71-9255

TRANSNUCLEAR WEST

January 30, 2001
NUH05-01-725
RMG-01-006

Mr. Timothy Kobetz
Spent Fuel Project Office, NMSS
U. S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Subject: Application for Amendment of the NUHOMS® MP187 Multi-Purpose Cask, 10 CFR71 Certificate of Compliance No. 9255, Amendment No. 6

References:
1. Certificate of Compliance (CoC) No. 9255 Rev. No. 5, dated 1/17/01
2. Application for Approval of Advanced NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Safety Analysis Report, ANUH-01.0150, Revision 0; Grenier to Baggett letter DCS-TNW0009-15, dated September 29, 2000

Dear Mr. Kobetz:

Transnuclear West Inc. (TN West) herewith submits its application to amend Reference 1 with Revision 11 of its Safety Analysis Report (SAR) for the NUHOMS® MP187 Multi-Purpose Cask. This application adds a fourth Dry Shielded Canister (DSC), designated the NUHOMS® 24PT1-DSC, and its associated payload (WE 14 X 14 spent fuel) to the authorized contents of the MP187 Multi-Purpose Cask. This DSC and payload are the subject of a 10 CFR72 application for spent fuel storage under a general license, submitted by TN West in Reference 2.

This submittal is organized in the following format to facilitate your staff's review:

Attachment A: Description, Justification and Evaluation of Amendment,
Attachment B: Suggested Changes to CoC 71-9255 Revision No. 5,
Attachment C: Changed MP187 SAR Pages, Document NUH-05-151, Revision 11,
Attachment D: Instructions for Update of MP187 SAR from Revision 10 to Revision 11, and
Attachment E: Affidavit pursuant to the requirements of 10 CFR 2.790.

Appendix A of the MP187 SAR, included in Attachment C, provides a complete evaluation of the NUHOMS® 24PT1-DSC and is prepared in a format consistent with Regulatory Guide 7.9.

Transnuclear West Inc.
39300 Civic Center Drive, Suite 280, Fremont, CA 94538
Phone: 510-795-9800 • Fax: 510-744-6002

NMSS01 Prop

Where analyses are bounded by the existing SAR, those sections of the SAR are referenced. This appendix provides the information required for your staff to review this application in accordance with NUREG-1617, Standard Review Plan for Transportation Packages for Spent Nuclear Fuel.

This submittal includes proprietary information which may not be used for any purpose other than to support your staff's review of the application. In accordance with 10 CFR 2.790, TN West is providing an affidavit (Attachment E) specifically requesting that you withhold this proprietary information from public disclosure.

Should you or your staff require additional information to support review of this application, please do not hesitate to contact Mr. U. B. Chopra (510-744-6053) or me (510-744-6020).

Sincerely,

Robert M. Grenier
President and Chief Operating Officer

Docket 71-9255

Enclosures: Ten (10) Proprietary copies of MP187 SAR, Revision 11 – Changed Pages
Three (3) Non-Proprietary copies of MP187 SAR, Revision 11 – Changed Pages

<u>ATTACHMENT A</u>

DESCRIPTION, JUSTIFICATION, AND EVALUATION OF AMENDMENT

ATTACHMENT A

DESCRIPTION, JUSTIFICATION AND EVALUATION OF AMENDMENT

1.0 INTRODUCTION

The purpose of this amendment application is to add a fourth Dry Shielded Canister (DSC), the NUHOMS® 24PT1-DSC, to the authorized contents of the NUHOMS®-MP187 Multi-Purpose Cask.

This section of the application provides (1) a brief description of the changes, (2) justification for the change, and (3) a safety evaluation for this change.

2.0 BRIEF DESCRIPTION OF THE CHANGE

2.1 Significant Changes to NUHOMS® Certificate of Compliance (CoC) 71-9255, Revision 5

The changes listed below are relative to Revision 5 of the CoC dated January 17, 2001:

- Revise Section 5.a.(2), Dry Shielded Canisters (DSCs), to add a description of the fourth DSC designated as 24PT1-DSC, as an allowable variation in DSC configurations under this CoC.
- Revise Section 5.a.(3), Drawings, to add Drawing NUH-05-4010, Revision 0 describing the 24PT1-DSC.
- Revise Section 5.b(1)(a) to include the 24PT1-DSC.
- Revise Section 5.b(1)(b) to include the 24PT1-DSC failed fuel can for storage of damaged fuel.
- Revise Section 5.b.(1)(c) to add the WE 14X14 fuel payload in conjunction with the 24PT1-DSC.
- Revise Section 5.b(1)(d) and (e) to note that the 24PT1-DSC may be used for transportation of WE 14X14 fuel assemblies with or without control components in the 24PT1-DSC.
- Revise Section 5.b(1)(f) to reflect a maximum total heat load for the 24PT1-DSC of 14kW. Also add a reference and include a Table 2 for specification of maximum burnup and minimum cooling times for WE 14X14 assemblies stored in the 24PT1-DSC.
- Revise Section 5.b(1)(g) to reflect a maximum individual assembly heat load of 0.583 kW for the 24PT1-DSC.
- Revise Section 5.b(1)(h) to specify the cooling time required for WE 14X14 fuel control components.

- Revise Section 5.b(2)(a) to reflect that two empty slots are allowable for the 24PT1-DSC if they are located on symmetrically opposite locations with respect to the 0°-180° and 90°-270° DSC axes.
- Clarify Section 6 to indicate that any WE 14X14 fuel assembly meeting the requirements of Section 5.b.(1) may be stored in any location in a 24PT1-DSC.

2.2 Changes to NUHOMS®-MP187 SAR Revision 10

Attachment C to this submittal contains revised pages that incorporate the NUHOMS® 24PT1-DSC into the MP187 SAR.

Attachment C includes a new Appendix A, "Evaluation Of Addition of NUHOMS® 24PT1-DSC to NUHOMS®-MP187 Multi-Purpose Cask Payload." Appendix A has been prepared in a format consistent with the current MP187 SAR. It provides a complete evaluation of the new DSC and its payload. It also documents the changes, where applicable, to the existing safety analyses provided in the MP187 SAR.

Revised pages to the Chapter 1 to reference the new Appendix A for the 24PT1-DSC as well as changes to the front sections of the SAR (cover page, table of contents, etc.) are also provided consistent with the Revision 11 SAR changes.

3.0 JUSTIFICATION OF CHANGE

The NUHOMS® 24PT1-DSC design is very similar to the FO-DSC configuration and utilizes failed fuel cans which are very similar to those used in the FF-DSC. This change is required to support San Onofre Nuclear Generating Station Unit 1 decommissioning efforts.

4.0 EVALUATION OF CHANGE

TN West has evaluated the NUHOMS® 24PT1-DSC for structural, thermal, shielding and criticality adequacy and has concluded that the addition of the new DSC payload to the MP187 Multi-Purpose Cask approved payloads has no significant effect on safety. This evaluation is documented in Appendix A to the MP187 SAR provided in Attachment C.

ATTACHMENT B

SUGGESTED CHANGES TO CERTIFICATE OF COMPLIANCE 71-9255 REVISION NO. 5

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9255	6	71-9255	USA/9255/B(U)F-85	1	OF	9

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

Transnuclear West Inc.
39300 Civic Center Drive
Suite 280
Fremont, California 94538-2324

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Transnuclear West Inc., consolidated application dated *January 30, 2001.*

4. CONDITIONS

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

5.

a. PACKAGING:

(1) Model No.: NUHOMS®-MP187 Multi-Purpose Cask

(2) Description:

The NUHOMS®-MP187 Multi-Purpose Cask (package) consists of an outer cask, into which one of *the four* different dry shielded canisters (DSC) is placed. During shipment, energy-absorbing impact limiters are utilized for additional package protection.

Cask

The purpose of the cask is to provide containment and shielding of the radioactive materials contained within the DSC during shipment. The cask is constructed of stainless steel and lead with a neutron shield of cementitious material. The inside cavity of the cask is a nominal 68 inches in diameter and 187 inches long. The bottom access closure is approximately 5 inches thick and 17 inches in diameter, secured by 12 1-inch diameter bolts. The top closure is approximately 6.5 inches thick and is secured by 36 2-inch diameter bolts. Both closures are sealed by redundant O-rings.

Containment is provided by a stainless steel closure lid bolted to the stainless steel cask. The containment system of the NUHOMS®-MP187 transportation cask consists of (a) the inner shell, (b) the bottom end closure plate, (c) the top closure plate, (d) the top closure inner O-ring seal, (e) the ram closure plate, (f) the ram closure inner O-ring seal, (g) the vent port screw, (h) the vent port O-ring seal, (i) the drain port screw, and (j) the drain port O-ring seal. No credit is given to the DSC as a containment boundary.

Shielding is provided by 4 inches of stainless steel, 4 inches of lead, and approximately 4.3 inches of neutron shielding. The overall length of the cask is approximately 200 inches; the outer diameter is approximately 93 inches. The maximum gross weight of the package, with impact limiters, is approximately 282,000 lbs. The total length of the package with the impact limiters attached is approximately 308 inches. Four removable trunnions (two upper and two lower) are provided for handling and lifting.

Dry Shielded Canisters (DSCs)

The purpose of the DSC, which is placed within the transport cask, is to permit the transfer of spent fuel assemblies, into or out of a storage module, a dry transfer facility, or a pool as a unit. The DSC also provides additional axial biological shielding during handling and transport. The DSC consists of a stainless steel shell and a basket assembly. The approximately 5/8-inch thick shell has an outside diameter of about 67 inches and an external length of about 186 inches. The DSC basket assembly provides criticality control and contains a storage position for each fuel assembly. The basket is composed of circular spacer discs machined from thick carbon steel plates. Axial support for the DSC basket is provided by four high strength steel support rod assemblies. Carbon steel components of each DSC basket assembly are electrolytically coated with a thin layer of nickel to inhibit corrosion.

On the bottom of each DSC is a grapple ring, which is used to transfer a DSC horizontally from the cask into and out of dry storage modules. Because of the nature of the fuel that is to be transported, three different types of DSCs are designed for the package. Variations in the DSC configurations are summarized below:

- Fuel-Only Dry Shielded Canister (FO-DSC)

  The FO-DSC has a cavity length of approximately 167 inches and has solid carbon steel shield plugs at each end. The FO-DSC is designed to contain up to 24 intact Babcock and Wilcox (B&W) pressurized water reactor (PWR) spent fuel assemblies. The FO-DSC basket assembly consists of 24 guide sleeve assemblies with integral borated neutron absorbing plates, 26 spacer discs, and 4 support rod assemblies.

- Fuel/Control Components Dry Shielded Canister (FC-DSC)

  The FC-DSC has an internal cavity length of approximately 173 inches to accommodate fuel with the B&W control components installed. To obtain the increased cavity length, the shield plugs are fabricated from a composite of lead and steel. The FC basket is similar to the FO-DSC except that the support rod assemblies and guide sleeves are approximately 6-inches longer. The FC-DSC is also designed to contain up to 24 intact B&W PWR spent fuel assemblies with control components.

- Failed Fuel Dry Shielded Canister (FF-DSC)

  The FF-DSC has an internal cavity length of approximately 173 inches to accommodate 13 damaged B&W PWR spent fuel assemblies. Because the cladding has been locally degraded, individual (screened) fuel cans are provided to confine any gross loose material, maintain the geometry for criticality control, and facilitate loading and unloading operations. The FF-DSC is similar to FC-DSC in most respects with the exception of the basket assembly.

- *Fuel/Control Components Dry Shielded Canister (24PT1-DSC)*

  The 24PT1-DSC has a cavity length of approximately 167 inches with a solid carbon steel shield plugs at each end. The 24PT1-DSC is designed to contain up to 24 WE 14x14 pressurized water reactor (PWR) spent fuel assemblies, including control components of which up to four may be damaged with the remainder intact. The 24PT1-DSC basket assembly consists of 24 guides sleeve assemblies with *integral borated neutron absorbing plates, 26 spacer discs, and 4 support rod assemblies. Screened individual fuel cans are provided for storage of damaged fuel within the guidesleeve assemblies. These failed fuel cans are similar in configuration to the FF-DSC failed fuel cans.*

Impact Limiters

The impact limiter shells are fabricated from stainless steel. Within that shell are closed-cell polyurethane foam and aluminum honeycomb material. The impact limiter is attached to the cask by carbon steel bolts. Each impact limiter is bolted to the cask body through the neutron shield top and bottom support rings. The weight of each impact limiter is approximately 15,800 lbs.

(3) Drawings

The package shall be constructed and assembled in accordance with the following Transnuclear West Drawing Numbers:

NUH-05-4000NP, Revision 7,
Sheets 1 through 2
MP187 Multi-Purpose Cask
General Arrangement

NUH-05-4001, Revision 13,
Sheets 1 through 6
MP187 Multi-Purpose Cask
Main Assembly

NUH-05-4002, Revision 4,
Sheets 1 and 2
MP187 Multi-Purpose Cask
Impact Limiters

NH-05-4003, Revision 8,
Sheets 1 and 2
NUHOMS®-MP187 Multi-Purpose Cask
On-Site Transfer Arrangement

NUH-05-4004, Revision 13,
Sheets 1 through 4
NUHOMS® FO-DSC & FC-DSC PWR Fuel
Main Assembly

NUH-05-4005, Revision 11,
Sheets 1 through 4
NUHOMS® FF-DSC PWR Fuel
Main Assembly

NUH-05-4006NP, Revision 6,
Sheets 1 and 2
NUHOMS®-MP187 Multi-Purpose
Transportation Skid/Personnel Barrier

NUH-05-4010, Revision 0
NUHOMS®-24PT1-DSC
Main Assembly

5.b Contents of Packaging:

(1) Type and Form of Material:

(a) Intact fuel assemblies - Assemblies containing fuel rods with no known or suspected cladding defects greater than hairline cracks or pinhole leaks are authorized when contained in the FO-DSC or the FC-DSC *or 24PT1-DSC.*

(b) Damaged fuel assemblies - Assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks or with cracked, bulging, or discolored cladding are authorized when contained in the FF-DSC *or in a Failed Fuel Can in the 24PT1-DSC.* Spent fuel, with plutonium in excess of 20 curies per package, in the form of debris, particles, loose pellets, and fragmented rods or assemblies are not authorized. Damaged fuel assemblies may be shipped with or without control components.

(c) The fuel authorized for shipment in the NUHOMS®-MP187 FO, FC or FF-DSCs is B&W 15x15 uranium oxide PWR fuel assemblies with a maximum initial pellet enrichment of 3.43% by weight of ^{235}U, and a total uranium content not to exceed 466 Kg per assembly.

The fuel authorized for shipment in the NUHOMS®-MP187 24PT1-DSC is Westinghouse (WE) 14x14 uranium oxide or mixed oxide PWR fuel assemblies as described in Table 2.

(d) Intact *B&W 15x15* fuel assemblies without control components shall be shipped only in the FO-DSC, intact *B&W fuel* assemblies with control components shall be shipped only in the FC-DSC.

(e) Intact WE 14x14 fuel assemblies with or without control components shall be shipped only in the 24PT1-DSC.

(f) The maximum burnup and minimum cooling times for the individual *B&W 15x15* fuel assemblies shall meet the requirements of Table 1. In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of 5.b(1)(g) and (h). The maximum total allowable cask heat load for the FO, FC and FF DSCs is 13.5 kW.

The maximum burnup and minimum cooling times for the individual WE 14x14 fuel assemblies shall meet the requirements of Table 2. In addition, the fuel shall have been decayed for a time sufficient to meet the thermal criteria of 5.b(1)(g) and (h). The maximum total allowable cask heat load for the 24PT1-DSC is 14 kW.

(g) The maximum assembly decay heat for B&W 15x15 fuel stored in the FO, FC or FF-DSC, including control components when present, of an individual assembly is 0.764 kW, referred to as Type I, or 0.563 kW, referred to as Type II.

The maximum assembly decay heat for WE 14x14 fuel stored in the 24PT1-DSC, including control components when present, of an individual assembly is 0.583 kW.

(h) Control components for B&W 15x15 fuel assemblies stored in the FO, FC and FF-DSCs shall be cooled for at least 8 years.

Control components for WE 14x14 fuel assemblies stored in the 24PT1-DSC shall be cooled for at least 10 years.

(2) Maximum quantity of material per package:

(a) For material described in 5.b(1) *to be stored in the FO, FC or FF-DSCs:* 24 PWR intact fuel assemblies or 13 damaged fuel assemblies, with no more than 15 damaged fuel rods per assembly. Where a DSC is to be loaded with fewer fuel assemblies than the DSC capacity, dummy fuel assemblies with the same nominal weight as a standard fuel assembly shall be installed in the unoccupied spaces.

For material described in 5.b(1) to be stored in the 24PT1-DSC: 22 to 24 PWR fuel assemblies of which up to 4 may be damaged fuel assemblies. Damaged fuel assemblies with no more than 14 damaged fuel rods per assembly may be stored in the four outer corner fuel assembly locations *along the 45°, 135°, 225°, 315° azimuth of the DSC. A DSC may include two empty slots if they are located on symmetrically opposite locations with respect to the 0°-180° and 90°-270° DSC axes. Any additional empty fuel slots are to be loaded with dummy fuel assemblies with the same nominal weight as a standard fuel assembly. Fuel spacers are to be located at the bottom and the top of each fuel assembly to center the fuel assemblies within the DSC. Failed fuel cans require only bottom spacers since a top spacer is integral to each fuel can.*

(b) For material described in 5.b(1): the approximate maximum payload (including control components when present) is 81,100 lbs.

Table 1 – FO, FC and FF-DSC Fuel Assembly Burnup vs. Cooling Time

Maximum Burn-up (MWD/MTIHM)*	Minimum Enrichment in the Active Fuel Region (w/o ^{235}U)	Minimum Required Type I Cooling Time (years)	Minimum Required Type II Cooling Time (years)	Maximum Burn-up (MWD/MTIHM)*	Minimum Enrichment in the Active Fuel Region (w/o ^{235}U)	Minimum Required Type I Cooling Time (years)	Minimum Required Type II Cooling Time (years)
<23,200	n/a	5	5	33,000	2.90	7	10
23,200	2.38	5	5	34,000	2.95	7	11
24,000	2.43	5	6	35,000	2.67	7	14
25,000	2.49	5	6	35,000	2.99	7	11
26,000	2.55	5	7	36,000	3.03	8	13
27,000	2.61	5	7	37,000	3.00	8	14
28,000	2.66	5	8	37,000	3.07	8	14
29,000	2.00	6	10	38,000	3.11	9	15
29,000	2.71	5	8	39,000	3.15	9	16
30,000	2.76	5	8	40,000	3.19	9	17
31,000	2.81	6	9				
32,000	2.86	6	10	* Megawatt Days per Metric Ton of Initial Heavy Metal			

Table 2 - 24PT1-DSC Fuel Assembly Burnup vs. Cooling Time

Fuel Type	*Maximum Enrichment (Weight %)*	*Minimum Enrichment (Weight %)*	*Maximum Burnup (Mwd/ Mtu)*	*Minimum Cooling Time[2] / Max Heat Load Per Cask / Max Assembly Heat Load (Incl. Control Components[2])*
WE 14x14 Stainless Steel Clad (SC) (May include Integral Fuel Burnable Absorber, boron coated fuel pellets)	*4.0 ^{235}U*	*3.80 ^{235}U*	*45,000*	*38[1] years/14 kW/ 0.583 kW*
		3.40 ^{235}U	*40,000*	
		3.16 ^{235}U	*35,000*	
WE 14x14 MOX	*0.71 ^{235}U 3.31 fissile Pu*	*2.78 Pu (64 rods) 3.05 Pu (92 rods) 3.25 Pu (24 rods)*	*25,000*	*20 years/14 kW/ 0.583 kW*

Notes:

1 Criteria for a shortened cooling time are provided in SAR Table A1.2-1.

2 Control component cooling time must be a minimum of 10 years.

5.c. Transport Index for Criticality Control

Minimum transport index to be shown on the label for nuclear criticality control: "0"

6. Type I fuel assemblies shall be loaded only into the four innermost cells of a DSC, while Type II assemblies may be loaded into any cell when using the FO-DSC or the FC-DSC. The FF-DSC *and 24PT1-DSC do not require restrictions with respect to placement of fuel assemblies.*

7. Fuel assemblies with missing fuel rods shall not be shipped unless dummy fuel pins that displace an equal amount of water have been installed in the fuel assembly.

8. For operating controls and procedures, in addition to the requirements of Subpart G of 10 CFR Part 71:

   a. Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed using the specifications contained within the application. At a minimum, those procedures shall include the following provisions:

      (1) a loading plan which has been independently verified and approved by a qualified individual other than the developer(s) which shall include:

         (a) hold points to verify that all fuel movements are performed under strict verbatim compliance with the fuel movement schedule;

         (b) videotaping and independent verification by ID number of each fuel assembly loaded; and

(c) a final independent verification of the fuel placement.

(2) procedures requiring that before shipment the licensee shall:

(a) perform a measured radiation survey to assure compliance with 49 CFR 173.441 and 10 CFR 71.47 and assure that the neutron measurement instruments are calibrated for the energy spectrum of neutrons being emitted from the package;

(b) verify that the surface removable contamination levels meet the requirements of 49 CFR 173.443 and 10 CFR 71.87; and

(c) leak test containment vessel seals to verify a leak rate of less than 1 x 10-7 standard cubic centimeters per second of helium (std-cc/sec). The leak test shall have a test sensitivity of at least 5 x 10-8 std-cc/sec and shall be conducted:

1) before first use of each package,

2) within the 12-month period prior to each shipment, and

3) after seal replacement.

(3) procedures that require that the package metallic seals be replaced after each use.

b. All fabrication acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for fabrication, acceptance testing, and maintenance shall be developed using the specifications contained within the application, and shall include the following provisions:

(1) With the exception of the weld between the inner shell and top forging, all longitudinal and circumferential inner shell welds, which form the containment boundary of the cask, shall be radiographically inspected (RT) with acceptance standards in accordance with the ASME Code, Section III, Division 1, NB-5320. The weld between the inner shell and top forging shall be verified by RT or ultrasonically inspected (UT). The substitution of UT for the examination of the completed weld may be made provided the examination is performed using detailed written procedures, proven by actual demonstration to the satisfaction of the inspector as capable of detecting and locating defects described in ASME Code, Section III, Division 1 Subsection NB-5000.

(2) The DSC outer top cover plate weld shall be verified by either volumetric or multilayer PT examination. If PT is used, at a minimum, it must include the root, each successive 1/4 inch weld thickness, and the final layer. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME B&PVC Section III, NB-5350. The inspection process, including findings (indications) shall be made a permanent part of the licensee's records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

(3) Before joining the structural shell to the inner shell, the upper lifting trunnions shall be load tested to 150% of their maximum working load or 188,000 lbs. minimum per trunnion, in accordance with the requirements of ANSI N14.6-1986.

(4) The cask containment boundary shall be pressure tested to 150% of the design pressure per 10 CFR 71.85(b). The minimum test pressure shall be 75 psig.

(5) The fabrication verification leak test for the inner shell shall be performed after initial fabrication, but before the lead pour, to verify that the leak rate from the cylindrical containment shell is less than 1 x 10 -7 std-cc/sec. A second fabrication verification leak test shall be performed on the finished cask to demonstrate a leak rate of less than 1 x 10 -7 std-cc/sec for the package. The results of both tests shall have at least, a sensitivity of 5 x 10 -8 std-cc/sec.

(6) The poured-lead shielding integrity of the MP187 cask body shall be confirmed via gamma scanning prior to installation of the neutron shield. The scan shall utilize, at a maximum, a 6x6-inch test grid. The minimum lead thickness in the main cask body, away from the trunnions and the top and bottom forgings, shall be 3.90 inches

(7) The neutron shield shall have a minimum thickness of 4.31 inches. Its integrity shall be confirmed through a strict combination of fabrication process control and verification by measurement. This may be done either at first use or with a check source using, at a maximum, a 6x6-inch test grid.

(8) The complete cask shall be subjected to a thermal heat rejection test to demonstrate satisfactory operation of the as-built shells, top lid, and shielding materials. This test may be performed without the ram closure installed. Acceptance criteria shall be calculated for the change in temperature across the cask wall based on the applied heat load and existing environmental conditions using the same analytical methods used to predict the cask performance for the normal and accident conditions.

(9) Foam shall be installed within the cask impact limiters and tested to ensure conformance with the required foam material properties.

(10) The neutron absorber plate's minimum acceptable areal boron content loading is 0.025 g/cm 2 Boron 10. The minimum Boron 10 content per unit area and the uniformity of dispersion within the sandwiched material shall be verified by testing each sheet with a sufficient sensitivity (at least to the 95/95 confidence level) to assure compliance with the drawings.

(11) The impact limiters shall be visually inspected within 1 year of use for water absorption or degradation. Each impact limiter shall also be weighed at the time of inspection. If the weight has increased more than 3%, the impact limiter shall be repaired or replaced.

9. This package is approved for exclusive use rail, truck or marine transport.

10. The package authorized by this certificate is hereby approved for use under the general license provisions of 10CFR71.12.

11. Expiration Date: September 10, 2003.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9255	6	71-9255	USA/9255/B(U)F-85	9	OF	9

REFERENCES

Transnuclear West Inc., consolidated Safety Analysis Report for the NUHOMS®-MP187 Multi-Purpose Cask, dated *January, 2001.*

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date:

ATTACHMENT C

CHANGED MP187 SAFETY ANALYSIS REPORT PAGES,

DOCUMENT NUH-05-151, REVISION 11

ATTACHMENT D

INSTRUCTIONS FOR UPDATE OF MP187 SAR FROM REVISION 10 TO REVISION 11

These instructions are to be used for incorporating changes to the MP187 SAR, Revision 10 to create the Revision 11 SAR, Proprietary Version

• Cover Sheet	Replace Revision 10 page with Revision 11 page
• Pages ii and iii	Replace Revision 10 pages with Revision 11 page
• Page ix	Replace Revision 10 page with Revision 11 page
• Page xxvii	Replace Revision 10 page with Revision 11 page
• Page xl and xli	Replace Revision 10 pages with Revision 11 page xli
• Page 1-1	Replace Revision 10 page with Revision 11 page

- Add new Volume, Proprietary Version of Appendix A to the SAR

These instructions are to be used for incorporating changes to the Non Proprietary Version of MP187 SAR, Revision 10, to create the Revision 11 SAR, Non-Proprietary Version.

• Cover Sheet	Replace Revision 10 page with Revision 11 page
• Pages ii and iii	Replace Revision 10 pages with Revision 11 page
• Page ix	Replace Revision 10 page with Revision 11 page
• Page xxv	Replace Revision 10 page with Revision 11 page
• Page xxxv and xxxvi	Replace Revision 10 page with Revision 11 page
• Page xli	Replace Revision 10 pages with Revision 11 page xli
• Page 1-1	Replace Revision 10 page with Revision 11 page

- Add new Volume, Non Proprietary Version of Appendix A to the SAR

ATTACHMENT E

AFFIDAVIT PURSUANT TO THE REQUIREMENTS OF 10 CFR 2.790

ATTACHMENT C

CHANGED MP187 SAFETY ANALYSIS REPORT PAGES,

DOCUMENT NUH-05-151, REVISION 11

ATTACHMENT D

INSTRUCTIONS FOR UPDATE OF MP187 SAR FROM REVISION 10 TO REVISION 11

These instructions are to be used for incorporating changes to the MP187 SAR, Revision 10 to create the Revision 11 SAR, Proprietary Version

- Cover Sheet — Replace Revision 10 page with Revision 11 page
- Pages ii and iii — Replace Revision 10 pages with Revision 11 page
- Page ix — Replace Revision 10 page with Revision 11 page
- Page xxvii — Replace Revision 10 page with Revision 11 page
- Page xl and xli — Replace Revision 10 pages with Revision 11 page xli
- Page 1-1 — Replace Revision 10 page with Revision 11 page

- Add new Volume, Proprietary Version of Appendix A to the SAR

These instructions are to be used for incorporating changes to the Non Proprietary Version of MP187 SAR, Revision 10, to create the Revision 11 SAR, Non-Proprietary Version.

- Cover Sheet — Replace Revision 10 page with Revision 11 page
- Pages ii and iii — Replace Revision 10 pages with Revision 11 page
- Page ix — Replace Revision 10 page with Revision 11 page
- Page xxv — Replace Revision 10 page with Revision 11 page
- Page xxxv and xxxvi — Replace Revision 10 page with Revision 11 page
- Page xli — Replace Revision 10 pages with Revision 11 page xli
- Page 1-1 — Replace Revision 10 page with Revision 11 page

- Add new Volume, Non Proprietary Version of Appendix A to the SAR

NUH-05-151
Revision 11
January 2001
File: NUH005.0151

SAFETY ANALYSIS REPORT

FOR THE

NUHOMS®-MP187

MULTI-PURPOSE CASK

Docket 71-9255

NON-PROPRIETARY
FOR INFORMATION ONLY

Prepared by:

Transnuclear West Inc.
Fremont, California

REVISION HISTORY

Rev. 0	September 1993	Initial Issue
Rev. 1	February 1995	Response to NRC Comments
Rev. 2	February 1996	Response to NRC Comments
Rev. 3	July 1997	Response to NRC Comments
Rev. 4	November 1997	Response to NRC Comments
Rev. 5	January 1998	Minor Editorial Corrections
Rev. 6	May 1998	Response to NRC Comments
Rev. 7	August 1998	Clarifications and Editorial
Rev. 8	August 1998	Change Company Name
Rev. 9	September 1998	Revise Cask Inspection Requirements
Rev. 10	November 2000	Incorporate Amendments
Rev. 11	January 2001	Incorporate 24PT1-DSC Payload

LIST OF EFFECTIVE PAGES

Page	Revision
i-iii	11
iv-viii	10
ix	11
x-xxiv	10
xxv	11
xxvi-xxxiv	10
xxxv-xxxvi	11
xli	11
1-1	11
1-2 - 1-3	10
1.1-1 - 1.1-5	10
1.2-1 - 1.2-15	10
1.3-i	10
1.3.1-1	10
1.3.2-1	10
2-1 - 2-2	10
2.1-1 - 2.1-34	10
2.2-1 - 2.2-7	10
2.3-1 - 2.3-19	10
2.4-1 - 2.4-8	10
2.5-1 - 2.5-18	10
2.6-1 - 2.6-227	10
2.7-1 - 2.7-475	10
2.8-1	10
2.9-1	10
2.10-i	10
2.10.1-1 - 2.10.1-7	10
2.10.2-1 - 2.10.2-37	10
2.10.3-1 - 2.10.3-43	10
2.10.4-1 - 2.10.4-25	10
2.10.5-1	10
2.10.7-1 - 2.10.7-14	10
2.10.8-1	10
2.10.9-1	10
2.10.10-1 - 2.10.10-74	10
2.10.11-1 - 2.10.11-41	10
2.10.12-1 - 2.10.12-25	10
3.1-1 - 3.1-5	10
3.2-1 - 3.2-20	10
3.3-1	10
3.4-1 - 3.4-15	10
3.5-1 - 3.5-17	10
3.6-i	10
3.6.1-1 - 3.6.1-3	10
3.6.2-1 - 3.6.2-13	10
3.6.3-1 - 3.6.3-15	10
3.6.4-1 - 3.6.4-14	10
3.6.5-1 - 3.6.5-17	10
3.6.6-1 - 3.6.6-11	10
4.1-1 - 4.1-4	10
4.2-1	10
4.3-1	10
4.4-i	10
4.4.1-1	10
4.4.2-1 - 4.4.2-2	10
4.4.3-1	10
4.4.4-1	10
5.1-1 - 5.1-5	10
5.2-1 - 5.2-2	10
5.3-1 - 5.3-14	10
5.4-1 - 5.4-38	10
5.5-i	10
5.5.1-1 - 5.5.1-2	10
5.5.2-1	10
5.5.3-1	10
5.5.4-1	10
5.5.5-1	10
5.5.6-1	10
5.5.7-1	10
5.5.8-1 - 5.5.8-21	10
6.1-1 - 6.1-6	10
6.2-1 - 6.2-2	10
6.3-1 - 6.3-11	10
6.4-1 - 6.4-16	10
6.5-1 - 6.5-9	10
6.6-i	10
6.6.1-1 - 6.6.1-2	10
6.6.2-1	10
7.1-1 - 7.1-19	10
7.2-1 - 7.2-7	10
7.3-1	10
7.4-i	10
7.4.1-1	10
7.4.2-1 - 7.4.2-6	10
8.1-1 - 8.1-18	10
8.2-1 - 8.2-4	10
8.3-i	10
8.3.1	10
8.3.2-1 - 8.3.2-3	10
8.3.3-1	10
Appendix A (195 pages)	11

Page

8.3.2 Lead Installation Procedure 8.3.2-1
8.3.3 NS-3 Specification 8.3.3-1

See Appendix A for a Table of Contents for Appendix A

Table	Title	Page
Table 6.4-5	MP187/FF-DSC KENO Results	6.4-10
Table 6.4-6	MP187/FF-DSC KENO Results (Cask Layer Removal)	6.4-11
Table 6.5-1	Benchmark Calculation Results	6.5-4

See Appendix A for a List of Tables for Appendix A

Figure	Title	Page
Figure 6.4-2	FF-DSC Broken Fuel Rod Models	6.4-13
Figure 6.4-3	FF-DSC Double-Ended Rod Break Models	6.4-14
Figure 6.4-4	NUHOMS®-MP187 Cask/FO-DSC Criticality Results	6.4-15
Figure 6.4-5	NUHOMS®-MP187 Cask/FF-DSC Criticality Results	6.4-16
Figure 6.5-1	Critical Benchmark Results	6.5-9
Figure 7.1-1	NUHOMS®-MP187 Cask Operating Procedures Flow Chart	7.1-15
Figure 7.1-2	NUHOMS®-MP187 Wet Loading Procedure	7.1-16
Figure 7.1-3	NUHOMS®-MP187 Dry Loading Procedure	7.1-18
Figure 7.1-4	NUHOMS®-MP187 Preparation for Transportation	7.1-19
Figure 7.4-1	Assembly Verification Leak Test Flow Chart	7.4.2-6
Figure 8.1-1	Thermocouple Locations for Thermal Acceptance Test	8.1-15
Figure 8.3-1	Thermocouple Locations for Lead Installation	8.3.2-2

See Appendix A for a List of Figures for Appendix A

SUMMARY OF CHANGES[1]

Section i	Updated for consistency with remainder of document
Section 1	Incorporated 24PT1-DSC payload reference to Appendix A. Clarified discussion of MP187 transfer cask usage.
Appendix A	Added Appendix A to provide analysis of the 24PT1-DSC payload.

[1] Revisions are denoted by vertical bars along the right margin. Corrected spell and punctuation have not been marked as revised text. Changes in pagination have also not been marked as revised text.

SUMMARY OF CHANGES[1]

Section i	Updated for consistency with remainder of document
Section 1	Incorporated 24PT1-DSC payload reference to Appendix A. Clarified discussion of MP187 transfer cask usage.
Appendix A	Added Appendix A to provide analysis of the 24PT1-DSC payload.

[1] Revisions are denoted by vertical bars along the right margin. Corrected spelling and punctuation have not been marked as revised text. Changes in pagination have also not been marked as revised text.

1. GENERAL INFORMATION

This section of the NUHOMS®-MP187 Multi-Purpose Cask 10CFR71 Safety Analysis Report (SAR) presents a general introduction to and a description of the NUHOMS®-MP187 Package. The NUHOMS®-MP187 Cask in this configuration is to be utilized for the off-site transportation of NUHOMS® Dry Shielded Canisters (DSCs) in accordance with 10CFR71 [1.1] and 49CFR173 [1.2]. The DSC is sometimes referred to generically as a type of Multi-Element Sealed Canister (MESC) or a Multi-Purpose Canister (MPC). The terminology used throughout this SAR is presented in Table 1.1-1. Illustrations of the NUHOMS®-MP187 Packaging are provided in Figure 1.1-1 through Figure 1.1-3.

The NUHOMS®-MP187 Cask is also used for on-site transfer of a DSC to and from NUHOMS® modular storage under a separate 10CFR72 license.

Chapters 1 through 8 of this SAR address the MP187 Package with the FO, FC or FF-DSC payload containing B&W 15x15 spent fuel assemblies. Appendix A to this SAR addresses the MP187 Package with the 24PT1-DSC payload containing Westinghouse 14x14 spent fuel assemblies.

