for the stress categories, per the ASME B&PV Code, Subsection NB, which include primary membrane (P_m), primary membrane-plus-bending ($P_m + P_b$), and primary membrane-plus-bending-plus-secondary ($P_m + P_b + Q$) stresses. Table 2.6.2-1 through Table 2.6.2-7 of the application document stress results with stress factors of safety all greater than unity. The staff reviewed the applicant's evaluation and determined that the analysis methods had been implemented appropriately and results interpreted properly. For this reason, the staff finds the results acceptable to demonstrate that the package meets the requirements of 10 CFR 71.71(c)(2) for the cold condition.

2.7.3 Reduced External Pressure

Section 2.6.3 of application noted that a drop in atmospheric pressure to 3.5 psia effectively results in an additional package internal pressure of 11.2 psig. Considering the maximum package cavity internal pressure of 112.4 psig, the applicant calculated the maximum package cavity internal pressure of 123.6 psig. Since 123.6 psig is less than the 135 psig evaluated in the previous SER sections for the heat and cold conditions, the applicant recalculated package body stresses. This resulted in adequate structural performance, such as the maximum primary membrane stress of 9.86 ksi with a factor of safety of 2.03 and membrane-plus-bending stress of 15.66 ksi with a factor of safety of 1.92, both of which are greater than unity. The staff reviewed the applicant's evaluation and determined the evaluation approach of comparing loading conditions was appropriate. For this reason, the staff finds the results acceptable to demonstrate that the package meets the requirements of 10 CFR 71.71(c)(3) for the reduced external pressure condition.

2.7.4 Increased External Pressure

Section 2.6.4 of the application noted that an increased external pressure of 20 psia has a negligible effect on the MAGNATRAN package. The applicant evaluated the increased pressure condition by noting that the pressure exerted on thick outer shell and end closures of the package is bounded by that of the deep immersion condition of package external pressure of 290 psi, which is reviewed in SER Section 2.9 below. On the neutron shield assembly enclosure sidewalls, the applicant applied a uniform pressure of 5.3 psig to calculate a minimum factor of safety of 1.14, which is greater than unity, for the membrane-plus-bending stress. The staff also reviewed the hand calculation results with the fixed-edge plate panel boundary condition, which is conservative. For this reason, the staff finds the evaluation results acceptable to demonstrate the package structural performance for meeting the requirements of 10 CFR 71.71(c)(4) for the increased external pressure condition.

2.7.5 Vibration

Section 2.6.5 of the application determines that resonant response of the package is insignificant, considering the periodic impulse load effect as the two closest rail car wheels pass over a rail junction. To perform a structural capability evaluation, the applicant used a 2-g cyclic load amplitude, which is the static, vertical component load factor, per 10 CFR 71.45(b), to compute stress factors of safety. Considering the allowable alternating stress intensity of 23,000 psi, which corresponds to 10^{11} cycles of load applications for austenitic stainless steel, the applicant computed the stress factors of safety of 1.76 and 7.2, which are greater than unity, for the package body and its shear ring, respectively. The staff reviewed the applicant's evaluation and determined that the 2-g load amplitude is conservative for the cyclic loading effect. For this reason, the staff finds the evaluation results acceptable to demonstrate the package structural performance for meeting the requirements of 10 CFR 71.71(c)(5) for the vibration test.

2.7.6 Water Spray

The applicant states in Section 2.6.6 of the application that water causes negligible corrosion of the stainless steel shell of the MAGNATRAN transportation package. Considering the specified package surface temperature between -20°F and 100°F during the water spray, the applicant noted that the thermal stress induced in the package components is less than the thermal stresses occurring during the extreme temperature conditions for the normal conditions of transport. This resulted in the determination that the water spray test has no adverse effect on the package. The staff reviewed the evaluation and concluded that the rationale for the determination was acceptable. For this reason, the staff finds the evaluation results acceptable to demonstrate the package structural performance for meeting the requirements of 10 CFR 71.71(c)(6) for the water spray test.

2.7.7 1-Foot Free Drop

Section 2.6.7 of the application evaluates the 1-ft free drop condition, by FEA and hand calculations of the packaging components by subjecting the package to a bounding side-drop deceleration of 15 g, corner- and end-drop deceleration of 20 g and other enveloping loading conditions associated with applicable temperature, pressure, and bolt pre-load conditions. As reviewed below, the evaluations consider the stress performance and other structural failure modes, as appropriate, of the principal packaging components, including the package body, TSC, fuel basket, and GTCC waste basket liner.

2.7.7.1 Package Body

In SER Section 2.6.1.2 above, the staff reviewed the applicant's 3-D FEA models for analyzing the package body subject to free drop and environmental loading conditions presented in Section 2.12.2.6 of the application. As noted in Section 2.6.7 of the application, the environmental loadings, which were evaluated individually and also for load combinations, include hot and cold ambient temperatures with corresponding internal pressure of up to 135 psig. The calculated stresses are post-processed for applicable stress categories, including the primary membrane (P_m), primary membrane-plus-bending ($P_m + P_b$) and primary membrane-plus-bending-plus-secondary peak stress ($P_m + P_b + Q$) stresses in accordance with the Regulatory Guide 7.6[10] criteria.

Sections 2.6.7.1 through 2.6.7.3 of the application present the 1-ft end-, side-, and corner-drop analyses, respectively. Tables 2.6.7-1 through 2.6.7-12 list the stress results for the top- and bottom-end drops. Tables 2.6.7-13 through 2.6.7-18 list the stress results for the side drop and Tables 2.6.7-19 through 2.6.7-30 for the top and bottom corner drops. The reported stress factors of safety for all drop orientations are greater than unity.

2.7.7.1.1 Closure Bolts

Section 2.6.7.6 of the application evaluates the package closure assembly bolts, per NUREG/CR-6007[11], to demonstrate structural adequacy for the 1-ft drop tests. The forces considered in the analysis include a bolt installation torque of 4,600 ± 200 ft-lb, an internal pressure on the inner lid of 135 psi, the O-ring compression force, and the inertia associated with the lid, canister, basket, and fuel during the end drop.

The forty-eight 2-inch-diameter SB-637, Grade N07718 closure bolts were evaluated for two failure modes. The tensile plus shear stress results were evaluated using the interaction equation, based on the sum of the squares of the stress ratios. The calculated ratio of 0.67 is less than 1 and is, therefore, acceptable.

For the combined state of stress that includes tensile, shear, and bending against the stress intensity limit of 1.35 S_m, the resulting stress factor of safety of 1.14 is greater than unity and is acceptable. Considering the thermal load and the preload corresponding to the maximum installation torque of 4,800 ft-lb, Section 2.6.7.6.1 of the application calculates a bolt fatigue life of 364 cycles of the torque and un-torque operation based on the ASME B&PV Code Section III, Appendix I alternating stress criteria.

2.7.7.1.2 Neutron Shield Assembly

In Section 2.6.7.7 of the application, NAC evaluated the neutron shield assembly for differential thermal expansion and the end-drop, and side-drop structural performance. The applicant noted the use of the 0.375-inch-thick expansion foam on either side of the NS-4-FR blocks, which permits the expansion of the NS-4-FR relative to the ASME SA-240, Type 304 stainless steel enclosure. For the 2-inch gap between each end of the NS-4-FR and the stainless steel shell, which is filled with expansion foam, the applicant calculated an equivalent pressure of 140 psi incident on the enclosure end plate. This pressure is demonstrated not to result in a shear failure of the enclosure end plate during the end drop. Section 2.6.7.7.3 of the application presents the FEA for calculating the enclosure sectional stresses for a 20-g side drop. The resulting loading on the bolt was used for evaluating the nut, aluminum fin, and lower and upper washer through hand calculations. Tables 2.6.7-38 through 2.6.7-40 of the application list the factors of safety, which are all greater than unity, for the calculated maximum stresses, primary membrane (P_m), primary membrane-plus-bending ($P_m + P_b$), and primary membrane-plus-bending-plus-secondary peak stress ($P_m + P_b + Q$). For the bearing stresses summarized in Table 2.6.7-41 for the lower washer, upper washer, nut, aluminum fin and NS-4-FR, the minimum factor of safety of 2.8 applies to the aluminum fin.

The staff verified the hand calculations evaluated above. The staff also determined that the FEA had been implemented appropriately by following common FEA practices, including application of boundary conditions and selection of package body section locations for post-processing stress results.

{proprietary information removed}

For this reason, the staff finds the evaluation results acceptably demonstrate that the package body structural performance meets the requirements of 10 CFR 71.71(c)(7) for the free drop condition.

2.7.7.2 Transportable Storage Canister

The MAGNATRAN transportation package provides the containment boundary during the transport. The TSC emplaced in the package only needs to serve as a handling component for the basket and contents during loading and unloading from the transport package. In Section

2.6.1.2 of the SER, the staff reviewed the ANSYS® FEA models used for analyzing the TSC subject to free drops and other environmental loading conditions. For the 72-inch diameter TSC, which measures either at 191.8-inch or 184.8-inch long, the applicant evaluated its structural performance using three FEA models: A, B, and C, each corresponds to one of the three lid configurations.

- Model A is used to analyze the configuration with the 9-inch thick lid,
- Model B with the composite lid configured with the ¾-inch square closure ring; and
- Model C with the composite lid and the 1-½-inches wide closure ring.

Section 2.6.12.3 of the application presents the thermal stress analysis of the TSC, in conjunction with the 20-g side-drop load and canister internal pressure of 120 psig, which is evaluated individually and also for load combinations.

Section 2.12.3.3 of the application noted the nominal diametric and axial clearances in the cavity between the TSC and the package inner shell of 0.25 inch and 1.00 inch, respectively. They are both larger than the calculated differential thermal expansions. As a result, the applicant reported that no thermal growth interference would occur between the TSC and the package inner shell.

The applicant performed thermal stress analysis of the canister by applying the temperatures generated in the thermal analyses. The applicant computed the thermal stresses, which are evaluated for the combined thermal and drop loading conditions. Sections 2.6.12.4 through 2.6.12.9 of the application evaluated the TSC for the individual 1-ft end-, side-, and corner-drop conditions in conjunction with an internal pressure of 120 psig, and the combined loading conditions associated with the thermal stresses. For the canister structural performance, the stresses calculated by FEA are post-processed for applicable stress categories at critical canister locations.

The stress categorization includes the primary membrane (P_m), primary membrane-plus-bending ($P_m + P_b$) and primary membrane-plus-bending-plus-secondary peak stress ($P_m + P_b + Q$) stresses in accordance with the Regulatory Guide 7.6 stress evaluation criteria. The section cut locations for linearized stress evaluation are shown in Figure 2.6.12-4 for Model A and in Figure 2.6.12-5 for Models B and C. Table 2.6.12-1 through Table 2.6.12-72 list the stress evaluation results, including the canister locations for which the maximum stresses are identified with corresponding factors of safety for the 1-ft drops. The stress factors of safety for all drop orientations are greater than unity.

The staff verified the hand calculations reviewed above. The staff also determined that the FEA had been implemented appropriately by following common FEA practices, including application of boundary conditions and selection of TSC shell section locations for post-processing stress results. For this reason, the staff finds the evaluation results acceptably demonstrate that the TSC structural performance meets the requirements of 10 CFR 71.71(c)(7) for the free drop condition.

2.7.7.3 Fuel Basket

The carbon steel PWR fuel basket assembly inside the TSC is comprised of an array of 21 square fuel tubes. {proprietary information removed} The 21 fuel tubes function as individual cells, as well as sidewalls for the developed cells for fuel assemblies. Together with the side and corner weldments, which also serve partially as sidewalls, the basket provides 37 PWR fuel loading positions.

Section 2.6.13.1 of the application describes the load paths and interfaces between basket components, which include {proprietary information removed}, contacts between tube corner flats, and bolted connections between the outer tubes and support weldments. Section 2.6.13.2 provides details of the two 3-dimensional periodic half-symmetry FEA models of the basket for the side drop conditions. The elements used include SOLID45 and BEAM4 for the fuel tubes and support weldment and CONTAC52, CONTA173, TARGE170, LINK10, and COMBIN40 elements to simulate interfaces among the fuel basket components. The two FEA models at the 0° and 45° azimuthal angles with respect to the basket axis are selected to maximize calculated stresses in the fuel tube sidewalls and the bending stresses in the tube corners, respectively. As presented in Section 2.6.1.3.2.1, the boundary conditions for the analyses include a side impact load of 15 g and CONTAC52 elements to simulate the interface of the basket and the TSC shell inside the transportation package.

Section 2.6.13.3 of the application considered a basket bounding weight of 22,500 pounds and used hand calculations to evaluate the basket components under a 10-g inertia load for the 1-ft end drop condition. For the most critically stressed components, the applicant calculated the bearing stress factors of safety of 1.20 and 2.25, which are greater than unity, on the fuel tube and the TSC bottom plate, respectively.

In Sections 2.6.13.4.1 through 2.6.13.4.3, the applicant used a 3-D FEA to evaluate the fuel tube, the corner support weldment, and the side support weldment, respectively, for a 15-g inertia load for the 1-ft side drop condition. The stresses calculated by FEA were post-processed into applicable stress categories at critical sections of the fuel tube and support weldment and evaluated against the stress allowables at 750°F. The stress categorization includes the primary membrane (P_m), primary membrane-plus-bending ($P_m + P_b$), and primary membrane-plus-bending-plus-secondary peak stress ($P_m + P_b + Q$) stresses in accordance with the ASME B&PV Code, Section III, Subsection NG stress criteria. Tables 2.6.13-1 through 2.6.13-20 list the stress evaluation results, including the locations for which calculated stresses were evaluated with corresponding factors of safety for the 1-ft drops. The stress factors of safety for basket components for all drop orientations are all greater than unity. As is also reported in the application, the interface forces used to evaluate fuel tube weldment attachment and the bearing stress on TSC shell all resulted in stress factors of safety greater than unity.

Section 2.6.13.5 of the application evaluated thermal expansion of the PWR basket using average axial temperatures at the center and at the outer radius of the basket of 650°F and 400°F, respectively. The applicant determined a bounding relative thermal expansion of 0.174 inches in the axial direction between the center and the edge of the basket. {proprietary information removed} , however, the applicant determined that no axial thermal stresses are produced by the axial expansion of the basket.

Section 2.6.13.7 of the application provides details of the LS-DYNA FEA for evaluating side drop of the neutron absorber retainer strips made of Type 304 stainless steel. The calculated maximum stress intensity of 24 ksi in the retainer has a stress factor of safety of 1.05. The induced thermal stress of 37,350 psi, based on hand calculations of thermal expansion of dissimilar materials, resulted in a stress factor of safety of 1.28.

The applicant evaluated the PWR damaged fuel basket by following essentially the same analysis approaches for the PWR fuel basket discussed above. Section 2.6.14 of the application notes that the PWR damaged fuel basket is also designed to accommodate up to

37 undamaged PWR fuel assemblies, including four damaged fuel can locations. This is accomplished by using 17, in lieu of 21, fuel tubes, four side support weldments, and four damaged fuel basket corner support weldments for loading the fuel. The fuel tubes and side support weldments of the damaged fuel basket are identical to those for the standard PWR fuel basket, while each damaged fuel basket corner support weldment is configured to accommodate one damaged fuel can. The structural evaluation for the PWR damaged fuel basket is performed using the criteria for Service Level A limits of ASME B&PV Code, Section III, Subsection NG. Sections 2.6.14.1 through 2.6.14.7 of the application replicated the evaluation of the PWR fuel basket for the PWR damaged fuel basket with applicable design attributes. The resulting stress factors of safety are all greater than unity.

Similarly, the applicant evaluated the BWR fuel basket by following essentially the same analysis approaches of those for the PWR fuel basket discussed above. Section 2.6.15 of the application notes that the BWR fuel basket constructed with 45 tubes to accommodate 87 BWR fuel assemblies is evaluated with the same Service Level A limits from ASME B&PV Code, Section III, Subsection NG. Sections 2.6.15.1 through 2.6.15.7 of the application replicated the evaluation of the PWR fuel basket for the BWR fuel basket with applicable design attributes. The resulting factors of safety are all greater than unity.

The staff verified the hand calculations and the stress results discussed above. The staff also determined that the FEA had been implemented appropriately by following common FEA practices, including application of boundary condition assumptions and selection of tube and weldment section cuts for post-processing stress results. For this reason, the staff finds the evaluation results acceptably demonstrate the fuel basket structural performance meets the requirements of 10 CFR 71.71(c)(7) for the free drop condition.

2.7.7.4 Greater-Than-Class C Waste Transportable Storage Canister and Waste Basket Liner

The GTCC TSC, which measures 184.8 inch in length, is the short version of the two TSC designs. The stainless steel GTCC waste basket liner is sized to dimensionally fit in a GTCC TSC.

Following the same evaluation approaches as those for the TSC reviewed in Section 2.7.7.2 of the SER above, the applicant essentially replicated the TSC evaluation for the GTCC TSC with all applicable design attributes. As noted in Section 2.6.12.6 of the application, for Finite Element Model B with the composite lid and the ¾-inch square closure ring, the attributes include: (1) a content weight of 55,000 lbs., (2) an internal pressure of 5 psig, (3) a heat load of 4 kW, and (4) the side-drop inertia of 15 g and end- and corner-drop inertia of 8 g.

Section 2.12.6.3 of the application computes component temperatures of the GTCC TSC for heat conditions with a 4 kW canister heat load. In addition to large stress factors of safety, the applicant stated that the differential thermal expansions of the GTCC-TSC and the package inner shell in both the radial and axial directions are bounded by those determined in Section 2.6.12.3 for the standard TSC. Because of the same materials and comparable dimensions for the TSCs of different length and the GTCC internal heat load of 4 kW is lower than the design basis of 23 kW for the standard TSC, the staff finds the thermal expansion evaluation results by the applicant acceptable.

Sections 2.6.16.4 through 2.6.16.9 of the application evaluated the GTCC TSC for the individual 1-ft end-, side-, and corner-drop conditions in conjunction with the 5 psig internal pressure and the combined loading conditions. For the canister structural performance, the stresses calculated by FEA are post-processed into applicable stress categories at critical canister locations. This stress categorization includes the primary membrane (P_m), primary membrane-plus-bending ($P_m + P_b$) and primary membrane-plus-bending-plus-secondary peak stress ($P + Q$) stresses in accordance with the ASME B&PV Code Section III, Subsection NB stress evaluation criteria. Tables 2.6.16-1 through 2.6.16-46 list the stress evaluation results, including the canister locations for which the maximum stresses were identified. The stress factors of safety for all drop orientations are greater than unity.

Section 2.6.16.13 of the application performed stress evaluation of the GTCC waste basket liner using the 3-D FEA. As noted in Section 2.6.16.1, the structural design criteria are based on the ASME B&PV Code Section III, Subsection NF, as summarized in Table 2.1.2-3 of the application, for the non-containment boundary components of the MAGNATRAN package. Tables 2.6.16-47 through 2.6.16-49 list the stress evaluation results for critical liner sections, all with large factors of safety.

The staff verified the hand calculations and stress results discussed above. The staff also determined that the FEA had been implemented appropriately by following common FEA practices, including application of boundary condition assumptions and selection of TSC shell section locations for post-processing stress results. For this reason, the staff finds the evaluation results acceptably demonstrate the structural performance of the GTCC TSC and its waste basket liner meet the requirements of 10 CFR 71.71(c)(7) for the free drop condition.

2.7.7.5 Package Cavity Spacer

Section 2.6.17 of the application presents design details of the package cavity spacer, which is a weldment made of ASME SA-240, Type 304, 3/8-inch stainless steel plate. The weldment consists of a 3/8-inch thick by 70.7-inch diameter stainless base with six raised cylinders of different diameters welded to it. The spacer is attached to the bottom of the closure lid by four 1-8 UNC-2A bolts of ASME SA-193, Gr B6 material.

Considering the loaded canister weight of 105,000 lbs., the applicant used a 1/8-periodic FEA model to perform stress analysis for a vertical-drop inertia load of 20 g. This resulted in a stress factor of safety of 2.33 for the shell membrane stress per the ASME B&PV Code, Section III, Subsection NF stress evaluation criteria. By hand calculations for the side-drop condition, the calculated shear stress of 8.2 ksi in the attachment bolts has a factor of safety of 2.48, which is greater than unity and is acceptable.

The staff reviewed the applicant’s evaluation and determined that the evaluation approaches are consistent with those previously approved for the NAC spent fuel transportation package designs. For this reason, the staff finds that the structural performance of the canister cavity spacer meets the requirements of 10 CFR 71.71(c)(7) for the free drop condition.

2.7.8 Corner Drop

The corner drop test, per 10 CFR 71.71(c)(8), does not apply because the package weighs more than 50 kg (110 lbs.) and is not of fiberboard or wood construction.

2.7.9 Compression

The compression test, per 10 CFR 71.71(c)(9), does not apply because the weight of the package exceeds 5,000 kg (11,000 lbs.).

2.7.10 Penetration

The staff reviewed the applicant's evaluation in Section 2.6.10 of the application and confirmed that the package has no unprotected valves or rupture disks that could be affected by the penetration test. On this basis, the staff concludes that the package meets the requirements of 10 CFR 71.71(c)(10) for the penetration test.

2.7.11 Fabrication Stresses

The applicant evaluated the lead pour fabrication stresses in the package inner shell in Section 2.6.11 of the application. Considering the hydrostatic pressure of the melted lead and the pressure resulting from lead shrinkage after cooldown, the applicant used a hand calculation to obtain a maximum hoop stress of 1,416 psi in the package inner shell. The staff verified the calculations and concludes that the lead pour operation will have negligible stress effects on the inner shell after the lead creep effect is considered. On this basis, the staff finds that the lead pour fabrication stresses need not be considered further for meeting the 10 CFR 71.71(b) requirements on the initial condition requirements for evaluating other tests and conditions.

2.8 Hypothetical Accident Conditions

Section 2.7 of the application presents structural analysis of the packaging components to demonstrate that the package has adequate structural integrity for the hypothetical accident conditions free drop, crush, puncture, thermal (fire) and water immersion tests and conditions.

2.8.1 30-Foot Free Drop

Section 2.7.1 of the application described FEA and hand calculations used to analyze the packaging components, including the package body, TSC, fuel basket, and GTCC waste basket liner for the 30-ft free drop conditions in conjunction with other applicable loadings associated with temperature, pressure, and bolt pre-load conditions.

2.8.1.1 Package Body

In Section 2.6.1.2 of this SER, the staff reviewed the FEA models for the package body subject to free drop and environmental loading conditions. The same FEA approach, which was presented in Section 2.12.2.6 of the application and used for the 1-foot drop, is also used for the 30-foot drop, with a few exceptions. The exceptions include primarily: (1) the stresses arising from thermal expansion are not considered for the stress evaluation, (2) a bounding 60-g inertial load is used for all drop conditions, and (3) the use of elastic-plastic analysis, per the ASME B&PV Code, Section III, Appendix F, for evaluating the package body sections undergoing material yielding. As a result, analyses to address load combinations were performed to also include four thermal and internal pressure combinations, considering the bounding internal pressures of 135 psig and 0 psig. The calculated stresses at 32 section cuts are post-processed for applicable stress categories of the primary membrane (P_m) and primary membrane-plus-bending ($P_m + P_b$) stresses.

Sections 2.7.1.1 through 2.7.1.3 of the application present the 30-ft end-, side-, and corner-drop analyses, respectively. Tables 2.7.1-1 through 2.7.1-8 list the stress results for the top and bottom end drops. Tables 2.7.1-9 through 2.7.1-12 list the stress results for the side drop, and Tables 2.7.1.13 through 2.7.1-20 for the top and bottom corner drops. The reported stress factors of safety for all drop orientations are greater than unity. Specifically, the minimum factors of safety for all drop orientations are 1.35 and 1.23 for P_m and $P_m + P_b$, respectively. The former occurs at the base of the package flange for the side drop and latter at the inner bottom forging for the bottom corner drop.

2.8.1.1.1 Lead Gamma Shield

Section 2.7.1.5 of the application assumes that the lead gamma shielding could slump and fill the annular gap between the package inner and outer shells due to the cooling of the lead after fabrication. The applicant used hand calculations to obtain a maximum gap of 0.87 inches at the top of the lead annulus during a package end drop and a maximum radial slump of 0.47 inches after a package side drop. The applicant considered these gaps for the shielding evaluation, which is reviewed in Section 5 of this SER.

2.8.1.1.2 Package Closure

Section 2.7.1.7 of the application analyzed the package closure assembly to demonstrate structural adequacy for two load cases: the design basis end-drop of 60 g and the puncture load applied at the center of the package lid.

For the failure mode associated with bolt tensile plus shear stresses for the end drop condition, the calculated interaction equation ratio is 0.39, which is less than 1.0. For the combined state of stress that includes tensile, shear, and bending, the calculated maximum bolt stress intensity of 116 ksi resulted in a factor of safety of 1.6 against the ultimate strength, S_u, of 185 ksi. For the puncture load having the same interaction equation ratio of 0.39, the large bolt bending resulted in a combined stress intensity of 145 ksi with a factor of safety of 1.28. The maximum bolt stresses of 116 ksi and 145 ksi are both less than the yield strength allowable of 150 ksi. This ensures that the package containment boundary seal is not broken during the 30-ft package top-end drop or the puncture drop accident condition.

2.8.1.1.3 Neutron Shield Assembly

Section 2.7.1.8 of the application evaluated the structural integrity of the neutron shield assembly for the end-drop and side-drop inertia load of 60 g. Using the FEA model presented in Section 2.6.7.7.4 of the application, large stress factors of safety are calculated for the nut and lower and upper washers of the retention stud assembly. For the tensile load in retention stud, which is welded to the package outer shell and is the most critically stressed, the applicant used a hand calculation to obtain a maximum tensile force of 44.6 ksi. The resulting greater than unity stress factor of safety of 1.05 against the allowable of 46.3 ksi ensures that the neutron shield is adequately supported during the 60-g side drop accident. As can be seen in license drawings,

{proprietary information removed}

For the free drop, the fins are not impacted and the bolts of the bolted neutron shield assemblies are not stressed such that the bolted neutron shield assemblies will detach from the package's outer shell to any extent. Therefore, the neutron shield assemblies that are not bolted to the outer shell will also remain attached to the package's outer shell.

2.8.1.1.4 Packaging Inner Shell

Section 2.7.14 of the application evaluated the buckling potential of the package inner shell using the ASME B&PV Code Case N-284-1 metal containment shell buckling design methods. The data considered for the buckling evaluation includes shell geometry parameters, shell material properties and the internal stresses associated with the longitudinal membrane, circumferential membrane, and in-plane shear stresses for the individual drop cases. The peak thermal stresses calculated were also added to the stress results for the 30-foot package end drop condition. The results of the buckling analysis of the package inner shell are summarized in Table 2.7.14-1 of the application with the interaction equation ratios among the applicable axial, hoop, and in-plane shear stresses all less than 1.0. This demonstrates that buckling of the transport package inner shell does not occur.

The staff verified the hand calculations and stress results evaluated above. The staff also determined that the FEA had been implemented appropriately by following common FEA practices, including application of boundary conditions and selection of package body section locations for post-processing stress results. For this reason, the staff finds the applicant's evaluation and analysis results acceptably demonstrate the package body structural performance meets the requirements of 10 CFR 71.73(c)(1) for the free drop condition.

2.8.1.2 Transportable Storage Canister

Sections 2.7.8 of the application evaluated the TSC for an accident internal pressure of 300 psig and the 30-ft free drop conditions with a 120 psig internal pressure. The evaluation was performed using the same FEA approach used for the normal conditions of transport conditions except that a design basis inertia load of 60 g, with different internal pressure, was considered; however, per the ASME B&PV Code, Section III, Appendix F provisions for accident conditions, the secondary, thermal stresses arising from thermal expansion are not considered for the stress evaluation. For the canister structural performance, the calculated stresses for either 16 or 15 section cuts are post-processed for appropriate stress categories of the primary membrane (P_m) and primary membrane-plus-bending ($P_m + P_b$) stresses in accordance with Regulatory Guide 7.6 stress evaluation criteria. The 15 and 16 section cut locations for linearized stresses are shown in Figure 2.6.12-4 for Model A and in Figure 2.6.12-5 for Models B and C, respectively.

Section 2.7.8.1 of the application evaluated the TSC for the accident internal pressure of 300 psig. Sections 2.7.8.2 through 2.7.8.4 evaluated the TSC for the end, side, and corner drops, respectively. As reported in Tables 2.7.8-1 through 2.7.8-45, the stress factors of safety for all drop orientations are greater than unity. Specifically, at an internal pressure of 120 psig, the calculated minimum factors of safety for all drop orientations are 1.19 and 1.43 for primary membrane (P_m) and primary membrane-plus-bending (P_m+P_b) stresses, respectively. As noted in Section 2.7.8.3 of the application, these minimum factors of safety are associated with the side drop condition and the stress results by averaging over the bearing region of an 18° impact limiter sector between the canister closure weld and canister shell.

Section 2.7.8.5 of the application evaluated the buckling potential for the TSC shell using the ASME B&PV Code Case N-284-1 metal containment shell buckling design methods. The evaluation followed the same approach used for evaluating the package inner shell. The data considered for the buckling evaluation includes shell geometry parameters, shell material properties, and the internal stresses associated with the longitudinal membrane, circumferential membrane, and in-plane shear stresses for the individual drop load cases. The results of the buckling analysis of the TSC shell are summarized in Table 2.7.8-46 with the interaction equation ratios of less than 1.0. This demonstrates that buckling of the TSC shell does not occur.

The staff reviewed the FEA stress results summarized above. The staff determined that the FEA has been implemented appropriately by following common FEA practices, including application of boundary conditions and selection of TSC shell section locations for post-processing stress results. For this reason, the staff finds the evaluation results acceptably demonstrates that the TSC structural performance meets the requirements of 10 CFR 71.73(c)(1) for the hypothetical accident conditions free drop condition.

2.8.1.3 Fuel Basket

2.8.1.3.1 Fuel Basket Structural Integrity

Section 2.7.9.1 of the application followed the FEA and hand calculation approaches used previously for normal conditions of transport to evaluate the basket components for the design basis inertia load of 60 g. For the most critically stressed components, the applicant calculated the membrane stress factor of safety of 1.17 in the standoff and the bearing stress factor of safety of 1.09 on the TSC base plate, which are greater than unity.

In Sections 2.7.9.2.1 through 2.7.9.2.3 of the application, the applicant used a 3-D FEA to evaluate the fuel tube, corner support weldment, and side support weldment, respectively, for a 60-g inertia load for the 30-ft side drop condition. The stresses calculated by the FEA were post-processed into applicable stress categories at critical sections of the fuel tube and the support weldments and evaluated against the stress allowables at 725°F for the fuel tube and 500°F for the support weldments. Since secondary stresses need not be considered for the accident conditions, the stress categorization includes only the primary membrane (P_m) and primary membrane-plus-bending ($P_m + P_b$) stresses in accordance with the ASME B&PV Code, Section III, Subsection NG stress criteria.

Tables 2.7.9-1 through 2.7.9-8 list the stress evaluation results, including the section cut locations for which stresses were evaluated with corresponding factors of safety. The stress factors of safety for basket components are all greater than unity. As is also reported in the application, the interface forces used to evaluate fuel tube weldment attachment and the bearing stress on TSC shell all resulted in stress factors of safety greater than unity.

Section 2.7.9.3 of the application provides the LS-DYNA FEA modeling details to evaluate side drop of the neutron absorber retainer strips made of Type 304 stainless steel for the PWR fuel tubes. The calculated maximum stress intensity of 33.8 ksi in the retainer strip has a stress factor of safety of 1.68. For the calculated maximum strain of 2.1% at the conical-shaped bearing surface of the retainer strip, the applicant noted its localized deformation nature for retaining the neutron absorber in place after the 30-ft side drop.

Section 2.7.10 of the application evaluated the PWR damaged fuel basket by following essentially the same approaches of those for the PWR fuel basket discussed above. As reviewed also in SER Section 2.7.7.3 above, the fuel tubes and side weldment for the damaged fuel basket are identical to those of standard PWR fuel basket, while each damaged fuel basket corner support weldment is configured to accommodate one damaged fuel can. The structural evaluation for the PWR damaged fuel basket is performed using the criteria for Service Level A limits of the ASME B&PV Code, Section III, Subsection NG. Using an inertia load of 60 g, Sections 2.7.10.1 and 2.7.10.2 of the application replicated the evaluation of the PWR fuel basket for the PWR damaged fuel basket with applicable design attributes. The resulting stress factors of safety are all greater than unity.

Similarly, the applicant evaluated the BWR fuel basket by following essentially the same analysis approaches of those for the PWR fuel basket discussed above. Using an inertia load of 60 g, Section 2.7.11 of the application evaluated the BWR fuel basket assembly with the same Service Level D limits of ASME B&PV Code, Section III, Subsection NG. Sections 2.7.11.1 through 2.7.11.3 of the application replicated the evaluation of the PWR fuel basket for the BWR fuel basket with applicable design attributes. The resulting factors of safety are all greater than unity.

The staff verified the hand calculations and stress results discussed above. The staff also determined that the FEA had been implemented appropriately by following common FEA practices, including application of boundary condition assumptions and selection of tube and weldment section cuts for post-processing stress results. For this reason, the staff finds the evaluation results acceptably demonstrate that the fuel basket structural performance meets the requirements of 10 CFR 71.73(c)(1) for the hypothetical accident conditions free drop condition.

2.8.1.3.2 Fuel Basket Geometric Stability

Section 2.7.13.1 of the application notes that the basket geometric configuration is maintained by three features of basket component design:

- the side and corner support weldments bolted to the fuel tubes at the basket periphery,
- {proprietary information removed}, and
- {proprietary information removed}.

To demonstrate geometric stability, the applicant evaluated the capability of the basket assembly to continue to retain its initial geometric configuration in that {proprietary information removed} . This is accomplished by evaluating transient responses of the basket models.

{proprietary information removed}

Figure 2.7.13-2 through Figure 2.7.13-4 present the LS-DYNA FEA models constructed primarily of brick and shell elements with contact/gap interfaces.

