TRANSNUCLEAR
AN AREVA COMPANY

August 5, 2010
E-29470

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Subject: Revision 3 to Transnuclear, Inc. (TN) Application for Amendment 11 to the Standardized NUHOMS® System, Response to First (Restart) Request for Additional Information (Docket No. 72-1004; TAC NO. L24080)

Reference: Letter from B. Jennifer Davis (NRC) to Donis Shaw (TN), "FIRST (RESTART) REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF AMENDMENT 11 TO THE STANDARDIZED NUHOMS® SYSTEM (TAC NO. L24080)," February 19, 2010

This submittal provides responses to the request for additional information (RAI) forwarded by the reference above. Enclosure 2 herein provides each of the NRC staff RAI followed by a TN response. Enclosure 3 provides a list of additional changes, not associated with the RAI. Enclosure 4 provides a list of Certificate of Compliance (CoC), Technical Specifications (TS), and updated final safety analysis report (UFSAR) pages associated with Revision 3. For the most part these are changed pages, but based on the RAIs, all TS pages are included, for continuity. Enclosure 6 provides the pages.

In the TS and UFSAR, changes made in response to Amendment 11 RAI No. 1, RAI No. 2, and new changes based on this current RAI are indicated by revision bars in the right margin and italics for inserted text. The new changes are shaded, to distinguish them from the RAI No. 1 and RAI No. 2 changes. Changed UFSAR pages are annotated as Revision 3. Page change instructions for CoC, TS, and UFSAR pages are provided in Enclosure 5.

CoC 1004 UFSAR Revisions 10 and 11 were issued on February 1, 2008 and February 1, 2010, respectively, both submittals occurring after the initial application for Amendment 11. Additionally, the update to Revision 11 incorporated CoC 1004 Amendment 10, which affected a great many UFSAR pages which are also being proposed for changes under Amendment 11. Based on this history, TN has taken steps to ensure that UFSAR pages with proposed Amendment 11 changes affected by the UFSAR update to Revisions 10 or 11 are included herein and reflect the latest UFSAR changes.

Enclosures 6 and 8 of this submittal include proprietary information which may not be used for any purpose other than to support NRC staff review of the application. In accordance with 10 CFR 2.390, I am providing an affidavit (Enclosure 1) specifically requesting that you withhold this proprietary information from public disclosure. Enclosure 7 provides non-proprietary versions of changed proprietary UFSAR pages in Enclosure 6. Enclosure 9 provides a non-proprietary version

NMSS01

of the calculation provided in Enclosure 8.

Should the NRC staff require additional information to support review of this application, please do not hesitate to contact Mr. Don Shaw at 410-910-6878 or me at 410-910-6881.

Sincerely,

Jayant Bondre

Jayant Bondre, PhD
Vice President - Engineering

cc: Jennifer Davis (NRC SFST), as follows, provided in a separate mailing:

- 11 copies of this cover letter and Enclosures 1 through 6
- *one copy of Enclosure 8*

Enclosures:

1. Affidavit Pursuant to 10 CFR 2.390
2. RAIs and Responses
3. List of Additional Changes Not Associated with the RAI
4. List of CoC, Technical Specifications, and UFSAR Pages Associated with Amendment 11, Revision 3
5. CoC, Technical Specifications, and UFSAR Page Change Instructions
6. Amendment 11 Revision 3 Changed Certificate of Compliance, Technical Specifications, and UFSAR pages (proprietary)
7. Amendment 11 Revision 3 Changed UFSAR Pages (non-proprietary versions)
8. TN Calculation NUH06L-0505, Revision 0 (proprietary version)
9. TN Calculation NUH06L-0505, Revision 0 (non-proprietary version)

AFFIDAVIT PURSUANT TO 10 CFR 2.390

Transnuclear, Inc.)
State of Maryland) SS.
County of Howard)

I, Jayant Bondre, depose and say that I am a Vice President of Transnuclear, Inc., duly authorized to execute 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 2.390 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in Enclosures 6, and 8 as listed below:

Enclosure 6:

- Portions of the UFSAR Chapter W.5

Enclosure 8:

- Portions of TN Calculation NUH06L-0505, Revision 0

These documents have been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Transnuclear, 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 2.390 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 involves certain safety analysis report shielding analysis information and a shielding analysis, all related to the design of the NUHOMS® OS197L transfer cask which are owned and have been held in confidence by Transnuclear, Inc.

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

3) Public disclosure of the information is likely to cause substantial harm to the competitive position of Transnuclear, Inc. because the information consists of descriptions of the design and analysis of dry spent fuel storage systems, the application of which provide a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Transnuclear, Inc., take marketing or other actions to improve their product's position or impair the position of Transnuclear, Inc.'s

product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

Further the deponent sayeth not.

Jayant Bondre
Vice President, Transnuclear, Inc.

Subscribed and sworn to me before this 5th day of August, 2010.

Notary Public

My Commission Expires 10/14/2012

ARLEASTRICE M. TAYLOR
NOTARY
PUBLIC
HOWARD COUNTY, MD

TECHNICAL SPECIFICATIONS – GENERAL

1-1 Technical Specifications 4.3.3(8), 4.3.3(9), 4.3.3(10) and 4.3.3(11) (former pages 4-26 and 4-27(added with December 21, 2007 responses to RAI #1)). These TS appear to have been removed without any specific discussion or justification. It is unclear whether they were removed purposefully with the August 14th responses to RAI #2, or inadvertently removed as a result of changed TS pages. (The subject TS were on pages that are now part of the ASME Code alternatives section. It is possible that a new page 4-32 could contain these TS, but was not provided with the August 14th responses to RAI #2.) Either provide the appropriate changed page(s), or provide the justification for removing the conditions from the TS. (See also RAIs 1-2, 2-1, 5-4, 5-7, 5-11, Q-3, Q-17, Q-35, and Q-38.)

It is inappropriate to remove conditions from the TS without explicitly providing an adequate justification.

Response to 1-1

A new TS page 4-33 which contained these TS was inadvertently not provided with the 8/14/09 responses to RAI #2. The TS pages provided with this submittal include these items.

1-2 Technical Specifications 5.5 Concrete Testing for HSM-H (former page 5.5-1). This TS appears to have been removed without any specific discussion or justification. It is unclear whether it was removed purposefully with the responses to RAI #1 and RAI #2, or inadvertently removed as a result of changed TS pages. The TS go from 5.4 (pages 5-13 and 5-14) to 5.6 on page 5-15. In addition, TS 5.5 appears in the Table of Contents (page ii), which was unchanged with the RAI #2 update. Either provide the appropriate changed page(s), or provide the justification for removing the conditions from the TS. (See also RAIs 1-1, 2-1, 5-11, Q-3, Q-17, Q-35, and Q-38.)

It is inappropriate to remove conditions from the TS without explicitly providing an adequate justification.

Response to 1-2

Section 5 pages were re-numbered in the 12/21/07 response to RAI No. 1. TS 5.5 and 5.6 were on Page 5-14 and 5-15, respectively, in that submittal. In the 8/14/09 response to RAI No. 2, a new TS Page 5-14 was provided, containing changed information for TS 5.4. At that time, TS 5.5 and 5.6 had shifted to Page 5-15 and 5-16, respectively, but those pages were inadvertently not provided in the 8/14/09 submittal. The TS pages provided with this submittal include these items.

CHAPTER 2 Structural Evaluation

2-1 Section W3.9. Provide analyses addressing the accidental drop for the OS197L TC supplemental shielding component, namely the top (outer) skid shielding, and include this analysis in the updated SAR. This component may be handled outside the fuel/reactor building.

The latest revision of this amendment in Section W3.9 indicates that the accidental drop of the top (outer) skid shielding is provided in Section W11.1.5. Reference to Section W11.1.5 is also made for the outer top shielding drop analyses on page 10-34 of the SAR (Tech Spec Bases). The staff could not locate this analysis in the current version of the application. (See also RAIs 5-11 and Q-38.)

This information is needed to confirm compliance with 10 CFR 72.236(l).

Response to 2-1

Section W.11.1.5, which is also referenced in Section W.3.9, is added to the SAR to document the accident drop of the top (outer) skid shielding. Section W.11.1.5 was included in the 12/21/07 RAI No. 1 response submittal but was inadvertently omitted in the 8/14/09 RAI No. 2 response submittal.

2-2 Section W11.1.3. Provide the drop accident analysis for the OS197L TC and include a discussion of consequences subsequent to the accidental drop, as applicable, in the updated SAR.

The current revision of this amendment in Section W11.1.3 indicates that the discussion of the OS197L TC drop accident analysis is included in Section W3.1.3. The staff could not locate this analysis in the current version of the application.

This information is needed by the staff to confirm compliance with 10 CFR 72.236(l).

Response to 2-2

The discussion of the OS197L TC drop accident analysis is provided in UFSAR Section W.3.7. Page W.11-2 is corrected to reference the correct UFSAR section. UFSAR Section W.3.7 provides the justification that the bounding 75 g drop accident accelerations used for the OS197 TC are also applicable to the OS197L TC. The OS197L TC structural shell is significantly thicker than the OS197 TC structural shell (2.68" for the OS197L TC versus 1.5" for the OS197 TC) and the top and bottom forging assemblies have not changed between the OS197 and OS197L TCs. Thus, the OS197L TC provides higher load capacity than the OS197 TC, and therefore, the drop accident stresses obtained for the OS197 TC are a conservative estimation of the OS197L TC drop accident stresses.

As with the OS197 TC and documented in UFSAR Section 8.2.5.3, complete loss of neutron shield is postulated for the OS197L TC as a consequence of the drop accident event. This event is evaluated in Chapters W.4 and W.5 for thermal and shielding effects. The post accident recovery actions as discussed in Section 8.2.5.4 are applicable to the OS197L TC. This paragraph is added to SAR Section W.3.7

CHAPTER 5 Shielding Evaluation, and CHAPTER 8 Radiation Protection Evaluation

Certificate of Compliance

5-1 Provide suggested text to revise the CoC to acknowledge shielding as a design function of the TC, specify the maximum loaded weight (as opposed to a range) of the TC for each series (OS200, OS197, OS197L, etc.), and provide construction materials. In addition, in Section 3.d, add the trailer shielding for the OS197L.

This information is needed to provide an adequate description of the system to which the certificate applies. (See also RAI 5-14.)

Response to 5-1

The changes described in this RAI are made to CoC Section 3.b and Section 3.d on TN's mark-up of suggested text.

Technical Specifications (TS)

5-2 Revise the TS so that in each place where Dry Storage Canisters (DSCs) are enumerated (2.1, 5.2.4(c), 5.2.4(e), 5.4.2, etc), all DSCs are included.

In the CoC the following DSCs are specified: 24P, 52B, 61BT, 32PT, 24PHB, 24PTH, 61BTH, and 32PTH1. In various parts of the TS, the following DSCs are included: 24P; 52B; 61BT; 32PT; 24PHB; 24PTH; 24PTH-S; 24PTH-L; 24PTH-S-LC; 24PTH-SLC; 61BTH; 61BTH, Type 1; 61BTH, Type 2; and 32PTH1. In the SAR the 24PT2 (S and L) is added. The CoC and TS need to include all the DSCs so that it is clear what fuel is allowed in each one and what conditions (e.g., dose rate limits) apply to each.

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 5-2

The designation 24PTH-SLC (now moved to the table notes) is corrected to 24PTH-S-LC on TS Pages 5-7 and 5-13.

DSC models are listed in the CoC. If the model number has a variant which specifically has certain limitations, then those are specifically called out in the TS. Information concerning the fuel types, dose rate limits or other technical specifications apply to all variants if they are not explicitly mentioned in the CoC or technical specifications. An example is the 24PTH DSC. In this case, 24PTH is the model number. The 24PTH-S, -L and –S-LC are variants with specific limitations, which are called out in the TS.

24PT2S and 24PT2L are variants of the 24P DSC model and they are not specifically called out in the TS because all of the requirements in the CoC and TS for the 24P DSC apply to the 24PT2S and 24PT2L.

Discussion to this effect is added to TS 2.1, to make this clear.

5-3 <u>TS 1.1 Definitions (page TS 1-2)</u>. "TRANSFER OPERATIONS" – revise the following sentence as indicated below:

... TRANSFER OPERATIONS begin after the TRANSFER CASK has been placed horizontal on the transfer trailer (and for the OS197L, the supplemental trailer shielding has been put in place) ready for TRANSFER OPERATIONS and end when the TC is at its destination and no longer secured on the transfer trailer. ...

The proposed wording is not consistent with the definition of "LOADING OPERATIONS" provided on the preceding page.

Response to 5-3

The sentence is revised to be consistent with the definition of "LOADING OPERATIONS".

5-4 TS 4.4 TRANSFER CASK Design Features (page TS-33). TS 4.4.1 (restriction of use to sites with limited crane capacity) was removed without any specific discussion or justification. Reinstate the condition or provide the justification for removing the condition that staff had previously indicated should be included in the TS.

It is inappropriate to remove conditions from the TS without explicitly providing an adequate justification.

Response to 5-4

The justification for removing the condition is as follows. In discussions with the NRC staff in an April 16, 2009 public meeting (ML091490353), TN indicated their intention to de-rate the OS197L TC for use with 100 ton or greater capacity cranes. The de-rate involves limiting the OS197L TC for use only with 32PT and 61BT DSCs, at a maximum heat load of 13 kW instead of the original 24 kW. The de-rate significantly lowers TC dose rates, making them comparable to other, approved transfer cask designs. Therefore, it is acceptable for sites with 100 ton or greater capacity cranes to use the OS197L TC.

5-5 TS 5.2.4d Radiation Protection Program (page TS 5-7, last paragraph). Revise the first sentence to read as follows: "If the required limits are not met, any available commercial decontamination technique may be used on the entire length of the DSC outer surface to reduce the DSC surface contamination levels to below the required limits."

This revision is needed to ensure there is no confusion with respect to the extent of the DSC surface that needs to be decontaminated in the event contamination levels are found to be above the specified levels.

Response to 5-5

The change requested by this RAI is made and included herein.

5-6 TS 5.2.4e Radiation Protection Program (page TS 5-7). Verify and justify (by providing a citation in the RAI response to the appropriate table in the SAR) the values specified in the dose rate limits tables. **(Note that this information was requested in RAI #2 as part of RAI 5.8, but was not satisfactorily addressed in the August 14th response.)**

The values in the TS need to have a clear link to analytical results. For example, in the non-OS197L table, an appropriate value for the axial surface dose rate for the 24P would seem to be 40 or 50 mrem/hour based on Table 7.3-2. If some other table (or text) justifies a higher value, please provide that citation in the RAI response. Similarly, in the OS197L table, an appropriate value for the axial surface dose rate would appear to be in the range of 120 to 130 mrem/hour based on Table M.5-5 which is cited in Section W.5.1.3 as bounding for both DSCs. In addition, based on Table W.5-3 or Table W.5-13, an appropriate value for the radial decontamination area surface dose rate limit would appear to be at most 60 mrem/hour. Based on these spot checks, all values in these tables need to be justified with respect to a value in an appropriate reference in the SAR.