ATTACHMENT

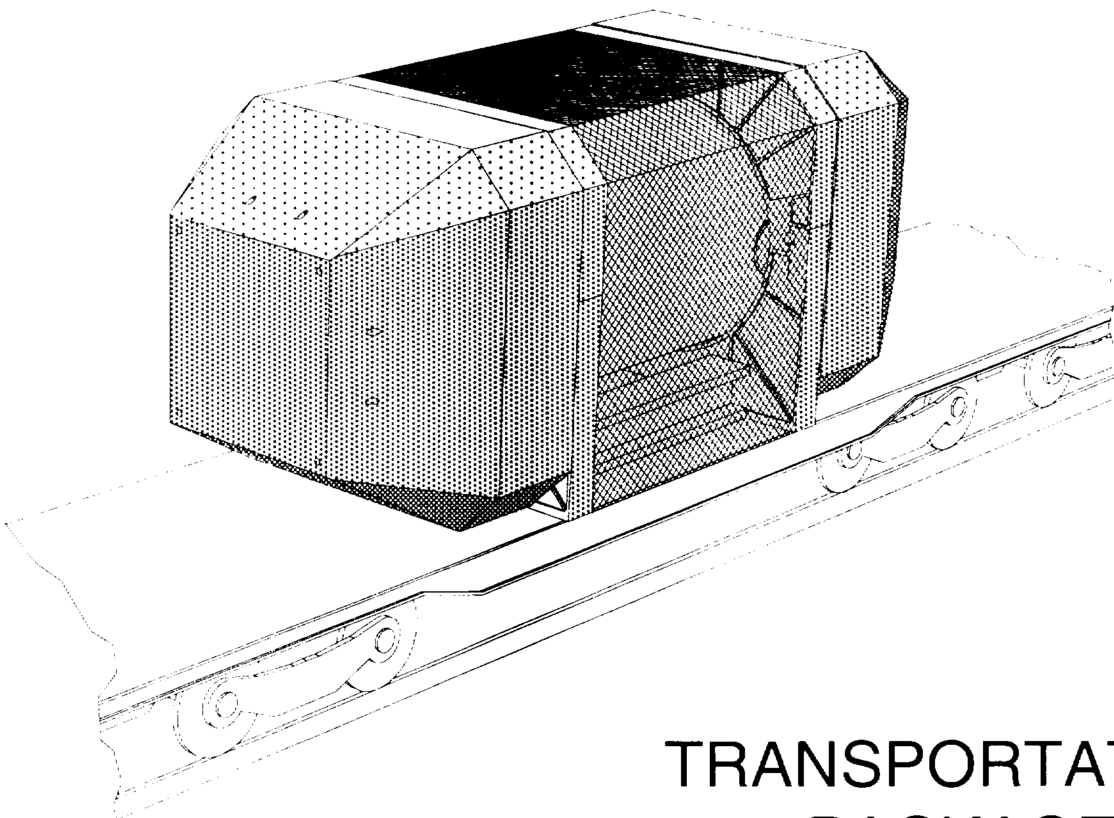
P116084
#1

BINDER/VOLUME #:

ATTACHMENT

TRANSNUCLEAR WEST

NUHOMS®-MP187 MULTI-PURPOSE CASK



TRANSPORTATION PACKAGE

SAFETY ANALYSIS REPORT

NON-PROPRIETARY
FOR INFORMATION ONLY

DOCUMENT NO. NUH-05-151

TABLE OF CONTENTS

Page

A1. GENERAL INFORMATION ... A1-1

A1.1 Introduction ... A1.1-1

A1.2 Package Description ... A1.2-1

A1.2.1 Packaging ... A1.2-1

A1.2.2 Operational Features ... A1.2-4

A1.2.3 Contents of Packaging ... A1.2-4

A1.3 Appendix ... A1.3-i

A1.3.1 References ... A1.3-i

A1.3.2 General Arrangement Drawings ... A1.3-i

A2. STRUCTURAL EVALUATION ... A2-1

A2.1 Structural Design ... A2.1-1

A2.1.1 Discussion ... A2.1-1

A2.1.2 Design Criteria ... A2.1-2

A2.2 Weights and Centers of Gravity ... A2.2-1

A2.3 Mechanical Properties of Materials ... A2.3-1

A2.4 General Standards for All Packages ... A2.4-1

A2.5 Lifting and Tie-Down Standards for All Packages ... A2.5-1

A2.6 Normal Conditions of Transport ... A2.6-1

A2.6.1 Heat ... A2.6-1

A2.6.2 Cold ... A2.6-1

A2.6.3 Pressure ... A2.6-1

A2.6.4 Vibration ... A2.6-1

A2.6.5 Water Spray ... A2.6-1

A2.6.6 Free Drop ... A2.6-1

A2.6.7 Corner Drop ... A2.6-2

A2.6.8 Compression ... A2.6-2

A2.6.9 Penetration ... A2.6-2

A2.6.10 Fabricaiton Stress Analysis ... A2.6-2

A2.6.11 A24PT1-DSC Normal Conditions of Transport ... A2.6-2

A2.6.12 Not Used ... A2.6-15

A2.6.13 Normal Conditions of Transport (NCT) Load Combination Evaluation ... A2.6-15

A2.7 Hypothetical Accident Conditions (HAC) ... A2.7-1

A2.7.6 24PT1-DSC Analysis – Hypothetical Accident Conditions ... A2.7-2

A2.7.7 Not Used ... A2.7-7

A2.7.8 HAC Load Combination Evaluation ... A2.7-7

TABLE OF CONTENTS
(Continued)

Page

A2.8 Special Form A2.8-1
A2.9 Fuel Rods A2.9-1
A2.10 Appendix A2.10-i
A2.10.1 References A2.10-i
A2.10.2 NUHOMS®-MP187 Cask Analyses A2.10-i
A2.10.3 24PT1-DSC Basket Analyses 2.10-i

A3. THERMAL EVALUATION A3-1
A3.1 Discussion A3.1-1
A3.2 Summary of Thermal Properties of Materials A3.2-1
A3.3 Technical Specifications of Components A3.3-1
A3.4 Thermal Evaluation for Normal Conditions of Transport A3.4-1
A3.4.1 Thermal Model A3.4-1
A3.5 Hypothetical Accident Thermal Evaluation A3.5-1
A3.6 Appendix A3.6-i
A3.6.1 References A3.6-i
A3.6.2 24PT1-DSC Pressure Analysis A3.6-i

A4. CONTAINMENT A4-1

A5. SHIELDING EVALUATION A5-1
A5.1 Discussion and Results A5.1-1
A5.2 Source Specification A5.2-1
A5.4.1 Gamma Source A5.2-3
A5.4.2 Neutron Source A5.2-19
A5.3 Model Specification A5.3-1
A5.3.1 Description of the Radial and Axial Shielding Configuration A5.3-1
A5.3.2 Shield Regional Densities A5.3-7
A5.4 Shielding Evaluation A5.4-1
A5.4.1 DORT-PC Input Parameters A5.4-1
A5.4.2 NUHOMS®-MP187 Package Dose Rate Results A5.4-7
A5.5 Appendix A5.5-1
A5.5.1 References A5.5-1
A5.5.2 SAS2H/ORIGEN-S Input Files A5.5-3
A5.5.3 DORT-PC and GIP Input Files A5.5-8

TABLE OF CONTENTS
(Continued)

Page

A6. CRITICALITY EVALUATION A6-1
A6.1 Discussion and Results A6.1-1
A6.1.1 NUHOMS®-MP187 Cask Design Features A6.1-1
A6.1.2 24PT1-DSC Design Features A6.1-1
A6.1.3 Fuel-Control Components A6.1-3
A6.1.4 Failed Fuel Can Design Features A6.1-3
A6.1.5 Criticality Analysis Summary and Results A6.1-4
A6.2 Package Fuel Loading A6.2-1
A6.3 Model Specification A6.3-1
A6.3.1 Description of Calculational Model A6.3-1
A6.3.2 NUHOMS®-MP187 Cask/24PT1-DSC Model A6.3-1
A6.3.3 NUHOMS®-MP187 Cask/24PT1-DSC Damaged Fuel Model A6.3-9
A6.3.4 Package Regional Densities A6.3-9
A6.4 Criticality Calculation A6.4-1
A6.4.1 Calculational Method A6.4-1
A6.4.2 Fuel Loading Optimization A6.4-1
A6.4.3 Criticality Results A6.4-4
A6.5 Critical Benchmark Experiments A6.5-1
A6.5.1 Benchmark Experiments and Applicability A6.5-1
A6.5.2 Results of the Benchmark Calculations A6.5-2
A6.6 Appendix A6.6-i
A6.6.1 References A6.6-1
A6.6.2 KENO Input Files [PRORRIETARY] A6.6-4
A7. OPERATING PROCEDURES A7-1
A7.1 Procedures for Loading the Package A7.1-1
A7.1.1 Preparation of the NUHOMS®-MP187 Cask for Use A7.1-1
A7.1.2 Wet Loading the NUHOMS®-MP187 Cask and DSC A7.1-1
A7.1.3 Transferring the DSC to an Onsite Storage Facility A7.1-3
A7.1.4 Placing the DSC in an HSM for Storage A7.1-3
A7.1.5 Loading the DSC into the Cask from an HSM A7.1-3
A7.1.6 Placing the DSC in Metal Cask Storage at an Onsite Facility A7.1-3
A7.1.7 Preparing the Cask for Transportation A7.1-3
A7.1.8 Placing the Cask on the Railcar A7.1-3
A7.1.9 Glossary A7.1-3
A8. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM A8-1

LIST OF TABLES

		Page
Table A1.1-1	Terminology and Notation Model: NUHOMS®-MP187	A1-2
Table A1.2-1	WE 14x14 Spent Fuel Assembly Description	A1.2-5
Table A1.2-2	Westinghouse Fuel Dimensions, Weights and Source Terms	A1.2-8
Table A2.1-1	Configuration of the FO-DSC to the 24PT1-DSC	A2.1-2
Table A2.2-1	Comparison of DSC Weights, cg and MOI	A2.2-1
Table A2.2-2	NUHOMS® System Mass Properties	A2.2-2
Table A2.3-1	NUHOMS® 24PT1-DSC Structural Component Materials	A2.3-1
Table A2.6-1	Spacer Disc Tributary Weight(2) Distribution [PROPRIETARY]	A2.6-4
Table A2.6-2	Weight Distribution of Westinghouse 14x14 Fuel Including Control Components[1]	A2.6-5
Table A2.6-3	Support Rod and Support Rod Spacer Sleeve Stresses Due to Preload	A2.6-10
Table A2.6-4	Fuel Spacer Vibration Stresses	A2.6-13
Table A2.6-5	Fuel Spacer 1-Foot Drop Stresses	A2.6-15
Table A2.7-1	Spacer Sleeve Stresses - 60g Bottom End Drop	A2.7-4
Table A3.6-1	Component Weight, Material, and Volume for a Single Fuel Assembly [PROPRIETARY]	A3.6-4
Table A3.6-2	Fuel Rod Data [PROPRIETARY]	A3.6-5
Table A3.6-3	Fission Gas Inventory per Assembly for WE 14x14 Fuel	A3.6-6
Table A3.6-4	Control Component Data [PROPRIETARY]	A3.6-7
Table A3.6-5	Design Basis Thermal Analysis Results for MP187 Package Pressure	A3.6-8
Table A5.1-1	Summary of Maximum Dose Rates (mrem/hr)	A5.1-4
Table A5.2-1	SAS2H/ORIGEN-S Input Cases for Shielding Evaluation	A5.2-3
Table A5.2-2	Summary of Fuel Assembly Material Weights Used in Source Term Calculations WE 14x14 SC Assembly Materials and Masses [PROPRIETARY]	A5.2-5
Table A5.2-3	Flux Fraction By Assembly Region	A5.2-6
Table A5.2-4	Control Component Regional Weights Thimble Plug Assemblies and Masses [PROPRIETARY]	A5.2-7
Table A5.2-5	Bounding Fuel In Core Region Gamma Source[1] (γ/s/assy)	A5.2-9
Table A5.2-6	NFAH Source Terms	A5.2-13
Table A5.2-7	Composite Gamma Source (γ/s/assy)	A5.2-17
Table A5.2-8	24PT1-DSC Design Basis Gamma Source Strength and Energy Spectrum Composite Gamma Source ($\gamma/s/cm^3$)	A5.2-18

LIST OF TABLES
(Continued)

		Page
Table A5.2-9	Design Basis Neutron Source Strength and Energy Spectrum[1] (neutrons/sec/assy)	A5.2-20
Table A5.3-1	Summary of NUHOMS®-MP187 Cask Shielding Material Densities [atoms/b·cm, (g/cc)]	A5.3-9
Table A5.4-1	Gamma-Ray Flux-to-Dose Rate Factors	A5.4-4
Table A5.4-2	Neutron Flux-to-Dose Rate Factors	A5.4-5
Table A5.4-3	Summary of Maximum Normal Conditions of Transport Dose Rates	A5.4-8
Table A5.4-4	Summary of Maximum Hypothetical Accident Dose Rates	A5.4-9
Table A6.1-1	Summary of Criticality Evaluation	A6.1-5
Table A6.2-1	Maximum Fuel Loading Parameters	A6.2-1
Table A6.2-2	Design Basis Fuel Parameters for Criticality Analysis	A6.2-2
Table A6.3-1	KENO Model Atom Densities	A6.3-10
Table A6.3-2	KENO Model Moderator Atom Densities	A6.3-11
Table A6.4-1	USL vs. Calculated k_{eff}	A6.4-5
Table A6.4-2	MP187/24PT1-DSC WE 14x14 SC Intact Fuel Results	A6.4-7
Table A6.4-3	MP187/24PT1-DSC-DSC WE 14x14 MOX Intact Fuel Results	A6.4-10
Table A6.4-4	MP187/24PT1-DSC WE 14x14 Damaged Fuel Results [PROPRIETARY] STAINLESS STEEL CLAD FUEL	A6.4-13
Table A6.5-1	Benchmark Results	A6.5-4
Table A6.5-2	USL-1 Results	A6.5-7
Table A6.5-3	Fuel Assembly Design Parameters Used in Criticality Benchmarks [PROPRIETARY]	A6.5-8
Table A6.5-4	Limiting Upper Subcritical Limit Based on Method 1 for the WE 14x14 SC Fuel Assemblies and the WE 14x14 MOX Fuel Assemblies [PROPRIETARY]	A6.5-9

LIST OF FIGURES

		Page
Figure A2.6-1	24PT1-DSC Basket Assembly Detail	A2.6-8
Figure A2.6-2	24PT1-DSC Spacer Disc Detail	A2.6-9
Figure A2.7-1	Support Rod Load Distribution - 65k Preload + Thermal + 60g BED [PROPRIETARY]	A2.7-5
Figure A5.3-1	NUHOMS®-MP187 Normal Operations DORT-PC 24PT1-DSC Top End Model [PROPRIETARY]	A5.3-3
Figure A5.3-2	NUHOMS®-MP187 Normal Operations DORT-PC 24PT1-DSC Bottom End Model [PROPRIETARY]	A5.3-4
Figure A5.3-3	NUHOMS®-MP187 Normal Operation Shielding Materials	A5.3-5
Figure A5.3-4	NUHOMS®-MP187 Hypothetical Accident Condition Shielding Materials	A5.3-6
Figure A5.4-1	NUHOMS®-MP187 24PT1-DSC Normal Operation Dose Rate Distribution	A5.4-10
Figure A5.4-2	NUHOMS®-MP187 Hypothetical Accident Dose Rate Distribution	A5.4-11
Figure A6.3-1	KENO Model and DSC Basket [PROPRIETARY]	A6.3-3
Figure A6.3-2	Exploded View of KENO Model	A6.3-5
Figure A6.3-3	Structure of KENO Model UNITS 33 and 34	A6.3-6
Figure A6.3-4	KENO Model UNITS 1-8, plus	A6.3-7
Figure A6.3-5	KENO Model of Design Basis Fuel Assemblies	A6.3-8
Figure A6.4-1	Single-Ended Rod Shear Example [PROPRIETARY]	A6.4-17
Figure A6.4-2	Double-Ended Rod Shear Example [PROPRIETARY]	A6.4-18

A1. GENERAL INFORMATION

Chapter 1 of the NUHOMS®-MP187 Multi-Purpose Cask 10CFR71 Safety Analysis Report (SAR) presents a general introduction to, and description of, the NUHOMS®-MP187 Package (MP187). Chapters 1 through 8 of this SAR address the MP187 Cask with FO, FC or FF DSC payloads for the storage and/or transportation of B&W 15x15 fuel assemblies. This Appendix A evaluates the 24PT1-DSC which is designed to store and transport Westinghouse (WE) 14x14 fuel, as an additional payload for the MP187 Cask.

The terminology specific to this Appendix is presented in Table A1.1-1.

NOTES:

1 - References to sections or chapters within this Appendix are identified with an "A" preceding the number (e.g., Section A1.2.1.2, Chapter A1 or Appendix A1.3.1). References to sections of the MP187 SAR outside of this Appendix (main body of SAR) simply refer to the applicable section (e.g., Section 1.2.12) or chapter (e.g., Chapter 2).

2 - Dimensions shown in figures in this Appendix are in units of inches unless otherwise noted.

Table A1.1-1

Terminology and Notation

Model: NUHOMS®-MP187

See Table 1.1-1 for Terminology and Notation applicable to all MP187 Packages.

DSC: The Dry Shielded Canister. The DSC consists of a cylindrical shell, top and bottom shield plugs, inner and outer bottom closure plates, inner and outer top cover plates, and the internal basket. There are four different types of DSCs: the FO-DSC, the FC-DSC, the FF-DSC and the 24PT1-DSC. All four DSCs have the same outside diameter and overall length.

Basket: A structural framework for the support of fuel assemblies which resides in the stainless steel canister shell assembly. For the FO-DSC, FC-DSC and 24PT1-DSC, the Basket Assembly consists of twenty-six carbon steel spacer discs with 24 openings each, supported in position by four high strength stainless steel support rods. Placed within each of the 24 openings is a composite guidesleeve consisting of a stainless steel inner sleeve, borated neutron absorbing material, and a stainless steel outer sleeve. For the FF-DSC, the Basket Assembly consists of 15 carbon steel spacer discs with 13 openings each, supported in position by four carbon steel support plates. Placed within each of the 13 openings is a stainless steel Failed Fuel Can which provides confinement for the damaged fuel assembly. For the 24PT1-DSC, up to four specially designed stainless steel Failed Fuel Cans may be placed in the outside corner spacer disc openings at the 45°, 135°, 225° and 315° azimuth locations to provide confinement for damaged fuel assemblies which may be stored in the 24PT1-DSC.

Table A1.1-1

Terminology and Notation

Model: NUHOMS®-MP187

(Concluded)

24PT1-DSC	A DSC designed to hold WE 14x14 stainless steel clad (SC) UO_2 and/or WE 14x14 zircalloy clad Mixed Oxide fuel (MOX) fuel assemblies including control components. These fuel assemblies are shorter than the B&W 15x15 fuel assemblies and therefore utilize top and bottom fuel spacers to center the fuel in the DSC. This DSC is also designed to hold up to four damaged fuel assemblies in Failed Fuel Cans. The 24PT1-DSC has the same internal cavity length as the FO-DSC.

A1.1 Introduction

This Appendix provides a detailed description of the 24PT1-DSC when used in conjunction with the MP187 Cask. The detailed description of the MP187 Cask design, testing, fabrication and operations is provided in Chapters 1 through 8 and is not affected by the 24PT1-DSC payload unless specifically noted in this Appendix. Therefore, unless required to clarify the text, or provide additional information, the MP187 Cask details are not repeated in this Appendix.

The 24PT1-DSC is designed to store and transport 24 WE 14x14 PWR fuel assemblies, either UO_2 (stainless steel clad) or Pu-UO_2 Mixed Oxide (MOX) Zircaloy clad fuel pellets, with or without integral control components. Provisions have been made to accommodate up to four stainless steel clad damaged fuel assemblies or one damaged Zircaloy clad MOX assembly in lieu of an equal number of undamaged fuel assemblies in the 24PT1-DSC. The 24PT1-DSC shell assembly is a high integrity stainless steel, welded pressure vessel, similar to the FO, FC and FF-DSCs described in Chapters 1 through 8. It provides confinement of radioactive materials; encapsulates the fuel in an inert atmosphere; and provides biological shielding during the DSC closure, transfer, storage, and transport operations. As an integral welded vessel, the canister shell assembly provides containment for the fuel; however, no credit other than biological shielding is taken for this additional containment boundary in this transportation SAR. The 24PT1-DSC basket consists of circular spacer disc plates which provide structural support for the fuel and the guidesleeves in the lateral direction. Axial support for the spacer discs is provided by four support rods, which extend over the full length of the cavity and bear on the canister top and bottom assemblies.

A1.2 Package Description

A basic description of the NUHOMS®-MP187 Package with the FO, FC and FF-DSCs is provided in Section 1.2 while this section provides a description of the 24PT1-DSC. General arrangement drawings for the MP187 Package are presented in Section 1.3.2 with drawings of the 24PT1-DSC provided in Section A1.3.2.

A1.2.1 Packaging

A1.2.1.1 Gross Weight

The gross weight provided in Section 1.2.1 bounds the 24PT1-DSC in that the FO/FC/FF-DSC payloads result in configurations that are heavier than the 24PT1-DSC as well as configurations that are lighter than the 24PT1-DSC.

A1.2.1.2 Materials of Construction, Dimensions, and Fabrication Methods

Materials of construction, dimensions and fabrication methods are provided in Section 1.2.1 for the MP187 Package including the MP187 Cask and impact limiters; a description of the 24PT1-DSC and its proposed payload is provided below.

A1.2.1.2.1 24PT1-Dry Shielded Canister

As described for the FO, FC and FF-DSCs; the 24PT1-DSC is a high integrity stainless steel, welded pressure vessel that provides confinement of the radioactive materials, encapsulates the fuel in an inert atmosphere, and provides axial biological shielding during DSC closure, transfer, storage, and transport operations. The 24PT1-DSC has an outside diameter of 67.19 inches and an overall length of 186.5 inches. The 24PT1-DSC has an internal cavity length of 167 inches and solid carbon steel shield plugs at each end. The shell and cover plates are fabricated from 5/8-inch Type 316 stainless steel instead of the Type 304 used for the FO, FC and FF-DSCs. The cylindrical shell, including the top and bottom cover plate assemblies, forms the pressure retaining boundary for the spent fuel and cover gas. The 24PT1-DSC is equipped with top and bottom shield plugs so that the occupational doses at the ends are minimized for drying, sealing, handling bottom shield plugs so that the occupational doses at the ends are minimized for drying, sealing, handling and transfer operations. Redundant welds join the shell and the top cover plate assemblies to form the confinement boundary. The cylindrical shell and inner bottom cover plate confinement boundary welds are made during fabrication of the DSC and are fully compliant to Subsection NB of the ASME Code. The top closure confinement welds are made after fuel loading. Both top plug penetrations (siphon and vent ports) are welded after DSC drying operations are complete. Other than for biological shielding purposes, the pressure containment capability of the DSC shell assembly is conservatively neglected for transportation.

The internal basket assembly is similar to the FO/FC-DSCs and contains 24 storage positions, one for each fuel assembly. The basket is composed of 26 circular spacer discs machined from carbon steel plates; the spacing of the discs has been modified from that used for the FO/FC DSCs to accommodate the WE 14x14 fuel. Axial support for the spacer discs is provided by four high strength stainless steel support rods which extend over the full length of the DSC cavity and bear on the canister top and bottom end assemblies. The 24PT1-DSC basket assembly includes 24 guidesleeve assemblies with integral borated neutron absorbing panels. The spacer discs have openings, that allow the guidesleeves and the fixed borated neutron absorber panels to pass through. No borated materials are used for structural load carrying members.

The DSC cavity is inerted with helium before being sealed. Carbon steel components are coated with a thin corrosion resistant layer of nickel to provide corrosion resistance for the short time that the DSC is in the spent fuel pool for fuel loading. After the DSC is drained, dried, inerted and sealed for storage, or transportation, there is no mechanism available for corrosion of the carbon steel components.

On the bottom of each DSC there is a grapple ring which allows the DSC to be transferred horizontally in and out of the NUHOMS® storage modules from the cask. The DSC basket is keyed to the DSC shell and the grapple ring is keyed to the cask bottom closure to maintain the basket-to-cask alignment during all operations.

The 24PT1-DSC is capable of storing and transporting up to 24 WE 14x14 intact fuel assemblies (UO_2 or MOX) including control components. Up to four damaged fuel assemblies may be placed in the outside corner spacer disc openings at the 45°, 135°, 225° and 315° azimuth locations.

The Failed Fuel Cans used in the 24PT1-DSC are similar to those used in the FF-DSC, described in Chapter 1 of the SAR, but with a smaller cross section to fit into the 24PT1-DSC guidesleeves and a different screen configuration at the bottom of the can (the FF-DSC has two screened openings on the bottom plate while the 24PT1-DSC has an opening in the bottom plate and two openings in opposing side plates near the bottom of the can). The 24PT1-DSC Failed Fuel Cans consist of seam-welded stainless steel tubes with a welded bottom lid assembly and a welded removable top lid assembly. The cans do not contain any borated neutron absorbing materials. They provide for the confinement of the fuel pellets/shards by means of fixed bottom screens and a removable top screen. The bottom lid and top lid stainless steel screens allow for dewatering of the Failed Fuel Cans. The bottom end of each can includes provisions for fuel support. The fuel can top lids are fitted with lifting provisions that interface with plant fuel handling equipment for placement of the lid and handling of the removable fuel can. The top lid also incorporates the top fuel spacer.

A1.2.1.3 Neutron Moderation and Absorption

Neutron moderation is provided in the 24PT1-DSC by geometric spacing of the guidesleeve assemblies and by borated neutron absorbing material contained in the guidesleeve assemblies. Geometric spacing of the guidesleeves and the fuel assemblies is maintained by the spacer discs. The borated neutron absorber plates are secured to the guidesleeves by oversleeves. No credit is taken for the borated material in the structural analysis of the guidesleeves.

A1.2.1.4 Heat Dissipation

There are no special devices utilized on the NUHOMS®-MP187 Package for the transfer or dissipation of heat. Heat dissipation from the package is entirely passive. The package maximum design decay heat load for this application is 14.0 kW for the 24PT1-DSC. A more detailed discussion of the package thermal characteristics (with the 24PT1-DSC) is provided in Chapter A3.

A1.2.2 Operational Features

The NUHOMS®-MP187 Package is not considered to be operationally complex and is similar to other licensed transportation packages such as the NUPAC-125B cask. All operational features are readily apparent from inspection of the General Arrangement Drawings provided in Section 1.3.2 (for the MP187 Cask) and Section A1.3.2 (for the 24PT1-DSC). Operational procedures are provided in Chapters 7 and A7.

A1.2.3 Contents of Packaging

The contents of the NUHOMS®-MP187 Packaging to be included in the 24PT1-DSC payload may consist of intact or damaged spent reactor fuel assemblies with or without control components. The fuel is described below.

A1.2.3.1 Description of Spent Fuel Assemblies

The fuel design included in the 24PT1-DSC payload is WE 14x14 Stainless Steel Clad (SC) UO_2 and Mixed Oxide Zircalloy Clad (MOX, UO_2 and PuO_2) fuel assemblies with/or without control components meeting the enrichment and burnup parameters listed in Table A1.2-1. Stainless steel clad fuel may also include Integral Fuel Burnable Absorber (IFBA) consisting of Boron coated fuel pellets.

Table A1.2-1

WE 14x14 Spent Fuel Assembly Description

Fuel Type	Maximum Enrichment (weight %)	Minimum Enrichment (weight %)	Maximum Burnup (MWd/ MTU)	Minimum Cooling Time[2] / Max Heat Load per cask / Max Assembly Heat Load (incl. control components[2])
WE 14x14 Stainless Steel Clad (SC) (may include Integral Fuel Burnable Absorber, boron coated fuel pellets)	4.0 ^{235}U	3.80 ^{235}U	45,000	38[1] years/14 kW/ 0.583 kW
		3.40 ^{235}U	40,000	
		3.16 ^{235}U	35,000	
WE 14x14 MOX	0.71 ^{235}U 3.31 fissile Pu	• 2.78 Pu (64 rods) • 3.05 Pu (92 rods) • 3.25 Pu (24 rods)	25,000	20 years/14 kW/ 0.583 kW

Notes:

[1] The 38 year cooling requirement may be shortened if it can be demonstrated using the methodology described in Chapter A5 that a payload, meeting all other requirements specified above, will result in calculated MP187 Package dose rates that are bounded by the results reported in Chapter A5. For this analysis a method of classification of fuel assemblies similar to that specified for the B&W 15x15 fuel, in Chapters 1 and 5, may be applied. Alternatively, the acceptability for transport of WE 14x14 stainless clad (SC) fuel with less than 38 years of cooling may be demonstrated by measured doses on the Package surface prior to shipment. This method must apply a 25% margin to account for potential errors in the value measured. The measured dose for the normal operation conditions with the added margin shall be compared to the 10CFR71 limits and is acceptable if within these limits (including margin).

[2] Control component cooling time must be a minimum of 10 years.

Intact fuel assemblies include those with minor cladding damage, including pinhole leaks and hairline cracks, provided the cladding integrity is sufficient to prevent significant fuel particulate release. The 24PT1-DSC may be loaded with fewer than 24 fuel assemblies as long as dummy fuel assemblies are installed in all unoccupied spaces, except that two spaces may remain empty. These two empty spaces must be located on opposing symmetrical locations with respect to the 0° - 180° and 90° - 270° axes. The dummy assemblies shall have the same nominal weight as a standard fuel assembly to maintain the symmetric weight distribution in the 24PT1-DSC and minimize loads on the DSC/cask alignment keys.

The 24PT1-DSC may include up to four damaged WE 14x14 stainless clad (SC) fuel assemblies together with intact WE 14x14 SC fuel and/or dummy fuel assemblies based on the limitations specified above. A single damaged WE 14x14 MOX fuel assembly may be stored in a 24PT1-DSC with no other damaged assemblies. Damaged fuel assemblies must be contained in 24PT1-DSC Failed Fuel Cans and are limited to the guidesleeves located in the four outside corner locations along the 45°, 135°, 225° and 315° azimuth locations. In all cases there may be two empty guidesleeves symmetrically located such that 22 guidesleeves contain intact fuel, dummy fuel assemblies or Failed Fuel Cans.

Damaged fuel stored in a 24PT1-DSC includes assemblies with known or suspected cladding defects greater than a hairline crack or a pinhole leak up to and including broken rods, portions of broken rods and rods with missing sections. Individual fuel rods or portions of fuel rods may be stored individually in a Failed Fuel Can. The Failed Fuel Cans in which the damaged fuel is stored confine gross fuel particles to a known, subcritical, volume during off-normal and accident conditions and facilitate handling and retrievability. The criticality analysis provided in Chapter A6 requires that cladding damage be limited to no more than 14 fuel pins in an assembly. Visual inspection of assemblies will be performed once, prior to placement of the fuel in the DSC, which may then be placed in storage and transported anytime thereafter without further fuel inspection.

Table A1.2-2 provides details of fuel dimensions, weights and shielding source terms.

A1.2.3.2 Maximum Payload Weight

The maximum weight of the payload for the FO, FC and FF-DSCs is provided in Section 1.2.3.5. The 24PT1-DSC maximum fuel payload is:

31,700 pounds

Weights of the loaded 24PT1-DSCs are provided in Section A2.2.

A1.2.3.3 Maximum Decay Heat

The maximum design basis decay heat for WE 14x14 fuel stored in the 24PT1-DSC is 14.0 kW. The maximum allowable decay heat for a single assembly, including control components, is 0.583 kW. Although this heat load is higher than the 13.5 kW maximum heat load for the MP187 B&W 15x15 fuel assemblies, the maximum component temperatures are lower for the 24PT1-DSC due to a reduced peaking factor as discussed in Chapter A3.

A1.2.3.4 Maximum Pressure Buildup

The MP187 Cask is designed for a maximum pressure of 64.7 psia. As shown in Chapter A3, the Maximum Normal Operating Pressure (MNOP) for the 24PT1-DSC payload is considerably less than the maximum pressure expected during the hypothetical accident conditions.

The 24PT1-DSC free volume and the radionuclides, available in the fuel pins for release used in the calculation of the maximum pressure, is specified in Chapter A3.

Table A1.2-2

Westinghouse Fuel Dimensions, Weights and Source Terms

Parameter	WE 14x14 SC(1)	WE 14x14 MOX(1)
Number of Rods	180	180
Cross Section (in)	7.763x7.763	7.763x7.763
Unirradiated Length (in)	138.5	138.5
Fuel Rod Pitch (in)	0.556	0.556
Fuel Rod O.D. (in)	0.422	0.422
Clad Material	Type 304 SS	Zircaloy-4
Clad Thickness (in)	0.0165	0.0243
Pellet O.D. (in)	0.3835	0.3659
Max. initial ^{235}U Enrichment (wt%)	4.0	Note 2
Theoretical Density (%)	93-95	91
Active Fuel Length (in)	120	119.4
Max. U Content (kg)	375	Note 3
Ave. U Content (kg)	366.3	Note 3
Assembly Weight (lbs)	1210	1150
Max. Assembly Weight incl. NFAH(4) (lbs)	1320	1320

Fuel Assembly Source Term

Parameter	WE 14x14 SC(1)	WE 14x14 MOX(1)
Gamma Source (γ/sec/assy)(6)	1.38E+15(5)	9.57E+14(5)
Neutron Source (n/sec/assy)(6)	1.03E+08(5)	4.90E+07(5)

NFAH Source Term

Parameter	Rod Cluster Control Assemblies (RCCAs)	Thimble Plugs (TPs)	Neutron Source Assemblies (NSAs)
Gamma Source (γ/sec/assy)(6)	7.60E+12	5.04E+12	1.20E+13
Decay heat (Watts)	1.90	1.2	1.66

(1) Nominal values shown unless stated otherwise
(2) Mixed-Oxide assemblies with 0.71 weight % U-235 and fissile Pu weight of 2.84 weight % (64 rods), 3.10 weight % (92 rods), and 3.31 weight % (24 rods)
(3) Total weight of Pu is 11.24 kg and the total weight of U is 311.225 kg
(4) Weights of TPAs and NSAs are enveloped by RCCAs
(5) Based on cooling time and burnups specified in Table A1.2-1
(6) Gamma/neutron source term by energy group is presented in Chapter A5.

A1.3 Appendix

A1.3.1 References

A1.3.2 General Arrangement Drawings

A1.3.1 References

A1.1 U.S. Government, Packaging and Transportation of Radioactive Material, Title 10 Code of Federal Regulations, Part 71, Office of the Federal Register, Washington, DC.

A1.2 U.S. Government, Shippers-General Requirements for Shipments and Packagings, Title 49 Code of Federal Regulations, Part 173, Office of the Federal Register, Washington, DC.

A1.3.2 General Arrangement Drawings

The following drawings are provided in this Appendix. The drawings for the MP187 Package are provided in Appendix 1.3.2 of the SAR:

1. NUH-05-4010, NUHOMS® 24PT1-DSC Main Assembly, Rev. 0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

ALL DIMENSIONS ARE APPLICABLE AT 70°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.	TRANSNUCLEAR WEST				
DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M-1994.					
BREAK AND DEBURR ALL SHARP EDGES	TITLE: SAFETY ANALYSIS REPORT GENERAL LICENSE NUHOMS® 24PT1-DSC MAIN ASSEMBLY				
3rd ANGLE PROJECTION					
DO NOT SCALE DRAWING	FILE NO. NUH005.4010	DWG. NO. NUH-05-4010NP	SCALE NONE	SHEET 1 OF 3	REV. NO. 0

97

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

ALL DIMENSIONS ARE APPLICABLE AT 70°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.

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

BREAK AND DEBURR ALL SHARP EDGES

3rd ANGLE PROJECTION

DO NOT SCALE DRAWING

TRANSNUCLEAR WEST

TITLE: SAFETY ANALYSIS REPORT
GENERAL LICENSE
NUHOMS® 24PT1-DSC
MAIN ASSEMBLY

FILE NO.	DWG. NO.	SCALE	SHEET	REV. NO.
NUH005.4010	NUH-05-4010NP	NONE	2 OF 3	0

2 1

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

ALL DIMENSIONS ARE APPLICABLE AT 70°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL MACHINING UNLESS NOTED OTHERWISE.	TRANSNUCLEAR WEST
DIMENSIONS ARE IN INCHES AND DEGREES UNLESS NOTED OTHERWISE. DIMENSIONING AND TOLERANCING IN ACCORDANCE WITH ASME Y14.5M–1994.	
BREAK AND DEBURR ALL SHARP EDGES	TITLE: SAFETY ANALYSIS REPORT GENERAL LICENSE NUHOMS® 24PT1–DSC MAIN ASSEMBLY
3rd ANGLE PROJECTION	
DO NOT SCALE DRAWING	FILE NO. NUH005.4010 | DWG. NO. NUH–05–4010NP | SCALE NONE | SHEET 3 OF 3 | REV. NO. 0

A2. STRUCTURAL EVALUATION

The structural evaluations presented herein demonstrate that the NUHOMS®-MP187 Package containing a 24PT1-DSC meets all applicable structural criteria of 10CFR71 [A2.3] as described in Section 2.1.2. The structural evaluation of the NUHOMS®-MP187 Cask and impact limiters for the normal transport and hypothetical accident conditions, presented in Chapter 2, is unchanged by the addition of the 24PT1-DSC payload.

A2.1 Structural Design

A2.1.1 Discussion

The NUHOMS®-MP187 Package (cask with impact limiters) may be loaded with any one of four DSC types. Chapter 2 of this Safety Analysis Report provides the justification and qualification for the transportation of the FO-DSC, the FC-DSC, and the FF-DSC. This Appendix addresses the fourth DSC type to be transported – the 24PT1-DSC. The 24PT1-DSC is designed to hold up to 24 Westinghouse (WE) 14x14 fuel assemblies as described in Section A.1. Also, symmetrically loaded Failed Fuel Cans may be placed in the 24PT1-DSC to transport damaged fuel assemblies with intact fuel assemblies. The damaged fuel assemblies (Failed Fuel Cans) must be located in the four outer corner fuel assembly locations along the 45°, 135°, 225°, 315° azimuths of the DSC.

As shown in Table A2.1-1, the 24PT1-DSC configuration is the same as the FO-DSC except that the 24PT1-DSC has a modified spacer disc spacing and support rod configuration (see Sections A2.6.11.A and A2.6.11.B for a discussion of the adequacy of the 24PT1-DSC configuration). Also, top and bottom fuel spacers are required to center the shorter WE 14x14 fuel. The maximum (hot, irradiated) fuel assembly length including control components plus the top and bottom fuel spacers is 166.05 inches (per Table 3.5-3 of [A2.2]). This is bounded (in terms of fuel assembly thermal expansion) by the B&W 15x15 hot irradiated fuel length of 166.89 inches, calculated in Section 2.6.11 for the FO-DSC, and the 24PT1-DSC cavity length of 167.00 inches. The 24PT1-DSC shell, outer cover plates and associated components are fabricated from SA240, Type 316 stainless steel to enhance corrosion resistance. Other minor changes in the DSC configuration, that do not impact the overall design, are detailed in the 24PT1-DSC drawing (NUH-05-4010) included in Section A1.3.2. A comparison of the FO-DSC and the 24PT1-DSC is provided in Table A2.1-1.

The configuration of the Failed Fuel Cans used in conjunction with the 24PT1-DSC is similar to that used in the FF-DSC. The major difference between the two cans is that the 24PT1-DSC Failed Fuel Can is designed to be installed inside a standard guidesleeve and is, therefore, smaller than the FF-DSC can. To accommodate the shorter WE 14x14 fuel, the 24PT1-DSC Failed Fuel Cans incorporate the top fuel spacer into the Failed Fuel Can lid. A bottom spacer, identical to that used for storage of intact fuel, is installed below the Failed Fuel Can to maintain the correct center of gravity for the canister. The 24PT1-DSC Failed Fuel Can has screened drain openings at the bottom of the can that differ slightly from those detailed for the FF-DSC. The bottom of each 24PT1-DSC Failed Fuel Can contains one drainage hole (vs. two for the FF-DSC) with two screened openings provided in the sidewalls of each 24PT1-DSC Failed Fuel Can. This change accommodates the different bottom end assembly configurations of the B&W 15x15 and WE 14x14 fuel. The Failed Fuel Can design length, including bottom spacer, is equal to an intact fuel assembly with top and bottom spacer length.