The PWR fuel assembly for the 0°, 22.5°, and 45° side drop analyses covered the range of drop orientations for which the most geometric instability potential for the fuel basket is expected. Based on the basket assembly fabrication process and tolerances, a bounding gap of {proprietary information removed} , is incorporated at specific locations as baseline to maximize effects of these gaps on the fuel tube relative displacements. In addition to the gap interfaces among tubes and support weldments, other key model attributes include the gaps between the basket assembly and its support weldments as accorded by the TSC shell and the implementation of gap elements between the canister and the inner shell of the transportation package.

Using the ANSYS® quasi-static analysis of the package body for a 60-g design-basis side impact, the applicant calculated the TSC shell deformation profile near the package mid-section for which the maximum shell ovalization is expected. This deformation profile is assumed to be a rigid surface boundary condition imposed by the package body inner shell. {proprietary information removed}

for determining the margin of safety against basket geometric instability as reviewed below.

{proprietary information removed}

In Section 2.7.13.1 of the application, NAC evaluated the analysis conservatism in the FEA modeling of the basket geometry stability. The ASME B&PV Code, Section III, Appendix F, Plastic Instability Load evaluation provides that the calculated instability load must be demonstrated with a factored load equal to or greater than {proprietary information removed} {proprietary information removed} . In evaluating the analysis results, the staff noted that the applicant did not explicitly perform an analysis for a factored load sufficient to result in an incipient collapse with gap changes among the fuel basket components equal to or greater than the {proprietary information removed}. Rather, by considering also the package inner shell profile as an amplified boundary condition by a factor of {proprietary information removed} {proprietary information removed} , to meet the intent of the ASME B&PV Code, Section III, Appendix F provision, is practically introduced for evaluating the basket fuel tube gap changes. On this basis, given further that the maximum gap change of {proprietary information removed} , the staff has reasonable assurance to conclude that the PWR fuel basket has adequate structural design margin for demonstrating its performance against fuel basket geometric instability.

{proprietary information removed}

The applicant evaluated the BWR fuel basket by following essentially the same approaches of those for the PWR fuel basket reviewed above, including identical boundary conditions and the acceleration time history applied at the package body inner shell.

{proprietary information removed}

{proprietary information removed}

Based on the evaluation above, the staff finds that NAC's evaluation results acceptably demonstrate that the fuel basket geometric stability for meeting the requirements of 10 CFR 71.73(c)(1) for the hypothetical accident conditions free drop condition.

2.8.1.4 Greater-Than-Class C Waste Transportable Storage Canister and Waste Basket Liner

Section 2.7.12 of the application followed essentially the same FEA approaches of those for the normal conditions of transport drops to evaluate the GTCC TSC for the 30-ft drops. The only major analysis attributes different from those for the normal conditions of transport evaluation include using the 250°F at-temperature stress allowables and inertia g-loads associated with the 30-ft drops.

Section 2.7.12-1 of the application reported large stress factors of safety for the canister subject to an internal pressure of 10 psig. Sections 2.7.12.2 through 2.17.12-4 evaluate the GTCC TSC for the 30-ft end, side, and corner drops, respectively. For the canister structural performance, the stresses calculated by FEA are post-processed into applicable stress categories at critical canister locations. The stress categorization includes primary membrane (P_m), and primary membrane-plus-bending ($P_m + P_b$) stresses in accordance with the ASME B&PV Code Section III, Subsection NB stress evaluation criteria. Tables 2.7.12-3 through 2.7.12-25 list the stress evaluation results, including the section cut locations for which the maximum stresses were identified with corresponding factors of safety. The stress factors of safety for all drop orientations are greater than unity.

In Section 2.7.12-5, from the buckling performance perspective, the applicant noted that design attributes of the closure lid weight and length of the canister shell of the short GTCC TSC are bounded by those of the long TSC. As a result, the TSC buckling capability as evaluated and demonstrated previously for the standard TSC also bounds the GTCC TSC buckling performance.

Section 2.7.12.6 of the application evaluated the GTCC waste basket liner for the 30-ft drop with no internal pressure applied. For the most critically stressed sections, the minimum factors of safety are 1.43 and 1.01 for the primary membrane (P_m) and primary membrane-plus-bending ($P_m + P_b$) stresses corresponding to the side drop the bottom-corner drop, respectively.

The staff verified the hand calculations and stress results described above. The staff also determined that the FEA has been implemented appropriately by following common FEA practices, including application of boundary condition assumptions and selection of TSC shell section locations for post-processing stress results. For this reason, the staff finds the evaluation results acceptably demonstrate that the structural performance of the GTCC TSC and its waste basket liner meets the requirements of 10 CFR 71.73(c)(1) for the hypothetical accident conditions free drop condition.

2.8.1.5 Package Cavity Spacer

The applicant continued to use the normal conditions of transport FEA approach to evaluate the package cavity spacer structural performance for hypothetical accident conditions. The design basis inertia load of 60 g was used for the top-end, bottom-end, and side-drop conditions. This resulted in a stress factor of safety of 1.17 for the shell membrane stress against the calculated maximum nodal stress of 23.9 ksi for the package top-end drop. Based on the closed-form solution of 29.3 ksi for local shell buckling, the applicant calculated a stress factor of safety of 1.23 against local buckling. Additionally, by a hand calculation for the side-drop condition, the resulting shear stress of 24.6 ksi in the attachment bolts has a factor of safety of 1.68.

The staff reviewed the applicant's evaluation and determined that the evaluation approaches are consistent with those previously approved for other NAC spent fuel transportation package designs. On this basis, the staff finds that the structural performance of the canister cavity spacer meets the requirements of 10 CFR 71.73(c)(1) for the hypothetical accident conditions free drop condition.

2.8.2 Crush

Because the package weighs more than 500 kg (1,100 lbs) and has an overall density that is greater than that of water, the crush test, per 10 CFR 71.73(c)(2), is not applicable.

2.8.3 40-Inch Puncture Test

The applicant evaluated effects of a 40-inch free drop of the package onto an upright 6-inch-diameter mild steel punch pin for three puncture locations: (1) the mid-point of the package side, (2) the center of the package lid, and (3) the center of the package bottom.

Using a total force corresponding to the steel bar material yielding with strain rate effect, the applicant calculated the puncture pin reaction force to be equivalent to a package deceleration of about 4.51 g. The localized puncture effects were evaluated by a stress analysis; however, using a general approach of applying a uniformly distributed pressure of 47,000 psi over a 6-inch-diameter region at the puncture pin location. Sections 2.7.3 of the application presents details of three ANSYS® FEA quarter-symmetry models for the package side, lid center, and bottom plate pin puncture drop conditions.

Tables 2.7.3-4 through 2.7.3-8 list the stress analysis results with all factors of safety greater than unity. Specifically, for the side, top end, and bottom end puncture drop, the calculated maximum stress factors of safety are 1.10, 2.60, and 1.83, respectively. Also, considering the pin perimeter as a shear plane through the package outer shell, the applicant calculated a puncture shear factor of safety 1.24. This demonstrates that the shear puncture failure would not occur to the 2.25-inch thick outer shell.

The staff reviewed the FEA stress results summarized above. The staff determined that the FEA had been implemented appropriately by following common FEA practices, including application of boundary condition assumptions and element selections, to demonstrate the structural capability of the package body outer shell in resisting pin puncture. For the neutron shield performance, Section 5.5.1.2 of the application states that the pin puncture on the neutron shield compartment may result in localized loss of neutron shielding material. This localized loss is an acceptable assumption because, for the given neutron shield design features with multiple constraints as structural support, any large tearing or dislodging of the neutron shield assemblies, {proprietary information removed} , would be unlikely to occur. On the basis of the evaluation above, the staff concludes that the structural performance of the package is adequate in meeting the requirements of 10 CFR 71.73(c)(3) for the puncture test.

2.8.4 Thermal

Tables 2.7.4-1 and 2.7.4-2 of the application summarize the maximum canister and package cavity pressures, respectively, for the package subject to the 1,475°F thermal fire test for 30 minutes. The calculated maximum canister internal pressure of 133.4 psig and package cavity internal pressure of 130.7 psig are enveloped by the common design-basis pressures of 300 psig for the structural evaluations.

The applicant reported minimum stress factors of safety at all canister and package section cut locations in Tables 2.7.4-3 through 2.7.4-4 with the minimum factors of safety of 6.11 and 5.44 for the primary membrane and (P_m) and primary membrane-plus-bending ($P_m + P_b$) stresses for the package containment boundary. Additionally, for the bounding internal pressure of 300 psig, the applicant calculated a reduction of closure lid bolt force of 14,668 lbs and noted that the preload reduction is bounded by a previously analyzed condition in which the decrease in bolt load due to the absence of the inertial load had been considered.

Section 2.7.4.3 of the application notes that stresses associated with differential thermal expansion are categorized as secondary and need not be evaluated for the fire accident, in accordance with the ASME B&PV Code stress criteria, because thermal induced stresses are self-limiting.

The staff reviewed the applicant's evaluation results and concludes that FEA approaches were implemented by following common practice. This demonstrates that the package meets the requirements of 10 CFR 71.73(c)(4) for the thermal test.

2.8.5 Immersion - Fissile Material

Section 2.7.5 of the application noted that the package is subject to a head of water of 0.9 m, which is equivalent to an external pressure of 1.3 psig. This pressure is negligible compared to an external pressure of 290 psig, for which the package is evaluated in Section 2.7.7 of the application and determined to be acceptable. The applicant assumed water inleakage for the criticality evaluation. The staff reviewed the applicant's evaluation and concludes that package meets the requirements of 10 CFR 71.73(c)(5) for the immersion test for fissile material.

2.8.6 Immersion - All Packages

The immersion test requirements are met because the effect of an external pressure of 21.7 psig caused by immersion under 50 feet of water is of negligible consequence, compared to that associated with an external pressure of 290 psig as discussed in SER Section 2.9, below. The staff reviewed the applicant's evaluation and concludes that the package meets the requirements of 10 CFR 71.73(c)(6) for the immersion test for all packages.

2.9 Special Requirements for Type B Packages Containing More Than 10^5A_2

Section 2.7.7 of the application evaluated the package to withstand an external pressure of 290 psi by performing an FEA of the package body. The finite element model used for the package drop analyses assuming no presence of internal pressure. As Summarized in Table 2.7.7-1 and Table 2.7.7-2, the critical factors of safety are large for primary membrane stress (P_m), and the primary membrane-plus-bending stress ($P_m + P_b$). The results demonstrate that the containment boundary is capable of withstanding the pressure of the deep immersion test without collapse, buckling, or inleakage of water, meeting the requirements 10 CFR 71.61.

2.10 Internal Pressure Test

Section 8.1.3.3 of the application states that the package containment boundary shall be hydrostatically pressure tested to 125% of the design pressure, which is 120 psig, in accordance with the ASME B&PV Code, Section III, Paragraph NB-6220 provision. Since the package containment is tested to a 150 psig internal pressure (1.25 x 120 = 150), which is greater than the 150% MNOP of 23.0 psig, the hydrostatic pressure test meets the requirements of 10 CFR 71.85(b) for the internal pressure test to verify structural integrity of the containment boundary.

2.11 Fuel Rod Performance

Section 2.11 of the application evaluates the fuel rod structural performance, per 10 CFR 71.55(e), to establish the fuel rod damage conditions after the tests for hypothetical accident conditions, consistent with those considered for demonstrating other safety functions of the MAGNATRAN package. The evaluation addresses primarily the fuel rod buckling capability under the package 30-ft end drop accident and only PWR fuel rod is evaluated for the 30-ft side drop. The staff consider this acceptable because, as noted in Table 2.3 of this SER, the {proprietary information removed} , which is below the 63g determined in a generic study (UCID-21246)[12] to be the minimum structural capacity without causing yielding in the PWR and BWR spent fuel rods. The fuel rod accident drop structural performance is reviewed as follows.

2.11.1 PWR Fuel Rod End Drop

Section 2.11.1 of the application uses a LS-DYNA FEA approach to evaluate the PWR fuel rod subject to a 30-ft end drop accident. A Westinghouse (WE) 17x17 fuel assembly is considered representative for demonstrating fuel rod end drop performance. Figures 2.11.1-1 through 2.11.1-3 depict the FEA models comprised primarily of shell and solid elements for a plane row of 17 fuel rods within the confines of the fuel tube and end fitting. In addition to removing the cladding oxidization of 120 microns and considering the material properties of the irradiated cladding at 572°F, other key model attributes include applying: (1) a terminal velocity of 527 in/sec for all nodes in conjunction, (2) the end fitting vertical motion for 0.08 sec corresponding to the package deceleration response time series with a maximum deceleration of 36 g, (3) a bow of 0.01 inch to initiate lateral deformation calculation for the fuel rod subject to axial loading, and (4) an internal rod pressure of 2,000 psi for determining the maximum fuel rod bending stresses.

The applicant evaluated two fuel configurations, the intact fuel and a case with damaged spacer grids for which the grid at the bottom row is removed from the intact model. By including a fuel cladding axial stress resulting from a rod internal pressure to the calculated maximum von Mises stresses associated with fuel rod bending, the applicant reported the maximum stresses of 42,079 psi and 39,326 psi, which correspond to the minimum factors of safety of 2.2 and 2.3, for the intact fuel and the damaged cases, respectively. The greater than unity factors of safety demonstrate the fuel rod elastic performance, which ensure that fuel rods will return to their original configuration after the 30-ft end drop accident.

2.11.2 BWR Fuel Rod End Drop

Section 2.11.2.1 of the application evaluates the buckling capability of BWR fuel rods by comparing the slenderness ratio, L/r, of a governing BWR fuel rod to that of the WE 17x17 fuel rod, which has been shown to behave elastically. For individual fuel rods, L is taken as the largest unsupported length between spacer grids and r the radius of gyration defined as the square root of I/A, where I is the cladding area moment of inertia and A the cladding cross-sectional area.

With the cladding oxidation properly accounted for in the evaluation of high burnup fuel, the applicant calculated the largest slenderness ratios of 279 and 185 for the PWR and BWR fuel rods, respectively. Since the BWR fuel is seen to have a relatively smaller slenderness ratio than the PWR fuel, it is less vulnerable to fuel rod buckling. On this basis, the staff finds the applicant's evaluation acceptable, which determined that the PWR fuel rod evaluation is bounding and there is no further evaluation need for BWR fuel.

Section 2.11.2.2 of the application evaluated the final axial position of the fuel assembly relative to its fuel pellets as a result from the 30-ft package top-end drop accident. The LS-DYNA FEA model used for the analysis consists primarily of four parts: (1) the fuel pellets as a lumped mass, (2) the end fitting assembly with a deformable bail handle, (3) the plenum spring connecting the mass and the end fitting, and (4) the canister lid as a rigid boundary. Other relevant modeling attributes include a terminal velocity of 527 in/sec and a bounding package body deceleration time history with a deceleration peak of 36 g. The FEA results established that, after the 30-ft package top-end drop, the final height of the bail handle was crushed to 0.96 inches and the length of the plenum spring was compressed from the original of 10.59 inches to 3.76 inches. The resulting fuel pellet locations are considered in other package safety evaluations in the SER, as appropriate.

2.11.3 Reactor Control Cluster Assemblies Spacer Drop

The applicant introduced a spacer between a 15x15 PWR fuel assembly and the canister lid to limit the axial movement of the RCCA for the criticality control for the package subject to the top end drop accident. Figures 2.11.3-1 and 2.11.3-3 provide schematics of the two essentially identical design configurations with difference only in the length of the top support legs to allow a total deformation not to exceed 1.38 inches between the RCCA holder and the spacer after the top end drop.

The LS-DYNA FEA of the RCCA spacer end drop evaluation assumed an initial gap of 4.4 in, which resulted in the permanent deformations of 0.01 in. and about 1.26 in. for the spacer and the RCCA holder, respectively. These permanent deformations are considered in other package safety evaluations in the SER, as appropriate.

2.11.4 PWR Fuel Rod Side Drop

Section 2.11.4 of the application uses an ANSYS® FEA approach to evaluate the PWR fuel rod subject to a side-drop inertia force of 60 g. For the same WE 17x17 fuel evaluated for the package end drop, the applicant considered the rod pitch and cladding oxidization of 120 microns to calculate a maximum lateral displacement of 2.33 in. The maximum lateral displacement is associated with the outer top most rod of a plane row of 17 rods, which deforms laterally before coming in contact with the next fuel rod. Figure 2.11.4-1 depicts the model schematic with the CONTAC52 elements interface to simulate an open gap of 2.33 in., which would close when the calculated lateral maximum displacement at any location of the fuel rod reaches the gap limit.

Although the maximum lateral displacement may vary for different fuel, the applicant assumed the same gap size in analyzing the three fuel types, Combustion Engineering (CE) 14 x14, WE15x15, and WE17x17 to result in the cladding stress factors of safety of 1.88, 1.45, and

1.50, respectively. The greater than unity factors of safety demonstrate the fuel rod elastic performance, which ensures that fuel rods will return to their original configuration after the 30-ft package side drop accident.

2.12 Review Findings

F2.1 The staff has reviewed the package structural design description and concludes that the description of the packaging and contents in the application meet the requirements of 10 CFR 71.31.

F2.2 The staff has reviewed the lifting and tie-down systems for the package and concludes that they meet 10 CFR 71.45 standards.

F2.3 The staff has reviewed the packaging structural evaluation and concludes that the application meets the requirements of 10 CFR 71.35.

F2.4 The application describes the materials that are part of the package and the suitability of those materials for their intended functions in sufficient detail to facilitate evaluation of their effectiveness. 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.

F2.5 The design of the MAGNATRAN package and the selection of materials adequately protect the spent fuel cladding against failure.

F2.6 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.

F2.7 The MAGNATRAN package employs only noncombustible materials that will help maintain safety control functions. The applicant has met the requirements of 10 CFR 71.43(f) and 71.51(a); the applicant demonstrated effective materials performance of packaging components under normal conditions of transport and hypothetical accident conditions.

F2.8 The MAGNATRAN package employs materials that are compatible with canister loading operations and facilities. These materials are not expected to degrade over time, or react with one another, during conditions of transportation. The applicant has met the requirements of 10 CFR 71.43(d). The applicant 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 prior to each shipment to verify its condition.

F2.9 The applicant has met the requirements of 10 CFR 71.43(f), 71.51(a), and 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.

F2.10 The staff has reviewed the packaging structural performance under the normal conditions of transport and concludes that there will be no substantial reduction in the effectiveness of the packaging.

F2.11 The staff has reviewed the packaging structural performance under the hypothetical accident conditions and concludes the packaging has adequate structural integrity to satisfy the subcriticality, containment, shielding, and temperature requirements of 10 CFR Part 71.

F2.12 The staff has reviewed the containment structure and concludes that it will meet the 10 CFR 71.61 requirements for irradiated nuclear fuel shipments.

F2.13 The staff has reviewed the containment structure and concludes that it will meet the 10 CFR 71.85(b) requirements for pressure test without yielding.

The staff reviewed the statements and representations in the application by considering the regulations, Regulatory Guides, applicable codes and standards, and acceptable engineering practices. On the basis of the review above, the staff concludes that the package structural performance has adequately been demonstrated to meet the requirements in 10 CFR Part 71.

2.13 References

1. Regulatory Guide 7.8, *Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material*, U.S. Nuclear Regulatory Commission, March 1989.

2. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, New York, NY.

3. NUREG-1617, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, U.S. Nuclear Regulatory Commission, March 2000.

4. NUREG/CR-6322, “Buckling Analysis of Spent Fuel Basket, U.S. Nuclear Regulatory Commission, May 1995.

5. U.S. Nuclear Regulatory Commission, Division of Spent Fuel Storage and Transportation Interim Staff Guidance – 11, Revision 3 – *Cladding Considerations for the Transportation and Storage of Spent Fuel*, U.S. NRC, November 2003.

6. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36, U.S. Nuclear Regulatory Commission, July 1980.

7. American National Standards Institute ANSI N14.6, *Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (45000 kg) or More for Nuclear Materials*, New York, NY, 1993.”

8. NUREG/CR-3019, Welding Criteria for Use in the Fabrication of Radioactive Material Shipping Containers, U.S. Nuclear Regulatory Commission, March 1984.

9. NUREG/CR-3854, “Fabrication Criteria for Shipping Containers,” U.S. Nuclear Regulatory Commission, March 1985.

10. Regulatory Guide 7.6, Revision 1, *Design Criteria for the Structural Analysis of Shipping Package Containment Vessels*, U.S. Nuclear Regulatory Commission, 1978.

11. NUREG/CR-6007, Stress Analysis of Closure Bolts for Shipping Casks, U.S. Nuclear Regulatory Commission, April 1992.

12. UCID-21246, Dynamic Impact Effects on Spent Fuel Assemblies, Lawrence Livermore National Laboratory, Lawrence Livermore National Laboratory, Livermore, CA, October 1987.

3.0 THERMAL REVIEW

3.1 Review Scope and Objective

The objective of this review is to verify that the package design meets the thermal requirements under normal conditions of transport and hypothetical accident conditions, per 10 CFR Part 71. The staff reviewed the thermal aspects of the MAGNATRAN transport cask (i.e., package) using NUREG-1617[1] "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" and evaluated the package design, including performing independent calculations, for normal conditions of transport and the thermal-related hypothetical accident condition.

3.2 Description of the Thermal Design

3.2.1 Packaging Design Features

The MAGNATRAN transport package, which is shipped horizontally, is designed to transport up to:

- 37 undamaged uranium PWR spent fuel assemblies in the 37 PWR basket assembly;
- 37 fuel assemblies in the 37 PWR damaged fuel basket, which may contain up to 4 damaged or high burnup fuel assemblies in four damaged fuel cans placed in the designated location in the fuel basket;
- 87 undamaged uranium BWR spent fuel assemblies in the 87 BWR basket assembly;
- 55,000 pounds of GTCC waste, placed within a GTCC waste basket liner.

The packaging is composed of the following major components:

- A transport package body includes an inner shell, outer structural shell, lead gamma shield, solid neutron shield, and containment boundary. Drawing No. 71160-500, Rev. 5P indicated there are six PWR configurations and four BWR configurations, depending on length of canister, presence of a canister spacer, type of lid, and number of BWR fuel assemblies. SAR Section 3.4.1.1.1 indicated that helium is backfilled inside the transport package and TSC.
- Impact limiters, consisting of large sections of relatively low thermal conductivity (i.e., thermal insulation) balsa wood and redwood, encased in stainless steel shells, and attached to the ends of the transport package body.
- Rotation trunnions and lifting trunnions.
- Passive fins (SAR Figure 1.2-1) on the MAGNATRAN transport package exterior. Details of the fins were provided in Drawing 71160-502, Rev. 6P.

SAR Chapter 1 indicated the content container as the MAGNASTOR TSC and the GTCC TSC, which are stainless steel canisters that provide confinement of radioactive material; the MAGNASTOR is certified for storage under 10 CFR Part 72. A PWR basket, BWR basket, or GTCC basket liner is located inside the TSC and provides structural support for the PWR, BWR, or GTCC content. According to SAR Section 3.1, PWR fuel assemblies have a design basis heat load of 23 kW and require a neutron absorber with Type 2 thermal conductivity (see SAR Table 3.2-12); a neutron absorber with a Type 1 thermal conductivity is limited to 22 kW. BWR fuel assemblies have a design basis heat load of 22 kW and require a neutron absorber with Type 1 thermal conductivity.

In addition, SAR Chapter 7 and Chapter 8 described thermally-related evaluations and operations of TSCs that must be performed, including the following:

- In SAR Section 7.1.1 NAC states that the transport vehicle and package (including personnel barrier) will be cleaned prior to transport. According to NAC in its response to Request for Supplemental Information No. 3-1 (ADAMS Accession No. ML13010A094) and Request for Additional Information No. 3-16 response (ADAMS Accession No. ML15296A085), cleaning will help to prevent dirt/debris buildup on the exterior that may affect the thermal performance of the package.

- In SAR Section 7.1.2 NAC states that there is a 41-hour time limit between lifting a TSC loaded with fuel out of the MAGNASTOR concrete cask and placement of the MAGNATRAN, in a horizontal orientation, on the transport vehicle. In addition, the applicant stated there was a 65-hour time limit from completion of TSC closure operations, including helium backfill and termination of external cooling of the TSC, through completion of the preparation of the MAGNATRAN for transport and placement in a horizontal orientation on the transport vehicle. These time limits would be implemented for all spent fuel decay heat loads.

- In SAR Section 7.2.2 NAC states that, once the MAGNATRAN package lid is removed, a TSC containing PWR or BWR fuel assemblies must be placed in a safe condition (i.e., in a MAGNASTOR transfer cask or equivalent transfer device). Specifically, the maximum time to complete the operational sequence from SAR Section 7.2.2 step 9 through step 18 shall be less than 6 hours, to ensure fuel clad temperatures do not exceed 400 °C.

- In SAR Section 7 (page 7-1) NAC states that the loading, preparation, and transfer procedures for TSCs loaded with spent fuel were provided as part of the MAGNASTOR FSAR. In addition, in its Request for Supplemental Information response 3-3 (ADAMS Accession No. ML13010A094) NAC states that the above time limits were analyzed as part of the MAGNASTOR transfer cask, which has been certified for storage operations under 10 CFR Part 72.

3.2.2 Codes and Standards

In SAR Section 2.1.4 NAC lists the codes and standards used for the packaging and SAR Section 2.2.1.2 referenced the ASME B&PV Code for structural material properties. In addition, details of the codes and standards applied to the MAGNATRAN transport package are listed in SAR Section 2, licensing drawings, and SAR Sections 8.1, 8.1.2, 8.1.3, 8.1.4, including the ASME B&PV Code and ASTM standards. The ASME B&PV Code is used for fabrication and inspection as well as welding per ASME B&PV Code, Section III, Subsection NB for the MAGNATRAN containment boundary. According to SAR Section 8.1.3.3, a hydrostatic pressure test (to 125% of the 120 psig design pressure) is performed in accordance with the ASME B&PV Code, Section III, Subsection NB.

SAR Section 8.1.4.2 and Section 8.1.5.2 discuss helium leakage rate testing per ANSI N14.5[2] that ensures the containment boundary would remain backfilled with helium, which is important for thermal performance. Additional discussion of codes and standards are described in the structural portion of this SER, above.

3.2.3 Content Heat Load Specification

According to SAR Section 1.3.2, the total decay heat of the MAGNATRAN transport package is limited to 23 kW for PWR fuel, 22 kW for BWR fuel, and 1.7 kW for GTCC waste. Individual PWR and BWR fuel assemblies are limited to 622 Watts (Type 2 neutron absorber) and 253 Watts, respectively. Likewise, the thermal analyses in SAR Section 3.4.1.1.1 and Section 3.4.1.2.1 indicated that the decay heat associated with PWR (with Type 2 neutron absorber) and BWR fuel assemblies (with Type 1 neutron absorber) would be 23 kW and 22 kW, respectively; thus, the thermal models are consistent with the content decay heats described in the General Information and Thermal Chapters, 1.0 and 3.0, respectively of the SAR. SAR Section 3.4.2 and SAR Table 3.4-1 indicated that PWR content was bounding.

3.2.4 Summary Tables of Temperatures

Package component temperature summary tables are presented in SAR Table 3.4-1 and Table 3.4-2 for normal conditions of transport; SAR Sections 2.6.1.1 and 2.6.2.1 also referenced the temperatures presented in Table 3.4-1 and Table 3.4-2. It is noted that the temperatures provided in SAR Table 3.4-1 are based on two ANSYS® models at normal conditions of transport; specifically, a full-length model for PWR content and a lower half-model for BWR content, which, according to SAR Table 3.4-1, assumed an O-ring temperature equivalent to the bounding PWR content. The components presented in the tables included cladding, basket, impact limiter, lead gamma shield, radial neutron shield, O-rings, average gas, and items associated with the package's containment boundary and structural components. The temperature limits for each component were listed in the tables. The tables indicated that all components were below their respective temperature limits for normal conditions of transport.

For the hypothetical accident conditions fire test, summary tables of temperatures are provided in SAR Table 3.5-1 for PWR fuel and SAR Table 3.5-2 for BWR fuel. In addition, SAR Figures 3.5-4, 3.5-5, 3.5-6, 3.5-7, 3.5-8, and 3.5-9 show transient temperature plots for lead gamma shield, exterior package surface, inner and outer package shells, lid inner O-ring, and lid coverplate inner O-ring, respectively, during the hypothetical accident condition; these plots are based on a non-physics, analytical methodology, as discussed in SER Section 3.4.1.

The component temperatures were below the listed temperature limits during the fire hypothetical accident condition, except the EPDM O-rings, neutron shield, aluminum fins, and aluminum personnel barrier; a discussion of this is provided in the paragraphs below. In addition, further discussion of the applicant's hypothetical accident condition thermal analyses and staff's independent analyses are discussed in SER Sections 3.4.1 and 3.4.2, below.

SAR Section 3.3.2 indicates that the EPDM O-ring safe operating temperature range is -40°F to 250 °F. However, in SAR Table 3.5-1 NAC notes that the package lid and coverplate metallic O-ring temperatures were 289 °F during the hypothetical accident condition fire transient. Based on the small temperature difference between the metallic O-ring and EPDM O-ring temperatures at normal conditions of transport (per SAR Table 3.4-1), this would indicate that the EPDM O-ring would reach, approximately, a 289 °F temperature because of the close proximity of the metallic and EPDM O-rings. Although 289 °F is above the 250 °F EPDM O-ring safe operating temperature, in SAR Section 3.3.2 NAC stated that the EPDM O-ring function is to provide a leak test volume; it is not used as part of the containment boundary, and therefore, its integrity during and after the hypothetical accident conditions fire is not required.

In SAR Section 3.3.2 NAC mentioned that, for thermal analysis purposes, the radial neutron shield is considered lost after the 30-minute hypothetical accident condition fire test because temperatures would exceed its maximum safe operating temperature. As NAC discussed in its response to Request for Additional Information No. 3-9 (ADAMS Accession No. ML14356A377), this is a conservative assumption because experimental data provided in "GA-A19897 Thermal Testing of Solid Neutron Shielding Material, by R.H. Boonstra, 1990" and "GA-A20770 Thermal Testing of Solid Neutron Shielding Materials" by R.H. Boonstra, 1992, indicated that the majority of the NS-4-FR material would remain intact after a 30-minute fire.

The applicant noted in its response to Request for Additional Information No. 3-6 (ML14356A377), that, although the aluminum fins would exceed their melting temperature during the hypothetical accident conditions fire, the aluminum fin is not a structural component and there would be no structural effects if the fin melted. Finally, the applicant noted in its response to Request for Additional Information No. 3-2 (ML15296A084) that the potential loss of the aluminum personnel barrier during a fire accident would have a negligible effect on the package. The applicant noted that the aluminum does not reach ignition temperature and that the heat input associated with the potential oxidation of aluminum was 122 Btu, which is a small quantity relative to the energy supplied by the 30-minute hypothetical accident conditions fire.

In SAR Sections 3.3.2 and 3.4.3 NAC stated that the minimum temperature of the package would be -40°F and indicated a low safe operating temperature of -40°F for package components, including the package lid and port coverplate metallic O-rings, lead gamma shield, radial NS-4-FR neutron shield, EPDM O-rings, adhesive used for the impact limiters, and silicone foam.

3.2.5 Summary Tables of Pressures in the Containment System

SAR Sections 3.4.4 and 3.5.4 present the summary of a method to determine the maximum internal pressure within the MAGNATRAN package during normal conditions of transport and hypothetical accident conditions, respectively. According to SAR Section 3.1, the TSC and transport package are both backfilled with helium, which has a relatively high thermal conductivity and aids in heat transfer. SAR Table 3.4-3 provided the maximum internal canister and transport package pressures under normal conditions of transport; SAR Section 3.4.4.1 and Calculation Package No. 71160-3022, Rev. 1 indicated that the pressure within the package body at normal conditions of transport was calculated to be 23 psig based on the helium backfill pressure and adjusting it to operational temperature.

The SAR indicated that the pressure within the cask could increase further if one accounted for the potential failure of the TSC confinement boundary such that, for a 3% fuel failure, the resulting 30% fission gas, 100% rod backfill, BPRA gas, integral fuel burnable absorber rod gas, and shim rod gas would pressurize the package. According to SAR Section 3.4.4.1, summing the previously mentioned gas molar quantities and applying the ideal gas law at a normal condition gas temperature

{proprietary information removed}

, which is below the package allowable pressure of 300 psig. Staff notes that the pressures at normal conditions of transport are reasonably consistent (within 2 psig) among the different SAR chapters.

3.3 Material Properties and Component Specifications

3.3.1 Material Properties

SAR Section 3.2 provided the thermal properties of the materials (e.g., thermal conductivity, emissivity) used in the MAGNATRAN transport package, including thermal properties for solid neutron shield, stainless steel, carbon steel, lead, cladding, helium, air, etc. SAR Section 3.2 also included viscosity, thermal conductivity, specific heat, and density values for air and helium (from which Prandtl numbers can be determined). It is noted that thermal conductivity is used for steady-state thermal analyses whereas thermal conductivity, density, and specific heat are used for transient thermal analyses. These properties were functions of temperature, except for the specific heat value for helium, which was held constant at 1.24 Btu/lbm °F. Thermo-mechanical properties, such as modulus of elasticity, Poisson's ratio, and coefficient of thermal expansion, were provided in SAR Section 2.2.1.2. The sources of the property data were referenced in Calculation Package No. 71160-2101, Rev. 7.

It is noted that a given material may have different emissivity and solar absorptivity values (recognizing that insolation is at a low wavelength of the emission spectrum), and therefore, thermal models should reflect appropriate emissivity and absorptivity values at the expected and maintained surface conditions (e.g., dull, polished). This practice was not apparent in some of the calculations presented in the application. For example, in SAR Table 3.2-2 NAC indicated a stainless steel emissivity of 0.36. Although this may be a reasonable value for solar absorptivity of polished stainless steel, there was no indication of the package's outer stainless steel material being polished or that it would remain in the polished condition during operation.

Likewise, the thermal model described in Appendix R of Calculation Package No. 71160-3014, Rev. 7 relied on an emissivity of 0.4 for aluminum, rather than the SAR value of 0.22. It is noted that Reference 28 (reported in Calculation Package No. 71160-2101, Rev. 6) indicated that an emissivity of 0.4 for aluminum is associated with a sand-blasted surface finish; however, there was no surface finish provided in the licensing drawings and no discussion in Appendix R of Calculation Package No.71160-3014, Rev. 7 to indicate that this surface condition would exist on the package surface. Since the results in Appendix R are not used to justify the SAR model, this emissivity assumption is not relevant to the application.