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 5-6

The transfer cask dose rate limits included in Technical Specification (TS) 5.2.4e are shown in Table 5-6-1 and Table 5-6-2 below. Table 5-6-1 lists the axial surface dose rates and Table 5-6-2 lists the radial surface dose rates. Also included in these tables are the citations to the Table or Figure in the UFSAR where these dose rates are calculated, including the calculated dose rates. Based on a comparison of the calculated and specified dose rates (for TS 5.2.4e), it is clear that only a small adjustment is made (less than 10%) to obtain the specified dose rates to account for uncertainty. The TS dose rate limits are revised based on the values shown in these tables. This information is also included in the TS Bases (Chapter 10, Section B.10.5.2.4.e of the UFSAR).

Table 5-6-1: Transfer Cask Axial Surface Dose Rates

DSC	TS Limit (mrem/hour)	Calculated (mrem/hour)	Configuration in UFSAR
24P	700	656	Table J.5-3, Shield Plug
52B	700	656	Bounded by 24P
61BT	800	< 800	Inner Cover Welding, Figure K.5-14
32PT	900	< 900	Inner Cover Welding, Figure M.5-27
24PHB	1400	1350 (250 + 1100)	Inner Cover Welding, Figure N.5-14 (neutron), Figure N.5-17 (gamma)
24PTH(1)	900	884	Welding, Table P.5-4
24PTH-S-LC	900	884	Bounded by 24PTH
61BTH	2200	2190	Welding, Table T.5-5
32PTH1	800	762	Welding, Table U.5-3
61BT(2)	800	< 800	Inner Cover Welding, Figure K.5-14
32PT(2)	900	< 900	Inner Cover Welding, Figure M.5-27

(1) Does not apply to 24PTH-S-LC

(2) Applicable only to the OS197L TC

Table 5-6-2: Transfer Cask Radial Surface Dose Rates

DSC	TS Limit (mrem/hour)	Calculated (mrem/hour)	Configuration in UFSAR
24P	600	592	Table 7.3-2, TRANSFER
52B	600	592	Bounded by 24P
61BT	1200	1156	Table K.5-2, TRANSFER
32PT	1000	950	Table M.5-5, TRANSFER
24PHB	1200	1193	Table N.5-3, TRANSFER
24PTH(1)	1500	1500	Table P.5-3, TRANSFER
24PTH-S-LC	600	577	Table P.5-5, TRANSFER
61BTH	1350	1320	Table T.5-4, TRANSFER
32PTH1	650	609	Table U.5-2, TRANSFER
61BT(2)	100	< 100	Figure W.5-2 (Maximum)
32PT(2)	100	< 100	Figure W.5-2 (Maximum)

(1) Does not apply to 24PTH-S-LC

(2) Applicable only to the OS197L TC

5-7 TS 5.4 HSM or HSM-H Dose Rate Evaluation Program (pages TS 5-13 through 14).

a) Replace the end shield wall dose rate limit and measurement in TS 5.4.1 and TS 5.4.2 that was removed in this revision.

b) Simplify the measurement process – specify the number and location of measurements, then compare each measurement to established limits. This may entail adding a separate dose rate limit for the bird screens.

c) Confirm the values in the dose rate limit table in TS 5.4.2 and justify in the RAI response by providing a supporting citation in the SAR for each value. Do this for all the values, including those that will be added based on the items above in this RAI.

Information was deleted from the TS without a justification being provided. Staff has reviewed the proposed method for verifying the TS compliance and determined that a method that directly compares measured to calculated dose rates is preferable. The dose rate values in the TS need to be easily traceable to values in the Updated Final Safety Analysis Report (UFSAR).

Response to 5-7

a) The end shield wall dose rate was removed from the Technical Specification (TS) because it was not representative of a single loaded HSM, but rather an array of HSMs. Further, as discussed with the staff on June 9, 2009, the radial dose rate TS limits for the Transfer Cask render the end shield wall dose rate limits redundant.

b) The measurement process is simplified, resulting in a separate dose rate limit for the bird screens. Please see the response to item c), below.

c) The HSM dose rate limits included in Technical Specification (TS) 5.4.2 are shown in Tables 5-7-1 through Table 5-7-3 below. Table 5-7-1 lists the dose rates at the outer surface of the door. Table 5-7-2 lists the HSM front surface dose rates. Table 5-7-3 lists the surface dose rates near the HSM front birdscreens. Also included in these tables are the citations to the Table or Figure in the UFSAR where these dose rates are calculated, including the calculated dose rates. Based on a comparison of the calculated and specified dose rates (for TS 5.4.2), it is clear that only a small adjustment is made (less than 10%) to obtain the specified dose rates to account for uncertainty. The minimum dose rate for measurement is set to 5 mrem/hour. This information is also included in the TS Bases (Chapter 10, Section B.10.5.4. of the UFSAR).

Table 5-7-1: HSM Door Outer Surface Dose Rates

DSC	HSM	TS Limit (mrem/hour)	Calculated (mrem/hour)	Configuration in UFSAR
24P	Standardized HSM	80	62.3	Table 7.3-2
52B	Standardized HSM	80	62.3	Bounded by 24P
61BT	Standardized HSM	200	194	Table K.5-2
32PT	Standardized HSM	200	185	Table M.5-3
24PHB	Standardized HSM	20	17.0	Table N.5-3
24PTH-S-LC	Standardized HSM	80	62.2	Table P.5-2
61BTH	Standardized HSM	100	88.4	Table T.5-3
24PTH	HSM-H	5	1.3	Table P.5-1
61BTH	HSM-H	5	0.35	Table T.5-1
32PTH1	HSM-H	5	0.7	Table U.5-1

Table 5-7-2: HSM Front Surface Dose Rates

DSC	HSM	TS Limit (mrem/hour)	Calculated (mrem/hour)	Configuration in UFSAR
24P	Standardized HSM	60	48.6	Table 7.3-2
52B	Standardized HSM	60	48.6	Bounded by 24P
61BT	Standardized HSM	150	118	Table K.5-3
32PT	Standardized HSM	100	95.5	Table M.5-4
24PHB	Standardized HSM	20	9.5	Table N.5-4
24PTH-S-LC	Standardized HSM	60	46.5	Table P.5-2
61BTH	Standardized HSM	30	25.6	Table T.5-3
24PTH	HSM-H	40	32.3	Table P.5-1
61BTH	HSM-H	20	9.61	Table T.5-1
32PTH1	HSM-H	20	15.2	Table U.5-1

Table 5-7-3: HSM Front Bird Screen Surface Dose Rates

DSC	*HSM*	*TS Limit (mrem/hour)*	*Calculated (mrem/hour)*	*Configuration in UFSAR*
24P	*Standardized HSM*	*350*	*326*	*Table 7.3-2*
52B	*Standardized HSM*	*350*	*326*	*Bounded by 24P*
61BT	*Standardized HSM*	*1300*	*1240*	*Table K.5-2*
32PT	*Standardized HSM*	*850*	*788*	*Table M.5-3*
24PHB	*Standardized HSM*	*550*	*480*	*Table N.5-4*
24PTH-S-LC	*Standardized HSM*	*550*	*492*	*Table P.5-2*
61BTH	*Standardized HSM*	*200*	*177*	*Table T.5-3*
24PTH	*HSM-H*	*1300*	*1237*	*Table P.5-1*
61BTH	*HSM-H*	*650*	*598*	*Table T.5-1*
32PTH1	*HSM-H*	*550*	*478*	*Table U.5-1*

Updated Final Safety Analysis Report

5-8 Section 1.3.2.1 On-Site TC (page 1.3-3). What is meant by the following statement (about three-fourths of the way down the page)? "In addition, the licensee may also elect to utilize a future transfer cask having a gross weight of about 113.4 Te (125 tons) which can be used on-site under 10CFR72, but is also suitable for future off-site shipment of intact NUHOMS® canisters under 10CFR71." Note that the current review is for use under 10 CFR Part 72, only.

Response to 5-8

UFSAR Section 1.3.2.1 is revised to delete the noted sentence.

5-9 Section 3.1.2.1 Handling and Transfer Equipment – On-Site Transfer Cask (page 3.1-4). In the last paragraph, revise the third sentence to encompass the OS197L transfer cask for which the existing wording is not correct. One possible solution would be to modify the sentence so that it reads something like: "The cask provides ... the cask; the exception to this being the OS197L transfer cask which requires supplemental shielding to provide an equivalent level of protection."

The SAR needs to be clear on the limitations and restrictions of the OS197L TC design.

Response to 5-9

The sentence is revised as suggested.

5-10 Section B.10.5.2.4d Maximum DSC Removable Surface Contamination, BASES (page 10-31). Add a sentence to the second paragraph to clarify the extent of decontamination needed in the unlikely event that removable contamination above the specified level is found in the area surveyed. One possible wording could be: However, in the unlikely event that contamination is found that exceeds the specified levels, the entire length of the DSC surface needs to be decontaminated.

This revision is needed to ensure there is no confusion with respect to the extent of the DSC surface that needs to be decontaminated in the event contamination levels are found to be above the specified levels.

Response to 5-10

The suggested sentence is added to the second paragraph on UFSAR Page 10-31.

5-11 Missing Section W.11.1.5 (RAI #1 responses, page W.11-4).

a) B.10.5.3.4 Supplemental Shielding Drop onto OS197L TC (BASES) (page 10-34). This paragraph refers the reader to Section W.11.1.5; however, there is no Section W.11.1.5 in the submittal. Please revise to refer to the appropriate section, or provide the information if not already included.

b) W.3.9 Structural Evaluation of OS197L TC Supplemental Shielding Components (page W.3-4). The last paragraph in this section refers the reader to Section W.11.1.5; however, there is no Section W.11.1.5 in the submittal. Please revise to refer to the appropriate section, or provide the information if not already included.

c) See also RAIs 2-1 and Q-38.

The submittal needs to be complete.

Response to 5-11

Please see the response to RAI 2-1.

5-12 Appendix W – General. For each section that consists of the statement "**No change**," provide the section (presumably in the main body of the SAR) that is being referenced.

This was addressed for the operations section in the August 14th response to RAI #2, question 5.25, but was not specifically requested for the entire appendix. However, since more instances of this ambiguous reference have been noted, the information is now being requested for all of Appendix W.

Response to 5-12

The sections with the "No Change" statement of Appendix W of the SAR are revised to identify the applicable sections of the main body of the SAR. In addition, TN will implement similar *revisions in various other appendices of the UFSAR through the provisions of the 10 CFR 72.48 process.*

5-13 Section W.1.1, Introduction, Section W.1.2.2 (pages W.1-2 and W.1-3). Remove the phrase "any alternate suitable targeting system," or specify how those alternate targeting systems can be verified to be suitable.

This information is requested to satisfy the requirements of 10 CFR 72.236(d).

Response to 5-13

The phrase "or any alternate suitable targeting system" on Pages W.1-2 and W.1-3 has been removed and the surrounding text is changed to requirements which are consistent with the language in Technical Specifications Section 4.4.2. Similar changes are made to the CoC (Condition 8, Loading Operations Item e.) and UFSAR Pages W.8-4, W.8-5, W.8-7, and W.8-14.

5-14 Section W.1.1, Introduction (page W.1-2). The last sentence of the second to last paragraph states, "[t]he OS197L TC can be configured to meet a gross weight limit of 85 tons." Clarify this sentence to indicate the empty weight (in tons), the nominal loaded weight (in tons), and the applicable 85-ton configuration.

This information is needed to provide an adequate description of the system to which the certificate applies. (See also RAI 5-1.)

Response to 5-14

Additional clarification is added to Section W.1.2 as requested.

5-15 Table W.1-2, OS197L TC UFSAR Sections Affected (page W.1-8). Provide a revised page W.1-8.

Given that Chapter 10 of the UFSAR was changed in response to the RAIs, it seems reasonable to assume that the corresponding table on page W.1-8 would be changed; however, a revised table was not provided.

Response to 5-15

Amendment 11 proposed UFSAR changes include adding Table 10-2, "Technical Specification Cross Reference Table between Amendment 10 and Amendment 11." Line items at the end of Table 10-2 account for situations which do not carry over from Amendment 10 Technical Specifications (TS) but which affect the Amendment 11 TS and therefore are of interest. One of those items is associated with the addition of the OS197L to the Standardized NUHOMS® System. Also, UFSAR Section B 10.5.3.4 was added as a result of adding the OS197L to the Standardized NUHOMS® System. UFSAR Table W.1-2 is updated to reflect these items.
Also, the entire UFSAR was re-reviewed for sections affected by the OS197L TC and additional changes were made to the table.

5-16 Chapter W.5. Submit the shielding calculation(s) that document(s) the evaluation(s) done for this chapter. At a minimum, this should include calculations NUH06L-0503 and NUH06L-0504.

There are several aspects of the shielding evaluation that are not adequately/clearly explained in Chapter W.5. For example, it is not clear that the openings in the decontamination area shield or the vents in the transfer trailer shielding were incorporated into the shielding models. There is not sufficient justification for the assertion that the dose rate distribution above the DSC is the same as (or bounded by) that in Appendix M. Also, given that the maximum gamma and neutron dose rates may not occur at the same location and Chapter W.5 does not contain the dose rate distributions, it is difficult for staff to verify the maximum reported dose rates.

This information is necessary to complete the review.

Response to 5-16

Please see the response to RAI 5-33 for information on Calculation NUH06L-0503. Calculation NUH06L-0503 has been superseded by TN calculation NUH06L.0506 to document a simplified occupational exposure calculation following crane malfunction event. TN calculation NUH06L.0506, Revision 0 was provided to the NRC staff in November 2009.

The initial Amendment 11 application was based on a heat load of 24 kW per DSC using the 32PT as the design basis DSC for the shielding evaluation. The shielding evaluation documented therein included design basis 32PT DSC source terms directly from UFSAR Chapter M.5. These source terms were considered generally bounding for the 24P, 52B, 61BT and 24PHB DSCs. Calculation NUH06L.0504 documents this shielding evaluation of the OS197L cask using the 32PT design basis source terms.

In the subsequent revision to the application (revision #2) as a response to RAI #2, the OS197L was de-rated to a decay heat limit of 13 kW/DSC and restricted to the 61BT and 32PT DSCs only. The revised shielding evaluation documented in UFSAR Chapter W.5 was revised to include the fuel qualification evaluations for reduced heat loads and calculation of design basis source terms. The dose rate evaluation was also revised for this purpose and ensured that the maximum calculated bare cask dose rate was less than 10,000 mrem/hour. Calculation NUH06L.0505 documents this source term and shielding evaluation of the OS197L cask with both the 32PT and 61BT DSCs. Therefore, this calculation is submitted as part of this response and is included as Enclosure 8. The representative computer files are included in Section W.5.6.