Table A2.1-1

Configuration of the FO-DSC to the 24PT1-DSC

Characteristic	FO-DSC Configuration	24PT1-DSC Configuration
Shell Outside Diameter x Length	67.19 " x 186.5"	67.19" x 186.5"
DSC Internal Cavity Length	167.0"	167.0"
Shell (SA240)	5/8" thick, Type 304	5/8" thick, Type 316
Support Rod Diameter/Preload	2"/80 +/- 5 kips	1.25"/40 +/- 15 kips
Support Rod Spacer Sleeves, Outside Dia./Inside Dia.	3" / 2.08"	3" / 1.33"
Spacer Discs (Number-Material)	26 – SA537, Cl 2	26 – SA537, Cl 2
Spacer Disc Spacing	See Section 1.3.2 for spacing	See Section A1.3.2 for spacing
Guidesleeves (inside width x length)	8.9" x 161.8"	8.9" x 165.25"
Top and Bottom Fuel Spacers	Not Required	See Section A1.3.2
Failed Fuel Cans	Not Permitted	Up to four, symmetrically loaded (similar to FF-DSC)

A2.1.2 Design Criteria

The applicable ASME Code Edition for the 24PT1-DSC is defined to be 1992, with Addenda through 1994 including Code Case N-595-1 [A2.1] (as opposed to 1992 with Addenda through 1993 for the FO, FC and FF-DSCs and MP187 Cask). There were no significant changes in requirements, methodology, allowable stresses, fabrication, inspection, or testing introduced in the 1994 Addenda. Hence, the criteria for the 24PT1-DSC are essentially the same as the FO, FC, and FF-DSC criteria as described in Section 2.1.2.

The brittle fracture and fatigue evaluations for the FO/FC-DSC spacer discs presented in Section 2.1.2.3.1.B and 2.2.3.1.C are applicable to the 24PT1-DSC since the FO/FC-DSC loads bound those for the 24PT1-DSC, and the same geometry is used for all three DSC spacer disc designs.

A2.2 Weights and Centers of Gravity

The structural analyses presented in Section 2.2 are applicable to the MP187 Cask loaded with a 24PT1-DSC, since the 24PT1-DSC weight, center of gravity (cg) and weight moment of inertia (MOI) are bounded by the FO, FC and FF-DSCs. The 24PT1-DSC weight is bounded by the FO/FC-DSC and FF-DSC weights to ensure that the effect of a lighter DSC (increased g loads during a drop) and a heavier DSC (higher stresses for non-drop loading conditions) envelope the 24PT1-DSC.

A comparison of the weight, cg and MOI for the four DSCs is provided in Table A2.2-1 below:

Table A2.2-1

Comparison of DSC Weights, cg and MOI

Parameter	FO-DSC	FC-DSC	FF-DSC	24PT1-DSC
Total DSC weight (lbs)	80,710	81,120	74,900	78,400[1,2]
DSC cg location, along center line of DSC, with respect to the outer surface of the MP187 Cask bottom cover plate (inches)	100.4	98.7	102.4	99.3
DSC MOI (lbm-in^2), relative to cg	3.36E+08	2.88E+08	3.03E+08	3.29E+08

[1] This weight is increased by 1,000 lbs. when four failed fuel cans are included in the DSC.

[2] This weight is made up of (rounded to nearest 100 lbs.):

Shell Assembly	15,600 lbs.
Basket Assembly	18,500 lbs.
Shield Plug	8,000 lbs.
Inner Cover Plate	700 lbs.
Outer Cover Plate	1,200 lbs.
Fuel Spacers	2,700 lbs.
WE 14x14 fuel/including NFAH	31,700 lbs.

The total weight of the NUHOMS®-MP187 transportation package, loaded with an FC-DSC or FF-DSC payload, ranges from 271,300 to 265,100 pounds and bounds the total weight of 268,800 pounds when loaded with a 24PT1-DSC payload. The weight, center of gravity and moment of inertia of the MP187 transportation package components are presented in Table

A2.2-2. As shown in this table, the MP187 loaded with a 24PT1-DSC payload is bounded by the FO/FC/FF-DSC payloads.

Table A2.2-2

NUHOMS® System Mass Properties

ITEM DESCRIPTION	WEIGHT (lb)	cg (in.)[1]	MOI (lbm-in^2)
MP187 Cask	158,600	99.7	7.02E+08
Bottom Impact Limiter	15,800	-1.5	4.48E+07
Top Impact Limiter	15,800	203.0	4.48E+07
MP187 Cask/FC- DSC (Loaded, Dry)	239,700	99.4	9.90E+08
MP187 Cask/FO- DSC (Loaded, Dry)	239,300	99.9	1.04E+09
MP187 Cask/FF- DSC (Loaded, Dry)	233,500	100.6	1.01E+09
MP187 Cask/24PT1-DSC (Loaded, Dry)	237,200	99.6	9.92E+08
MP187 Cask/FC-DSC/Impact Limiter (W/ Fuel, Dry)	271,300	99.5	1.41E+09
MP187 Cask/F0-DSC/Impact Limiter (W/ Fuel, Dry)	270,900	100.0	1.46E+09
MP187 Cask/FF-DSC/Impact Limiter (W/ Fuel, Dry)	265,100	100.6	1.43E+09
MP187 Cask/24PT1-DSC/Impact Limiter (W/ Fuel, Dry)	268,800	99.8	1.41E+09
MP187 Cask/FC-DSC (W/ Fuel, Water & W/ Cover Plates)	244,400	-	-
MP187 Cask/FO-DSC (W/ Fuel, Water & W/ Cover Plates)	243,800	-	-
MP187 Cask/FF-DSC (W/ Fuel, Water & W/ Cover Plates)	239,500	-	-
MP187 Cask/24PT1-DSC (W/ Fuel, Water & Cover Plates)	241,500	-	-

Note:
(1) Longitudinal centers of gravity are measured from the outer surface of the MP187 Cask bottom cover plate.

A2.3 Mechanical Properties of Materials

The materials used in the fabrication of a 24PT1-DSC are shown in Table A2.3-1 and are the same as those for the FO-DSC reported in Section 2.3.2, with the exception of the shell and cover plates which are fabricated from SA 240, Type 316 material. The SA-240, Type 316 material provides slightly higher allowables; however, since the DSC shell and cover plates are not credited in the MP187 package structural analyses, this has no effect on the analyses presented. The difference in thermal properties between the two materials is negligible, given the high thermal conductivity of the steel with respect to air, or helium gas, used in the thermal analyses. The difference in the coefficient of thermal expansion between the two materials is also negligible. Therefore, the effect of this material change on the thermal analyses is negligible.

Table A2.3-1

NUHOMS® 24PT1-DSC Structural Component Materials

NUHOMS® 24PT1-DSC Component	Material Specification	Type/Grade
Shell	SA-240	Type 316
Spacer Disc	SA-537	Class 2
Support Rod & Support Rod Sleeves	SA-564	Type 630
Guidesleeves	SA-240	Type 304
Oversleeves	SA-240	Type 304
Failed Fuel Can	SA-240	Type 304
Top and Bottom Shield Plugs	A36	-
Inner and Outer Bottom Cover Plates	SA-240	Type 316
Inner and Outer Top Cover Plates	SA-240	Type 316

The neutron absorber panels used in the 24PT1-DSC are the same material used in the FO/FC-DSCs and are described in Section 2.3.5.

A2.4 General Standards for All Packages

The discussion provided in Section 2.4 for the FO-DSC is also applicable to the 24PT1-DSC.

A discussion of the susceptibility of packaging materials to chemical, galvanic or other reactions is provided in Section 2.4.4. This discussion is applicable to the packaging with a 24PT1-DSC payload.

A2.5 Lifting and Tie-Down Standards for All Packages

The discussion provided in Section 2.5 is applicable to the MP187 Cask with a 24PT1-DSC payload.

A2.6 Normal Conditions of Transport

A2.6.1 Heat

No change to MP187 Cask analysis provided in Section 2.6.1 for the 24PT1-DSC payload. Analyses for FO/FC/FF-DSC configurations bound the 24PT1-DSC payload.

A2.6.2 Cold

No change to MP187 Cask analysis provided in Section 2.6.2 for the 24PT1-DSC payload. Analyses for FO/FC/FF-DSC configurations bound the 24PT1-DSC payload.

A2.6.3 Pressure

No change to MP187 Cask analysis provided in Section 2.6.3 for the 24PT1-DSC payload. Analyses for FO/FC/FF-DSC configurations bound the 24PT1-DSC payload.

A2.6.4 Vibration

No change to MP187 Cask analysis provided in Section 2.6.4 for the 24PT1-DSC payload. Analyses for FO/FC/FF-DSC configurations bound the 24PT1-DSC payload.

A2.6.5 Water Spray

No change to MP187 Cask analysis provided in Section 2.6.5 for the 24PT1-DSC payload. Analyses for FO/FC/FF-DSC configurations bound the 24PT1-DSC payload.

A2.6.6 Free Drop

No change to MP187 Cask analysis provided in Section 2.6.6 for the 24PT1-DSC payload. Analyses for FO/FC/FF-DSC configurations bound the 24PT1-DSC payload.

A2.6.7 Corner Drop

No change to MP187 Cask analysis provided in Section 2.6.7 for the 24PT1-DSC payload. Analyses for FO/FC/FF-DSC configurations bound the 24PT1-DSC payload.

A2.6.8 Compression

No change to MP187 Cask analysis provided in Section 2.6.8 for the 24PT1-DSC payload. Analyses for FO/FC/FF-DSC configurations bound the 24PT1-DSC payload.

A2.6.9 Penetration

No change to MP187 Cask analysis provided in Section 2.6.9 for the 24PT1-DSC payload. Analyses for FO/FC/FF-DSC configurations bound the 24PT1-DSC payload.

A2.6.10 Fabricaiton Stress Analysis

No change to MP187 Cask analysis provided in Section 2.6.10 for the 24PT1-DSC payload. Analyses for FO/FC/FF-DSC configurations bound the 24PT1-DSC payload.

A2.6.11 24PT1-DSC Normal Conditions of Transport

The NUHOMS®-MP187 transportation package with a 24PT1-DSC payload meets the performance requirements specified in Subpart E of 10CFR71 when subjected to normal conditions of transport, as specified in 10CFR71.71 [A2.3].

As discussed in Chapter A3 the thermal heat load for the 24PT1-DSC is enveloped by the FO/FC-DSC with the same external design conditions. Therefore, the design basis temperatures and thermal gradients for the 24PT1-DSC are enveloped by the FO/FC-DSC and the thermal stress analyses results presented in Section 2.6 are directly applicable to the 24PT1-DSC.

Figure A2.6-1 provides an elevation and end view of the 24PT1-DSC basket assembly.

Sections A2.6.11.A through A2.6.11.E below provide an overview of the analyses for the spacer discs, support rods, guidesleeves, Failed Fuel Cans and fuel spacers.

A2.6.11.A Spacer Discs

The 24PT1-DSC spacer disc detail is shown in Figure A2.6-2 (the difference in support rod hole diameter, which is the only difference with the FO/FC-DSC is indicated on this figure). The 24PT1-DSC spacer disc design loads and load distribution are shown in Table A2.6-1 below. Table A2.6-2 provides the WE 14x14 fuel weight distribution used in determining the load distribution for the spacer discs. By comparing these loads to the FO/FC-DSC design loads shown in Tables 2.10.3.1a and 2.10.3.1b it can be seen that the FO/FC loads bound the 24PT1-DSC. Therefore, the FO/FC spacer disc analyses presented in Section 2.6.11 bound the 24PT1-DSC spacer discs and are directly applicable.

Table A2.6-1

Spacer Disc Tributary Weight[2] Distribution

Disc No.	Centerline Location from bottom inside of DSC (inches)	Tributary Length (span between spacer disc centerlines) (inches)	Spacer Disc Weight (lbs)	Fuel Weight[1] (lbs)	Failed Fuel Cans Weight (lbs)	Guidesleeve Weight (lbs)	Support Rod Assembly Weight (lbs)	TOTAL Weight W_{TOT} (lb)
1			418.8	610.5	0.0	237.7	50.4	1,317.4
2			418.8	512.8	0.0	199.6	42.3	1,173.6
3			418.8	628.8	0.0	199.6	42.3	1,289.5
4			418.8	1,388.6	27.1	209.2	44.3	2,088.0
5			418.8	1,546.4	30.1	232.9	49.4	2,277.7
6			418.8	1,641.1	32.0	247.2	52.4	2,391.5
7			418.8	1,672.7	32.6	251.9	53.4	2,429.4
8			418.8	1,704.2	33.2	256.7	54.4	2,467.3
9			418.8	1,704.2	33.2	256.7	54.4	2,467.3
10			418.8	1,704.2	33.2	256.7	54.4	2,467.3
11			418.8	1,704.2	33.2	256.7	54.4	2,467.3
12			418.8	1,704.2	33.2	256.7	54.4	2,467.3
13			418.8	1,704.2	33.2	256.7	54.4	2,467.3
14			418.8	1,704.2	33.2	256.7	54.4	2,467.3
15			418.8	1,704.2	33.2	256.7	54.4	2,467.3
16			418.8	1,704.2	33.2	256.7	54.4	2,467.3
17			418.8	1,704.2	33.2	256.7	54.4	2,467.3
18			418.8	1,672.7	32.6	251.9	53.4	2,429.4
19			418.8	1,641.1	32.0	247.2	52.4	2,391.5
20			418.8	1,641.1	32.0	247.2	52.4	2,391.5
21			418.8	1,641.1	32.0	247.2	52.4	2,391.5
22			418.8	1,266.0	30.8	237.7	50.4	2,003.6
23			418.8	426.2	29.5	228.2	48.4	1,151.0
24			418.8	427.2	28.0	216.3	45.8	1,136.2
25			418.8	508.4	129.4	202.0	42.8	1,301.4
26			418.8	845.4	210.8	329.1	69.8	1,873.9
			10,889.0	35,113.0	1,011.0	6,352.0	1,346.0	54,710.0

Note: [1] Includes weight of top and bottom fuel spacers.
[2] Weights shown are bounding weights for the DSC configuration with WE 14x14 SC or MOX fuel.

Table A2.6-2

Weight Distribution of Westinghouse 14x14 Fuel Including Control Components[1]

Region	Component	Component Span (in)	Component Weight (lbs)	Piece Line Load (lb/in)	Region Line Load (lb/in)
Bottom Region	Bottom Nozzle	3.188	17.4	5.46	5.46
Above Active Fuel Region	Cladding	126.130	236.2	1.87	2.94
	Guide Tubes	126.130	15.8	0.13	
	Spacer Plenum	126.130	1.5	0.01	
	Spacer-Incore	126.130	7.7	0.06	
	Control Components	126.130	110.0	0.87	
Active Fuel Region	Active Fuel	120.000	909.0	7.58	10.52
Top Region	Holddown Spring	6.770	0.9	0.13	3.13
	Top Nozzle	6.770	20.3	3.00	
Fuel Assembly Total (w/Control Components)			1318.8		

1 Weight distribution provided is bounding for WE 14x14 SC and WE 14x14 MOX fuel

The 24PT1-DSC support rods, guidesleeves, Failed Fuel Cans and fuel spacers are discussed below. These sections supplement the information presented in Section 2.6.11 for the FO/FC/FF DSC.

Guidesleeve weight and fuel weight used in the analysis for the 24PT1-DSC are bounded by the FO-DSC configuration, therefore the FO-DSC spacer disc analyses are bounding for the 24PT1-DSC.

A2.6.11.B Support Rods

The configuration of the support rod assembly in the 24PT1-DSCs is similar to the configuration used in the FO-DSC described in Section 2.6. However, as listed in Table A2.1-1 and summarized below, the differences between the 24PT1-DSC and the FO-DSC show that the 24PT1-DSC structural capacity is higher:

- The nominal OD of the support rods has been decreased.
- The support rod spacer sleeve inside diameter has been decreased to match the decreased support rod OD.

- Preload in the rod assembly has been decreased. Analyses contained in this chapter conservatively assume 65 kips maximum preload and zero preload to envelope actual conditions. The actual conditions will be 40 kips ±15 kips.

The net result of these changes is a significant increase in the capacity of the support rod assemblies. This results from moving material from the support rods (where the most significant load is the initial preload) to the spacer sleeves which carry all loads from the spacer discs.

A2.6.11.C Guidesleeves

With the following exceptions, which serve to enhance the fabricability and structural capacity of the guidesleeves; the structure of the guidesleeve assemblies (guidesleeves, oversleeves, poison sheets, and associated welds) in the 24PT1-DSC are identical to the guidesleeve assemblies in the FO-DSC:

- corner fillet welds are eliminated and guidesleeves are fabricated using full penetration welds; thus increasing the capacity of the sleeve,
- an option has been added for using fillet welds in place of the spot welds for connecting the oversleeves to the guidesleeves,
- the top edge of the oversleeve is connected to the "shim" plates instead of directly to the guidesleeve. This increases the strength of the connection and simplifies fabrication, and
- the location of the neutron absorbing sheets is adjusted to ensure coverage of the active fuel region under all conditions.

A2.6.11.D Failed Fuel Cans

The Failed Fuel Cans are fabricated using the same material (grade and thickness) as the guidesleeves and are sized to fit inside the guidesleeves. Thus, under all loading conditions, the Failed Fuel Cans act in parallel with the guidesleeves. Since the materials have equal stiffness (same thickness and material properties), the induced loads will be distributed equally between the guidesleeve and Failed Fuel Can. Thus, for all load conditions, the guidesleeve results presented in Section 2.6 also apply to the Failed Fuel Cans and no additional analysis is necessary.

A2.6.11.E Fuel Spacers

Fuel Spacers are used in the 24PT1-DSC to center the fuel within the canister. Fuel Spacers are not used in the FO/FC/FF-DSCs. The fuel spacers were designed for all normal conditions of transport including handling/vibration loads and the 1-foot side drop. Although the spacers are not pressure boundary components, pressure loads were applied along with the normal conditions of transport mechanical loads. Analyses of the fuel spacers are included in the following sections.

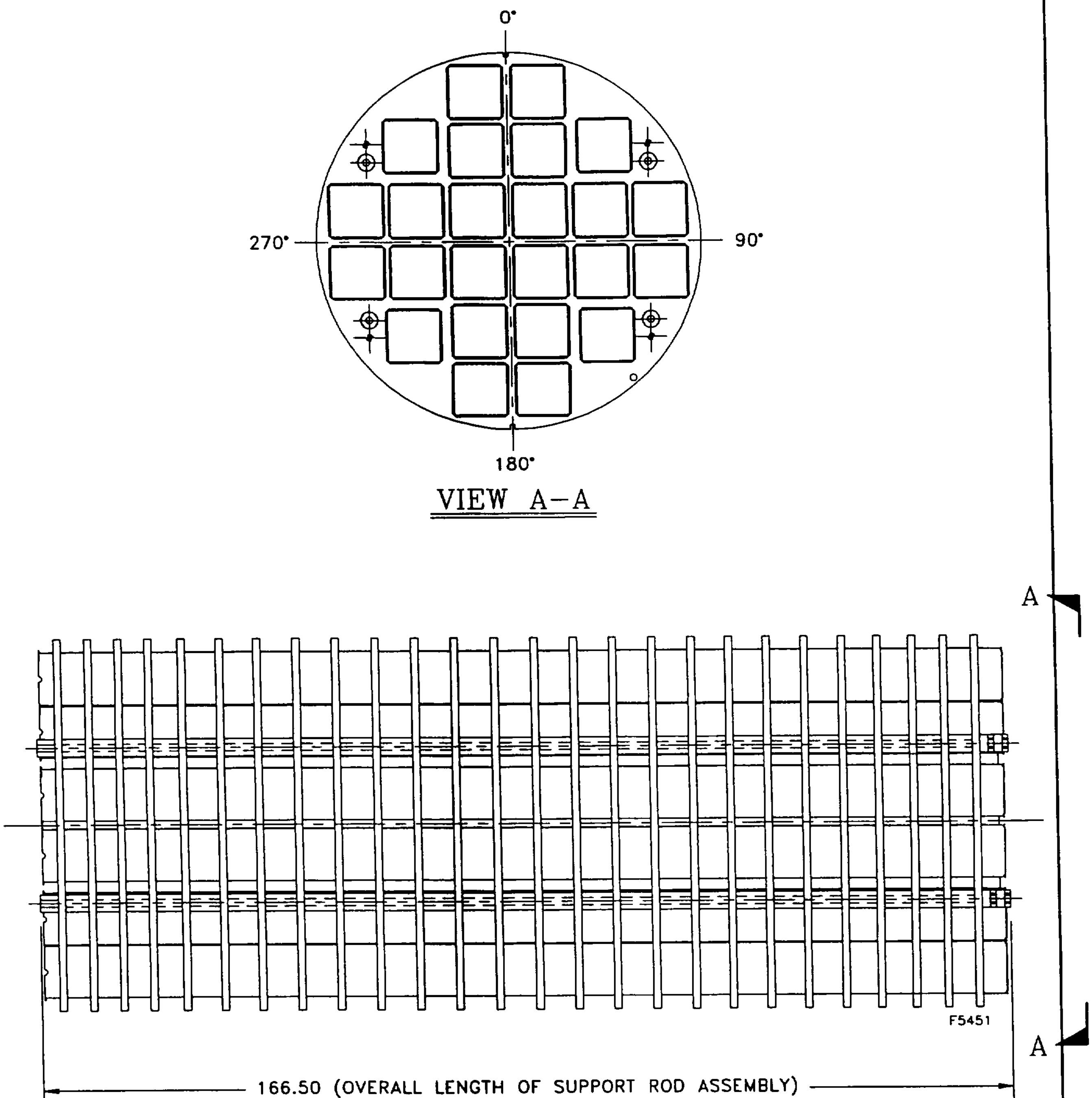


Figure A2.6-1

24PT1-DSC Basket Assembly Detail

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure A2.6-2

24PT1-DSC Spacer Disc Detail

The following sections provide the results for the 24PT1-DSC support rods, guidesleeves, fuel spacers, and Failed Fuel Cans for the normal conditions of transport that are unique to the 24PT1-DSC. All other basket components are enveloped by the analyses presented in Section 2.6.11 and are not repeated here.

A2.6.11.1 Basket Fabrication Stress Analysis

Stresses from the imposed preload are considered in all evaluations of the support rod assemblies. The preload is applied using methods which minimize residual twisting of the rod (e.g., using a tensioner to apply the initial stretch to the rod).

Stresses in the spacer sleeves and support rod from the enveloping preload of 65 kips are provided in Table A2.6-3:

Table A2.6-3

Support Rod and Support Rod Spacer Sleeve Stresses Due to Preload

Component	Axial Preload Stress* (ksi)	Allowable Stress (Level A, 600°F) (ksi)	Stress Ratio
Support Rod (Threaded Section)	$\frac{65\ kip}{A_{ten}} = 60.6$	65.7	0.92
Support Rod (Nominal Section)	$\frac{65\ kip}{A_{Nom}} = 53.0$	55.7	0.95
Spacer Sleeve	$\frac{65\ kip}{A_{Nom}} = 11.5$	(53.6 min) varies with sleeve length	< 0.22

* The analysis preload of 65 kips was selected to maximize support rod stresses (i.e., support rod stress ratio approximately 1.0). The actual applied preload is significantly less (40 +/- 15 ksi).

A2.6.11.2 Thermal

A2.6.11.2.1 Spacer Disc Thermal Expansion

The analysis presented in Section 2.6.11.2.1 is applicable to the 24PT1-DSC since the spacer disc configuration, loading and thermal conditions are bounded by the FO/FC-DSC analyses.

A2.6.11.2.2 Spacer Disc Thermal Stress Analysis

The analysis presented in Section 2.6.11.2.2 is applicable to the 24PT1-DSC since the spacer disc configuration, loading and thermal conditions are bounded by the FO/FC-DSC analyses.

A2.6.11.2.3 Support Rod Thermal Expansion

As described in 2.6.11.2.3, clearance between the support rods and DSC cavity is large such that thermal growth of the rods will not be restrained by the DSC cavity. Therefore, no additional evaluation is required.

A2.6.11.2.4 Support Rod Thermal Stress Analysis

Support rod assembly stresses are affected by thermal loads in two ways:

- As temperature increases, the spacer discs expand at a greater rate than the support rod/spacer sleeves. This increases tension in the rod and compression in the sleeves.
- As temperature increases, the elongation applied by the initial preload remains constant. However the axial stiffness of the rod assembly decreases (since E decreases with temperature). Thus, the loads (stresses) in the rod assembly decrease with increasing temperature.

The net result of these two effects is a small reduction in preload at elevated temperatures. These effects are included in the analyses of the rod assemblies, as appropriate.

A2.6.11.2.5 Guidesleeve Thermal Expansion

As described in A2.6.11.A and A2.6.11.C, the spacer disk and guidesleeve design for the 24PT1-DSC is bounded by the FO-DSC. As described in 2.6.11.2.5, clearance is provided in the FO-DSC to allow free expansion of the guidesleeve.

A2.6.11.2.6 Guidesleeve Thermal Stress Analysis

Since the guidesleeves are free to expand, they will not experience any significant stresses due to thermal loading during the normal conditions of transport.

A2.6.11.2.7 Fuel Spacer Thermal Analysis

Clearance between the fuel spacers and the guidesleeves is provided to allow for free expansion of the spacers. Therefore, there are no significant thermal stresses induced in the fuel spacers.

A2.6.11.3 Vibration

A2.6.11.3.1 Spacer Disc Vibration Analysis

The analysis presented in Section 2.6.11.3.1 is applicable to the 24PT1-DSC since the spacer disc configuration, loading and thermal conditions are bounded by the FO/FC-DSC analyses.

A2.6.11.3.2 Support Rod Vibration Stress Analysis

Vibration stresses in the support rod assemblies are calculated using the methodology described in Section A2.7.6.3 for lateral loads. Controlling stresses occur in the threaded section of the support rod and are dominated by the initial preload. The calculations use an enveloping value of 65 kips, Level A allowable stresses at 600°F, and reduced section properties at the threaded section. As shown below, the bending and tensile stresses are combined to determine the interaction ratio for vibration loading (note that the tensile stress is due to preload and the additional stress due to vibration, f_b, is very small):

Rod Interaction Ratio : Vibration

$$\frac{f_b}{F_b} + \frac{f_{tb}}{F_{tb}} \le 1.0$$

$$= \frac{.03\ ksi}{69.6\ ksi} + \frac{60.6\ ksi}{65.7\ ksi}$$

$$= 0.92$$

A2.6.11.3.3 Guidesleeve Vibration Stress Analysis

As described in Section A2.6.11.C, the guidesleeve assembly in the 24PT1-DSC is essentially the same as the FO-DSC guidesleeve assembly. Therefore, the results described in 2.6.11.3.3 are applicable.

A2.6.11.3.4 DSC Anti-Rotation Shear Key Analyses

The analysis provided in Section 2.6.11.3.4 for the FO/FC/FF-DSCs is applicable to the 24PT1-DSC since the payload weight is less than the FO/FC DSCs and the DSC grapple ring configuration is the same.

A2.6.11.3.5 Fuel Spacer Vibration Analysis

The vibration analysis of the fuel spacers conservatively applies the inertial weight of a fuel assembly to the spacer (2.5g in the axial direction). Under these loading conditions, stresses in the tube section of the fuel spacer are provided in Table A2.6-4.

Table A2.6-4

Fuel Spacer Vibration Stresses

Stress Category	Calculated Stress Intensity (ksi)	Allowable Stress Intensity (Level A @ 700°F) (ksi)	Stress Ratio (ksi)
P_m	1.31	16.0	.08
$P_m + P_b$	8.95	24.0	.37

A2.6.11.4 1-Foot Drop

A2.6.11.4.1 Spacer Disc 1-Foot Drop

See Section 2.6.11.4.1 for the FO/FC-DSC spacer disc analysis. This analysis bounds the 24PT1-DSC due to the enveloping loads and identical spacer disc geometry.

A2.6.11.4.2 Support Rod 1-Foot Drop Stress Analysis

As described in Section 2.6.11.4.2, an equivalent static load of 25.5g is used for the 24PT1-DSC support rod 1-foot drop analysis. The support rod assemblies are loaded in a manner similar to the horizontal dead weight analysis.

The support rod stresses are dominated by the initial preload, assuming an enveloping preload value of 65 kips, allowable stresses at 600°F, and reduced section properties at the threaded section. The bending and tensile stresses are combined to determine the interaction ratio for the 1 foot drop (note that the tensile stress is due to preload and the additional stress due to the drop, f_b, is very small):

Rod Interaction Ratio : 1 - Foot Drop

$$\frac{f_b}{F_b} + \frac{f_{tb}}{F_{tb}} \leq 1.0$$

$$= \frac{0.26\,ksi}{69.6\,ksi} + \frac{60.6\,ksi}{65.7\,ksi}$$

$$= 0.93$$

A2.6.11.4.3 Guidesleeve 1-Foot Drop

As described in Section A2.6.11.C, the guidesleeve assembly in the 24PT1-DSC is the same as the FO guidesleeve assembly. Therefore, the results described in Section 2.6.11.4.3 are applicable.

A2.6.11.5 Guidesleeve 1-Foot Drop Weld Evaluation

The 24PT1-DSC guidesleeves are fabricated using full penetration butt welds to join the tube material. Since the strength of these welds is equal to, or greater than, the strength of the base metal, no additional evaluation is required.

A2.6.11.6 Fuel Spacer 1-Foot Drop

Stresses in the fuel spacer for a 1-foot drop loading are calculated using a simple linear elastic ANSYS model. Using an inertial load of 25.5g, stresses in the tube section of the fuel spacer are presented in Table A2.6-5.

Table A2.6-5

Fuel Spacer 1-Foot Drop Stresses

Stress Category	Calculated Stress Intensity (ksi)	Allowable Stress Intensity (Level A @ 700°F) (ksi)	Stress Ratio
P_m	0.70	16.0	.04
$P_m + P_b$	6.67	24.0	.28

A2.6.12 Not Used

A2.6.13 Normal Conditions of Transport (NCT) Load Combination Evaluation

A2.6.13.1 Cask NCT Load Combinations

The load combinations provided in Section 2.6.13.1 bound the 24PT1-DSC payload due to the FO/FC/FF-DSCs bounding weight, cg and MOI.

A2.6.13.2 24PT1-DSC NCT Load Combination Evaluation

This section provides the load combination results for the support rods, guidesleeves, fuel spacers, and the 24PT1-DSC Failed Fuel Cans for the normal conditions of transport that are unique to the 24PT1-DSC. All other basket components are enveloped by the load combination results presented in Section 2.6.11 and are not repeated here.

A2.6.13.2.1 Spacer Disc Stress Analysis – Vibration Plus Thermal

Since the vibration and thermal stresses identified in Section 2.6 are bounding for the 24PT1-DSC, the load combinations provided in Section 2.6.13.2.1 are also bounding for the 24PT1-DSC.

A2.6.13.2.2 Support Rod Stress Analysis - Vibration Plus Thermal

The vibration stress analysis of the support rod assemblies described in Section A2.6.11.3.2 includes the enveloping preload of 65 kips and allowable stresses at 600°F. Since preload will decrease with temperature, the results described in Section A2.6.11.3.2 envelope the stresses from the vibration plus thermal load combination.

A2.6.13.2.3 Guidesleeve Stress Analysis - Vibration Plus Thermal

As noted in Section A2.6.11.2.6, the guidesleeves do not experience significant thermal stresses. As described in Section A2.6.11.3.3, vibration stresses calculated for the FO-DSC guidesleeves are applicable to the 24PT1-DSC guidesleeves. Therefore, the results described in Section 2.6.13.2.3 also apply to the 24PT1-DSC guidesleeves.

A2.6.13.2.A Fuel Spacer Stress Analysis - Vibration Plus Thermal

As described in Section A2.6.11.2.7, thermal stresses in the fuel spacers are negligible. The vibration loads described in Section A2.6.11.3.5 represent the combined vibration plus thermal stress for the fuel spacers.

A2.6.13.2.4 Spacer Disc Stress Analysis - 1-Foot Drop Plus Thermal

The analysis provided in Section 2.6.13.2.4 bounds the 24PT1-DSC configuration since the spacer disc loading for the FO/FC DSCs bounds the 24PT1-DSC.

A2.6.13.2.5 Support Rod Stress Analysis - 1-Foot Drop Plus Thermal

The 1-foot drop stress analysis of the support rod assemblies are described in Section A2.6.11.4.2. The analysis includes the enveloping preload of 65 kips and allowable stresses at 600°F. Since preload will decrease with temperature, the results described in Section A2.6.11.4.2 envelope the stresses from the 1-foot drop plus thermal load combination.

A2.6.13.2.6 Guidesleeve Stress Analysis - 1-Foot Drop Plus Thermal

As noted in Sections A2.6.11.2.5 and A2.6.11.2.6, the guidesleeves do not experience significant thermal stresses. As described in Section A2.6.11.3.3, vibration stresses for the FO-DSC guidesleeves are directly applicable to the 24PT1-DSC guidesleeves. Therefore, the results described in Section 2.6.13.2.6 are applicable to the 24PT1-DSC guidesleeves.

A2.6.13.2.7 Fuel Spacer Stress Analysis - 1-Foot Drop Plus Thermal

As described in Section A2.6.11.2.7 the thermal stresses in the fuel spacers are negligible. Therefore, the vibration loads described in Section A2.6.11.3.5 also provide the combined 1-foot drop plus thermal load combination results.

A2.7 Hypothetical Accident Conditions (HAC)

The NUHOMS®-MP187 Package loaded with a 24PT1-DSC and WE 14x14 spent fuel meets the performance requirements specified in Subpart E of 10CFR71 [A2.3] when subjected to hypothetical accident conditions as specified in 10CFR71.73.

Sections 2.7.1 through 2.7.5 provide details of the MP187 Cask and are unchanged by the addition of the 24PT1-DSC as the FO/FC-DSC Hypothetical Accident Condition loads bound the 24PT1-DSC payload. In addition, the conclusions described in the "Summary of Damage" presented in Section 2.7.6 envelope the 24PT1-DSC results and are directly applicable as written.

Sections 2.7.6.1 and 2.7.6.2 present the analyses and results for the FO/FC spacer disc drop loads and buckling evaluation. The accelerations, due to the regulatory drop loads, developed in Appendices 2.10.9 and 2.10.10 are unchanged by the addition of the 24PT1-DSC as the weights, mass moment of inertias, and center of gravity are bounded by the design values selected. The results apply to the 24PT1-DSC spacer disc as they have the same geometry as the FO/FC-DSC and the applied loads are less, as discussed in Section A2.6.11.A above. See Section A2.1 for a detailed comparison of the 24PT1-DSC support rods, guidesleeves and Failed Fuel Cans to those used in the FO/FC DSCs.

The changes in the 24PT1-DSC support rods and guidesleeves result in minor changes to the text presented in Sections 2.7.6.3 through 2.7.6.6; details are provided below. Since the guidesleeves are longer than the FO-DSC guidesleeves, extension angles are not required for the 24PT1-DSC to maintain the location of the poison sheets relative to the fuel. Therefore; the discussion presented in Section 2.7.6.7 is not applicable. Sections 2.7.6.8 and 2.7.6.9 bound the 24PT1-DSC oversleeve and guidesleeve weld. The 24PT1-DSC Failed Fuel Can HAC results are addressed below.

The 24PT1-DSC fuel spacers are provided for Normal Conditions of Transport and are not required to resist the HAC drop loads. To ensure that the 24PT1-DSC will remain sub-critical for all events, the guidesleeves extend the full length of the cavity and the neutron absorber extends the full length of the active fuel rods in the event that the fuel spacers are crushed to a length equal to the volume of material present in the fuel spacer (2.86 inches). This is a very conservative assumption as the crushed length of the square tubes is unlikely to be crushed to this extent by the drop decelerations.

A2.7.6 24PT1-DSC Analysis – Hypothetical Accident Conditions

A2.7.6.1 Spacer Disc Stress Analysis

The analysis provided in Section 2.7.6.1 bounds the 24PT1-DSC spacer discs since the FO/FC spacer disc loading bounds the 24PT1-DSC spacer disc loading and the geometry of the spacer discs is the same.

A2.7.6.2 Spacer Disc Stability Analysis

The analysis provided in Section 2.7.6.2 bounds the 24PT1-DSC spacer discs since the FO/FC spacer disc loading bounds the 24PT1-DSC spacer disc loading and the geometry of the spacer discs is the same.

A2.7.6.3 Support Rod Stress Analysis

The support rod assemblies are evaluated by the methodologies described below for lateral and axial load directions. All evaluations consider the initial preload and thermal stresses as appropriate.

A2.7.6.3.1 Evaluation for Lateral Loads

The support rod assemblies are evaluated for lateral loads (such as side drops) by modeling the assembly as a beam and using hand calculations. The critical location is the cantilever section at the top spacer disc. Stresses are calculated assuming enveloping maximum (65 kip) and minimum (0 kip) preload, evaluations based on enveloping preload to provide the controlling stresses. A conservative side drop (horizontal) load of 120g equivalent acceleration is used for the side drop evaluation. This conservatively bounds the FO/FC rod design acceleration discussed in Section 2.7.6.3.1.

A2.7.6.3.2 Evaluation for Axial Loads

Analysis for axial loading is based on a combination of linear elastic ANSYS analyses and hand calculations. A simple ANSYS beam model is used to determine the load distribution in the support rod system. The loads imposed on the model are inertial acceleration, and spacer disc loads and bending moments from spacer discs are conservatively assumed to be "fixed" at the support rod locations to maximum the support rod stresses. Hand calculations are used to calculate the stresses and to demonstrate the acceptability of the system using criteria from ASME, Section III, Subsection NF and Appendix F [A2.1].

Analyses were performed for the following load conditions/combinations (application of enveloping preload provides conservative results by maximizing the axial compression stresses in the spacer sleeves).

1. 65k Preload Only
2. 65k Preload + 600°F
3. 65k Preload + 600°F + 1.0g Axial Load
4. 65k Preload + 600°F + 2.5g Axial Load
5. 65k Preload + 600°F + 60.0g Axial (end drop) Load

Results shown in Figure A2.7-1 illustrate the load distribution under axial loads (results shown are from a 60g bottom end drop). From left to right, Figure A2.7-1 shows the axial compression loads in the spacer sleeves, the bending loads in the spacer sleeves, and the remaining axial tension in the support rod.

