Enclosure 7 to TN E-29128

Replacement and New NUHOMS®-MP197 Safety Analysis Report Pages, Revision 7 (for the Non-proprietary version)

NON-PROPRIETARY

AREVA

TRANSNUCLEAR INC.

NUHOMS®-MP197 TRANSPORT PACKAGING SAFETY ANALYSIS REPORT

Revision 7

TRANSNUCLEAR INC.
7135 Minstrel Way • Suite 300 • Columbia, MD 21045

CHAPTER 7
OPERATING PROCEDURES

This chapter contains NUHOMS®-MP197 loading and unloading procedures that are intended to show the general approach to cask operational activities. A separate Operations Manual (OM) will be prepared for the NUHOMS®-MP197 to describe the operational steps in greater detail. The OM, along with the information in this chapter, will be used to prepare the site-specific procedures that will address the particular operational considerations related to the cask.

7.1 Procedures for Loading the Package

The NUHOMS®-MP197 Cask will be used to transport fuel off-site. This mode of use requires (1) preparation of the cask for use; (2) verification that the fuel assemblies to be loaded meet the criteria set forth in this document; and (3) installation of a DSC and fuel assemblies into the cask.

Offsite transport involves (1) preparation of the cask for transport; (2) assembly verification leakage-rate testing of the package containment boundary; (3) placement of the cask onto a transportation vehicle; and (4) installation of the impact limiters.

During shipment, the packaging contains up to 61BWR spent fuel assemblies in the NUHOMS®-61BT DSC. Procedures are provided in this section for transport of (1) the cask/DSC directly from the spent plant fuel pool and (2) transport of a NUHOMS®-61BT DSC after storage in a NUHOMS® Horizontal Storage Module (HSM). A glossary of terms used in this section is provided in Section 7.1.6.

7.1.1 Preparation of the NUHOMS®-MP197 Cask for Use

Procedures for preparing the cask for use after receipt at the site are provided in this section.

a. Remove the impact limiter attachment bolts from each impact limiter and remove the impact limiters from the cask. Wash the cask and impact limiters to remove mud dirt & grime and touch-up paint as required.

b. Anytime prior to removing the lid, sample the cask cavity atmosphere through the vent port. Flush the cask interior gases to the site radwaste systems if necessary.

c. Remove the personnel barrier(s) from the transport frame.

d. Remove the transportation skid tie down straps.

7.1.2 Wet Loading the NUHOMS®-MP197 Cask and 61BT DSC

The procedure for wet loading the cask and 61BT DSC is summarized in this section. This procedure is intended to describe the type and quality of work performed to load and seal a DSC. *Site specific conditions and requirements may require the use of different equipment and ordering of steps other than those described below to accomplish the same objectives or acceptance criteria which must be met to ensure the integrity of the package.* The NUHOMS®-MP197 Cask is designed to transport one NUHOMS-61BT DSC containing 61 BWR fuel assemblies. All fuel assembly locations are to be loaded with design basis fuel assemblies. Verification that the burnup, enrichment, and cooling time of the assemblies are all within acceptable ranges will be performed by site personnel, prior to shipment, as discussed below. All basket compartments must be filled with a fuel assembly or a dummy fuel assembly as specified in the C of C.

7.1.2.1 Preparation of the Transport Cask and DSC

a. Verify the basket type A, B, or C, by inspecting the last digit of the serial number on the grapple ring at the bottom of the DSC.

b. Verify that the fuel assemblies to be placed in the DSC meet the maximum burnup, maximum initial enrichment, minimum cooling time, and maximum decay heat limits for fuel assemblies as specified in Section 1.2.3 of this document and the C of C. The enrichment limit must correspond to the basket type verified in step "a".

c. Using a suitable prime mover, position the cask and onsite transfer trailer below the plant crane.

d. Remove the onsite transfer trailer trunnion block covers.

e. Engage the cask front trunnions with the lifting yoke using the plant crane, rotate the cask to a vertical orientation, lift the cask from the onsite transfer trailer, and place the cask in the plant decon area.

f. Place scaffolding around the cask so that the top closure lid and surface of the cask are easily accessible to personnel.

g. Remove the top closure lid and examine the cask cavity for any physical damage and ready the cask for service.

h. Examine the DSC for any physical damage which might have occurred since the receipt inspection was performed. The DSC is to be cleaned and any loose debris removed.

i. Using a crane, lower the DSC into the cask cavity by the internal lifting lugs and rotate the DSC to match the cask and DSC alignment marks.

j. Fill the cask-DSC annulus with clean, demineralized water. Place the inflatable seal into the upper cask liner recess and seal the cask-DSC annulus by pressurizing the seal with

q. Move the onsite transfer trailer and cask to a low-dose maintenance area.

r. Inspect the cask hardware (including vent/drain/test port plugs) for damage that may have occurred during transportation. Repair or replace as necessary.

7.2.3 Unloading the NUHOMS®-MP197 Cask to a Fuel Pool

The procedure for unloading the cask and DSC into a fuel pool is summarized in this section. This procedure is intended to describe the type and quality of work performed to unload a DSC. *Site specific conditions and requirements may require the use of different equipment and ordering of steps other than those described below to accomplish the same objectives or acceptance criteria which must be met to ensure the integrity of the package.*

a. Tow the onsite transfer trailer to the fuel receiving area.

b. Remove the onsite support skid pillow block covers.

c. Using the cask lifting yoke, engage the front trunnions, rotate the cask to a vertical orientation, lift the cask from the onsite support skid, and place the cask in the decon pit.

d. Sample the cask cavity atmosphere through the vent port. Flush the cask interior gases to the site radwaste systems if necessary.

e. Remove the bolts from the cask lid and lift the lid from the cask.

f. Remove and discard the cask lid seals.

g. Locate the DSC siphon and vent ports using the indications on the DSC outer top cover plate.

h. Drill a hole in the DSC outer top cover plate and remove the siphon closure plug to expose the siphon port quick connect.

i. Drill a hole in the DSC outer top cover plate and remove the vent closure plug to expose the vent port quick connect.

j. Sample the DSC cavity atmosphere. If necessary, flush the DSC cavity gases to the site radwaste systems.

k. Fill the DSC with fuel pool or equivalent water through the siphon port with the vent port open and routed to the plant's off-gas system or other appropriate system.

Master Table of Contents

Chapter A.1

Appendix A.1.4.1

Appendix A.1.4.2

Appendix A.1.4.3

Appendix A.1.4.4

Appendix A.1.4.5

Appendix A.1.4.6

Appendix A.1.4.7

Appendix A.1.4.8

Appendix A.1.4.9

Appendix A.1.4.9A

Appendix A.1.4.10

Chapter A.2

Appendix A.2.13.1

Appendix A.2.13.2

Appendix A.2.13.3

Appendix A.2.13.4

Appendix A.2.13.5

Appendix A.2.13.6

Appendix A.2.13.7

Appendix A.2.13.8

Appendix A.2.13.9

Appendix A.2.13.10

Appendix A.2.13.11

Appendix A.2.13.12

Appendix A.2.13.13

Chapter A.3

Chapter A.4

Chapter A.5

Chapter A.6

Appendix A.6.5.1

Appendix A.6.5.2

Appendix A.6.5.3

Appendix A.6.5.4

Appendix A.6.5.5

Appendix A.6.5.6

Appendix A.6.5.7

Chapter A.7

Appendix A.7.7.1

Appendix A.7.7.2

Appendix A.7.7.3

Appendix A.7.7.4

Appendix A.7.7.5

Appendix A.7.7.6

Appendix A.7.7.7

Appendix A.7.7.8

Appendix A.7.7.9

Chapter A.8

Chapter A.1
General Information

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

Chapter A.1
General Information

NOTE: References in this Chapter are shown as [1], [2], etc. and refer to the reference list in Section A.1.3.

A.1.1 Introduction

This Appendix A to the Safety Analysis Report (SAR) for the NUHOMS®-MP197 Transport Cask presents the evaluation of a modified version of the *NUHOMS®* MP197. This modified version is a Type B(U) spent fuel transport packaging developed by Transnuclear, Inc. and designated as Model Number NUHOMS®-MP197HB packaging. Appendix A of this SAR describes the design features and presents the safety analyses which demonstrate that the NUHOMS®-MP197HB packaging complies with applicable requirements of 10 CFR 71 [1]. The format and content of Appendix A follow the guidelines of Regulatory Guide 7.9 [2].

The NUHOMS®-MP197HB packaging consists of the NUHOMS®-MP197HB Transport Cask, which is utilized for the off-site transport of one of several NUHOMS® Dry Shielded Canisters (DSCs) or a secondary container with dry irradiated and/or contaminated non-fuel bearing solid materials in accordance with 10 CFR 71 [1]. The packaging is intended to be shipped as exclusive use. The criticality safety index (CSI) for nuclear criticality control for the packaging when transporting fuel is determined to be zero (0) in accordance with 10 CFR 71.59 *[1]*. See Chapter A.6 for details of this determination.

Transnuclear, Inc. has an NRC approved quality assurance program (Docket Number 71-0250) which satisfies the requirements of 10 CFR 71 Subpart H [1].

A.1.2 Package Description

A.1.2.1 Packaging

The NUHOMS®-MP197HB packaging can be used to transport several types of Boiling Water Reactor (BWR) fuel assemblies with or without fuel channels or Pressurized Water Reactor (PWR) fuel assemblies with or without control components. The fuel assemblies are contained in a single NUHOMS® DSC. In addition the NUHOMS®-MP197HB packaging can be used to transport a secondary container with dry irradiated and/or contaminated non-fuel bearing solid materials. The NUHOMS®-MP197HB packaging is designed for a maximum heat load of 32 kW depending on the NUHOMS® DSC being transported and cask configuration. The fuels that may be transported in the NUHOMS®-MP197HB packaging are presented in Section A.1.2.3. The dry irradiated and/or contaminated non-fuel bearing solid materials that may be transported in the NUHOMS®-MP197HB packaging are also presented in Section A.1.2.3.

The NUHOMS®-MP197HB packaging is shown in Figure A.1-1 and consists of the following components:

- A NUHOMS®-MP197HB transport cask consists of a containment boundary, structural shell, gamma shielding material, and solid neutron shield. The containment boundary consists of a cylindrical shell, bottom plate with a ram access penetration, cask body flange, bottom and top cover plates (lids) with associated seals and bolts, and vent and drain port closure bolts and seals. The transport cask cavity also contains an inert gas atmosphere.

- Because there are two different Outside Diameters (ODs) for the DSCs and secondary containers, an aluminum inner sleeve is used for smaller diameter DSCs and secondary containers. The inner sleeve is designed with slots to accommodate the existing rails inside the cask and to provide rails inside the sleeve on which the smaller diameter DSCs or secondary containers slide during horizontal loading or unloading of the cask.

- To accept the varying lengths of the DSCs and secondary containers, stainless steel or aluminum spacers are provided to limit axial movement of the payload.

- *For a* NUHOMS®-69BTH *DSC* with heat load greater than 26 kW, removable external fins are provided as an option for the cask. *The use of these fins is optional.* The aluminum fins, if used, are attached to an outer aluminum sleeve which is fabricated in two halves which are bolted together around the cask between the impact limiters.

- Sets of removable front and rear trunnions which are bolted to the outer shell of the cask provide support, lifting, and rotation capability for the NUHOMS®-MP197HB cask.

- Impact limiters consisting of balsa and redwood encased in stainless steel shells are attached to each end of the NUHOMS®-MP197HB cask during shipment. A thermal shield is provided between each impact limiter and the cask to minimize heat transfer to the impact limiters. Each impact limiter is held in place by twelve (12) attachment bolts.

- A personnel barrier is mounted to the transport frame to prevent unauthorized access to the cask body.

- There are nine DSC designs authorized for transport in the NUHOMS®-MP197HB packaging. All of the DSCs consist of a cylindrical shell, and top and bottom shielded closure assemblies. Details for each DSC type are provided in Appendices A.1.4.1 through A.1.4.9. After loading, each DSC is vacuum dried and back-filled with an inert gas. Each DSC includes a fuel basket assembly, located inside the DSC. The basket assembly locates and supports the fuel assemblies, transfers heat to the DSC wall, and provides neutron absorption to satisfy nuclear criticality requirements. For some DSC designs, a basket hold down ring is installed on top of the basket, after fuel loading, to prevent axial motion of the basket within the DSC.

- The dry irradiated and/or contaminated non-fuel bearing solid materials are contained in a secondary container (Radioactive Waste *Canister* (RWC)). The safety analysis of this configuration takes no credit for the containment provided by the *RWC.*

A.1.2.1.1 NUHOMS®-MP197HB Transport Cask

The cask is fabricated primarily of nickel-alloy steel (NAS). Other materials include the cast lead shielding between the containment boundary inner shell and the structural shell, the O-ring seals, the borated resin neutron shield and the carbon steel closure bolts. Socket headed cap screws (bolts) are used to secure the lid to the cask body and the ram access closure plate to the bottom of the cask. The body of the cask consists of a 1.25 inch thick, 70.50 inch inside diameter NAS inner (containment) shell and a 2.75 inch thick, 84.50 inch outside diameter NAS structural shell which sandwich the 3.00 inch thick cast lead shielding material.

The overall dimensions of the NUHOMS®-MP197HB packaging are 271.25 inches long and 126.00 inches in diameter with both impact limiters installed. The transport cask body is 210.25 inches long and 84.50 inches in diameter. The cask diameter including the radial neutron shield is 97.75 inches or 104.25 inches with the fins. The *minimum length of* cask cavity is 199.25 inches and 70.50 inches in diameter without the sleeve or 68 inches with the sleeve. Detailed design drawings for the NUHOMS®-MP197HB packaging are provided in Appendix A.1.4.10, Section A.1.4.10.1. The materials used to fabricate the packaging are shown in the Parts List on Drawing MP197HB-71-1002. Where more than one material has been specified for a component, the most limiting properties are used in the analyses in the subsequent sections of this appendix to the SAR.

The maximum gross weight of the loaded package is 152.0 tons including a maximum payload of 56.0 tons. Table A.1-1 summarizes the dimensions and weights of the NUHOMS®-MP197HB packaging components. Trunnions, attached to the cask body, are provided for lifting and handling operations, including rotation of the packaging between the horizontal and vertical orientations. The NUHOMS®-MP197HB packaging is transported in the horizontal orientation, on a specially designed shipping frame, with the lid end facing the direction of travel.

DSCs with a spent fuel payload are shipped dry in a helium atmosphere. Both the transport cask cavity and the DSC cavity are filled with helium. The heat generated by the spent fuel assemblies is rejected to the environment by conduction, convection and radiation. No forced cooling is required.

RWCs with dry irradiated and/or contaminated non-fuel bearing solid materials are shipped dry in an air, nitrogen or inert gas environment. When a wet load procedure (i.e., in-pool) is followed for cask loading, the *RWC* and cask cavities are drained and dried in order to ensure that free liquids do not remain in the package during transport. The heat generated by the contents of the *RWC* is rejected to the environment by conduction, convection and radiation. No forced cooling is required.

A. Containment Vessel

The cask containment boundary consists of the inner shell, a 6.50 inch thick bottom plate with a 28.88 inch diameter, 2.50 inch thick ram access closure plate, a cask body flange, a 4.50 inch thick lid with lid bolts, vent and drain port closures and bolts, and O-ring seals for each of the penetrations. A 70.50 inch diameter, 199.25 inch long cavity is provided.

The containment vessel prevents leakage of radioactive material from the cask cavity. It also maintains an inert atmosphere (helium) in the cask cavity. Helium *within the DSCs* assists in heat removal and provides a non-reactive environment to protect fuel assemblies against fuel cladding degradation. To preclude air in-leakage, the cask cavity is pressurized with helium to above atmospheric pressure.

The inner containment shell is SA-203, Grade E, and the bottom and top flange materials are SA-350-LF3. The lid is constructed from SA-350-LF3 or SA-203, Grade E. The NUHOMS®-MP197HB packaging containment vessel is designed, fabricated, examined and tested in accordance with the requirements of Subsection NB [3] of the ASME Code to the maximum practical extent. In addition, the design meets the requirements of Regulatory Guides 7.6 [5] and 7.8 [6]. *Alternatives* to the ASME Code are discussed in Chapter A.2, Appendix A.2.13.13. The construction of the containment boundary is shown in Drawings MP197HB-71-1002, -1003, -1004, -1005 and -1006 provided in Appendix A.1.4.10, Section A.1.4.10.1. The design of the containment boundary is discussed in Chapter A.2 and the fabrication requirements (including examination and testing) of the containment boundary are discussed in Chapter A.4.

B. Gamma and Radial Neutron Shielding

The lead and steel shells of the transport cask provide shielding between the fuel and the exterior surface of the package for the attenuation of gamma radiation (Drawings MP197HB-71-1002, -1003, -1004, -1005 and -1006).

Neutron shielding is provided by a borated resin compound surrounding the outer shell. The resin compound is cast into long, slender aluminum containers. The containers are constructed from 6063-T651 aluminum. The total thickness of the resin and aluminum is 6.25 inches. The array of resin-filled containers is enclosed within a 0.375 inch thick outer steel shell

A.1.2.3.2 *Radioactive Waste Canister*

The NUHOMS®-MP197HB packaging is also licensed to transport a RWC. The RWC is designed to carry dry irradiated and/or contaminated non-fuel-bearing solid materials. Details of the RWC are provided in Appendix A.1.4.9A.

A.1.3 References

1. 10 CFR 71, Packaging and Transportation of Radioactive Material.

2. USNRC Regulatory Guide 7.9, "Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material," Rev. 2, March 2005.

3. American Society of Mechanical Engineers, ASME Boiler And Pressure Vessel Code, Section III, Division 1 - Subsection NB, 2004 edition including 2006 Addenda.

4. Not Used.

5. USNRC Regulatory Guide 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessel," Rev. 1, March 1978.

6. USNRC Regulatory Guide 7.8, "Load Combinations for the Structural Analysis of Shipping Cask," Rev. 1, March 1989.

7. ANSI N14.6-1993, "American National Standard For Radioactive Materials-Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More," American National Standards Institute, Inc., New York, New York.

8. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Subsection NF, 2004 edition including 2006 *Addenda.*

A.1.4 Appendices

A.1.4.1 NUHOMS®-24PT4 DSC
A.1.4.2 NUHOMS®-32PT DSC
A.1.4.3 NUHOMS®-24PTH DSC
A.1.4.4 NUHOMS®-32PTH DSC
A.1.4.5 NUHOMS®-32PTH1 DSC
A.1.4.6 NUHOMS®-37PTH DSC
A.1.4.7 NUHOMS®-61BT DSC
A.1.4.8 NUHOMS®-61BTH DSC
A.1.4.9 NUHOMS®-69BTH DSC
A.1.4.9A Radioactive Waste Canister
A.1.4.10 Drawings of Transport Packaging, DSCs*, and RWC.*

Table A.1-1
Nominal Dimensions and Weights of the NUHOMS®-MP197HB Packaging

Nominal Dimensions (in.)	
NUHOMS®-MP197HB packaging overall length with impact limiters and thermal shield	271.25
NUHOMS®-MP197HB packaging overall length without impact limiters and thermal shield	210.25
NUHOMS®-MP197HB cask impact limiter outside diameter	126.00
NUHOMS®-MP197HB cask outside diameter (w/o impact limiters and fins)	97.75
NUHOMS®-MP197HB cask outside diameter with fins (w/o impact limiters)	104.25
NUHOMS®-MP197HB cask cavity inner diameter	70.50
NUHOMS®-MP197HB cask cavity length (minimum)	199.25
NUHOMS®-MP197HB cask inner shell radial thickness	1.25
NUHOMS®-MP197HB cask lead gamma shield radial thickness	3.00
NUHOMS®-MP197HB cask body outer shell	2.75
NUHOMS®-MP197HB cask lid thickness	4.50
NUHOMS®-MP197HB cask bottom thickness	6.50
NUHOMS®-MP197HB cask resin and aluminum box thickness	6.25
Nominal Weights (lb x 1000)	
Weight of Contents *(maximum)*	112.0
Empty weight of NUHOMS®-MP197HB Packaging without lid or impact limiters	157.5
Cask lid	6.0
Outer sleeve with fins	3.1
Weight of impact limiters, thermal shield, and attachments	25.0
Total loaded weight of NUHOMS-MP197HB® Packaging (without transport skid)	303.6

Table A.1-2
DSC* Configuration *in the NUHOMS®-MP197HB Package

DSC Type	Sub Type	Bottom Spacer Required	Sleeve Required	Fins Recommended	Detailed Contents Description in Appendix
NUHOMS®-24PT4	—	Yes	Yes	No	A.1.4.1
NUHOMS®-32PT	S-100	Yes	Yes	No	A.1.4.2
	S-125	Yes	Yes	No	
	L-100	Yes	Yes	No	
	L-125	Yes	Yes	No	
NUHOMS®-24PTH	-S	Yes	Yes	No	A.1.4.3
	-L	Yes	Yes	No	
	-S-LC	Yes	Yes	No	
NUHOMS®-32PTH	—	Yes	No	No	A.1.4.4
	Type 1	Yes	No	No	
NUHOMS®-32PTH1	-S	Yes	No	No	A.1.4.5
	-M	Yes	No	No	
	-L	No	No	No	
NUHOMS®-37PTH	-S	Yes	No	No	A.1.4.6
	-M	Yes	No	No	
NUHOMS®-61BT	—	Yes	Yes	No	A.1.4.7
NUHOMS®-61BTH	*Type 1*	Yes	Yes	No	A.1.4.8
	Type 2				
NUHOMS®-69BTH	—	Yes	No	*Optional*(1)	A.1.4.9

(1) For Heat Loads Greater than 26kW

Notes to Figure A.1-1

A. Some details exaggerated for clarity.

B. Components are listed below:

1 Impact Limiter
2 *Transport Cask Cavity*
3 *Transport Cask Slide Rail*
4 Hold Down Ring (if required)
5 Transport Cask Lid
6 Transport Cask Inner Shell
7 Transport Cask Gamma (Lead) Shield
8 Transport Cask Outer Shell
9 Transport Cask Neutron (Resin) Shield
10 Transport Cask Shield Shell
11 Transport Cask Bottom
12 Transport Cask Bearing Block
13 Impact Limiter Attachment Bolt
14 Thermal Shield
15 Trunnion

Appendix A.1.4.1
NUHOMS®-24PT4 DSC

NOTE: References in this Appendix are shown as [1], [2], etc. and refer to the reference list in Section A.1.4.1.4.

A.1.4.1.1 NUHOMS®-24PT4 DSC Description

The NUHOMS®-24PT4 DSC consists of a DSC shell assembly and a basket assembly. The shell assembly consists of a cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies and top and bottom outer top cover plates. *The DSC shell assembly is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NB [1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13.* The *maximum* length and the outer diameter of the 24PT4 DSC are *approximately* 196.3 inches and 67.*2* inches, respectively. The 24PT4 DSC assembly and details are shown in the drawings in Section A.1.4.10.2 of Appendix A.1.4.10. The shell assembly is a high integrity stainless steel welded pressure vessel that provides confinement of radioactive materials, encapsulates the fuel in an inert atmosphere (the canister is back-filled with helium before being seal welded closed), and provides biological shielding (in the axial direction). The 24PT4 DSC has double redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the 24PT4 DSC. The top end closure welds are made after fuel loading. Both top plug penetrations (siphon and vent ports) are redundantly sealed after the 24PT4 DSC drying operations are complete.

The canister is designed to contain the fuel basket and fuel assemblies, and is completely supported by the transport cask. Under normal transport conditions, the canister rests on four canister rails, attached to the inside surface of the aluminum inner sleeve of the transport cask.

A.1.4.1.2 NUHOMS®-24PT4 Fuel Basket

The basket structure is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NG[1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The details of the 24PT4 fuel basket are shown in the drawings in Section A.1.4.10.2 of Appendix A.1.4.10. The 24PT4 basket is designed to accommodate 24 intact and/or damaged PWR fuel assemblies. The 24PT4 can hold up to 12 damaged fuel assemblies in specially designed Failed Fuel Cans with the balance being loaded with intact fuel. The basket structure consists of circular spacer discs which provide radial support to the guidesleeves and fuel assemblies. Poison plates are placed around the guidesleeves for criticality control.

The guidesleeves are open at each end. Therefore, longitudinal fuel assembly/failed fuel loads are applied directly to the canister/cask body and not the basket structure. The fuel assemblies are laterally supported by the guidetubes/failed fuel can in the circular spacer discs and the canister shell. The guidesleeves are laterally supported by the circular spacer discs and the canister shell. The spacer discs establish and maintain basket orientation. Axial support for the basket assembly is provided by four support rods.

*S*hear key*s*, welded to the inner wall of the 24PT4 DSC, mate with notch*es* in *top and bottom* spacer discs to prevent the basket from rotating during normal operations.

Table A.1.4.1-2
PWR Fuel Specifications of Damaged Fuel to be Transported in the 24PT4 DSC

Fuel Design	Damaged CE 16x16 PWR fuel assembly or equivalent reload fuel that is enveloped by the fuel assembly design characteristics as listed in Table A.1.4.1-3 and the following requirements:
Fuel Damage	Damaged fuel may include assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial and/or missing rods (i.e., extra water holes). Damaged fuel assemblies shall be encapsulated in individual Failed Fuel Cans and placed in Zones A and/or B as shown in Figure A.1.4.1-1. Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a Rod Storage Basket are also considered as damaged fuel. Loose fuel debris, not contained in a Rod Storage Basket may also be placed in a Failed Fuel Can for storage, provided the size of the debris is larger than the Failed Fuel Can screen mesh opening. Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met.
Physical Parameters[1]	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500[2] [3]
Max. U Content (kg)	455.5
No. of Assemblies per DSC	≤ 12 damaged assemblies, balance intact.
Fuel Cladding	Zircaloy-4 or ZIRLO™
Reconstituted Fuel Assemblies	Damaged fuel rods replaced by either stainless rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly).
Nuclear and Radiological Parameters	
Maximum Initial ^{235}U Enrichment (wt %)	Per Table A.1.4.1-4 and Figure A.1.4.1-1
Fuel Assembly Average Burnup and Minimum Cooling Time[4] *[5]*	Per Table A.1.4.1-5 and decay heat restrictions below
Decay Heat[4]	Per Figures A.1.4.1-2, A.1.4.1-3 or A.1.4.1-4

Notes:
(1) Nominal values shown unless stated otherwise.
(2) Does not include weight of Poison Rodlets (25 lbs each) installed in accordance with Table A.1.4.1-4.
(3) Includes the weight of fuel assembly Poison Rods installed for 10CFR50 criticality control in spent fuel pool racks.
(4) Minimum cooling time is the longer of that given in Table A.1.4.1-5 for a given burnup and enrichment of a fuel assembly and that calculated via the decay heat equation based on the restrictions provided in Figures A.1.4.1-2, A.1.4.1-3 or A.1.4.1-4.
(5) An additional cooling time of 8 years is required for damaged fuel assemblies in addition to that obtained from Table A.1.4.1-5, when 5 or more damaged fuel assemblies are loaded.

Table A.1.4.1-5
PWR Fuel Qualification Table for the 24PT4 DSC
(Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU)	Initial Enrichment																													
	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
15	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
20	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
25	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
28	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
30	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
32	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
34	8.5	8.5	8.0	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
36	10.5	10.0	9.5	9.0	8.5	8.0	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
38											7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
39											8.0	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
40											8.5	8.5	8.0	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
41											9.5	9.0	8.5	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
42											10.0	9.5	9.0	9.0	8.5	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0
43																8.5	8.5	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5
44																9.5	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.0	8.0	8.0	8.0	8.0
45																					9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5
48																					11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5
51																					13.0	13.0	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5
54																					16.0	15.0	14.5	14.0	13.5	13.0	13.0	12.5	12.5	12.5	12.5
57																					19.0	18.5	18.0	17.0	16.5	16.0	15.5	15.0	14.5	14.0	13.5
60																					23.0	22.0	21.5	20.5	20.0	19.5	18.5	18.0	17.5	17.0	16.5

Notes:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup conservatively applied in determination of actual values for these two parameters.
- For reconstituted fuel assemblies with irradiated stainless steel rods, increase the cooling time by 1 year for fuel assemblies in the 12 peripheral locations of the canister with cooling times less than 10 years. For fuel assemblies with cooling times greater than 10 years or in the center of the basket, no adjustment is required.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for transport.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for transport.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transport after 7-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for transport after a 7.0-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).
- *When loading five or more damaged fuel assemblies per DSC, an additional cooling time of 8 years is required for only damaged fuel assemblies.*

Table A.1.4.1-6
PWR Assembly Decay Heat for Heat Load Configurations

The Decay Heat (DH) in watts is expressed as:

$$F1 = -44.8 + 41.6*X1 - 37.1*X2 + 0.611*X1^2 - 6.80*X1*X2 + 24.0*X2^2$$
$$DH = F1*Exp(\{[1-(1.8/X3)]* -0.575\}*[(X3-4.5)^{0.169}]*[(X2/X1)^{-0.147}]) + 20$$

where,

- F1 Intermediate Function
- X1 Assembly Burnup in GWD/MTU
- X2 Initial Enrichment in wt. % U-235
- X3 Cooling Time in Years (minimum 7 years)

Note: Even though a minimum cooling time of 7 years is used, the minimum cooling time requirement with five or more damaged fuel assemblies from shielding requirements is per Table A.1.4.1-5.

Appendix A.1.4.2
NUHOMS®-32PT DSC

NOTE: References in this Appendix are shown as [1], [2], etc. and refer to the reference list in Section A.1.4.2.4.

A.1.4.2.1 NUHOMS®-32PT DSC Description

Each NUHOMS®-32PT DSC consists of a DSC shell assembly and a basket assembly. The shell assembly consists of a cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies and outer top cover plate. *The DSC shell assembly is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NB [1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13.* As shown in Table A.1.4.2-1, the 32PT DSC system consists of four design configurations or Types as follows:

- 32PT-S100, Short Canister
- 32PT-L100, Long Canister
- 32PT-S125, Short Canister
- 32PT-L125, Long Canister

Table A.1.4.2-1 provides the overall lengths and outer diameters for each 32PT DSC configuration. The shell assemblies are high integrity stainless steel welded pressure vessels that provide confinement of radioactive materials, encapsulate the fuel in an inert atmosphere (the canister is back-filled with helium before being seal welded closed), and provide biological shielding (in axial direction). The 32PT DSCs have double redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the 32PT DSCs. The top end closure welds are made after fuel loading. Both top plug penetrations (siphon and vent ports) are redundantly sealed after the 32PT DSC drying operations are complete.

The canister is designed to contain its fuel basket and fuel assemblies, and is completely supported by the transport cask. Under normal transport conditions, the canister rests on four canister rails, attached to the inside surface of the aluminum inner sleeve of the transport cask.

A.1.4.2.2 NUHOMS®-32PT Fuel Basket

The basket structures are designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NG[1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The overall lengths and diameters of the baskets for each canister configuration are provided in Table A.1.4.2-1. The details of the 32PT fuel baskets are shown in the drawings in Section A.1.4.10.3 of Appendix A.1.4.10. The 32PT baskets are designed to accommodate 32 intact PWR fuel assemblies with or without Control Components (CCs). The basket structure consists of a grid assembly of welded stainless steel plates or tubes that accommodate aluminum and/or poison plates and *surrounded by* support rails.

The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the canister/cask body and not the fuel basket structure. The fuel assemblies are laterally supported by the stainless steel grid/tube assembly. The basket is laterally supported by

Table A.1.4.2-1
Nominal Dimensions and Weight of the NUHOMS®-32PT DSC

	32PT DSC Design Configuration			
	32PT-S100	**32PT-S125**	**32PT-L100**	**32PT-L125**
Canister Length (in.)	186.*55* *maximum*	186.*55* *maximum*	192.*55* *maximum*	192.*55* *maximum*
Outside Diameter (in.)	67.25	67.25	67.25	67.25
Cavity Length (in.)	169.6	167.1	175.6	173.1
Cavity Diameter (in.)	*66.19*	*66.19*	*66.19*	*66.19*
Basket Length (in.)	168.*6*	166.*1*	174.*6*	172.*1*
Basket Diameter (in.)	65.94	65.94	65.94	65.94

Table A.1.4.2-7
Acceptable Average Initial Enrichment / Minimum Burnup Combinations - NUHOMS®-32PT
Part 1 of 2

	WE 17x17, WE 15x15, B&W 15x15, CE 14x14 and CE 15x15 Assembly Classes					
Enrichment (wt. % U-235)	*16 PP NO PRA 20PP NO PRA*	*24 PP NO PRA*	*20 PP 04 PRA*	*24 PP 04 PRA*	*24 PP 08 PRA 20 PP 08 PRA*	*24 PP 16 PRA 20 PP 16 PRA*
1.30	*fresh*	-	-	-	-	-
1.40	-	*fresh*	*fresh*	-	-	-
1.55	-	-	-	*fresh*	-	-
1.65	-	-	-	-	*fresh*	-
	Burnup (GWD/MTU) 40 Years Decay		*Burnup (GWD/MTU) 30 Years Decay*			*Burnup (GWD/MTU) 15 Years Decay*
2.00	*20*	*19*	*18*	*18*	*11*	*fresh*
2.25	*25*	*23*	*19*	*19*	*17*	*8*
2.50	*30*	*27*	*23*	*22*	*19*	*12*
2.75	*32*	*31*	*27*	*26*	*21*	*16*
3.00	*36*	*34*	*31*	*30*	*24*	*19*
3.25	*39*	*38*	*33*	*33*	*29*	*20*
3.50	*41*	*39*	*37*	*36*	*31*	*22*
3.75	*44*	*42*	*39*	*39*	*33*	*26*
4.00	-	*45*	*42*	*41*	*37*	*29*
4.20	-	-	*44*	*43*	*39*	*31*
4.40	-	-	-	-	*40*	*32*
4.60	-	-	-	-	*42*	*34*
4.80	-	-	-	-	*44*	*37*
5.00	-	-	-	-	*45*	*39*

Table A.1.4.2-7
Acceptable Average Initial Enrichment / Minimum Burnup Combinations - NUHOMS®-32PT

Part 2 of 2

	WE 14x14 Assembly Class				
Enrichment (wt. % U-235)	*16 PP NO PRA*	*24 PP NO PRA*	*20 PP NO PRA*	*20 PP 04 PRA 20 PP 08 PRA 24 PP 04 PRA 24 PP 08 PRA*	*20 PP 16 PRA 24 PP 16 PRA (see note)*
1.50	*fresh*	-	*fresh*	-	-
1.60	-	*fresh*	-	-	-
1.75	-	-	-	*fresh*	-
	Burnup GWD/MTU), 40 Years Decay			*Burnup (GWD/MTU), 15 Years Decay*	
2.00	*18*	*14*	*19*	*9*	*fresh*
2.25	*19*	*19*	*20*	*16*	*8*
2.50	*21*	*20*	*24*	*19*	*12*
2.75	*26*	*23*	*29*	*20*	*16*
3.00	*30*	*28*	*31*	*25*	*19*
3.25	*32*	*31*	*34*	*29*	*20*
3.50	*35*	*32*	*38*	*31*	*22*
3.75	*38*	*36*	*39*	*34*	*26*
4.00	*40*	*39*	*42*	*38*	*29*
4.20	*42*	*40*	*45*	*39*	*31*
4.40	*45*	*42*	-	*41*	*32*
4.60	-	*45*	-	*43*	*34*
4.80	-	-	-	*45*	*37*
5.00	-	-	-	-	*39*

Notes:

- *Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these parameters (uncertainty in enrichment to be added and uncertainty in burnup to be subtracted)*
- *Interpolation can be performed to determine the burnup for enrichment values (between 2.00 wt. % U-235 and 5.00 wt. % U-235) that are not explicitly shown herein. Alternatively, the burnup value corresponding to the next higher enrichment may be utilized.*
- *Extrapolation shall not be performed to determine burnup requirements.*
- *The burnup of the "fresh" assemblies is 0. For a given configuration, the enrichment corresponding to "fresh" in this Table is the maximum enrichment above which a burnup value is needed for fuel assemblies to qualify for transportation.*
- *An additional burnup of 3 GWD/MTU is required for loading fuel assemblies with control rod insertion deeper than 20 cm inside the active fuel during depletion.*

This page intentionally left blank

Appendix A.1.4.3
NUHOMS®-24PTH DSC

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

Appendix A.1.4.3
NUHOMS®-24PTH DSC

NOTE: References in this Appendix are shown as [1], [2], etc. and refer to the reference list in Section A.1.4.3.4.

A.1.4.3.1 NUHOMS®-24PTH DSC Description

Each NUHOMS®-24PTH DSC consists of a DSC shell assembly and a basket assembly. The shell assembly consists of a cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies and outer top cover plate. *The DSC shell assembly is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NB [1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13.* As shown in Table A.1.4.3-1, the 24PTH DSC system consists of three design configurations as follows:

- 24PTH-S, Short Canister including "F" version
- 24PTH-L, Long Canister including "F" version
- 24PTH-S/LC, short canister with long cavity including "F" version

Table A.1.4.3-1 provides the overall lengths and outer diameters for each 24PTH DSC configuration. The shell assemblies are high integrity stainless steel welded pressure vessels that provide confinement of radioactive materials, encapsulate the fuel in an inert atmosphere (the canister is back-filled with helium before being seal welded closed), and provide biological shielding (in axial direction). The 24PTH DSCs have double redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the 24PTH DSCs. The top end closure welds are made after fuel loading. Both top plug penetrations (siphon and vent ports) are redundantly sealed after the 24PTH DSC drying operations are complete.

The canister is designed to contain its fuel basket and fuel assemblies, and is completely supported by the transport cask. Under normal transport conditions, the canister rests on four canister rails attached to the inside of the aluminum inner sleeve of the transport cask.

A.1.4.3.2 NUHOMS®-24PTH DSC Fuel Basket

The basket structures are designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NG[1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The overall lengths and diameters of the baskets for each canister configuration are provided in Table A.1.4.3-1. The details of the 24PTH fuel baskets are shown in the drawings in Section A.1.4.10.4 of Appendix A.1.4.10. The 24PTH baskets are designed to accommodate 24 intact or up to 12 damaged, with up to 8 failed fuel cans loaded with failed fuel with the remainder intact, PWR fuel assemblies with or without Control Components (CCs). The basket structure consists of a welded assembly of stainless steel tubes with the space between adjacent tubes filled with aluminum and neutron poison plates and surrounded by support rails.

The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the canister/cask body and not the fuel basket structure. The fuel assemblies are laterally supported by the stainless steel tube assembly. The basket is laterally supported by the basket rails and the canister shell. The stainless steel and aluminum basket rails are oriented

parallel to the axis of the canister, and are attached to the periphery of the basket to provide support, and to establish and maintain basket orientation.

*S*hear key*s*, welded to the inner wall of the DSC, mate with notch*es* in the basket support rails to prevent the basket from rotating during normal operations.

Aluminum and/or neutron absorbing poison plates are sandwiched between the fuel compartments. The poison plates are constructed of either borated aluminum or Metal Matrix Composites (MMCs) or Boral® that provide criticality control and together with the aluminum plates provide a heat conduction path from the fuel assemblies to the canister wall. Table A.1.4.3-6 provides the minimum B10 content as a function of basket type and poison plate material. Table A.1.4.3-7 provides the maximum allowable heat load for the various 24PTH DSC configurations for transport.

The failed fuel assemblies are to be placed in individual Failed Fuel Cans (FFCs). Each FFC is constructed of sheet metal and is provided with a welded bottom closure and a removable top closure which allows lifting of the FFC with the enclosed damaged assembly/debris. The FFC is provided with screens at the bottom and top to contain fuel debris and allow fill/drainage of water from the FFC during loading operations. The FFC is protected by the fuel compartment tubes and its only function is to confine the failed fuel.

A.1.4.3.3 NUHOMS®-24PTH DSC Contents

Each of the NUHOMS®-24PTH configurations is designed to transport intact (including reconstituted) and/or damaged and/or failed PWR fuel as specified in Table A.1.4.3-2 and Table A.1.4.3-4. The fuel to be transported is limited to a maximum assembly average initial enrichment of 5.0 wt. % U-235. The maximum allowable assembly average burnup is limited to 62 GWd/MTU and the minimum cooling time *requirements are given in Table A.1.4.3-2.* The 24PTH DSC *is* also designed to transport Control Components (CCs) with thermal and radiological characteristics as listed in Table A.1.4.3-3. The CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources.

Partial Length Shield Assemblies (PLSAs) for the Westinghouse 15x15 class, where part of the active fuel is replaced with steel are also included as authorized contents.

Reconstituted assemblies containing up to 10 replacement stainless steel rods per assembly or unlimited number of lower enrichment UO_2 rods are acceptable for storage in 24PTH DSC as intact fuel assemblies. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is given in Table A.1.4.3-2.

The NUHOMS®-24PTH DSCs can also accommodate up to a maximum of 12 damaged fuel assemblies placed in cells located at the outer edge of the DSC as shown in Figure A.1.4.3-6. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater hairline cracks, or pinhole leaks. The extent of damage in the fuel rods is to be limited such that a *fuel assembly needs to be handled by normal means*. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.

The NUHOMS®-24PTHF DSC, an alternative version of NUHOMS®-24PTH DSC, is designed to accommodate up to a maximum of 8 failed fuel assemblies encapsulated in individual failed fuel cans and placed in cells located at the outer edge of the DSC as shown in Figure A.1.4.3-6. Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means.

Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a Rod Storage Basket are also considered as *failed* fuel. Loose fuel debris, not contained in a Rod Storage Basket may also be placed in a Failed Fuel Can for storage, provided the size of the debris is larger than the Failed Fuel Can screen mesh opening.

Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its contents shall be less than 1682 lbs.

A 24PTH DSC containing less than 24 fuel assemblies may contain either empty slots or dummy fuel assemblies in the empty slots. The dummy assemblies are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a fuel assembly.

A.1.4.3.4 References

1. American Society of Mechanical Engineers, ASME Boiler And Pressure Vessel Code, Section III, Division 1 - Subsections NB, NG and NF, 1998 edition including 2000 Addenda.

Table A.1.4.3-1
Key Design Parameters of the NUHOMS®-24PTH System

Parameter	24PTH DSC Type		
	24PTH-S	**24PTH-L**	**24PTH-S-LC**
DSC Length (in)	186.55 (Maximum)	192.55 (Maximum)	186.*67* (Maximum)
DSC Outside Diameter (in)	67.*19*	67.*19*	67.*19*
DSC Cavity Length (in)	169.6*0*	175.*10*	173.28
Basket Length (in)	168.*60*	174.10	172.28
Basket Diameter (in)	65.94	65.94	65.94

Note: Unless stated otherwise, nominal values are provided.

Table A.1.4.3-2
PWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-24PTH DSC
(concluded)

Control Components (CCs)	• Up to 24 CCs are authorized for storage in 24PTH-S, 24PTH-L, and 24PTH-S-LC DSCs. • Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources. • Design basis thermal and radiological characteristics for the CCs are listed in Table A.1.4.3-3.
Nominal Assembly Width for Intact and Damaged Fuel Assemblies Only	8.536 inches
No. of Intact Assemblies	≤24
No. and Location of Damaged Assemblies	Up to 12 damaged fuel assemblies. Balance may be intact fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration. Damaged fuel assemblies are to be placed in Locations A and/or B as shown in Figure A.1.4.3-6. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
No. and Location of Failed Assemblies	Up to 8 failed fuel assemblies. Balance may be intact and/or damaged fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration. Failed fuel assemblies are to be placed in Location A as shown in Figure A.1.4.3-6. Failed fuel assembly/fuel debris is to be encapsulated in an individual Failed Fuel Can (FFC) provided with a welded bottom closure and a removable top closure.
Maximum Assembly plus CC Weight	1682 lbs
THERMAL/RADIOLOGICAL PARAMETERS: Fuel Assembly Average Burnup and minimum Cooling Time(1) *(2)*	Per Table A.1.4.3-5, Table A.1.4.3-8, *Table A.1.4.3-8A* and decay heat and burnup credit restrictions below.
Maximum Decay Heat(1) Limits for Zones 1, 2, 3, and 4 Fuel	Per Figure A.1.4.3-1 or Figure A.1.4.3-2 or Figure A.1.4.3-3 or Figure A.1.4.3-4 or Figure A.1.4.3-5.
Decay Heat(1) per DSC	Type 1 Basket ≤ 26.0 kW for 24PTH-S and 24PTH-L DSCs with decay heat limit for Zones 1, 2, 3 and 4 as specified in Figure A.1.4.3-1, or Figure A.1.4.3-2, Figure A.1.4.3-3 or Figure A.1.4.3-4.
	Type 2 Basket Same as Type 1 Basket except ≤26.0 kW/DSC and ≤ 1.3 kW/fuel assembly for 24PTH-S and 24PTH-L DSCs. ≤ 24.0 kW for 24PTH-S-LC DSC with decay heat limits as≤ 24.0 kW for 24PTH-S-L DSC (Type 2 Basket) specified in Figure A.1.4.3-5.
Burnup Credit Restrictions(1)	Per Table A.1.4.3-8 *for intact fuel assemblies and per Table A.1.4.3-8A for Damaged or Failed fuel assemblies.*

Notes:

(1) Minimum cooling time is the longer of that given in Table A.1.4.3-5; that calculated via the decay heat equation given in Table A.1.4.3-9 based on the restrictions provided in Figures A.1.4.3-1, A.1.4.3-2, A.1.4.3-3 or A.1.4.3-4; and Table A.1.4.3-8 *or Table A.1.4.3-8A.*

(2) An additional cooling time of 8 years is required for damaged fuel assemblies in addition to that obtained from Table A.1.4.3-5, when 5 or more damaged fuel assemblies are loaded.

Notes: Table A.1.4.3-5:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial assembly average enrichment either less than 0.7 or greater than 5.0 wt.% U-235 is unacceptable for Transport.
- Fuel with a burnup greater than 62 GWd/MTU is unacceptable for transport.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transport after 10-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for transport after 10-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).
- Even though cooling times less than 15 years are shown in this table, the minimum cooling time requirement for criticality from Table A.1.4.3-8 *and Table A.1.4.3-8A* for transportation is 15 years.
- *When loading five or more damaged fuel assemblies per DSC, an additional cooling time of 8 years is required for only damaged fuel assemblies.*

Table A.1.4.3-8
Acceptable Average Initial Enrichment / Minimum Burnup Combinations *for* NUHOMS®-24PTH – *Intact Fuel Assemblies*

(Part 1 of 2)

	WE 17x17, WE 15x15, BW 15x15, CE 14x14, CE 15x15 and CE 16x16 assembly classes				
Enrichment (wt. % U-235)	*Type A*	*Type B*	*Type C*	*Type A*	*Type B*
1.55	*fresh*	-	-	*fresh*	-
1.65	-	*fresh*	-	-	*fresh*
1.80	-	-	*fresh*	-	-
	Burnup (GWd/MTU), 15 years decay			*Burnup (GWd/MTU), 30 years decay*	
2.00	*18*	*14*	*8*	*17*	*12*
2.25	*19*	*19*	*15*	*19*	*18*
2.50	*24*	*21*	*19*	*21*	*19*
2.75	*28*	*24*	*20*	*25*	*21*
3.00	*32*	*28*	*23*	*30*	*26*
3.25	*35*	*31*	*28*	*31*	*30*
3.50	*39*	*34*	*31*	*35*	*32*
3.75	*41*	*38*	*33*	*38*	*35*
4.00	*44*	*39*	*36*	*40*	*37*
4.20	*47*	*43*	*38*	*42*	*39*
4.40	*50*	*45*	*41*	*45*	*41*
4.60	-	*48*	*43*	*48*	*43*
4.80	-	*50*	*45*	*50*	*45*
5.00	-	-	*47*	-	*47*

Table A.1.4.3-8
Acceptable Average Initial Enrichment / Minimum Burnup Combinations *for* NUHOMS®-24PTH – *Intact Fuel Assemblies*

(Part 2 of 2)

	WE 14x14 assembly class	
Enrichment (wt. % U-235)	*Type A*	*Type B*
1.80	*fresh*	-
1.95	-	*fresh*
	Burnup (GWd/MTU), 30 Years decay	*Burnup (GWd/MTU), 15 Years decay*
2.00	*6*	*5*
2.25	*11*	*9*
2.50	*17*	*14*
2.75	*19*	*18*
3.00	*20*	*19*
3.25	*24*	*21*
3.50	*28*	*25*
3.75	*31*	*29*
4.00	*32*	*31*
4.20	*34*	*33*
4.40	*37*	*35*
4.60	*39*	*37*
4.80	*41*	*39*
5.00	*42*	*41*

Notes:

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these parameters (uncertainty in enrichment to be added and uncertainty in burnup to be subtracted)
- Interpolation can be performed to determine the burnup for enrichment values (between 2.00 wt. % U-235 and 5.00 wt. % U-235) that are not explicitly shown herein. Alternatively, the burnup value corresponding to the next higher enrichment may be utilized.
- Extrapolation shall not be performed to determine burnup requirements.
- The burnup of the "fresh" assemblies is 0. For a given configuration, the enrichment corresponding to "fresh" in this Table is the maximum enrichment above which a burnup value is needed for fuel assemblies to qualify for transportation.
- *An additional burnup of 3 GWD/MTU is required for loading fuel assemblies with control rod insertion deeper than 20 cm inside the active fuel during depletion.*

Table A.1.4.3-8A
Acceptable Average Initial Enrichment / Minimum Burnup Combinations for NUHOMS®-24PTH – Damaged Fuel Assemblies

(Part 1 of 2)

	WE 17x17, WE 15x15, BW 15x15, CE 14x14, CE 15x15 and CE 16x16 assembly classes				
Enrichment (wt. % U-235)	*Type A*	*Type B*	*Type C*	*Type A*	*Type B*
1.55	*fresh*	-	-	*fresh*	-
1.65	-	*fresh*	-	-	*fresh*
1.80	-	-	*fresh*	-	-
	Burnup (GWd/MTU), 15 Years decay			*Burnup (GWd/MTU), 30 Years decay*	
2.00	*19*	*16*	*10*	*19*	*14*
2.25	*21*	*21*	*17*	*21*	*20*
2.50	*26*	*23*	*21*	*23*	*21*
2.75	*30*	*26*	*22*	*27*	*23*
3.00	*34*	*30*	*25*	*32*	*28*
3.25	*37*	*33*	*30*	*33*	*32*
3.50	*41*	*36*	*33*	*37*	*34*
3.75	*43*	*40*	*35*	*40*	*37*
4.00	*46*	*41*	*38*	*42*	*41*
4.20	*49*	*45*	*40*	*44*	*43*
4.40	-	*47*	*43*	*47*	*45*
4.60	-	*50*	*45*	*50*	*47*
4.80	-	-	*47*	-	*49*
5.00	-	-	*49*	-	-

Table A.1.4.3-8A
Acceptable Average Initial Enrichment / Minimum Burnup Combinations for NUHOMS®-24PTH – Damaged Fuel Assemblies

(Part 2 of 2)

	WE 14x14 assembly class	
Enrichment (wt. % U-235)	*Type A*	*Type B*
1.80	*fresh*	*-*
1.95	*-*	*fresh*
	Burnup (GWd/MTU), 30 Years decay	*Burnup (GWd/MTU), 15 Years decay*
2.00	*10*	*8*
2.25	*15*	*13*
2.50	*20*	*18*
2.75	*24*	*20*
3.00	*28*	*23*
3.25	*30*	*27*
3.50	*32*	*31*
3.75	*34*	*32*
4.00	*36*	*34*
4.20	*38*	*37*
4.40	*41*	*39*
4.60	*42*	*41*
4.80	*45*	*44*
5.00	*46*	*46*

Notes:

- *Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these parameters (uncertainty in enrichment to be added and uncertainty in burnup to be subtracted)*
- *Interpolation can be performed to determine the burnup for enrichment values (between 2.00 wt. % U-235 and 5.00 wt. % U-235) that are not explicitly shown herein. Alternatively, the burnup value corresponding to the next higher enrichment may be utilized.*
- *Extrapolation shall not be performed to determine burnup requirements.*
- *The burnup of the "fresh" assemblies is 0. For a given configuration, the enrichment corresponding to "fresh" in this Table is the maximum enrichment above which a burnup value is needed for fuel assemblies to qualify for transportation.*
- *An additional burnup of 3 GWD/MTU is required for loading fuel assemblies with control rod insertion deeper than 20 cm inside the active fuel during depletion.*

Table A.1.4.3-9
PWR Assembly Decay Heat for Heat Load Configurations

The Decay Heat (DH) in watts is expressed as:

$$F1 = -44.8 + 41.6*X1 - 37.1*X2 + 0.611*X1^2 - 6.80*X1*X2 + 24.0*X2^2$$
$$DH = F1*Exp(\{[1-(1.8/X3)]* -0.575\}*[(X3-4.5)^{0.169}]*[(X2/X1)^{-0.147}]) + 20$$

where,

- F1 Intermediate Function
- X1 Assembly Burnup in GWD/MTU
- X2 Initial Enrichment in wt. % U-235
- X3 Cooling Time in Years (minimum 10 years)

Note: Even though a minimum cooling time of 10 years is used, the minimum cooling time requirement for criticality from Table A.1.4.3-8 *and Table A.1.4.3-8A* is 15 years.

Appendix A.1.4.4
NUHOMS®-32PTH DSC

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

Appendix A.1.4.4
NUHOMS®-32PTH DSC

NOTE: References in this Appendix are shown as [1], [2], etc. and refer to the reference list in Section A.1.4.4.4.

A.1.4.4.1 NUHOMS®-32PTH DSC Description

Each NUHOMS®-32PTH DSC consists of a DSC shell assembly and a basket assembly. The shell assembly consists of a cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies and outer top cover plate. *The DSC shell assembly is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NB [1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13.* As shown in Table A.1.4.4-1, the 32PTH DSC consists of two design configurations as follows:

- 32PTH
- 32PTH Type 1

Table A.1.4.4-1 provides the overall lengths and outer diameters for each 32PTH DSC configuration. The shell assemblies are high integrity stainless steel welded pressure vessels that provide confinement of radioactive materials, encapsulate the fuel in an inert atmosphere (the canister is back-filled with helium before being seal welded closed), and provide biological shielding (in axial direction). The 32PTH DSCs have double redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the 32PTH DSCs. The top end closure welds are made after fuel loading. Both top plug penetrations (siphon and vent ports) are redundantly sealed after the 32PTH DSC drying operations are complete.

The canister is designed to contain its fuel basket and fuel assemblies, and is completely supported by the transport cask. Under normal transport conditions, the canister rests on four canister rails attached to the inside of the transport cask.

A.1.4.4.2 NUHOMS®-32PTH DSC Fuel Basket

The basket structures are designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NG[1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The overall lengths and diameters of the baskets for each canister configuration are provided in Table A.1.4.4-1. The details of the 32PTH fuel baskets are shown in the drawings in Section A.1.4.10.5 of Appendix A.1.4.10. The 32PTH baskets are designed to accommodate 32 intact or up to 16 damaged with the remainder intact PWR fuel assemblies with or without Control Components (CCs). The basket structure consists of a welded assembly of stainless steel tubes with the space between adjacent tubes filled with aluminum and neutron poison plates and surrounded by support rails.

The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the canister/cask body and not the fuel basket structure. The fuel assemblies are laterally supported by the stainless steel tube assembly. The basket is laterally supported by the basket rails and the canister shell. The stainless steel and aluminum basket rails are oriented

parallel to the axis of the canister, and are attached to the periphery of the basket to provide support, and to establish and maintain basket orientation.

Blocks (32PTH DSC) and shear key*s (32PTH Type 1 DSC) are used* to prevent the basket from rotating during normal operations.

Aluminum and/or neutron absorbing poison plates are sandwiched between the fuel compartments. The poison plates are constructed of either borated aluminum or Metal Matrix Composites (MMCs) or Boral® that provide criticality control and together with the aluminum plates provide a heat conduction path from the fuel assemblies to the canister wall. Table A.1.4.4-6 provides the minimum B10 content as a function of basket type and poison plate material.

A.1.4.4.3 NUHOMS®-32PTH DSC Contents

The NUHOMS® 32PTH DSC and the NUHOMS® 32PTH Type 1 DSC are designed for the transport of 32 intact and/or up to 16 damaged with remaining intact PWR fuel assemblies as specified in Table A.1.4.4-2 and Table A.1.4.4-3. The fuel to be transported is limited to a maximum assembly average initial enrichment of 5.0 wt. % ^{235}U. The maximum allowable assembly average burnup is limited to 60 GWd/MTU and the minimum cooling time *requirements are given in Table A.1.4.4-2.* The fuel assemblies may be transported with or without Control Components (CCs). The CC thermal and radiological characteristics are listed in Table A.1.4.4-4.

The 32PTH DSC may transport up to 32 PWR fuel assemblies arranged in accordance with a heat load zoning configuration as shown in Figure A.1.4.4-1, with a maximum decay heat of 1.5 kW per assembly and a maximum heat load of 26 kW per DSC.

The 32PTH DSC can accommodate up to 16 damaged fuel assemblies which include assemblies with missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of the damage is to be limited such that a fuel assembly can be handled by normal means. Damaged fuel assemblies shall be placed into the sixteen inner most basket fuel compartments, as shown in Figure A.1.4.4-2, which contain top and bottom end caps that confine any loose material and gross fuel particles to a known, sub-critical volume during normal and accident conditions.

Table A.1.4.4-1
Key Design Parameters of the NUHOMS®-32PTH System

Parameter	**32PTH**	**32PTH Type 1**
DSC Length (in)	185.75 (Maximum)	193.00 (Maximum)
DSC Outside Diameter (in)	69.75	69.75
DSC Cavity Length (in)	164.5	171.63
DSC Shell Thickness (in)	0.5	0.5
Basket Length (in)	162.00	169.00
Basket Diameter (in)	68.50	68.50

Note: Unless stated otherwise, nominal values are provided.

Table A.1.4.4-2
PWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-32PTH DSC
(concluded)

No. and Location of Damaged Assemblies	Up to 16 damaged fuel assemblies with the balance intact fuel assemblies, or dummy assemblies. Damaged fuel assemblies are to be placed in the center 16 locations as shown in Figure A.1.4.4-2. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
Maximum Assembly plus CC Weight	1585 lbs
THERMAL/RADIOLOGICAL PARAMETERS: Fuel Assembly Average Burnup and minimum Cooling Time(1)	 Per Table A.1.4.4-5; Table A.1.4.4-8, *Table A.1.4.4-8A* and decay heat and burnup credit restrictions below.
Decay Heat(1)	Per Figure A.1.4.4-1
Burnup Credit Restrictions(1)	Per Table A.1.4.4-8 *for Intact Fuel Assemblies and per Table A.1.4.4-8A for Damaged Fuel Assemblies*

Notes:
(1) Minimum cooling time is the longer of that given in Table A.1.4.4-5; that calculated via the decay heat equation given in Table A.1.4.4-7; based on the restrictions provided in Figures A.1.4.4-1; and Table A.1.4.4-8 *or Table A.1.4.4-8A.*

Notes: Table A.1.4.4-5:

- BU = Assembly average burnup.
- Use burnup and enrichment to look-up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 0.3 or greater than 5.0 wt.% U-235 is unacceptable for Transport.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for transport.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transport after 10-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for transport after 10-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).
- Even though cooling times less than 15 years are shown in this table, the minimum cooling time requirement for criticality from Table A.1.4.4-8 *and Table 1.4.4-8A* for transportation is 15 years.

Table A.1.4.4-7
PWR Assembly Decay Heat for Heat Load Configurations

The Decay Heat (DH) in watts is expressed as:

$$F1 = -44.8 + 41.6*X1 - 37.1*X2 + 0.611*X1^2 - 6.80*X1*X2 + 24.0*X2^2$$
$$DH = F1*Exp(\{[1-(1.8/X3)]* -0.575\}*[(X3-4.5)^{0.169}]*[(X2/X1)^{-0.147}]) + 20$$

where,

F1 Intermediate Function
X1 Assembly Burnup in GWD/MTU
X2 Initial Enrichment in wt. % U-235
X3 Cooling Time in Years (minimum 10 years)

Note: Even though a minimum cooling time of 10 years is used, the minimum cooling time requirement for criticality from Table A.1.4.4-8 *and Table 1.4.4-8A* is 15 years.

Table A.1.4.4-8
Acceptable Average Initial Enrichment/Minimum Burnup Combinations - NUHOMS®-32PTH –*Intact Fuel Assemblies*

Enrichment (wt. % U-235)	*WE 17x17, WE 15x15, CE 14x14 and CE 16x16 fuel assembly classes*									
	Type A	*Type B*	*Type C*	*Type D*	*Type E*	*Type A*	*Type B*	*Type C*	*Type D*	*Type E*
1.45	*fresh*	-	-	-	-	*fresh*	-	-	-	-
1.55	-	*fresh*	-	-	-	-	*fresh*	-	-	-
1.60	-	-	*fresh*	-	-	-	-	*fresh*	-	-
1.70	-	-	-	*fresh*	-	-	-	-	*fresh*	-
1.80	-	-	-	-	*fresh*	-	-	-	-	*fresh*
	Burnup (GWD/MTU), 15 years decay					***Burnup (GWD/MTU), 30 years decay***				
2.00	*20*	*16*	*14*	*11*	*7*	*19*	*15*	*13*	*9*	*6*
2.25	*23*	*19*	*19*	*17*	*14*	*20*	*19*	*18*	*15*	*12*
2.50	*29*	*22*	*20*	*19*	*19*	*24*	*20*	*19*	*19*	*18*
2.75	*31*	*27*	*25*	*22*	*20*	*29*	*24*	*23*	*20*	*19*
3.00	*36*	*31*	*30*	*26*	*23*	*32*	*28*	*27*	*24*	*20*
3.25	*39*	*33*	*32*	*30*	*27*	*35*	*31*	*30*	*28*	*24*
3.50	*41*	*38*	*36*	*32*	*30*	*39*	*34*	*33*	*31*	*28*
3.75	*45*	*40*	*39*	*36*	*32*	*41*	*37*	*35*	*33*	*31*
4.00	*50*	*43*	*42*	*39*	*35*	*44*	*39*	*39*	*36*	*33*
4.20	-	*46*	*44*	*41*	*38*	*46*	*41*	*40*	*38*	*35*
4.40	-	-	*46*	*43*	*39*	*49*	*44*	*42*	*39*	*37*
4.60	-	-	*49*	*45*	*41*	-	*46*	*44*	*40*	*39*
4.80	-	-	-	*47*	*43*	-	*49*	*47*	*43*	*40*
5.00	-	-	-	*50*	*45*	-	-	*50*	*45*	*42*

Notes:

- *Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these parameters (uncertainty in enrichment to be added and uncertainty in burnup to be subtracted)*
- *Interpolation can be performed to determine the burnup for enrichment values (between 2.00 wt. % U-235 and 5.00 wt. % U-235) that are not explicitly shown herein. Alternatively, the burnup value corresponding to the next higher enrichment may be utilized.*
- *Extrapolation shall not be performed to determine burnup requirements.*
- *The burnup of the "fresh" assemblies is 0. For a given configuration, the enrichment corresponding to "fresh" in this Table is the maximum enrichment above which a burnup value is needed for fuel assemblies to qualify for transportation.*
- *An additional burnup of 3 GWD/MTU is required for loading fuel assemblies with control rod insertion deeper than 20 cm inside the active fuel during depletion.*

Table A.1.4.4-8*A*
Acceptable Average Initial Enrichment/Minimum Burnup Combinations - NUHOMS®-32PTH – *Damaged Fuel Assemblies*

Enrichment (wt. % U-235)	***WE 17x17, WE 15x15, CE 14x14 and CE 16x16 fuel assembly classes***									
	Type A	***Type B***	***Type C***	***Type D***	***Type E***	***Type A***	***Type B***	***Type C***	***Type D***	***Type E***
1.50	*fresh*	-	-	-	-	*fresh*	-	-	-	-
1.60	-	*fresh*	-	-	-	-	*fresh*	-	-	-
1.65	-	-	*fresh*	-	-	-	-	*fresh*	-	-
1.75	-	-	-	*fresh*	-	-	-	-	*fresh*	-
1.80	-	-	-	-	*fresh*	-	-	-	-	*fresh*
	Burnup (GWD/MTU), 15 years decay					***Burnup (GWD/MTU), 30 years decay***				
2.00	*23*	*19*	*19*	*15*	*12*	*19*	*18*	*16*	*13*	*10*
2.25	*28*	*23*	*20*	*19*	*18*	*23*	*19*	*19*	*19*	*17*
2.50	*31*	*28*	*26*	*23*	*21*	*29*	*25*	*22*	*20*	*19*
2.75	*37*	*32*	*31*	*28*	*25*	*33*	*30*	*28*	*25*	*22*
3.00	*41*	*37*	*35*	*33*	*30*	*37*	*33*	*31*	*29*	*26*
3.25	*44*	*41*	*39*	*36*	*33*	*40*	*37*	*35*	*32*	*31*
3.50	*49*	*45*	*43*	*39*	*37*	*44*	*39*	*39*	*35*	*33*
3.75	-	*49*	*47*	*43*	*40*	*48*	*42*	*41*	*39*	*37*
4.00	-	-	-	*46*	*43*	-	*46*	*44*	*41*	*39*
4.20	-	-	-	*49*	*46*	-	*49*	*47*	*44*	*41*
4.40	-	-	-	-	*50*	-	-	*50*	*47*	*44*
4.60	-	-	-	-	-	-	-	-	*50*	*47*
4.80	-	-	-	-	-	-	-	-	-	*50*
5.00	-	-	-	-	-	-	-	-	-	-

Notes:

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these parameters (uncertainty in enrichment to be added and uncertainty in burnup to be subtracted)
- Interpolation can be performed to determine the burnup for enrichment values (between 2.00 wt. % U-235 and 5.00 wt. % U-235) that are not explicitly shown herein. Alternatively, the burnup value corresponding to the next higher enrichment may be utilized.
- Extrapolation shall not be performed to determine burnup requirements.
- The burnup of the "fresh" assemblies is 0. For a given configuration, the enrichment corresponding to "fresh" in this Table is the maximum enrichment above which a burnup value is needed for fuel assemblies to qualify for transportation.
- *An additional burnup of 3 GWD/MTU is required for loading fuel assemblies with control rod insertion deeper than 20 cm inside the active fuel during depletion.*

Appendix A.1.4.5
NUHOMS®-32PTH1 DSC

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

Appendix A.1.4.5
NUHOMS®-32PTH1 DSC

NOTE: References in this Appendix are shown as [1], [2], etc. and refer to the reference list in Section A.1.4.5.4.

A.1.4.5.1 NUHOMS®-32PTH1 DSC Description

Each NUHOMS®-32PTH1 DSC consists of a DSC shell assembly and a basket assembly. The shell assembly consists of a cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies and outer top cover plate. *The DSC shell assembly is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NB [1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13.* As shown in Table A.1.4.5-1, the 32PTH1 DSC system consists of three design configurations as follows:

- 32PTH1-S, Short DSC
- 32PTH1-M, Medium DSC
- 32PTH1-L, Long DSC

Table A.1.4.5-1 provides the overall lengths and outer diameters for each 32PTH1 DSC configuration. The shell assemblies are high integrity stainless steel welded pressure vessels that provide confinement of radioactive materials, encapsulate the fuel in an inert atmosphere (the canister is back-filled with helium before being seal welded closed), and provide biological shielding (in axial direction). The 32PTH1 DSCs have double redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the 32PTH1 DSCs. The top end closure welds are made after fuel loading. Both top plug penetrations (siphon and vent ports) are redundantly sealed after the 32PTH1 DSC drying operations are complete.
The canister is designed to contain its fuel basket and fuel assemblies, and is completely supported by the transport cask. Under normal transport conditions, the canister rests on four canister rails attached to the inside of the MP197HB transport cask.

A.1.4.5.2 NUHOMS®-32PTH1 DSC Fuel Basket

The basket structures are designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NG[1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The overall lengths and diameters of the baskets for each canister configuration are provided in Table A.1.4.5-1. The details of the 32PTH1 fuel baskets are shown in the drawings in Section A.1.4.10.6 of Appendix A.1.4.10. The 32PTH1 baskets are designed to accommodate 32 intact, or up to 16 damaged with the remainder intact, PWR fuel assemblies with or without Control Components. The basket structure consists of a welded assembly of stainless steel tubes with the space between adjacent tubes filled with aluminum and neutron poison plates and surrounded by support rails.

The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the canister/cask body and not the fuel basket structure. The fuel assemblies are laterally supported by the stainless steel tube assembly. The basket is laterally supported by the basket rails and the canister shell. The stainless steel and aluminum basket rails are oriented

parallel to the axis of the canister, and are attached to the periphery of the basket to provide support, and to establish and maintain basket orientation.

*S*hear key*s*, welded to the inner wall of the DSC, mate with notch*es* in the basket support rails to prevent the basket from rotating during normal operations.

Aluminum and/or neutron absorbing poison plates are sandwiched between the fuel compartments. Table A.1.4.5-6 provides the minimum B10 content as a function of basket type and poison plate material. Table A.1.4.5-7 provides the maximum allowable heat load for the various 32PTH1 DSC configurations for transport.

A.1.4.5.3 NUHOMS®-32PTH1 DSC Contents

Each of the three alternate DSC configurations is designed to transport intact (including reconstituted) and/or damaged PWR fuel assemblies as specified in Table A.1.4.5-2 and Table A.1.4.5-4. The fuel to be transported is limited to a maximum assembly average initial enrichment of 5.0 wt.% U-235. The maximum allowable assembly average burnup is limited to 62 GWd/MTU and the minimum cooling time *requirements are given in Table A.1.4.5-2.* Each of the DSC types is designed to transport Control Components (CCs) with thermal and radiological characteristics as listed in Table A.1.4.5-3. The CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs) and Neutron Sources.

Reconstituted assemblies containing up to 10 replacement irradiated stainless steel rods per assembly, or unlimited number of lower enrichment UO_2 rods instead of Zircaloy clad enriched UO_2 rods, or Zr rods or Zr pellets, or unirradiated stainless steel rods are acceptable for storage in the 32PTH1 DSC as intact fuel assemblies. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four with irradiated stainless steel replacement rods or 32 with lower enrichment UO_2 replacement rods.

The NUHOMS®-32PTH1 DSCs can also accommodate up to a maximum of 16 damaged fuel assemblies placed in the center cells of the DSC as shown in Figure A.1.4.5-1 through Figure A.1.4.5-3. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks, or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is able to be handled by normal means. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.

A 32PTH1 DSC containing less than 32 fuel assemblies may contain dummy fuel assemblies in the empty slots. The dummy assemblies are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a fuel assembly.

Table A.1.4.5-1
Key Design Parameters of the NUHOMS®-32PTH1 System

Parameter	**32PTH1 DSC Type**		
	32PTH1-S	**32PTH1-M**	**32PTH1-L**
DSC Length (in)	185.75 (Maximum)	193.00 (Maximum)	198.50 (Maximum)
DSC Outside Diameter (in)	69.75	69.75	69.75
DSC Cavity Length (in)	164.*38*	17*1.63*	181.*38*
Basket Length (in)	162.00	169.00	178.75
Basket Diameter (in)	68.50	68.50	68.50

Note: Unless stated otherwise, nominal values are provided.

Table A.1.4.5-2
PWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-32PTH1 DSC
(concluded)

THERMAL/RADIOLOGICAL PARAMETERS:	
Fuel Assembly Average Burnup and minimum Cooling Time(1)	Per Table A.1.4.5-5; Table A.1.4.5-8, *Table A.1.4.5-8A* and decay heat and burnup credit restrictions below.
Decay Heat(1)	Per Figure A.1.4.5-1 or Figure A.1.4.5-2 or Figure A.1.4.5-3.
Burnup Credit Restrictions(1)	Per Table A.1.4.5-8 *for Intact Fuel Assemblies and Per Table A.1.4.5-8A for Damaged Fuel Assemblies*

Note:
(1) Minimum cooling time is the longer of that given in Table A.1.4.5-5; that calculated via the decay heat equation given in Table A.1.4.5-9 based on the restrictions provided in Figures A.1.4.5-1, A.1.4.5-2, or A.1.4.5-3; and Table A.1.4.5-8 *or Table A.1.4.5-8A*.

Notes, Table A.1.4.5-5:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 0.7 or greater than 5.0 wt.% U-235 is unacceptable for Transport.
- Fuel with a burnup greater than 62 GWd/MTU is unacceptable for transport.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transport after 10-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for transport after 10-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).
- Even though cooling times less than 15 years are shown in this table, the minimum cooling time requirement for criticality from Table A.1.4.5-8 *and Table A.1.4.5-8A* for transportation is 15 years.

Table A.1.4.5-8
Acceptable Average Initial Enrichment/Minimum Burnup Combinations - NUHOMS®-32PTH1 – *Intact Fuel Assemblies*

(Part 1 of 2)

Enrichment (wt. % U-235)	WE 17x17, WE 15x15, BW 15x15, CE 14x14, CE 15x15 and CE 16x16 fuel assembly classes									
	Type A	Type B	Type C	Type D	Type E	Type A	Type B	Type C	Type D	Type E
1.45	fresh	-	-	-	-	fresh	-	-	-	-
1.55	-	fresh	-	-	-	-	fresh	-	-	-
1.60	-	-	fresh	-	-	-	-	fresh	-	-
1.70	-	-	-	fresh	-	-	-	-	fresh	-
1.80	-	-	-	-	fresh	-	-	-	-	fresh
	Burnup (GWD/MTU), 15 years decay					Burnup (GWD/MTU), 30 years decay				
2.00	20	16	14	11	7	19	15	13	9	6
2.25	23	19	19	17	14	20	19	18	15	12
2.50	29	22	20	19	19	24	20	19	19	18
2.75	31	27	25	22	20	29	24	23	20	19
3.00	36	31	30	26	23	32	28	27	24	20
3.25	39	33	32	30	27	35	31	30	28	24
3.50	41	38	36	32	30	39	34	33	31	28
3.75	45	40	39	36	32	41	37	35	33	31
4.00	50	43	42	39	35	44	39	39	36	33
4.20	-	46	44	41	38	46	41	40	38	35
4.40	-	-	46	43	39	49	44	42	39	37
4.60	-	-	49	45	41	-	46	44	40	39
4.80	-	-	-	47	43	-	49	47	43	40
5.00	-	-	-	50	45	-	-	50	45	42

Table A.1.4.5-8
Acceptable Average Initial Enrichment/Minimum Burnup Combinations - NUHOMS®-32PTH1 – *Intact Fuel Assemblies*

(*Part 2 of 2*)

Enrichment (wt. % U-235)	*WE 14x14 assembly class*					
	Type A	*Type B*	*Type C*	*Type A*	*Type B*	*Type C*
1.70	*fresh*	-	-	*fresh*	-	-
1.85	-	*fresh*	-	-	*fresh*	-
1.90	-	-	*fresh*	-	-	*fresh*
	Burnup (GWD/MTU), 15 years decay			***Burnup (GWD/MTU), 30 years decay***		
2.00	*11*	*6*	*5*	*10*	*5*	*5*
2.25	*17*	*12*	*10*	*16*	*10*	*9*
2.50	*19*	*17*	*15*	*19*	*16*	*14*
2.75	*22*	*19*	*19*	*20*	*19*	*18*
3.00	*25*	*21*	*20*	*24*	*20*	*19*
3.25	*30*	*25*	*23*	*28*	*24*	*21*
3.50	*32*	*29*	*26*	*31*	*26*	*24*
3.75	*35*	*31*	*30*	*33*	*30*	*28*
4.00	*39*	*34*	*32*	*36*	*31*	*31*
4.20	*40*	*36*	*34*	*38*	*36*	*32*
4.40	*42*	*39*	*37*	*39*	*37*	*34*
4.60	*45*	*40*	*39*	*41*	*38*	*36*
4.80	*48*	*42*	*40*	*43*	*39*	*39*
5.00	*50*	*44*	*42*	*45*	*41*	*40*

Notes:

- *Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these parameters (uncertainty in enrichment to be added and uncertainty in burnup to be subtracted).*
- *Interpolation can be performed to determine the burnup for enrichment values (between 2.00 wt.% U-235 and 5.00 wt.% U-235) that are not explicitly shown herein. Alternatively, the burnup value corresponding to the next higher enrichment may be utilized.*
- *Extrapolation shall not be performed to determine burnup requirements.*
- *The burnup of the "fresh" assemblies is 0. For a given configuration, the enrichment corresponding to "fresh" in this Table is the maximum enrichment above which a burnup value is needed for fuel assemblies to qualify for transportation.*
- *An additional burnup of 3 GWD/MTU is required for loading fuel assemblies with control rod insertion deeper than 20 cm inside the active fuel during depletion.*

Table A.1.4.5-8*A*
Acceptable Average Initial Enrichment/Minimum Burnup Combinations - NUHOMS®-32PTH1 –
Damaged Fuel Assemblies

(Part 1 of 2)

Enrichment (wt. % U-235)	WE 17x17, WE 15x15, BW 15x15, CE 14x14, CE 15x15 and CE 16x16 fuel assembly classes									
	Type A	Type B	Type C	Type D	Type E	Type A	Type B	Type C	Type D	Type E
1.50	fresh	-	-	-	-	fresh	-	-	-	-
1.60	-	fresh	-	-	-	-	fresh	-	-	-
1.65	-	-	fresh	-	-	-	-	fresh	-	-
1.75	-	-	-	fresh	-	-	-	-	fresh	-
1.80	-	-	-	-	fresh	-	-	-	-	fresh
	Burnup (GWD/MTU), 15 years decay					Burnup (GWD/MTU), 30 years decay				
2.00	23	19	19	15	12	19	18	16	13	10
2.25	28	23	20	19	18	23	19	19	19	17
2.50	31	28	26	23	21	29	25	22	20	19
2.75	37	32	31	28	25	33	30	28	25	22
3.00	41	37	35	33	30	37	33	31	29	26
3.25	44	41	39	36	33	40	37	35	32	31
3.50	49	45	43	39	37	44	39	39	35	33
3.75	-	49	47	43	40	48	42	41	39	37
4.00	-	-	50	46	43	-	46	44	41	39
4.20	-	-	-	49	46	-	49	47	44	41
4.40	-	-	-	-	50	-	-	50	47	44
4.60	-	-	-	-	-	-	-	-	50	47
4.80	-	-	-	-	-	-	-	-	-	50
5.00	-	-	-	-	-	-	-	-	-	-

Table A.1.4.5-8A
Acceptable Average Initial Enrichment/Minimum Burnup Combinations - NUHOMS®-32PTH1 – Damaged Fuel Assemblies

(Part 2 of 2)

Enrichment (wt. % U-235)	WE 14x14 assembly class									
	Type A	Type B	Type C	Type D	Type E	Type A	Type B	Type C	Type D	Type E
1.70	fresh	-	-	-	-	fresh	-	-	-	-
1.85	-	fresh	-	-	-	-	fresh	-	-	-
1.90	-	-	fresh	-	-	-	-	fresh	-	-
1.75	-	-	-	fresh	-	-	-	-	fresh	-
1.80	-	-	-	-	fresh	-	-	-	-	fresh
	Burnup (GWD/MTU), 15 years decay					Burnup (GWD/MTU), 30 years decay				
2.00	20	18	16	15	12	19	16	14	13	10
2.25	26	20	19	19	18	22	19	19	19	17
2.50	31	26	24	23	21	28	23	21	20	19
2.75	35	31	29	28	25	31	28	26	25	22
3.00	39	35	34	33	30	35	32	31	29	26
3.25	43	39	38	36	33	39	35	34	32	31
3.50	47	42	40	39	37	41	39	38	35	33
3.75	-	47	44	43	40	47	41	40	39	37
4.00	-	-	48	46	43	50	45	43	41	39
4.20	-	-	-	49	46	-	48	46	44	41
4.40	-	-	-	-	50	-	-	50	47	44
4.60	-	-	-	-	-	-	-	-	50	47
4.80	-	-	-	-	-	-	-	-	-	50
5.00	-	-	-	-	-	-	-	-	-	-

Notes:

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these parameters (uncertainty in enrichment to be added and uncertainty in burnup to be subtracted).
- Interpolation can be performed to determine the burnup for enrichment values (between 2.00 wt.% U-235 and 5.00 wt.% U-235) that are not explicitly shown herein. Alternatively, the burnup value corresponding to the next higher enrichment may be utilized.
- Extrapolation shall not be performed to determine burnup requirements.
- The burnup of the "fresh" assemblies is 0. For a given configuration, the enrichment corresponding to "fresh" in this Table is the maximum enrichment above which a burnup value is needed for fuel assemblies to qualify for transportation.
- *An additional burnup of 3 GWD/MTU is required for loading fuel assemblies with control rod insertion deeper than 20 cm inside the active fuel during depletion.*

Table A.1.4.5-9
PWR Decay Heat for Heat Load Configurations

The Decay Heat (DH) in watts is expressed as:

$$F1 = -44.8 + 41.6*X1 - 37.1*X2 + 0.611*X1^2 - 6.80*X1*X2 + 24.0*X2^2$$
$$DH = F1*Exp(\{[1-(1.8/X3)]* -0.575\}*[(X3-4.5)^{0.169}]*[(X2/X1)^{-0.147}]) + 20$$

where,

F1 Intermediate Function
X1 Assembly Burnup in GWD/MTU
X2 Initial Enrichment in wt. % U-235
X3 Cooling Time in Years (minimum 10 years)

Note: Even though cooling times less than 15 years are shown in this table, the minimum cooling time requirement for criticality from Table A.1.4.5-8 *and Table A.1.4.5-8A* for transportation is 15 years.

Appendix A.1.4.6
NUHOMS®-37PTH DSC

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

Appendix A.1.4.6
NUHOMS®-37PTH DSC

NOTE: References in this Appendix are shown as [1], [2], etc. and refer to the reference list in Section A.1.4.6.4.

A.1.4.6.1 NUHOMS®-37PTH DSC Description

Each NUHOMS®-37PTH DSC consists of a DSC shell assembly and a basket assembly. The shell assembly consists of a cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies and outer top cover plate. *The DSC shell assembly is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NB [1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13.* As shown in Table A.1.4.6-1, the 37PTH DSC system consists of two design configurations as follows:

- 37PTH-S, Short Canister
- 37PTH-M, Medium Canister

Table A.1.4.6-1 provides the overall lengths and outer diameters for each 37PTH DSC configuration. The shell assemblies are high integrity stainless steel welded pressure vessels that provide confinement of radioactive materials, encapsulate the fuel in an inert atmosphere (the canister is back-filled with helium before being seal welded closed), and provide biological shielding (in axial direction). The 37PTH DSCs have double redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the 37PTH DSCs. The top end closure welds are made after fuel loading. Both top plug penetrations (siphon and vent ports) are redundantly sealed after the 37PTH DSC drying operations are complete.

The canister is designed to contain its fuel basket and fuel assemblies, and is completely supported by the transport cask. Under normal transport conditions, the canister rests on four canister rails attached to the inside of the transport cask.