Note that UFSAR Chapter W.5 is also revised to include additional details, including justification for using axial dose rate results from UFSAR Chapter K.5 and Chapter M.5.

5-17 Section W.5.1.2, Bounding Dose Rates as a Function of Distance (pages W.5-4, W.5-38, and W.5-40). Confirm what is plotted in Figure W-5-2 and revise the text, referenced table, and figure, appropriately.

The last paragraph indicates that the plotted values in Figure W.5-2 for the case "*Below Cask Support Skid when Cask is on Trailer with Inner and Outer Trailer Top Shielding*" come from Table W.5-14. However, Table W.5-14 indicates that its data represents the configuration "Prior to Installation of Outer Top Trailer Area Shielding."

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 5-17

Section W.5.1.2 is modified and includes description of the dose rates plotted in Figure W.5-2. The plot for the case "Below Cask Support Skid when Cask is on Trailer with Inner and Outer Trailer Top Shielding" shown in Figure W.5-2 is based on dose rates calculated at an elevation below the cask support skid. The dose rate distribution at this elevation is not significantly dependent on the outer top trailer shielding. Therefore, the dose rates shown in Table W.5-14 are applicable.

5-18 Section W.5.2, Source Specification (pages W.5-6 and W.5-7). Present a more illustrative example to support the assertion made regarding the bounding primary gamma source term in the 1.0 to 1.66 MeV and 2.0 to 3.0 MeV ranges.

The UFSAR states: "Consider a set of burnup and enrichment combinations with cooling times greater than a certain value in an FQT for a given decay heat restriction. The bounding primary gamma radiation source occurs at the lowest enrichment, and the lowest burnup combination of that set." The applicant's example set selection on page W.5-7 uses a burnup enrichment combination at cooling times exceeding 10.0 years. However for the FQT referenced, W.2-8, this threshold would include all the fuel presented in that table. Since this comparison varies burnup, enrichment and cooling time, it is not immediately clear how this set selection follows the guidance in NUREG-1536 SRP for Dry Cask Storage which states: "the shielding source term ... should be based on the lowest enrichment (for a given burnup)."

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 5-18

The FQT presented in UFSAR Table W.2-8 provides the acceptable combinations of burnup, enrichment and cooling time such that the resulting decay heat is approximately the same (within 5 watts). As described in the bulleted list in UFSAR Section W.5-2, the primary gamma source term at longer cooling times is dominated by the radiological sources in the 1.00 to 1.66 MeV energy range. Therefore, in order to determine the bounding source terms for shielding for a given burnup, not only is it based on the lowest enrichment, but also on minimum cooling time. Therefore, the bounding source terms are determined using combinations of enrichment and cooling time to maximize the contribution from the primary gamma source terms since these overwhelmingly dominate the dose rate on and around the bare cask.

UFSAR Section W.5.2 is modified to include additional clarity in the description of the source term selection methodology.

5-19 Section W.5.4, Shielding Evaluation (page W.5-14). Provide a representative input file in Chapter W.5 for the model used in the shielding evaluation.

This information is needed for staff to properly assess the method used in the shielding evaluation.

Response to 5-19

Representative input files are included in revised UFSAR Chapter W.5, Section W.5.6 for both the source term and shielding evaluations.

5-20 Section W.5.4.6.1, Source Term Assumptions (page W.5-17, second bullet). Revise the SAR here, and in other locations as needed, to correctly reflect the current operations. Further, determine whether the assertion made is still valid, and if so, provide the appropriate justification.

This write-up contradicts the August 14th response to RAI #2, question 5-15. The RAI response indicates water no longer needs to be drained from neutron shield; however, the SAR text contradicts this. (See also RAIs 5-24, 5-25, 5-28, 8-1, and Q-29.)

Response to 5-20

All operations are performed with water in the neutron shield. The assumption (second bullet) in Section W.5.4.6.1 is provided to show that the dose rates are dominated by the primary gamma sources. The second bullet is revised to clarify the assumption and to delete the reference to "draining of the neutron shielding."

5-21 Section W.5.4.7, Summary of the Calculational MCNP Models (page W.5-18, first paragraph). Review and revise the next-to-last sentence of this paragraph to take into account the data in Tables W.5-8 and W.5-11. Further, review the last sentence and revise as needed.

The next-to-last sentence of this paragraph contains the phrase "all dose rate components" to justify the use of only the 32PT DSC analyses. However, examination of Tables W.5-8 and W.5-11 shows that dose rates for the 61BT exceed those for the 32PT near the top of the TC. The discussion should take this into account. In addition, in the last sentence of the first paragraph, the word "satiations" seems out of place. Perhaps "situations" was intended.

Response to 5-21

UFSAR Section W.5.4.7 is revised to include the discussion on dose rate comparisons of the 32PT and 61BT contents. The sentence containing Satiations is revised.

5-22 Section W.5.4.7, Summary of the Calculational MCNP Models (page W.5-18, second paragraph). Review the figures cited in this paragraph, determine what figures were intended to be cited, and revise this paragraph accordingly.

In roughly the middle of the paragraph Figures W.5-1 and W.5-2 are cited as showing the quarter symmetry of the MCNP model. It does not appear that Figure W.5-2 is the correct figure to reference here. In the last sentence of the second paragraph, an incorrect figure appears to be referenced. Figure W.5-1 shows the shielding configuration. Figure W.5-2 has plots of dose rates and appears to be the correct figure to cite here.

Response to 5-22

The cross-reference to the figures in UFSAR Section W.5.4.7 is updated appropriately.

5-23 Section W.5.4.7, Summary of the Calculational MCNP Models (page W.5-18, numbered list item number 2). Revise this paragraph as needed so that it correctly reflects operational conditions and distinguishes between modeling assumptions and actual conditions as appropriate.

This item states that the DSC is assumed to be dry when placed in decontamination area shield – then further states that this is the expected configuration. This contradicts operating procedures on page W.8-16 that indicate the DSC contains water when lifted from pool.

Response to 5-23

The DSC cavity is full of water during decontamination operations. The DSC cavity is dry following welding and sealing operations. However, for simplicity, dose calculations are performed only with the dry DSC configuration and conservatively applied to the wet (flooded) DSC configurations. The description in UFSAR Section W.5.4.7 is revised to reflect the correct operational configuration and the corresponding evaluated configuration (from a shielding standpoint).

5-24 Section W.5.4.7, Summary of the Calculational MCNP Models (page W.5-19, numbered list item number 6). Revise this paragraph as needed to provide a clear, error-free description of the configuration modeled.

This discussion contains several ambiguous, unclear, or incorrect sections. For example, the first sentence presumably indicates the shielding model has 2.5 inches of shielding above the top half (radially) of the TC and 5.5 inches of supplemental shielding in the sides of the transfer trailer; however, this is unclear. It seems that this configuration applies to the last row of Table W.5-3, but that is not actually stated. At the end of the third line the text "at the bottom of the cask" is ambiguous. If the intent is to indicate that the model did not consider any shielding located underneath the cask (in its horizontal orientation on the trailer), then it should be more clearly stated. There is a reference to filling the neutron shield with water, but the August 14th responses to the RAIs indicate the neutron shield will not be drained. Also, this paragraph cites Figure

W.5-7, but there is no such figure in the application. (See also RAIs 5-20, 5-25, 5-28, 5-30, 8-1, Q-29, and Q-33.)

Response to 5-24

UFSAR Section W.5.4.7 is revised to provide a clearer description of the modeled configuration. The text describing the "bottom of the cask" is revised to indicate that the calculational model did not consider any shielding beneath the cask in the horizontal orientation on the transfer trailer except for the 0.25" plate representing the trailer platform. The reference to the "filling" of the neutron shield is modified to indicate a water-filled neutron shield. This paragraph is also revised to reference the multi-part Figure W.5-5.

5-25 Section W.5.4.8.3, Removable Two Piece Neutron Shield Dose ... (page W.5-21). Revise the discussion in the last paragraph of this section, citing appropriate sources.

The last paragraph of this section contains a flawed/misleading discussion of the dose rate effects of the seam in the neutron shield. Dose rates on the bare cask surface are compared to values from Table W.5-12 which is characterized (in this paragraph) as having no water in the neutron shield. However, the heading of Table W.5-12 clearly states that it is for normal conditions with water in the neutron shield. It appears that the appropriate values for comparison can be found in the fourth configuration presented in Table W.5-4. (See also RAIs 5-20, 5-24, 5-28, 8-1, and Q-29.)

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 5-25

The discussion in the last paragraph of UFSAR Section W.5.4.8.3 is revised to provide the comparison from the results shown in UFSAR Table W.5-4.

5-26 Section W.5.4.9, Accident Models (page W.5-22). Review all cited tables for correctness of entries and correct any entries found to be in error, including those mentioned below. When citing sections in the SAR, provide citations to specific sub-sections or tables, not general citations to entire sections.

In the third bulleted item in the discussion of Table W.5-4, Tables W.5-7 and W.5-8 are cited. Several values in these three tables that appear to be identical are not equal. The discussion following the bulleted list cites Section M.5 as the source for the UFSAR values, but Table W.5-4 cites Table M.11-2.

Response to 5-26

The discussion in UFSAR Section W.5.4.9 is revised for correctness of the referenced citations.

The accident condition dose rates for the OS197 TC with the 32PT DSC are evaluated in UFSAR Section M.5 (described in Section M.5.4.5, numbered item #5) and the results are *reported in UFSAR Table M.5-3. These results are also reproduced in UFSAR Table M.11-2. Therefore, both references are appropriate.*

5-27 Section W.5.4.10.2, Cask Decontamination (page W.5-23). Revise this discussion to take into account the fact that the OS197L sits inside a supplemental shield structure and discuss the operational differences that will affect how decontamination tasks are accomplished. Also, justify the use of Table W.5-8 dose rates and the assertion that the use of the axial dose rates from Appendix M is conservative.

While the radiation fields outside the OS197L supplemental shielding may be comparable to or bounded by those associated with the OS197, the OS197L is not accessible for decontamination in the same manner as the OS197 since the OS197L sits inside the supplemental shielding. This discussion needs to take this into consideration. Additionally, it is not apparent that it is appropriate to use Table W.5-8 exclusively in this discussion when Table W.5-11 has much higher dose rates at the top of the TC. A justification also needs to be provided to support the assertion that the axial dose rates from Appendix M bound those expected from the configuration in Appendix W. **Note that this information was requested in RAI #2 as part of RAI 5.32, but was not satisfactorily addressed in the August 14th response.**

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 5-27

The discussion in UFSAR Section W.5.4.10.2 is expanded to include the operational differences between the OS197 TC and the OS197L TC within the supplemental shielding.

The discussion is also modified to provide additional justification for the applicability of the axial dose rates calculated with the OS197 TC from UFSAR Chapter M.5 and Chapter K.5 to the OS197L cask.

5-28 Section W.5.4.10.4, TC Placement in the Transfer Trailer (page W.5-24). Revise this section, and the rest of Appendix W, to correctly reflect the status of the neutron shield with respect to whether or not it contains water.

This write-up contradicts the August 14th response to RAI #2, question 5-15: SAR text indicates the neutron shield is empty, and later that the neutron shield is to be filled after the TC is placed on the trailer – However, the RAI response indicates water no longer needs to be drained from neutron shield. Determine which approach is correct and ensure it is correctly implemented throughout the SAR. (See also RAIs 5-20, 5-24, 5-25, 8-1, and Q-29.)

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 5-28

UFSAR Section W.5.4.10.4 and the entire UFSAR Appendix W are revised to be consistent with TN's approach previously stated in the RAI No. 2, Question 5.15 response that "the neutron shield is filled with water at all times."

5-29 Tables W.5-3 and W.5-4 (pages W.5-29 and W.5-30). For each configuration presented in these tables, provide the source of the data presented, similar to what is done in the first row.

This is needed to enable staff to identify and confirm the data presented.

Response to 5-29

The information presented in UFSAR Table W.5-3 and Table W.5-4 is modified to provide the source of the data for each of the entries as appropriate.

5-30 Table W.5-3 footnote 3, and Table W.5-4 footnote 4 (pages W.5-29 and W.5-30). Revise these footnotes to clarify what configuration is represented by the reported values. If the reported dose rates do not reflect those expected at distances out to the side of the transfer trailer, then state this clearly.

It is unclear if the values presented in the last row of Table W.5-3 (and the next-to-last row of Table W.5-4) are meant to represent dose rates out to the side of the 5.5 inches of shielding in the trailer sides, or above the 2.5 inches of shielding over the TC. (See also RAI 5-24.)

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 5-30

The footnotes in UFSAR Table W.5-3 and Table W.5-4 are revised to provide the required clarification.

5-31 Table W.5-4 (page W.5-30). Confirm the values presented in the last row of this table. Revise this table (and/or other tables) as necessary, or justify the different values presented.

Three of the four total dose rate values for the bare cask in the last configuration in this table do not agree with the dose rates in Table W.5-7 from which they presumably were taken.

Response to 5-31

The dose rates reported in UFSAR Table W.5-4 for the OS197L TC are the calculated maximum dose rates due to either a 61BT DSC or a 32PT DSC as appropriate. Also, please see the response to RAI 5-29, above.

5-32 Tables W.5-6, W.5-7, W.5-9, W.5-10, W.5-12, W.5-13, and W.5-14 (pages W.5-32, W.5-34, W.5-36, W.5-37, and W.5-38). Explain the method used to determine the relative error for the total dose rates (neutron plus gamma).

It is not apparent what method was used to determine these relative error values, and it does not appear that a standard error propagation method was used. **Note that this information was requested in RAI #2 as part of RAI 5.58, but was not satisfactorily addressed in the August 14th response.**

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 5-32

TN assumes that reference to "RAI 5.58" above is meant to be "RAI 5.48."

Section W.5.4.5 is modified to include the method (in a new item #6) employed to determine the relative errors associated with the total dose rates.

5-33 Calculation NUH06L-0503

Update the responses to questions in the March 27th RAI that indicated that calculation NUH06L-0503 had been revised in response to the RAI. **(See August 14th responses to RAI #2, questions 5.28, 5.29, and 5.31.)**

These questions relate to TN calculation NUH06L-0503. Each response began with "Calculation NUH06L-0503 is revised to provide guidance to the general licensee during off-normal conditions such as crane malfunction, by including a sample calculation." When staff requested a copy of the revised calculation, TN indicated that the 0503 calculation had only been revised to add a reference to another calculation, NUH06L-0506, which was later provided.

Response to 5-33

A new TN calculation NUH06L.0506 that superseded TN calculation NUH06L-0503 was prepared to document a simplified occupational exposure calculation after a discussion with the staff. Revised responses to RAI #2 questions that referenced calculation NUH06L-0503 are shown below.