3.3.2 Technical Specifications of Components

SAR Section 4.5 included manufacturer's performance sheets for the O-Flex^{TM} metal O-rings and the Parker Hannifin Corporation EPDM O-rings. In addition, Appendix G of Calculation Package No. 71160-5508, Rev. 0 provided formulation specifications for NS-4-FR and boron carbide. Thermal testing information of the solid neutron shielding material was provided in document GA-A20770 and document GA-A19897 as part of the SAR Revision 15A submittal (October 2015 (ADAMS Accession No. ML15296A084)).

In SAR Section 8.1.5.3.8 NAC described the thermal conductivity testing for the metal matrix borated aluminum neutron absorber and included a table of the radial and axial thermal conductivities for the Type 1 and Type 2 neutron absorbers. The results of thermal conductivity testing of the Type 2 neutron absorber and property data for Type 1 neutron absorber were provided in Appendix E and Appendix A, respectively, in Calculation Package No. 71160-2101, Rev 8. As stated in Appendix J of Calculation Package No. 71160-3014, Rev. 9, some of the thermal models conservatively included a 10% penalty in the thermal conductivity of the neutron absorber, whereby the thermal conductivity was decreased by 10% from the measured values.

In addition, the results of testing a sample of fabricated and installed neutron absorbers for the gap between the basket tube and neutron absorber and the gap between the neutron absorber and stainless steel retainer were presented in NAC Calculation Package No. 71160-R-01, Rev.0 (MAGNASTOR Tube to Poison Gap Assessment at Hitachi-Zosen). A summary of that document and its relation to the SAR model is discussed in SER Section 3.4.1.

In SAR Section 3.3.2 NAC indicated a -40 °F lower safe operating temperature limit for the package lid and port coverplate metallic O-rings, lead gamma shield, radial NS-4-FR neutron shield, EPDM O-rings, adhesive used for the impact limiters, and silicone foam. Thermal properties of the FiberFrax and Silicone Foam HT-800 were provided in Calculation Package No. 71160-2101, Rev. 8.

3.3.3 Thermal Design Limits of Package Materials and Components

SAR Section 3.3.2 and SAR Table 3.4-1 provided the temperature limits at normal conditions for the components that could affect containment, shielding, and criticality functions. These components included metallic O-rings, neutron shielding, lead gamma shielding, and fuel cladding. As mentioned in SER Section 3.2.4 above, the SAR tables indicated these components were within their allowable values.

SAR Table 3.5-1 and SAR Table 3.5-2 indicated that metallic O-rings and lead gamma shielding temperatures after the hypothetical accident conditions fire test were 289 °F and 536 °F, respectively. These temperatures are below their allowable values of 500 °F and 600 °F, respectively, as reported in SAR Table 3.5-1.

The independent thermal analysis (described in SER Section 3.4.2) showed that the metallic O-ring and lead gamma shield temperatures during the hypothetical accident conditions fire test were 273 °F and 316 °F, respectively. In addition, in SAR Table 3.5-1 and Table 3.5-2, NAC indicated that the maximum fuel temperature for the hypothetical accident condition fire test was predicted to be 902 °F which is less than the 1058 °F allowable value presented in SAR Table 3.5-1 and Table 3.5-2. Further discussion of the applicant's hypothetical accident condition thermal analyses, as well as staff's independent analyses, are provided in SER Sections 3.4.1 and 3.4.2. In addition, SER Section 3.2.4 has discussed the performance of the radial neutron shield, aluminum personnel barrier, and EPDM O-ring during the hypothetical accident conditions fire.

3.4 Thermal Evaluation Methods

3.4.1 Evaluation by Analyses

The applicant used the ANSYS® FEA code to model the MAGNATRAN transportation package and to perform thermal analyses for normal conditions of transport and hypothetical accident conditions. In addition, the applicant generated a short axial section FLUENT computational fluid dynamics model of a fin arrangement to calculate the heat transfer coefficients applied to the outer package (180° axisymmetric model). As noted in SAR Section 1.4.2, these codes and their use would fall under NAC's quality assurance program, which is applied to the design and analysis of the package and the packaging's fabrication, assembly, testing, maintenance, and repair.

According to SAR Section 3.4.1, separate FEA models were generated by the applicant to model the transportation package at normal conditions of transport; one modeled PWR fuel and the other modeled BWR fuel. The applicant's response to Request for Additional Information

No. 3-3 (ADAMS Accession No. ML14356A385) provided a discussion to show that the PWR and BWR thermal models bound the six PWR and four BWR configurations (mentioned in SER Section 3.2.1), respectively. The PWR model was three-dimensional, full-length, and half-symmetry (180°) whereas the BWR model was three-dimensional, bottom half-length, and a full 360° extent. According to SAR Section 3.4.2, the PWR content was the bounding fuel. Therefore, the applicant assumed the temperature of the O-rings (which were not included in the BWR model) for the BWR content was the same as the temperature of the O-rings for the PWR content.

The models consisted of the fuel and fuel assembly basket, TSC, shell and structural components of the MAGNATRAN transport cask, and helium within the TSC and cask cavity; the fuel assembly was modeled using the effective thermal conductivity methodology as described in SAR Sections 3.4.1.1.2 and 3.4.1.2.2. The MAGNATRAN package model included the inner shell, lead, outer shell, neutron shield, copper and aluminum cooling fins, and neutron shield shell. According to SAR Sections 3.4.1.1.3 and 3.4.1.2.3, the neutron shield assembly models included NS-4-FR material and the orthotropic effective material properties of the silicone foam, thermal insulation, and stainless steel. SAR Figure 3.4-3 and SAR Figure 3.4-7 provided the design basis fuel assembly axial power distribution for PWR and BWR fuel, respectively.

According to SAR Sections 3.1, 3.2.2.2, 3.2.3, and 3.4.1, the model's exterior thermal boundary conditions at normal conditions for determination of maximum temperatures included a 100 °F ambient temperature, natural convection heat transfer, radiation heat transfer from the package to the ambient, and insolence on the curved package surface and outward facing fins. In SAR Sections 3.1, 3.4.1.1.1 and 3.4.1.2.1 NAC indicated that insolation boundary conditions were derived from the 12-hour 10 CFR Part 71 values of 2950 Btu/ft^2 for horizontal flat surfaces and 1475 Btu/ft^2 for curved surfaces. The insolation actually applied was averaged over 24 hours, considering the MAGNATRAN package has large thermal mass, and reduced by the fraction defined by the reported material's emissivity, recognizing that the models assumed absorptivity was equal to emissivity. The emissivity values applied to the exterior and interior components were defined in the tables presented in SAR Section 3.2. It was noted in SAR Section 3.4.1 that the large impact limiters, which are to be fabricated from relatively low thermal conductivity wood, were modeled as adiabatic.

SAR Section 3.4.2 indicated that the hot and cold steady-state analyses were based on the maximum allowable decay heat, circumferentially-varying convection heat transfer coefficient on the package surface, including the inward and outward facing fins, and assumed ambient conditions of 100 °F (with insolation) and -40 °F (without insolation); further discussion about the insolation boundary conditions is provided in later paragraphs. The eight circumferentially-applied heat transfer coefficients along the package's axial length were provided in Calculation Package No. 71160-3045, Rev. 2. As described in the main part of the calculation package, these heat transfer coefficients were determined from a three-dimensional (180° extent) periodic axial section FLUENT computational fluid dynamics model based on the heat flux from BWR fuel with a 1.22 peaking factor. According to the April 2018 submittal (ADAMS Accession No. ML18102A852), the coefficients were determined based on the average conditions of the package surface, both along the axial and circumferential directions in order to represent the average temperature over the surface. In addition, the April 2018 submittal indicated that the computational fluid dynamics-based heat transfer coefficients were similar to those computed from a bare tube empirical relation, the ratio of computational fluid dymanics-based coefficient-to-empirical coefficient varying between 0.6 and 1.35. Recognizing the presence of recirculation flow patterns adjacent to the fin structure and the uncertainty as to an appropriate turbulence modeling scheme, Calculation Package No. 71160-3045, Rev. 2, Appendix F indicated that a sensitivity analysis showed similar results when four different turbulence models were used.

As noted above, although velocity vector plots show recirculation zones within the flow area between inward and outward facing fins, Appendix H presented results which indicate improved heat transfer from the three-dimensional model with fins when compared to the results of a two-dimensional FLUENT model without fins. According to the response to RAI 3-1 (ADAMS Accession No. ML14356A385) and Calculation Package No. 71160-3045, Appendix H, the FLUENT analysis showed a fin effectiveness of 3.1, which would tend indicate that the fin material and geometry were designed to result in a net improvement in heat transfer.

According to Drawing No. 71160-502, the fin materials are comprised of copper and aluminum which have relatively high thermal conductivities and, when coupled with the relatively low heat transfer coefficient of the low velocity buoyant air stream, support there being a fin effectiveness equal to or greater than one. Therefore, a fin effectiveness value of at least one and considering that the FLUENT-based heat transfer coefficients are similar to the bare tube correlation coefficients, indicate that the fins should not impede heat transfer.

As noted above, SAR Section 3.2-3 also included an empirical equation for the convection heat transfer coefficient. According to the October 2017 response submittal (ADAMS Accession No. ML18311A259), the ANSYS model peak cladding temperature (PCT) decreased by 14 °F when using the FLUENT-based heat transfer coefficients rather than those from the empirical equation; it is noted that an independent COBRA-SFS model showed approximately 3 °F change in PCT when comparing empirical-based heat transfer coefficients and the FLUENT-based heat transfer coefficients. Although there is an uncertainty associated with this difference in sensitivity, both models show that the reported PCT values would be below the 752 °F allowable temperature limit.

Four separate two-dimensional models were generated by the applicant to model the effective conductivities of the PWR fuel assembly, BWR fuel assembly, PWR neutron absorber, and BWR neutron absorber. The effective conductivities were then applied to the three-dimensional ANSYS® package models with PWR and BWR fuel contents.

Specifically, in SAR Sections 3.4.1 and 3.4.1.1.2, NAC indicated that the fuel assemblies were modeled as a homogeneous material with effective temperature-dependent thermal conductivity. The effective thermal conductivity was determined from a separate two-dimensional FEA model of the UO_2 fuel pellets, zircaloy cladding, and gas between the fuel and cladding. The neutron absorber between the fuel assembly and fuel was also modeled as a homogeneous material with effective thermal conductivity properties in the parallel and axial directions.

As mentioned above, effective thermal conductivity results from the two-dimensional neutron absorber model were used as inputs to the three-dimensional ANSYS® models. The October 2017 response submittal indicated that the three-dimensional ANSYS® model PCT decreased by 16 °F when the Type 2 neutron absorber thermal conductivity was applied rather than the Type 1 neutron absorber thermal conductivity. An independent COBRA-SFS model showed an approximately 3 °F change in PCT when comparing Type 1 and Type 2 thermal conductivities. Although there is an uncertainty associated with this difference in sensitivity, both models show that the reported PCT values would be below the 752 °F allowable temperature limit.

The applicant provided results in Appendix P of Calculation Package No. 71160-3014, Rev. 7 that recorded the change in predicted PCT when the model had 556,000 elements, 1,216,000 elements, and 2,644,000 elements. Table P-2 indicated the increased element number ratios among the three models for the transverse X, transverse Y, and axial directions as 1.3, 1.3, and 2.0, respectively, for the content. Likewise, the increased element number ratios among the three models for the radial, circumferential, and axial directions were 1.3, 1.3, and 2.0, respectively, for the cask body. Correspondingly, PCT decreased from 735 °F to 727 °F to 725 °F; this indicates a smaller rate of change in PCT with increased element number. Figure P-5 also showed the decreasing trend of the PCT with increasing element number of the refined grids.

In addition to the above-mentioned grid refinement analysis, the applicant provided Appendix J of Calculation Package No. 71160-3014, Rev. 7. This analysis showed the effect of grid refinement in the neutron absorber region. When a 1/8-sectional model with simplified assumptions (e.g., one-element thick, limited to the content/basket region of the canister, imposed constant temperature at the canister shell) was refined from 3,899 elements to 3,071,834 elements, it was found that PCT increased by 4°F.

Likewise, another Appendix J 1/8-sectional model showed that PCT could increase by 4 °F if the 0.002-inch gap between the fuel tube and neutron absorber was replaced by a 0.004-inch gap and the 0.002-inch gap between the neutron absorber and stainless steel retainer was replaced by a 0.01-inch gap.

Another analysis showed that 1/8-sectional models using a k-effective thermal conductivity approach for modeling the neutron absorber resulted in fuel temperatures similar to a 1/8-sectional model using an explicit approach to modeling the neutron absorber (i.e., actually modeling the neutron absorber, stainless steel retainer, and gaps between retainer and basket).

These 1/8-sectional models indicated that PCT could be slightly higher than that reported in the full-scale models described in the SAR and the calculation packages.

Appendix W of Calculation Package No. 71160-3014, Rev. 9 provided a mesh sensitivity of the radiation matrices used at the surface of the package thermal model. Table W-1 indicated there was less than 1 °F temperature change when the number of shell elements used for the radiation matrix increased from 2850 to 24,200.

Likewise, the energy balance was 0.0001 W and the highest temperature convergence found during the sensitivity analysis was 0.005 °F. In addition, Appendix X of Calculation Package No. 71160-3014, Rev. 9 presented the sensitivity analysis of changing the number of sampling zones (20, 30, and 40) used in the radiation matrix and the number of modeling rays that determine the view factors. This sensitivity analysis showed negligible change in the package's inner shell and surface temperatures as well as negligible differences in the numerical values of the view factors.

As depicted in licensing drawing 71160-502, Rev. 6P, the outer package's finned arrangement consists of copper fins that are attached at different angles from the package surface and form alternate pairs of inward facing and outward facing fins. The package surface material between the in-ward facing fins is stainless steel and the package surface material between outward facing fins is aluminum. It is apparent from the above-mentioned drawing that the inward facing fins have a view that is mostly limited to the stainless steel package surface that is at a higher temperature (approximately 218 °F, per SAR Table 3.4-1) than the ambient surrounding temperature of 100 °F. This limited view would tend to reduce the radiant heat transfer to the ambient compared to the outward facing fins that have a direct view of the cooler ambient temperature.

As noted in Appendix Q of Calculation Package No. 71160-3014, Rev. 9, the SAR model did not include insolation to the inward facing fins and did not include the effect of the fin's stainless steel emissivity. When the base thermal model included the effect of the stainless steel fins and a revised radiation matrix to more accurately represent the

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. These models were based on the 10 CFR 71.71(c)(1) insolation of 400 g-cal/cm^2 (12-hour period); the applied flux was averaged over 24 hours and considered the surface's emissivity.

Appendix Q described another model that was based on the regulation's 200 g-cal/cm^2 insolation (12-hour period); similarly, this flux was averaged over 24 hours and considered the surface's emissivity. {proprietary information removed}.
Appendix Q stated that one conservatism in the model was not accounting for radiation heat transfer from one of the 15 sets of fins (although this was not quantified). Further, Appendix Q presented additional results assuming a simplification of the fin arrangement's "self-shielding effects"; however, in addition to being inconsistent with the regulations, such a simplification did not quantify or address the solar flux reflections and radiation heat transfer exchange throughout the outer surfaces of the model.

It is also necessary to

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SAR Table 3.2-13 presented the gaps between the packaging components in the MAGNATRAN model. These gaps included the space between basket slots at corners, between the TSC and MAGNATRAN inner shell, between the lead gamma shield and inner shell, between basket and canister bottom plate, and between the canister bottom plate and packaging bottom plate. SAR Section 3.2.2.3 provided the heat transfer equation for the gaps, which indicated that conduction and radiation processes are modeled. RAI 3-7 response (ADAMS Accession No. ML14356A385) and RAI 3-11 response (ADAMS Accession No. ML15296A084) presented detailed discussions for the choice of gaps and their assumptions, including the use of ANSYS models to determine contact locations of the basket and canister as well as the canister and cask inner shell.

In addition, the SAR model assumed

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and operation. The choice of gap sizes associated with the neutron absorber is non-negligible, considering results found in Calculation Package No. 71160-3014, Rev. 7,
{proprietary information removed}
1/8-sectional model.

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The applicant noted in Calculation Package No. 71160-3014, Rev. 8 Appendix U that the SAR thermal model (base case) is based on {proprietary information removed}.
The analysis assumed that the basket was specifically oriented 45 ° (see SAR Figure 2.6.13-4) to model horizontal and vertical gaps between adjacent fuel tubes (vertical gaps formed by side-by-side fuel tubes and horizontal gaps by fuel tubes placed on top and below each other).

It is noted that, although no analyses were provided, the April 2018 (ADAMS Accession No. ML18102A852), and August 2018 (ADAMS Accession No. ML18262A298) submittals indicated that the deadweight load of the fuel assemblies would tend to result in all adjacent tube corners being in contact.
{proprietary information removed}
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{proprietary information removed}

assumptions as in Case 1; this indicates an approximate 8 °F increase in temperature from the SAR model results.

The applicant qualitatively noted, in the August 2018 submittal, that the {proprietary information removed} is a bounding condition, considering that basket fabrication from individual fuel tubes would tend to require larger contact widths in order for the basket to fit into the TSC. Although an objective analysis to demonstrate this was not provided, the applicant's Appendix U, Case 2 analysis showed that {proprietary information removed} .

SAR Section 3.4.1.1.1 listed three model assumptions that the applicant considered conservative, as described below:

- One model assumption was to locate the fuel assembly at the center of the basket tube such that there would be no gap between the assembly and the tube as it rested in the horizontal orientation.
- A second model assumption was to remove convection heat transfer within the horizontal package and TSC.
- A third model assumption was to include a gap between the lead and the package inner shell.

Although there was no quantification of the assumptions, staff concludes these assumptions tend to be conservative. In addition, Appendix J of Calculation Package No. 71160-3014, Rev. 7 reported that the SAR analysis assumed a 10% penalty in the neutron absorber thermal conductivity such that the model's thermal conductivity is 10% lower than the lowest measured thermal conductivity. When removing this penalty, Appendix P of Calculation Package No. 71160-3014, Rev. 7 showed that {proprietary information removed} using the minimum thermal conductivity test data. Likewise, for the 1/8-sectional model, Appendix J reported PCT values that {proprietary information removed}, depending on using the averaged test data or the minimum test data for the neutron absorber conductivity.

SAR Section 3.5.1.1 states that the thermal analysis for the hypothetical accident condition fire test used the ANSYS® FEA solver and was based on a three-dimensional, full-length, one-fifteenth symmetry model. SAR Section 3.5.2 indicated that, during the 30-minute fire test, the package surface absorptivity was 0.9 (applicant assumed absorptivity was equal to emissivity) and that the fire emissivity was 1.0. SAR Section 3.5.1.1 noted that a fire-related convection coefficient of 0.01833 Btu/hr-in^2-F was applied to all package outer surfaces, including to both sides of the model's finned surfaces.

In addition, SAR Section 3.2.2.2 and RAI response 3.7 (ADAMS Accession No. ML15296A084) described how the above-mentioned heat transfer coefficient was modified to include radiation heat transfer. SAR Table 3.2-13 indicated a 0.015-inch gap was modeled between the lead and the inner shell. SAR Section 3.5.2 and the response to RAI 3-9 (ADAMS Accession No. ML14356A385) indicated that the results in SAR Section 3.5 are bounding for a package after undergoing the drop and puncture tests defined in 10 CFR 71.73.

In the SAR model, the applicant did not model the TSC and its content for the hypothetical accident condition fire. Rather, one model applied the heat flux (based on the decay heat) to the MAGNATRAN's inner shell for the PWR content; another model applied the heat flux that corresponded to the BWR content. To obtain an estimate of the basket and cladding temperatures, their normal conditions of transport temperatures were added to the increase of the package's maximum inner shell temperature calculated from the SAR model's fire transient analysis.

Staff notes that the applicant's methodology does not consider the important relation of content decay heat and thermal mass during the heat-up and cooldown phases of the hypothetical accident condition fire test and thus do not provide, *a priori*, known accurate or bounding results. Likewise, depending on the relative relation between the decay heat and thermal mass, cladding temperatures can increase and reach a peak value beyond the heat-up phase; this is not modeled in the applicant's methodology and there was no explanation provided which indicated that the methodology is, *a priori*, accurate or bounding.

In order to demonstrate that the SAR's HAC fire model is conservative, the applicant performed an additional fire analysis on a three-dimensional, 180° extent ANSYS® model that included the TSC and its PWR fuel contents. This model was described in Appendix E of Calculation Package No. 71160-3015, Rev. 7 and was based on the normal conditions of transport model found in Appendix M (Rev. 6) of Calculation Package No. 71160-3014, which had a normal conditions of transport PCT {proprietary information removed}.

Thermal and cooldown conditions similar to those described for the SAR's hypothetical accident conditions thermal model were applied to the Appendix E model. According to Table E-1, the PCT was reported

{proprietary information removed}

PCT reported in the SAR, thus demonstrating that the SAR's fire analysis model is conservative.

It is noted that, although the results indicate that the SAR thermal fire model is conservative for the particular conditions studied (e.g., decay heat, thermal mass), it has not been demonstrated that the SAR method will guarantee bounding values for other conditions. Therefore, future changes in decay heat and thermal mass may warrant additional analyses (similar to the Appendix E model in Calculation Package No. 71160-3015, Rev. 7) to demonstrate that the SAR model results are bounding.

3.4.2 Independent Analysis

The application relied on the results of both partial (e.g., 1/8-sectional, two-dimensional) and full (3D) ANSYS® models as described in the SAR and multiple calculation packages. Often, the ANSYS® models in the calculation packages were used to explain the continued reliance on the SAR ANSYS® model as being representative. For example:

- The October 2018 (ADAMS Accession No. ML18311A259) response listed six sensitivity models (from Appendix J, Appendix P, Appendix Q, and Appendix U in Calculation Package No. 71160-3014, Rev. 8) and their corresponding impact on PCT in order to explain the appropriateness of the SAR thermal model.
- The submitted models often were generated by resuming an ANSYS® database (.db) file and continuing to build the model in APDL, rather than generating an ANSYS® model file with a complete ANSYS® database file that contained only the geometry and mesh, in addition to including relevant APDL scripts (with comments) showing the material properties, heat load, and applied boundary conditions.
- As a result, some values or boundary conditions used in the models were mixed between the .db and APDL scripts, which had variability in the level of documentation and ability for review.

As part of the staff's review, attempts were made to evaluate numerous models using the resumed database file and APDL scripts; however, there were issues and uncertainties with many of these sensitivity models as a result of the review. For example: the ANSYS® model described in the SAR {proprietary information removed} (per Drawing No. 71160-551). The aforementioned issues and uncertainties with the modeling approach made it more challenging for the staff to make a safety finding regarding the MAGNATRAN design.

To help understand the impacts of these uncertainties on the thermal performance of the MAGNATRAN package, staff worked with Pacific Northwest National Laboratory to generate a COBRA-SFS model of the MAGNATRAN transport package and fuel assemblies inside a TSC. The TSC was modeled to include 37 PWR fuel assemblies with a decay heat of 23 kW. The information provided in the application was used to generate the COBRA-SFS model in which individual fuel assemblies were modeled.

The COBRA-SFS model assumed Type 2 neutron absorber {proprietary information removed} (these dimensions are discussed in Calculation Package No. 71160-3014, Rev. 7, Appendix J). In addition, the model assumed {proprietary information removed} between fuel tubes, as noted in licensing Drawing No. 71160-551, Rev. 10P.

The COBRA-SFS model assumed a 0.01-inch gap in the vertical regions that exist between fuel tubes and direct contact in the horizontal regions that exist between fuel tubes; these gap values were the boundary conditions for the SAR model described as "base case" in Appendix U of Calculation Package 71160-3014, Rev. 9. The computational fluid dynamics-based heat transfer coefficients applied to the package exterior's outward facing fins were as described in Calculation Package No. 71160-3045, Rev. 2. As with the SAR model, the COBRA-SFS model averaged the 400 g-cal/cm^2 (over a 12-hour period, per 10 CFR 71.71(c)(1)) insolation over 24 hours. As done in the SAR, the solar input applied to the package was based on using a material's emissivity value, described in the SAR's thermal chapter, as an absorptivity. As mentioned in SER Section 3.4.1, the inward facing fins have boundary conditions different from the outward facing fins. As a conservatism, there was no heat transfer from the inward facing fins.

Results from the independent analyses showing the package component temperatures are provided below. It is noted that, as discussed in SER Section 3.4.1, the PCT from the applicant's SAR thermal model and calculation package thermal models ranged from 725 °F to 737 °F.

Table: COBRA-SFS and SAR Results, peak component temperatures at normal conditions

Component	SAR results: Table 3.4-1 (°F)	COBRA-SFS component peak temperatures at normal conditions (conditions described above) (°F)
package lid and cover plate metallic O-ring		213
package lid and cover plate EPDM O-ring		213
package outer surface		220
radial neutron shield		229
lead gamma shield		260
basket		709
canister shell		306
canister bottom plate		309
maximum fuel rod (PCT)		731
package bottom plate		266
bottom forging		266
package inner shell		267
package outer shell		255
impact limiter wood		265
average gas temperature in canister		464

{proprietary information removed from SAR results column}
In addition, sensitivity analyses using the COBRA-SFS model described above were performed. The following are some examples of how the COBRA-SFS sensitivity results affected PCTs:

- One result showed that PCT would decrease by approximately {proprietary information removed}.

- Another result showed that PCT could increase by {proprietary information removed}

- Another run showed that PCT would only increase by approximately {proprietary information removed}.

As discussed above, the geometry of the fuel tube, including the contact width and gaps between the tubes, influences the conduction heat transfer from the hot contents to the package exterior. This is to be expected because, as noted in SAR Section 3.2, heat transfer within the horizontally oriented package is primarily by conduction and radiation.

Using COBRA-SFS, a sensitivity analysis was performed to determine the impact of changing

{proprietary information removed}

As previously mentioned, SAR Section 3.5 indicated that the applicant's model did not include the TSC and its content when modeling the hypothetical accident condition fire exposure. Therefore, the applicant's SAR methodology did not consider the important inter-related effects of content decay heat and thermal mass during the heat-up and cooldown phases of the hypothetical accident condition fire exposure. As mentioned earlier, depending on the relative relation between the decay heat and thermal mass, cladding temperatures can increase and reach a peak value beyond the heat-up phase. This was not modeled accurately in the applicant's SAR model methodology and there was no explanation provided which indicated that, *a priori*, the methodology and the results always would be bounding.

Therefore, a COBRA-SFS model was also used to model the MAGNATRAN transport package under the hypothetical accident condition fire exposure conditions. The initial condition of the HAC thermal analysis was based on the COBRA-SFS steady-state normal conditions of transport thermal analysis described earlier.

Results from the hypothetical accident conditions analysis indicated that the maximum temperature of the lead gamma shielding was

{proprietary information removed}

This PCT result is different from the applicant's SAR Section 3.5 hypothetical accident condition cladding temperature for the fire test transient, which indicated a peak temperature {proprietary info removed}, and, according to Figures 3.5-4 through Figure 3.5-9, show package peak temperatures occur within less than 10 hours. The large difference in the temperature transients is due to the differences between the physics-based COBRA-SFS approach and the applicant's non-physics-based approach of adding a normal conditions-derived temperature difference to the fuel and basket temperature for the fire hypothetical accident condition fire exposure. It is noted, however, the ANSYS® model described in Calculation Package 71160-3015, Rev. 7, which modeled the decay heat and the thermal mass effects of the content, showed that the maximum fuel temperature of {proprietary information removed} hours after the start of the fire.

Thermal model results from the application show, and the independent analyses confirm, that package component and cladding temperature would be below allowable temperatures for the specific conditions analyzed, including content containing PWR fuel with Type 2 neutron absorber and 23 kWs of decay heat. The uncertainties in the applicant's SAR model discussed in this SER should be considered for any future changes in package design or operating conditions.

3.4.3 Evaluation of Accessible Surface Temperature

The main accessible surface of the MAGNATRAN package is the personnel barrier that surrounds the package body, as shown in SAR Figure 1.2-2. According to licensing Drawing No. 71160-511 Rev. 1, this personnel barrier consists of an aluminum mesh with an opening size of 1 inch or smaller and an opening percentage of 65% or greater; the barrier is located 7.1 inches from the package fin tips.

The applicant's Appendix M, Calculation Package No. 71160-3045, Revision 2 presented the analysis and assumptions in determining the temperature of the personnel barrier. The applicant's analysis indicated that the personnel barrier, which is located away from the thermal boundary layer (rising heated air surrounding the package), has a screen temperature of 162.85 °F for the design-basis heat load of 23 kW, which is below the 185 °F regulatory limit for exclusive use shipment in 10 CFR 71.43(g). SAR Section 7.1.1 indicated that the exterior of the transport vehicle and package, which includes the personnel barrier, are to be cleaned to prevent dirt/debris buildup on the exterior that would affect heat transfer from the package.

3.5 Thermal Evaluation for Normal Conditions of Transport

3.5.1 Heat and Cold

SAR Section 3.4.1.1.1 indicated that the ANSYS® models were run as a steady-state analysis to determine the package component temperatures at normal conditions of transport. The results of the applicant's thermal models, described above, were discussed in SAR Section 3.4.2, Section 3.4.3, Section 3.4.4, Section 3.4.5 and Section 3.4.6; these sections indicated there would be no degradation of the package or contents during normal conditions of transport. For cold conditions, the applicant stated that the -40 °F minimum temperatures of the package components would occur assuming zero decay heat and a -40 °F ambient temperature. The applicant indicated in SAR Section 3.3.2 that the metallic and EPDM O-rings, lead gamma shield, and radial NS-4-FR neutron shield had a safe operating temperature of -40 °F.

SAR Table 3.4-1 showed that at a 100 °F ambient temperature, with insolation, and maximum decay heat, the temperature of the containment vessel, fuel/cladding, O-rings, radial neutron shield, lead gamma shield, and impact limiters were below their allowable values. For example, the package lid/coverplate metallic and EPDM O-rings were predicted to be 186 °F which is below the specified temperature limit. The average gas temperature inside the TSC was 459 °F and 509°F for PWR and BWR fuel, respectively.

3.5.2 Maximum Normal Operating Pressure

SAR Section 3.4.4 presented the analyses for determining the MNOP within the MAGNATRAN package body during normal conditions of transport. SAR Section 3.4.4.1 indicated that the

MNOP within the package body was calculated to be 23 psig based on the helium backfill pressure and adjusting the temperature to the normal conditions of transport temperatures. The SAR indicated that the pressure within the transport package could increase if one was to account for the failure of the TSC confinement boundary and 3% fuel failure, such that 30% fission gas, 100% rod backfill, BPRA gas, integral fuel burnable absorber gas, and shim rod gases would pressurize the cask.

According to SAR Section 3.4.4.1, summing the previously mentioned gas molar quantities and applying the ideal gas law at a gas temperature of 515 °F (slightly higher than the ANSYS calculated number provided in SAR Table 3.4-1),
{proprietary information removed}
. Both pressures are below the 120 psig design pressure (per SAR Section 8.1.3.3) and below the allowable pressure of 300 psig, as denoted in SAR Table 3.5-3.

The discussions presented in the SAR indicate that combustible gas generation is not an issue for the MAGNATRAN package contents. For example, SAR Section 1.3.2 states that, for the GTCC waste canister, which contains metallic (non-radiolysis) components, "…no hydrogen generation occurs as a result of residual water …" due to content loading in the canister and the subsequent drying process.

For spent fuel content, SAR Section 7 stated that "… loading, preparation, and transfer procedures, and requirements for TSCs containing spent nuclear fuel are provided in the MAGNASTOR Final Safety Analysis Report (FSAR) …". It is noted that MAGNASTOR FSAR Revision 1 (ADAMS Accession No. ML110130432) stated on page 8.10-6 that "… hydrogen gas concentrations exceeding 2.4% are removed by flushing air, nitrogen, argon or helium into the region below the closure lid or by evacuating the hydrogen using a vacuum pump."

3.5.3 Maximum Thermal Stresses

The applicant indicated that thermal stress calculations for normal conditions of transport were provided in SAR Section 2.6. Further review of thermal stress is considered in the SER in Chapter 2, "Structural Evaluation."

3.6 Thermal Evaluation Hypothetical Accident Conditions

3.6.1 Initial Conditions

The applicant performed a transient thermal analysis to evaluate the package with maximum decay heat under the hypothetical accident condition fire test. The pre-fire condition, based on normal conditions of transport, included a 100 °F ambient temperature and insolation, with radiation heat transfer and convection heat transfer at the cask surface. Unlike the normal conditions of transport pressure, which assumed 3% fuel rod failure, SAR Section 3.5.4 assumed 100% fuel rod failure when calculating pressure within the package during the hypothetical accident conditions fire test.

According to SAR Section 3.5, the MAGNATRAN transport package was analyzed in a form consistent with the damage sustained in the free-drop and puncture tests for hypothetical accident conditions. In addition, the applicant stated in RAI response 3-9 (ADAMS Accession No. ML14356A385) and SAR Section 2.12.2.3 that the impact limiters on the package were not damaged sufficiently during the free drop and puncture test analyses to warrant a reduction in thermal effectiveness during the fire accident condition. Appendix I of Calculation Package No. 71160-3015, Rev. 3 indicated that only a thin layer of the impact limiter's wood near the surface would char during the fire transient. Although the SAR model did not address the effects of charring, according to the response to RAI 3-9 the applicant analyzed this effect, and per Appendix H of Calculation Package No. 71160-3015, Rev. 3, showed that temperatures of the O-rings, lead, and package lid changed very little (less than 2 °F) from the model that assumed adiabatic conditions at the impact limiter location.

3.6.2 Fire Test Simulation

SAR Section 3.5.1.1 indicated that the ambient temperature boundary condition was 1475 °F during the 30-minute fire simulation. The SAR stated that the emissivity of the fire simulation was assumed to be 1.0 and the absorptivity of the package surface was assumed to be 0.9. According to SAR Section 3.5.1.1, the package's exterior convection coefficient during the 30-minute fire was 0.01833 Btu/hr-in^2-F.