A.1.4.6.2 NUHOMS®-37PTH DSC Fuel Basket

The basket structures are designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NG[1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The overall lengths and diameters of the baskets for each canister configuration are provided in Table A.1.4.6-1. The details of the 37PTH fuel baskets are shown in the drawings in Section A.1.4.10.7 of Appendix A.1.4.10. The 37PTH baskets are designed to accommodate 37 intact, or up to 4 damaged with the remainder intact, PWR fuel assemblies with or without Control Components. The basket structure consists of a welded assembly of stainless steel *plates or tubes that accommodate aluminum and/or poison plates* and surrounded by support rails.

The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the canister/cask body and not the fuel basket structure. The fuel assemblies are laterally supported by the stainless steel grid. The basket is laterally supported by the basket rails and the canister shell. The stainless steel and aluminum basket rails are oriented parallel to the axis of the canister, and are attached to the periphery of the basket to provide support, and to establish and maintain basket orientation.

*S*hear key*s*, welded to the inner wall of the DSC, mate with notch*es* in the basket support rails to prevent the basket from rotating during normal operations.

Each fuel compartment accommodates aluminum and/or absorbing poison plates. The poison plates are constructed from borated aluminum, or an aluminum/B4C metal matrix composite with a minimum B10 areal density of 0.020 gm/cm^2, and provide a heat conduction path along with the aluminum from the fuel assemblies to the canister wall, as well as criticality control. Alternatively, Boral® can be employed as the poison material, with a minimum B10 areal density of 0.025 gm/cm^2.

A.1.4.6.3 NUHOMS®-37PTH DSC Contents

Each of the two alternate DSC configurations is designed to transport intact (including reconstituted) and/or damaged PWR fuel assemblies as specified in Table A.1.4.6-2 and Table A.1.4.6-4. The fuel to be transported is limited to a maximum assembly average initial enrichment of 5.0 wt. % U-235. The maximum allowable assembly average burnup is limited to 62 GWd/MTU and the minimum cooling time *requirements are given in Table A.1.4.6-2.* Each of the DSC types is designed to transport Control Components (CCs) with thermal and radiological characteristics as listed in Table A.1.4.6-3. The CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs) and Neutron Sources.

Reconstituted assemblies containing up to 10 replacement irradiated stainless steel rods or stainless steel clad rods per assembly or an unlimited number of lower enrichment UO_2 rods, or Zircaloy (including other Zirconium based alloy) rods or Zr pellets, or unirradiated stainless steel rods are acceptable for storage in the 37PTH DSC as intact fuel assemblies. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The nominal volume of the replacement rods is equivalent to the replaced fueled rods in the active fuel region of the fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four with irradiated stainless steel replacement rods or 37 with UO_2 replacement rods.

The NUHOMS®-37PTH DSCs can also accommodate up to a maximum of four damaged fuel assemblies placed in the four cells of the DSC shown in Figure A.1.4.6-1. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks, or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is able to be handled by normal means. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.

Table A.1.4.6-1
Key Design Parameters of the NUHOMS®-37PTH System

Parameter	37PTH DSC Type	
	37PTH-S	37PTH-M
DSC Length (in)	182.00 (Maximum)	189.25 (Maximum)
DSC Outside Diameter (in)	69.75	69.75
DSC Cavity Length (in)	164.38	171.63
Basket Length (in)	162.00	169.00
Basket Diameter (in)	68.50	68.50

Note: Unless stated otherwise, nominal values are provided.

Table A.1.4.6-6
Acceptable Average Initial Enrichment / Minimum Burnup Combinations - NUHOMS®-37PTH – *Intact and Damaged Fuel Assemblies*

Enrichment (wt. % U-235)	***WE 17x17, WE 15x15, CE 14x14, CE 15x15 and CE 16x16 assembly classes***		***WE 14x14 assembly class***
1.65	*fresh*	*fresh*	-
1.90	-	-	*fresh*
	Burnup (GWD/MTU), 15 years decay	***Burnup (GWD/MTU), 30 years decay***	***Burnup (GWD/MTU), 15 years decay***
2.00	*14*	*12*	*5*
2.25	*19*	*18*	*10*
2.50	*20*	*19*	*15*
2.75	*25*	*22*	*19*
3.00	*30*	*27*	*20*
3.25	*32*	*31*	*24*
3.50	*37*	*32*	*28*
3.75	*39*	*36*	*31*
4.00	*42*	*39*	*33*
4.20	*44*	*40*	*35*
4.40	*47*	*42*	*38*
4.60	*50*	*44*	*39*
4.80	-	*47*	*40*
5.00	-	*50*	*43*

Notes:

- *Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these parameters (uncertainty in enrichment to be added and uncertainty in burnup to be subtracted)*
- *Interpolation can be performed to determine the burnup for enrichment values (between 2.00 wt. % U-235 and 5.00 wt. % U-235) that are not explicitly shown herein. Alternatively, the burnup value corresponding to the next higher enrichment may be utilized.*
- *Extrapolation shall not be performed to determine burnup requirements.*
- *The burnup of the "fresh" assemblies is 0. For a given configuration, the enrichment corresponding to "fresh" in this Table is the maximum enrichment above which a burnup value is needed for fuel assemblies to qualify for transportation.*
- *An additional burnup of 3 GWD/MTU is required for loading fuel assemblies with control rod insertion deeper than 20 cm inside the active fuel during depletion.*

This page intentionally left blank.

This page intentionally left blank.

Appendix A.1.4.7
NUHOMS®-61BT DSC

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

Appendix A.1.4.7
NUHOMS®-61BT DSC

NOTE: References in this Appendix are shown as [1], [2], etc. and refer to the reference list in Section A.1.4.7.4.

A.1.4.7.1 NUHOMS®-61BT DSC Description

Each NUHOMS®-61BT DSC consists of a DSC shell assembly and a basket assembly. The shell assembly consists of a cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies and outer top cover plate. *The DSC shell assembly is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NB [1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13.* The *maximum* length and the outer diameter of the 61BT DSC *are* approximately *196.0* inches and 67.3 inches respectively. The shell assembly is a high integrity stainless steel welded pressure vessel that provides confinement of radioactive materials, encapsulates the fuel in an inert atmosphere (the canister is back-filled with helium before being seal welded closed, and provides biological shielding (in axial direction). The 61BT DSC has double redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the 61BT DSC. The top plug penetrations (siphon and vent ports) are redundantly sealed after the 61BT DSC drying operations are complete.

The canister is designed to contain the fuel basket and fuel assemblies, and is completely supported by the transport cask. Under normal transport conditions, the canister rests on four canister rails attached to the inside surface of the aluminum inner sleeve of the MP197HB Transport Cask.

A.1.4.7.2 NUHOMS®-61BT Fuel Basket

The basket structure is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NG[1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The overall length and outer diameter of the basket, including the hold down ring, is approximately 178.*5* inches and 66.0 inches respectively. The details of the 61BT fuel baskets are shown in the drawings in Section A.1.4.10.8 of Appendix A.1.4.10. The 61BT basket is designed to accommodate 61 intact, or up to 16 damaged, with the remainder intact, BWR fuel assemblies with or without fuel channels. The basket structure consists of a welded assembly of stainless steel tubes (fuel compartments) separated by poison plates and surrounded by larger stainless steel boxes and support rails.

The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the canister/cask body and not on the fuel basket structure. The fuel assemblies are laterally supported by the stainless steel structural boxes. The basket is laterally supported by the basket rails and the canister shell. The stainless steel basket rails are oriented parallel to the axis of the canister, and are attached to the periphery of the basket to provide support, and to establish and maintain basket orientation.

A shear key, welded to the inner wall of the DSC, mates with a notch in one of the basket support rails to prevent the basket from rotating during normal operations. Also a hold down ring is installed above the basket to prevent the basket from moving axially during transport.

The poison plates are constructed from borated aluminum, or an aluminum/B4C metal matrix composite (MMC), or Boral® and provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control.

A.1.4.7.3 NUHOMS®-61BT DSC Contents

The NUHOMS®-61BT DSC is designed to transport 61 intact, or up to 16 damaged and the remainder intact, for a total of 61, standard BWR fuel assemblies with or without fuel channels. The NUHOMS®-61BT DSC can transport intact or damaged BWR fuel assemblies with the characteristics described in Table A.1.4.7-1. Damaged BWR fuel assemblies are fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks.

The NUHOMS®-61BT DSC may transport BWR fuel assemblies with a maximum decay heat of 300 watts/assembly, or a total of 18.3 kW. *The heat load zoning configuration for NUHOMS®-61BT DSC is uniform as shown in Figure A.1.4.7-1.*

The design characteristics of fuel assemblies considered are listed in Table A.1.4.7-2

The NUHOMS®-61BT DSC has three basket configurations, based on the boron content in the poison plates. The maximum lattice average enrichment authorized for Type A, B and C NUHOMS®-61BT DSCs is 3.7, 4.1 and 4.4 wt. % U-235, respectively.

Intact BWR fuel assemblies may be transported in any of the three NUHOMS®-61BT DSC Types provided the loading meets the maximum lattice average enrichment limit for the NUHOMS®-61BT DSC type, as given on Table A.1.4.7-3. Damaged BWR fuel assemblies may only be transported in Type C NUHOMS®-61BT DSCs with end caps installed on each four compartment assembly, where a damaged fuel assembly is authorized.

Fuel assemblies with various combinations of burnup, enrichment and cooling time can be transported in the NUHOMS®-61BT DSC as long as the fuel assembly parameters fall within the design limits specified in Table A.1.4.7-1, Table A.1.4.7-3, and Table A.1.4.7-4.

A.1.4.7.4 References

1. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Subsections NB, NG and NF, 1998 edition including 1999 Addenda.

Table A.1.4.7-1
BWR Fuel Specification for Fuel to be Transported in the NUHOMS®-61BT DSC

PHYSICAL PARAMETERS: Fuel Design	Intact or damaged unconsolidated 7x7, 8x8, 9x9, or 10x10 intact BWR fuel assemblies manufactured by General Electric or Exxon/ANF or reload fuel manufactured by the same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table A.1.4.7-2.
Fuel Damage(3)	Damaged BWR fuel assemblies are 7x7 and 8x8 fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in the fuel rods is to be limited such that a fuel assembly needs to be handled by normal means. Damaged fuel may only be transported in the "Type C" NUHOMS®-61BT Canister. Damaged fuel is restricted to the 7x7 and 8x8 designs only. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.
Channels	Fuel may be transported with or without fuel channels, *channel fasteners, or finger springs*
No. of Intact Assemblies	≤61
No. and Location of Damaged Assemblies	Up to sixteen (16) damaged fuel assemblies may be accommodated in the four corner 2x2 compartment assemblies with endcaps installed on each end of the compartment.
Maximum Assembly plus fuel channel weight	705 lbs
THERMAL/RADIOLOGICAL PARAMETERS(1): Maximum Initial ^{235}U Enrichment (wt. %)	Per Table A.1.4.7-3
Fuel Assembly Average Burnup and minimum Cooling Time(1) *(4)*	Per Table A.1.4.7-4 and decay heat restrictions below
Decay Heat(1)(2)	0.300 kW/Assembly calculated per Table A.1.4.7-5

(1) Minimum cooling time is the longer of that given in Table A.1.4.7-4; that calculated via the decay heat equation given in Table A.1.4.7-5 to meet the 0.300 kW/assembly limit.

(2) For FANP9 9x9-2 fuel assemblies, the maximum decay heat is limited to 0.21 kW/assembly.

(3) For damaged fuel assemblies, the maximum initial lattice average enrichment and maximum pellet enrichment is limited to 4.0 wt.% U-235 and 4.4 wt.% U-235 respectively.

(4) An additional cooling time of 8 years is required for damaged fuel assemblies in addition to that obtained from Table A.1.4.7-4, when 5 or more damaged fuel assemblies are loaded.

Table A.1.4.7-4
BWR Fuel Qualification Table for the NUHOMS®-61BT DSC
(Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU	Initial Enrichment																													
	1.4	1.5	1.6	1.7	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4
10	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
15	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
20	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
25	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
28	**Not Acceptable or Not Analyzed**				7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
30					7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
32					7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
34					7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
36					7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
38					7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
39					7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
40					7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 1.4 and greater than 4.4 wt.% U-235 is unacceptable for transportation.
- Fuel with a burnup greater than 40 GWd/MTU is unacceptable for transportation.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transportation after 7 years cooling.
- Example: An assembly with an initial enrichment of 4.15 wt.% U-235 and a burnup of 31.5 GWd/MTU is acceptable for transport after a 7-year year cooling time as defined by 4.1 wt. % U-235 (rounding down) and 32 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).
- *When loading five or more damaged fuel assemblies per DSC, an additional cooling time of 8 years is required for only damaged fuel assemblies.*

Table A.1.4.7-5
BWR Assembly Decay Heat for Heat Load Configurations

The Decay Heat (DH) in watts is expressed as:

$$F1 = -59.1 + 23.4*X1 - 21.1*X2 + 0.280*X1^2 - 3.52*X1*X2 + 12.4*X2^2$$
$$DH = F1*Exp(\{[1-(1.2/X3)]* -0.720\}*[(X3-4.5)^{0.157}]*[(X2/X1)^{-0.132}]) + 10$$

where,

F1 Intermediate Function
X1 Assembly Burnup in GWD/MTU
X2 Initial Enrichment in wt. % U-235
X3 Cooling Time in Years (minimum 7 years)

Note: Even though a minimum cooling time of 7 years is used, the minimum cooling time requirement with five or more damaged fuel assemblies from shielding requirements is per Table A.1.4.7-4.

			Z1	*Z1*	*Z1*			
	Z1 *	*Z1* *	*Z1*	*Z1*	*Z1*	*Z1* *	*Z1* *	
	Z1 *	*Z1* *	*Z1*	*Z1*	*Z1*	*Z1* *	*Z1* *	
Z1	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*
Z1	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*
Z1	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*	*Z1*
	Z1 *	*Z1* *	*Z1*	*Z1*	*Z1*	*Z1* *	*Z1* *	
	Z1 *	*Z1* *	*Z1*	*Z1*	*Z1*	*Z1* *	*Z1* *	
			Z1	*Z1*	*Z1*			

** Denotes only locations where damaged fuel assembly can be transported*

	Zone 1
Maximum Decay Heat (kW/FA)[(1)]	*0.30*
Maximum Decay Heat per Zone (kW)	*18.3*
Maximum Decay Heat per DSC (kW)	*18.3*

(1) Decay heat per fuel assembly shall be determined per Table A.1.4.7-5.

Figure A.1.4.7-1
Heat Load Zoning Configuration for 61BT DSCs

Appendix A.1.4.8
NUHOMS®-61BTH DSC

NOTE: References in this Appendix are shown as [1], [2], etc. and refer to the reference list in Section A.1.4.8.4.

A.1.4.8.1 NUHOMS®-61BTH DSC Description

Each NUHOMS®-61BTH DSC consists of a DSC shell assembly and basket assembly. The shell assembly consists of a cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies and outer top cover plate. *The DSC shell assembly is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NB [1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13.* The 61BTH DSC system consists of three design configurations, depending upon the type of fuel and heat load, as follows:

- 61BTH Type 1
- 61BTH Type 2
- 61BTHF, accommodates up to 4 Failed Fuel Cans with Failed Fuel

Table A.1.4.8-1 provides the overall lengths and outer diameters for each 61BTH DSC configuration. The shell assemblies are high integrity stainless steel welded pressure vessels that provide confinement of radioactive materials, encapsulate the fuel in an inert atmosphere (the canister is back-filled with helium before being seal welded closed), and provide biological shielding (in the axial direction). The 61BTH DSCs have double redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the 61BTH DSCs. The top end closure welds are made after fuel loading. Both top plug penetrations (siphon and vent ports) are redundantly sealed after the 61BTH DSC drying operations are complete.

The canister is designed to contain its fuel basket and fuel assemblies, and is completely supported by the transport cask. Under normal conditions, the canister rests on four canister rails attached to the inside surface of the aluminum inner sleeve of the transport cask.

A.1.4.8.2 NUHOMS®-61BTH DSC Fuel Basket

The basket structure is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NG[1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The overall lengths and outer diameters of the baskets, including the hold down rings, are provided in Table A.1.4.8-1. The details of the 61BTH fuel baskets are shown in the drawings in Section A.1.4.10.9 of Appendix A.1.4.10. The 61BTH baskets are designed to accommodate 61 intact, or up to 16 damaged with up to four (4) Failed Fuel Cans (FFCs) loaded with failed fuel with the remainder intact BWR fuel assemblies with or without fuel channels. The basket structure consists of a welded assembly of stainless steel tubes (fuel compartments) separated by poison plates and surrounded by larger stainless steel boxes and support rails.

The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the canister/cask body and not the fuel basket structure. The fuel assemblies are laterally supported by the stainless steel structural boxes. The basket is laterally supported by the basket rails and the canister shell. The stainless steel basket rails are oriented parallel to the axis of the canister, and are attached to the periphery of the basket to provide support, and to establish and maintain basket orientation.

The failed fuel assemblies are to be placed in individual Failed Fuel Cans (FFCs). Each FFC is constructed of sheet metal and is provided with a welded bottom closure and a removable top closure which allows lifting of the FFC with the enclosed damaged assembly/debris. The FFC is provided with screens at the bottom and top to contain fuel debris and allow fill/drainage of water from the FFC during loading operations. The FFC is protected by the fuel compartment tubes and its only function is to confine the failed fuel.

*S*hear key*s*, welded to the inner wall of the DSC, mate with notch*es* in the basket support rails to prevent the basket from rotating during normal operations. Also a hold down ring is installed above the basket to prevent the basket from moving axially during transport.

The NUHOMS®-61BTH DSC is designed with six alternate basket configurations based on the boron content in the poison plates as listed in Table A.1.4.8-4 or Table A.1.4.8-5 (designated as "A" for the poison plates with the lowest B10 loading to "F" for the highest B10 loading). Three alternate poison materials are allowed: (a) Borated Aluminum alloy, (b) Boron Carbide/Aluminum Metal Matrix Composite (MMC), or (c) Boral®. The poison plates provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control.

A.1.4.8.3 NUHOMS®-61BTH DSC Contents

Each of the NUHOMS®-61BTH DSC Type 1 and Type 2 configurations is designed to transport intact (including reconstituted) and/or damaged BWR fuel assemblies as specified in Table A.1.4.8-2 and Table A.1.4.8-3. In addition, the 61BTHF can transport up to four failed fuel assemblies placed in Failed Fuel Cans as described in Table A.1.4.8-2. The fuel to be transported is limited to a maximum lattice average initial enrichment of 5.0 wt. % ^{235}U. The maximum allowable fuel assembly average burnup is limited to 62 GWd/MTU and the minimum cooling time *requirement is given in Table A.1.4.8-2.*

Reconstituted fuel assemblies containing up to four replacement irradiated stainless steel rods per assembly or 61 lower enrichment UO_2 rods instead of Zircaloy clad enriched UO_2 rods are acceptable for storage in 61BTH DSCs as intact fuel assemblies. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four with irradiated stainless steel rods or 61 with UO_2 rods or Zr rods or Zr pellets or unirradiated stainless steel rods.

The NUHOMS®-61BTH DSCs can also accommodate up to a maximum of 16 damaged fuel assemblies placed in the 2x2 compartments located at the outer edge of the DSC as shown in Figure A.1.4.8-9. Damaged BWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel rods is to be limited such that the fuel assembly will still be able to be handled by normal means. Missing fuel rods are allowed. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.

The NUHOMS®-61BTHF DSC, an alternative version of NUHOMS®-61BTH DSC discussed in Section A.1.4.8.2 is designed to accommodate up to a maximum of four failed fuel assemblies in Failed Fuel Cans placed in cells located at the outer edge of the DSC as shown in Figure A.1.4.8-9. Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means.

Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a Rod Storage Basket are also considered as *failed* fuel. Loose fuel debris, not contained in a Rod Storage Basket may also be placed in a Failed Fuel Can for storage, provided the size of the debris is larger than the Failed Fuel Can screen mesh opening.

Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its content shall be less than 705 lbs.

A 61BTH DSC containing less than 61 fuel assemblies may contain dummy fuel assemblies in the empty slots. The dummy assemblies are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a fuel assembly.

The NUHOMS®-61BTH DSC may transport up to 61 BWR fuel assemblies arranged in any of the eight alternate heat load zoning configurations shown in Figure A.1.4.8-1 through A.1.4.8-8.

A.1.4.8.4 References

1. American Society of Mechanical Engineers, ASME Boiler And Pressure Vessel Code, Section III, Division 1 - Subsections NB, NG and NF, 1998 edition including 2000 Addenda.

Table A.1.4.8-1
Key Design Parameters of the NUHOMS®-61BTH System

Parameter	61BTH Type 1 DSC	61BTH Type 2 DSC
DSC Length (in.)	196.04 (Maximum)	196.04 (Maximum)
DSC Outside Diameter (in.)	67.25	67.25
DSC Cavity Length (in.)	179.50	179.50
Basket length (including holddown ring) (in.)	178.50	178.50
Basket OD (in.)	66.00	66.00

Note: Unless stated otherwise, nominal values are provided.

Table A.1.4.8-2
BWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-61BTH DSC

PHYSICAL PARAMETERS: Fuel Class	Intact or damaged or failed 7x7, 8x8, 9x9 or 10x10 BWR assemblies manufactured by General Electric or Exxon/ANF or FANP or ABB or reload fuel manufactured by same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table A.1.4.8-3. Damaged fuel assemblies beyond the definition contained below are not authorized for transport in damaged fuel locations shown in Figure A.1.4.8-9.
Damaged Fuel	Damaged BWR fuel assemblies are assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel rods is to be limited such that the fuel assembly will still be able to be handled by normal means. Missing fuel rods are allowed. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.
Failed Fuel	Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly can not be handled by normal means. Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a Rod Storage Basket are also considered as *failed* fuel. Loose fuel debris, not contained in a Rod Storage Basket may also be placed in a Failed Fuel Can for storage, provided the size of the debris is larger than the Failed Fuel Can screen mesh opening. Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its content shall be less than 705 lbs.
RECONSTITUTED FUEL ASSEMBLIES: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with unlimited number of low enriched UO2 rods or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	 4 4 61
No. of Intact Assemblies	≤61

Table A.1.4.8-2
BWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-61BTH DSC
(concluded)

No. and Location of Damaged Assemblies	Up to 16 damaged fuel assemblies, with balance intact or dummy assemblies, are authorized for transport in 61BTH DSC. Damaged fuel assemblies may only be transported in the 2x2 compartments as shown in Figure A.1.4.8-9. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
No. and Location of Failed Assemblies	Up to 4 failed fuel assemblies. Balance may be intact and/or damaged fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration. Failed fuel assemblies are to be placed as shown in Figure A.1.4.8-9. Failed fuel assembly/fuel debris is to be encapsulated in an individual Failed Fuel Can (FFC) provided with a welded bottom closure and a removable top closure.
Channels	Fuel may be transported with or without channels, channel fasteners, or finger springs.
Maximum Assembly Weight with Channels	705 lbs
THERMAL/RADIOLOGICAL PARAMETERS[(1)]: Maximum Initial ^{235}U Enrichment (wt. %)	Per Table A.1.4.8-4 or Table A.1.4.8-5.
Fuel Assembly Average Burnup and minimum Cooling Time*[(2)]*	Type 1 Per Table A.1.4.8-6.
	Type 2 Per Table A.1.4.8-7.
Decay Heat per DSC	≤22.0 kW for Type 1 DSC, per Figures A.1.4.8-1 through A.1.4.8-4
	≤24.0 kW for Type 2 DSC, per Figures A.1.4.8-1 through A.1.4.8-8
Minimum B10 Content in Poison Plates	Per Table A.1.4.8-4 or Table A.1.4.8-5.

Notes:

(1) Minimum cooling time is the longer of that given in Table A.1.4.8-6, *Table* A.1.4.8-7, and that calculated via the decay heat equation *given in Table A.1.4.8-8* based on the restrictions provided in Figures A.1.4.8-1 *through* A.1.4.8-8.

(2) An additional cooling time of 8 years is required for damaged fuel assemblies in addition to that obtained from Table A.1.4.8-6 or Table A.1.4.8-7, when 5 or more damaged fuel assemblies are loaded.

Notes: Tables A.1.4.8-6 and Table A.1.4.8-7:

- Burnup = Assembly Average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with a lattice average initial enrichment less than 0.9 (or less than the minimum provided above for each burnup) or greater than 5.0 wt.% U-235 is unacceptable for transportation.
- Fuel with a burnup greater than 62 GWd/MTU is unacceptable for transportation.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transportation after 7-years cooling.
- For reconstituted fuel assemblies with irradiated stainless steel rods, increase the cooling time by 1 year for fuel assemblies in the 24 peripheral locations of the canister with cooling times less than 10 years. No adjustment of cooling time is required for fuel assemblies in other locations or for those that have cooled for more than 10 years.
- The cooling times for failed, damaged and intact assemblies are identical. *However, when loading five or more damaged fuel assemblies per DSC, an additional cooling time of 8 years is required for only damaged fuel assemblies.*
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for transport after a 7-year year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).

Table A.1.4.8-8
BWR Assembly Decay Heat for Heat Load Configurations

The Decay Heat (DH) in watts is expressed as:

$$F1 = -59.1 + 23.4*X1 - 21.1*X2 + 0.280*X1^2 - 3.52*X1*X2 + 12.4*X2^2$$
$$DH = F1*Exp(\{[1-(1.2/X3)]* -0.720\}*[(X3-4.5)^{0.157}]*[(X2/X1)^{-0.132}]) + 10$$

where,

F1 Intermediate Function
X1 Assembly Burnup in GWD/MTU
X2 Initial Enrichment in wt. % U-235
X3 Cooling Time in Years (minimum 7 years)

Note: Even though a minimum cooling time of 7 years is used, the minimum cooling time requirement with five or more damaged fuel assemblies from shielding requirements is per Table A.1.4.8-6 for Type 1 DSC and A.1.4.8-7 for Type 2 DSC.

Appendix A.1.4.9
NUHOMS®-69BTH DSC

NOTE: References in this Appendix are shown as [1], [2], etc. and refer to the reference list in Section A.1.4.9.4.

A.1.4.9.1 NUHOMS®-69BTH DSC Description

Each NUHOMS®-69BTH DSC consists of a DSC shell assembly and a basket assembly. The shell assembly consists of a cylindrical shell, the inner cover plates of the top and bottom shield plug assemblies and outer top cover plate. *The DSC shell assembly is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NB [1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13.* The *maximum* length and the outer diameter of the 69BTH DSC *are* approximately 197.0 inches and 69.8 inches, respectively. The shell assembly is a high integrity stainless steel welded pressure vessel that provides confinement of radioactive materials, encapsulates the fuel in an inert atmosphere (the canister is back-filled with helium before being seal welded closed) and provides biological shielding (in axial direction). The 69BTH DSC has double redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the 69BTH DSC. The top plug penetrations (siphon and vent ports) are redundantly sealed after the 69BTH DSC drying operations are complete.

The canister is designed to contain the fuel basket and fuel assemblies, and is completely supported by the transport cask. Under normal transport conditions, the canister rests on four canister rails attached to the inside surface of the transport cask.

A.1.4.9.2 NUHOMS®-69BTH Fuel Basket

The basket structure is designed, fabricated and inspected in accordance with ASME B&PV Code Subsection NG[1]. Alternatives to the code are provided in Chapter A.2, Appendix A.2.13.13. The overall length and outer diameter of the basket, including the hold down ring, are approximately 178.6 inches and 68.4 inches respectively. The details of the 69BTH fuel basket is shown in the drawings in Section A.1.4.10.10 of Appendix A.1.4.10. The 69BTH basket is designed to accommodate 69 intact, or up to 24 damaged with the remainder intact BWR fuel assemblies with or without fuel channels. The basket structure consists of a welded assembly of stainless steel tubes (fuel compartments) separated by poison plates and surrounded by larger stainless steel boxes and support rails.

The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the canister/cask body and not the fuel basket structure. The fuel assemblies are laterally supported by the stainless steel structural boxes. The basket is laterally supported by the basket rails and the canister shell. The aluminum basket rails are oriented parallel to the axis of the canister, and are attached to the periphery of the basket to provide support, and to establish and maintain basket orientation.

*S*hear key*s*, welded to the inner wall of the DSC, mate with notch*es* in the basket support rails to prevent the basket from rotating during normal operations. Also a hold down ring is installed above the basket to prevent the basket from moving axially during transport.

The NUHOMS®-69BTH DSC is designed with six alternate basket configurations based on the boron content in the poison plates as listed in Table A.1.4.9-3 (designated as "A" for the poison plates with the lowest B10 loading to "F" for the highest B10 loading). Three alternate poison materials are allowed: (a) Borated Aluminum alloy, (b) Boron Carbide/Aluminum Metal Matrix Composite (MMC), or (c) Boral®. The poison plates provide a heat conduction path from the fuel assemblies to the canister wall, as well as the necessary criticality control.

A.1.4.9.3 NUHOMS®-69BTH DSC Contents

The NUHOMS®-69BTH DSC is designed to transport 69 intact, or up to 24 damaged and the remainder intact, standard BWR fuel assemblies with or without fuel channels. The NUHOMS®-69BTH DSC can transport intact or damaged BWR fuel assemblies with the characteristics described in Table A.1.4.9-1, which include a variety of cooling times, enrichment and maximum bundle average burnup. Damaged BWR fuel assemblies are fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that the fuel assembly will still be able to be handled by normal means. Missing fuel rods are allowed.

The fuel assemblies considered are listed in Table A.1.4.9-2.

A.1.4.9.4 References

1. American Society of Mechanical Engineers, ASME Boiler And Pressure Vessel Code, Section III, Division 1 - Subsections NB, NG and NF, 2004 edition including 2006 Addenda.

Table A.1.4.9-1
BWR Fuel Specification for the Fuel to be Transported in the NUHOMS®-69BTH DSC

PHYSICAL PARAMETERS: Fuel Class	Intact or damaged 7x7, 8x8, 9x9 or 10x10 BWR assemblies manufactured by General Electric or Exxon/ANF or FANP or ABB or reload fuel manufactured by same or other vendors that are enveloped by the fuel assembly design characteristics listed in Table A.1.4.9-2. Damaged fuel assemblies beyond the definition contained below are not authorized for transport.
Damaged Fuel	Damaged BWR fuel assemblies are assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that the fuel assembly will still be able to be handled by normal means. Missing fuel rods are allowed. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.
RECONSTITUTED FUEL ASSEMBLIES: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with unlimited number of low enriched UO2 rods or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	4 4 69
No. of Intact Assemblies	≤69
No. and Location of Damaged Assemblies	Up to 24 damaged fuel assemblies, with balance intact or dummy assemblies, are authorized for transport in 69BTH DSC. Damaged fuel assemblies may only be transported in the four outer "6-compartment" arrays as shown in Figure A.1.4.9-1. The DSC basket cells which accommodate damaged fuel assemblies are provided with top and bottom end caps.
Channels	Fuel may be transported with or without channels, channel fasteners, or finger springs.
Maximum Assembly Weight with Channels	705 lbs
THERMAL/RADIOLOGICAL PARAMETERS: Maximum Initial ^{235}U Enrichment (wt. %)	Per Table A.1.4.9-3.
Allowable Heat Load Zoning Configurations for each 69BTH DSC	Per Figure A.1.4.9-2 or Figure A.1.4.9-3 or Figure A.1.4.9-4 or Figure A.1.4.9-5.
Fuel Assembly Average Burnup and minimum Cooling Time(1)	Per Table A.1.4.9-4
Decay Heat per DSC	Per Figure A.1.4.9-2 or Figure A.1.4.9-3 or Figure A.1.4.9-4 or Figure A.1.4.9-5.
Minimum B10 Content in Poison Plates	Per Table A.1.4.9-3.

(1) An additional cooling time of 8 years is required for damaged fuel assemblies in addition to that obtained from Table A.1.4.9-4, when ***five*** *or more damaged fuel assemblies are loaded.*

Table A.1.4.9-3
BWR Fuel Assembly Initial Lattice Average Enrichment v/s Minimum B10 Requirements for the NUHOMS®-69BTH DSC Poison Plates

Basket Type	**Maximum Lattice Average Enrichment(1) (wt% U-235)**	**Minimum B10 Areal Density, gram/cm²**	
		Borated Aluminum/MMC	**Boral®**
A	3.7	*0.021*	*0.025*
B	4.1	*0.031*	*0.037*
C	4.4	*0.039*	*0.047*
D	4.6	*0.046*	*0.055*
E	4.8	*0.053*	*0.064*
F	5.0	*0.061*	*0.073*

	Maximum Lattice Average Initial Enrichment(1) (wt.% U-235)			
Basket Type	Intact Assemblies	Up to 4 Damaged Assemblies(2)	5 to 8 Damaged Assemblies(2)	9 to 24 Damaged Assemblies(2)
A	3.70	3.70	*3.30*	*2.80*
B	4.10	4.10	*3.60*	3.00
C	4.40	4.20	*3.60*	*3.10*
D	4.60	4.40	3.70	3.20
E	4.80	4.40	3.70	3.20
F	5.00	4.80	3.90	*3.40*

(1) For LaCrosse fuel assemblies, the enrichment shall be reduced by 0.1 wt. % U-235.
(2) Allowable locations in basket per Figure A.1.4.9-1.

Notes, Table A.1.4.9-4:

- Burnup = Assembly Average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with a lattice average initial enrichment less than 0.9 (or less than the minimum provided above for each burnup) or greater than 5.0 wt.% U-235 is unacceptable for transportation.
- Fuel with a burnup greater than 70 GWd/MTU is unacceptable for transportation.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for transportation after 6-years cooling.
- For reconstituted fuel assemblies with irradiated stainless steel rods, increase the cooling time by 1 for fuel assemblies in the 24 peripheral locations of the canister with cooling times less than 10 years. No adjustment of cooling time is required for fuel assemblies in other locations or for those that have cooled for more than 10 years.
- The cooling times for damaged and intact assemblies are identical. *However, when loading five or more damaged fuel assemblies per DSC, an additional cooling time of 8 years is required for only damaged fuel assemblies.*
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for transport after a 6-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table (other considerations not withstanding).

Table A.1.4.9-5
BWR Assembly Decay Heat for Heat Load Configurations

The Decay Heat (DH) in watts is expressed as:

$$F1 = -59.1 + 23.4*X1 - 21.1*X2 + 0.280*X1^2 - 3.52*X1*X2 + 12.4*X2^2$$
$$DH = F1*Exp(\{[1-(1.2/X3)]* -0.720\}*[(X3-4.5)^{0.157}]*[(X2/X1)^{-0.132}]) + 10$$

where,

F1 Intermediate Function
X1 Assembly Burnup in GWD/MTU
X2 Initial Enrichment in wt. % U-235
X3 Cooling Time in Years (minimum 6 years)

Note: Even though a minimum cooling time of 6 years is used, the minimum cooling time requirement with five or more damaged fuel assemblies from shielding requirements is per Table A.1.4.9-4.

Appendix A.1.4.9A
Radioactive Waste Canister

TABLE OF CONTENTS

LIST OF TABLES

Appendix A.1.4.9A Radioactive Waste Canister

NOTE: References in this appendix are shown as [1], [2], etc., and refer to the reference list in Section A.1.4.9A.4.

A.1.4.9A.1 Radioactive Waste Canister Description

The radioactive waste canister (RWC) is designed to contain dry irradiated and/or contaminated non-fuel-bearing solid materials (described further in paragraph A.1.4.9A.3), and is completely supported by the transport cask. Under normal transport conditions, the canister rests on four canister rails, attached to the inside surface of the aluminum inner sleeve of the NUHOMS®-MP197HB transport cask. The RWC is designed to transport its payload dry and in an air or inert gas environment. When a wet-load procedure (i.e., in-pool) is followed for cask loading, the RWC and transport cask cavities are drained and dried in order to ensure that free liquids do not remain in the package during transport. The heat generated by the contents of the RWC is transferred through the transport cask to the environment by conduction, convection and radiation. No forced cooling is required.

Each RWC system includes an outer cylindrical shell assembly. The shell assembly consists of a cylindrical shell, top shield plug, outer top cover plate, bottom shield plug, and outer bottom cover plate. As shown in Table A.1.4.9A-1, the RWC system consists of two design configurations:

- *Welded Top Shield Plug Design (RWC-W)*
- *Bolted Top Shield Plug Design (RWC-B)*

Table A.1.4.9A-1 provides the overall dimensions for each RWC configuration. The details of each configuration are included in the drawings contained in Section A.1.4.10.11 of Appendix A.1.4.10.

The RWC shell assemblies are stainless steel welded vessels that provide confinement of radioactive materials, encapsulate the contents in an air or inert atmosphere, and provide biological shielding. The RWC shell has redundant seal welds that join the shell and the top and bottom cover plate assemblies to seal the canister. The bottom end assembly welds are made during fabrication of the RWC shell. The top end closure welds are made after content loading. Both top plug penetrations (siphon and vent ports) are sealed after the RWC drying and backfilling operations are complete.

The RWC cylindrical shell, outer top cover plate and outer bottom cover plate are fabricated from ASTM A240 type 304 stainless steel. The bottom and top shield plugs are fabricated from ASTM A240 Type F304 or ASTM A182 Type 304 stainless steel. All RWC welding procedures, welders, and welding are performed in accordance with the requirements of AWS D1.1-98 [1] and AWS D1.6-99 [2]. All inspections are performed in accordance with AWS D1.1-98 [1] and AWS D1.6-99 [2].

Material properties used are listed in Chapter A.2, Table A.2-4. All structural components and payloads are the same or similar alloys of stainless steel and therefore, are not subject to chemical or galvanic interaction. Similarly, no hydrogen gas generation is expected.

A.1.4.9A.2 RWC Inner Liner

The inner liner assembly is a stainless steel welded cylinder with a bottom plate that is used with the RWC-W. The bottom plate is designed with drain holes to allow liquid from the inner liner to drain to the bottom of the RWC for dewatering.

All inner liner welding procedures, welders, and welding are performed in accordance with the requirements of AWS D1.6-99 [2]. All inspections are performed in accordance with AWS D1.6-99 [2]. The overall length and diameter of the liner are provided in Table A.1.4.9A-2. Details of the inner liner are shown in the drawings contained in Section A.1.4.10.11 of Appendix A.1.4.10.

Four lifting lugs are provided on the inner liner for lifting the inner liner either empty or loaded. The lugs are designed, fabricated and tested to the requirements of ANSI N14.6 [3]. The inner liner is manufactured with a keyway for alignment in the outer RWC-W canister.

A.1.4.9A.3 RWC Contents

The NUHOMS®-MP197HB packaging is designed to transport a payload of up to 56.0 tons of dry irradiated and/or contaminated non-fuel bearing solid materials in the RWC. The safety analysis of the cask takes no credit for the containment provided by the RWC.

The quantity of radioactive material is limited to a maximum of 8,182 A_2. The radioactive material is typically in the form of neutron activated metals, or metal oxides in solid form. Surface contamination may also be present on the irradiated components. When a wet-load procedure (i.e., in-pool) is followed for cask loading, the cask cavity and RWC are drained and dried to ensure that there are no free liquids in the package during transport.

The payload will vary from shipment to shipment. Typical composition of the payload consists of the following components either individually or in combinations:

1. *BWR Control Rod Blades*
2. *BWR Local Power Range Monitors (LPRMs)*
3. *BWR Fuel Channels*
4. *BWR Poison Curtains*
5. *PWR Burnable Poison Rod Assemblies (BPRAs)*
6. *PWR and BWR Reactor Vessel and Internals*

The typical cobalt-60 specific activity ranges for these items are as follows:

1. *Control Rod Blades*	$1.3x10^{-4} - 1.1x10^{-2}$ Ci/g
2. *LPRMs*	$1.0x10^{-2} - 4.8x10^{-2}$ Ci/g
3. *Fuel Channels*	$7.8x10^{-5} - 2.0x10^{-4}$ Ci/g
4. *Poison Curtains*	$6.2x10^{-4} - 4.0x10^{-2}$ Ci/g
5. *BPRAs*	$3.8x10^{-4} - 1.3x10^{-3}$ Ci/g

6. *Reactor Vessel and Internals* $2.0x10^{-5} - 1.3x10^{-2}$ *Ci/g*

Components with high specific activity are generally placed near the center of the RWC. For each shipment, the RWC is normally filled to capacity, which prevents shifting of the contents during transport. If the RWC is not full, appropriate component spacers or shoring is used to prevent significant movement of the contents.

The RWC assembly provides a minimum steel thickness of 1.75 inches in the radial direction. The RWC assembly provides a minimum steel thickness of 5.75 inches below the payload and a minimum steel thickness of 7.00 inches above the payload in the axial direction.

A.1.4.9A.4 References

1. *American Welding Society, D1.1-98, Structural Welding Code-Steel*
2. *American Welding Society, D1.6-99, Structural Welding Code-Stainless Steel*
3. *ANSI N14.6, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More, 1993*

Table A.1.4.9A-1
Nominal Dimensions of the RWC

	RWC Design Parameters	
	RWC-W	***RWC-B***
Shell Thickness (in)	*1.25*	*1.75*
Canister Length (in.)	*186.50*	*186.50*
Outside Diameter (in.)	*67.19*	*67.19*
Cavity Length (in.)	*167.30*	*167.30*
Cavity Diameter (in.)	*64.69*	*63.69*

Table A.1.4.9A-2
Nominal Dimensions of the RWC Inner Liner

	RWC-W Inner Liner Design Parameters
Shell Thickness (in.)	*0.50*
Outside Length (in.)	*166.30*
Outside Diameter (in.)	*63.69*
Cavity Length (in.)	*162.11*
Cavity Diameter (in.)	*62.69*

Appendix A.1.4.10
Drawings of Transport Packaging and DSCs

TABLE OF CONTENTS

Appendix A.1.4.10
NUHOMS®-MP197HB SAR Drawings

The following drawings for the NUHOMS®-MP197HB Cask are included in Section A.1.4.10.1.

Drawing Number	Title
MP197HB-71-1001 Rev *1*	NUHOMS®-MP197HB Packaging Transport Configuration (2 sheets)
MP197HB-71-1002 Rev *1*	NUHOMS®-MP197HB Packaging Parts List (2 sheets)
MP197HB-71-1003 Rev *1*	NUHOMS®-MP197HB Packaging General Arrangement (1 sheet)
MP197HB-71-1004 Rev *1*	NUHOMS®-MP197HB Packaging Cask Body Assembly (1 sheet)
MP197HB-71-1005 Rev *1*	NUHOMS®-MP197HB Packaging Cask Body Details (3 sheets)
MP197HB-71-1006 Rev 0	NUHOMS®-MP197HB Packaging Lid Assembly & Details (1 sheet)
MP197HB-71-1007 Rev 0	NUHOMS®-MP197HB Packaging Regulatory Plate (1 sheet)
MP197HB-71-1008 Rev *1*	NUHOMS®-MP197HB Packaging Impact Limiter Assembly (1 sheet)
MP197HB-71-1009 Rev *1*	NUHOMS®-MP197HB Packaging Impact Limiter Details (1 sheet)
MP197HB-71-1011 Rev 0	NUHOMS®-MP197HB Packaging Transport Configuration Outer Sleeve With Fins Option (1 sheet)
MP197HB-71-1014 Rev 0	NUHOMS®-MP197HB Packaging Internal Sleeve Design (2 sheets)

The following drawings for the NUHOMS® 24PT4 DSC are included in Section A.1.4.10.2.

Drawing Number	Title
NUH24PT4-71-1001 Rev 0	NUHOMS® 24PT4 Transportable Canister For PWR Fuel Basket Assembly (5 sheets)
NUH24PT4-71-1002 Rev 0	NUHOMS® 24PT4 Transportable Canister For PWR Fuel Main Assembly (8 sheets)
NUH24PT4-71-1003 Rev 0	NUHOMS® 24PT4 Transportable Canister For PWR Fuel Failed Fuel Can (4 sheets)

The following drawings for the NUHOMS® 32PT DSC are included in Section A.1.4.10.3.

Drawing Number	Title
NUH32PT-71-1000 Rev 0	NUHOMS® 32PT Transportable Canister For PWR Fuel Summary Dimensions (1 sheet)
NUH32PT-71-1001 Rev 0	NUHOMS® 32PT Transportable Canister For PWR Fuel Main Assembly (5 sheets)
NUH32PT-71-1002 Rev 0	NUHOMS® 32PT Transportable Canister For PWR Shell Assembly (3 sheets)
NUH32PT-71-1003 Rev 0	NUHOMS® 32PT Transportable Canister For PWR Fuel "A" Basket Assembly (16 Poison/16 Compartment Plates) (8 sheets)
NUH32PT-71-1004 Rev 0	NUHOMS® 32PT Transportable Canister For PWR Fuel Aluminum Transition Rail – R90 (2 sheets)
NUH32PT-71-1005 Rev 0	NUHOMS® 32PT Transportable Canister For PWR Fuel Aluminum Transition Rail – R45 (1 sheet)
NUH32PT-71-1006 Rev 0	NUHOMS® 32PT Transportable Canister For PWR Fuel "A/B/C/D" Basket Assembly (20 Poison/12 Compartment Plates) (6 sheets)
NUH32PT-71-1007 Rev 0	NUHOMS® 32PT Transportable Canister For PWR Fuel "A/B/C/D" Basket Assembly (24 Poison/8 Compartment Plates) (8 sheets)

The following drawings for the NUHOMS® 24PTH DSC are included in Section A.1.4.10.4.

Drawing Number	Title
NUH24PTH-71-1000 Rev 0	NUHOMS® 24PTH Transportable Canister For PWR Fuel Main Assembly (5 sheets)
NUH24PTH-71-1001 Rev 0	NUHOMS® 24PTH Transportable Canister For PWR Fuel Basket-Shell Assembly (4 sheets)
NUH24PTH-71-1002 Rev 0	NUHOMS® 24PTH Transportable Canister For PWR Shell Assembly (4 sheets)
NUH24PTH-71-1003 Rev *1*	NUHOMS® 24PTH Transportable Canister For PWR Fuel Basket Assembly (*8* sheets)
NUH24PTH-71-1004 Rev 0	NUHOMS® 24PTH Transportable Canister For PWR Fuel Transition Rails (4 sheets)
NUH24PTH-71-1008 Rev 0	NUHOMS® 24PTHF Transportable Canister For PWR Fuel Failed Fuel Can (2 sheets)
NUH24PTH-71-1009 Rev 0	NUHOMS® 24PTHF Transportable Canister For PWR Fuel Basket Assembly (8 sheets)

The following drawings for the NUHOMS® 32PTH DSC and the 32PTH Type 1 DSC are included in Section A.1.4.10.5.

Drawing Number	Title
NUH32PTH-71-1001 Rev *1*	NUHOMS®32PTH Transportable Canister for PWR Fuel Parts List (1 Sheet)
NUH32PTH-71-1002 Rev *1*	NUHOMS®32PTH Transportable Canister for PWR Fuel Main Assembly (1 Sheet)
NUH32PTH-71-1003 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Siphon Pipe Details (1 Sheet)
NUH32PTH-71-1004 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Inner Top Cover Details (2 sheets)
NUH32PTH-71-1005 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Outer Top Cover Details (1 Sheet)
NUH32PTH-71-1006 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Shell Assembly (1 Sheet)
NUH32PTH-71-1007 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Shell Bottom Details (1 Sheet)
NUH32PTH-71-1008 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Grapple Ring Details (1 Sheet)
NUH32PTH-71-1009 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Basket Assembly (1 Sheet)
NUH32PTH-71-1010 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Basket Assembly Details (1 Sheet)
NUH32PTH-71-1011 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Basket Assembly Details (1 Sheet)
NUH32PTH-71-1012 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Basket Assembly – Details (1 Sheet)
NUH32PTH-71-1013 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Basket Rail A180 (1 Sheet)
NUH32PTH-71-1014 Rev 0	NUHOMS®32PTH Transportable Canister for PWR Fuel Basket Rail A90 (1 Sheet)
NUH32PTH-71-1015 Rev 0	*NUHOMS®32PTH Transportable Canister for PWR Fuel Damaged Fuel End Caps (1 Sheet)*
NUH32PTH Type 1-71-1000 Rev 0	NUHOMS® 32PTH Type 1 Transportable Canister For PWR Fuel Main Assembly (4 sheets)
NUH32PTH Type 1-71-1001 Rev *1*	NUHOMS® 32PTH Type 1 Transportable Canister For PWR Fuel Basket Shell Assembly (*4* sheets)
NUH32PTH Type 1-71-1002 Rev 0	NUHOMS® 32PTH Type 1 Transportable Canister For PWR Fuel Shell Assembly (4 sheets)

Drawing Number	Title
NUH32PTH Type 1-71-1003 Rev *1*	NUHOMS® 32PTH Type 1 Transportable Canister For PWR Fuel Basket Assembly (7 sheets)
NUH32PTH Type 1-71-1004 Rev *1*	NUHOMS® 32PTH Type 1 Transportable Canister For PWR Fuel Transition Rails (*4* sheets)
NUH32PTH Type 1-71-1010 Rev 0	NUHOMS® 32PTH Type 1 Transportable Canister For PWR Fuel Alternate Top Closure (6 sheets)

The following drawings for the NUHOMS® 32PTH1 DSC are included in Section A.1.4.10.6.

Drawing Number	Title
NUH32PTH1-71-1000 Rev 0	NUHOMS® 32PTH1 Transportable Canister For PWR Fuel Main Assembly (4 sheets)
NUH32PTH1-71-1001 Rev 0	NUHOMS® 32PTH1 Transportable Canister For PWR Fuel Basket Shell Assembly (5 sheets)
NUH32PTH1-71-1002 Rev 0	NUHOMS® 32PTH1 Transportable Canister For PWR Fuel Shell Assembly (4 sheets)
NUH32PTH1-71-1003 Rev *1*	NUHOMS® 32PTH1 Transportable Canister For PWR Fuel Basket Assembly (*8* sheets)
NUH32PTH1-71-1004 Rev 0	NUHOMS® 32PTH1 Transportable Canister For PWR Fuel Transition Rails (7 sheets)
NUH32PTH1-71-1010 Rev 0	NUHOMS® 32PTH1 Transportable Canister For PWR Fuel Alternate Top Closure (6 sheets)

The following drawings for the NUHOMS® 37PTH DSC are included in Section A.1.4.10.7.

Drawing Number	Title
NUH37PTH-71-1001 Rev *1*	NUHOMS® 37PTH Transportable Canister For PWR Fuel Main Assembly (4 sheets)
NUH37PTH-71-1002 Rev *1*	NUHOMS® 37PTH Transportable Canister For PWR Fuel Basket Shell Assembly (5 sheets)
NUH37PTH-71-1003 Rev *1*	NUHOMS® 37PTH Transportable Canister For PWR Fuel Shell Assembly (4 sheets)
NUH37PTH-71-1004 Rev *1*	NUHOMS® 37PTH Transportable Canister For PWR Fuel Alternate 2 Top Closure (6 sheets)
NUH37PTH-71-1011 Rev *1*	NUHOMS® 37PTH Transportable Canister For PWR Fuel Basket Assembly (7 sheets)
NUH37PTH-71-1012 Rev *1*	NUHOMS® 37PTH Transportable Canister For PWR Fuel Transition Rails (7 sheets)
NUH37PTH-71-1015 Rev 0	*NUHOMS® 37PTH Transportable Canister For PWR Fuel Damaged Fuel End Caps (1 sheet)*

The following drawings for the NUHOMS® 61BT DSC are included in Section A.1.4.10.8.

Drawing Number	Title
NUH61BT-71-1000 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Parts List (1 sheet)
NUH61BT-71-1001 Rev *1*	NUHOMS® 61BT Transportable Canister For BWR Fuel Basket Assembly (1 sheet)
NUH61BT-71-1002 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Basket Details (1 sheet)
NUH61BT-71-1003 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel General Assembly (1 sheet)
NUH61BT-71-1004 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel General Assembly (1 sheet)
NUH61BT-71-1005 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Shell Assembly (1 sheet)
NUH61BT-71-1006 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Shell Assembly (1 sheet)
NUH61BT-71-1007 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Canister Details (1 sheet)
NUH61BT-71-1008 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Canister Details (1 sheet)
NUH61BT-71-1009 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Basket Details (1 sheet)
NUH61BT-71-1010 Rev 0	NUHOMS® 61BT Transportable Canister For BWR Fuel Additional Basket Details – Damaged Fuel (4 sheets)

The following drawings for the NUHOMS® 61BTH DSC are included in Section A.1.4.10.9.

Drawing Number	Title
NUH61BTH-71-1000 Rev 0	NUHOMS® 61BTH Type 1 Transportable Canister For BWR Fuel Main Assembly (5 sheets)
NUH61BTH-71-1100 Rev *1*	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Main Assembly (7 sheets)
NUH61BTH-71-1101 Rev 0	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Shell Assembly (2 sheets)
NUH61BTH-71-1102 Rev *1*	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Basket Assembly (8 sheets)
NUH61BTH-71-1103 Rev 0	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Transition Rails (2 sheets)
NUH61BTH-71-1104 Rev 0	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Damaged Fuel End Caps (1 sheet)
NUH61BTH-71-1105 Rev 0	NUHOMS® 61BTHF Type 2 Transportable Canister For BWR Fuel Failed Fuel Can (2 sheets)
NUH61BTH-71-1106 Rev *1*	NUHOMS® 61BTH Type 2 Transportable Canister For BWR Fuel Top Grid Assembly Alternate 3 (2 sheets)

The following drawings for the NUHOMS® 69BTH DSC are included in Section A.1.4.10.10.

Drawing Number	Title
NUH69BTH-71-1001 Rev *1*	NUHOMS® 69BTH Transportable Canister For BWR Fuel Main Assembly (4 sheets)
NUH69BTH-71-1002 Rev *1*	NUHOMS® 69BTH Transportable Canister For BWR Fuel Basket – Shell Assembly (4 sheets)
NUH69BTH-71-1003 Rev *1*	NUHOMS® 69BTH Transportable Canister For BWR Fuel Shell Assembly (4 sheets)
NUH69BTH-71-1004 Rev *1*	NUHOMS® 69BTH Transportable Canister For BWR Fuel Alternate Top Closure (6 sheets)
NUH69BTH-71-1011 Rev *1*	NUHOMS® 69BTH Transportable Canister For BWR Fuel Basket Assembly (5 sheets)
NUH69BTH-71-1012 Rev *1*	NUHOMS® 69BTH Transportable Canister For BWR Fuel Transition Rail Assembly And Details (6 sheets)
NUH69BTH-71-1013 Rev *1*	NUHOMS® 69BTH Transportable Canister For BWR Fuel Holddown Ring Assembly (2 sheets)
NUH69BTH-71-1014 Rev *1*	NUHOMS® 69BTH Transportable Canister For BWR Fuel Damaged Fuel Modification (1 sheet)
NUH69BTH-71-1015 Rev *1*	NUHOMS® 69BTH Transportable Canister For BWR Fuel Damaged Fuel End Caps (1 sheet)

The following drawings for the Radioactive Waste Canister are included in Section A.1.4.10.11.

Drawing Number	*Title*
NUHRWC-71-1001 Rev 0	*NUHOMS® System RWC Canister - Welded Top Shield Plug Design Main Assembly (5 sheets)*
NUHRWC-71-1002 Rev 0	*NUHOMS® System RWC Canister - Welded Top Shield Plug Design Inner Liner (3 sheets)*
NUHRWC-71-1003 Rev 0	*NUHOMS® System RWC Canister - Bolted Top Shield Plug Design Main Assembly (4 sheets)*

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED FOR FABRICABILITY ENHANCEMENTS	04/07/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. Patent No. 4,780,269
Transnuclear, Inc.

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT

NUHOMS® MP197HB PACKAGING
TRANSPORT CONFIGURATION

DRAWING NO.	SCALE	SHEET
MP197HB-71-1001	NONE	1 OF 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER NRC RAI #1 ITEMS 4-3 AND 7-3 AND FABRICABILITY ENHANCEMENTS	04/07/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. Patent No. 4,780,269
Transnuclear, Inc.
This drawing may not be disclosed to others in whole or in part, or used for other than the transmitted purpose without written permission of Transnuclear, Inc.

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT

NUHOMS® MP197HB PACKAGING PARTS LIST

DRAWING NO. MP197HB-71-1002 | SCALE NONE | SHEET 1 OF 2

DRAWING NO. MP197HB-71-1002 SHEET 2 OF 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
MP197HB-71-1002	2 OF 2	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED FOR FABRICABILITY ENHANCEMENTS	04/07/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. Patent No. 4,780,269
Transnuclear, Inc.

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® MP197HB PACKAGING
GENERAL ARRANGEMENT

DRAWING NO. MP197HB-71-1003 | SCALE NONE | SHEET 1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER FABRICABILITY ENHANCEMENTS; EDITORIAL CORRECTIONS	04/07/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269

TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® MP197HB PACKAGING
CASK BODY ASSEMBLY

DRAWING NO. MP197HB-71-1004 | SCALE NONE | SHEET 1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED FOR FABRICABILITY ENHANCEMENTS AND PER NRC RAI #1 5-1	04/07/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. Patent No. 4,780,269 Transnuclear, Inc.

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT

NUHOMS®MP197HB PACKAGING
CASK BODY DETAILS

DRAWING NO. MP197HB-71-1005 | SCALE NONE | SHEET 1 OF 3

8 7 6 5 4 3 2 1

H G F E D C B A

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

8 7 6 5 4 3 2 1

DRAWING NO.	SHEET	REVISION
MP197HB-71-1005	2 OF 3	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
MP197HB-71-1005	3 OF 3	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER NRC RAI #1	04/13/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® MP197HB PACKAGING
IMPACT LIMITER ASSEMBLY

DRAWING NO.	SCALE	SHEET
MP197HB-71-1008	NONE	1 OF 1

DRAWING NO. MP197HB-71-1008 SHEET 1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER NRC RAI #1	04/13/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT

NUHOMS® MP197HB PACKAGING
IMPACT LIMITER DETAILS

DRAWING NO. MP197HB-71-1009 | SCALE NONE | SHEET 1 OF 1

NUH24PTH-71-1003 1 OF 8

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED FOR FABRICABILITY ENHANCEMENTS	04/07/10
0	FIRST ISSUE	03/25/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 24PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
BASKET ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH24PTH-71-1003	NONE	1 OF 8

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH24PTH-71-1003	2 OF 8	1

DRAWING NO. NUH24PTH-71-1003 SHEET 2 OF 8

NUH24PTH-71-1003 3 OF 8

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH24PTH-71-1003 SHEET 3 OF 8 REVISION 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH24PTH-71-1003	4 OF 8	1

DRAWING NO. NUH24PTH-71-1003 SHEET 4 OF 8

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH24PTH-71-1003	5 OF 8	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH24PTH-71-1003 SHEET 8 OF 8 REVISION 1

A.1.4.10.5 NUHOMS® 32PTH DSC DRAWINGS

This section contains drawings for the NUHOMS® 32PTH DSC.

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER NRC RAI #1 2-25	03/23/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®32PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
PARTS LIST

DRAWING NO.	SCALE	SHEET
NUH32PTH-71-1001	NONE	1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER NRC RAI #1 2-33	04/07/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. Patent No. 4,780,269
Transnuclear, Inc.

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®32PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH32PTH-71-1002	NONE	1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
0	INITIAL ISSUE PER NRC RAI #1 2-3	04/08/10

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. Patent No. 4,780,269
Transnuclear, Inc.

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®32PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
DAMAGED FUEL END CAPS

DRAWING NO.	SCALE	SHEET
NUH32PTH-71-1015	NONE	1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	EDITORIAL CORRECTIONS	04/07/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®32PTH TYPE 1
TRANSPORTABLE CANISTER FOR PWR FUEL
BASKET SHELL ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH32PTH TYPE 1-71-1001	NONE	1 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH TYPE 1-71-1001	2 OF 4	1

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH TYPE 1-71-1001	3 OF 4	1

NUH32PTH TYPE 1-71-1001 4 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH TYPE 1-71-1001	4 OF 4	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

1	REVISED PER RAI #1 2-3	04/07/10
0	FIRST ISSUE	03/26/09
REVISION	DESCRIPTION	DATE

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®32PTH TYPE 1
TRANSPORTABLE CANISTER FOR PWR FUEL
BASKET ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH32PTH TYPE 1-71-1003	NONE	1 OF 7

8 7 6 5 4 3 2 1

DRAWING NO. NUH32PTH TYPE 1-71-1003 | SHEET 2 OF 7

H G F E D C B A

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH TYPE 1-71-1003	2 OF 7	1

8 7 6 5 4 3 2 1

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH32PTH TYPE 1-71-1003 SHEET 3 OF 7

DRAWING NO. NUH32PTH TYPE 1-71-1003 | SHEET 3 OF 7 | REVISION 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

NUH32PTH TYPE 1-71-1003 | 4 OF 7

DRAWING NO.	SHEET	REVISION
NUH32PTH TYPE 1-71-1003	4 OF 7	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH TYPE 1-71-1003	5 OF 7	1

DRAWING NO. NUH32PTH TYPE 1-71-1003 SHEET 5 OF 7

8 7 6 5 4 3 2 1

H G F E D C B A

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH TYPE 1-71-1003	6 OF 7	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

1	REVISED FOR FABRICABILITY ENHANCEMENTS	04/07/10
0	FIRST ISSUE	03/25/09
REVISION	DESCRIPTION	DATE

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

THIS DRAWING MAY NOT BE DISCLOSED TO OTHERS IN WHOLE OR IN PART, OR USED FOR OTHER THAN THE TRANSMITTED PURPOSE WITHOUT WRITTEN PERMISSION OF TRANSNUCLEAR, INC.

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 32PTH TYPE 1
TRANSPORTABLE CANISTER FOR PWR FUEL
TRANSITION RAILS

DRAWING NO.	SCALE	SHEET
NUH32PTH TYPE 1-71-1004	NONE	1 OF 4

DRAWING NO. NUH32PTH TYPE 1-71-1004 SHEET 1 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH TYPE 1-71-1004	2 OF 4	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH TYPE 1-71-1004	3 OF 4	1

NUH32PTH TYPE 1-71-1004 4 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH TYPE 1-71-1004	4 OF 4	1

A.1.4.10.6 NUHOMS® 32PTH1 DSC DRAWINGS

This section contains drawings for the NUHOMS® 32PTH1 DSC.

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER RAI #1 2-3	04/07/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®32PTH1
TRANSPORTABLE CANISTER FOR PWR FUEL
BASKET ASSEMBLY

DRAWING NO. NUH32PTH1-71-1003 | SCALE NONE | SHEET 1 OF 8

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-71-1003	2 OF 8	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-71-1003	3 OF 8	1

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-71-1003	4 OF 8	1

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-71-1003	6 OF 8	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

8 7 6 5 4 3 2 1

H G F E D C B A

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH32PTH1-71-1003	8 OF 8	1

8 7 6 5 4 3 2 1

DRAWING NO. NUH32PTH1-71-1003 SHEET 8 OF 8

A.1.4.10.7 NUHOMS® 37PTH DSC DRAWINGS

This section contains drawings for the NUHOMS® 37PTH DSC.

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

1	REVISED FOR FABRICABILITY ENHANCEMENTS & EDITORIAL CORRECTIONS	04/07/10
0	FIRST ISSUE	04/07/09
REVISION	DESCRIPTION	DATE

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

THIS DRAWING MAY NOT BE DISCLOSED TO OTHERS IN WHOLE OR IN PART, OR USED FOR OTHER THAN THE TRANSMITTED PURPOSE WITHOUT WRITTEN PERMISSION OF TRANSNUCLEAR, INC.

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®37PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH37PTH-71-1001	NONE	1 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH37PTH-71-1001 | SHEET 2 OF 4 | REVISION 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1001	3 OF 4	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1001	4 OF 4	1

DRAWING NO. NUH37PTH-71-1001 SHEET 4 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED FOR FABRICABILITY ENHANCEMENTS & EDITORIAL CORRECTIONS	04/07/10
0	FIRST ISSUE	03/25/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®37PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
BASKET SHELL ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH37PTH-71-1002	NONE	1 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1002	2 OF 5	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1002	5 OF 5	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

1	REVISED FOR FABRICABILITY ENHANCEMENTS & EDITORIAL CORRECTIONS	04/07/10
0	FIRST ISSUE	03/25/09
REVISION	DESCRIPTION	DATE

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®37PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
SHELL ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH37PTH-71-1003	NONE	1 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1003	2 OF 4	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1003	3 OF 4	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1003	4 OF 4	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED FOR FABRICABILITY ENHANCEMENTS & EDITORIAL CORRECTIONS	04/07/10
0	FIRST ISSUE	04/09/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®37PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
ALTERNATE TOP CLOSURE

DRAWING NO.	SCALE	SHEET
NUH37PTH-71-1004	NONE	1 OF 6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1004	2 OF 6	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1004	3 OF 6	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH37PTH-71-1004 SHEET 4 OF 6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1004	5 OF 6	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1004	6 OF 6	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

1	REVISED FOR FABRICABILITY ENHANCEMENTS & EDITORIAL CORRECTIONS	04/07/10
0	FIRST ISSUE	03/25/09
REVISION	DESCRIPTION	DATE

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®37PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
BASKET ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH37PTH-71-1011	NONE	1 OF 7

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH37PTH-71-1011 | SHEET 2 OF 7 | REVISION 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1011	3 OF 7	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1011	5 OF 7	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1011	7 OF 7	1

NUH37PTH-71-1012 | 1 OF 7

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED FOR FABRICABILITY ENHANCEMENTS & EDITORIAL CORRECTIONS	04/07/10
0	FIRST ISSUE	04/09/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®37PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
TRANSITION RAILS

DRAWING NO.	SCALE	SHEET
NUH37PTH-71-1012	NONE	1 OF 7

DRAWING NO.	SHEET
NUH37PTH-71-1012	2 OF 7

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1012	2 OF 7	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1012	3 OF 7	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

8 | 7 | 6 | 5 | 4 | 3 | 2 | 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1012	5 OF 7	1

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-71-1012	6 OF 7	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH37PTH-71-1012 | SHEET 7 OF 7 | REVISION 1

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
0	FIRST ISSUE PER NRC RAI #1 2-3	04/07/10

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®37PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
DAMAGED FUEL END CAPS

DRAWING NO.	SCALE	SHEET
NUH37PTH-71-1015	NONE	1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REMOVE NON REQUIRED PT	3/23/10
0	FIRST ISSUE	3/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 61BT
TRANSPORTABLE CANISTER FOR BWR FUEL
BASKET ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH61BT-71-1001	NONE	1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER NRC RAI #1 2-33	04/02/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 61BTH TYPE 2
TRANSPORTABLE CANISTER FOR BWR FUEL
MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH61BTH-71-1100	NONE	1 OF 7

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

NUH61BTH-71-1100 2 OF 7

DRAWING NO.	SHEET	REVISION
NUH61BTH-71-1100	2 OF 7	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH61BTH-71-1100	3 OF 7	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH61BTH-71-1100	5 OF 7	1

NUH61BTH-71-1100 6 OF 7

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH61BTH-71-1100	6 OF 7	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH61BTH-71-1100	7 OF 7	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REMOVE NON REQUIRED PT	04/12/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 61BTH TYPE 2
TRANSPORTABLE CANISTER FOR BWR FUEL
BASKET ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH61BTH-71-1102	NONE	1 OF 8

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH61BTH-71-1102	2 OF 8	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH61BTH-71-1102 SHEET 3 OF 8

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH61BTH-71-1102	4 OF 8	1

NUH61BTH-71-1102 5 OF 8

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH61BTH-71-1102	5 OF 8	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH61BTH-71-1102 SHEET 6 OF 8 REVISION 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH61BTH-71-1102	7 OF 8	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH61BTH-71-1102	8 OF 8	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	INCLUDE FABRICABILITY ENHANCEMENTS	04/02/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 61BTH TYPE 2
TRANSPORTABLE CANISTER FOR BWR FUEL
TOP GRID ASSEMBLY ALTERNATE 3

DRAWING NO. NUH61BTH-71-1106 | SCALE NONE | SHEET 1 OF 2

NUH61BTH-71-1106 2 OF 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH61BTH-71-1106	2 OF 2	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER NRC RAI #1 2-33, FABRICABILITY ENHANCEMENTS & EDITORIAL CORRECTIONS	04/14/10
0	FIRST ISSUE	04/07/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 69BTH
TRANSPORTABLE CANISTER FOR BWR FUEL
MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH69BTH-71-1001	NONE	1 OF 4

DRAWING NO. NUH69BTH-71-1001 SHEET 2 OF 4

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1001	2 OF 4	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

8 7 6 5 4 3 2 1

H G F E D C B A

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1001	4 OF 4	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED FOR FABRICABILITY ENHANCEMENTS & EDITORIAL CORRECTIONS	04/14/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®69BTH
TRANSPORTABLE CANISTER FOR BWR FUEL
BASKET-SHELL ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH69BTH-71-1002	NONE	1 OF 4

8 7 6 5 4 3 2 1

DRAWING NO. NUH69BTH-71-1002 SHEET 2 OF 4

H G F E D C B A

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1002	2 OF 4	1

8 7 6 5 4 3 2 1

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH69BTH-71-1002 | SHEET 3 OF 4 | REVISION 1

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1002	4 OF 4	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED FOR FABRICABILITY ENHANCEMENTS	04/14/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®69BTH
TRANSPORTABLE CANISTER FOR BWR FUEL
SHELL ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH69BTH-71-1003	NONE	1 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1003	2 OF 4	1

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1003	3 OF 4	1

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH698TH-71-1003	4 OF 4	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED PER NRC RAI #1 2-33, FOR FABRICABILITY ENHANCEMENTS & EDITORIAL CORRECTIONS	04/14/10
0	FIRST ISSUE	04/10/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 69BTH
TRANSPORTABLE CANISTER FOR BWR FUEL
ALTERNATE 2 TOP CLOSURE

DRAWING NO.	SCALE	SHEET
NUH69BTH-71-1004	NONE	1 OF 6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1004	2 OF 6	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1004	3 OF 6	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1004	4 OF 6	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1004	5 OF 6	1

8 | 7 | 6 | 5 | 4 | 3 | 2 | 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1004	6 OF 6	1

8 | 7 | 6 | 5 | 4 | 3 | 2 | 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

1	REVISED FOR FABRICABILITY ENHANCEMENTS AND EDITORIAL CORRECTIONS	04/14/10
0	FIRST ISSUE	04/01/09
REVISION	DESCRIPTION	DATE

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 69BTH
TRANSPORTABLE CANISTER FOR BWR FUEL
BASKET ASSEMBLY

DRAWING NO. NUH69BTH-71-1011 | SCALE NONE | SHEET 1 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1011	2 OF 5	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1011	3 OF 5	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1011	4 OF 5	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1011	5 OF 5	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

1	REVISED FOR FABRICABILITY ENHANCEMENTS AND EDITORIAL CORRECTIONS	4/14/10
0	FIRST ISSUE	3/26/10
REVISION	DESCRIPTION	DATE

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®69BTH
TRANSPORTABLE CANISTER FOR BWR FUEL
TRANSITION RAIL ASSEMBLY AND DETAILS

DRAWING NO. NUH69BTH-71-1012 | SCALE NONE | SHEET 1 OF 6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO. NUH69BTH-71-1012 | SHEET 2 OF 6 | REVISION 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO	SHEET	REVISION
NUH69BTH-71-1012	5 OF 6	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1012	6 OF 6	1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED FOR FABRICABILITY ENHANCEMENTS	04/14/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®69BTH
TRANSPORTABLE CANISTER FOR BWR FUEL
HOLDDOWN RING ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH69BTH-71-1013	NONE	1 OF 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH69BTH-71-1013	2 OF 2	1

DRAWING NO. NUH69BTH-71-1013 SHEET 2 OF 2

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

1	REVISED FOR FABRICABILITY ENHANCEMENTS	04/14/10
0	FIRST ISSUE	03/26/09
REVISION	DESCRIPTION	DATE

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 69BTH
TRANSPORTABLE CANISTER FOR BWR FUEL
DAMAGED FUEL MODIFICATION

DRAWING NO.	SCALE	SHEET
NUH69BTH-71-1014	NONE	1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
1	REVISED FOR FABRICABILITY ENHANCEMENTS	04/14/10
0	FIRST ISSUE	03/26/09

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. Patent No. 4,780,269
Transnuclear, Inc.

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®69BTH
TRANSPORTABLE CANISTER FOR BWR FUEL
DAMAGED FUEL END CAPS

DRAWING NO. NUH69BTH-71-1015 | SCALE NONE | SHEET 1 OF 1

A.1.4.10.11 Radioactive Waste Canister Drawing

This section contains drawings for the Radioactive Waste Canister.

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
0	FIRST ISSUE PER RAI #1 2-6	04/07/10

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® SYSTEM
RWC CANISTER-WELDED TOP SHIELD PLUG DESIGN
MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUHRWC-71-1001	NONE	1 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

8 7 6 5 4 3 2 1

H G F E D C B A

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUHRWC-71-1001	3 OF 5	0

NUHRWC-71-1001 4 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUHRWC-71-1001	4 OF 5	0

NUHRWC-71-1001 5 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUHRWC-71-1001	5 OF 5	0

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
0	FIRST ISSUE PER RAI #1 2-6	04/07/10

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269
TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® SYSTEM
RWC CANISTER-WELDED TOP SHIELD PLUG DESIGN
INNER LINER

DRAWING NO. NUHRWC-71-1002 | SCALE NONE | SHEET 1 OF 3

NUHRWC-71-1002 1 OF 3

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUHRWC-71-1002	2 OF 3	0

PROPRIETARY AND
SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUHRWC-71-1002	3 OF 3	0

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
0	FIRST ISSUE PER RAI 1# 2-6	04/07/10

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ANSI Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269 TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® SYSTEM
RWC CANISTER-BOLTED TOP SHIELD PLUG DESIGN
MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUHRWC-71-1003	NONE	1 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

8 7 6 5 4 3 2 1

DRAWING NO. NUHRWC-71-1003 SHEET 3 OF 4

H G F E D C B A

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

H G F E D C B A

DRAWING NO.	SHEET	REVISION
NUHRWC-71-1003	3 OF 4	0

8 7 6 5 4 3 2 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUHRWC-71-1003	4 OF 4	0

assembly is provided by four support rods. The support rods extend over the full length of the cavity with allowance provided for thermal growth of the support rods in the axial direction.

The basket structure is open at each end. Therefore, longitudinal fuel assembly loads are applied directly to the DSC end plates and not to the fuel basket structure. The fuel assemblies are laterally supported in the fuel compartments, and the basket is laterally supported by the support rails and the DSC shell.

A.2.1.1.4 Radioactive Waste *Canister*

A radioactive waste *canister* (RWC) is also included as an authorized payload of the MP197HB cask. As described in Chapter A.1, *Appendix* A.1.*4.9A*, the RWC is bounded by the DSCs in terms of weight and decay heat. Also, the RWC does not provide containment for its contents. Therefore, the analyses provided in this Chapter for the DSCs are considered bounding for the RWC and no analyses for the RWC are necessary.

A.2.1.2 Design Criteria

The packaging consists of the following major components:

- Cask Body
- Impact Limiters
- Payload (DSC or RWC)

The structural design criteria for these components are described below.

A.2.1.2.1 Basic Design Criteria

Cask Containment Vessel

The containment vessel consists of the inner shell including the flange inside of the lid inner O-ring, the bottom, ram access closure plate, and the lid. The lid and ram access closure plate bolts and seals are also part of the containment vessel as are the drain and vent port plug bolts and seals. The containment vessel is designed to the maximum practical extent as an ASME Class I component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [3]. The Subsection NB rules for materials, design, fabrication and examination are applied to all of the above components to the maximum practical extent. In addition, the design meets the requirements of Regulatory Guides 7.6 [5] and 7.8 [6]. Alternatives to the ASME Code are discussed in Section A.2.1.4 and Appendix A.2.13.13 of this Chapter.

The acceptability of the containment vessel under the applied loads is based on the following criteria:

- Title 10, Chapter 1, Code of Federal Regulations, Part 71
- Regulatory Guide 7.6 Design Criteria
- ASME Code Design Stress Intensities
- Preclusion of Fatigue Failure
- Preclusion of Brittle Fracture

The stresses due to each load are categorized as to the type of stress induced, such as membrane or bending, and the classification of stress, such as primary or secondary. Stress limits for containment vessel components, other than bolts, for NCT (ASME Level A) and HAC (ASME Level D) are given in Table A.2-1.

The primary membrane stress and primary membrane plus bending stress are limited to S_m (S_m is the code allowable stress intensity) and 1.5 S_m, respectively, at any location in the cask for NCT (ASME Level A).

The HAC events are evaluated as short duration, Level D conditions. The stress criteria are taken from Section III, Appendix F of ASME Code [3]. For elastic quasi-static analysis, the primary membrane stress intensity (P_m) is limited to the smaller of the 2.4 S_m or 0.7 S_u, and membrane plus bending stress intensities ($P_m + P_b$) are limited to the smaller of the 3.6 S_m or S_u. For the elastic-plastic analysis, the primary membrane stress intensity (P_m) is limited to 0.7 S_u, and membrane plus bending stress intensities ($P_m + P_b$) are limited to 0.9 S_u.

The allowable stress limits for the containment bolts are listed in Table A.2-2.

The allowable stress intensity value, S_m, as defined by the Code, is taken at the maximum temperature calculated for each service load condition.

Cask Non-Containment Structure

Certain components such as the outer shell, the shield shell and the trunnions are not part of the cask containment vessel but do have structural functions. These components, referred to as non-containment structures, are required to withstand the containment environmental loads, and in some cases share the loads with the containment vessel. The outer shell stress limits are the same as those given in Table A.2-1 for the containment structure. The neutron shield shell is designed, fabricated and inspected in accordance with the ASME Code Subsection NF [3], to the maximum practical extent. Structural and structural attachment welds are examined by the *PT or MT* method, in accordance with Section V, Article 6 of the ASME Code [8]. The *PT or MT* examination acceptance standards are in accordance with Section III, Subsection NF, Paragraph NF-5350 *or NF-5340* [3].

Seal welds are examined visually, or by *PT or MT* method, in accordance with Section V of the ASME Code [8]. Electrodes, wire, and fluxes used for fabrication comply with the applicable requirements of the ASME Code, Section II, Part C [9].

The welding procedures, welders and weld operators are qualified in accordance with Section IX of the ASME Code [10].

The radial neutron shield, including the *carbon* steel enclosure, has not been designed to withstand all of the HAC loads. The shielding may degrade during the fire or due to the 40 inch drop onto the puncture bar. Therefore a bounding shielding analysis, assuming that up to 75% of the exterior neutron shielding is removed, has been performed. This analysis shows that the accident dose rates are not exceeded. These accident shielding analyses are described in Chapter A.5.

Dry Shielded Canister (DSC)

The NUHOMS® MP197HB is designed to carry several different DSCs. Some of these are currently licensed under 10CFR72 for storage of spent nuclear fuel. A table of DSCs with the licensing information is provided below.

DSC Design	Applicable Storage License CoC	ASME B&PV Code Year
NUHOMS® 32PTH	1030	1998 w/ 2000 Addenda
NUHOMS® 32PTH1	1004	1998 w/ 2000 Addenda
NUHOMS® 37PTH	Note (1)	2004 w/2006 Addenda
NUHOMS® 69BTH	Note (1)	2004 w/2006 Addenda
NUHOMS® 24PT4	1029	1992 thru1994 Addenda
NUHOMS® 24PTH	1004	1998 w/ 2000 Addenda
NUHOMS® 32PT	1004	1998 w/ 2000 Addenda
NUHOMS® 61BT	1004	1998 w/ 1999 Addenda
NUHOMS® 61BTH	1004	1998 w/ 2000 Addenda
NUHOMS® 61BTH *with failed fuel, (61BTHF)*	Note (1)	2004 w/2006 Addenda
NUHOMS® 24PTH *with failed fuel, (24PTHF)*	Note (1)	2004 w/2006 Addenda

Note: (1) These DSCs are currently not a part of CoC 1004 but will be added at a later date via amendment.

DSC Shell Assembly

The components of each DSC including the shell assembly, the top outer/inner cover plates, the inner bottom cover plate, the siphon vent block, and the siphon/vent port cover plate are designed, fabricated and inspected in accordance with the ASME Code Subsection NB to the maximum practical extent (Code alternatives are given in Section A.2.1.4 and Appendix A.2.13.13 below). Note that the applicable Code year varies depending on the storage license of the specific DSC. The basis for the allowable stresses is Article NB-3200 for NCT (Level A) loads, and Appendix F for HAC (Level D) loads. Stress limits for NCT (Level A) and HAC (Level D) are given in Table A.2-1. When evaluating the results from the non-linear elastic-plastic analysis for the accident conditions, the general primary membrane stress intensity, P_m, shall not exceed 0.7 S_u and the maximum stress intensity at any location ($P_m + P_b$) shall not exceed 0.9 S_u.

DSC Basket

The baskets for all of the DSCs are designed, fabricated and inspected in accordance with the ASME Code Subsection NG to the maximum practical extent with the applicable Code year again depending on the storage license of the applicable DSC. (Code alternatives are given in Section A.2.1.4 and Appendix A.2.13.13 below).

The basket is designed to meet the heat transfer, nuclear criticality, and the structural requirements. The basket structure must provide sufficient rigidity to maintain a subcritical configuration under all applied loads. The 304 stainless steel members in DSC baskets are the primary structural components. The neutron poison plates are the primary heat conductors, and provide the necessary criticality control.

A.2.1.4 Identification of Codes and Standards for Package Design

The cask containment boundary and the shell, top outer/inner plates, inner bottom cover plate, siphon vent block, and siphon/vent port cover plate of the DSC are designed, fabricated and inspected in accordance with the ASME Code Subsection NB to the maximum practical extent. The basket is designed, fabricated and inspected in accordance with ASME Code Subsection NG to the maximum practical extent. Other cask components (such as the shield shell and neutron shielding) and DSC components (such as outer bottom cover, top and bottom shield plugs) are not governed by the ASME Code. The ASME code alternatives for cask, canister, and basket are specified in Appendix A.2.13.13.

A.2.2 Materials

A.2.2.1 Material Properties and Specifications

A.2.2.1.1 Cask Material Properties

This section provides the mechanical properties of materials used in the structural evaluation of the NUHOMS®-MP197HB cask. Table A.2-4 lists the materials selected, the applicable components, and the minimum yield, ultimate, and design stress values specified by the ASME Code, Section II, Part D [9]. *Mechanical properties of lead used for gamma shielding are listed in Table A.2-5.*

A.2.2.1.2 DSC Material Properties

The material properties of the stainless steel used in the DSCs are taken from the ASME Code. However, the various DSC designs have been designed and fabricated to several different ASME code year editions. The specific DSCs and applicable ASME code year are given in Section A.2.1.2. The material properties are listed with specific references in Appendices A.2.13.7 and A.2.13.8.

A.2.2.1.3 Impact Limiter Material Properties

Mechanical properties of the energy absorbing wood used in the impact limiters are specified in Appendix A.2.13.12.