Previous RAI 5.28

Justify assumption 4.5 used to determine worker doses for crane failure in calculation

NUH06L-0503.

Assumption 4.5 states that typical distances between the workers and the cask are assumed to be on the order of 10 meters, and that these distances are justified since this represents approximately twice the length of the cask. Explain why the length of the cask impacts the assumption of where the workers may be with respect to the cask.

This information is necessary to demonstrate compliance with 10 CFR 72.236(d).

Response to Previous RAI 5.28

TN calculation NUH06L.0506 was prepared to document a simplified occupational exposure calculation which supercedes calculation NUH06L-0503. Calculation NUH06L.0506 provides guidance to the general licensee during off-normal condition such as crane malfunction by including a sample calculation. This sample calculation can be used as a guidance to evaluate the occupational exposure to recover from off-normal crane malfunction event based on the site specific conditions.

Previous RAI 5.29

Revise Section 10 of the SAR to include the additional considerations for enhancing radiation protection that are listed in section 6.0 of calculation NUH06L-0503.

The third paragraph of Section 6.0 of calculation NUH06L-0503 states some additional considerations for enhancing radiation protection. This paragraph should be reflected in Section 10 of the SAR, to help ensure that plant personnel take all necessary precautions when using the high-dose rate OS197L TC.

This information is necessary to ensure compliance with 10 CFR 72.236(d).

Response to Previous RAI 5.29

TN calculation NUH06L.0506 was prepared to document a simplified occupational exposure calculation which supercedes calculation NUH06L-0503. Calculation NUH06L.0506 provides guidance to the general licensee during off-normal condition such as crane malfunction by including a sample calculation. This sample calculation can be used as a guidance to evaluate the occupational exposure to recover from off-normal crane malfunction event based on the site specific conditions. Chapter W.10 is revised to include this change and the additional considerations for enhancing radiation protection are included as necessary.

Previous RAI 5.31

In Calculation NUH061-0503, clarify/revise the tables and discussions on pages 17 - 21 relating to scenarios 1 and 2.

The discussions indicate that scenario 2 results in higher doses than scenario 1, but the doses shown in the tables and text do not support this statement (specifically, 970 for scenario 1 is greater than the 956 given for scenario 2). Additionally, provide further explanation for how the backscatter correction factors were chosen. It is not inherently clear that the selected backscatter factors are appropriate. Further, clarify the discussion to expand on the assumptions used for calculating worker doses presented in Tables 5-4, 5-5, and 5-6. It is not clear what assumptions were made regarding the location of the workers, particularly for the steps involving traversing to the crane bridge and to the crane. Provide a fuller explanation of the assumptions used to determine what dose rates the workers were exposed to and for what periods of time.

This information is necessary to ensure compliance with 10 CFR 72.236(d).

Response to Previous RAI 5.31

TN calculation NUH06L.0506 was prepared to document a simplified occupational exposure calculation which supercedes calculation NUH06L-0503. Calculation NUH06L.0506 provides guidance to the general licensee during off-normal condition such as crane malfunction by including a sample calculation. This sample calculation can be used as a guidance to evaluate the occupational exposure to recover from off-normal crane malfunction event based on the site specific conditions.

CHAPTER 8 Radiation Protection Evaluation

8-1 Section W.8.1.1, Preparation of the TC and DSC, Step 18.b (page W.8-14). Revise this step to be consistent with the commitment made in the August 14^{th} response to RAI 5.15.

In the response to RAI 5.15, the applicant stated that the neutron shield would contain water at all times. Step 18.b appears to contradict or put qualifications on this commitment. (See also RAIs 5-20, 5-24, 5-25, 5-28 and Q-29.)

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 8-1

Step 18.b of UFSAR Section W.8.1.1 has been revised to remove qualifications relating to meeting crane capacity limits. The step now states "Fill the TC liquid neutron shield."

8-2 Section W.8.1.2, DSC Fuel Loading (page W.8-14). Revise the note at the beginning of this section to make it clear that both remote crane operations and remote targeting are required with the OS197L system.

The note at the beginning of this section contains the phrase "**or** other mitigating ALARA practices" [emphasis added]. This SAR needs to be clear that both remote crane operations and remote targeting are needed with the OS197L system. **Note that this should have been corrected as part of the August 14^{th} response to question 5.43 in the March 27th RAI.**

Response to 8-2

The wording in the note at the beginning of Section W.8.1.2 in the UFSAR, page W.8-14 is changed to make it clear that both remote crane operations and remote targeting are needed with the OS197L system (the word "or" is changed to "and").

8-3 Section W.8.1.3, DSC Drying and Backfilling (page W.8-17). Revise step 4 to include a statement to indicate that the entire DSC length must be decontaminated if the top 12" is found to be contaminated above the stated levels.

This revision is needed to ensure there is no confusion with respect to the extent of the DSC surface that needs to be decontaminated in the event contamination levels are found to be above the specified levels.

Response to 8-3

Step 4 is revised to indicate that actions in accordance with Technical Specification 5.2.4 d potentially involve decontamination of the entire length of the DSC outer surface.

8-4 Section W.8.4.1.1, Functional Description (page W.8-26). Expand the description of the OS197L transfer system to include the supplemental shielding.

The description given in the first paragraph of this section does not make it apparent that the OS197L TC includes the supplemental shielding for the decontamination area and the transfer trailer as part of the transfer system.

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 8-4

The description of the OS197L TC provided in UFSAR Section W.8.4.1.1 is revised to reference Sections W.1.2 and W.1.2.1.1, which provide detailed descriptions of the OS197L, including the supplemental shielding for the decontamination area and the transfer trailer as part of the transfer system.

CHAPTER 9 Acceptance Criteria and Maintenance Program

9-1 Section W.9, Acceptance Criteria and Maintenance Program (page W.9-1). Specify the locations in the UFSAR where the acceptance criteria and maintenance requirements pertaining to the OS197L are found.

The text in this section is similar to the "No change" statements in other sections. If this section is referring to other parts of the UFSAR, it should indicate the specific sections that are being cited.

Response to 9-1

SAR Chapter W.9 is revised to indicate the acceptance criteria and maintenance program requirements for the OS197L TC.

CHAPTER 10 Radiation Protection

10-1 Section W.10.2, Inspection of Decontamination Shield Openings... (page W.10-4). Review, and revise as necessary, the dose rates and dose estimates in this section. Justify the chosen dose rates. Also, justify the assertion that the dose rate at the top of the OS197L TC (outside the decontamination area shield) drops below 100 mrem/hour for distances greater than 10 cm. Specify where in Section W.5 each of these results is located. Provide the supporting calculation document if needed.

An examination of Tables W.5-8 and W.5-11 indicates that (for short distances) dose rates increase as distance from the side of the TC increases, calling into question whether the use of the bare cask surface dose rates is conservative. Additionally, the dose rates at the level of the top openings appear to significantly exceed the chosen value, especially for the 61BT DSC. Further, it is not clear that Section W.5 supports the assertion that the dose rate at a distance of 10 cm outside the decontamination area shielding at the level of the top openings is below 100 mrem/hour. There is no indication that the model of the decontamination area shielding was sufficiently detailed to include the top and bottom openings. Further, this paragraph was not revised from the previous submittal, even though the projected dose rates changed significantly due to the limitations of the OS197L contents.

This information is requested to satisfy the requirements of 10 CFR 72.236.

Response to 10-1

The decontamination area shield is 6" thick and the dose rate distribution on and around the shield is conservatively bounded by the shielding analysis performed with the trailer shielding configuration. UFSAR Section W.10.2 is revised to include the necessary justification for the dose rates employed in the occupational exposure calculation associated with the inspection of the decontamination shield openings for blockage. Further, UFSAR Section W.5.4.10.2 is revised to include the dose rate calculation employed for this configuration.

The following RAIs reflect staff Quality Assurance concerns and questions.

In the March 27th RAI, in addition to the technical issues, staff identified numerous quality-related problems. In the August 14th response to RAI #2, question 5.55, TN indicated "[t]he revised Chapter W.5 is reviewed for elimination of typos and editorial inconsistencies." Despite this, staff noted a large number of editorial problems in the current submittal; quality-related RAIs are included here. Collectively, these quality-related problems are of concern to the staff. TN is requested to document these quality-related problems in their corrective action system, identify the cause of these conditions, and identify the corrective action(s) taken to prevent repetition. In addition, TN should justify that the Quality Assurance program satisfies the requirements of 10 CFR 72.152 for Document Control, and 10 CFR 72.146 for Design Control.

Q-1 Document Quality. The requirements in 10 CFR 72.146(b) state that the applicant shall establish measures for the identification and control of design interfaces and for coordination among participating design organizations. These measures must include the establishment of written procedures among participating design organizations for the review, approval, release, distribution, and revision, of documents involving design interfaces. The requirements in 10 CFR 72.146(b) further state that, for the verifying or checking process, the licensee shall designate individuals or groups other than those who were responsible for the original design, but who may be from the same organization. The applicant shall apply design control measures to items such as the following: criticality physics, radiation, shielding, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for in-service inspection, maintenance, and repair; features to facilitate decontamination; and delineation of acceptance criteria for inspections and tests. In addition, the requirements in 10 CFR 72.152 state that the applicant shall establish measures to control the issuance of documents such as instructions, procedures, and drawings, including changes, which prescribe all activities affecting quality. These measures must assure that documents, including changes, are reviewed for adequacy, and are approved for release by authorized personnel.

The applicant should demonstrate that appropriate design control measures have been established and that all values and analyses associated with the thermal and containment design (not limited to these RAIs) are accurate and reliable.

This information is required by the staff to determine compliance with 10 CFR 72.146 and 72.152.

Response to Q-1

The quality related issues raised by the staff represent a violation of TN Quality Assurance Program Implementing Procedure (TIP 5.4, Revision 5), "Control of Licensing Documents." Section 3.6 of TIP 5.4 assigns and describes the responsibility for control of licensing documents to every level of the TN organization.

TN has opened corrective action report (CAR) to address the quality related questions documented in the February 19, 2010 letter from the staff. Each of these quality related questions are being addressed and resolved in accordance with the process of TIP 16.1, "Corrective Action." As part of the CAR resolution process TN actions include a determination of the apparent cause, identification of corrective actions and identification of actions to prevent recurrence.

Specifically, as part of the CAR resolution TN has taken or identified the following immediate corrective actions and actions to prevent recurrence:

1. *Quality verification checklists, with detailed acceptance criteria and signature blocks, were developed and put into use for all regulatory submittals. While not yet formal, these records are being entered into TN's Records Management System. Final resolution of this CAR will involve formal incorporation into TN procedures.*

2. *A classroom training session was conducted by the VP Engineering and the Licensing Manager to Transnuclear, Inc. personnel. This training session was held on 3/17/10 and emphasized the following:*

   i. *The history of Amendment 11 quality issues and concerns by the staff, particularly in the area of design control and accuracy of licensing documents*
   ii. *The NRC staff expectations of licensing submittal documents*
   iii. *Lessons learned focusing on accuracy and clarity of design and licensing documents and attention to detail*

3. *An organizational meeting was held by the VP Engineering with Transnuclear, Inc. Engineering personnel. This meeting occurred on 3/26/10 and emphasized feedback from engineering personnel to identify weak links and areas of improvement in the*

process of document preparation and review, editorial verification and document production.

4. *A new TN position was created, with flexible roles, one of which is to perform as a Technical Editor for regulatory submittals.*

5. *An independent consultant performed an assessment of TN's control of licensing documents and provided a report.*

6. *TN's design control process is also being reviewed and assessed for effectiveness.*

7. *Closeout activities for this CAR involve assessing lessons learned from the ongoing quality verifications, recommendations from the assessments described above, and identification of additional necessary training, procedure changes, etc.*

Q-2 TS Table of Contents (pages iv, v and vi). With the replacement of pages v and vi, according to RAI #2 responses, pages iv and v no longer align. (Page iv ends with Table 1-3f; page v begins with Table 1-5d.

Provide up-to-date Table of Contents pages for the TS.

Response to Q-2

The TS table of contents has been reconciled. The entire table of contents is included herein.

Q-3 TS Section 4.3 Storage Location Design Features (page 4-25). Existing page 4-25, from the responses to RAI #1, was not replaced with the responses to RAI #2. This page appears to be out of place, as it is within the ASME Code Alternatives section (pages 4-6 to 4-29). In addition, this page appears to be replaced by RAI #2 changed page 4-30. (See also RAIs 1-1, 1-2, 2-1, 5-11, Q-17, Q-35, and Q-38.)

Provide actual replacement page for 4-25.

Response to Q-3

The actual replacement page for 4-25 is included herein, containing ASME Code Alternatives for the NUHOMS®-61BTH DSC Confinement Boundary (Concluded).

Q-4 TS 5.2.4a Radiation Protection Program (page TS 5-5). In item number 2 in the third paragraph, delete the word "rate."

Response to Q-4

The word "rate" has been deleted as directed.

Q-5 TS 5.2.4d Radiation Protection Program – contamination limits (page TS 5-7). In the second paragraph, insert a period between "TC" and "Decon as necessary." It appears that punctuation is missing.

Response to Q-5

The two sentences involved are revised as follows:

Before: "If contamination levels are still not met, remove the fuel assemblies from the DSC and put them back in the fuel pool and remove the DSC from the TC Decon as necessary."

After: "If contamination levels are still not met, remove the fuel assemblies from the DSC and put them back in the fuel pool, remove the DSC from the TC and decontaminate as necessary."

Q-6 TS, Table 1-1l PWR Fuel Specification (page TS T-13). Numbers in second column do not align with criteria in first column.

Response to Q-6

The numbers are shifted left and aligned properly.

Q-7 TS, Table1-6d (page TS T-80). Revise the footnote to this table to refer to the notes on the preceding page in the Technical Specifications, instead of the page in the UFSAR.

Response to Q-7

The note is revised to refer to Table 1-6c.

Q-8 Section 4.7.3.3 Transport Cask Lifting Yoke, and text just preceding (page 4.7-7). Determine whether the use of the term "transport cask" in the heading of this section, at the end of the preceding paragraph, and elsewhere in the UFSAR is as intended.

There are specifications in the UFSAR for the transfer cask and associated equipment. If the term "transport cask" is used interchangeably with "transfer cask," the UFSAR should make that clear.

Response to Q-8

The term "transport cask" has been replaced with "transfer cask" throughout the UFSAR, as appropriate (Pages vi, xx, 4.7-7, 4.7-15, 8.2-26, M.2-5, and P.2-10). Additionally, the term "transport trailer" and "transport" were assessed in the UFSAR. For pages included herein, those terms are changed to "transfer trailer" and "transfer" as appropriate and for the remainder of the UFSAR those terms will be assessed and changed as appropriate through the provisions of the 10 CFR 72.48 process.