The post fire cool-down was modeled as a 64-hour transient. The cool-down ambient conditions were 100 °F with insolation. The emissivity values of the stainless steel, copper, and aluminum exterior surfaces were 0.36, 0.65, and 0.22, respectively. The FLUENT-based heat transfer coefficient described in SAR Section 3.2.3 was applied to the package surface after the 30-minute fire; no artificial cooling was applied. Per SAR Section 3.5.1.1 and 3.5.2, for thermal analysis purposes, the NS-4-FR neutron shield was assumed lost during the fire because it reached a temperature above its allowable value; the volume previously occupied by the neutron shield was replaced with air properties.

3.6.3 Maximum Temperatures and Pressure

Results of the applicant's thermal analysis for the hypothetical accident conditions fire test were presented in SAR Section 3.5.3 and SAR Table 3.5-2. The components presented in the Table 3.5-1 for PWR fuel, and SAR Table 3.5-2 for BWR fuel, included cladding, shielding, metallic O-rings, average helium gas, and items associated with the package's structural components.

In addition, SAR Figures 3.5-4 through 3.5-9 showed the transient temperature plots for lead, exterior package surface, inner and outer package shells, package lid and its inner O-ring, and coverplate inner O-ring, during hypothetical accident conditions including the resulting cool-down period; it is noted that these transient profiles of temperature reflect the non-physics based methodology of not including the thermal mass associated with the content in the analysis. The temperature limits for each component were listed in the above-mentioned tables; components were below these temperature limits for hypothetical accident conditions (see additional discussion in SER Section 3.2.4). For example, the lead gamma shield remained below its allowable temperature of 600 °F.

Section 3.5.4 of the SAR presented the analyses for determining the maximum internal pressure within the MAGNATRAN package body during the hypothetical accident conditions fire test. As mentioned previously, SAR Section 3.4.4.1 indicated that the pressure at normal conditions within the cask body was calculated to be 23 psig based on the helium backfill pressure and adjusting the temperature to operation condition.

The SAR indicated that the pressure within the package could increase if one was to account for the failure of the TSC confinement boundary and 100% fuel failure at accident conditions. According to SAR Section 3.4.4.1 and Section 3.5.4, the applicant summed the molar quantities of the previously mentioned gases and applied the ideal gas law at a hypothetical accident condition gas temperature {proprietary info removed}; the maximum pressure within the package was calculated to be {proprietary information removed}. SAR Section 3.5.4 indicated that this is below the package allowable pressure of 300 psig.

3.6.4 Maximum Thermal Stresses

In response to RAI 3-10 (ADAMS Accession No. ML15296A084) regarding the potential for adverse consequences to the package due to thermal gradients during the hypothetical accident condition fire test, the applicant stated that the Calculation Package No. 71160-2108, Appendix L analysis of the package lid bolt loads indicated the integrity of the closure assembly would be maintained during the fire transient. The applicant also stated that there would be no significant change at the interface between the lead shielding and the interfaces with the inner shell or outer shell. Further review of thermal stresses was considered in the SER structural evaluation.

3.7 Appendix

The applicant provided numerous calculations and their appendices throughout the review to support the information provided in Chapter 3 of the SAR and NAC's responses to NRC's requests for additional information. The calculations and other supporting information may be found in earlier revisions of a calculation package. In addition to the SAR and NAC's responses to NRC's requests for additional information, the calculations and their appendices provided information necessary for the staff to make its safety findings regarding the adequacy of the design with respect to the requirements in 10 CFR Part 71.

- Calculation Package No. 71160-2101, Revision No. 6 (submitted with letter dated November 26, 2012), Revision No. 7 (submitted with letters dated December 1, 2014, and October 15, 2015), Revision No. 8 (submitted with letter dated January 11, 2017) "Structural and Thermal Material Properties – MAGNASTOR/ MAGNATRAN Cask Systems."
- Calculation Package No. 71160-2108, Revision No. 2 (submitted with letter dated November 26, 2012), Revision No. 3 (submitted with letter dated December 1, 2014) "Transport Cask Structural Evaluation."
- Calculation Package No. 71160-2121, Revision No. 0 (submitted with letter dated October 15, 2015) "Gap between the Lead and Steel Shells of the Transport Cask."
- Calculation Package No. 71160-3001, Revision No. 3 (submitted with letter dated October 15, 2015) "Effective Property Calculation of Loaded PWR Basket of NewGen System."
- Calculation Package No. 71160-3002, Revision No. 1 (submitted with letter dated October 15, 2015) "Effective Property Calculation of Loaded BWR Basket of NewGen System."
- Calculation Package No. 71160-3011, Revision No. 0 (submitted with letter dated November 26, 2012), Revision No. 1 (submitted with letter dated December 1, 2014), Revision No. 2 (submitted with letter dated January 11, 2017) "Effective Property Calculation of PWR/BWR Fuel Assembly and Poison Plate for Transport Condition of the NAC-MAGNATRAN System."
- Calculation Package No. 71160-3013, Revision No. 0 (submitted with letter dated November 26, 2012), Revision No. 1 (submitted with letter dated December 1, 2014), Revision No. 2 (submitted with letter dated October 15, 2015), Revision No. 3 (submitted with letter dated January 11, 2017) "Thermal Evaluation of MAGNATRAN Transport Cask/BWR Canister."
- Calculation Package No. 71160-3014, Revision No. 2 (submitted with letter dated November 26, 2012), Revision No. 3 (submitted with letter dated December 1, 2014),

Revision No. 4 (submitted with letter dated October 15, 2015), Revision No. 5 (submitted with letter dated January 11, 2017), Revision No. 6 (submitted with letter dated October 6, 2017), Revision No. 7 (submitted with letter dated April 9, 2018), Revision No. 8 (submitted with letter dated August 30, 2018), Revision No. 9 (submitted with letter dated October 30, 2018) "Thermal Evaluation of MAGNATRAN Transport Cask/PWR Canister."

- Calculation Package No. 71160-3015, Revision No. 1 (submitted with letter dated November 26, 2012), Revision No. 3 (submitted with letter dated December 1, 2014), Revision No. 5 (submitted with letters dated October 15, 2015 and May 9, 2016), Revision No. 6 (submitted with letter dated January 11, 2017), Revision No. 7 (submitted with letter dated October 6, 2017) "MAGNATRAN PWR/BWR Cask/Basket Hypothetical Fire Accident Analyses."
- Calculation Package No. 71160-3020, Revision No. 14 (submitted with letters dated December 1, 2014 and October 15, 2015) "MAGNASTOR Transfer Cask and PWR Canister Transient Analysis."
- Calculation Package No. 71160-3022, Revision No. 1 (submitted with letter dated December 1, 2014) "MAGNATRAN Internal Pressure Evaluation."
- Calculation Package No. 71160-3045, Revision No. 0 (submitted with letter dated November 26, 2012), Revision No. 1 (submitted with letter dated December 1, 2014), Revision No. 2 (submitted with letter dated October 15, 2015) "Evaluation of the Convection Film Coefficient for the MAGNATRAN Cask Surface."
- Document No. Package No. 71160-R-01, Rev. 0 (submitted with letter dated April 9, 2018), "MAGNASTOR Tube to Poison Gap Assessment at Hitachi-Zosen."

3.7.1 Computer Program Description

The applicant provided a brief description of the ANSYS® finite element analysis code. SAR Section 3.4 indicated that ANSYS is a general-purpose finite element analysis program and subsequent sections of the SAR described the types of elements that were used to generate the models. As noted in SAR Section 1.4.2, the ANSYS® code and its use would fall under NAC's quality assurance program, which is applied to the design and analysis of the package and the packaging's fabrication, assembly, testing, maintenance, and repair.

3.8 Evaluation Findings

F3.1 The staff has reviewed the package description and evaluation and has reasonable assurance that the information provided demonstrates that the thermal requirements of 10 CFR Part 71 are met.

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 method used in the thermal evaluation for normal conditions of transport and has reasonable assurance that the models are described in sufficient detail to permit an independent review 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 and conditions 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 and conditions specified in 10 CFR 71.73.

3.9 References

1. NUREG-1617, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, U.S. Nuclear Regulatory Commission, March 2000.

2. American National Standards Institute ANSI N14.5, *American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials, New York, NY, 1997.*

4.0 CONTAINMENT REVIEW

The objective of the review is to verify that the MAGNATRAN transportation package containment design is adequately described and evaluated under normal conditions of transport and hypothetical accident conditions, as required per 10 CFR Part 71. This was achieved by reviewing and confirming information presented in the SAR Chapters including: Structural Analysis (Chapter 2), Materials (Chapter 2), Thermal Analysis (Chapter 3), Package Operations (Chapter 7), and Acceptance Tests and Maintenance (Chapter 8).

4.1 Description of the Containment System

4.1.1 Containment Vessel

The containment system boundary consists of the containment shell, baseplate (bottom forging), the closure flange (top forging), the closure lid, and the closure lid access port cover plate. The closure lid subcomponents include 48 lid bolts and the closure lid access port plug. The closure lid containment boundary seals include the closure lid inner seal and the closure lid access port plate inner seal. Materials of construction for all containment components, except for the metallic O-rings, are 304 stainless steel with the exception of the closure lid, which is fabricated of 17-4 PH stainless steel.

The containment system components are designed and fabricated in accordance with the requirements of the ASME B&PV Code[1], Section III, Subsection NB, as described in Section 8.1.1(c) of the application.

The applicable ASME code requirements and alternatives to specific code requirements are presented in the SAR in Table 2.1.4-1. In Chapter 7, Package Operations, closure bolt torque values and lubrication requirements are provided in Table 7.1-1 of the SAR.

The applicant states that the closure lid containment boundary and boundary sealing surfaces are not subject to corrosion due to the presence of the cavity helium backfill and a dry loading environment. The applicant also stated that the materials of construction are highly corrosion resistant such that no galvanic, chemical, or other reactions will occur between the seal and the packaging or its contents during transportation. This has been confirmed by the materials review.

All of the containment system components, as identified above, are shown in the drawings and the information provided by the applicant in the Chapter 4, “Containment Evaluation,” of the SAR regarding components of the containment system is consistent with that presented in the structural and thermal evaluation sections of the application.

4.1.2 Containment Penetrations

The containment boundary penetrations are the closure lid and the closure lid access port. The containment penetrations are designed and tested to ensure that the radionuclide release rates specified in 10 CFR 71.51 will not be exceeded.

<u>Seals:</u>

The package closure lid has two concentric seals to form the closure with the containment closure flange surface; an outer non-containment EPDM seal and an inner metallic seal. An inter-seal test port provides access to the volume between the two lid seals. Following leakage rate testing of the closure lid inner seal, the closure lid port cover plate is closed by bolting the port cover plate to the mating surface with two concentric seals; an outer non-containment EPDM seal and an inner metallic seal. An inter-seal test port provides access to the volume between the two port cover seals.

The inner metallic containment seals are spring energized Helicoflex seals having material options that include stainless steel 321 Alloy 600 or Alloy X750 for springs and silver or nickel plating on the seal jacket. Critical characteristics of the metallic seals are provided in Appendix 4.5.2 of the application. Non-containment EPDM seal critical characteristics are provided in Appendix 4.5.3 of the application. Tolerances for the containment boundary groove dimensions have been provided on the licensing drawings.

Welds

The package containment system boundary welds consist of full penetration welds forming the package inner containment shell and the full penetration welds connecting the inner containment shell to the top and bottom forgings. Post-weld examinations are performed using liquid penetrant testing and radiographic examination. As specified, all containment system boundary welds are fabricated and inspected in accordance with the ASME B&PV Code Section III, Subsection NB (NB-5350, NB-5320).

4.2 Containment Under Normal Conditions of Transport

The containment system of the package is designed to be leaktight as defined in ANSI N14.5-1997[2], i.e., there is no seal leakage greater than 2 x 10^{-7} ref-cm^3/s of helium with a test sensitivity of 1 x 10^{-7} ref-cm^3/s of helium as is described in Section 8.1.4.1 of the application.

Fission Gas Products

For normal conditions of transport, a 3% rod failure was assumed to be the maximum fission gas release and this gas volume is available for release to the TSC cavity. The evaluation of the MAGNATRAN package assumes that the TSC will release pressurized gasses into the package internal cavity, thereby increasing the pressure above the MNOP of 23 psig. As the MAGNATRAN transport package is evaluated with a leaktight containment boundary, explicit release fraction calculations are unnecessary, per the ANSI N14.5-1997 definition of leaktight.

Summary of Pressures and Temperatures

The maximum normal condition pressure for the package containment vessel was calculated to be 112.4 psig and was based on the hypothetical release of contents of the TSC, as indicated above. This calculated cavity pressure, was lower than the design internal pressure of 120 psig (135 psig was used for analysis).

The normal conditions of transport maximum temperatures of the containment shell, bottom forging, package bottom, and closure lid, as illustrated in Table 3.4-1 of the application, do not result in stresses that exceed the ASME B&PV Code allowable limits presented in Section 2.6 of the application. Inner and outer seal temperatures are also below allowable values.

Section 2.6.7.6 of the SAR, *Closure Analysis*, demonstrated that lid seal elements remained closed (i.e., the loading in the elements representing the seal remains compressive) and the bolts remained elastic.

Containment of Radioactive Material

The regulatory limit (10 CFR 71.51) for the release of radioactive material under normal conditions of transport is 10^{-6} A_2/hr. A leaktight containment boundary meets this requirement per ANSI N14.5-1997.

Summary

Since the thermal and structural evaluations demonstrated no reduction in the effectiveness of the package or components, there will be no release of radioactive material, which exceed NRC regulations under normal conditions of transport.

4.3 Containment Under Hypothetical Accident Conditions

The containment system of the package is designed to be leaktight, as defined in ANSI N14.5-1997, (i.e., there is no seal leakage greater than 2 x 10^{-7} ref-cm^3/s of helium with a test sensitivity of 1 x 10^{-7} ref-cm^3/s of helium) as described in Section 8.1.4.1 of the application.

Fission Gas Products

The maximum fission gas release for hypothetical accident conditions was assumed to be 100% rod failure and this gas volume is available for release to the TSC cavity. The evaluation of the MAGNATRAN package assumes that the TSC will release pressurized gasses into the package internal cavity, thereby increasing the pressure above the MNOP of 23 psig. As the MAGNATRAN transport package is evaluated with a leak-tight containment boundary, explicit release fraction calculations are unnecessary, per the ANSI N14.5-1997 definition of leak-tight.

Summary of Pressures and Temperatures

The maximum pressure for the package containment vessel after evaluation of the tests for hypothetical accident conditions was calculated to be 256.3 psig, based on the hypothetical failure of the TSC boundary, as indicated above. This calculated cavity pressure, was lower than the internal pressure of 300 psig used for the applicant's analysis, as illustrated in Section 3.5.4 and Table 3.5-3 of the application. The staff noted that in Section 2.7.4.1, Summary of Temperatures and Pressures, the applicant incorrectly indicated that the fire accident pressure evaluation was performed using only 3% failed rods.

The hypothetical accident conditions maximum temperatures of the containment shell, inner closure lid, outer closure lid, containment baseplate, and lid and cover plate metallic seals do not exceed the temperature limits illustrated in Table 3.5-1 of the application.

Section 2.7 of the application illustrates that all containment system boundary components are within their ASME B&PV Code allowable stress limits and the metallic lid seals will remain compressed during all hypothetical accident conditions.

Containment of Radioactive Materials

The regulatory limit (10 CFR 71.51) for the release of radioactive material under hypothetical accident conditions is less than 10 A_2 in 1 week for krypton-85 and 1.0 A_2/week for all other radioactive isotopes. A leaktight containment boundary meets the requirement per ANSI N14.5-1997.

Summary

Since the thermal and structural evaluations demonstrated no reduction in the effectiveness of the package or components, there will be no release of radioactive material exceeding NRC requirements under hypothetical accident conditions.

4.5 Evaluation Findings

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

References:

1. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 3, *Containment Systems and Transport Packagings For Spent Nuclear Fuel and High Level Radioactive Waste*, New York, NY.

2. American National Standards Institute ANSI N14.5, *American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials, New York, NY, 1997.*

5.0 SHIELDING REVIEW

The objective of this review is to verify that the package design meets the external radiation requirements in 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions.

For the shielding review, the staff evaluated the capability of the MAGNATRAN shielding features to provide adequate protection against direct radiation from its contents. This review includes the staff's evaluation of the descriptions of the proposed contents, the package shielding features and the calculation of the dose rates from both gamma and neutron radiation at locations near the package and at distances away from the package during transportation for both normal conditions of transport and hypothetical accident conditions.

5.1 Description of the Shielding Design

5.1.1 Packaging Design Features

The package design is characterized by the licensing drawings, which are incorporated by reference into the CoC. The staff reviewed Chapter 1 of the application and the licensing drawings to understand the package's features, particularly as they relate to shielding.

The package contents are loaded into a welded canister (the TSC) that is loaded into the package body. Spent fuel contents are loaded into a basket in the TSC that is formed by carbon steel tubes or channels and support weldments. Neutron absorber plates are attached to the walls of the channels and are credited in the shielding analysis. GTCC waste contents are loaded into a GTCC waste basket liner and placed into the TSC. The liner is open on the top (i.e., does not have a lid) and has drain holes in the bottom.

The TSC has a stainless steel base and shell. The TSC lid is either a single solid stainless steel lid or a thinner stainless steel lid with an attached carbon steel shield plate. For some of the vent and drain ports, a shield collar is attached to the underside of the lid or shield plate. The TSC has two lengths. When the shorter length TSC is used, the position of the TSC is maintained in the package body by the use of a cavity spacer, as shown on Drawing No. 71160-506, Rev. 1, attached to the underside of the lid and comprised of a steel plate and concentric steel rings.

The package body uses steel and lead as the gamma shielding. Axial gamma shielding is accomplished with the thick steel of the lid, lower forging and bottom plate. Thick lead and steel shells provide the primary radial gamma shielding. Neutron shielding is provided by the neutron shield assemblies for the radial surface and by the wood in the impact limiters for the axial surfaces. A personnel barrier is also attached to the package between the impact limiters.

The neutron shield assemblies do not extend the entire length of the package. There is a large gap between the top end of the neutron shield assemblies and the top impact limiter. There is also a small gap between these assemblies and the bottom impact limiter. However, the neutron shield assemblies extend from the base of the package cavity to within several inches of the top of the cavity, as seen in the licensing drawings. Exceptions are where the two steel rotation trunnions are located on the package body. Some of the assemblies are bolted onto the outer steel shell of the package body, while others are fixed to the outer steel shell by the copper fins and the neighboring assemblies that are bolted to the package body.

The neutron shield assemblies also include the steel shell encasing the neutron shield material and thermal insulator and expansion foam. Thus, along with the bolt penetrations for the bolted assemblies, the rotation trunnions, and the copper fins between neighboring assemblies, these materials represent areas for neutron streaming. The gap between the bottom impact limiter and the neutron shield assemblies is also a streaming path. There is also a large gap between the top impact limiter and the neutron shield assemblies. The package surface in this gap area is at the surface of the outer steel shell rather than the surface of the neutron shield assemblies.

The lead gamma shielding covers all but the top few inches of the package's cavity. Given the shape and manner of attaching the impact limiters to the package body, there are areas in the impact limiters of reduced shielding and streaming paths (e.g., the holes for the rods that attach the impact limiter to the package body, a several-inch tall ring of void space on the underside of the impact limiter through which the attachment rods pass), which can be seen from the licensing drawings.

Through its review of the drawings and Chapter 1, the staff identified the package design features that affect shielding. The staff also identified the surfaces where dose rates must meet regulatory limits (see Section 5.1.2 of this SER). The staff also identified the features that could be radiation streaming paths. The staff's review of the drawings included identification of the material specifications, particularly for the neutron shield material, and the dimensions of the package features. The staff's review of the dimensions included the tolerances. As discussed later in this chapter of the SER, the staff used this information to ensure the package models in the applicant's shielding analysis used appropriate material properties and package dimensions and tolerances to demonstrate the package's shielding capability.

In reviewing the drawings, the staff identified that they contained appropriate minimum material specifications for non-standard shielding materials, namely the NS-4-FR neutron shielding. The staff had concerns about the initial lack of dimensional tolerances in the drawings. Without the tolerances, it was difficult for the staff to identify the design's minimum shielding capability, to confirm that the shielding analysis demonstrated that minimum shielding capability, and to confirm the adequacy of the shielding acceptance and maintenance tests, including their acceptance criteria (see the respective sections in Section 8 of this SER).

To address the staff's concerns, the applicant provided tolerances on dimensions of packaging components that are important in regard to the shielding performance of the package. The tolerances enable the establishment of minimum thicknesses for several packaging components (e.g., the lead shielding and the neutron shielding) as well as the minimum axial extent of some components that can potentially impact the package dose rates where the packaging's shielding properties change (e.g., the axial extent of the lead, resulting in more of the package cavity not being shielded by lead in the radial direction). The staff reviewed the applicant's proposed tolerances and finds that most tolerances are consistent with the dimensions used in the shielding analysis.

Some tolerances, however, do not appear to be consistent with the analysis. This was particularly so for the axial tolerances of the lead and neutron shield assemblies. For these cases, the applicant provided additional analyses or other justification for why those tolerances should be acceptable. The staff reviewed this information and determined that the impact of the tolerances versus the dimensions used in the main analysis has no or has a limited impact on the package dose rates. For instances where some limited impact is identified, the impact is compensated by other conservatisms in the analysis or occurs at locations on the package with significant margin to the regulatory radiation limit for normal conditions of transport. The tolerances do not alter the maximum package dose rates. Thus, the staff finds these proposed tolerances to also be acceptable.

For other package component dimensions, the applicant did not add tolerances. These include dimensions such as the length of the cavity spacer and the TSC lids or lid components that are forgings. The staff reviewed these items for potential impact to the shielding evaluation, considering the applicant's reasoning for not adding tolerances to these other component dimensions. This review included consideration of effects on dose rates due to reduced component thickness and the impact of small changes in these dimensions on the possible configuration of the package contents versus package shielding. Based on those considerations as well as the margins to regulatory radiation limits, the expected size of tolerances on component thickness, the amount of remaining material for shielding for some locations around the package, and the tolerances for those dimensions for which the drawings include tolerances, the staff finds that not specifying tolerances on the remaining dimensions is acceptable. The staff also finds, based on the package design, as described in the drawings, specification of a tolerance is not needed on some of these non-toleranced dimensions because they would not be expected to alter the position of the contents with respect to other shielding (e.g., versus the axial end of the lead or neutron shield) to an extent that would affect package dose rates and compliance with regulatory limits as evaluated later in this SER chapter.

Therefore, based on its review, as described here, the staff finds that the package has been adequately described to enable a proper evaluation of the shielding design. The staff also finds that the technical drawings adequately capture the design specifications, including material and dimension specifications together with needed tolerances, to ensure the fabricated package will meet the shielding design as analyzed in the application.

5.1.2 Summary Table of Maximum Radiation Levels

The package is designed and analyzed for exclusive use shipment on an open conveyance with an enclosure; thus, the dose rate limits for exclusive use shipments in 10 CFR 71.47 apply. For the MAGNATRAN, the enclosure is the personnel barrier that extends the entire length of the package between the impact limiters. The conveyance in the analysis is 124 inches wide. Given the enclosure only covers the radial package surfaces between the impact limiters, the limit of 1000 mrem/hr only applies to those radial surfaces of the package. The limit of 200 mrem/hr applies to all the other package surfaces (i.e., the impact limiter surfaces).

The staff notes that the width of the conveyance is the same as the diameter of the personnel barrier and is a few inches smaller than the widest diameter of the impact limiters. No dimensions for the axial placement of the package versus the ends of the conveyance were provided; therefore the limits for the conveyance edge and 2 meters from the conveyance edge are applied to the impact limiters' axial surfaces and 2 meters from those surfaces.

Tables 5.1-3 and 5.1-4 of the application summarize the maximum dose rates for normal conditions of transport and hypothetical accident conditions, respectively, for undamaged spent fuel contents. Tables 5.1-5 and 5.1-6 summarize the maximum dose rates for normal conditions of transport and hypothetical accident conditions, respectively, for damaged PWR spent fuel contents. Tables 5.1-7 and 5.1-8 summarize the maximum dose rates for normal conditions of transport and hypothetical accident conditions, respectively, for GTCC waste contents. The tables include the maximum dose rates on the package surfaces, at 1 meter from the package surfaces, and at 2 meters from the package surfaces or conveyance edge for locations along the package radial side, axial top, and axial bottom. Tables 5.1-9 and 5.1-10 summarize the spent fuel contents that result in the maximum dose rates. Figures 5.1-1 through 5.1-6 of the application show some of the maximum dose rates relative to the package for the respective contents for the respective conditions (normal conditions of transport and hypothetical accident conditions).

For an exclusive use shipment, the applicant needs to provide the maximum normal conditions of transport dose rates for the enclosure surface and the package surfaces that are outside of the enclosure (namely the radial surfaces of the impact limiters for the MAGNATRAN) and demonstrate they are less than the appropriate regulatory limit. The applicant modeled the impact limiters at a diameter equal to that of the neutron shield assemblies' outer diameter and calculated radial dose rates at 1 foot from the neutron shield surface. This 1-foot location is the location of the personnel barrier (the package enclosure) and the width of the conveyance. While not included in the tables and figures identified earlier in this section, the applicant provided information regarding the 1-foot location dose rates in the description of the shielding analysis results for the PWR and BWR spent fuel contents, including damaged PWR fuel (see Sections 5.8.3, 5.8.4 and 5.8.10 of the application). The information also addresses dose rate peaking due to axial and radial variations in the package's shielding and the loading configurations. The applicant did not provide the dose rates at this 1–foot distance for the GTCC waste contents; however, the dose rates for the package surface already meet the regulatory limit for the enclosure surface. Thus, the maximum dose rates determined by the applicant demonstrate compliance with 10 CFR 71.47 for exclusive use shipments with an enclosure.

5.2 Radiation Source

5.2.1 Spent Fuel Contents

The proposed package contents include PWR and BWR spent fuel and PWR non-fuel hardware (NFH). PWR contents include both damaged and undamaged fuel, with the exception that damaged CE 16x16 fuel (assembly type 16a in the applicant's analysis) is not allowed. BWR contents are limited to undamaged fuel. Damaged PWR fuel is limited to four corner locations of the damaged fuel basket and can only be loaded in the shorter length TSCs. PWR and BWR contents are limited to assemblies with zirconium-based alloy cladding. This includes instrument tubes, guide tubes, and water holes. BWR assemblies may be loaded without channels or with zirconium-based alloy channels.

Undamaged spent fuel assemblies may include solid filler, or replacement, rods in the place of fuel rods. These rods may include steel rods or unenriched rods. These rods may be irradiated for PWR assemblies; however, such assemblies are limited to one per TSC and up to five irradiated steel rods in that assembly. There is no limit for unenriched rods. The replacement rods for BWR assemblies are limited to un-irradiated rods.

The applicant performed an analysis for a PWR assembly with irradiated steel replacement rods, but not for irradiated unenriched rods. While the staff finds it reasonable to expect that, as the applicant stated, the number of unenriched replacement rods will be limited, the staff finds that this introduces some uncertainty into the analysis. The staff performed a depletion calculation of a sample case to evaluate the impact of these kinds of replacement rods on the spent fuel source term. The staff used a Babcock and Wilcox (B&W) 15x15 assembly at one of the proposed combinations of burnup, enrichment and cooling time limits. The staff calculated the source terms using the 2-D depletion sequence, t-depl, in the SCALE 6.1 code for both a completely enriched assembly and an assembly with a reasonably conservative number of unenriched replacement rods. The comparison of the source terms resulted in a few percent difference between the two cases. The staff included this result in its evaluation of uncertainties and conservatisms described later in this chapter.

The PWR contents also include high burnup fuel (maximum assembly average burnup exceeding 45 GWd/MTU) up to a maximum assembly average burnup of 60 GWd/MTU. High burnup fuel is treated as damaged fuel. Therefore, any limitations for damaged fuel and for the remaining package contents when damaged fuel is present also apply to high burnup fuel and to TSCs containing high burnup fuel regardless of the actual condition of the high burnup fuel. These conditions include the requirements that high burnup fuel is loaded in a damaged fuel can and that the additional cooling time needed for loading damaged fuel is applied to the contents of TSCs containing the high burnup fuel. High burnup CE 16x16 fuel is not an allowed package content since damaged CE 16x16 fuel is not allowed.

Decay heat is one of the properties of the spent fuel contents that the applicant used to characterize the allowable contents for the package. For BWR spent fuel, the maximum package decay heat is 22 kW, which equates to 253 W per basket location. The applicant proposed two decay heat limits for PWR spent fuel, depending on the thermal conductivity of the neutron absorber panels in the basket, 23 kW (622 W per basket location, including the NFH contribution) and 22 kW (595 W per basket location, including the NFH contribution). The shielding evaluation, as described in this chapter of the SER, is for the 23 kW PWR basket. The analysis remains acceptable or is bounding for the 22kW basket because the lower decay heat limit requires longer cooling times. These longer cooling times result in lower source terms than used in the analyses. For some spent fuel contents, the dose rate limits are more restrictive than the decay heat limits for either heat load basket, in which case, the source term for those contents in the 22 kW basket is the same as for those contents in the 23 kW basket. Unique differences for evaluations and contents limits for the 22 kW basket PWR contents, such as burnup, enrichment, and cooling time limits and differences in requirements for non-fuel hardware, are described in Section 5.4.10 of this SER.

5.2.2 Non-Fuel Hardware Contents

The PWR NFH contents may include irradiated and un-irradiated items. The irradiated NFH contents include control element assemblies (CEAs) and other reactor control components (all of which are collectively referred to as CEAs in this SER chapter), BPRAs, guide tube thimble plug devices (GTPDs), neutron sources and source assemblies (NSAs), and hafnium absorber assemblies. Individual components (e.g., rods) from these devices may also be loaded into the package; however, only the components from a single NFH device may be loaded with an assembly. CEAs may be loaded only in the nine central basket locations. Only one NSA is allowed in a TSC and may only be loaded in one of the nine central basket locations. With the exception of NSAs, PWR NFH contributes only to the gamma source in the package. NSAs contribute to both the neutron source and the gamma source. NSA content limits are tied to the limits for BPRAs and GTPDs depending on whether or not the NSA includes absorber rods. Only GTPDs without any type of absorber rods or water displacement rods may be loaded in the package. NSAs for CE fuel types (CE 14x14 and CE 16x16) are not allowed package contents.

While BWR fuel channels may also be considered NFH, the proposed contents include only zirconium-based alloy channels, activation of which is very limited because the quantity of impurities in zirconium-based alloys that can be activated is not significant. Thus, their contribution to dose rates is not significant, and the staff finds it acceptable to not include BWR channels in the NFH definition and the shielding analysis.

5.2.3 Greater-Than-Class C Waste Contents

Finally, the contents also include GTCC waste. This waste is restricted to solid, irradiated, and contaminated hardware, with fissile material less than Type A quantity and that does not exceed the limits in 10 CFR 71.15. This waste consists of items such as sections of core baffle plates and angles, baffle formers, lower core plates, and miscellaneous related hardware. The activity in the waste is characterized based on the measured activity of the three isotopes nickel-63 (Ni-63), iron-55 (Fe-55) and cobalt-60 (Co-60). The first two decay by beta emission and electron capture with some low energy x-rays and so contribute negligibly to package dose rates. Thus, for purposes of the shielding analysis, the waste is evaluated in terms of the specific activity of the Co-60.

A package may contain up to 55,000 lbs of GTCC waste. Other limits include a maximum decay heat of 1.7 kW, a 85,760 Ci total limit for Co-60, a maximum contents-average specific activity limit of 2.7 curie (Ci) Co-60 per pound, and a maximum localized peak specific activity of 16.1 Ci Co-60 per pound. The activity and contents-average specific activity limits are the most restrictive, with only 31,763 lbs of GTCC waste resulting in the total Ci of Co-60 at the specific activity limit. Specification of both a contents-average and a localized peak specific activity limit accounts for the variability of the specific activity of the different GTCC waste items that may be transported. Reasonable application of the concept of a localized peak will ensure that variability in the contents' specific activity is appropriately dealt with by package users, including variability from item-to-item and, for large components, variability from one area of the component to another (due to differences in the irradiation field at different areas of the component).

5.2.4 Source Calculation Method

5.2.4.1 Computer Codes

The applicant calculated the radiation source terms using the SAS2H module of the SCALE code, version 4.4. The applicant used the 44-group ENDF/B-V cross section library, which also includes ENDF/B-VI data for a limited number of nuclides. SAS2H performs one-dimensional depletion analyses for spent fuel with ORIGEN-S. While the cross section library and the code version are old, both have been used in shielding analyses for NRC-approved packages. Given the similarities between those packages and their contents with the proposed MAGNATRAN package and contents, the staff finds that use of this code, including version, and selected cross section libraries are acceptable for fuel at burnups up to 45 GWd/MTU.

The validation data for SAS2H and the chosen cross section library is quite limited for high burnup fuel. The applicant stated that there is nothing to indicate there will be trends in the analysis results for high burnup fuel. However, because of the limited nature of validation data at burnups higher than 47 GWd/MTU for PWR fuel, the applicant reduced the allowable heat load for the proposed high burnup fuel contents by 5%. So, for high burnup fuel, the minimum required cooling time is the more limiting of: 1) the cooling time that is necessary to meet the 2-meter normal conditions of transport dose rate limit, which the applicant reduced by different amounts for the different fuel types, and 2) the cooling time that is necessary to meet the reduced allowable heat load. For many of the high burnup contents, the reduced decay heat limit results in longer required minimum cooling times for the fuel. However, there are some high burnup contents that are still limited by dose rates and the minimum cooling times remained unchanged. Thus, the reduced heat load does not introduce any additional margin for these few high burnup contents in terms of meeting the applicant-imposed dose rate limits, which are less than the regulatory limits.