A.2.2.2 Chemical, Galvanic, or Other Reactions

The materials of the NUHOMS®-MP197HB cask have been reviewed to determine whether chemical, galvanic or other reactions among the materials, contents and environment might occur during any phase of loading, unloading, handling or transport.

- The materials from which NUHOMS®-MP197HB transportation packaging is fabricated will not experience significant chemical, galvanic, or other reactions in air, helium, or water environments. The exterior of the cask is carbon steel that with the exception of the trunnion bearing surfaces and surfaces contact to the impact limiters is painted using an epoxy, acrylic urethane, or equivalent enamel coating. The paint is selected to be

A.2.6.14 Summary of Normal Condition Cask Body Structural Analysis

The following table lists the highest stress ratio in each cask component and also identifies the load combination tables where these stress intensities are given in Appendix A.2.13.1. The stress limits based on the Section A.2.1.2 structural design criteria are also listed in the table.

Comparison of the Maximum Stress Intensities with Allowables

Component	**Calculated Stress Intensity (ksi)**	**Allowable Stress Intensity (ksi)**	**Stress Ratio**	**Stress Result Table**
Outer Shell	28.2	$P_l + P_b = 34.7$	81.4%	A.2.13.1-31
Inner Shell	14.5	$P_m = 23.1$	62.8%	A.2.13.1-30
Lid	21.6	$P_l + P_b = 34.7$	62.3%	A.2.13.1-29
Top Flange	27.6	$P_l + P_b = 32.6$	84.7%	A.2.13.1-30
Bottom Flange	37.9	$P + Q = 65.3$	58.1%	A.2.13.1-30
Bottom Plate	29.2	$P + Q = 65.3$	44.7%	A.2.13.1-24
RAM Closure Plate	20.4	$P + Q = 69.3$	29.4%	A.2.13.1-27

The stress values in the RAM closure plate shown in the above table were calculated for the case when plate is made of carbon steel. If stainless steel is used to fabricate RAM closure plate then the increase in primary stresses would be insignificant due to the small, below 2%, difference in Young's modulus and the primary stress will remain well below the stress limit of the SA-240 Type 304 stainless. The thermal stresses also would remain unchanged due to diametric clearance of 0.06" between RAM closure plate OD and Bottom forging ID, and 0.06" clearance of bolt holes.

From the analysis results presented in the above table, it can be shown that the NCT loads will not result in any structural damage to the cask and that the containment function of the cask will be maintained.

A.2.6.15 Structural Evaluation of the Basket/Canister Shell under Normal Condition Loads

A.2.6.15.1 Basket Stress Analysis

The loading conditions considered in the evaluation of the fuel basket consist of inertial loads resulting from normal inertial loading (1foot drop) and thermal loads. The inertial loads of significance for the basket analysis are those transverse to the cask and basket longitudinal axes, so that the loading from the fuel assemblies is applied normal to the basket plates and transferred to the cask wall by the basket.

The structural adequacy of the basket plates in the NUHOMS® basket assembly under a NCT free drop is analyzed in Appendix A.2.13.8. The baseline g loads and drop orientations used for the structural analysis of the basket are described in Appendix A.2.13.12. The baseline g loads are multiplied by the dynamic load factor calculated in Appendix A.2.13.9. The maximum g loads used for the basket structural evaluations are summarized in Table A.2.13.8-1 of Appendix A.2.13.8. The stress analysis of the basket due to inertial and thermal loads is also described in detail in Appendix A.2.13.8. Based on the results of analyses shown in Appendix A.2.13.8, all the basket designs meet the ASME Code Subsection NG requirements [3]. Therefore, the basket is structurally adequate and it will properly support and position the fuel assemblies under normal loading conditions.

A.2.6.15.2 DSC Shell Stress Analysis

The MP197HB DSC shell assemblies each consist of a cylindrical shell, top outer/inner cover plates, bottom inner/outer cover plates and bottom and top shield plugs. Each DSC shell assembly functions to support a basket assembly and confine associated fuel assemblies that are contained within the DSC shell assembly. The confinement vessel for each of the MP197HB DSCs consists of a shell which is a welded, stainless steel cylinder with a stainless steel bottom closure assembly, and a stainless steel top closure assembly. Additional details, geometry, and shell and plate thicknesses are provided in Appendix A.1.4.10.

Multiple DSC shell assembly designs are evaluated in Appendix A.2.13.7. Each design is categorized into one of four groups based on similarity of geometry, plate thicknesses and compartment payload. For each group, the bounding payload weight (basket plus fuel assembly) is used for the analyses.

Finite element analyses are performed in order to quantify stresses in the DSCs generated by transport loads. The applied loads considered are top end, bottom end, and side drops combined with internal and external pressures and temperature distributions (thermal expansion stresses). Several three-dimensional finite element models are used to evaluate stresses for the normal and accident loads: 180° 3D models are used for side drop analyses; 2D axisymmetric models are used for end drop and thermal expansion analyses.

The loading conditions considered in the evaluation of the DSC shell consist of inertial loads resulting from normal condition inertial loading (1foot drop), internal /external pressures and thermal loads. The inertial loads of significance for the DSC shell analysis are those transverse to the cask and DSC shell longitudinal axes, so that the loadings from the fuel assemblies and basket are transferred to the cask wall by the DSC shell.

The baseline g loads and drop orientations used for structural analysis of the DSCs are described in Appendix A.2.13.12. The baseline g loads are multiplied by the dynamic load factor calculated in Appendix A.2.13.9. The maximum g loads used for the DSC shell assembly structural evaluations are summarized in Section A.2.13.7.3 of Appendix A.2.13.7. The stress analysis of the DSC shell assembly due to inertial and thermal loads is also described in detail in Appendix A.2.13.7. Based on the results of analyses shown in Appendix A.2.13.7, all the DSC shell assembly designs meet the ASME Code Subsection NB requirements [3].

A nonlinear finite element analysis is performed in order to evaluate the buckling capacity of the inner shell of the NUHOMS®-MP197HB cask. A 2-dimensional axisymmetric ANSYS [7] finite element model is constructed for this purpose. The results of the finite element analysis provide both stresses and displacements generated during the end drop event. The allowable buckling load is 215g and the maximum lead slump is 0.32 inches. The detailed analysis is provided in Appendix A.2.13.3.

A.2.7.2 Crush

This test does not apply to the NUHOMS®-MP197HB Package since the package weight is in excess of 5,000 kg (11,000 lb).

A.2.7.3 Puncture

An evaluation of the puncture drop as specified by 10CFR71.73(c)(3) is presented below. This is considered to be the worst case scenario of the package dropped from a distance of 40 in. onto a vertical puncture bar. The specified puncture bar is a 6 in. diameter solid, vertical, cylindrical, mild steel bar. The impact limiters will protect the ends of the cask body during this event. Consequently, the most severe damage to the cask body, resulting from the puncture drop, will occur on the outer cylindrical shell midway between the impact limiters. For this load condition it is conservatively assumed that the cask outer shell surface impacts the puncture bar directly.

Required Thickness

The required thickness t_{req} to preclude puncture is calculated using the Nelms equation for lead backed shells [4], which is given by:

$$t_{req} = \left[\frac{W}{S_u}\right]^{0.71}$$

Where W is the weight of the package (conservatively taken equal to 310,000 lb) *and the material property of S_u is taken at 352 °F*:

$$t_{req} = \left[\frac{310{,}000}{70{,}000}\right]^{0.71} = 2.88\,\text{in}.$$

The thickness of the outer shell is 2.75 in, and the thickness of the neutron shield shell is equal to 0.375 in. The total steel thickness is therefore equal to 2.75 + 0.375 = 3.125 in, which is greater than the required thickness computed above. Therefore, the outer shell combined with the neutron shield shell will preclude penetration of the bar during the postulated puncture event.

A.2.7.6 Immersion–All Packages

The immersion loading condition results in an external pressure applied to the cask body corresponding to a 50 foot head of water (21.7 psig). Assuming a 0 psia cask cavity pressure, the resulting maximum external pressure loading is 36.4 psi (21.7 + 14.7). The cask body stresses for this immersion condition (36.4 psi external pressure) is enveloped by the deep water immersion condition (water pressure of 290 psi) described in Section A.2.7.7 below.

A.2.7.7 Deep Water Immersion Test

10CFR 71.61 requires that the package be subjected to an external water pressure of 290 psig for a period of not less than one hour without collapse, buckling, or inleakage of water. The load combination performed to evaluate this event is included in Table A.2-9.

Table A.2.13.1-46 of Appendix A.2.13.1 lists the combined stress intensities for this accident event.

A.2.7.8 Summary of Damage

A.2.7.8.1 Summary of Accident Condition Cask Body Structural Analysis

The following table lists the highest stress ratio in the cask components and also identifies the load combination tables from Appendix A.2.13.1 where these stresses are shown. Also listed in the tables are the stress limits based on the Section A.2.1.2 structural design criteria.

Comparison of the Maximum Stress Intensities with the Allowables (Elastic-Plastic Analysis)

Component	**Maximum Stress Intensity (ksi)**	**Allowable Stress Intensity (ksi)**	**Stress Ratio**	**Stress Result Table**
Outer Shell	40.5	$P_m = 49.0$	82.6%	A.2.13.1-43
Inner Shell	41.6	$P_m = 49.0$	84.9%	A.2.13.1-44
Lid	32.9	$P_l + P_b = 63.0$	52.2%	A.2.13.1-45
Top Flange	38.8	$P_m = 49.0$	79.2%	A.2.13.1-44
Bottom Flange	35.9	$P_m = 49.0$	73.4%	A.2.13.1-43
Bottom Plate	30.5	$P_l = 63.0$	48.4%	A.2.13.1-38
RAM Closure Plate	17.6	$P_m = 63.0$	28.0%	A.2.13.1-43

The stress values in the RAM closure plate shown in the above table were calculated for the case when plate is made of carbon steel. If stainless steel is used to fabricate RAM closure plate then the increase in primary stresses would be insignificant due to the small, below 2%, difference in Young's modulus and the primary stress will remain well below the stress limit of the SA-240 Type 304 stainless.

From the analysis results presented in the above table, it can be shown that the accident loads will not result in any structural damage to the cask.

A.2.7.8.1.1 Basket Stress Analysis

The structural adequacy of the basket plates in the NUHOMS® DSC fuel basket under HAC free end and side drop load cases are performed in Appendix A.2.13.8. Based on the results of analyses shown in Appendix A.2.13.8, all the basket designs meet the ASME Code Subsection NG and Appendix F requirements [3]. The basket is structurally adequate and will properly support and position the fuel assemblies under accident loading conditions.

A.2.7.8.1.2 DSC Shell Stress Analysis

The g loads and drop orientations used for the structural analysis of the *DSC Shell* are described in Appendix A.2.13.7. Based on the results of analyses shown in Appendix A.2.13.7, all the DSC shell assembly designs meet the ASME Code Subsection NB and Appendix F requirements [3].

A.2.8 Accident Conditions for Air Transport of Plutonium

This section does not apply to the NUHOMS®-MP197HB Packaging because the package will not be transported by air.

A.2.9 Accident Conditions for Fissile Material Packages for Air Transport

This section does not apply to the NUHOMS®-MP197HB Packaging because the package will not be transported by air.

A.2.10 Special Form

This section does not apply to the NUHOMS®-MP197HB Packaging because the payloads are not considered to be special form.

A.2.11 Fuel Rods

As discussed in Chapter A.4, containment of the radioactive material is provided by the cask containment boundary. Analyses of the cask boundary for 10CFR71 NCT and HAC requirements demonstrate that the cask remains leak tight.

The structural adequacy of the fuel rod in the NUHOMS® DSC fuel basket under HAC free end and side drop load cases are performed in Appendix A.2.13.11. Based on the results of analyses shown in Appendix A.2.13.11, the integrity of the fuel rod will not be breached during the normal and hypothetical accident loads.

A.2.12 References

1. 10 CFR PART 71, Packaging and Transportation of Radioactive Material.
2. American National Standards Institute, ANSI N14.6, American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials, 1993.
3. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, and Appendices, 2004 including 2006 addenda.
4. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 3, Subsection WB, 2004 including 2006 addenda.
5. USNRC, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," Regulatory Guide 7.6, Revision 1, March 1978.
6. USNRC, "Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material," Regulatory Guide 7.8, Revision 1, March 1989.
7. ANSYS Release 10.0A1.
8. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section V, 2004 including 2006 addenda.
9. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section II, Part C and Part D, 2004 including 2006 addenda.
10. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section IX, 2004 including 2006 addenda.
11. NUREG-0933, Generic Issue 196, "Boral Degradation."
12. Larrabee, C. P. and Coburn, S. K. "The Atmospheric Corrosion of Steels as Influenced by Changes in Chemical Composition," p 276-285, Proc. First Int. Congress on Metallic Corrosion, Butterworths, London, 1962.
13. WRC Bulletin 107, March 1979 Revision "Local Stresses in Spherical and Cylindrical Shells Due to External Loadings."
14. ANSI N14.23, "Draft American National Standard Design Basis for Resistance to Shock and Vibration of Radioactive Material Packages Greater Than One Ton in Truck Transport," May, 1980.
15. USNRC, "Shock and Vibration Environments for Large Shipping Containers on Rail Cars and Trucks," NUREG 766510, June, 1977.
16. USNRC, "Stress Analysis of Closure Bolts for Shipping Casks," NUREG/CR-6007 Lawrence Livermore National Laboratory, 1992.
17. *Tietz, T. E., "Determination of the Mechanical Properties of a High Purity Lead and a 0.058 % Copper-Lead Alloy," WADC Technical Report 57-695, ASTIA Document No. 151165, Stanford Research Institute, Menlo Park, CA, April 1958.*
18. *H.J. Rack, G.A. Knorovsky, "An Assessment of Stress-Strain Data Suitable for Finite-Element Elastic-Plastic Analysis of Shipping Containers," Sandia Laboratories, NUREG/CR-0481, SAND77-1872 R-7, 1978.*
19. *R.A. Robinson, et. al., " A survey of Strain-Rate Effects for some common Structural Materials used in Radioactive Material Packaging and Transportation Systems", Report BMI-1954, August 1976, Battelle Columbus Laboratories.*

Table A.2-4
Cask Material Properties

Material	Class	Temp. (°F)	S_y (ksi)	S_u (ksi)	S_m (ksi)	E (10^{-6} psi)	$\alpha x10^{-6}$ (in/in/°F)
SA-540, Gr. B23, Cl 1 *and Gr. B24, Cl 1*[(1)] (Bolt)	Sect III Class 1	70	150.0	165.0	50.0	27.8	6.4
		200	144.0	165.0	47.8	27.1	6.7
		300	140.3	165.0	46.2	26.7	6.9
		400	137.9	165.0	44.8	26.2	7.1
		500	136.0	165.0	43.4	25.7	7.3
		600	133.4	165.0	41.4	25.1	7.4
		700	129.0	158.6	---	24.6	7.6
SA-240, Type 304	Sect III Class 1	70	30.0	75.0	20.0	28.3	8.5
		200	25.0	71.0	20.0	27.5	8.9
		300	22.4	66.2	20.0	27.0	9.2
		400	20.7	64.0	18.6	26.4	9.5
		500	19.4	63.4	17.5	25.9	9.7
		600	18.4	63.4	16.6	25.3	9.8
		700	17.6	63.4	15.8	24.8	10.0
SA-182, Type F304 >5"	Sect III Class 1	70	30.0	70.0	20.0	28.3	8.5
		200	25.0	66.3	20.0	27.5	8.9
		300	22.4	61.8	20.0	27.0	9.2
		400	20.7	59.7	18.6	26.4	9.5
		500	19.4	59.2	17.5	25.9	9.7
		600	18.4	59.2	16.6	25.3	9.8
		700	17.6	59.2	15.8	24.8	10.0
SA-182, Type F304 ≤5"	Sect III Class 1	70	30.0	75.0	20.0	28.3	8.5
		200	25.0	71.0	20.0	27.5	8.9
		300	22.4	66.2	20.0	27.0	9.2
		400	20.7	64.0	18.6	26.4	9.5
		500	19.4	63.4	17.5	25.9	9.7
		600	18.4	63.4	16.6	25.3	9.8
		700	17.6	63.4	15.8	24.8	10.0
SA-182, Type F304N	Sect III Class 1	70	35.0	80.	23.3	28.3	8.5
		200	28.6	80	23.3	27.6	8.9
		300	25.0	76.1	22.5	27.0	9.2
		400	22.6	73.2	20.3	26.5	9.5
		500	21.0	71.2	18.9	25.8	9.7
		600	19.9	69.7	17.9	25.3	9.8
		700	19.1	68.6	17.2	24.8	10.0

(1) SA-540 Gr. B24 Cl 1 bolt material is specified for the 1/3 scale benchmark analyses.

Table A.2-5
Mechanical Lead Properties

Static Mechanical Lead Properties

Temp. (°F)	Static Stress Properties (ksi) *[17]*			E *[18]* (10^6 psi)	Coef. of Thermal Exp *[18]* (10^{-6} in/in/°F)
	Yield (S_Y)		Ultimate (S_U)		
	Tension	Compression	Tension		
-99	-	-	-	2.50	15.28
70	-	-	-	2.34	16.07
100	0.584	0.490	1.570	2.30	16.21
175	0.509	0.428	1.162	2.20	16.58
250	0.498	0.391	0.844	2.09	16.95
325	0.311	0.320	0.642	1.96	17.54
440	-	-	-	1.74	18.50
620	-	-	-	1.36	20.39

Dynamic Stress-Strain Lead Properties

Strain (in/in)	Stress (ksi) *[19]*				
	At 100 °F	At 230 °F	At 300 °F	At 350 °F	At 500 °F
0.00048	1.14	1.06	1.00	0.97	0.86
0.03	2.2	2.0	1.7	1.5	1.1
0.1	3.3	2.8	2.38	2.1	1.26
0.3	4.9	3.2	2.72	2.4	1.44
0.5	5.6	3.6	3.06	2.7	1.62

B. ANSYS Models–Description

Four 3D ANSYS finite element models were constructed to represent the MP197HB cask body in the analyses. Due to symmetry considerations, all four models consist of 180° representations of the cask with symmetry boundary conditions. Figure A.2.13.1-2 shows the basic structural components of the MP197HB cask used in the development of the finite element models. The main features of each of the four models are summarized below.

Model MOD20	**Model MOD20S**	**Model MOD20SP**	**Model MOD20TI**
Used in HAC Evaluations	Used in NCT Evaluations	Used in NCT Evaluations	Used in NCT Evaluations
All HAC cases	Side drop cases End drop cases 3 g lifting case	Bolt preload case	Rail car shock cases Rail car vibration cases 1g gravity case
Includes main structural components	Includes main structural components	Includes main structural components	Added saddles, straps, impact limiter steel blocks, neutron shield shell components
Nonlinear spring model of bolts in tensile and shear directions	Nonlinear spring model of bolts in tensile and shear directions	LINK8 model with imposed preload in tensile direction Nonlinear spring model in shear direction	Nonlinear spring model of bolts in tensile and shear directions
Linear elastic material properties for steel components for elastic analysis. Bilinear kinematic hardening model for steel components, with 5% tangent modulus for elastic-plastic analysis	Linear elastic material properties for steel components for elastic analysis.	Linear elastic material properties for steel components for elastic analysis.	Linear elastic material properties for steel components for elastic analysis.
Multi-linear kinematic hardening model for lead component	Bilinear kinematic hardening model for lead component, with 1% tangent modulus	Bilinear kinematic hardening model for lead component, with 1% tangent modulus	Bilinear kinematic hardening model for lead component, with 1% tangent modulus

All four models use ANSYS structural solid elements SOLID45 for modeling the cask structural components. The interfaces between the steel and lead gamma shielding, between lid and the top flange, and between RAM closure plate and bottom plate are modeled with surface to surface contact elements CONTA173 and TARGE170.

The analyses for NCT cases are based on linear elastic material properties for the steel components. The analyses for HAC cases are based on elastic material properties for the elastic analyses and elastic-plastic material properties for the non-linear (elastic-plastic) analyses. For the elastic-plastic analyses, a bilinear kinematic hardening material law is used, assuming a tangent modulus of 5% of the material elastic modulus. Lead material *properties from Table A.2-5 in Chapter A.2 are used.*

The finite element model (MOD20) used for the HAC accident drop analyses is shown on Figures A.2.13.1-3 and A.2.13.1-4.

A separate model, (MOD20TI), is developed to analyze the cask for shock, vibration and gravity loadings while in the transport configuration. In the transport configuration, the cask is oriented horizontally and secured to the transport skid at the bearing block in the longitudinal direction, supported at two saddles and held by tie down straps.

Drop Orientation		g Loads *Including -40 °F Effect (App. A.2.13.12)*	g Loads used in cask body stress calculation
30 feet end drop		*52* g	55
30 feet side drop		*55* g	55
30 feet CG over corner drop		*39* g	45
30 feet slap down (10°)	1st impact	*23* g	Translation =45 g Rotational α = 551 rad/sec^2 to bound both 1st and 2nd impacts
		α =*122* rad/sec^2	
	2nd impact	*31*	
		α =*170* rad/sec^2	
30 feet slap down (20°)	1st impact	*18*	Translation =45 g Rotational α = 507 rad/sec^2 to bound both 1st and 2nd impacts
		α = *72* rad/sec^2	
	2nd impact	28	
		α = *160.9* rad/sec^2	
1 foot end drop		18	25
1 foot side drop		*13*	25

E. Method of Load Application to the Cask Body

The weight of the payload, the impact limiters, and the reaction loads at the interfaces of the cask with the impact limiters are modeled as radially applied pressures. Co-sinusoidal distribution functions are used to represent the circumferential distribution of these loads.

The magnitude and the distribution of the loads applied to the model are such as to ensure a conservative estimation of the resulting stresses. The following sections describe the details of the modeling of the pressure distribution for the HAC or NCT events.

In general, loads resulting from component weights are applied as pressures in the form of cosine functions in the circumferential direction, within a ±75° angle sector, and are generally assumed uniform in the axial direction. A ramped pressure distribution profile was used to represent the DSC loading in the corner drop case.

Loads due to reaction forces are modeled as applied pressures on the interface surfaces between the cask body and impact limiters in the form of cosine × cosh functions in the circumferential direction, within a ±90° angle sector.

Pressure loads acting in the axial direction are in general uniform, except for pressure due to the corner drop. For the corner drop, pressures acting in axial direction are modeled as varying linearly in the direction transverse to the impact plane, with the maximum pressure at the bottom location and the minimum pressure (zero magnitude) at the top location.

A.2.13.1.12 References

1. 10CFR PART 71, Packaging and Transportation of Radioactive Material.
2. Not used.
3. Regulatory Guide 7.8, "Load Combinations for the Structural Analysis of Shipping casks for Radioactive Material."
4. ASME Code Section III, Division 1, Subsection NB and Appendix F, 2004, including 2006 addenda.
5. NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," March 2000.
6. Regulatory Guide 7.9, "Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material."
7. ANSYS Computer Code and User's Manual, Release 10A1.
8. Not used.
9. Not Used.
10. *Not Used.*
11. Gordon, J. L., "OUTCUR: An Automated Evaluation of Two-Dimensional Finite Element Stresses" ASME Paper No. 76-WA/PVP-16, ASME Winter Annual Meeting (December 1976).

- Temperature load
- Impact load
- Puncture load
- External pressure loads
- Load combinations for normal and accident conditions
- Bolt stresses and allowable stresses
- Bearing stress
- Lid bolt fatigue
- Thread engagement length evaluation

The design parameters for the closure lid analysis taken from [1] are summarized in Table A.2.13.2-1. The lid bolt data and material allowables are presented in Tables A.2.13.2-2 to A.2.13.2-4. Material properties and allowable stresses for NCT and HAC analyses are based on 350 °F, which bound -40 °F, -20 °F, and 100 °F ambient conditions.

The following load cases are considered in the analysis:

1. Preload + Temperature Load (NCT);
2. Internal Pressure + 30 Foot Corner Drop (HAC);
3. Internal Pressure + Puncture Load (HAC).

A.2.13.2.2 Lid Bolt Load Calculations

A.2.13.2.2.1 Bolt Preload

The method of analysis is described in Table 4.1 of [1].

A bolt torque range of *950* to 1,040 ft.lb is required to ensure leak tightness against normal and accident loadings.

Bolt preload for the minimum torque is:

$$F_a = \frac{Q}{K \times D_b} = \frac{950 \times 12}{0.135 \times 1.5} = 56{,}296 \text{ lb/bolt}$$

Bolt preload for the maximum torque is:

$$F_a = \frac{Q}{K \times D_b} = \frac{1{,}040 \times 12}{0.135 \times 1.5} = 61{,}630 \text{ lb/bolt}$$

Residual torsional moment for the minimum torque is:

$$M_{tr} = 0.5 \times Q = 0.5 \times 950 \times 12 = 5{,}700 \text{ in.lb/bolt}$$

Residual torsional moment for the maximum torque is:

$$M_{tr} = 0.5 \times Q = 0.5 \times 1{,}040 \times 12 = 6{,}240 \text{ in.lb/bolt}$$

Residual tensile bolt force:

$$F_{ar} = F_a = 61{,}630 \text{ lb/bolt}$$

A.2.13.2.2.2 Gasket Seating Load

The analysis is described in Table 4.2 of [1].

An elastomer O–ring is used, therefore the gasket seating load is negligible, and $F_s = 0$.

A.2.13.2.2.3 Internal Pressure Load

The analysis is described in Table 4.3 of [1].

The axial force per bolt due to internal pressure is:

$$F_a = \frac{\pi \times D_{lg}^2 \times (P_{li} - P_{lo})}{4N_b}$$

D_{lg} for outer seal is conservatively taken equal to the outer diameter of the outer seal groove and is equal to (see Appendix A.1.4.12):

$$D_{lg} = 72.267 \text{ (seal groove I. D.)} + 2 \times 0.235 \text{ (groove width)} = 72.737 \text{ in.}$$

Therefore:

$$F_a = \frac{\pi \times 72.737^2 \times (30 - 0)}{4 \times 48} = 2{,}598 \text{ lb/bolt}$$

The fixed edge closure lid force is:

$$F_f = \frac{D_{lb} \times (P_{li} - P_{lo})}{4} = \frac{74.81 \times (30 - 0)}{4} = 562 \text{ lb/in}$$

A.2.13.2.7.3 Results

Figure A.2.13.2-1 plots the decompression of the seal as a function of circumferential location. As can be seen there is no decompression of the seal; *based on preload of 50 kips (minimum preload is 56.3 kips).* Figure A.2.13.2-2 shows the zoomed deformation plot near the lid-flange region.

From the analysis results it can be concluded that there is no decompression in the seal during the CG over corner drop impact scenario with internal pressure and bolt preload. Since the seal exists all along the circumference of the MP197HB Transport cask, the internal contents will not leak during the worst case loading condition.

A.2.13.2.8 Minimum Engagement Length for Bolt and Flange

For the 1 ½–6UNC bolt, the material is SA–540 Gr. B23 Cl. 1 which has a yield strength of 139.1 ksi and a tensile strength of 165.0 ksi at 350°F. The flange material is SA–350 LF3 or SA–203 Gr. E, with S_u = 70.00 ksi (at 350°F).

The minimum engagement length L_e for the bolt and flange is ([3], page 1490):

$$L_e = \frac{2 \times A_t}{3.1416 \times K_{n\max} \times \left[\frac{1}{2} + .57735 \times n \times \left(E_{s,\min} - K_{n\max}\right)\right]}$$

A_t is the tensile–stress area of the screw and is given by the following formula:

$$A_t = \pi \times \left(\frac{E_{s,\min}}{2} - \frac{0.16238}{n}\right)^2$$

According to [3]:

n = number of threads per inch = 6.

$E_{s,\min}$ = minimum pitch diameter of external threads = 1.3772 in.

Therefore, A_t = 1.375 in^2.

$K_{n\max}$ = maximum minor diameter of internal threads = 1.3500 in.

$D_{s\min}$ = minimum major diameter of external threads = 1.4703 in.

Substituting the values given above:

So:

$$J = \frac{1.178 \times 165.0}{1.569 \times 70.0} = 1.77$$

Therefore, the minimum required engagement length $Q = J \times L_e$ = 1.77 x 0.73 = 1.29 in.

The actual minimum engagement length is equal to:

4.00 (bolt length) – 1.25 (thickness of the cover plate under the screw head) – 0.177 (washer thickness) = 2.57 in > 1.50 in (ram cover plate bolts inserts) > 1.29 in.

The above calculation bounds the minimum required engagement length if inserts are used because S_u of inserts is higher than the S_u for the lid, thus lowering the J value.

A.2.13.2.10 Conclusions

A lid bolt torque range of *950* to 1,040 ft.lb is required.

A ram closure plate bolt torque range of 100 to 125 ft.lb is required.

For the required preloads:

1. Bolt stresses meet the acceptance criteria of NUREG/CR-6007 "Stress Analysis of Closure Bolts for Shipping Casks."
2. A positive (compressive) load is maintained during all load combinations, except, for the lid bolts, for the accident condition impact plus pressure load case. A more detailed analysis is performed to evaluate the closure of the lid during this event and shows that there is no decompression in the seal during this event, and therefore no leak of the contents during the worst case loading condition.
3. The bolt and flange thread engagement lengths are acceptable.

The MP197HB cask lid bolts will not fail due to fatigue for 250 round trip shipments.

Table A.2.13.2-1
Design Parameters for Bolts Analysis

	Parameter	Lid	Ram access cover plate
α_b	Thermal coefficient of expansion of the bolts (in/in/°F)	7.0×10^{-6}	
α_c, α_l	Thermal coefficients of expansion of the cask, closure lid and ram cover plate (in/in/°F)	7.5×10^{-6}	
ai	Maximum rigid–body impact acceleration of the cask for Hypothetical Accident Conditions – 30 ft C. G. over corner drop (g)	40	
xi	Impact angle between the cask axis and target surface for Hypothetical Accident Conditions – 30 ft C. G. over corner drop	60.3°	
D_b	Nominal diameter of closure bolt (in)	1.50	1.00
D_{lb}	Bolt circle diameter (in)	74.81	27.00
D_{lg}	Outer seal diameter (in)	72.737	25.258
D_{li}	Inner edge diameter (in)	70.44	22.00
D_{lo}	Outer edge diameter (in)	77.18	28.88
D_{pb}	Puncture bar diameter (in)	6.0	
DLF	Dynamic Load Factor	1.1	
E_b	Young's modulus of bolt material (ksi)	26.45×10^3	
E_c, E_l	Young's modulus of cask flange, cask bottom, closure lid and ram cover plate material (ksi)	26.45×10^3	
K	Nut factor for empirical relation between applied torque and achieved preload	0.135	
L_b	Bolt length between top and bottom surfaces of cover plate at bolt circle (in)	2.41	1.25
N_b	Total number of closure bolts	48	12
N_{ul}	Poisson's ratio of cover plates	0.3	
P_{li}	Pressure inside the cask (psig)	30	
P_{lo}	Pressure outside the cask (psig)	290	
Q	Applied preload bolt torque (ft.lb)	*950*–1,040	100–125
S_{ub}	Ultimate strength of bolt material (ksi)	165.0	
S_{ul}	Ultimate strength of cover plates material (ksi)	70.0	
S_{yb}	Yield strength of bolt material (ksi)	139.1	
S_{yl}	Yield strength of cover plates material (ksi)	32.6	34.8
t_c	Thickness of cask wall (in)	7.0	
t_l	Thickness of plates at center (in)	4.5	5.00
t_{lf}	Thickness of plates flange (in)	3.94	2.50
W_c	weight of contents (lbs)	118,500	N/A
W_l	weight of plate (lbs)	6,100	530

Table A.2.13.2-5
Lid Bolts Individual Summary

Load Case	**Applied Load**		**Non-Prying Tensile Force F_a (lb/bolt)**	**Torsional Moment M_t (in.lb/bolt)**	**Prying Force F_f (lb/in)**	**Prying Moment M_f (in.lb/in)**
Preload	Residual torque	Minimum	*56,296*	*5,700*	0	0
		Maximum	61,630	6,240	0	0
Gasket	Seating load		0	0	0	0
Internal Pressure	30 psig internal		2,598	0	562	5,247
Thermal	350°F		0	0	0	0
Impact	30 ft accident conditions drop		132,944	0	27,152	253,905
Puncture	Drop on six inch diameter rod		0	0	0	0
External pressure	290 psig external		0	0	-5,424	-50,719

Table A.2.13.2-6
Lid Bolts Normal and Accident Load Combinations

Load Case	**Combination Description**		**Non-Prying Tensile Force F_a (lb/bolt)**	**Torsional Moment M_t (in.lb/bolt)**	**Prying Force F_f (lb/in)**	**Prying Moment M_f (in.lb/in)**
1	Preload + Temperature (Normal Conditions)	A. Min. Torque	*56,296*	*5,700*	0	0
		B. Max. Torque	61,630	6,240	0	0
2	Pressure + Accident Impact (Accident Conditions)		135,542	0	27,714	259,152
3	Pressure + Puncture (Accident Conditions)		0	0	–5,424	–50,719

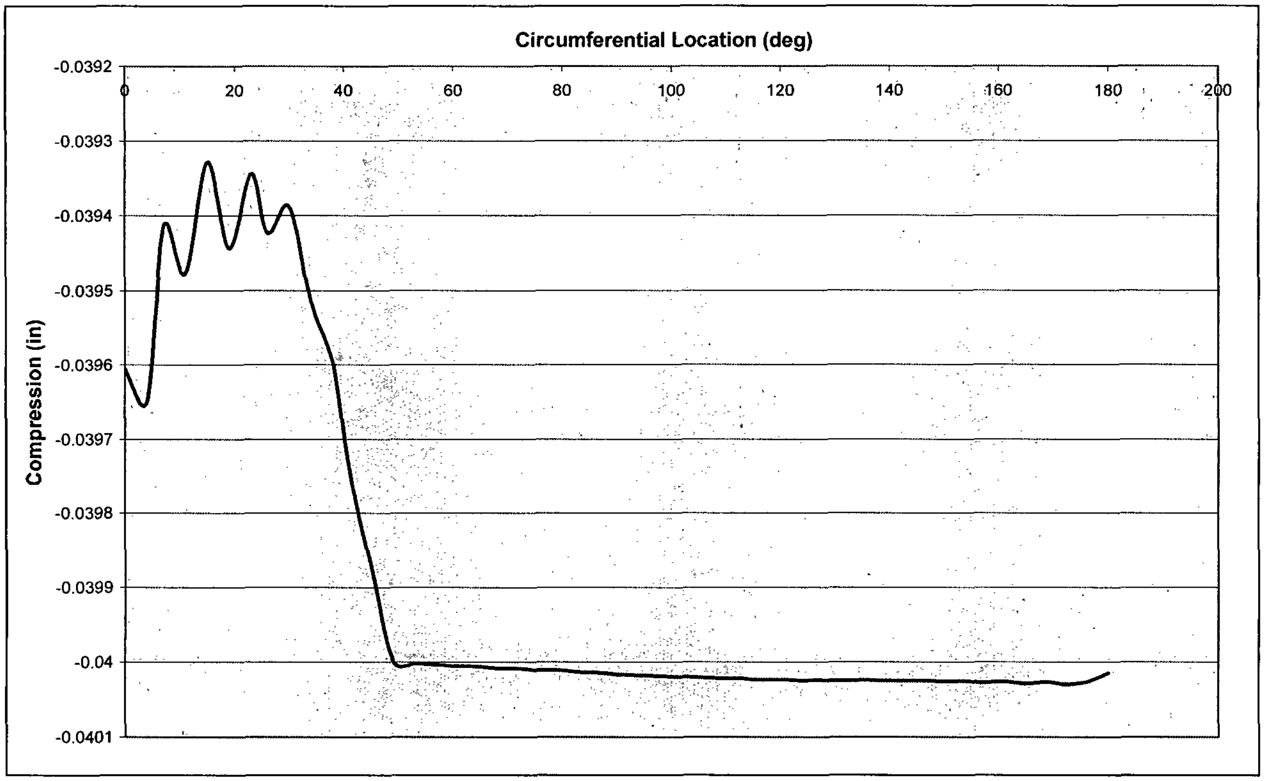


Figure A.2.13.2-1
MP197HB Transport Cask (CG Over Corner Lid Drop–Hot) Seal Decompression as a Function of Circumferential Location

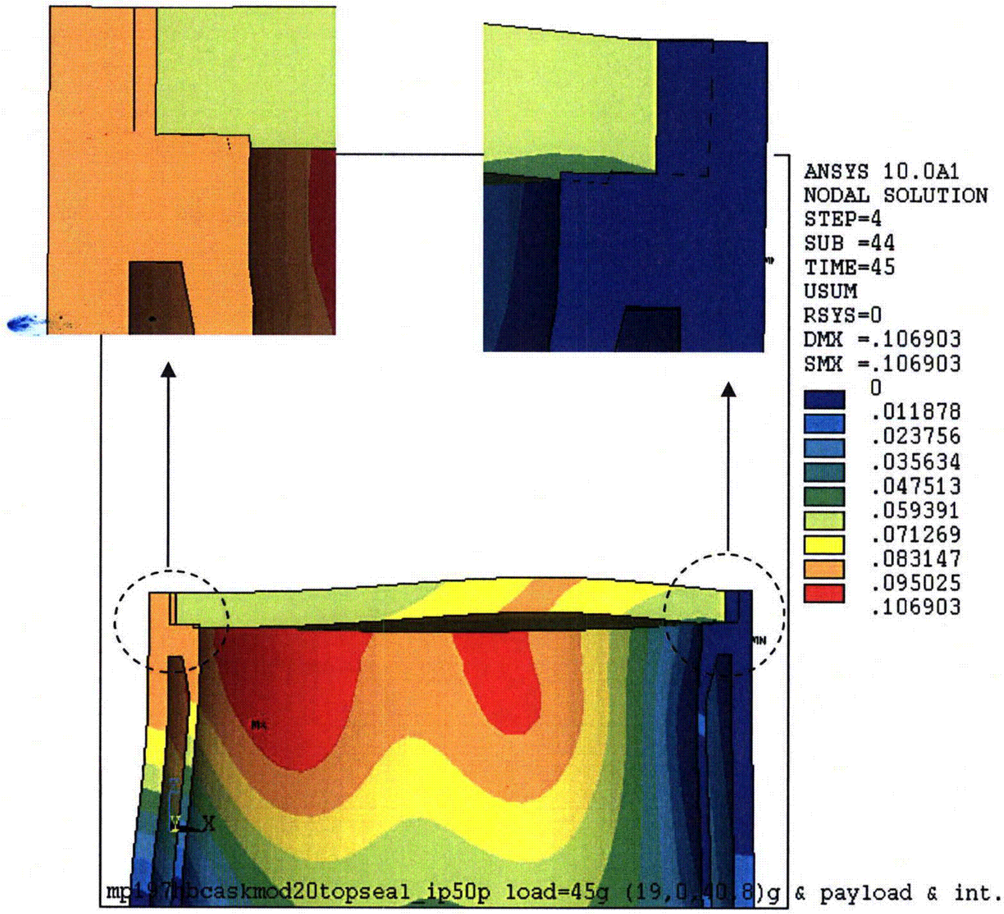


Figure A.2.13.2-2
Deformation Plot Near Lid–Flange Interface

A.2.13.7.3 Canister Structural Analysis

Finite element analyses are performed in order to quantify stresses in the DSCs generated by transport loads. The applied loads considered are normal and accident condition top end, bottom end, and side drops, combined with internal and external pressures and temperature distributions (thermal expansion stresses). Several finite element models are used to evaluate stresses for the normal and accident loads: 180 degree 3D models are used for side drop analyses; 2D axisymmetric models are used for end drop and thermal expansion analyses. Elastic material properties are used for normal condition stress analyses. Elastic-plastic material properties are used for normal condition limit load analyses and accident condition stress analyses.

A. Material Properties

Steel Material Properties for Group 1 through 4 DSCs

Material properties and allowable stresses for normal (NCT) and accident (HAC) drop analyses are based on 500 °F which bound -40 °F, -20 °F, and 100 °F ambient conditions. Thermal expansion analyses are based on temperature-dependent material properties shown in Tables A.2.13.7-1 and A.2.13.7-2 [3].

For the accident condition side drop cases where elastic-plastic analyses are performed, the tangent modulus is taken as 5% of the elastic modulus.

Lead Material Properties for Group 4 DSCs

The Group 4 DSCs have lead shield plugs instead of steel shield plugs. Material properties for the normal and accident drop analyses are based on 500 °F which bounds the maximum DSC temperatures. For accident condition load cases, dynamic stress-strain properties are used for lead. Material properties of lead are shown in Table A.2.13.7-3 [6], *[7]*.

B. Design Criteria

The steel component stresses are compared with the allowable stresses set forth by ASME B&PV Code Subsection NB [1]. The allowable stress values at 500 °F for the steel components are summarized in Table A2.13.7-4. Closure weld stress allowables are based on ISG-15 [5], which requires a design stress reduction factor to account for weld imperfection or flaws and recommends a stress reduction factor of 0.8 based on multi-level PT examination. The corresponding values at 500 °F are summarized in Table A.2.13.7-5. If the allowable stress limits are exceeded, a simplified fatigue analysis per NB- 3228.5 [1] is performed.

C. Loading Conditions

The load cases considered are normal and hypothetical accident condition drops, pressure loads, and temperature distributions (thermal expansion stresses). The normal condition drop loads are combined with internal and external pressure and the 100° F and -20° F ambient environment thermal loads. The accident condition drop loads are combined with internal and external pressure. The following tables summarize both normal and accident condition DSC individual load cases.

A.2.13.7.6 References

1. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, see Chapter A.2, Section A.2.1.2.1 for applicable editions.

2. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Appendices, see Chapter A.2, Section A.2.1.2.1 for applicable editions.

3. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section II, Part D, see Chapter A.2, Section A.2.1.2.1 for applicable editions.

4. ANSYS Computer Code and User's Manual, Release 10A1.

5. ISG-15, Rev. 0, "Materials Evaluation."

6. *R.A. Robinson, et. al., " A survey of Strain-Rate Effects for some common Structural Materials used in Radioactive Material Packaging and Transportation Systems", Report BMI-1954, August 1976, Battelle Columbus Laboratories.*

7. *Tietz, T. E., "Determination of the Mechanical Properties of a High Purity Lead and a 0.058 % Copper-Lead Alloy," WADC Technical Report 57-695, ASTIA Document No. 151165, Stanford Research Institute, Menlo Park, CA, April 1958.*

8. *H.J. Rack, G.A. Knorovsky, "An Assessment of Stress-Strain Data Suitable for Finite-Element Elastic-Plastic Analysis of Shipping Containers," Sandia Laboratories, NUREG/CR-0481, SAND77-1872 R-7, 1978.*

Table A.2.13.7-3
Material Properties for B-29 Lead

Static Mechanical Lead Properties

Temp. (°F)	Static Stress Properties (ksi) *[7]*			E *[8]* (10^6 psi)	Coef. of Thermal Exp *[8]* (10^{-6} in/in/°F)
	Yield (S_Y)		Ultimate (S_U)		
	Tension	**Compression**	**Tension**		
-99	-	-	-	2.50	15.28
70	-	-	-	2.34	16.07
100	0.584	0.490	1.570	2.30	16.21
175	0.509	0.428	1.162	2.20	16.58
250	0.498	0.391	0.844	2.09	16.95
325	0.311	0.320	0.642	1.96	17.54
440	-	-	-	1.74	18.50
620	-	-	-	1.36	20.39

Dynamic Stress-Strain Lead Properties

Strain (in/in)	Stress (ksi) [6]				
	100 °F	230 °F	300 °F	350 °F	500 °F
0.00048	1.14	1.06	1.00	0.97	0.86
0.03	2.2	2.0	1.7	1.5	1.1
0.1	3.3	2.8	2.38	2.1	1.26
0.3	4.9	3.2	2.72	2.4	1.44
0.5	5.6	3.6	3.06	2.7	1.62

- Fuel weight
  - The fuel weight used in the side drop analyses correspond to the maximum fuel weight in the active fuel region.
  - For the slap down drops the region of maximum g loads are at basket ends where the fuel assembly weight is lower than in the active fuel region and thus, lower stresses are expected.

The baseline g loads for the end and side drop analyses are calculated in Section A.2.13.12.10 of the Appendix A.2.13.12. These baseline g loads are conservatively using the wood temperature effect at -40°F (10CFR Part 71 only require -20°F). For the basket end and side drop analyses, first, the g loads are multiplied by appropriate dynamic load factors and then these g loads are increased by additional factors and used for basket structural evaluation. The calculated maximum g loads and g loads used for the analysis of each basket type are summarized in Table A.2.13.8-1.

Thermal stresses for -20 °F and 100 °F ambient conditions are calculated for all baskets. Furthermore thermal stresses for the -40 °F ambient conditions are also calculated for the 69BTH and 37PTH baskets.

A.2.13.8.4 Design Criteria

The basis for the basket stress allowables is the ASME Code, Section III, Subsection NG [1]. The primary membrane stress intensity and membrane plus bending stress intensities are limited to code allowable stress intensity, S_m, and 1.5 S_m, respectively, at any location in the basket for Normal Conditions of Transport (Service Level A) load combinations. If the limits on primary membrane stress intensity and primary membrane plus primary bending stress intensity are not satisfied at a specific location the primary membrane stress intensity and primary membrane plus primary bending stress are qualified by means of limit analysis (NG-3228.2) to demonstrate that the specified loadings do not exceed two-thirds of the lower bound collapse load.

The ASME Code provides a basic $3S_m$ limit on primary plus secondary stress intensity for Level A conditions. That limit is specified to prevent ratcheting of a structure under cyclic loading and to provide controlled linear strain cycling in the structure so that a valid fatigue analysis can be performed. If the primary plus secondary stress intensity exceeded $3S_m$, the criteria of ASME Code NG-3228.3 (Simplified Elastic-Plastic Analysis) are used in lieu of elastic analysis methods. The primary plus secondary stresses are also qualified by demonstrating that shakedown to elastic action occurs, as defined in NG-3228.1.

The baskets are evaluated under Hypothetical Accident Condition (Level D Service) loadings in accordance with the Level D Service limits for components in Appendix F of Section III of the Code [2]. The HAC free drops are evaluated as short duration Level D conditions. For elastic analysis, the primary membrane stress is limited to the smaller of $2.4S_m$ or $0.7S_u$ and membrane plus bending stress intensities are limited to the lesser of 3.6 S_m or S_u. The maximum primary shear stress is limited to 0.42 S_u. The results from the non-linear elastic-plastic analysis for the accident conditions, the general primary membrane stress intensity, Pm, shall not exceed the greater of $0.7S_u$ or $S_y + 1/3\ (S_u - S_y)$ and the maximum stress intensity at any location (P_l or $P_l + P_b$) shall not exceed 0.9 Su.

Table A.2.13.8-5
NUH61BT/61BTH Type 1 Normal Condition - Maximum Stress Summary

Drop Orientation	Component	Stress Category	Maximum Stress (ksi)	Allowable Stress (ksi)
0 degrees	Fuel Compartments	P_m	7.98	16.40
		$P_m + P_b$	13.81	24.60
		$P_m + P_b + Q$	34.01	49.20
	Canister	P_m	1.26	17.50
		$P_m + P_b$	8.08	26.25
		$P_m + P_b + Q$	-	52.50
	Rails	P_m	15.32	17.50
		$P_m + P_b$	26.13	26.25
		$P_m + P_b + Q$	27.53	52.50
30 degrees(1)(2)	Fuel Compartments	P_m	7.99	16.40
		$P_m + P_b$	43.32	24.60
		$P_m + P_b + Q$	63.52	49.20
	Canister	P_m	1.04	17.50
		$P_m + P_b$	13.71	26.25
		$P_m + P_b + Q$	-	52.50
	Rails	P_m	12.80	17.50
		$P_m + P_b$	31.63	26.25
		$P_m + P_b + Q$	33.03	52.50
45 degrees(1)	Fuel Compartments	P_m	6.72	16.40
		$P_m + P_b$	40.12	24.60
		$P_m + P_b + Q$	60.31	49.20
	Canister	P_m	1.16	17.50
		$P_m + P_b$	12.19	26.25
		$P_m + P_b + Q$	-	52.50
	Rails	P_m	10.26	17.50
		$P_m + P_b$	41.85	26.25
		$P_m + P_b + Q$	43.25	52.50
180 degrees(1)	Fuel Compartments	P_m	7.88	16.40
		$P_m + P_b$	28.67	24.60
		$P_m + P_b + Q$	48.87	49.20
	Canister	P_m	3.22	17.50
		$P_m + P_b$	47.99	26.25
		$P_m + P_b + Q$	-	52.50
	Rails	P_m	19.16	17.50
		$P_m + P_b$	45.40	26.25
		$P_m + P_b + Q$	46.80	52.50

Notes:
(1) *The exceedances for stresses are localized and have characteristics of secondary stresses. Limit analyses (NG-3228.2) are performed for these cases.*
(2) Simplified Elastic-Plastic Analyses performed for these cases

Table A.2.13.8-6
NUH61BTH Type 2 Normal Condition - Maximum Stress Summary

Drop Orientation	Component	Stress Category	Maximum Stress (ksi)	Allowable Stress (ksi)
0 degrees	Fuel Compartments	P_m	8.57	16.40
		$P_m + P_b$	14.57	24.60
		$P_m + P_b + Q$	34.77	49.20
	Canister	P_m	1.81	17.50
		$P_m + P_b$	6.32	26.25
		$P_m + P_b + Q$	-	52.50
	Rails	P_m	13.78	17.50
		$P_m + P_b$	24.42	26.25
		$P_m + P_b + Q$	25.82	52.50
30 degrees(1)	Fuel Compartments	P_m	9.96	16.40
		$P_m + P_b$	31.79	24.60
		$P_m + P_b + Q$	51.99	49.20
	Canister	P_m	1.98	17.50
		$P_m + P_b$	12.44	26.25
		$P_m + P_b + Q$	-	52.50
	Rails	P_m	10.33	17.50
		$P_m + P_b$	33.03	26.25
		$P_m + P_b + Q$	34.43	52.50
45 degrees(1)	Fuel Compartments	P_m	9.83	16.40
		$P_m + P_b$	31.93	24.60
		$P_m + P_b + Q$	52.12	49.20
	Canister	P_m	1.46	17.50
		$P_m + P_b$	13.98	26.25
		$P_m + P_b + Q$	-	52.50
	Rails	P_m	9.61	17.50
		$P_m + P_b$	30.51	26.25
		$P_m + P_b + Q$	31.91	52.50
180 degrees(1)	Fuel Compartments	P_m	9.07	16.40
		$P_m + P_b$	16.29	24.60
		$P_m + P_b + Q$	36.68	49.20
	Canister	P_m	3.04	17.50
		$P_m + P_b$	31.60	26.25
		$P_m + P_b + Q$	-	52.50
	Rails	P_m	14.12	17.50
		$P_m + P_b$	31.10	26.25
		$P_m + P_b + Q$	32.48	52.50

Notes:
(1) *The exceedances for stresses are localized and have characteristics of secondary stresses. Limit analyses (NG-3228.2) are performed for these cases.*

Table A.2.13.8-12
NUH69BTH Normal Condition - Maximum Stress Summary
(concluded)

Drop Orientation	Component	Stress Category	Maximum Stress (ksi)	Allowable Stress (ksi)
180 degrees(1)	Fuel Compartments	P_m	7.09	16.0
		$P_m + P_b$	21.27	24.0
		$P_m + P_b + Q$	30.99	47.4
	Fuel Compartment Wrap Plates	P_m	7.64	16.0
		$P_m + P_b$	37.09	24.0
		$P_m + P_b + Q$	46.81	47.4
	Canister	P_m	3.18	17.5
		$P_m + P_b$	30.41	26.3
		$P_m + P_b + Q$	50.91	52.5
	Aluminum Rails	Bearing Stress	2.63	5.50

Notes:
(1) *The exceedances for stresses are localized and have characteristics of secondary stresses. Limit analyses (NG-3228.2) are performed for these cases.*

Table A.2.13.8-16
NUH24PTH Normal Condition - Maximum Stress Summary (concluded)

Drop Orientation	Component	Stress Category	Maximum Stress (ksi)	Allowable Stress (ksi)
180 degrees(1)	Fuel Compartments	P_m	31.41	15.60
		$P_m + P_b$	23.18(2)	23.40
		$P_m + P_b+Q$	31.03(2)	46.80
	Canister	P_m	5.18	15.60
		$P_m + P_b$	24.03	23.40
		$P_m + P_b+Q$	43.5	46.80
	Basket Straps	P_m	4.91	15.60
		$P_m + P_b$	26.17	23.40
		$P_m + P_b+Q$	31.53	46.80
	R45 Transition Rails	P_m	20.61	15.60
		$P_m + P_b$	23.41	23.40
		$P_m + P_b+Q$	32.13	46.80
	R90 Transition Rails	Bearing	3.77	5.2

Note:
(1) *The exceedances for stresses are localized and have characteristics of secondary stresses. Limit analyses (NG-3228.2) are performed for these cases.*
(2) Maximum stress excludes the local peak concentrated stresses.

Table A.2.13.8-21
NUH32PTH1 Type 1 Normal Condition - Maximum Stress Summary

Drop Orientation	Component	Stress Category	Maximum Stress Intensity (ksi)	Allowable Stress Intensity (ksi)
0 degrees[2]	DSC Shell	P_m	0.88	17.5
		$P_m + P_b$	9.63	26.25
	Transition Rails	*Bearing Stress*	2.50	4.85
	Fuel Compartments	P_m	10.54[1]	15.80
		$P_m + P_b$	14.33	23.70
		$P_m + P_b + Q$	25.93	47.40
	Basket Straps	P_m	15.58	15.80
		$P_m + P_b$	16.61	23.70
		$P_m + P_b + Q$	28.21	47.40
30 degrees[2]	DSC Shell	P_m	1.17	17.5
		$P_m + P_b$	15.58	26.25
	Transition Rails	*Bearing Stress*	2.57	4.85
	Fuel Compartments	P_m	16.39[1]	15.80
		$P_m + P_b$	25.33[1]	23.70
		$P_m + P_b + Q$	36.93	47.40
	Basket Straps	P_m	8.78[1]	15.80
		$P_m + P_b$	28.68	23.70
		$P_m + P_b + Q$	40.28	47.40
45 degrees[2]	DSC Shell	P_m	1.11	17.5
		$P_m + P_b$	14.19	26.25
	Transition Rails	*Bearing Stress*	2.69	4.85
	Fuel Compartments	P_m	14.16[1]	15.80
		$P_m + P_b$	25.45[1]	23.70
		$P_m + P_b + Q$	37.05	47.40
	Basket Straps	P_m	7.82[1]	15.80
		$P_m + P_b$	28.68	23.70
		$P_m + P_b + Q$	40.28	47.40
180 degrees[2]	DSC Shell	P_m	2.11	17.5
		$P_m + P_b$	8.25	26.25
	Transition Rails	*Bearing Stress*	2.76	4.85
	Fuel Compartments	P_m	10.21[1]	15.80
		$P_m + P_b$	14.03[1]	23.70
		$P_m + P_b + Q$	25.63	47.40
	Basket Straps	P_m	8.01[1]	15.80
		$P_m + P_b$	16.81	23.70
		$P_m + P_b + Q$	28.41	47.40

Notes:

(1) Maximum stress intensity excluding point contact loaded elements

(2) *The exceedances for stresses are localized and have characteristics of secondary stresses. Limit analyses (NG-3228.2) are performed for these cases.*

Table A.2.13.8-22
NUH32PTH/NUH32PTH1 Type 2 Normal Condition - Maximum Stress Summary

Drop Orientation	Component	Stress Category	Maximum Stress Intensity (ksi)	Allowable Stress Intensity (ksi)
0 degrees(2)	DSC Shell	P_m	1.15	17.5
		$P_m + P_b$	6.13	26.25
	Transition Rails	P_m	11.14	17.05
		$P_m + P_b$	33.80(1)	25.58
		$P_m + P_b + Q$	48.30	51.15
	Fuel Compartments	P_m	9.32	15.80
		$P_m + P_b$	20.23(1)	23.70
		$P_m + P_b + Q$	29.43	47.40
	Basket Straps	P_m	13.56	15.80
		$P_m + P_b$	20.30	23.70
		$P_m + P_b + Q$	29.50	47.40
30 degrees(2)(3)	DSC Shell	P_m	2.35	17.5
		$P_m + P_b$	12.72	26.25
	Transition Rails	P_m	10.93	17.05
		$P_m + P_b$	22.96(1)	25.58
		$P_m + P_b + Q$	37.46	51.15
	Fuel Compartments	P_m	20.10	15.80
		$P_m + P_b$	29.02(1)	23.70
		$P_m + P_b + Q$	38.22	47.40
	Basket Straps	P_m	11.50(1)	15.80
		$P_m + P_b$	41.11(1)	23.70
		$P_m + P_b + Q$	50.31	47.40
45 degrees(2)(3)	DSC Shell	P_m	1.32	17.5
		$P_m + P_b$	9.95	26.25
	Transition Rails	P_m	8.89	17.05
		$P_m + P_b$	25.20	25.58
		$P_m + P_b + Q$	39.7	51.15
	Fuel Compartments	P_m	15.92	15.80
		$P_m + P_b$	26.58(1)	23.70
		$P_m + P_b + Q$	35.78	47.40
	Basket Straps	P_m	7.61(1)	15.80
		$P_m + P_b$	39.01(1)	23.70
		$P_m + P_b + Q$	48.21	47.40
180 degrees(2)(3)	DSC Shell	P_m	6.72	17.5
		$P_m + P_b$	48.21	26.25
	Transition Rails	P_m	14.93	17.05
		$P_m + P_b$	45.97(1)	25.58
		$P_m + P_b + Q$	60.47	51.15
	Fuel Compartments	P_m	9.99	15.80
		$P_m + P_b$	11.47(1)	23.70
		$P_m + P_b + Q$	20.67	47.40
	Basket Straps	P_m	13.82	15.80
		$P_m + P_b$	22.15(1)	23.70
		$P_m + P_b + Q$	31.35	47.40

Notes:
(1) Maximum stress intensity excluding point contact loaded elements
(2) *The exceedances for stresses are localized and have characteristics of secondary stresses. Limit analyses (NG-3228.2) are performed for these cases.*
(3) $P_m + P_b + Q$ may exceed the 3 S_m stress limit provided requirements (a) through (f) of NG-3228.3 [7] are met.

Table A.2.13.8-29
NUH37PTH Normal Condition - Maximum Stress Summary

Drop Orientation	Component	Stress Category	Maximum Stress (ksi)	Allowable Stress (ksi)
0 degrees	Fuel Compartments	P_m	10.97	16.4
		$P_m + P_b$	23.92	24.6
		$P_m + P_b + Q$	31.43	49.2
	Canister	P_m	0.60	17.5
		$P_m + P_b$	13.61	26.25
	Rails	Max. Bearing stress[3]	5.38	5.5
30 degrees[1]	Fuel Compartments	P_m	8.11	16.4
		$P_m + P_b$	43.87	24.6
		$P_m + P_b + Q$	45.26	49.2
	Canister	P_m	2.72	17.5
		$P_m + P_b$	24.86	26.25
	Rails	Max. Bearing stress[3]	3.89	5.5
45 degrees[1]	Fuel Compartments	P_m	7.40	16.4
		$P_m + P_b$	41.10	24.6
		$P_m + P_b + Q$	48.61	49.2
	Canister	P_m	2.75	17.5
		$P_m + P_b$	19.87	26.25
	Rails	Max. Bearing stress[3]	3.45	5.5
90 degrees	Fuel Compartments	P_m	11.0	16.4
		$P_m + P_b$	23.96	24.6
		$P_m + P_b + Q$	31.47	49.2
	Canister	P_m	0.59	17.5
		$P_m + P_b$	13.59	26.25
	Rails	Max. Bearing stress[3]	5.37	5.5
180 degrees[1]	Fuel Compartments	P_m	8.07	16.4
		$P_m + P_b$[2]	31.37	24.6
		$P_m + P_b + Q$	38.88	49.2
	Canister	P_m	4.04	17.5
		$P_m + P_b$	31.03	26.25
	Rails	Max. Bearing stress[3]	4.6	5.5

Notes:
(1) *The exceedances for stresses are localized and have characteristics of secondary stresses. Limit analyses (NG-3228.2) are performed for these cases.*
(2) The stress values exclude the corner stress concentration elements occurring at the bottom ends of compartments.
(3) Maximum bending stress from the aluminum rails are reported.

Table A.2.13.8-33
NUH24PT4 Normal Condition – Spacer Disk Maximum Stress Summary

Drop Orientation	Stress Category	Linearzed Stress (ksi)	Allowable Stress (ksi)
0 degrees	P_m	12.5	26.7
	$P_m + P_b$	56.3*	40.1
30 degrees	P_m	11.8	26.7
	$P_m + P_b$	90.7*	40.1
45 degrees	P_m	6.8	26.7
	$P_m + P_b$	93.1*	40.1
180 degrees	P_m	18.44	26.7
	$P_m + P_b$	62.20*	40.1

Note: * *The exceedances for stresses are localized and have characteristics of secondary stresses. Limit analyses (NG-3228.2) are performed for these cases.*

A.2.13.9.4 Dynamic Load Factor Calculations

ANSYS transient dynamic analysis was performed using a finite element model consisting of a damped spring oscillator, COMBIN14, and structural mass, MASS21, elements to calculate DLFs for 1' and 30' end and side drops. The finite element model is shown in Figure A.2.13.9-5. The acceleration time history of the 1 ft and 30 ft end and side drops are applied to the mass element and DLF is calculated by:

$DLF = u_{max\ dynamic} / u_{max\ static}$

where,
$u_{max\ static}$ – calculated by max g load x mass / stiffness
$u_{max\ dynamic}$ – calculated by ANSYS

Unit mass is assumed for all analyses, and the stiffness and damping (7% damping is used) are calculated based on the desired frequency of the spring using the following equations:

$$k = m(2\pi f)^2$$

where,
m – mass
f – desired frequency

$$c = 2\zeta\sqrt{(km)}$$

where,
ζ – damping ratio
k – stiffness
m – mass

DLFs are calculated for each drop condition for a frequency range from 5 to 200 Hz and are shown in Figure A.2.13.9-6. The DLFs for all baskets, canisters, and fuel claddings are summarized in Table A.2.13.9-6 and Table A.2.13.9-7.

The LSDYNA analyses in Appendix A.2.13.12 have been updated in Rev. 7, however since the effect on acceleration time histories is minimal, the results for the DLF are unaffected.

Proprietary information on pages A.2.13.11-i, -ii, and A.2.13.11-1 to A.2.13.11-17 withheld pursuant to 10 CFR 2.390

Proprietary information on pages A.2.13.11-21 to A.2.13.11-35 withheld pursuant to 10 CFR 2.390

Table A.2.13.11-9

DELETED

Table A.2.13.11-10
DELETED

Table A.2.13.11-11
DELETED

Table A.2.13.11-12
DELETED

Proprietary information on pages A.2.13.11-40 to A.2.13.11-52 withheld pursuant to 10 CFR 2.390

Figure A.2.13.11-15
DELETED

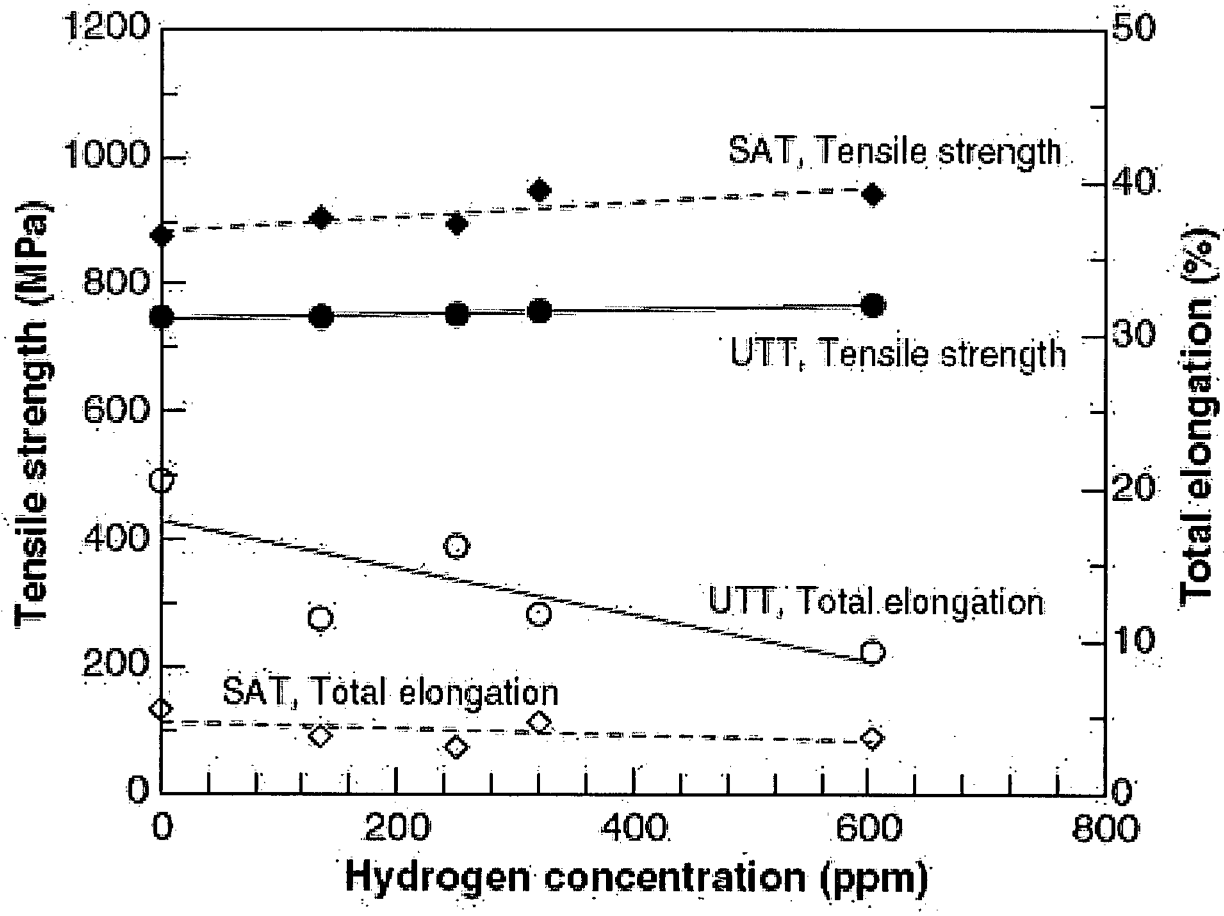


Figure A.2.13.11-16
(reproduced from Chu's Figure 7)
Effect of hydrogen concentration on the mechanical properties of SRA fuel cladding specimens tested under uniaxial tension and slotted arc tension; all hydrides in specimens were circumferentially aligned

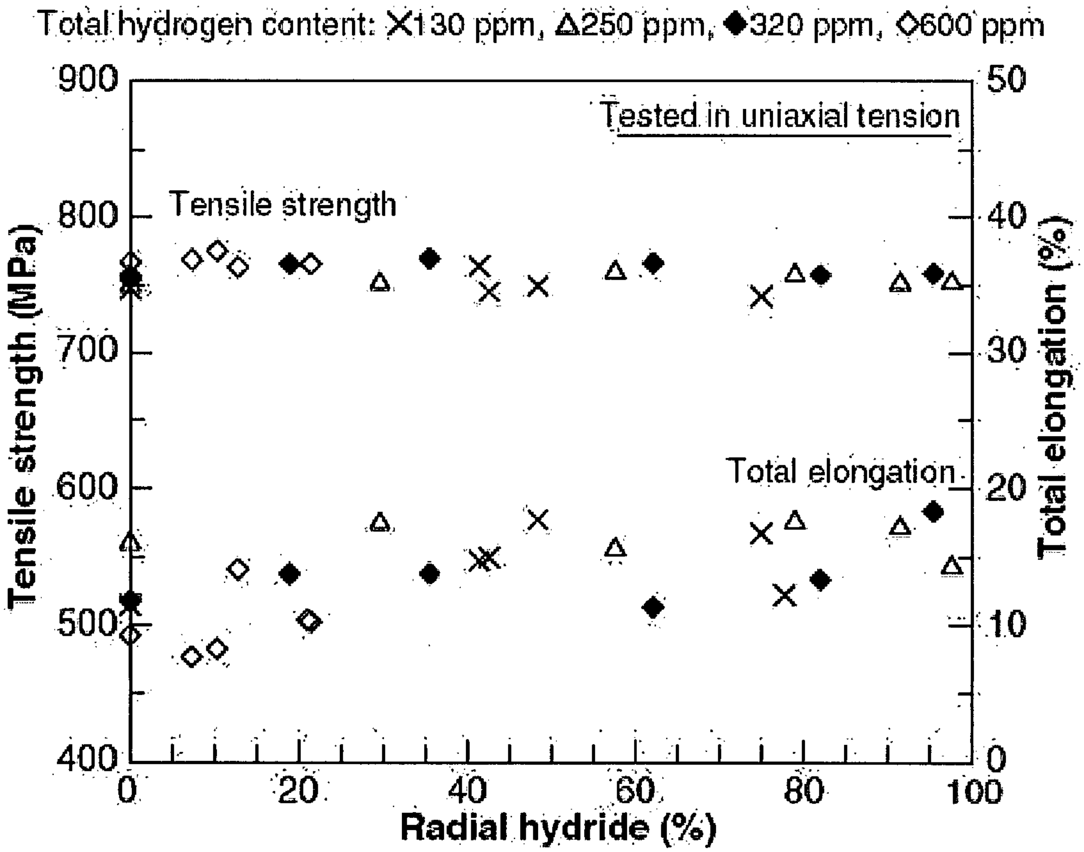


Figure A.2.13.11-17
(reproduced from Chu's Figure 8)
Effect of radial hydrides on the mechanical properties of SRA fuel cladding specimens tested under uniaxial tension at room temperature

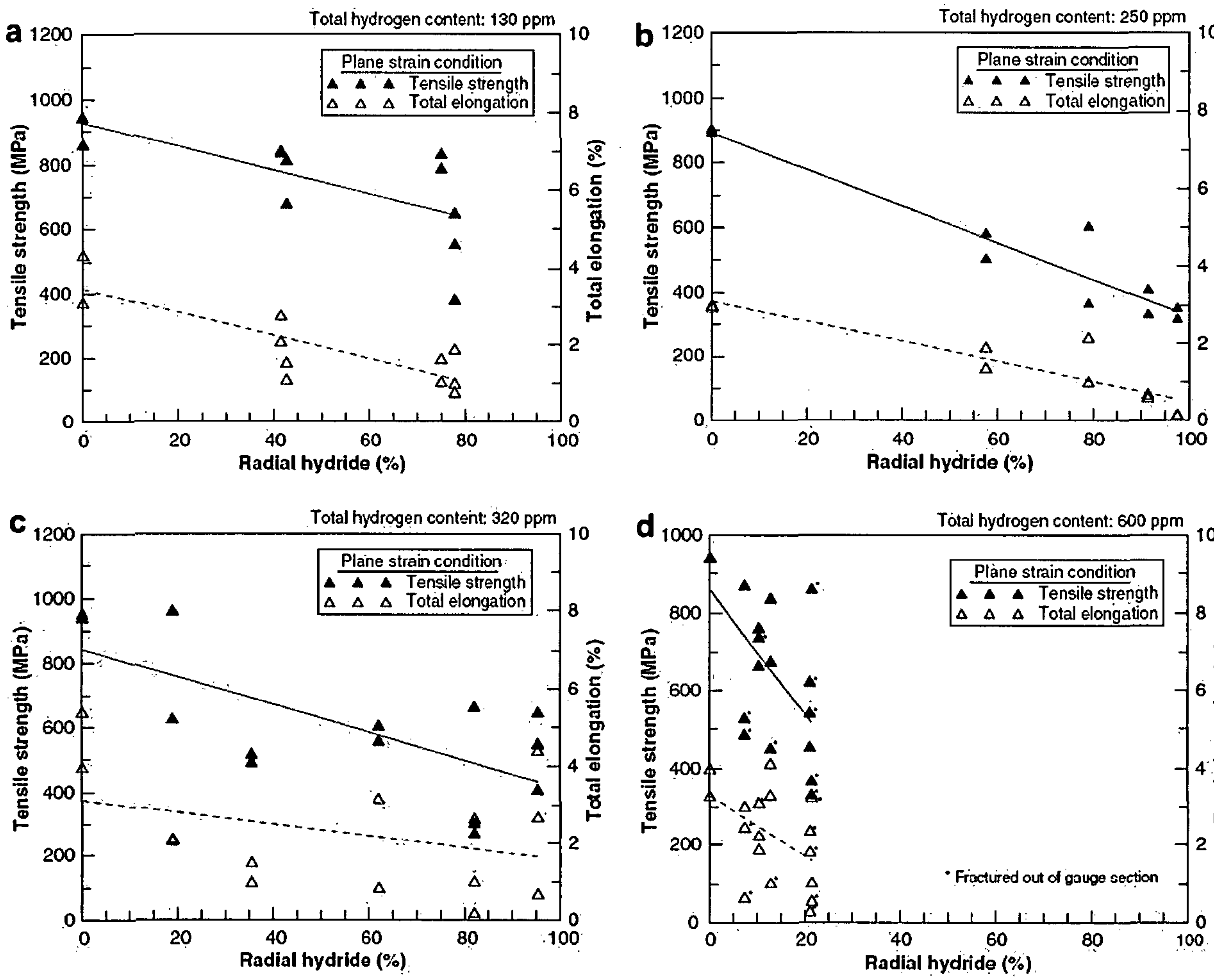


Figure A.2.13.11-18
(reproduced from Chu's Figure 9)
Effect of radial hydrides on the mechanical properties of cladding specimens with various hydrogen content levels tested under slotted arc tension at room temperature: (a) 130 ppm, (b) 250 ppm, (c) 320 ppm, and (d) 600 ppm.

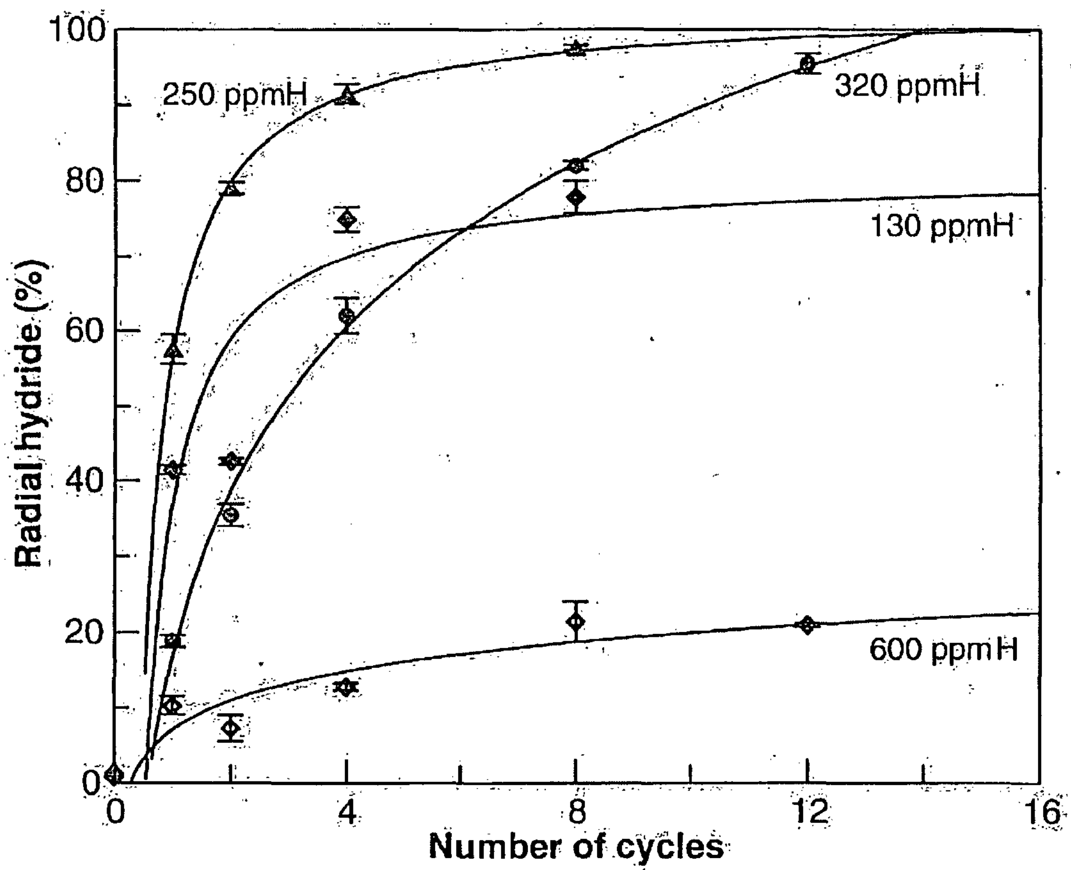


Figure A.2.13.11-19
(reproduced from Chu's Figure 5)
Effect of the Thermal Cycle Number on the Hydride Reorientation of Cladding Tubes with Various Hydrogen Concentrations

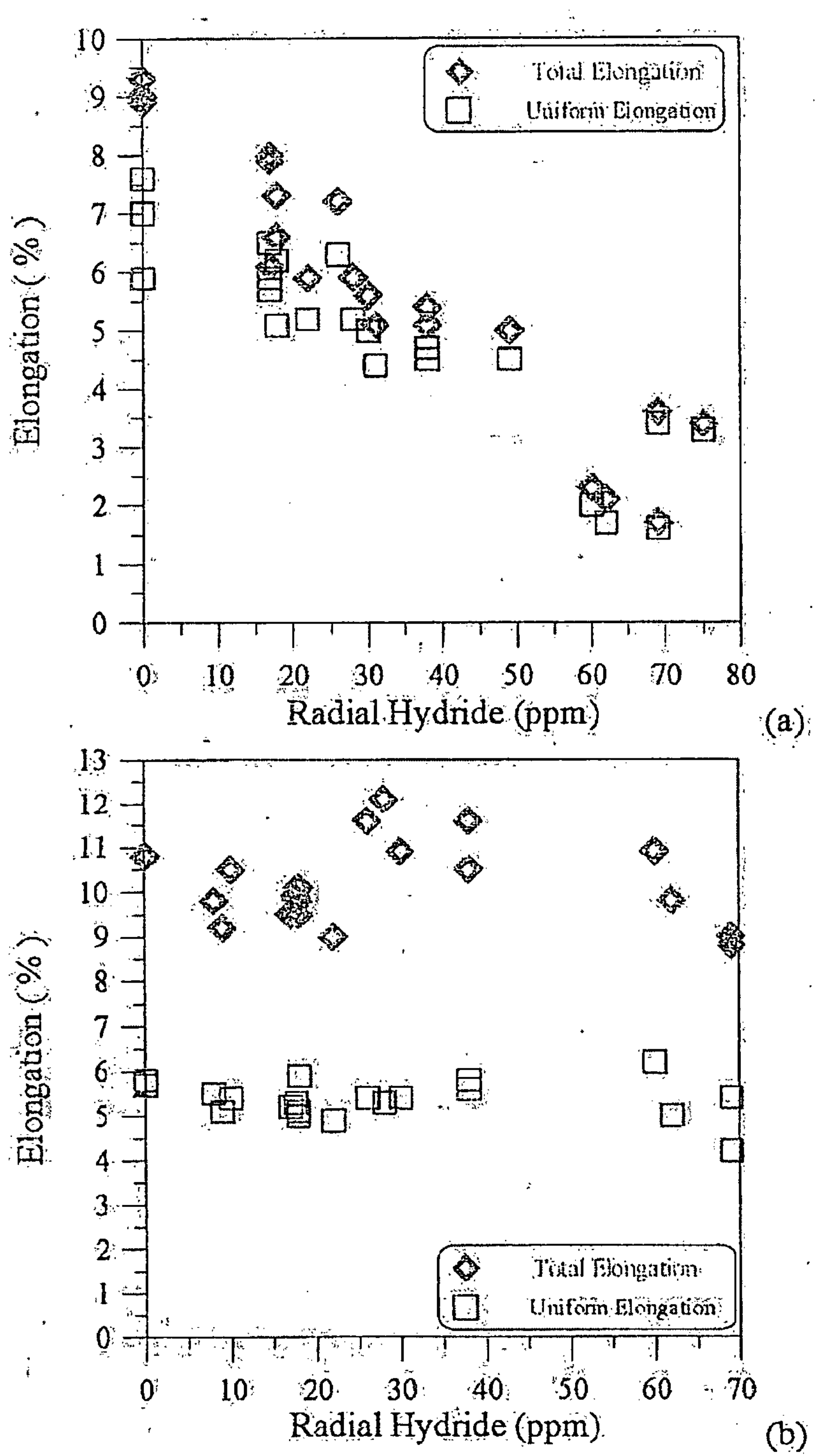


Figure A.2.13.11-20
(reproduced from Yagnik's Figure 9)
Effect of radial hydrides on the elongation of Zircaloy-4 cladding specimens with ~200 ppm H; SAT at 1(a) room temperature; (b) 300 °C

Figure A.2.13.11-21
DELETED

Figure A.2.13.11-22
DELETED

Figure A.2.13.11-23
DELETED

Figure A.2.13.11-24
DELETED

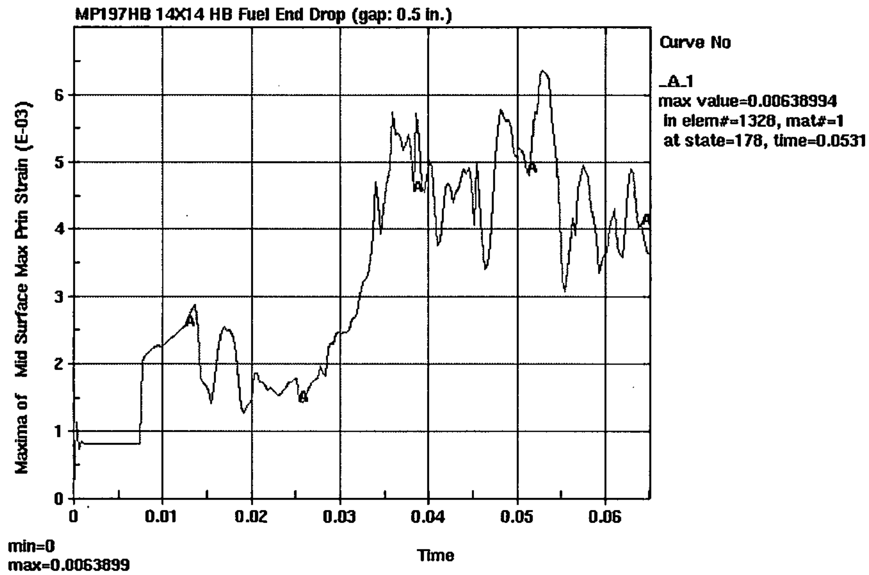


Figure A.2.13.11-25
Maximum Principal Strain Time-History

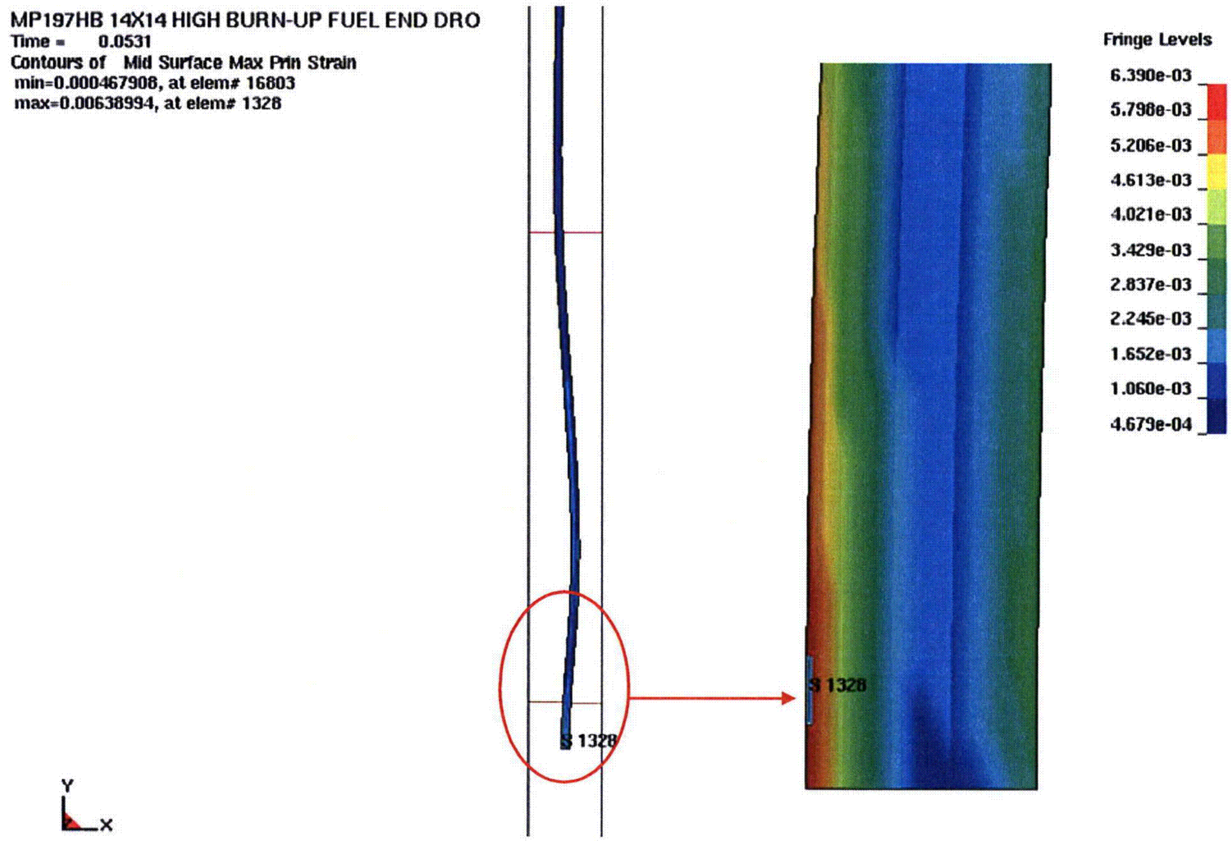


Figure A.2.13.11-26
Maximum Principal Strain Contour Plot

A.2.13.12.3 MP197 1/3 Scale LS-DYNA Benchmark Analysis

This section describes the series of analysis used to benchmark the impact limiter analysis methodology against the results of the 1/3 scale impact limiter drop testing program documented in NUHOMS®-MP197 Transport Package, main SAR. For this purpose a finite element model of the 1/3 scale MP197 cask mockup equipped with impact limiters is developed using the LS-DYNA computer program [2]. The LS-DYNA finite element model is analyzed for the same three hypothetical accident conditions as those in the test program, as summarized in Section A.2.13.12.2: 30' End Drop (-20°F), 30' Side Drop (Room Temperature), and 30' 20° Slap Down (Room Temperature). The finite element analysis results are compared to the actual drop test results documented in Appendix 2.10.9 of the NUHOMS®-MP197 Transport Package, main SAR, to benchmark the adequacy of the finite element model.