A2.7.6.3.3 Support Rod Assembly Analysis for Drop Loads

Stresses for the postulated drop conditions are calculated as described in Sections A2.7.6.3.1 and A2.7.6.3.2 above. For the postulated drops, the controlling stresses are in the spacer sleeves under loading from a 60g end drop. Results for the 60g bottom end drop (BED) are provided in Table A2.7-1. Load distribution for this analysis is shown in Figure A2.7-1.

Table A2.7-1

Spacer Sleeve Stresses - 60g Bottom End Drop

[Rods_BF1] 60g Bottom End Drop - Fixed, Level D at 600°F E:\Rods\[SCE_Rods1.xls]Sleeve_Stresses

Sleeve Number	Elements 1	Elements 2	Maximum Moment	Maximum Axial Load	Sleeve Stresses: Bending f_b	Sleeve Stresses: Axial, f_a	Equation 20: Term 1 f_a/F_a	Equation 20: Term 2 F'_e	Equation 20: Term 2 $C_m f_b/xF_b$	Equation 20: Result	Equation 21: Term 1 $f_a/x(.60S_y)$	Equation 21: Term 2 f_b/F_b	Equation 21: Result	Equation 22 Result
End Sections (Threaded, No Preload Effects)														
1	28	–	9.4 in-kip	194.4 kip	3.69 ksi	34.23 ksi	0.423	14819	0.0148	0.438	0.372	0.0246	0.397	0.448
27	29	–	0.0 in-kip	0.2 kip	0.00 ksi	0.04 ksi	0.001	5623	0.0000	0.001	0.000	0.0000	0.000	0.001
Preloaded Sections														
1	28	30	19.1 in-kip	238.3 kip	7.48 ksi	41.96 ksi	0.519	14819	0.0301	0.549	0.456	0.0500	0.506	0.569
2	31	32	20.9 in-kip	231.6 kip	8.21 ksi	40.78 ksi	0.510	8335	0.0331	0.543	0.443	0.0549	0.498	0.565
3	33	34	19.9 in-kip	224.4 kip	7.82 ksi	39.50 ksi	0.494	8335	0.0315	0.525	0.429	0.0523	0.482	0.546
4	35	36	20.2 in-kip	217.1 kip	7.92 ksi	38.23 ksi	0.478	8335	0.0319	0.510	0.416	0.0530	0.469	0.531
5	37	38	20.4 in-kip	209.9 kip	8.00 ksi	36.96 ksi	0.465	6586	0.0323	0.497	0.402	0.0535	0.455	0.518
6	39	40	19.9 in-kip	202.6 kip	7.80 ksi	35.68 ksi	0.453	4839	0.0315	0.484	0.388	0.0521	0.440	0.505
7	41	42	20.2 in-kip	195.3 kip	7.92 ksi	34.38 ksi	0.436	4839	0.0320	0.468	0.374	0.0529	0.427	0.489
8	43	44	20.0 in-kip	187.9 kip	7.83 ksi	33.09 ksi	0.421	4409	0.0316	0.453	0.360	0.0523	0.412	0.473
9	45	46	20.0 in-kip	180.5 kip	7.86 ksi	31.79 ksi	0.404	4409	0.0318	0.436	0.346	0.0526	0.398	0.457
10	47	48	20.0 in-kip	173.2 kip	7.85 ksi	30.49 ksi	0.388	4409	0.0317	0.420	0.331	0.0524	0.384	0.440
11	49	50	20.0 in-kip	165.8 kip	7.85 ksi	29.19 ksi	0.371	4409	0.0317	0.403	0.317	0.0525	0.370	0.424
12	51	52	20.0 in-kip	158.4 kip	7.85 ksi	27.89 ksi	0.355	4409	0.0317	0.387	0.303	0.0525	0.356	0.407
13	53	54	20.0 in-kip	151.0 kip	7.85 ksi	26.59 ksi	0.338	4409	0.0317	0.370	0.289	0.0525	0.341	0.391
14	55	56	20.0 in-kip	143.7 kip	7.85 ksi	25.30 ksi	0.322	4409	0.0317	0.353	0.275	0.0525	0.327	0.374
15	57	58	20.0 in-kip	136.3 kip	7.85 ksi	24.00 ksi	0.305	4409	0.0316	0.337	0.261	0.0525	0.313	0.358
16	59	60	20.0 in-kip	128.9 kip	7.85 ksi	22.70 ksi	0.289	4409	0.0316	0.320	0.247	0.0524	0.299	0.341
17	61	62	20.0 in-kip	121.5 kip	7.86 ksi	21.40 ksi	0.272	4409	0.0317	0.304	0.233	0.0525	0.285	0.325
18	63	64	20.0 in-kip	114.2 kip	7.83 ksi	20.10 ksi	0.256	4409	0.0316	0.287	0.218	0.0524	0.271	0.308
19	65	66	20.2 in-kip	106.8 kip	7.91 ksi	18.80 ksi	0.238	4839	0.0318	0.270	0.204	0.0528	0.257	0.291
20	67	68	20.0 in-kip	99.4 kip	7.84 ksi	17.51 ksi	0.222	4839	0.0316	0.254	0.190	0.0524	0.243	0.274
21	69	70	20.1 in-kip	92.1 kip	7.87 ksi	16.21 ksi	0.206	4839	0.0317	0.237	0.176	0.0526	0.229	0.258
22	71	72	19.9 in-kip	84.7 kip	7.82 ksi	14.92 ksi	0.189	4839	0.0315	0.221	0.162	0.0523	0.214	0.241
23	73	74	20.3 in-kip	77.4 kip	7.96 ksi	13.62 ksi	0.172	5911	0.0320	0.204	0.148	0.0532	0.201	0.225
24	75	76	20.4 in-kip	70.1 kip	7.99 ksi	12.34 ksi	0.156	5911	0.0321	0.188	0.134	0.0534	0.187	0.209
25	77	78	19.6 in-kip	62.8 kip	7.71 ksi	11.05 ksi	0.138	7819	0.0310	0.169	0.120	0.0515	0.172	0.190
26	79	80	40.0 in-kip	55.5 kip	15.68 ksi	9.78 ksi	0.122	8335	0.0630	0.185	0.106	0.1048	0.211	0.227
27	81	29	0.0 in-kip	48.3 kip	0.01 ksi	8.50 ksi	0.107	5623	0.0001	0.107	0.092	0.0001	0.093	0.107
									Maximum Interaction Values:	0.549			0.506	0.569

Note: Reference to Equations 20, 21 and 22 above is for Interaction Equations from ASME B&PV, Div. I, Sec. III, Subsection NF-3322.

Figure A2.7-1

Support Rod Load Distribution - 65k Preload + Thermal + 60g BED

A2.7.6.4 Support Rod Stability Analysis

Stability of the support rod assemblies is demonstrated by the results described in Table A2.7-1. No additional evaluation is required.

A2.7.6.5 Guidesleeve Stress Analysis

As described in Section A2.6.11.C, the guidesleeve assembly in the 24PT1-DSC is the same as the FO-DSC guidesleeve assembly. Therefore, the results described in Section 2.7.6.5 are directly applicable to the 24PT1-DSC guidesleeves

A2.7.6.6 Guidesleeve Stability Analysis

As described in Section A2.6.11.C, the guidesleeve assembly in the 24PT1-DSC is the same as the FO-DSC guidesleeve assembly. Therefore, the results described in Section 2.7.6.6 are directly applicable to the 24PT1-DSC guidesleeves.

A2.7.6.7 Not Used

A2.7.6.8 Oversleeve Structural Evaluation

A2.7.6.8.A Oversleeve to Sleeve Welds

The oversleeve to guidesleeve connection in the 24PT1-DSC utilizes plug welds as described in Section 2.7.6.8. Fillet welds of equivalent strength may be substituted with no additional analysis required.

A2.7.6.8.B Oversleeves

The 24PT1-DSC oversleeve is the same as the FO-DSC oversleeve. Therefore, the results in Section 2.7.6.8 are directly applicable to the 24PT1-DSC oversleeves.

A2.7.6.9 Guidesleeve 30 Foot Drop Weld Evaluation

As shown in the drawings in Section A1.3.2, the 24PT1-DSC guidesleeves are fabricated with full penetration butt welds. Since the strength of these welds is equal to, or greater than, the strength of the base metal, no additional evaluation is required.

A2.7.7 Not Used

A2.7.8 HAC Load Combination Evaluation

The cask load combinations presented in Section 2.7.8 for the FO/FC-DSC payloads bound the 24PT1-DSC payload.

A2.8 Special Form

Special form materials are not part of the payloads considered for the MP187 Cask with 24PT1-DSC payload.

A2.9 Fuel Rods

As discussed in Chapter A4, containment of radioactive materials is provided by the cask containment boundary. Analyses of the cask containment boundary for all normal conditions of transport and hypothetical accident conditions defined by the Part 71 Regulations [A2.3] demonstrated that the cask remains leak tight while transporting a 24PT1-DSC payload.

A2.10 Appendix

A2.10.1 References

A2.10.2 NUHOMS®-MP187 Cask Analyses

A2.10.3 24PT1-DSC Basket Analyses

A2.10.1 References

A2.1 ASME Boiler and Pressure Vessel Codes, 1992 Edition with addenda through 1994, including Code Case N-595-1

A2.2 Transnuclear West Safety Analysis Report for the Standardized Advanced NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, ANUH-01.0150, Revision 0

A2.3 10CFR71, "Packaging and Transportation of Radioactive Material"

A2.10.2 NUHOMS®-MP187 Cask Analyses

Appendices 2.10.2, 2.10.5, 2.10.6, 2.10.7, 2.10.8, 2.10.9, 2.10.10, 2.10.11 and 2.10.12 for the MP187 Cask are unchanged as the FO/FC-DSC design basis loads bound the 24PT1-DSC loaded with WE 14x14 fuel.

A2.10.3 24PT1-DSC Basket Analyses

Sections of Appendices 2.10.3 and 2.10.11 are applicable to the 24PT1-DSC as referenced (indirectly via references to Sections 2.6 and 2.7) in Sections A2.6 and A2.7.

A3. THERMAL EVALUATION

The purpose of the thermal analyses presented herein is to demonstrate that the NUHOMS®-MP187 Cask with a 24PT1-DSC payload provides suitable heat dissipation under the 10CFR71 [A3.2] normal conditions of transport and hypothetical accident conditions.

A3.1 Discussion

The 24PT1-DSC contains up to 24 WE 14x14 PWR fuel assemblies (See Section A1.2.3.1 for fuel assembly and non-fuel assembly hardware, NFAH, data). The maximum design basis decay heat utilized in this analysis, which is the maximum heat load allowed per 24PT1-DSC including control components, is 14 kW.

See Section A3.4 for a discussion of heat loads and results, including a comparison to the analysis provided in Chapter 3.

A3.2 Summary of Thermal Properties of Materials

As discussed in Chapter A2, the differences in 24PT1-DSC configuration and materials relative to the FO/FC-DSCs are minor. The use of SA 240 Type 316 for 24PT1-DSC shell and cover plates in lieu of SA 240 Type 304, used for the FO/FC-DSC, has a negligible effect on thermal analysis results since the key thermal parameters (thermal conductivity, specific heat and thermal expansion coefficient) are nearly identical. Any non-conservative differences in these properties are enveloped by the conservatism in the FO/FC heat load with respect to the 24PT1-DSC heat load, as discussed in Section A3.4.1.A.

A3.3 Technical Specifications of Components

The discussion provided in Section 3.3 is applicable to the MP187 with a 24PT1-DSC payload.

A3.4 Thermal Evaluation for Normal Conditions of Transport

A3.4.1 Thermal Model

A3.4.1.A MP187 Cask and DSC Shell Thermal Analysis Heat Load

The thermal analysis of the MP187 Cask and FO/FC-DSC shell assembly presented in Chapter 3 uses a design decay heat load of 13.5 kW multiplied by a uniformly applied peaking factor of 1.2. The DSC basket analysis conservatively models one section of the DSC basket (between the center line of one spacer disc and the next, including all basket, shell and cask components within this cross section of the MP187 package) and ignores all axial heat transfer. The total applied heat load used in the DSC and cask thermal analyses is therefore 16.2 kW (13.5 x 1.2).

The WE 14x14 fuel design decay heat load is 14.0 kW. Section 5.4.1.1 of [A3.1] demonstrates that the WE 14x14 fuel has a maximum peaking factor of 1.07. Based on this data, this SAR evaluation uses a peaking factor of 1.08 resulting in a total applied heat load of 15.1 kW (14 x 1.08) for DSC and cask thermal analyses with WE 14x14 fuel assembly payload. Therefore, the 24PT1-DSC total applied heat load of 15.1 kW is bounded by the 16.2 kW total applied heat load for the FO/FC-DSC analyses in Chapter 3 and reanalysis of the cask and DSC shell assembly is not needed to qualify the 24PT1-DSC.

A3.4.1.B DSC Basket Thermal Analysis Heat Load

As discussed above, the WE 14x14 payload total applied heat load (15.1 kW) is bounded by the B&W 15x15 payload total applied heat load (16.2 kW) developed in Chapter 3 of this SAR.

The volumetric heat density for the WE 14x14 fuel assembly is calculated as follows:

$$Q = \frac{((14\text{ kW}/24) \times 1.08) \times (3412\text{ Btu/hr/kW}) \times (1\text{ hr} / 60\text{ min})}{(8.9)^2\text{ in}^2 \times 120\text{ in}}$$

$$Q = 0.0037\text{ Btu/min-in}^3$$

Where 8.9 in. is the inner width of the square guidesleeve and 120 in. is the active fuel length.

This volumetric heat density is bounded by the B&W 15x15 fuel heat density of 0.004642 Btu/min-in^3 used in Section 3.6.4 for the FO/FC-DSC basket thermal analysis. Therefore, the basket thermal analysis in Chapter 3 bounds the 24PT1-DSC payload.

A3.4.1.C WE 14x14 Fuel - Fuel Clad Temperature

The short term fuel cladding temperature limit for the WE 14x14 MOX (Mixed Oxide, zircalloy clad fuel) is the same as that for the B&W zircalloy fuel (1058° F) provided in Chapter 3. The short term fuel cladding temperature limit for the WE 14x14 stainless steel clad (SC) fuel is higher than that of the zircalloy fuel as discussed in [A3.1], Section 3.5.1. The acceptance criteria for Zircalloy fuel clad temperature calculated in Chapter 3 for the B&W 15x15 fuel bounds that of the stainless clad WE 14x14 fuel. The 669° F maximum normal conditions of transport fuel clad temperature and the 790° F maximum hypothetical accident fuel clad temperature calculated in Chapter 3 bound the WE 14x14 fuel clad temperature since the DSC heat load for the WE 14x14 fuel (including peaking factor) is less than the B&W 15x15 fuel, as discussed above. Therefore, no further analysis is required to justify the WE 14x14 fuel temperatures.

A3.4.1.D Thermal Analysis Conclusions

The thermal analysis results for fuel clad temperatures and for all Package components (used in structural thermal stress analyses) with the FO/FC-DSC payload bound the MP187 package with a 24PT1-DSC payload.

A3.4.1.E Pressure Analysis

See Appendix A3.6.2.

A3.5 Hypothetical Accident Thermal Evaluation

As discussed in Section A3.4, the thermal analysis for the FO/FC-DSC in Chapter 3 is bounding for the 24PT1-DSC. See Appendix A3.6.2 for the 24PT1-DSC pressure analyses.

A3.6 Appendix

A3.6.1 References

A3.6.2 24PT1-DSC Pressure Analysis

A3.6.1 References

A3.1 Standardized Advanced NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Safety Analysis Report, ANUH-01.0150, Revision 0, NRC Docket 72-1029

A3.2 Code of Federal Regulations, Title 10, Part 71, Packaging and Transportation of Radioactive Material

A3.6.2 Pressure Analysis

This section demonstrates that the 24PT1-DSC/MP187 Cask internal pressure for normal conditions of transport and hypothetical accident conditions when transporting WE 14x14 fuel assemblies is bounded by the FO/FC-DSC containing B&W 15x15 fuel assemblies.

A3.6.2.1 DSC/Cask Annulus Volume

The DSC/cask annulus volume and associated operating pressure is the same as that for the FO/FC-DSC configuration since the DSC outside envelope dimensions are the same and the cask/DSC annular gap helium pressure applied (Chapter 7 and A7) is the same.

A3.6.2.2 24PT1-DSC Free Volume

Since the same components are used in the FO/FC and 24PT1-DSCs, the DSC basket volume calculated in Section 3.6.6.2.1.(B) is applicable to the 24PT1-DSC basket (69,519 in^3; volume of spacer discs, 38,194 in^3; plus volume of support rods, 4,708 in^3; plus volume of the guidesleeve assemblies, 26,617 in^3); this is the volume of the spacer discs, support rods, support rod sleeves, guidesleeves, poison and poison oversleeves). The total 24PT1-DSC internal volume is the same as the FO-DSC shown in Section 3.6.6.2.1.(B) (570,301 in^3). The volume of the WE 14x14 fuel assemblies (24 assemblies, excluding control components, to maximize fill gas quantity) is 81,535 in^3. This volume is calculated as follows:

$$V_{fuel\ assy} = n_{fr}\,[\,V_{fuel\ rod}\,] + V_{hardware}$$

$$= 3{,}397 \quad in.^3$$

where

n_{fr} = number of fuel rods per assembly

= 180

and where $V_{fuel\ rod}$ and $V_{hardware}$ are calculated below.

$$V_{fuel\ rod} = \pi\ (OD/2)^2 \times L$$

$$= 17.64 \quad in.^3$$

where

$OD = 0.422$ in.

$L = 126.13$ in.

$$V_{hardware} = \Sigma\ [\ W_{hardware,i} / \gamma_{hardware,i}\ ]$$

$$= 221.84\ in.^3$$

$$V_{24\ FAs} = n_{fa}\ V_{fuel\ assy}$$

$$= 81{,}535 \quad in.^3$$

where

n_{fa} = number of fuel assemblies

= 24

Table A3.6-1

Component Weight, Material, and Volume for a Single Fuel Assembly

Component	Weight (W)		Material	γ	Volume W/ γ
	(kg)	(lbs.)		(lbs/in³)	(in³)
Bottom Nozzle			SS 304	0.285	
Guide Tubes			SS 304	0.285	
Holddown spring			Inconel 718	0.296	
Spacer-Plenum			Inconel 718	0.296	
Spacer-Incore			Inconel 718	0.296	
Top Nozzle			SS 304	0.285	
TOTAL	hardware =	63.59		$V_{hardware}$ =	221.84

The volume of the 48 fuel spacers used in the 24PT1-DSC is 48,528 in³. Reducing the free volume by the volume of the four 24PT1-DSC Failed Fuel Cans and 24 RCCAs yields the largest DSC pressure, and is therefore bounding. The volume of a Failed Fuel Can is 908 in³. The volume of 24 RCCAs is 7803 in³.

The total free volume for the 24PT1-DSC is therefore:

$$V_{24PT1\text{-}DSC} = (570{,}301 - 69{,}519 - 81{,}535 - 48{,}528 - (4 \times 908) - 7803) = 359{,}284 \text{ in}^3$$

A3.6.2.3 Quantity of Helium Fill Gas in 24PT1-DSC Cavity

The resulting fill gas volume, based on the initial maximum DSC fill pressure of 3 psig, and conservatively using the same method and helium gas temperature used in Section 3.6.6.2.2.(A) is:

$$n_{He} = \frac{(3 + 14.7\,\text{psia})(6894.8\,\text{Pa/psi})(359{,}284\,\text{in}^3)(1.6387\,\text{x}\,10^{-5}\,\text{m}^3/\text{in}^3)}{(8.314\,\text{J/mol}-\text{K})(400°\text{F} + 460°\text{R})(5/9\,\text{K/R})}$$
$$= 181 \text{ moles}$$

A3.6.2.4 Quantity of Helium Fill Gas in WE 14x14 Fuel Rods

The fuel rod fill gas is calculated below:

$$n_{He} = \frac{(315 + 14.7\,\text{psia})(6894.8\,\text{Pa/psi})(24\,\text{x}\,180\,\text{x}\,1.22\,\text{in}^3)(1.6387\,\text{x}\,10^{-5}\,\text{m}^3/\text{in}^3)}{(8.314\,\text{J/mol}-\text{K})(80°\text{F} + 460°\text{R})(5/9\,\text{K/R})}$$
$$= 79 \text{ moles} \quad (\text{use } 80 \text{ moles})$$

This is based on the fuel assembly data presented in **Table A3.6-2** below:

Table A3.6-2

Fuel Rod Data

Number of Fuel Rods per Assembly	180

A3.6.2.5 Quantity of Fission Product Gases in the Fuel Rods

The fission gas inventory for the WE 14x14 SC fuel is presented in Table A3.6-3 below:

Table A3.6-3

Fission Gas Inventory per Assembly for WE 14x14 Fuel

Isotope	#Moles
Ar 36	9.94E-08
Ar 38	9.24E-06
Ar 39	3.10E-08
Ar 40	5.98E-08
H 1	2.45E-01
H 2	3.89E-04
H 3	8.62E-03
H 4	6.62E-01
I127	1.69E-01
I129	7.04E-01
Kr 80	3.68E-07
Kr 81	3.52E-08
Kr 82	4.95E-03
Kr 83	2.30E-01
Kr 84	6.67E-01
Kr 85	6.69E-02
Kr 86	1.02E+00
Ne 20	1.26E-05
Ne 21	4.95E-07
Ne 22	7.36E-07
Xe128	1.41E-02
Xe129	1.05E-04
Xe130	3.33E-02
Xe131	1.39E+00
Xe132	4.36E+00
Xe134	5.60E+00
Xe136	8.46E+00
Total	**2.36E+01**

The total number of fission gas moles (assuming 30% of fission gases are available for release) contributing to 24PT1-DSC cavity pressure is therefore;

$$n_{fg} = 23.6\,moles \times 24 \times 0.3 = 170\,moles$$

The stainless steel clad (SC) fuel fission gas quantity bounds the MOX fuel due to the significantly lower burnup of the MOX fuel (25,000 MWd/MTU for MOX compared to 45,000

MWd/MTU for the SC fuel). Therefore, it is conservative to use the total gas moles for the WE 14x14 SC fuel throughout this calculation.

A3.6.2.6 Quantity of Gas Contained in Control Components

The only control component for the WE 14x14 fuel that contains internal pressure is the neutron source assembly (NSA). The data contained in Table A3.6-4 is used to calculate the quantity of gas in the NSAs.

Table A3.6-4

Control Component Data

The neutron source gas moles are calculated below.

$n_{He,ns} =$

$= 2.45$ moles

A3.6.2.7 24PT1-DSC in MP187 Cask Pressure Analysis

A3.6.2.7.1 Normal Conditions of Transport

The Helium average temperature for normal conditions of transport from Table 3.6-6, Case 2 is 351°F for the DSC/cask annulus. Using the methodology of Section 3.6.6.2.2.(D), and based on the same DSC/cask annulus volume (19,517 in^3) and the same moles of gas in the annulus (13 moles) and the gas totals calculated above, the maximum normal condition of transport pressure with fuel rods and DSC pressure boundary failed is:

$$P_{nor} = \frac{(1.4504\,x\,10^{-4})(181+80+170+2.45+13\,\text{mol})(8.314\,\text{J/mol}-\text{K})(351+460°\text{R})}{(359{,}812+19{,}517\,\text{in}^3)(1.6387\,x\,10^{-5}\,\text{m}^3/\text{in}^3)(1.8\,\text{R/K})}$$

$$P_{nor} = 39\,psia\,(24.3\,psig)$$

A3.6.2.7.2 Hypothetical Accident Conditions

For the hypothetical accident condition (bounding case of post fire steady state condition) the Helium average temperature is 536°F for the DSC/cask annulus (Table 3.6-7). The maximum accident condition pressure is:

$$P_{acc} = \frac{(1.4504\,x\,10^{-4})(181+80+170+2.45+13\,\text{mol})(8.314\,\text{J/mol}-\text{K})(536+460°\text{R})}{(359{,}812+19{,}517\,\text{in}^3)(1.6387\,x\,10^{-5}\,\text{m}^3/\text{in}^3)(1.8\,\text{R/K})}$$

$$P_{acc} = 47.9\,psia\,(33.2\,psig)$$

A3.6.2.7.3 Comparison of 24PT1-DSC Payload Versus FO/FC-DSC Payload Pressures

A summary of the applicable 24PT1-DSC and FO/FC-DSC pressure analysis results is provided in Table A3.6-5 below. The data provided in this table demonstrates that the 24PT1-DSC pressure is bounded by the FO/FC-DSC analysis. Therefore, the pressure used in the cask structural analyses provided in Chapter 2 of this SAR bounds the 24PT1-DSC payload pressures.

Table A3.6-5

Design Basis Thermal Analysis Results for MP187 Package Pressure

DSC	Maximum Parameter Value	NOC, Intact Fuel Rods	NOC, All Fuel Rods Ruptured	HAC, Final Steady State, All Fuel Rods Ruptured	MP187 Cask Pressure Limit (pressure used in structural analysis)
FO/FC	Cask Internal Pressure (psia)	22.5	46.2	56.7	64.7
24PT1	Cask Internal Pressure (psia)	22.5	39.0	47.9	64.7

A4. CONTAINMENT

The containment boundary for the MP187 Package includes the cask inner shell, bottom forging, the ram closure plate port, the top end forging, top closure plate, the vent and drain ports and their associated seals.

The containment boundary is not impacted by the addition of the 24PT1-DSC to the MP187 Package since the 24PT1-DSC configuration, materials, weight and center of gravity are bounded by the FO/FC/FF-DSCs. Therefore, the analysis provided in Chapter 4 of the SAR is applicable to the 24PT1-DSC.

The spent fuel contents of the 24PT1-DSC are described in Chapter A1.

A5. SHIELDING EVALUATION

The shielding analysis of the NUHOMS®-MP187 Cask with a 24PT1-DSC payload (WE 14x14 stainless steel clad (SC) UO_2 fuel, and WE 14x14 zirconium clad mixed oxide fuel (MOX) is evaluated in this chapter. The configuration of the 24PT1-DSC is described in Chapter A1. The following demonstrates that the shielding analysis for the MP187 Cask with a 24PT1-DSC (WE 14x14 fuel) payload meets all 10CFR71 [A5.1] shielding criteria. The design basis fuel parameters to be evaluated are specified below (data taken from Chapter A1):

Table A5-1

Design Basis Fuel Parameters

Fuel Type	Maximum Enrichment (weight %)	Minimum Enrichment (weight %)	Maximum Burnup (MWd/ MTU)	Minimum Cooling Time / Max Heat Load per cask / Max Assembly Heat Load (incl. control components[1])
WE 14x14 SC (may include Integral Fuel Burnable Absorber, Boron Coated Fuel Pellets)	4.0 U-235	3.80 U-235	45,000	38 years/14 kW/ 0.583 kW
		3.40 U-235	40,000	
		3.16 U-235	35,000	
WE 14x14 MOX	0.71 U-235 3.31 fissile Pu	• 2.78 Pu (64 rods) • 3.05 Pu (92 rods) • 3.25 Pu (24 rods)	25,000	20 years/14 kW/ 0.583 kW

1 – Control component cooling time is a minimum of 10 years

The analysis in this chapter compares the source term for 10 year cooled SC fuel with each of the burnup and minimum enrichments specified above. These source terms are compared to define the bounding SC fuel burnup/enrichment combination (the bounding configuration is 3.80 weight % U-235, with 45,000 MWd/MTU burnup). The bounding SC fuel is decayed to 38 years and compared to the MOX fuel configuration specified above. The bounding SC, 38 year cooled fuel configuration, is used as the design basis source term.

The shielding analysis is performed for the minimum enrichments specified in Table A5-1 to bound all fuel assemblies with enrichments up to the maximum enrichments specified in Table A A5-1.

The following section demonstrates that the MP187 Transportation Packaging (cask with impact limiters) provides adequate shielding to ensure compliance with the external dose rate requirements specified in 10CFR Part 71.47 [A5.1], 10CFR Part 71.51 [A5.1], and 49CFR Part 173.441 [A5.2] for the fuel parameters stated above in Table A5-1. As stated in Chapter 7, the dose rates surrounding the package are measured prior to each shipment and verified to be in compliance with 10CFR Part 71.47 and 49CFR173.441. The MP187 shielding evaluation results presented in Table A5.1-1 show that the predicted dose rates with the least margin relative to the regulatory limits are those for normal conditions of transport. As shown in this chapter, the MP187 Package meets the dose rate limits of 10CFR Part 71.47 and 49CFR173.441 for normal conditions of transport and the limits of 10CFR71.51 after sustaining damage from the hypothetical accident conditions. The shielding evaluation presented herein demonstrates that the package meets the regulatory requirements for normal conditions of transport and hypothetical accident conditions.

A5.1 Discussion and Results

The Packaging provides neutron and gamma-ray radiation shielding for up to 24 PWR spent fuel assemblies. Gamma-ray shielding is provided by lead and stainless steel shells that make up the cask wall. Neutron shielding is provided by a cementitious castable material contained within a stainless steel jacket surrounding the cylindrical portion of the cask body. Gamma shielding in the cask ends is provided by stainless steel and the top and bottom end assemblies of the DSC.

The 24PT1-DSC design basis payload neutron and gamma-ray sources are determined to be 2.460×10^9 neutrons/sec/DSC and 3.444×10^{16} γ/sec/DSC, respectively, as derived in Section A5.2.2 (neutron) and Section A5.2.1.4 (gamma).

The NUHOMS®-MP187 package is transported by exclusive use shipment, inside a closed transport vehicle (enclosure provided by the personnel barrier). The applicable 10CFR Part 71.47 external radiation requirements for normal conditions during shipment include [A5.17]:

1. Radiation levels must not exceed 10 mSv/hr (1000 mrem/hr) on the external surface of the package.

2. Radiation levels must not exceed 2 mSv/hr (200 mrem/hr) at any point on the outer surface of the transport vehicle.

3. Radiation levels must not exceed 0.1 mSv/hr (10 mrem/hr) at any point two meters from the outer lateral surfaces of the vehicle or, in the case of a flat-bed style vehicle, at any point two meters from the vertical planes projected by the outer edges of the vehicle and on the lower external surface of the vehicle.

4. Radiation levels must not exceed 0.02 mSv/hr (2 mrem/hr) in any normally occupied space.

In the shielding evaluations, the external surface of the packaging is defined as the radial surface of the cask neutron shield panel and the surfaces of the impact limiters. The outer surface of the vehicle is bounded by the impact limiters and the personnel barrier, which is mounted on the skid and extends to the same radius as the impact limiters as shown on the drawings provided in Section 1.3.2. The 10CFR Part 71.51 external radiation requirements after the 10CFR Part 71.73 hypothetical accident conditions are that there shall be [A5.1],

> ". . . no external radiation dose rate exceeding 10 mSv/h (1 rem/hr) at one meter from the external surface of the package."

During the hypothetical accident conditions defined in 10CFR Part 71.73, the cask is assumed to be separated from the skid. The cask neutron shield and neutron shield jacket are assumed to be lost during the accident, as are the impact limiters. The cask lead shielding is assumed to have slumped, producing relatively unshielded gaps in the cask wall. The suitability of the cask cavity closure during the hypothetical accident conditions is documented in Chapters 2 and 4.

The results of the shielding evaluation performed for the NUHOMS®-MP187 Package with a loaded 24PT1-DSC are summarized in Table A5.1-1. The basis for the values reported in Table A5.1-1 is described in Section A5.4.2. The bounding dose rate for the MP187 Package is the two meter dose rate along the package side. The dose rate determined for this bounding dose rate is 8.28 mrem per hour (10CFR71.47 limit is 10 mrem per hour). The equivalent design basis FO/FC DSC dose rates are documented in Chapter 5, Table 5.1-1. The bounding 24PT1-DSC dose rate (normal conditions of transport, 2 meter side dose) is bounded by the respective FO/FC-DSC dose rate. The analysis performed is very conservative assuring that actual doses will be significantly below these calculated value. Significant conservatisms applied in this analysis include:

- Impact limiter thickness assumed equal to the minimum thickness with no credit taken for the actual square cross-section.
- Worst case nuclear data used for source term calculations.

- Source terms used in shielding analysis are based on the design basis fuel assembly which exceeds the maximum allowable heat load of 14 kW.
- No credit taken for self shielding of the control components.
- Conservative neutron shield properties used including a 10% reduction in hydrogen and a 50% reduction in boron.
- The methodology used to perform the shielding calculations has been thoroughly benchmarked against measured data and shown to provide significantly conservative (factor of two or more) results as documented in Reference [A5.15].
- The calculated fuel source term peaking factor is conservatively applied to the entire length of the active fuel.

Because the dose rates at the top and bottom ends of the Package are less than 2 mrem/hr, the radiation level in occupied positions of the vehicle will also be less than 2 mrem/hr. The MP187 Package meets all of the applicable external radiation criteria as documented in the remainder of this chapter. All Package dose rates are measured immediately prior to shipment and the Package will not be shipped if the 10CFR71 limits are exceeded.

Table A5.1-1

Summary of Maximum Dose Rates

(mrem/hr)

24PT1-DSC

	Package Surface		
	Side (mrem/hr)	Top (mrem/hr)	Bottom (mrem/hr)
Normal Conditions			
Gamma	2.12E+01	2.61E-01	3.86E-01
Neutron	1.46E+02	9.52E-01	1.42E+00
Total*	1.49E+02	1.19E+00	1.63E+00
10CFR71 Limit	1.00E+03	1.00E+03	1.00E+03
	Vehicle Outer Surface		
	Side (mrem/hr)	Top (mrem/hr)	Bottom (mrem/hr)
Normal Conditions			
Gamma	8.53E+00	2.61E-01	3.86E-01
Neutron	5.51E+01	9.52E-01	1.42E+00
Total*	5.75E+01	1.19E+00	1.63E+00
10CFR71 Limit	2.00E+02	2.00E+02	2.00E+02
	2 Meters From Vehicle Outer Surface		
	Side (mrem/hr)	Top (mrem/hr)	Bottom (mrem/hr)
Normal Conditions			
Gamma	1.60E+00	9.59E-02	1.29E-01
Neutron	7.43E+00	4.14E-01	5.93E-01
Total*	8.28E+00	5.02E-01	6.77E-01
10CFR71 Limit	1.00E+01	1.00E+01	1.00E+01
	1 Meter From Surface of Package		
	Side (mrem/hr)	Top (mrem/hr)	Bottom (mrem/hr)
Hypothetical Accident Conditions			
Gamma	1.4E+ 1	3.1E- 1	5.5E- 1
Neutron	2.7E+ 2	2.0E+ 1	6.2E+ 1
Total*	2.8E+ 2	2.0E+ 1	6.2E+ 1
10CFR71 Limit	1.00E+3	1.00E+ 3	1.00E+3

* Peak Gamma and Neutron doses do not necessarily occur at the same location. Therefore, the maximum total dose rate is not necessarily the sum of the maximum gamma and neutron doses.

A5.2 Source Specification

The neutron and gamma-ray radiological source strengths and the gamma-ray source spectrum for the design basis fuel assembly are determined using the SAS2H (ORIGEN-S) computer code in the SCALE 4.4 package [A5.3]. The design basis 24PT1-DSC payload neutron and gamma-ray sources are 2.460×10^9 neutrons/sec/DSC and 3.444×10^{16} γ/sec/DSC, respectively. The fuel parameters used to develop the design basis source are taken from Section A1.2.3. These parameters include a maximum burnup of 45,000 MWd/MTU and a minimum cooling time of 38 years for WE 14x14 SC fuel and 25,000 MWd/MTU and minimum cooling time of 20 years for WE 14x14 MOX fuel. The fuel assembly heat load is limited to 14 kW total per DSC and 0.583 kW per assembly. The WE 14x14 design basis fuel assembly source term was chosen by comparing the three SC fuel assembly configurations specified in Chapter A1 (35, 40 and 45 GWd/MTU) and the MOX fuel assemblies, along with the three control components allowed: rod cluster control assembly (RCCA), thimble plug (TP); and neutron source assembly (NSA). A 10 year cooling time was conservatively assumed for the control components and fuel assembly end fittings (outside core region of the fuel assembly). The largest source for each of the fuel and control components for each gamma and neutron energy group was selected to generate the design basis source term.

The SAS2H/ORIGEN-S [A5.3] is used to calculate the gamma and neutron sources for the design basis fuel assemblies. The design basis source term for the NUHOMS®-MP187 Cask shielding analysis with a 24PT1-DSC payload is represented by the combination of fuel and control components (non-fuel assembly hardware, NFAH), with the neutron and gamma source terms described below. SAS2H/ORIGEN-S maps the results directly to the results CASK-81 [A5.4] energy structure for the external dose rate calculations.

For the acceptable fuel parameters defined in Section A1.2.3, the source term calculations described below determine bounding neutron and gamma-ray sources.

Acceptable fuel burnups range from 0 MWd/MTU to 45,000 MWd/MTU. Acceptable fuel cooling times are greater than or equal to 38 years for SC fuel assemblies and greater than or equal to 20 years for MOX fuel assemblies with burnup limited to 25,000 MWd/MTU. The initial enrichments of the fuel assemblies vary according to the burnup of the assembly. The choice of initial enrichment for each case has a significant effect on the neutron source (lower enrichments produce greater sources). The minimum enrichments were therefore used for these analyses.

Eight SAS2H/ORIGEN-S cases are run to determine the bounding fuel assembly and control component parameters permitted as a payload for the 24PT1-DSC (per Chapter A1). These cases are:

1. SC fuel – 35 GWd/MTU, 3.16 U-235 minimum enrichment (wt. %), 10 year cooled
2. SC fuel – 40 GWd/MTU, 3.40 U-235 minimum enrichment (wt. %), 10 year cooled
3. SC fuel – 45 GWd/MTU, 3.80 U-235 minimum enrichment (wt. %), 10 year cooled
4. SC fuel – 45 GWd/MTU, 3.80 U-235 minimum enrichment (wt. %), 38 year cooled
5. MOX fuel – 25 GWd/MTU, with minimum enrichments as specified in Chapter A1, 20 year cooled
6. Rod Cluster Control Assembly (RCCA) NFAH – 10 year cooled
7. Thimble Plug (TP) NFAH – 10 year cooled
8. Neutron Source Assembly (NSA) NFAH – 10 year cooled

These cases represent the minimum cooling time and maximum burnup specifications for each fuel/NFAH combination. The gamma and neutron sources for each case are extracted from the ORIGEN-S output and used to develop the design basis sources discussed below. A summary of the SAS2H/ORIGEN-S cases is provided in Table A5.2-1.

