71-9255

TRANSNUCLEAR WEST

August 24, 2001
NUH05-01-837
RMG-01-042

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

Subject: Response to Request for Additional Information (RAI) and Submittal of Amendment No. 6, Revision 1 of the NUHOMS® MP187 Multi-Purpose Cask, 10 CFR71 Certificate of Compliance No. 9255 (TAC No. L23264)

References:
1. Request for Additional Information Regarding Approval to Store Spent Nuclear Fuel in the NUHOMS® MP187 Multi-Purpose Cask (TAC No. L23264)
2. Grenier to Kobetz Letter NUH05-01-725, Dated 1/30/01, Application for Amendment of the NUHOMS® MP187 Multi-Purpose Cask, 10 CFR 71 Certificate of Compliance No. 9255, Amendment No. 6

Dear Mr. Kobetz:

Transnuclear West Inc., (TN West) herewith submits the responses to the RAI (Reference 1) and Revision 12 of the MP187 Multi-Purpose Cask SAR. This package includes:

1. Affidavit for withholding proprietary information (Attachment A)
2. Responses to the RAI (Attachment B)
3. Suggested changes to CoC 71-9255 Revision No. 5, revised to address changes associated with this RAI response (Attachment C)
4. Instructions for incorporating changed pages to update the MP187 SAR (Attachment D)
5. Revision 12 replacement pages for the proprietary version of the NUHOMS® MP187 Multi-Purpose Cask SAR (Attachment E)
6. Revision 12 replacement pages for the non-proprietary version of the NUHOMS® MP187 Multi-Purpose Cask SAR (Attachment F).

Revision 12 of the SAR reflects the changes resulting from the RAI responses. In addition, TN West has revised the SAR to include the latest 24PT1-DSC drawings. The MOX assembly maximum heat load, the 24PT1-DSC maximum heat load when the payload consists of at least one MOX assembly, and the short term temperature limit for the stainless steel cladding

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

NMSS01 Prop

have also been revised. These additional changes make the SAR consistent with the Advanced NUHOMS Storage Application (Docket 72-1029).

Please contact Mr. U. B. Chopra (510-744-6053) or me (510-744-6020) if you have any questions regarding this transmittal.

Sincerely,

Robert M. Grenier
President and Chief Operating Officer

Docket 71-9255

cc: File: NUH05.0003
SCE-01-0007.01

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 document included in Attachment E of this submittal and as listed below:

- Safety Analysis Report for the NUHOMS® -MP187 Cask, Revision 12.

This document has been appropriately designated as proprietary.

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.

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

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

Subscribed and sworn to me before this 24th day of August, 2001, by Robert M. Grenier.

Notary Public

TRANSNUCLEAR WEST

August 24, 2001
NUH05-01-837
RMG-01-042

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

Subject: Response to Request for Additional Information (RAI) and Submittal of Amendment No. 6, Revision 1 of the NUHOMS® MP187 Multi-Purpose Cask, 10 CFR71 Certificate of Compliance No. 9255 (TAC No. L23264)

References:
1. Request for Additional Information Regarding Approval to Store Spent Nuclear Fuel in the NUHOMS® MP187 Multi-Purpose Cask (TAC No. L23264)
2. Grenier to Kobetz Letter NUH05-01-725, Dated 1/30/01, Application for Amendment of the NUHOMS® MP187 Multi-Purpose Cask, 10 CFR 71 Certificate of Compliance No. 9255, Amendment No. 6

Dear Mr. Kobetz:

Transnuclear West Inc., (TN West) herewith submits the responses to the RAI (Reference 1) and Revision 12 of the MP187 Multi-Purpose Cask SAR. This package includes:

1. Affidavit for withholding proprietary information (Attachment A)
2. Responses to the RAI (Attachment B)
3. Suggested changes to CoC 71-9255 Revision No. 5, revised to address changes associated with this RAI response (Attachment C)
4. Instructions for incorporating changed pages to update the MP187 SAR (Attachment D)
5. Revision 12 replacement pages for the proprietary version of the NUHOMS® MP187 Multi-Purpose Cask SAR (Attachment E)
6. Revision 12 replacement pages for the non-proprietary version of the NUHOMS® MP187 Multi-Purpose Cask SAR (Attachment F).

Revision 12 of the SAR reflects the changes resulting from the RAI responses. In addition, TN West has revised the SAR to include the latest 24PT1-DSC drawings. The MOX assembly maximum heat load, the 24PT1-DSC maximum heat load when the payload consists of at least one MOX assembly, and the short term temperature limit for the stainless steel cladding

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

have also been revised. These additional changes make the SAR consistent with the Advanced NUHOMS Storage Application (Docket 72-1029).

Please contact Mr. U. B. Chopra (510-744-6053) or me (510-744-6020) if you have any questions regarding this transmittal.

Sincerely,

Robert M. Grenier
President and Chief Operating Officer

Docket 71-9255

cc: File: NUH05.0003
SCE-01-0007.01

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 document included in Attachment E of this submittal and as listed below:

- Safety Analysis Report for the NUHOMS® -MP187 Cask, Revision 12.

This document has been appropriately designated as proprietary.

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
President and Chief Operating Officer
Transnuclear West Inc.

Subscribed and sworn to me before this 24th day of August, 2001, by Robert M. Grenier.

Notary Public

ATTACHMENT B

RESPONSES TO RAI

Attachment B

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
TRANSNUCLEAR WEST INC., TAC NO. L23264

Chapter 1 General

Question 1-1

Clarify the Pu enrichments in the mixed oxide fuel descriptions.

The fissile Pu enrichments listed in Table A.1.2-1, "WE 14x14 Spent Fuel Assembly Descriptions," and Table A.5.1, "Design Basis Fuel Parameters," are different from the fissile Pu enrichments listed in Table A.5.2-1, "SAS2/ORIGEN Input Cases for Shielding Evaluation," and Table 6.2-1, "Maximum Fuel Loading Parameters." Source term information needs to be clearly specified for the staff to evaluate the external radiation doses per 10 CFR 71.47.

Response Question 1-1

The fissile Pu enrichment specified in Tables A1.2-1, A5-1, A5.2-1 and A6.2-1 has been clarified.

The MOX fuel assemblies each contain fuel rods with different plutonium enrichments. Minimum enrichments are used for shielding analyses and maximum enrichments are used for criticality analyses. The corrected Pu enrichments are shown below.

Number of Fuel Rods	Minimum Fissile Pu Weight Percent*	Maximum Fissile Pu Weight Percent*	Total Pu Weight Percent
64	2.78	2.84	3.30
92	3.05	3.10	3.65
24	3.25	3.31	3.85

*Fissile Pu weight percent = (weight of Pu^{239} + weight of Pu^{241})/(total weight of Pu + Total weight of U)

Maximum fissile Pu weight percent values have been added to SAR Table A1.2-1 and the minimum enrichment values for Pu have been clarified to specify that these values are for fissile Pu. The same changes have been made to SAR Table A5-1. The values specified in SAR Table A5.2-1 are minimum initial enrichment. The description of these values has also been clarified to specify that they are fissile Pu enrichment values. The enrichment values specified in SAR Table A6.2-2 for criticality analysis are for maximum initial enrichment. This table has been revised to clarify the values specified. SAR Figure A6.3-5 has also been revised to specify the maximum fissile Pu enrichment.

Question 1-2

Clarify footnote 1 to Table A.1.2-1, "WE 14x14 Spent Fuel Assembly Descriptions."

The footnote indicates that stainless steel clad fuel assemblies that have been cooled less than the minimum cooling time of 38 years can be loaded for shipment. Based upon the shielding evaluation provided in Chapter A5 of the Safety Analysis Report (SAR), the minimum cooling time has been determined to be 38 years for the stainless steel clad fuel assemblies. If the 38 year cooling requirement needs to be shortened, then the evaluation in the SAR should take this into account and shorter cooling times should be determined. This information is required by staff to assess compliance with 10 CFR 71.33 and 71.47.

Response Question 1-2

A new section has been added to the SAR (Section A5.2.3) to address the mechanism by which payloads with shorter decay times may be qualified in the 24PT1-DSC/MP187 cask for transportation. Shorter decay times are allowed if analyses consistent with those provided in Chapter A5 and Chapter 5 of the SAR are used to demonstrate that the dose rate with shorter decay times will not result in a dose rate that is higher than that specified in Chapter A5 of the SAR. Measured data may also be used in qualifying the alternate decay time payload with appropriate correction factors to the dose rate to account for uncertainties in the measurements. The detailed methodology for qualifying these payloads is discussed in the new SAR Section A5.2.3.

NOTE: In addition to the changes implemented to SAR Chapter 1 in response to RAI 1-1, TN West has included the revised DSC drawings NUH-05-4040 and NUH-05-4010NP in SAR Section 1.5.2 to make the DSC configuration identical to that described in the Advanced NUHOMS® Storage System Application [Docket No. 72-1029]. These changes enhance fabricability, correct minor interference errors, provide clarification to the notes shown on these drawings and do not have any impact on the analysis presented in the SAR.

Chapter 2 Structural

Question 2-1

Demonstrate that the brittle fracture and fatigue evaluations for the FO/FC-DSC spacer discs presented in sections 2.1.2.3.1.B and 2.12.3.2.C are applicable to the 24PT1-DSC.

FO/FC-DSC loads are both heavier than the 24PT1-DSC. Since neither load is lighter than the 24PT1-DSC it is not clear how their evaluations bound the 24PT1-DSC. This is required for the staff to assess compliance with 10 CFR 71.41(a).

Response to Question 2-1

Brittle Fracture

As described in Sections 2.1.2.3.1.B, 2.3.2 and A.2.1.2 the nil-ductility requirements for the spacer disc material are identical for the FO/FC-DSC and the 24PT1-DSC spacer discs. The material selected for the 24PT1-DSC is the same as that used for the FO/FC-DSC. The nil-ductility requirements are independent of the applied stress level and are not altered by any reduction in stress level of the 24PT1-DSC spacer discs.