A. Description of Finite Element Model

The finite element analysis model is a representation of the surrogate 1/3 scale MP197 cask used for the actual drop tests with an impact limiter installed on each end of the cask. The impact limiter model incorporates the individual balsa or redwood sections that make up the impact limiter and the stainless steel cover shell. The impact limiter incorporates a thermal shield consisting of an aluminum plate interfacing with the cask and aluminum spacer blocks that create a thermal gap with the cask lid at each end. The finite element model includes the aluminum spacer blocks. The aluminum plate itself is not modeled as it would have negligible contribution to the structural performance of the cask. Additional features such as the impact limiter attachment bolts and alignment tubes are also included in the LS-DYNA model. The impact surface, which consists of a steel plate over a thick concrete pad is also included in the model.

The impact limiter wood sections, the concrete pad, *the* steel plate, *and the cask model* are modeled *with default* LS-DYNA *constant stress* solid elements. The impact limiter shell is modeled with fully integrated shell elements.

The impact limiter attachment bolts and associated inner and outer bolt tubes are modeled as a combination of spring elements and beam elements. The outer alignment tubes and inner welded bolt tubes are modeled as beam elements with the proper dimensions. The inner bolt tube is welded to the stainless steel impact limiter shell. The section of bolt extending from the bolt boss to the impact limiter shell is modeled as a beam section. The beam is fixed to the bolt boss on one end. The other end is fixed to the impact limiter shell in a way that approximates the bolt penetrating the shell. For the end drop condition, this end of the beam is coupled to the shell in the X and Y directions. For the side drop and slap-down conditions, this end of the beam is coupled to the shell in the Y and Z directions. The remaining section of bolt is modeled as a non-linear spring. The spring has a tensile stiffness and a negligibly small compression stiffness. This is done in order to model the bolt slipping through the tube during compression. The small compression spring rate prevents instabilities within LS-DYNA. Bolts located on the axis of symmetry are modeled with modified dimensions that have half the moment of inertia and half the cross sectional area.

Only one-half of the cask, impact limiters, steel plate and concrete are modeled, as the entire arrangement is reasonably symmetric about the x-y plane. The distribution of the thermal shield spacer blocks is not symmetric to the plane of symmetry. However, the slight asymmetry is negligible since the effect of the blocks is still captured.

The three analyzed drop accident conditions are as follows:

1. 30' End Drop (-20°F)
2. 30' Side Drop (Room Temperature)
3. 30' 20° Slap Down (Room Temperature)

Drop models have consistent *material properties for the* room temperature *and chilled (-20 °F) cases. The* wood properties *for* the chilled end drop *are* modified to simulate a -20°F environment. The slap down scenario has the cask initially rotated 20° from the horizontal side drop position and includes an impact limiter at both ends.

The finite element model and different drop orientations can be seen in Figure A.2.13.12-1 through Figure A.2.13.12-4.

B. Cask Model and Steel Plate Material

The cask model and steel impact plate are modeled as A36 steel with the following properties:

$E = 27.7\times10^6$ psi
$v = 0.3$
$S_y = 30.0$ ksi
Tangent Modulus, $E_T = 2\%\ E = 5.540\times10^5$ psi

Material properties at 400 °F are used; material properties at room temperature and at -20 °F would have negligible effect on the results.

C. Impact Limiter Shell Material

The impact limiter shell is modeled as SA-240 Type 304 stainless steel with the following properties *applicable to the -20 °F and room temperature cases.*

$E = 28.3\times10^6$ psi
$v = 0.3$
$S_y = 30.0$ ksi
Tangent Modulus, $E = 1.0\text{X}10^5$ psi
Mass density, $\rho_{eff} = 9.405\times10^{-4}$ lb sec^2/in^4

The density of the impact limiter shell is adjusted to account for the weight of those components not included in the model, such as the lifting lugs. The measured weight of the actual 1/3 scale cask mockup with both impact limiters installed is 9,750 lbs.

$S_y = 30{,}000$ psi.
Tangent Modulus, $E_T = 30 \times 10^4$ psi.

E. Impact Limiter Wood Segment Material

The impact limiter wood is modeled using the Mat_Modified_Honeycomb material model in LS-DYNA (material type 126), which models crushable materials with anisotropic behavior. References [4] and [5] document the use of *Mat _Honeycomb* material model in LS-DYNA to model wood in a similar application. Since the crush strength of wood is not isotropic, separate material properties are used in directions parallel and perpendicular to the wood grain. Strain rate effects are neglected in the analysis. Table A.2.13.12-3 summarizes the wood segment material properties. Table A.2.13.12-4 summarizes the wood segment material properties increased by 20% for the -20° F temperature condition.

The pressure versus volumetric strain relationship defines the crush strength of the wood segments. Table A.2.13.12-5 shows the pressure versus strain curves for redwood and balsa parallel and perpendicular to the wood grain for the average room temperature wood properties. Table A.2.13.12-6 shows the same for the 20% increased wood properties for the -20° F ambient condition. The pressure strain curve is assumed to be initially linear with a slope of the modulus of elasticity E, and then flat up to a locking strain of 0.8 for balsa and 0.6 for redwood.

F. Bolt and Alignment Tube Material

There are 12 bolts that attach each impact limiter to the cask. The following elastic, linearly plastic material properties are used for the bolts in the finite element model *for the -20 °F and room temperature cases.*

SA-540 G*rade* B24 C*lass* 1
$E = 27.8 \times 10^6$ psi
$v = 0.3$
$S_y = 150.0$ ksi
Tangent Modulus, $E_T = 2\%\ E = 5.54 \times 10^5$ psi

The bolts that attach the impact limiters to the cask are modeled as circular cross section beams. For the Side Drop and Slap Down conditions, the bolt tube sections at the symmetry plane are modeled as hollow circular cross section beams with modified dimensions to represent half the area and moment of inertia.

The springs representing the sections of bolts within the impact limiter were given spring rates representing that of the bolt. The length of bolt used for the spring rate calculation is 10". For the Side Drop and Slap Down conditions, the springs at the plane of symmetry were given half the spring rate. The springs at the plane of symmetry for the End Drop condition were left unchanged as the bolts would be in compression and therefore would have negligible affect on the analysis.

Bolt alignment tubes are modeled as tube beams. The alignment tubes modeled at the plane of symmetry are also full-size as their strength difference is minimal in this analysis. The following elastic, linearly plastic material properties are used for the bolt tubes *for the -20 °F and room temperature cases.*

SA-249 Type 304

$E = 28.3\times10^6$ psi
$v = 0.3$
$S_y = 30.0$ ksi
Tangent Modulus, $E = 10^5$ psi

G. Boundary Conditions and Initial Conditions

One-half of the cask and impact limiters are modeled with symmetry boundary conditions used to simulate the full structure. The lowest point of the impact limiter shell is initially placed within 0.25" of the steel impact plate to minimize run time. An initial velocity corresponding to the drop height is applied to the model. The initial velocity computed for a 30' drop is 527.45 in/sec.

The automatic *general* contact definition in LS-DYNA (*Contact_Automatic_General*) is applied to model the contact between any two parts of the model. A conservatively low coefficient of friction of 0.25 is applied between all the contact surfaces. The use of a low value for the coefficient of friction is conservative because less energy is absorbed due to friction resulting in greater impact acceleration forces.

An interior contact (Contact_Interior) contact definition is applied to the elements modeling the wood sections to prevent *elements from inverting and becoming negative volumes. Hourglass controls are applied to control the underintegrated solid elements used in the model.*

Non-reflecting boundaries are applied to the bottom and sides of the modeled concrete not aligned with the plane of symmetry (bottom, left side, right side, and back) to prevent artificial stress waves from reflecting.

A.2.13.12.4 1/3 Scale Impact Limiter Benchmark Analysis Results

Table A.2.13.12-7 summarizes the results of the LS-DYNA analysis in terms of peak filtered accelerations, maximum crush depths, and impact durations for the three drop cases analyzed and compare them with similar parameters obtained from the test results. The maximum accelerations from the tests and the LS-DYNA analyses model are divided by three to represent the accelerations of a full scale MP197 cask. Impact durations and crush depths are unchanged (compare 1/3 scale analysis results to1/3 scale test results). Figures A.2.13.12-5 through A.2.13.12-8 show the deformed shapes of the impact limiters for each drop case analyzed. *Figures A.2.13.12-57 to A.2.13.12-59 show the energy plots (total, kinetic, internal, hourglass) for the 1/3 scale benchmark analyzed cases.*

The acceleration time histories from the LS-DYNA analyses and from the 1/3 scale testing are shown in the figures summarized in the table below:

Drop Test Case	Figures from LS-DYNA Analysis	Figures from 1/3 Scale Testing	
Side Drop	Figure A.2.13.12-9	Figure A.2.13.12-10	Figure A.2.13.12-11
End Drop	Figure A.2.13.12-12	Figure A.2.13.12-13	Figure A.2.13.12-14
20° Slap Down	Figure A.2.13.12-15 (1st Impact)	Figure A.2.13.12-17 (1st Impact)	
	Figure A.2.13.12-16 (2nd Impact)	Figure A.2.13.12-18 (2nd Impact)	

As described in Appendix 2.10.9 of NUHOMS®-MP197 Transport Package, main SAR, twelve (12) accelerometers were used to measure the response deceleration time histories for each test. The accelerometers were placed along the test body at 0°, 90°, 180°, and 270° orientations at approximately the center of gravity location and adjacent to each impact limiter. *The location of the 12 accelerometers is shown in Figure 2.10.9-7 of Appendix 2.10.9. The locations correspond to the locations at which the LS-DYNA analysis results are compared, as shown in Figure A.2.13.12-56.*

The test response acceleration time histories shown in Figures A.2.13.12-10, -11, -13, -14, -17, and -18 are representative of the test case indicated. The peak g load calculated from the LS-DYNA analysis *obtained at the locations shown in Figure A.2.13.12-56* is compared with the measured averaged g loads from all the accelerometers.

The analysis results shown in Table A.2.13.12-7 agree well with the measured results of the impact limiter drop tests. The impact time durations match closely with those of the measured results. Wood crush depths are also similar. Therefore, it can be concluded that the LS-DYNA model and analysis methodology implemented as described in this section accurately predicts the response of the impact limiters during the accidental drop.

surface. Only ½ of the cask, impact limiters, steel plate and concrete are modeled as the entire arrangement is reasonably symmetric about the x-y plane.

The impact limiter wood sections, the concrete pad, steel plate, *and cask model* are modeled in LS-DYNA using *constant stress* elements. The impact limiter shell is modeled with fully integrated shell elements.

Additional features, such as the impact limiter attachment bolts and associated outer alignment and inner welded tubes are modeled as a combination of spring elements and beam elements. The alignment tubes and welded tubes are modeled as beam elements welded to the stainless steel impact limiter shell.

The drop scenarios analyzed include the hypothetical accident condition drop with a drop height of 30 feet and the 1-foot normal condition of transport drop. Consistent wood properties are used for all analysis cases except for the 30 feet end drop for which modified wood properties to simulate -20°F and -40°F environments are used. The Slap Down cases are 20° and 10° from the horizontal side drop position. The CG over corner drop is 65° from the horizontal side drop position.

Figures A.2.13.12-19 through A.2.13.12-24 show the MP197HB finite element models for the different drop orientations analyzed.

B. Material Properties

The material properties used in this analysis are the same as those used for the 1/3 Scale MP197 Drop Analysis Benchmark. The material properties required to perform the analysis include modulus of elasticity, E, shear modulus, G, Poison's Ratio, v, and material density, ρ, for the cask model, impact limiter shell, wood segments, steel plate and concrete. *The mechanical properties used are the same as those used in the 1/3 scale benchmark analysis in Section A.2.13.12.2.*

C. Cask Model and Steel Plate Material

The following elastic, linearly plastic material properties for mild steel (A-36) are used for the steel plate and cask model.

$E = 27.7\times10^6$ psi
$v = 0.3$
$S_y = 30.0$ ksi
Tangent Modulus, $E_T = 2\%\ E = 5.540\times10^5$ psi

Material properties at 400 °F are used ; material properties at room temperature and at -20 °F would have negligible effect on the results.

D. Cask Model

The cask model has the same mass, center of gravity, and moment of inertia as the actual cask. The critical moment of inertia is about Y-axis which is perpendicular to the plane of symmetry of the model. The moments of inertia about the other two axes do not influence this analysis due to the boundary conditions.

The dimensions and density of the modeled cask are adjusted to have the same mass, center of gravity, and moment of inertia as the actual cask. The geometry of the modeled cask is similar to the actual cask except that the center section has a smaller diameter. The reduced diameter area is offset from the geometric center to affect the center of gravity.

E. Impact Limiter Shell

The following elastic, linearly plastic material properties are used for the impact limiter shell.

Stainless Steel (SA-240 Type 304) *at room temperature and -20 °F*

$E = 28.3\times10^6$ psi
$v = 0.3$
$S_y = 30.0$ ksi
Tangent Modulus = 10^5 psi

The density of the impact limiter shell is adjusted to calibrate the weight of the impact limiter model. This is to take into account for the mass of those impact limiter parts not included in the model.

F. Concrete

The concrete material model and properties are identical to those used for the 1/3 scale MP197 model in Section A.2.13.12.3.

G. Impact Limiter Wood Segments

The impact limiter wood is modeled using the Mat_Modified_Honeycomb material model (Material type 126) in LS-DYNA [2], which models crushable materials with anisotropic behavior such as wood. Since the crush strength of wood is not isotropic, separate material properties are used in directions parallel and perpendicular to the wood grain. Wood crush strengths and density are taken from the MP197HB 1/3 Scale Benchmark Analysis.

H. Bolt and Alignment Tubes

There are 12 bolts that attach each impact limiter to the cask model. The following elastic, linearly plastic material properties are used for the bolts.

SA-540 G*rade* B23 C*lass* 1 *at room temperature and -20 °F*
$E = 27.8\times10^6$ psi
$v = 0.3$
$S_y = 150.0$ ksi
Tangent Modulus, $E_T = 2\%\ E = 5.56\times10^5$ psi

Bolts at the symmetry plane are modeled as hollow circular cross section beams with modified dimensions to represent approximately half the area and moment of inertia.

Each bolt has a bolt alignment tube and a welded bolt tube. The following elastic, linearly plastic material properties are used for the bolt tubes:

SA-312 Type 304 *at room temperature and -20 °F*

$E = 28.3 \times 10^6$ psi
$\nu = 0.3$
$S_y = 30.0$ ksi
Tangent Modulus, $E = 10^5$ psi

Bolt alignment tubes are modeled as tube beams. Bolt alignment tubes at the plane of symmetry represent approximately half the area and moment of inertia.

Welded bolt alignment tubes are modeled as tube beams. Welded bolt alignment tubes at the plane of symmetry represent approximately half the area and moment of inertia.

I. Boundary and Initial Conditions

Because of symmetric one-half of the cask and impact limiters are modeled with symmetry boundary conditions used to simulate the full structure. The initial velocity is computed by equating potential and kinetic energies. For a 30 foot drop, the initial velocity is 527.5 in/sec. For a 1 foot drop, the initial velocity is 96.3 in/sec.

An automatic surface to surface (contact_automatic_single_surface) contact definition is applied between all parts where contact is feasible. *An interior* (contact_interior) contact definition *is* also applied to the wood parts to prevent elements from inverting and becoming negative volumes. *Hourglass controls are applied to all materials.* A conservatively low coefficient of friction (static and kinetic) of 0.25 is applied between all contact surfaces.

Non-reflecting boundaries are applied to the bottom and sides of the modeled concrete not aligned with the plane of symmetry (bottom, left side, right side, and back) to prevent artificial stress waves from *reflecting*. Both dilatation and shear waves are damped.

J. Data Reduction

The following table lists the duration of the analysis for each drop condition. The time step was automatically chosen by the LS-DYNA program based on the minimum model element size.

Drop Condition	Run Duration (sec)
30' End Drop (Room temperature)	0.06
30' End Drop (-20°F)	0.06
30' End Drop (-40°F)	0.06
30' Side Drop	0.06
30' Slap Down 20°	0.2
30' Slap Down 10°	0.2
30' CG Over Corner Drop	0.1
1' Normal Condition End Drop	0.04
1' Normal Condition Side Drop	0.06

The resulting nodal acceleration time histories are computed by LS-DYNA. The nodal accelerations obtained at the center of the cask model are averaged for all drop conditions except for the slap down analysis cases where the accelerations are averaged for both the top and bottom of the cask due to the dual impacts. Figures A.2.13.12-25 and A.2.13.12-26 show the regions of nodes averaged.

For the Slap Down 20° scenario, the cask is 17.9° counterclockwise from horizontal during the peak acceleration of the first impact and 3.5° clockwise from horizontal during the peak acceleration of the second impact.

For the Slap Down 10° scenario, the cask is 8.48° counterclockwise from horizontal during the peak acceleration of the first impact and 2.35° clockwise from horizontal during the peak acceleration of the second impact.

The time step in the LS-DYNA analysis is 50 μsec. Therefore, by the Nyquist theorem, the frequency content of the nodal acceleration data ranges from zero Hz, up to the following maximum frequency, f_{max}.

$$f_{max} = \frac{1}{2}\frac{1}{50\times10^{-6}\text{ sec}} = 10\text{ kHz}$$

The natural frequencies of the MP197HB cask model, which can be excited by an impact event, are significantly lower than 10kHz. These natural modes of the transport cask involve small displacements (and therefore low stresses) at frequencies higher than that of the rigid body motion of the transport cask. These high frequency accelerations mask the true rigid body motion of the transport cask, because both the low frequency rigid body acceleration and the high frequency natural vibration accelerations superimpose. The net acceleration is contained in the raw data computed by LS-DYNA. Therefore, filtering is necessary to remove these high frequency accelerations.

The averaged raw data for each cross section is filtered using a low pass Butterworth filter with different cutoff frequencies depending on the orientation of the model in order to recover the actual rigid body acceleration of the cask model. A 180 Hz cutoff frequency is used for the End Drop, *side drop,* and the CG Over Corner drop *conditions*. A 120 Hz cutoff frequency is used for the Slap Down runs. The cutoff frequencies are conservative because they will filter out some but not all of the high vibration modes of the cask model in their respective locations and drop orientations. Therefore, the response predicted by the filtered results includes more dynamics than simply the rigid body motion of the transport cask.

A.2.13.12.6 Analysis Results

Table A.2.13.12-8 summarizes the results of the LS-DYNA analysis in terms of peak filtered accelerations, impact durations, maximum crush depths, and provides the corresponding time history plot for the drop scenarios analyzed. Impact limiter crush depth is based on the deformation of the cask into the impact limiter in the vertical direction.

Impact durations are visually determined from the acceleration time history plots (Figures A.2.13.12-27 through A.2.13.12-36).

Figure A.2.13.12-37 through Figure A.2.13.12-46 show the maximum impact limiter deformation plots. *Energy plots (total, kinetic, internal, hourglass) are presented in Figures A.2.13.12-60 to A.2.13.12-66.*

A.2.13.12.7 Additional Analysis Results for Slap Down Drop Analyses

The peak rotational acceleration is calculated for the Slap Down conditions based on the cask interface forces and rotational centers. Figure A.2.13.12-47 shows the locations of nodes groups selected to determine acceleration during peak acceleration of the first and second impacts. Figures A.2.13.12-48 and A.2.13.12.-49 show the plot results for the 20° and 10° slap down drop cases respectively.

The rotational centers are at 172.86" from the bottom for the first impact and 55.69" from the bottom for the second impact of the Slap Down 20°. The rotational centers are at 159.61" from the bottom for the first impact and 46.33" from the bottom for the second impact of the Slap Down 10°.

Axial and shear bolt forces at various positions around the impact limiter are plotted for the Slap Down 20° and Slap Down 10° cases in Figures A.2.13.12-50 and A.2.13.12-51 respectively.

Rigid body accelerations for the entire cask model are shown in Figures A.2.13.12-52 and A.2.13.12-53 for the Slap Down 20° and Slap Down 10° runs respectively. The following table shows the cask rigid body accelerations and rotational acceleration at the CG of the cask body.

Cask Rigid Body Accelerations and Rotational Acceleration at CG of the Cask Body
(10° and 20° Slap Down)

Drop Orientation		**Peak rigid body deceleration at CG of the cask**
30 feet slap down (10°)	1st impact	Translation (*20.2*g)
		Rotational (α = *106.8* rad/sec^2)
	2nd impact	Translation (*25.7*g)
		Rotational (α = *144.9* rad/sec^2)
30 feet slap down (20°)	1st impact	Translation (*15.8*g)
		Rotational (α = *62.9* rad/sec^2)
	2nd impact	Translation (*24.1*g)
		Rotational (α = *138.7* rad/sec^2)

A.2.13.12.8 End Drop Analysis Based on -40 °F Wood Properties

An additional analysis was performed using the wood properties at -40°F temperature. For the -40°F chilled case, the wood modulus of elasticity and shear modulus are increased by 30% [*3*] from the room temperature. The results of this analysis are shown on Table A.2.13.12-8.

A.2.13.12.9 Sensitivity Studies of MP197HB Impact Limiter Analysis

Sensitivity studies are performed to determine how the analysis results are affected by various model characteristics. The sensitivity studies performed are summarized as follows:

A. Effect of Wood Element Mesh Density and Effect of Bolts on Calculated Impact Load

Two sensitivity studies are performed to evaluate the sensitivity of the analysis results to changes in mesh discretization of the wood elements in the impact limiter model and changes in the modeling of the attachment bolts. The first study run is an end drop at room temperature with double wood elements in the lower impact limiter. The second study run is a side drop with all bolts and bolt alignment tubes removed in order to evaluate the affect of the bolts and their influence on the peak accelerations. The following table summarizes the results of these sensitivity analyses. The comparison indicates that the wood mesh density used is sufficiently accurate. The study run without bolts indicates that the bolt model functions properly and does not significantly stiffen the impact limiter.

Summary of Model Geometry and Friction Coefficient Sensitivity Studies

	First Impact Peak Acceleration	Second Impact Peak Acceleration
0.125 μ	42.5 *g*	48.6 *g*
0.25 μ	40.5 *g*	50.2 *g*
0.375 μ	40.0 *g*	49.2 *g*
Original Slap Down 10°	40.4 *g*	53.2 *g*

Based on the results shown in this table, the friction coefficient does not significantly affect the results. The results from the original are bounding.

A.2.13.12.10 Baseline g Loads for Structural Evaluations

Based on the LS-DYNA calculated impact accelerations shown in Table A.2.13.12-8, the *effect of* wood propert*ies* at -20 °F temperature *is to* increase the g-load*s* approximately by *11%* (*49.7/44.9*=1.*11*). *The effect of* wood propert*ies* at -40 °F temperature *is to* increase the g-load*s* approximately by *16%* (*52.0/44.9*=1.*16*). *Therefore, the g-loads resulting from the room temperature case for all drop orientations are increased by 11% and 16% to account for the effect of wood properties at -20 °F and -40 °F, respectively. The table below shows the analysis-based peak rigid body accelerations and the g-loads increased by the above factors. The table below also shows the baseline g-loads used for the structural evaluations.*

Drop Orientation		Peak rigid body deceleration	-20°F Case Factor	-40°F Case Factor	Factored g-loads (-20°F Case)	Factored g-loads (-40°F Case)	g-loads Used In Structural Evaluations
30' End Drop		44.9 g	1.11	1.16	49.8	52.1	55.0
30' Side Drop		47.3 g	1.11	1.16	52.5	54.9	55.0
30' CG Over Corner Drop		33.4 g (Vertical Direction)	1.11	1.16	37.1	38.7	40.0
30' Slap (20°)	1st impact	Translation (15.8g)	1.11	1.16	17.5	18.3	22.0
		Rotational (α = 62.9 rad/sec^2)	1.11	1.16	α = 69.8 rad/sec^2	α = 73.0 rad/sec^2	α = 93 rad/sec^2
	2nd Impact	Translation (24.1g)	1.11	1.16	26.8	28.0	28.0
		Rotational (α = 138.7 rad/sec^2)	1.11	1.16	α = 154.0 rad/sec^2	α = 160.9 rad/sec^2	α = 164 rad/sec^2
30' Slap (10°)	1st impact	Translation (20.2g)	1.11	1.16	22.4	23.4	25.0
		Rotational (α = 106.8 rad/sec^2)	1.11	1.16	α = 118.5 rad/sec^2	α = 123.9 rad/sec^2	α = 132 rad/sec^2
	2nd Impact	Translation (25.7g)	1.11	1.16	28.5	29.8	32.0
		Rotational (α = 144.9 rad/sec^2)	1.11	1.16	α = 160.8 rad/sec^2	α = 168.1 rad/sec^2	α = 166 rad/sec^2
1' End Drop		15.4 g	1.11	1.16	17.1	17.9	18.0
1' Side Drop		11.0 g	1.11	1.16	12.2	12.8	19.0

A.2.13.12.11 Impact Limiter Bolt Evaluation

The purpose of this section is to determine the stresses in the NUHOMS® – MP197HB impact limiter attachment bolts and blocks.

The worst loading occurs in the top impact limiters attachment blocks during the second impact of a shallow angle slap–down drop.

Twelve impact limiter attachment bolts take the moment applied during a shallow angle slap–down drop.

The impact limiter bolts are evaluated using two approaches. The first evaluation is made using the maximum tensile force *from the LS-DYNA analysis taken from* Figure A.2.13.12-51 for a bolt during a shallow angle slap–down drop, which is *about 200*,000 lb. The maximum tensile force F in a bolt considered in this case will be conservatively taken equal to 210,000 lb.

In order to benchmark the *above* method, a *second* evaluation is made where the tensile force in the bolts is calculated by considering the equilibrium of moments each bolt is *subjected* to, conservatively assuming that:

1. The lateral force exerted on the impact limiter by the cask comes from the full weight of the cask (306,500 lbs) and is centered in the middle of the impact limiter cavity (at a distance 15.375 in from the bottom of the impact limiter);
2. The g load for the second impact of a shallow angle slap–down drop is conservatively taken equal to 35 g (see Section A.2.13.12.10, baseline g load is 32 g);
3. The friction coefficient μ between the cask and the impact limiter and between the impact limiter and the impact surface is 0.42 [6], based on hard steel against hard steel friction properties.

Since the bottom impact limiter crushes during the first impact, the top impact limiter impacts with a slight angle during the second impact, and its bottom edge crushes first. Therefore, the reaction force on the impact limiter is located close to the bottom edge of the impact limiter. However, to maximize bolt forces, we conservatively assume that the reaction force is exerted on the other side of the lateral force exerted by the cask, at a distance from that lateral force equal to 10% of the depth of the impact limiter cavity.

Material mechanical properties for the impact limiter and attachment bolts are taken at 200°F, and at 300°F for the attachment bolts blocks and for the cask shell. However, material properties used for checking thread engagement length are taken at room temperature.

Nut factor for empirical relation between the applied torque and achieved preload is 0.135 for neolube lubricant.

Table A.2.13.12-3
Wood Segment Room Temperature Material Properties

	Redwood	Balsa
Density, ρ	3.445×10-5 lbm/in^3 (23 lb/ft^3)	1.647×10-5 lbm/in^3 (11 lb/ft^3)
Shear Modulus, G parallel to grain	*9,801* psi	*1,513* psi
Shear Modulus, G perpendicular to grain	*68,599* psi	*11,194* psi
Elastic Modulus parallel to grain	*942*,000 psi	*303*,000 psi
Elastic Modulus perpendicular to grain	*75,135* psi	*4,537* psi

Note: The properties listed in this table are originated from [3] and are adjusted to benchmark against the 1/3 scale impact limiter drop test results.

Table A.2.13.12-4
Wood Segment Material Properties - 20 °F

	Redwood	Balsa
Density, ρ	3.445×10-5 lbm/in^3 (23 lb/ft^3)	1.647×10-5 lbm/in^3 (11 lb/ft^3)
Shear Modulus, G parallel to grain	*11,759* psi	*1,816* psi
Shear Modulus, G perpendicular to grain	*82,319* psi	*13,433* psi
Elastic Modulus parallel to grain	1,*132*,000 psi	*364*,000 psi
Elastic Modulus perpendicular to grain	*90,160* psi	*5,444* psi

Table A.2.13.12-5
Pressure vs. Volumetric Strain for 100% Average Room Temperature Wood Properties

Redwood Parallel to Grain	
Volumetric Strain ($\Delta V/V$)	**Pressure (psi)**
-1	*-381,788*
-0.6	*-5,094*
-0.0043	*-5,034*
0.000	0
0.0043	*5,034*
0.6	*5,094*
0.601	*1*
1	*1*

Redwood Perpendicular to Grain	
Volumetric Strain ($\Delta V/V$)	**Pressure (psi)**
-1	*-339,384*
-0.6	*-573*
-0.0059	*-552*
0.000	0
0.0059	*552*
0.6	*573*
0.601	*1*
1	*1*

Balsa Parallel to Grain	
Volumetric Strain ($\Delta V/V$)	**Pressure (psi)**
-1	*-62,072*
-0.8	*-1,499*
-0.0036	*-1,381*
0.000	0
0.0036	*1,381*
0.8	1,*499*
0.801	*1*
1	*1*

Balsa Perpendicular to Grain	
Volumetric Strain ($\Delta V/V$)	**Pressure (psi)**
-1	*-60,836*
-0.8	*-292*
-0.049	*-278*
0.000	0
0.049	*278*
0.8	*292*
0.801	*1*
1	*1*

Note: The properties listed in this table are originated from [3] and are adjusted to benchmark against the 1/3 scale impact limiter drop test results.

Table A.2.13.12-6
Pressure vs. Volumetric Strain for Wood Properties -20 °F

Redwood Parallel to Grain	
Volumetric Strain (ΔV/V)	**Pressure (psi)**
-1	*-458,145*
-0.6	*-6,114*
-0.0043	*-6,042*
0.000	0
0.0043	*6,042*
0.6	*6,114*
0.601	*1*
1	*1*

Redwood Perpendicular to Grain	
Volumetric Strain (ΔV/V)	**Pressure (psi)**
-1	*-407,260*
-0.6	*-686*
-0.0059	*-662*
0.000	0
0.0059	*662*
0.6	*686*
0.601	*1*
1	*1*

Balsa Parallel to Grain	
Volumetric Strain (ΔV/V)	**Pressure (psi)**
-1	*-74,449*
-0.8	*-1,760*
-0.0036	*-1,658*
0.000	0
0.0036	*1658*
0.8	*1760*
0.801	*1*
1	*1*

Balsa Perpendicular to Grain	
Volumetric Strain (ΔV/V)	**Pressure (psi)**
-1	*-73,004*
-0.8	*-350*
-0.049	*-334*
0.000	0
0.049	*334*
0.8	*350*
0.801	*1*
1	*1*

Table A.2.13.12-7
Peak Nodal Accelerations, Wood Crush Depths, and Impact Duration Comparisons
(1/3 Scale Test Results vs. 1/3 Scale LS-DYNA Analysis)

		Test Results	**LS-DYNA Model**
End Drop (-20°F)	Acceleration	65g	65.8g
	Impact Duration	0.010 sec.	0.012 sec.
	Wood Crush Depth	2.5"	2.5"
Side Drop	Acceleration	61g	61.5g
	Impact Duration	0.012 sec.	0.014 sec.
	Wood Crush Depth	2.69"-2.75"	2.75"
20° Slap Down 1st Impact	Acceleration at Center of Cask	17g	18.2g
	Acceleration at Bottom of Cask	36g	34.9g
	Impact Duration	0.016 sec.	0.023 sec.
	Wood Crush Depth Bottom Limiter	4.92"	5.5"
20° Slap Down 2nd Impact	Acceleration at Center of Cask	32g	41.1g
	Acceleration at Top of Cask	73g	78.4g
	Impact Duration	0.009 sec.	0.012 sec.
	Wood Crush Depth Upper Limiter	2.42"(1)	3.0"

NOTE:
(1) Re-examination of the MP197 test results: It shows that the crush depth is 2.42" instead of 4.72" as specified in Appendix 2.10.9 of the MP197 main SAR.

Table A.2.13.12-8
Summary of the Full Scale Impact Limiter Analysis Results

Drop Scenario		Peak Acceleration (g)	Impact Duration (sec)	Impact Limiter Crush Depth (in)	Time History Figure Number
30' End Drop (Room Temp)		*44.9*	0.*05*	10.2"	A.2.13.12.2-27
30' End Drop (-20°F)		*49.7*	0.*045*	9.3"	A.2.13.12.2-28
30' End Drop (-40°F)		*52.0*	0.*045*	9.0"	----
30' CG Over Corner Drop		*33.4*	0.090	27.2"	A.2.13.12.2-30
30' Side Drop		*47.3*	0.055	11.5"	A.2.13.12.2-29
30' Slap Down 20°	**1st Impact**	*28.1*	0.075	15"	A.2.13.12.2-31
	2nd Impact	*52.2*	0.045	11.3"	A.2.13.12.2-32
30' Slap Down 10°	**1st Impact**	*39.0*	0.060	13.4"	A.2.13.12.2-33
	2nd Impact	*55.2*	0.050	13.9"	A.2.13.12.2-34
1' Normal Condition End Drop		*15.4*	0.*035*	1.1"	A.2.13.12.2-35
1' Normal Condition Side Drop		*11.0*	0.045	1"	A.2.13.12.2-36

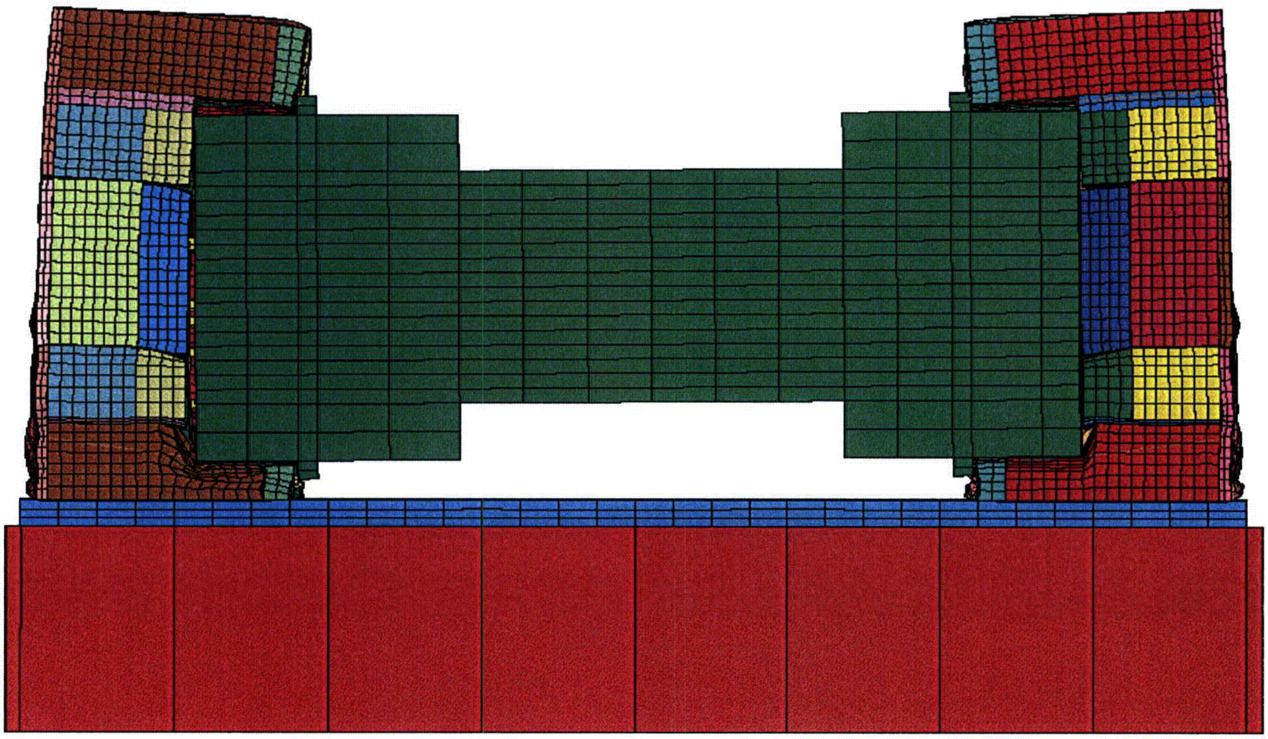


Figure A.2.13.12-5
Plot of Maximum Deformation for 1/3 Scale Side Drop

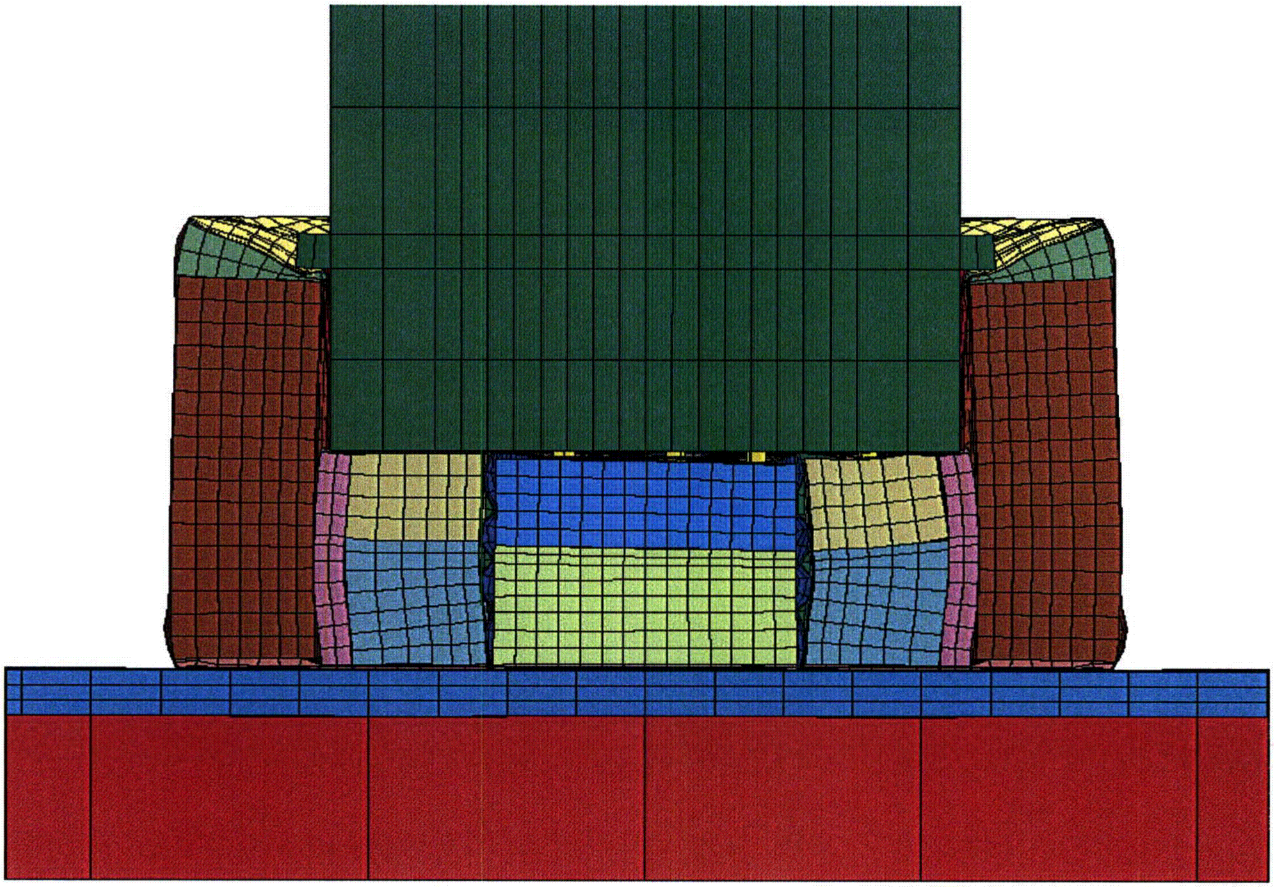


Figure A.2.13.12-6
Plot of Maximum Deformation for 1/3 Scale End Drop (-20°F)

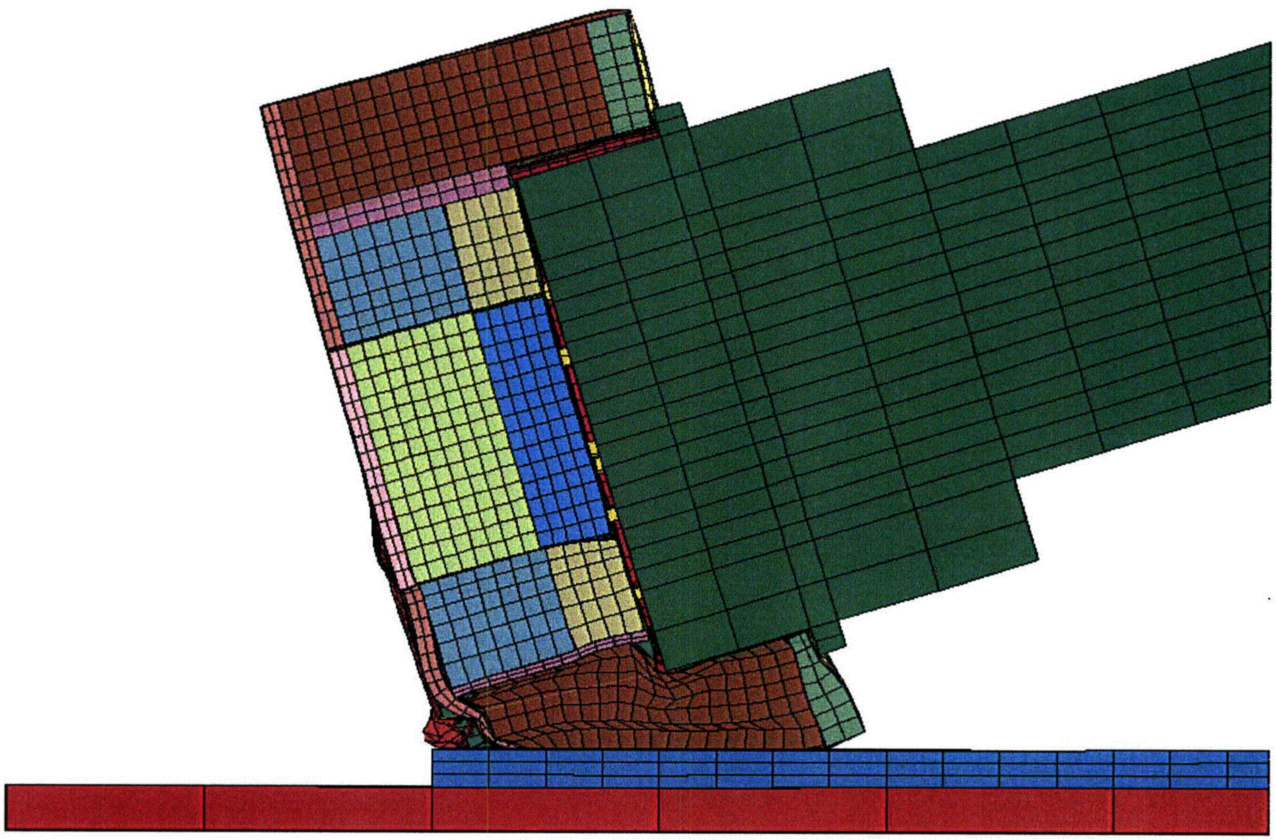


Figure A.2.13.12-7
Plot of Maximum Deformation for 1/3 Scale 20° Slap Down Drop
(First Impact)

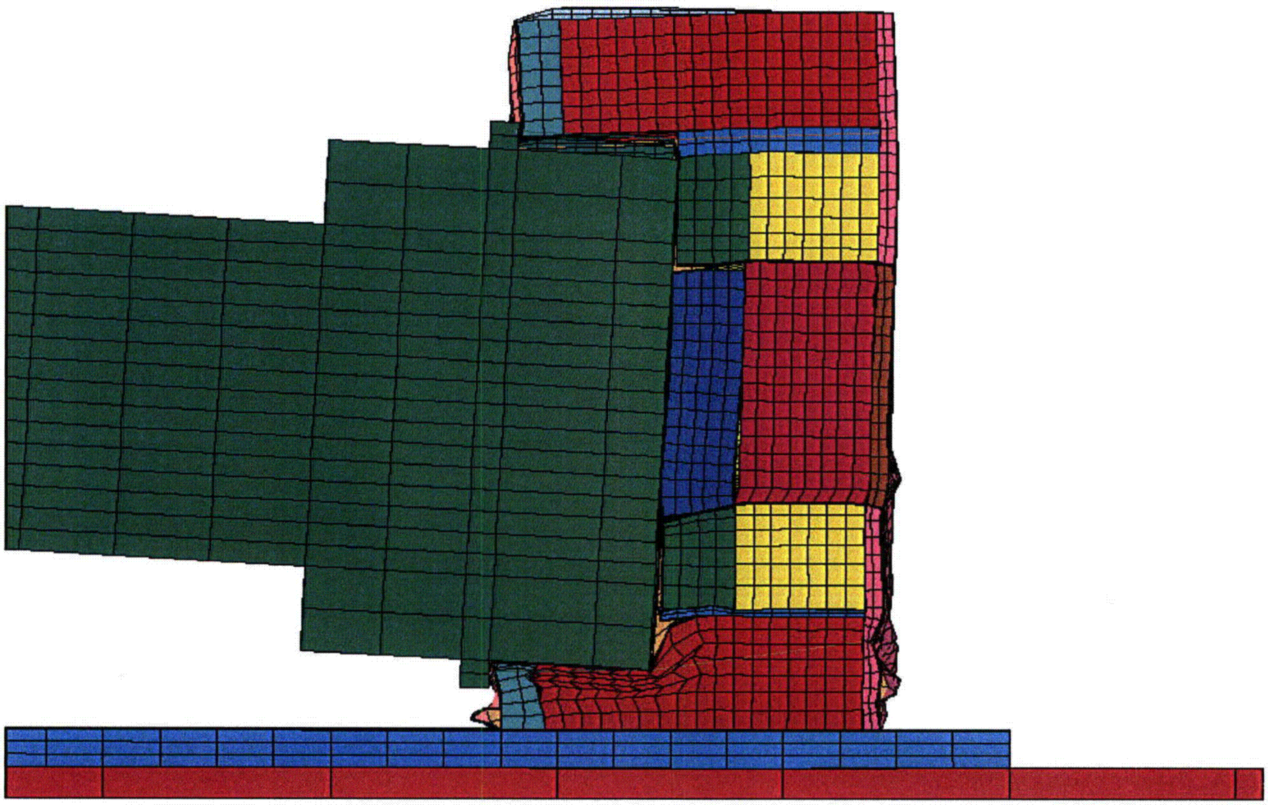
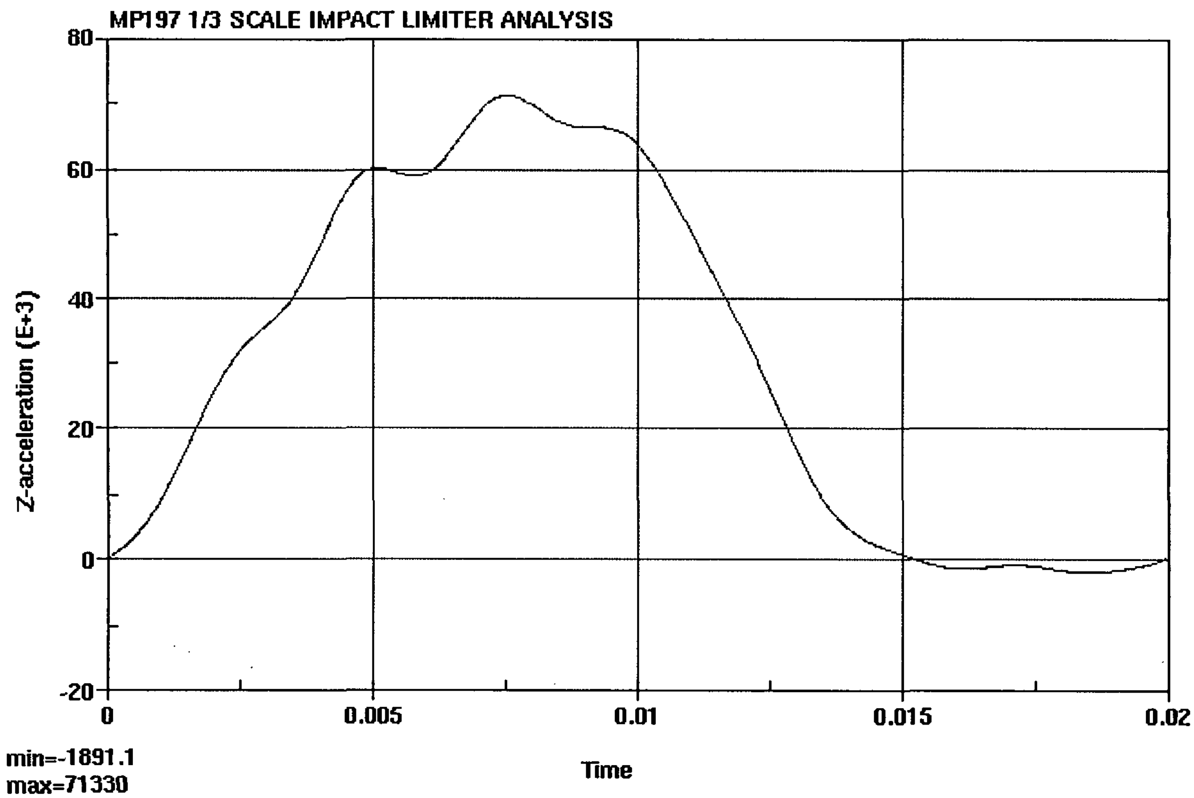
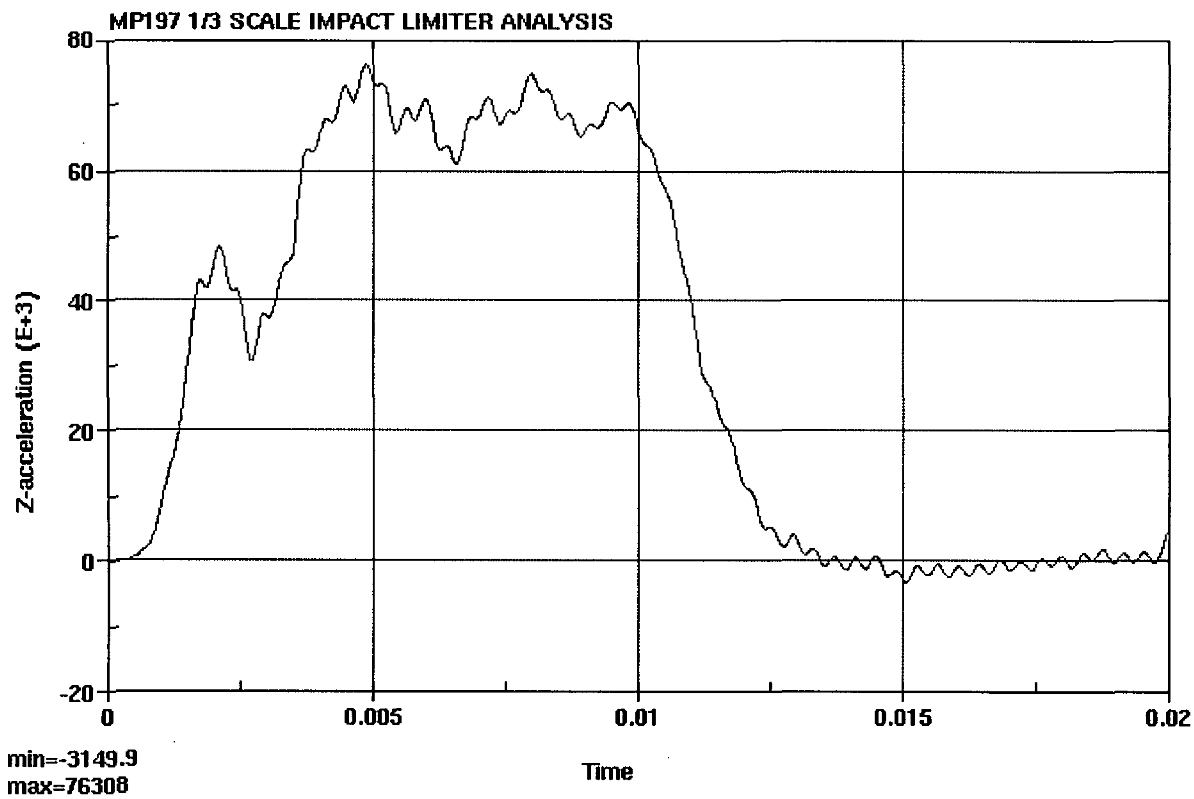


Figure A.2.13.12-8
Plot of Maximum Deformation for 1/3 Scale 20° Slap Down Drop
(Second Impact)



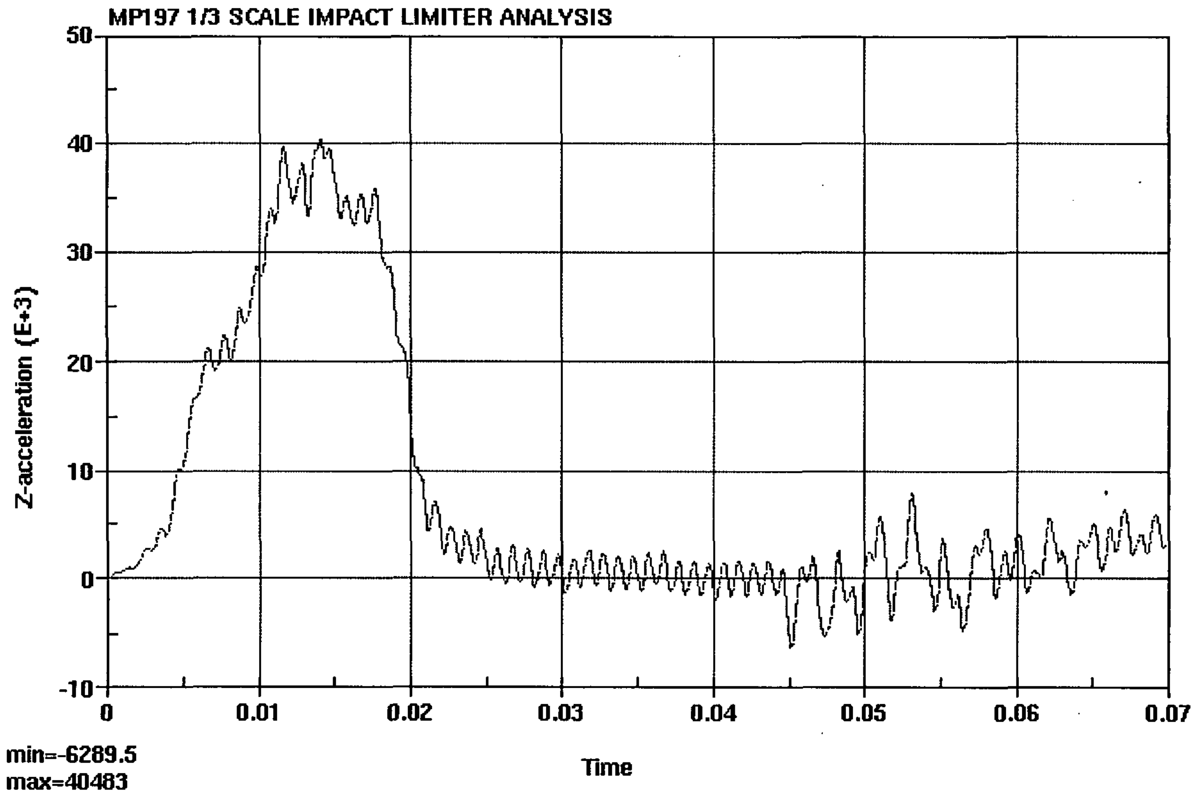
Note: The acceleration unit is in/sec^2 and unit for time is sec.

Figure A.2.13.12-9
1/3 Scale Side Drop Acceleration Time History
(From LS-DYNA)



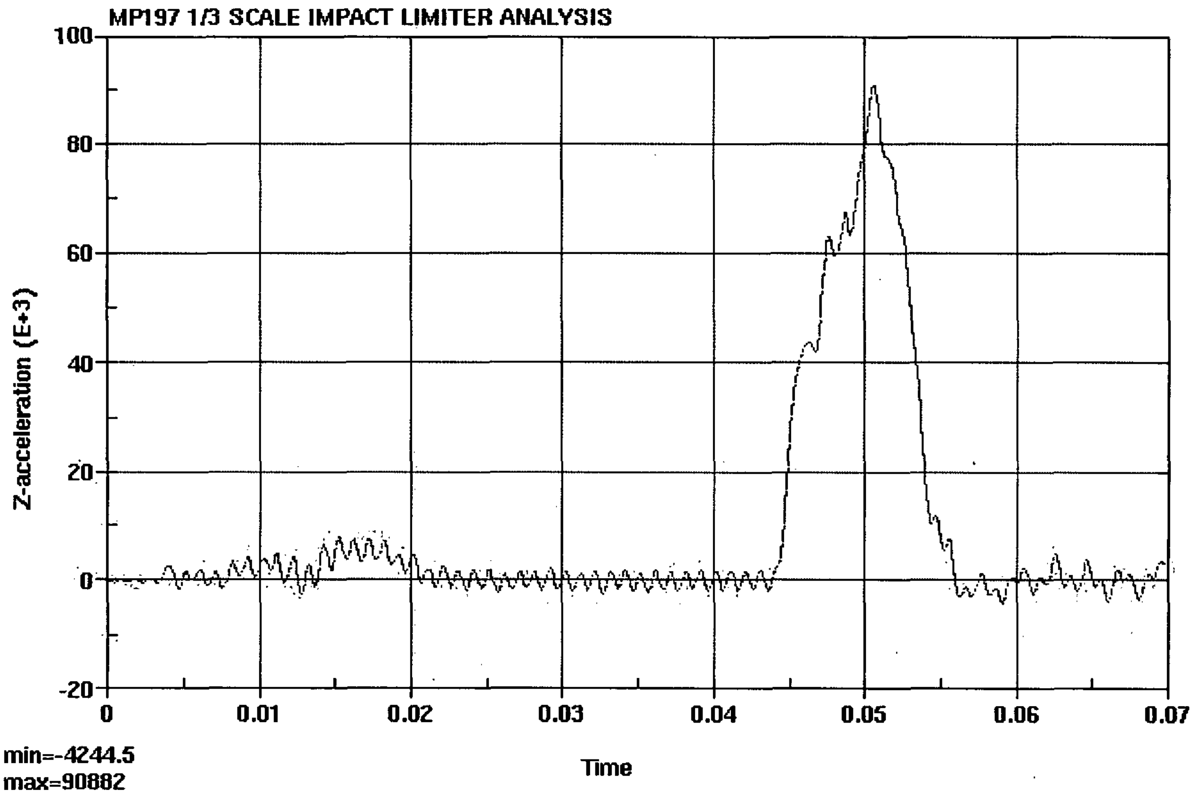
Note: The acceleration unit is in/sec^2 and unit for time is sec.

Figure A.2.13.12-12
1/3 Scale End Drop (-20°F) Acceleration Time History
(From LS-DYNA)



Note: The acceleration unit is in/sec^2 and unit for time is sec.

Figure A.2.13.12-15
1/3 Scale 20° Slap Down Drop Acceleration Time History
(First Impact-From LS-DYNA)



Note: The acceleration unit is in/sec^2 and unit for time is sec.

Figure A.2.13.12-16
1/3 Scale 20° Slap Down Drop Acceleration Time History
(Second Impact-From LS-DYNA)

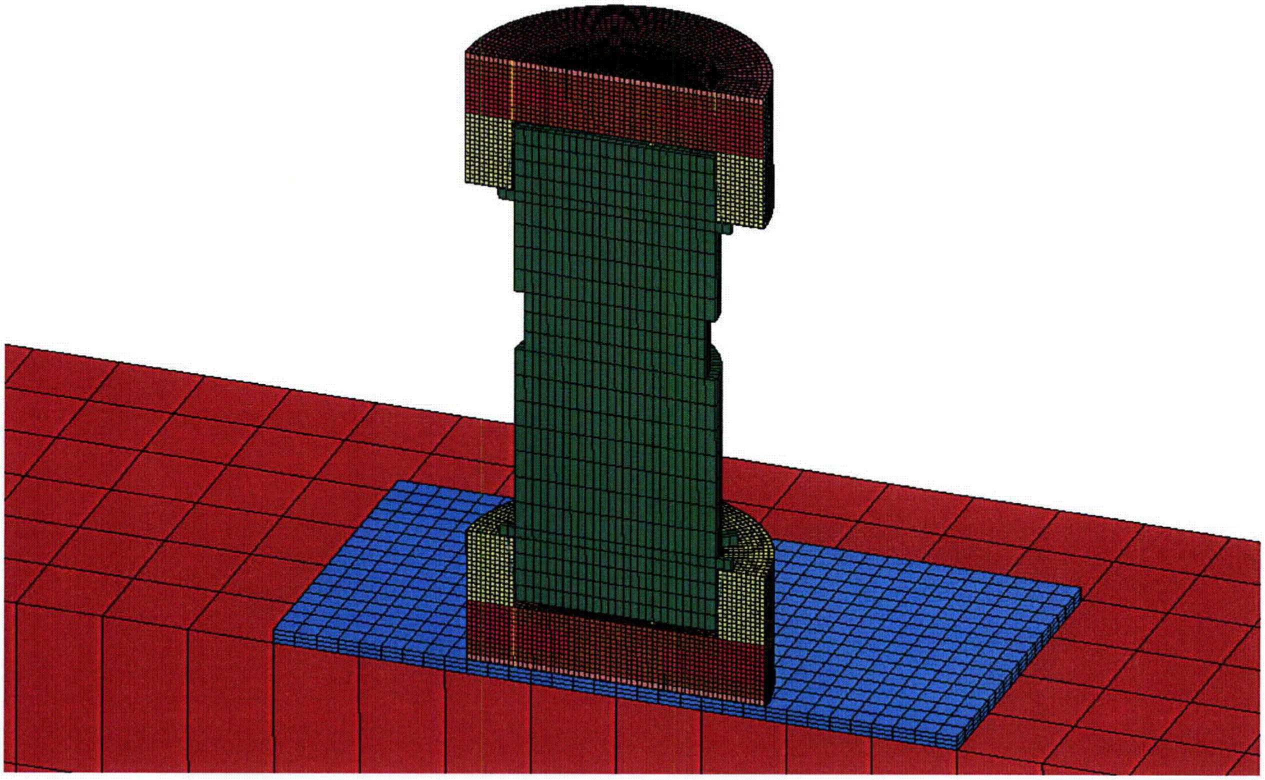


Figure A.2.13.12-19
Full Scale MP-197HB Impact Limiter Finite Element Model Overview

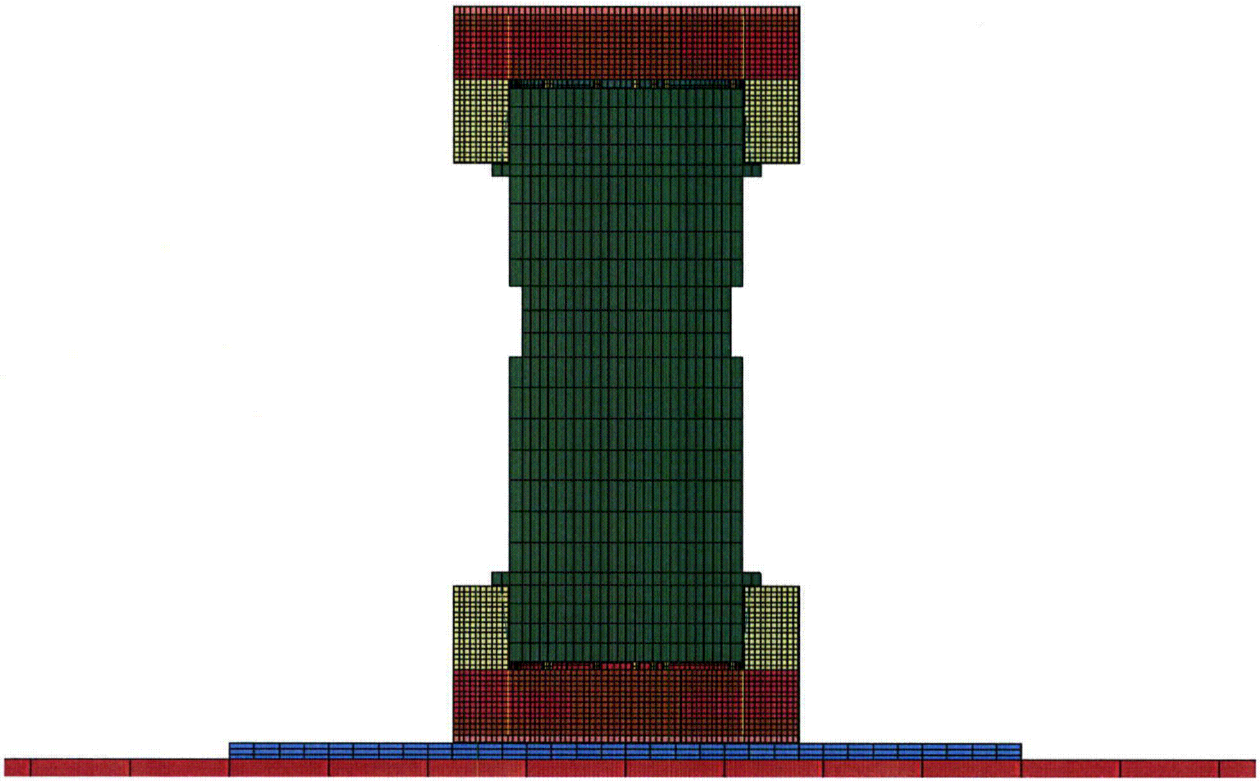


Figure A.2.13.12.-20
Full Scale MP-197HB Impact Limiter Finite Element Model for End Drop Orientation

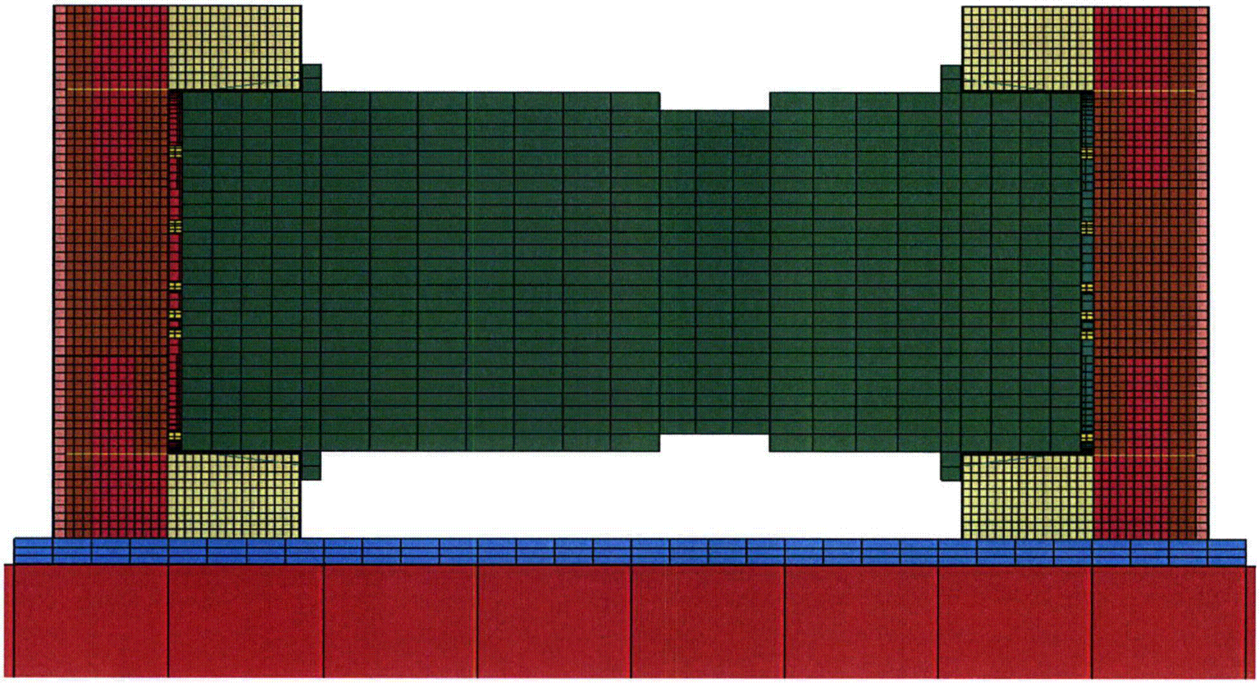


Figure A.2.13.12-21
Full Scale MP-197HB Impact Limiter Finite Element Model for Side Drop Orientation

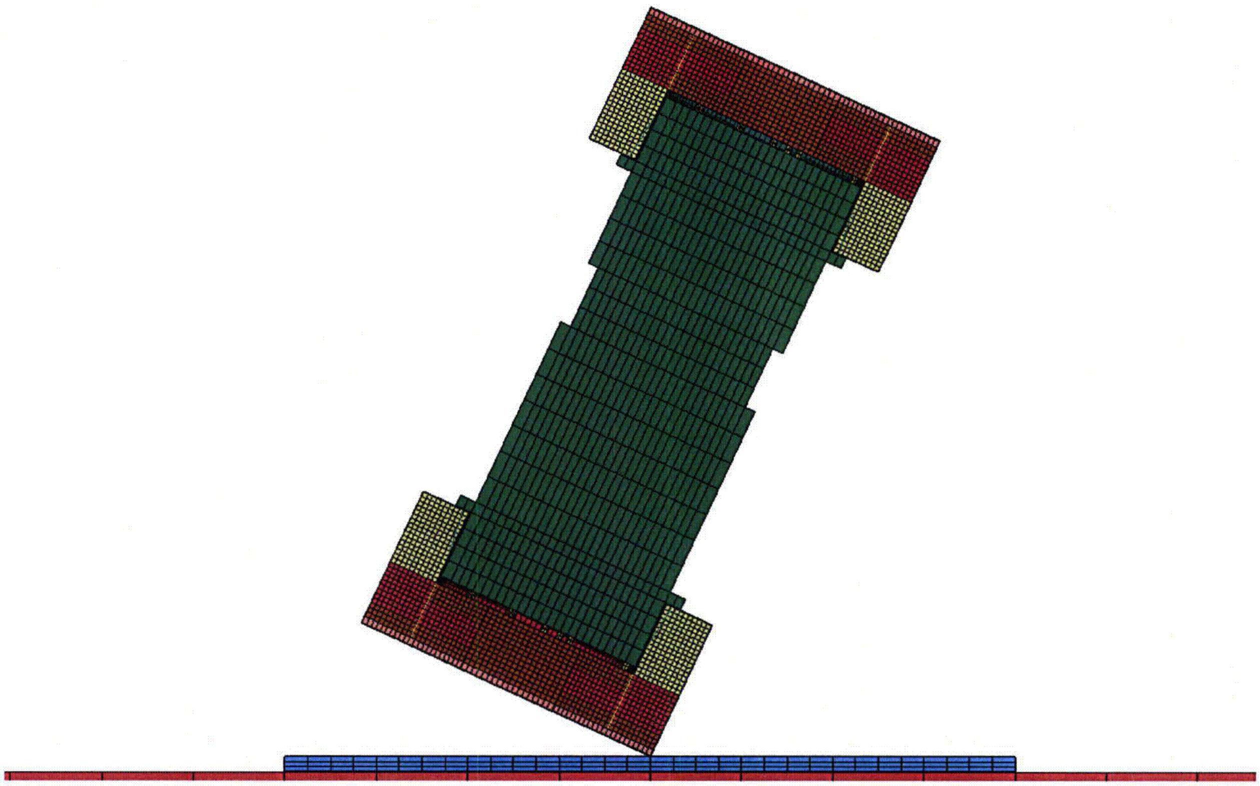


Figure A.2.13.12-22
Full Scale MP-197HB Impact Limiter Finite Element Model for CG Over Corner Orientation

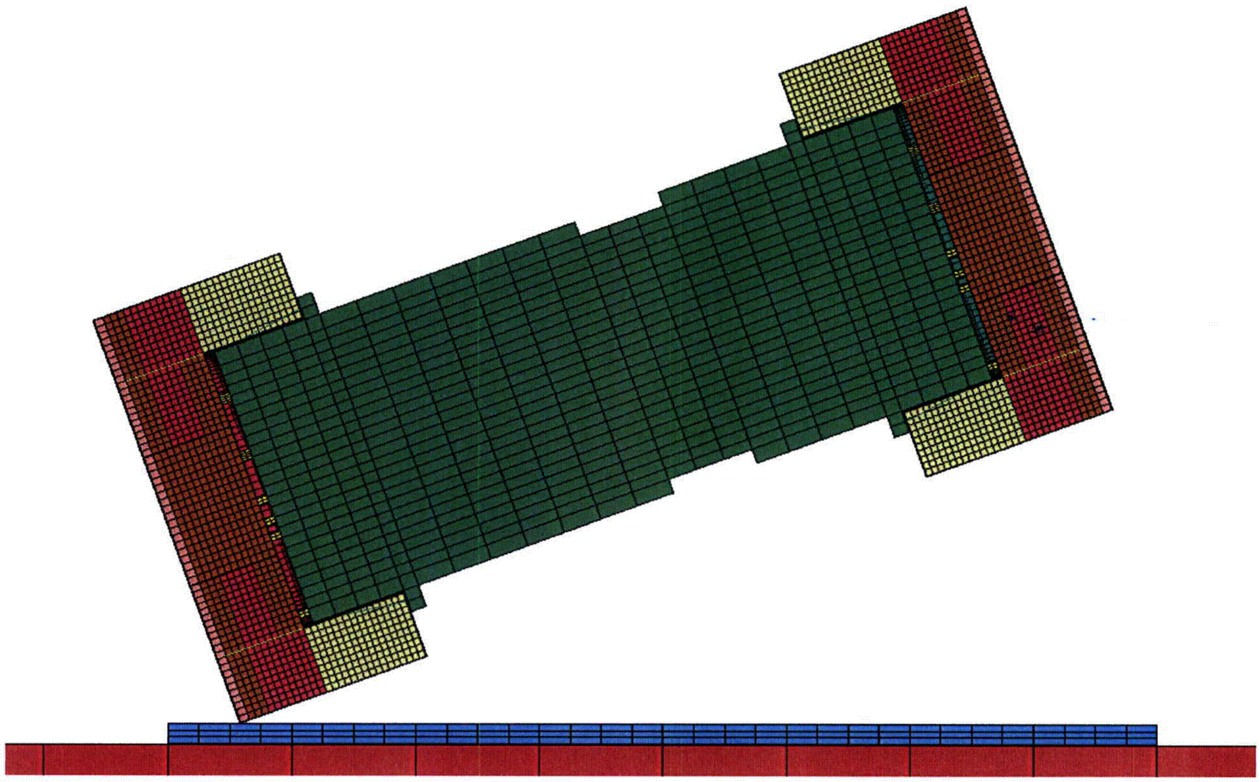


Figure A.2.13.12-23
Full Scale MP-197HB Impact Limiter Finite Element Model for 20° Slap Down Orientation

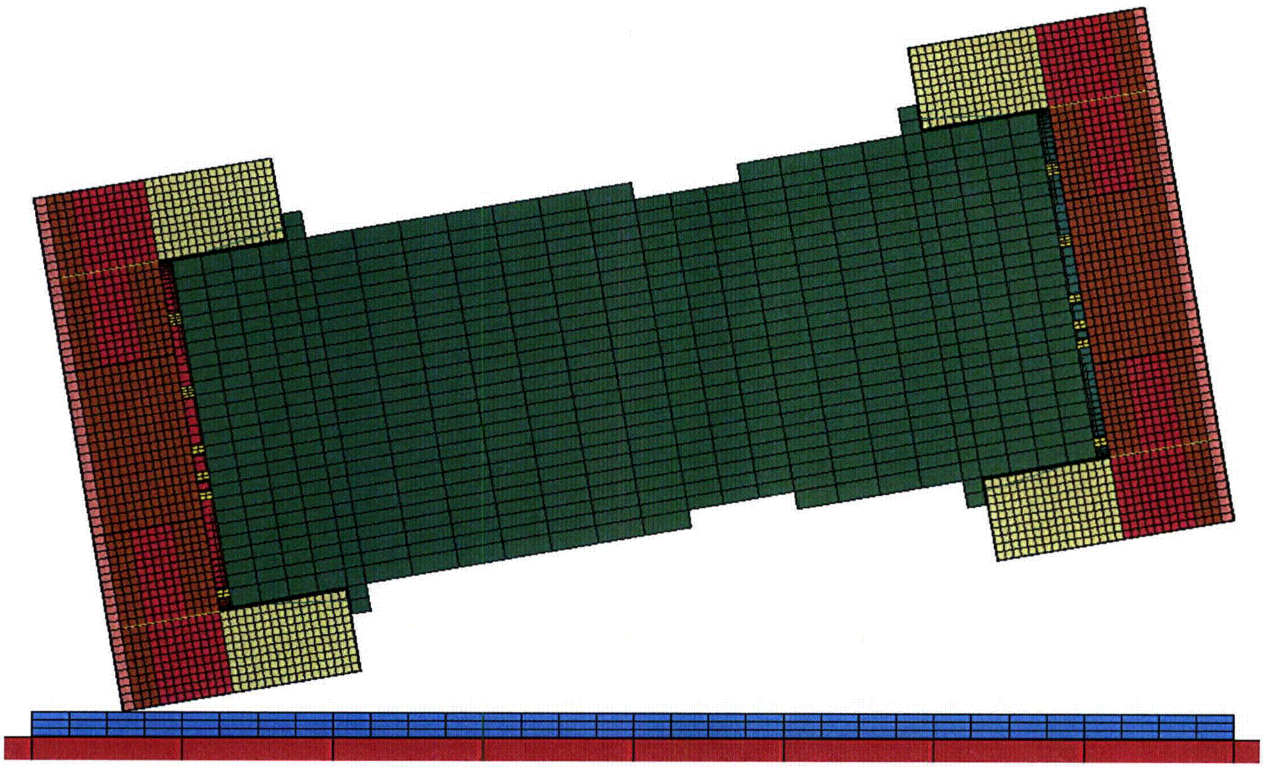


Figure A.2.13.12-24
Full Scale MP-197HB Impact Limiter Finite Element Model for 10° Slap Down Orientation