Table A5.2-1

SAS2H/ORIGEN-S Input Cases for Shielding Evaluation

Case	Component	Burnup (MWd/MTU)	Initial Enrichment (wt. % ^{235}U)	Required Cooling Time (years)
I	SC fuel	35,000	3.16	10
II	SC fuel	40,000	3.40	10
III	SC fuel	45,000	3.80	10
IV	SC fuel	45,000	3.80	38
V	MOX fuel	25,000	.71 for U-235, Pu initial enrichment: 2.78% - 64 rods 3.05% - 92 rods 3.25% - 24 rods	20
VI	RCCA	165,000 (associated fuel assembly burnup over life of NFAH)	3.80 for associated fuel assembly	10
VII	TP	165,000 (associated fuel assembly burnup over life of NFAH)	3.80 for associated fuel assembly	10
VIII	NSA	60,000 (associated fuel assembly burnup over life of NFAH)	3.80 for associated fuel assembly	10

A5.2.1 Gamma Source

SAS2H/ORIGEN-S [A5.3] is used to calculate both the neutron and gamma-ray source terms for the NUHOMS®-MP187 Package. A description of the fuel assembly models and data reduction is provided in this section. Data reduction that applies specifically to the neutron source is described in Section A5.2.2. The ORIGEN-S code computes the radioactivity of fuel assemblies that have undergone irradiation in a nuclear reactor and subsequent decay after removal from the reactor core. It has the ability to compute the isotopic fractions, radioactivity, decay thermal power, toxicity, neutron absorption, neutron emission, and photon emission for various isotopes in the fuel assembly. SAS2H/ORIGEN-S results are used in spent fuel shipping package analyses to develop neutron and gamma ray radioactive decay source strengths and to develop decay thermal powers. SCALE 4.4 which contains the SAS2H/ORIGEN-S modules is an industry standard code that is distributed by Oak Ridge National Laboratory's Radiation Shielding Information Center (ORNL/RSIC).

A5.2.1.1 Description of Fuel Assembly ORIGEN Models

The fuel assembly material weights used in the SAS2H/ORIGEN-S models are shown in Table A5.2-2. The detailed material compositions and impurities are given in the sample input files reproduced in Section A5.5.2. The fuel assembly SAS2H/ORIGEN-S model is split into four distinct regions: active fuel, gas plenum, top nozzle, and bottom nozzle. Additional SAS2H/ORIGEN-S runs are made to calculate the activation source term in the design basis control components.

The power irradiation in SAS2H/ORIGEN-S is performed for the fuel assembly materials in the active fuel region. The light element material compositions are modified by the flux factors specified in Table A5.2-3. The fractions of the in-core flux that are applied to the top nozzle, plenum, and bottom nozzle zones for the calculation of activation products in these zones are taken from Reference [A5.6]. The assumed operating histories are listed in the SAS2H/ORIGEN-S input files. Analysis burnups, enrichments, and cooling times are based on the data specified in Table A5.2-1.

Table A5.2-2

Summary of Fuel Assembly Material Weights Used in Source Term Calculations

WE 14x14 SC Assembly Materials and Masses

WE 14x14 MOX Assembly Materials and Masses

Table A5.2-3

Flux Fraction By Assembly Region

Region	Flux Factor
Bottom	0.20
In-Core	1.00
Plenum	0.20
Top	0.10
Above Top	0.01

A sample SAS2H/ORIGEN-S input file for the design basis fuel assembly is reproduced in Section A5.5.2. After the assembly is irradiated in the manner described above, the materials in the assembly are decayed for 38 years (SC) or 20 years (MOX). Analyses were also performed for various enrichments/burnups with a 10-year cooling time to identify the bounding enrichment/burnup combination.

A5.2.1.2 Description of Control Component ORIGEN Models

The three NFAH (control components) discussed above were irradiated as described in Table A5.2-1 to establish their associated source terms. The control component material compositions are shown in Table A5.2-4.

Table A5.2-4

Control Component Regional Weights

A5.2.1.3 Selection of Design Basis Fuel Parameters

As stated in Section A5.2 the largest fuel and control component source term for gamma and neutron sources was selected for each gamma/neutron energy group to generate the design basis fuel assembly for this analysis. The two-dimensional code DORT-PC is used to perform the final shielding calculations for the design basis source term as described in Section A5.4.

A5.2.1.4 Design Basis Gamma Source

The calculated volumetric gamma source strengths and energy spectra for the top nozzle, plenum, in-core, and bottom nozzle regions and for the whole assembly are shown in Table A5.2-8. The energy structure is that used by the CASK-81 cross-section library. The design basis gamma source is chosen by first identifying the bounding SC fuel assembly in-core source term by comparing the 35 GWd/MTU, 40 GWd/MTU and 45 GWd/MTU; 10 year cooled fuel assembly source terms (See Table A5.2-5). The 45 GWd/MTU is the bounding in-core source term. This source term was decayed for 38 years and compared to the MOX source term. The MOX source term is bounding for energy groups 31, 32 and 33 only (by a maximum of 54% of the SC source term). However, since the MOX (20 year decay) neutron source term (Table A5.2-9) is approximately 50% of the SC (38 year decay) neutron source term; and as the bounding dose rate (2 meter side dose, normal operating condition) is approximately 90% due to the neutron dose rate contribution, the SC 38 year cooled in-core source term is bounding. For the top nozzle, plenum and bottom nozzle regions, the maximum gamma dose for each region and each energy group, between the three 10-year cooled SC fuel burnups evaluated and the 20 year cooled MOX fuel, is used to conservatively define the bounding source term. The NFAH gamma source is added to the fuel gamma dose by including the largest gamma source of the four NFAH per region for each energy group (NFAH gamma sources are conservatively based on 10 year cooling). Table A5.2-5 highlights the gamma sources considered with the bounding fuel and NFAH source, for each region and energy group. The total design basis gamma source strength in the package containing 24 WE 14x14 fuel assemblies in a 24PT1-DSC is 3.444×10^{16} photons per second.

Table A5.2-5

Bounding Fuel In Core Region Gamma Source[1] (γ/s/assy)

Cask 81 Energy Group	14x14 SC 35 GWd/MTU 10 Years	14x14 SC 40 GWd/MTU 10 Years	14x14 SC 45 GWd/MTU 10 Years	14x14 SC 45 GWd/MTU 38 Years	14x14 MOX 25 GWd/MTU 20 Years
23	8.169E+04	1.236E+05	1.633E+05	5.759E+04	2.679E+04
24	3.848E+05	5.820E+05	7.691E+05	2.713E+05	1.263E+05
25	1.962E+06	2.967E+06	3.921E+06	1.384E+06	6.444E+05
26	4.889E+06	7.393E+06	9.770E+06	3.449E+06	1.608E+06
27	2.736E+08	3.249E+08	3.685E+08	1.022E+07	5.042E+06
28	2.184E+09	2.577E+09	2.919E+09	1.917E+08	2.773E+07
29	3.687E+10	4.239E+10	4.762E+10	1.827E+09	8.798E+08
30	7.945E+10	9.092E+10	1.027E+11	3.509E+10	1.647E+10
31	4.815E+13	5.410E+13	5.888E+13	1.724E+12	2.445E+12
32	1.865E+14	2.111E+14	2.316E+14	9.691E+12	1.495E+13
33	5.158E+13	6.393E+13	7.635E+13	3.955E+12	5.509E+12
34	1.086E+15	1.250E+15	1.416E+15	6.684E+14	5.043E+14
35	9.708E+13	1.191E+14	1.413E+14	9.760E+12	6.739E+12
36	2.243E+13	2.542E+13	2.876E+13	1.375E+13	6.457E+12
37	3.457E+13	3.953E+13	4.496E+13	2.004E+13	1.026E+13
38	1.180E+14	1.355E+14	1.544E+14	6.424E+13	3.512E+13
39	1.635E+14	1.854E+14	2.093E+14	1.151E+14	7.935E+13
40	8.211E+14	9.362E+14	1.060E+15	5.023E+14	2.910E+14

1 – See Section A5.2.1.4 discussion for the methodology used to identify bounding source term.

Table A5.2-5

Bounding Fuel Top Region Gamma Source (γ/s/assy)

(Continued)

Group	35 GWd	40 GWd	45 GWd	MOX
23	0.000E+00	0.000E+00	0.000E+00	0.000E+00
24	0.000E+00	0.000E+00	0.000E+00	0.000E+00
25	0.000E+00	0.000E+00	0.000E+00	1.800E-13
26	0.000E+00	0.000E+00	0.000E+00	2.825E+03
27	6.378E-12	1.224E-11	2.055E-11	1.822E+06
28	2.130E+04	2.373E+04	2.559E+04	3.198E-04
29	1.374E+07	1.530E+07	1.650E+07	7.677E+10
30	1.002E-03	1.296E-03	1.593E-03	2.719E+11
31	5.789E+11	6.449E+11	6.953E+11	1.176E+08
32	2.050E+12	2.284E+12	2.462E+12	1.096E+08
33	1.184E+09	1.347E+09	1.492E+09	9.249E+05
34	1.922E+08	2.151E+08	2.342E+08	1.464E+07
35	6.974E+06	7.769E+06	8.377E+06	1.121E+07
36	1.104E+08	1.229E+08	1.326E+08	2.251E+08
37	8.421E+07	9.380E+07	1.011E+08	9.327E+08
38	1.695E+09	1.888E+09	2.035E+09	7.654E+09
39	7.024E+09	7.824E+09	8.436E+09	0.000E+00
40	5.578E+10	6.214E+10	6.700E+10	0.000E+00
Total	2.695E+12	3.002E+12	3.237E+12	3.577E+11

Table A5.2-5

Bounding Fuel Plenum Region Gamma Source (γ/s/assy)

(Continued)

Group	35 GWd	40 GWd	45 GWd	MOX
23	0.000E+00	0.000E+00	0.000E+00	0.000E+00
24	0.000E+00	0.000E+00	0.000E+00	0.000E+00
25	0.000E+00	0.000E+00	0.000E+00	0.000E+00
26	0.000E+00	0.000E+00	0.000E+00	0.000E+00
27	2.188E-11	4.169E-11	6.963E-11	6.597E-13
28	2.834E+04	3.156E+04	3.402E+04	2.193E+03
29	1.828E+07	2.035E+07	2.194E+07	1.414E+06
30	3.214E-03	4.161E-03	5.117E-03	3.613E+00
31	7.701E+11	8.576E+11	9.245E+11	5.959E+10
32	2.727E+12	3.037E+12	3.274E+12	2.110E+11
33	1.457E+09	1.648E+09	1.815E+09	3.484E+08
34	6.136E+08	6.870E+08	7.477E+08	5.048E+08
35	9.278E+06	1.033E+07	1.114E+07	2.963E+08
36	1.468E+08	1.635E+08	1.763E+08	2.466E+07
37	1.122E+08	1.250E+08	1.347E+08	1.221E+07
38	2.256E+09	2.512E+09	2.708E+09	2.320E+08
39	9.347E+09	1.041E+10	1.122E+10	7.303E+08
40	7.495E+10	8.348E+10	9.002E+10	6.984E+09
Total	3.586E+12	3.993E+12	4.305E+12	2.797E+11

Table A5.2-5

Bounding Fuel Bottom Region Gamma Source (γ/s/assy)

(Concluded)

Group	35 GWd	40 GWd	45 GWd	MOX
23	0.000E+00	0.000E+00	0.000E+00	0.000E+00
24	0.000E+00	0.000E+00	0.000E+00	0.000E+00
25	0.000E+00	0.000E+00	0.000E+00	0.000E+00
26	0.000E+00	0.000E+00	0.000E+00	0.000E+00
27	2.480E-15	4.040E-15	5.800E-15	0.000E+00
28	3.073E+04	3.423E+04	3.690E+04	1.571E-16
29	1.982E+07	2.207E+07	2.379E+07	4.077E+03
30	1.087E-05	1.161E-05	1.224E-05	2.629E+06
31	8.352E+11	9.301E+11	1.003E+12	8.860E-08
32	2.957E+12	3.294E+12	3.550E+12	1.108E+11
33	1.673E+09	1.909E+09	2.123E+09	3.924E+11
34	3.494E+06	3.891E+06	4.195E+06	1.777E+07
35	1.006E+07	1.121E+07	1.208E+07	4.636E+05
36	1.592E+08	1.773E+08	1.911E+08	1.335E+06
37	1.213E+08	1.351E+08	1.457E+08	2.112E+07
38	2.444E+09	2.721E+09	2.934E+09	1.610E+07
39	1.013E+10	1.128E+10	1.216E+10	3.242E+08
40	7.984E+10	8.890E+10	9.581E+10	1.344E+09
Total	3.887E+12	4.329E+12	4.666E+12	5.049E+11

With the exception of the Rod Cluster Control Assemblies (RCCAs), the non-fuel assembly hardware is handled in an identical fashion. The RCCAs are withdrawn from the fuel assemblies during irradiation in the core, making the RCCA regions (above top, top, plenum, core) different than those in the fuel assembly itself. In order to add the RCCA sources to those of the fuel assembly, the data provided in Table A5.2-5 is mapped to the fuel assembly source regions. This is done using the following mapping function:

Mapped RCCA Region	Function
Top	=(AboveSource)(FuelTopHeight)/(RCCAAbove)
Plenum	=(AboveSource)(FuelPlenumHeight)/(RCCAAbove)
Core	=(AboveSource)(RCCAAbove - FuelTopHeight - FuelPlenumHeight)/(RCCAAbove) + TopSource + PlenumSource + FuelSource

Where AboveSource, TopSource, PlenumSource, and FuelSource are the groupwise RCCA sources from Table A5.2-5, RCCAAbove is the height of the RCCA above top nozzle region, and FuelTopHeight is the height of the fuel top nozzle region (8.6 in) and FuelPlenumHeight is the height of the fuel plenum region (6 in). The bounding NFAH source calculations are shown in Table A5.2-6 below.

Table A5.2-6

NFAH Source Terms

Source Term for Rod Cluster Control Assembly, Eleven Cycles In-core, 10 Year Cooled (γ/s/assy)

Group	Above Top Region γ/s	Top Region γ/s	Plenum Region γ/s	Fuel Region γ/s
23	0.000E+00	0.000E+00	0.000E+00	0.000E+00
24	0.000E+00	0.000E+00	0.000E+00	0.000E+00
25	0.000E+00	0.000E+00	0.000E+00	0.000E+00
26	0.000E+00	0.000E+00	0.000E+00	0.000E+00
27	1.077E-11	4.939E-16	6.937E-16	1.064E-14
28	6.764E+03	1.050E+02	1.481E+02	5.308E+04
29	4.362E+06	6.774E+04	9.552E+04	3.423E+07
30	2.052E-04	3.504E-07	4.968E-07	8.601E-06
31	1.838E+11	2.854E+09	4.025E+09	1.442E+12
32	6.509E+11	1.011E+10	1.425E+10	5.108E+12
33	1.921E+08	6.119E+07	8.622E+07	1.537E+09
34	3.751E+07	1.194E+04	1.684E+04	6.035E+06
35	2.214E+06	3.439E+04	4.849E+04	1.738E+07
36	3.504E+07	5.441E+05	7.672E+05	2.750E+08
37	2.672E+07	4.147E+05	5.848E+05	2.096E+08
38	5.379E+08	8.359E+06	1.178E+07	4.221E+09
39	2.230E+09	3.490E+07	4.912E+07	1.750E+10
40	1.784E+10	3.734E+08	5.248E+08	1.385E+11

Table A5.2-6

NFAH Source Terms

(Continued)

Bounding NFAH Top Region Gamma Source (γ/s/assy)

Group	RCCA	Thimble	Neutron
23	0.000E+00	0.000E+00	0.000E+00
24	0.000E+00	0.000E+00	0.000E+00
25	0.000E+00	0.000E+00	0.000E+00
26	0.000E+00	0.000E+00	0.000E+00
27	1.009E-12	2.153E-11	2.671E-11
28	6.338E+02	1.991E+04	2.194E+04
29	4.088E+05	1.284E+07	1.415E+07
30	1.923E-05	4.108E-04	5.072E-04
31	1.722E+10	5.410E+11	5.961E+11
32	6.099E+10	1.916E+12	2.111E+12
33	1.800E+07	4.065E+08	7.160E+08
34	3.515E+06	7.574E+07	5.671E+07
35	2.075E+05	6.517E+06	7.181E+06
36	3.284E+06	1.031E+08	1.136E+08
37	2.504E+06	7.863E+07	8.663E+07
38	5.041E+07	1.583E+09	1.744E+09
39	2.090E+08	6.563E+09	7.231E+09
40	1.672E+09	5.210E+10	5.711E+10
Total	8.018E+10	2.518E+12	2.774E+12

Table A5.2-6

NFAH Source Terms

(Continued)

Bounding NFAH Plenum Region Gamma Source (γ/s/assy)

Group	RCCA	Thimble	Neutron
23	0.000E+00	0.000E+00	0.000E+00
24	0.000E+00	0.000E+00	0.000E+00
25	0.000E+00	0.000E+00	0.000E+00
26	0.000E+00	0.000E+00	0.000E+00
27	7.041E-13	4.235E-15	6.135E-15
28	4.422E+02	1.996E+04	2.165E+04
29	2.852E+05	1.287E+07	1.396E+07
30	1.342E-05	3.519E-06	4.635E-06
31	1.202E+10	5.424E+11	5.883E+11
32	4.255E+10	1.921E+12	2.083E+12
33	1.256E+07	6.068E+08	8.885E+08
34	2.453E+06	2.269E+06	2.461E+06
35	1.448E+05	6.535E+06	7.087E+06
36	2.291E+06	1.034E+08	1.121E+08
37	1.747E+06	7.880E+07	8.547E+07
38	3.517E+07	1.587E+09	1.721E+09
39	1.458E+08	6.580E+09	7.135E+09
40	1.167E+09	5.212E+10	5.610E+10
Total	5.594E+10	2.524E+12	2.737E+12

Table A5.2-6

NFAH Source Terms

(Concluded)

Bounding NFAH Core Region Gamma Source (γ/s/assy)

Group	RCCA	Neutron Source
23	0.000E+00	0.000E+00
24	0.000E+00	0.000E+00
25	0.000E+00	0.000E+00
26	0.000E+00	0.000E+00
27	9.067E-12	1.951E-14
28	5.902E+04	5.144E+04
29	3.806E+07	3.317E+07
30	1.820E-04	1.459E-05
31	1.604E+12	1.398E+12
32	5.680E+12	4.950E+12
33	1.846E+09	2.751E+09
34	3.761E+07	5.848E+06
35	1.932E+07	1.684E+07
36	3.057E+08	2.665E+08
37	2.330E+08	2.031E+08
38	4.693E+09	4.090E+09
39	1.946E+10	1.696E+10
40	1.544E+11	1.337E+11
Total	7.465E+12	6.506E+12

The design basis gamma source for each region is calculated by summing the shaded fuel and NFAH cells from Table A5.2-5 and A.5.2-6. This results in the γ source term per assembly shown in Table 5.2-7. The sources input to the DORT-PC models are then calculated by multiplying these results by the number of fuel assemblies (24) and dividing by the applicable region volume. The resulting, design basis gamma source term is given in Table A5.2-8. These sources are used directly in the DORT-PC models.

Table A5.2-7

Composite Gamma Source

(γ/s/assy)

CASK-81 Group	Top Nozzle	Gas Plenum	Active Fuel	Bottom Nozzle	Total
23	0.000E+00	0.000E+00	5.759E+04	0.000E+00	5.759E+04
24	0.000E+00	0.000E+00	2.713E+05	0.000E+00	2.713E+05
25	1.800E-13	0.000E+00	1.384E+06	0.000E+00	1.384E+06
26	2.825E+03	0.000E+00	3.449E+06	0.000E+00	3.452E+06
27	1.822E+06	7.034E-11	1.022E+07	5.800E-15	1.204E+07
28	4.752E+04	5.567E+04	1.917E+08	3.690E+04	1.919E+08
29	7.678E+10	3.590E+07	1.865E+09	2.379E+07	7.871E+10
30	2.719E+11	3.613E+00	3.509E+10	2.629E+06	3.069E+11
31	1.291E+12	1.513E+12	3.328E+12	1.003E+12	7.135E+12
32	4.573E+12	5.357E+12	1.537E+13	3.550E+12	2.885E+13
33	2.208E+09	2.704E+09	3.957E+12	3.924E+11	4.355E+12
34	3.099E+08	7.502E+08	6.684E+14	1.777E+07	6.684E+14
35	1.839E+07	3.034E+08	9.760E+12	1.208E+07	9.760E+12
36	3.387E+08	2.884E+08	1.375E+13	1.911E+08	1.375E+13
37	1.019E+09	2.202E+08	2.004E+13	1.457E+08	2.005E+13
38	9.399E+09	4.429E+09	6.425E+13	2.934E+09	6.426E+13
39	1.567E+10	1.836E+10	1.151E+14	1.216E+10	1.152E+14
40	1.241E+11	1.461E+11	5.025E+14	9.581E+10	5.028E+14
Total	6.366E+12	7.043E+12	1.416E+15	5.057E+12	1.435E+15

Table A5.2-8

24PT1-DSC Design Basis Gamma Source Strength and Energy Spectrum

Composite Gamma Source

(γ/s/cm^3)

CASK-81 Group	Top Nozzle	Gas Plenum	Active Fuel	Bottom Nozzle	Total
23	0.000e+00	0.000e+00	3.120e-01	0.000e+00	3.120e-01
24	0.000e+00	0.000e+00	1.470e+00	0.000e+00	1.470e+00
25	1.215e-17	0.000e+00	7.495e+00	0.000e+00	7.495e+00
26	1.907e-01	0.000e+00	1.868e+01	0.000e+00	1.887e+01
27	1.230e+02	6.804e-15	5.537e+01	8.632e-19	1.783e+02
28	3.207e+00	5.385e+00	1.039e+03	5.491e+00	1.053e+03
29	5.182e+06	3.473e+03	1.011e+04	3.541e+03	5.199e+06
30	1.835e+07	3.495e-04	1.901e+05	3.913e+02	1.854e+07
31	8.716e+07	1.463e+08	1.803e+07	1.492e+08	4.007e+08
32	3.086e+08	5.182e+08	8.326e+07	5.284e+08	1.438e+09
33	1.490e+05	2.615e+05	2.144e+07	5.839e+07	8.024e+07
34	2.091e+04	7.257e+04	3.621e+09	2.645e+03	3.621e+09
35	1.241e+03	2.935e+04	5.287e+07	1.798e+03	5.290e+07
36	2.286e+04	2.790e+04	7.448e+07	2.844e+04	7.456e+07
37	6.879e+04	2.130e+04	1.086e+08	2.168e+04	1.087e+08
38	6.343e+05	4.285e+05	3.480e+08	4.366e+05	3.495e+08
39	1.057e+06	1.776e+06	6.238e+08	1.810e+06	6.284e+08
40	8.376e+06	1.414e+07	2.722e+09	1.426e+07	2.759e+09
Total	4.296e+08	6.813e+08	7.673e+09	7.525e+08	9.537e+09

A5.2.2 Neutron Source

The neutron source strength for the design basis fuel assembly is calculated using the same SAS2H/ORIGEN-S runs used to calculate the gamma source. All input parameters and assumptions are the same as those used for the gamma-ray calculations described above.

The neutron source results from spontaneous fission and (α,n) reactions in the active fuel region. The total neutron source for the design basis case is 2.460×10^9 neutrons/sec/DSC as calculated by the SAS2H/ORIGEN-S code.

The spectrum for spontaneous fission neutrons from ^{244}Cm is assumed for the neutron source in this analysis [A5.7]. The ^{244}Cm spontaneous fission spectrum was chosen because it represents more than 90% of the total neutron source in the package. The ^{244}Cm spectrum group fractions for the CASK-81 energy structure are then multiplied by the total source strength to determine the groupwise source strengths for the cask to be used in the shielding calculations. The total neutron source strength and energy spectrum for each of the five burnup cases is shown in Table A5.2-9.

The design basis neutron source is the 38 year cooled WE 14x14 SC with a burnup of 45,000 GWd/MTU. This burnup/minimum enrichment combination fuel bounds all other burnup/enrichments considered when 10 year cooled fuel is compared. This fuel also bounds the MOX 20 year cooled fuel as shown in Table A5.2-9.

Table A5.2-9

Design Basis Neutron Source Strength and Energy Spectrum[1] (neutrons/sec/assy)

Cask 81 Energy Group	Normalized Cm-244 Fission Source	14x14 SC 35 GWd/MTU 10 Years	14x14 SC 40 GWd/MTU 10 Years	14x14 SC 45 GWd/MTU 10 Years	14x14 SC 45 GWd/MTU 38 Years	14x14 MOX 25 GWd/MTU 20 Years
1	2.018E-04	2.880E+04	4.347E+04	5.739E+04	2.068E+04	9.878E+03
2	1.146E-03	1.635E+05	2.468E+05	3.259E+05	1.175E+05	5.610E+04
3	4.471E-03	6.380E+05	9.631E+05	1.272E+06	4.583E+05	2.189E+05
4	1.768E-02	2.523E+06	3.808E+06	5.028E+06	1.812E+06	8.654E+05
5	4.167E-02	5.946E+06	8.976E+06	1.185E+07	4.271E+06	2.040E+06
6	5.641E-02	8.050E+06	1.215E+07	1.604E+07	5.782E+06	2.761E+06
7	1.197E-01	1.708E+07	2.578E+07	3.404E+07	1.227E+07	5.859E+06
8	9.616E-02	1.372E+07	2.071E+07	2.735E+07	9.856E+06	4.707E+06
9	2.256E-02	3.219E+06	4.859E+06	6.416E+06	2.312E+06	1.104E+06
10	1.227E-01	1.751E+07	2.643E+07	3.490E+07	1.258E+07	6.006E+06
11	2.110E-01	3.011E+07	4.545E+07	6.001E+07	2.163E+07	1.033E+07
12	1.794E-01	2.560E+07	3.864E+07	5.102E+07	1.839E+07	8.782E+06
13	1.138E-01	1.624E+07	2.451E+07	3.236E+07	1.166E+07	5.571E+06
14	1.301E-02	1.857E+06	2.802E+06	3.700E+06	1.334E+06	6.368E+05
15	6.555E-05	9.354E+03	1.412E+04	1.864E+04	6.719E+03	3.209E+03
16	4.765E-06	6.800E+02	1.026E+03	1.355E+03	4.884E+02	2.332E+02
17	3.134E-07	4.472E+01	6.751E+01	8.913E+01	3.212E+01	1.534E+01
18	4.527E-08	6.460E+00	9.751E+00	1.287E+01	4.640E+00	2.216E+00
19	9.759E-09	1.393E+00	2.102E+00	2.775E+00	1.000E+00	4.777E-01
20	1.521E-09	2.170E-01	3.276E-01	4.326E-01	1.559E-01	7.445E-02
21	3.353E-10	4.785E-02	7.222E-02	9.536E-02	3.437E-02	1.641E-02
22	9.683E-11	1.382E-02	2.086E-02	2.754E-02	9.925E-03	4.740E-03
Totals	**1.000E+00**	**1.427E+08**	**2.154E+08**	**2.844E+08**	**1.025E+08**	**4.895E+07**

[1] See Section A5.2.2 discussion for the methodology used to identify bounding source term.

A5.3 Model Specification

The shielding analysis was performed to verify that the MP187 Package containing a 24PT1-DSC with 24 WE 14x14 fuel assemblies meets the shielding criteria specified in 10CFR Part 71 [A5.1] and 49CFR Part 173 [A5.2]. Analytical models of the MP187 Package were developed to determine the dose rates around the Package for both normal conditions of transport and hypothetical accident conditions. The 2-D discrete ordinates code DORT-PC [A5.9] was used to calculate the neutron and gamma-ray dose rates as described in Section A5.4. The models used for the 24PT1-DSC payload are the same models used for the MP187 Cask in Chapter 5 with changes to the DSC and fuel areas to reflect the differences in the payload.

A5.3.1 Description of the Radial and Axial Shielding Configuration

The radial and axial shielding configuration of the Package is shown in Figure A5.3-3 and Figure A5.3-4. Figure A5.3-1 and Figure A5.3-2 show the overall dimensions for the 24PT1-DSC. The DSC/cask are modeled symmetrically in cylindrical coordinates and nominal dimensions were used to generate the model. A single model is used to calculate radial and axial, neutron and gamma-ray dose rates around the NUHOMS®-MP187 Cask containing a 24PT1-DSC. The DORT-PC model of the Package includes irregularities in the lead shielding to represent the tapered ends of the lead column. The impact limiters have a square cross-section that cannot be modeled accurately in cylindrical coordinates. The impact limiter geometry is therefore approximated using cylinders with radii such that they would be completely enclosed by the actual impact limiter geometry. The trunnions and the drain, vent, and test ports have been neglected. Because the upper trunnions extend through the neutron shield, the dose rates around these trunnions are discussed separately in Section A5.4.1.8. The cask drain, vent and test ports contain steel plugs during shipment and do not represent a shine path. The support rods in the DSC have also been neglected. The spacer discs are neglected in the homogenized fuel region, but are explicitly modeled in the gap between the fuel region and the DSC shell. A gap is modeled in the cask bottom cover. This gap is not currently shown on the cask drawings, but represents a conservatism in the shielding calculations. The Package shielding normal conditions of transport materials are shown in Figure A5.3-3.

The neutron shielding and neutron shield jacket are assumed to be lost during the hypothetical accident conditions due to the postulated cask drop and subsequent fire. This assumption is made purely for conservatism as the neutron shield is expected to survive the hypothetical accident. Also, the impact limiters are assumed to be lost during the hypothetical accident even though there is no physical mechanism that could remove the impact limiter inner shells. The cask lead shielding is assumed to slump 2.5 inches at each end during the postulated drop, producing air gaps in the cask shielding. The suitability of the cask cavity closure during the hypothetical accident conditions is documented in Chapters 2 and 4. During the accident, the DSC is assumed to slide to the top of the cask cavity, and the fuel is assumed to slide to the top of the DSC cavity. These assumptions are made to maximize the streaming through the lead slump gap at the top of the DSC. The Package shielding materials for the hypothetical accident conditions are shown in Figure A5.3-4.

Neutron and gamma-ray dose rates were calculated in various locations, radially and axially, around the package. These locations include the Package surface, the surface of the transport vehicle, and two meters from the surface of the transport vehicle along its top, bottom, and sides. The maximum dose rates reported in Table A5.1-1 are the maximum dose rates observed at any location, radially and axially, along the surface. The Package surface is defined by the surface of the impact limiters and the neutron shield during normal conditions of transport. The Package surface is defined by the cask structural shell after the hypothetical accident. The outer surface of the vehicle is defined by the impact limiters and the personnel barrier, which is mounted on the skid and extends to the same radius as the impact limiters as shown on the drawings provided in Section 1.3.2.

Figure A5.3-1

NUHOMS®-MP187 Normal Operations DORT-PC 24PT1-DSC Top End Model

Figure A5.3-2

NUHOMS®-MP187 Normal Operations DORT-PC 24PT1-DSC Bottom End Model

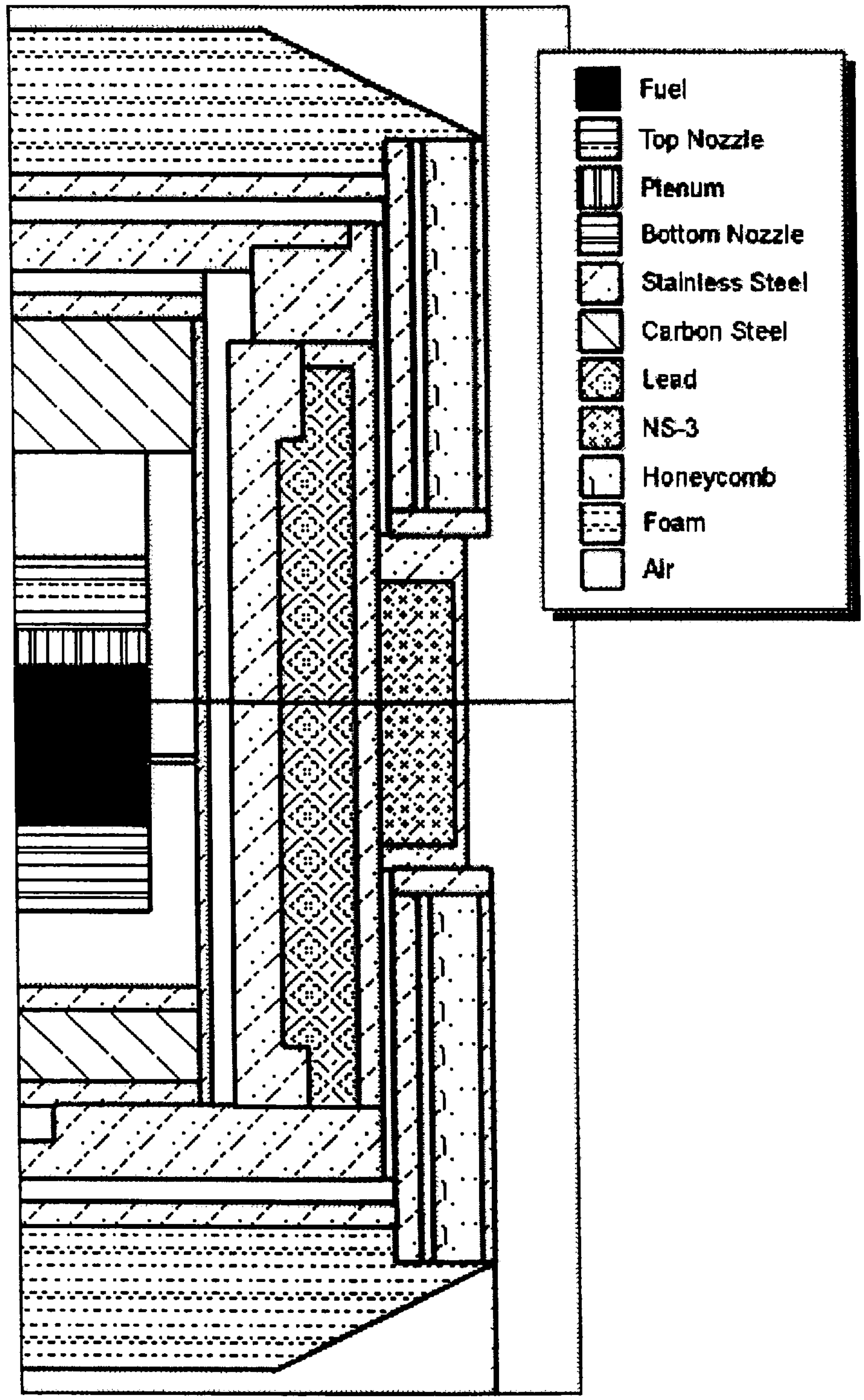


Figure A5.3-3

NUHOMS®-MP187 Normal Operation Shielding Materials

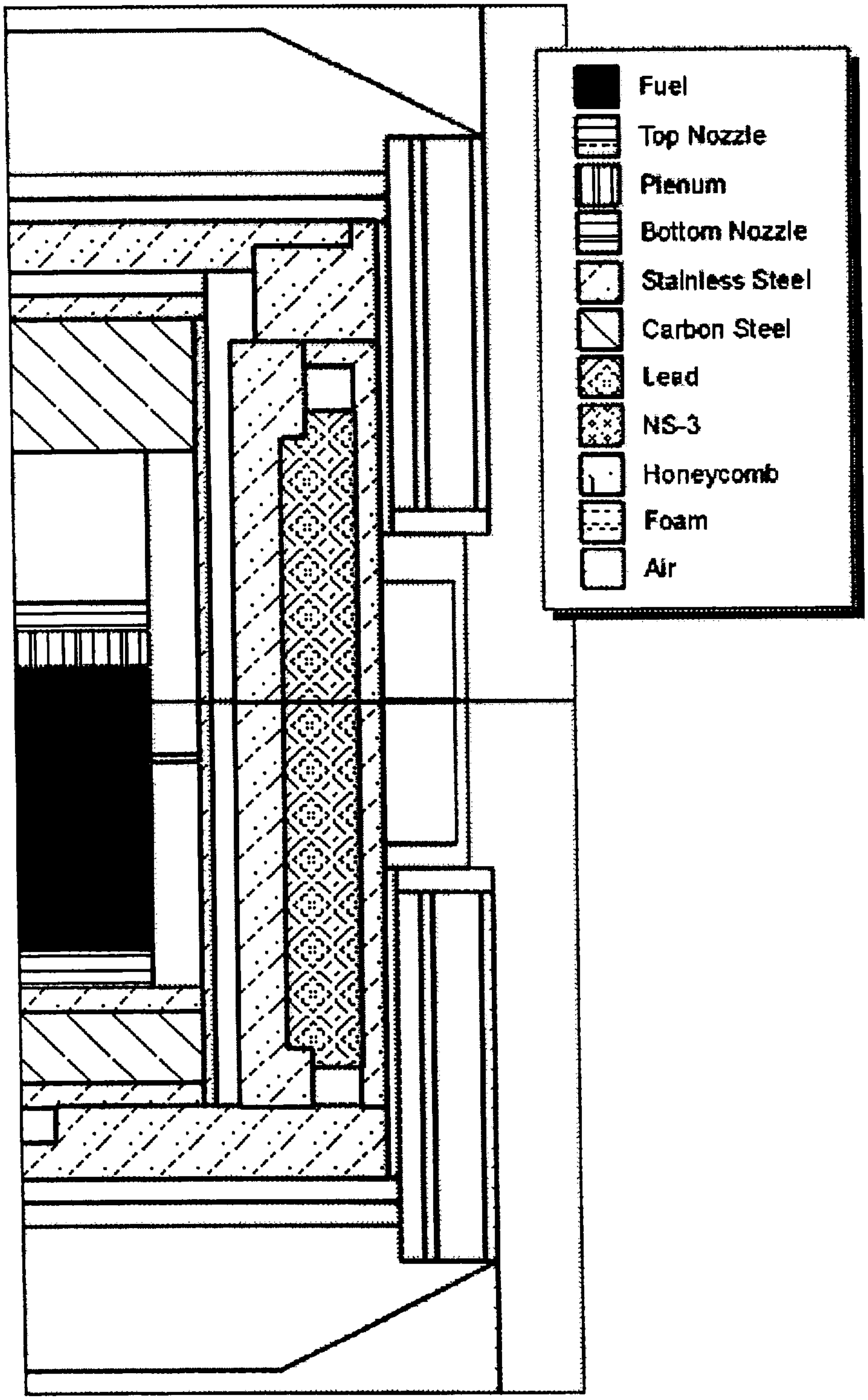


Figure A5.3-4

NUHOMS®-MP187 Hypothetical Accident Condition Shielding Materials

A5.3.2 Shield Regional Densities

The material densities for all of the constituent nuclides in the materials used in the analytical models of the Package are given in this section. Shield regional densities for the in-core, top nozzle, and bottom nozzle regions are calculated using the material weights given in Table A5.2-2. The materials in these regions are assumed to be evenly distributed over the entire volume of the region (smeared). This is a common modeling technique for shielding calculations.