The staff reviewed the applicant's approach for addressing calculation of high burnup fuel source terms and the associated uncertainties. The staff also reviewed the documents and data referenced by the applicant. In terms of the conclusions in the documents referenced by the applicant, it is important to recognize that, as stated in Section 6 of NUREG/CR-7012[1], while no measurable dependence on burnup has been observed in the bias for calculated nuclide concentrations, "the bias associated with the application will change as the relative importance of different isotopes changes as a function of fuel enrichment, burnup, and cooling time". Thus, a more correct evaluation of uncertainties in depletion results for high burnup fuel should include this consideration. The staff does recognize that the applicant's approach (reducing allowable decay heat by 5%) is consistent with the approach that has been used in analyses for high burnup fuel in other packages and in dry storage systems. Considering these points, the staff included a 5% uncertainty for high burnup fuel analyses in its evaluation of uncertainties and conservatisms, as described in Section 5.4.9 of this SER. In this evaluation for high burnup fuel, the staff considered both damaged and undamaged fuel conditions. Based on considerations of the available validation data, the applicant's approach to address the uncertainties associated with estimating high burnup fuel source terms, and the staff's inclusion of this uncertainty in its evaluations, the staff finds the use of the identified code, including code version, and cross section library acceptable to calculate the high burnup fuel source terms for this application.

The applicant also used this code and cross section library to determine the gamma sources from the assembly hardware and PWR NFH. To do this, an amount of cobalt is added to the SAS2H model for irradiation during the depletion analysis. The resulting amount of Co-60 is used along with flux scaling factors for the different assembly regions where the assembly hardware and different parts of the NFH are located in the assembly (i.e., top and bottom nozzles, upper and lower plena, active fuel). The applicant scaled the results by the amount of cobalt that is estimated to be in the assembly hardware and the parts of the NFH in each assembly region. The flux scaling factors are given in Section 5.2.1 of the application. The staff finds these flux factors are acceptable because they are consistent with those used in the shielding analyses for other spent fuel packages and technical reports regarding the flux variation in fuel assemblies during irradiation. The staff's evaluation of the cobalt mass in the different hardware and NFH components is described in Section 5.2.5.2 of this SER.

The applicant used ORIGEN-S to calculate the GTCC waste source term by decaying the nuclides in the waste (see Section 5.2.3 of this SER) from an initial time when the waste was characterized to a point where the Co-60 specific activity meets the limit of 2.7 Ci/lb. The applicant used the resulting source spectrum in its shielding calculations. This is a valid use of the ORIGEN-S code as the code is designed to be able to perform decay calculations and provide energy spectra for radionuclides. Thus, the staff finds the applicant's use of the code in this manner to be acceptable.

5.2.4.2 Additional Method Features

In addition to the computer code, the applicant's method also includes their approach to axial blankets, axial burnup profile, axial source profile, and assembly classification.

The proposed spent fuel contents, both PWR and BWR, include assemblies with axial blankets. These blankets may be low-enriched, unenriched and/or annular blankets. Unenriched blankets are limited to a nominal length of 6 inches or less on each end of the assembly. The applicant did not propose a limit on the nominal length of the low-enriched or annular blankets, stating that these other blanket types would not cause the same source effects that have been postulated for unenriched blankets.

In its review regarding blanketed fuel, the staff took into consideration typical characteristics of these other types of axial blankets. The staff understands that 6 inches is a common length for axial blankets. Also, low-enriched blankets are usually at about 2% enrichments though they more recently may be up to enrichments of around 2.6%. Further, the axial burnup profiles for low-enriched blankets are not distinguishable from non-blanketed assemblies. Also, annular blankets are typically associated with integral fuel burnable absorbers. Based on its review, including these considerations for the different types of axial blankets and the expected impacts on depletion versus non-blanketed fuel and fuel with unenriched axial blankets, the staff has reasonable assurance that the any effects of these blankets is adequately captured by an analysis with unenriched blankets and that a maximum nominal length is only needed for the unenriched blankets.

The effects of the unenriched blankets result in a few percent increase in the source term versus un-blanketed fuel, which the staff included in its evaluation of uncertainties described in Section 5.4.9. The applicant uses an assembly average minimum enrichment and an assembly average maximum burnup. The presence of blankets impacts the calculation of both values. It also impacts the axial burnup profile. Including the blankets in the average enrichment lowers the enrichment throughout the fuel assembly, which acts to increase the source term. However, including the blankets in the average burnup also lowers the burnup throughout the assembly, particularly in the active enriched zone where the burnup will peak, which acts to decrease the source term. Staff evaluations indicate that the source term increases by a few percent for depletion calculations with the blankets explicitly modeled versus included in an averaged enrichment.

The axial burnup profile and axial source profile do not affect the calculation of the source term in the depletion calculation. However, they do affect the source term as it is used in the shielding model to calculate dose rates. The applicant used these two profiles to adjust the source term, specifically the axial distribution and total source strength in the shielding model. These profiles are described in Sections 5.3 and 5.4 of the application. The staff reviewed these descriptions. The peaking factors for both PWR and BWR burnup profiles and the application of the burnup profiles to determine the axial source profile are consistent with analyses the staff has seen for other approved spent fuel packages. Based on this consistency and the data used to derive the burnup and source profiles, the staff finds the profiles used in this application to be acceptable. Considerations for how the axial profiles are applied for damaged fuel are described in Section 5.8.10 of the application. Based on its experience, the staff finds these considerations to be reasonable and acceptable for treatment of damaged fuel source terms.

For spent fuel with burnups less than 30 GWd/MTU, the burnup profile will have a higher peak relative to the lower burned ends than for spent fuel with burnups greater than 30 GWd/MTU. However, the applicant applied the burnup profiles for burnups greater than 30 GWd/MTU to spent fuel with burnups less than this value (i.e., less than 30 GWd/MTU). The applicant provided information to compare different burnup profiles for PWR spent fuel with burnups as low as 5 GWd/MTU in Figure 5.3-5. Based on this figure, the difference in the peaks of the burnup profiles used in the analysis versus the profiles for these lower burnups is about 4% for PWR spent fuel. The applicant states that the peaking for these lower burnups is less than 10%. Based on this information, the staff finds that the applicant’s choice of burnup profiles is also acceptable for these lower burnups. The staff considers the effects of the burnup profile as an uncertainty. This uncertainty is included in the evaluation described in Section 5.4.9 below.

The applicant divided the spent fuel into different hybrid types based on key characteristics of the fuel. These characteristics include the fuel rod array size, fuel length, and maximum uranium mass loading. The hybrid types used for the shielding analysis do not necessarily align neatly with the assembly types that show in the contents specification in the tables in Chapter 1 of the application. Thus, it is not always readily apparent that the contents descriptions are adequately covered by the shielding analysis. Due to this, the staff performed a limited number of independent calculations for the contents at the specifications listed in the Chapter 1 tables and compared the source terms to the source terms for the hybrid assemblies used in the shielding analysis. This comparison indicates that the spent fuel contents as specified in Chapter 1 are adequately covered by the hybrid assembly types used in the shielding analysis. Thus, the staff finds this part of the evaluation to be acceptable.

5.2.5 Gamma Source

The gamma source for the spent fuel TSCs is the combination of the spent fuel gammas, activated assembly hardware, activated NFH (for PWR fuel only), and secondary gammas from the neutron-gamma reactions in the package. For GTCC waste TSCs, the source is the Co-60 nuclide from activation of the items that are included in the waste.

5.2.5.1 Spent Fuel Source Calculation

The calculation of the spent fuel source term is fairly straightforward. The applicant re-bins the gamma energy spectrum from the default energy structure of the SAS2H output to the default 22-group MCBEND structure. However, the applicant does this within the SCALE code, which is an appropriate way to do the restructuring of the gamma spectrum. The applicant provided examples of the gamma spectra for some of the bounding spent fuel source terms, which are displayed in terms of particles per second per assembly in the 22-group energy structure.

5.2.5.2 Assembly Hardware and Non-Fuel Hardware Source Calculation

The calculation of the activation source is not as straightforward and affects the assembly hardware and the PWR NFH source. This calculation involves consideration of the neutron flux at the different axial zones for the assembly hardware and the portions of the NFH, the mass of materials in those different axial zones, the cobalt impurity levels in the hardware and NFH materials, and operation considerations for the NFH. All of these considerations are handled outside of the depletion calculation done with SAS2H. A mass of cobalt is included in the SAS2H spent fuel depletion calculations for a hybrid assembly type. This can be any mass (e.g., 1 gram or 1 kilogram) so that the curies of Co-60 per gram of cobalt can be determined. The applicant scaled the resulting Co-60 source based on the preceding considerations for the different axial zones for the assembly hardware and the NFH. For CEAs, the source includes the activation of the silver (Ag), indium (In), and cadmium (Cd) absorber material. Thus, similar appropriate considerations are required for calculation of the activation source of CEA components for these elements in addition to cobalt.

The applicant's analysis of the assembly hardware and the NFH breaks down these items by different axial zones (i.e., the different assembly regions listed in Section 5.2.4.1 of this SER). The neutron flux varies depending on the zone. The staff reviewed the applicant's axial flux factors for the different assembly types and finds they are consistent with the factors used in the analyses for other spent fuel packages. The applicant's method for determining the source in each axial zone is also consistent with that done in other spent fuel package analyses and is an acceptable method. However, the staff had concerns with input parameters for the method and the application of the results for some NFH to other NFH, specifically the assumed cobalt impurity levels and NFH hardware masses, as explained in the following sections.

5.2.5.2.1 Cobalt Impurity Levels

One concern is the assumed cobalt impurity levels in the steel and Inconel components. A significant NFH source is Co-60; so, the cobalt level could impact compliance with dose rate limits. The applicant assumes the impurity level is 0.8 g cobalt per 1 kg of steel and Inconel. While this is consistent with assemblies and NFH manufactured after the late 1980s, it is not consistent with fuel and NFH made prior to that time. For fuel and NFH made prior to that time, PNL-6906[2], indicates that the cobalt levels in steel and Inconel could be up to 2 to 3 times as much. The applicant addressed the appropriateness of the 0.8 g cobalt per 1 kg of steel and Inconel for assembly hardware, noting that design basis fuel is either short cooling time-low burnup fuel or long cooling time-high burnup fuel. The first type will have been manufactured after the late 1980s. Dose rates for the second type are dominated by the fuel source. The data provided by the applicant as part of its shielding calculation package files supports these conclusions and is consistent with the staff's expectations. However, the staff also included the impacts of increasing the cobalt levels for both the assembly hardware and the NFH as part of its evaluations of uncertainties described in Section 5.4.9 below. This was done in part because the cooling times were extended to allow NFH to be loaded with the assembly for at least some of the assembly types.

5.2.5.2.2 Non-Fuel Hardware Masses

The staff used DOE/RW-0184[3] to review the hardware masses for the different zones for the NFH contents. Although this document is dated, the information in it provides a good reference point for hardware masses for the different types of NFH. Based on this review, the staff identified that some of the applicant's selected steel and Inconel masses for different assembly types' NFH and AgInCd masses for CEAs were not bounding. Additionally, the applicant did not explicitly evaluate NSAs, instead using the other NFH types to bound the source from NSAs. The staff's review also identified that the masses for NSAs were not always bounded by the masses in the other NFH types. The staff considered these differences as part of its evaluation of uncertainties described in Section 5.4.9 of this SER.

5.2.5.3 Non-Fuel Hardware Burnup and Cooling Time Limits

The applicant did not propose any burnup and cooling time limits that are unique for NSAs. Instead, the applicant proposed that the GTPD limits be applied to NSAs without absorber rods and that the BPRA limits be applied to NSAs with absorber rods. The staff's review, however, identified that while GTPDs do not have material in the active fuel zone of an assembly, NSAs do, including NSAs that do not have absorber rods. This fact further illustrates how NSAs are not always bounded by the other NFH types. However, the staff considered these differences in its evaluation of uncertainties (Section 5.4.9 of this SER). Based on that evaluation and the information provided by the applicant (e.g., proposed limits on numbers and locations of NSAs), the staff finds use of the GTPD and BPRA limits for NSAs to be acceptable. As noted previously, NSAs for CE fuel types are not authorized package contents; therefore, burnup and cooling time limits are not provided for these NSAs.

The applicant also proposed limits for BPRAs and GTPDs in terms of maximum burnup (or exposure) and minimum cooling time or maximum activity in terms of curies of Co-60. BPRAs and GTPDs, and by extension non-CE NSAs, that meet either limit may be loaded into the package. In its review, the staff performed a simple confirmatory calculation that indicates that the proposed activity limits are consistent with the burnup and cooling time limits for these NFH types. This calculation used the applicant's cobalt impurity levels and material masses in the different axial zones. Given this result and that the analysis method is based on using steel and Inconel masses that are bounding for these NFH types on an axial zone basis, the staff finds that this approach is acceptable.

5.2.5.4 GTCC Waste Source Calculation

The staff's review of the applicant's characterization and calculation of the GTCC waste gamma source is described in Sections 5.2.3 and 5.2.4.1 of this SER.

5.2.6 Neutron Source

The neutron source for the package contents includes the fuel and the NSAs (PWR only).

5.2.6.1 Spent Fuel Source Calculation

As with the fuel gamma source term, the calculation of the fuel neutron source term is fairly straightforward. The applicant re-bins the energy spectrum from the default energy structure of the SAS2H output to the default 28-group MCBEND structure. The applicant does this within the SCALE code, which is an appropriate way to do the restructuring of the spectrum. The applicant provided examples of the neutron source spectra for some of the bounding spent fuel source terms, which are displayed in terms of particles per second per assembly for each energy bin.

5.2.6.2 Non-Fuel Hardware Source Calculation

The applicant did not explicitly evaluate the neutron source from NSAs other than to note that the magnitude of the source is similar to that of an irradiated assembly. The applicant's basis for not explicitly evaluating the NSA source is, at least in part, that NSAs are restricted to interior TSC basket locations and only one NSA per TSC.

Based on its experience, the staff finds that the characterization of the NSA neutron source is acceptable. NSAs include primary and secondary sources. Secondary sources have also been referred to as regenerative sources as they rely on activation of the materials in the source to generate photoneutrons. These sources often have very short half-lives and decay quickly after removal from a reactor. Further, activation in the TSC is such that these kinds of sources will contribute negligibly to the total neutron source. Primary sources do not require activation to generate neutrons and can have significant half-lives. These kinds of sources are of similar magnitude to a fuel assembly neutron source. Therefore, consideration of a NSA with a source equivalent to a single assembly will capture the different NSA types that a licensee may wish to transport.

5.2.6.3 Greater-Than-Class C Waste Source Calculation

The applicant did not calculate a neutron source for the GTCC waste contents since the amount of fissile material in a package containing GTCC waste is limited to amounts that do not exceed the limits in 10 CFR 71.15. The GTCC waste contents may not include any other kind of neutron source material. In its reviews of other packages with similar contents limits, the staff has found that neutron sources from such materials are only expected to result in dose rates that are a small fraction of the respective regulatory limits. Based on this experience and the significant margins to the limits for the MAGNATRAN loaded with GTCC waste, the staff finds it acceptable not to evaluate a neutron source term for the GTCC waste contents.

5.3 Shielding Model

Section 5.5 of the application includes most of the model description. The description focuses on undamaged spent fuel contents as well as the packaging for both normal conditions of transport and hypothetical accident conditions. Additional details on the model for the contents and the packaging are included in Section 5.8 of the application for the different contents (e.g., undamaged PWR and BWR fuel, damaged PWR fuel, GTCC waste). The staff reviewed the shielding model description as described in the following sections. As part of its review the staff compared the model to the licensing drawings in Chapter 1 of the application and conferred with the structural and thermal reviewers to ensure the shielding model is consistent with or bounding for the package as designed and as it behaves under normal conditions of transport and hypothetical accident conditions.

5.3.1 Configuration of Source and Shielding

5.3.1.1 Source Configuration

The package uses two lengths of TSC. The limits on package contents include conditions for what can be loaded in the different lengths of TSC. BWR/2-3 assemblies, corresponding to hybrid BWR assembly types 7a, 8a and 9a, can only be loaded in the shorter length TSC. The BWR/4-6 assemblies, corresponding to the remaining hybrid BWR assembly types, are loaded only in the longer length TSC. Undamaged PWR fuel can only be loaded in the shorter length TSC except for CE 16x16 type fuel, which may be loaded into either length of TSC. Damaged PWR fuel and GTCC waste may only be loaded in the shorter length TSC. The staff reviewed the applicant's models, including a variety of the sample input files provided as part of Calculation Package No 71160-5508, Revision 0 and finds that the models are consistent with these proposed CoC conditions.

5.3.1.1.1 Non-Fuel Hardware Configuration

The proposed contents include allowance for the loading of either a complete NFH assembly or individual components of a NFH assembly with a spent fuel assembly. This means, for example, that a PWR assembly may contain just the individual absorber rods from a BPRA and not the entire BPRA. However, the individual NFH components must all be from the same NFH assembly. Also, since the components may not extend the whole length of the assembly, their location may shift along the assembly length. The shielding model only includes complete NFH assemblies.

The applicant qualitatively evaluated the condition of the axial movement of loose NFH rods or rodlets, focusing on changes in the top zones since the package body top region produces maximum dose rates. The applicant determined that the ends of the rods/rodlets in the upper assembly axial zones represents less material and therefore less source when compared to the hub structure of the NFH despite the rods/rodlets having higher activation than the hub structure. The staff recognizes the material masses will be less as described by the applicant. However, less material also means less self-shielding. Therefore, the staff compared the source from the different axial zones of the NFH on a per inch basis. For some NFH types (for NFH designs used with some of the hybrid fuel assembly types), this comparison indicated that the dose rates would be less than for the modeled full NFH assembly. The comparison indicated that for others, the dose rates may increase over the modeled configuration. For those situations, the staff considered the estimated change in dose rates as an uncertainty and included that in its uncertainty evaluation described in Section 5.4.9.

5.3.1.1.2 Fuel Burnup and Source Profiles

The burnup and source profiles affect the physical distribution of the spent fuel source within the active fuel zone as well as the total source strength that would be entered into the shielding model. The staff reviewed a selection of the applicant's input files and determined that the modeled spent fuel neutron and gamma source's physical distributions account for the impacts of the burnup and source profiles described for undamaged fuel. The applicant adjusted the total source strength to account for the impacts of these profiles. The applicant applied the fuel source (gamma and neutron) distribution to the volume defined by the assembly cross-section area and the active fuel height. The applicant applied the activated hardware and NFH sources uniformly over the volumes of the fuel assembly's axial fuel zones. For NFH types that are limited to only certain TSC basket locations, the applicant only applied their sources to assemblies in the allowed locations.

5.3.1.1.3 Damaged Pressurized-Water Reactor Spent Fuel Configuration

The applicant's analysis for damaged fuel contents addresses two scenarios: damaged fuel migrating into the end hardware zones (plenum and nozzle areas of the assembly); and damaged fuel from one area of the active fuel region migrating into another area of the active fuel region. The applicant combined the results of these two scenarios to cover a third scenario of active fuel collecting in both the end hardware and active fuel regions. The damaged fuel source in each assembly region is based on the damaged fuel filling all the interstitial void space within each region. In the active fuel region, this results in a source term that is about three times the undamaged source term in this region. The active fuel region source retains its axial profile. The applicant modeled damaged fuel in the end hardware regions at the peak axial burnup. Section 5.8.10 of the application describes the details of the source model for the damaged fuel analysis.

As part of its review of the damaged fuel analysis, the staff considered the descriptions of the application of the burnup profile and the increased source and source material for the fuel source added to the different axial zones of the assembly. Staff experience is that damaged fuel analyses based on maximizing dose rates do not completely fill the interstitial voids within the different axial zones. Also, these analyses indicate dose rate increases versus undamaged fuel dose rates of about 20% to 30%. Competing effects of increased source and increased self-shielding due to increased source material influence the determination of the configuration of material and source resulting in maximum dose rates. The staff considered this experience in its evaluation of the applicant's treatment of damaged fuel by applying a similar dose rate increase to undamaged fuel in package models using the same assumptions as for the damaged fuel. A comparison of the dose rates from that evaluation with the applicant's dose rates indicates that the applicant's method is reasonable. Based on these considerations, the staff finds the applicant's modeling of the damaged fuel source to be acceptable.

5.3.1.1.4 Greater-Than-Class C Waste Configuration

The GTCC waste limits are based on a maximum activity and maximum specific activity limits (see SER Section 5.2.3). Thus, the GTCC source is modeled at the maximum contents-average specific activity throughout the cavity of the GTCC waste liner in the TSC. Though the applicant did not model the waste at the maximum localized peak specific activity, the applicant addressed the effects on dose rates in a bounding manner (see SER Section 5.4.8).

5.3.1.2 Transportable Storage Canister Configuration

The applicant shifted the TSC and the contents within the respective cavities to maximize the dose rates for the different regulatory limit locations around the package. For dose rates at and above the top axial end of the package and for most of the radial dose rates, the applicant shifted the TSC and the contents to the top end of the package and TSC cavities. This shift is appropriate for the radial dose rates because the package's neutron and lead shielding do not cover several inches of the upper part of the package cavity. Thus, radial dose rates are expected to peak at this area of the package, which is demonstrated by the applicant's analysis. The applicant shifted the TSC and its contents toward the package and TSC base to calculate dose rates at and below the package's axial base and radial dose rates around the rotation trunnions and the gap between the neutron shield and the bottom impact limiter. The staff confirmed these shifts were done as part of its review of the applicant's input files. The staff finds the shifts are appropriate because they result in package and source configurations that maximize dose rates for the different package features.

The applicant used two models of the TSC in the analysis. The models differed in crediting the neutron absorber plates that are part of the spent fuel TSC baskets. For the undamaged analysis, the applicant's model does not credit the absorber plates. However, the model for the damaged PWR fuel contents does credit the absorber plates. Neglecting these absorber plates is a common practice for shielding analyses. The cavity of the package is dry, so the effectiveness of the absorbers should be minimal for reducing neutron dose rates. However, the staff finds that including the absorber plates in the model is acceptable because they are in fact in the actual TSC basket.

The applicant also modeled the TSC with the stainless steel lid with carbon steel shield plate instead of the all stainless steel lid and shield plate. The basis for this selection is the lower density of the carbon steel versus stainless steel. Thus, the lid in the model will be less effective in terms of shielding. Also, for the GTCC TSC, the applicant's model includes the GTCC waste liner.

When the short TSC is used, a cavity spacer is attached to the underside of the package lid. This spacer maintains the short TSC's position within the package cavity. The applicant's shielding model credits the spacer only for maintaining the TSC's position. The applicant conservatively omitted the spacer from the model. Based on the structural and thermal evaluations, the staff finds that crediting the spacer's function (maintaining the short TSC's position) is acceptable.

The applicant's analysis does not include tolerances for the TSC and its internals (e.g., the basket) that are credited in the models. The applicant stated that the tolerances on the components of the TSC that result in reduced component thicknesses are such that the impact is negligible on dose rates. The staff considered the tolerances described for the TSC components and evaluated the potential increase in dose rates from application of these tolerances. The staff's evaluation confirmed that the impact of the identified TSC component tolerances would be small and that conservatisms in the analysis model are sufficient to counter the impact of these tolerances. Thus, the staff finds it acceptable that the analysis models do not include the TSC components' tolerances.

5.3.1.3 Packaging Configuration

The package design includes a number of features that act as radiation streaming paths or as areas of reduced shielding. The impact limiters have holes for the attachment rods. The impact limiters also have an annular void area between them and the package lid and base. There is a gap between the neutron shield and the bottom impact limiter. The rotation trunnions are a neutron streaming path in the neutron shield. There are also copper fins that are between the different neutron shield assemblies. For neutron shield assemblies that are bolted to the package body, the bolt locations are streaming paths. There are areas in the neutron shield assemblies where expansion foam and thermal insulator material exists instead of neutron shield material. The foam and insulator material also are on radial surfaces of the neutron shield assemblies; so, the neutron shield material does not fill the entire cavity of the shield assembly in the radial direction. Also, as noted, the neutron shield and the lead gamma shield do not extend the whole axial length of the package cavity. This area is also not covered by the upper impact limiter; therefore, the package surface is the surface of the steel body (the top forging's outer surface).

5.3.1.3.1 Impact Limiters

The impact limiters and their streaming paths are only relevant for the normal conditions of transport analysis. The applicant's normal conditions of transport models significantly reduce the size of the impact limiters versus their actual size. Radially, the impact limiters do not extend beyond the radius of the neutron shield assemblies. Axially, the impact limiters only extend to the length of the outermost radial surface's length. The staff considered the impacts of the normal conditions of transport conditions in 10 CFR 71.71 in its review of the model's impact limiters. Based on the structural evaluation of those impacts, the staff finds the model to be acceptable and conservative since, in at least some of the drop orientations, the amount of impact limiter material removed in the shielding model is much more than the amount of crush due to the normal conditions of transport tests.

The applicant's models for the spent fuel contents include the attachment rod holes, though they are modeled to be smaller than indicated in the drawings. The staff's evaluation of the dose rates and the analysis described later in this SER chapter include consideration of the sizes of the holes. Additionally, the applicant modeled the annular gap between the impact limiters and the package lid and base as existing over the whole surface between the impact limiters and the ends of the package (see Figure 5.5-1 of the application). The applicant's GTCC model does not include these voids in the impact limiters; however, given the significant margins in the axial dose rates for the GTCC package, the staff finds that to be acceptable.

5.3.1.3.2 Neutron Shield Assemblies Region

The applicant explicitly modeled the copper fins and the rotation trunnions. The applicant also included the gap between the neutron shield and the bottom impact limiter as well as the bolts and bolt penetrations in the neutron shield assemblies that are bolted to the package body. The applicant modeled the expansion foam and thermal insulator as void. Thus, the staff finds the models adequately capture these design features, which contribute to radiation streaming. The outer surface of the neutron shield assemblies is rounded in the model. Therefore, the amount of material is further slightly reduced versus reality for some neutron shield assemblies. However, for some areas, the amount of material is slightly increased versus the actual design. This effect should be small, given the difference is only a slight increase versus the actual design and other model features that are conservative in nature.

5.3.1.3.3 Hypothetical Accident Conditions

The hypothetical accident conditions models differ from the normal conditions of transport models in a few ways. First, the impact limiters are assumed to be gone. The removal of the impact limiters is an acceptable model assumption since the impact limiters experience significant damage due to the hypothetical accident conditions in 10 CFR 71.73. Removal of the impact limiters from the model alleviates the need to justify any credit for impact limiters under hypothetical accident conditions.

Additionally, the applicant concurrently modeled both radial and axial lead slump. Axial slump is applied to both the top and the bottom ends of the lead shielding. The amount of slump is based on the volume of space the applicant has determined to exist in the lead shield cavity after the lead has been poured and cooled. The applicant applied available void space to each type of slump, which results in a model with three times the amount of possible slump. The staff finds this approach to be acceptable because it enables the applicant to address the impacts of the different ways the lead may slump, applying the total amount of slump possible to each slump location.

The applicant's models also include a 6-inch diameter hole through the entire thickness of the neutron shield in line with the radial lead slump. This hole captures the effects of a puncture test on the neutron shield. Thus, the applicant's model captures the cumulative impacts of both changes to the package shielding. The staff reviewed the structural impacts of the hypothetical accident conditions tests and finds that these shielding model features are adequate to capture those effects. The applicant credits the neutron shield in the hypothetical accident conditions models, though with degraded material properties. The staff's evaluation of this feature is discussed in Section 5.3.2.4 below.

Sections 2.11.2 and 2.11.3 of the application discuss deformation of the upper assembly hardware for BWR assemblies and the RCCA and spacer for PWR assemblies, respectively, due to a 30-foot axial end drop for the package. The applicant's shielding analyses for hypothetical accident conditions did not address this deformation and the accompanying axial shift of the fuel assembly toward the top end of the package and TSC cavities. However, the staff performed some simple evaluations to determine whether or not the spatial compaction of the radiation source terms for the affected assembly and RCCA items and the axial shifting of these items along with the rest of the assembly and RCCA components could result in dose rates that exceed the limits for hypothetical accident conditions. The staff's evaluations considered the effects on both axial and radial dose rates. The effects on dose rates are at locations around the package where the applicant's analyses show substantial margins to the limits. Based on its evaluations, the staff finds that the axial dose rates would remain substantially below the regulatory limits. The staff also finds that there would still be significant margin to the regulatory limits for the affected radial dose rates, though they would be noticeably reduced from what the applicant's current analysis indicates. Thus, given the remaining margins in the staff's evaluation and the other considerations identified in this section of the SER, the staff finds that the applicant's shielding analysis for hypothetical accident conditions is adequate for the package design and contents as they are currently defined in the application though it does not include the effects of the deformation discussed in Sections 2.11.2 and 2.11.3 of the application.

5.3.1.3.4 Packaging Tolerances

The applicant's shielding models also account for package tolerances as described here and in Section 5.1.1 of this SER. The applicant implemented tolerances in a way that reduces the thicknesses of the package materials. The applicant did not apply tolerances to the neutron shield materials because, it argued, that modeling the thermal insulator and expansion foam on the neutron shield assemblies' radial surface as void is sufficient to cover any impacts of the dimensional tolerances for the neutron shield material. As noted, the applicant did not apply tolerances to the TSC components. The staff reviewed the package licensing drawings and the shielding models with respect to package tolerances (see Section 5.1.1 of this SER). Based on this review the staff finds that the models use package dimensions that are consistent with, or acceptable for, the application of the package tolerances in the drawings and that minimize the package's shielding capability.

5.3.2 Material Properties

5.3.2.1 Greater-Than-Class C Contents

The applicant modeled GTCC waste as stainless steel that uniformly fills the GTCC waste liner cavity. The applicant reduced the density to a value that is consistent with the waste mass that yields the maximum Co-60 activity, which is 32,000 lbs. Using the applicant's package drawings and the mass of waste, the staff confirmed that the applicant used an appropriate mass density for the GTCC waste contents and correctly specifies the material properties for stainless steel. With regard to the mass needed to achieve the maximum Co-60 activity at the specified maximum contents-average specific activity limit, the staff notes that the actual mass of waste material is a little less than 32,000 lbs. However, staff evaluation indicates that the effect on dose rates is small (less than 1%). Therefore, the staff finds the modeled waste mass is acceptable.

5.3.2.2 Spent Nuclear Fuel Contents

The applicant homogenized the fuel assembly materials over the volume of the assembly for each axial zone of the assembly. The applicant adjusted the densities of the materials accordingly. The applicant modeled the fuel as fresh UO_2 fuel. Staff analyses for other applications have indicated that modeling the fuel as fresh fuel versus irradiated fuel may reduce dose rates by a few percent due to fresh fuel having greater self-shielding. For undamaged fuel, the applicant modeled the fuel at 5 wt% uranium-235 (U-235). For the damaged fuel analysis, the applicant reduced the enrichment to 2 wt% for both the damaged fuel and the undamaged fuel included in the analysis. Since neutron dose rates are a substantial fraction of the total dose rate and the undamaged fuel models produced a conservative sub-critical multiplication, the inclusion of neutron absorber plates and the reduced enrichment resulted in significant reduction in dose rates for many of the spent fuel contents. Even at 2 wt%, the modeled sub-critical multiplication is still conservative. The staff considered both of these aspects of the materials properties included in the models. Section 5.4.9 below provides information on how these effects were included in the staff's evaluation.

The staff reviewed the materials properties as they appear in the applicant's input files. The staff calculated densities and mass fractions for both undamaged and damaged fuel using the assembly descriptions in Calculation Package Nos. 1160-5001, Rev. 0 and 71160-5002, Rev. 0 along with the description of the damaged fuel modeling in Section 5.8.10 of the application. The staff compared its calculated properties with those in the input files. The staff's calculated values showed good agreement with the properties in the input files. Therefore, the staff has reasonable assurance that the applicant's calculations use appropriate materials properties, including densities and mass fractions, for the spent fuel contents for fresh fuel compositions. Materials properties were added to the appropriate assembly axial zones for NFH calculations.

5.3.2.3 Package Materials Except Neutron Shield

The applicant modeled the lead shield as 100% pure lead. This introduces a small uncertainty in the gamma dose rates for the package as the lead purity may be as low as 99.9% according the specifications in the drawings. As noted, the applicant modeled the thermal insulator and the expansion foam in the neutron shield assemblies as void; therefore, no review of the properties of these materials was necessary in terms of the shielding analysis. The applicant also did not model the aluminum cooling fins. The applicant modeled the impact limiters as entirely balsa wood. The staff confirmed that the input models use material properties for balsa wood that are consistent with properties defined in the SCALE code manual and are acceptable. Other than the neutron shield material, all other items were modeled as stainless steel or carbon steel, as appropriate.

5.3.2.4 Neutron Shield Material

The neutron shield is a proprietary material called NS-4-FR. The licensing drawings, specifically Drawing No. 71160-502, Rev. 6P, specify important properties of the neutron shield material, including the minimum B_4C content, minimum hydrogen content, and the minimum overall material density. The staff confirmed that the input files for normal conditions of transport model the NS-4-FR with properties that are consistent with the specifications in the design drawings.

The neutron shield is credited in both the hypothetical accident conditions and normal conditions of transport models. To account for changes to the shield material after the hypothetical accident conditions tests, the applicant modeled the material at half its mass density and with the hydrogen, nitrogen and oxygen components removed from the material. The applicant stated that modeling the material with these properties for the hypothetical accident conditions was conservative based on thermal tests of the material. The applicant indicated that these tests had shown charring due to a fire to occur at depths up to small fractions of an inch in the material.

In its review of these tests, the staff found that the most relevant test report is the GA-A19897 report, which includes testing of the same material that is used in the MAGNATRAN package. The staff also found that the configurations of the neutron shielding material in the tests are similar to the configuration for the MAGNATRAN neutron shield material for a hypothetical accident conditions thermal test after a puncture test. The staff also identified that the thermal test conditions in the report were adequately similar to the hypothetical accident conditions thermal test conditions (e.g., the temperatures). Thus, the staff finds that the test results are appropriate to use for determining the behavior of the MAGNATRAN neutron shielding. The staff also identified in the test reports the changes in the neutron shield properties reported as a result of the tests and used these reports to develop confirmatory analyses that included reduced thickness of the neutron shield and an enlarged puncture in the neutron shield versus those modeled by the applicant. The results of these confirmatory analyses indicated that dose rates would increase by a few percent, still leaving significant margins to the hypothetical accident conditions dose rate limits. The descriptions of the neutron shield material from core samples taken from the tested material (away from the puncture area) also indicate that a significant thickness of the material does not, at least by visual examination, change as a result of the thermal test. This indicates that some uniform amount of neutron shielding material would remain throughout the neutron shield compartment after a hypothetical accident conditions thermal test. Thus, based on its review, including the confirmatory analyses, the staff finds the applicant’s crediting of the neutron shielding with the material properties used in the hypothetical accident conditions analyses to be acceptable. As already stated, the staff notes that there is significant margin to the hypothetical accident conditions dose rate limits. Thus, while some neutron shield material is needed to meet the limits (based on staff's independent evaluations), the margins provide additional assurance that the package can meet the hypothetical accident conditions dose rate limits.