Fatigue

As described in SAR Section 2.1.2.3.2.C, the fatigue evaluation for the 24PT1-DSC is performed for the vibration and thermal loads applied to the most highly stressed FO/FC-DSC spacer disc. The assumed dead weight load distribution analyzed for the FO/FC-DSC spacer discs is 2607 lbs, as discussed in SAR Section 2.6.11.3.1. This load is greater than the 24PT1-DSC spacer disc dead weight load of 2467.3 lbs, presented in SAR Table A2.6-1. As the loading configuration, geometry, and materials for the FO/FC-DSC and the 24PT1-DSC spacer discs are the same, the calculated frequencies and the number of fatigue cycles will be the same. Therefore, the FO/FC-DSC vibration stresses presented in SAR Section 2.6.11.3.1 (6.5 ksi maximum) are greater than the 24PT1-DSC vibration stresses.

As described in SAR Section A3.4.1.D, the thermal analysis results for the 24PT1-DSC are enveloped by the FO/FC-DSC results used in the structural thermal analysis. Therefore, the reported thermal resultant stresses for the FO/FC-DSC spacer discs are conservative when applied to the 24PT1-DSC spacer discs.

As the FO/FC-DSC vibration and thermal stress resultants are greater than the 24PT1-DSC stresses, the fatigue evaluation presented in SAR Section 2.1.3.1.B for the FO/FC-DSC envelops the 24PT1-DSC spacer disc fatigue effects.

SAR Section A2.1.2 has been revised to correct the reference to the applicable fatigue evaluation section for the FO/FC-DSC. The reference to Section 2.2.3.1.C has been revised to 2.1.2.3.2.C.

Question 2-2

Explain why fuel spacers are required in the 24PT1-DSC but are not required to resist hypothetical accident condition drop loads. Also, clarify the statements contained in the last paragraph of Section A2.7(P.A2.7-1) pertaining to fuel spacer crush length.

This information is required by staff to assess compliance with 10 CFR 71.41(a).

Response to Question 2-2

The primary function of the fuel spacers is to ensure that the active fuel region of the fuel assembly does not extend beyond the ends of the poison sheets and to ensure that the spent fuel assemblies are loaded into the cask in a manner that will maintain the package center of gravity and mass moment of inertia under normal operating conditions and hypothetical accident drop scenarios. This ensures a known location of each part of the fuel assembly in the criticality analyses. For the hypothetical accident end drop loads, the fuel is in a known configuration and the fuel spacers act as secondary impact limiting devices. For the hypothetical horizontal orientation and slapdown drop loads, the fuel spacers are primarily loaded in the perpendicular direction and no crushing will occur. Therefore, the fuel location is maintained in the as-analyzed condition for both primary and secondary slapdown, horizontal drop events.

Due to the uncertainty in determining an accurate crushed length of a fuel spacer following an end drop accident, it is conservatively assumed that the spacers are crushed to a length established by the solid volume of the fuel spacer parts. Therefore, the 14-inch long uncrushed tube, plus end plates of the spacer, is reduced to a length of 2.28 inches. This is conservative since complete crushing of the spacer eliminating all internal voids, is not credible in a cask drop accident. Therefore, using the 2.28-inch length ensures that the location of the poison plates conservatively covers the active fuel locations of the fuel assembly for the worst case top and bottom end drop orientations.

SAR Section A2.7 has been revised to specify the minimum crushed spacer height.

Question 2-3

Provide the most critical buckling load in the support rod that results from a 60g bottom end drop. Also provide the allowable buckling load for the support rods.

This information was not contained in Table A2.7-1 as stated in Section A2.7.6.4. This information is required by staff to assess compliance with 10 CFR 71.41(a).

Response to Question 2-3

The support rod assembly consists of a support rod which passes through all spacer discs and through spacer sleeves located between spacer discs.

SAR Section A2.7.6.3 has been clarified to show that the support rods carry only tensile loads and are not subjected to buckling conditions. The spacer sleeves carry all compressive loads. In the extreme case where there is zero preload, the support rods are subjected to a compressive load due to their own dead weight. This would result in a compressive stress that varies from zero at the top, to a maximum of 2.9 ksi at the base of the support rod. Any tendency to buckle will be resisted by the confinement provided by the support rod spacer sleeves. The support rod spacer sleeves carry all compressive loads and, as shown in Table A2.7-1, have a maximum interaction ratio of 0.569. During a hypothetical end drop the vertical deflections of the DSC shell, end shield plugs and cover plates are limited so that there are no additional compressive loads imposed on the

support rod assemblies.

Sections A2.6 and A2.7 of the SAR have been revised to clarify that the support rods carry tension and the support rod spacer sleeves carry all the compressive loads.

Chapter 3 Thermal

Question 3-1

Provide an analysis that accounts for the orthotropic nature of all the materials in the HEATING7.2 thermal model for the MP-187, or provide justification why it is not necessary to account for the orthotropic nature of the materials in the model.

It is not clear to the staff whether the analysis in the SAR took into account the orthotropic nature of the materials used to fabricate the cask. This information is required by staff to assess compliance with 10 CFR 71.35.

Response to Question 3-1

The thermal analysis for the 24PT1-DSC as discussed in Chapter A3 is bounded by the thermal analysis of the MP187 cask with the FO/FC-DSCs. Section 3 of the MP187 SAR provides a summary of the thermal properties of materials used in the thermal analysis of the package. This section documents the significant conservatisms incorporated in the thermal properties used for the various package materials. The package component susceptible to varying properties of material along the orthogonal axes (orthotropic material properties) is the neutron shield. The neutron shield is provided with a stainless steel outer shell, a stainless steel inner shell (cask structural shell) and a series of Aluminum/Stainless steel stiffeners between these two shells with NS-3 neutron shield material filling the remaining volume of the neutron shield. The calculation of thermal properties for the neutron shield includes the following significant conservative assumptions that preclude the need to explicitly model orthotropic material properties of the neutron shield:

1. During Normal Operating Conditions (NOC) no credit is taken for the NS-3, it is assumed to be replaced with air. This significantly increases the thermal resistance of the NS-3 shield thus increasing the MP187 and 24PT1-DSC temperatures.
2. The 24 stiffeners between the inner and outer shell of the neutron shield are assumed to be welded only to the inner shell (although the stainless steel stiffeners are welded to both shells). An air gap is assumed between the aluminum stiffeners and the outer shell. A "series and parallel conductors" equivalent thermal resistance methodology is used to determine the effective conductivity of the neutron shield annulus including NS-3 and stiffeners. The geometry assumed for calculating the effective thermal conductivity of

material in the neutron shield cavity is shown in Figure 3.2-1 of the SAR with the resulting thermal conductivities presented in SAR Table 3.2-1.

3. For the 30-minute fire transient, the increased thermal resistance of the NS-3 due to charring of the material at high temperatures is ignored and the NS-3 material properties are used for heat transfer into the package during the fire. This assumption significantly increases the heat transfer into the package due to fire during the hypothetical fire accident scenario.
4. For the POSTFIRE transient, although some NS-3 is expected to remain after the fire, the thermal properties of the NS-3 are assumed to be equal to that of air. The aluminum stiffeners are also assumed to be melted and their volume replaced with air. This again increases the temperatures of the MP187 and the 24PT1-DSC since less heat transfer out of the package is allowed.
5. The effect of the drop condition on the impact limiter foam and honeycomb materials is exaggerated by assuming that the damage is sustained by all portions of the impact limiter and not limited to one side as would be expected in a drop condition. This damage is assumed to exist on both impact limiters during the fire transient.

In addition to the above conservatisms, Section 8.1.6 of the SAR requires thermal acceptance tests to be performed prior to first use of an MP187 cask to demonstrate that the material properties assumed for the normal conditions of transport bound the actual as-built cask thermal response.

No SAR changes are required.

Question 3-2

Verify, through a sensitivity study, that the thermal models of the MP-187 presented in the SAR are not mesh dependent, (i.e., a refinement of the finite difference mesh will not greatly effect the temperatures produced by the analysis).

In finite difference and finite element analyses, where a mesh must be developed to determine temperatures, the staff requests that applicants review their models for mesh dependency. This information is required by staff to assess compliance with 10 CFR 71.35.

Response to Question 3-2

The mesh used in the finite difference analysis is considered appropriate for the system ensuring that the maximum temperatures identified are not in a region where a localized thermal gradient is missed. A review of the thermal analysis results in Appendices 3.6.2 and 3.6.3 confirms that the maximum temperatures in a region do not vary significantly from adjacent temperatures. Furthermore, the conservative neutron shield material properties discussed in the response to RAI Question 3-1 above assure that the maximum cask and 24PT1-DSC temperatures calculated are conservatively high. In addition,

Section 8.1.6 of the SAR requires thermal acceptance tests to be performed prior to first use of an MP187 cask to demonstrate that the temperatures obtained are consistent with the thermal analysis performed.

A confirmatory analysis performed by the NRC in evaluating the MP187 cask for the B&W fuel payload and reported in the MP187 Safety Evaluation Report (SER) states "...cask temperatures were in agreement with the applicant's results." The thermal analysis performed for the MP187 for the B&W fuel bounds the analysis for the Westinghouse 14 x 14 fuel. Therefore, the MP187 SER (CoC No. 9255) statement above is applicable to this payload.

No SAR changes are required.

Question 3-3

Provide a sensitivity study of the thermal model by minimizing or maximizing the voids shown in Figure 3.4-1 of the SAR and record the effect of these changes on component temperatures.

The thermal model presented in the SAR did not take into account the most limiting gaps that could be present in a manufactured MP-187. The thermal model should represent the most limiting gaps included on manufacturing tolerances. This information is needed to assure compliance with 10 CFR 71.35.

Response to Question 3-3

The following gaps are assumed to exist in the MP187 cask/24PT1-DSC model to increase fuel and component temperatures during normal conditions and during post fire conditions:

1. The 24PT1-DSC internal volume is assumed to contain air rather than helium thus reducing the thermal heat transfer out of the 24PT1-DSC and increasing 24PT1-DSC component temperatures.
2. An air gap between the poison and the guidesleeve and between the poison and the oversleeve is assumed.
3. An air gap is assumed between the oversleeves and the spacer discs.
4. An air gap is assumed between the spacer disc and the 24PT1-DSC shell ignoring any contact between the spacer discs and the shell.
5. An air gap is assumed between the lead and the inner surface of the cask structural shell.
6. An air gap is assumed between the neutron shield stiffeners and the inner surface of the neutron shield outer shell (as discussed in question 3-1).