The basket assembly support rods are assumed to provide no shielding. The DSC guidesleeves and neutron absorber plates are assumed to completely cover the active fuel region in the model. The top nozzle and bottom nozzle regions are assumed to be uncovered. Minor impurities in the fuel, DSC, and the cask materials have been neglected.

The fissile nuclides present in significant quantities in the fuel are included in the homogenized fuel region to provide a source for subcritical multiplication. Because the fixed neutron source (spontaneous fission and (α,n) reactions) produces the majority of the neutron dose rate on the Package surface, the overall neutron dose rates are thus maximized by using the largest possible fixed neutron source. This occurs at the greatest fuel burnup allowed by the Package design. The fissile content in the fuel is significantly depleted at these high burnups and use of the initial enrichment for subcritical multiplication is, therefore, conservative.

The neutron shield density and material composition are shown in Table A5.3-1. Variations in the composition of the neutron shield over the life of the Package are accounted for by neglecting 10% of the hydrogen weight and 50% of the boron weight. Variations in the actual neutron shield boron loading have only a small impact on the Package dose rates. A 50% reduction in the boron weight conservatively bounds any depletion that may occur during the life of the Package.

The neutron shield is completely encased in a sealed, stainless-steel annulus. When the neutron shield is exposed to high temperatures, some steam can be evolved from the material. Because the neutron shield is completely encased in a sealed, stainless-steel annulus, none of this steam is released from the cask and the total neutron shield weight remains constant. A 10% loss of hydrogen is assumed, however, to account for stratification of steam released by the solid material. Long term thermal testing of the neutron shield has been performed as discussed in Reference [A5.8]. Blocks of neutron shield material were held at a constant temperature of 250°F for an extended period of time. The test results show that the neutron shield material weight stabilized after a two month period and that the total reduction in hydrogen weight is less than 9.3 percent. The maximum neutron shield temperature during normal operation in the NUHOMS®-MP187 Cask is less than 250°F as documented in Chapter 3. The assumed hydrogen loss of 10 weight percent, therefore, results in conservative material properties for the shielding calculations.

The mass and atom densities of the remaining materials used in the shielding calculations are given in Table A5.3-1.

Table A5.3-1

Summary of NUHOMS®-MP187 Cask Shielding Material Densities

[atoms/b·cm, (g/cc)]

Element	Stainless Steel	Carbon Steel	Lead	Neutron Shield(1)	Aluminum Honeycomb	Foam	Air
H	-	-	-	4.180×10^{-2} (0.070)	-	1.004×10^{-2} (0.017)	-
B	-	-	-	7.131×10^{-4} (0.013)	-	-	-
C	-	-	-	7.871×10^{-3} (0.157)	-	7.220×10^{-3} (0.144)	-
N	-	-	-	-	-	8.255×10^{-4} (0.019)	3.587×10^{-5} (8.343×10^{-4})
O	-	-	-	3.442×10^{-2} (0.915)	-	2.168×10^{-3} (0.058)	9.534×10^{-6} (2.533×10^{-4})
Al	-	-	-	8.528×10^{-3} (0.382)	6.428×10^{-3} (0.288)	-	-
Si	-	-	-	1.155×10^{-3} (0.054)	-	-	-
Ca	-	-	-	1.351×10^{-3} (0.090)	-	-	-
Cr	1.728×10^{-2} (1.492)	-	-	6.104×10^{-4} (0.053)	-	-	-
Fe	6.073×10^{-2} (5.632)	8.465×10^{-2} (7.85)	-	2.242×10^{-3} (0.208)	-	-	-
Ni	7.447×10^{-3} (0.726)	-	-	2.631×10^{-4} (0.026)	-	-	-
Pb	-	-	3.296×10^{-2} (11.34)	-	-	-	-

(1) Neutron shield material densities include an assumed 10% hydrogen loss and a 50% reduction in boron. Neutron shield material densities include the stainless steel and aluminum stiffeners present in the neutron shield annulus.

Table A5.3-1

Summary of NUHOMS®-MP187 Cask Shielding Material Densities

(atoms/b-cm)

(Concluded)

Region Volumes, cm^3		1.61E+05	4.96E+06	2.48E+05	3.56E+05
Element	Atomic Weight	Bottom Nozzle Region	In-Core Region	Plenum Region	Top Nozzle Region
H	1.008	0	0	0	0
B-10	10.013	2.007E-04	2.009E-04	2.009E-04	2.009E-04
C	12.011	9.927E-04	9.934E-04	9.933E-04	9.933E-04
N	14.0067	0	0	0	0
O	15.9994	0	7.936E-03	0	0
Na	22.99	0	0	0	0
Mg	24.305	0	0	0	0
Al	26.9814	6.902E-04	6.906E-04	6.906E-04	6.906E-04
Si	28.086	0	0	0	0
Ca	40.078	0	0	0	0
Ti	47.88	0	0	0	0
Cr	51.9961	3.371E-03	1.488E-03	1.565E-03	2.156E-03
Mn	54.938	0	0	0	0
Fe	55.847	1.185E-02	5.231E-03	5.502E-03	7.580E-03
Ni	58.69	1.454E-03	6.417E-04	6.750E-04	9.298E-04
Cu	63.646	0	0	0	0
Zr	91.224	0	0	0	0
Mo	95.94	0	0	0	0
Sn	118.71	0	0	0	0
Pb	207.2	0	0	0	0
U-235	235.044	0	1.520E-04	0	0
U-238	238.051	0	3.799E-03	0	0

A5.4 Shielding Evaluation

The neutron and gamma-ray dose rates for the Package are estimated using the DORT-PC computer code [A5.9]. The DORT-PC input parameters and results are discussed in the following sections.

A5.4.1 DORT-PC Input Parameters

A5.4.1.1 The DORT-PC Computer Code

DORT-PC determines the fluence of particles throughout one-dimensional or two-dimensional geometric systems by solving the Boltzmann transport equation using either the method of discrete ordinates or a diffusion theory approximation. Particles can be generated by either particle interaction with the transport medium or extraneous sources incident upon the system. Anisotropic cross-sections can be expressed in a Legendre expansion of arbitrary order. DORT-PC is an industry standard code distributed by ORNL/RSIC.

The DORT-PC code implements the discrete ordinates method as its primary mode of operation. Balance equations are solved for the flow of particles moving in a set of discrete directions in each cell of a space mesh and in each group of a multigroup energy structure. Iterations are performed until all implicitness in the coupling of cells, directions, groups, and source regeneration has been resolved.

DORT-PC was chosen for this application because of its ability to solve two dimensional, cylindrical, deep penetration, radiation transport problems. The package has thick multilayered shields which are difficult to analyze with point-kernel codes. The cask geometry is too complicated to be treated adequately with a one-dimensional discrete ordinates code.

A5.4.1.2 Source Distribution

Four DORT-PC runs were made for each case, normal operating and hypothetical accident, to account for the contributions from the three different source regions and for neutron and gamma sources. In the first two runs, the gamma and neutron sources were placed throughout the active fuel region of the DORT-PC model. The remaining runs accounted for the gamma contributions from the top nozzle and bottom nozzle regions. The results of these four runs were then summed to determine the total neutron and gamma-ray dose rates around the package.

The ^{235}U fission spectrum [A5.10] is input into the 1* array of the DORT-PC input file to account for subcritical multiplication, increasing the neutron source in the active fuel region. Axial peaking is accounted for in the active fuel region by inputting a relative flux factor at each node in the 97* array. The appropriate flux factor for the design basis case is 1.12 for gamma-rays, conservatively assumed to apply to the entire length of the active fuel region. The flux factor for neutrons has been raised to the fourth power to account for the variation of neutron sources with fuel burnup and is conservatively assumed to apply to the entire length of the active fuel region. Differences between PWR fuel designs, including the locations of the grid spacers and associated hardware, is accounted for by conservatively applying the maximum peaking factor of 1.12 across the entire active fuel region.

A5.4.1.3 Model Geometry

The Package geometry shown in Figure A5.3-1 through Figure A5.3-4 is used in the DORT-PC computer models. The Package is modeled in cylindrical (R-Z) coordinates and one model is used for both radial and axial dose rates. A total of 46 zones are defined by a mesh of 86 intervals in the radial direction and 193 intervals in the ax+ial direction. A reflective boundary condition is placed on the central axis of the cask (left boundary). The boundary conditions on the remaining boundaries are voids.

A total of four DORT-PC runs were made to calculate the normal conditions of transport and hypothetical accident dose rates around the package. The first two runs for each case include the in-core region gamma and neutron sources, respectively. The third run for each case includes only the top nozzle source. The final run includes only the bottom nozzle source. Each run includes the full cask geometry. The normal conditions of transport case materials are as shown in Figure A5.3-3 and the hypothetical accident case materials are as shown in Figure A5.3-4.

A5.4.1.4 Cross Section Data

The cross section data used in this analysis is taken from the CASK-81 22 neutron, 18 gamma-ray energy group, coupled cross-section library [A5.4]. CASK-81 is an industry standard cross section library compiled for the purpose of performing calculations of spent fuel shipping casks and is distributed by ORNL/RSIC. The cross section data allows coupled neutron/gamma-ray runs to be made that account for secondary gamma radiation (n,γ).

Microscopic P_3 cross sections were taken from the CASK-81 library and mixed using the GIP-PC computer program distributed with DORT-PC [A5.9] to provide macroscopic cross sections for the materials in the cask model. The GIP input file is reproduced in Section A5.5.3. The material compositions used in the GIP input file are listed in Table A5.3-1.

An additional element and material, "fluxdosium," is included in the cross section data and mixing table in the GIP input file. Fluxdosium is used to provide flux-to-dose rate conversion factors as described below for use in activity calculations. The presence of fluxdosium in the cross-section data does not impact the actual flux calculations.

A5.4.1.5 Flux-to-Dose Rate Conversion Factors

The flux distribution calculated by the DORT-PC code is converted to dose rates using the flux-to-dose rate conversion factors provided in ANSI/ANS-6.1.1-1977 [A5.11]. The gamma-ray and neutron flux-to-dose rate conversion factors for the CASK–81 energy groups are shown in Table A5.4-1 and A5.4-2, respectively.

The dose rate at each node in the DORT-PC model is calculated using the activity calculation feature of DORT-PC. The "cross section" data for one material in the input file contains only flux-to-dose rate conversion factors. This material, "fluxdosium," is specified for activity calculations which determines the gamma and neutron dose rate at each node.

Table A5.4-1

Gamma-Ray Flux-to-Dose Rate Factors

Energy Group	E_{Lower} (MeV)	E_{Upper} (MeV)	Flux-to Dose Factor (μRem/hr)/(γ/s/cm^2)
23	8.00e+00	1.00e+01	8.7716
24	6.50e+00	8.00e+00	7.4785
25	5.00e+00	6.50e+00	6.3748
26	4.00e+00	5.00e+00	5.4136
27	3.00e+00	4.00e+00	4.6221
28	2.50e+00	3.00e+00	3.9596
29	2.00e+00	2.50e+00	3.4686
30	1.66e+00	2.00e+00	3.0192
31	1.33e+00	1.66e+00	2.6276
32	1.00e+00	1.33e+00	2.2051
33	8.00e-01	1.00e+00	1.8326
34	6.00e-01	8.00e-01	1.5228
35	4.00e-01	6.00e-01	1.1725
36	3.00e-01	4.00e-01	0.87594
37	2.00e-01	3.00e-01	0.63061
38	1.00e-01	2.00e-01	0.38338
39	5.00e-02	1.00e-01	0.26693
40	1.00e-02	5.00e-02	0.93477

Table A5.4-2

Neutron Flux-to-Dose Rate Factors

Energy Group	E_{Lower} (MeV)	E_{Upper} (MeV)	Flux-to Dose Factor (µRem/hr)/(n/s/cm^2)
1	12.2e+01	1.49e+01	194.49
2	1.00e+01	1.22e+01	159.71
3	8.18e+00	1.00e+01	147.06
4	6.36e+00	8.18e+00	147.73
5	4.96e+00	6.36e+00	153.39
6	4.06e+00	4.96e+00	150.62
7	3.01e+00	4.06e+00	138.92
8	2.46e+00	3.01e+00	128.43
9	2.35e+00	2.46e+00	125.27
10	1.83e+00	2.35e+00	126.32
11	1.11e+00	1.83e+00	128.94
12	5.50e-01	1.11e+00	116.85
13	1.11e-01	5.50e-01	65.209
14	3.35e-03	1.11e-01	9.1878
15	5.83e-04	3.35e-03	3.7134
16	1.01e-04	5.83e-04	4.0086
17	2.90e-05	1.01e-04	4.2946
18	1.07e-05	2.90e-05	4.4761
19	3.06e-06	1.01e-05	4.5673
20	1.12e-06	3.06e-06	4.5355
21	4.14e-07	1.12e-06	4.3701
22	1.00e-08	4.14e-07	3.7142

A5.4.1.6 Quadrature Data

The DORT-PC runs use a 402 direction radially biased quadrature set based on the half-symmetric S_{10} set and the 210 direction upward biased set discussed in Reference [A5.12]. In two-dimensional systems such as that which is used to describe the package, the discrete ordinates method can lead to solutions with flux distortions. This is particularly true in poorly scattering media (such as air and the impact limiter materials) that contain isolated sources. Because the cask structural shell directly adjacent to the top and bottom end fittings exhibits localized high dose rates, flux oscillations could be observed at the one and two meter distances required for this evaluation. The potential for flux oscillations is reduced by using a more detailed, radially biased quadrature set in the top and bottom nozzle source runs. The quadrature data is provided in the DORT-PC input files listed in Section A5.5.3

A5.4.1.7 DORT-PC Input Files

The DORT-PC input files are included in Section A5.5.3. The DORT-PC input parameters are discussed in the preceding sections for the eight normal operation and hypothetical accident runs. DORT-PC is run using the quadrature sets described above and a P_3 order of scattering.

A5.4.1.8 Evaluation of Cask Penetrations

Several penetrations exist through the NUHOMS®-MP187 Cask shielding. These include the upper and lower trunnions, the shear key, the test ports, and the drain and vent ports. An evaluation of the impact of these penetrations on the shielding effectiveness of the cask is provided in Chapter 5 for the FO/FC DSCs. Since the Chapter 5 results for the B&W 15x15 fuel are bounding for the WE 14x14 fuel in this area of the Cask, the analyses provided for these penetrations in Chapter 5 bounds the effect on MP187 transportation of the 24PT1-DSC with the WE 14x14 payload.

A5.4.2 NUHOMS®-MP187 Package Dose Rate Results

The DORT-PC computer run dose rate results for the MP187 Cask with a 24PT1-DSC are included in Section A5.5.3. The results of each of the four runs are summed node-by-node for both the normal conditions of transport and the hypothetical accident cases. The maximum gamma, neutron, and total dose rates at the Package side, top, and bottom for the normal conditions of transport case are listed in Table A5.4-3. Note that at some locations the sum of the maximum gamma-ray and neutron dose rates does not equal the maximum total dose rate. This implies that the maximum gamma, neutron, and total dose rates occur at different locations around the Package. As shown in Table A5.4-3, the 10CFR Part 71.47 [A5.1] limits at the surface of the Package, at the surface of the vehicle, and at a distance of two meters from the surface of the vehicle are not exceeded. The limit for occupied locations in the vehicle is also met because the Package top and bottom surface dose rates are less than the two mrem/hr requirement. The Packaging therefore provides suitable shielding during normal conditions of transport to ship 24 PWR fuel assemblies with sources bounded by those discussed in Section A5.2 in accordance with 10CFR Part 71.

Table A5.4-3

Summary of Maximum Normal Conditions of Transport Dose Rates

	Package Surface		
	Side (mrem/hr)	Top (mrem/hr)	Bottom (mrem/hr)
Normal Conditions			
Gamma	2.12E+01	2.61E-01	3.86E-01
Neutron	1.46E+02	9.52E-01	1.42E+00
Total*	1.49E+02	1.19E+00	1.63E+00
10CFR71 Limit	1.00E+03	1.00E+03	1.00E+03
	Vehicle Outer Surface		
	Side (mrem/hr)	Top (mrem/hr)	Bottom (mrem/hr)
Normal Conditions			
Gamma	8.53E+00	2.61E-01	3.86E-01
Neutron	5.51E+01	9.52E-01	1.42E+00
Total*	5.75E+01	1.19E+00	1.63E+00
10CFR71 Limit	2.00E+02	2.00E+02	2.00E+02
	2 Meters From Vehicle Outer Surface		
	Side (mrem/hr)	Top (mrem/hr)	Bottom (mrem/hr)
Normal Conditions			
Gamma	1.60E+00	9.59E-02	1.29E-01
Neutron	7.43E+00	4.14E-01	5.93E-01
Total*	8.28E+00	5.02E-01	6.77E-01
10CFR71 Limit	1.00E+01	1.00E+01	1.00E+01

* Peak Gamma and Neutron doses do not necessarily occur at the same location. Therefore, the maximum total dose rate is not necessarily the sum of the maximum gamma and neutron doses.

The maximum gamma, neutron, and total dose rates at the Package side, top, and bottom for the hypothetical accident case are listed in Table A5.4-4. As shown in Table A5.4-4, the 10CFR Part 71.51 [A5.1] limit at a distance of one meter from the surface of the Package is not exceeded. The Packaging therefore provides suitable shielding during the hypothetical accident conditions to ship 24 PWR fuel assemblies with sources bounded by those discussed in Section A5.2 in accordance with 10CFR Part 71. Because the hypothetical accident dose rates are less than 50% of the limit while the maximum normal conditions of transport dose rates are at the allowable, the normal conditions of transport case is bounding for the NUHOMS®-MP187. Because the dose rates are surveyed prior to transport and verified to meet the 10CFR71 and 49CFR173 limits for normal conditions of transport, the Package will meet all applicable dose limits under both normal and accident conditions.

Table A5.4-4

Summary of Maximum Hypothetical Accident Dose Rates

	Package Surface		
	Side (mrem/hr)	Top (mrem/hr)	Bottom (mrem/hr)
Accident Conditions			
Gamma	1.22E+02	6.06E+00	2.03E+00
Neutron	8.10E+02	5.28E+01	1.83E+02
Total*	8.17E+02	5.35E+01	1.85E+02
10CFR71 Limit	-	-	-
	1 Meter From Surface of Package		
	Side (mrem/hr)	Top (mrem/hr)	Bottom (mrem/hr)
Accident Conditions			
Gamma	1.37E+01	3.06E-01	5.54E-01
Neutron	2.74E+02	2.01E+01	6.17E+01
Total*	2.78E+02	2.04E+01	6.22E+01
10CFR71 Limit	1.00E+03	1.00E+03	1.00E+03

* Peak Gamma and Neutron doses do not necessarily occur at the same location. Therefore, the maximum total dose rate is not necessarily the sum of the maximum gamma and neutron doses.

Figure A5.4-1 shows the total dose rate along the length of the Package for normal conditions of transport. Neutron, gamma, and total dose rates are shown at the Package surface and at a distance of two meters from the vehicle surface. The total dose rate is also shown along the vehicle surface. The peak dose rates are seen to occur just above and below the neutron shield during normal conditions of transport. The hypothetical accident dose rates are shown in Figure A5.4-2. The peak dose rates during the hypothetical accident occur due to streaming through the postulated gaps in the lead shielding.

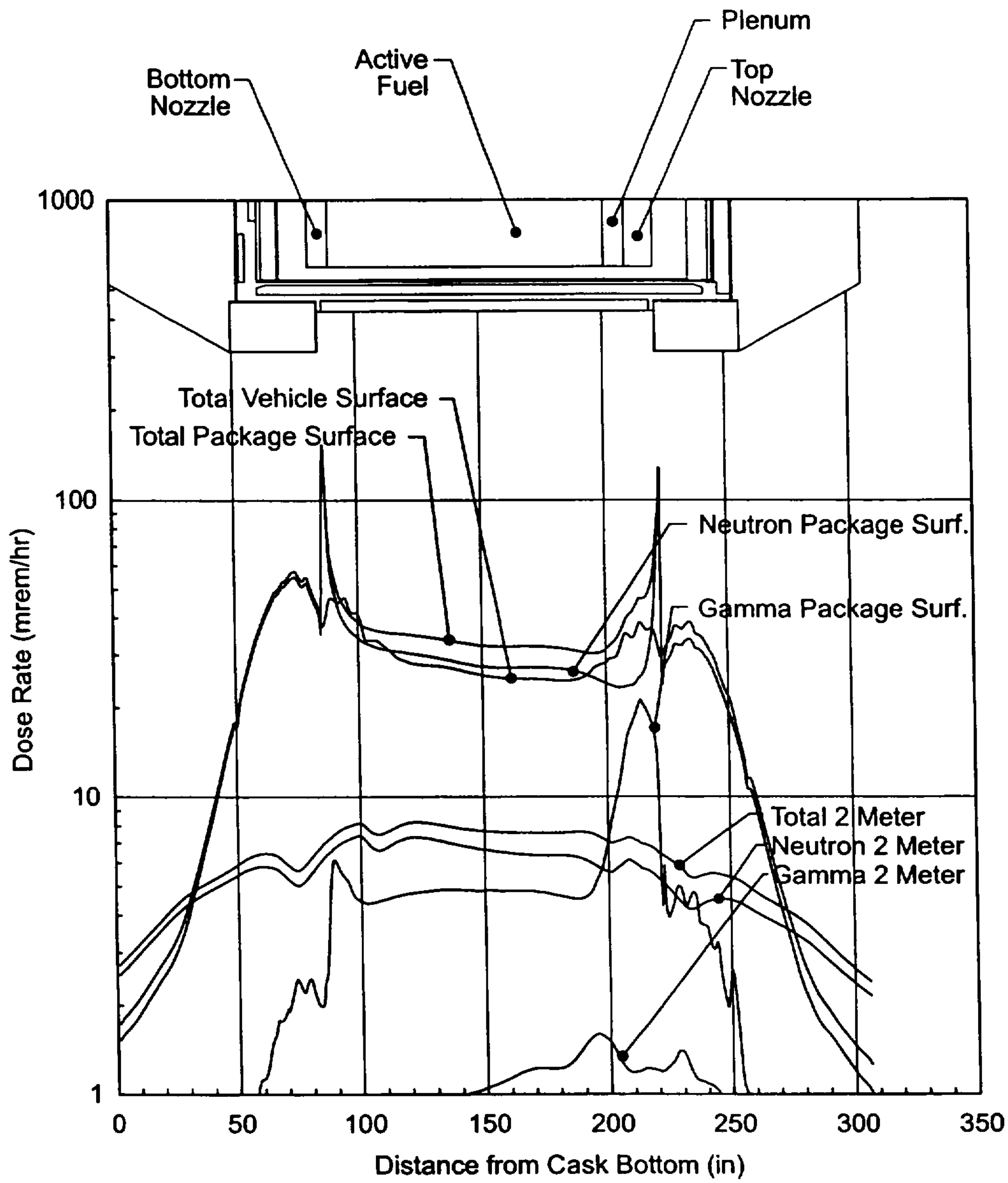


Figure A5.4-1

NUHOMS®-MP187 24PT1-DSC Normal Operation Dose Rate Distribution

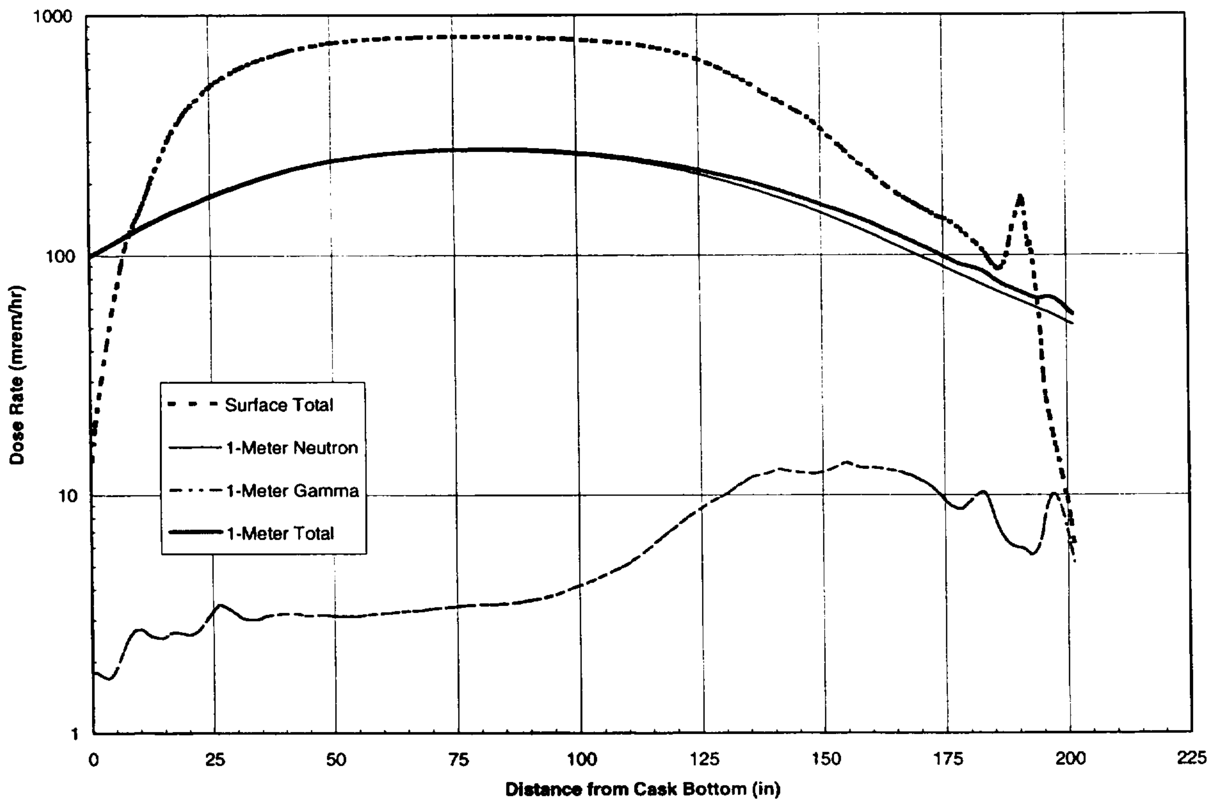


Figure A5.4-2

NUHOMS®-MP187 Hypothetical Accident Dose Rate Distribution

A5.5 Appendix

A5.5.1 References

A5.5.2 SAS2H/ORIGEN-S Input Files

A5.5.3 DORT-PC and GIP Input Files

A5.5.1 References

A5.1 U.S. Government, "Packaging and Transportation of Radioactive Material," Title 10 Code of Federal Regulations, Part 71, Office of the Federal Registrar, Washington DC.

A5.2 U.S. Government, "Shippers – General Requirements for Shipments and Packagings," Title 49 Code of Federal Regulations, Part 173, Section 441, Office of the Federal Registrar, Washington DC.

A5.3 Oak Ridge National Laboratory, RSIC Computer Code Collection, "SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluations for Workstations and Personal Computers," NUREG/CR-0200, Revision 6, ORNL/NUREG/CSD-2/V2/R6.

A5.4 "CASK-81 - 22 Neutron, 18 Gamma-Ray Group, P_3, Cross Sections for Shipping Cask Analysis", DLC-23, Oak Ridge National Laboratory, RSIC Data Library Collection, June 1987.

A5.5 Luksic, A., "Spent Fuel Assembly Hardware: Characterization and 10CFR61 Classification for Waste Disposal," Volume 1, Pacific Northwest Laboratory, PNL-6906, June 1989.

A5.6 "Characteristics of Potential Repository Wastes", DOE/RW-0184-R1, Office of Civilian Radioactive Waste Management, July 1992.

A5.7 "MCNP 4 - Monte Carlo Neutron and Photon Transport Code System", CCC-200A/B, Oak Ridge National Laboratory, RSIC Computer Code Collection, October 1991.

A5.8 "Moderate Temperature (250°F) Weight Loss of NS-3," Technical Report No. NS-3-029, Bisco Products, Inc., Park Ridge, Illinois, Revision 0, April 1985 (Appendix 5.5.6).

A5.9 "DORT-PC - Two-Dimensional Discrete Ordinates Transport Code System", CCC-532, Oak Ridge National Laboratory, RSIC Computer Code Collection, October 1991.

A5.10 Watt, B. E., "Energy Spectrum of Neutrons and Thermal Fission of ^{235}U," Phys. Rev 87, 1037 (1952).

A5.11 "American National Standard Neutron and Gamma-Ray Flux-to-Dose-Rate Factors", ANSI/ANS-6.1.1-1977, American Nuclear Society, La Grange Park, Illinois, March 1977.

A5.12 Jenal, J. P., P. J. Erickson, W. A. Rhoades, D. B. Simpson, and M. L. Williams, "The Generation of a Computer Library for Discrete Ordinates Quadrature Sets", ORNL/TM-6023, Oak Ridge National Laboratory, October 1977.

A5.13 "ANISN-ORNL - One-Dimensional Discrete Ordinates Transport Code System with Anisotropic Scattering", CCC-254, Oak Ridge National Laboratory, RSIC Computer Code Collection, April 1991.

A5.14 "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages," DOE/RW-0472, Office of Civilian Radioactive Waste Management, Revision 0, May 1995.

A5.15 Jones, K.B., and B.D. Thomas, "Dry Fuel Storage Cask Shielding Benchmarks," Proceedings of the Sixth Annual International High Level Radioactive Waste Management Conference, Las Vegas, Nevada, May 1995.

A5.16 Hermann, O. W., and S. M. Bowman, M. C. Brady, and C. V. Parks, "Validation of the Scale System for PWR Spent Fuel Isotopic Composition Analysis," ORNL/TM-12667, Oak Ridge National Laboratory, March 1995.

A5.17 "Evaluation of Shielding Analysis Methods in Spent Fuel Cask Environments", EPRI TR-104329, Electric Power Research Institute, May 1995.

A5.5.2 SAS2H/ORIGEN-S Input Files

A5.5.3 DORT-PC and GIP Input Files

A6. CRITICALITY EVALUATION

This chapter describes the design elements of the NUHOMS®-MP187 Package which are important to safety and necessary to comply with the criticality control performance requirements specified in Sections 71.55 and 71.59 of 10CFR Part 71 [A6.1] as applicable to the 24PT1-DSC payload.

The results of detailed analyses are presented which demonstrate that the NUHOMS®-MP187 Package with the 24PT1-DSC payload (as defined in Chapter A1) is critically safe, under normal conditions of transport and hypothetical accident conditions, considering a variety of mechanical uncertainties.

A6.1 Discussion and Results

A6.1.1 NUHOMS®-MP187 Cask Design Features

The NUHOMS®-MP187 Package, with the 24PT1-DSC payload, is designed to provide criticality control through a combination of mechanical and neutronic isolation of fuel assemblies. Unlike traditional spent fuel shipping packages, the NUHOMS®-MP187 Cask is designed to carry a payload of canisterized fuel. The 24PT1-DSCs are suitable for use in the Advanced NUHOMS® dry storage system, and for transportation in the NUHOMS®-MP187 Cask. This Appendix/Chapter addresses the 24PT1-DSC payload and two design basis fuel assembly types (Westinghouse 14x14 Stainless Steel Clad Uranium Dioxide Fuel Assemblies, WE 14x14 SS, and Westinghouse 14x14 Zirconium Clad Mixed Oxide Fuel Assemblies, WE 14x14 MOX).

A6.1.2 24PT1-DSC Design Features

The principal performance features of the NUHOMS®-MP187 Cask and 24PT1-DSC as they relate to criticality control are:

A. The package is designed such that it would be subcritical if water were to leak into or out of the canister. No credit is taken for the containment capability of the canister to exclude moderator from the fuel matrix (10CFR71.55 b).

B. The criticality analyses have been performed with consideration for the most reactive credible configuration consistent with the chemical and physical form of the material (10CFR71.55b1), moderation by water to the most reactive credible extent (10CFR71.55b2), and close full reflection by water on all sides or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging (10CFR71.55b3).

C. Any number of undamaged, or damaged (10CFR71.73) packages will remain subcritical in any arrangement with close full water reflection and optimum interspersed hydrogenous moderation. Therefore, the transport index for nuclear criticality control is zero (10CFR71.59b).

The 24PT1-DSC support structure is composed of four axially oriented support rods and twenty-six spacer discs. This basket assembly provides positive location for twenty-four fuel assemblies under both normal conditions of transport (NOC) and post-hypothetical accident conditions (HAC). The basket assembly utilizes fixed neutron absorbers which isolate each fuel assembly. Guidesleeves are designed to permit unrestricted flooding and draining of fuel cells.

The absorber panel material was chosen due to its desirable neutron attenuation, low density, and minimal thickness. It has been used for applications and in environments comparable to those found in spent fuel storage and transportation since the early 1950s (the U.S. Atomic Energy Commission's AE-6 Water-Boiler Reactor). In the 1960s, it was used as a poison material to ship irradiated fuel rods from Canada's Chalk River laboratories to Savannah River. More than 12,000 British Nuclear Fuels, Ltd. (BNFL) flasks containing the material have been used to transport fuel to BNFL's reprocessing plant in Sellafield.

The neutron absorber panels are composed of boron carbide and 1100 alloy aluminum. Boron carbide provides the necessary content of the neutron absorbing B10 isotope in a chemically inert, heat resistant, highly crystalline and extremely hard form. Boron carbide contained in the panels does not react under these conditions. The boron carbide core is tightly held within an 1100 aluminum alloy matrix and further protected by solid 1100 aluminum alloy cladding plates. Although 1100 alloy aluminum is a chemically reactive material, it behaves much like an inert material when properly applied. Proper application includes due consideration to the formation of a highly protective aluminum oxide layer and allowance for creation of the reaction by-product hydrogen.

Aluminum reacts with water to produce hydrogen (H_2) and an impervious tightly adhering layer of hydrated aluminum oxide ($Al_2O_3 \bullet 3H_2O$) called bayerite which protects the surface from further attack.

$$2\,Al + 6\,H_2O \rightarrow Al_2O_3 \bullet 3H_2O + 6H^+ + 6\,\text{electrons}$$

Initially, the DSC basket will be submerged in the spent fuel pool. During this period, aluminum in the panels will react with water in the manner noted above to form a small amount of hydrogen gas and produce a stable bayerite layer on all surfaces of the panel. The bayerite layer formed on the panel during pool immersion persists through DSC drying, sealing, storage, and eventual shipping; preventing further corrosion or hydrogen production. A complete corrosion discussion for the package components is provided in Section 2.4.4.

Leaching of the boron carbide along the unsealed edges of the panels is expected to occur to an insignificant degree. There are three reasons why this is anticipated to be insignificant. First, the panel core is a sintered Al/B_4C material. Only the boron carbide particles exposed by saw cut are available for leaching. Second, the immersion environment is relatively benign and the time is brief (a few hours or days). The material has been commonly used in U.S. spent fuel racks for many years and, in fact, has gained a reputation for not leaching, as other neutron absorbing materials have done. And third, direct experimental observations of accelerated aging tests performed at the University of Michigan [A6.10] showed no indications of boron degradation. The test specimens were exposed to high neutron and gamma irradiation in a reactor pool environment for over nine years. Subsequent neutron radiography showed no signs of reduced neutron attenuation anywhere on the test specimens.

A6.1.3 Fuel-Control Components

The 24PT1-DSC is designed to accommodate fuel control components. No credit is taken for the presence of control hardware in the criticality analysis.

A6.1.4 Failed Fuel Can Design Features

The 24PT1-DSC is designed to accommodate a screened Failed Fuel Can to contain damaged fuel assemblies and portions of damaged fuel pins. This Failed Fuel Can is placed within the 24PT1-DSC guidesleeves above a fuel assembly spacer. The locations of Failed Fuel Cans within the 24PT1-DSC are limited to the outermost fuel assembly locations along the 45°, 135°, 225° and 315° azimuths (up to four damaged WE 14x14 SC fuel assemblies may be stored in a DSC with the balance intact WE 14x14 SC fuel assemblies; no more than one damaged WE

14x14 MOX fuel assembly may be stored in a DSC with the balance intact WE 14x14 SC fuel assemblies.) The Failed Fuel Cans are intended to package fuel with gross cladding defects. Fuel assemblies to be stored and/or transported are to be visually inspected to document that cladding damage is limited to no more than 14 fuel pins with known or suspected cladding damage greater than hairline cracks and pinhole leaks. The potential does exist, however, for individual pellets to escape the cladding during transportation. This material would still be confined by the Failed Fuel Cans. Each assembly is placed in a separate, removable can with a fixed mesh screen on the bottom and similarly screened lid on top. The Failed Fuel Cans are designed to permit unrestricted flooding and draining of fuel cells.

The criticality analysis for the Failed Fuel Cans uses the same criteria as that of the intact fuel configuration plus additional considerations arising from mechanical uncertainties of damaged fuel after transport or hypothetical accident conditions.

A6.1.5 Criticality Analysis Summary and Results

The calculated maximum k_{eff}, including 2σ uncertainty, for the NUHOMS®-MP187 Package with the 24PT1-DSC payload is 0.8677 for the intact WE 14x14 SC fuel assemblies, 0.9368 for the damaged fuel geometry WE 14x14 SC fuel assemblies (analysis assumes 24 damaged fuel assemblies, while the C of C limits the payload to no more than 4 damaged fuel assemblies per DSC), 0.9111 for the intact WE 14x14 MOX fuel assemblies and 0.8921 for the damaged fuel geometry WE 14x14 MOX fuel assemblies (analysis assumes 4 damaged fuel assemblies, while the C of C limits the payload to no more than 1 damaged MOX fuel assembly per DSC); including all biases and uncertainties applicable to the calculational methodology and the design. The associated Upper Subcritical Limit has been calculated as 0.9396 (See Section A6.5 for details). The results are summarized in Table A6.1-1.