The staff also considered the structural impacts of the hypothetical accident conditions tests. Other than the puncture test damaging the neutron shield assemblies, which the hypothetical accident conditions models address, the staff has reasonable assurance that neutron shield assemblies will not experience greater damage than was assumed in the hypothetical accident conditions shielding models for the puncture test. This includes the loss of neutron shield assemblies that are not bolted to the package body. This finding is based on the configuration of the package, with these neutron shield assemblies being held in place by the copper fins and the adjacent assemblies, which are bolted to the package body. It is also based on considerations of the orientations of the different hypothetical accident conditions tests that could challenge the package's configuration for retaining the assemblies and the kind of damage and the extent of that damage that those tests could cause.

5.4 Shielding Evaluation

5.4.1 Computer Code

The applicant performed the shielding analysis using the MCNP6 code. MCNP is a three-dimensional, continuous-energy Monte Carlo code that can be used to determine dose rate fields for complex and non-symmetric configurations and geometries. The MCNP code has been used extensively in radiation transport calculations by a wide array of users. Thus, it is a well-supported and validated code. Based on the code's capabilities and its validation and usage history, the staff finds that it is an acceptable code for calculating dose rates for the present application. The staff's evaluation of the other codes used in the analysis for calculating source terms has previously been described (see Section 5.2.4.1).

5.4.2 Direct Calculation and Response Function Methods

The applicant uses two approaches to calculate dose rates, referred to as the direct calculation method and the response function method. As implied in the method's name, the direct calculation method uses MCNP to calculate the dose rates for the full source definition (i.e., the complete energy spectrum and total source strength are included in the input file) for the package contents. Thus, the code does the entire calculation to determine dose rates. The applicant used this method for calculations to validate the response method and for calculating dose rates for packages containing damaged PWR fuel, reconstituted PWR assemblies, and GTCC waste.

The applicant uses a response function method for the majority of the shielding analysis. In this method, the MCNP code is used to determine the dose rate per starting particle for each

- source type,
- source location,
- package configuration (including material property changes of the packaging or the source),
- source energy bin in the energy spectrum, and
- detector location of interest.

The results of these calculations are collected and used to determine the dose rates for the sources of different package contents through simple multiplication of the relevant response functions with the contents' source spectra.

For the MAGNATRAN package, the applicant calculated a set of response functions for each hybrid assembly type. The applicant assumed uniform loading of assemblies in terms of the assembly type, burnup, enrichment, and cooling time. Therefore, the same source is modeled in all TSC basket locations. The applicant's calculations are limited to gamma and neutron energies that capture 99% of the dose rate from each of these particle types. These two aspects of the method greatly reduced the necessary number of MCNP calculations. Furthermore, the different dose rate locations of interest are all included in the same calculation, which further reduces the number of calculations by producing several response functions with a single calculation. Calculations are performed for each assembly axial zone, for each particle type and each particle energy associated with each assembly axial zone. The applicant performed calculations for both normal conditions of transport and for hypothetical accident conditions, since the package properties differ between the two conditions. Also, the applicant performed additional calculations to determine the response functions for PWR NFH. The NFH response function calculations credit the presence of the NFH materials.

This method for analyzing dose rates is not uncommon for spent fuel packages and spent fuel dry storage systems. As long as the method is used in a way that correctly accounts for the various factors that influence the calculated response functions, the staff finds it is a valid analysis method. Thus, in its review, the staff considered the description of how the response functions were developed to ensure that the applicant had calculated and used appropriate response functions for the different package contents and the different package configurations. Based on that review, the staff has reasonable assurance that the applicant did calculate and use appropriate response functions. The staff noted the small difference in dose rates due to the truncation of the source spectra considered in the analysis and included it in the staff's evaluation of uncertainties described in Section 5.4.9.

5.4.3 Detectors

The applicant used surface detectors in its analysis, both for radial and axial dose rates. The axial surface detectors are located at the package's axial end surfaces (bottom and top) and at 1 foot, 1 meter, 2 meters, and 4 meters from each axial surface. For the normal conditions of transport analysis, the surface is the impact limiter, whereas in the hypothetical accident conditions analysis, the surface is the package lid and base. The extent of the detectors increases with greater distance from the package surface. This is appropriate to ensure dose rates from different aspects of the package are adequately captured. The axial surface detectors are segmented radially. Thus, the radial variations in the dose rates can be determined.

5.4.3.1 Axial Detectors and Streaming Paths

The staff reviewed the applicant's definitions of the axial surface detectors. The staff finds that the detectors include locations relevant for determining compliance with regulatory dose rate limits for both normal conditions of transport and hypothetical accident conditions. The staff considered the sizes of the radial segments of the detectors and determined that they were acceptable. This determination is based on consideration of practical sizes of actual detectors and the expectations regarding detectability of radial variations at the different distances from the package.

The staff does note, however, that the segmentation should have also considered azimuthal variations too. This is because the impact limiters have streaming paths, namely the attachment rod holes that are features with limited radial and azimuthal size. Additionally, for TSCs containing damaged fuel, the damaged fuel is limited to four basket locations. Thus, azimuthal segmentation is necessary to ensure identification of the maximum dose rates on the package end surfaces and at 2 meters from the package surfaces. These holes can be significant streaming paths for neutrons. Thus, the staff evaluated the potential impact on dose rates of these streaming paths for both undamaged and damaged fuel contents. NSAs are considered too, but only as an impact on the peak dose rates. With an NSA present in the package, the peak axial dose rates will still meet the limits with large margins. NSAs should have minimal impact on dose rates at the attachment rod holes since only one is allowed in a package and the allowed basket locations are not near the rod holes.

Initially the staff considered estimating the dose rates at the rod holes in the impact limiters by using the dose rates calculated for hypothetical accident conditions. The staff considered this an option, though a fairly conservative one, because the source term for both normal conditions of transport and hypothetical accident conditions was the same for the top dose rates. The source term for the bottom dose rates was also the same between the two conditions. The staff looked at the hypothetical accident conditions neutron dose rates for the detector radial segments that covered the locations of the rod holes and accounted for the differences in axial positions of the hypothetical accident conditions detectors versus the normal conditions of transport detectors. The staff added these neutron dose rates to the normal conditions of transport dose rates at the radial segments covering the rod holes. The surface dose rates still met limits; however, the 2-meter dose rates exceeded the limits; this was still true for the bottom dose rates even when accounting for the over-prediction in sub-critical multiplication in the undamaged analysis.

Thus, the staff considered whether or not there might be another, more realistic, method to estimate the dose rates for the rod holes. The staff selected the applicant's mesh tally analysis of the rotation trunnions as a possible option. The trunnions are larger in diameter, but the configuration is similar to the rod holes. Both will have steel (though the holes are modeled as simply void), and both are neutron streaming paths. For this evaluation, the staff used the relative difference between the trunnion dose rates from the mesh tally analysis and the dose rates averaged around the package at the location of the rotation trunnion. The staff then increased the bottom and top surface and 2-meter dose rates by these relative differences (package surface and 2 meters). This evaluation indicated that there were significant margins to all the dose rate limits for both undamaged and damaged fuel cases. Given the similarities in the configurations of the rotation trunnions and the impact limiters' rod holes, and the results of this evaluation, the staff finds there is reasonable assurance that dose rates due to streaming through the rod holes will not result in dose rates that exceed the normal conditions of transport regulatory limits.

5.4.3.2 Radial Detectors and Streaming Paths

The staff reviewed the definitions of the radial surface detectors. Similar to the axial surface detectors, the radial detectors are located at the package surface and at distances of 1 foot, 1 meter and 2 meters from the package surface. The latter location, however, is at 2 meters from the vehicle edge. The applicant defines the vehicle width to be 124 inches. The axial extent of these surface detectors increases with increasing distance from the package to ensure package dose rates are adequately captured all around the package. These surface detectors are segmented axially to enable calculation of the changes in dose rate along the surface of the package and identify the maximum dose rates. The staff reviewed the sizes of the axial segments of the radial detectors and determined that they were acceptable based on consideration of practical sizes of actual detectors and expectations regarding the ability to detect axial variations at the evaluated distances from the package. However, the staff finds they are inadequate to evaluate the azimuthal variations in dose rates due to streaming paths and other azimuthal variations in the package shielding and non-uniform contents loading (e.g., damaged fuel). The staff included the effects of this azimuthal variation in its uncertainty evaluation, described below.

Also, the surface detector at the package surface is actually located at the surface of the package's neutron shield assemblies, and is not located on the package surface above the neutron shield (i.e., at the top forging). Therefore, the applicant added two additional surface detectors for this part of the package. One is at the top forging's surface and the other is at 1 meter from the top forging's surface. The axial extent of these detectors is the same as the exposed surface of the top forging in the normal conditions of transport model. The detectors are also azimuthally segmented to capture variations in dose rates around the package surface, but they are not axially segmented.

The staff finds the segmentation results in detector sizes (for areas above the neutron shield) that are too large when considering the sizes of actual detectors. Lack of axial segmentation also smears dose rates across zones of different shielding; therefore, it is not clear that the peak dose rate is calculated. However, given the large margins to the normal conditions of transport limit for the enclosed package surface and the hypothetical accident conditions limit, the staff finds these two detectors' definitions are adequate for the current application. Additionally, though large, the azimuthal segmentation is adequate to capture variations in surface and 1-meter dose rates at this part of the package due to damaged fuel.

The applicant also added three other surface detectors. Two are at the package surface and are segmented to capture azimuthal variation in package dose rates due to the rotation trunnions and the heat fins, which are neutron streaming paths through the neutron shield assembly area of the package surface. Each is about 1 foot tall. Given the size of the rotation trunnions and the axial extents of the heat fins, the staff finds these detector segment sizes, though large, to be acceptable. The detectors are segmented azimuthally. The staff finds this segmentation to be a little larger than the staff would expect. The staff noted the applicant provided analyses of dose rates for these features using mesh tallies. Thus, discussion of the streaming impacts of these features is included as part of the evaluation of the mesh tally analysis in Section 5.4.3.3 of this SER.

The third additional surface detector is located at 1 meter from the package surface near the package's axial mid-plane. The detector is 6 inches in height and is azimuthally segmented. The purpose of this detector is to capture the dose rate impacts of the 6-inch hole due to the puncture test on the neutron shield combined with the radial lead slump. The azimuthal segmentation results in detectors that are larger than the staff would expect; however, given the size of the features for which the azimuthally segmented detector is used, the staff finds that the size is acceptable.

The applicant's analysis did not indicate any noticeable effect of the puncture and radial lead slump on package dose rates. The staff performed independent calculations to investigate that result in terms of the neutron dose rates using a simple model and point detectors in MCNP5, an earlier version of the MCNP code than the version used by the applicant. The detectors were placed at 1 meter from the center of the puncture in the neutron shield. The results of these calculations indicated that dose rates only increased by a few percent versus the dose rates due to changes in the neutron shield material for hypothetical accident conditions. This result increased the staff's confidence in the applicant's results. The staff finds that, with the significant margins to the hypothetical accident conditions dose rate limits, this small effect of the puncture and lead slump is negligible.

The applicant also added two other surface detectors to evaluate the azimuthal variations in dose rates near the top of the package's neutron shielding. These variations arise due to differences in the materials in the top area of the neutron shield compartments, with alternating compartments having a relatively large void region in this area. One detector was placed at the package surface and another was placed at 1 foot from the package surface. Given the results from these two detectors, the applicant determined that a similar detector was not needed at the 2-meter location due to no resolvable azimuthal dose rate variation or peaking being expected at that distance. While the application did not include details of the azimuthal segmentation nor the axial extent of the detectors, the staff expects that they are similar to the other surface detectors that the applicant used to evaluate azimuthal dose rate variations for package features of limited axial extent. Given the size of the shield compartment variations, the staff finds that such detector sizes would be acceptable for this evaluation. The applicant also used these detectors to evaluate the dose rate variations at the 1-foot (the enclosure surface) location that arise from the damaged fuel contents for the damaged fuel TSC. The staff considered the results from these detectors in its evaluations of the package dose rates and the analysis uncertainties and conservatisms described in Section 5.4.9 of the SER. The staff used the results of the 1-foot azimuthal dose rates to estimate an uncertainty to apply to the maximum dose rates at that location. The staff also included a small uncertainty, of a few percent, in its uncertainty evaluation for the maximum 2-meter dose rates. Based on the trend in variation shown for the surface and 1-foot locations, the staff determined that an estimated variation of a few percent in dose rates would be adequate to cover any azimuthal dose rate variation at the 2-meter location due to the variation in the neutron shielding in the top area of the shield components.

5.4.3.3 Mesh Tallies and Streaming Paths

The applicant included an analysis of the impacts of streaming through the heat fins and the rotation trunnions. The analysis also addressed streaming through the bolts and the expansion foam and thermal insulator near the bolts in the bolted neutron assemblies and through the gap between the bottom impact limiter and the neutron shield assemblies. The staff reviewed the mesh detector sizes for the different analyses. Based on considerations of actual detector sizes and expectations regarding detectability of dose rate variations at the package surface and at 2 meters from the vehicle edge, the staff finds the detector sizes to be adequate for this analysis. The results of the analysis indicate small variations in dose rates versus the azimuthally averaged dose rates at 2 meters from the vehicle. These variations range from a few percent for the bolts to about 10% for the trunnions when comparing dose rates for the same configuration (i.e., the TSC and contents are shifted down in the package cavity). The staff included these effects in its evaluation of uncertainties described below. With the significant margins to the surface dose rate limit for enclosed package surfaces, the staff focused mainly on the 2-meter dose rate limits with only some consideration given to the dose rates for the package's enclosure surface.

The staff notes that the discussion of dose rates from streaming paths should not be tied to any kind of discussion of whole body exposure (see page 5.8.12-4 of the application). The discussion is not relevant because the dose rate limits are not defined in terms of whole body exposure. The important point should be on the size of detectors. Detectors in analyses should be reasonably sized. The reasonableness of their size should be justified in terms of comparisons to the sizes of actual detectors that are used for dose rate measurements and the package features being evaluated. Thus, it would seem that streaming from a gap of about 1.5 inches (the size of the gap between the bottom impact limiter and the lower trunnion), as described in this part of the application, would be detected by a reasonably sized detector and should therefore be included in the package's dose rate analysis. For this particular case, the dose rate is not the maximum surface dose rate and is significantly below the dose rate limit for package surfaces within an enclosure. Therefore, the dose rate from this streaming path does not impact compliance with the regulatory limit.

5.4.4 Dose Conversion Factors

The output from the MCNP code's detectors, also referred to as tallies, must be modified to obtain the results in terms of dose rates. The applicant performed this modification by including the flux-to-dose (rate) conversion factors from the ANSI/ANS 6.1.1-1977[4] standard in the input files. Use of the conversion factors from this revision of this standard is consistent with the staff's review guidance. Therefore, the staff finds the applicant's conversion factors to be acceptable.

5.4.5 Low-Enriched Spent Nuclear Fuel Contents

For low-enriched assemblies down to 1.3% U-235, the applicant listed minimum cooling time and maximum assembly average burnup limits that apply to all hybrid assembly types in Tables 5.8-14b and 5.8-23b in the application. The calculations supporting the limits in these two tables are based on a bounding hybrid assembly type. The staff identified the bounding assembly type from a comparison of the burnup, enrichment and cooling time limits in these two tables for the 30 GWd/MTU burnup limit versus the limits for the combination in Tables 5.8-15 and 5.8-24 of the application. The bounding hybrid assembly types are the 15b (B&W 15x15) and the 8b (BWR/4-6 8x8) for PWR and BWR assemblies, respectively. The staff finds that this is consistent with staff's experience and analyses for other spent fuel packages and for spent fuel dry storage systems regarding the assembly types that are bounding for shielding.

5.4.6 Non-Fuel Hardware and Reconstituted Pressurized-Water Reactor Assemblies

The applicant performed calculations using the response method to determine the dose rates from NFH. The applicant only performed calculations for BPRAs, GTPDs and CEAs. The applicant performed source term calculations to show that hafnium absorber assemblies are bounded by BPRAs. The burnup and cooling times used for this calculation are included as limits for hafnium absorber assemblies in the contents specifications, which are only allowed contents for Westinghouse assemblies. Some CEAs may also use hafnium (Hf) as the primary absorber. The applicant's analysis shows that the source term from Hf CEAs is bounded by the source terms from other CEAs.

The applicant provided a qualitative evaluation of NSAs to show that NSAs, based on hardware similarities with BPRAs and GTPDs and the proposed limits on allowed numbers and locations, are bounded by these other two NFH types. This does not apply to NSAs from CE assembly types since BPRAs and GTPDs for CE assembly types were not analyzed; therefore, the CoC is conditioned such that CE NSAs are not authorized contents.

Based on its analysis, the applicant developed proposed limits for the different NFH types. These include combinations of allowable maximum burnup (exposure) and minimum cooling time. As an option to these limits the applicant also specified the limits for BPRAs and GTPDs and for B&W and Westinghouse NSAs in terms of total Co-60 activities. Section 5.2.2 of this SER describes the proposed limits on the allowed numbers and locations of the NFH contents.

The PWR contents also include reconstituted assemblies that have fuel rods that were replaced by irradiated solid steel rods. Only one such assembly may be loaded in a TSC, but it may be located in any basket location. Also, the assembly's irradiated steel rods (limited to five) are limited to a maximum burnup of 32.5 GWd/MTU and a minimum cooling time of 21 years. Thus, the minimum cooling time for a spent fuel assembly containing these replacement rods is either 21 years or the cooling time specified in the loading tables in Section 5.8.3 of the application (and incorporated into the CoC), as modified for other considerations (e.g., additional cooling time for loading NFH), plus 1 year, whichever is greater.

In its review of the NFH analysis, the staff evaluated the reported dose rates. Based on experience with spent fuel dry storage systems, the staff considered that the dose rates were unexpectedly small for the proposed NFH limits. Thus, the staff compared the shielding features of a familiar dry storage system with those of the MAGNATRAN. The comparison was with the storage system's transfer cask, since the gamma shielding materials and configuration are similar to those of the MAGNATRAN. The staff did the comparison using the total number of half-value layers through the base and radial side of the MAGNATRAN package and the selected transfer cask. The comparison was performed at two gamma energies. The comparison showed that the MAGNATRAN had significantly more shielding through the base and the side versus the transfer cask. The staff estimated the effect of this difference on dose rates. The estimated effect approximated the difference in dose rates between the MAGNATRAN and the transfer cask for the similar NFH limits. Thus, the staff has reasonable assurance that the applicant's NFH dose rates are reasonable.

5.4.7 Analysis Results – Burnup, Enrichment, Cooling Time Limits

The results of the applicant's analysis are a set of tables of limits in terms of maximum assembly average burnup, minimum assembly average enrichment and minimum cooling time. These tables include Tables 5.8-14b, 5.8-15, 5.8-16, 5.8-23b, and 5.8-24 of the application. Tables 5.8-15, 5.8-16 and 5.8-24 of the application are divided into ranges of maximum assembly average burnup. For each burnup range, a minimum cooling time is specified for each range of minimum assembly average enrichment for each hybrid assembly type. Enrichments for which cooling times are not provided are not allowed contents.

The applicant calculated these cooling times based on meeting the dose rate and decay heat limits but did not include NFH for PWR assemblies. For dose rates, the applicant determined the most restrictive regulatory limit is at 2 meters from the vehicle edge. Further, to account for uncertainties in its analysis, the applicant imposed a dose rate limit of 9.5 mrem/hr at this location, a 5% margin. Thus, the cooling times are based on not exceeding this 9.5 mrem/hr dose rate at 2 meters from the vehicle edge. For some PWR fuel types, the applicant imposed an even lower dose rate limit of either 8.5 mrem/hr or 8.6 mrem/hr.

Including NFH requires that the minimum assembly cooling time be increased in order to meet the applicant-imposed dose rate limits that apply to the different assembly types. The amount of increased cooling time varied for the different PWR assembly types. Thus, the applicant also calculated the amount of additional cooling time for the different PWR assembly types when loaded with the different types of NFH. Reconstituted PWR assemblies also require additional cooling time. The content specifications in the CoC include these additional cooling time requirements that are noted in Section 5.4.6 of this SER. As indicated earlier, the inclusion of damaged fuel also requires additional cooling time for the PWR spent fuel contents, both damaged and undamaged. When multiple damaged fuel types are loaded in a single TSC, the longer additional cooling time is the additional cooling time that is to be applied to all of the PWR spent fuel contents in that TSC. When the contents include combinations of these contents types, the additional cooling time requirements associated with each content type must be included in the total cooling time requirements for the assemblies. For example, the required minimum cooling time for undamaged low burnup PWR assemblies that contain NFH and that are loaded into a TSC that contains damaged fuel is the summed total of the minimum cooling time specified in Tables 5.8-14b and 5.8-15 of the application plus the additional cooling times for both NFH (application Table 5.8-34) and damaged fuel (application Table 5.8-49).

Based on the evaluation of the contents, package shielding design, and the package dose rates, as described throughout this chapter of the SER, the staff finds the proposed contents limits to be acceptable.

5.4.8 Greater-Than-Class C Analysis

The applicant's analysis indicates that the maximum dose rates for GTCC waste for both normal conditions of transport and hypothetical accident conditions are at the package's side, near or above the cavity's mid-plane. The staff performed a simple analysis to confirm this for hypothetical accident conditions. Since the source is uniformly distributed within the package cavity, the staff anticipated that dose rates may actually peak near, or above, the area of axial lead slump. The staff evaluated a detector placed 1 meter from the package surface above the package cavity mid-plane and another detector placed 1 meter from the package surface above the area of axial lead slump. The calculations indicated that dose rates from the axial lead slump zone could be as significant as the peak dose rate at the package cavity's mid-plane. However, given the significant margins to the limits, the staff finds the applicant's determination is acceptable.

The staff also notes that the applicant's model for the GTCC analysis includes a few simplifications versus the models for the spent fuel contents. For example, the model uses nominal material thicknesses. The staff finds this to be acceptable given the significant margins to the regulatory dose rate limits. Even for conditions such as described below for localized peak specific activities and loose contamination where these margins are significantly reduced, the staff finds the GTCC analysis models to be acceptable based on conservatisms in the model.

The applicant's analysis models only included GTCC waste at the maximum contents-average specific activity. To address localized peaking of the specific activity, the applicant increased the calculated GTCC waste dose rates by the ratio of the peak limit to the contents-average limit. Doing this is equivalent to assuming the entire waste contents have a specific activity equal to the localized peak limit. Tables 5.8-54 and 5.8-55 of the application include notes to describe the dose rate impacts. In actuality, only localized areas or volumes of the contents will be allowed to have specific activities up to this limit; thus, the staff finds this approach to be bounding and acceptable for addressing the amount of non-uniformity to be allowed per the proposed contents limits. The hypothetical accident conditions dose rates remain significantly below the regulatory limit. The normal conditions of transport dose rates also have large margins to and remain below the limits, though the margin is significantly reduced when evaluated for localized peaking of the waste's specific activity, particularly versus the 2-meter dose rate limit.

The GTCC waste includes contaminated hardware that may include removable, or loose, contamination that might be able to collect together in the package. This contamination may include fines resulting from cutting of the waste into pieces small enough to fit into the GTCC waste basket liner in the TSC. The staff finds that these fines do not pose a concern since they are expected to have the same specific activity of the localized areas of the components from which they derive. The applicant, in response to staff's questions, also provided information to demonstrate that contamination, including loose contamination, is not a shielding concern.

The staff considered the information the applicant provided to evaluate the potential impacts of removable contamination that collected into one location within the package. The applicant's evaluation is based on contamination information for spent fuel assemblies and includes justification for why it is applicable to the GTCC waste items. Based on the information available to the staff regarding the kinds of contamination the proposed GTCC waste contents could include, the staff finds that the applicant's estimates are reasonable and acceptable for evaluating the impacts on package dose rates. Using these estimates, the staff estimated how much removable contamination could be in the GTCC contents. Conservative estimates showed a small impact on normal conditions of transport dose rates, but the dose rates still met regulatory limits even when combined with dose rates from the evaluation for the waste being at the localized peak specific activity. For hypothetical accident conditions, the package configuration is such that the material could not collect next to the areas of lead slump. Therefore, while the dose rates around the lead slump areas would likely increase, the staff has reasonable assurance that the resulting dose rates will remain below the hypothetical accident conditions limit with significant margin.

5.4.9 Uncertainties and Conservatisms

The applicant identified and evaluated a variety of uncertainties and conservatisms in the analysis. The applicant's evaluations include quantitative analysis that the applicant used to support its conclusions. This analysis includes estimates of the conservatism in the sub-critical multiplication due to modeling the spent fuel as fresh fuel at 5% U-235 enrichment (versus 2% U-235 enrichment) and without the basket's neutron absorber plates for the undamaged fuel contents. It also includes the dose rate analysis using mesh tallies for streaming through the rotation trunnions and other features in the neutron shield assemblies.

The applicant identified that there is some uncertainty in the depletion code and determination of heat loads for high burnup fuel. Therefore, the applicant de-rated the allowable heat load for high burnup fuel to be 5% less than the allowable heat load for low burnup fuel. The staff reviewed the applicant's approach for high burnup fuel as described in Section 5.2.4.1 of this SER. The staff notes that this 5% is consistent with what other applicants have identified as the uncertainty in high burnup heat load calculations in their analyses. While this reduction in the allowed decay heat did increase the minimum cooling times for many of the proposed high burnup fuel contents, others were not affected since they are still limited by dose rates and not decay heat. Thus, for these high burnup fuel contents the staff notes that the decreased heat load does not introduce any conservatism to compensate for uncertainties in the dose rate analysis due to source term calculations. The staff notes that for nearly all of these contents, they are dose-rate limited at the lower, applicant-imposed dose rates of 8.5 and 8.6 mrem/hr. Thus, as part of its confirmatory evaluations, the staff considered a 5% uncertainty in the dose rates due to the depletion uncertainty for high burnup fuel.

With regard to the sub-critical multiplication, the applicant's identified amount of conservatism is the difference between what is modeled for undamaged fuel versus damaged fuel. The degree of conservatism is much less for the damaged fuel analysis, which uses a smaller enrichment and takes credit for the neutron absorber panels in the TSC basket. Any remaining conservatism in the modeled sub-critical multiplication has not been identified, calculated, or justified.

The applicant identified other uncertainties in the analysis, which include: azimuthal variation in dose rates for the damaged fuel loading; the use of fresh fuel compositions to model the spent fuel materials; differences between the response function method and the direct calculation method for dose rates; truncation of the source spectra; dose rate peaking due to azimuthal variations and streaming paths in the package shielding; effects of axial blankets; the burnup profiles used for low (<30 GWd/MTU) burnup fuel analysis; and the source term calculations. The applicant concluded that the effects of these uncertainties are not significant.

The applicant built other conservatisms into the analytical model, which include: loss of neutron material from the model due to rounding of the outer surface of the neutron shield assemblies in the model; not taking credit for loss of fuel material and source that was replaced by steel rods in the reconstituted PWR assembly; not crediting reduction in end hardware dose rates due to shielding by fuel material in the hardware zones in the damaged fuel analysis; 5% reduction of the most restrictive dose rate limit (reduced 10 mrem/hr to 9.5 mrem/hr at 2-meter location or less for some assemblies); and conservative inputs in the source term calculations (e.g., constant power applied to the entire irradiation of the assembly and use of constant soluble boron concentration in the moderator).

The staff evaluated the model effects on the neutron shield. The staff's evaluation indicated that the applicant modeled the neutron shield material at the minimum thickness specified in the drawings for some of the neutron shield assemblies and at the maximum thickness specified in the drawings for other neutron shield assemblies. Thus, the staff finds that it is not clear that the shielding model results in the loss of material, at least in any consistent manner. Thus, the staff does not find that the model is necessarily conservative with respect to the neutron shield material. Thus, the model of the neutron shield introduces some uncertainties. However, the staff recognized that other aspects of the model introduce some conservatisms (e.g., thermal insulation and expansion foam modeled as void) among other considerations, which offset the uncertainties from the model of the neutron shield.

The staff used the information in the application and its experience with the shielding analyses for other spent fuel packages and for spent fuel dry storage systems to evaluate the impact of uncertainties and conservatisms on the package dose rates. In addition to the uncertainties identified by the applicant, the staff also identified the following as uncertainties or included their effects in the analysis of the uncertainties: cobalt impurity levels in assembly hardware and NFH; differences in bounding NFH masses; loading of individual NFH assembly components (including axial movement of these components); differences in NSAs versus other NFH types (both gamma and neutron source contributions); and irradiated unenriched fuel replacement rods. Many of these items are discussed in the preceding sections of this SER chapter. For NSAs, the staff considered the limitations on the allowable locations in the TSC basket and the effects of those limitations on the NSAs' contributions to package dose rates.

With regard to azimuthal variations, the package and the applicant's models include features that act as streaming paths or areas of differences in the shielding. There are also non-uniform aspects to the contents loading specifications, as already described. The applicant defined detectors so as to enable evaluation of the dose rate impacts of these variations. The staff evaluation of the impacts to dose rates of these variations was derived from consideration of the applicant's results and the results of staff's confirmatory calculations and experiences with other packages. The staff considered the results of azimuthal dose rate calculations from the azimuthally segmented surface detectors as well as the mesh tally analysis results.

In its evaluation, the staff determined the relative impact of the different uncertainties on dose rates and used these impacts to increase the dose rates reported in the application and the files provided with the shielding Calculation Package No. 71160-5508, Rev. 0. The staff evaluated the impacts for several hybrid assembly types, focusing mainly on the PWR contents since these contents had the greatest variety and the most factors for introducing uncertainty into the analysis. However, the staff also evaluated the effects on the BWR dose rates.

The staff also included the effects of conservatisms on the dose rates when necessary to show dose rates were below limits. For example, for undamaged fuel, the relative difference in total dose rates from the differences in models to address sub-critical multiplication (undamaged model versus damaged model) was included. Though not quantified in the application, the staff considered the effects of crediting the damaged fuel can materials and the thicker basket weldment walls for the damaged fuel locations. The staff recognized other conservatisms in the models and weighed these qualitatively in its evaluation of the analysis method's uncertainties. The staff also recognized there are conservatisms in the staff's evaluation of the impacts of the identified uncertainties. The results of this evaluation indicate that the package will meet the dose rate limit at 2 meters from the vehicle edge.

The staff's evaluation focused mainly on the 2-meter dose rates because the limit at this location was shown to be the most restrictive. The staff also considered the effects of uncertainties and conservatisms on the dose rates at the package's enclosure surface. The staff performed this evaluation for the enclosure surface dose rates in a manner similar to its evaluation of the 2-meter dose rates, accounting for differences in the effects of the uncertainties at the closer distance to the package. The staff also evaluated the effects of items such as NSAs' neutron sources, streaming paths in the impact limiters, and non-uniform contents loading (e.g., damaged fuel) on dose rates at the axial ends of the package. The staff also evaluated the effects of not including the tolerances for the TSC components in the models. Based on the results of these evaluations and the evaluation for the dose rates at 2 meters from the vehicle edge and at the package's enclosure surface described here, the staff has reasonable assurance that the package meets the regulatory dose rate limits.

5.4.10 Pressurized-Water Reactor Spent Fuel in 22kW Basket

As noted in Section 5.2.1, the PWR spent fuel contents have two separate limits for package decay heat. The limits are based on different neutron absorber panel thermal conductivities in the spent fuel basket (see the thermal chapter of this SER). Previous references to tables for allowable burnup, enrichment and cooling times and for the added decay time for loading non-fuel hardware with the spent fuel are for the 23kW basket. Tables 5.8-58, 5.8-59, and 5.8-60 provide the allowable minimum assembly average enrichment, minimum cooling time, and maximum assembly average burnup for the PWR spent fuel loaded into the 22kW basket. Additionally, Table 5.8-61 shows the additional assembly cooling time that is required to load non-fuel hardware with the spent fuel in the 22kW basket. All other requirements and limits that apply to the 23kW basket PWR contents apply as written, with the same numerical values for numerical limits, to the 22kW basket PWR contents. For example, loading damaged fuel in the 22kW basket requires that the cooling times for all contents, including the damaged fuel, be extended by the amount shown in Table 5.8-49.

The staff evaluated the information in Tables 5.8-58 through 5.8-61 and the information in Section 5.8.14 of the application. Nearly all of the minimum cooling times are longer than the cooling times for the 23kW basket by amounts that appear to be consistent with the amount of time needed to reduce decay heat by 1kW based on the differences in the spent fuel sources for different burnup and enrichment combinations. Further, review of the shielding Calculation Package No. 71160-5508, Rev. 1 showed that some of the contents are limited by dose rates in the 23kW basket such that a smaller increase in cooling time was needed to meet the 22kW limit than would have been needed if they were limited by decay heat in the 23kW basket. Additionally, the staff noted that some cooling times are the same as for the 23kW basket contents. Review of the shielding calculation package shows that these contents continue to be limited by dose rates. As with the 23kW basket contents, a 5% decay heat penalty is applied to the high burnup fuel contents in the 22kW basket. The reasons for this penalty are described previously and remain valid for the 22kW basket. Thus, the allowable decay heat limit for high burnup fuel in the 22kW basket is 20.9kW per package (565W per basket location). Based on its evaluation, the staff finds these limits to be acceptable.

The staff evaluated the need for modifications to the additional cooling time requirements applied to the spent fuel for loading non-fuel hardware, damaged fuel, and reconstituted fuel. Changes were required only for some fuel types for loading non-fuel hardware; no changes were made to the burnup and cooling time requirements of the non-fuel hardware items. The staff noticed that the items for which changes were made contributed greater amounts of decay heat. Others contributed little; therefore, the staff finds it reasonable that no changes to the added cooling times would be needed for those non-fuel hardware types to meet decay heat limits in the 22kW basket. For damaged fuel and reconstituted fuel nothing is added beyond the assembly itself; therefore, there is no need for changes to the additional cooling time requirements for these two items, which were developed to ensure compliance with dose rate limits. Thus, based on its review, the staff finds the proposed additional cooling time requirements for loading these items to be acceptable for the 22kW basket.