Q-9 Chapter 7, Radiation Protection (page 7.1-1). Revise the wording in the first sentence on this page so that it makes sense. It appears that the words "and transfer in the standardized cask, OS197 or OS197H TCs" got inserted after "-52B" when they should have been inserted after "Model 102."

Response to Q-9

The wording is revised accordingly.

Q-10 Section W.1.2.2, Operational Features (page W.1-3). At the end of the second paragraph in Section W.1.2.2, Technical Specification "5.3.2.4.a" should be "5.2.4.a."

Response to Q-10

The change is made as directed.

Q-11 Section W.1.4, Generic Cask Arrays (page W.1-4). Revise this section to clarify what is not being changed. Cite the specific section to which this refers.

Response to Q-11

Please see the response to RAI 5-12.

Q-12 Table W.1-1, Comparison of Key Parameters... (page W.1-6). The "OS197L TC" column of the "Top Cover Assembly" row has a reference to the use of the interim aluminum cover. Based on information in the RAI response, it appears this note should be deleted.

Response to Q-12

The reference to the interim aluminum cover is deleted.

Q-13 Table W.1-2, OS197L TC UFSAR Sections Affected (page W.1-7). In the last column of the last row delete "needs to be."

Response to Q-13

The entries in row 31 are revised to indicate that the yoke shown in Figure 4.2-15a is an alternate lifting yoke design and does not necessarily need to be tied to the OS197L TC.

Q-14 Figure W.1-1, OS197L Configuration (page W.1-9). Is "LIGHT" or "LIQUID" the correct word in the label on the neutron shield assembly?

Response to Q-14

The term "LIGHT" is used to describe the neutron shield assembly associated with the lightweight OS197 (OS197L) cask.

Q-15 Section W.2.1, Spent Fuel To Be Stored (page W.2-1). Near the end of the second line in the first paragraph of Section W.2.1, delete the word "are."

Response to Q-15

The word "are" is deleted as directed.

Q-16 Notes for Tables W.2-7 and W.2-8 (page W.2-12). Correct the fifth bulleted item as needed.

The enrichment limitations state that "fuel with an initial enrichment less than 21.01 and greater than 5.0 wt % U-235 is unacceptable for storage." The first number in the range limitation seems to be a typo since 21.01 is already greater than 5.0.

Response to Q-16

The fifth bulleted item on page W.2-12 is corrected as follows: "Fuel with an initial enrichment less than 1.1 and greater than 5.0 wt.% U-235 is unacceptable for storage." This same change is also made to the fifth bullet of the notes for Technical Specifications Table 1-6c on Page T-79.

Q-17 Pages W.3-6 to W.3-12. The List of Changed CoC, Technical Specifications and UFSAR Pages Associated with Amendment 11, Revision 2 for RAI #2 does not specify that these pages should be removed with the change; however, page W.3-6 appears to be extraneous given changed page W.3-5.

Clarify which of these pages are integral to Chapter W.3, and provide new/revised pages as appropriate. In addition, provide revised list of changed pages indicating deletions as appropriate. (See also RAIs 1-1, 1-2, 2-1, 5-11, Q-3, Q-35, and Q-38.)

Response to Q-17

With the changes to Chapter W.3 for Amendment 11, Revision 2 for RAI #2, the information which provides the reference list for Chapter W.3 on page W.3-6 (December 2007, Revision 1) now appears on page W.3-5 (August 2009, Revision 2). As the RAI indicates, the submittal should have provided direction as to what action to take regarding Pages W.3-6 and beyond.

As it turns out, information added to Chapter W.3 for the current submittal causes the reference list for Chapter W.3 to again appear on Page W.3-6. The new Page W.3-6 is enclosed herein with the set of changed UFSAR pages. A revised list of changed pages is also included.

Q-18 Section W.5.1.2, Bounding Dose Rates as a Function of Distance (page W.5-3). The second paragraph states three shielding configurations are presented in the plot. However, the first item contains two shielding configurations. Revise this paragraph to clearly describe the number of shielding configurations considered.

Response to Q-18

A total of four different shielding configurations are evaluated although five different plots are shown in Figure W.5-2. For one of the shielding configurations, dose rates are calculated at two different locations. Section W.5.1.2 is revised to provide a clear description of these configurations and the dose rate results. Please see the response to RAI 5-17.

Q-19 Section W.5.2.1, Methodology for Determination of Bounding ... (page W.5-9), and Section W.5.2.3, Bounding Radiological Sources... (page W.5-13). It appears that Tables K.2-11 and M.2-5 ["M.2.5" in SAR text] have been duplicated in Appendix W as Tables W.2-4 and W.2-7, respectively. Given that, it would seem that it would be better to refer to the version of each table that has been specifically modified for Appendix W instead of referring back to the original tables.

Response to Q-19

The relevant portions of SAR Sections W.5.2.1 and W.5.2.3 are revised to cross-reference the tables included in Chapter W.2.

Q-20 Section W.5.2.2.1, Bounding Radiological Sources for FAs in ... (page W.5-10). The words "...on the surface of the cask containing a 32PT DSC at NC" appear a little beyond halfway down the paragraph. Define "NC." This term is not in the list of abbreviations.

Response to Q-20

The language in Section W.5.2.2.1 is modified to clarify the description of the bounding radiological source terms for FAs in the 32PT DSC.

Q-21 Section W.5.2.2.1, Bounding Radiological Sources for FAs in ...(page W.5-11), and Section W.5.2.2.2, Bounding Radiological Sources for FAs in ...(page W.5-12). On these pages there are five instances of the U content of 32PT design basis fuel assembly being specified as 0.475 **kg** instead of 0.475 **MT** per fuel assembly. Correct these and ensure there are no other instances of this (or any similar) error.

Response to Q-21

The noted discrepancy is corrected to specify an initial loading of 0.475 MTU.

Q-22 Section W.5.2.4, Spectral Distributions of Neutron Source Terms ... (page W-5-13). The reference number cited [5.20] is incorrect. It appears that "5.20" should be "5.2." Correct this. In addition, provide the page number and/or section number in this reference where this information is found.

Response to Q-22

The noted typo discrepancy in UFSAR Section W.5.2.4 is corrected to cite the correct reference [5.2]. This information is provided in Volume I, Appendix H, Page H-3 under Item 2, Spontaneous Fission.

Q-23 Section W.5.4.2, Spatial Source Distribution (page W.5-15). Revise the section number for the referenced section. The section cited, W.5.2.4, is incorrect. Section W.5.2.4 deals with the neutron spectrum. The correct section to cite appears to be W.5.2.5 which does deal with axial peaking.

Response to Q-23

UFSAR Section W.5.4.2 is corrected to cite reference to Section W.5.2.5, which relates to axial peaking.

Q-24 Section W.5.4.6.1 Source Term Assumptions, first bullet (page W.5-17). Revise the section number for the referenced section. The section cited, W.5.2.3, is incorrect. Section W.5.2.3 deals with bounding sources. The correct section to cite appears to be W.5.2.4 which does deal with the neutron spectrum.

Response to Q-24

UFSAR Section W.5.4.6.1 is corrected to cite reference to Section W.5.2.4.

Q-25 Page W.5-20.

In the first paragraph, "on an around" should be "on **and** around"

In the third paragraph, "and he other" should be "and **the** other"

In the fifth paragraph, "shells is modeled" should be "shells **are** modeled." Also, "neuron shield" should be "neu**t**ron shield."

Reword the second and last sentences in the last paragraph on this page to be clearer. This may be as simple as moving "bounding" from behind to in front of "results," but the current wording in this paragraph is unclear.

Response to Q-25

Section W.5.4.8.1 has been revised as recommended in the RAI; "bounding" is moved from behind to in front of "results."

Q-26 Section W.5.4.10.3, Welding Operations (page W.5-23). Revise this paragraph to correctly reflect expected conditions. If any conditions are assumed for ease of analysis, be sure to distinguish between what is expected and what is assumed.

In this paragraph, the SAR states that the DSC is dry – this is inconsistent with discussions of operations in Chapter W.8. This may be meant to indicate that the DSC is **assumed** to be dry for dose rate calculations, or it may be a carry-over from an earlier version. If the DSC is dry at this point in the operation, then the operations section needs to be revised.

Response to Q-26

UFSAR section W.5.4.10.3 is revised to state that the DSC cavity is assumed to be dry. Please see the response to RAI 5-23 above.

Q-27 Section W.5.4.10.2, Cask Decontamination (page W.5-23). In the last sentence of this paragraph, the word "is" is used twice. In both instances, the correct verb would be "**are**."

Response to Q-27

The two instances of "is" are changed to "are."

Q-28 Section W.5.4.10.4, TC Placement in the Transfer Trailer (page W.5-24). The first paragraph of this section cites Figure W.5-7. There is no such figure in Section W.5. Correct this here and anywhere else it occurs.

Response to Q-28

Chapter W.5 is revised to ensure that the cross-reference information is accurate and appropriate. Specifically Section W.5.4.10.4 is revised to cite Figure W.5-5.

Q-29 Section W.5.4.10.5, Cask Transfer to ISFSI Operations (page W.5-25). Review the citations to various tables and sections. Revise as necessary.

The citation to Table W.5-9, appears to be incorrect. Table W.5-9 has dose rates for the bare OS197L TC. The correct table to cite appears to be W.5-4 which has a set of entries for a shielded OS197L without water in the neutron shield. (See also RAIs 5-20, 5-24, 5-25, 5-28, and 8-1.)

Response to Q-29

Chapter W.5 is revised to ensure that the cross-reference information is accurate and appropriate. Specifically for Section W.5.4.10.5, the correct citations are Table W.5-4, Table W.5-6 and Table W.5-9 instead of Table W.5-9, Table W.5-11 and Table W.5-14, respectively.

Q-30 Tables in Appendix W. The tables in Appendix W that present dose rates by component (i.e., gamma and neutron) plus total should have the same footnote that similar tables in other appendices have; specifically, "Gamma and Neutron dose rate peaks do not always occur at same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate."

Response to Q-30

The various tables in Appendix W that present dose rates by component are revised to include the note regarding the location of gamma and neutron dose rate peaks as appropriate.

Q-31 Table W.5-3, Summary On OS197L TC Normal... (pages W.5-29 and W.5-30). The title of this table should probably read, "Summary **Of** OS197L...."

The heading in the 4.57 meters column is missing a closing parenthesis.

The text "Table W.5-14" that appears three times in the footnotes is in a different font (and size) than the rest of the footnote text.

In footnote 2 to this table, it seems that "OS197L" needs to be inserted between "bare" and "cask" to make this a clear statement. Revise this accordingly.

Response to Q-31

In the titles of both Table W.5-3 and W.5-4, "on" is changed to "of." The closing parenthesis is added to the 4.57 meters column. The font style and size of "Table W.5-14" in the footnotes is changed to be consistent. In Table W.5-4 footnote 2, "OS197L" is inserted between "bare" and "cask." Various additional editorial and formatting changes are also made to these two pages.

Q-32 Tables W.5-3 through W.5-14 (pages W.5-29 through W.5-38). In these tables dose rates are presented with anywhere from two to seven significant figures. The large number of significant digits frequently used gives the impression that there is a level of certainty in the modeling results that is not warranted.

Response to Q-32

The various tables in Appendix W that present dose rates are revised to ensure that the dose rates are shown with an appropriate number of significant digits.

Q-33 Table W.5-13, OS197L Radial Dose Rates with...(page W.5-37). The note following this table refers to Figure W.5-7. This figure is not included in the document. Correct this note by either including the figure, or citing the correct figure. (See also RAI 5-24.)

Response to Q-33

The correct figure is the multi-part Figure W.5-5. The note in Table W.5-13 is revised to indicate the correct figure.

Q-34 W.8.1.1, Preparation of the TC and DSC, Step 11 (page W.8-14). Revise this step to make its meaning clear.

This step contains the following text: "water from the fuel pool and equivalent source or demineralized water." It seems that this may have been meant to read as: "water from the fuel pool, an equivalent source, or demineralized water."

Response to Q-34

Step 11 is changed as suggested.

Q-35 Page W.8-29. The List of Changed CoC, Technical Specifications and UFSAR Pages Associated with Amendment 11, Revision 2 for RAI #2 does not specify that this page should be removed with the change; however, page W.8-29 appears to be extraneous given changed page W.8-28. (See also RAIs 1-1, 1-2, 2-1, 5-11, Q-3, Q-17, and Q-38.)

Provide revised list of changed pages indicating deletions as appropriate.

Response to Q-35

Enclosure 5, "CoC, Technical Specifications, and UFSAR Page Change Instructions," directs that UFSAR Page W.8-29 (December 2007, Revision 1) be removed from the Amendment 11 pages.

Q-36 Section W.10.1, Recovery/Repair Operations ...(page W.10-2). Revise step #5 to correctly reflect the expected steps.

Step 5 specifies that the worker stays on the cask top while it is lowered and tilted on its side. In addition to being very questionable with respect to safety, this is inconsistent with #5 on the next page that indicates that the distance between the worker and the cask is increasing at this point.

Response to Q-36

The steps on Page W.10-2 are revised to address this item.

Q-37 Section W.11.1.4, Loss of Neutron Shield (page W.11-2). Correct the description of the middle configuration in the provided table.

As submitted, the description of the second configuration in the table yields the same geometry as the bare cask in the third configuration listed. By referring back to Table W.5-4, it appears that the second configuration was meant to be with the inner supplemental shield, but without the outer shield.

Response to Q-37

The description of the second configuration provided in the Table in Section W.11.1.4 is revised to indicate "with supplemental inner and without outer shielding."

Q-38 Pages W.11-3 to W.11-4. The List of Changed CoC, Technical Specifications and UFSAR Pages Associated with Amendment 11, Revision 2 for RAI #2 does not specify that page W.11-4 should be removed with the change; however, the transition between new page W.11-3 and RAI #1 page W.11-4 appears to indicate that something is missing here, or that page W.11-4 is no longer necessary.

Provide new/revised pages as appropriate. In addition, provide revised list of changed pages indicating deletions as appropriate. (See also RAIs 1-1, 1-2, 2-1, 5-11, Q-3, Q-17, and Q-35.) +

Response to Q-38

Enclosure 5, "CoC, Technical Specifications, and UFSAR Page Change Instructions," directs that UFSAR Page W.11-4 (December 2007, Revision 1) be removed from the Amendment 11 pages.