The gaps used are based on nominal dimensions of the associated components. These gaps ignore the normal horizontal position of the cask in which gravity eliminates many

of the above gaps. Therefore, the use of the nominal gaps with a gap between each adjacent surface provides a conservative analysis by maximizing thermal resistance between the fuel and the ambient environment in all directions. The use of nominal gaps in this uniformly distributed configuration is more conservative than using the maximum gaps in the normal horizontal cask configuration.

The effect of the gaps present in the fabricated MP187 package are confirmed by the thermal performance tests required by Section 8.1.6 of the SAR. These tests are performed prior to the first use of the MP187 package and serve to demonstrate that the cask configuration assumed for the normal conditions of transport bound the actual as-built cask thermal response. The conservatisms in material properties discussed in the response to RAI Question 3-1 above in conjunction with the confirmation of the system thermal response during normal conditions ensure that the hypothetical accident condition temperatures predicted are conservative.

As discussed in the response to RAI Question 3-2 above, the NRC confirmatory analysis reported in the SER for the B&W payload for the MP187 cask applies to the Westinghouse 14 x 14 fuel payload and therefore confirms the adequacy of the assumptions used in the analysis model.

No SAR changes are required.

Note: As described above, none of the Chapter 3 RAI responses require a SAR revision. However, TN West has revised affected pages of SAR Chapters 1, 3, and 5 to limit the MOX fuel assembly maximum heat load to 0.294 kW and the maximum 24PT1-DSC heat load to 13.706 kW if the canister payload consists of at least one MOX assembly. In addition, the short term fuel cladding limit for the WE stainless steel clad (SC) fuel is revised to 806 degrees F. These changes make the SAR analysis consistent with the Advanced NUHOMS® Storage Application [Docket No. 72-1029].

Chapter 5 Shielding

Question 5-1

Provide a summary of the dose rates from the MOX fuel.

Chapter A5 states that the bounding fuel is the WE 14x14, stainless steel clad fuel with a 3.80 weight percent U-235 enrichment and 45,000 MWD/MTU burnup. The shielding evaluation needs to contain additional information about the MOX fuel and dose rates from it. This information is required by staff to assess compliance with 10 CFR 71.33 and 71.47.

Response to Question 5-1

The MOX fuel source term and associated payload limitations have been revised to reflect an increased decay time for the MOX fuel (increased from 20 years to 30 years) in order to clearly demonstrate that the MOX fuel source term is bounded by the UO_2 source term. The dose rates reported are therefore applicable to UO_2 and MOX fuel payloads.

The fuel source term has also been revised to incorporate an uncertainty of 0.04 weight percent enrichment for a WE 14x14 stainless clad fuel. This reduced minimum enrichment results in the requirement of a 39 year cooling time for this fuel.

Chapter A5 of the SAR has been revised to reflect the above changes. Table A1.2-1 of the SAR has also been revised to reflect the new minimum cooling time required for fuel assemblies with a reduced minimum enrichment.

Question 5-2

With regards to Section A5.2:

a. *Provide information about the CASK-81 energy library to demonstrate that this is an appropriate library to use for determining external dose rates for the MP-187. Also, provide the energy ranges for each energy group from CASK-81 for both gammas and neutrons.*

   According to ORNL/M-5003, "The Radioactive Materials Packaging Handbook," printed in 1988, the CASK library was developed for depleted uranium shipping casks with a water-filled cavity. At the time CASK was developed, the energy groupings were based on typical fuel spectra for that time and the data source for this library is now considered old. Additionally, CASK has been shown to underestimate the contribution to the dose from secondary gammas produced from neutron interactions. This information is required by staff to assess compliance with 10 CFR 71.33 and 71.47.

b. *Provide the input parameters (or input fill) used to determine the source term from the MOX fuel. This information is required by staff to assess compliance with 10 CFR 71.47.*

Response to Question 5-2

a. The CASK-81 library used for this application has been previously used and approved in the original MP187 cask application, CoC 71-9255 Revisions 1 through 5 for the B&W fuel payload (which consists of a canister and fuel configuration that is very similar to the WE 14 x 14 fuel configurations). This library has also been used and approved in support of TNW CoC 72-1004, including a recently reviewed amendment for the addition of the 61BT DSC payload. Site licenses for the Duke Oconee plant and the Constellation Nuclear Calvert Cliffs Power Plant also use this same library.

The CASK-81 library energy ranges for each gamma and neutron group are identified in Tables A5.4-1 and A5.4-2 of the SAR.

Data obtained from dose measurements for NUHOMS® system fuel loading activities have consistently demonstrated that the associated analyses using the CASK-81 library produce conservative results with respect to measured dose rates.

No SAR changes are required.

b. The input file used to generate the MOX fuel source term has been added, see SAR Section A5.5.2.3.

Chapter 6 Criticality

Question 6-1

Revise the criticality analysis to include any change in k_{eff} due to hypothetical accident conditions.

Section A6.4.3, which discusses the applicant's criticality analysis results, states that there was an increase in k_{eff} for other MP-187 package configurations due to guidesleeve deformation and cask layer removal, but this is ignored for the MP-187 when using the 24PT1 basket. These accident conditions are applicable to the MP-187 when using the 24PT1 basket. Also, 10 CFR 71.55 requires consideration of the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents.

Response to Question 6-1

SAR Table A6.4-1 has been revised to more clearly define the comparison of the effect of the hypothetical accident condition for the 24PT1-DSC and the FO/FC-DSC. The margin between the maximum k_{eff} and the USL is compared with the hypothetical accident condition criticality analyses performed for the more reactive B&W fuel payload. This comparison demonstrates that the margin available is significantly larger than the increase in k_{eff} due to the hypothetical accident conditions analyzed in Chapter 6.

Question 6-2

Describe in Chapter A6 how use of the fuel spacers will be confirmed. Also justify that the fuel spacers are not required to withstand hypothetical accident conditions.

The DSC inner cavity length is approximately 167 inches, the fuel assembly length is approximately 137 inches (120 inches active fuel length), and the neutron poison is 142

inches long (reference drawing NUH-05-4010). If the fuel spacers were missing or crushed and there is axial shifting of the fuel assembly, there may not be adequate neutron poison coverage. Consideration of the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents is required by 10 CFR 71.55

Response to Question 6-2

The bottom fuel spacers are installed in the 24PT1-DSC cavity during the fabrication process. The TNW approved QA program, meeting the requirements of 10 CFR 71, used by the fabricator assures their presence within the cavity. Additionally, this would be verified during receipt inspection of the 24PT1-DSC, or during the fuel loading process where the fuel handling machine will typically include the ability to monitor the vertical location of the fuel assembly. If a basket cell without a fuel spacer is encountered then the fuel would need to be lowered an additional 14 inches to release it. In this case, the assembly would be removed and a spacer installed. For the top fuel spacers, 24 fuel spacers must be loaded after the fuel and any misloading would be obvious by spacers remaining on the fuel deck. Also, a visual inspection is performed prior to installing the shield plug. Any missing spacers would show as a hole in the top surface of the cavity.

Figure 6.2-1 demonstrates the extreme locations of the active fuel after top and bottom end hypothetical drop accident events. In determining the worst case locations, the fuel spacers are assumed to be reduced to a 2.28 inch length which is equal to the volume of metal contained within the spacer. See additional discussion of the conservatisms associated with this value in the response to RAI Question 2-2. This ensures that under all postulated drop events the active fuel elements are adequately covered by the installed poison plates.

SAR Section A6.1 has been revised to discuss the effect of spacer crushing on poison plate location.

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Figure 6.2-1
Worst Case Fuel vs. Poison Plate Location during End Drop Conditions

Question 6-3

Revise the criticality analysis, including all appropriate tables, to consider uncertainty in the initial enrichment.

The initial enrichment uncertainty was not addressed. This is required for the staff to assess compliance with 10 CFR 71.55.

Response to Question 6-3

Chapter A6 has been revised to incorporate a maximum initial enrichment for the stainless steel clad fuel of 4.05%. The initial enrichment for the MOX fuel already included appropriate uncertainties in initial enrichment. The analysis of the revised maximum initial enrichment is provided in SAR Section A6.4.3.3.

Question 6-4

Describe in greater detail in Chapter A6 how the Upper Subcritical Limit (USL) was determined.

The validation does not discuss any bias and uncertainty associated with the USL determination, nor does it appear to include any uncertainty due to modeling approximations. Only biases that increase k_{eff} should be applied. This is required for the staff to assess compliance with 10 CFR 71.55.

Response to Question 6-4

SAR Section A6.5.1 has been revised to incorporate additional discussion of the method used for calculation of the Upper Subcritical Limit (USL). The methodology used is based on NUREG/CR-6361, USL method 1.

Question 6-5

Describe which factors were considered when selecting the experiments used to validate the computer code used for the criticality safety analysis.

The bias and uncertainty should be established using critical experiments that are applicable to the package design. It is not clear why certain experiments were used. This is needed to determine the applicability of the critical experiments to the application and to assess compliance with 10 CFR 71.55.