5.5 Evaluation Findings

Based on its review of the information and representations provided in the application and the staff's independent, confirmatory calculations, the staff has reasonable assurance that the proposed package design and contents satisfy the shielding requirements and dose rate limits in 10 CFR Part 71 and that the external radiation levels will not significantly increase during normal conditions of transport consistent with the tests specified in 10 CFR 71.71.

5.6 References

1. I.C. Gauld, G. Ilas, and G. Radulescu, *Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel*, NUREG/CR-7012 (ORNL/TM-2010/41), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, Oak Ridge, TN (2011).

2. A. Luksic, *Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal*, PNL-6906, Pacific Northwest Laboratory, Richland, WA, June 1989.

3. DOE/RW-0184, Appendix 2E, *Physical Descriptions of LWR Nonfuel Assembly Hardware*, U, S, Department of Energy, July 1992.

4. ANSI/ANS 6.1.1-1977, *American National Standard for Neutron and Gamma-Ray Flux to Dose Factors*, La Grange Park, IL, 1977.

6.0 CRITICALITY REVIEW

This section presents the findings of the criticality safety review for an application to authorize the Model No. MAGNATRAN transportation package using a criticality analysis using credit for fuel burnup. The staff evaluated the package for its ability to meet the fissile material requirements of 10 CFR Part 71, including the general requirements for fissile material packages in 10 CFR 71.55, and the standards for arrays of fissile material packages in 10 CFR 71.59. The staff reviewed the criticality safety analysis of the package presented in the SAR, and also performed independent calculations to confirm the applicant's results. The staff's review considered the criticality safety requirements of the radioactive material transportation regulations in 10 CFR Part 71, as well as the review guidance presented in NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel,"[1] and Interim Staff Guidance 8, Revision 3 (ISG-8), "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Packages."[2]

6.1 Description of the Criticality Design

6.1.1 Packaging Design Features

The Model No. MAGNATRAN package design consists of a cylindrical, steel shell containment system, with a flat bottom and two bolted closure lids at the top, and with three different available internal basket structures for maintaining the position of the spent fuel contents. Fuel assemblies are supported in the MAGNATRAN basket by a series of steel fuel tubes, and "developed cells" formed by the fuel tubes, as shown in Figure 6.3.1-1 of the SAR. Neutron absorber sheets, consisting of either borated aluminum, aluminum/B_4C metal matrix composite, or Boral®, are attached to the inside of the fuel tubes, such that there is one sheet between each pair of adjacent fuel assemblies. Criticality safety is maintained by the spent fuel and basket geometry, fixed neutron absorber plates between the fuel assemblies, and in the case of PWR fuel, credit for the net depletion of fissile nuclides and the buildup of stable or long-lived neutron absorbing nuclides with fuel irradiation (burnup credit).

The three available TSCs for spent fuel are designed to hold PWR, BWR, and damage fuel PWR fuel. The PWR TSC can transport up to 37 undamaged PWR spent fuel assemblies, the BWR TSC can transport up to 87 undamaged BWR spent fuel assemblies, and the PWR damaged fuel TSC can transport up to 37 undamaged PWR spent fuel assemblies, with up to four damaged fuel cans in four corner locations of the basket. Damaged fuel cans may contain an undamaged PWR spent fuel assembly, a damaged PWR spent fuel assembly, or fuel debris equivalent to one PWR spent fuel assembly. An optional 82-assembly configuration, shown in Figure 6.1.1-1 of the SAR, maintains five empty fuel assembly locations, and allows for higher initial maximum planar average enrichment BWR fuel assemblies to be transported. For WE15x15 fuel assemblies in the PWR undamaged or damaged fuel baskets, three under-loaded configurations, with one, two, or four empty fuel locations, allow for fuel assemblies with lower assembly-average burnups (see Figure 6.10.1-17 for empty fuel locations). Also, for WE15x15 fuel assemblies in the PWR basket, up to nine low burnup fuel assemblies may be transported in the center nine fuel locations, provided they include depleted RCCAs.

6.1.2 Summary of Criticality Evaluations

The applicant provided a summary of criticality evaluations in Section 6.1.2 of the SAR. Detailed results for the PWR undamaged fuel basket, PWR damaged fuel basket, and BWR basket are contained in Sections 6.10.1, 6.10.2, and 6.10.3 of the SAR, respectively. All reported results include the calculated k_{eff} plus two standard deviations. The applicant demonstrated that all design-basis criticality calculations result in k_{eff}s that are less than the corresponding Upper Subcritical Limit (USL) determined in Section 6.8 of the SAR. Each USL includes the code bias and bias uncertainty, and for calculations that include credit for the burnup of the fuel, include an additional bias and bias uncertainty for the depletion code calculation, an additional burnup-dependent bias for minor actinide and fission product validation, and an additional bias to cover uncertainty related to the depletion power density assumption.

6.1.3 Criticality Safety Index

The applicant demonstrated that infinite arrays of MAGNATRAN packages containing undamaged fuel are adequately subcritical under normal conditions of transport and hypothetical accident conditions. Therefore, the CSI, determined in accordance with 10 CFR 71.59(b), is 0.0 for PWR and BWR canisters containing undamaged fuel. For damaged fuel contained in a PWR damaged fuel canister, a single package was demonstrated to be subcritical, and the corresponding CSI is 100.0.

6.2 Spent Nuclear Fuel Contents

The MAGNATRAN package is designed to transport a maximum of 87 BWR and 37 PWR fuel assemblies in the PWR, PWR damaged fuel, and BWR TSCs. Specific TSC contents are summarized below:

PWR TSC – Designed to transport up to 37 undamaged PWR spent fuel assemblies. PWR fuel assemblies allowed for transport are divided into 12 assembly groups, as defined by the assembly loading criteria given in Table 6.1.2-1 of the SAR. The maximum initial enrichment for PWR fuel represents the peak fuel rod enrichment for the assembly. Assemblies may also include non-fuel hardware, as defined in Section 1.1 of the SAR. Parameters which define the fuel loading curves, which give the minimum required fuel burnup as a function of initial peak fuel rod enrichment, are given in Tables 6.10.1-12, 6.10.1-13, and 6.10.1-14 for the 0.036 g/cm^3 ^{10}B neutron absorber, 0.030 g/cm^3 ^{10}B neutron absorber, and 0.027 g/cm^3 ^{10}B neutron absorber, respectively. The seven loading curves shown, for BW15x15, BW17x17, CE14x14, CE16x16, WE14x14. WE15x15, and WE17x17 fuel assemblies, are the most limiting for each fuel manufacturer and array size out of the 12 assembly groups. The minimum cooling time for all loading curves is 15 years. Alternative loading curve criteria for WE15x15 fuel in under-loaded configurations, with one, two, or four empty fuel locations, are shown in Table 6.10.1-16, for fuel assemblies with lower assembly-average burnups (see Figure 6.10.1-17 for empty fuel locations). Also, for WE15x15 fuel assemblies in the PWR basket, up to nine low burnup fuel assemblies may be transported in the center nine fuel locations, provided they include fresh or depleted RCCAs (see Section 6.10.1.7 of the SAR).

- PWR damaged fuel TSC – Designed to transport up to 37 undamaged PWR spent fuel assemblies, with up to 4 DFCs in the four corner locations of the basket. DFCs may contain an undamaged PWR spent fuel assembly, a damaged PWR spent fuel assembly, or fuel debris equivalent to one PWR spent fuel assembly. Parameters which define the fuel loading curves are given in Tables 6.10.2-5, 6.10.2-6, and 6.10.2-7 for the 0.036 g/cm^3 ^{10}B neutron absorber, 0.030 g/cm^3 ^{10}B neutron absorber, and 0.027 g/cm^3 ^{10}B neutron absorber, respectively. Damaged fuel in a damaged fuel can is limited to 4.05 weight percent initial enrichment and must have a burnup of at least 5 GWd/MTU and cooling time of at least 15 years. Alternative loading curve criteria for WE15x15 fuel in under-loaded configurations, with one, two, or four empty fuel locations, are shown in Table 6.10.2-11, for fuel assemblies with lower assembly-average burnups (see Figure 6.10.1-17 for empty fuel locations). Also, for WE15x15 fuel assemblies in the PWR basket, up to nine low burnup fuel assemblies may be transported in the center nine fuel locations, provided they include fresh or depleted RCCAs (see Section 6.10.1.7 of the SAR).

- BWR TSC – Designed to transport up to 87 undamaged BWR spent fuel assemblies, with or without zirconium-based alloy channels. BWR fuel assemblies allowed for transport are divided into 20 assembly groups, as defined by the assembly loading criteria given in Table 6.1.2-2 of the SAR. The maximum initial enrichment for BWR fuel represents the peak planar-average enrichment for the assembly. BWR fuel assemblies may contain partial length rods, limited to the assemblies and locations shown in Figure 6.2.1-1 of the SAR. The maximum initial enrichment for each assembly type, in the 0.027 g/cm^3 ^{10}B neutron absorber basket, 0.0225 g/cm^3 ^{10}B neutron absorber basket, and 0.02 g/cm^3 ^{10}B neutron absorber basket, is given in Table 6.1.2-3 for fuel assemblies with a minimum 6-inch axial blanket, and in Table 6.1.2-4 for fuel assemblies with no axial blanket. Alternative maximum initial enrichments are given in the same tables for an under-loaded configuration of 82 BWR fuel assemblies. Figure 6.1.1-1 shows the empty fuel locations for the 82-assembly BWR basket configuration.

For all TSCs, low-enriched, unenriched (less than 6 inches), and/or annular blankets are allowed. Undamaged fuel assemblies contain a full, nominal set of rods. Removed rods must be replaced with filler rods, which displace an equal or greater amount of water than the original rod. Undamaged fuel must meet all fuel-specific and system-related functions, and cannot contain fuel rods with breaches greater than pinholes leaks or hairline cracks, or grid, grid strap and/or grid spring damage such that the unsupported length of fuel rods exceeds 60 inches. Damaged fuel cans in the PWR damaged fuel TSC may contain undamaged fuel assemblies, damaged fuel assemblies, or fuel debris equivalent to up to one PWR fuel assembly. PWR fuel assemblies with burnups greater than 45 GWd/MTU must be placed in a DFC in the PWR damaged fuel TSC. PWR or BWR fuel assemblies with burnups greater than 45 GWd/MTU are not authorized in either the PWR or BWR TSCs.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The applicant evaluated three-dimensional models of a single package and arrays of packages under hypothetical accident conditions. The hypothetical accident conditions model bounds the normal conditions of transport model, and is, therefore, used for all criticality evaluations. The applicant explicitly modeled the fuel rods and cladding, guide tubes, water gaps, and neutron absorber in the basket. For all cases where the containment is flooded, the fuel-to-clad gap is also conservatively assumed to be flooded with fresh water. The applicant modeled the package body conservatively neglecting the neutron shielding material, allowing more neutron communication between packages in an array, as well as better neutron reflection from the package wall in the single package. Additionally, the applicant ignored the impact limiters, allowing for closer package spacing. Preferential flooding was not considered for undamaged fuel, due to the holes present at the top and bottom of the basket cell walls to prevent preferential flooding. Preferential flooding was considered for damaged fuel cans in the PWR damaged fuel TSC, since they may drain at a slower rate due to the screens at the bottom of the damaged fuel can. The applicant also evaluated partial flooding, where the basket is flooded to varying depths, with the remaining volume filled with low density water simulating steam.

The applicant modeled all fuel at a conservative pellet stack density of 96% of the theoretical maximum uranium dioxide (UO_2) density, with no allowance for pellet dishing and chamfer. With the exception of channels in the case of BWR fuel, fuel structural material (e.g., spacer and mixing grids) is conservatively ignored. This material would serve to absorb neutrons and displace water, both of which would decrease system reactivity. Fuel assembly end fittings were modeled as a mixture of steel and water. The applicant also considered the most reactive combination of material and fabrication tolerances for the fuel and basket and evaluated eccentric positioning of fuel assemblies within the basket guide tubes.

Both BWR and PWR fuel may contain unenriched or low enriched solid or annular top and bottom axial blankets. For PWR fuel, the applicant demonstrates that assuming the fuel is fully enriched is conservative, and there are therefore no limits associated with axial blankets for criticality purposes. For BWR fuel, axial blankets may affect the allowed maximum initial planar-average enrichment, in the top-end drop scenario where a small amount of active fuel may become uncovered by the neutron absorber plates. For unblanketed fuel, this configuration causes a significant increase in reactivity, and the maximum initial planar-average must be reduced. For blanketed fuel, where the uncovered fuel is unenriched or low enriched, a higher initial planar-average enrichment is shown to be acceptable. Fuel assemblies with blankets smaller than 6 inches are considered to be unblanketed, while fuel with larger blankets is bounded by the 6-inch blanket evaluation.

The applicant neglected the presence of any integral fuel burnable absorbers (e.g., BWR gadolinia, PWR erbium or boron), as they would only serve to decrease system reactivity. The presence of non-fuel hardware during transportation is conservatively ignored, as hardware would displace water and reduce system reactivity. The effects of the presence of BPRAs and RCCAs during irradiation are considered for PWR fuel in the burnup credit analysis, as exposure to BPRAs and RCCAs can increase discharged fuel reactivity. Additionally, depleted RCCAs are modeled inserted into the center nine fuel assemblies in the PWR and PWR damaged fuel TSCs for WE15x15 fuel configurations with under-burned fuel assemblies.

The applicant's structural analysis in Section 2.0 of the SAR demonstrates that undamaged spent fuel remains intact under normal conditions of transport and hypothetical accident conditions. Therefore, undamaged spent fuel assemblies were modeled with an as-built, intact geometry in the PWR, PWR damaged fuel, and BWR TSCs. Fuel with greater than 45 GWd/MTU burnup is not allowed in the PWR or BWR TSCs, or in undamaged fuel locations in the PWR damaged fuel TSC. PWR fuel with greater than 45 GWd/MTU must be treated as damaged, and transported in a damaged fuel can in one of the four corner locations of the PWR damaged fuel TSC.

To determine the most reactive configuration, the applicant modeled damaged PWR fuel in damaged fuel cans in the four corner locations of the PWR damaged fuel TSC in three different configurations: clad rods, unclad rods, and homogenized fuel. The clad rod configuration consists of an essentially undamaged fuel assembly, with fuel in cladding with the undamaged fuel assembly rod pitch. The unclad rod configuration moves the clad material outside the lattice area of the fuel assembly and allows the rod pitch to expand within the physical limits of the damaged fuel can. The homogenized fuel configuration considers fuel material and water mixed homogeneously within the damaged fuel can. The applicant also evaluated fuel assemblies in the damaged fuel can with missing rods. The most reactive of these configurations was used by the applicant to determine the loading curve for the remaining 33 undamaged fuel assemblies in the PWR damaged fuel TSC.

For undamaged, under-burned WE15X15 fuel transported with fresh or burned RCCAs, the applicant modeled the under-burned fuel at an initial enrichment of 4.05 weight percent, an assembly-average burnup of 12 GWd/MTU, and a cooling time of 15 years. The applicant modeled the RCCA assuming full assembly power with insertion in the core for an equivalent exposure of 200 GWd/MTU. The applicant models the RCCA such that 5 inches of active fuel is uncovered at the bottom of the assembly, in order to account for uncovered fuel at full insertion and deformation in a top end drop. The remaining 28 undamaged fuel assemblies in the PWR TSC, or 24 undamaged fuel assemblies and 4 damaged fuel cans in the PWR damaged fuel SC, are modeled at their required burnups and initial enrichments for standard loadings. For under-burned fuel with RCCA credit, transportation is limited to the 0.036 g $^{10}B/cm^2$ areal density neutron absorber basket in the PWR or PWR damaged fuel TSCs.

6.3.2 Material Properties

Fresh fuel compositions were modeled as UO_2 with 96% of theoretical density. The ^{234}U and ^{236}U which are present in fresh fuel were conservatively ignored. For burned fuel compositions, the applicant modeled the fuel with the following 12 actinides and 16 fission products incorporated into the fuel matrix:

Actinides:		Fission Products:		
^{234}U	^{239}Pu	^{95}Mo	^{143}Nd	^{152}Sm
^{235}U	^{240}Pu	^{99}Tc	^{145}Nd	^{151}Eu
^{236}U	^{241}Pu	^{101}Ru	^{147}Sm	^{153}Eu
^{238}U	^{242}Pu	^{103}Rh	^{149}Sm	^{155}Gd
^{237}Np	^{241}Am	^{109}Ag	^{150}Sm	
^{238}Pu	^{243}Am	^{133}Cs	^{151}Sm	

Section 6.10.1 of the SAR discusses the calculations to determine burned fuel compositions.

Table 6.3.2-1 of the SAR provides the criticality model material number densities and compositions of the fuel assemblies. Table 6.3.2-2 of the SAR provides the criticality model material number densities and compositions of the basket, TSC, and package. Fixed neutron absorber plates in the BWR TSC are modeled at areal densities of 0.027, 0.0225, and 0.020 g$^{10}B/cm^2$, while absorber plates in the PWR and PWR damaged fuel TSCs are modeled at 0.036, 0.030, and 0.027 g $^{10}B/cm^2$. These represent effective absorber areal densities as only 75% or 90% of the minimum required ^{10}B is credited, depending on absorber type (e.g., Boral® or borated aluminum). Table 6.1.1-1 provides actual required minimum ^{10}B content for 75% or 90% credit.

6.3.3 Computer Codes and Cross-Section Libraries

The applicant used the MCNP5 v1.30 three-dimensional Monte Carlo code, along with the continuous-energy ENDF/B-VI cross-section library for fresh BWR fuel criticality calculations. For PWR spent fuel compositions, as well as fresh PWR fuel compositions at 0 burnup, the applicant used the MCNP6.1 three-dimensional Monte Carlo code, along with the continuous-energy ENDF/B-VII cross-section data.

For isotopic depletion analyses to determine burned fuel compositions, as well as to determine the composition of burned RCCAs, the applicant used the TRITON two-dimensional depletion sequence of the SCALE 6.1 code system, with the 238-group ENDF/B-VII cross-section library.

The staff performed confirmatory criticality calculations of the MAGNATRAN package using the CSAS6 sequence of the SCALE 6.1 code system, with KENO VI and the ENDF/B-VII continuous-energy cross section library. The staff also performed confirmatory PWR fuel and RCCA isotopic depletion analyses, using the TRITON sequence of SCALE 6.1 with the 238-group ENDF/B-VII cross-section library.

6.3.4 Demonstration of Maximum Reactivity

The applicant performed multiple sensitivity studies for the single package and array configurations of the MAGNATRAN package to determine the most reactive condition. These studies included variation of internal and external moderation, partial flooding of the containment, eccentric positioning of fuel assemblies in the basket, and variation of basket structural and neutron absorbing materials according to the worst-case combination of material and fabrication tolerances. For all TSCs, the maximum reactivity was obtained with an infinite array of packages (except the PWR damaged fuel TSC, which was modeled as a single package), the hypothetical accident conditions model, maximum guide tube material with minimum tube interface width, minimum neutron absorber width and maximum thickness, fuel assemblies shifted to center, a dry TSC-to-package gap, and a dry exterior. The staff finds these results are consistent with analyses performed for similar spent fuel transportation systems and with the results of the staff's confirmatory analyses.

For PWR fuel, the applicant states that the presence of low-enriched, axial blankets with solid pellets is bounded by the full-length enrichment analysis. Annular pellets could potentially increase reactivity, but they are located in a high axial leakage location at the ends of the fuel assembly. The applicant demonstrated that blanketed fuel is less reactive by performing an analysis of a 12-inch fully enriched annular blanket on a WE17x17H1 fuel assembly. The results indicate that assuming fully-enriched solid rods for the full active length is more reactive, which is consistent with the fact that blanketed fuel has less fissile material to begin with, and that the lower fissile mass for the fuel assembly means that the corresponding burnup in the central regions of the fuel must be higher to reach same assembly average burnup.

For the PWR damaged fuel TSC, the unclad rod scenario was shown to be the most reactive for both fresh and burned fuel compositions, and was therefore used for subsequent loading curve determination. The staff finds that the unclad rod scenario bounds expected damaged fuel assembly configurations with reasonable assurance, and that this configuration is consistent with most reactive damaged fuel scenarios from similar, previously approved spent fuel transportation packages. The applicant also determined that a preferentially flooded damaged fuel can resulted in a higher system reactivity, due to increased neutron communication between packages in the array. The applicant changed the array analysis to a single package, which was adequately subcritical. The corresponding CSI is 100 for the PWR damaged fuel TSC.

For the BWR TSC, the applicant demonstrated that assuming a uniform maximum planar average enrichment is more reactive than the actual variable pin enrichments that exist in BWR fuel assemblies. Additionally, the applicant demonstrated that, for fuel assemblies with partial length rods, it is conservative to assume those rod locations are empty. The staff finds the results of these analyses are consistent with previously approved analyses of BWR fuel assemblies and are therefore acceptable.

6.3.5 Confirmatory Analyses

The staff performed confirmatory criticality evaluations of the MAGNATRAN package, for both fresh and burned fuel configurations. The staff also performed confirmatory isotopic depletion calculations to determine burned fuel and depleted RCCA compositions. Since the staff determined that the applicant's assumptions were conservative, as demonstrated in the SAR and consistent with staff's experience modeling similar spent fuel transportation systems, the staff used assumptions similar to the applicant's (i.e., the hypothetical accident conditions model, maximum guide tube material with minimum tube interface width, minimum neutron absorber width and maximum thickness, fuel assemblies shifted to center, a dry TSC-to-package gap, and a dry exterior). The staff calculated k_{eff} values for select configurations which were within the margin of error of those calculated by the applicant, and confirmed that the package will meet the criticality safety requirements of 10 CFR Part 71.

6.4 Single Package Evaluation

6.4.1 Configuration

The applicant evaluated the MAGNATRAN package using the bounding configuration determined in earlier sensitivity studies for an infinite array of packages. This configuration included maximum guide tube material with minimum tube interface width, minimum neutron absorber width and maximum thickness, fuel assemblies shifted towards the center, most reactive assembly geometric conditions, optimum internal moderation, most reactive damaged fuel geometry, and consideration of axial blankets during the top end drop (BWR). BWR evaluations included fresh fuel compositions, while PWR evaluations included actinide and fission product burnup credit. The applicant modeled a single package under hypothetical accident conditions by removing the reflecting boundary condition from the array model and replacing it with full water reflection. The applicant used the hypothetical accident conditions results to bound the package under normal conditions of transport, as the fully flooded hypothetical accident conditions condition is more reactive.

The staff finds this acceptable because the fully flooded and reflected condition of the single package model under hypothetical accident conditions bounds the consideration of a flooded single package required by 10 CFR 71.55(b). The staff further finds this acceptable because this hypothetical accident conditions model also bounds consideration of the package under normal conditions of transport, as required by 10 CFR 71.55(d)(1). The structural analysis in Section 2.0 of the SAR demonstrates that, under normal conditions of transport: 1) the geometric form of the package contents will not be substantially altered, per 10 CFR 71.55(d)(2), 2) there will be no leakage of water into the containment system, per 10 CFR 71.55(d)(3), and 3) there will be no substantial reduction in the effectiveness of the packaging, per 10 CFR 71.55(d)(4). The staff finds the assumed hypothetical accident conditions condition also satisfies 10 CFR 71.55(e), in that: 1) the fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents, 2) moderation is assumed at the most reactive credible extent, and 3) there is full water reflection on all sides.

6.4.2 Results

Results are given in Sections 6.10.1, 6.10.2, and 6.10.3 of the SAR for the PWR, PWR damage fuel, and BWR TSCs, respectively. The maximum system reactivity for all three TSCs, in the single package configuration described above, is less than the calculated USL for the code and cross section library used for the evaluation. Therefore, the staff finds with reasonable assurance that the MAGNATRAN package meets the requirements of 10 CFR 71.55 for single package evaluations and is adequately subcritical under all conditions.

6.5 Evaluation of Package Arrays

6.5.1 Configuration

The applicant reflected the most reactive single package models on all sides to produce infinite arrays of packages, conservatively ignoring the impact limiter and neutron shield. For the normal conditions of transport analysis, the interior of the package is considered dry, as the structural analysis in Section 2.0 of the SAR demonstrates no leakage under normal conditions of transport. Under hypothetical accident conditions, the structural analysis also demonstrates no leakage; however, the applicant conservatively considers the interior of the TSC to be at optimum moderation, which is fully flooded. The applicant varied the water density between packages to find the most reactive condition, which was a dry exterior. The applicant used the subcritical arrays to determine the CSI for the package, per the requirements of 10 CFR 71.59.

6.5.2 Results

Results of the package array evaluations are given in Sections 6.10.1, 6.10.2, and 6.10.3 of the SAR for the PWR, PWR damaged fuel, and BWR TSCs, respectively. All infinite arrays under normal conditions of transport, with a dry package interior, are shown to be well below the calculated USL. Infinite array k_{eff}s for the PWR and BWR TSCs under hypothetical accident conditions are not significantly different than those for the water-reflected single package models. This indicates that the package is neutronically isolated, as expected due to the large amount of structural and shielding material between fissile material in adjacent packages. The corresponding CSI for the PWR and BWR TSCs is 0.0.

For the PWR damaged fuel TSC under hypothetical accident conditions, the case of preferential flooding of the damaged fuel cans (i.e., fully flooded damaged fuel cans with the rest of the TSC dry) was not adequately subcritical in an infinite array. Therefore, the applicant limited the calculation to a single package, resulting in a corresponding CSI of 100.

6.6 Benchmark Evaluations

6.6.1 Experiments and Applicability

Benchmark evaluations related to burnup credit in the PWR and PWR-DF TSCs are discussed in Section 6.7 of this SER. For the fresh fuel calculations in the PWR and PWR damaged fuel TSCs, the applicant used MCNP6.1 with the continuous energy ENDF/B-VII cross-section library. For the BWR TSC, the applicant used MCNP5 version 1.30 with the continuous energy ENDF/B-VI cross-section library. The applicant benchmarked both codes and cross-section libraries against 186 low enriched uranium (LEU) critical experiments, selected for LEU pins in rectangular arrays, with reactivity controlled by soluble boron or borated plates. Although the MAGNATRAN system will not rely on soluble boron for criticality control, the staff finds it acceptable to include such experiments, as it increases the number of experiments that include ^{10}B, and the USL calculated based on soluble boron concentration is not the limiting USL (see Table 6.8.1-1). The relevant parameters of these critical experiments, and the resulting validation statistics, are given in Table 6.8.1-3 of the SAR. The applicant included the uncertainties reported for each experiment in the validation statistics.

Trending analyses were performed on the critical experiment results based on the following parameters:

- ^{235}U enrichment,
- Fuel rod pitch,
- Fuel pellet outer diameter,
- Fuel rod outer diameter,
- Hydrogen to ^{235}U atomic ratio,
- Soluble boron concentration,
- Cluster gap spacing,
- Neutron absorber ^{10}B loading, and
- Energy of the average lethargy of fission

No significant correlations were identified.

6.6.2 Bias Determination

The applicant determined the bias and bias uncertainty for the fresh fuel calculations based on the k_{eff} results of the selected benchmark critical experiments, using the methodology recommended in NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages."[3] For BWR calculations using MCNP5 version 1.30 with the ENDF/B-VI continuous energy cross-section library, the applicant determined a lower limit constant USL of 0.9376. For PWR fresh fuel calculations using MCNP6.1 with the ENDF/B-VII continuous energy cross-section library, the applicant determined a lower limit constant USL of 0.9427. The range of applicability for these USLs, as defined by the relevant parameters of the critical experiments evaluated, is given in Table 6.8.1-1 of the SAR.

The staff reviewed the fresh fuel benchmarking analysis performed by the applicant and determined that the USLs were determined in accordance with relevant NRC guidance. The critical experiments chosen are appropriate for the system being evaluated, and the resulting USLs are bounding and acceptable.

6.7 Burnup Credit

6.7.1 Limits for the Licensing Basis

The MAGNATRAN package burnup credit criticality evaluations assume a maximum of 5.0 wt.% ^{235}U initial enrichment, credit a maximum burnup of 45 GWd/MTU, and take credit for a maximum of 15 years cooling time, all of which meet the guidance for the licensing basis limits in ISG-8, Rev. 3. Additionally, the 28 actinides and fission products credited by the applicant are identical to those recommended by ISG-8, Rev. 3. Therefore, the staff finds the proposed licensing basis limits to be acceptable.

6.7.2 Licensing-Basis Model Assumptions

The applicant used the t-depl sequence of the SCALE 6.1 code system, with the TRITON two-dimensional lattice physics code and the 238-group ENDF/B-VII cross-section library, for all fuel depletion calculations, as well as to determine burned RCCA compositions for use with under-burned WE15x15 fuel. The applicant estimated the isotopic composition of burned fuel as a function of burnup, enrichment, and cooling time, which was subsequently used in the criticality calculations. The model assumptions for the isotopic depletion analysis that affect the reactivity of the package include the core specific power, moderator temperature, fuel temperature, and soluble boron concentration during irradiation. These core operating parameters are given in Table 6.10.1-4 of the SAR. For moderator temperature, fuel temperature, and soluble boron concentration, the applicant determined the bounding irradiation parameters by comparison to similar values determined in NUREG/CR-6665[4], "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel. The applicant conservatively increased the recommended values from the NUREG/CR (610K from 600K for moderator temperature, 1100K from 1000K for fuel temperature, and 1000 ppmb from 750 ppmb for soluble boron concentrations). As it is difficult to determine or define a bounding power density for assembly irradiation for actinide and fission product burnup credit, the applicant assumed an average power density of 60 MW/MTU and provided an analysis to determine a bias to cover potential non-conservatism in some cases. The applicant demonstrated that {proprietary information removed} . The staff finds the applicant's treatment of power density during depletion to be conservative and acceptable.

The applicant considered the effects of exposure to BPRAs (including wet annular burnable absorbers when appropriate), RCCAs (or CEAs), and integral fuel burnable absorbers, all of which can increase a fuel assembly's discharge reactivity significantly. The applicant considered different exposures for different assembly types. For the CE14x14 and CE16x16 fuel types, the applicant depleted the fuel with the CE type CEA fully inserted for the full burnup credited. Since the CEA has the largest effect on discharge reactivity, this depletion assumption bounds all BPRA and integral fuel burnable absorbers use for these assembly types. For the WE and BW fuel types, BPRAs are assumed to be present for all three cycles of irradiation. This bounds potential integral fuel burnable absorber rods use, as BPRAs are typically inserted for only the first cycle. The staff finds this conclusion is consistent with the conclusions of NUREG/CR-6760, "Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit."[5]

For CEA exposure in WE and BW fuel types, the applicant assumes minor insertions up to 20 centimeters at full power, and demonstrates that assuming BPRA exposure for 3 cycles bounds this level of insertion (note that CEAs cannot be inserted into assemblies that contain a BPRA). The staff finds this result is consistent with the conclusions of NUREG/CR-6761, "Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit,"[6] and NUREG/CR-6759, "Parametric Study of the Effect of Control Rods for PWR Burnup Credit,"[7] and is conservative, given that PWR reactors typically operate with CEAs fully withdrawn.

For the burnup credit criticality calculations in the MAGNATRAN package, the applicant considered the effects of axial burnup distribution. The applicant assumed a burnup-dependent, 18-node axial profile, as shown in Table 6.10.1-5. The staff compared the applicant's axial profiles to the conservative profiles determined in NUREG/CR-6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses,"[8] and determined that the profiles are consistent with, or more conservative than, the profiles given in NRC guidance.

For RCCAs credited for underburned fuel, the applicant modeled RCCAs with absorber pellets consisting of 80% silver (Ag), 15% indium (In), and 5% cadmium (Cd) in a stainless steel cladding, using the SCALE 6.1/TRITON depletion code and the 238-group ENDF/B-VII cross-section library. Key absorber dimensions are given in Table 6.10.1-17. To achieve the 200 GWd/MTU burnup assumed in the analysis, the applicant performed the depletion analysis for an RCCA in a fresh WE15x15 fuel assembly out to 45 GWd/MTU, then moved the burned RCCA compositions into another fresh WE15x15 fuel assembly and performed another

45 GWd/MTU depletion. This process was repeated until the 200 GWd/MTU equivalent burnup was achieved. Only initial Ag, In, and Cd isotopes are considered in the subsequent criticality model, and metastable isotopes are not included. The RCCA compositions included in the MCNP criticality model are shown in Table 6.10.1-18 of the SAR.

The staff finds this analysis to be acceptable because the assumed 200 GWd/MTU is conservative, as U.S. PWR plants typically operate with the RCCAs completely withdrawn. The tips of the RCCAs may sometimes be inserted, and RCCAs may occasionally be fully inserted at full power in order to hold down reactivity in a damaged fuel assembly. However, actual RCCA equivalent burnups are typically significantly less than 200 GWd/MTU.

6.7.3 Code Validation – Isotopic Depletion

ISG-8, Rev. 3 states that applicants may use pre-calculated values for isotopic depletion code bias and bias uncertainty, provided the applicant uses the SCALE code system with TRITON and the 238-group ENDF/B-VII cross section library, and that the evaluated system is neutronically similar to the GBC-32 system evaluation in NUREG/CR-7108, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Isotopic Composition Predictions.”[9] The applicant used the code and cross section library referenced in ISG-8, and provided information in Section 6.8.1.2.3 of the SAR that demonstrates that the MAGNATRAN system is similar to the GBC-32 system.

For validation of the RCCA depletion analysis, there is no publicly available radiochemical assay data, which is typically used for depletion validation. In order to account for this lack of proper code validation, the applicant introduces significant conservatisms into the analysis, including:

- The RCCAs are assumed to be depleted to 200 GWd/MTU. PWRs are typically operated with all RCCAs withdrawn, meaning that typical RCCAs will experience very little, if any, depletion.

- Credited isotopes are only those isotopes initially present in material (i.e., no generated isotopes, only loss of original absorbers).

- Requirements for underburned fuel assemblies (no more than 4.05 weight percent enrichment, at least 12 GWd/MTU burnup, and transported in a TSC with a minimum neutron absorber areal density of 0.036 g $^{10}B/cm^2$) are such that the resulting k_{eff}s are significantly less than the standard fuel loading (see Table 6.10.1-22).