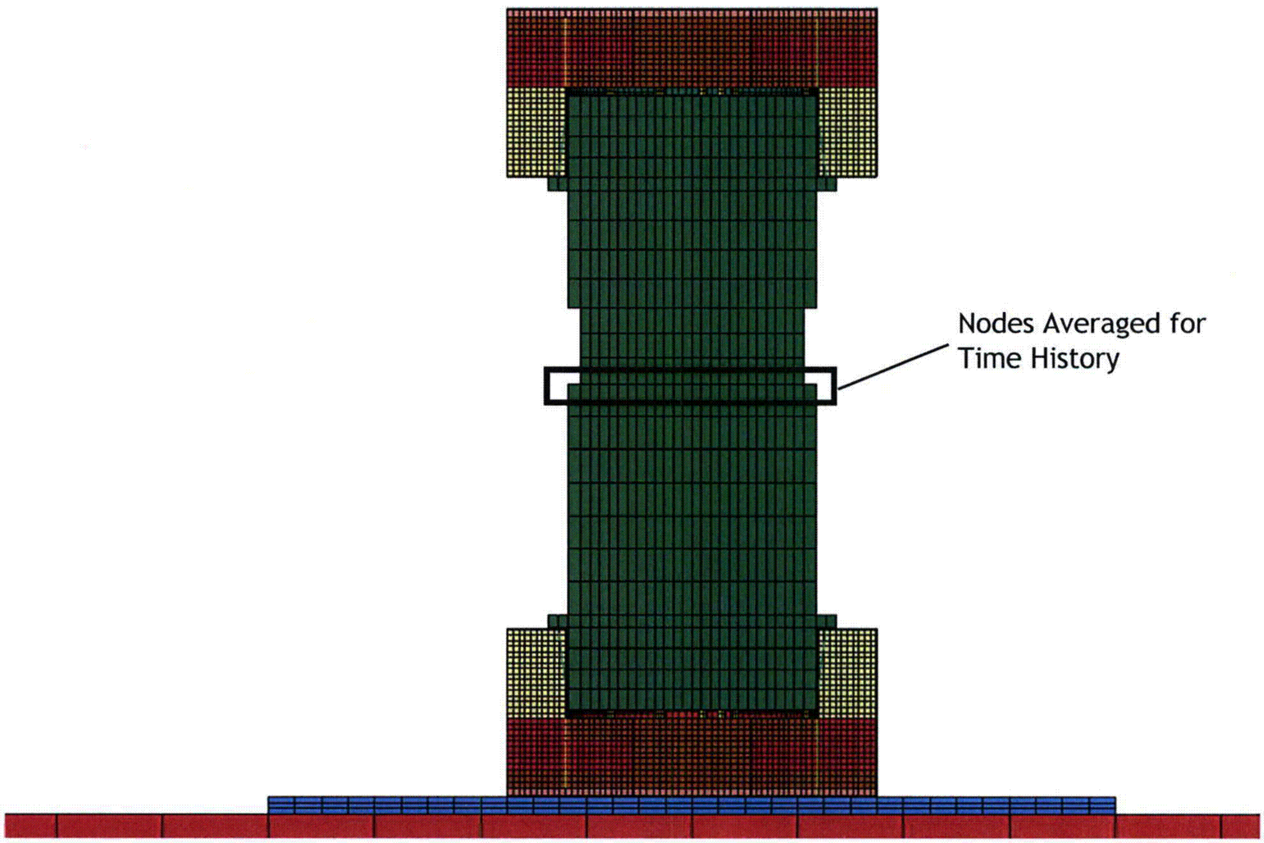


Figure A.2.13.12-25
Region of Nodes Averaged for Time History (All cases other than Slap Down)

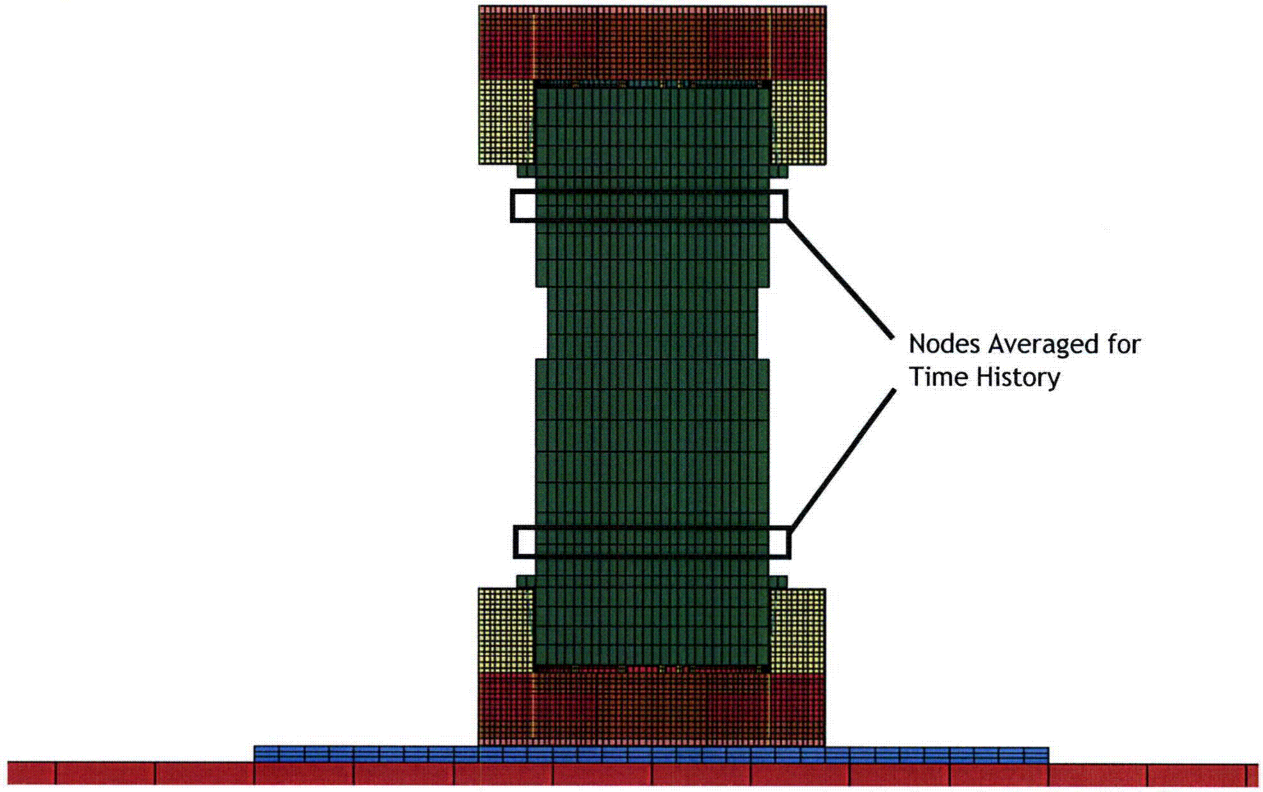


Figure A.2.13.12-26
Region of Nodes Averaged for Time History (Slap Down Cases)

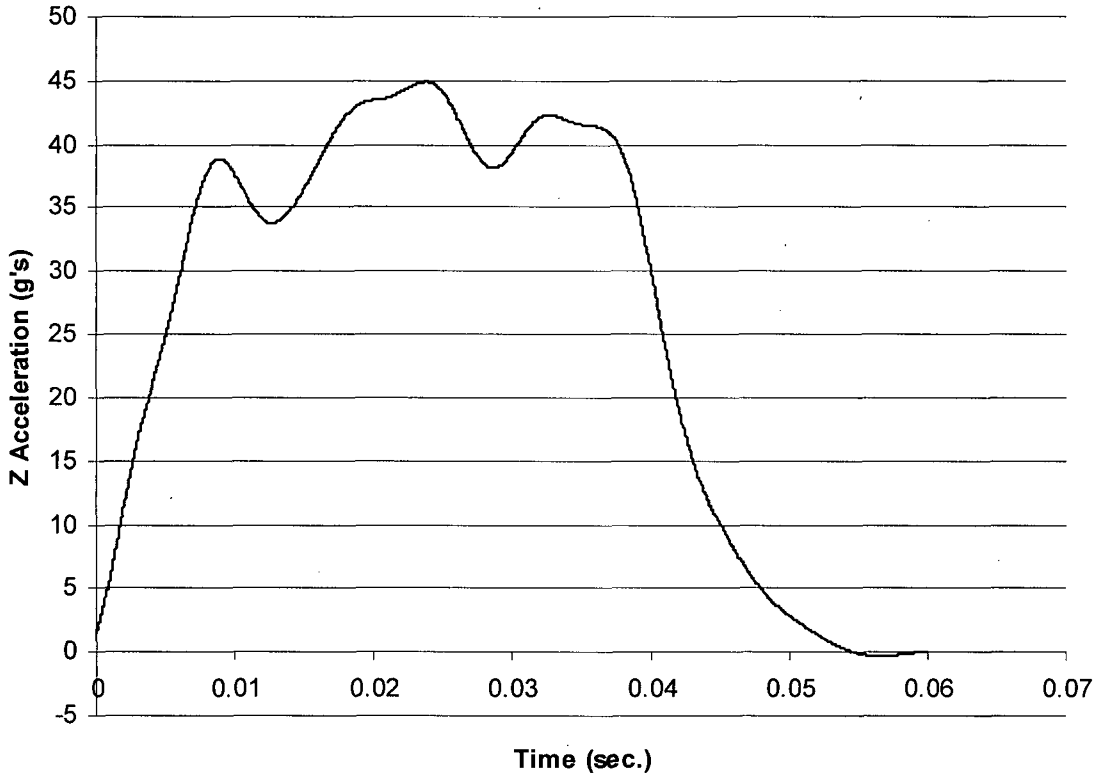


Figure A.2.13.12-27
Full Scale 30' End Drop Acceleration Time History (Room Temperature)

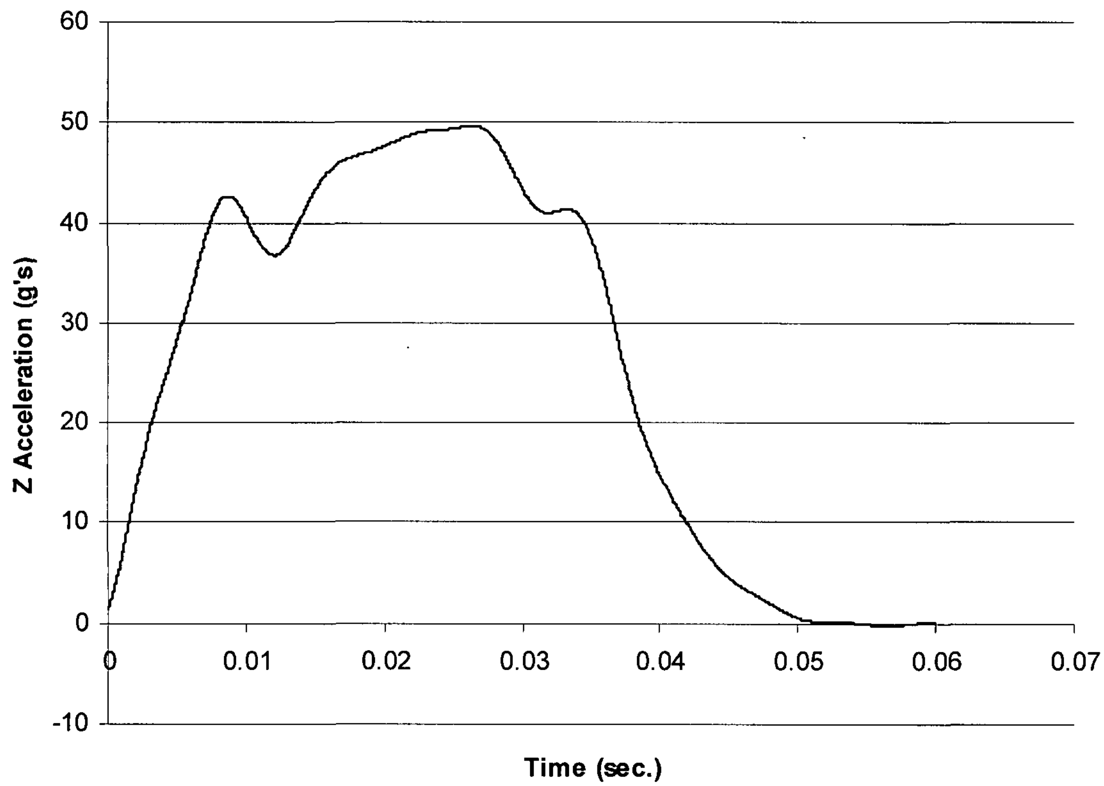


Figure A.2.13.12-28
Full Scale 30' End Drop Acceleration Time History (-20°F)

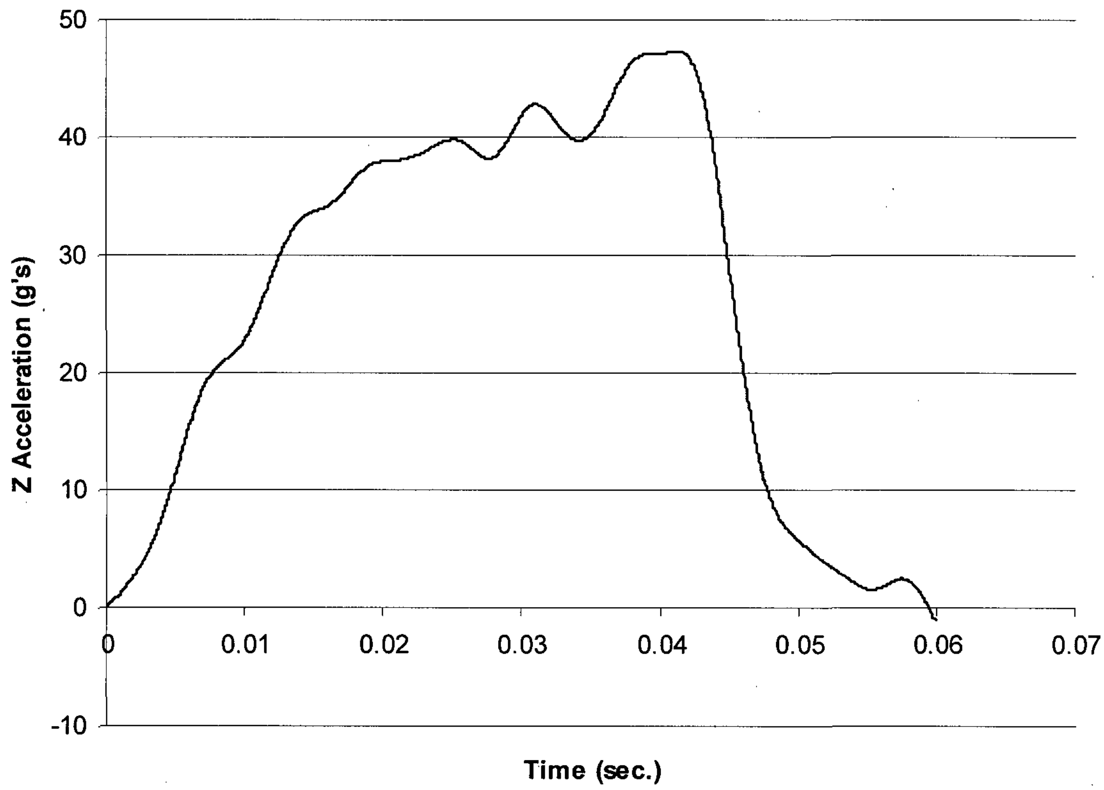


Figure A.2.13.12-29
Full Scale 30' Side Drop Acceleration Time History

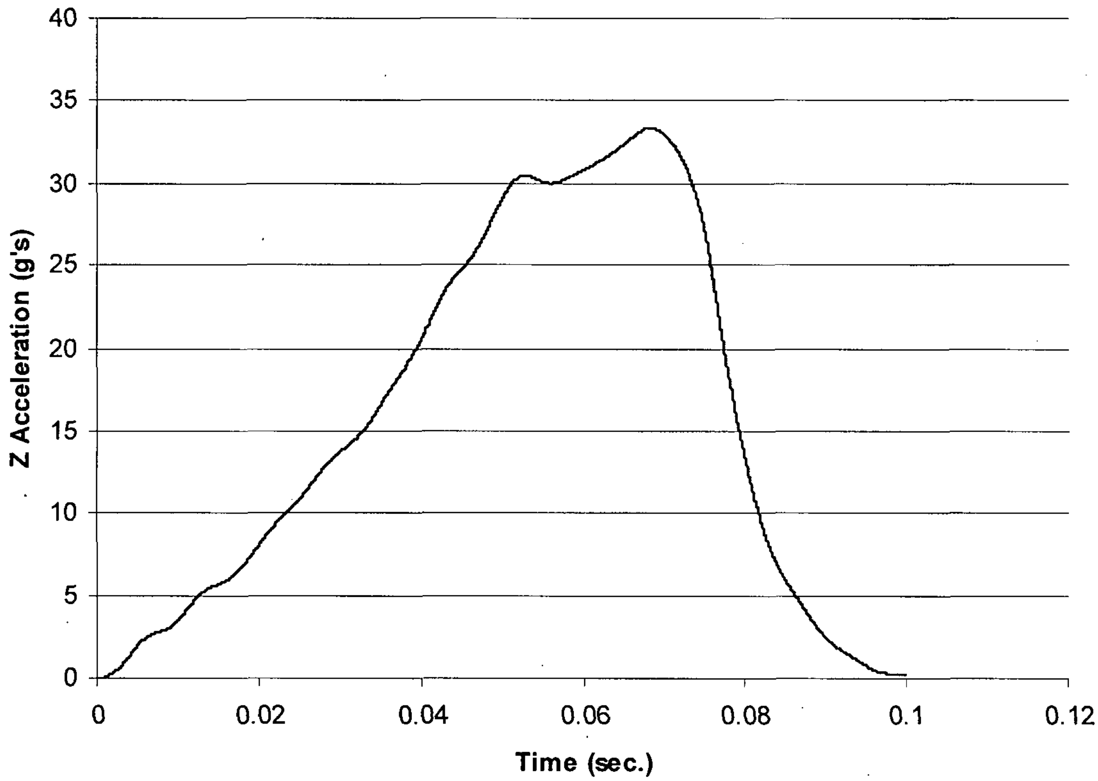


Figure A.2.13.12-30
Full Scale 30' CG Over Corner Drop Acceleration Time History

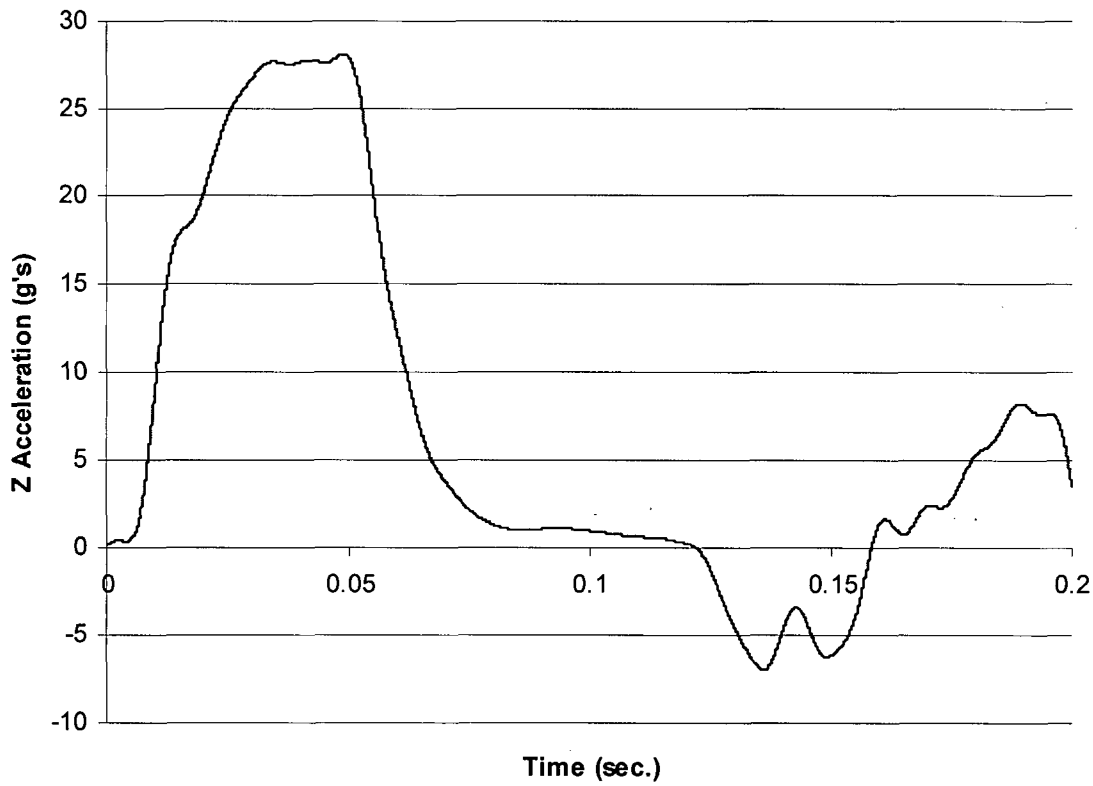


Figure A.2.13.12-31
Full Scale 30' 20° Slap Down Drop Acceleration Time History (First Impact)

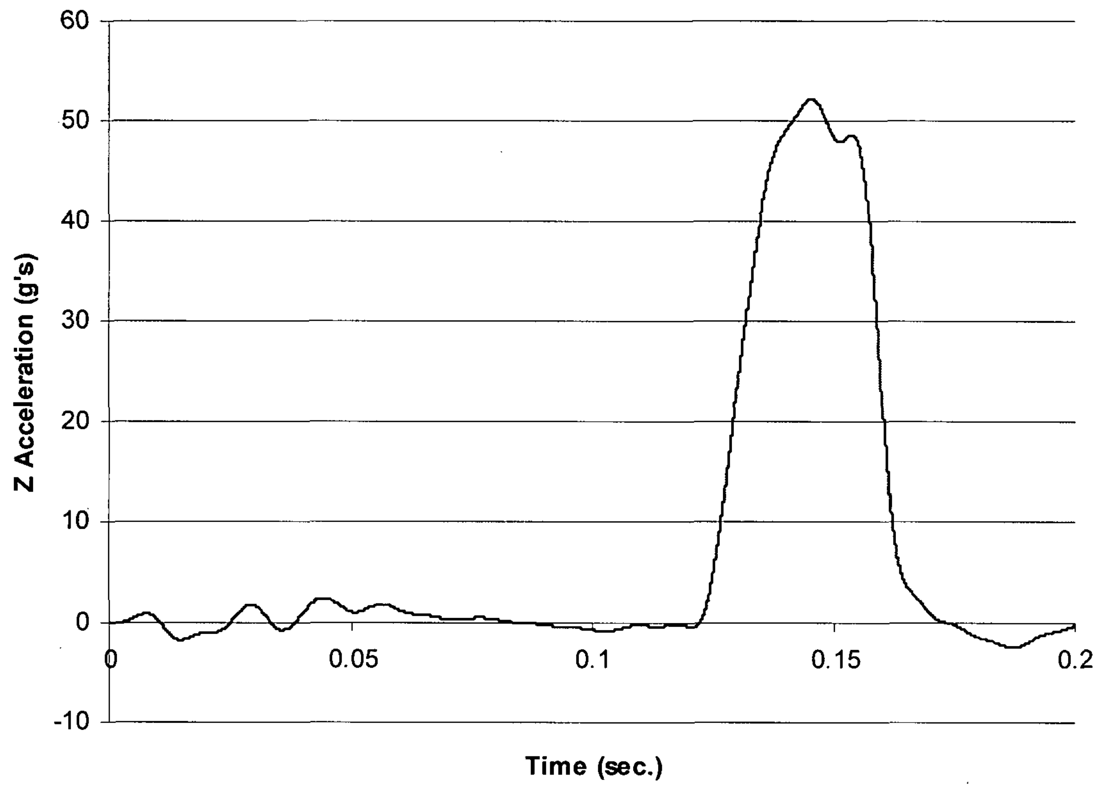


Figure A.2.13.12-32
Full Scale 30' 20° Slap Down Drop Acceleration Time History (Second Impact)

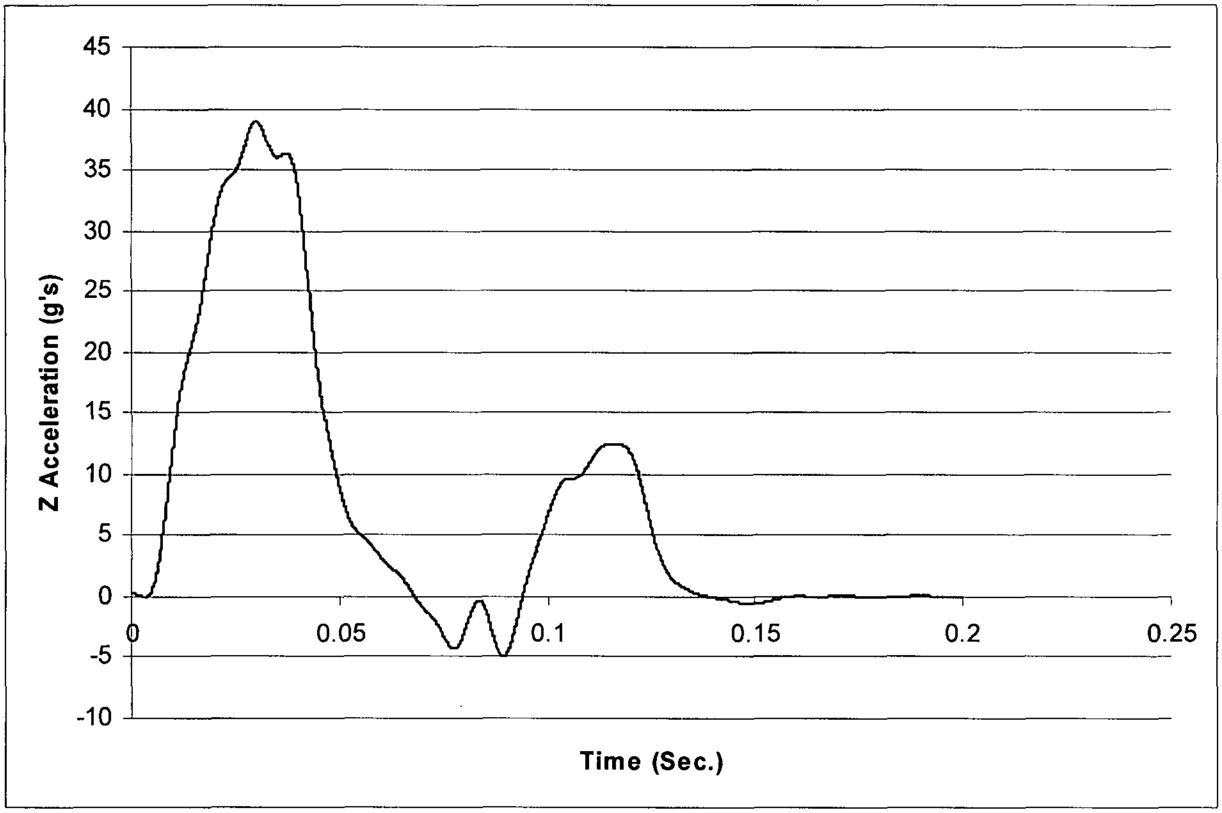


Figure A.2.13.12-33
Full Scale 30’ 10° Slap Down Drop Acceleration Time History (First Impact)

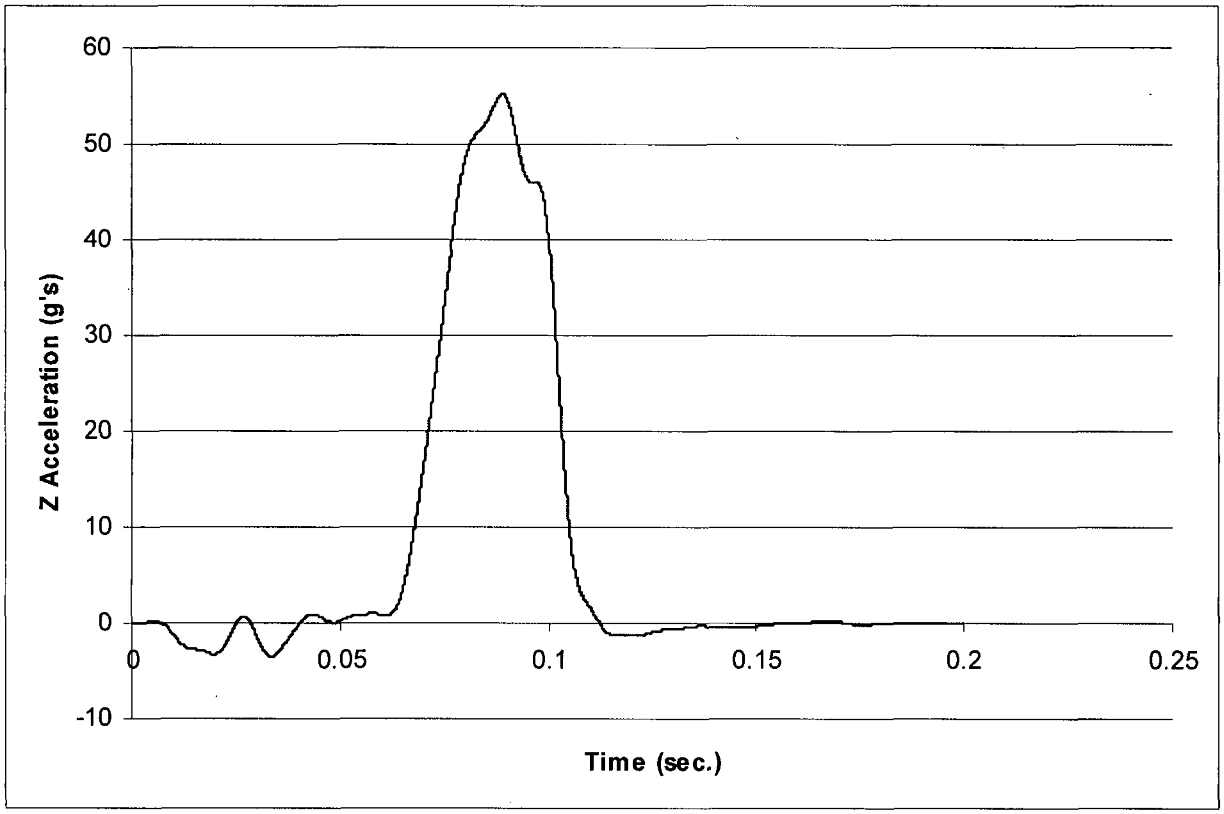


Figure A.2.13.12-34
Full Scale 30' 10° Slap Down Drop Acceleration Time History (*Second* Impact)

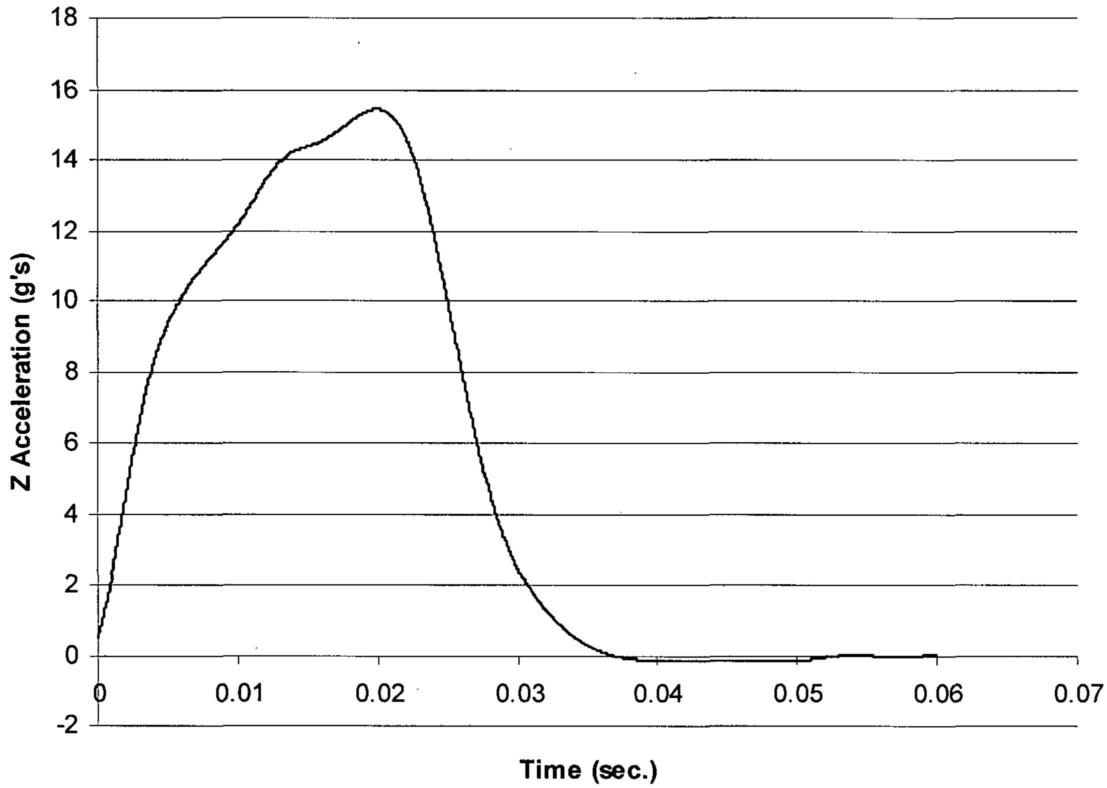


Figure A.2.13.12-35
Full Scale 1 Foot Normal Condition End Drop Acceleration Time History

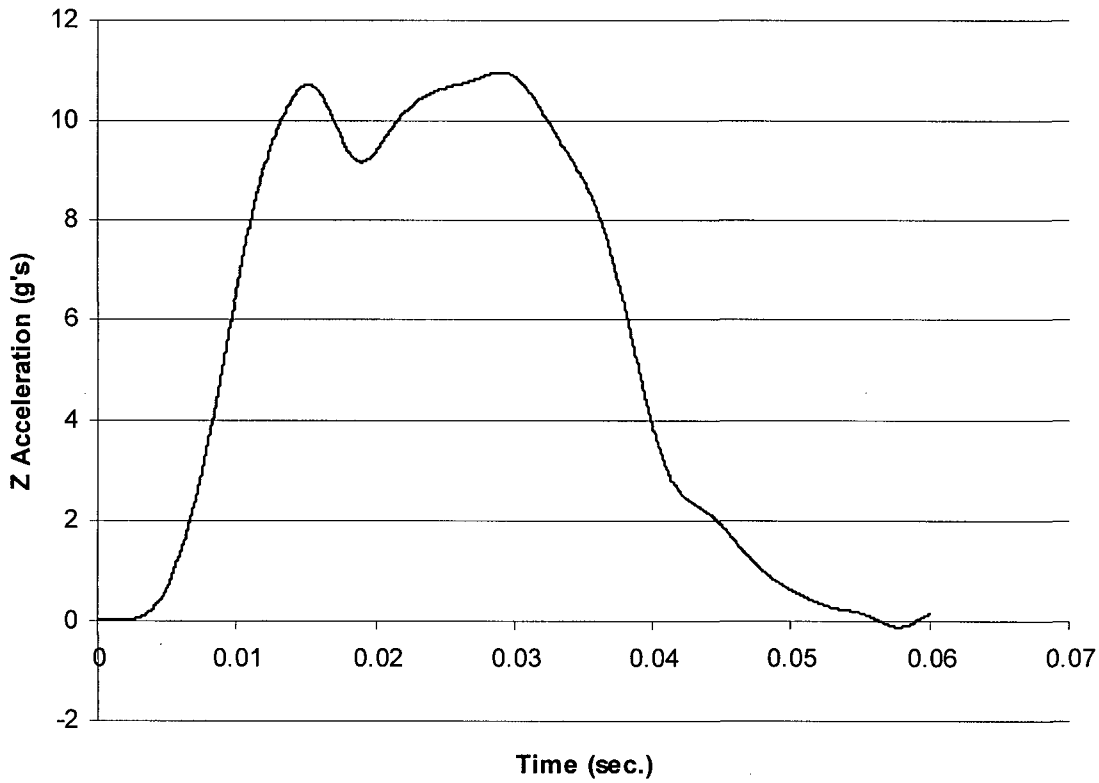


Figure A.2.13.12-36
Full Scale 1 Foot Normal Condition Side Drop Acceleration Time History

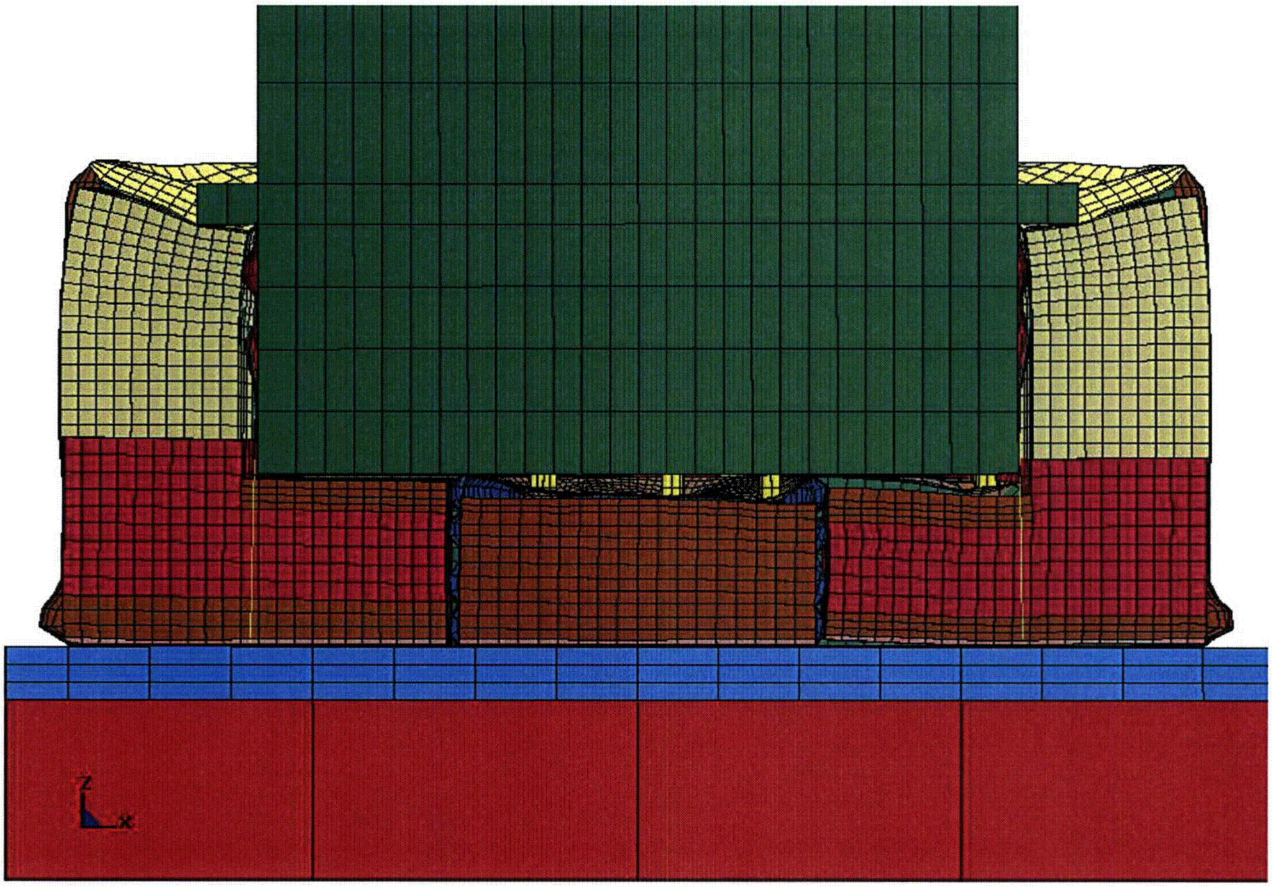


Figure A.2.13.12-37
Plot of Maximum Deformation for 30' End Drop- Full Scale

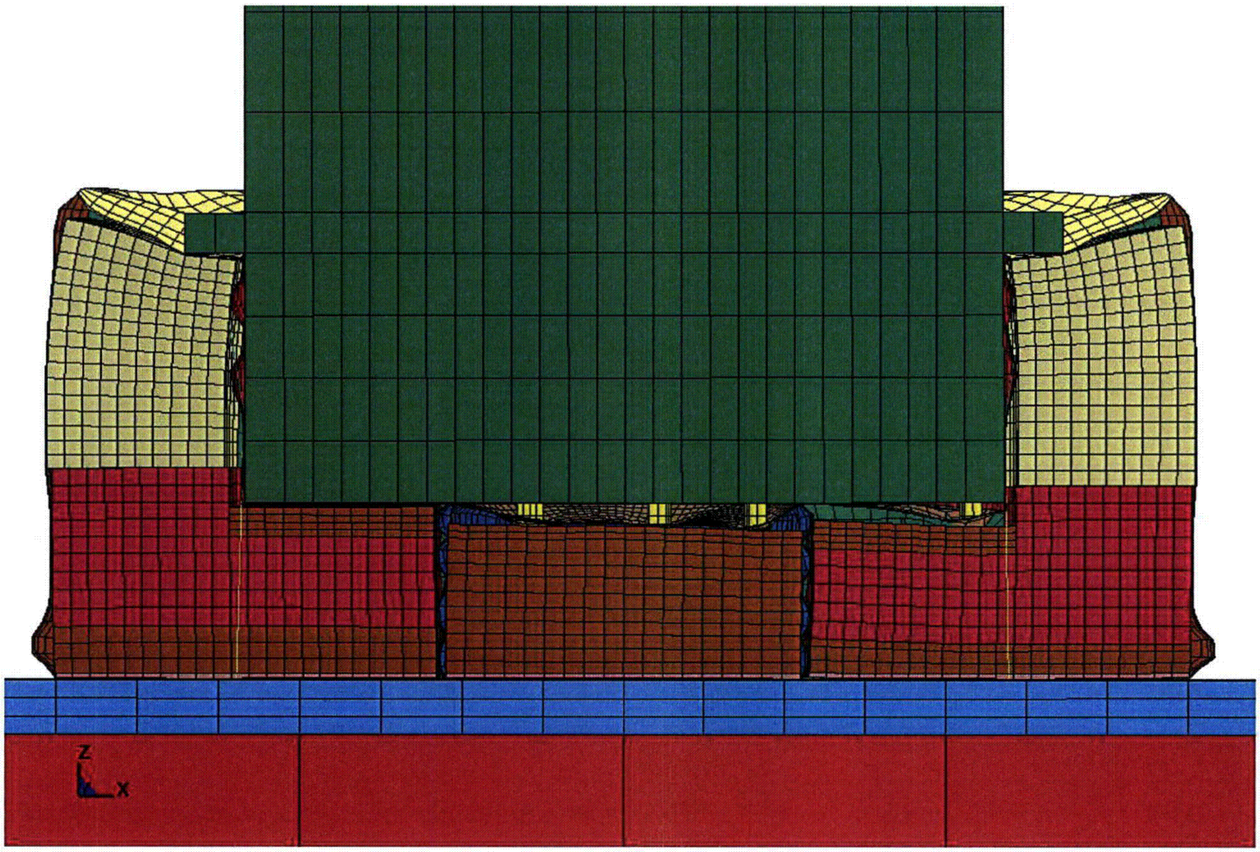


Figure A.2.13.12-38
Plot of Maximum Deformation for 30' End Drop (-20ºF) – Full Scale

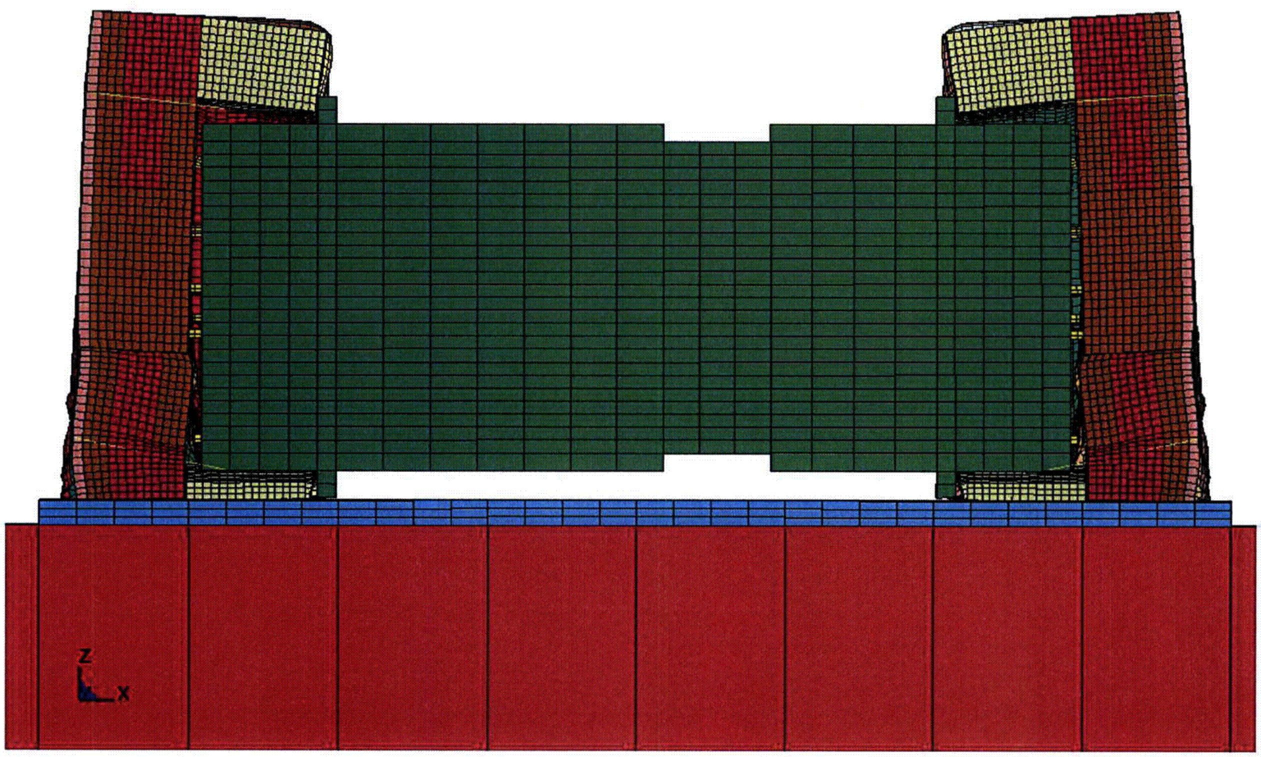


Figure A.2.13.12-39
Plot of Maximum Deformation for 30' Side Drop – Full Scale

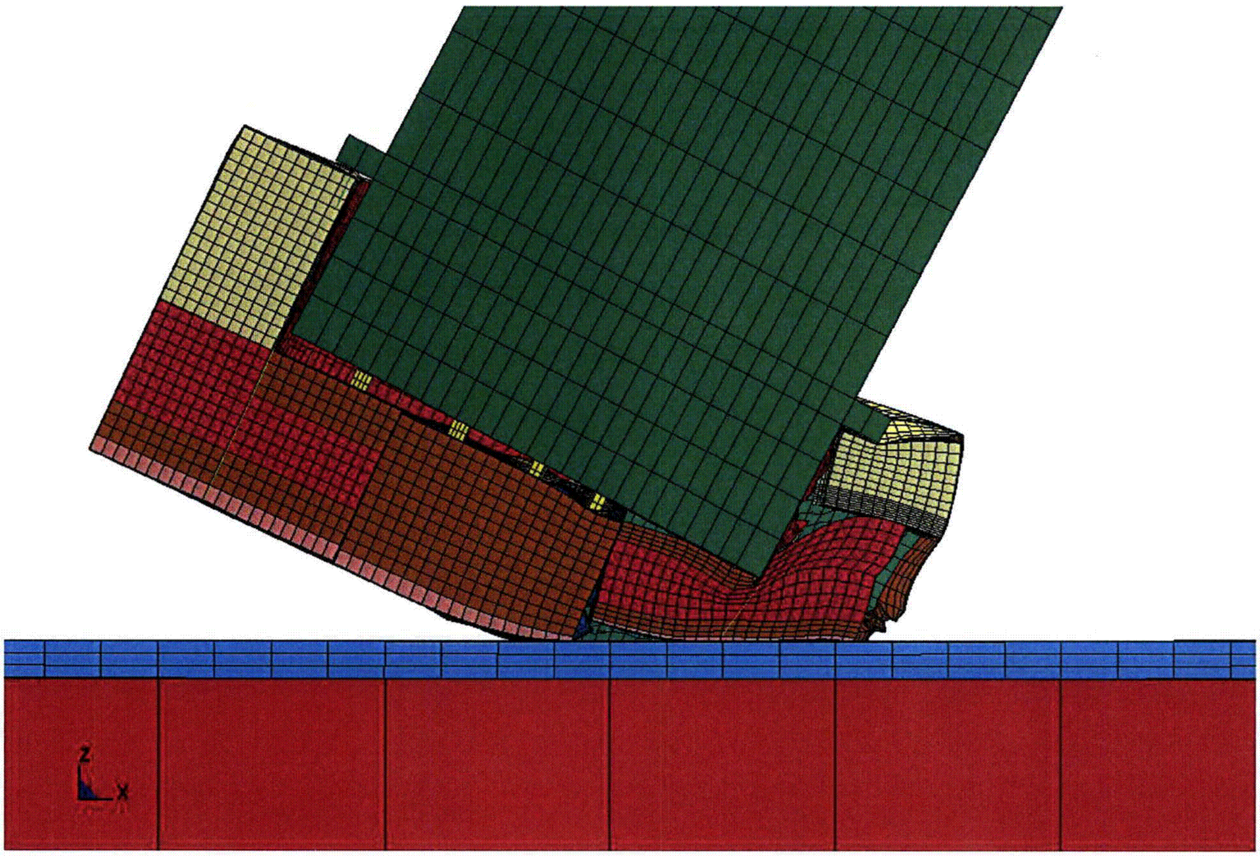


Figure A.2.13.12-40
Plot of Maximum Deformation for 30' CG Over Corner Drop – Full Scale

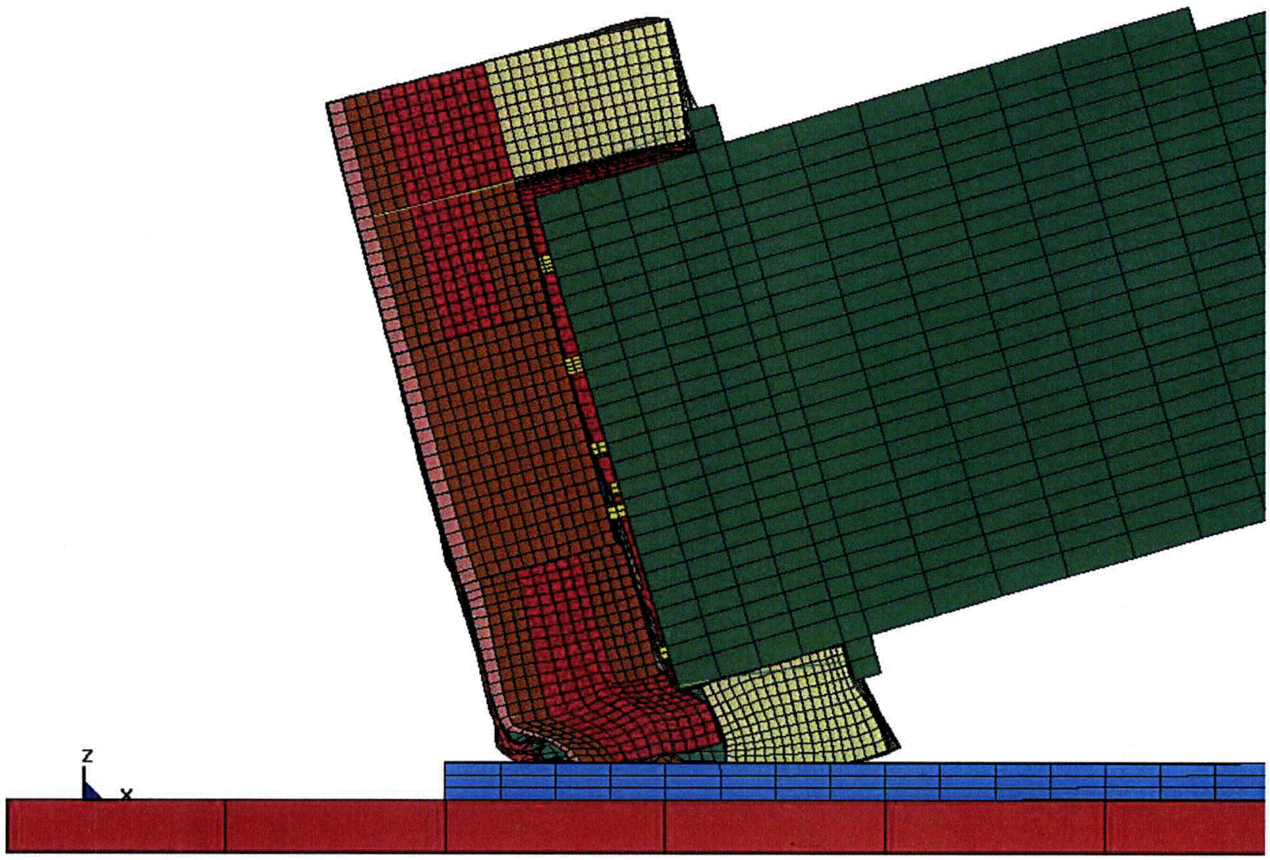


Figure A.2.13.12-41
Plot of Maximum Deformation for 30' 20° Slap Down Drop – Full Scale (First Impact)

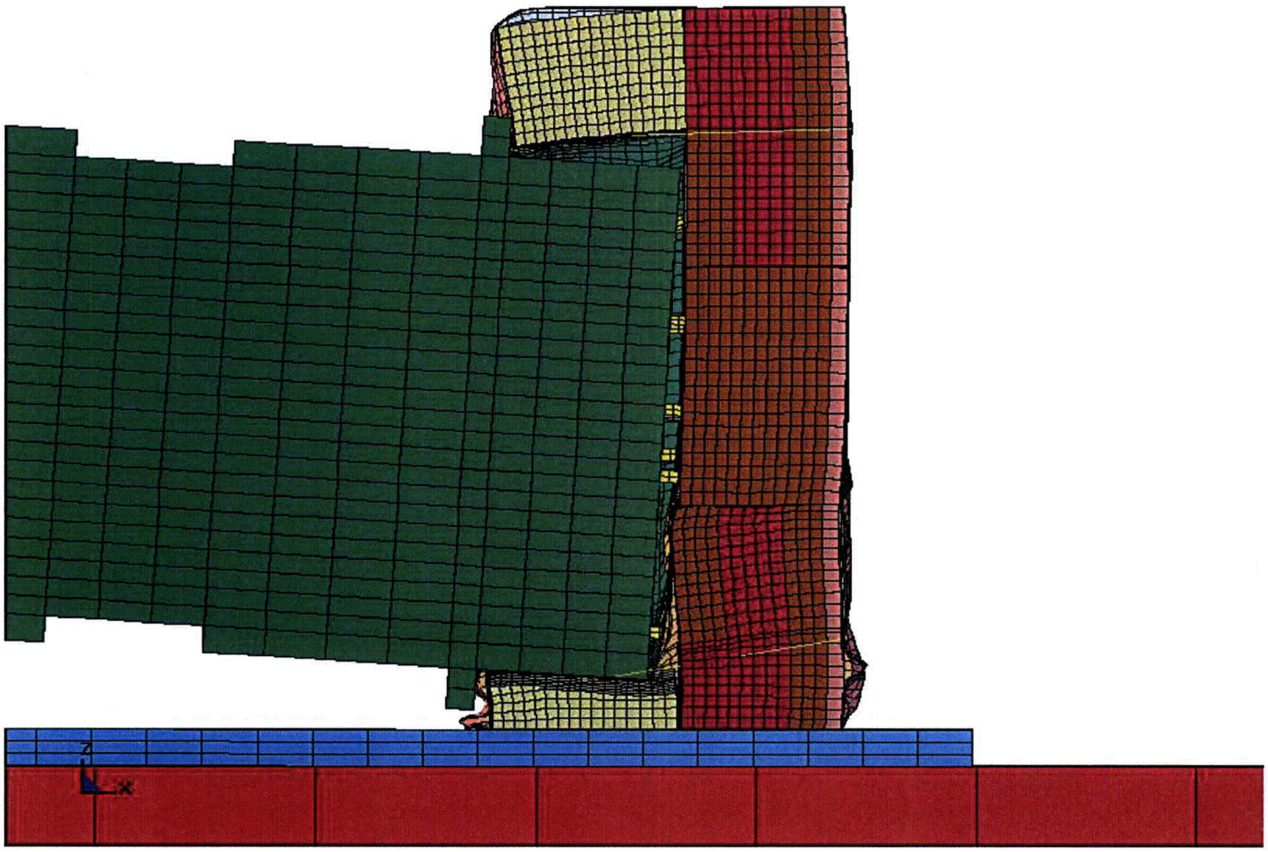


Figure A.2.13.12-42
Plot of Maximum Deformation for 30' 20° Slap Down Drop – Full Scale (Second Impact)

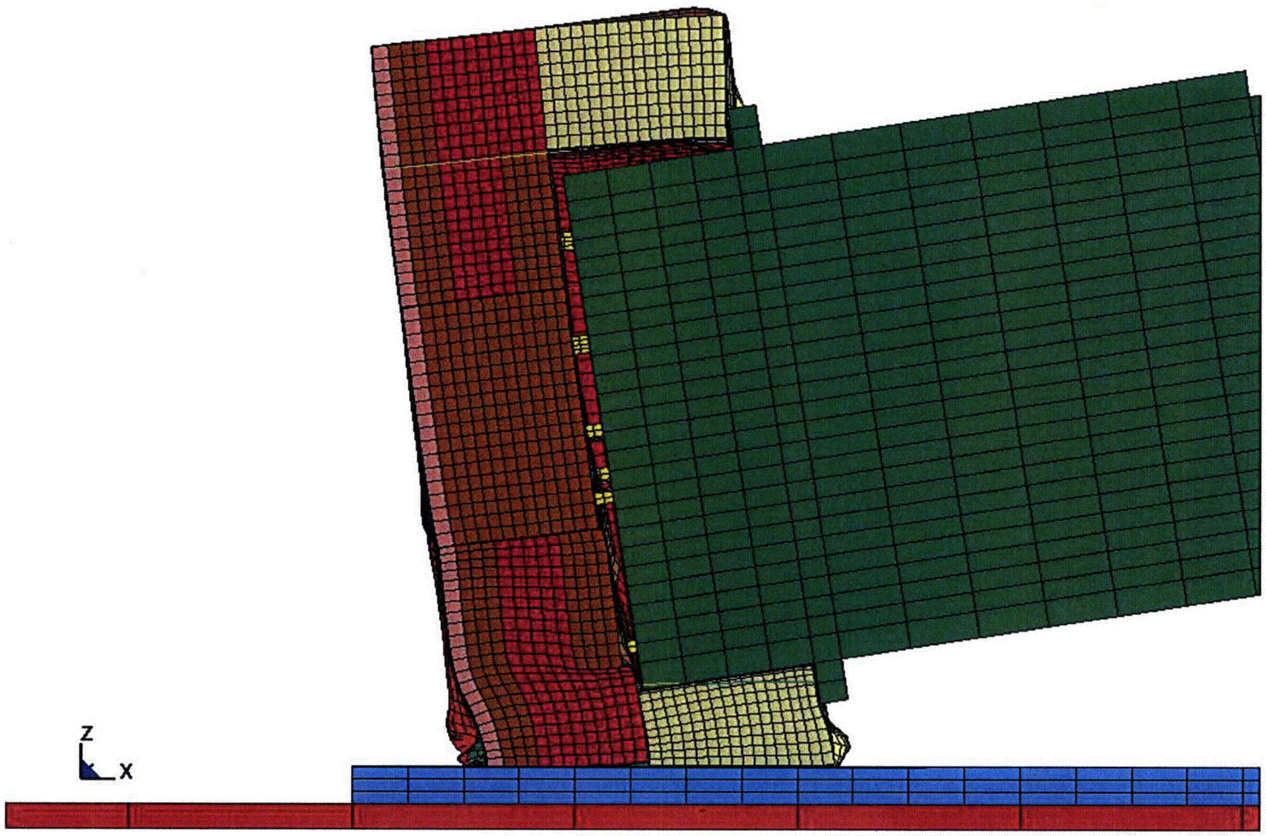


Figure A.2.13.12-43
Plot of Maximum Deformation for 30' 10° Slap Down Drop–Full Scale (First Impact)

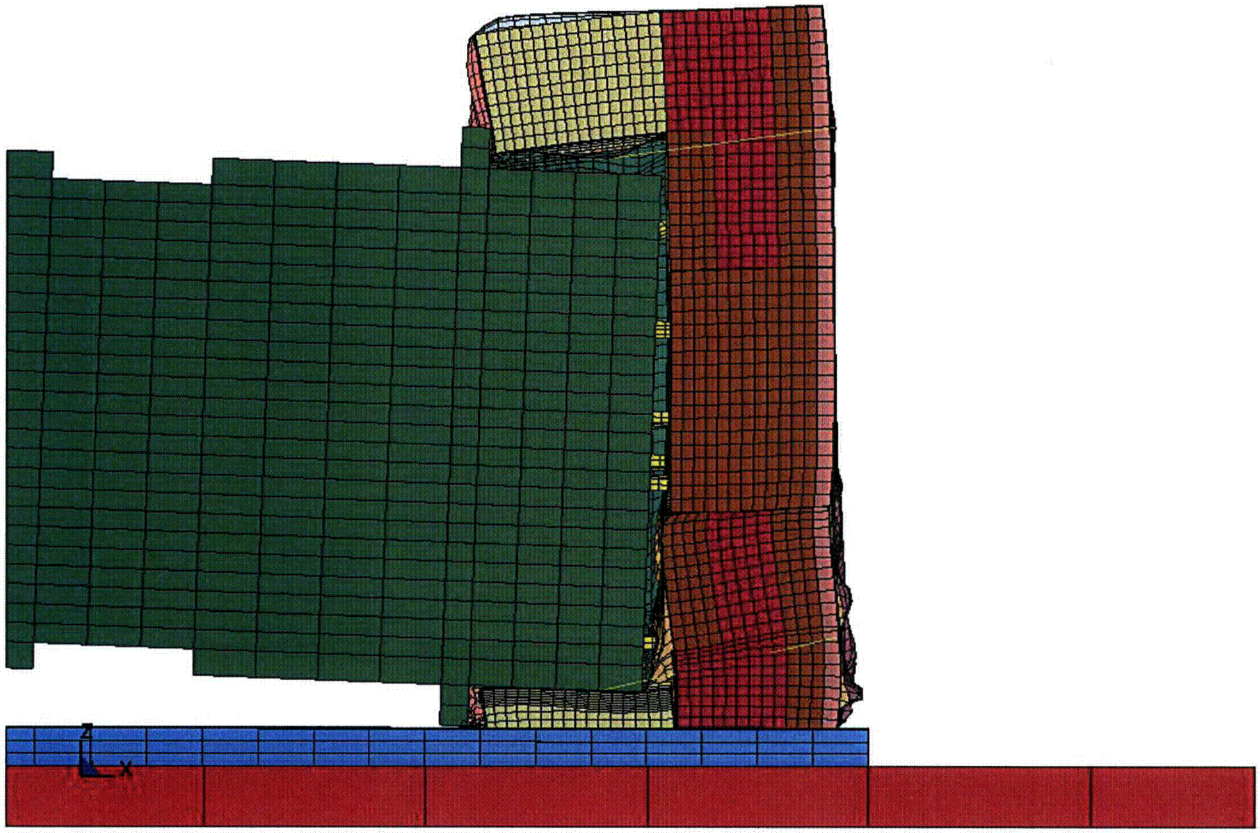


Figure A.2.13.12-44
Plot of Maximum Deformation for 30' 10° Slap Down Drop–Full Scale (Second Impact)

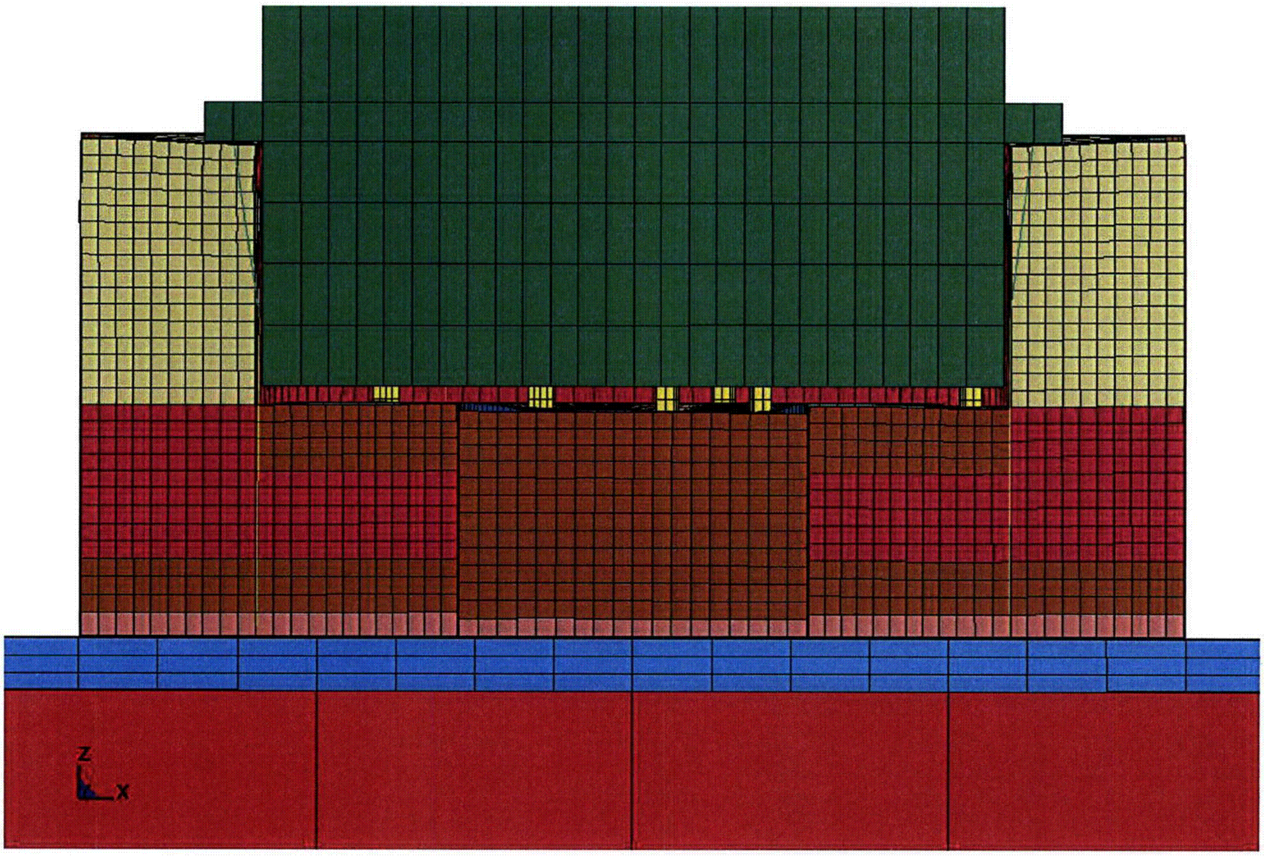


Figure A.2.13.12-45
Plot of Maximum Deformation for 1 Foot Normal Condition End Drop – Full Scale

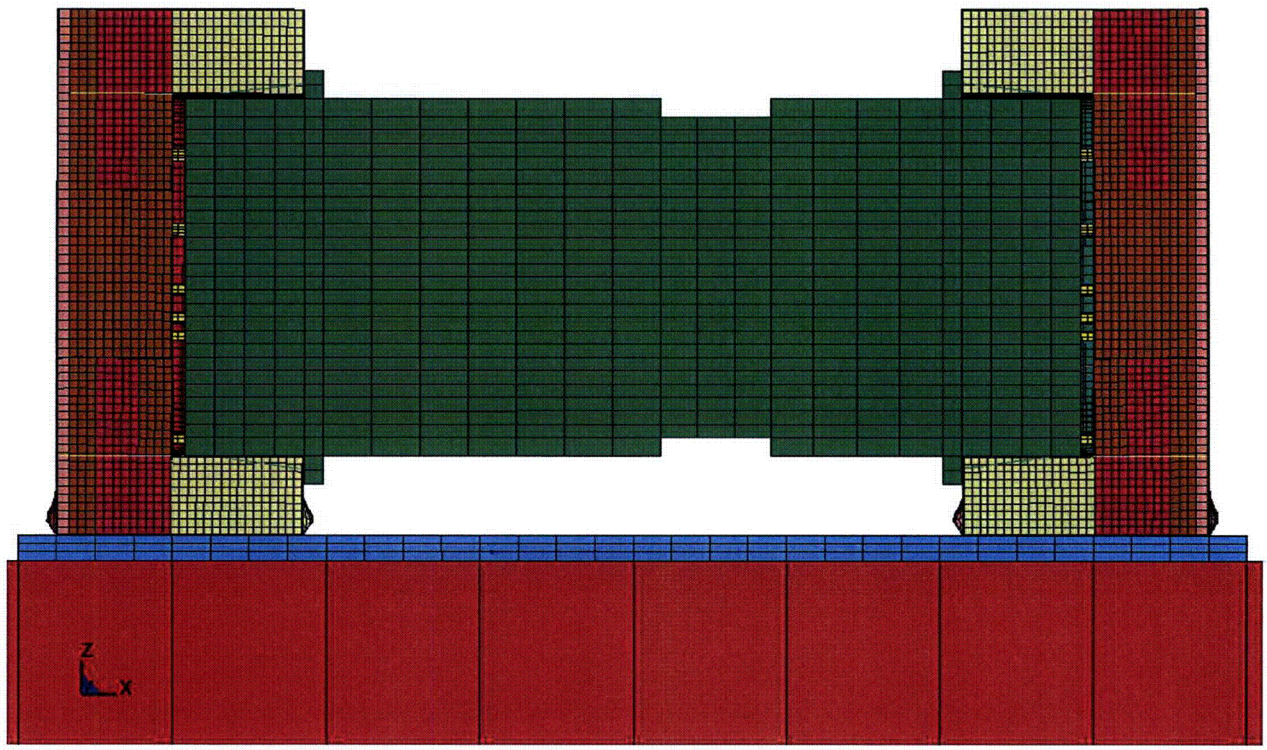


Figure A.2.13.12-46
Plot of Maximum Deformation for 1 Foot Normal Condition Side Drop – Full Scale

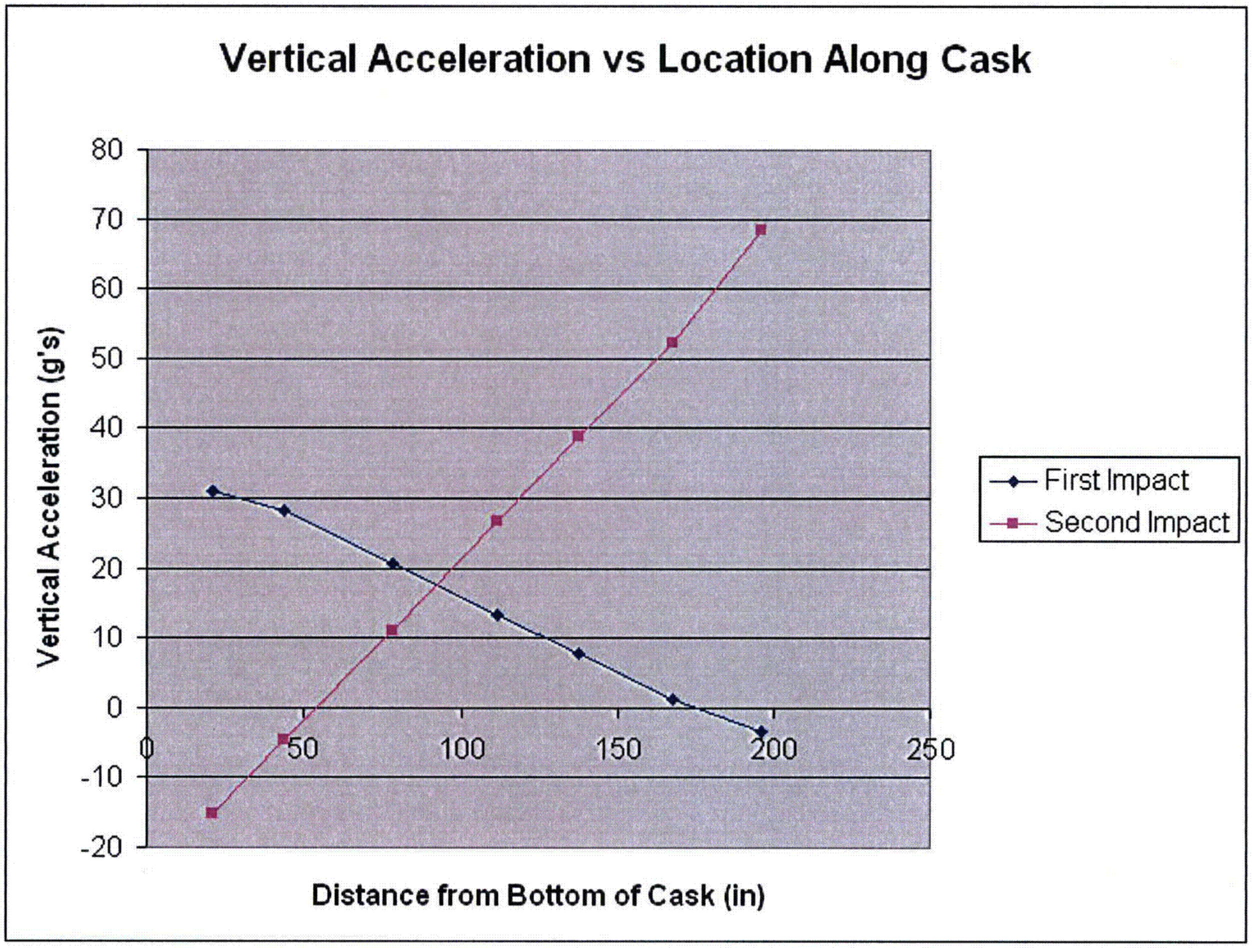


Figure A.2.13.12-48
Vertical Acceleration Along Cask During First And Second Impact to Compute Rotational Acceleration for 20° Slap Down Drop

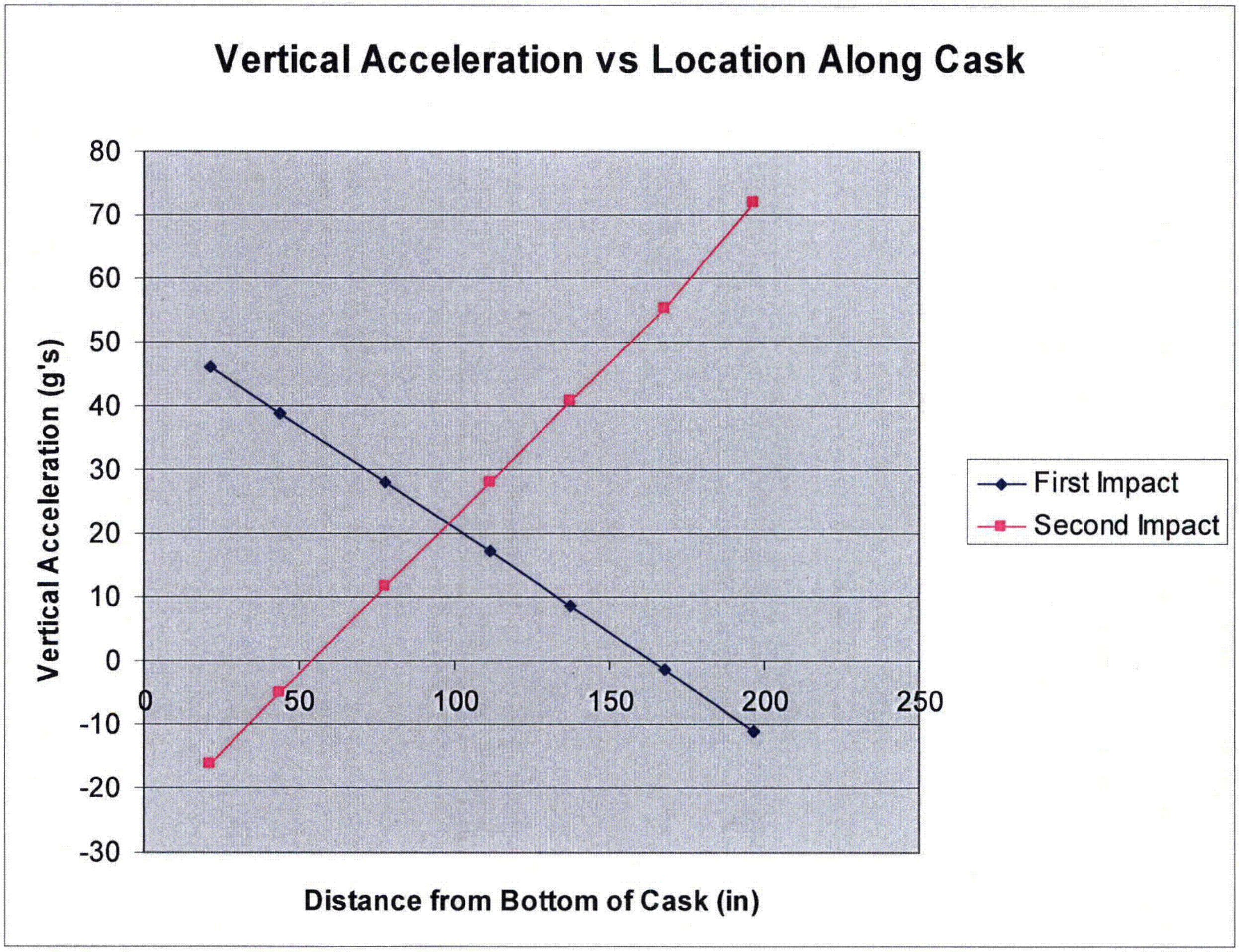


Figure A.2.13.12-49
Vertical Acceleration Along Cask During First And Second Impact to Compute Rotational Acceleration for 10° Slap Down

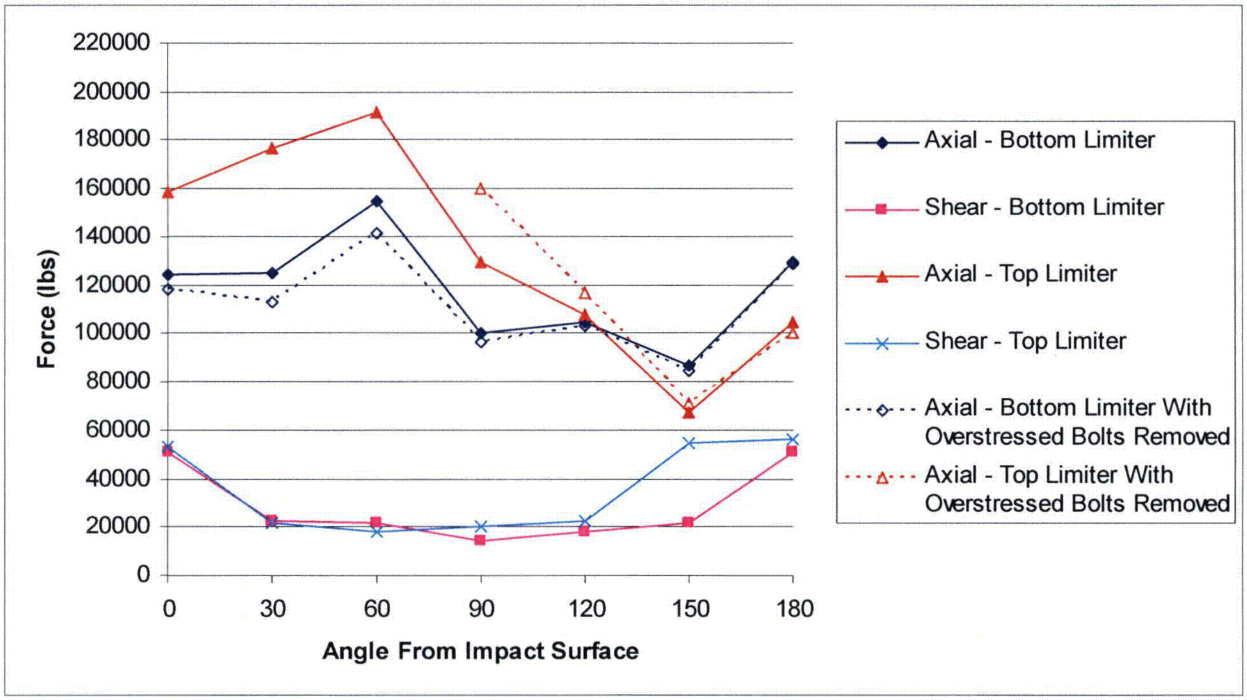


Figure A.2.13.12-50
Bolt Forces Around Top and Bottom Impact Limiters for 20° Slap Down Drop

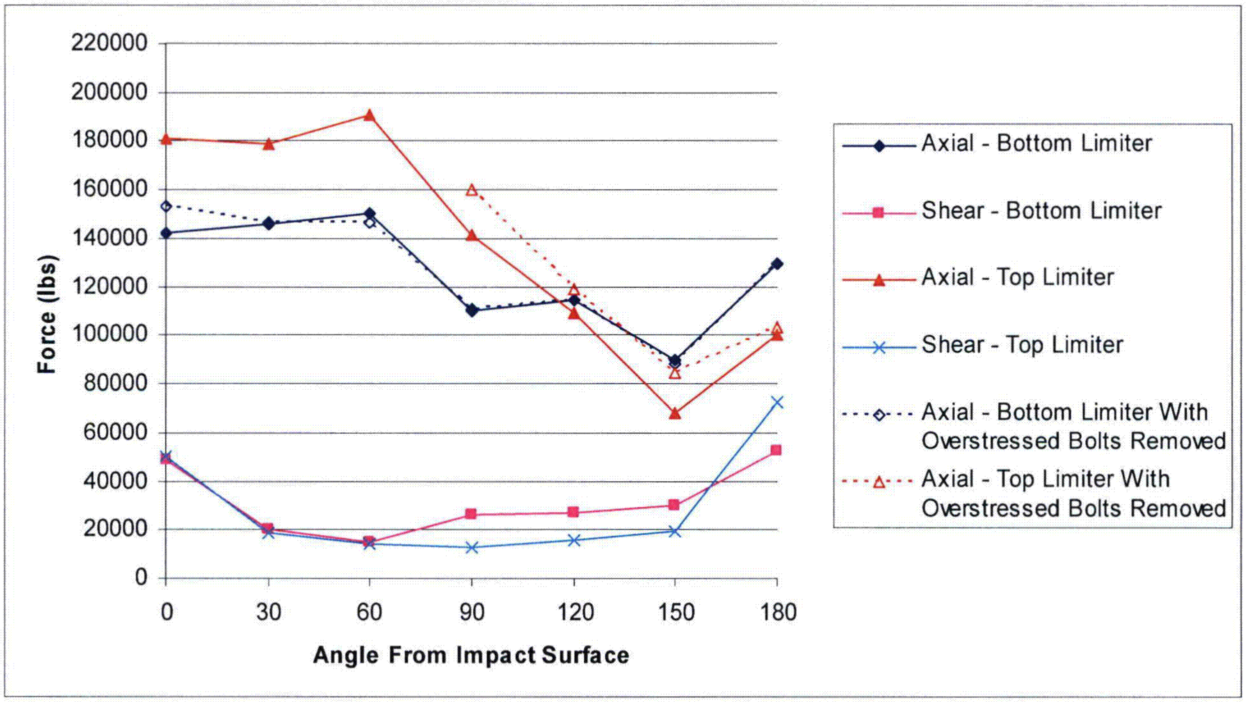


Figure A.2.13.12-51
Bolt Forces Around Top and Bottom Impact Limiters for 10° Slap Down Drop