Response to Question 6-5

The benchmark analysis performed included many experiments associated with various parameters. This set of benchmarks is considered generic and is used for application of

the computer code to various system designs. Included in this set are 15 benchmarks that contain materials not present in the system analyzed for this application. The benchmarks are:

Case No.	Case Designation	Enrich. (wt %)	Pitch (cm)	H2O/Fuel Volume	Description
31	BW1810A	2.46	1.636	1.841	UO2-Gd2O rods, water holes, borated H2O
32	BW1810B	2.46	1.636	1.841	UO2-Gd2O rods, water holes, borated H2O
33	BW1810C	2.46 & 4.02	1.636	1.841 & 1.532	UO2-Gd2O rods, water holes, borated H2O
34	BW1810D	2.46 & 4.02	1.636	1.841 & 1.532	UO2-Gd2O rods, water holes, borated H2O
35	BW1810E	2.46 & 4.02	1.636	1.841 & 1.532	UO2-Gd2O rods, water holes, borated H2O
39	BW1810I	2.46	1.636	1.841	Water holes, Ag-In-Cd rods, borated H2O
44	DSN399-1	4.74	1.6	3.807	Hafnium separator plates
45	DSN399-2	4.74	1.6	3.807	Hafnium separator plates
46	DSN399-3	4.74	1.6	3.807	Hafnium separator plates
47	DSN399-4	4.74	1.6	3.807	Hafnium separator plates
166	W3269A	5.7	1.422	1.93	Ag-In-Cd rods
167	W3269B1	3.7	1.105	1.432	Ag-In-Cd rods
168	W3269B2	3.7	1.105	1.432	Ag-In-Cd rods
169	W3269B3	3.7	1.105	1.432	Ag-In-Cd rods
170	W3269C	2.72	1.524	1.494	Ag-In-Cd rods

All other benchmarks used to develop the USL contain materials found in the analyzed system and can be used to determine bias associated with the cross section library. These experiments are presented in a revised SAR Table A6.5-1. Little to no effect was found when these 15 experiments were removed from the USL.

In addition, two parameters were added to investigate the calculational bias associated with the Boral neutron absorbing material. Boron weight percent and B-10 areal density are added to the list of parameters used to determine the calculational bias in the modeling system.

SAR Table A6.5-2 has been revised to document the changes in the USL reflecting the removal of the 15 benchmark experiments noted above and adding the Boron parameters. Correlation coefficients for the 8 parameters listed in SAR Table A6.5-2 are included in the last row of SAR Table A6.5-1. The limiting USLs for all 8 parameters are presented in SAR Table A6.5-4.

Section A6.5 has been revised to reflect these changes.

Question 6-6

Discuss in Chapter A6 the failed fuel can screen mesh size and either justify that uneven draining is not credible or provide an analysis considering uneven flooding. Also, revise Note 31 on drawing NUH-05-4010, to require a minimum mesh size.

Uneven draining may be possible for screens with a mesh size of 350 or less because the water surface tension may be capable of supporting water. Uneven draining in the canister may be more reactive that a fully flooded cask. This is required for the staff to assess compliance with 10 CFR 71.55.

Response to Question 6-6

The screen mesh size is specified in NUH-05-4010, sheet 1, Bill of Material, Item 43, SAR Section A1.3.2. A discussion of the screen mesh size specified in the SAR 24PT1-DSC drawing has been added to SAR Section A6.3.2. This screen mesh size is the same as that used and approved for the MP187 cask with the FO/FC-DSC payload.

Question 6-7

Justify that the postulated failure scenarios for damaged fuel is bounding.

The criticality safety analysis in Section A6.3 of the SAR does not provide justification that the scenarios considered for failed fuel bound other possible scenarios such as missing fuel pins. This is required for the staff to assess compliance with 10 CFR 71.55.

Response to Question 6-7

In addition to the criticality analysis model description provided in SAR Section A6.3, a discussion of the criticality calculation scenarios analyzed is provided in SAR Section A6.4.2 for the damaged fuel configuration.

Additional discussion of the damaged fuel analysis has been provided in SAR Section A6.4.2 to further demonstrate that the postulated failure scenarios for damaged fuel are bounding.

ATTACHMENT C

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

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 *December 13, 2000, as supplemented.*

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, *four* 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.

- *24PT1 Dry Shielded Canister (24PT1-DSC)*

  The 24PT1-DSC has a cavity length of approximately 167 inches with a solid carbon steel shield plug at each end. The 24PT1-DSC is designed to contain up to 24 Westinghouse (WE) 14x14 PWR spent fuel assemblies, including control components. Fuel assemblies may be damaged or intact as described in 5.b(2)(a). 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 failed 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 1
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, 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 *a Failed Fuel Can inside* the FF-DSC *or 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 WE 14x14 UO2 stainless steel clad (SC) or zircaloy clad mixed oxide (MOX) 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. Control Components integral to WE 14x14 assemblies in a 24PT1-DSC include Rod Cluster Control Assemblies, Thimble Plug Assemblies and Neutron Source Assemblies.

(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 per Table 2.

(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 per Table 2.

(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 WE 14x14 SC fuel assemblies with the balance intact WE 14x14 SC or MOX fuel assemblies. No more than one damaged WE 14x14 MOX fuel assembly can be stored per 24PT1-DSC with the balance intact WE 14x14 SC 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 failed 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 / Max Heat Load Per Cask / Max Assembly Heat Load (Incl. Control Components[2])*
WE 14x14 Stainless Steel Clad (SC)	*4.05 ^{235}U*	*3.76 ^{235}U*	*45,000*	*39[1] years/14 kW/ 0.583 kW*
(May include Integral Fuel Burnable Absorber, boron coated fuel pellets)		*3.36 ^{235}U*	*40,000*	
		3.12 ^{235}U	*35,000*	
WE 14x14 MOX	*0.71 ^{235}U 2.84 fissile Pu (64 rods) 3.10 fissile Pu (92 rods) 3.31 fissile Pu (24 rods)*	*2.78 fissile Pu (64 rods) 3.05 fissile Pu (92 rods) 3.25 fissile Pu (24 rods)*	*25,000*	*30 years/13.706 kW/ 0.294 kW*

Notes:

[1] Criteria for a shortened cooling time are provided in SAR Section A5.2.3.

[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 *has no Type I or II placement restrictions. The 24PT1-DSC requires restrictions on the location of damaged fuel assemblies per Section 5.b.(2).*

7. Fuel assemblies with missing fuel rods shall not be shipped *in a FO, FC or a FF-DSC* 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.

REFERENCES

Transnuclear West Inc., consolidated Safety Analysis Report for the NUHOMS®-MP187 Multi-Purpose Cask, dated *December 13, 2000, as supplemented January 30 and August 24, 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 D

INSTRUCTIONS FOR UPDATE OF MP187 SAR FROM REVISION 11 TO REVISION 12

NON-PROPRIETARY
FOR INFORMATION ONLY

These instructions are to be used for incorporating changes to the MP187 SAR, Revision 11 to create the Revision 12 SAR, Proprietary Version.

MP187 SAR – Volume 1:

- Cover Sheet — Replace Revision 11 page with Revision 12 page
- Pages ii and iii — Replace Revision 11 pages with Revision 12 pages
- Page xli — Replace Revision 11 page with Revision 12 page

MP187 SAR – Volume 4:

- See List of Effective Pages for pages to be replaced or added in Appendix A.
- Replace drawing NUH-05-4010, Rev. 0 in Section A1.3.2 with NUH-05-4010, Rev. 1

These instructions are to be used for incorporating changes to the Non-Proprietary Version of MP187 SAR, Revision 11, to create the Revision 12 SAR, Non-Proprietary Version.

MP187 SAR – Volume 1:

- Cover Sheet — Replace Revision 11 page with Revision 12 page
- Pages ii and iii — Replace Revision 11 pages with Revision 12 page
- Page xxxvi — Replace Revision 11 page with Revision 12 page

MP187 SAR – Volume 4:

- See List of Effective Pages for pages to be replaced or added in Appendix A.
- Replace drawing NUH-05-4010NP, Rev. 0 in Section A1.3.2 with NUH-05-4010NP, Rev. 1

ATTACHMENT F

CHANGED MP187 SAFETY ANALYSIS REPORT PAGES,

DOCUMENT NUH-05-151, REVISION 12

[NON-PROPRIETARY]

(3 copies)

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
Rev. 12	*August 2001*	*Response to NRC RAIs for 24PT1-DSC*

LIST OF NON-PROPRIETARY EFFECTIVE PAGES

Page	Revision
i-iii	*12*
iv-viii	10
ix	11
x-xxvi	10
xxv	11
xxvi-xxxiv	10
xxxv-xxxvi	11
xli	*12*
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	
(All pages of Appendix A are Revision 11 except for the following pages:)	
Ai – Avi	*12*
A1.2-2 – A1.2-3	*12*
A1.2-5	*12*
A1.3-2	*12*
A2.1-3	*12*
A2.6-2	*12*
A2.6-5 - A2.6-6	*12*
A2.6-9 - A2.6-12	*12*
A2.6-14	*12*
A2.6-16 - A2.6-17	*12*
A2.7-1 - A2.7-4	*12*
A2.7-6	*12*
A3.4-1 – A3.4-2	*12*
A5-1	*12*
A5.2-1 - A5.2-3	*12*
A5.2-6	*12*
A5.2-8 – A5.2-12	*12*
A5.2-19 – A5.2-21	*12*
A6.1-2 - A6.1-2a	*12*
A6.1-4 - A6.1-5	*12*
A6.1-6a	*12*
A6.2-1 - A6.2-2	*12*
A6.3-4	*12*
A6.3-8	*12*
A6.4-5 - A6.4-6a	*12*
A6.4-16a - A6.4-16d	*12*
A6.5-1 - A6.5-9	*12*
A8-1	*12*

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. *Revised Appendix A to address NRC RAI for 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.