List of Additional Changes Not Associated with the RAI

Item Number	Discussion	Areas Affected
1	For consistency with Transnuclear's CoC 1030 Amendment 1 and CoC 9302 Revision 3 ongoing licensing actions, changes are made regarding neutron absorber tests.	UFSAR Chapters K.9, M.9, P.9, T.9 and U.9
2	Quality technical and editorial reviews of the Technical Specifications and the UFSAR revealed instances where changes are warranted involving spelling, verb tense, punctuation, consistency, etc.	Various, as indicated on Enclosure 4
3	For consistent treatment of reconstituted fuel assemblies with control components, the second bulleted note in TS Tables 1-2d through 1-2h is added to additional TS tables associated with the 32PT DSC.	TS Tables 1-2i through 1-2m.
4	The value in UFSAR Table K.9-3 for B10 content is changed from 0.039 g/cm^2 to 0.038 g/cm^2 to be consistent with TS Tables 1-1v and 1-1w.	UFSAR Table K.9-3
5	Review of the ASME Code Exceptions Tables resulted in the need to make certain changes, as follows: • Updated references to UFSAR sections in NCA-1140 lines • Added or modified lines regarding NDE personnel • Changed NG to NG/NF for certain lines • Changed certain NG-3352 lines for welding requirements • Added UFSAR Table 4.9-1 for the OS200 and OS200FC transfer casks • Added NC-1140 line to the Tech Spec table for the OS200 and OS200FC transfer casks • Added mention of ASME Code requirements in UFSAR Chapter U.2 design criteria subsections.	TS Pages 4-8, 4-9, 4-11, 4-12, 4-14, 4-15, 4-17, 4-19, 4-21, 4-23, 4-25, 4-26, 4-27, 4-28 UFSAR Tables 4.8-1, 4.8-2, 4.9-1, K.3.1-2, K.3.1-3, M.3.1-1, M.3.1-2, P.3.1-1, P.3.1-2, T.3.1-2,T.3.1-3, U.3.1-1, U.3.1-2, U.3.1-3 UFSAR Pages U.2-7, U.2-8, and U.2-12
6	Technical Specifications Table 1-1dd has two instances where "WE" is corrected to "CE"	Technical Specifications Table 1-1dd

List of CoC, Technical Specifications, and UFSAR Pages Associated with Amendment 11, Revision 3

Page number	Reason for change
Certificate of Compliance	**Certificate of Compliance**
CoC page 1	5-1
CoC page 2	5-1
CoC page 3	No change, included for continuity
CoC Inserts	5-1
Technical Specifications	**Technical Specifications**
cover page	No change, included for continuity
TS i	Q-2
TS ii	Q-2, text shift
TS iii	Q-2, Additional Item 2
TS iv	Q-2
TS v	Q-2
TS vi	Q-2
1-1	No change, included for continuity
1-2	5-3, Additional Item 2
1-3	No change, included for continuity
1-4	No change, included for continuity
1-5	No change, included for continuity
1-6	No change, included for continuity
1-7	No change, included for continuity
1-8	No change, included for continuity
1-9	No change, included for continuity
1-10	No change, included for continuity
2-1	5-2
2-2	No change, included for continuity
3-1	No change, included for continuity
3-2	No change, included for continuity
3-3	No change, included for continuity
3-4	No change, included for continuity
3-5	No change, included for continuity
3-6	No change, included for continuity
3-7	No change, included for continuity
3-8	Additional Item 2
3-9	Additional Item 2
3-10	No change, included for continuity
3-11	No change, included for continuity
3-12	No change, included for continuity
4-1	No change, included for continuity
4-2	Additional Item 2
4-3	Additional Item 2
4-4	No change, included for continuity
4-5	No change, included for continuity
4-6	Additional Item 2
4-7	No change, included for continuity
4-8	Additional Item 2, Additional Item 5
4-9	Additional Item 2, Additional Item 5
4-10	Additional Item 2
4-11	Additional Item 2, Additional Item 5
4-12	Additional Item 2
4-13	Additional Item 2
4-14	Additional Item 5
4-15	Additional Item 2, Additional Item 5

List of CoC, Technical Specifications, and UFSAR Pages Associated with Amendment 11, Revision 3

Page number	Reason for change
4-16	Additional Item 2
4-17	Additional Item 5
4-18	Additional Item 2
4-19	Additional Item 5
4-20	Additional Item 2
4-21	Additional Item 5
4-22	Additional Item 2, Additional Item 5
4-23	Additional Item 5
4-24	Additional Item 2
4-25	Additional Item 5
4-26	Additional Item 2, Additional Item 5
4-27	Additional Item 2, Additional Item 5
4-28	Additional Item 5
4-29	Additional Item 2, Additional Item 5
4-30	Additional Item 2, Additional Item 5
4-31	Additional Item 2
4-32	shifted text
4-33	1-1, shifted text
4-34	Additional Item 2, shifted text
5-1	Additional Item 2
5-2	Additional Item 2
5-3	Additional Item 2
5-4	Additional Item 2
5-5	Q-4
5-6	No change, included for continuity
5-7	5-2, 5-5, Q-5
5-8	5-6
5-9	No change, included for continuity
5-10	No change, included for continuity
5-11	No change, included for continuity
5-12	Additional Item 2
5-13	5-2, 5-6, 5-7
5-14	5-6, 5-7
5-15	1-2
5-16	1-2
T-1	Additional Item 2
T-2	Additional Item 2
T-3 to T-8	No change, included for continuity
T-9	Q-6
T-10 to T-12	No change, included for continuity
T-13	Q-6
T-14 to T-30	No change, included for continuity
T-31	Additional Item 6
T-32 to T-43	No change, included for continuity
T-44	Additional Item 3
T-45	Additional Item 3
T-46	Additional Item 3
T-47	Additional Item 3
T-48	Additional Item 3
T-49 to T-78	No change, included for continuity
T-79	Q-16
T-80	Q-7
F-1 to F-30	No change, included for continuity

List of CoC, Technical Specifications, and UFSAR Pages Associated with Amendment 11, Revision 3

Page number	Reason for change
UFSAR pages	**UFSAR pages**
vi (TOC)	Q-8, Additional Item 2
xx (TOC)	Q-8, Additional Item 2
1.2-3	UFSAR Rev. 11, Additional Item 2
1.2-11	UFSAR Rev. 11, Additional Item 2
1.3-3	5-8, UFSAR Rev. 11
1.3-4	UFSAR Rev. 11, Additional Item 2
1.3-7	UFSAR Rev. 11, Additional Item 2
1.3-10	UFSAR Rev. 11, Additional Item 2
3.1-4	5-9, UFSAR Rev. 11, Additional Item 2
3.3-35	UFSAR Rev. 10, Additional Item 2
4.2-9	UFSAR Rev. 11, Additional Item 2
4.2-10	UFSAR Rev. 11, Additional Item 2
4.2-26a	Q-13
4.7-7	Q-8, Q-13, Additional Item 2
4.7-10	UFSAR Rev. 11
4.7-15	Q-8
4.8-3	Additional Item 5
4.8-5	Additional Item 5
4.8-6	Additional Item 5
4.9-3	Additional Item 5
4.9-4	Additional Item 5
5.1-1	UFSAR Rev. 11, Additional Item 2
5.1-2	UFSAR Rev. 10
5.1-10	UFSAR Rev. 10, Additional Item 2
5.1-11	UFSAR Rev. 10, Additional Item 2
5.4-1	UFSAR Rev. 11
7.1-1	Q-9, UFSAR Rev. 11
8.2-26	2-8, Additional Item 2
8.2-44	UFSAR Rev. 11
10-3	Additional Item 2
10-4	Additional Item 2
10-5	Additional Item 2
10-7	Additional Item 2
10-8	Additional Item 2
10-9	Additional Item 2
10-9a	text shift
10-30	Additional Item 2
10-31	5-10, 5-6
10-32	Additional Item 2
10-33	Additional Item 2
10-34	shifted text
10-35	5-7
K.2-14	Additional Item 2
K.3.1-6	Additional Item 5
K.3.1-7	Additional Item 5
K.3.1-8	Additional Item 5
K.4-17	Additional Item 2
K.4-25	UFSAR Rev. 11, Additional Item 2
K.6-51	Additional Item 2

List of CoC, Technical Specifications, and UFSAR Pages Associated with Amendment 11, Revision 3

Page number	Reason for change
K.8-3	UFSAR Rev. 11
K.8-8	UFSAR Rev. 11, Additional Item 2
K.8-9	UFSAR Rev. 11, Additional Item 2
K.8-10	UFSAR Rev. 11, Additional Item 2
K.8-11	UFSAR Rev. 10
K.9-2	Additional Item 1
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enters near the bottom of the HSM, circulates and rises around the DSC and exits through shielded openings near the top of the HSM. The cross-sectional areas of the air inlet and outlet openings, and the interior flow paths are designed to optimize ventilation air flow in the HSM for decay heat removal including worst case extreme summer ambient conditions. The thermal performance features of the NUHOMS® system are described in Chapters 4 and 8.

External Atmosphere Criteria: Given the corrosion resistant properties of materials and the coatings used for construction of the NUHOMS® system components, and the warm, dry environment which exists within the HSM, no limits on the range of acceptable external atmospheric conditions are required. All components are either stainless steel, are coated with inorganic coatings, or are galvanized. Hence, all metallic materials are protected against corrosion. The interior of the HSM is a concrete surface and is void of any substance which would be conducive to the growth of any organic or vegetative matter. The design of the HSM also provides for drainage of ambient moisture which further eliminates any need for external atmospheric limitation.

The ambient temperatures selected for the design of the NUHOMS® system range from -40°F to 125°F, with a lifetime average ambient temperature of 70°F. The extreme ambient temperatures of -40°F and 125°F are expected to last for a short period of time, i.e., on the order of hours. The minimum and maximum average ambient temperatures of 0°F and 100°F are expected to last for longer periods of time, i.e., on the order of days.

1.2.3 Operating and Fuel Handling Systems

Some handling equipment and support systems within the plant needed to implement the NUHOMS® system are covered by the licensee's 10CFR50 operating license. The on-site transfer cask is designed to satisfy a range of plant specific conditions and requirements. The general operations for a typical NUHOMS® system installation are summarized in Table 1.2-3. A more detailed procedure for this sequence of operations is provided in Section 5.1, Appendices K, L, M, N, P, T, U *and W*. The majority of the fuel handling operations involving the DSC and transfer cask (i.e. fuel loading, draining and drying, *transfer* trailer loading etc.) utilize procedures similar to those already in place at reactor sites for SFA shipment. The remaining operations (canister sealing, cask-HSM alignment and DSC transfer) are unique to the NUHOMS® system.

1.2.4 Safety Features

The principal safety features of a NUHOMS® ISFSI include the high integrity containment for the confinement of spent fuel materials, the axial shielding provided by the DSC, and the extensive biological shielding and protection against extreme natural phenomena provided by the massive reinforced concrete HSM. The shielding materials incorporated into the DSC and HSM designs reduce the gamma and neutron flux emanating from the SFAs so that the dose rate at the ISFSI fence is within 10CFR72 limits and is ALARA. The radiological safety features of the NUHOMS® system are described in Chapters 3 and 7.

The DSC and HSM are designed and constructed in accordance with industry accepted codes and practices for important to safety systems under an approved Quality Assurance program as

Table 1.2-3
NUHOMS® System Operations Overview(1), (2), (3)
(concluded)

16. Weld the outer top cover plate to the DSC shell and perform NDE.
17. Install the transfer cask top cover plate.
18. Lift and downend the transfer cask onto the *transfer* trailer.
19. Ready the HSM to receive the DSC.
20. Ready the cask for *transfer* and tow the *transfer* trailer to the HSM.
21. Position the transfer cask with the HSM access opening.
22. Remove the transfer cask top cover plate.
23. Align and secure the transfer cask to the HSM.
24. Set-up and ready the hydraulic ram for DSC transfer.
25. Push the DSC into the HSM.
26. Retract the ram and disengage the transfer cask from the HSM.
27. Install the DSC axial retainer and the HSM door.

---

(1) See Section 5.1 for more detailed system operation description.
(2) See Appendices K, M, N, P, T and U for the operations overview of the NUHOMS®-61BT, 32PT, 24PHB, 24PTH, 61BTH and 32PTH1 system, respectively.
(3) Use of the OS197L TC requires significant variations from the steps listed here. See Appendix W.

controlled access. The necessary civil work required to prepare the ISFSI site is the same as that for an ISFSI utilizing vertical storage casks.

Two alternate designs of the standardized HSM are available for licensees' use: the original HSM, now designated as HSM Model 80 and HSM Model 102. HSM Model 102 design is similar to HSM Model 80 design except for the following two features:

- The steel encased composite door of HSM Model 80 design is replaced by a two foot thick reinforced concrete door with a steel liner on its inside surface. The steel liner mitigates DSC damage from spalled concrete due to tornado generated missile impact.
- The inlet and outlet vents, which are formed in concrete for HSM Model 80, are lined with 1½" steel plates.

The above features included with HSM Model 102 are improvements to the original HSM Model 80 design that increase the shielding capabilities of the HSM. The heat transfer capability (decay heat rejection from the DSC to the HSM and heat removal from the HSM by natural convection) of both HSM Model 80 and HSM Model 102 designs are equivalent. Appendix E drawings show both models. Each model can store a DSC with maximum weight up to 102 kips which includes 24P, 52B, 24PT2 and 61BT DSCs.

1.3.2 Transfer Systems Descriptions

1.3.2.1 On-Site TC

The transfer cask used in the NUHOMS® system provides shielding and protection from potential hazards during the DSC closure operations and transfer to the HSM. Five alternate configurations of the transfer cask are available for the licensees' use. The basic configuration, where the cask is provided with a solid neutron shield, is described herein as the "Standardized Cask." An alternate configuration, where a liquid neutron shield is provided instead, is described in this SAR as the "OS197, OS197H, OS197L, or OS200 Cask."

The configuration of the OS197 is a slightly modified version of the NRC approved cask (with a liquid neutron shield) as described in the NUHOMS®–24P Topical Report (1.10). The standardized transfer cask documented in this SAR has a gross weight of less than 90.7 Te (100 tons) and is limited to on-site use under 10CFR72. The OS197 and OS197H transfer casks, which are also limited to on-site use under 10CFR72, have a maximum gross weight of 94.6 Te (104.25 tons) and 113.4 Te (125 Tons), respectively. Where applicable, any other NRC licensed NUHOMS® transfer or transportation cask is acceptable for use with the standardized NUHOMS® system subject to an application specific safety evaluation.

The third configuration of the transfer cask, designated as OS197FC/OS197H FC, or OS197FC-B/OS197HFC-B is a modified version of OS197/OS197H equipped with a modified lid to allow air circulation through the TC/DSC annulus, and is described in Appendix P and Appendix T.

The fourth configuration of the transfer cask, designated as OS197L TC, is a version of the OS197 TC designed for *heat loads of 13 kW maximum. It* is described in Appendix W.

The fifth configuration is designated as OS200TC. This is a larger diameter transfer cask designed to accommodate the 32 PTH1 DSC. The OS200TC is described in Appendix U.