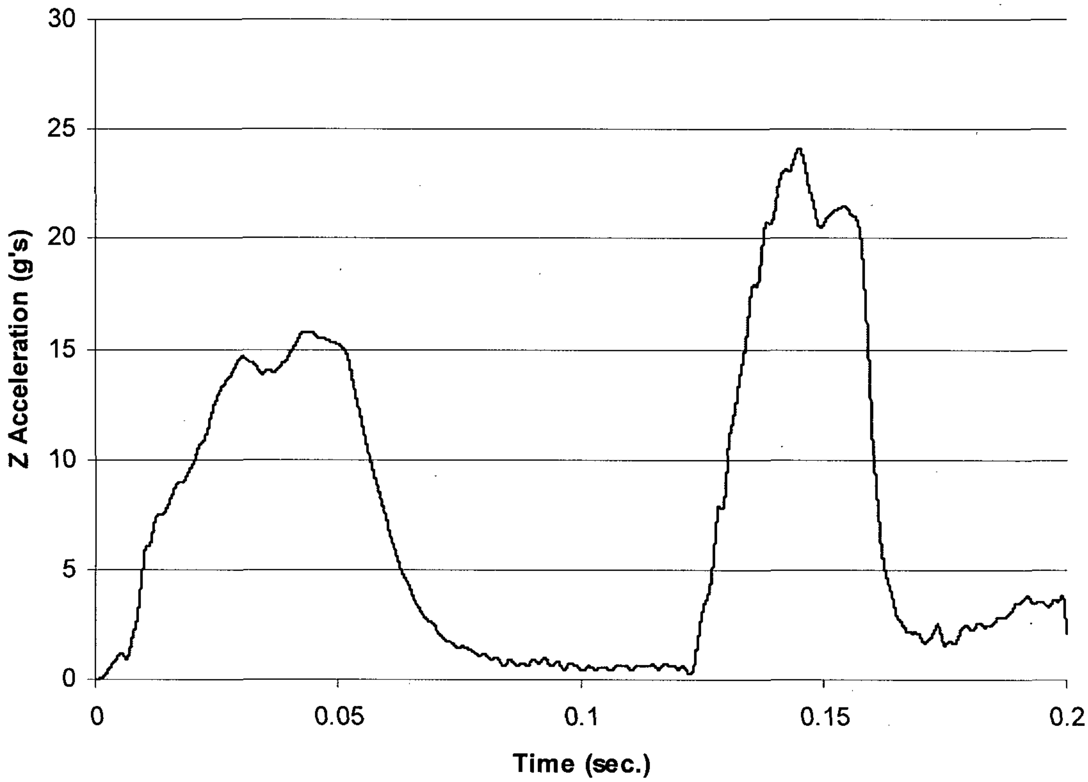


Figure A.2.13.12-52
Cask Rigid Body Resultant Acceleration for 20° Slap Down Drop – Full Scale

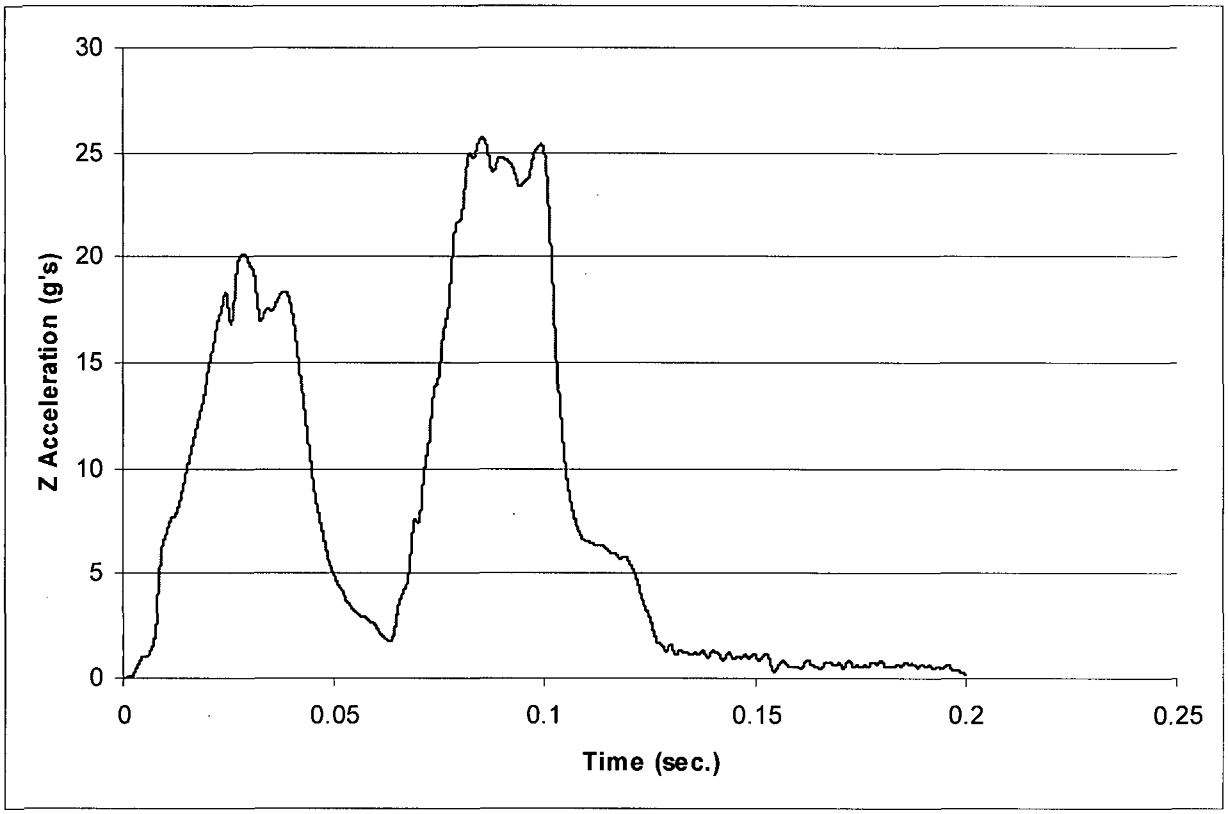


Figure A.2.13.12-53
Cask Rigid Body Resultant Acceleration for 10° Slap Down Drop–Full Scale

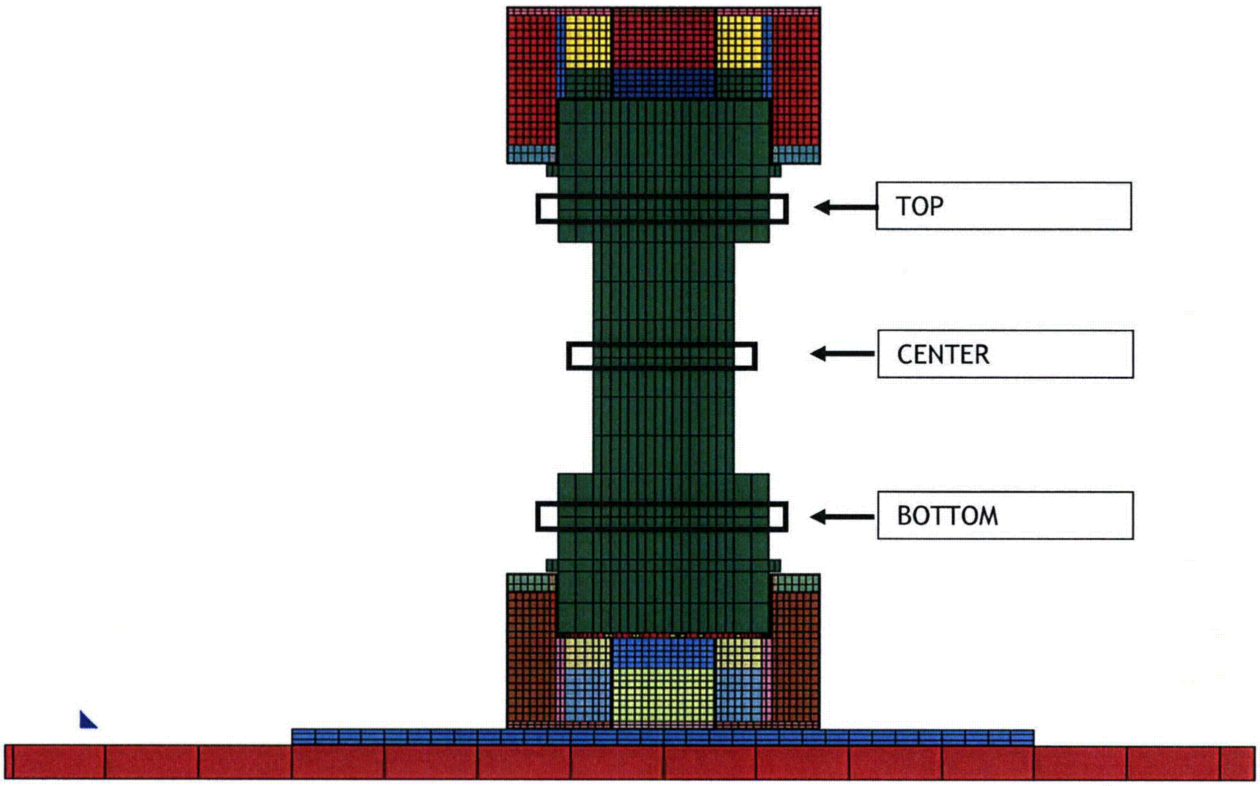


Figure A.2.13.12-56
Location of Nodes Selected to Average Accelerations

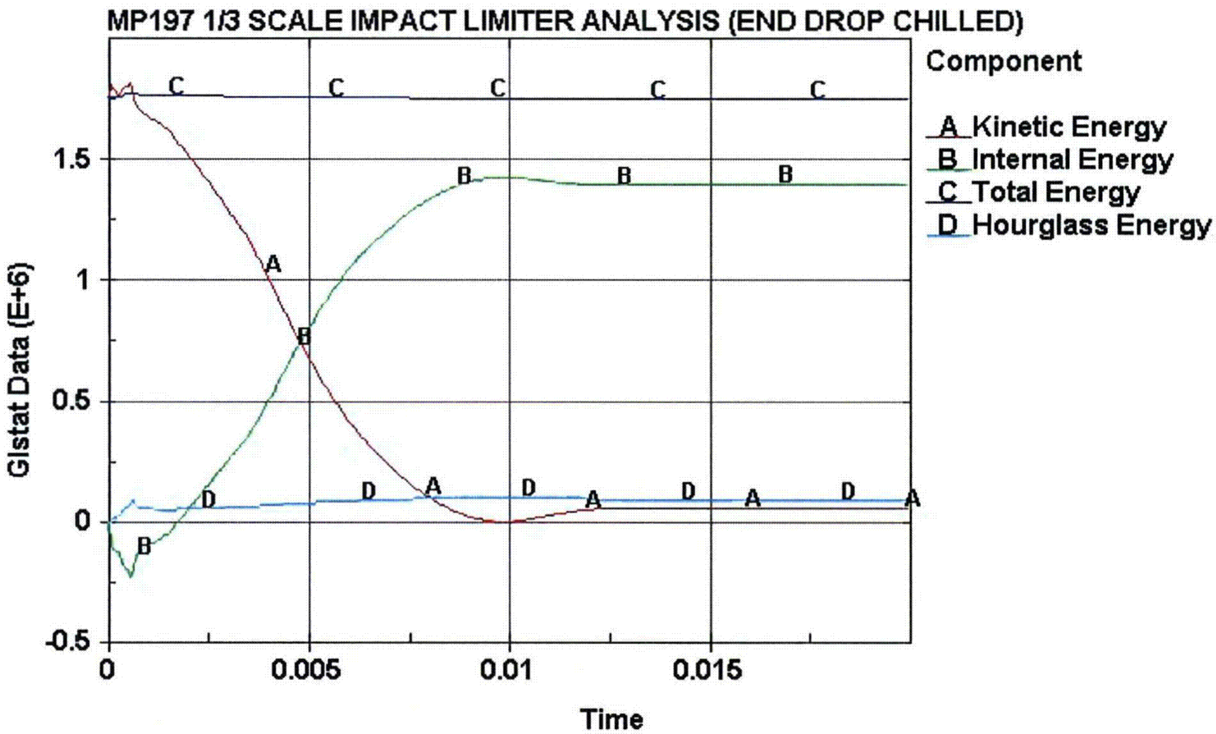


Figure A.2.13.12-57
LS-DYNA Energy Plots (in-lbf) 30' End Drop–1/3 Scale Benchmark Model

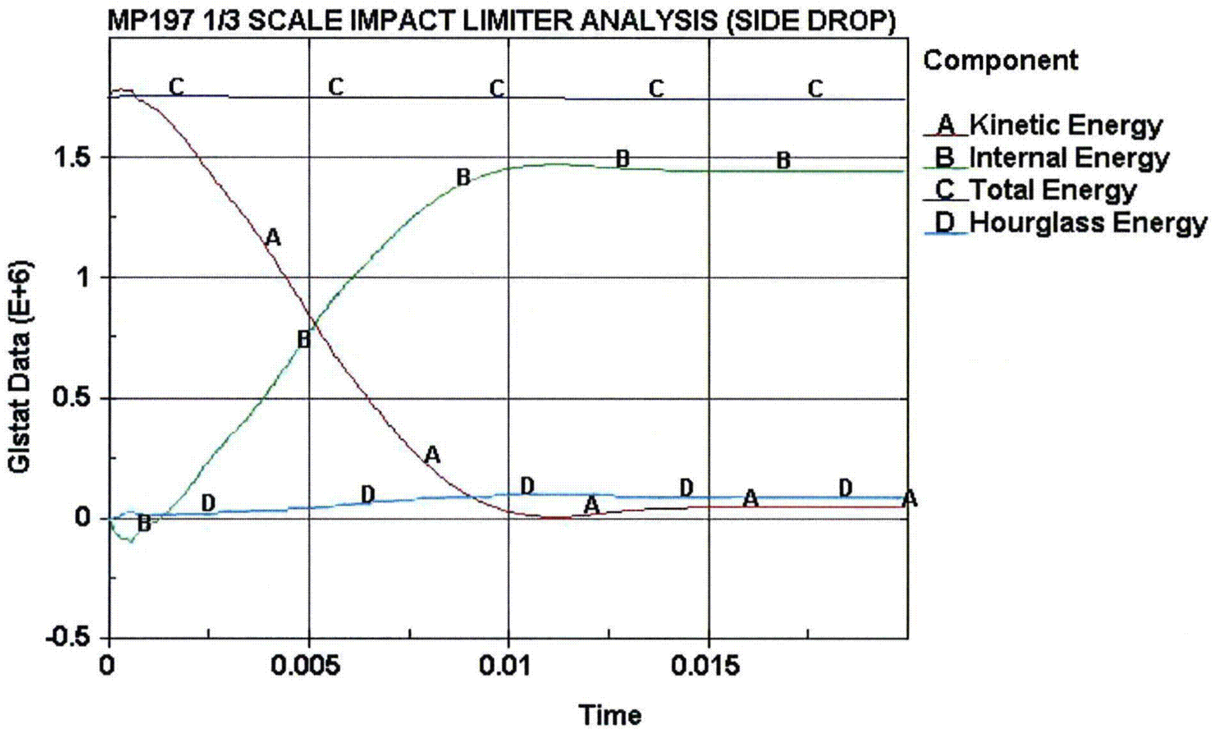


Figure A.2.13.12-58
LS-DYNA Energy Plots (in-lbf) 30' Side Drop–1/3 Scale Benchmark Model

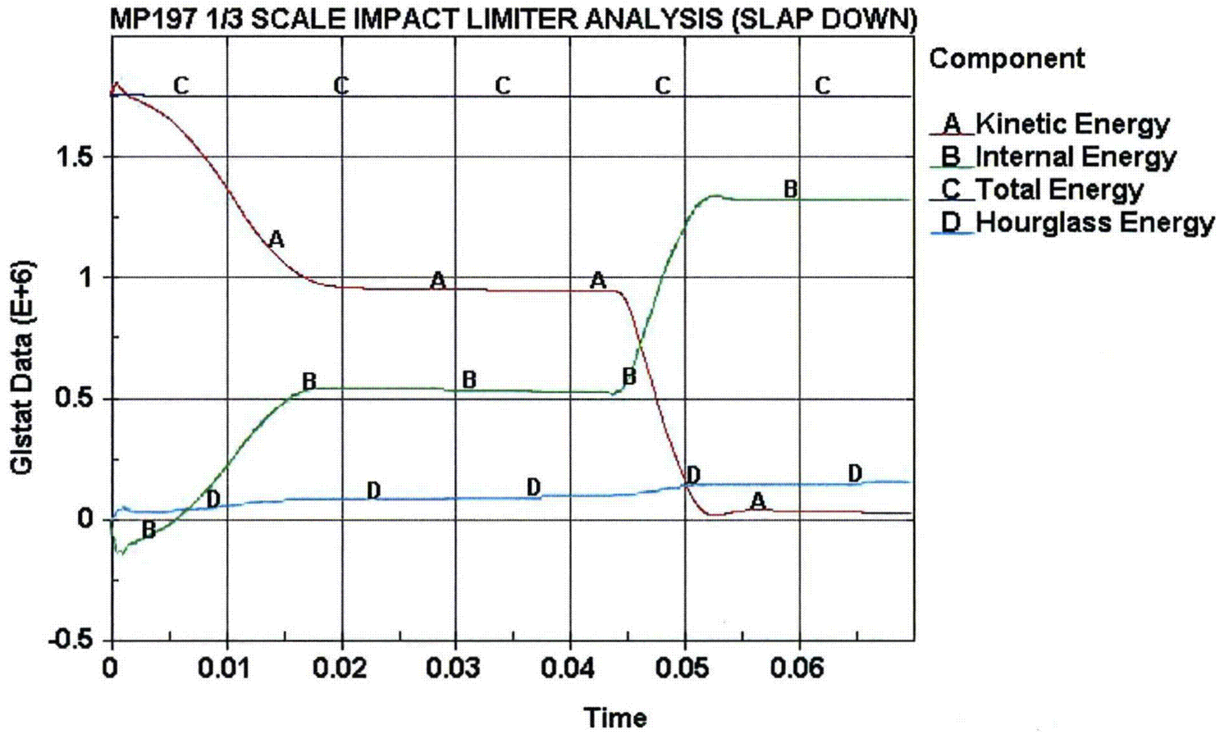


Figure A.2.13.12-59
LS-DYNA Energy Plots (in-lbf) 30' Slap Down–1/3 Scale Benchmark Model

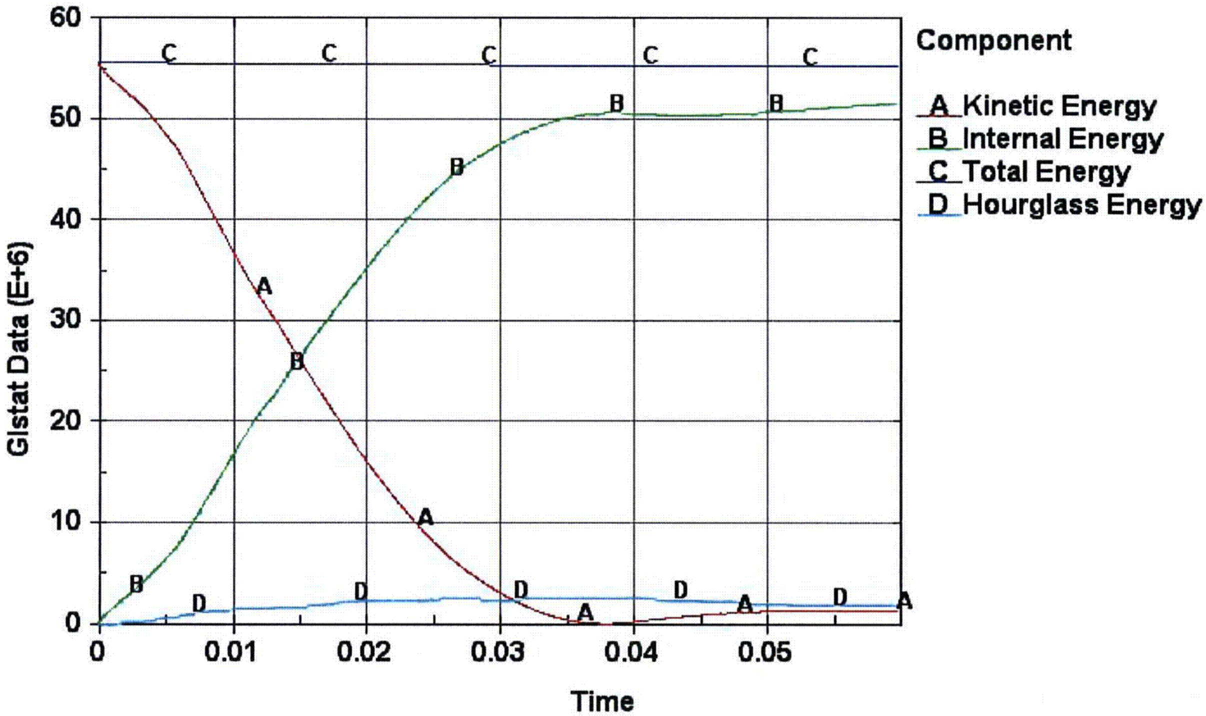


Figure A.2.13.12-60
LS-DYNA Energy (in-lbf) Plots 30' End Drop (Room Temperature)–Full Scale Model

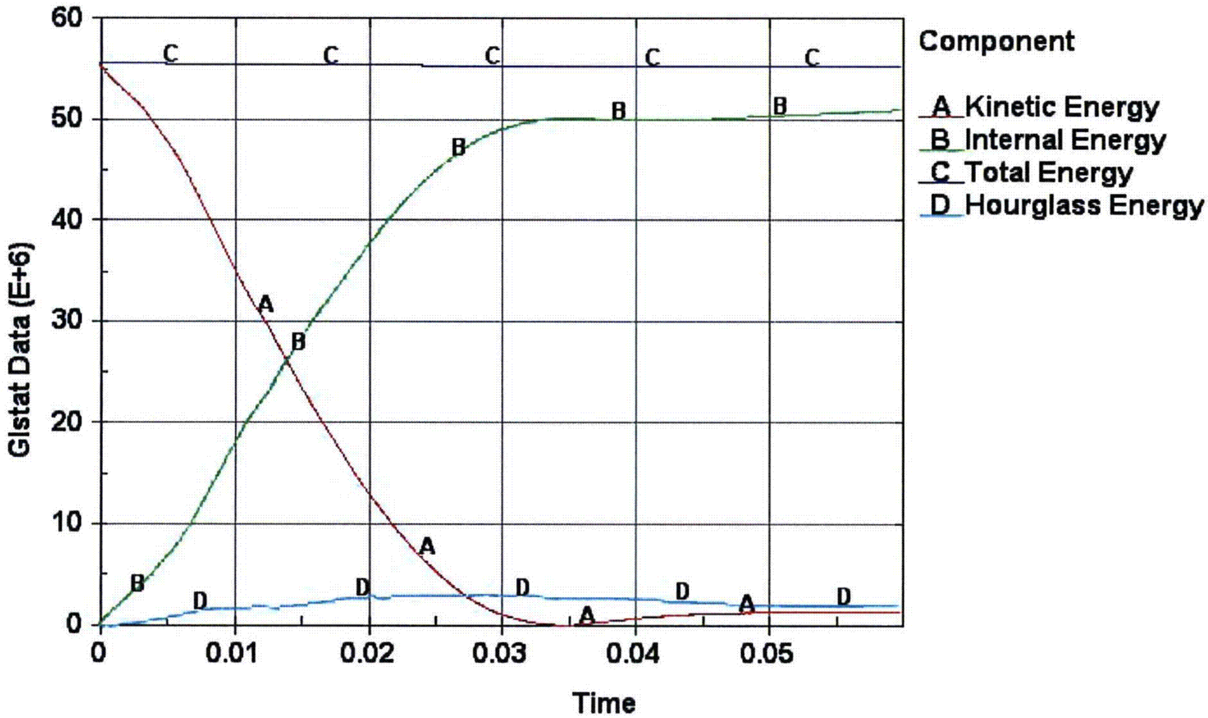


Figure A.2.13.12-61
LS-DYNA Energy (in-lbf) Plots 30' End Drop (-20 °F) –Full Scale Model

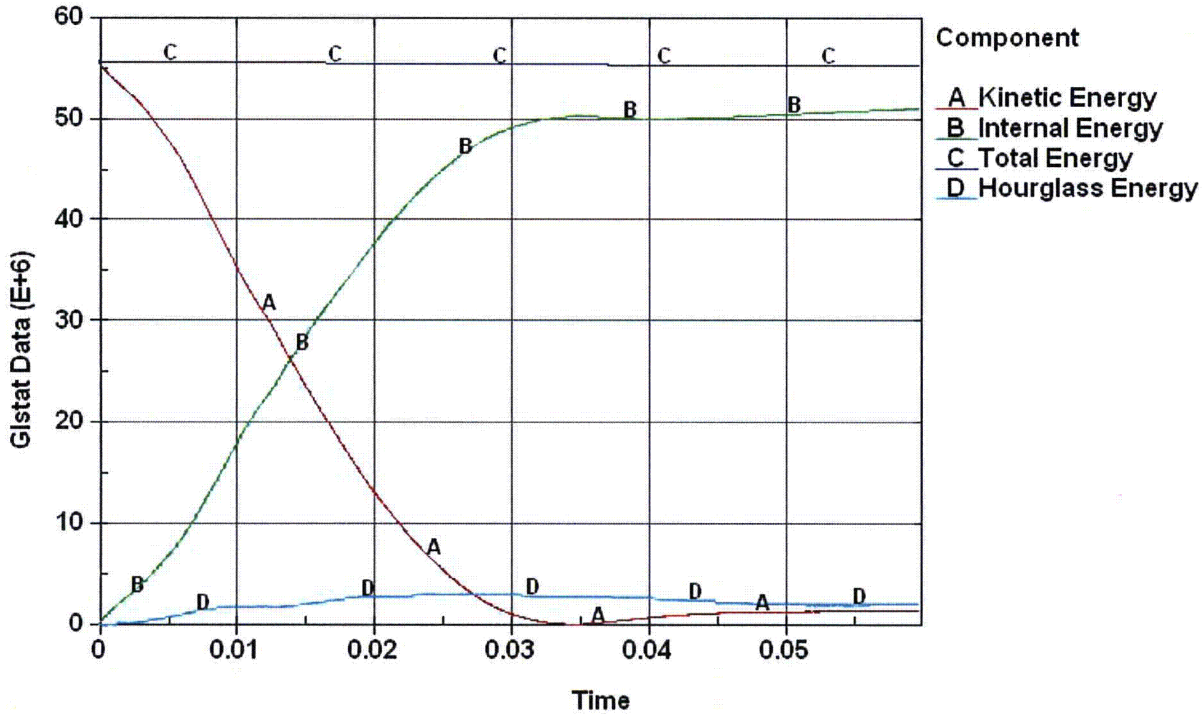


Figure A.2.13.12-62
LS-DYNA Energy (in-lbf) Plots 30' End Drop (-40 °F)–Full Scale Model

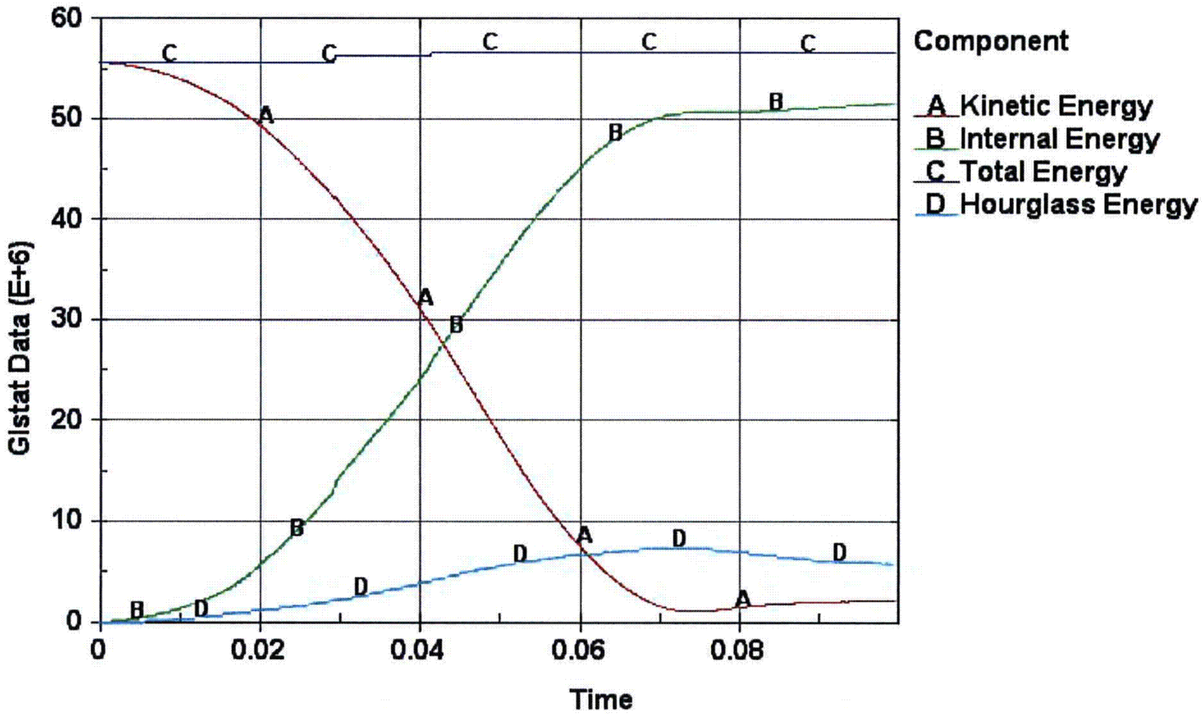


Figure A.2.13.12-63
LS-DYNA Energy (in-lbf) Plots 30' CG Over Corner–Full Scale Model

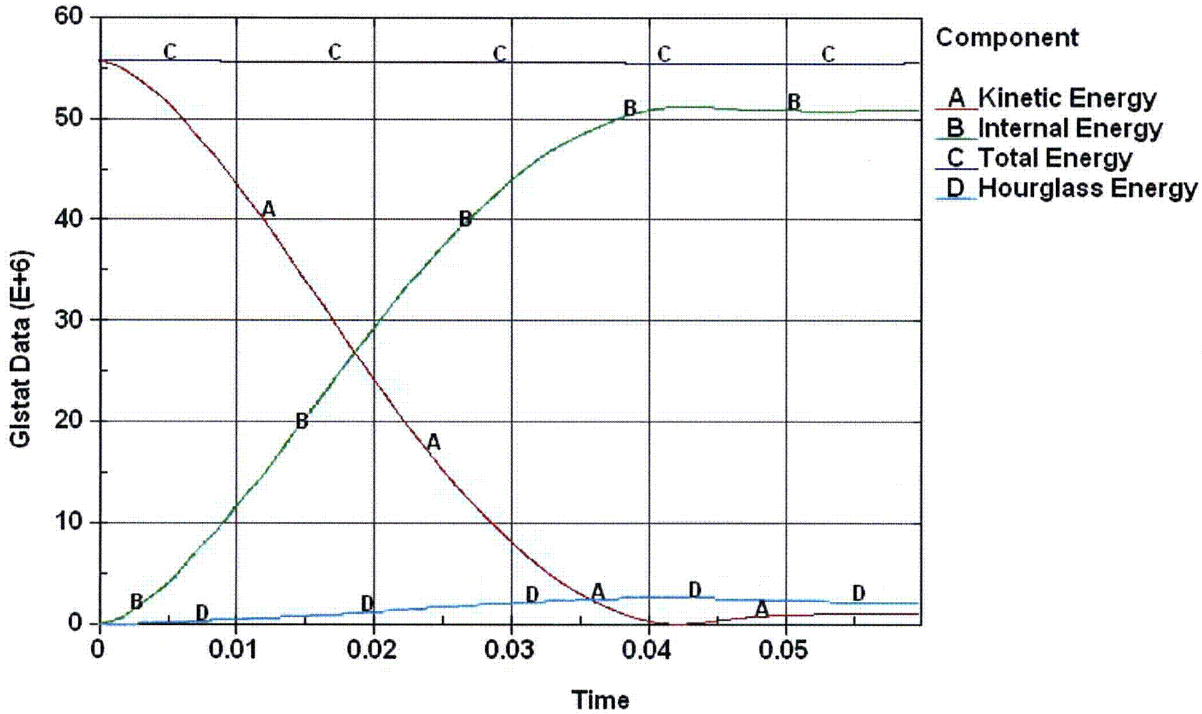


Figure A.2.13.12-64
LS-DYNA Energy (in-lbf) Plots 30' Side Drop–Full Scale Model

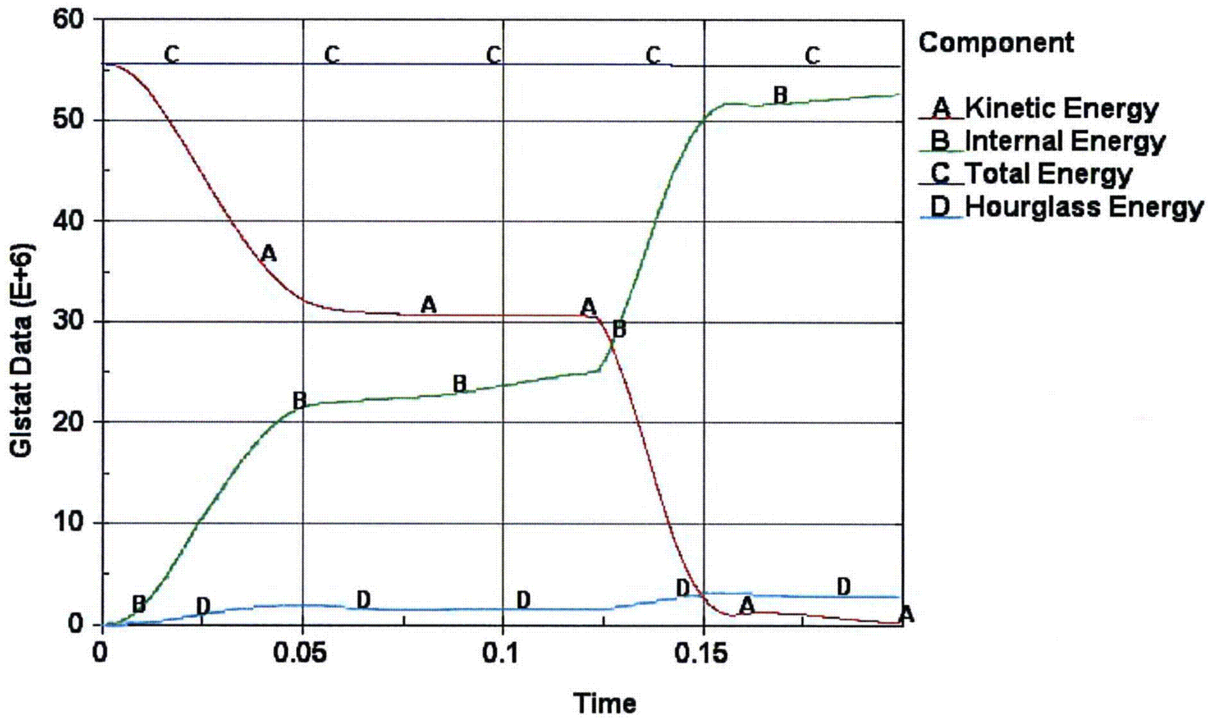


Figure A.2.13.12-65
LS-DYNA Energy (in-lbf) Plots 30' Side Drop Slap Down 20°–Full Scale Model

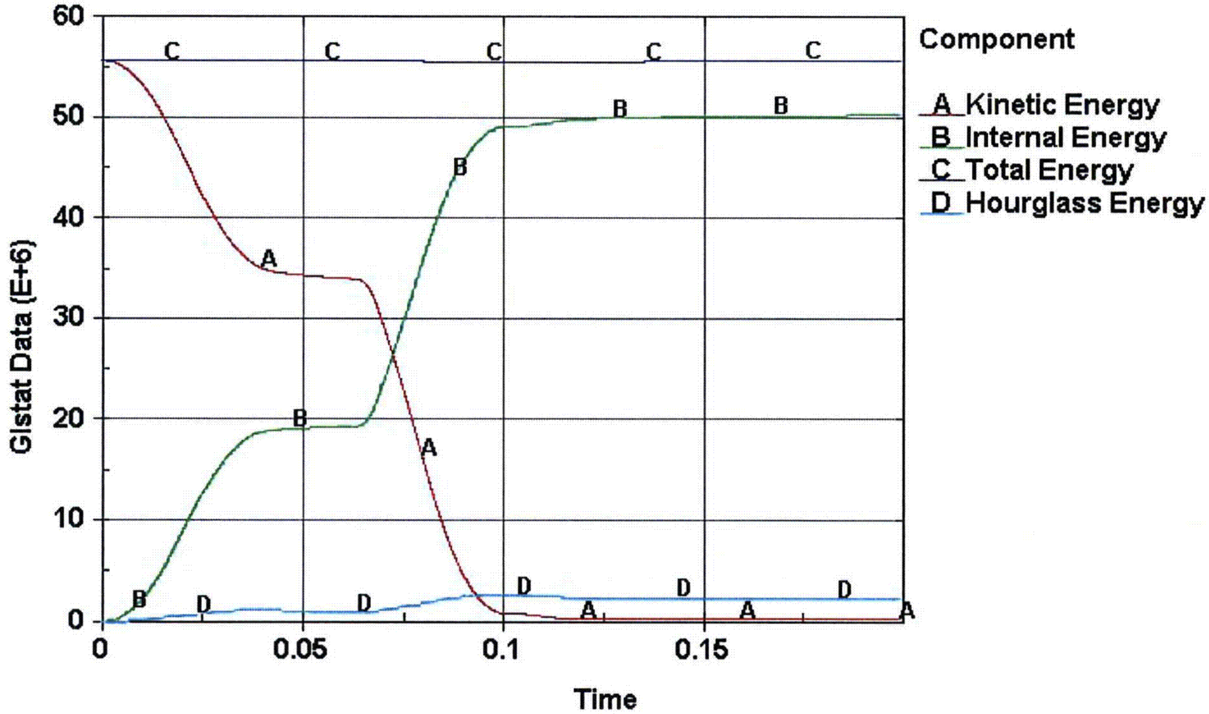


Figure A.2.13.12-66
LS-DYNA Energy (in-lbf) Plots 30' Side Drop Slap Down 10° –Full Scale Model

A.2.13.13.2 References

1. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section II, Section III, and Appendices, see Chapter A.2, Section A.2.1.2.1 for applicable editions.
2. 10 CFR PART 71, Packaging and Transportation of Radioactive Material.
3. Cases of ASME Boiler and Pressure Vessel Code, N-595-1 and N-595-2.
4. 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste."
5. AWS D1.3-98, "Structural Welding Code Steel."
6. ANSI N14.5-1997, "Leakage Tests on Packages for Shipment."
7. Interim Staff Guidance - 15, Rev. 0, "Materials Evaluation."
8. 49 CFR Part 173, Shippers — General Requirements for Shipments and Packaging
9. SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing."
10. *Regulatory Guide 7.11, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1m)," June 1991.*
11. *Regulatory Guide 7.12, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness Greater than 4 Inches (0.1m) But not Exceeding 12 Inches (0.3m)," June 1991.*

Table A.2.13.13-1
ASME Code Alternatives for the NUHOMS®-MP197HB Cask Containment Boundary

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NCA	All	Not compliant with NCA [1].
NB-1100	Requirements for Code Stamping of Components.	The NUHOMS®-MP197HB cask containment boundary is designed & fabricated in accordance with the ASME Code, Section III, Subsection NB [1] to the maximum extent practical. However, Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.
NB-1131	The design specification shall define the boundary of a component to which other components are attached.	A code design specification is not prepared for the NUHOMS®-MP197HB cask. A TN design criteria is prepared in accordance with TN's QA program.
NB-2130 NB-4121	Material must be supplied by ASME approved material suppliers. Material Certification by Certificate Holder.	All materials designated as ASME on the SAR drawings are certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability & certification are maintained in accordance with TN's NRC approved QA program.
NB-7000	Overpressure Protection.	No overpressure protection is provided for the NUHOMS®-MP197HB cask. The function of the NUHOMS®-MP197HB cask is to contain radioactive materials under normal, off-normal, and hypothetical accident conditions postulated to occur during transportation. The NUHOMS®-MP197HB cask is designed to withstand the maximum internal pressure considering 100% fuel rod failure at maximum accident temperature. The NUHOMS®-MP197HB cask is pressure tested in accordance with the requirements of 10CFR71 [2] and TN's approved QA program.
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000.	The NUHOMS®-MP197HB cask nameplates provide the information required by 10CFR71 and 49CFR173 [8] as appropriate. Code stamping is not required for the NUHOMS®-MP197HB cask. QA Data packages are prepared in accordance with the requirements of 10CFR71 and TN's approved QA program.
NB-3122.1	*No structural strength shall be attributed to cladding*	*The thickness of the weld overlay/cladding is included in the analytical models described in Chapter A.2 to calculate the behavior of the MP197HB cask for NCT and HAC loading conditions. Weld overlay/cladding will meet the normal ASME Code weld qualification requirements. In addition, a testing program to demonstrate that the weld overlay/cladding bond strength is adequate to transmit all loads, and that the HAZ meets the nil ductility requirements of RGs 7.11 [10] and 7.12 [11] will be performed prior to cask fabrication. An annual UT inspection of the weld overlay/cladding surface will also be performed to ensure that the interface bond remains intact.*

Table A.2.13.13-1
ASME Code Alternatives for the NUHOMS®-MP197HB Cask Containment Boundary
(Concluded)

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NB-5221	Volumetric (RT) inspection of CAT. B weld required	Fabrication sequence for inner and outer shells makes RT of 2 closure welds very difficult. May need to use UT or multilayer MT/PT to provide volumetric examination.

Chapter A.3
Thermal Evaluation

TABLE OF CONTENTS

MP197HB TC Design Features

DSC type	DSC OD (in.)	TC Sleeve	TC External Fins	Max. DSC Heat Load for Transport (kW)
69BTH	69.75	No	No	26.0
			Yes (optional) (3)	29.2
			Yes (optional) (3)	32.0
61BTH Type 1 (1)	67.25	Yes	No	22.0
61BTH Type 2 (1)	67.25	Yes	No	24.0
61BT	67.25	Yes	No	18.3
37PTH	69.75	No	No	22.0
32PTH/32PTH Type 1	69.75	No	No	26.0
32PTH1 Type 1	69.75	No	No	26.0
32PTH1 Type 2	69.75	No	No	24.0
32PT	67.19	Yes	No	24.0
24PTH-S or -L (w/ Al inserts)(1)	67.19	Yes	No	26.0
24PTH-S or –L (w/o Al inserts)(1)	67.19	Yes	No	26.0
24PTH-S-LC (1)	67.19	Yes	No	24.0 (2)
24PT4	67.19	Yes	No	24.0

Notes:
(1) 61BTHF and 24PTHF *DSCs* have the same dimensions and use the same MP197HB features as DSC types 61BTH and 24PTH, respectively.
(2) The 24PTH-S-LC DSC is allowed for 24 kW heat load. The analyis assumes a heat load of 26 kW for conservatism.
(3) The external fins are optional to enhance heat transfer and provide larger margin for the cladding temperature.

The MP197HB TC consists of multiple shells which conduct the decay heat to the cask outer surface. The other thermal design feature of the cask is the conduction path created by the aluminum boxes that contain the neutron shielding material as described in Chapter A.5. The neutron shielding material is provided by a resin compound cast into long slender aluminum boxes placed around the gamma shield shell and enclosed within a steel shell (shield shell). The aluminum boxes are designed to fit tightly against the steel shell surfaces, thus improving the heat transfer across the neutron shield.

The external circular fins can be used optinally for the transport of DSCs with heat load exceeding 26 kW.

Heat dissipates from the packaging outer surfaces via natural convection and radiation. The outer surface of the shield shell is painted white to enhance the thermal radiation exchange with ambient.

The design of the steel-encased wood impact limiters is described in Chapter A.1, Section A.1.2. These components are included in the thermal analysis because of their contribution as a thermal insulator. The impact limiters provide protection to the lid and bottom regions from the external heat input due to fire during the HAC thermal event.

A personnel barrier prevents access to the outer surfaces of the cask body. The barrier, which consists of a stainless steel mesh attached to stainless steel tubing, encloses the wood are assumed to be decomposed or charred after fire accident. Therefore, the maximum temperatures for these components are irrelevant for HAC. The maximum fuel cladding, gamma shield and seal temperatures remain below the allowable limits and ensure the integrity of the fuel cladding and the containment boundary for HAC.

A.3.1.4 Summary Tables of Maximum Pressures

The maximum internal pressures inside MP197HB TC cavity are calculated in Section A.3.3.3 for NCT and Section A.3.4.3 for HAC. The maximum internal pressures of the MP197HB TC cavity are summarized in Table A.3–20. The maximum pressures inside canister cavities of 69BTH and 37PTH DSCs are listed in Table A.3–22. The maximum internal pressures for all DSCs proposed for transport in MP197HB TC are summarized in Table A.3–23. These pressures remain below the design pressures for NCT and HAC *considered for the structural evaluation.*

A.3.2 Material Properties and Component Specifications

A.3.2.1 Material Properties

The following tables provide the thermal properties of materials used in the analysis of the MP197HB TC with DSCs. *Each table is valid for all the models unless it is specifically noted in the title of the table.*

1. Bounding PWR Fuel Assembly for 37PTH *DSC*

Calculation of the effective properties for homogenized PWR fuel assemblies in a 37PTH DSC are discussed in Section A.3.6.5.1. The bounding effective properties for PWR fuel assemblies in 37PTH DSC are listed below.

Homogenized PWR Fuel Assemblies in Four Corner Fuel Compartments in 37PTH DSC

Temp (°F)	Transverse Conductivity (Btu/hr-in-°F)	Temp (°F)	Axial Conductivity (Btu/hr-in-°F)	Temp (°F)	Specific Heat (Btu/lbm-°F)	Density (lbm/in^3)
178	0.0168	200	0.0456	80	0.05924	0.1114
267	0.0195	300	0.0481	260	0.06538	
357	0.0230	400	0.0506	692	0.07255	
448	0.0273	500	0.0527	1502	0.07779	
541	0.0323	600	0.0548			
635	0.0380	800	0.0594			
730	0.0444					
826	0.0513					

Homogenized PWR Fuel Assemblies in Other Fuel Compartments in 37PTH DSC

Temp (°F)	Transverse Conductivity (Btu/hr-in-°F)	Temp (°F)	Axial Conductivity (Btu/hr-in-°F)	Bounding effective specific heat and density are the same as those for corner fuel assemblies.
138	0.0174	200	0.0454	
233	0.0199	300	0.0478	
328	0.0238	400	0.0503	
423	0.0285	500	0.0524	
519	0.0340	600	0.0545	
616	0.0403	800	0.0591	
714	0.0473			
812	0.0552			

Note:
(1) Only 95% of the axial effective conductivity calculated in [1, Appendix M, Section M.4.8] for 32PT DSC is considered in the 37PTH DSC model for conservatism.

2. Bounding BWR Fuel Assembly for 69BTH *DSC*

Calculation of the effective properties for homogenized BWR fuel assemblies in 69BTH DSC are discussed in Section A.3.6.5.2. The bounding effective properties for BWR fuel assemblies in 69BTH DSC are listed below.

Homogenized BWR Fuel Assemblies in 69BTH DSC

Temperature (°F)	k, Transverse (Btu/hr-in-°F)	k, Axial (Btu/hr-in-°F)	ρ (lb_m/in^3)	C_p (Btu/lb_m-°F)
200	0.0157	0.0402	0.103	0.0575
300	0.0181			
400	0.0211			
500	0.0246			
600	0.0285			
700	0.0328			
800	0.0375			

3. SA-240, Type 304 Stainless Steel [10]

Temperature (°F)	**k (Btu/hr-in-°F)**	**ρ (lbm/in³)**	**C_p (Btu/lb_m-°F)**
70	0.717	0.284	0.116
100	0.725		0.117
200	0.775		0.122
300	0.817		0.125
400	0.867		0.129
500	0.908		0.131
600	0.942		0.133
700	0.983		0.135
800	1.025		0.136
900	1.058		0.137
1000	1.092		0.138

4. SA-240 Type 316 [2, Section 4.2.c]

Temperature (°F)	**k (Btu/hr-in-°F)**	**ρ (lbm/in³)**	**C_p (Btu/lb_m-°F)**
70	0.642	0.285	0.117
100	0.658		0.118
200	0.700		0.121
300	0.750		0.126
400	0.792		0.126
500	0.833		0.130
600	0.875		0.132
700	0.917		0.134
800	0.958		0.135
900	1.000		0.137
1000	1.033		0.137

8. Aluminum, Type 1100 [10]

Temperature (°F)	k (Btu/hr-in-°F)	ρ (lb_m/in^3)	C_p (Btu/lb_m-°F)
70	11.092	0.098	0.214
100	10.983[(1)]		0.216
150	10.833		0.219
200	10.708		0.222
250	10.608		0.224
300	10.517		0.227
350	10.442		0.229
400	10.375		0.232

Note: [(1)] *The input files of the ANSYS models for the baskets contain a thermal conductivity of 11.150 Btu/hr-in-°F (133.8 Btu/hr-ft-°F) instead of 10.983 Btu/hr-in-°F. Since this value is used only at 100°F and since* the basket temperature is over 150°F for all analyzed cases, this value does not affect the results in this SAR.

9. Aluminum, Type 6063 [10]

Temperature (°F)	k (Btu/hr-in-°F)	ρ (lb_m/in^3)	C_p (Btu/lb_m-°F)
70	10.067	0.098	0.213
100	10.025		0.215
150	9.975		0.218
200	9.917		0.221
250	9.875		0.223
300	9.842		0.226
350	9.833		0.228
400	9.800		0.230

10. Aluminum, Type 6061 [10]

Temperature (°F)	k (Btu/hr-in-°F)	ρ (lb_m/in^3)	C_p (Btu/lb_m-°F)
70	8.008	0.098	0.213
100	8.075		0.215
150	8.167		0.218
200	8.250		0.221
250	8.317		0.223
300	8.383		0.226
350	8.442		0.228
400	8.492		0.230

11. Gamma Shield, ASTM B29 Lead [24]

Temperature (°F)	k (Btu/hr-in-°F)	ρ (lb_m/in^3)	C_p (Btu/lb_m-°F)
-100	1.767	0.413	0.030
-10	1.733	0.411	0.030
80	1.700	0.409	0.031
260	1.637	0.406	0.032
440	1.579	0.402	0.033
620	1.512	0.398	0.034

12. Neutron Shield Resin (Vyal B) [17]

Temperature (°F)	k (Btu/hr-in-°F)	ρ (lb_m/in^3)	C_p (Btu/lb_m-°F)
104	0.039	0.06	0.256
140			0.260
176			0.282
212			0.301
284			0.358
320			0.380

13. Trunnion Plug Resin (Polypropylene) [5]

Temperature (°F)	k (Btu/hr-in-°F)	ρ (lb_m/in^3)	C_p (Btu/lb_m-°F)
All temperatures	0.0067	0.032	0.46

14. Wood [20]

Minimum conductivity, k_{min} = 0.0019 Btu/hr-in-°F for NCT and cool-down period, see MP197 SAR, Section 3.2, Item 8		
Maximum conductivity, k_{max} = 0.0378 Btu/hr-in-°F during fire period, see MP197 SAR, Section 3.2, Item 8		
Temperature (°F)	**ρ (lb_m/in^3) (1)**	**C_p (Btu/lb_m-°F)**
100	0.007	0.312
200	0.006	0.363
300	0.005	0.414
400	0.005	0.466
500	0.004	0.517
600	0.004	0.568

Note: (1) The wood density is calculated based on thermal diffusivity using $\alpha = \frac{k}{\rho c_P}$ with

α = 0.00025 in^2/s (0.9 in^2/hr) based on Wood Handbook [20],
k = conductivity =0.0019 (Btu/hr-in-°F),
ρ = density (lbm/in^3), and
c_P = specific heat (Btu/lbm-°F).

20. Effective Conductivity for Dummy Aluminum Assemblies *in 69BTH DSC*

(See Section A.3.3.1.5 for calculation of effective properties)

a_{dummy} =	5.875	in
t_{gap} =	0.0625	in
w_{comp} =	6	in

Temp	k_{Al6061} [10]
(°F)	(Btu/hr-in-°F)
70	8.008
100	8.075
200	8.250
300	8.383
400	8.492
650	8.492 (1)

Temp	k_{He} (2)	Temp	k_{He}
(°F)	(Btu/hr-in-°F)	(°F)	(Btu/hr-in-°F)
-10	0.0064	70	0.0071
80	0.0072	100	0.0074
260	0.0086	200	0.0081
440	0.0102	300	0.0090
620	0.0119	400	0.0098
980	0.0148	650	0.0121
1340	0.0174		

Temp	R_{th_He1}	R_{th_Al6061}	R_{th_He2}	$R_{th,tr,dummy}$	$k_{eff,tr,dummy}$	$k_{eff,ax,dummy}$
(°F)	(Btu/hr-in-°F)	(Btu/hr-in-°F)	(Btu/hr-in-°F)	(Btu/hr-in-°F)	(Btu/hr-in-°F)	(Btu/hr-in-°F)
70	1.4648	0.1249	13218.8	3.0546	0.327	7.678
100	1.4162	0.1238	12779.5	2.9562	0.338	7.742
200	1.2807	0.1212	11557.4	2.6827	0.373	7.910
300	1.1632	0.1193	10496.3	2.4456	0.409	8.037
400	1.0581	0.1178	9548.5	2.2340	0.448	8.142
650	0.8579	0.1178	7741.9	1.8336	0.545	8.142

Notes:

(1) Al6061 conductivity increases at higher temperatures. Increasing of the Al6061 conductivity is conservatively ignored for calculation of effective conductivity of aluminum dummy assembly.

(2) See Section A.3.2.1, material # 15 for helium properties.

Applicable Code Years for Each Canister Design

Canister Design	Applicable Storage License	ASME B&PV Code Year
NUHOMS® 32PTH	1030	1998 w/ 2000 Addenda
NUHOMS® 32PTH1	1004	1998 w/ 2000 Addenda
NUHOMS® 37PTH	*Note (1)*	2004 w/2006 Addenda
NUHOMS® 69BTH	*Note (1)*	2004 w/2006 Addenda
NUHOMS® 24PT4	1029	1992 thru1994 Addenda
NUHOMS® 24PTH	1004	1998 w/ 2000 Addenda
NUHOMS® 32PT	1004	1998 w/ 2000 Addenda
NUHOMS® 61BT	1004	1998 w/ 1999 Addenda
NUHOMS® 61BTH	1004	1998 w/ 2000 Addenda
NUHOMS® 61BTH *with failed fuel (61BTHF)*	*Note (1)*	2004 w/2006 Addenda
NUHOMS® 24PTH *with failed fuel (24PTHF)*	*Note (1)*	2004 w/2006 Addenda

Note (1): These DSCs are currently not a part of CoC 1004 but will be added at a later date via amendment.

The shell and cover plates of all DSC types except for 24PT4 consist of stainless steel SA-240, type 304 (SS304). There are no changes in thermal conductivity of SS304 in ASME 1998 to 2006 in temperature range from 70 to 700°F. This range properly covers the DSC shell temperature for all DSC types in this calculation.

The shield plugs of all DSC types except for 24PT4 consist of carbon steel A36. The changes in the A36 conductivity between ASME code years 1998 to 2006 are limited to ±0.9%. This small change has no significant effect on the thermal evaluation.

The thermal properties for 24PT4 DSC are taken from UFSAR for standardized advanced NUHOMS® system [2]. These properties are based on ASME code 1992 through 1994 addenda and used in this calculation without any changes.

A.3.2.2 Component Specifications

The components for which thermal technical specification are necessary are the MP197HB containment seals and poison plates used in DSC basket.

A.3.2.2.1 MP197HB TC

The seals used in the packaging are the Fluorocarbon seals (Viton O-rings). The seals will have a minimum and maximum temperature rating of -40°F and 400°F, respectively.

A.3.2.2.2 69BTH DSC

The 69BTH DSC design allows the use of different neutron absorber materials based on the heat load zoning configuration (HLZC). Boral, Metal Matrix Composite (MMC), or Borated Aluminum can be used as poison materials for HLZC # 1, # 2 and # 3 in 69BTH basket. For 69BTH basket with HLZC # 4, only borated aluminum can be used as poison material. The HLZCs for 69BTH are described in Section A.3.3.1.4

A.3.2.2.3 37PTH DSC

The 37PTH DSC design allows the use of different neutron absorber materials. Boral plates paired with Al1100 plates or single plates of metal matrix composite (MMC) or conductivities are conservative since the number and the assumed gaps between the internal sleeve pieces are larger than those considered for the proposed internal sleeve.

The material properties used in the MP197HB model are listed in Section A.3.2.1.

The seal o-rings are not explicitly considered in the models. The maximum seal temperatures are retrieved from the models by selecting the nodes at the locations of the corresponding seal o-rings.

The geometry of the TC model and the gaps are shown in Figure A.3–2 through Figure A.3–5. Mesh sensitivity of the MP197HB model is discussed in Appendix A.3.6.2.1.

Typical boundary conditions for TC model under NCT are shown in Figure A.3–6 through Figure A.3–8.

A.3.3.1.2 Calculation of Maximum Accessible Surface Temperature

A personnel barrier installed on the transport skid between the two impact limiters of MP197HB TC limits the accessible packaging surfaces to the impact limiter and barrier outer surfaces. The personnel barrier has an open area of at least 80%. Radiation heat transfer between the cask and the barrier will be minimal due to the small radiation view factor between the cask and the barrier. Due to large distance between the barrier and cask outer surface, the free convection heat transfer around TC remains undisturbed. The transport configuration is shown in the drawings in Chapter A.1, Appendix A.1.4.10.

The TC model described in Section A.3.3.1.1 is run without insolance to determine the accessible surface temperature of the impact limiters in the shade. A heat load of 32 kW and boundary conditions at 100°F and no insolance are considered in the cask model to bound the maximum accessible surface temperature under shade.

The maximum accessible surface temperature of impact limiters under these conditions is 121°F. The maximum temperature of the cask outer surface is 302°F and belongs to a part of shield shell uncovered by the external fins in the model. The maximum temperature of the personnel barrier is calculated based on the maximum temperature of the cask outer surface using the following methodology.

The personnel barrier is exposed to thermal radiation from the cask shield shell / finned shell and dissipates heat via thermal radiation and natural convection to ambient. Since the personnel barrier is far apart from the cask shield shell, it is not exposed to the hot air streams from the cask. *This assumption is justified in Section A.3.3.1.2.1.*

The heat balance for the personnel barrier is shown schematically in Figure A.3–9. The following conservative assumptions are considered to simplify the heat balance.

- Convection heat dissipation from the barrier is omitted completely.
- Radiation heat dissipation to ambient from barrier surfaces facing the cask is omitted.
- The maximum cask outer surface temperature is considered for the cask entire outer surface facing the barrier.
- An emissivity of 0.9 is considered for the cask outer surface. Based on discussion in Section A.3.2.1, the emissivities for shield shell and finned shell are

$$F_{1-2} = \frac{1}{l_{PB}} \int_0^{l_{PB}} \left(\frac{Y}{X^2 + Y^2} \right) dx$$

Considering $L=l_{PB}/r$ and $dX=dx/r$, gives the view factor of the personnel barrier to the cask.

$$F_{1-2} = \frac{1}{L} \int_0^{L} \left(\frac{Y}{X^2 + Y^2} \right) dX = \frac{1}{L} \left(\tan^{-1} \frac{L}{Y} \right) = \frac{r}{l_{PB}} \tan^{-1} \left(\frac{l_{PB}}{y} \right) = 0.685$$

Since the personnel barrier has an open area of 80%, a factor of 0.2 should be considered to calculate the view factor of the personnel barrier mesh to the cask.

$$F_{PB-shell} = F_{1-2} \times 0.2 = 0.137$$

The substitution of the above values in the heat balance of the personnel barrier gives the maximum temperature of the personnel barrier as 152°F.

An additional analysis is performed considering insolation for TC model with 32 kW heat load, 100°F ambient, and no external fins. Under these conditions, the maximum cask shield shell temperature is 335°F. Using the same methodology as described above gives a maximum personnel barrier temperature of 163°F.

A.3.3.1.2.1 Personnel Barrier and Hot Stream from Cask

The assumption that the personnel barrier is not exposed to hot air stream from the cask shield shell can be justified by calculation of the thermal boundary layer thickness around the lower half of the cask. This calculation demonstrates that the thermal boundary layer thickness is smaller than the shortest distance between the personnel barrier and the cask and therefore the personnel barrier remains out of the hot air stream from the cask.

The large diameter of the cask and the relative large temperature difference between the cask outer surface and ambient temperature suggest that the free convection over the cylinder is a turbulent flow.

The theoretical and experimental studies of the free convection and its related thermal boundary layer thickness are widely available and well documented ([43], [45], [46], and [47]). These correlations can be used to determine the free convection thermal boundary layer thickness over the horizontal MP197HB cask. These studies show that the thickness of the free convection thermal boundary layer is inversely proportional to a power of the local Nusselt number for laminar or turbulent flows.

$$\frac{\delta}{x} = \frac{C\,f(Pr)}{Nu_x^{\ m}} \qquad (1)$$

δ = local thermal boundary layer thickness
x = local position
f(Pr) = a function of Prandtl number
Nu_x = local Nusselt number
m and c = constant values

For instance, the theoretical calculation in reference [46] determines the following equation for a free convection laminar flow over flat vertical plates.

$$\frac{\delta}{x} = \frac{2}{Nu_x}$$

This equation means that c = 2, f(Pr) =1, and m = 1 in equation (1).

The correlations to determine the free convection thermal boundary layer thickness and local Nusselt number over a vertical flat plate in turbulent flow are documented in references [44] and [45]. These correlations are shown below.

$$\frac{\delta_x}{x} = 0.565\, Gr_x^{-0.1}\, Pr^{-8/15}\left(1 + 0.494\, Pr^{2/3}\right)^{0.1} \qquad [44] \qquad (2)$$

$$Gr_x = \frac{g\beta\Delta T\, x^3}{\upsilon^2}$$

$$Nu_x = 0.0295\left[\frac{Pr^7}{\left(1 + 0.494\, Pr^{2/3}\right)^6}\right]^{1/15} Gr_x^{2/5} \qquad [45]$$

$$Nu_L = 0.834\, Nu_x \qquad [45] \qquad (3)$$

An examination of the above equations shows that the thermal boundary layer thickness is reversely proportional to the $Nu_x^{0.25}$ *for turbulent free convection over a vertical flat plate.*

Considering the relationship between the thermal boundary layer thickness and the local Nusselt number, the boundary layer thickness over a horizontal cylinder in free convection turbulent flow can be determined using the correlations over a vertical flat plate and the inverse ratio of the local Nusselt numbers. Since the correlations for the average Nusselt numbers of free convection for vertical flat plates and horizontal cylinders are known better than the local Nusselt numbers, the ratio of the local Nusselt numbers are extended to include the average Nusselt numbers and avoid elimination of any functions related to Prandtl number.

$$\frac{\delta_{D0}}{\delta_x} = \frac{\delta_L}{\delta_x}\cdot\frac{\delta_D}{\delta_L}\cdot\frac{\delta_{D0}}{\delta_D}$$

$$\frac{\delta_{D0}}{\delta_x} = \left(\frac{Nu_x}{Nu_L}\right)^p \left(\frac{Nu_L}{Nu_D}\right)^q \left(\frac{Nu_D}{Nu_{D0}}\right)^r \qquad (4)$$

δ_{D0} = thermal boundary layer thickness at midsection of a horizontal cylinder (α = 0)
δ_x = thermal boundary layer thickness at height of x for a vertical flat plate
Nu_x = local Nusselt number for a vertical flat plate at height x
Nu_L = average Nusselt number for a vertical flat plate at height L
Nu_{D0} = local Nusselt number at midsection of a horizontal cylinder (α = 0)
Nu_D = average Nusselt number for a horizontal cylinder with outer diameter of D
p, q, and r = constant parameters

Based on the discussion for the thermal boundary layer thickness over a vertical flat plate above, the constant parameter p in equation (4) is 0.25.

An extensive study on free convection over large diameter, horizontal cylinders conducted in reference [48] shows that the onset of turbulent transition occurs at a point passing the midsection of the cylinder by five degree even for large Rayleigh numbers so that the free convection over the lower half of the cylinder remains laminar.

Since the personnel barrier designed for MP197HB cask is extended only to the midsection of the cask, the correlations for free convection laminar flow over horizontal cylinders can be used to determine the thermal boundary layer thickness at this location.

The free convection local Nusselt number over a horizontal cylinder in laminar flow in air is given in reference [47] as follows.

$$Nu_{D\alpha} = 0.604\, Gr_D^{1/4}\, \phi(\alpha) \qquad [47]$$

$$Gr_D = \frac{g \beta \Delta T\, D^3}{\upsilon^2}$$

α	-90°	-60°	-30°	0°	30°	60°	75°	90°
$\phi(\alpha)$	0.76	0.75	0.72	0.66	0.58	0.46	0.36	0
	Bottom half			Top half				

The local Nusselt number at the midsection of the cylinder at α = 0 is:

$$Nu_{D0} = 0.604\, Gr_D^{1/4} \times 0.66$$

Based on the above correlation, the average Nusselt number over the horizontal cylinder is:

$$Nu_D = \frac{1}{180}\int_{-90}^{90} Nu_{D\alpha}\, d\alpha = 0.604\, Gr_D^{1/4} \times \frac{1}{180}\int_{-90}^{90} \phi(\alpha)\, d\alpha$$

The integration of $\phi(\alpha)$ over the range of -90° to 90° performed using the data in the above table gives:

$$Nu_D = 0.604\, Gr_D^{1/4} \times 0.6025$$

Comparison of the correlations for Nu_{D0} and Nu_D gives:

$$Nu_D = \frac{0.6025}{0.66} Nu_{D0} = 0.913\, Nu_{D0} \qquad (5)$$

Since the equation (5) is based on free convection laminar flow, the constant parameter r in equation (4) is equal to 1.

The ratio of the average Nusselt numbers for vertical flat plate and horizontal cylinder can be determined using the corresponding correlations shown in Section A.3.3.1.1.

The thickness of the boundary layer at the midsection of the cask can be determined by substitution of the correlations for the local and average Nusselt numbers into equation (4). The height (L) of the vertical flat plate can be set equal to the outer diameter of the cask (D) for this evaluation. Average of the cask outer surface and ambient temperatures are considered in calculation of the Nusselt numbers.

As seen above, the correlations for the local and average Nusselt numbers depend on the Grashof and Prandtl numbers, which in turn depend on the cask outer surface temperature. A sensitivity analysis is performed to cover the effects of a wide range of cask temperatures from 200°F to 500°F on the thickness of free convection thermal boundary layer at the midsection of the cask. This sensitivity analysis starts with variation of Grashof and Prandtl numbers as summarized in the following table.

Variation of Grashof, Prandtl, and Rayleigh Numbers

T_∞ =100°F
For L=D

T_{cask}	T_{cask}	T_∞	T_{avg}	β	ν	Gr	Pr	Ra
(F)	(K)	(K)	(K)	(1/K)	(m^2/s)	(---)	(---)	(---)
200	367	311	339	2.95E-03	2.235E-05	4.931E+10	0.71	3.490E+10
300	422	311	367	2.73E-03	2.849E-05	5.609E+10	0.70	3.948E+10
400	478	311	394	2.54E-03	3.516E-05	5.135E+10	0.70	3.602E+10
500	533	311	422	2.37E-03	4.232E-05	4.415E+10	0.70	3.091E+10

The values in the above table are calculated based on air properties shown in Section A.3.2.1, Item 16, at the average air temperature. As seen in the above table, the Grashof number varies between 5.6E10 and 4.4E10, the Prandtl number varies between 0.70 and 0.71, and the Rayleigh number varies between 3.1E10 and 4.0E10 for the cask outer surface temperatures between 200°F and 500°F. Since the variation of Prandtl is relative small, an average Prandtl number of 0.70 is considered in the sensitivity analysis to calculate the average Nusselt numbers for a vertical flat plate (Nu_L) and a horizontal cylinder (Nu_D). To bound the variation of the Rayleigh number conservatively, the average Nusselt numbers Nu_L and Nu_D are evaluated for a wider range between 1E10 and 1E11.

As shown in Section A.3.3.1.1, the correlation for Nu_L depends on C_t^V and f factors, which in turn depend on Prandtl number and surface temperature, respectively. C_t^V is a weak function of Prandtl number. The variation of the f factor is determined for a cask outer surface temperature from 200°F to 500°F. The variations of these values are summarized in the following table.

Variation of C_t^V and f in Calculation of Nu_L

T_∞ =100°F

Pr	C_t^V	T_{cask}	f
(---)	(---)	(°F)	(---)
0.70	0.103	200	1.014
0.71	0.103	300	1.028
Average	0.103	400	1.042
		500	1.056
		Average	1.035

The average values of C_t^V and f from the above table are considered in calculation of Nu_L. The variation of the average Nusselt numbers Nu_L and Nu_D and their ratios summarized in the table below.

Variation of Nu_L/Nu_D

Vertical Flat Plate
Pr = 0.70

Ra	Nu^T	Nu_l	C_t^V	f	Nu_t	Nu_L
(---)	(---)	(---)	(---)	(---)	(---)	
1.00E+10	162.86	163.9	0.103	1.035	208.5	216.0
2.00E+10	193.67	194.7	0.103	1.035	275.0	280.5
5.00E+10	243.53	244.5	0.103	1.035	384.0	388.1
8.00E+10	273.89	274.9	0.103	1.035	452.4	456.1
1.00E+11	289.61	290.6	0.103	1.035	488.5	492.0

Horizontal Cylinder
Pr = 0.70

Ra	Nu^T	F	Nu_l	$\overline{C}_t$	Nu_t	Nu_D	Nu_L/Nu_D
(---)	(---)	(---)	(---)	(---)	(---)	(---)	(---)
1.00E+10	125.73	0.940	126.7	0.103	221.9	222.0	0.973
2.00E+10	149.51	0.942	150.5	0.103	279.6	279.6	1.003
5.00E+10	188.00	0.944	188.9	0.103	379.5	379.5	1.023
8.00E+10	211.44	0.945	212.4	0.103	443.8	443.8	1.028
1.00E+11	223.58	0.945	224.5	0.103	478.1	478.1	1.029
						Average	1.011

As shown in the above table, the ratio of Nu_L to Nu_D varies between 0.973 and 1.029 for the range of considered Rayleigh numbers. An average value of 1.011 is considered for this ratio to use in equation (4). Since this ratio is close to one, the constant parameter q in equation (4) does not have any significant effect and can be omitted.

Substitution of the local and average Nusselt number ratios from equation (3), equation (5), and the table of Nu_L/Nu_D variations into equation (4) gives:

$$\delta_{D0} = (1/0.834)^{0.25} \times 1.011 \times 0.913\ \delta_x$$

Using δ_x from equation (2) in the above equation determines the range of boundary layer thickness at the midsection of the cask as summarized in the following table.

Thickness of the Thermal Boundary Layer

T_∞=100°F
L = D = 97.75"

T_{cask}	T_{cask}	T_∞	T_{avg}	Gr	Pr	δ_x/L	δ_x	δ_{D0}
(°F)	(K)	(K)	(K)	(---)	(---)	(---)	(in)	(in)
200	367	311	339	4.931E+10	0.71	0.060	5.9	**5.7**
300	422	311	367	5.609E+10	0.70	0.059	5.8	**5.6**
400	478	311	394	5.135E+10	0.70	0.060	5.9	**5.7**
500	533	311	422	4.415E+10	0.70	0.061	6.0	**5.7**

As seen in the above table, the variation of the thermal boundary layer thickness is small and its maximum value at the midsection of the cask is 5.7" for a uniform cask surface temperature of 500°F. Based on data shown in Section A.3.3.1.2, the shortest distance between the outer surface of the cask and the personnel barrier is over 9".

Shortest distance = distance to cask centerline – cask OD/2 = 58 – 97.75 / 2 = 9.125"

The conservatively evaluated boundary layer thickness of 5.7" is much smaller than shortest distance between the outer surface of the cask and the personnel barrier. Therefore, the personnel barrier remains out of the hot air streams flowing around the cask outer surface.

A.3.3.1.3 Effective Thermal Properties in MP197HB TC Model

1) Effective Heat Transfer Coefficient for External Fins

To reduce the complexity of the TC model, an effective heat transfer coefficient is calculated for the external fins based on the geometry shown in the drawings in Chapter A.1, Appendix A.1.4.10. Circular external fins are welded to an aluminum shell which will be installed over the outer surface of the TC shield shell to enhance heat dissipation for heat loads over 26 kW. An effective heat transfer coefficient is calculated for the external fins which includes the convection and radiation heat transfer to ambient. The following dimensions are considered for the fins.

- Fin height = 3.0"
- Fin thickness = 0.156"
- Fin pitch = 1.0"

A sub-model of the TC outer surface is developed for this purpose using ANSYS [27].

This sub-model considers a 30 degree segment of the aluminum shell with three circular external fins. Figure A.3–10 shows the sub-model of the finned shell.

Convection boundary conditions are applied over the outer surfaces of the fins in the model using surface load (SF) commands in ANSYS [27]. SHELL57 elements are overlaid on the external surfaces of the fins to create radiation super-element. The radiation shell elements are shown in Figure A.3–10. Thermal radiation from the outer surfaces is modeled using /AUX12 processor. Ambient temperatures of 100°F, -20°F, and -40°F are considered for convection and radiation.

An emissivity of 0.70 is considered for the anodized aluminum for exposed finned surfaces as shown in Section A.3.2.1.

$$k_{eff,axl} = k_{eff,axl,1} \times \frac{\theta_{Al}}{\theta_{nom}}$$

with $\theta_{nom} = \frac{360}{40} = 9^\circ$

$$\theta_{Al} = \theta_{nom} - \frac{gap_{rad}}{D_{o,sleeve}/2} \times \frac{180}{\pi} = 8.6^\circ$$

gap_{rad} = radial gap =0.25"
$D_{o,sleeve}$ = 70.5"

Radial Effective Conductivity

Forty aluminum segments in the radial direction build up parallel resistances perpendicular to the direction of heat flow from the center to the periphery. Again, the total radial effective conductivity can be set proportional to the ratio of the angle for one aluminum segment (θ_{Al}) to the nominal angle of each segment (θ_{nom}).

$$k_{eff,rad} = k_{Al} \times \frac{\theta_{Al}}{\theta_{nom}}$$

The effective conductivity values calculated for the sleeve are summarized in Section A.3.2.1, material # 31.

A.3.3.1.4 69BTH DSC Model

The following assumptions and conservatism are considered for the 69BTH DSC model:

The fuel assemblies contained in the DSC basket are intact fuel assemblies. Since the damaged fuel assemblies are loaded in the outermost fuel compartment cells, they do not affect the maximum temperatures or the maximum temperature gradients in this evaluation. *A sensitivity analysis is conducted to bound the effects of the damaged fuel assemblies on the thermal performance of the MP197HB cask considering the worst case condition, in which the high burnup damaged fuel assemblies becomes rubble. This sensitivity analysis is discussed in Section A.3.6.9.*

No convection is considered within the canister cavity.

Only helium conduction is considered from the basket upper surface to the canister top shield plug.

Radiation is considered only implicitly between the fuel rods and the fuel compartment walls in the calculation of effective fuel conductivity. No other radiation heat exchange is considered within the basket model.

Active fuel length for BWR fuel assemblies is 144" [11] and starts about 7.5" from the bottom of the basket [11]. The total length of the basket assembly is 176.5".

The following gaps are considered in the DSC canister/basket model at thermal equilibrium:

a) 0.30" diametrical hot gap between the basket outer surface and the canister inner surface. This gap is justified in Section A.3.6.7.3.

b) 0.125" axial gap between the bottom of the basket and the DSC bottom inner cover plate

c) 0.01" gap between any two adjacent plates or components in the cross section of the basket.

d) 0.125" gap in axial direction between the aluminum rail pieces.

e) 0.01" gap between the sections of the paired aluminum and poison plates in axial direction.

f) 0.1" gap between the two small aluminum rails at the basket corners.

g) 0.1" gap between the two pieces of large aluminum rails at 0° -180° and 90° - 270° orientations.

h) 0.0625" gap between DSC shield plugs and DSC cover plates for calculation of effective conductivities in axial direction.

No gap is considered between the paired poison and aluminum plates. The 0.01" gaps considered on either side of the paired plates account for the thermal resistance between the multiple plates. This assumption is justified in Section A.3.6.7.4.

The gaps considered between the aluminum rail segments are larger than the nominal cold gaps and are therefore conservative. The axial gaps considered between the aluminum rail pieces in the axial direction are larger than the tolerances considered for the rails and are therefore conservative.

The benchmarking of finite element models against test data in [35] shows that the 0.01" gaps considered between adjacent plates or components in the cross section of the basket account conservatively for the tolerances and contact resistances.

In the fabrication, the diametrical gap between the basket and canister shell assigned as gap "a" above is controlled by dimensional inspections of the diameters of the basket and canister shell.

The structure of the 69BTH basket is similar to the 61BT and 61BTH baskets approved in accordance with 10 CFR 72 regulations. The uniform gap of 0.01" assigned as gap "c" above considered in the thermal model between any two adjacent components in the cross section of the 69BTH basket has the same size as the corresponding gaps considered in the 61BT and 61BTH baskets. In practical terms, fabrication of the 69BTH basket requires very tightly compressed assembly in order to fit the basket into the shell. Interfaces are formed as components and parts are assembled. The fit between mating components, for example between fuel compartment tubes and adjacent sheets, cannot practically be measured. Fabrication methods provide for the tightest practical assembly of these parts.

The gaps between adjacent components are related only to the flatness and roughness tolerances of the plates. The micro gaps related to these tolerances are non-uniform and provide interference contact at some areas and gaps on the other areas as shown schematically in the figure in Section A.3.6.7.2. For the purpose of thermal evaluation, surfaces of intermittent contact between adjacent components are conservatively *modeled as a uniform gap of 0.01". As shown in SAR Section A.3.6.7.4, the assumed gap size of 0.01" is approximately two times larger than the contact resistances between the adjacent components. It should be noted that for conservatism no contact pressure was considered between the components. This assumption implies that no friction exists between the components within the basket, which adds to the conservatism considered in the size of this uniform gap. In reality, there is sufficient friction that 61BT baskets have been lifted during fabrication using only the friction on the perimeter of the four-compartment subassemblies.*

The 0.01" axial gaps between the sections of the paired aluminum and poison sheets assigned as gap "e" above are shown in SAR Figure A.3-17. The 0.1" gaps between the rail segments assigned as gaps "f" and "g" above are shown in SAR Figure A.3-15. These gaps are not located in the primary heat flow paths. A sensitivity analysis is performed to determine the effect of these gaps on the thermal performance. The results of this sensitivity analysis show that doubling the size of these gaps increases the maximum temperatures by less than 1 °F. Therefore, the effects of these gaps on the thermal performance are insignificant.

The thickness of paired aluminum and poison plates within the wrapped compartment blocks of 69BTH basket is 0.25". This thickness is reduced to 0.21" to accommodate for the size of the gaps and maintain the outer basket diameter contained within the DSC inner diameter. An effective conductivity is calculated for these plates in Section A.3.3.1.5 to maintain the conductivity of plates within the basket. All other dimensions are based on nominal dimensions for 69BTH basket.

Paired aluminum and poison plates are considered as one homogenized material in the 69BTH basket model. The effective conductivities for paired aluminum poison plates are calculated in Section A.3.3.1.5.

To reduce the complexity of the 69BTH basket model, the contact resistances between the DSC shield plugs and DSC cover plates are integrated into the bottom shield plug and top inner cover plate. Axial effective conductivities are calculated for top and bottom shield plugs of DSC in Section A.3.3.1.3 and listed in Section A.3.2.1. The conductivities of these components remain unchanged in the radial direction.

Decay heat load is applied as heat generation boundary conditions over the elements representing homogenized fuel assemblies. The base heat generation rate is multiplied by peaking factors along the axial fuel length to represent the axial decay heat profile. A correction factor is used to avoid degradation of decay heat load due to imperfections in application of peaking factors. The heat generation rates used in this analysis is calculated as follows.

cladding temperatures loaded in the DSCs. Therefore, no additional analysis is required for the secondary containers.

As calculated in Section A.3.3.1.2, the maximum accessible surface temperatures for impact limiter and personnel barrier are 121°F and *163*°F, respectively. These temperatures are well below the limit of 185°F defined in Section A.3.1.

The thermal analysis of NCT demonstrates that the MP197HB TC with up to 32 kW heat load meets all applicable requirements. The highest maximum temperatures are summarized in Table A.3–11.

The maximum temperatures calculated using conservative assumptions are well below specified limits. The seal O-rings are not explicitly considered in the models. The maximum seal temperatures are retrieved from the models by selecting the nodes at the locations of the corresponding seal O-rings. The maximum seal temperature (382°F, 194°C) for NCT is below the long-term limit of 400°F (204°C) specified for continued seal function.

The maximum neutron shield temperature is 290°F (143°C) for NCT, which is below the long term limit of 320°F (160°C). No degradation of the neutron shielding is expected.

The maximum temperature of gamma shield is 397°F (203°C) for NCT, which is well below the melting point of lead (621°F, 327°C).

The predicted maximum fuel cladding temperature *of 674*°F *(357*°C) *for the maximum heat load of 32kW* is well within the allowable fuel temperature limit of 752°F (400°C) for NCT.