NON-PROPRIETARY 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 Fabrication Stress Analysis A2.6-2

A2.6.11 24PT1-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

A2.8 Special Form A2.8-1

A2.9 Fuel Rods A2.9-1

A2.10 Appendix A2.10-i

NON-PROPRIETARY TABLE OF CONTENTS

Continued

Page

A2.10.1 References A2.10-i
A2.10.2 NUHOMS®-MP187 Cask Analyses A2.10-i
A2.10.3 24PT1-DSC Basket Analyses A2.10-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
A5.2.1 Gamma Source A5.2-3
A5.2.2 Neutron Source A5.2-19
A5.2.3 Spent Fuel Payloads A5.2-21
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-i
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

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

NON-PROPRIETARY TABLE OF CONTENTS

Continued

Page

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....A6.5-3
A6.6 Appendix....A6.6-1
A6.6.1 References....A6.6-1
A6.6.2 KENO Input Files....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

NON-PROPRIETARY 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	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	Support Rod Assembly Spacer Sleeve Stresses - 60g Bottom End Drop	A2.7-4
Table A3.6-1	Component Weight, Material, and Volume for a Single Fuel Assembly	A3.6-4
Table A3.6-3	Fuel Rod Data	A3.6-5
Table A3.6-5	Fission Gas Inventory per Assembly for WE 14x14 Fuel	A3.6-6
Table A3.6-7	Control Component Data	A3.6-7
Table A3.6-9	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	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	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 (γ/s/cm^3)	A5.2-18
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

NON-PROPRIETARY LIST OF TABLES

Continued

		Page
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	A6.4-8
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 STAINLESS STEEL CLAD FUEL	A6.4-13
Table A6.4-5	Bounding SC Fuel Criticality Analysis Analyzed for 4.05 weight % ^{235}U Damaged Fuel Assemblies with External Moderator Density Varying for Most Reactive Rod Pitch Case	A6.4-16a
Table A6.4-6	Clad OD Sensitivity Evaluation	A6.4-16b
Table A6.4-7	Clad Thickness Sensitivity Evaluation	A6.4-16c
Table A6.4-8	Pellet Diameter Sensitivity Evaluation	A6.4-16d
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	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	A6.5-9

NON-PROPRIETARY 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	2.7-5
Figure A5.3-1	NUHOMS®-MP187 Normal Operations DORT-PC 24PT1-DSC Top End Model	AZ5.3-3
Figure A5.3-2	NUHOMS®-MP187 Normal Operations DORT-PC 24PT1-DSC Bottom End Model	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 A6.1-1	Worst Case Fuel vs. Poison Plate Location during End Drop Conditions	A6.1-6a
Figure A6.3-1	KENO Model and DSC Basket	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	A6.4-17
Figure A6.4-2	Double-Ended Rod Shear Example	A6.4-18

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 *stainless clad* damaged fuel

assemblies *or one zircaloy clad MOX assembly* 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.

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)	4.05 ^{235}U	3. 76 ^{235}U	45,000	39[1] years/14 kW/ 0.583 kW
		3. 36 ^{235}U	40,000	
		3.12 ^{235}U	35,000	
WE 14x14 MOX	0.71 ^{235}U • *2.84 Fissile Pu (64 rods)* • *3.10 Fissile Pu (92 rods)* • *3.31 Fissile Pu (24 rods)*	• 2.78 *Fissile* Pu (64 rods) • 3.05 *Fissile* Pu (92 rods) • 3.25 *Fissile* Pu (24 rods)	25,000	30 years/*13.706* kW/ 0.*294* kW

Notes:

[1] The 39 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 39 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.

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-05-4010NP NUHOMS® 24PT1-DSC Main Assembly, Rev. *1*

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

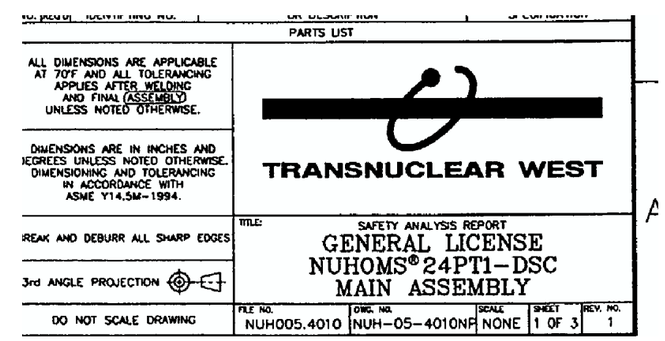


FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

ALL DIMENSIONS ARE APPLICABLE AT 70°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL ASSEMBLY 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-05-4010NF / SCALE NONE / SHEET 2 OF 3 / REV. NO. 1

7

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

ALL DIMENSIONS ARE APPLICABLE AT 70°F AND ALL TOLERANCING APPLIES AFTER WELDING AND FINAL (ASSEMBLY) 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-05-4010NF / SCALE NONE / SHEET 3 OF 3 / REV. NO. 1

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.*1*.2.3.*2*.C are applicable to the 24PT1-DSC since the FO/FC-DSC loads *exceed* those for the 24PT1-DSC, and the same geometry *and materials are* used for all three DSC spacer disc designs.

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 Fabrication 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.

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 rod *assemblies*, 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 Rod *Assemblies*

The configuration of the support rod assembl*ies* 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. *The OD of the spacer sleeve has not been changed which results in a larger cross-sectional area for the spacer sleeve.*

- *The applied support rod* preload *has* been decreased. Analyses contained in this chapter conservatively assume 65 kips maximum *support rod* preload and zero *support rod* preload to envelope *the* actual conditions. The *specified preload is* 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 *support rod* spacer sleeves which carry all *of the compressive* loads from the spacer discs *to the ends of the* 24PT1-*DSC cavity.*

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

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Figure A2.6-2

24PT1-DSC Spacer Disc Detail

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 Assembly Thermal Expansion

As described in 2.6.11.2.3, clearance between the support rod *assemblies* and the 24PT1-DSC cavity is large such that thermal growth of the rods *and support rod spacer sleeves* will not be restrained by the 24PT1-DSC cavity. Therefore, no additional evaluation is required.

A2.6.11.2.4 Support Rod Assembly 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 *support* rod and compression in the *support rod spacer* sleeves.
- As temperature increases, the elongation applied by the initial preload remains constant. However the axial stiffness of the *support* rod assembly decreases (since E decreases with temperature). Thus, the loads (stresses) in the *support* 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 *Assembly* 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 *prestress* value of 65 kips, Level A allowable stresses at 600°F, and reduced section properties at the threaded section. As shown below, the *support rod* bending and tensile stresses *from Table A2.6-3* 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):

Support Rod Interaction Ratio : Vibration

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

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

$$= 0.92$$

For the support rod spacer sleeves the axial prestress compressive loading is 11.5 ksi (Table A2.6-3) with no significant bending stresses and so the interaction ratio is less than 0.22.

A2.6.11.4.2 Support Rod *Assembly* 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 *assembly* 1-foot drop analysis. *The support rod spacer sleeves carry only compressive loading from the initial prestress.*

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 *support rod* 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):

Support 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.

A2.6.13.2.1 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 *Assembly* 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 *Assembly* 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 rod *assemblies*, guidesleeves and Failed Fuel Cans to those used in the FO/FC DSCs.

The changes in the 24PT1-DSC support rod *assemblies* 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 *hypothetical accident* 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 *of 2.28 inches which is* equal to the volume of material present in the fuel spacer.

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 *Assembly* Stress Analyses

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 *support rod* as a beam *loaded by the spacer sleeves* and using hand calculations. The critical location is the cantilever section at the top spacer disc. *The support rod s*tresses are calculated based on *the* enveloping preload, *maximum 65 kip and minimum 0 kip,* 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 *Support Rod Assembly* Evaluation for Axial Loads

Analysis *of the support rod assemblies* 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 *assemblies*. The loads imposed on the model are inertial acceleration, *axial support rod* loads and bending moments from *the* spacer discs. *The spacer discs* are conservatively assumed to be "fixed" at the support rod locations to *maximize* the support rod *assembly* stresses. Hand calculations are used to calculate the stresses and to demonstrate the acceptability of the *support rod assemblies* 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 *support rod spacer sleeve compressive* 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 *compressive* stresses *occur* 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

Support Rod Assembly 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	Elements		Maximum	Maximum	Sleeve Stresses		Equation 20				Equation 21			Equation
					Bending	Axial,	Term 1	Term 2		Result	Term 1	Term 2	Result	22
Number	1	2	Moment	Axial Load	f_b	f_a	f_a/F_a	F'_e	$C_m f_b/xF_b$		$f_a/x(.60S_y)$	f_b/F_b		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.

A2.7.6.4 Support Rod *Assembly* Stability Analysis

Stability of the support rod *spacer sleeves* is demonstrated by the results described in Table A2.7-1. No additional evaluation *of the support rods is required as the applied loads applied to the rods are tensile. For the extreme case where there is zero prestress the support rods would be loaded by their self weight with a maximum compressive stress of 2.9 ksi developed at the bottom (drop) end of the rod. In this case the support rods are confined within the support rod spacer sleeves and lateral deflection is not possible.*

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.

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 *maximum 24PT1-DSC* 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 *806°F* as discussed in [A3.1], Section 3.5.1. 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.

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 evaluated are specified below:

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)	4.05 ^{235}U	3.76 ^{235}U	45,000	39 years/14 kW/ 0.583 kW
		3.36 ^{235}U	40,000	
		3.12 ^{235}U	35,000	
WE 14x14 MOX	0.71 ^{235}U • *2.84 Fissile Pu (64 rods)* • *3.10 Fissile Pu (92 rods)* • *3.31 Fissile Pu (24 rods)*	• 2.78 *Fissile* Pu (64 rods) • 3.05 *Fissile* Pu (92 rods) • 3.25 *Fissile* Pu (24 rods)	25,000	30 years/*13.706* kW/ *0.294* kW

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

The analysis in this chapter compares the source term for *39* 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.*76* weight % U-235, with 45,000 MWd/MTU burnup *with 39 years cooling. A more conservative source term based on* SC, 38 year cooled fuel *with a U-235 enrichment of 3.8 weight %,* is used as the design basis source term. *The dose rate calculated based on this bounding source term will be applicable to UO_2 and MOX fuel assemblies.*

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 39 years for WE 14x14 SC fuel and 25,000 MWd/MTU and minimum cooling time of 30 years for WE 14x14 MOX fuel. 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 39 years for SC fuel assemblies and greater than or

equal to 30 years for MOX fuel assemblies with burnup limited to 25,000 MWd/MTU. The choice of initial enrichment for each case has a significant effect on the neutron source *as* lower enrichments produce greater sources. The minimum enrichments *are* 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, 3.12 U-235 minimum enrichment (wt. %), *39* year cooled
2. SC fuel – 40 GWd/MTU, 3. *36* U-235 minimum enrichment (wt. %), *39* year cooled
3. SC fuel – 45 GWd/MTU, 3. *76* U-235 minimum enrichment (wt. %), *39* year cooled
4. SC fuel – 45 GWd/MTU, 3.8 U-235 minimum enrichment (wt. %), 38 year cooled *(used to develop the enveloping design basis source term)*
5. MOX fuel – 25 GWd/MTU, with minimum enrichments as specified in Chapter A1, 30 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

Fuel assembly end fitting source terms for 10 year cooled, 3.8 weight % U-235 with 45 GWd/MTU burnup are conservatively used in conjunction with the above cases. 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)	*Minimum* Initial Enrichment (wt. % ^{235}U)	Required Cooling Time (years)
I	SC fuel	35,000	3.12	39
II	SC fuel	40,000	3.36	39
III	SC fuel	45,000	3.76	39
IV	SC fuel	45,000	3.8	38
V	MOX fuel	25,000	.71 for U-235, *Fissile* Pu initial enrichment: 2.78% - 64 rods 3.05% - 92 rods 3.25% - 24 rods	30
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

**The use of nominal enrichment for these runs versus the minimum enrichment specified for the fuel assemblies has a negligible effet on the associated NFAH gamma doses.*

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

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 *39* years (SC) or *30* years (MOX). Analyses were also performed for various enrichments/burnups 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.