The standardized transfer cask for the NUHOMS® system, has a 4.75m (186.75 inches) long inner cavity, a 1.73 m (68 inches) inside diameter and a maximum payload capacity of 44,430 kg (97,950 pounds) wet and 42,321 kg (93,300 pounds) dry. These maximum payload capacities are based on standardized transfer cask weights of 102,050 (without top lid) and 106,700 pounds (with top lid), respectively. A cask collar is used to extend the transfer cask cavity length by .25 m (10 inches) for use with the longer DSCs for BWR fuel. The OS197 and OS197H transfer casks with a longer cavity length of 196.75 inches (no cask collar) may be used for DSCs with BWR fuel and when combined with a cask spacer may also be used to load DSCs with PWR fuel. The transfer cask is designed to meet the requirements of 10CFR72 for on-site transfer of the DSC from the plant's fuel pool to the HSM. As shown in Figure 1.3-6, the transfer cask is constructed from two concentric cylindrical steel shells with a bolted top cover plate and a welded bottom end assembly. The annulus formed by these two shells is filled with cast lead to provide gamma shielding. The transfer cask also includes an outer steel jacket which is filled with a hydrogen rich solid material or water for neutron shielding. The top and bottom end assemblies also incorporate a solid neutron shield material.

The transfer cask is designed to provide sufficient shielding to ensure that dose rates are ALARA. Two lifting trunnions are provided for handling the transfer cask in the plant's fuel/reactor building using a lifting yoke and an overhead crane. Lower support trunnions are provided on the cask for pivoting the transfer cask from/to the vertical and horizontal positions on the support skid/transfer trailer. A cover plate is provided to seal the bottom hydraulic ram access penetration of the cask during fuel loading.

1.3.2.2 Transfer Equipment

Note: For transfer of the OS197L TC listed in Section 1.3.2.1 above, a modified version of the transfer equipment described in the following paragraphs is provided. See Appendix W.1 for a brief description of the details.

Transfer Trailer: The NUHOMS® transfer trailer consists of a heavy industrial trailer with a payload capacity of 113.4 Te (125 tons). The trailer transfers the cask support skid and the loaded transfer cask between the plant's fuel/reactor building and the ISFSI. The trailer is designed to ride as low to the ground as possible to minimize the HSM height and the transfer cask height during transfer and DSC transfer operations. Figure 1.3-7 shows the heavy haul industrial trailer used with the standardized NUHOMS® system. The trailer is equipped with four hydraulic leveling jacks to provide vertical travel for alignment of the cask with the HSM. The trailer is towed by a conventional heavy haul truck tractor or other suitable prime mover. The nominal trailer bed height during canister transfer to the HSM is such that the transfer cask is typically not elevated more than 1.68 m (5'-6") above grade as measured from the lowest point on the cask. This is well below the 2.0 m (80 inch) drop height used as the accident drop design basis of the cask and canister.

Cask Support Skid: The NUHOMS® system cask support skid is similar in design and operation to other transfer cask skids used for shipment of fuel. The key differences are:

1. There is no ancillary equipment mounted on the skid, except as described on pages P.1-3 and P.1-4.
2. The skid is mounted on a surface with sliding support bearings and hydraulic positioners to provide alignment of the cask with the HSM. Brackets with locking bolts are provided to prevent movement during trailer towing.
3. The hydraulic ram is mounted on the skid or, as an option, the ram can be set-up using a frame structure bolted to the cask bottom and a rear support tripod.

between the edge of the cover plate and the DSC shell. This weld provides the inner seal for the DSC.

DSC Drying and Backfilling: The initial blow-down of the DSC is accomplished by pressurizing the vent port with helium. The remaining liquid water in the DSC cavity is forced out the siphon tube and routed back to the fuel pool or to the plant's liquid radwaste processing system, as appropriate. The DSC is then evacuated to remove the residual liquid water and water vapor in the DSC cavity. When the system pressure has stabilized, the DSC is backfilled with helium and re-evacuated. The second backfill and evacuation ensures that the reactive gases remaining are less than 0.25% by volume. After the second evacuation, the DSC is again backfilled with helium and slightly pressurized. A helium leak test of the inner seal weld is then performed. The helium pressure is then reduced, the helium lines removed, and the siphon and vent port penetrations seal welded closed.

Outer DSC Sealing: After helium backfilling, the DSC outer top cover plate is installed by placing a second seal weld between the cover plate and the DSC shell. Together with the inner seal weld, this weld provides a redundant seal at the upper end of the DSC. The lower end has redundant seal welds which are installed and tested during fabrication. The NUHOMS®-61BT, 32PT, 24PHB, 24PTH, *61BTH and 32PTH1* DSCs are designed and tested to be leak tight per ANSI N14.5-1997 as described in Appendices K, M, N, P, *T and U*, respectively.

Cask/DSC Annulus Draining and Top Cover Plate Placement: The transfer cask is drained, removing the demineralized water from the cask/DSC annulus. A swipe is then taken over the DSC exterior at the DSC top cover plate and the upper portion of the DSC shell. Clean demineralized water is flushed through the cask/DSC annulus to remove any contamination left on the DSC exterior as required. The transfer cask top cover plate is then put in place using the plant's crane. The cask lid is bolted closed for subsequent handling operations.

Placement of Cask on *Transfer* Trailer Skid: The transfer cask is then lifted onto the cask support skid. The plant's crane is used to downend the cask from a vertical to a horizontal position. Trunnions must be seated completely onto the trunnion bearings. The trunnion closure plates are then installed (optional for the OS197 TC and the OS197H TC).

Transfer of Loaded Cask to HSM: Once the cask is securely installed onto the skid, loaded and secured, the *transfer* trailer is towed to the ISFSI along a predetermined route on a prepared road surface. Upon entering the ISFSI secured area, the transfer cask is generally positioned and aligned with the particular HSM in which a DSC is to be transferred.

Cask/HSM Preparation: At the ISFSI with the transfer cask generally positioned in front of the HSM, the cask top cover plate is removed. The transfer trailer is then backed into close proximity with the HSM and the HSM door is removed. The skid positioning system is used for the final alignment and docking of the cask with the HSM.

Loading DSC into HSM: After final alignment of the transfer cask, HSM, and hydraulic ram; the DSC is pushed into the HSM by the hydraulic ram (located at the rear of the cask).

Table 1.3-1
Components, Structures and Equipment for the Standardized NUHOMS® System
(concluded)

On-Site TC (OS197, OS200)

- Cask Structural Shell Assembly
- Bolted Top Cover Plate[4]
- Upper Lifting Trunnions
- Lower Support Trunnions
- Lead Gamma Shielding
- Inner Liner
- Outer Jacket
- Neutron Shielding
- Ram Access Penetration Cover Plate
- *Supplemental Shielding (OS197L TC Only; see Appendix W)*

***Transfer* Trailer**

- Heavy-Haul Industrial Trailer
- Cask Support Skid
- Skid Positioning System

Hydraulic Ram System

- Hydraulic Cylinder and Supports
- Hydraulic Power Supply
- Grapple Assembly

(4) See Appendices P and T for modifications made to the OS197 TC top cover to accommodate higher heat loads for PWR and BWR fuel, respectively (OS197FC, OS197 FC-B)

Once inside the HSM, the DSC and its payload of SFAs is in passive dry storage. Safe storage in the HSM is assured by a natural convection heat removal system, and massive concrete walls and slabs which act as biological radiation shields. The storage operation of the HSMs and DSCs is totally passive. No active systems are required.

3.1.2.1 Handling and Transfer Equipment

The handling and transfer equipment required to implement the NUHOMS® system includes a cask handling crane at the reactor fuel pool, a cask lifting yoke, a transfer cask, a cask support skid and positioning system, a low profile heavy haul *transfer* trailer and a hydraulic ram system. This equipment is designed and tested to applicable governmental and industrial standards and is maintained and operated according to the manufacturer's specifications. Performance criteria for this equipment, excluding the fuel/reactor building cask handling crane, is given in the following sections. The criteria are summarized in Table 3.1-7.

The handling and transfer equipment and operational steps required when using the OS197L TC for loading and transfer operations involve the use of supplemental shielding, remote crane operations, and optical targets as described in Appendix W.

On-Site Transfer Cask: The on-site transfer cask used for the NUHOMS® system has certain basic features. The DSC is transferred from the plant's fuel pool to the HSM inside the transfer cask. The cask provides neutron and gamma shielding adequate for biological protection at the outer surface of the cask, *the exception to this being the OS197L transfer cask, which requires supplemental shielding to provide an approximately equivalent level of protection*. The cask is capable of rotation, from the vertical to the horizontal position on the support skid. The cask has a top cover plate which is fitted with a lifting eye allowing removal when the cask is oriented horizontally. The cask is capable of rejecting the design basis decay heat load to the atmosphere assuming the most severe ambient conditions postulated to occur during normal, off-normal and accident conditions. For the NUHOMS®-24P, 24PHB DSC or the NUHOMS®-24PT2 DSC, the standardized transfer cask has a cylindrical cavity of 1.73 m (68 inches) diameter and 4.75 m (186.75 inches) in length and a maximum dry payload capacity of 42,321 Kg (93,300 pounds). For the NUHOMS®-52B or NUHOMS®-61BT, the standardized transfer cask is fitted with an extension collar to accommodate the longer BWR DSC and fuel. Alternatively, the OS197 and OS197H transfer casks with a full length cavity of 5.0 m (196.75 inches) may be used for the NUHOMS®-24P, 24PHB (with cask spacer), NUHOMS®-52B, NUHOMS®-61BT DSCs, NUHOMS®-24PT2 DSC (with cask spacer) or NUHOMS®-32PT DSC (with cask spacer). The OS197 and OS197H casks can carry a maximum dry payload of 44,100 kg (97,250 lb) and 52,600 kg (116,000 lb), respectively. These payload capacities are based on a transfer cask weight of 111,250 pounds. The cask and the associated lifting yoke are designed and operated such that the consequences of a postulated drop satisfy the current 10 CFR 50 licensing bases for the vast majority of plants. See Appendix T for the description of OS197FC-B transfer cask and Appendix U for description of the OS200/*OS200FC* transfer cask. Appendix W provides a detailed description of the OS197L transfer cask.

shipping cask externals (3.22) Table V, 10CFR71.87(i)(1). Surface swipes of the DSC exterior are taken while in the cask decon area to assure that the maximum DSC removable contamination does not exceed:

Beta/gamma emitters	2,200 dpm/100 cm^2
Alpha emitters	220 dpm/100 cm^2

Transfer cask external contamination is minimized by the use of smooth, easily decontaminated surface finishes to minimize personnel radiation exposures during cask handling operations outside the spent fuel pool. There are no explicit surface contamination limits in 10CFR72 for an on-site transfer cask, and the controlling values will be determined by the specific licensee's radiation protection (RP) program.

Containment of radioactive material associated with spent fuel assemblies is provided by fuel cladding, the DSC stainless steel shell, and double seal welded inner and outer closures.

3.3.7.2 Radioactive Waste Treatment

No radioactive waste is generated during the storage period for the NUHOMS® DSC. Radioactive wastes generated during DSC loading operations (contaminated water from the spent fuel pool and potentially contaminated helium from the DSC cavity) are treated using existing plant system and procedures as described in Chapter 6.

3.3.7.3 On-site Waste Storage

The requirements for on-site waste storage are satisfied by existing plant facilities for handling and storage of waste from the spent fuel pool and dry active wastes as described in Chapter 6.

3.3.8 Industrial and Chemical Safety

No hazardous chemicals or chemical reactions are involved in the NUHOMS® system loading and storage operations. Industrial safety relating to handling of the cask and DSC are addressed by the licensee's procedures which meet the Occupational Safety and Health Administration (OSHA) requirements.

OR

- The coarse and fine aggregates to be one or a mix of the following: limestone, dolomite, marble, basalt, granite, rhyolite, gabbro. Determination of the aggregate constituents shall be done in accordance with the same methods described above.

For all PWR and BWR HSM components the above aggregate requirements can be waived if the criteria established by Appendix D for strength reduction is further validated by strength tests performed on the actual concrete mix to be used for construction subjected to elevated temperatures established by the design. Alternatively the minimum compressive strength requirements for the concrete may be increased to account for an appropriate reduction in concrete strength. This approach removes the need to reevaluate the HSM design analyses.

4.2.3.3 On-Site Transfer Cask

The on-site transfer cask *(TC)* is a nonpressure-retaining cylindrical vessel with a welded bottom assembly and bolted top cover plate. *Section 1.3.2.1 describes the four alternate TC configurations provided for use with the NUHOMS® system.* The *TC* is designed for on-site *transfer* of the DSC to and from the plant's spent fuel pool and the ISFSI as shown in Figure 4.2-10 and Figure 4.2-11 *(OS197 TC is shown). For all the TC configurations except the OS197L TC, t*he *TC* provides the principal biological shielding and heat rejection mechanism for the DSC and SFAs during handling in the fuel/reactor building, DSC closure operations, *transfer* to the ISFSI, and transfer to the HSM. The *TC* also provides primary protection for the loaded DSC during off-normal and drop accident events postulated to occur during the *transfer* operations. The *standardized TC* is illustrated in Figure 1.3-6.

Appendix W, Section W.1.2 provides a detailed description of the OS197L TC, which is shown in Figure W.1-1. The OS197L TC relies on the use of supplemental shielding to provide biological shielding in conjunction with remote operations during handling in the fuel/reactor building, transfer to the ISFSI, and transfer to the HSM operations. Drawings for the OS197L transfer cask, including the supplemental shielding components are provided in Appendix W, Section W.1.5.

The *standardized (with solid neutron shield)* transfer cask may be fitted with a shielded collar to extend the cask cavity length to accommodate the longer NUHOMS®-52B DSC as shown in Figure 4.2-12. The collar is a heavy forged steel ring with a bolt circle to match that of the transfer cask top flange and cover plate. Alternatively, a *TC* with a longer cavity length may be used *(e.g., OS197)* for DSCs with PWR (with cask spacer) or BWR fuel.

The transfer cask to be used by a utility may be any one of the designs documented in Appendix E, including the standardized cask, OS197, OS197H, or OS197L, Appendix P (OS197FC/ *OS197HFC*) or Appendix T (OS197FC-B) and Appendix U (OS200 transfer cask for transfer of the bigger diameter 32PTH1 DSC) *or Appendix W (OS197L TC)*. The licensee may also use any other previously NRC reviewed and approved design such as the transfer cask designs documented in the NUHOMS®-24P Topical Report [4.13], the Oconee Nuclear Station ISFSI Safety Analysis Report [4.16], and the Calvert Cliffs ISFSI Safety Analysis Report [4.17], provided it is demonstrated prior to use that the limiting conditions of use as described in CoC 1004 can be met.