The temperature distributions for NCT with 100°F ambient and insolation are shown in Figure A.3–29 to Figure A.3–34.

Under the minimum ambient temperature of -40°F (-40°C), the resulting packaging component temperatures will approach -40°F if no credit is taken for the decay heat load. Since the package materials, including containment structures and the seals, continue to function at this temperature, the minimum temperature condition has no adverse effect on the performance of the MP197HB TC.

The maximum component temperatures for ambient temperatures of -40°F and -20°F with maximum decay heat and no insulation are calculated for 69BTH DSC and 37PTH DSC to use for structural evaluations. These temperatures are listed in Table A.3–12 and Table A.3–13.

The average temperatures of helium gas in TC cavity, and the average temperatures of fuel assemblies and helium within 37PTH and 69BTH DSC cavities for NCT are listed in Table A.3–14. These temperatures are used to evaluate the maximum internal pressures within TC and DSC cavities.

Thermal stresses for the MP197HB TC loaded with DSC are discussed in Chapter A.2. The maximum normal operating pressure for the MP197HB TC is discussed in Section A.3.3.3. The performance of the MP197HB TC loaded with DSCs for HAC is discussed in Section A.3.4.

5) Quantity of Gases released as a Result of Irradiation

For the 69BTH DSC, the quantity of gases released as a result of irradiation for the "generic" BWR fuel assembly is 20.2 g-moles as shown in Appendix T, Section T.4.6.6.4 associated with Amendment 10 to Part 72 CoC 1004 for the Standardized NUHOMS® System [3]. The total gas moles (per fuel assembly) are then multiplied by the 30% release fraction to obtain the number of moles contributing to 69BTH DSC cavity gas pressure.

The amount of gas released from one assembly because of irradiation is adjusted to burnup of 70,000 MWD/MTU as follows.

$\boldsymbol{n}_{ig\ 1\ FA}$ = *20.2 g-moles*70,000/62,000 * 0.3 = 6.84 g-moles*

Table A.3–22 presents the amount of gas released into the DSC cavity by fuel assemblies n_{ig} for normal and accident conditions assuming a 30% gas release from the fuel pellets [9] and a 3% and 100% rod rupture percentage, respectively.

6) Maximum Normal Operating Pressure Calculation

Calculation of the maximum pressure in the 69BTH DSC HLZC #1 for normal conditions of transport is shown below. In accordance with [9], 3% of the fuel rods are assumed to be ruptured. The total amount of gas in the DSC cavity is therefore:

$$n_{DSC-NCT} = n_{he\ initial} + n_{he\ fuel\ rod\ release} + n_{ig},$$

$$n_{DSC-NCT} = 134.39 + 3.27 + 14.16 = 151.82 \quad g - moles$$

Ruptured pins will vent plenum gas until it comes into equilibrium with the DSC pressure; therefore, the plenum volume within the ruptured pins can be included in the total DSC internal volume. For a 3% pin rupture the additional volume is therefore:

$$V_{pin\ plenum} = 100\,pins/assy \cdot 2.136\,in^3/pin \cdot 69\,assy/basket \cdot 0.03 = 442.15\ \ in^3.$$

The maximum normal operating pressure (MNOP) for this configuration is then,

$$P_{DSC-NCT} = \frac{\left(1.4504 \cdot 10^{-4}\,\frac{psia}{Pa}\right)(151.82\,g-moles)(8.314\,J/(mol \cdot K))(984\,°R)(5/9\,K/°R)}{(258413\,in^3 + 442.15\,in^3)(1.6387 \cdot 10^{-5}\,m^3/in^3)},$$

$$P_{DSC-NCT} = 23.59\,psia\,(8.89\,psig)$$

The maximum pressures are summarized in Table A.3–22. As seen from Table A.3–22, the maximum internal pressures in the 69BTH DSC *calculated based on thermal conditions are lower than the design pressures considered for the structural evaluation.*

A.3.3.3.3 37PTH DSC Operating Pressure

The maximum internal pressure for the 37PTH DSC within the MP197HB TC for NCT is determined based on the maximum allowable heat load of 22 kW discussed in Section A.3.3.1.6 and maximum burnup of 65 GWD/MTU. Although BW 15x15 fuel assembly is not allowed for transport in the 37PTH DSC, it is considered as the limiting fuel assembly type in this evaluation for conservatism.

6) Maximum Normal Operating Pressure Calculation

Calculation of the maximum pressure in 37PTH DSC follows the same methodology as used for 69BTH DSC in Section A.3.3.3.2.

The maximum pressures for 37PTH DSC are summarized in Table A.3–22. As seen from Table A.3-22, the maximum internal pressures in 37PTH DSC *calculated based on thermal conditions are lower than the design pressures considered for the structural evaluation.*

A.3.3.3.4 Internal Pressure for DSCs analyzed for Storage/Transfer

The maximum internal pressures for 61BTH, 61BT, 32PTH, 32PTH1, 32PT, 24PTH, and 24PT4 DSCs for storage and transfer conditions under 10 CFR 72 requirements are determined in [1], [2],[3], and [4].

Based on discussions in Section A.3.3.2 the maximum fuel cladding and basket component temperatures for 61BTH, 61BT, 32PTH, 32PTH1, 32PT, 24PTH, and 24PT4 DSC in MP197HB TC for transport conditions (10 CFR 71) are bounded by temperatures for storage and transfer conditions (10 CFR 72) and no DSC thermal analysis is required. It is also applicable to the average gas temperatures in DSC cavity. Therefore, the internal pressure in a DSC for NCT with 3% ruptured fuel rods can be evaluated interpolating between the internal pressures calculated for normal storage/transfer conditions with 1% ruptured fuel rods and off-normal storage/transfer conditions with 10% ruptured fuel rods. Higher average helium temperatures for accident storage and transfer conditions compared to normal temperatures provide additional conservatism in the internal pressure calculation for NCT.

Since 100% percent of ruptured fuel rods is assumed for both transfer accident conditions and transport HAC and maximum DSC shell temperatures for transfer accident conditions bound those for transport HAC as noted in Section A.3.4 (which results in lower average DSC helium temperature), the internal pressures calculated for storage licensed DSCs for transfer accident conditions bound the internal pressures for transport in MP197HB TC during HAC.

As seen from Table A.3-23, the maximum internal pressures in storage licensed DSCs *calculated based on thermal conditions are lower than the design pressures considered for the structural evaluation.*

A.3.3.3.5 Operating Pressures for 61BTHF and 24PTHF DSCs

As shown in Chapter 4, Table 4-28 of [4] for 32PTH DSC, a loading configuration with 50% intact and 50% damaged fuel assemblies results in negligible increase (~0.1 psi) in maximum internal pressure for bounding accident conditions.

The number of failed and damaged fuel assemblies for 61BTHF and 24PTHF DSCs is 16 and 12, respectively. Therefore, the maximum pressures calculated for 61BTH DSC and 24PTH DSC reported in Table A.3–23 can be used for 61BTHF and 24PTHF DSCs. As seen from Table A.3–23, the maximum internal pressures in storage licensed DSCs *calculated based on thermal conditions are lower than the design pressures considered for the structural evaluation.*

A.3.3.4 Thermal Evaluation for Loading/Unloading Operations

Vacuum drying is considered as a normal condition for wet loading operations. The fuel transfer operations for wet loading occur when the MP197HB and the loaded DSC are in the spent fuel pool. The fuel is always submerged in free-flowing pool water permitting heat dissipation. After completion of fuel loading, the TC and DSC are removed from the pool and the DSC is drained, dried, sealed and backfilled with helium. These operations occur when the annulus between the TC and DSC remains filled with water.

The water in the annulus is replenished with fresh water to prevent boiling and maintain the water level if excessive evaporation occurs. Presence of water within the annulus maintains the maximum DSC shell temperature below the boiling temperature of water in open atmosphere (212°F).

Water in the DSC cavity is forced out of the cavity (blowdown operation) before the start of vacuum drying. Helium is used as the medium to remove water and subsequent vacuum drying occurs with a helium environment in the DSC cavity. The vacuum drying operation does not reduce the pressure sufficiently to reduce the thermal conductivity of the helium in the canister cavity *(*[3], Appendix T, Section T.4 based on [5], [32], and [33]*)*.

With helium being present during vacuum drying operations, the maximum temperatures including the maximum fuel cladding temperature are bounded by those calculated for transport operation if the DSC shell temperature under NCT is higher than the DSC shell temperature of 212°F maintained during vacuum drying. As shown in Table A.3–8 and Table A.3–9 for all DSCs in MP197HB TC, all DSC shell minimum temperatures are higher than 212°F. Therefore, no additional thermal evaluation is needed.

Presence of helium during blowdown and vacuum drying operations eliminates the thermal cycling of fuel cladding during helium backfilling of the DSCs subsequent to vacuum drying. Therefore, the thermal cycling limit of 65°C (117°F) for short term operations set by ISG-11 [7] is satisfied for vacuum drying operation in MP197HB.

The bounding unloading operation considered is the reflood of the DSCs with water. For unloading operations, the DSC is filled with the spent fuel pool water through its siphon port. During this filling operation, the DSC vent port is maintained open with effluents routed to the plant's off-gas monitoring system.

The maximum fuel cladding temperature during reflooding event is significantly less than the vacuum drying condition owing to the presence of water/steam in the canister cavity. Based on the above rational, the maximum cladding temperature during unloading operation is bounded by the maximum fuel cladding temperature for vacuum drying operation.

Initially, the pool water is added to the canister cavity containing hot fuel and basket components, some of the water will flash to steam causing internal cavity pressure to rise. This steam pressure is released through the vent port. The procedures specify that the flow rate of the reflood water be controlled such that the internal pressure in the canister cavity does not exceed *the maximum pressure specified for reflooding* *operations as noted in Chapter A.7, Appendices A.7.7.1 through A.7.7.9*. This is assured by monitoring the maximum internal pressure in the canister cavity during the reflood event. The reflood for the DSC is considered as a Service Level D event and the design pressures of the DSCs are well above *15 psig for the 32PTH DSC and 20 psig for the other DSCs (see Chapter A.7, Appendices A.7.7.1 through A.7.7.9)*. Therefore, there is sufficient margin in the DSC internal pressure during the reflooding event to assure that the canister will not be over pressurized.

The effects of the thermal loads on the fuel cladding during reflooding operations are evaluated in Appendix T, Section T.4.7.3 and Appendix U, Section U.4.7.3 for BWR and PWR fuel assemblies respectively, associated with Amendment 10 to Part 72 CoC 1004 for the Standardized NUHOMS® System [3]. Since the same fuel assemblies are handled in the DSCs contained in MP197HB, these evaluations remain valid for this calculation.

The decay heat generation rates used in the transient model are listed below.

Decay Heat Generation Rate

DSC Type	Heat Load (kW)	Heat Load (Btu/hr)	D_i (in)	L_b (in)	Decay heat Generation Rate (Btu/hr-in^3)
69BTH	26.0	88,720	68.75	164	0.1457
	32.0	109,194	68.75	164	0.1794
24PTH	26.0	88,720	66.19	168.60	0.1529

All the assumptions and conservatism described in Section A.3.3.1.1 for the MP197HB model are valid for determination of initial conditions.

A.3.4.2 Fire Test Conditions

No fire test is performed. Instead, the fire conditions are simulated using the finite element model of the MP197HB TC.

Based on the requirements in 10 CFR 71, part 73 [6], a fire temperature of 1475 °F, fire emissivity of 0.9 and a period of 30 minutes are considered for the fire conditions. A bounding forced convection coefficient of 4.5 Btu/hr-ft^2-°F is considered during burning period based on data from reference [13]. Surface emissivity of 0.8 is considered for the packaging surfaces exposed to fire based on 10 CFR 71, part 73 [6].

The total heat transfer coefficient during fire is determined using the following equations.

$$h_{t,fire} = h_{r,fire} + h_{c,fire}$$

Where,

$h_{r,fire}$ = fire radiation heat transfer coefficient (Btu/hr-in^2-°F)

$h_{c,fire}$ = forced convection heat transfer coefficient during fire = 4.5 Btu/hr-in^2-°F

The radiation heat transfer coefficient, $h_{r,fire}$, is given by the equation:

$$h_{r,fire} = \varepsilon_w \, F_{wf} \left[\frac{\sigma(\varepsilon_f T_f^{\,4} - T_w^{\,4})}{T_f - T_w} \right] \quad \text{Btu/hr-in}^2\text{-°F}$$

where,

ε_w = TC outer surface emissivity = 0.8 [6]

ε_f = fire emissivity = 0.9 [6]

F_{wf} = view factor from TC surface to fire = 1.0

$\sigma = 0.1714 \times 10^{-8}$ Btu/hr-ft^2-°R^4

T_w = surface temperature (°R)

T_f = fire temperature = 1475°F = 1,935°R

The sensitivity study that documents the effects of fire emissivity of 1.0 on the thermal performance of the MP197HB TC is documented in Appendix A.3.6.8.

The following gaps are reduced from 0.0625" under NCT to 0.01" under HAC to maximize the heat input from the fire toward the cask after free drop:

a) 0.01" axial gap between thermal shield and impact limiter case
b) 0.01" axial gap between thermal shields and cask top or bottom end surface

$$P_{DSC-HAC} = \frac{\left(1.4504 \cdot 10^{-4} \frac{psia}{Pa}\right)(715.39\, g - moles)(8.314\, J/(mol \cdot K))(992\,°R)(5/9\, K/°R)}{(258415\, in^3 + 14738.40\, in^3)(1.6387 \cdot 10^{-5}\, m^3/in^3)},$$

$$P_{DSC-HAC} = 106.17\, psia\ (91.47\, psig).$$

The maximum internal pressures inside 37PTH and 69BTH DSCs are summarized in Table A.3–22. The maximum internal pressures are 102.64 psig and 95.54 psig for the 37PTH DSC and 69BTH DSC under HAC, respectively. These maximum *calculated* internal DSC pressures are below the design *pressure of 140 psig specified in Section A.3.1 for HAC structural evaluations of the* 69BTH and 37PTH DSCs.

Based on discussions in Section A.3.3.3.4, the maximum internal pressures for DSCs analyzed for accident storage/transfer conditions under 10 CFR 72 requirements remain bounding for transport conditions under HAC. The evaluated DSC internal pressures for DSCs to be transported within MP197HB TC are summarized in Table A.3–23. As seen in Table A.3–23, the maximum *calculated* internal pressures under HAC remains below the corresponding design *pressures considered for the structural evaluations for HAC* for all DSC types.

A.3.4.4 Maximum Thermal Stresses

Thermal stresses for the MP197HB TC loaded with DSCs are discussed in Chapter A.2.

A.3.4.5 Accident Conditions for Fissile Material Packages for Air Transport

The MP197HB TC is not designed for air transportation. Therefore, the accident conditions for air transport are irrelevant.

A.3.5 References

1. Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUH-003, Rev. *11*.
2. Updated Final Safety Analysis Report for the Standardized Advanced NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, ANUH-01.0150, Rev. 3.
3. *Same as Reference 1.*
4. Final Safety Analysis Report for NUHOMS® HD Horizontal Modular Storage System for Irradiated Nuclear Fuel, Rev. *2*.
5. Perry, Chilton, Chemical Engineers' Handbook, 5th Edition, 1973.
6. U.S. Code of Federal Regulations, Part 71, Title 10, "Packaging and Transportation of Radioactive Material."
7. U.S. NRC, Spent Fuel Project Office, Interim Staff Guidance, ISG-11, Rev 3, Cladding Considerations for the Transportation and Storage of Spent Fuel.
8. U.S. Code of Federal Regulations, Part 72, Title 10, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.
9. U.S. Nuclear Regulatory Commission, Standard Review Plan for Transportation Packages for Spent Nuclear Fuel, Final Report, NUREG-1617, January 2000.
10. ASME Boiler and Pressure Vessel Code, Section II, Part D, Material Properties, 2004.
11. Moore, R. S., Notz, K. J., Physical Characteristics of GE BWR Fuel Assemblies, Oak Ridge National Laboratory, Report No. ORNL/TM-10902, June 1989.
12. Report, Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages, Office of Civilian Radioactive Waste Management, DOE/RW-0472, Revision 2, September 1998.
13. Gregory, et al., Thermal Measurements in a Series of Long Pool Fires, SANDIA Report, SAND 85-0196, TTC-0659, 1987.
14. Mitchell S. Sweet, Fire Performance of Wood: Test Methods and Fire Retardant Treatments, Fire Safety of Wood Products, USDA Forest Service, http://www.fpl.fs.fed.us/documnts/pdf1993/sweet93a.pdf
15. Bucholz, J. A., Scoping Design Analysis for Optimized Shipping Casks Containing 1-, 2-, 3-, 5-, 7-, or 10-Year old PWR Spent Fuel, Oak Ridge National Laboratory, January, 1983, ORNL/CSD/TM-149.
16. NOT USED

37. U.S. Department of Energy, "Characteristics of Spent Nuclear Fuel, High- Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation, Appendix 2A," U.S Department of Energy, Office of Civilian Radioactive Waste Management, DOE/RW-0184, Volume 3 of 6, December, 1987.
38. M. M. Yovanovich, J. R. Culham, P. Teertstra, "Calculating Interface Resistance," Electronics Cooling, Vol. 3, No. 2, May 1997.
39. Amiss, J. M., et al., "Machinery's Handbook," 24th Edition, Industrial Press, 1992 - Fig. 5, pg 672.
40. Aluminum Association, Inc., "Aluminum Standards and Data," 10th Edition, 1990 - Table 2.1, pg 33.
41. ASME Boiler and Pressure Vessel Code, Section II, Part A, "Ferrous Material Specification," 1998 and 2000 addenda - Table 2.
42. Gordon England Company, "Microhardness Test," http://www.gordonengland.co.uk/hardness/microhardness.htm
43. *Eckert, "Introduction to the Transfer of Heat and Mass," 1st Edition, 1950.*
44. *Eckert and Jackson, "Analysis of Turbulent Free-Convection Boundary Layer on Flat Plate," NASA Technical Reports, Report Number NACA-TN-2207, 1950.*
45. *Kays, Crawford, and Weigand, "Convective Heat and Mass Transfer," 4th Edition, 2005.*
46. *Holman, "Heat Transfer," 8th Edition, 1997.*
47. *Kreith, "Principles of Heat Transfer," 3rd Edition, 1973.*
48. *Misumi, Suzuki, and Kitamura, "Fluid Flow and Heat Transfer of Natural Convection around Large Horizontal Cylinders: Experiments with Air," Heat Transfer --Asian Research, 32 (4), 2003.*
49. *T. Eric Skidmore, "Performance Evaluation of O-Ring Seals in Model 9975 Packaging Assemblies (U)," Savannah River Technology Center, WSRC-TR-98-00439, December 1998.*
50. *Oak Ridge National Laboratory, "Effect of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks," by K.R. Elam, J.C. Wagner and C.V. Parks, NUREG/CR-6835 (ORNL/TM-2002/255), September 2003.*
51. *H. J. Rack, G. A. Knorovsky, "An Assessment of Stress-Strain Data Suitable for Finite-Element Elastic-Plastic Analysis of Shipping Containers," Sandia Laboratories, NUREG/CR-0481, SAND77-1872 R-7, 1978.*

Proprietary information on pages A.3-97 to A.3-102 withheld pursuant to 10 CFR 2.390

between ~28.0" to ~100.0" for PWR fuel assemblies. These locations are measured from the bottom of active fuel length. The active fuel length starts approximately 7.5" and 4.0" measured from the bottom of the fuel assembly for BWR and PWR fuel assemblies, respectively.

To quantify the differences between the DSC shell temperatures under NCT and normal transfer conditions, the maximum DSC shell temperature for each DSC type under NCT is retrieved at the above locations from the MP197HB TC model and shown in Table A.3–8 and Table A.3–9. These temperatures give the maximum DSC shell temperature in the region of highest peaking factors, where the maximum fuel cladding temperatures are expected

The maximum DSC shell temperatures for NCT under 10 CFR 71 requirements are compared to the corresponding data for transfer conditions under 10 CFR 72 requirements in Table A.3–24.

As shown in Table A.3–24, the maximum DSC shell temperature for NCT at the mid section, where the highest peaking factors are located, is 7 to 17°F lower than the absolute maximum DSC shell temperature for the NCT. Both of these values are lower than the maximum DSC shell temperature for normal transfer conditions.

Since the DSC shell temperatures for NCT at the ends and at mid section are lower than those for the normal transfer conditions, the DSC shell temperature profile for normal transfer conditions gives the bounding values for the basket and fuel cladding temperatures.

For DSC types, 61BTH Type 2, 32PTH, 32PTH Type 1, 32PTH Type 1 and 2, 24PTH-S, and 24PTH-L the maximum heat loads for transport conditions are lower than the maximum heat loads for transfer conditions. Therefore for these DSC types, even lower basket and fuel cladding temperatures are expected for NCT.

Based on this discussion, the thermal analysis results for DSCs in 10 CFR 72 SARs ([1], [2], [3] and [4]) under normal transfer conditions are applicable for NCT and represent the bounding fuel cladding and basket component temperatures.

Based on the comparison shown in the lower half of Table A.3–24, the maximum DSC shell temperatures for HAC under 10 CFR 71 requirements are also lower than the corresponding ones for accident transfer conditions under 10 CFR 72 requirements. The same arguments as above are therefore valid for HAC of transport as well. Therefore, the thermal analysis results for DSCs in 10 CFR 72 SARs ([1], [2], [3] and [4]) under accident transfer conditions are bounding for HAC and no further thermal analyses are required for these DSC types.

To provide additional assurance that the above arguments are valid and the fuel cladding and the basket component temperatures in 10 CFR Part 72 SARs represent the bounding values for transport conditions, the DSC type 24PTH-S (without Al inserts) is selected for evaluation under NCT.

Among the DSC types previously evaluated for storage applications and proposed for transport in MP197HB, DSC type 24PTH-S (without Al inserts) has the smallest margin

(19°F) for the maximum fuel cladding temperature under storage conditions and has the second highest heat load for transportation conditions (26 kW) after the 69BTH DSC.

Consistent with the approach described in Section A.3.3.1.4, the DSC shell temperature profile for 24PTH DSC is retrieved from the cask model and applied as boundary conditions to the detailed model of the 24PTH-S DSC/basket. The DSC/basket model of 24PTH-S is identical to the model previously used for storage conditions in 10 CFR 72 UFSAR [1], Appendix P. A uniform heat load zone configuration with the maximum heat load of 26 kW is applied in the DSC model. The results of this case are compared with the results used in the SAR to demonstrate the conservative nature of the approach.

Comparisons of the maximum DSC component temperatures are listed in the following table.

Comparison of the Maximum Temperatures for 24PTH-S DSC

DSC Type	*24PTH-S (w/o Al inserts)*		
HLZC	*Uniform (1.3 kW/FA)*	*Uniform (1.08 kW/FA)*	
Operating Condition	*Normal Transfer 31.2 kW UFSAR [1], Tables P.4-10, -14 and -16*	*NCT 26 kW*	*Additional Thermal Margin*
	$T_{Transfer}$ (°F)	T_{NCT} (°F)	($T_{Transfer} - T_{NCT}$) (°F)
Fuel Cladding	733	664	+69
Fuel Compartment	682	616	+66
Al/Poison	681	615	+66
DSC Shell	475	463	+12

As seen in the above table, the maximum fuel cladding and basket component temperatures for the DSC type 24PTH-S (without Al inserts) under NCT are more than 60°F lower than the bounding values listed in the UFSAR [1]. This large difference demonstrates that the comparison of the DSC shell temperatures as discussed above is a conservative approach to bound the maximum fuel cladding and basket component temperatures for transport conditions.

A.3.6.4 Acceptance Criteria for Coating Damages for MP197HB TC

During handling and operation of MP197HB transport cask (TC), the painted surfaces of the shield shell for the un-finned cask or the anodized/painted surfaces of the finned aluminum shell for the finned cask can be scratched, peeled off, or physically damaged. The emissivity and solar absorptivity of the painted and anodized surfaces are considered as inputs for the thermal evaluation. Physical damages on the coating

As seen, the axial effective fuel conductivity from [3], Appendix T, Section T.4.8 remains the bounding value to be used in the thermal analysis.

Effective Density and Specific Heat

The effective density (ρ_{eff}) and specific heat ($c_{P,\ eff}$) calculated for fuel assemblies FANP 9x9, LaCrosse, and SVEA-92 are compared to the bounding values from [3], Appendix T, Section T.4.8 in the following table.

Fuel Assembly Effective Density and Specific Heat

	FANP 9x9 (9x9-81)	LaCrosse (10x10-100/0)	SVEA-92 (ABB-10-2)	Bounding Values (2)
No of fuel rods	96 (1)	81	96	
OD fuel rod (in)	0.395	0.424	0.378	
Clad thickness (in)	0.0210	0.03	0.0243	
No of water tubes	4 (1)	0	0.59	
Pellet OD (in)	0.3465	0.3565	0.3224	
Fuel length (in)	85	150	150.59	
Cladding area (in^2)	2.47	3.01	3.18	
UO_2 area (in^2)	9.05	8.09	7.84	
Compartment area (in^2)	36.0	36.0	36.0	
Cladding volume (in^3)	210	451	479	
UO_2 volume (in^3)	769	1213	1180	
Compartment volume (in^3)	3060	5400	5421	
$Density_{eff}$ (lbm/in^3)	0.119	0.109	0.107	0.103
$c_{P,\ eff}$ (Btu/lbm-°F)	0.0658	0.0578	0.0579	0.0575

Notes:

(1) Fuel assembly FANP 9x9 can optionally contain up to four water rods. To determine the lowest possible density and specific heat, four water rods are considered for fuel assembly FA FANP 9x9.

(2) Bounding values are from [3], Appendix T, Section T.4.8

As seen, the effective density and specific heat from [3], Appendix T, Section T.4.8 remain the bounding values to be used in the thermal analysis.

The effective conductivities along with specific heat and density used for BWR fuel assemblies are summarized in Section A.3.2.1 material # 2.

A.3.6.6 Thermal Analysis of 24PTHF and 61BTHF DSCs in the MP197HB TC

The 24PTHF and 61BTHF DSCs are proposed for transportation of damaged and failed fuel assemblies in the MP197HB. The failed fuel assemblies are to be encapsulated in individual failed fuel cans (FFCs) that are designed to fit into the 61BTHF and 24PTHF basket fuel compartments. The 24PTHF and 61BTHF DSCs have the same basket configurations as those for 24PTH and 61BTH DSCs except for additional FFCs to store failed fuel assemblies.

Damaged FAs are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. Damaged FA may be stored in the certain basket locations and does not require a separate FFC.

(5) Effect of aluminum inserts are omitted in calculation of this temperature for conservatism. The slight difference between this temperature and the one evaluated in Table A.3–10 has insignificant effect on thermal/structural performance.

The maximum temperatures for Failed Fuel Canisters

Transport Condition	Normal Condition of Transport (NCT)		Accident Condition of Transport (HAC)	
DSC Type	24PTHF	61BTHF	24PTHF	61BTHF
Heat Load	26 kW	24 kW	26 kW	24 kW
Component	T_{max} (°F)	T_{max} (°F)	T_{max} (°F)	T_{max} (°F)
Failed Fuel Compartment	601	606	645	645
Failed Fuel Canister	608	615	653	652

A.3.6.7 Justification of Hot Gaps

The following hot gaps assumed in the MP197HB TC, 69BTH DSC, and 37PTH DSC models are justified in this section.

a) The radial gap of 0.025" assumed between the gamma shield and the cask outer shell in MP197HB TC model.

b) The radial gap of 0.01" assumed between the finned aluminum shell and the cask shield shell in MP197HB TC model.

c) The diametrical hot gap of 0.30" between the basket outer surface and the DSC shell inner surface in 69BTH DSC model.

d) The 0.01" gaps considered on either side of the paired poison and aluminum plates in 69BTH DSC/basket model.

A.3.6.7.1 Gap between Gamma Shield and Cask Outer Shell

A radial air gap of 0.025" is assumed between the gamma shield (lead) and the TC outer shell within the finite element model of MP197HB described in Section A.3.3.1.1. This air gap is due to the differential thermal expansion of the cask body and the gamma shield.

The following assumptions are made for the verification of the gap:

- The cask body nominal dimensions *are taken* at 70°F.
- During the lead pour the cask body and lead *temperatures* are *held above the melting point of lead* at 620°F.
- *Because of the controlled cooling process used after lead pour is completed, the lead solidifies from the bottom upward and thus is always covered with molten lead. Any void volume or gap due to contraction of the solidifying lead is filled with molten lead which then solidifies.*
- The inner diameter of the gamma shell (lead) is equal to the outer diameter of the inner cask shell at thermal equilibrium.

The average coefficients of thermal expansion for SA-203, Gr. E and lead are listed in the following table.

Thermal Expansion Coefficients

Temperature (°F)	SA203, Gr. E α (in/in-°F) [10]
70	6.40E-06
200	6.70E-06
300	6.90E-06
400	7.10E-06
500	7.30E-06
600	7.40E-06
650	7.60E-06

Temperature (°F)	Lead α (in/in-°F) [*51*]
70	16.07 E-6
100	16.21 E-6
175	16.58 E-6
250	16.95 E-6
325	17.54 E-6
440	18.50 E-6
620	20.39 E-6

The density of lead as a function of temperature is listed below.

Density of Lead

Temperature (K)	Density [24] (kg/m3)	Temperature (°F)	Density (lbm/in^3)
50	11,570	-370	0.4180
100	11,520	-280	0.4162
150	11,470	-190	0.4144
200	11,430	-100	0.4129
250	11,380	-10	0.4111
300	11,330	80	0.4093
400	11,230	260	0.4057
500	11,130	440	0.4021
600	11,010	620	0.3978

The volume within the "lead cavity" is calculated by determining the cask body dimensions at 620°F. As no gaps will be present between the lead and the cask body, this volume is also equal to the volume of lead at 620°F. The mass of the lead *in the* lead cavity at 620°F is then determined.

The dimensions of the "lead cavity" for operating conditions are calculated based on cask body temperature at NCT. A temperature of 360°F is considered for the cask body. This temperature is lower than the maximum cask inner shell temperature shown in Table A.3–11 for 32 kW heat load. Since the gap size increases at lower temperatures, the above chosen value is conservative. From the mass of the lead and its density at 360°F, the lead volume at NCT is determined.

The length of the gamma shield at the cask body temperature is calculated based on thermal expansion coefficients listed in the above table. The lead volume is used to determine the maximum size of the air gap adjacent to the lead.

Determination of Lead Mass

α_{CS} = 7.44 x 10^{-6} in/in-°F @ 620°F (via linear interpolation from expansion coefficients table, above)

ρ_{lead} = 0.3978 lbm/in^3 @ 620°F (from lead density table, above)

A.3.6.9 Sensitivity Study for Effects of High Burnup Damaged Fuel Assemblies

The cladding of high burnup damaged fuel assemblies can experience further damages during NCT. To bound the effect of these damages, a sensitivity analysis is conducted considering the worst case condition, in which the high burnup damaged fuel assemblies become rubbles. Following the rationale in NUREG/CR-6835 [50], it is assumed that the fuel rods do not shatter into very small pieces and the fuel rubble is not in a tightly compacted mass. Instead, the fuel rubble is assumed to be 50% void by volume. Since the end drop is the most critical condition under NCT and the end caps and the fuel compartment walls constrain the damaged fuel assembly, the fuel rubble is assumed to be contained within the original active fuel volume, albeit in the lower portion of the original volume. Consistent with NUREG/CR-6835, the axial-burnup variation in the rubble is also assumed to be uniform.

The height of the fuel rubbles with the assumption of 50% void by volume is determined based on the volume of the fuel rods. The bounding fuel rubble height is 108" for the fuel assemblies.

The 69BTH DSC with the bounding heat load of 32 kW is considered for this sensitivity analysis. In the sensitivity run, the heat generation rate corresponding to the damaged fuel assemblies is applied uniformly over the fuel rubble height of 108" concentrated at the rear bottom of the 69BTH DSC with a peaking factor of one.

Conductivity of helium is considered for the fuel rubble for conservatism.

The DSC shell temperature retrieved from the cask model described in Section A.3.3.1.1 is applied as boundary conditions for the 69BTH DSC model, which is consistent with the approach described in Section A.3.3.1.4.

The maximum component temperatures resulting from the sensitivity analysis are compared to the corresponding values for 69BTH with intact fuel assemblies in the following table.

Comparison of the Maximum Fuel Temperatures for 69BTH DSC with Intact and Damaged Fuel Assemblies under NCT

DSC Type	$T_{max,\ Fuel}$* (°F)	$T_{max,\ Comp}$ (°F)	$T_{max,\ Al/Poison}$ (°F)	$T_{max,\ Rail}$ (°F)
69BTH, 32 kW w/ intact FAs (see Table A.3-10)	674.3	638.3	621.8	534.3
69BTH, 32 kW w/ intact and damaged FAs	679.6	644.3	628.0	537.1
Difference	+ 5.3	+ 6.0	+ 6.2	+ 2.8

** Fuel cladding temperature limit is 752° F*

As seen in the above table, the maximum fuel cladding temperature changes approximately by 5°F. Considering the large margin of 78°F for the fuel cladding temperature, this small change does not have any significant effect on the thermal performance of the cask and DSC.

A.3.6.10 Sensitivity Analysis for HAC using Coupled Model

The analysis for HAC described in Section A.3.4 is based on a combination of transient calculations of the MP197HB TC (including a homogenized basket) and steady state calculations of the DSC including basket components described in Section A.3.3.1 as separate thermal analyses. These models are used to calculate the component maximum temperatures (including cladding temperatures). To justify the approach considered in the HAC analysis, a coupled transient model is prepared to include the TC, DSC, and basket in one single model assigned here as coupled model.

The coupled model is created by introducing the elements and nodes from the 69BTH basket model described in Section A.3.3.1.4 into the TC model with crushed impact limiters described in Section A.3.4. The 69BTH basket and the MP197HB TC thermal models have dissimilar meshes since the mesh density of the 69BTH basket model is much finer than the mesh density of the TC model. The mesh density of the DSC shell is refined to provide adequate interfaces between the TC and the basket meshes. These two dissimilarly meshed models of the TC and 69BTH basket are tied together using the DSC shell nodes and constraint equations via the "CEINTF" command in ANSYS.

To ensure the correct application of the constraint equations, the same fine meshed DSC shell used in the coupled model described above was introduced into the coarser meshed model of TC described in Section A.3.3.1.1 and the constrained equations were applied at the intersection of the fine/coarse meshes. The results of this model were compared to the result of the TC model for NCT. The comparison showed that the maximum temperatures of the TC components remain virtually unchanged (the changes are within ±0.1°F) and the maximum DSC shell temperature changes by approximately by 1°F.

The coupled model of the TC, DSC, and basket includes the MP197HB TC and the bounding DSC (69BTH with 32 kW heat load) and considers the homogenized fuel assemblies within compartments. All the basket components (including back-filled gas and aluminum transition rails) are explicitly modeled in the coupled model considering the same assumption described in Section A.3.3.1.4 for 69BTH basket. The geometry of the coupled model is shown in Figure A.3-53.

Decay heat load is applied as heat generation boundary conditions over the elements representing homogenized fuel assemblies. The base heat generation rate is multiplied by peaking factors along the axial fuel length to represent the axial decay heat profile consistent with the approach described in Section A.3.3.1.4. The peaking factors remain identical to those shown in Table A.3-2.

The ambient boundary conditions for the coupled model are identical to those described in Section A.3.4 for the TC model under HAC.

The time temperature histories resulting from the coupled model are shown in Figure A.3-54 and Figure A.3-55. The maximum component temperatures from the coupled HAC thermal analysis are compared with the corresponding temperatures from the decoupled HAC analysis described in Sections A.3.4 in the following table.

Comparison of Maximum Temperatures of MP197HB TC for HAC with Coupled/Decoupled Models

DSC type	69BTH			
Heat Load	32 kW			
Inner sleeve	No			
External fins	Yes (Melted)			
Component	$T_{max, coupled}$ (°F)	$T_{max, decoupled}$ [(1)] (°F)	ΔT (°F)	Limit (°F)
Fuel Cladding	680	693	-13	1058 [7]
Fuel Compartment	650	658	-8	
Aluminum / Poison Plates	649	657	-8	
Basket Rails	548	557	-9	
DSC shell	521	537	-16	
Cask inner shell	499	497	2	---
Gamma shield	567	571	-4	621 [5]
Outer shell	716	720	-4	---
Shield shell	1440	1440	0	---
Cask lid	306	315	-9	---
Cask bottom plate	419	416	3	---
Cask lid seal	314	323	-9	400 [18], [19]
Vent & test seal @ top	304	313	-9	
Ram plate seal	387	380	7	
Test seal @ bottom	388	382	6	
Drain port seal @ bottom	392	388	4	
Helium in TC Cavity	387	389	-2	---

(1) For the maximum temperatures of the decoupled HAC analysis fire emissivity of 1.0, see Table A.3-17 and Table A.3-18 for 69BTH DSC (32 kW heat load) and Section A.3.6.8 for MP197HB TC.

As seen in the above table, all the maximum temperatures remain below the allowable limits in the coupled model.

The fuel cladding temperature resulting from the coupled model is 680°F and is lower by 13°F compared to the results from the decoupled models and is well below the accident temperature limit of 1058°F.

The maximum seal temperature for fluorocarbon seals is 392°F at drain port resulting from the coupled model. Although the maximum seal temperature increases by 4°F compared to the results from the decoupled models, the maximum seal temperature remains below the long-term limit of 400°F specified for continued seal function. Parker O-ring [18] gives a short term temperature limit of 482°F for fluorocarbon seals. The short term temperature limit was verified in [49] for this seal compound. The maximum seal temperatures resulting from the coupled or decoupled models remain well below the short term limit.

The maximum temperature of gamma shield (lead) is 567°F in the coupled model, which is 4°F lower than the corresponding value from the decoupled model and remains well below the lead melting point of 621°F.

Based on the above discussion, the differences between the maximum temperatures of critical components resulting from the coupled and decoupled models are limited to a *few degrees. This comparison shows that although the decoupled models do not include the axial profile of the decay heat load directly, the approach followed in the decoupled models captures the transient behavior of the TC and DSC during the fire and cool-down stages with adequate accuracy.*

Table A.3–1
DSC Shell Nominal Dimensions

Parameter (1)	69BTH	61BT / 61BTH Type 1	61BTH Type 2	37PTH(2)	32PTH / 32PTH1(3)	32PT(4)	24PTH(5)	24PT4
Outer Top Cover	2.00	1.25	1.50	2.00	2.00	1.50	1.50	1.25
Inner Top Cover	2.00	0.75	1.25	2.00	2.00	1.25	1.25	6.75 (6)
Top Shield Plug	5.75	7.00	6.25	5.75	8.00	7.50	6.25	
Total Top End	9.75	9.00	9.00	9.75	12.00	10.25	9.00	8.00
Inner Bottom Cover	2.25 (8)	0.75	1.75	2.25 (8)	2.25	1.75	1.75	2.00
Bottom Shield Plug	3.00 (8)	5.00	4.00	3.00 (8)	4.50	5.25	4.00	4.75 (6)
Outer Bottom Cover	2.00	1.75	1.75	2.00	2.00	1.75	1.75	
Total Bottom End	7.25	7.50	7.50	7.25	8.75	8.75	7.50	6.75
Cavity Length	178.41	179.50	179.50	164.38	164.38	167.10	169.60	180.20
DSC Length (w/o grapple) (7)	195.41	196.00	196.00	181.38	185.13	186.10	186.10	194.95
Basket height	164	164	164	162	162.00	166.10	168.60	179.13

Note:
(1) 61BTHF and 24PTHF *DSCs* have the same dimensions as DSC types 61BTH and 24PTH, respectively.
(2) The shortest cavity length for 37PTH baskets belongs to 37PTH-S.
(3) The shortest cavity length for 32PTH, 32PTH, type 1 and 32PTH1 Type 1 & 2 baskets belongs to 32PTH1-S.
(4) The shortest cavity length for 32PT baskets belongs to 32PT-S125.
(5) The shortest cavity length for 24PTH baskets belongs to 24PTH-S.
(6) Shield plugs of the 24PT4 DSC are lead encapsulated in plates of stainless steel 316, see [2].
(7) The canister length in the model is the sum of total top end, total bottom end, and cavity length.
(8) The thicknesses of inner bottom cover plate and bottom shield plug for 69BTH and 37PTH DSCs are designed as 1.75" and 3.5", respectively. Since the bottom shield plug (carbon steel) has a higher conductivity than the inner bottom cover plate (stainless steel), considering a smaller thickness of 3" for bottom shield plug and higher thickness of 2.25" for inner bottom cover plate is conservative which increases the thermal resistance across these two plates.

Table A.3–10
Maximum Fuel Cladding and Basket Component Temperatures for NCT

DSC Type	$T_{max, Fuel}$ (°F)	Reference for bounding fuel cladding temperature	$T_{max, Comp}$ (°F)	$T_{max, Al/Poison}$ (°F)	$T_{max, Rail}$ (°F)	Reference for bounding basket component temperatures
69BTH, 32 kW[3]	674	–	638	622	534	–
	650	–	612	612	507	–
69BTH, 29.2 kW	651	–	622	621	481	–
69BTH, 26 kW	658	–	643	643	475	–
61BTH Type 1	< 706	[3], Table T.4-12	< 683	< 682	< 565	[3], Table T.4-13
61BTH Type 2	< *721*	[3], Table T.4-12[4]	< *692*	< *692*	< 549	[3], Table T.4-14[4]
61BT	< 638	[1], Table K.4-2	< 615	< 615	< 493	[1], Table K.4-2
37PTH	660	–	649	648	443	–
32PTH, 32PTH Type 1	< 723	[4], Table 4-1	< 697	< 696	< 561	[4], Table 4-1
32PTH1 Type 1	< 713	[3], Table U.4-15	< 677	< 676	< 520	[3], Table U.4-16
32PTH1 Type 2	< 728	[3], Table U.4-15	< 648	< 648	< 529	[3], Table U.4-17
32PT	< 720	[1], Table M.4-2	< 705	< 705	< 471	[1], Table M.4-3
24PTH-S or –L w/ Al Inserts	< 733	[1], Table P.4-14	< 680	< 679	< 576 (1)	[1], Table P.4-16
24PTH-S or –L w/o Al Inserts	< 733	[1], Table P.4-14	< 682	< 681	< 576 (1)	[1], Table P.4-16
24PTH-S-LC	< 714	[1], Table P.4-14	< 674	< 673	< 500 (1)	[1], Table P.4-17
24PT4	< 707	[2], Table A4.4-7	< 670	< 670	(2)	[2], Table A4.4-6

Notes:

(1) This value is the maximum rail temperature for rail R90 taken from evaluations of 24PTH DSC under normal transfer conditions.

(2) Based on [2], Table A.4.4-6, the maximum spacer disc and support rod temperatures for 24PT4 DSC under normal transfer conditions are 663°F and 574°F. These temperatures are the bounding values for NCT.

(3) The maximum temperatures for TC without external fins are presented in the first row and the maximum temperatures for TC with external fins are presented in the second row.

(4) The maximum temperatures for 61BTH Type 2 DSC are increased from values in [3] due to allowance of six shims between the basket and the rails per note 5 of Drawing NUH61BTH-71-1102 in Appendix A.1.4.10.9.

Table A.3–11
Maximum/Minimum Component Temperatures for NCT

Conditions	**100°F with Insolation (1)**				**-40°F No Insolation (2)**	
Heat load	26 < Q ≤ 32 kW		Q ≤ 26 kW		0	
Inner sleeve	No	No	No	Yes	N/A	
External fins	No	Yes	No	No	N/A	
Component	T_{max} (°F)	T_{max} (°F)	T_{max} (°F)	T_{max} (°F)	T_{min} (°F)	Allowable Range (°F)
Fuel Cladding	674	651	< 728	< 733	-40	752 max. [7]
Fuel Compartment	638	622	< 697	< 692	-40	(4)
Al/Poison Plates	622	621	< 696	< 692	-40	(4)
Basket Rails	534	507	< 561	< 576	-40	(4)
DSC shell	510	484	451	464	-40	(4)
Inner sleeve	N/A	N/A	N/A	347	-40	(4)
Cask inner shell	398	367	351	344	-40	(4)
Gamma shield	397	366	349	343	-40	621 max. [5]
Outer shell	382	352	337	335	-40	(4)
Shield shell	335	305	299	295	-40	(4)
Finned Shell	N/A	229	N/A	N/A	-40	(4)
Cask lid (5)	295	267	319	336	-40	(4)
Cask bottom plate	383	353	338	322	-40	(4)
Neutron Shield Resin(3)	276	290	288	285	-40	320 max. [17]
Trunnion Plug Resin(3)	303	277	272	268	-40	445 max. [25]
Seals	382	352	337	336	-40	400 max. [18, 19]
Wood in Impact limiter	208	302	291	289	-40	320 max. [20]

Notes:
(1) These temperatures are the highest values taken from Table A.3–8, Table A.3–9, and Table A.3–10.
(2) These temperatures are based on assuming no credit for decay heat and a daily average ambient temperature of -40°F.
(3) The resin temperature is the volumetric, average temperature at the hottest cross section.
(4) The components perform their intended safety function within the operating range.
(5) The maximum cask lid temperatures for 26 kW < Q ≤ 32 kW and Q ≤ 26 kW belong to DSCs loaded with BWR and PWR fuel assemblies, respectively. Since a spacer is used for PWR DSCs, the heat load of the PWR fuel assemblies is closer to the cask lid. Due to this configuration, the maximum cask lid temperature for Q ≤ 26 kW is higher than for 26 kW < Q ≤ 32 kW. See Table A.3-8 and Table A.3-9 for details.

Table A.3–18
Maximum Fuel Cladding and Basket Component Temperatures for HAC

DSC Type	$T_{max, Fuel}$ (°F)	Reference for bounding fuel cladding temperature	$T_{max, Comp}$ (°F)	$T_{max, Al/Poison}$ (°F)	$T_{max, Rail}$ (°F)	Reference for bounding basket component temperatures
69BTH, 32 kW	693	–	658	657	557	–
69BTH, 29.2 kW	693	–	667	666	529	–
69BTH, 26 kW	668	–	653	653	487	–
61BTH Type 1	< 749	[3], Table T.4-21	< 727	< 727	< 609	[3], Table T.4-22
61BTH Type 2	*< 830*	[3], Table T.4-21*(6)*	*< 805*	*< 804*	*< 650*	[3], Table T.4-23*(6)*
61BT	< 809	[1], Section K.4.6.4	< 787	< 787	< 772	[1], Table K.4-1(3)
37PTH	671	–	661	660	459	–
32PTH, 32PTH Type 1	< 1036	[4], Table 4-5	< 1021	< 1021	< 878	[4], Table 4-5
32PTH1 Type 1(1)	< 796	[3], Table U.4-24	< 766	< 766	< 609	[3], Table U.4-25
32PTH1 Type 2(2)	< 858	[3], Table U.4-24	< 831	< 830	<689	[3], Table U.4-26
32PT	< 863	[1], Table M.4-13	< 852	< 852	< 631	[1], Table M.4-14
24PTH-S or –L w/ Al inserts	< 843	[1], Table P.4-25	< 802	< 801	< 716(4)	[1], Table P.4-27
24PTH-S or –L w/o Al inserts	< 843	[1], Table P.4-25	< 802	< 801	< 716(4)	[1], Table P.4-27
24PTH-S-LC	< 747	[1], Table P.4-25	< 709	< 708	< 716(4)	[1], Table P.4-28
24PT4	< 805	[2], Table A4.4-7	< 768	< 768	(5)	[2], Table A4.4-6

Notes:
(1) This value is the maximum fuel cladding temperature for 32PTH1 Type 1 DSC with 31.2 kW heat load under transfer accident conditions.
(2) This value is the maximum fuel cladding temperature for 32PTH1 Type 2 DSC with 31.2 kW heat load under transfer accident condition.
(3) Based on discussion in [1], Appendix K, Section K.4.6.5, the maximum temperatures for transfer accident conditions are bounded by the maximum temperatures for blocked vent accident conditions.
(4) This value is the maximum rail, R90, temperature for transfer accident conditions with 40.8 kW heat load.
(5) Based on [2], Table A.4.4-6, the maximum spacer disc and support rod temperatures for 24PT4 DSC under accident transfer conditions (loss of sunshade and neutron shield) are 761°F and 673°F. These temperatures are the bounding values for transport HAC.
(6) The maximum temperatures for 61BTH Type 2 DSC are increased from values in [3] due to allowance of six shims between the basket and the rails per note 5 of Drawing NUH61BTH-71-1102 in Appendix A.1.4.10.9.

Table A.3–22
Maximum Internal Pressures of 69BTH and 37PTH DSCs in MP197HB TC

Maximum Pressures for Normal Conditions of Transport											
DSC	HLZC #	DSC cavity volume	Helium fill amount	Plenum volume	Plenum helium amount	CC gas amount	Fission products amount	Total gas amount	$T_{av\ He}$	*Calculated* Pressure	*Design* Pressure
		in^3	g-moles	in^3	g-moles	g-moles	g-moles	g-moles	°F	psig	psig
69BTH	1	258,413	134.4	442.2	3.3	0.0	14.16	151.82	524	8.89	15
	2	228,101	118.6	390.9	2.9	0.0	12.52	134.03	517	8.74	
	3	228,101	118.6	390.9	2.9	0.0	12.52	134.03	527	8.96	
	4	199,970	104.0	333.2	2.5	0.0	10.67	117.13	538	9.15	
37PTH	37 FAs w/CC	257,646	138.1	306.1	17.0	2.5	0.0	157.6	500	9.28	15
Maximum Pressures for Hypothetical Accident Conditions											
DSC	HLZC #	DSC cavity volume	Helium fill amount	Plenum volume	Plenum helium amount	CC gas amount	Fission products amount	Total gas amount	$T_{av\ He}$	*Calculated* Pressure	*Design* Pressure
		in^3	g-moles	in^3	g-moles	g-moles	g-moles	g-moles	°F	psig	psig
69BTH	1	258,413	134.4	14,738.4	108.9	0.0	472.09	715.39	532	91.47	140
	2	228,101	118.6	13,029.6	96.3	0.0	417.36	632.26	525	90.88	
	3	228,101	118.6	13,029.6	96.3	0.0	417.36	632.26	568	95.54	
	4	199,970	104.0	11,107.2	82.1	0.0	355.78	541.85	581	94.52	
37PTH	37 FAs w/CC	257,646	138.1	10,204.9	568.1	82.9	0.0	789.2	514	102.64	140

Table A.3–23
Maximum Internal Pressure in DSCs for Transport in MP197HB TC

DSC	Operating conditions	*Design* Pressure psig	*Calculated* Pressure psig
69BTH	NCT	15	9.15
	HAC	140	95.54
61BTH/61BTHF Type 1	NCT	10	8.10
	HAC	65	56.10
61BTH Type 2	NCT	15	8.60
	HAC	120	68.70
61BT	NCT	10	9.56
	HAC	65	46.00
37PTH	NCT	15	9.28
	HAC	140	102.64
32PTH/32PTH Type 1	NCT	15	7.47
	HAC	120	91.00
32PTH1 Type 1/Type 2	NCT	15	10.74
	HAC	140	126.34
32PT	NCT	15	8.04
	HAC	105	101.68
24PTH/24PTHF (all types)	NCT	15	8.57
	HAC	120	97.20
24PT4	NCT	20	18.57
	HAC	100	80.70

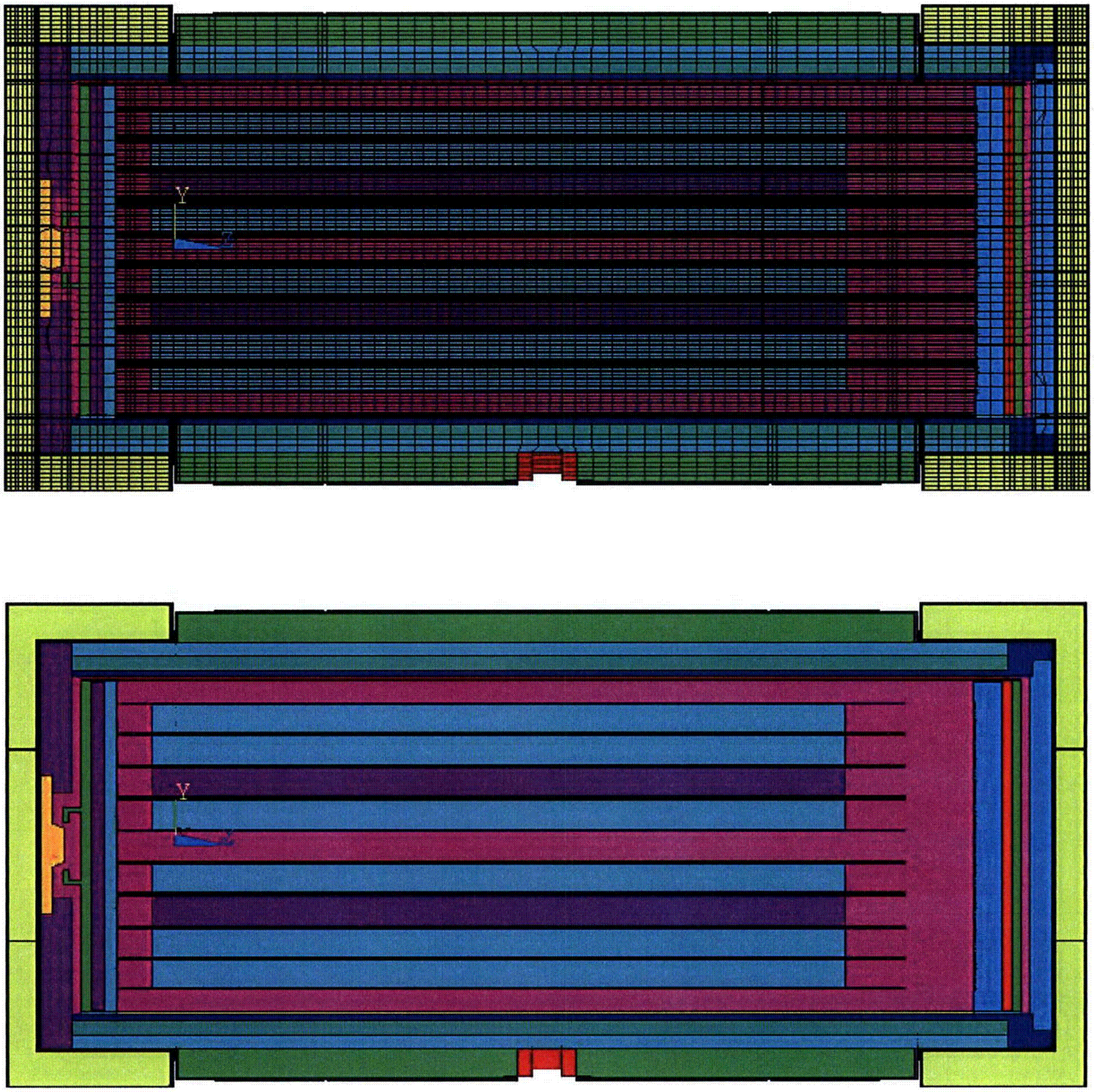


Figure A.3-53
Coupled Model of MP197HB TC and 69BTH DSC

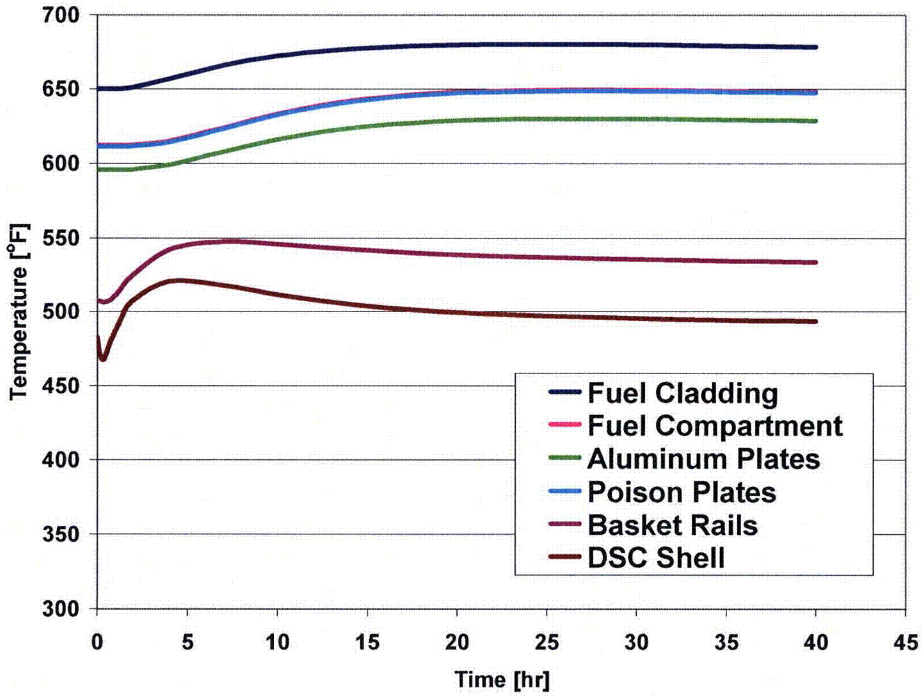


Figure A.3-54
Temperature Time Histories for Coupled Model, 69BTH DSC Components

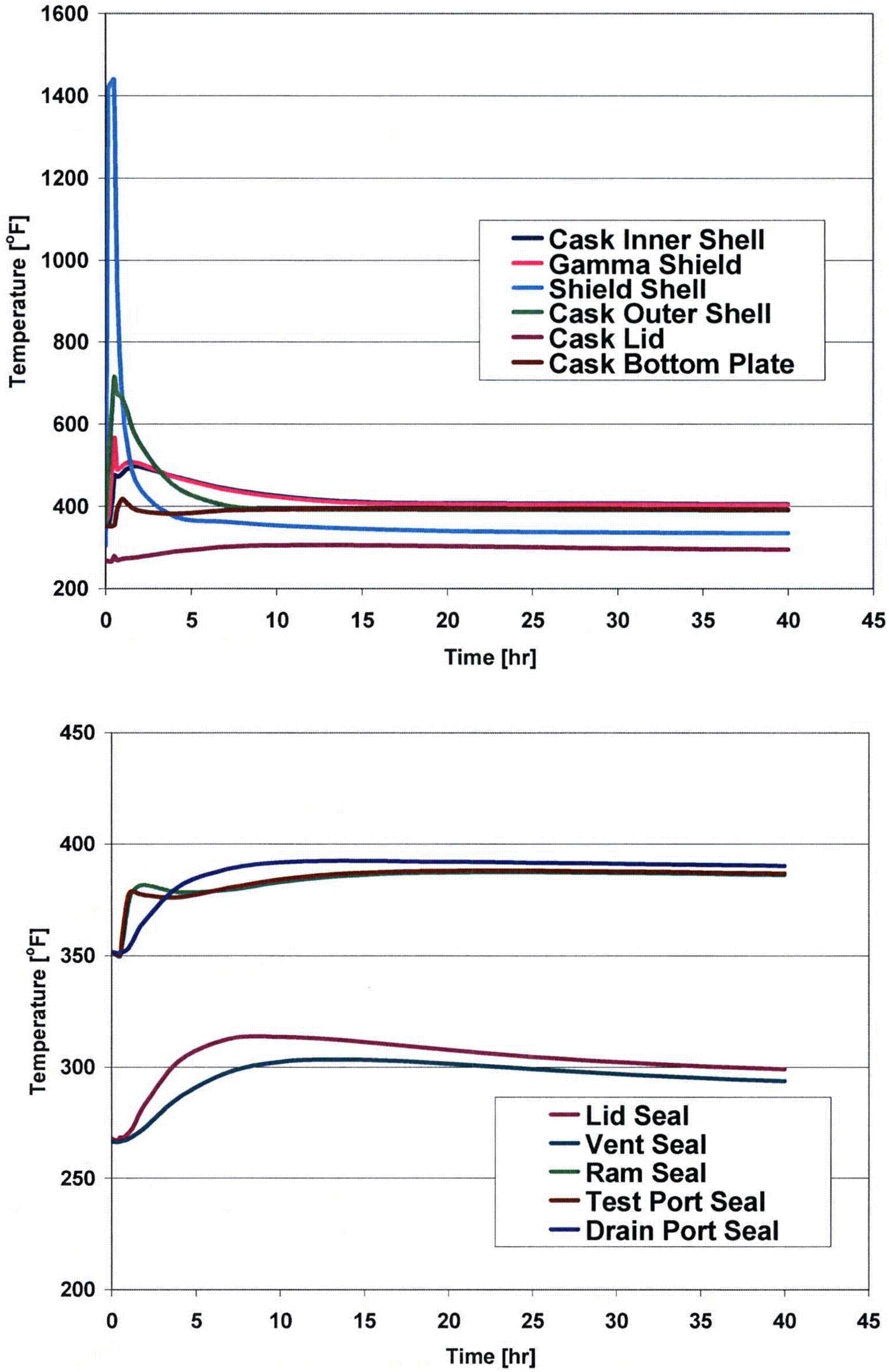


Figure A.3-55
Temperature Time Histories for Coupled Model, MP197HB TC Components

Chapter A.4
Containment

NOTE: References in this Chapter are shown as [1], [2], etc. and refer to the reference list in Section *A.4.5.*

A.4.1 Description of Containment System

A.4.1.1 Containment Boundary

The containment boundary for the NUHOMS®-MP197HB cask consists of a cylindrical inner shell, a bottom plate with a RAM access closure plate with seal, a cask body flange, a top lid with seal, and vent and drain port closure bolts and seals. The containment boundary is shown in Figure A.4-1. The construction of the containment boundary is shown on the drawings provided in Appendix A.1.4.10, Section A.1.4.10.1. The containment vessel prevents leakage of radioactive material from the cask cavity. It also maintains an inert atmosphere (helium) in the cask cavity.

Additionally, each of the welded canisters (DSCs) with used fuel as authorized contents contains helium. Thus, the welded canister also provides a containment function. Helium assists in heat removal and provides a non-reactive environment to protect fuel assemblies against fuel cladding degradation which might otherwise lead to gross rupture.

A.4.1.1.1 Containment Vessel

The NUHOMS®-MP197HB containment vessel consists of the inner shell, a 6.50 inch thick bottom plate with a 28.88 inch diameter, 2.50 inch thick RAM access closure plate, a cask body flange, a 4.50 inch thick lid with lid bolts, vent and drain port closures and bolts, and double O-ring seals for each of the penetrations. A 70.50 inch diameter by 199.25 inch long cavity is provided.

The inner containment shell is SA-203 Grade E nickel-alloy steel, and the bottom, and cask body flange materials are SA-350-LF3. The lid is constructed from SA-203 Grade E or SA-350-LF3. The NUHOMS®-MP197HB packaging containment vessel is designed, fabricated, examined and tested in accordance with the requirements of Subsection NB of the ASME Code [1] to the maximum practical extent. In addition, the design meets the requirements of Regulatory Guides 7.6 [2] and 7.8 [3]. Alternatives to the ASME Code are discussed in Section A.2.1.4. The design of the containment boundary is discussed in Chapter A.2.

The cask design, fabrication and testing are performed under Transnuclear's Quality Assurance Program which conforms to the criteria in Subpart H of 10CFR71.

The materials of construction meet the requirements of Section III, Subsection NB-2000 and Section II, material specifications or the corresponding ASTM Specifications. The containment vessel is designed to the ASME Code, Section III, Subsection NB, Article 3200.

All the seals used in the NUHOMS®-MP197HB cask containment boundary are static face seals. The seal areas are designed for no significant plastic deformation under normal and accident loads as shown in Chapter A.2. The bolts are torqued to maintain a seal load during all load conditions as shown in Appendix A.2.13.2. The seals used for all of the penetrations are fluorocarbon elastomer O-rings. All seal contact surfaces are stainless steel and are machined to a 32 RMS or finer surface finish. The dovetail grooves in the cask lid and the RAM closure plate are intended to retain the seals during installation. The volume of the grooves is controlled to allow the mating metal surfaces to contact under bolt loads, thereby providing uniform seal deformation in the final installation condition.

A fluorocarbon elastomeric seal was chosen for use on the MP197HB package because it has acceptable characteristics over a wide range of parameters. The fluorocarbon compound specified is VM835-75 or equivalent which meets the military rubber specification MIL-R-83485. (Note that this specification has been superceded by AMS-R-83485). Fluorocarbon O-rings are used in applications where temperatures are between -15°F and 400°F. The VM835-75 compound as listed on page 8-4 of the Parker O-ring Handbook [5] is specially formulated for use at temperatures as low as -40 °F while maintaining the upper temperature limit of 400°F.

A.4.1.1.4 Closure

The containment vessel contains an integrally-welded bottom closure and a bolted and flanged top closure plate (lid). The lid plate is attached to the cask body with forty eight (48), SA-540, Grade B23, Class 1, 1 ½ inch diameter bolts. Closure of the RAM closure plate is accomplished by twelve (12), SA-540, Grade B23, Class 1, 1 inch diameter cap screws. The bolt torque required for the lid and RAM closure plate are provided in Drawing MP197HB-71-1002 in Appendix A.1.4.10, Section A.1.4.10.1. The closure bolt analysis is presented in Appendix A.2.13.2.

Closure of each of the vent and drain ports is accomplished by a single 3/4 inch *brass or A193 B8* bolt with *a* seal under the head of the bolt.

A.4.2 Containment under Normal Conditions of Transport (Type B Packages)

A.4.2.1 Containment of Radioactive Material

As described earlier, the NUHOMS®-MP197HB is designed and tested for a leak rate of 1×10^{-7} ref cm^3/s, defined as "leak tight" per ANSI N14.5. Additionally, the structural and thermal analyses presented in Chapters A.2 and A.3, respectively, verify that there is no release of radioactive materials under any of the normal and accident conditions of transport.

A.4.2.2 Pressurization of Containment Vessel

The NUHOMS®-MP197HB contains either a canister (DSC) containing irradiated fuel or a secondary container containing dry irradiated and/or contaminated non-fuel bearing solid materials.

The DSCs are sealed (welded) canisters which have been tested to a "leak tight" criteria. Therefore, the pressure in the NUHOMS®-MP197HB when loaded with a DSC is from helium that has been backfilled into an evacuated cask cavity to a pressure of 3.5 psig at the end of

A.4.5 References

1. American Society of Mechanical Engineers, ASME Boiler And Pressure Vessel Code, Section III, Division 1 – Subsection NB, 2004 edition including 2006 Addenda.
2. USNRC Regulatory Guide 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessel," Rev. 1, March 1978.
3. USNRC Regulatory Guide 7.8, "Load Combinations for the Structural Analysis of Shipping Cask," Rev. 1, March 1989.
4. ANSI N14.5-1997, "American National Standard for Radioactive Material – Leakage Tests on Packages for Shipment," February 1998.
5. "Parker O-Ring Handbook," Publication No. ORD-5700, *2007 Edition,* Parker Seals *www.parkerorings.com.*

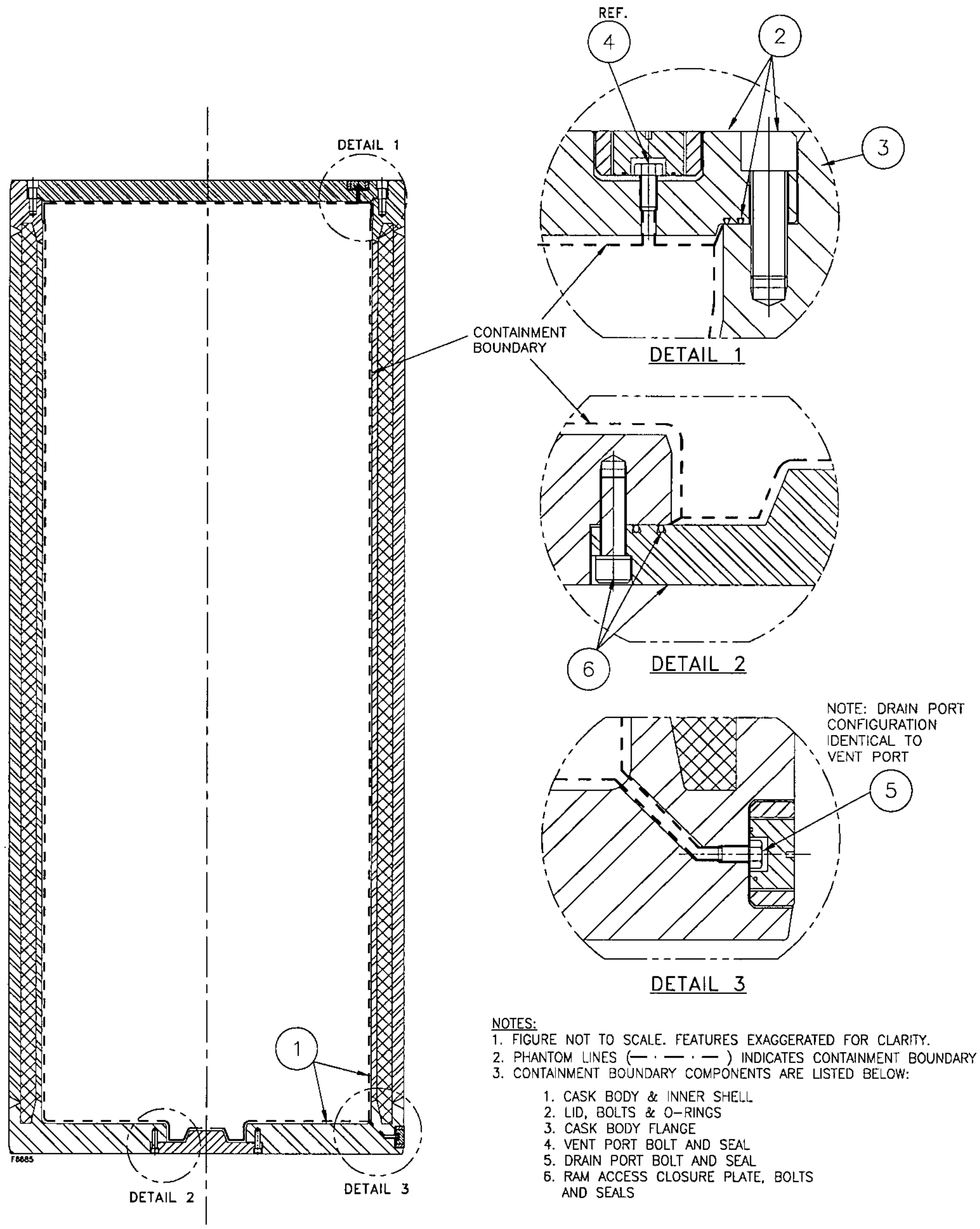


Figure A.4-1
NUHOMS®-MP197HB Containment Boundary Components

Chapter A.5
Shielding Evaluation

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

Chapter A.5
Shielding Evaluation

NOTE: References in this Chapter are shown as [1], [2], etc. and refer to the reference list in Section A.5.6.

A.5.1 Discussion and Results

Section A.5.2 describes the source specification and Section A.5.3 describes the shielding analysis model. Results of the bounding shielding analysis are described and summarized in Section A.5.4. Fuel qualification is described in Section A.5.5.

Shielding for the MP197HB transportation package is provided mainly by the cask body. Shielding against gamma radiation is provided by the lead and stainless steel shells that comprise the cask wall. For the neutron shielding, a borated VYAL-B resin compound surrounds the cask body radially. Gamma shielding in the cask ends is provided by the steel top and bottom assemblies of the transportation cask and axial ends of the DSCs. Additional shielding is provided by the steel outer shell surrounding the resin layer, the steel and aluminum structure of the fuel basket and optional heat dissipation fins surrounding the cask side between impact limiters.

For transport, wood filled impact limiters are installed on either end of the cask and provide additional shielding for the ends and some radial shielding for the areas at either end of the radial neutron shield. Sketches and screenshots from MCNP geometry plotter on Figure A.5-1 through Figure A.5-9 show modeled shielding configuration of the package. Table A.5-6 lists the thickness of major shielding components. Their elemental composition and density is in Table A.5-18.

The MP197HB cask is designed to transport one of several NUHOMS® DSCs loaded with used fuel assemblies or dry irradiated and/or contaminated non-fuel bearing solid materials in a *radioactive waste canister (RWC)* in accordance with the requirements of the 10 CFR 71. The authorized contents acceptable for transport are described in Chapter A.1, Section A.1.2.3, including appendices A.1.4.1 through A.1.4.9*A*. A complete list of the NUHOMS® DSCs authorized for transport is provided in Chapter A.1, Section A.1.2.3.1. Chapter A.1, Section A.1.2.3.2 *(also in Appendix A.1.4.9A)* provides a description of the irradiated and/or contaminated non-fuel bearing solid materials authorized for transport in *the RWC*. The MP197HB transportation cask loaded with any of these authorized contents is referred to as transportation package or simply as the cask in this Chapter.

A.5.1.1 NUHOMS® DSC Contents

For DSCs loaded with irradiated PWR and BWR fuel, the B&W 15x15 Mark B10 and the GE-2, 3 7x7 Type G2A fuel assembly contains the maximum heavy metal weight in their type, nearly 490 and 198 kgU , respectively. Because of this they result in bounding neutron and gamma source terms for PWR and BWR type of assemblies, respectively. Therefore B&W 15x15 Mark B-10 and the GE-2, 3 7x7 are identified as the Design Basis (DB) PWR and BWR fuel assembly (FA) in the shielding evaluation of MP197HB transportation package, respectively. Hardware source term per DSC is limited to that specified in the above appendices for the DSCs containing PWR fuel assemblies.

Therefore, a separate material composition and irradiation history is not necessary for characterizing all of these CCs. A description of the source term calculation for the CCs is provided in Section A.5.2.2.2. A discussion of the adjustment to the cooling times for qualification of fuel assemblies containing CCs is provided in Section A.5.5.2.

Reconstituted fuel assemblies are those where one or more fuel rods are replaced with "reconstituted" rods that displace the same amount of moderator in the active fuel region. Table A.5-9 of the SAR provides material details of a reconstituted fuel assembly where the rods are replaced with solid stainless steel after one cycle of irradiation. This assembly undergoes two additional cycles of irradiation where the source terms of the original and reconstituted fuel assemblies are compared. The summary of these evaluations is discussed in Section A.5.5.3.

Partial length shielding assemblies (PLSAs) are only authorized for the 24PTH DSC and are Westinghouse 15x15 design fuel assemblies that consist mostly of stainless steel. They are restricted to a maximum burnup of 40 GWD/MTU and a maximum MTU loading of 0.330 as shown on Chapter A.1, Appendix A.1.4.3. Therefore, they are bounded by the design basis B&W 15x15 fuel assemblies.

Fuel qualification tables are established for each DSC to assure compliance with maximum transportation dose rates criteria. Radiological source terms are calculated using SAS2H/ORIGEN-S modules of SCALE [1]. Two depletion models are used for radiological source calculation. They are based on the DB BWR and PWR FA parameters and characteristics. Radiological sources from these two depletion models are used for qualification of BWR and PWR assemblies for transportation, respectively. The model presented in Appendix P, Section P.5.5.1 of reference [18] is used for DB PWR FA. SAS2H\ORIGEN-S depletion model for DB BWR FA is developed using design characteristics and components presented in Table A.5-7 and Table A.5-8.

To perform fuel qualification for the transportation in the cask, response functions for all the transportation cask\DSC shielding configurations are established. The response functions were also used for determination of the bounding cask\DSC shielding configuration for hypothetical accident conditions (HAC). Details on the fuel qualification for transportation purpose and calculation of response functions are provided in Section A.5.5.1.

The initial uranium loading (MTU) of the fuel assemblies is only employed to determine the bounding fuel assembly design from a shielding standpoint. The design basis DSC from a shielding standpoint is dependent on the number of fuel assemblies, fuel assembly source terms, geometry and material layout of the basket. The response function and fuel qualification calculations are performed to ensure that all the DSCs are similar from a shielding standpoint.

The DSCs that house the PWR fuel assemblies are constrained by higher cooling times (15 to 30 years) associated with burnup credit. This will generally result in lower fuel assembly source terms. The 69BTH DSC contains the maximum number of fuel assemblies with the lowest allowable cooling time of 6 years. This ensures that the source terms for 69BTH DSC are bounding for shielding.

The cask shielding performance is based on a bounding shielding evaluation. The MP197HB cask loaded with 69BTH DSC containing DB BWR assemblies results in bounding dose rates. Because of that, DB BWR assembly is simply referred to as DB FA for the shielding evaluation in the text of this chapter. The DB FA for normal conditions of transport (NCT) dose rate analysis has an initial enrichment of 3.8 wt. % U-235 bundle-average burnup of 55,000 MWD/MTU with a 7 3/4 year decay time. *If between 5 and 24 damaged fuel assemblies are present, the damaged fuel assemblies shall have an additional 7¼ years of cooling time (15 years total cooling time). This additional cooling time also applies to damaged fuel loaded into the other DSCs authorized for transport in the MP197HB, as the 69BTH DSC contains the largest number of damaged fuel assemblies. The maximum number of damaged fuel assemblies and their position within the basket for the various DSCs are summarized in Table A.5-4.* The DB FA with an enrichment of 4.3 wt. % U-235 and a bundled-average burnup of 70,000 MWD/MTU and 21.0 year decay time generates radiological sources for the shielding performance evaluation of the cask at HAC.

NCT configurations are modeled with the neutron shield and impact limiters intact. *Three NCT configurations are analyzed: (1) no damaged fuel assemblies, (2) up to 4 damaged assemblies, (3) between 5 and 24 damaged assemblies. Damaged fuel assemblies are modeled with the active fuel length reduced to 75% of the nominal value. These shielding calculations are* performed using the Monte Carlo computer code MCNP [5]. Dose rates on the side, top and bottom of the MP197HB package are calculated for the various sources described in Section A.5.2 and summed to give a total gamma and neutron dose rate.

Three HAC configurations are analyzed: (1) no damaged fuel assemblies, (2) up to 24 damaged assemblies reduced to 50% of the nominal fuel assembly volume, and (3) up to 24 damaged assemblies reduced to 75% of the nominal fuel assembly volume. HAC shielding evaluation assumes that 75% of the neutron shield is lost. The impact limiters are assumed to be crushed 12" axially and the wood is removed. In addition, the top and bottom 0.375 inches of lead (axial direction) is removed to account for lead slump. Finally, the lead gamma shield radial thickness is reduced by 0.1". These assumptions result in a more severe degradation of the cask shielding properties than the accident conditions shown in Chapter A.2. Tests have shown that the neutron shielding material retains more than 60% of its principal contents (hydrogen, boron) following a design basis fire accident and a 25% credit employed in the shielding calculations is conservative. Shielding calculations for the HAC are also performed using the MCNP code.

Fuel qualification tables (FQTs) are established for each DSC type and authorized loading configuration. The applicable FQTs for each DSC are presented in Chapter A.1, Appendix A.1.4.1.1 through Appendix A.1.4.1.9. The individual FQTs for each DSC/loading configuration span the authorized burnup and initial enrichment combinations allowed for the fuel to be transported. The transportation FQTs are determined to provide minimum required cooling time to comply only with the dose rates limit criteria for transportation. It is important to remember however that the minimum cooling times provided in these FQTs may not be bounding for criteria other than shielding. These tables only provide the minimum required cooling time to meet the transportation dose rate limits. Other considerations are decay heat limits and burnup credit limits depending on the specific DSC and loading configuration.

To evaluate the decay heat for a given assembly, one must use the applicable decay heat equations (DHE) provided in Section A.5.5.4. Two sets of DHEs are established; one set for BWR (maximum of 0.198 MTU) fuel and one set for PWR (maximum of 0.490 MTU) fuel.

The expected dose rates (for NCT and HAC) from the MP197HB package with a DSC loaded with spent fuel are summarized in Tables A.5-1 through A.5-3. *Results are provided for cases with and without damaged fuel assemblies.* Maximum dose rates at various distances from the cask are provided in Tables A.5-21 through A.5-26 *only for configurations without damaged fuel assemblies*. The spatial distributions of the dose rates at various distances from the cask are shown on plots of Figure A.5-10 through Figure A.5-15 *for the NCT configuration without damaged assemblies*. The results show that the MP197HB transportation package complies with dose rates restrictions of 10 CFR 71 during NCT and HAC.

Two sets of package surface dose rates are shown in Table A.5-1. The first set under the column "Transport Package Surface" provides maximum dose rates at the external surface of the package. The surfaces where the maximum dose rates are calculated in the axial direction (top and bottom ends) are located at the outer surface of the impact limiters. The surfaces where the maximum dose rates are calculated in the radial direction (side) are located at the outer surface of the cask. Table A.5-23 does not include the surface dose rate since this is very conservative and is not an accessible surface during NCT. The second set under the column "Vehicle Edge" provides the maximum dose rates at the external surface of the package in the transport configuration. All the maximum dose rates in the radial and axial direction are calculated at the outer surface of the impact limiters. Note 1 of the Table A.5-1 provides the appropriate clarification.

The radial dose rates for these two sets of dose rates are different because they are calculated at different locations while the axial dose rates are identical.

Dose rates restriction criteria for the transportation are the following:

- External dose rate at any point on the outer accessible surface of the vehicle under normal conditions: 200 mrem/hr (max).
- External dose rate at any point 2 m from the outer lateral surfaces of the vehicle under normal conditions: 10 mrem/hr (max).
- External dose rate at any point 1 m from *the surface of the package* under hypothetical accident conditions: 1000 mrem/hr (max).
- The transportation package must not result in dose rates greater than 2.0 mrem/hr at occupied locations near the package during the transportation. Ends of the conveyance are considered the occupied locations.
- *External dose rate at any point on the outer accessible surface of the package under normal conditions: 1000 mrem/hr (max).*

A.5.1.2 Irradiated/Contaminated Waste

The NUHOMS®-MP197HB is designed for shipment of various types of irradiated and contaminated reactor hardware. The payload will vary from shipment to shipment and will consist predominately of components specified in a numbered list of Section A.5.2. The NUHOMS®-MP197HB is designed to transport a payload of 55.0 tons of dry irradiated and/or contaminated non-fuel bearing solid materials in *the RWC*. The safety analysis of the cask takes no credit for the containment provided by *the RWC*.

The quantity of radioactive material is limited to a maximum of 8,182 A_2 (90,000 Ci of Co60). The radioactive material is primarily in the form of neutron activated metals, or metal oxides in solid form. Surface contamination may also be present on the irradiated components. When a wet load procedure (i.e., in-pool) is followed for cask loading, the cask cavity and *RWC* are drained and dried to ensure that free liquids do not remain in the package during transport.

All these materials are basically neutron activated hardware that are mostly composed of steel and are defined by a Co-60 source spectrum. Therefore, detailed material composition and irradiation history is not necessary for the characterization of this waste. As discussed above, since the maximum quantity of the radioactive material is restricted to 90,000 Ci of Co-60 (equivalent) and the total payload mass is restricted to 55 tons, the shielding calculations are performed using this maximum allowable source terms. A list of the various types of irradiated hardware materials (not necessarily an exhaustive list) is provided in Section A.5.2. A discussion of the source specification based on the maximum allowable limits is provided in Section A.5.2.2.3.