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 identifying the bounding SC fuel assembly in-core source term by comparing the 35 GWd/MTU, 40 GWd/MTU and 45 GWd/MTU; *39* year cooled fuel assembly source terms (See Table A5.2-5). *A bounding source term based on* 45 GWd/MTU, *3.8 % enrichment and 38 year cooling is conservatively used as* the bounding in-core source term. This source term was decayed for 38 years and compared to the MOX source term. 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 *30* 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 $3.4\mathit{44}\times10^{16}$ photons per second, based on the bounding source term shown in Table A5.2-5.

Table A5.2-5

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

Cask 81 Energy Group	*14x14 MOX 25 GWd/MTU 30 Years cooling*	14x14 SC 35 GWd/MTU, *3.12 % enrichment, 39 Years cooling*	14x14 SC 40 GWd/MTU *3.36% enrichment 39 Years cooling*	14x14 SC 45 GWd/MTU *3.76% enrichment 39 Years cooling*	14x14 SC 45 GWd/MTU 38 Years *cooling* *Bounding Source Term Used in Dose Analyses*
23	*1.910E+04*	*2.857E+04*	*4.293E+04*	*5.654E+04*	5.759E+04
24	*9.007E+04*	*1.346E+05*	*2.023E+05*	*2.664E+05*	2.713E+05
25	*4.600E+05*	*6.866E+05*	*1.032E+06*	*1.359E+06*	1.384E+06
26	*1.148E+06*	*1.712E+06*	*2.572E+06*	*3.386E+06*	3.449E+06
27	*3.410E+06*	*5.076E+06*	*7.624E+06*	*1.004E+07*	1.022E+07
28	*2.247E+07*	*1.080E+08*	*1.459E+08*	*1.897E+08*	1.917E+08
29	*6.496E+08*	*1.383E+09*	*1.567E+09*	*1.774E+09*	1.827E+09
30	*1.263E+10*	*2.661E+10*	*3.013E+10*	*3.410E+10*	3.509E+10
31	*7.657E+11*	*1.252E+12*	*1.410E+12*	*1.545E+12*	1.724E+12
32	*5.600E+12*	*6.742E+12*	*7.807E+12*	*8.798E+12*	9.691E+12
33	*2.537E+12*	*2.726E+12*	*3.219E+12*	*3.737E+12*	3.955E+12
34	*3.979E+14*	*5.075E+14*	*5.801E+14*	*6.529E+14*	6.684E+14
35	*3.875E+12*	*7.333E+12*	*8.334E+12*	*9.454E+12*	9.760E+12
36	*5.006E+12*	*1.044E+13*	*1.182E+13*	*1.337E+13*	1.375E+13
37	*7.690E+12*	*1.513E+13*	*1.719E+13*	*1.947E+13*	2.004E+13
38	*2.565E+13*	*4.843E+13*	*5.503E+13*	*6.236E+13*	6.424E+13
39	*7.514E+13*	*8.932E+13*	*1.007E+14*	*1.128E+14*	1.151E+14
40	*2.306E+14*	*3.810E+14*	*4.330E+14*	*4.893E+14*	5.023E+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[1]	40 GWd[1]	45 GWd[1]	MOX[2]
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

1 *Conservatively based on 10 year cooled fuel at the applicable nominal enrichment.*
2 *Conservatively based on 20 year cooled fuel at minimum enrichment.*

Table A5.2-5

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

(Continued)

Group	35 GWd[1]	40 GWd[1]	45 GWd[1]	MOX[2]
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

1 *Conservatively based on 10 year cooled fuel at the applicable nominal enrichment.*
2 *Conservatively based on 20 year cooled fuel at minimum enrichment.*

Table A5.2-5

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

(Concluded)

Group	35 GWd[1]	40 GWd[1]	45 GWd[1]	MOX[2]
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

1 Conservatively based on 10 year cooled fuel at the applicable nominal enrichment.
2 Conservatively based on 20 year cooled fuel at minimum enrichment.

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,

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 *as shown in* Table A5.2-9 *below.* This fuel also bounds the MOX 30 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 MOX 25 GWd/MTU, 30 Years*	14x14 SC 35 GWd/MTU *3.12 weight % enrichment, 39 Years cooling*	14x14 SC 40 GWd/MTU *3.36 weight % enrichment, 39 Years cooling*	14x14 SC 45 GWd/MTU *3.76 weight % enrichment, 39 Years cooling*	14x14 SC 45 GWd/MTU *3.8 weight % enrichment, 38 Years cooling*
1	2.018E-04	7.299E+03	1.040E+04	1.549E+04	2.032E+04	2.068E+04
2	1.146E-03	4.145E+04	5.908E+04	8.797E+04	1.154E+05	1.175E+05
3	4.471E-03	1.617E+05	2.305E+05	3.432E+05	4.502E+05	4.583E+05
4	1.768E-02	6.395E+05	9.114E+05	1.357E+06	1.780E+06	1.812E+06
5	4.167E-02	1.507E+06	2.148E+06	3.199E+06	4.196E+06	4.271E+06
6	5.641E-02	2.040E+06	2.908E+06	4.330E+06	5.680E+06	5.782E+06
7	1.197E-01	4.330E+06	6.171E+06	9.188E+06	1.205E+07	1.227E+07
8	9.616E-02	3.478E+06	4.957E+06	7.381E+06	9.683E+06	9.856E+06
9	2.256E-02	8.160E+05	1.163E+06	1.732E+06	2.272E+06	2.312E+06
10	1.227E-01	4.438E+06	6.325E+06	9.418E+06	1.236E+07	1.258E+07
11	2.110E-01	7.632E+06	1.088E+07	1.620E+07	2.125E+07	2.163E+07
12	1.794E-01	6.489E+06	9.248E+06	1.377E+07	1.807E+07	1.839E+07
13	1.138E-01	4.116E+06	5.866E+06	8.735E+06	1.146E+07	1.166E+07
14	1.301E-02	4.706E+05	6.707E+05	9.986E+05	1.310E+06	1.334E+06
15	6.555E-05	2.371E+03	3.379E+03	5.032E+03	6.601E+03	6.719E+03
16	4.765E-06	1.724E+02	2.456E+02	3.658E+02	4.798E+02	4.884E+02
17	3.134E-07	1.134E+01	1.616E+01	2.406E+01	3.156E+01	3.212E+01
18	4.527E-08	1.637E+00	2.334E+00	3.475E+00	4.559E+00	4.640E+00
19	9.759E-09	3.530E-01	5.031E-01	7.491E-01	9.827E-01	1.000E+00
20	1.521E-09	5.501E-02	7.841E-02	1.168E-01	1.532E-01	1.559E-01
21	3.353E-10	1.213E-02	1.728E-02	2.574E-02	3.376E-02	3.437E-02
22	9.683E-11	3.502E-03	4.992E-03	7.433E-03	9.751E-03	9.925E-03
Totals	**1.000E+00**	***3.617E+07***	***5.155E+07***	***7.676E+07***	***1.007E+08***	**1.025E+08**

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

A5.2.3 *Spent Fuel Payloads*

Table A1.2-1 specifies a minimum cooling time of 39 years for WE 14x14 stainless steel clad fuel and 30 years for WE 14x14 MOX fuel. Fuel with shorter cooling times may be transported if a dose rate analysis is performed under a QA program approved by the NRC for 10CFR71 or 10CFR50, using the methodology described in Sections A5.2.1, A5.2.2, A5.3 *and* A5.4 *for fuel meeting all requirements of Table A1.2-1, except for fuel cooling time.*

The analysis models and computer codes must be the same as those specified in the SAR sections identified above.

Fuel loading using a zoning configuration to isolate the "hot" fuel assemblies in the center of the 24PT1-DSC, similar to that described in Section 5.2 of this SAR, may also be used. The methodology and analysis models, including the same computer codes, specified in Section 5.2 must be used.

The results of the above dose rate analyses for fuel with shorter cooling times must be bounded by the analysis results reported in Table A5.4-3 *and* Table A5.4-4 *in order to qualify such a payload for transportation in the MP187 cask.*

Another alternative for qualification of a payload with cooling times less than the cooling times specified in Table A1.2-1 is by direct measurement of the packaging dose rates corresponding to the Normal Conditions of Transport shown in Table A5.4-3. *Measured data are to be increased by 25% to account for potential measurement inaccuracy. An evaluation of the sensitivity and accuracy of the instrumentation used must demonstrate that the measurements, when multiplied by a factor of 1.25, will envelope instrument sensitivity and inaccuracy. The correction factor used for the measured data may not be less than 1.25 regardless of instrument accuracy. The normal condition analysis is the most limiting dose case, with the least margin between the analysis and the regulatory limit. The measured dose rates multiplied by 1.25 must therefore be less than the normal dose rates presented in Table A5.1-1. Verification based on this case will ensure that accident dose rate limits, with conservative assumptions with respect to the loss of the neutron shield, are met.*

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 location ensures that the active fuel is covered under all conditions including a hypothetical accident in which the fuel spacers are assumed to crush completely (the 14 inch spacer crusher to a minimum height of 2.28"). Figure A6.1-1 summarizes the fuel, poison and spacer locations during a hypothetical top and bottom end drop.

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 ($Al_2O_3 \cdot 3H_2O$) called bayerite which protects the surface from further attack.