The staff finds, with reasonable assurance, that these conservatisms in the analysis for under-burned WE15x15 fuel with burned RCCA credit is adequate to cover any potential biases and uncertainties associated with the RCCA depletion analysis.

6.7.4 Code Validation – K_{eff} Determination

The applicant benchmarked MCNP6.1 and the ENDF/B-VII cross-section library against a series of 215 critical experiments containing mixed uranium and plutonium isotopes. These experiments included 59 experiments containing fresh mixed uranium and plutonium oxide fuel rods, as well as 156 experiments detailed in NUREG/CR-6979, “Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data.”[10] NUREG/CR-6979 demonstrates that HTC experiments are applicable for validation of burned commercial UO_2 fuel. This code and cross-section library validation analysis is applicable to the major actinide (^{235}U, ^{238}U, ^{238}Pu,

^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am) contribution to the criticality bias and bias uncertainty.

The applicant performed a trending analysis on a number of relevant parameters, including:

- energy of the average lethargy of fission,
- Water to fuel volume ratio,
- ^{234}U to ^{238}U atomic ratio,
- ^{235}U to ^{238}U atomic ratio,
- ^{238}Pu to ^{238}U atomic ratio,
- ^{239}Pu to ^{238}U atomic ratio,
- ^{240}Pu to ^{238}U atomic ratio,
- ^{241}Pu to ^{238}U atomic ratio,
- ^{242}Pu to ^{238}U atomic ratio, and
- ^{241}Am to ^{238}U atomic ratio.

No significant trends were identified. The resulting USL, determined in accordance with NUREG/CR-6361, is {proprietary information removed}. The range of applicability for this USL is given in Table 6.8.1-7 of the SAR.

For validation of the minor actinide and fission product component of the criticality code and cross-section library bias, the applicant used the methodology detailed in ISG-8, Rev. 3. Since the applicant: 1) used MCNP with the ENDF/B-VII cross-section library, 2) validated the major actinide component of the criticality code bias using applicable mixed uranium and plutonium oxide and HTC experiments, 3) demonstrated that the MAGNATRAN system is similar to the GBC-32 system evaluated in NUREG/CR-7109, "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions,"[11] and 4) demonstrated that the fission product and minor actinide worth is {proprietary information removed} . The applicant determined this bias as shown in Table 6.10.1-33 of the SAR and applied the resulting values to the USL for burnup credit criticality analysis.

For validation of MCNP for criticality analyses with burned RCCAs, the applicant modeled 41 critical experiments containing Ag-In-Cd rods in LEU lattices. These experiments make up only two sets of experiments, all with the same rods, and are therefore highly correlated. Additionally, the resulting distribution of k_{eff} values cannot be shown to be normal. {proprietary information removed} , and the lower USL for burned fuel is used instead. However, the high USL indicates that the MCNP/ENDF/B-VII calculation will likely predict RCCA Ag-In-Cd effects well and will not significantly affect the burnup credit USL when modeling RCCA material. Although this method of validation would not be acceptable for cases where there was little margin or model conservatism, the applicant has demonstrated a significant amount of conservatism in the criticality analysis for burned RCCAs. Therefore, the staff finds that using the burned fuel USL for criticality calculations that include RCCAs is acceptable.

The staff reviewed the burned fuel benchmarking analysis performed by the applicant and determined that the biases and USLs were determined in accordance with relevant NRC guidance. The staff also reviewed the selected critical experiments and finds they are appropriate for the system being evaluated and that the resulting USLs are bounding and acceptable.

6.7.5 Loading Curve and Burnup Verification

The applicant determined loading curves for the PWR and PWR damaged fuel TSCs such that system k_{eff} will be below the USL for all burnup and initial enrichment combinations. The applicant first determined the maximum enrichment at zero burnup that could be shown to be below the fresh fuel USL. Then the applicant calculated system k_{eff}, as a function of burnup and initial enrichment, for various state points, as shown in Table 6.10.1-35 of the SAR. The applicant then applied a linear fit of the k_{eff} data, within each burnup range (less than 18 GWd/MTU, between 18 and 30 GWd/MTU, and greater than 30 GWd/MTU). The applicant applied a linear fit bias, to ensure that no under-prediction of k_{eff} occurs on the loading curve, regardless of the linearity of the data.

The applicant generated loading curves for each fuel assembly hybrid type, and then collapsed these curves into limiting curves for each lattice configuration (WE14x14, WE15x15, WE17x17, CE14x14, CE16x16, BW15x15, and BW17x17). The resulting loading curves are shown in tabular form for all neutron absorber ^{10}B loadings and lattice configurations in Tables 6.10.1-12, 6.10.1-13, and 6.10.1-14 of the SAR for the PWR TSC, and in Tables 6.10.2-5, 6.10.2-6, and 6.10.2-7 of the SAR for the PWR damaged fuel TSC. Separate loading curves for under-loaded WE15x15 fuel in the 36, 35, and 33 fuel assembly configurations in the PWR TSC are shown in Table 6.10.1-16. The staff reviewed the loading curves generated by the applicant and concluded that they are determined in a conservative manner, and are therefore acceptable.

6.7.5.1 Misload Analyses

The applicant evaluated potential misloaded fuel assemblies in the PWR and PWR damaged fuel TSCs according to the recommendations for misload analyses contained in ISG-8, Rev. 3. The applicant demonstrated that 90% of the discharged fuel population in the fuel database is acceptable for loading in the MAGNATRAN system, and therefore, no moderately underburned multiple-assembly misload needs to be evaluated. The applicant performed misload analyses for a single highly underburned fuel assembly of multiple fuel types, using the underburned assembly parameters identified in Table 6.10.1-30 of the SAR.

{proprietary information removed}

The staff reviewed the applicant’s misload analysis, and finds that it is consistent with the recommendations in ISG-8, Rev. 3, and that the criticality design of the package adequately protects against the consequences of fuel assembly misloads.

6.7.5.2 Administrative Procedures

The applicant included additional administrative procedures for loading PWR and PWR damaged fuel transportable storage canisters evaluated for criticality safety using burnup credit. These procedures include:

- Verification of the location of high reactivity fuel (i.e., fresh or severely underburned fuel) in the spent fuel pool prior to and after transportable storage canister loading to ensure appropriate fuel assemblies have been loaded,
- Qualitative visual verification that a fuel assembly has been burned, performed prior to or during transportable storage canister loading operations,

- Verification of the transportable storage canister or package fuel inventory and loading records, performed under a 10 CFR 71 quality assurance program prior to shipment for previously loaded transportable storage canisters (i.e., transportable storage canisters transferred from 10 CFR 72 storage facilities),
- Fuel assemblies without visible identification are only to be loaded after quantitative burnup measurement of the fuel assembly, and
- The documented fuel assembly burnup must be decreased to account for reactor record or measurement uncertainty prior to comparison to minimum requirements in the CoC.

The staff finds these additional procedures are comparable to those recommended in ISG-8, Rev. 3, and are acceptable for reducing the likelihood and severity of misload events.

6.8 Findings

The staff finds the applicant has demonstrated that the MAGNATRAN package, when loaded with fuel assemblies meeting the characteristics of the contents described in Section 1.3 of the SAR, will be adequately subcritical under all conditions. Therefore, the applicant has shown, and the staff finds that, the MAGNATRAN package meets the fissile material requirements of §71.55 for single packages, and §71.59 for arrays of packages with a CSI of 0.0 for the PWR and BWR transportable storage canisters, and 100 for the PWR damaged fuel transportable storage canister.

6.9 References

1. NUREG-1617, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, U.S. Nuclear Regulatory Commission, March 2000.

2. U.S. Nuclear Regulatory Commission, *Division of Spent Fuel Storage and Transportation Interim Staff Guidance – 8, Revision 3 – Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Packages*, U.S. NRC, September, 2012.

3. J. J. Lichtenwalter, S. M. Bowman, M. D. DeHart, and C. M. Hopper, *Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages*, NUREG/CR-6361, (ORNL/TM-13211), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 1997.

4. C. V. Parks, M. D. DeHart, and J. C. Wagner, *Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel* NUREG/CR-6665,(ORNL/TM-1999/303, U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, February 2000.

5. C. E. Sanders and J. C. Wagner, *Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit*, NUREG/CR-6760, (ORNL/TM-2000/321), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 2002.

6. J. C. Wagner and C. V. Parks, *Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit*, NUREG/CR-6761, (ORNL/TM-2000/373), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 2002.

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8. J.C. Wagner, M.D. DeHart, and C.V. Parks, *Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses*, NUREG/CR-6801, (ORNL/TM-2001/273), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, March 2003.

9. G. Radulescu, I.C. Gauld, G. Ilas, and J.C. Wagner, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Isotopic Composition Predictions*, NUREG/CR-7108, (ORNL/TM-2011/509), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, April 2012.

10. D.E. Mueller, K.R. Elam, and P.B. Fox, *Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data*, NUREG/CR-6979, (ORNL/TM-2007/083), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, September 2008.

11. J.M. Scaglione, D.E. Mueller, J.C. Wagner, and W.J. Marshall, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions*, NUREG/CR-7109, (ORNL/TM-2011/514), U.S. Nuclear Regulatory Commission, Oak Ridge National Laboratory, April 2012.

7.0 OPERATING PROCEDURES

The staff reviewed the package loading instructions. Chapter 7 of the application included loading and unloading instructions based on the type of payload. The applicant provided procedures for preparing the MAGNATRAN package for loading.

7.1 Package loading procedures

This section describes the procedures for receipt of an empty MAGNATRAN package, preparation of the package for loading and loading of the contents, closure, and preparation of the loaded package for transport. The package operations for spent fuel contents are based on using transportable storage canisters that have already been loaded as part of dry storage operations under 10 CFR Part 72; however, a summary of operations for loading a transportable storage canister with spent fuel is included in the application to cover package operations when the TSC has not been stored pursuant to 10 CFR Part 72 prior to transport. Acceptance criteria for completion of an operation are included where appropriate.

7.1.1 Receipt and Preparation for Loading

The procedures in this section include steps needed for preparation of the package for loading. These steps include performing radiation surveys upon receipt of the empty package, visual inspection of the transport vehicle and packaging to ensure it is not damaged, cleaning of the packaging and transport vehicle, moving the package for loading.

7.1.2 Loading a Transportable Storage Canister into the Package

Prior to loading into the package, TSCs containing spent nuclear fuel will be evaluated to ensure that the neutron poisons in the basket are acceptable for transport. In addition, TSCs containing spent fuel that have been in dry cask storage will be evaluated to ensure that the TSC meets the conditions in the CoC. Package loading procedures include removing the lid and coverplate; loading both a TSC that has been in dry storage under 10 CFR Part 72 and a TSC that is placed into the package immediately after being loaded and closed. When placing a TSC in the package immediately after loading, the operating procedures contain appropriate conditions, based on the analysis in the SAR, to ensure that the maximum fuel cladding temperature is kept below 400 °C.

7.1.3 Preparation for Transport

After loading of the TSC into the package, the metallic O-ring is replaced and the outer EPDM O-ring is inspected for damage, and if needed, replaced. Operations include the package cavity spacer for packages loaded with short TSCs to ensure that the package contents are maintained in a position consistent with the shielding analysis. Placement of the lid and inspection and placement of the lid bolts, including torque values, are also specified. The package is then evacuated to a pressure of <3 Torr and backfilled with helium to a pressure of 17.5 (+2.5, -0) psia. The metallic port cover seal is replaced and the outer EPDM O-ring is inspected and replaced, if damaged. The port coverplate is replaced and the package is leak tested to ensure it is leaktight in accordance with ANSI N14.5[1].

The trunnions are reinstalled, as are the impact limiters. The package is decontaminated, the tamper indicating seal is installed and the final visual inspection of the loaded package is performed. Radiation and contamination surveys are completed. Radiation and contamination limits for different package configurations (empty package and loaded package) are provided.

The staff reviewed the limits described in the operations for the different package configurations and finds the appropriate limits are described for each package configuration.

The staff reviewed the applicant's description of package operations to ensure that they result in the package being used in accordance with the shielding design specified in the technical drawings and appropriate regulatory radiation limits. The package operations descriptions should contain the essential elements of operations for using the package. Where alternates to sequences or operations are acceptable, the operations descriptions should include these alternate sequences and operations. The staff finds that, based on its review, the operations descriptions in the application are consistent with the technical design and shielding analysis. The staff finds that, based on its review, the operations descriptions in the application are consistent with these considerations.

7.1.4 Loading a Transportable Storage Canister with Spent Fuel

The procedures for loading a TSC immediately before transport include visual inspection of the TSC; placing the TSC inside the transfer cask and transferring it to the spent fuel pool; loading the TSC in accordance with the CoC, including ensuring that the contents are appropriate for loading. The staff reviewed the descriptions of the contents limits included in the operations chapter. The staff finds that the application of the contents limits is appropriately described. This includes the assembly burnup, which is the maximum assembly average burnup, and the application of the limit on GTCC waste specific activity.

In addition, the staff reviewed the package components descriptions and, based on comparisons with the design drawings, finds the descriptions are consistent with the design description.

After loading the TSC, the lid is placed on the TSC and welded into place. The TSC is vacuum dried. The vent and drain ports are installed.

7.1.5 Loading a Transportable Storage Canister with Greater-Than-Class C Waste

The GTCC waste TSC and GTCC waste basket liner are visually inspected, then the liner is placed into the designated loading location. The GTCC waste is loaded into the waste liner and the basket liner is loaded into a TSC located in a transfer cask. The lid is placed onto the TSC and the TSC is removed from the pool. The lid is placed onto the TSC and welded in place and the TSC is vacuum dried.

The staff also evaluated the operations and sequences based on the design description, including the design drawings. As part of this evaluation, the staff noticed that the operations descriptions included certain options for the installation and welding of the closure ring and port cover plates for the GTCC transportable storage canister that are precluded by the package shipping configuration drawing (incorporated by reference into the CoC). Thus, the applicant removed discussion of these options from the package operations and made the descriptions consistent with the package shipping configuration design drawing. Based on this evaluation, the staff finds the sequences and operations are consistent with the design description.

As part of this same evaluation, the staff identified that the GTCC transportable storage canister assembly drawing also includes options for the installation of the closure ring and port covers that are precluded by the package shipping configuration drawing. The applicant did not revise the GTCC transportable storage canister assembly drawing to remove these options; however, because the shipping configuration drawing (Drawing No. 71160-500, Rev. 5P) precludes implementation of those options, the options are not allowed.

7.2 Package Unloading

NAC provided procedures for receipt and dry unloading of a package.

7.2.1 Receipt of the Package from Carrier

After receipt, the radiation and contamination surveys are performed to ensure they are within the limits in NRC regulations. The personnel barrier is removed and radiation and contamination surveys are performed on the newly-accessible package surfaces. The exterior of the package is visually inspected and cleaned. The tamper-indicating devise is inspected to ensure it is intact and then removed. The upper and lower impact limiters are removed. The trunnion plugs are removed, and their recesses are inspected for damage. The package is lifted and rotated into a vertical orientation.

7.2.2 Removal of Contents

The lid port cover is removed, and it and the bolt threads are inspected. A sample of the cavity gas inside the package body is taken and evaluated to ensure that it is acceptable for the facility's systems. After venting of the interior of the package body, the package lid bolts are removed and inspected. After removal of the lid, the TSC is removed and transferred to the transfer cask. A contamination survey of the packaging inner shell is completed. The lid and its bolts are replaced.

7.3 Preparation of Empty Packaging for Transport

The procedures for transporting an empty package include decontaminating the package inside and out to ensure that the limits in Title 49 of the Code of Federal Regulations, 173.428, "Empty Class 7 (Radioactive) Materials Packaging" are met. The upper and lower impact limiters are installed, and visual inspections of the packaging are performed, and the appropriate labels are applied.

7.5 Findings

The staff reviewed the Operating Procedures in Chapter 7 of the application to verify that the package will be operated in a manner that is consistent with its design evaluation. On the basis of its evaluation, the staff concludes that the combination of the engineered safety features and the operating procedures provide adequate measures and reasonable assurance for safe operation of the proposed package in accordance with 10 CFR Part 71. The CoC states that the package must be prepared for shipment and operated in accordance with the Operating Procedures specified in Chapter 7 of the application.

7.5 References

1. American National Standards Institute ANSI N14.5, *American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials, New York, NY, 1997.*

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

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

8.1 Acceptance Tests

8.1.1 Visual Inspections and Measurements

After fabrication, tests and measurements are performed, following package specific procedures, to ensure that the package was fabricated in accordance with the applicable design criteria provided in the application and the drawings referenced in the CoC. Packaging components are measured and inspected visually for defects and to ensure material compliance with the applicable code specifications for each component. Other tests listed in Section 8.1.1 of the SAR are discussed below.

8.1.2 Welding and Weld Examinations

Welds are inspected as discussed in Section 2.3, “Codes, Standards, Fabrication and Examination,” above, to ensure compliance with applicable codes.

8.1.3 Structural and Pressure Tests

Three structural and pressure tests are performed on the package: The structural and pressure tests include: (1) load testing of the lifting trunnions; (2) load testing of the rotation trunnions; and (3) hydrostatic pressure testing of the transport cask containment boundary.

The lifting trunnions are tested using a test load equivalent to 300% of the maximum service load (weight of the heaviest, loaded transport cask during in-plant handling) which is applied to the pair of trunnions. The test load is held for a minimum of 10 minutes. After completion of the load test, the trunnions are removed from the package body and all trunnion and bolt load-bearing surfaces are visually inspected for permanent deformation, galling or cracking using the acceptance criteria for visual inspections in the ASME B&PV Code, Section III, Subsection NF, Article NF-5360[1]. In addition, the trunnion load-bearing surfaces are also liquid penetrant (PT) examined in accordance with the ASME B&PV Code, Section V, Articles 1 and 6, with acceptance per Section III, Subsection NF, Article NF-5350.

The rotation trunnions are tested utilizing a vertical load equivalent to 150% of the maximum design transport weight of the MAGNATRAN package. After completion of the rotation trunnion load test, all rotation trunnion welds and loadbearing surfaces shall be visually inspected for permanent deformation, galling or cracking. A liquid penetrant examination of the rotation trunnion welds and load-bearing surfaces is performed in accordance with the ASME B&PV Code, Section V, Article 6, using the acceptance standards in the ASME B&PV Code, Section III, Subsection NF, Article NF-5350.

The MAGNATRAN package containment boundary components are pressure tested to ensure compliance with 10 CFR 71.85(b). Each MAGNATRAN package is pressurized to 150 (+20, -0) psig and held for a minimum of 10 minutes. Following the completion of the 10-minute hold time, and while the containment boundary is under pressure, a visual inspection of the containment vessel welds will be conducted to detect evidence of leakage. In addition, all containment boundary components, welded joints, connections and regions of high stress will be visually examined to verify that no permanent deformation or breach of the containment boundary resulted from the test.

All accessible containment boundary welds will be liquid penetrant examined per the ASME B&PV Code Section V, Article 6, with acceptance criteria per ASME B&PV Code, Section III, Subsection NB, Article NB-5350. Any identified defects must be repaired and the hydrostatic test repeated, in accordance with the original test requirements and acceptance criteria, prior to final acceptance.

8.1.4 Leakage Tests

Fabrication leakage rate tests demonstrate that the containment system provides the required level of containment. Fabrication leakage rate tests are performed on the containment system boundary (consisting of the containment shell and containment baseplate (bottom forging), the containment closure flange (top forging), and the closure lid, and the closure lid access port cover plate) as described in Section 8.1.4 of the application, which also specifies the allowable leakage rates and test sensitivities. The package containment components are tested to the helium leaktight criterion as specified in ANSI N14.5-1997[2].

Pre-shipment leakage rate tests are performed before each shipment, after the contents are loaded and the containment system is assembled. Pre-shipment leakage rate tests are performed on all containment seals, as identified in Section 8.2.2.3 of the application. The allowable leakage rates and test sensitivities are described in Section 8.2.2.4 of the application and all seals are tested to the helium leaktight criterion because all metallic seals are replaced prior to each shipment.

Periodic leakage rate tests demonstrate that the containment capabilities of the MAGNATRAN package built to an approved design have not deteriorated over an extended period of use. Periodic leakage rate tests are performed on all containment seals as is described in Section 8.2.2.1 of the application. The periodic leakage rate tests are valid for one year.

Maintenance leakage rate tests confirm that any maintenance, repair, or replacement of components has not degraded the containment system and are performed prior to returning a package to service. Maintenance leakage rate tests are performed on the containment system boundary consisting of the containment shell and containment baseplate (bottom forging), the containment closure flange (top forging), and the closure lid, and the closure lid access port cover plate as illustrated in Section 8.2.2.2 of the application.

The allowable leakage rates and test sensitivities are presented in Section 8.2.2.4 of the application and all containment components are tested to the helium leaktight criterion. Further, as identified in the application, helium leakage test procedures shall be prepared and approved by qualified personnel in accordance with the requirements of SNT-TC-1A[3] for Level III non-destructive testing engineer for leak testing.

8.1.5 Component and Material Tests

8.1.5.1 Gaskets

The package lid and lid port coverplate metallic O-rings are replaced and helium leak tested in accordance with ANSI N14.5-1997 prior to each shipment, except when the package is transported empty, in accordance with 49 CFR 173.428. The outer EPDM O-rings are replaced annually, or as required based on visual inspections performed as required in Chapter 7, “Operating Procedures.”

8.1.5.2 Neutron Absorber Tests

Qualification tests are used to demonstrate suitability and durability for a specific application. The applicant presented specifications that will be used to qualify a new borated material or changes to an existing borated material. Qualification testing is required for neutron absorber material specifications that were:

(1) not previously qualified;
(2) previously qualified, but manufactured by a new supplier; and
(3) previously qualified, but with changes in key process controls.

Key process controls for producing the neutron absorber material used for qualification testing shall be the same as those used for commercial production. Qualification testing shall demonstrate consistency between lots (2 minimum). The applicant has stated that non-conforming material shall be evaluated within NAC International’s quality assurance program.

Qualification tests of the neutron absorber material, described by the applicant in the SAR, include the following:

- Thermal conductivity qualification testing of the neutron absorber materials shall conform to ASTM E1225 ASTM E1461, or an equivalent method.
- Yield strength qualification testing of the neutron absorber shall conform to ASTM Test Method B557/B557M, E8 or E21.
- The ^{10}B areal density is measured through neutron attenuation testing using a collimated thermal neutron beam. Based on the MAGNASTOR required minimum effective areal density of ^{10}B – 0.036, 0.030 or 0.027 g/cm^2 for the PWR basket and 0.027, 0.0225 or 0.020 g/cm^2 for the BWR basket, the minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level (also expressed as 95/95 level) or better.

The required minimum actual ^{10}B loading in a neutron absorber sheet is determined based on the effectiveness of the material—75% for Boral and 90% for borated aluminum alloys and for borated metal matrix composite. All neutron absorber material acceptance verification will be conducted in accordance with the NAC International’s quality assurance program.

The Boral neutron absorbing material is an aluminum matrix material formed from aluminum and boron-carbide. The mixing of the aluminum and boron-carbide powder forming the neutron absorber material is controlled to assure the required ^{10}B areal density. The constituents of the neutron absorber material shall be verified by chemical testing and by dimensional measurement to ensure the quality of the finished plate or sheet. The results of all neutron absorber material tests and inspections, including the results of wet chemistry coupon testing, are documented and become part of the quality assurance records documentation package for the fuel tube and basket assembly.

8.1.6 Shielding Tests

The staff reviewed the acceptance tests important to ensuring the shielding in the as-fabricated package meets the design specified in the technical drawings and evaluated in the shielding analysis, at the time of fabrication. These tests include dimension and material specifications and tests described in Sections 8.1.1, 8.1.5.1, and 8.1.8 of the application and the neutron and gamma shielding tests described in Section 8.1.6 of the application.

The package shielding design includes the package's steel shells, lids and base. It also includes the radial lead shielding and the neutron shield assemblies that are attached to the package's outer steel shell. The shielding effectiveness of the package is ensured, in part, by confirming that the package components are fabricated to the dimensional and material specifications, including tolerances, described in the design drawings that are part of the CoC. Shielding effectiveness is also ensured, in part, by confirming the package components do not have any defects and that the package and its components are fabricated and assembled properly.

The staff reviewed the acceptance tests described in Sections 8.1.1, 8.1.5.1, and 8.1.8 of the application and finds that these tests are adequate for these purposes for most of the package components that provide or relate to the package's shielding function. This includes the integrity of the neutron shield assemblies' steel enclosure shell.

Additional tests are needed for the lead gamma shield and the neutron shield material. The lead gamma shield is poured into the cavity formed by the package's inner and outer steel shells and the top forging and package base. This process must be done with good controls to ensure against formation of voids in the lead and that the lead adequately fills the cavity and provides the required shielding.

Section 8.1.6.1 describes the acceptance testing that is done to confirm the adequacy of the lead shielding. The scan uses a Co-60 source large enough to provide count rates that are at least three times background. The test also specifies requirements for scan path spacing, scanning speed, and the scanning grid pattern. The staff reviewed these proposed requirements and, based on its understanding of radiation measurement equipment features and properties and interactions with health physicists, the staff finds these proposed requirements to be acceptable.

The gamma scan is conducted over the surface of the package where the lead shielding is located. The scan is conducted prior to the attachment of neutron shield assemblies and cooling fins. The acceptance criterion is that the count rates must not exceed the count rates measured on a mockup that represents the same configuration as for the gamma scan with steel and lead layers that are at the minimum thicknesses, including application of tolerances that minimize the package steel and lead thicknesses, allowed in the package's design drawings. The staff finds this acceptance criterion to be adequate because it is the same configuration as the package's configuration for the gamma scan and it is tied to the package design and its allowable dimensions and tolerances.

Areas of the package body that exceed the acceptance criterion will be evaluated to determine the appropriate corrective action. Corrective actions include repairs of that area or component of the package. Once repaired, the repaired areas and the areas surrounding the repaired area that may have also been affected by the repair will be retested. For some types of repairs, package areas surrounding the repaired area may be affected; thus, the staff finds that a retest of the package that includes the areas surrounding the repaired area is appropriate and necessary. Based on the inclusion of those areas in the retest, the staff finds the description of repair acceptance testing to be acceptable.

For the neutron shield, the acceptance tests in Section 8.1.6.2 include a chemical analysis to confirm the composition and density of the neutron shield material. The tests description also includes statements that the installation will be done using qualified procedures. While these kinds of tests and procedures are useful, they are, by themselves, not sufficient to show the neutron shielding of the as-fabricated package performs as designed.

Therefore, the acceptance tests include a shielding effectiveness test in Section 8.1.6.3. This test includes both the neutron and the gamma shielding of the package. This test is performed upon the first loading of the package with spent fuel contents. The test includes gamma and neutron measurements at multiple distinct locations along the axial height of the neutron shield as well as multiple areas above and below the neutron shield on the package side and at points on the package's top and bottom surfaces (i.e., on the impact limiters).

These measurements are taken on the package surface and at 2.3 meters from the package surface (equivalent to 2 meters from the vehicle edge for the package side). These measured dose rates are then compared to calculated dose rates for the package containing those specific contents. For the package to be acceptable for use, the measured dose rates must not exceed the calculated dose rates. Measurements that exceed the calculated dose rates will require appropriate corrective actions to be taken.

For the calculated dose rates, the parameters of the loaded contents (e.g., fuel type, enrichment, burnup, cooling time) are used to determine the neutron and gamma source terms. These source terms are then used in the calculation of gamma and neutron dose rates. The calculations use a package configuration that represents the minimum shielding effectiveness of the package, including geometric and material tolerances, as described in Chapter 5 of the application.

Based on its review, it is not clear to the staff that referencing Chapter 5 of the application is sufficient to specify the package configuration needed to determine this test's acceptance criteria (i.e., the calculated dose rates). This is in part due to that aspect of the application not being incorporated by reference (into the CoC). Thus, the staff has added a condition to the CoC to state the package configuration for these calculations be one that represents the minimum shielding effectiveness of the package, including tolerances, as defined by the package design drawings in the CoC. The configuration currently used in the Chapter 5 analyses is consistent with that condition; however, this CoC condition ensures the method for determining the acceptance criteria is part of the CoC and remains consistent with the design as it has been evaluated and approved for the current application.

The staff finds the use of calculated dose rates for the particular set of loaded contents to be acceptable as the test acceptance criteria. The basis of this finding is that these criteria, as described above, represent the package's as-designed shielding capability and demonstrate that for the specific set of loaded contents, the package's shielding performs as designed. Use of regulatory dose rate limits as acceptance criteria and the pre-shipment measurements to meet the requirements in 10 CFR 71.87(j) does not necessarily do this, particularly for contents that are not expected to result in package dose rates at or near the regulatory limits. This kind of acceptance test, while not ideal, is consistent with the acceptance tests the staff has approved for other certified spent fuel package designs; thus, the staff finds the proposed test to be acceptable.

8.1.7 Thermal Acceptance Test

SAR Section 8.1.5.3.8 described the thermal conductivity testing for the metal matrix and borated aluminum neutron absorbers and included a table of the acceptable radial and axial thermal conductivities for the Type 1 and Type 2 neutron absorbers.

SAR Section 8.1.7 provided a high-level overview of a thermal acceptance test for each MAGNATRAN transport cask, in which electric heaters would apply a heat flux equivalent to 23 kW to the inner surface of the package cavity. According to the SAR, this test would verify that the fabricated and assembled MAGNATRAN transport cask has the heat rejection capabilities (e.g., considers fin performance and actual gap dimensions) evaluated in the SAR's thermal analyses using the SAR's thermal models. Twelve thermocouples, located on the package's inner and external surfaces, as denoted in SAR Figure 8.1-1, would record temperatures.

These temperatures would be compared with calculated temperatures from the three-dimensional, full length, ANSYS® model described in the SAR's thermal chapter with boundary conditions that reflect test conditions, including application of a uniform heat flux applied to the cask inner shell, convection film coefficient based on the ambient temperature, and no solar insolation. SAR Sections 8.1.7.2 and 8.1.7.3 note that the package thermal test acceptance is based on comparing temperatures, temperature gradients, and heat rejection rates with the computational model.

Although SER Section 3.4.1 notes potential issues with the thermal models described in the application, the applicant's quality assurance program (discussed in SAR Section 1.4.2) will be applied to the thermal acceptance tests; thus, the applicant is committed to ensure that the detailed implementation of the thermal acceptance test will result in valid results from which technically rigorous comparisons with correct calculated results can be made.

8.2 Maintenance Programs

Table 8.2-1 provides a list of the MAGNATRAN maintenance schedule, including activity and frequency. The tests include visual and weld examinations, leak testing, and part replacement as appropriate.

8.2.1 Structural and Pressure Tests

The two lifting trunnions and associated bolts, lifting trunnion recesses, and the two rotation trunnions will be visually inspected prior to each shipment. In addition, the lifting and rotation trunnions will be inspected annually (a period not to exceed 14 months while the package is in use) in accordance with Paragraph 6.3.1(b) of ANSI N14.6[4], which includes visual inspection of all trunnion welds, load-bearing surfaces and attachment bolting for permanent deformation, galling or cracking; liquid penetrant examinations of welds and load-bearing surfaces will be performed in accordance with the ASME B&PV Code, Section V, Article 6. Liquid penetrant acceptance standards are those listed in ASME B&PV Code, Section III, Division 1, Subsection NF, Article NF-5350.

8.2.2 Leakage Tests

Periodic and maintenance leak rate testing is discussed above in Section 8.1.4

8.2.3 Miscellaneous Tests

The staff reviewed the maintenance programs that are important to ensuring the shielding in the as-fabricated package will continue to meet the design specified in the technical drawings and evaluated in the shielding analysis over the course of its service life. These programs include those described in Sections 8.2.3 and 8.2.5 of the application related to the package's shielding components.

While the package's gamma and neutron shielding are not expected to degrade over time, the applicant did include a test, in the Section 8.2.3 maintenance programs, to ensure continued effectiveness of the package's shielding. The test is to be conducted every 5 years or prior to the next loaded shipment. The test is the same as the shielding effectiveness test described in Section 8.1.6.3 of the application, just with the calculated dose rates (i.e., acceptance criteria) based on the contents in the package at the time of the maintenance test.

The staff condition regarding the package configuration for calculating the dose rates for the Section 8.1.6.3 acceptance test also applies to the dose rate calculations for the Section 8.2.3 maintenance test. This test is consistent with the maintenance tests for other certified spent fuel packages. Based on that consideration and the staff's evaluation of the acceptance test, the staff finds the proposed maintenance test to be acceptable.

Section 8.2.3 also includes examination of the sealed neutron shield assemblies every 5 years or upon identification of potential component damage. This test is the same test that is described in Section 8.1.8 of the application. In Addition, Section 8.2.5 describes other inspection and maintenance activities related to damage and repair of damage. The repaired areas of the package must conform to the CoC drawings and CoC conditions.

8.3 Evaluation Findings

Based on the statements and representations in the application, the staff concludes that the acceptance tests and maintenance program for the packaging meet the requirements of 10 CFR Part 71. Further, the CoC has been conditioned to specify that each package must meet the acceptance tests and maintenance program in Chapter 8 of the application as described above.

8.4 References

1. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, New York, NY.

2. American National Standards Institute ANSI N14.5, American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials, New York, NY, 1997.

3. American Society for Nondestructive Testing (SNT), Recommended Practice No. SNT-TC-1A, "Personnel Qualification and Certification in Nondestructive Testing," Columbus OH, 2016.

4. American National Standards Institute ANSI N14.6, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (45000 kg) or More for Nuclear Materials, New York, NY, 1993."

9.0 CONDITIONS

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

Condition No. 6 states that in addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented, except that the minimum component thicknesses for the mockup in Section 8.1.6.1 and the minimum shielding effectiveness configuration for calculating the dose rates used as acceptance criteria for the tests in Sections 8.1.6.3 and 8.2.3 are defined by the component dimensions and tolerances in the drawings listed in Condition 5.(a)(3).

Condition No. 7 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 No. 8 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 No. 9 states that transport by air is not authorized.

Condition No. 10 states that the package authorized by this CoC is hereby approved for use under the general license provisions of 10 CFR 71.17.

Condition No. 11 states that the CoC's expiration date is April 30, 2024.

CONCLUSION

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

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