The decay heat load of the radioactive material is expected to be less than 5 kW, which is well below the 26 kW limit for the cask.

NCT and HAC dose rates from the transportation cask containing the irradiated waste canister are summarized in Table A.5-1, Table A.5-2, Table A.5-5 and also Table A.5-27, Table A.5-28.

Dose rate distributions at various distances from the cask are plotted on Figure 5-14 and Figure 5-15.

A.5.2 Source Specification

There are five principal sources of radiation associated with transport of spent nuclear fuel that are of concern for radiation protection:

- Primary gamma radiation from spent fuel,
- Primary neutron radiation from spent fuel (both alpha-n reactions and spontaneous fission),
- Gamma radiation from activated fuel structural materials and fuel inserts,
- Capture gamma radiation produced by attenuation of neutrons by shielding material of the cask, and
- Neutrons produced by sub-critical multiplication in the fuel.

The MP197HB package is designed to transport LWR class of fuel assemblies in various DSC designs or dry irradiated and/or contaminated non-fuel bearing solid material contained in *the RWC*. The fuel assemblies as a function of DSC type are given in Chapter A.1, Appendix A.1.4.1.1 through Appendix A.1.4.1.9. The various fuel assembly designs were separated according to fuel assembly array, the maximum metric tons of uranium, and the number of guide /instrument tubes. These parameters are the significant contributors to the SAS2H/ORIGEN-S model. The largest uranium loading results in the largest source term at any chosen enrichment and burnup, thus the B&W 15x15 Mark B10 and GE-2, 3 7x7 Type G2A are bounding assembly types for PWR and BWR spent fuel assemblies, respectively. It was established that the cask loaded with 69BTH DSC containing bounding BWR assemblies results in the bounding dose rates.

Fuel Qualification Tables (FQTs) for shielding are established to assure that the applicable dose rate restrictions are satisfied. To establish these FQTs, the minimum cooling times were determined for each burn-up and enrichment combination for all DSCs such that the resulting dose rates are below the maximum NCT dose rate limit at two meters from surface of the cask. Cooling times for all of the low burnup FQT entries are established such that the resulting dose rates are well below the dose rate limits. For those BECT combinations where the calculated cooling times result in dose rates that are at or slightly below the limit, the source terms at NCT are considered equivalent. In addition, the contribution of the neutron and gamma components to the total dose rate from these BECT combinations is also calculated as part of this evaluation. Therefore, any one BECT combination from these "equivalent" combinations can be employed in the NCT evaluation. The BECT combination resulting in the highest neutron radiation component of the total dose rate at NCT can be employed to determine the design basis source terms for HAC. This combination is designated for HAC source terms since the HAC shielding configuration is based on substantial loss of neutron shielding. This is also discussed in Section A.5.3.3.1 and Section A.5.5. of the SAR. The FQTs shown for the various DSCs in Appendix 1.4 also include the additional cooling time requirements from the criticality analyses.

Table A.5-7 and Table A.5-8 provide characteristics and components of the design basis BWR fuel assembly. The SAS2H/ORIGEN-S modules of the SCALE code are used to generate gamma and neutron source terms for the design basis fuel assembly. The design basis radiological sources for NCT and HAC are due to DB FA irradiated at a constant specific power of 12.4 and 15.8 MW/assembly to a total bundle average burnup of 55,000 and 70,000 MWD/MTU, respectively. A three-cycle operating history is utilized. The assembly is burned for 292 effective full power days per cycle, for the duration of 3 cycles with 72 day down time after each cycle except for no down time in the last cycle.

The source terms used in the bounding shielding evaluation are generated for the fuel assembly active fuel region, the plenum region, and the top and bottom end fitting regions. The fuel assembly hardware materials and masses on a per assembly basis are listed in Table A.5-8. Table A.5-10 provides the material composition of fuel assembly hardware materials. Cobalt impurities are included in the SAS2H model.

The masses for the materials in the top end fitting, the plenum, and the bottom fitting regions are multiplied by 0.1, 0.2 and 0.15, respectively [4] in the BWR FA model. These factors are used to correct for the spatial and spectral changes of the neutron flux outside of the active fuel zone. The material compositions of the fuel assembly hardware are included in the SAS2H/ORIGEN-S model on a per assembly basis.

burnup is more and more dominated by spontaneous fission neutrons. Reviewing the output from the SAS2H runs, the neutron source term is due almost entirely to the spontaneous fission of ^{244}Cm (more than 90% of all neutrons both spontaneous fission and (α, n)) with (α, O–18) of lesser importance. Reference [13] documents that SAS2H tends slightly to over predict the concentration of ^{244}Cm when burnup is varied during the sensitivity study. Therefore, as the ^{244}Cm isotope accounts for more than 90% of the total neutron source term, the uncertainty in the neutron source and associated neutron dose rates is expected to be less than ±11%.

As documented in Reference [15] and as observed in preparing the fuel qualification tables, the gamma dose rate increases nearly linearly with burnup relative to the direct gamma component and the neutron dose rate increases with burnup to the fourth power. Therefore, as burnups go beyond 45 GWd/MTU, the contribution from neutron (and associated n, γ) components to the total dose rates measured on the surfaces of the cask increases in relative importance to that of the gamma component.

As discussed above, any impact of uncertainties in source terms is not expected to be significantly greater than 10 %, for the transportation system. Therefore, depletion calculations with SAS2H for calculation of some terms are appropriate for fuel burned above 45 GWd/MTU.

Uncertainties in radiological source terms are defined by the ability of the SAS2H code to accurately predict the isotopic concentration of nuclides in the fuel. The results of the isotopic assay evaluations and their comparison with measurements are summarized in various NUREG /ORNL documents. These evaluations suggest that in some cases, the concentrations of some of major gamma emitters may be under-predicted while concentrations for others major gamma emitters are over-predicted. The primary concern from a shielding standpoint is the net effect for the prediction of isotopic concentrations that contribute most to the dose rates. Section 3.3.3.4 of reference [2] indicates that the use of the SAS2H/ORIGEN-S methodology with the ENDF/B-V cross section library typically result in predictions that are in agreement with available measured data to within 10%. Further, sensitivity studies from reference [12] have indicated that that no significant change in the nature of the comparison is expected with burnups up to 75 GWd/MTU. A study of the various NUREG/ORNL documents listed in Section A.5.6 suggests that SAS2H prediction of the major gamma emitters is such that the resulting total gamma source term (and the effect on gamma dose rates) is conservative. For example, an estimate using data on major gamma radiation emitters from ORNL/TM-13315 [7] suggests that a component of the design basis dose rate due to primary gamma radiation source is overestimated up to 20%. That justifies not applying any uncertainties to the design basis primary gamma radiation source terms or adjustments to the primary gamma radiation dose rates due to such uncertainties. Finally, since only the net effect for the prediction of isotopes that contributes the most to radiological sources is important, another method of validation of source term calculations is to measure the dose rate from those sources. Numerous measurements from various Transnuclear's spent fuel storage and transportation systems have indicated that the measured dose rates are typically lower than calculated dose rates.

In summary, the source terms calculated by the SAS2H/ORIGEN-S modules are conservative because of the use of bounding material specification parameters employed in the models (maximize Co-60, maximize initial Uranium content). The comparison to isotopic assay data indicates that the SAS2H methodology is capable of accurately predicting the isotopic composition of nuclides important from a source term standpoint and do not show any *significant bias with burnup [8]. Therefore, no other adjustments are made to the source terms calculated from SAS2H or dose rates due to such sources.*

The NUHOMS®-MP197HB is also designed for shipment of various types of irradiated and contaminated reactor hardware. The payload will vary from shipment to shipment and will consist predominately of the following components either individually or in combinations:

1. BWR Control Rod Blades
2. BWR Local Power Range Monitors (LPRMs)
3. BWR Fuel Channels
4. BWR Poison Curtains
5. PWR Burnable Poison Rod Assemblies
6. Reactor Vessel and Internals (PWR and BWR)

The typical Cobalt 60 specific activity ranges for these items are as follows:

1.	Control Rod Blades	$1.3\times10^{-4} - 1.1\times10^{-2}$ Ci/g
2.	LPRMs	$1.0\times10^{-2} - 4.8\times10^{-2}$ Ci/g
3.	Fuel Channels	$7.8\times10^{-5} - 2.0\times10^{-4}$ Ci/g
4.	Poison Curtains	$6.2\times10^{-4} - 4.0\times10^{-2}$ Ci/g
5.	BPRAs	$3.8\times10^{-4} - 1.3\times10^{-3}$ Ci/g
6.	Reactor Vessel and Internals	$2.0\times10^{-5} - 1.3\times10^{-2}$ Ci/g

Components with high specific activity are generally placed near the center of the cask. For each shipment, the cask is normally filled to capacity, which prevents shifting of the contents during transport. If the container is not full, appropriate component spacers or shoring is used to prevent significant movement of the contents.

A.5.2.1 Axial Source Distribution

Axial peaking factors used for the neutron and gamma sources in BWR fuel are provided in Table A.5-15. These peaking factors are directly obtained from those employed for BWR fuel assemblies for the MP197 cask shown in Table 5.2-7. Table A.5-16 provides the PWR [3] fuel peaking factors used to generate the PWR FQTs. The peaking factors for both neutron and gamma sources are given as a function of active fuel height. These factors are used to describe radiological source terms strength distribution along axis of fuel region in MCNP models for bounding shielding evaluation and calculation of response functions employed during the qualification of assemblies for transportation.

The factors in Table A.5-15 are based on typical axial burnup distributions for BWR assemblies and typical axial water density distribution that occurs during core operation. Using the base SAS2H/ORIGEN-S input for the 7x7 BWR, selected as the DBFA for the bounding MP197HB cask shielding performance evaluation, neutron and gamma source terms are generated for *each* axial zone as a function of burnup and moderator density. *The gamma and neutron peaking factors are generated from these source terms.* This estimates both the non-linear behavior of the neutron source with burnup and the core operating moderator density effects on the actinide isotopics (neutron source). This axial distribution is conservative at high burnup because the burnup distribution will flatten out with increased burnup resulting in a reduction in the overall peaking factor.

The PWR gamma and neutron peaking factors in Table A.5-16 are developed using a slightly different method. The PWR gamma peaking factors are assumed to be the same as the burnup profile (note that this assumption is verified in Table A.5-15, in which the gamma peaking factors are computed explicitly.) The burnup profile is obtained from Table 2 of NUREG/CR-6801 [3] for burnups >46 GWD/MTU. The non-normalized PWR neutron peaking factor is assumed to be the fourth power of the axial burnup profile, which is a reasonable approximation. For comparison with Table A.5-15, the PWR neutron peaking factors are also presented as a normalized distribution in Table A.5-16.

The PWR burnup profile was obtained from NUREG/CR-6801, which was generated for burnup credit criticality analysis. In burnup credit analyses, axial burnup profiles are highly peaked to minimize burnup at the ends, which is the most reactive condition. These highly peaked profiles are conservative for shielding applications since they maximize the dose rates in the mid-plane of the active fuel. Therefore the PWR axial burnup profile employed is acceptable.

The BWR axial burnup profile is identical to the one employed in the 61BTH DSC for storage from Chapter T.5 of the UFSAR [18]. It is based on a low burnup BWR fuel assembly with a significantly peaked axial burnup profile which maximizes the dose rates in the mid-plane of the active fuel. Therefore the BWR axial burnup profile employed is acceptable.

The ratio of the true total neutron source in an assembly to the neutron source calculated by SAS2H/ORIGEN-S for an average assembly burnup is 1.326 and 1.152 for BWR and PWR assemblies, respectively. Therefore, the neutron source per fuel assembly as reported in Table A.5-11, *Table A.5-11a,* and Table A.5-12 *are* multiplied by 1.326 when used in MCNP models for bounding shielding evaluation. The response function entries corresponding to neutron radiation source are multiplied by 1.326 and 1.152 for the cask containing DSCs with BWR and PWR assemblies, respectively.

The gamma and neutron peaking factors may be used to compute the number of particles in each axial zone. The number of particles in each axial zone is the total source strength × fractional zone width × normalized peaking factor. The fractional zone widths are given in Table A.5-15 and Table A.5-16. The number of particles in each zone is input to MCNP in the shielding models. It is not necessary to input the actual number of particles in each zone on the MCNP input card because MCNP will renormalize the distribution, although the relative number of particles between each zone must match the true particle distribution.

A.5.2.2 Gamma Source

A.5.2.2.1 Used Fuel in DSCs

The primary gamma radiation source terms for the design basis spent fuel assembly for NCT *(up to 4 damaged assemblies), NCT damaged fuel (between 5 and 24 damaged fuel assembles),* and HAC shielding evaluation are provided in Table A.5-11, *Table A.5-11a,* and Table A.5-12 *respectively. The source terms in Table A.5-11 apply to both undamaged and damaged NCT assemblies when limited to a maximum of 4 damaged fuel assemblies. When modeling 24 damaged fuel assemblies, the source terms in Table A.5-11 apply to intact fuel only, while the source terms in Table A.5-11a apply to damaged fuel.* The spectral

Total Primary Gamma Radiation Source Term Strength for Bounding Shielding Evaluation

Normal conditions of transport total primary gamma radiation source terms strength, *(up to 4 damaged assemblies),* S_G^{NCT}, is:

S_G^{NCT} = sum[4.*6037E*+12, 2.*5916E*+15, 1.*5932E*+12, 3.*7889E*+12] gammas/sec/assy x 69 assy) = 1.*7951E*+17 gammas per second per cask.

Normal conditions of transport total primary gamma radiation source terms strength for 45 intact assemblies and 24 damaged assemblies, $S_G^{NCT\_DAM}$*, is:*

$S_G^{NCT\_DAM}$ *= 45 assy*sum[4.6037E+12, 2.5916E+15, 1.5932E+12, 3.7889E+12] + 24 assy*sum[1.7676E+12, 1.6720E+15, 5.9492E+11, 1.4569E+12] = 1.5729E+17 gammas per second per cask.*

Hypothetical accident conditions total primary gamma radiation source terms strength, S_G^{HAC}, is:

S_G^{HAC} = sum[1.0069E+12, 1.7524E+15, 3.3439e+11, 8.2723e+11] gammas/sec/assy x 69 assy) = 1.2107e+17 gammas per second per cask

A.5.2.2.2 Control Components (CCs)

Radiological source in Table A.5-13 represents any CC provided that the source term for the CC in question is bounded by the source term provided in this table. The source in this table is referred to as design basis CC source.

DB PWR FA burnable absorber assemblies with burnup between 36,000 MWd/MTU and 45,000 MWd/MTU are bounded by the design basis CC source after 8 years decay. All other BPRAs irradiated between 36,000 MWd/MTU and 45,000 MWd/MTU would require 13 years of decay to be bounded by the design basis CC source. All other CCs would need to be examined on a case by case basis.

Combinations of radiological sources due DB PWR assembly and the DB CC source result in bounding dose rates when evaluating shielding performance of MP197HB cask loaded with DSCs containing PWR FAs with DB CC sources.

Guidelines for adjustment of FQT cooling times due to presence of DB CC sources are provided in Section A.5.5.2.

A.5.2.2.3 Irradiated/Contaminated Waste

Maximum of total activity associated with irradiated\contaminated hardware specified in the numbered list on Section A.5.2 is 8182 A_2*11 (A_2 value of Co^{60}) =90,000 Ci. It is assumed in MCNP models that radiological source associated with such an activity is uniformly smeared throughout a cylindrical volume specified in item 3 of a numbered list in Section A.5.3.1. The source is assumed to be isotropic and due to Co-60 radiation.

Co-60 emits two photons per disintegration, one at 1.17 and one at 1.33 MeV. Therefore radiological source terms net strength is calculated as:

S_G^{GTCC} = (90,000 Ci) x (2 photons/disintegration) x (3.7e+10 disintegrations per second per Curie) = 6.66e+15 photons *per second per cask.*

A.5.2.3 Neutron Source

Table A.5-11, *Table A.5-11a,* and Table A.5-12 provide the total neutron source for the design basis fuel assembly under the irradiation/decay history described above in Section A.5.2. The sources are for the bounding shielding performance evaluation under NCT *(with up to 4 damaged assemblies), NCT damaged assemblies (when 24 damaged assemblies are modeled),* and HAC, respectively. The magnitude of the neutron source is provided as the final row in the gamma source term tables.

One SAS2H/ORIGEN-S run is required for each burnup/initial enrichment/cooling time combination to determine the total neutron source term for the in-core regions. At discharge the neutron source is almost equally produced from ^{242}Cm and ^{244}Cm. The other strong contributor is ^{252}Cf, which is approximately 1/10 of the Cm intensity, but its share vanishes after 6 years of cooling time because the half-life of ^{252}Cf is 2.65 years. The half-lives of ^{242}Cm and ^{244}Cm are 163 days and 18 years, respectively. Contributions from the next strongest emitters, ^{238}Pu and ^{240}Pu, are lower by a factor of 1000 and 100, respectively, relative to ^{244}Cm. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables, ^{244}Cm represents more than 90% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein.

Effect of subcritical neutron multiplication and source terms strength variation along FA axis due to a variation of an axial burn-up profile in assembly's active region are not accounted for when using SAS2H\ORIGEN-S [1] depletion model. However, these effects are accounted for in the shielding analysis and calculation by applying correction factors when describing the source in MCNP input decks. Neutron source terms for use in the MCNP shielding models are calculated by multiplying the fuel assembly source by the number of assemblies in the DSC fuel compartments. The magnitude of the neutron source is also increased to account for the axial distribution in the fuel, as explained in Section A.5.2.1. The effect of the neutron subcritical multiplication and correction to the neutron source strength due to axial burn-up profile variation is also accounted in calculation of the response functions used for fuel qualification for the transportation purpose.

To conservatively account for subcritical multiplication inside the DSC, the neutron source terms and the (n,γ) sources are multiplied by $1/(1-k_{eff})$. k_{eff} is the effective neutron multiplication factor determined using criticality codes that involved more detailed analysis and treatment of fuel region during criticality calculations. A k_{eff}= 0.40 is used for the bounding shielding evaluation and calculating response function entries relevant to MP197HB cask containing 69BTH DSC. A k_{eff}= 0.42 is conservatively used when calculating response function entries for the cask containing other DSCs.

The fixed source spectrum in MCNP is assumed to follow a ^{244}Cm spontaneous fission spectrum for all of the shielding calculations. It is based on the following relationship:

$$p(E) \sim exp(-E/a)sinh(bE)^{1/2}$$

with input parameters $a=0.906$ MeV and $b=3.848$ (MeV)$^{-1}$, as given in the MCNP manual [5].

Total Neutron Source Term

Normal conditions of transport total neutron radiation source term strength, *(up to 4 damaged assemblies),* S_N^{NCT}, is

S_N^{NCT} = (3.21E+08 neutron/sec/assy x 69 assy) x 1.326 /(1-0.40)

= 4.8949e+10 neutrons per second per cask

Normal conditions of transport total neutron radiation source term strength for 45 intact assemblies and 24 damaged assemblies, $S_N^{NCT\_DAM}$*, is*

$S_N^{NCT\_DAM}$ *= [(3.21E+08 neutron/sec/assy x 45 assy) + (2.438E+08 neutrons/sec/assy x 24 assy)] x 1.326 /(1-0.40)*

= 4.485e+10 neutrons per second per cask

Hypothetical accident conditions total neutron radiation source term strength, S_N^{HAC}, is

S_N^{HAC} = (4.00E+08 neutron/sec/assy x 69 assy) x 1.326 /(1-0.40)

= 6.0996e+10 neutrons per second per cask

A.5.3 Model Specification

The 3-D Monte Carlo computer code MCNP [5] is used for calculating response functions, the gamma and neutron radiation dose *rates* for the bounding shielding analysis of the cask. This section provides details of the geometry, material, source term configurations, physics and tallies description employed in the shielding models to determine the dose rates and response functions used for qualification of fuel assemblies for transportation based solely on the dose rate limits.

Three NCT configurations are analyzed: (1) no damaged fuel assemblies, (2) up to 4 damaged assemblies, and (3) between 5 and 24 damaged assemblies. Damaged fuel assemblies are modeled with the active fuel length reduced to 75% of the nominal value. This value was selected to bound any reasonable fuel damage during transport. The position of the top and bottom nozzles and plenum regions are assumed to remain in place, see Figure A.5-6a and Figure A.5-6b for 24-damaged and 4-damaged assembly models, respectively. Because the volume of the active fuel has been reduced, the density increases accordingly.

Fuel damage, if any, will occur only for fuel assemblies classified as damaged prior to loading. Vibration loads or off-normal loading will not cause intact fuel to become damaged. Therefore, it is assumed that only fuel classified as damaged could rubblize during NCT. Because rubblizing the fuel increases the side dose rate in the limiting location, models are run with both 4 and 24 damaged fuel assemblies. With 4 damaged assemblies, the design basis source is modeled in all 69 fuel locations. The axial source distributions for the intact fuel assemblies are obtained from Table A.5-15. For the damaged assemblies, this distribution is modified to employ a three-zone profile. The three-zone axial profile is applied to the rubblized fuel, combining Zones 1 and 2 as the bottom zone, Zones 3 though 10 as the middle zone, and Zones 11 and 12 as the top zone. This profile reflects that under NCT, large scale relocation of fuel is not anticipated.

When 24 fuel assemblies are rubblized to reduce the radial dose rate, additional cooling time is applied to the damaged fuel assemblies. Because the dose rates are driven by the outer damaged assemblies, the additional cooling time lowers the radial dose rates considerably.

A.5.3.1 Description of Radial and Axial Shielding Configuration

The model geometry of the shielding configuration can be viewed on Figure A.5-1 through Figure A.5-9. Thicknesses of the major shielding components of the cask and 69BTH DSC are summarized in Table A.5-6.

MCNP models were constructed for each DSC payload within the MP197HB cask when calculating entries of response functions used for fuel qualification for transportation. The models are created for the same cask geometry and material specification as in Figure A.5-1 through Figure A.5-9 and Table A.5-6.

Two types of base models were constructed for both NCT and HAC bounding shielding evaluations and calculation of response functions. The first one corresponds to the neutron transport problem and the second is the gamma.

Variance reduction was accomplished by means of importance zoning in all MCNP models. The importance function was created to keep balance of the particles (per volume) throughout the problem geometry. The process used to do this was an iterative approach starting with basic attenuation factors for the shielding materials. The neutron importance function developed was also applied to the secondary gammas.

Sections 5.3.1.1 and 5.3.1.2 describe the shielding model developed for the MP197HB under NCT and HAC, respectively. Described models were used to calculate the axial and radial dose rates in the bounding shielding evaluation. Similar models are used for calculating response functions, with the differences in the description of DSC basket compartments, fuel regions, shielding materials densities and the thickness on ends of the cask, axial burn-up profile variation (BWR vs. PWR) of the radiological source. Such a difference is due to the fact that the cask is designed for the transportation of the various DSCs designed for BWR and PWR FAs. Presented description of NCT model is applicable to the models employed for calculating response functions used for transportation qualification of BWR assemblies with the difference in number and arrangement of fuel compartments and aluminum transition rails in DSCs. PWR model is very similar with the exception for burn-up profile, fuel region materials densities and composition.

Geometry of the irradiated waste canister and the volume occupied by the irradiated\contaminated hardware are specified in MCNP models using the following assumptions:

1. The canister modeled as a carbon steel cylinder with 70.50" diameter and 189.19" height. The cylinder is centered at the cask axis and it is 2.71" from the cask bottom plug.

2. Thickness of the cylindrical shell on side of the canister is 1.75". Thickness of shield plugs on bottom and top of the canister is 5.75" and 7.00", respectively. *See Appendix A.1.4.9A*

3. Radioactive waste occupies only portion of the inner volume of the canister. It is assumed that the waste is distributed within a cylindrical volume with 66.0" diameter and 168" height. Bottom of that cylindrical region is in contact with the bottom plug of the canister. The rest of the inner volume of the canister is occupied with air.

A.5.3.1.1 NCT Radial and Axial Shielding Configuration

The geometry of NCT model for the bounding shielding evaluation is a complete three dimensional simulation of the MP197HB transportation package loaded with 69BTH DSC containing design basis BWR assemblies. The cask, the DSC and its contents are modeled with a discrete representation of the basket and fuel structure. Each fuel assembly is divided into four axial zones. The bottom zone represents the lower end fittings, the middle zone the active fuel region and the upper zones represent the plenum and upper end fittings, respectively. The modeled active fuel length is 144 inches *for intact fuel* and the plenum length is 12.93 inches. *For damaged fuel in the NCT models, the active fuel length is modeled as 108 inches.* The modeled bottom end fitting and top end fitting lengths are 6.65 inches and 12.62 inches, respectively. The fuel, end fittings and plenum are homogenized within the each assembly envelope and the axial length of their respective zones. All of the above is applicable to MCNP models used for calculation of the response functions with the exception for axial extents of fuel regions, which are different in the models corresponding to the cask with DSCs containing PWR fuel.

A fuel basket assembly is designed to locate and support fuel assemblies. For NUHOMS® 69BTH, the basket structure consists of welded stainless steel tubes (fuel compartments) separated by aluminum-poison plates. Fuel compartments are arranged in 2 arrays (full 3×3 fuel compartment array and partial 3x3 fuel compartment array) surrounded by stainless steel wrap. These compartment arrays are separated by 0.375" thick aluminum plates. Solid aluminum transition rails centers fuel compartments clusters inside of the DSC, see sketch of the geometry in Figure A.5-4 and screen shot from MCNP geometry plotter in Figure A.5-6. Fuel compartments are modeled as square stainless steel tubes separated by aluminum sheets. Thickness of the compartments and aluminum plates in MCNP model is 0.20" and 0.12", respectively. This description also applies for the MCNP model used for calculation of response function related to MP197HB\69BTH DSC shielding configuration. This is a conservative modeling approach for representing DSC fuel compartments in the shielding calculations.

The fuel pins and fuel assembly hardware (end fittings and plenum materials) are homogenized within the each assembly envelope and the axial length of their respective axial zones in MCNP models. This is a conservative modeling approach for representing fuel regions in the shielding calculations.

The densities of the homogenized regions are calculated by summing all of the material in the region and dividing by the volume. This volume spans the volume of a cuboid that spans the outer envelop of the fuel assembly. The spacing between the fuel pins or other components of the fuel assembly is included in the homogenized volume. As an example, the total mass of materials in the active fuel zone for the BWR fuel assembly (from Table A.5-8, excluding the channel) is 276,870 grams and the total volume enveloped by the fuel assembly (5.44 inches wide by 144 inches high) is 69,833 cm^3. The density of the homogenized region is calculated to be 3.96 grams/cm^3 as shown in Table A.5-17.

The borated neutron shielding material (VYAL-B) is a vinylester resin mixed with alumina hydrate and zinc borate which are added for their fire retardant properties. The approximate elemental composition of the VYAL-B resin is shown in Table A.5-6. The neutron shielding material is embedded into 0.12" thick aluminum boxes. There are 60 such boxes around the side of the cask perimeter between impact limiters. Sides of the boxes adjacent to 2.50" thick cask Outer Shell and 0.375" Shield Shell are modeled as 0.125" thick aluminum cylindrical shells. Note thickness of the shells should be 0.12". One millimeter more in aluminum thickness, however, has a negligible, if any, impact on dose rates. The other two sides of the aluminum boxes are homogenized with the neutron shielding material. Mass of the homogenized aluminum is 0.753 kg. Density and composition of the neutron shielding material homogenized with the aluminum is presented in Table A.5-18.

Optional fins for excessive heat dissipation on outer surface of the cask are not modeled. MCNP simulation suggests that it may result in up to 15% (depending on burn-up\enrichment combination from FQT) of conservatism in calculated dose rates.

Trunnion plugs are assumed to be made from the same neutron shielding material that is on the side of the cask. The same amount of the neutron shielding material of the plugs is assumed lost during HAC on the cask side. The plug is encased with 0.0625" thick steel shell. The shell portion at the bottom of the plug is placed on top of the plug in MCNP models. This preserves the total thickness of the steel on top and the bottom of the plug. Assumed configuration of the trunnion plugs in MCNP models is shown on Figure A.5-7.

Geometry of the grapple ring cut-out on the cask bottom and shear key cut out on the side in MCNP models are shown on sketches of Figure A.5-8 and Figure A.5-9. *When the cask is in the vertical position the shear key cut-out is closed with the shear key plug made of steel and the same neutron shielding material as on the cask side. Prior to being rotated to the horizontal position and placed on the transport platform, the shear key plug is removed. When the cask is positioned horizontally on the transporter the shear key cut out fits over the steel shear key on the transporter platform. When the cask is removed from the transport platform, the shear key plug is reinstalled.*

The cask is secured in a horizontal position on a skid attached to a railcar or other trailer with a deck or floor during transportation. However the cask and its content are modeled as a stand alone entity, without any surroundings in computational models for bounding shielding evaluation and determination of the response functions. Therefore effect of transportation equipment on dose rate distributions below the cask (which can be especially important for the close, less than 2 meters distances) is conservatively not accounted for in the current analysis.

The impact limiters are modeled as wood surrounded by a 0.25 in. thick steel shell. The interior steel gussets are conservatively neglected. Wood thickness between end of the impact limiters and the cask ends is 26.25" in MCNP models. The outer diameter is 125.53".

A.5.3.1.2 HAC Radial and Axial Shielding Configuration under Hypothetical Accident Conditions

HAC models are similar to the NCT with exceptions highlighted in Section A.5.1.1. The same amount, 75%, of the neutron shielding material of the trunnion plugs is assumed lost during HAC at the cask side.

For HAC, models are developed for 69 intact assemblies, and 45 intact/24 damaged assemblies (see Figure A.5-6c). Because the margin to the dose rate limit is large, the design basis HAC source term is used in both models for all fuel assemblies. In the HAC models with damaged fuel, it is assumed that the entire fuel assembly may rubblize up to either 50% or 75% of the total fuel assembly volume, including the plenum and end fittings. The material description is simply a homogenization of the four homogenized regions in the standard fuel assembly. The source is then combined into one homogenized axial zone with a uniform axial distribution to simulate severe fuel relocation.

4.3 wt. % and 21.0 years cooled FAs loaded into 69BTH DSC should result in 9.40 mrem/hr dose rates at the location of interest at NCT. 96 % of that dose rate is due to neutron (including the secondary gamma, from (n,γ) interactions) radiation. Radiological sources from DB assembly with such parameters are used for calculation HAC dose rates in the bounding shielding analysis. HAC source is shown in Table A.5-12.

Sampling of axial position of radiological source particles in in-core region of fuel assemblies is governed by burn-up profile function. Burn-up profile functions for the BWR and PWR assemblies are shown in Table A.5-15 and Table A.5-16, respectively.

The MCNP shielding evaluation and calculation of the response functions accounts for axial burn-up variation and subcritical neutron multiplication as described in Section A.5.2.1 and Section A.5.2.2, respectively.

For the dose calculation around the MP197HB, the source is divided into four separate regions: fuel, plenum, top end fitting, and bottom end fitting. The model is utilized in two separate computer runs consisting of contributions from the following sources:

- Primary gamma radiation from the active fuel and from activated hardware within the top end fitting, plenum region and bottom end fitting (axial and radial directions).

- Neutron radiation from the active fuel region and secondary gamma radiation from neutron, mainly (n,γ), interactions.

The sources in the active fuel region (gamma and neutron) are modeled as uniform radially but vary axially. The sources in the structural hardware regions (plenum, top end fitting, and bottom end fitting) are modeled as uniform both radially and axially. The results from the individual runs are summed to provide the total gamma, neutron and total dose for the package.

A.5.3.3.2 Irradiated/Contaminated Waste

The source strength with its spectrum is specified in Section A.5.2.2.3. Geometry of the source region, its materials and composition are specified Section A.5.3.1 and Section A.5.3.2, respectively.

A.5.3.4 Physics Specification

Upper energy limit for detailed photon physics treatment during MCNP calculation is set to 20.0 MeV. Photons with energy less than 0.001 MeV are cut off. This covers energy spectrum from the fuel assemblies in storage or qualified for transportation. Physics photon treatment accounts for coherent scattering and Doppler energy broadening. It does not account for bremmstrahlung dose rate peaking due to radiation streaming is not expected may have larger size, but it does not exceed 46x39 cm. Such size is sufficient to spot other potential regions where dose rate peaking can occur and verify the designation of the "critical" regions described.

4. The dose rate distribution is flattened at distances greater than 2-meters and eventually becomes flat at distances are in the order of the cask size or larger. Therefore larger mesh tallies segments are used at distances greater than 2 meters from the cask.
5. Flux-to-dose conversion factors (from Reference [6]) in Table A.5-19 and Table A.5-20 are used in MCNP input decks for calculating dose rates and response functions.
6. It is assumed that occupied position during the cask transport is at the ends of the transportation platform or rail car. As discussed above, the distances from the ends of the impact limiters correspond *to* rail car decks that are 40 and 50 feet long assuming the cask is centered on the railcar. (These correspond to the 2.7 and 4.3 meter locations from ends of impact limiters at NCT or 3.0 and 4.6 meter locations from ends of impact limiters at HAC.)
7. The outer diameter of the impact limiters is 126". The diameter of the cask body and impact limiters in MCNP models is 98.5" and 125.53", respectively. The width of the rail car is expected to be at least 130 inches. It is also expected that there is a protective screen around the package during the transportation. The personnel barrier can extend as far as 6" from the side of the impact limiters. Given these assumptions the dose rates reported at various distances from the side of the package are 8 inches closer to the package than in reality. This results in an over estimate of the NCT dose rates at the side of the cask shown in Table A.5-23 and Table A.5-28.
8. When the package is secured on a railcar and ready for transport an imaginary horizontal plane (through the cask axis and the trunnions is at 90 degrees angular coordinate in the cylindrical coordinate system) is used for the definition of the tallies for calculating dose rates at various radial distances along the cask side. Dose rates along the transportation package side reported in the current section are at fixed radial distances. This implies an additional conservatism since only dose rates on a vertical plane from the transportation platform represent the interest.
9. Notes in item 7 through item 8 above are also applicable to discussion in Section A.5.4, FQT calculations described in Section A.5.5, Section A.5.5.1 through Section A.5.5.3.

A.5.4 Shielding Evaluation

A.5.4.1 Used Fuel in DSCs

Dose rates (for NCT and HAC) from the MP197HB package with a DSC payload are summarized in Table A.5-1 through Table A.5-3. *Results are provided for cases with and without damaged fuel assemblies.* Maximum dose rates at various distances from the cask are provided in Table A.5-21 through Table A.5-26 *only for configurations without damaged fuel* *assemblies.* Also spatial distributions of the dose rates at various distances from the cask are shown on plots of Figure A.5-10 through Figure A.5-15 *for the NCT configuration without damaged assemblies.* The dose rates at 2 meters from the vehicle edge are calculated assuming that the vehicle edge is 8 inches away from the edge of the impact limiters. *These dose rates are shown in Table A.5-1, Table A.5-1a, and Table A.5-1b.* The results show that the MP197HB transportation package comply with dose rates restrictions during NCT and HAC.

Due to statistical nature of simulations with MCNP, dose rates are determined with uncertainties. The uncertainties are less than 3% for the total dose rates at the locations where dose rates criteria for the condition of transport are defined in Section A.5.1.

The statistical uncertainties are generally less than 3% for the majority of tallies except for some local tally bins for the accident evaluations. Table A.5-22 and Table A.5-24 through Table A.5-26 show some gamma dose rates where the calculated dose rates have the larger uncertainty. Note however that contribution of the primary gamma radiation to the total dose rate is very small in comparison with the total dose rate. Also, *the* location of maximum of total dose rate does not coincide with *the* location of maximums of different dose rate components due to neutron, primary and secondary gamma radiation sources. For the accident evaluation, the neutron dose rates have the highest relative uncertainty which are around 2%.

The dose rate location terminology is illustrated in Figure A.5-16 and is described below:

- *The radial dose rates refer to those calculated from the cask side.*
- *The "Cask Body Shield Shell" surface refers to the outer side surface of the cask body and is shown as location 1 in Figure A.5-16.*
- *The "Package Side Perimeter" surface refers to the outer side surface of the cask body between the impact limiters and the outer side surface of the impact limiters (where impact limiters are present) and is shown as location 2 in Figure A.5-16.*
- *The "Side of Impact Limiters (ILs)" surface refers to the outer side surface of the impact limiters and is shown as location 3 in Figure A.5-16.*
- *The axial dose rates refer to the dose rates calculated from the Top and Bottom Ends of the Impact Limiters and are shown as location 4 in Figure A.5-16.*

Dose rates at various distances measured from ends of impact limiters are also presented for NCT and HAC. The reported dose rates are due to the cask loaded with 69BTH DSC containing radiological sources due to DB FAs.

Note, based on the definition of the FQTs for the transportation purpose, NCT dose rates at two meters from side of the impact limiters due to assemblies at FQT cooling times are less than or equal to 9.9 mrem/hr, regardless of the DSC type contained in the cask. Therefore, any burn-up, enrichment, cooling time combination from FQT resulting in 9.9 mrem/hr could have been used for the bounding shielding evaluation. This ensures that there is adequate margin to the regulatory dose rate limits at two meters from the edge of the vehicle which is approximately 8 inches away from the edge of the impact limiters. The maximum regulatory dose rate at two meters from the vehicle edge is 9.37 mrem/hour. *The FQT methodology employed herein ensures* *that this maximum regulatory dose rate calculated using the HAC source terms will be less than or equal to 9.37 mrem/hour.*

Due to shielding properties of the cask and concentration of most of the radiological source strength in in-core region, dose rate values along the cask side between impact limiters are larger than dose rates at the same distances from ends of impact limiters, regardless of the fuel assembly types loaded, conditions of transport (NCT or HAC) and DSC types. Therefore in general, if dose rates along the side of the package comply with conditions of transport dose rate restrictions specified at the end of Section A.5.1, it indicates that shielding performance of the cask meets the regulatory requirements.

An evaluation of the contribution from neutron radiation source to the total dose rate was performed using response functions. It showed that the neutron radiation dose rates for MP197HB cask loaded with 69BTH DSC containing neutron radiation source from Table A.5-12 (the one that was subsequently used for HAC shielding evaluation) results in bounding neutron radiation dose rates. The cask configuration for HAC results in the dose rates that are nearly totally dominated by neutron radiation source at high burnups. Therefore it was concluded that radiological source from Table A.5-12 will result in bounding HAC dose rates.

When other DSCs are loaded in the cask, presented dose rates at various distances from ends of impact limiters may be slightly different. This is due to differences in the DSCs shield plug thicknesses and source terms distribution and strength in axial exposure regions. However the evaluation presented herein bounds the other cases.

The dose rate distribution on side of the cask body displayed on Figure A.5-10 demonstrates variation of the shielding properties on side of the cask. The distribution can be mapped on surface of the cask using a cylindrical coordinate system with an axis coincident with the DSC\cask axis. Polar (angular) coordinate is measured in rotations from an imaginary plane through sheer key and DSC\cask axis. The center of the shear key is at 0.0 or 1.0 rotations angular coordinate in that system. Centers of the trunnions are at 0.25 and 0.75 rotations. Dose rate variation can be understood if looking on modeled geometry of the cask shown on sketch of Figure A.5-1. It can be seen from the figure that active fuel region on bottom of the cask extends axially beyond the neutron shield on side of the cask. Also the in-core region faces trunnion attachment blocks on bottom portion of the cask side (not shown on Figure A.5-1). The attachment blocks cut large pieces of the neutron shielding material out side of the cask. Because they are made from steel they provide poor protection from neutron radiation. These two geometry features cause peak 1 and 2 on Figure A.5-10. Naturally, if there were no trunnion attachment blocks and neutron shielding would extend throughout an entire length of the active region, dose rate on side of the cask would be highest at the mid of the in-core region, which is marked with 3 on the figure. Solid aluminum transition rails around peripheral DSC fuel compartments provide fair additional shielding, especially against gamma radiation. If looking on Figure A.5-1 one can notice that Top Nozzle region extends above solid aluminum transition rails in the MCNP model for the bounding shielding evaluation. This causes a fourth peak on Figure A.5-10.

NCT maximum dose rates are summarized in Table A.5-1, *Table A.5-1a, and Table A.5-1b. These tables present* dose rates on *the* surface of the transport package, edge of the *vehicle* and 2 meters from the edge *of the vehicle for cases without damaged fuel assemblies, with 4 damaged fuel assemblies, and with 24 damaged fuel assemblies*. For the HAC, Table A.5-2 *and Table A.5-2a summarize* the maximum calculated doses at 1 m from the cask body shield shell *with no damaged assemblies and with 24 damaged assemblies. Damaged cases are performed for both 50% and 75% fuel rubblization by volume for the damaged fuel assemblies. The degree of rubblization has almost no effect on the maximum radial dose rates, although the bottom dose rates are slightly larger for 50% rubblization.*

The effect of damaged fuel assemblies for the DSCs where they are loaded in the central compartments (described in Table A.5-4) is insignificant and is not modeled. The dose rates for an individual at the end of the rail car are presented in Table A.5-3. These results are presented as a function of the length of the rail car. Comparing these dose rates to the maximum dose rates in Table A.5-21 and Table A.5-22, a minimum distance of 3.3 m from the outer surface of the impact limiter is required to ensure that the dose rate at the occupied locations is below the limit of 2.0 mrem/hour.

The dose rate analysis was performed using MCNP5 [5]. Selected inputs for MCNP are included in Section A.5.7.

A.5.5 Fuel Qualification

As stated previously, the MP197HB cask is designed to transport one of several NUHOMS® DSCs loaded with used fuel assemblies or dry irradiated and/or contaminated non-fuel bearing solid materials in *the RWC* in accordance with 10 CFR 71. To comply with thermal requirements for the total heat load per DSC, fuel assemblies can be distributed within their fuel compartments. These compartments are grouped within certain radial zones which are referred to as heat zones. An evaluation was performed for every heat zone to determine the fuel assembly parameters of burnup, percent initial enrichment and cooling time that would result in decay heats per assembly not exceeding certain limit. Minimum required cooling times to meet certain decay heat restrictions can be found as solutions of decay heat equations (DHEs) specified in Section A.5.5.4.1 and A.5.5.4.2.

The MP197HB transportation package must also comply with dose rates restrictions during NCT and HAC. Those criteria are summarized at the end of Section A.5.1.

An evaluation was performed for fuel assembly parameters of burnup, percent initial enrichment and cooling time that would result in normal conditions of transport dose rate at 2.0 meters from the transportation package side not exceeding 10.0 mrem/hr. The results are expressed in tabular format showing the minimum required cooling times as a function of enrichment and burnup. Such tables are referred to as Fuel Qualification Tables (FQTs) based on shielding evaluation or transportation FQTs. Note, that after the cooling times to meet 10.0 mrem/hr dose rate requirement were determined, they were rounded up to the nearest 1/2-year in the final transportation FQTs. For example 5.1 is rounded up to 5.5, 8.8 is rounded up to 9.0, etc. At the end, the cooling times in the final transportation FQTs actually stand for minimum required cooling time not to exceed 9.9 mrem/hr NCT dose rate at 2 meters from the package.

Due to shielding properties of the cask and concentration of the most of the radiological source strength in in-core region, dose rate values along the cask side between ends of impact limiters are larger than dose rates at the same distances from ends of impact limiters, regardless of the fuel assembly types loaded, conditions of transport (NCT or HAC) and DSC types. The results from the bounding shielding evaluation verify this assumption. Therefore, if dose rates along the side of the package at two meters from impact limiters comply with NCT dose rate restrictions specified at the end of Section A.5.1, dose rates at the same distance from the cask ends will also be compliant.

It follows from the bounding evaluations discussed in Section A.5.4 that compliance with the NCT 10.0 mrem/hr dose rate restriction at 2 meters from the package outer surface also results in compliance with the rest of conditions of transport specified at the end of Section A.5.1.
The transportation FQTs are, therefore, a set of acceptable combinations of burnup, enrichment, and cooling times such that the expected NCT dose rates at two meters from the cask are the same, regardless of the canister type contained inside of the cask. Therefore, no bounding set of NCT source terms are generated – rather all the source terms are expected to result in similar dose rates – bounded only by the maximum dose rate limits.

Proprietary information on pages A.5-33, A.5-34, and A.5-34a, withheld pursuant to 10 CFR 2.390

A.5.6 References

1. SCALE 4.4, "Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers," CCC-545, ORNL.

2. *B. L. Broadhead, "Recommendations for Shielding Evaluations for Transport and Storage Packages," NUREG/CR-6902, ORNL/TM–2002/31, May 2003.*

3. NUREG/CR-6801 (ORNL/TM-2001/273), March 2003, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses."

4. Luksic, "Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal," PNL-6906, UC-85, June 1989.

5. "MCNP/MCNPX – Monte Carlo N-Particle Transport Code System Including MCNP5 1.40 and MCNPX 2.5.0 and Data Libraries," CCC-730, Oak Ridge National Laboratory, RSICC Computer Code Collection, January 2006.

6. "American National Standard for Calculation and Measurement of Direct and Scattered Gamma Radiation from LWR Nuclear Power Plants," ANSI/ANS-6.6.1-1977, American Nuclear Society, Illinois, 1977.

7. *O. W. Hermann and M. D. DeHart, "Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel." ORNL/TM-13315, Oak Ridge National Laboratory, September 1998.*

8. *"Isotopic Analysis of High-Burnup PWR Spent Fuel Samples From the Takahama-3 Reactor," ORNL/TM-2001/259 (NUREG/CR-6798), CE Sanders and IC Gauld, January 2003*

9. DOE Report, "Characteristics of Potential Repository Wastes," DOE/RW-0184-R1, Volume 1, July 1992.

10. EIE Service Report, "Spent Nuclear Fuel Discharges from U.S. reactors 1993," SR/CNEAF/95-01, February 1995.

11. O.W. Hermann, C.V.Parks, and J.P.Renier "Technical Support for Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data," NUREG/CR-5625, ORNL-6698, July 1994.

12. U. S. Nuclear Regulatory Commission, "Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High Burnup LWR Fuel," NUREG/CR-6701, Published January 2001, ORNL/TM-2000/277.

13. B. D Murphy, "Prediction of Isotopic Composition of UO2 Fuel from a BWR: Analysis of the DU1 Sample from the Dodewaard Reactor," ORNL/TM-13687, October 1998.

14. Japan Atomic Energy Research Institute, "Technical Development on Burnup Credit for Spent LWR Fuels," JAERI-Tech 2000-071, September 21, 2000.

15. U. S. Nuclear Regulatory Commission, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High Burnup LWR Fuel," NUREG/CR-6700, Published January 2001, ORNL/TM-2000/284.

16. O. W. Hermann, S. M. Bowman, M. C. Brady, C. V. Parks, "Validation of the SCALE

System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, March 1995.

17. M. D. DeHart and O. W. Hermann, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, September 1996.

18. Transnuclear Inc, "Updated Final Safety Analysis Report (UFSAR) for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 1*1*.

Proprietary information on pages A.5-36, to A.5-105 withheld pursuant to 10 CFR 2.390

Table A.5-1
Summary of MP197HB NCT Dose Rates
(Exclusive Use Package for Transportation)

Radiation	Transport Package Surface, mSv/h (mrem/h)			Vehicle Edge, mSv/h (mrem/h) (1)			2 Meters from Vehicle Edge, mSv/h (mrem/h) (2)		
	Top	Side	Bottom	Top	Side	Bottom	Top	Side	Bottom
Gamma	0.03 (2.7)	0.59 (58.7)	0.16 (16.0)	0.03 (2.7)	0.23 (22.8)	0.16 (16.0)	0.01 (0.6)	0.051 (5.1)(3)	0.02 (1.8)
Neutron	0.02 (2.4)	1.51 (151)	0.16 (16.0)	0.02 (2.4)	0.58 (57.8)	0.16 (16.0)	0.01 (0.8)	0.043 (4.3)(3)	0.02 (2.4)
Total(4)	0.05 (5.1)	1.91 (191)	0.32 (32.0)	0.05 (5.1)	0.69 (68.7)	0.32 (32.0)	0.01 (1.2)	0.094 (9.4)(3)	0.04 (4.2)
Gamma from irradiated/ contaminated waste canister	0.37 (37.1)	0.30 (30.2)	0.72 (72.2)	0.37 (37.1)	0.24 (24.1)	0.72 (72.2)	0.077 (7.7)	0.083 (8.3)	0.083 (8.3)
Limit	10 (1000)	10 (1000)	10 (1000)	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

(1) Dose rates are calculated at the edges of Impact Limiters (ILs).
(2) Dose rates are calculated at a distance measured from ILs.
(3) Dose rates are not calculated at a distance measured from ILs.
(4) Location of the maximum total dose rate does not coincide with the position of maximum neutron or maximum gamma dose rates. Therefore, the maximum total dose rate is less than or equal to the sum of the maximum neutron and maximum gamma dose rate.

Table A.5-1a
Summary of MP197HB NCT Dose Rates with 4 Damaged Fuel Assemblies
(Exclusive Use Package for Transportation)

Radiation	Transport Package Surface, mSv/h (mrem/hr)			Vehicle Edge, mSv/h (mrem/hr)(1)			2 Meters from Vehicle, mSv/h (mrem/hr)(2)		
	Top	Side	Bottom	Top	Side	Bottom	Top	Side	Bottom
Gamma	0.027 (2.7)	0.30 (29.6)	0.16 (15.7)	0.027 (2.7)	0.22 (21.6)	0.16 (15.7)	0.005 (0.5)	0.054 (5.4)(3)	0.018 (1.8)
Neutron	0.025 (2.5)	0.64 (63.7)	0.17 (16.6)	0.025 (2.5)	0.31 (30.6)	0.17 (16.6)	0.008 (0.8)	0.048 (4.8)(3)	0.025 (2.5)
Total(4)	0.052 (5.2)	0.75 (75.2)	0.32 (32.3)	0.052 (5.2)	0.41 (41.3)	0.32 (32.3)	0.012 (1.2)	0.094 (9.4)(3)	0.043 (4.3)
Limit	10 (1000)	10 (1000)	10 (1000)	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

(1) Dose rates are calculated at the edges of impact limiters (ILs).
(2) Dose rates are calculated at a distance measured from ILs.
(3) Dose rates are not calculated at a distance measured from ILs.
(4) Location of the maximum total dose rate does not coincide with the position of maximum neutron or maximum gamma dose rates. Therefore, the maximum total dose rate is less than or equal to the sum of the maximum neutron and maximum gamma dose rate.

Table A.5-1b
Summary of MP197HB NCT Dose Rates with 24 Damaged Fuel Assemblies
(Exclusive Use Package for Transportation)

Radiation	Transport Package Surface, mSv/h (mrem/hr)			Vehicle Edge, mSv/h (mrem/hr)(1)			2 Meters from Vehicle, mSv/h (mrem/hr)(2)		
	Top	Side	Bottom	Top	Side	Bottom	Top	Side	Bottom
Gamma	0.027 (2.7)	0.21 (21.3)	0.16 (15.6)	0.027 (2.7)	0.20 (19.6)	0.16 (15.6)	0.005 (0.5)	0.039 (3.9)(3)	0.017 (1.7)
Neutron	0.028 (2.8)	0.63 (63.4)	0.17 (17.1)	0.028 (2.8)	0.30 (30.3)	0.17 (17.1)	0.007 (0.7)	0.046 (4.6)(3)	0.025 (2.5)
Total(4)	0.055 (5.5)	0.74 (74.1)	0.33 (32.8)	0.055 (5.5)	0.39 (38.7)	0.33 (32.8)	0.01 (1.0)	0.079 (7.9)(3)	0.043 (4.3)
Limit	10 (1000)	10 (1000)	10 (1000)	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

(1) Dose rates are calculated at the edges of impact limiters (ILs).
(2) Dose rates are calculated at a distance measured from ILs.
(3) Dose rates are not calculated at a distance measured from ILs.
(4) Location of the maximum total dose rate does not coincide with the position of maximum neutron or maximum gamma dose rates. Therefore, the maximum total dose rate is less than or equal to the sum of the maximum neutron and maximum gamma dose rate.

Table A.5-2
Summary of MP197HB HAC Dose Rates
(Exclusive Use Package for Transportation)

Radiation	1 Meter from Vehicle Edge, mSv/h (mrem/h) [(1)]		
	Top	**Side**	**Bottom**
Gamma	0.02 (1.5)	0.18 (17.9)	0.02 (1.8)
Neutron	0.17 (17.4)	2.94 (294)	0.56 (55.8)
Total	0.19 (18.5)	3.12 (312)	0.57 (57.3)
Gamma from irradiated waste canister	0.28 (28.4)	0.49 (49.1)	0.37 (37.3)
Limit	10 (1000)	10 (1000)	10 (1000)

(1) Dose rates are calculated at a distance measured from cask body shield shell.

Table A.5-2a
Summary of MP197HB HAC Dose Rates-Damaged Fuel

	50% Rubblization *1 Meter from Package Surface, mSv/hr (mrem/hr)*			***75% Rubblization*** *1 Meter from Package Surface, mSv/hr (mrem/hr)*		
Radiation	***Top***	***Side***	***Bottom***	***Top***	***Side***	***Bottom***
Gamma	*0.011 (1.1)*	*0.17 (17.1)*	*0.19 (1.9)*	*0.012 (1.2)*	*0.17 (17.3)*	*0.019 (1.9)*
Neutron	*0.17 (17.3)*	*2.88 (288)*	*2.19 (219)*	*0.19 (19.1)*	*2.94 (294)*	*1.88 (188)*
Total[(1)]	*0.18 (18.4)*	*3.02 (302)*	*2.21 (221)*	*0.20 (20.2)*	*3.08 (308)*	*1.90 (190)*
Limit	*10 (1000)*	*10 (1000)*	*10 (1000)*	*10 (1000)*	*10 (1000)*	*10 (1000)*

(1) Location of the maximum total dose rate does not coincide with the position of maximum neutron or maximum gamma dose rates. Therefore, the maximum total dose rate is less than or equal to the sum of the maximum neutron and maximum gamma dose rate.

Table A.5-4
DSC Damaged Fuel Limits

DSC Type	***Maximum Number of Damaged Fuel Assemblies***	***Location of Damaged Fuel***
24PT4	*12*	*Peripheral compartments*
32PT	*0*	*N/A*
24PTH	*12*	*Peripheral compartments*
32PTH	*16*	*Central compartments*
32PTH1	*16*	*Central compartments*
37PTH	*4*	*4 corners in periphery of the basket compartments*
61BT	*16*	*4 corners in periphery of the basket compartments*
61BTH	*16*	*4 corners in periphery of the basket compartments*
69BTH	*24*	*4 corners in periphery of the basket compartments*

Table A.5-7
DB BWR Fuel Assembly Design Characteristics

Parameter	Value	Reference
• Number of fuel rods positions per assembly	49	[9]
• Typical number of fuel rods per assembly	49	[9]
• Maximum uranium loading per assembly	198.0 kg	(see Note 1)
• Channel avg. ^{10}B content.	*50* ppm	*(see Note 3)*
• Water temperature, °K	558	[11]
• Channel water temperature, °K	552	[11]
• Water vol-avg. density, g/cm^3	0.4323	[11]
• Channel water density, g/cm^3	0.669	[11]
• Water temperature at bottom region, °K	552	(see Note 1)
• Water density at bottom region, g/cm^3	0.743	(see Note 1)
• Water temperature at plenum/top region, °K	558	(see Note 1)
• Water density at plenum/top region, g/cm^3	0.264	(see Note 1)
• Rod pitch, cm (in)	1.87452 (0.738)	[9]
• Pellet Outer Diameter LTL, cm (in)	1.23698 (0.487)	[9]
• Cladding (Rod) Inner Diameter, cm (in)	1.26746 (0.499)	[9]
• Cladding (Rod) Outer Diameter, cm (in)	1.43002 (0.563)	[9]
• Cladding material	Zircaloy	[9,11]
• Clad temperature, °K	620	[11]
• Effective fuel temperature, °K	840	[11]
• Assembly width, in	(2)5.44-6.515	[9,10]
• Assembly length, in	(2)171.13-175.87	[9,10]
• Active fuel length, cm. (in)	365.76(144)	[9,11]
• Rod length cm (in)	406.4 (160)	[9]
• Plenum Length, in	(2)11.24-16.0	[9]

Notes:
(1) Presented value(s) are representative or bounding. They are set based on studying of data in numerous references.

(2) These are typical dimensions for 7x7 BWR assemblies. Actual dimensions for DB FA used in the analysis are shown in Table A.5-4. Fuel assembly width is 5.44' in MCNP models.

(3) A density multiplier of 7.15e-6 is used in the SAS2H model to obtain a soluble boron concentration of approximately 50 ppm.

Table A.5-11a
NCT Radiological Source for Damaged Fuel

NCT Radiological Source, g/(sec*FA): 55 GWD/MTU, 3.8 wt. %, after 15 years of cooling(1)						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	In-Core	Plenum	Top Nozzle
0.00E+00	to	5.00E-02	3.7829E+10	5.8239E+14	1.8648E+10	3.0983E+10
5.00E-02	to	1.00E-01	4.6093E+09	1.1509E+14	1.5566E+09	3.7989E+09
1.00E-01	to	2.00E-01	1.2240E+09	8.0929E+13	9.5873E+08	9.9126E+08
2.00E-01	to	3.00E-01	6.1930E+07	2.4192E+13	5.3605E+07	5.0005E+07
3.00E-01	to	4.00E-01	9.9245E+07	1.5832E+13	1.6392E+08	7.7575E+07
4.00E-01	to	6.00E-01	6.0414E+08	2.9198E+13	3.1194E+09	4.0344E+08
6.00E-01	to	8.00E-01	3.2433E+08	7.7619E+14	1.6148E+09	2.5749E+08
8.00E-01	to	1.00E+00	8.9326E+07	1.7278E+13	2.5075E+07	1.0226E+08
1.00E+00	to	1.33E+00	1.3434E+12	2.6548E+13	4.4353E+11	1.1075E+12
1.33E+00	to	1.66E+00	3.7937E+11	4.3262E+12	1.2525E+11	3.1277E+11
1.66E+00	to	2.00E+00	6.1375E+00	4.1475E+10	3.1756E+01	4.0919E+00
2.00E+00	to	2.50E+00	9.0029E+06	2.7898E+09	2.9724E+06	7.4225E+06
2.50E+00	to	3.00E+00	1.3960E+04	2.2345E+08	4.6090E+03	1.1509E+04
3.00E+00	to	4.00E+00	5.1263E-12	3.2824E+07	2.0575E-15	2.8163E-11
4.00E+00	to	5.00E+00	0.0000E+00	8.3891E+06	0.0000E+00	0.0000E+00
5.00E+00	to	6.50E+00	0.0000E+00	3.3668E+06	0.0000E+00	0.0000E+00
6.50E+00	to	8.00E+00	0.0000E+00	6.6046E+05	0.0000E+00	0.0000E+00
8.00E+00	to	1.00E+01	0.0000E+00	1.4023E+05	0.0000E+00	0.0000E+00
Total Gamma, g/(sec*FA)			1.7676E+12	1.6720E+15	5.9492E+11	1.4569E+12
Total Neutron, n/(sec*FA)			2.438E+08			

(1) This source term for damaged fuel is used only in the NCT analysis with 24 damaged fuel assemblies. For the NCT analysis with 4 damaged fuel assemblies, source terms from Table A.5-11 are used for damaged fuel.

Table A.5-15
BWR Axial Peaking Factors

Zone	Fraction of Core Height	Fractional Width of Zone	Burnup Profile	Gamma Peaking Factor (normalized)	Neutron Peaking Factor (normalized)
1	0.05	0.05	0.2357	0.2256	0.0018
2	0.1	0.05	0.7746	0.7674	0.1683
3	0.2	0.1	1.0750	1.0854	0.8447
4	0.3	0.1	1.1836	1.2027	1.3859
5	0.4	0.1	1.2000	1.2223	1.5288
6	0.5	0.1	1.2000	1.2244	1.5775
7	0.6	0.1	1.1912	1.2164	1.5624
8	0.7	0.1	1.1515	1.1227	1.3842
9	0.8	0.1	1.0766	1.0964	1.0707
10	0.9	0.1	0.8973	0.9053	0.5047
11	0.95	0.05	0.6330	0.6255	0.1093
12	1.0	0.05	0.2410	0.2303	0.0028
			Average	1.000	1.000
Ratio of the "true" total assembly source strength to the source strength computed with an average assembly burnup				1.0	1.326

Table A.5-16
PWR Axial Peaking Factors

Zone	*Active Fuel Zone Center (% of Height)*	*Fractional Width of Zone*	*Burnup Profile*	*Gamma Peaking Factor (normalized)*	*Neutron Peaking Factor (not-normalized)*	*Neutron Peaking Factor (normalized)*
1	2.78	0.056	0.573	0.573	0.108	0.0936
2	8.33	0.055	0.917	0.917	0.707	0.614
3	13.89	0.056	1.066	1.066	1.291	1.121
4	19.44	0.055	1.106	1.106	1.496	1.299
5	25	0.056	1.114	1.114	1.540	1.337
6	30.56	0.055	1.111	1.111	1.524	1.323
7	36.11	0.056	1.106	1.106	1.496	1.299
8	41.69	0.056	1.101	1.101	1.469	1.276
9	47.22	0.055	1.097	1.097	1.448	1.257
10	52.78	0.056	1.093	1.093	1.427	1.239
11	58.33	0.055	1.089	1.089	1.406	1.221
12	63.89	0.057	1.086	1.086	1.391	1.207
13	69.44	0.054	1.081	1.081	1.366	1.185
14	75	0.057	1.073	1.073	1.326	1.151
15	80.56	0.054	1.051	1.051	1.220	1.059
16	86.11	0.057	0.993	0.993	0.972	0.844
17	91.67	0.055	0.832	0.832	0.479	0.416
18	97.22	0.056	0.512	0.512	0.069	0.0597
		Average	1.000	1.000	1.152	1.000

Note: The ratio of the "true" total assembly neutron source strength to the neutron source strength computed with an average assembly burnup is 1.152.

Table A.5-23
NCT Maximum Dose Rate at Various Radial Distances from Side of Impact Limiter

Axial Distance from Side of IL,s m	Primary Gamma Radiation		Secondary Gamma Radiation		Total Gamma Radiation		Neutron Radiation		Total	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
Package Side Perimeter	20.3	0.02	13.3	0.004	22.8	0.02	57.8	0.01	68.7	0.01
0	20.8	0.02	9.3	0.003	21.6	0.02	27.6	0.01	38.6	0.01
1	6.1	0.03	4.1	0.003	9.2	0.01	8.2	0.01	17.1	0.01
2	3.2	0.03	2.2	0.004	5.5	0.02	4.7	0.01	9.9	0.01
2.7	2.5	0.02	1.6	0.004	4.1	0.02	3.4	0.01	7.4	0.01
4.3	1.5	0.03	0.8	0.005	2.3	0.02	1.8	0.01	4.2	0.01

Note: Location of Maximum of Total in dose rate spatial distribution may not coincide with positions of neutron, primary and secondary gamma radiation dose rates maximums. Because of that the maximum of Total is less than or equal to the sum of maximums from neutron, primary and secondary gamma radiation dose rates.

Table A.5-26
HAC Maximum Dose Rate at Various Radial Distances from Side of Cask

Radial Distance from [1]Impact Limiters, m	Primary Gamma Radiation		Secondary Gamma Radiation		Total Gamma Radiation		Neutron Radiation		Total	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
Package *Side* Perimeter	15.5	0.11	44.5	0.005	55.2	0.02	847.8	0.01	903.0	0.01
0	7.9	0.16	27.8	0.005	33.6	0.02	557.8	0.005	591.4	0.005
1	4.3	0.09	13.7	0.004	17.9	0.02	293.7	0.004	311.7	0.004
2	2.6	0.12	6.5	0.006	9.1	0.03	147.4	0.005	156.5	0.005
2.7	1.7	0.12	3.7	0.007	5.3	0.04	87.7	0.01	93.0	0.01
4.3	1.1	0.17	1.9	0.01	3.0	0.06	46.5	0.01	49.2	0.01

(1) HAC dose rates for distances equal to 1 and 2 meters correspond to radial distances measured from the cask body (Shield Shell), not from side of Impact Limiters.

Note: Location of Maximum of Total in dose rate spatial distribution may not coincide with positions of neutron, primary and secondary gamma radiation dose rates maximums. Because of that, the maximum of Total is less than or equal to the sum of maximums from neutron, primary and secondary gamma radiation dose rates.

Table A.5-28
Summary on Maximum Dose Rate at Various Radial Distances from Side of the Cask Containing Irradiated Waste Canister

Radial Distance from Side of ILs or *Body* (1), m	Normal Conditions of Transport (NCT)		Hypothetical Accident Condition	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
Shield Shell	64.0	0.01	103.0	0.01
Package *Side* Perimeter	30.2	0.01	103.0	0.01
0	24.1	0.01	74.3	0.01
1	12.7	0.005	49.1	0.01
2	8.33	0.01	31.2	0.01
2.7	6.4	0.01	20.9	0.01

(1) HAC dose rates for distances equal to 1 and 2 meters correspond to radial distances measured from the cask body (Shield Shell), not from side of Impact Limiters.

Proprietary information on pages A.5-133a, through -133d withheld pursuant to 10 CFR 2.390

Table A.5-23
NCT Maximum Dose Rate at Various Radial Distances from Side of Impact Limiter

Axial Distance from Side of IL,s m	Primary Gamma Radiation		Secondary Gamma Radiation		Total Gamma Radiation		Neutron Radiation		Total	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
Package Side Perimeter	20.3	0.02	13.3	0.004	22.8	0.02	57.8	0.01	68.7	0.01
0	20.8	0.02	9.3	0.003	21.6	0.02	27.6	0.01	38.6	0.01
1	6.1	0.03	4.1	0.003	9.2	0.01	8.2	0.01	17.1	0.01
2	3.2	0.03	2.2	0.004	5.5	0.02	4.7	0.01	9.9	0.01
2.7	2.5	0.02	1.6	0.004	4.1	0.02	3.4	0.01	7.4	0.01
4.3	1.5	0.03	0.8	0.005	2.3	0.02	1.8	0.01	4.2	0.01

Note: Location of Maximum of Total in dose rate spatial distribution may not coincide with positions of neutron, primary and secondary gamma radiation dose rates maximums. Because of that the maximum of Total is less than or equal to the sum of maximums from neutron, primary and secondary gamma radiation dose rates.

Table A.5-26
HAC Maximum Dose Rate at Various Radial Distances from Side of Cask

Radial Distance from [(1)]Impact Limiters, m	Primary Gamma Radiation		Secondary Gamma Radiation		Total Gamma Radiation		Neutron Radiation		Total	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
Package *Side* Perimeter	15.5	0.11	44.5	0.005	55.2	0.02	847.8	0.01	903.0	0.01
0	7.9	0.16	27.8	0.005	33.6	0.02	557.8	0.005	591.4	0.005
1	4.3	0.09	13.7	0.004	17.9	0.02	293.7	0.004	311.7	0.004
2	2.6	0.12	6.5	0.006	9.1	0.03	147.4	0.005	156.5	0.005
2.7	1.7	0.12	3.7	0.007	5.3	0.04	87.7	0.01	93.0	0.01
4.3	1.1	0.17	1.9	0.01	3.0	0.06	46.5	0.01	49.2	0.01

(1) HAC dose rates for distances equal to 1 and 2 meters correspond to radial distances measured from the cask body (Shield Shell), not from side of Impact Limiters.

Note: Location of Maximum of Total in dose rate spatial distribution may not coincide with positions of neutron, primary and secondary gamma radiation dose rates maximums. Because of that, the maximum of Total is less than or equal to the sum of maximums from neutron, primary and secondary gamma radiation dose rates.

Table A.5-28
Summary on Maximum Dose Rate at Various Radial Distances from Side of the Cask Containing Irradiated Waste Canister

Radial Distance from Side of ILs or *Body* (1), m	Normal Conditions of Transport (NCT)		Hypothetical Accident Condition	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
Shield Shell	64.0	0.01	103.0	0.01
Package *Side* Perimeter	30.2	0.01	103.0	0.01
0	24.1	0.01	74.3	0.01
1	12.7	0.005	49.1	0.01
2	8.33	0.01	31.2	0.01
2.7	6.4	0.01	20.9	0.01

(1) HAC dose rates for distances equal to 1 and 2 meters correspond to radial distances measured from the cask body (Shield Shell), not from side of Impact Limiters.

Proprietary information on pages A.5-133a, through -133d withheld pursuant to 10 CFR 2.390

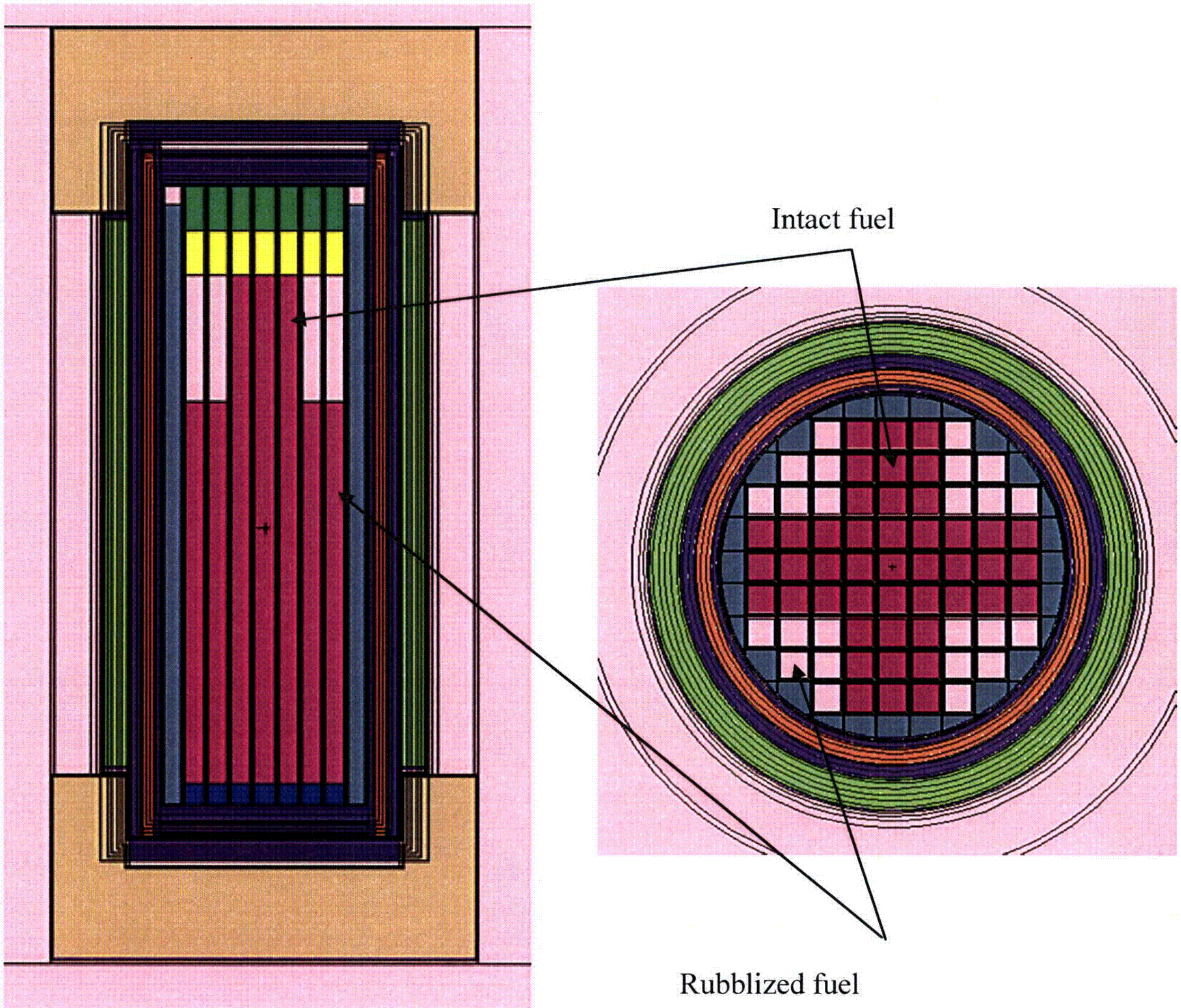


Figure A.5-6a
NCT Model Geometry with 24 Damaged Fuel Assemblies

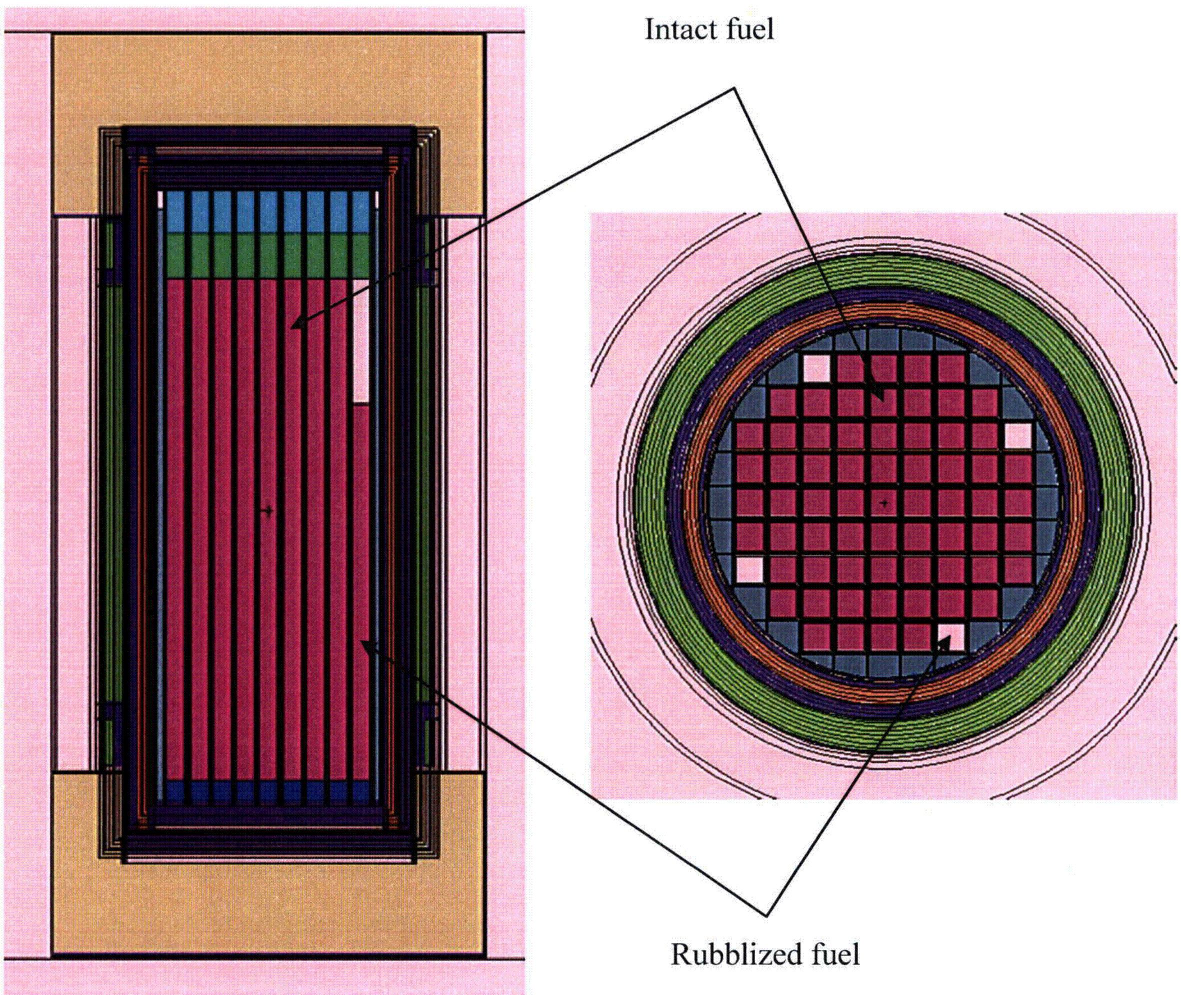


Figure A.5-6b
NCT Model Geometry with 4 Damaged Fuel Assemblies

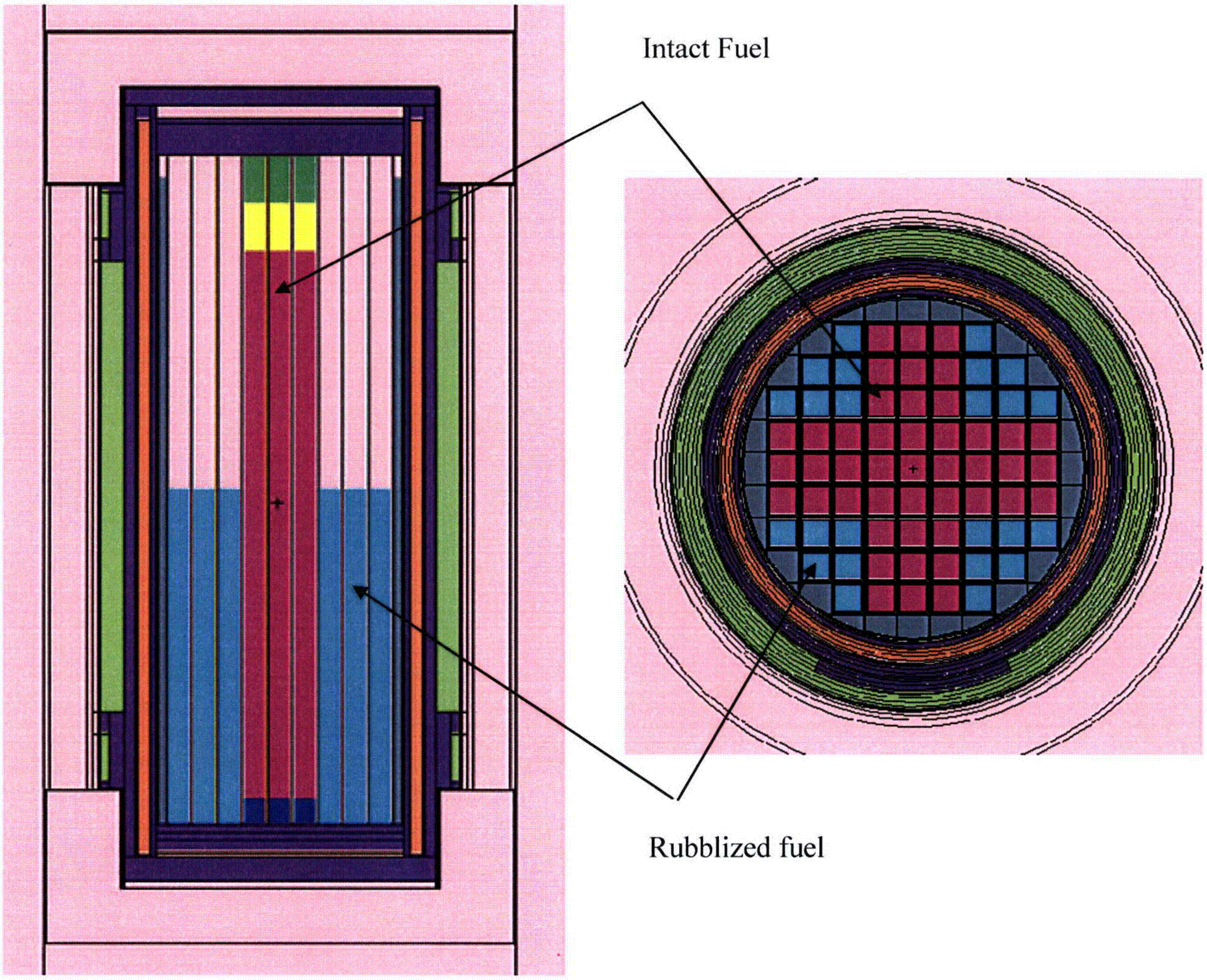


Figure A.5-6c
HAC Model Geometry with 24 Damaged Fuel Assemblies (50% Rubble)

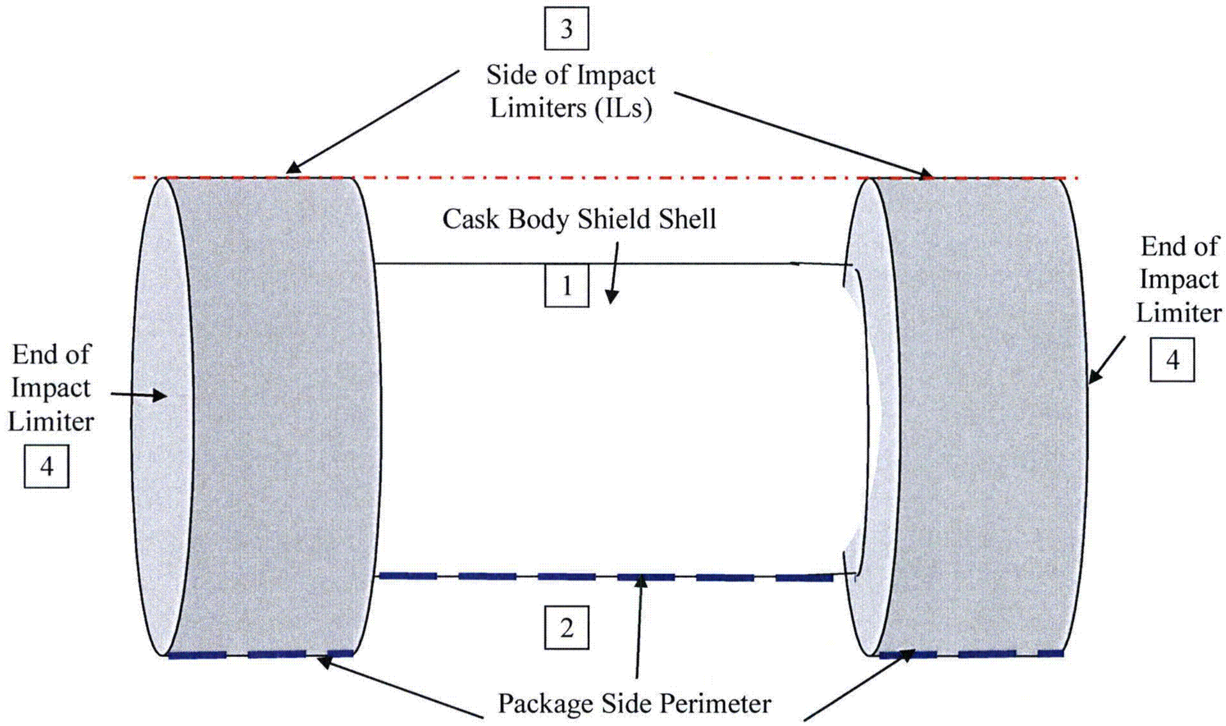


Figure A.5-16
Dose Rate Location Terminology