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

14x14 MOX fuel assembly may be stored in a *24PT1*-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, 0.93*92* 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 *24PT1*-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 *20 intact SC plus* 4 damaged *MOX* fuel assemblies, while the C of C limits the payload to no more than 1 damaged MOX fuel assembly per *24PT1*-DSC), including all biases and uncertainties applicable to the calculational methodology and the design. The associated Upper Subcritical Limit *is calculated* as 0.*9401* (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: *SC-all assemblies damaged, MOX-4 assemblies damaged*); ($k_{eff} + 2\sigma$)	yes	0.9392 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].

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

Figure A 6.1-1

Worst Case Fuel vs. Poison Plate Location during End Drop Conditions

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 4.05 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: ≤ 4.05 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
Maximum Initial Enrichment, wt. %	^{235}U: 4.05	^{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

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.

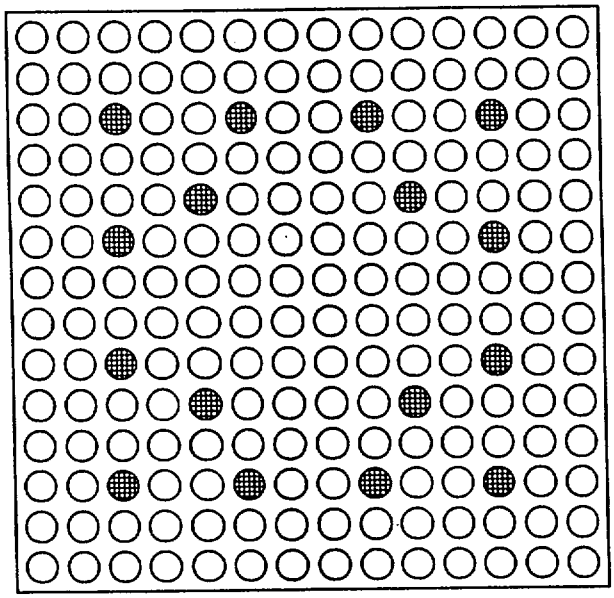
Figure A6.3-3 shows the structure of UNITs 33 and 34: the cask slices. Note that the difference between the two UNITs is that UNIT 33 is a spacer disc (steel surrounding fuel assemblies) and UNIT 34 has steel support rods only (water surrounding fuel assemblies). Also, for the HAC cases, there is no 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.

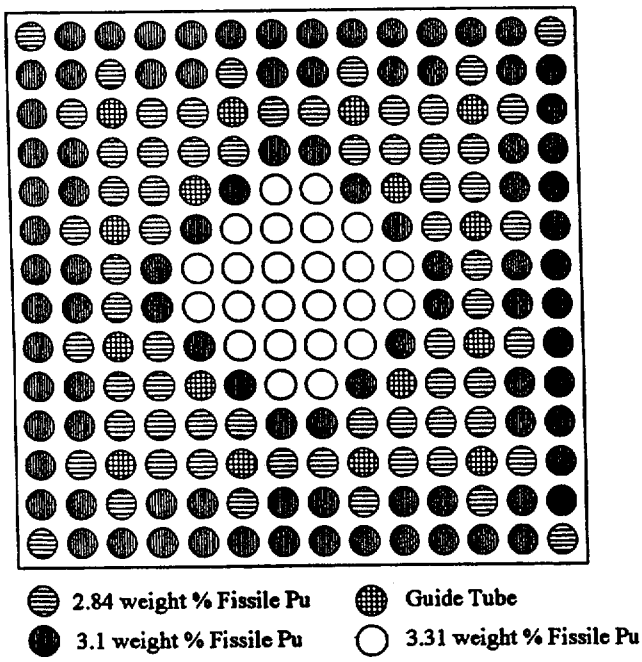
Figure A6.3-4 shows more closely the way in which UNITs 1-8 are constructed. Each HOLE is identified by UNIT number and its own particular coordinate origin.

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.

The analyses performed are based on a completely flooded 24PT1-DSC. Slots are provided at the bottom of the guidesleeves and openings are provided at the bottom and sides (near the bottom) of the failed fuel cans to ensure uniform draining and filling of all areas of the 24PT1-DSC. The failed fuel can openings are screened to contain potentially loose pellets or debris. The 6x6 mesh size and 0.047 in. wire diameter used for these screens will not obstruct uniform draining and filling.



WE 14x14 SC Fuel Assembly Configuration



WE 14x14 MOX Fuel Assembly Configuration

Figure A6.3-5

KENO Model of Design Basis Fuel Assemblies

Table A6.4-1

USL vs. Calculated k_{eff}

Fuel Configuration Analyzed (Associated Upper *Subcritical* Limit, USL)	*Intact Fuel, Normal Conditions of Operation, Limiting* k_{eff} + 2σ (Margin to USL)	*Δk_{eff} for Guidesleeve Displacement Based on B&W 15x15 Intact Fuel Analysis*	*Δk_{eff} for Cask Layer Removal Based on B&W 15x15 Intact Fuel Analysis*
FO/FC with B&W 15x15 fuel (w/ USL = 0.95)	0.9374 (margin to USL = 0. 0126)	*.00705*	*.00167*
24PT1-DSC with WE 14x14 SC fuel (w/ USL = 0.9*401*)	0.8666 (margin to USL = 0.073*5*)		
24PT1-DSC with WE 14x14 MOX fuel (w/USL = 0.9*401*)	0.9111 (margin to USL = 0.02*90*)		

The guidesleeve deformation parametric study in Chapter 6 demonstrates that a small increase in keff 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 keff 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.

Table A6.4-1 above summarizes the margin to the USL for the B&W fuel normal conditions analysis, from Chapter 6, as well as the margin to the USL for the WE 14x14 SC and MOX fuel analyses. The Hypothetical Accident Conditions Δkeff for the B&W fuel is also presented in this table. The available margin for the WE 14x14 fuel is significantly higher than the Δkeff for conservative analyses with respect to damaged fuel configurations allowed by the C of C (analyses are for 4 MOX damaged fuel assemblies with the balance intact SC fuel assemblies, while the C of C limits the number of MOX damaged fuel assemblies to 1, and for 24 damaged SC fuel assemblies while the C of C limits the number of damaged SC fuel assemblies to 4).

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 *per 24PT1-DSC*) with 0.80 g/cc moderator interspersed between an infinite array of packages, is 0.9392 ($k_{eff} + 2\sigma$). The highest calculated k_{eff} for the WE 14x14 MOX damaged fuel *(analyzed for 20 intact SC plus 4 damaged MOX fuel assemblies, with a C of C limit of 1 damaged MOX fuel assembly per 24PT1-DSC)* is for 0.6000 g/cc moderator interspersed between an infinite array of packages. The reactivity is 0.8921 ($k_{eff} + 2\sigma$).

A6.4.3.3 *Evaluation of Effect of Uncertainty in Maximum Initial Enrichment*

The maximum initial enrichment used in the criticality analyses for UO_2 fuel (4.0%) does not include uncertainties (manufacturing tolerance) in maximum initial enrichment. However, the

maximum initial enrichment for MOX fuel used in the criticality analysis does account for potential uncertainty in Pu maximum initial enrichment.

To address the effect of potential $U0_2$ fuel assembly initial enrichment uncertainty, an evaluation of the effect of an increased initial enrichment from 4.0 weight % to 4.05 weight %, has been performed. The results of this analysis are presented in Table A6.4-5. These results when compared to the equivalent case analyzed for 4.0 weight %, Table A6.4-4, show that the increase in enrichment from 4.0 weight % to 4.05 weight % results in an increase in k_{eff} (incl. 2σ) from 0.9368 to 0.9392. The increased k_{eff} remains less than the USL of 0.9401. Based on these results, storage of fuel of up to 4.05 weight % enrichment is acceptable.

A6.4.3.4 *Effect of Fuel Parameter Tolerances on Reactivity*

To evaluate the effect of fuel parameter tolerances on reactivity, the fuel parameters used in the criticality analyses have been reviewed to identify sensitivity studies needed. A review of fuel parameters identified in Tables A6.2-1 and A6.2-2 indicates that all parameters listed with the exception of pellet diameter, clad outer diameter (OD) and clad thickness are enveloped by the criticality analyses performed. To evaluate the effect of tolerances in clad OD, clad thickness and fuel pellet diameter on reactivity, sensitivity analyses are performed to evaluate system reactivity as a function of these parameters. These analyses are performed for the SC fuel assemblies since the damaged SC assembly analysis is the bounding analysis. In addition, changes in the configuration of the stainless steel cladding have a more significant effect on reactivity than the changes in zirconium cladding, which is used in the MOX fuel. The CSAS25 models used to perform this sensitivity analysis are based on the model used to calculate k_{eff} for the fuel assemblies centered in the guidesleeves, the results of which are reported in Table A6.4-2. This model is revised to include 4.05 wt. % enriched fuel and the revised clad and pellet dimensions. The results of the evaluation are presented in Tables A6.4-6, A6.4-7 and A6.4-8. The results demonstrate that the calculated changes in reactivity between the various cladding and pellet dimensions are not significant.