Table A6.1-1

Summary of Criticality Evaluation

	REQUIRED	CALCULATED[1]
NORMAL CONDITIONS		
Number of undamaged packages calculated to be subcritical	infinite	infinite
Optimum interspersed hydrogenous moderation, ($k_{eff} + 2\sigma$)	yes	0.8666 for SC fuel, 0.9111 for MOX fuel
Closely and fully reflected by water	yes	yes
Package cavity size, cm^3	--	1.11E+07
ACCIDENT CONDITIONS		
Number of damaged packages calculated to be subcritical	infinite	infinite
Optimum interspersed hydrogenous moderation, optimum water reflection, bounding case (damaged fuel case); ($k_{eff} + 2\sigma$)	yes	0.9368 for SC damaged fuel, 0.8921 for MOX damaged fuel
Package cavity size, cm^3	--	1.11E+07
Transport Index	--	0

Notes:
[1] All k_{eff}s include 2σ uncertainty. See Section A6.4.3 for summary details.

The criticality analysis was performed in accordance with the requirements of:

- ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors." [A6.2],
- ANSI/ANS-8.17-1984, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors." [A6.3],
- USNRC Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety," Revision 1, May, 1977. [A6.4],
- ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety." [A6.5],
- USNRC NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages", March 1997 [A6.17].

Guidance has been taken from USNRC Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," Proposed Revision 2, December 1981 [A6.6], as it applies to the calculation of k_{eff} for a transportation cask.

A6.2 Package Fuel Loading

The NUHOMS®-MP187 Cask is designed to accommodate the 24PT1-DSC with or without Failed Fuel Cans, as described above. The design basis fuel for the 24PT1-DSC is Westinghouse 14x14 stainless steel clad fuel with a maximum fuel enrichment of 4.0 wt. % ^{235}U and Westinghouse 14x14 zirconium clad mixed oxide fuel with 0.71 wt. % ^{235}U and Pu enrichment as shown in Table A6.2-1. The fuel loading parameters as they relate to criticality are summarized in Table A6.2-1. Since no credit for burnup was assumed in the criticality calculations, unirradiated fuel is qualified for shipment.

Table A6.2-1

Maximum Fuel Loading Parameters

Parameter	Value
Number of Assemblies, 24PT1-DSC	≤ 24
Number of fuel assemblies in Failed Fuel Cans in 24PT1-DSC	≤ 4 WE 14x14 SC or ≤ 1 WE 14x14 MOX
Enrichment, wt. % ^{235}U	WE 14x14 SC: ≤ 4.0 WE 14x14 MOX: ≤ 0.71
Maximum Fissile Pu Enrichment in WE 14x14 MOX fuel assemblies, wt. %	64 rods – 2.84 wt. % 92 rods – 3.10 wt. % 24 rods – 3.31 wt. %
Minimum Burnup	0
Design Basis Fuel	WE 14x14 SC and WE 14x14 MOX
Maximum Number of Failed Rods (in Failed Fuel Cans)	14/assy

The design properties of the reference fuel are given in Table A6.2-2.

Table A6.2-2

Design Basis Fuel Parameters for Criticality Analysis

Parameter	WE 14x14 SC Fuel	WE 14x14 MOX Fuel
Fuel Pellet Outside Diameter, in.	0.3835	0.3659
Fuel Clad Thickness (nominal), in.	0.0165	0.0243
Fuel Clad Outside Diameter (nominal), in.	0.422	0.422
Fuel Rod Pitch, in.	0.556	0.556
Active Fuel Height, in.	120	120
Enrichment, wt. %	4.0: ^{235}U	^{235}U: 0.71 Fissile Pu : 64 rods – 2.84 92 rods – 3.10 24 rods – 3.31
Pellet Density, %Theoretical Dens.	95	91
Rod Array (NxN Rods)	14	14
Fueled Rod Locations	180	180

A6.3 Model Specification

A6.3.1 Description of Calculational Model

CSAS25 (KENO V.a) of SCALE 4.4 [A6.8] is used to perform the criticality evaluation. The criticality calculations are done using full-cask KENO V.a [A6.8] models. The models are described in detail below and in the Appendix 6.6.2.

The safety requirements of ANSI/ANS-8.17 [A6.3] prescribe that all applicable biases and uncertainties must be investigated and statistically attached to the nominal case k_{eff}. Rather than a statistical approach, this criticality analysis models the system with all the important parameters concurrently in their worst-case state:

- Maximum fabrication thickness and minimum boron content for all the neutron absorber plates (this combination is the worst case since aluminum displaces moderator and is not a strong absorber),
- Minimum fabrication width for all the neutron absorber plates,
- Minimum fabrication thickness for all steel guide tubes and steel absorber wrappers,
- Only 75% credit taken for the boron in neutron absorber plates,
- Worst-case fuel assembly position (includes DSC fabrication tolerances and an allowance for fuel assembly bow and twist),
- Maximum enrichment.

A6.3.2 NUHOMS®-MP187 Cask/24PT1-DSC Model

The KENO models consist of 560 axial layers stacked into an array. The layers consist of partial spacer disc and partial moderator regions inside and outside of the active fuel region. At the top and bottom of the model are the DSC steel cylinder representing the shield plugs. The length of the active fuel layers is equivalent to the greatest common denominator of the spacer disc and moderator region axial lengths. For example, five 0.25" layers of the spacer disc are stacked to make an equivalent 1.25" spacer disc region. The center to center spacing of the spacer disc intervals varies over a range starting at 5.5 and ending at 6.75 inches. However, some of these intervals occur in non-fuel areas. This axially finite arrangement is shown in Table A6.3-1. By specifying specular reflection on the $\pm x$ and $\pm y$ directions (perpendicular to cask centerline) of these array layers, the model represents an infinite array of casks.

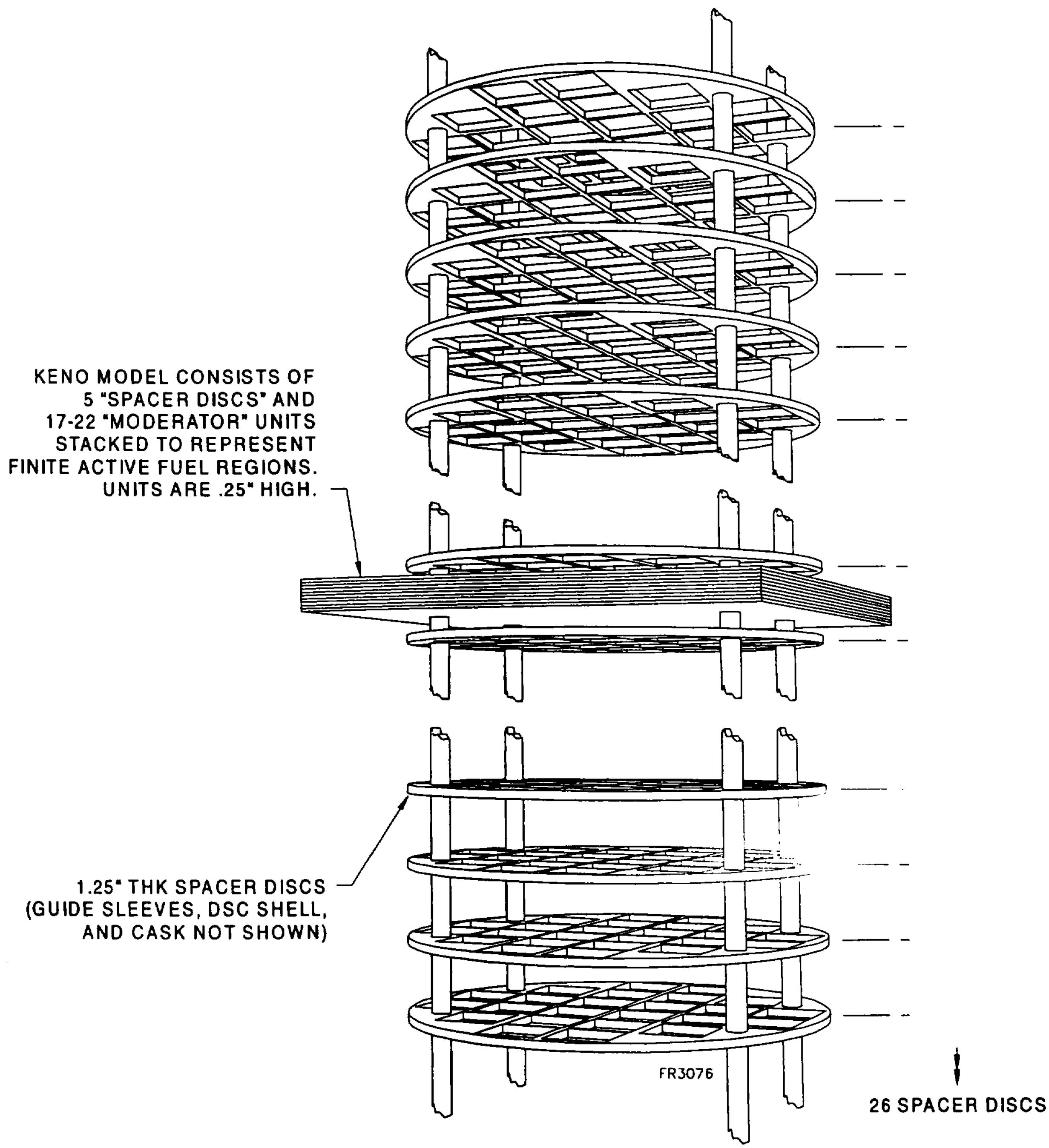


Figure A6.3-1

KENO Model and DSC Basket

Figure A6.3-2 shows the KENO model in an exploded view. UNIT 33 is a slice through the cask at the DSC spacer disc level. UNIT 34 is a similar slice, but in between the spacer discs.

Figure A6.3-3 shows the structure of UNITs 33 and 34: the cask slices. Note that the difference between the two UNITs is that UNIT 33 is a spacer disc (steel surrounding fuel assemblies) and UNIT 34 has steel support rods only (water surrounding fuel assemblies). Also, for the HAC cases, there is no guidesleeve deformation within the spacer disc (Unit 33) region. The fuel assemblies are identified in Figure A6.3-3 by the position numbers (1-24) used to refer to their unique locations. UNIT numbers 1-8 represent the active fuel assemblies in the spacer disc region and UNIT numbers 82-89 represent the active fuel assemblies in the moderator region. The fuel assemblies are inserted into the model using KENO V.a HOLE capability.

A detail of the guidesleeve assembly is shown in the enlarged section of Figure A6.3-3. These models include all major components of the guidesleeve assembly: the square tube, absorber sheets (4 per tube), and the oversleeves which hold the sheets in place. Note that the guidesleeves on the outer periphery of the basket (12 total) only have two absorber sheets per tube.

Figure A6.3-4 shows more closely the way in which UNITs 1-8 are constructed. Each HOLE is identified by UNIT number and its own particular coordinate origin.

UNIT 32 is a cross section of the design basis WE 14x14 fuel assembly. It is illustrated in Figure A6.3-4, which also shows the locations of the fuel assembly guide tubes and the UNIT origin for insertion as a HOLE.

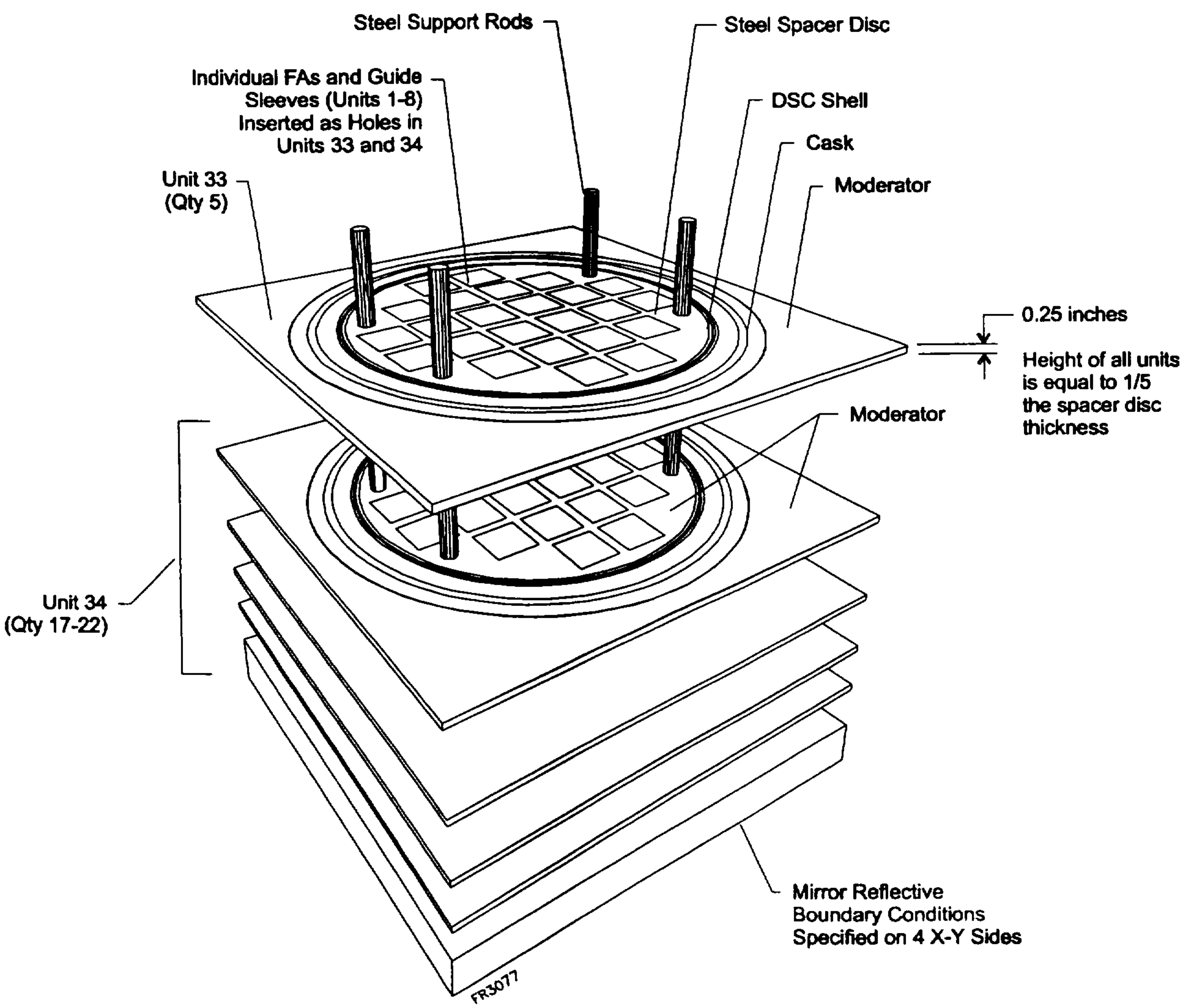


Figure A6.3-2

Exploded View of KENO Model

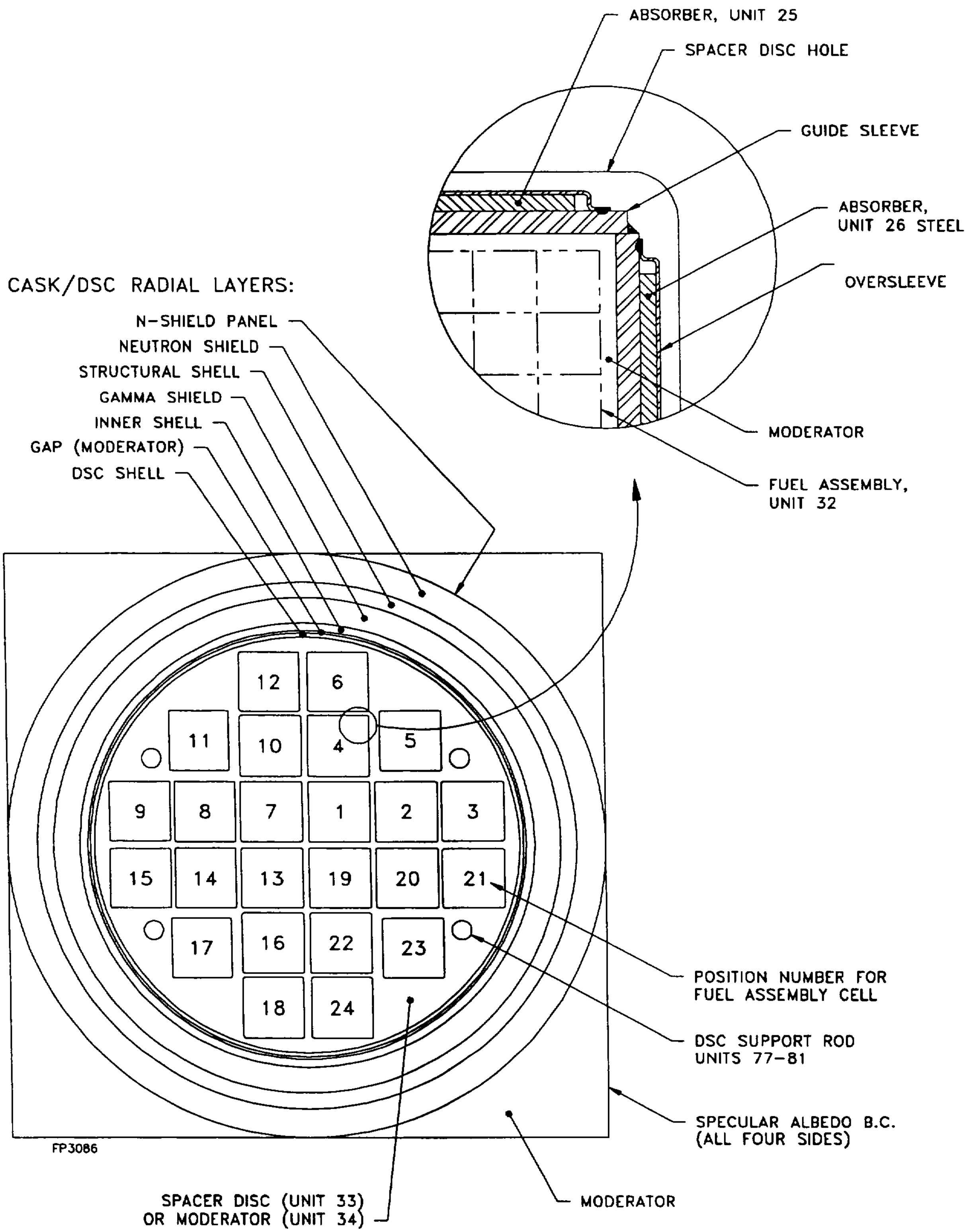


Figure A6.3-3

Structure of KENO Model UNITS 33 and 34

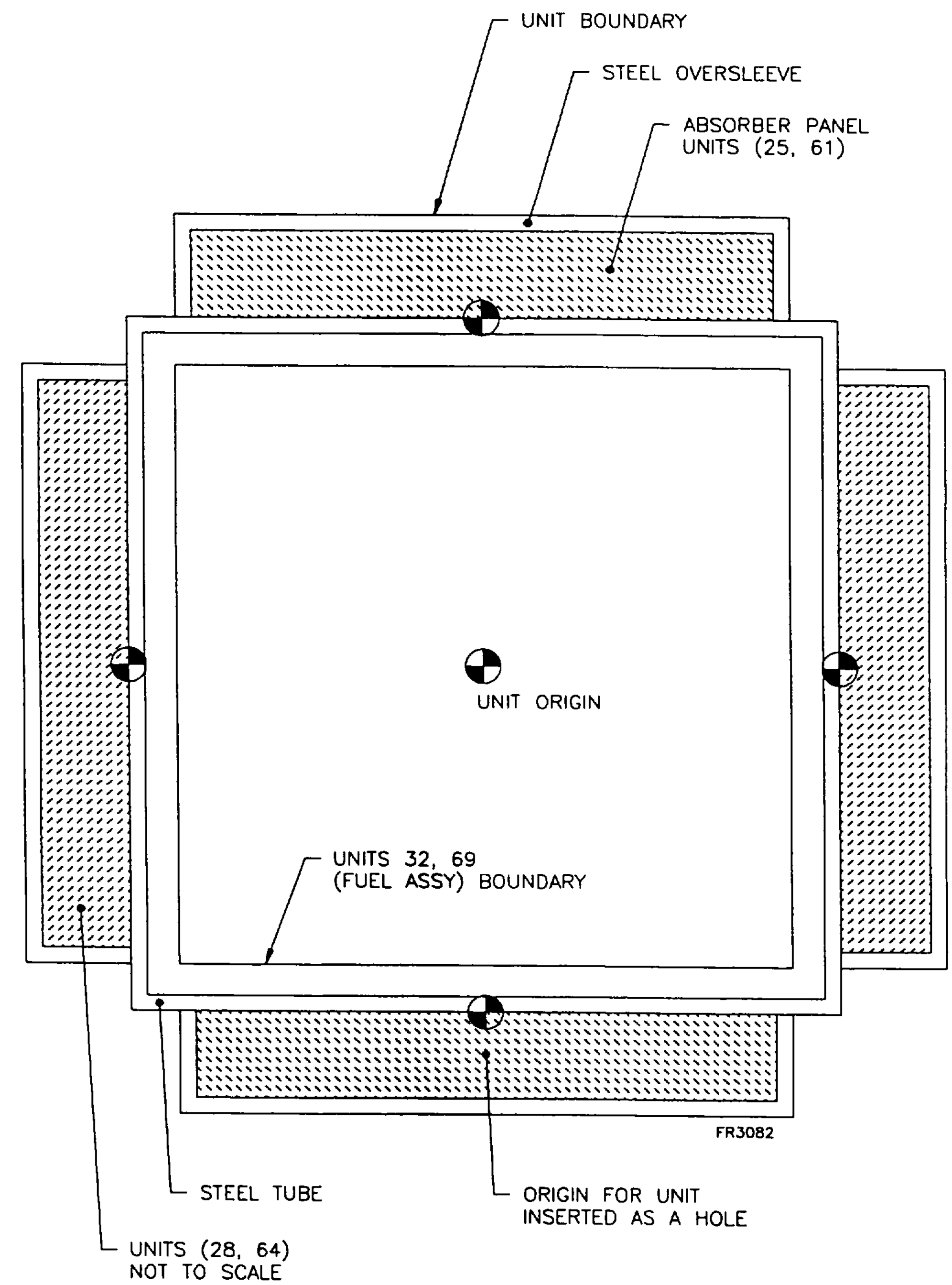
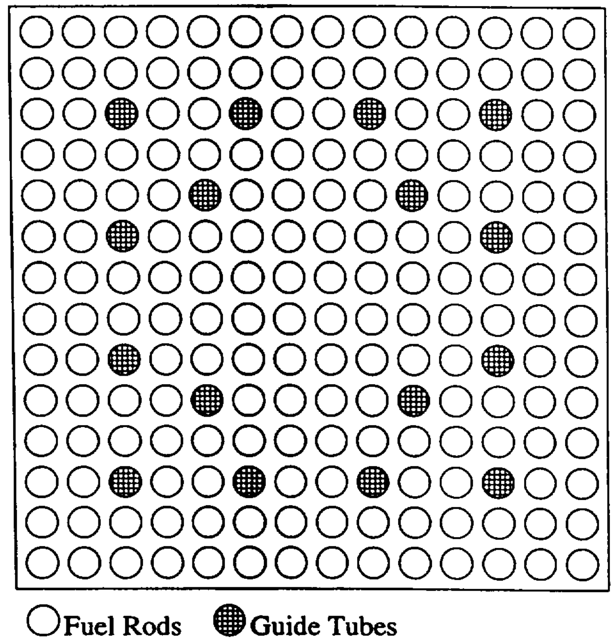
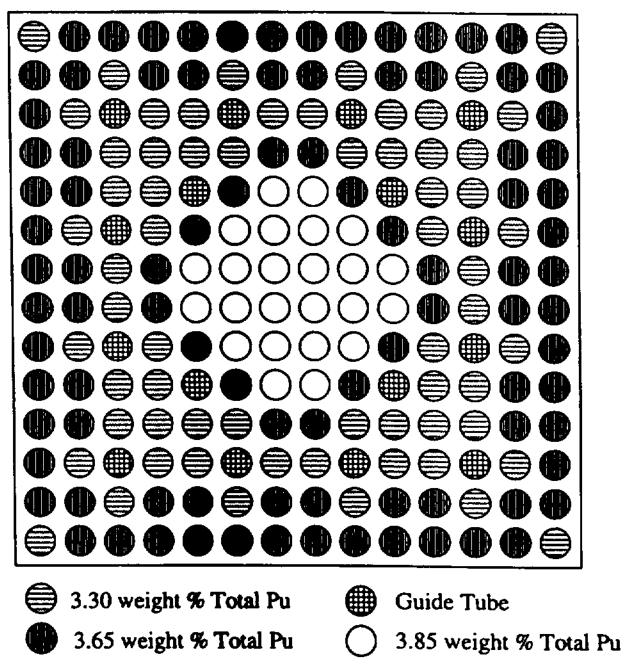


Figure A6.3-4

KENO Model UNITS 1-8, plus



WE 14x14 SC Fuel Assembly Configuration



WE 14x14 MOX Fuel Assembly Configuration

Figure A6.3-5

KENO Model of Design Basis Fuel Assemblies

A6.3.3 NUHOMS®-MP187 Cask/24PT1-DSC Damaged Fuel Model

The NUHOMS®-MP187 Cask/24PT1-DSC KENO model for analysis of failed fuel contents is constructed in the same "slice of the cask" style as the MP187/24PT1-DSC intact fuel model discussed above. The major differences is that the damaged fuel is contained within stainless steel Failed Fuel Cans located within the DSC guidesleeves. These Failed Fuel Cans, however, are conservatively not included in the criticality analysis model.

A6.3.4 Package Regional Densities

Table A6.3-1 and Table A6.3-2 summarizes the calculated atom densities used in the KENO models.

Table A6.3-1

KENO Model Atom Densities

Fuel Pellet (1) – MOX, 24 rods per assembly		Fuel Pellet (11) – MOX, 64 rods per assembly	
Element	atom/b-cm	Element	atom/b-cm
Oxygen	4.65E-02	Oxygen	4.62E-02
^{234}U	1.22E-06	^{234}U	1.22E-6
^{235}U	1.60E-04	^{235}U	1.60E-04
^{238}U	2.21E-02	^{238}U	2.21E-02
^{239}Pu	8.14E-04	^{239}Pu	6.93E-04
^{240}Pu	1.35E-04	^{240}Pu	1.15E-04
^{241}Pu	5.25E-05	^{241}Pu	4.47E-05
^{242}Pu	8.09E-06	^{242}Pu	6.89E-06
Fuel Pellet (12) – MOX, 92 rods per assembly		**Fuel Pellet (1) – Stainless Steel Clad UO2**	
Element	atom/b-cm		
Oxygen	4.64E-02	Oxygen	4.65E-02
^{234}U	1.22E-06	^{235}U	9.41E-04
^{235}U	1.60E-04	^{238}U	2.23E-02
^{238}U	2.21E-02		
^{239}Pu	7.71E-04		
^{240}Pu	1.28E-04		
^{241}Pu	4.98E-05		
^{242}Pu	7.66E-06		
Zircaloy (10)		**Absorber Plate (6)**	
Element	atom/b-cm	Element	atom/b-cm
Chromium	7.60E-05	Aluminum	3.93E-02
Iron	1.49E-04	Boron	2.49E-02
Zirconium	4.25E-02	Carbon	7.67E-03
Tin	4.83E-04		
Hafnium	2.21E-06		
C-Steel (4)		**Lead (7)**	
Element	atom/b-cm	Element	atom/b-cm
Iron	8.35E-02	Lead	3.30E-02
Carbon	3.93E-03		
S_Steel (2)		**NS-3 (8)**	
Element	atom/b-cm	Element	atom/b-cm
Carbon	3.19E-04	Aluminum	7.03E-03
Silicon	1.70E-03	Calcium	1.48E-03
Phosphorus	6.95E-05	Carbon	8.25E-03
Chromium	1.75E-02	Hydrogen	5.10E-02
Manganese	1.74E-03	Iron	1.06E-04
Iron	5.85E-02	Oxygen	3.78E-02
Nickel	7.74E-03	Silicon	1.27E-03

( ) identify mixture number in KENO analysis

Table A6.3-2

KENO Model Moderator Atom Densities

Density (g/cc)	Hydrogen (at/b-cm)	Oxygen (at/b-cm)
1.00000	6.69e-02	3.34e-02
0.90000	6.02e-02	3.01e-02
0.80000	5.35e-02	2.67e-02
0.70000	4.68e-02	2.34e-02
0.60000	4.01e-02	2.01e-02
050000	3.34e-02	1.67e-02
0.40000	2.68e-02	1.34e-02
0.30000	2.01e-02	1.00e-02
0.20000	1.34e-02	6.69e-03
0.10000	6.69e-03	3.34e-03
0.0500	3.34e-03	1.67e-03
0.0001	6.69e-06	3.34e-06

A6.4 Criticality Calculation

A6.4.1 Calculational Method

Criticality calculations for the NUHOMS®-MP187 Package with a 24PT1-DSC payload are performed using CSAS25 (KENO V.a) with the 44-neutron group library based on ENDF-B Version 5 cross section data that is part of the SCALE 4.4 code package [A6.8].

The major assumptions made in the KENO modeling are:

- unirradiated fuel – no credit taken for fissile depletion or fission product poisoning,
- no credit taken for fuel control components,
- fuel is intact with no gross damage or missing rods (applies to intact fuel analyses only),
- the fuel enrichment is modeled as uniform throughout the fuel rods. The maximum pellet enrichment is assumed to exist everywhere,
- fuel and cask are modeled as having finite length (water reflection is specified top and bottom) in all models,
- only 75% credit is taken for boron in neutron absorber panels, and
- all fuel rods are assumed to be filled with 100% moderator in the fuel cladding gap.

A6.4.2 Fuel Loading Optimization

In order to assure that the requirements of 10CFR71, Sections 71.55 and 71.59, are satisfied, the KENO models were specified with either 100% specular albedo, or infinite water conditions on all four sides. All void regions of the Package have been modeled with optimum moderation, including the fuel pellet-clad gaps. Further discussion regarding the models can be found in Section A6.3.2 and in the Appendix 6.6.2.

A6.4.2.1 Fuel Loading Optimization – Failed Fuel Considerations

A6.4.3 Criticality Results

Reactivity calculations were performed in four sets of parametric studies for the cask with 24 intact WE 14x14 SC fuel assemblies, the cask with 24 intact WE 14x14 MOX fuel assemblies, the cask with 24 damaged WE 14x14 SC fuel assemblies and the cask with 20 intact WE 14x14 SC fuel assemblies and four damaged WE 14x14 MOX fuel assemblies in the outermost 45°, 135°, 225° and 315° azimuths locations. The parametric studies were designed to meet the range of conditions specified in 10CFR71.55 and 71.59. Each of the intact fuel studies discussed above consists of four parametric studies which include both normal conditions of transport and hypothetical accident conditions with both internal and external moderator variation (independent of each other). The inner (radially) spacer disc "cutouts" are modeled uniquely, compared to the outer cutouts as far as size is concerned. The tolerances of the cutout center locations as well as the cutout size tolerances were considered in order to be as conservative as possible. However, the minimum allowable ligament size was the controlling factor and was always maintained. The ligament represents the steel region between cut outs. Parmetric studies of guidesleeve deformation and removal of cask layers performed for the FO/FC DSCs in Chapter 6 are not included for the 24PT1-DSC. The analyses presented here for the 24PT1-DSC show that the intact fuel analyses performed in Chapter 6 for B&W fuel bound the Westinghouse fuel analyses. Data provided below for this comparison is $k_{eff} + 2\sigma$ for optimal moderation and normal operating condition:

Table A6.4-1

USL vs. Calculated k_{eff}

Fuel Configuration Analyzed (associated Upper Safety Limit, USL)	k_{eff} + 2σ (margin to USL)
FO/FC with B&W 15x15 fuel (w/ USL = 0.95)	0.9374 (margin to USL = 0. 0126)
24PT1-DSC with WE 14x14 SC fuel (w/ USL = 0.9396)	0.8666 (margin to USL = 0.0730)
24PT1-DSC with WE 14x14 MOX fuel (w/USL = 0.9396)	0.9111 (margin to USL = 0.0285)

The guidesleeve deformation parametric study in Chapter 6 demonstrates that a small increase in k_{eff} on the order of 0.00705 results from deformation up to 0.18 inches. This deformation which was bounding for the FO/FC DSCs is also bounding for the 24PT1-DSC as discussed in Chapter A2. The modeling of this deformation was very conservative in that it assumes that fuel assemblies are moved to the closest possible position based on minimum spacer disc ligament considering maximum spacer disc hole, minimum guidesleeve thickness, maximum poison plate thickness, minimum poison plate boron concentration and minimum poison plate wrapper thickness. The configuration modeled is not possible since it assumes that in a side drop the 12 fuel assemblies below the DSC horizontal midplane move upward (gravity is reversed for these fuel assemblies). These assumptions resulted in minimal increases in reactivity for the more reactive B&W system, thereby ensuring that the B&W analyses performed in Chapter 6 bound analyses for WE 14x14 fuels evaluated in this Appendix. Therefore, guidesleeve deformation is not included in the 24PT1-DSC analyses in this Appendix. The cask layer removal parametric study performed in Chapter 6 also demonstrates that a small increase in k_{eff} on the order of 0.00167 results from a non-mechanistic removal of the structural shell. The minimal increases in reactivity for the more reactive B&W system demonstrate that condition would be bounded by the B&W criticality analyses in Chapter 6.

The results of the studies are shown in tabular form at the end of this Section.

A6.4.3.1 NUHOMS®-MP187 Cask with 24PT1-DSC Intact Fuel Summary

The highest calculated k_{eff} for the WE 14x14 SC fuel is for the Hypothetical Accident Condition with 0.80 g/cc moderator interspersed between an infinite array of packages. The reactivity is 0.8677 ($k_{eff} + 2\sigma$). The highest calculated k_{eff} for the WE 14x14 MOX fuel is for the Normal Operating Condition with 0.0001 g/cc moderator interspersed between an infinite array of packages. The reactivity is 0.9111 ($k_{eff} + 2\sigma$).

A6.4.3.2 NUHOMS®-MP187 Cask/24PT1-DSC Damaged Fuel Summary

The highest calculated k_{eff} for the WE 14x14 SC damaged fuel (analyzed for 24 damaged fuel assemblies, with a C of C limit of 4 damaged fuel assemblies) with 0.80 g/cc moderator interspersed between an infinite array of packages, is 0.9368 ($k_{eff} + 2\sigma$). The highest calculated k_{eff} for the WE 14x14 MOX damaged fuel is for 0.6000 g/cc moderator interspersed between an infinite array of packages. The reactivity is 0.8921 ($k_{eff} + 2\sigma$).

Table A6.4-2

MP187/24PT1-DSC WE 14x14 SC Intact Fuel Results

Normal Operating Condition: Assembly Position Case Results			
k_{eff}	+/- 1σ	k_{eff} + 2σ	Description
0.8636	0.0011	0.8659	The fuel assembly is located in the corner of each guidesleeve closest to the 24PT1-DSC centerline.
0.8581	0.0013	0.8608	The fuel assemblies are centered in each guidesleeve.
0.8382	0.0011	0.8404	The fuel assemblies are moved radially outwards from the center of the 24PT1-DSC
0.8475	0.0011	0.8497	The fuel assemblies are moved towards the upper left corner of each guidesleeve.

Normal Operating Condition: Internal Moderator Density Varying Assuming the Inward Assembly Position			
k_{eff}	+/- 1σ	k_{eff} + 2σ	Internal Moderator (H_2O) Density, g/cc
0.4122	0.0006	0.4133	0.0001
0.4762	0.0006	0.4774	0.05
0.5101	0.0007	0.5114	0.1
0.5660	0.0009	0.5678	0.2
0.6161	0.0010	0.6181	0.3
0.6634	0.0010	0.6655	0.4
0.7055	0.0012	0.7078	0.5
0.7424	0.0011	0.7446	0.6
0.7767	0.0013	0.7792	0.7
0.8088	0.0011	0.8111	0.8
0.8379	0.0013	0.8406	0.9
0.8636	0.0011	0.8659	1.0

Table A6.4-2

MP187/24PT1-DSC WE 14x14 SC Intact Fuel Results

(continued)

Normal Operating Condition: External Moderator Density Varying Assuming the Inward Assembly Position			
k_{eff}	+/- 1σ	k_{eff} + 2σ	External Moderator ($H_2$0) Density, g/cc
0.8605	0.0011	0.8628	0.0001
0.8623	0.0013	0.8649	0.05
0.8629	0.0013	0.8654	0.1
0.8644	0.0011	0.8665	0.2
0.8633	0.0010	0.8653	0.3
0.8639	0.0011	0.8661	0.4
0.8625	0.0014	0.8652	0.5
0.8617	0.0011	0.8640	0.6
0.8620	0.0012	0.8643	0.7
0.8593	0.0012	0.8618	0.8
0.8616	0.0013	0.8641	0.9
0.8641	0.0012	0.8666	1.0

Hypothetical Accident Condition: Internal Moderator Density Varying Assuming the Inward Assembly Position			
k_{eff}	+/- 1σ	k_{eff} + 2σ	Internal Moderator ($H_2$0) Density, g/cc
0.4118	0.0005	0.4128	0.0001
0.4764	0.0007	0.4777	0.05
0.5111	0.0006	0.5124	0.1
0.5644	0.0008	0.5661	0.2
0.6160	0.0011	0.6182	0.3
0.6623	0.0011	0.6646	0.4
0.7058	0.0010	0.7077	0.5
0.7429	0.0010	0.7449	0.6
0.7782	0.0012	0.7805	0.7
0.8088	0.0012	0.8113	0.8
0.8360	0.0012	0.8385	0.9
0.8631	0.0010	0.8651	1.0

Table A6.4-2

MP187/24PT1-DSC WE 14x14 SC Intact Fuel Results

(concluded)

Hypothetical Accident Condition: External Moderator Density Varying Assuming the Inward Assembly Position			
k_{eff}	+/- 1σ	k_{eff} + 2σ	External Moderator (H_2O) Density, g/cc
0.8625	0.0013	0.8651	0.0001
0.8622	0.0013	0.8647	0.05
0.8640	0.0013	0.8665	0.1
0.8635	0.0011	0.8657	0.2
0.8620	0.0011	0.8642	0.3
0.8626	0.0011	0.8647	0.4
0.8621	0.0012	0.8645	0.5
0.8622	0.0012	0.8647	0.6
0.8624	0.0011	0.8645	0.7
0.8650	0.0014	0.8677	0.8
0.8635	0.0011	0.8658	0.9
0.8599	0.0011	0.8621	1.0

Table A6.4-3

MP187/24PT1-DSC-DSC WE 14x14 MOX Intact Fuel Results

Normal Operating Condition: Assembly Position Study			
k_{eff}	+/- 1σ	k_{eff} + 2σ	Description
0.9061	0.0014	0.9089	The fuel assembly is located in the corner of each guidesleeve closest to the 24PT1-DSC centerline.
0.8952	0.0012	0.8976	The fuel assemblies are centered in each guidesleeve.
0.8771	0.0012	0.8795	The fuel assemblies are moved radially outwards from the center of the 24PT1-DSC
0.8875	0.0018	0.8911	The fuel assemblies are moved towards the upper left corner of each guidesleeve.