Table A6.4-5

Bounding SC Fuel Criticality Analysis Analyzed for 4.05 weight % ^{235}U

Damaged Fuel Assemblies with External Moderator Density Varying for Most Reactive Rod Pitch Case

K_{eff}	+/- 1σ	K_{eff} +2σ	External Moderator (H_2O) Density, g/cc
0.9348	0.0013	0.9374	0.0001
0.9364	0.0012	0.9389	0.05
0.9341	0.0012	0.9365	0.10
0.9365	0.0012	0.9389	0.20
0.9348	0.0013	0.9374	0.30
0.9364	0.0013	0.9390	0.40
0.9338	0.0011	0.9360	0.50
0.9362	0.0012	0.9386	0.60
0.9368	0.0012	0.9392	0.70
0.9365	0.0012	0.9389	0.80
0.9345	0.0012	0.9369	0.90
0.9354	0.0011	0.9376	1.00

Table A6.4-6

Clad OD Sensitivity Evaluation

Fuel Clad OD: 4.05 wt.% U-235 Fuel Centered In Guide Tube			
K_{eff}	+/- 1σ	K_{eff} +2σ	Clad OD (inches)[1]
0.8645	0.0011	0.8667	0.415
0.8653	0.0013	0.8679	0.418
0.8640	0.0013	0.8666	0.420
0.8625	0.0013	0.8651	0.422
0.8646	0.0012	0.8670	0.424
0.8642	0.0012	0.8666	0.426
0.8631	0.0012	0.8655	0.429

1. *Min. clad thickness is used for this evaluation.*

Table A6.4-7

Clad Thickness Sensitivity Evaluation

K_{eff}	+/- 1σ	K_{eff} +2σ	Clad OD (inches)	Clad ID (inches)
0.8631	0.0012	0.8654	0.429	.3976
0.8599	0.0012	0.8623	0.429	.3970
0.8591	0.0013	0.8617	0.429	.3964
0.8588	0.0011	0.8610	0.429	.3960
0.8575	0.0013	0.8602	0.429	.3954
0.8563	0.0011	0.8584	0.429	.3948
0.8559	0.0012	0.8583	0.429	.3944

Table A6.4-8

Pellet Diameter Sensitivity Evaluation

K_{eff}	+/- 1σ	K_{eff} +2σ	Pellet OD (inches)
0.8626	0.0010	0.8646	0.3831
0.8610	0.0010	0.8631	0.3833
0.8619	0.0012	0.8642	0.3834
0.8630	0.0012	0.8653	0.3836
0.8628	0.0013	0.8655	0.3838
0.8621	0.0013	0.8647	0.3839
0.8615	0.0011	0.8638	0.3841

A6.5 Critical Benchmark Experiments

A6.5.1 Benchmark Experiments and Applicability

A series of 122 benchmark experiments were simulated in calculations with the SCALE 4.4 PC/CSAS25 package [A6.8] using the 44-group cross-section library. 110 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:

A. water moderation

B. boron neutron absorbers

C. unirradiated light water reactor type fuel (no fission products or "burnup credit")

D. close reflection

E. near room temperature (vs. reactor operating temperature)

F. Uranium oxide and MOX fuels.

Verification and validation (V&V) of the software is performed in accordance with QA program requirements. 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.

Criticality codes are verified by comparing benchmark calculations to actual critical benchmark experiments. The difference between the calculated reactivity and the experimental reactivity is referred to as 'calculational' bias. This bias may be a function of system parameters such as fuel lattice separation, fuel enrichment, neutron absorber properties, reflector properties, or fuel/moderator volume ratio; or, there may be no specific correlation with system parameters. The benchmark data used for determination of the USL is provided in Table A6.5-1. *The set of criticality experiments used as benchmarks are representative of the composition, configuration,*

and nuclear characteristics of the system modeled. Eight parameters were selected in order to demonstrate the applicability of the SCALE 44-group ENDF/B-V cross-section library for the range of conditions spanned by the calculation models. The results of these evaluations are provided in Table A6.5-2. Only those experiments with the parameter in question were used to determine the USL for that parameter. The methodology used to calculate the USL is based on 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 Δk) to the linear fit of results of the experimental benchmark data developed. The USL from the data set with the best correlation is used as the acceptance criteria for subsequent criticality evaluations. Since there was not a strong correlation for any of the data sets, i.e., the correlation was essentially random, the lowest possible USL-1 result was used as the USL.

The uncertainty due to modeling approximations does not impact the calculated k_{eff}. Worst case tolerances (as specified in the design drawings presented in Chapter1) are used in the analysis to maximize k_{eff}. Only the tolerances of those dimensions that had a positive effect on K_{eff} were included in the SCALE geometry models.

A6.5.1.1 Uranium Oxide Fuel Assemblies

The 110 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 0.67. 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 criticality evaluation presented here used the same cross section library, fuel materials and similar material/geometry options that were used in the 122 benchmark calculations as shown in Table A6.5-3. The modeling techniques and the applicable parameters listed in Table A6.5-3 for the actual criticality evaluations fall within the range of those addressed by the benchmarks in Table A6.5-2. 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.*9401* based on comparisons to the most limiting assembly parameters.

Table A6.5-1

Benchmark Results

Run ID	U Enrich. Wt%	Pu Enrich. Wt%	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	Boron Wt%	B-10 Areal Density (g/cm^2)	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	1.6140		0.9962	0.0010
BW1484A2	2.46		1.636	1.841	4.908	35.1629	0.1000		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
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
BW1810J	2.46		1.636	1.841		33.1403			0.9983	0.0011
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	28.7000	0.0670	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	28.7000	0.0670	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)

Run ID	U Enrich. Wt%	Pu Enrich. Wt%	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	Boron Wt%	B-10 Areal Density (g/cm^2)	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	28.7000	0.0669	0.9988	0.0012
P3314BC	4.31		1.892	1.6	2.83	33.2284	31.8800	0.0262	0.9992	0.0012
P3314BF1	4.31		1.892	1.6	2.83	33.2505	32.7400	0.0236	1.0037	0.0013
P3314BF2	4.31		1.892	1.6	2.83	33.2184	32.7400	0.0471	1.0009	0.0013
P3314BS1	2.35		1.684	1.6	3.86	34.8594	1.0500	0.0045	0.9956	0.0013
P3314BS2	2.35		1.684	1.6	3.46	34.8356	1.6200	0.0069	0.9949	0.0010
P3314BS3	4.31		1.892	1.6	7.23	33.4247	1.0500	0.0045	0.9970	0.0013
P3314BS4	4.31		1.892	1.6	6.63	33.4162	1.6200	0.0069	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	30.3600	0.0408	1.0029	0.0013
P3602BS1	2.35		1.684	1.6	4.8	34.7750	1.0500	0.0045	1.0027	0.0012
P3602BS2	4.31		1.892	1.6	9.83	33.3679	1.0500	0.0045	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)

Run ID	U Enrich. Wt%	Pu Enrich. Wt%	Pitch (cm)	H_2O/fuel volume	Separation of assemblies (cm)	AEG	Boron Wt%	B-10 Areal Density (g/cm^2)	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	35.6300		1.0012	0.0013
P71F14F3	4.31		1.891	1.6	5.19	32.8237	29.2200		1.0009	0.0014
P71F14V3	4.31		1.891	1.6	5.19	32.8597	29.2200		0.9972	0.0014
P71F14V5	4.31		1.891	1.6	5.19	32.8609	29.2200		0.9993	0.0013
P71F214R	4.31		1.891	1.6	5.19	32.8778	29.2200		0.9969	0.0012
PAT80L1	4.74		1.6	3.807	4.9	35.0253	22.2000	0.0460	1.0012	0.0012
PAT80L2	4.74		1.6	3.807	4.9	35.1136	22.2000	0.0460	0.9993	0.0015
PAT80SS1	4.74		1.6	3.807	4.9	35.0045	22.2000	0.0460	0.9988	0.0013
PAT80SS2	4.74		1.6	3.807	4.9	35.1072	22.2000	0.0460	0.9960	0.0013
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	0.34	-0.26	0.41	0.24	0.67	-0.04	0.36	0.05	N/A	N/A

Table A6.5-2

USL-1 Results

Parameter	Range of applicability	USL-1
U Enrichment (wt% U-235)	2.4 2.8 3.3 3.8 – 5.7	*0.9422* *0.9428* *0.9434* *0.9436*
Pu Enrichment (wt% Pu)	2.0 – 6.6	*0.9417*
Fuel Rod Pitch (cm)	1.07 1.1 1.4 1.6 1.9 – 2.6	*0.9401* *0.9405* *0.9418* *0.9430* *0.9436*
Water/Fuel Volume Ratio	0.38 1.9 3.3 – 11	*0.9412* *0.9423* *0.9423*
Assembly Separation (cm)	1.6 4.4 7.1 9.8 – 21	*0.9409* *0.9425* *0.9440* *0.9441*
Average Energy Group Causing Fission (AEG)	30 – 37	*0.9431*
B-10 Areal Density (g/cm^2)	*4.50E-3* *1.34E-2 – 2.23E-2* *3.12E-2* *4.02E-2 – 4.91E-2* *5.80E-2 – 6.69E-2*	*0.9434* *0.9435* *0.9436* *0.9437* *0.9438*
Boron Content (wt% B)	*0.1* *5.2* *10* *15* *20* *25* *31 – 36*	*0.9421* *0.9425* *0.9429* *0.9433* *0.9437* *0.9441* *0.9445*

Table A6.5-3

Fuel Assembly Design Parameters Used in Criticality Benchmarks

Parameter	WE 14x14	MOX 14x14
Fuel Material -	UO_2	MOX
U-enrichment (wt% U-235)	4.05	0.71
Pu-enrichment (wt% Pu)	0.00	3.3 - 3.85
Clad Material	Type 304 SS	Zircaloy-4
Fuel Rod Pitch (cm)	1.4 (1.07 - 1.66)*	1.4 (1.07 – 1.66)*
Water/Fuel Volume Ratio	1.5 (0.3 - 2.5)*	1.6 (0.4 – 2.7)*
Assembly Separation (cm)	2.5-4.2	2.5-4.2
Average Energy Group Causing Fission (AEG)	33 (34)*	32 (33)*
B-10 Areal Density (g/cm^2)	0.019	0.019
Absorber Material - B (wt %)	18.9	18.9

* (Damaged fuel)

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

Parameter	WE 14x14 SC	WE 14x14 MOX	Most Limiting USL-1
U-enrichment (weight % U-235)	4.05	0.71	0.9436
Pu-enrichment (weight % Pu)	0.00	3.3 - 3.85	0.9417
Fuel Rod Pitch (cm)	1.4 (1.07 - 1.66)*	1.4 (1.07 – 1.66)*	0.9401
Water/Fuel Volume Ratio	1.5 (0.3 - 2.5)*	1.6 (0.4 – 2.7)*	0.9412
Assembly Separation (cm)	2.5 - 4.2	2.5 – 4.2	0.9409
Average Energy Group Fission (AEG)	33 (34)*	32 (33)*	0.9431
B-10 Areal Density (g/cm2)	*0.019*	*0.019*	*0.9434*
Absorber Material – B (wt. %)	*18.9*	*18.9*	*0.9433*

*(Damaged fuel)

A8. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The acceptance tests and maintenance program described in Chapter 8 are *applicable to the MP187 Cask and 24PT1-DSC payload.*