Normal Operating Condition: Internal Moderator Density Varying Assuming the Inward Assembly Position			
k_{eff}	+/- 1σ	k_{eff} + 2σ	Internal Moderator (H_20) Density, g/cc
0.4269	0.0005	0.4279	0.0001
0.5223	0.0008	0.5239	0.1
0.5799	0.0009	0.5817	0.2
0.6339	0.0011	0.6361	0.3
0.6819	0.0011	0.6841	0.4
0.7272	0.0011	0.7294	0.5
0.7694	0.0011	0.7716	0.6
0.8067	0.0012	0.8091	0.7
0.8461	0.0011	0.8483	0.8
0.8764	0.0013	0.8790	0.9
0.9061	0.0014	0.9089	1.0

Table A6.4-3

MP187/24PT1-DSC-DSC WE 14x14 MOX Intact Fuel Results

(continued)

Normal Operating Condition: External Moderator Density Varying Assuming the Inward Assembly Position			
k_{eff}	+/- 1σ	k_{eff} + 2σ	External Moderator (H_2O) Density, (g/cc)
0.9087	0.0012	0.9111	0.0001 g/cc
0.9068	0.0013	0.9094	0.05 g/cc
0.9058	0.0013	0.9084	0.1 g/cc
0.9062	0.0013	0.9088	0.2 g/cc
0.9065	0.0012	0.9089	0.3 g/cc
0.9071	0.0013	0.9097	0.4 g/cc
0.9047	0.0012	0.9071	0.5 g/cc
0.9077	0.0013	0.9103	0.6 g/cc
0.9067	0.0012	0.9091	0.7 g/cc
0.9052	0.0012	0.9076	0.8 g/cc
0.9073	0.0012	0.9097	0.9 g/cc
0.9069	0.0014	0.9097	1.0 g/cc

Hypothetical Accident Condition: Internal Moderator Density Varying Assuming the Inward Assembly Position			
k_{eff}	+/- 1σ	k_{eff} + 2σ	Internal Moderator (H_2O) Density, (g/cc)
0.4276	0.0005	0.4286	0.0001
0.4856	0.0005	0.4866	0.05
0.5217	0.0007	0.5231	0.1
0.5792	0.0009	0.5810	0.2
0.6326	0.0009	0.6344	0.3
0.6836	0.0011	0.6858	0.4
0.7290	0.0010	0.7310	0.5
0.7691	0.0012	0.7715	0.6
0.8086	0.0012	0.8110	0.7
0.8426	0.0011	0.8448	0.8
0.8775	0.0011	0.8797	0.9
0.9070	0.0011	0.9092	1.0

Table A6.4-3

MP187/24PT1-DSC-DSC WE 14x14 MOX Intact Fuel Results

(concluded)

Hypothetical Accident Condition: External Moderator Density Varying Assuming the Inward Assembly Position			
k_{eff}	**+/- 1σ**	**k_{eff} + 2σ**	**External Moderator (H_20) Density, (g/cc)**
0.9054	0.0014	0.9082	0.0001
0.9071	0.0012	0.9095	0.05
0.9061	0.0014	0.9089	0.1
0.9056	0.0013	0.9082	0.2
0.9071	0.0011	0.9093	0.3
0.9057	0.0012	0.9081	0.4
0.9086	0.0011	0.9108	0.5
0.9065	0.0011	0.9087	0.6
0.9047	0.0012	0.9071	0.7
0.9072	0.0012	0.9096	0.8
0.9063	0.0011	0.9085	0.9
0.9070	0.0011	0.9092	1.0

Table A6.4-4

MP187/24PT1-DSC WE 14x14 Damaged Fuel Results

STAINLESS STEEL CLAD FUEL

Variation of the Rod Pitch Study

k_{eff}	1 sigma	k_{eff} + 2 sigma	Rod Pitch (inches)
0.6366	0.0010	0.6385	0.422
0.6719	0.0013	0.6744	0.440
0.7462	0.0010	0.7482	0.480
0.8101	0.0012	0.8126	0.520
0.8571	0.0011	0.8593	0.556
0.8843	0.0011	0.8865	0.580
0.9118	0.0011	0.9139	0.610
0.9332	0.0011	0.9354	0.652

k_{eff}	1 sigma	k_{eff} + 2 sigma
0.8559	0.0011	0.8580
0.8608	0.0014	0.8635
0.8647	0.0013	0.8673
0.8642	0.0012	0.8665
0.8576	0.0010	0.8597

k_{eff}	1 sigma	k_{eff} + 2 sigma
0.8562	0.0012	0.8585
0.8584	0.0011	0.8607
0.8594	0.0013	0.8620

Table A6.4-4

MP187/24PT1-DSC WE 14x14 Damaged Fuel Results

STAINLESS STEEL CLAD FUEL

(continued)

Internal Moderator Density Varying for Most Reactive Rod Pitch Case

k_{eff}	1 sigma	k_{eff} + 2 sigma	Internal Moderator (H_20) Density, g/cc
0.4061	0.0004	0.4070	0.0001
0.4821	0.0008	0.4836	0.05
0.5285	0.0007	0.5298	0.10
0.6097	0.0010	0.6116	0.20
0.6786	0.0010	0.6805	0.30
0.7363	0.0010	0.7383	0.40
0.7852	0.0012	0.7875	0.50
0.8236	0.0011	0.8257	0.60
0.8583	0.0013	0.8610	0.70
0.8872	0.0011	0.8895	0.80
0.9119	0.0012	0.9143	0.90
0.9332	0.0011	0.9354	1.00

External Moderator Density Varying for Most Reactive Rod Pitch Case

k_{eff}	1 sigma	k_{eff} + 2 sigma	External Moderator (H_20) Density, g/cc
0.9333	0.0012	0.9356	0.0001
0.9319	0.0011	0.9340	0.05
0.9336	0.0011	0.9358	0.10
0.9305	0.0011	0.9327	0.20
0.9329	0.0013	0.9355	0.30
0.9323	0.0012	0.9347	0.40
0.9316	0.0012	0.9340	0.50
0.9328	0.0010	0.9348	0.60
0.9319	0.0013	0.9345	0.70
0.9340	0.0014	0.9368	0.80
0.9340	0.0012	0.9365	0.90
0.9330	0.0014	0.9358	1.00

Table A6.4-4

MP187/24PT1-DSC WE 14x14 Damaged Fuel Results

MIXED OXIDE FUEL

(continued)

MP187/24PT1-DSC WE 14x14 MOX Damaged Fuel Results - Rod Pitch

Rod Pitch, in	k_{eff}	σ	$k_{eff} + 2\sigma$
0.4220	0.8606	0.0012	0.8630
0.4400	0.8626	0.0010	0.8646
0.4800	0.8614	0.0011	0.8636
0.5200	0.8618	0.0012	0.8642
0.5560	0.8628	0.0010	0.8648
0.5800	0.8676	0.0011	0.8698
0.6100	0.8707	0.0011	0.8729
0.6520	0.8870	0.0010	0.8890

Distance Moved by Dislocated Rods, in	k_{eff}	σ	$k_{eff} + 2\sigma$
0.220	0.8694	0.0011	0.8716
0.450	0.8734	0.0011	0.8756
0.670	0.8741	0.0010	0.8761
0.890	0.8737	0.0013	0.8763
1.116	0.8747	0.0011	0.8769

Rods Dislocated	k_{eff}	σ	$k_{eff} + 2\sigma$
1/3	0.8708	0.0011	0.8730
1/4	0.8689	0.0010	0.8709
1/2	0.8714	0.0014	0.8742

Table A6.4-4

MP187/24PT1-DSC WE 14x14 Damaged Fuel Results

(concluded)

MIXED OXIDE FUEL

Most Reactive Configuration

Internal Moderator Varying for Bounding Rod Pitch Case

Density, g/cm³	k_{eff}	σ	$k_{eff} + 2\sigma$
0.0001	0.4119	0.0005	0.4129
0.1000	0.5166	0.0007	0.5180
0.2000	0.5749	0.0008	0.5765
0.3000	0.6265	0.0009	0.6283
0.4000	0.6743	0.0010	0.6763
0.5000	0.7171	0.0010	0.7191
0.6000	0.7523	0.0010	0.7543
0.7000	0.7887	0.0011	0.7909
0.8000	0.8218	0.0010	0.8238
0.9000	0.8524	0.0014	0.8552
1.0000	0.8870	0.0010	0.8890

External Moderator Varying for Bounding Rod Pitch Case

Density, g/cm³	k_{eff}	σ	$k_{eff} + 2\sigma$
0.0001	0.8863	0.0012	0.8887
0.1000	0.8875	0.0010	0.8895
0.2000	0.8846	0.0011	0.8868
0.3000	0.8837	0.0013	0.8863
0.4000	0.8833	0.0010	0.8853
0.5000	0.8862	0.0011	0.8884
0.6000	0.8897	0.0012	0.8921
0.7000	0.8866	0.0012	0.8890
0.8000	0.8862	0.0012	0.8886
0.9000	0.8881	0.0011	0.8903
1.0000	0.8874	0.0011	0.8896

Figure A6.4-1

Single-Ended Rod Shear Example

Figure A6.4-2

Double-Ended Rod Shear Example

A6.5 Critical Benchmark Experiments

A6.5.1 Benchmark Experiments and Applicability

A series of 137 benchmark experiments were simulated in calculations with the SCALE 4.4 PC/CSAS25 package [A6.8] using the 44-group cross-section library. 125 of these calculations were with uranium oxide fuel, while twelve were with mixed uranium/plutonium MOX fuel.

The benchmark problems used in this verification are representative of benchmarks of commercial light water reactor (LWR) fuels with the following characteristics:

- water moderation,
- boron neutron absorbers,
- unirradiated light water reactor type fuel (no fission products or "burnup credit"),
- close reflection,
- near room temperature (vs. reactor operating temperature), and
- Uranium oxide and MOX fuels.

Verification and validation (V&V) of the software is performed in accordance with QA program requirements (see Chapter 13). Sample cases are defined in the V&V report which are run prior to performance of an analysis on a new computer hardware configuration to assure consistency with the hardware used for V&V and benchmarking.

A6.5.1.1 Uranium Oxide Fuel Assemblies

The 125 uranium oxide experiments were chosen to model a wide range of uranium enrichments, fuel pin pitches, assembly separation, concentration of soluble boron and control elements in order to test the codes ability to accurately calculate k_{eff}. These experiments are discussed in detail in Reference [A6.17]. The input decks were in general taken from Reference [A6.17], however each input deck was modified to be consistent with use of CSAS25. The case ID names are identical to those used in Reference [A6.17].

A6.5.1.2 MOX Fuel Assemblies

In order to verify and validate the CSAS25 module for MOX fuel; twelve additional critical benchmark experiments were included in the evaluation. These experiments are discussed in Reference [A6.17]. The input decks for the twelve MOX cases were in general taken from Reference [A6.17], however each input deck was modified to be consistent with use of CSAS25. The case ID names are identical to those used in Reference [A6.17].

The MOX experiments include variations on fuel rod arrangement, fuel enrichment (both uranium and plutonium), fuel rod pitch, fuel rod diameter, fuel-to-moderator ratio, absorber materials and soluble boron concentration.

A6.5.2 Results of the Benchmark Calculations

A summary of all of the pertinent parameters for each experiment along with the results of each case is included in Table A6.5-1. The best correlation (linear regression correlation for each parameter vs. k_{eff}) is observed for fuel assembly separation distance, with a correlation of 0.65. All other parameters show much lower correlation ratios indicating no real correlation. All parameters were evaluated for trends and to determine the most conservative USL. Since there was no observable correlation, the worst case USL was selected for the identified parameters.

The USL is calculated in accordance to NUREG/CR-6361 [A6.17]. USL Method 1 (USL-1) applies a statistical calculation of the bias and its uncertainty plus an administrative margin (0.05) to the linear fit of results of the experimental benchmark data. The basis for the administrative margin is from Reference [A6.18]. Results from the USL evaluation are presented in Table A6.5-2.

The criticality evaluation presented here used the same cross section library, fuel materials and similar material/geometry options that were used in the 137 benchmark calculations as shown in Table A6.5-4. The modeling techniques and the applicable parameters listed in Table A6.5-4 for the actual criticality evaluations fall within the range of those addressed by the benchmarks in Table A6.5-1. The results from the comparisons of physical parameters of each of the fuel assembly types to the applicable USL value are presented in Table A6.5-4. The minimum value of the USL-1 was determined to be 0.9396 based on comparisons to the most limiting assembly parameters.

Table A6.5-1

Benchmark Results

Run ID	U Enrich. weight %	Pu Enrich. weight %	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	k_{eff}	1σ
B1645SO1	2.46		1.41	1.015		32.8194	0.9967	0.0009
B1645SO2	2.46		1.41	1.015		32.7584	1.0002	0.0011
BW1231B1	4.02		1.511	1.139		31.1427	0.9966	0.0012
BW1231B2	4.02		1.511	1.139		29.8854	0.9972	0.0009
BW1273M	2.46		1.511	1.376		32.2106	0.9965	0.0009
BW1484A1	2.46		1.636	1.841	1.636	34.5304	0.9962	0.0010
BW1484A2	2.46		1.636	1.841	4.908	35.1629	0.9931	0.0010
BW1484B1	2.46		1.636	1.841		33.9421	0.9979	0.0010
BW1484B2	2.46		1.636	1.841	1.636	34.5820	0.9955	0.0012
BW1484B3	2.46		1.636	1.841	4.908	35.2609	0.9969	0.0011
BW1484C1	2.46		1.636	1.841	1.636	34.6463	0.9931	0.0011
BW1484C2	2.46		1.636	1.841	4.908	35.2422	0.9939	0.0012
BW1484S1	2.46		1.636	1.841	1.636	34.5105	1.0001	0.0010
BW1484S2	2.46		1.636	1.841	1.636	34.5569	0.9992	0.0010
BW1484SL	2.46		1.636	1.841	6.544	35.4151	0.9935	0.0011
BW1645S1	2.46		1.209	0.383	1.778	30.1040	0.9990	0.0010
BW1645S2	2.46		1.209	0.383	1.778	29.9961	1.0037	0.0011
BW1810A	2.46		1.636	1.841		33.9465	0.9984	0.0008
BW1810B	2.46		1.636	1.841		33.9631	0.9984	0.0009
BW1810C	2.46		1.636	1.841		33.1569	0.9992	0.0010
BW1810D	2.46		1.636	1.841		33.0821	0.9985	0.0013
BW1810E	2.46		1.636	1.841		33.1600	0.9988	0.0009
BW1810F	2.46		1.636	1.841		33.9556	1.0031	0.0011
BW1810G	2.46		1.636	1.841		32.9409	0.9973	0.0011
BW1810H	2.46		1.636	1.841		32.9420	0.9972	0.0011
BW1810I	2.46		1.636	1.841		33.9655	1.0037	0.0009
BW1810J	2.46		1.636	1.841		33.1403	0.9983	0.0011
DSN399-1	4.74		1.6	3.807	1.8	33.9520	1.0036	0.0015
DSN399-2	4.74		1.6	3.807	5.8	34.4207	0.9989	0.0016
DSN399-3	4.74		1.6	3.807		35.3140	1.0024	0.0015
DSN399-4	4.74		1.6	3.807		35.3784	0.9977	0.0013
EPRU65	2.35		1.562	1.196		33.9106	0.9960	0.0011
EPRU65B	2.35		1.562	1.196		33.4013	0.9993	0.0012
EPRU75	2.35		1.905	2.408		35.8671	0.9958	0.0010
EPRU75B	2.35		1.905	2.408		35.3043	0.9996	0.0010
EPRU87	2.35		2.21	3.687		36.6129	1.0007	0.0011
EPRU87B	2.35		2.21	3.687		36.3499	1.0007	0.0011
NSE71SQ	4.74		1.26	1.823		33.7610	0.9979	0.0012
NSE71W1	4.74		1.26	1.823		34.0129	0.9988	0.0013
NSE71W2	4.74		1.26	1.823		36.3037	0.9957	0.0010
P2438BA	2.35		2.032	2.918	5.05	36.2277	0.9979	0.0013
P2438SLG	2.35		2.032	2.918	8.39	36.2889	0.9986	0.0012
P2438SS	2.35		2.032	2.918	6.88	36.2705	0.9974	0.0011
P2438ZR	2.35		2.032	2.918	8.79	36.2840	0.9987	0.0010
P2615BA	4.31		2.54	3.883	6.72	35.7286	1.0019	0.0014
P2615SS	4.31		2.54	3.883	8.58	35.7495	0.9952	0.0015
P2615ZR	4.31		2.54	3.883	10.92	35.7700	0.9977	0.0014
P2827L1	2.35		2.032	2.918	13.27	36.2526	1.0057	0.0011
P2827L2	2.35		2.032	2.918	11.25	36.2908	0.9999	0.0012

Table A6.5-1

Benchmark Results

(Continued)

CaseID	U Enrich. weight %	Pu Enrich. weight %	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	k_{eff}	1σ
P2827L3	4.31		2.54	3.883	20.78	35.6766	1.0092	0.0012
P2827L4	4.31		2.54	3.883	19.04	35.7131	1.0073	0.0012
P2827SLG	2.35		2.032	2.918	8.31	36.3037	0.9957	0.0010
P3314BA	4.31		1.892	1.6	2.83	33.1881	0.9988	0.0012
P3314BC	4.31		1.892	1.6	2.83	33.2284	0.9992	0.0012
P3314BF1	4.31		1.892	1.6	2.83	33.2505	1.0037	0.0013
P3314BF2	4.31		1.892	1.6	2.83	33.2184	1.0009	0.0013
P3314BS1	2.35		1.684	1.6	3.86	34.8594	0.9956	0.0013
P3314BS2	2.35		1.684	1.6	3.46	34.8356	0.9949	0.0010
P3314BS3	4.31		1.892	1.6	7.23	33.4247	0.9970	0.0013
P3314BS4	4.31		1.892	1.6	6.63	33.4162	0.9998	0.0012
P3314SLG	4.31		1.892	1.6	2.83	34.0198	0.9974	0.0012
P3314SS1	4.31		1.892	1.6	2.83	33.9601	0.9999	0.0012
P3314SS2	4.31		1.892	1.6	2.83	33.7755	1.0022	0.0012
P3314SS3	4.31		1.892	1.6	2.83	33.8904	0.9992	0.0013
P3314SS4	4.31		1.892	1.6	2.83	33.7625	0.9958	0.0011
P3314SS5	2.35		1.684	1.6	7.8	34.9531	0.9949	0.0013
P3314SS6	4.31		1.892	1.6	10.52	33.5333	1.0020	0.0011
P3314W1	4.31		1.892	1.6		34.3994	1.0024	0.0013
P3314W2	2.35		1.684	1.6		35.2167	0.9969	0.0011
P3314ZR	4.31		1.892	1.6	2.83	33.9954	0.9971	0.0013
P3602BB	4.31		1.892	1.6	8.3	33.3221	1.0029	0.0013
P3602BS1	2.35		1.684	1.6	4.8	34.7750	1.0027	0.0012
P3602BS2	4.31		1.892	1.6	9.83	33.3679	1.0039	0.0012
P3602N11	2.35		1.684	1.6	8.98	34.7438	1.0023	0.0012
P3602N12	2.35		1.684	1.6	9.58	34.8391	1.0030	0.0012
P3602N13	2.35		1.684	1.6	9.66	34.9337	1.0013	0.0012
P3602N14	2.35		1.684	1.6	8.54	35.0282	0.9974	0.0013
P3602N21	2.35		2.032	2.918	11.2	36.2821	0.9987	0.0011
P3602N22	2.35		2.032	2.918	10.36	36.1896	1.0025	0.0011
P3602N31	4.31		1.892	1.6	14.87	33.2094	1.0057	0.0013
P3602N32	4.31		1.892	1.6	15.74	33.3067	1.0093	0.0012
P3602N33	4.31		1.892	1.6	15.87	33.4174	1.0107	0.0012
P3602N34	4.31		1.892	1.6	15.84	33.4683	1.0045	0.0013
P3602N35	4.31		1.892	1.6	15.45	33.5185	1.0013	0.0012
P3602N36	4.31		1.892	1.6	13.82	33.5855	1.0004	0.0014
P3602N41	4.31		2.54	3.883	12.89	35.5276	1.0109	0.0013
P3602N42	4.31		2.54	3.883	14.12	35.6695	1.0071	0.0014
P3602N43	4.31		2.54	3.883	12.44	35.7542	1.0053	0.0015
P3602SS1	2.35		1.684	1.6	8.28	34.8701	1.0025	0.0013
P3602SS2	4.31		1.892	1.6	13.75	33.4202	1.0035	0.0012
P3926L1	2.35		1.684	1.6	10.06	34.8519	1.0000	0.0011
P3926L2	2.35		1.684	1.6	10.11	34.9324	1.0017	0.0011
P3926L3	2.35		1.684	1.6	8.5	35.0641	0.9949	0.0012
P3926L4	4.31		1.892	1.6	17.74	33.3243	1.0074	0.0014
P3926L5	4.31		1.892	1.6	18.18	33.4074	1.0057	0.0013
P3926L6	4.31		1.892	1.6	17.43	33.5246	1.0046	0.0013
P3926SL1	2.35		1.684	1.6	6.59	33.4737	0.9995	0.0012
P3926SL2	4.31		1.892	1.6	12.79	33.5776	1.0007	0.0012

Table A6.5-1

Benchmark Results

(Concluded)

CaseID	U Enrich. weight %	Pu Enrich. weight %	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	k_{eff}	1σ
P4267B1	4.31		1.8901	1.59		31.8075	0.9990	0.0010
P4267B2	4.31		0.89	1.59		31.5323	1.0033	0.0010
P4267B3	4.31		1.715	1.09		30.9905	1.0050	0.0011
P4267B4	4.31		1.715	1.09		30.5061	0.9996	0.0011
P4267B5	4.31		1.715	1.09		30.1011	1.0004	0.0011
P4267SL1	4.31		1.89	1.59		33.4737	0.9995	0.0012
P4267SL2	4.31		1.715	1.09		31.9460	0.9988	0.0016
P62FT231	4.31		1.891	1.6	5.19	32.9196	1.0012	0.0013
P71F14F3	4.31		1.891	1.6	5.19	32.8237	1.0009	0.0014
P71F14V3	4.31		1.891	1.6	5.19	32.8597	0.9972	0.0014
P71F14V5	4.31		1.891	1.6	5.19	32.8609	0.9993	0.0013
P71F214R	4.31		1.891	1.6	5.19	32.8778	0.9969	0.0012
PAT80L1	4.74		1.6	3.807	4.9	35.0253	1.0012	0.0012
PAT80L2	4.74		1.6	3.807	4.9	35.1136	0.9993	0.0015
PAT80SS1	4.74		1.6	3.807	4.9	35.0045	0.9988	0.0013
PAT80SS2	4.74		1.6	3.807	4.9	35.1072	0.9960	0.0013
W3269A	5.7		1.422	1.93		33.1480	0.9988	0.0012
W3269B1	3.7		1.105	1.432		32.4055	0.9961	0.0011
W3269B2	3.7		1.105	1.432		32.3921	0.9963	0.0011
W3269B3	3.7		1.105	1.432		32.2363	0.9944	0.0011
W3269C	2.72		1.524	1.494		33.7727	0.9989	0.0012
W3269SL1	2.72		1.524	1.494		33.3850	0.9981	0.0014
W3269SL2	5.7		1.422	1.93		33.0910	1.0005	0.0013
W3269W1	2.72		1.524	1.494		33.5114	0.9966	0.0014
W3269W2	5.7		1.422	1.93		33.1680	1.0014	0.0014
W3385SL1	5.74		1.422	1.932		33.2387	1.0009	0.0012
W3385SL2	5.74		2.012	5.067		35.8818	0.9997	0.0013
EPRI70UN	0.71	2	1.778	1.2		31.6775	0.9983	0.0012
EPRI70B	0.71	2	1.778	1.2		30.9021	1.0009	0.0012
EPRI87UN	0.71	2	2.2098	2.53		33.3230	1.0096	0.0011
EPRI87B	0.71	2	2.2098	2.53		31.6775	0.9983	0.0012
EPRI99UN	0.71	2	2.5146	3.64		35.1817	1.0063	0.0011
EPRI99B	0.71	2	2.5146	3.64		34.4098	1.0095	0.0011
SAXTON52	0.71	6.6	1.3208	1.68		30.2980	1.0020	0.0014
SAXTON56	0.71	6.6	1.4224	2.16		31.4724	1.0010	0.0014
SAXTON56B	0.71	6.6	1.4224	2.16		31.0038	0.9994	0.0013
SAXTN735	0.71	6.6	1.8669	4.7		34.1848	1.0007	0.0016
SATN792	0.71	6.6	2.01168	5.67		34.6401	1.0026	0.0013
SAXTN104	0.71	6.6	2.6416	10.75		35.8333	1.0054	0.0014
Correlation	0.31	-0.26	0.43	0.25	0.65	-0.01	N/A	N/A

Table A6.5-2

USL-1 Results

Parameter	Range of Applicability	Upper Subcritical Limit as Determined by the USL-1 Method for the 6 Parameters that Could Show Correlation to k_{eff}
U Enrichment (weight % ^{235}U)	2.4 2.8 3.3 3.8 – 5.7	0.9424 0.9430 0.9435 0.9438
Pu Enrichment (weight % Pu)	2.0 – 6.6	0.9417
Fuel Rod Pitch (cm)	0.89 1.1 1.4 1.6 1.9 – 2.6	0.9396 0.9408 0.9421 0.9433 0.9439
Water/Fuel Volume Ratio	0.38 1.9 3.3 – 11	0.9414 0.9425 0.9426
Assembly Separation (cm)	1.6 4.4 7.1 9.8 – 21	0.9410 0.9425 0.9440 0.9441
Average Energy Group Causing Fission (AEG)	30 – 37	0.9433

Table A6.5-3

Fuel Assembly Design Parameters Used in Criticality Benchmarks

Table A6.5-4

Limiting Upper Subcritical Limit Based on Method 1 for the WE 14x14 SC Fuel Assemblies and the WE 14x14 MOX Fuel Assemblies

A6.6 Appendix

A6.6.1 References

A6.6.2 KENO Input Files

A6.6.1 References

A6.1 Code of Federal Regulations, Title 10, Part 71, "Packaging and Transportation of Radioactive Material."

A6.2 ANSI/ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

A6.3 ANSI/ANS-8.17-1984, "Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors."

A6.4 USNRC Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety," Revision 1, May, 1977.

A6.5 ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety."

A6.6 USNRC Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," Proposed Revision 2, December 1981 (for guidance only).

A6.7 "Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 5, TNW, August 2000.

A6.8 "SCALE, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vol. I-III, September 1998.

A6.9 Transnuclear West Report / Project No.: SCE-01.0602, "Verification and Validation Document: SCALE 4.4 PC; CSAS25 For Uranium Oxide and Uranium Plutonium Mixed Oxide (MOX) Fuel", Rev. 1.

A6.10 Burn, Reed R., "Boral Accelerated Radiation Aging Tests," Nuclear Reactor Laboratory, University of Michigan, Ann Arbor, Michigan, May 9, 1990.

A6.11 Bierman, S.R., et al., "Criticality Experiments with Subcritical Clusters of 2.35 Wt% and 4.31 Wt% ^{235}U Enriched UO2 Rods in Water With Steel Reflecting Walls," Battelle Pacific Northwest Laboratories, NUREG/CR-1784, April, 1981.

A6.12 Bierman, S.R., et al., "Critical Separation Between Subcritical Clusters of 4.29 Wt% ^{235}U Enriched UO2 Rods in Water With Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories, NUREG/CR-0073, May 1978.

A6.13 Bierman, S.R., et al., "Criticality Experiments with Subcritical Clusters of 2.35 Wt% and 4.31 Wt% ^{235}U Enriched UO2 Rods in Water With Uranium or Lead Reflecting Walls," Battelle Pacific Northwest Laboratories, NUREG/CR-0796, August, 1981.

A6.14 Bierman, S.R., et al., "Critical Separation Between Subcritical Clusters of 2.35 Wt% ^{235}U Enriched UO2 Rods in Water with Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories, PNL-2438, October, 1977.

A6.15 Baldwin, M.N., et al., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, Babcock & Wilcox Company, Lynchburg, Virginia, July, 1979.

A6.16 Bierman, S.R., "Reactivity Measurements on an Experimental Assembly of 4.31 Wt% ^{235}U Enriched UO2 Fuel Rods Arranged in a Shipping Cask Geometry," Battelle Pacific Northwest Laboratories, PNL-6838, October, 1989.

A6.17 U.S. Nuclear Regulatory Commission, "Criticality Benchmark Guide for Light-Water Reactor fuel in Transportation and Storage Packages," NUREG/CR-6361, ORNL/TM-13211, March 1997.

A6.18 U.S. Nuclear Regulatory Commission, "Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages," NUREG/CR-5661, ORNL/TM-11936, April 1997.

A6.6.2 KENO Input Files

A7. OPERATING PROCEDURES

The procedures associated with the MP187 Cask operation for the 24PT1-DSC are identical to the FO/FC DSCs described in Chapter 7. The changes in Package loading procedures associated with specific requirements of the 24PT1-DSC as they deviate from the procedures specified in Chapter 7 are identified below. The sections not described below remain unchanged and are as specified in Chapter 7.

A7.1 Procedures for Loading the Package

A7.1.1 Preparation of the NUHOMS®-MP187 Cask for Use

No change

A7.1.2 Wet Loading the NUHOMS®-MP187 Cask and DSC

The NUHOMS®-MP187 Cask is designed to transport one 24PT1-DSC containing up to twenty-four intact WE 14x14 PWR fuel assemblies, with up to 4 damaged WE 14x14 PWR fuel assemblies or one damaged MOX assembly, placed in 24PT1-DSC Failed Fuel Cans, with the balance of the assemblies intact. Two empty slots are allowed when placed in symmetrically opposite locations in the DSC. All remaining fuel assembly locations are to be loaded with design basis fuel assemblies or dummy assemblies of the same weight.

a. Verify that the fuel assemblies to be placed in the 24PT1-DSC meet the maximum burnup, maximum initial enrichment, minimum cooling time and decay heat limits specified in Section A1.2.3. The potential of fuel misloading is essentially eliminated through the implementation of multiple procedural and administrative barriers. The controls instituted to ensure that fuel assemblies are loaded into a known cell location within a DSC will typically take the following form:

    - A cask/DSC loading plan is developed to compare various parameters including maximum burn, maximum initial enrichment, and minimum cooling time requirements.

    - The loading plan is independently verified and approved.

    - A fuel movement schedule is then written, verified and approved based upon the loading plan. All fuel movements from any rack location are performed under strict verbatim compliance of the fuel movement schedule.

- All fuel assemblies are videotaped and independently verified by ID number to match the movement schedule prior to the placement of the shield plug.

- A third verification is performed in preparation for the DOE reporting requirements, as required. This third verification independently verifies that fuel in the DSC is placed per the original cask loading plan.

Note: Damaged fuel assemblies to be loaded in the 24PT1-DSC Failed Fuel Cans must meet the criteria in Section A1.2.3 for damaged fuel, having no more than 14 known or suspected damaged fuel rods.

Subsections A7.1.2.b through A7.1.2.j remain unchanged as specified in Section 7.1.2.

k. Lower the cask and 24PT1-DSC into the fuel pool and load the fuel assemblies, dummy fuel assemblies or damaged fuel assemblies (up to four damaged WE 14x14 fuel assemblies, or one MOX assembly, to be placed within 24PT1-DSC Failed Fuel Cans which are located inside their respective guidesleeves).

Note: 1. Bottom fuel spacers are to be placed in each fuel assembly location prior to placing the DSC in the cask and top fuel spacers shall be installed after loading of fuel but prior to installation of the top shield plug.

Note: 2. Damaged fuel assemblies may be placed into their removable Failed Fuel Cans directly in the 24PT1-DSC and then the can lids installed, or loaded in a separate area and the loaded Failed Fuel Can placed into the 24PT1-DSC. No top fuel spacer is used when loading a Failed Fuel Can.

Subsections A7.1.2.*l* through A7.1.2.q remain unchanged as specified in Section 7.1.2.

r. Backfill the DSC cavity with helium from 0 psig to 3 psig.

s. Install and seal weld the DSC outer cover plate.

t. Install and weld the outer top cover plate and outer top cover plate leak test port. No helium leak testing of closure welds is required since no credit for the DSC shell boundary is taken in this safety analysis report.

Subsections A7.1.2.u through A7.1.2.v remain unchanged as specified in Section 7.1.2.

A7.1.3 Transferring the DSC to an Onsite Storage Facility

No change

A7.1.4 Placing the DSC in an HSM for Storage

No change

A7.1.5 Loading the DSC into the Cask from an HSM

No change

A7.1.6 Placing the DSC in Metal Cask Storage at an Onsite Facility

No change

A7.1.7 Preparing the Cask for Transportation

No change

A7.1.8 Placing the Cask on the Railcar

No change

A7.1.9 Glossary

No change

A8. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The acceptance tests and maintenance program described in Chapter 8 are associated with the MP187 Cask and do not include tests or maintenance requirements for the DSC. Acceptance tests for the DSC are identified in Chapter A7 and under the associated storage system testing requirements. Since the 24PT1-DSC payload does not alter the Cask configuration and associated testing, the requirements specified in Chapter 8 are applicable to the MP187 Cask with the 24PT1-DSC payload.

AFFIDAVIT PURSUANT
TO 10 CFR 2.790

Transnuclear West Inc.)
State of California) SS.
County of Alameda)

I, Robert M. Grenier, depose and say that I am President and Chief Operating Officer of Transnuclear West Inc., duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in the documents included in Attachments C of this submittal. The specific sections/tables considered proprietary are:

1. Section A1.3.2, Drawing NUH-05-4010, Rev. 0
2. Table A2.6-1
3. Figure A2.7-1
4. Table A3.6-1
5. Table A3.6-2
6. Table A3.6-4
7. Table A5.2-2
8. Table A5.2-4
9. Figure A5.3-1
10. Figure A5.3-2
11. Section A5.5.2
12. Section A5.5.3
13. Figure A6.3-1
14. Section A6.4.2.1
15. Table A6.4-4
16. Figure A6.4-1
17. Figure A6.4-2
18. Table A6.5-3
19. Table A6.5-4
20. Section A6.6.2

The proprietary sections of these documents have been so designated in the Attachments.

I have personal knowledge of the criteria and procedures utilized by Transnuclear West Inc. in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1) The information sought to be withheld from public disclosure is design drawings and analyses of the NUHOMS® Cask, which is owned and has been held in confidence by Transnuclear West Inc.

2) The information is of a type customarily held in confidence by Transnuclear West Inc. and not customarily disclosed to the public. Transnuclear West Inc. has a rational basis for determining the types of information customarily held in confidence by it.

3) The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.

4) The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

5) Public disclosure of the information is likely to cause substantial harm to the competitive position of Transnuclear West Inc. because:

   a) A similar product is manufactured and sold by competitors of Transnuclear West Inc.

   b) Development of this information by Transnuclear West Inc. required thousands of man-hours and hundreds of thousands of dollars. To the best of my knowledge and belief, a competitor would have to undergo similar expense in generating equivalent information.

   c) In order to acquire such information, a competitor would also require considerable time and inconvenience related to the development of a design and analysis of a dry spent fuel storage system.

   d) The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

   e) The information consists of description of the design and analysis of a dry spent fuel storage and transportation system, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Transnuclear West Inc., take marketing or other actions to improve their product's position or impair the position of Transnuclear West's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

   f) In pricing Transnuclear West's products and services, significant research, development, engineering, analytical, licensing, quality assurance and other costs and expenses must be included. The ability of Transnuclear West's competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

Further the deponent sayeth not.

SHANNON L. CHRISTENSEN
COMM. # 1264196
NOTARY PUBLIC-CALIFORNIA
ALAMEDA COUNTY
COMM. EXP. MAY 18, 2004

Robert M. Grenier

Robert M. Grenier
President and Chief Operating Officer
Transnuclear West Inc.

Subscribed and sworn to me before this 30th day of January, 2001, by Robert M. Grenier.

Shannon L. Christensen

Notary Public

ATTACHMENT C

CHANGED MP187 SAFETY ANALYSIS REPORT PAGES,

DOCUMENT NUH-05-151, REVISION 11

ATTACHMENT D

INSTRUCTIONS FOR UPDATE OF MP187 SAR FROM REVISION 10 TO REVISION 11

These instructions are to be used for incorporating changes to the MP187 SAR, Revision 10 to create the Revision 11 SAR, Proprietary Version

• Cover Sheet	Replace Revision 10 page with Revision 11 page
• Pages ii and iii	Replace Revision 10 pages with Revision 11 page
• Page ix	Replace Revision 10 page with Revision 11 page
• Page xxvii	Replace Revision 10 page with Revision 11 page
• Page xl and xli	Replace Revision 10 pages with Revision 11 page xli
• Page 1-1	Replace Revision 10 page with Revision 11 page

- Add new Volume, Proprietary Version of Appendix A to the SAR

These instructions are to be used for incorporating changes to the Non Proprietary Version of MP187 SAR, Revision 10, to create the Revision 11 SAR, Non-Proprietary Version.

• Cover Sheet	Replace Revision 10 page with Revision 11 page
• Pages ii and iii	Replace Revision 10 pages with Revision 11 page
• Page ix	Replace Revision 10 page with Revision 11 page
• Page xxv	Replace Revision 10 page with Revision 11 page
• Page xxxv and xxxvi	Replace Revision 10 page with Revision 11 page
• Page xli	Replace Revision 10 pages with Revision 11 page xli
• Page 1-1	Replace Revision 10 page with Revision 11 page

- Add new Volume, Non Proprietary Version of Appendix A to the SAR