The standardized *TC,* OS197, *OS197H, OS197FC, OS197FC-B and OS200 TCs* are constructed from three concentric cylindrical shells to form an inner and outer annulus. These are filled with lead and a neutron absorbing material. The two inner shells are welded to heavy forged ring assemblies at the top and bottom ends of the cask as shown in Figure 1.3-6. Rails fabricated from a non-galling, wear resistant material coated with a high contact pressure dry film lubricant are provided to facilitate DSC transfer. All surfaces exposed to fuel pool water are stainless steel. The transfer cask structural shell and the bolted top cover plate may be fabricated from carbon or stainless steel. The transfer cask carbon steel structural shell and top cover plate are coated with a durable epoxy paint which is shop applied in accordance with the manufacturer's standards. This coating system is suitable for immersion service with a continuous temperature of 250°F with intermittent temperatures to 400°F.

The method used to cast the transfer cask lead shielding will vary between fabricators. Only one transfer cask need be utilized for each ISFSI. Transfer casks for different ISFSIs may be supplied by different fabricators. Each fabricator is required to submit detailed procedures for the lead pour consistent with the requirements delineated on the Appendix E drawings. These procedures include specific locations and sealing of pour holes, temporary bracing, and controlled cooling methods for the lead, all of which must meet the applicable codes and standards.

As shown in Figure W.1-1, the OS197L transfer cask is constructed from a single, thicker (2.68"nominal thickness) structural shell in lieu of the concentric nominal ½" thick inner liner and the nominal 1.5" thick outer structural shell with lead shielding in the annular space in the OS197 transfer cask. To compensate for the lack of lead shielding, the OS197L transfer cask requires the use of supplemental shielding (made up of thick steel plates, as shown in Figures W.1-2 and W.1-3) in conjunction with remote operations and use of optical targets. The supplemental shielding consists of a 6" thick carbon steel upper shielding bell and a lower shielding sleeve which enclose the cask in the decontamination area, and 5.5" combined thickness carbon steel plates/covers which are attached to or supported by the transfer trailer skid and which enclose the transfer cask while on the transfer trailer.

The transfer cask neutron shield cavity is fabricated as a pressure vessel since it is desirable to have this cavity remain leak tight to prevent intrusion of contaminated spent fuel pool water. Also, the support members for the outer shell of the solid neutron shield are angled at 45° with respect to the transfer cask structural shell to further enhance shielding and decay heat removal. Solid neutron shielding materials are also incorporated into the top and bottom end closures to provide effective radiological protection.

Two trunnion assemblies are provided in the upper region of the cask for lifting of the transfer cask and DSC inside the plant's fuel/reactor building, and for supporting the cask on the skid for *transfer* to and from the ISFSI. An additional pair of trunnions in the lower region of the cask are used to position the cask on the support skid, serve as the rotation axis during down-ending of the cask, and provide support for the bottom end of the cask during *transfer* operations. The trunnion assemblies may be fabricated either as (1) a one solid forged piece or, (2) as a hollow trunnion piece welded to a trunnion sleeve assembly with the hollow space filled with NS-3 neutron shielding material. There are no testing requirements per the ASME Code for the transfer cask trunnions. Neither the transfer cask nor the trunnions are special lifting devices per ANSI N14.6. Nonetheless, for transfer casks fabricated under the general license, a one-time pre-service load test of the trunnions is performed at a load equal to 150% of the design load followed by an examination of all accessible

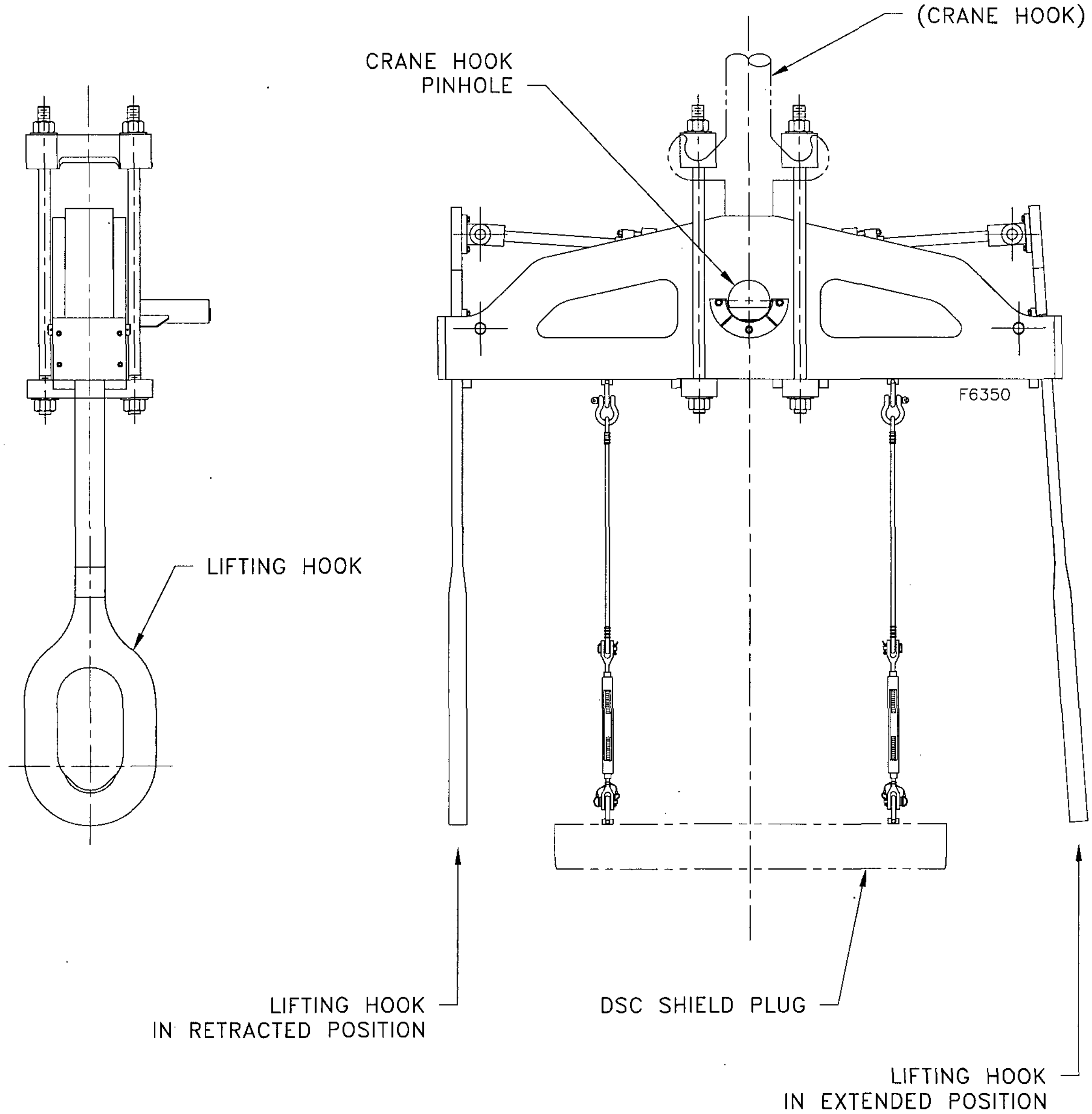

Figure 4.2-15a
<u>NUHOMS® Transfer Cask Alternate Lifting Yoke</u>

Although the loss of the neutron shield is highly unlikely, the loss of neutron shield accident case is analyzed in Section 8.2.5.3. The transfer cask *(all the TCs listed in Section 1.3.2.1 except the OS197L TC)* is designed to provide adequate shielding to maintain the maximum radiation surface dose to less than 5 *rem*/hr combined gamma and neutron for a cask drop accident event assuming a complete loss of neutron shielding. *See Appendix W, Chapter W.5 for the consequences of a loss of neutron shield accident* when using the OS197L TC.

The transfer cask is designated important to safety since it provides biological shielding and structural protection for the DSC from impact loads. The codes and standards used to design and fabricate the *transfer* cask are presented in Section 4.7.4.

4.7.3.3 *Transfer* Cask Lifting Yoke

The lifting yoke may be a fixed arm J-hook configuration (Figure 4.2-15) or a swing arm *type* configuration (*such as the one shown in* Figure 4.2-15a) may be used.

The fixed arm lifting yoke is a passive, open hook design with two parallel lifting beams fabricated from thick, high-strength carbon steel plate material, with a decontaminable coating.

It is designed to support a loaded transfer cask. A lifting pin connects the fuel/reactor building cask handling crane hook and the lifting yoke. The lifting yoke is shown in Figure 4.2-15.

The swing arm yoke is similar in configuration except the arms are allowed to pivot and are operated by a water filled hydraulic system *or a pneumatic air-operated system*. Check valves are included in the design to ensure that the swing arm yoke can not fail in any operational mode that could result in an uncontrolled release of the onsite transfer cask.

It is designed to be compatible with the fuel/reactor building crane hook and load block. The lifting yoke engages the outer shoulder of the transfer cask lifting trunnions. To facilitate ease of shipment and maintenance, all yoke subcomponent structural connections are bolted.

The lifting yoke is designated "safety related" since it is in the direct load path of the cask. The codes and standards used to design and fabricate the lifting yoke are presented in Section 4.7.4.

4.7.3.4 *Transfer* Trailer

The *transfer* trailer is designed for use with the NUHOMS® transfer equipment. Its function is to move the cask and cask support skid from the fuel/reactor building to the pressure and flow control devices ensure that the maximum design forces and speeds of the hydraulic ram are not exceeded. System pressure gauges are provided to monitor the insertion operation and to verify that design force limits are not exceeded.

4.7.3.7 Ram Support Assembly

The ram hydraulic cylinder may be permanently mounted on the cask transfer trailer using a steel support assembly, or it may be installed on an adjustable tripod as shown in Figure 1.3-9 and attached to the cask by a steel support frame. In either case, the hydraulic ram push loads are transmitted through the support assembly to the transfer cask, through the cask to the cask restraints and into the HSM front wall embedments.

4.7.3.8 Cask Support Skid

The cask support skid is a structural steel frame fabricated from standard wide flange members, built up box beam cross members and trunnion support towers. The cask support skid, shown in Figure 1.3-8, is designed according to the AISC code for its operating loads. During cask loading and trailer towing operations, the cask support skid is rigidly attached to the transfer trailer by four bolted brackets. During cask alignment, the bolts are removed, and the alignment system is used to move the cask support skid into position. For this operation, the skid is supported by the skid positioning system bearing pads located on the trailer frame cross members. The transfer cask is supported on the front and rear trunnion support tower pillow blocks. For cask downending, the lower trunnions are engaged into the front pillow blocks, and the top section of the blocks installed. The cask lifting yoke and fuel/reactor building crane are then used to lower the upper trunnions into the rear trunnion supports. The yoke is then removed and the upper trunnion tower closure plates are installed (optional for the OS197 TC and the OS197H TC).

A modified version of the cask support skid is provided for transfer of the OS197L TC, as described in Appendix W, Chapter W.1.

4.7.4 Transfer Equipment

Applicable sections of the following codes and standards are specified for the design, construction, and testing of the NUHOMS® ISFSI transfer equipment components.

4.7.4.1 Transfer Cask and Lifting Yoke

A. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC, "Class 2 Components", 1983 Edition through Winter 1985 Addenda, used as a guide for design and fabrication.

B. ASME Boiler and Pressure Vessel Code, Section III, Division 1, Appendices

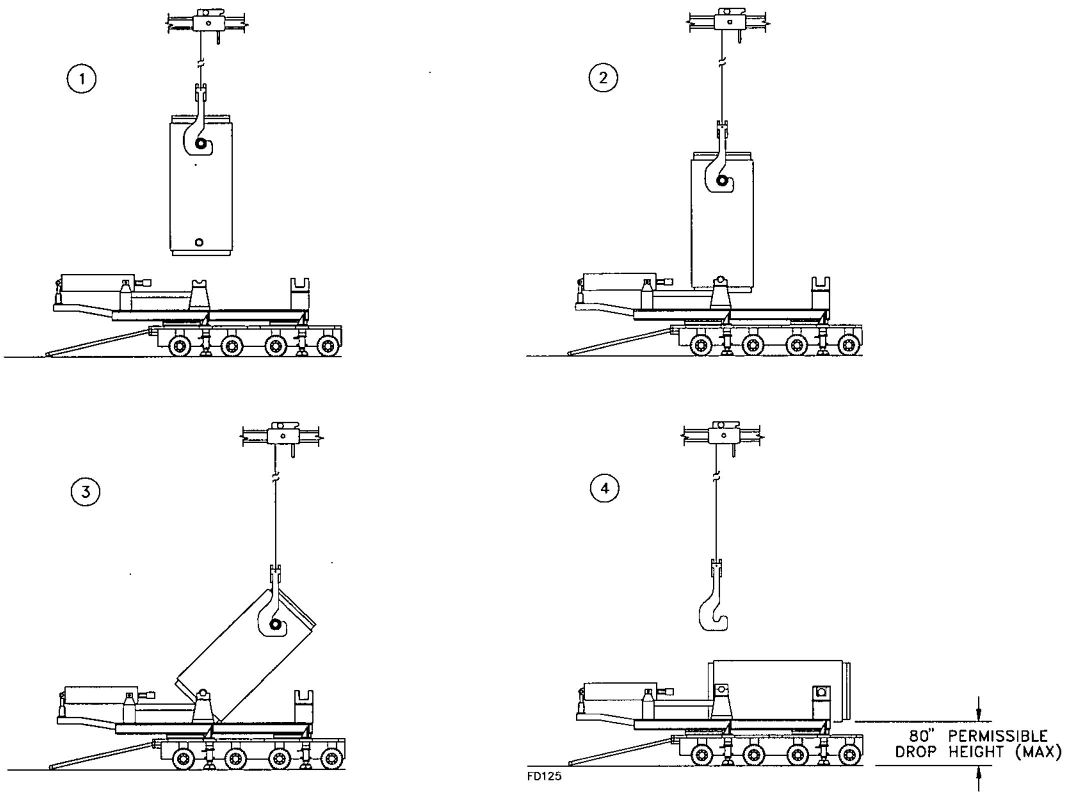

Figure 4.7-4
NUHOMS® *Transfer* Cask Downending Sequence

Table 4.8-1
<u>*ASME Code Alternatives for NUHOMS®-24P, 24PHB*</u>
<u>**and 52B DSC Pressure Boundary Components**</u>

Reference ASME Code Section/Article	***Code Requirement***	***Alternatives, Justification and Compensatory Measures***
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section 4.2.1 may be used for construction, but in no case earlier than 3 years before that specified in Section 4.2.1.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section 4.2.1 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NB-1100	*Requirements for Code Stamping of Components, Code reports and certificates, etc.*	*Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.*
NB-1132	*Attachments with a pressure retaining function, including stiffeners, shall be considered part of the component.*	*Bottom shield plug and outer bottom cover plate are outside code jurisdiction; these components together are much larger than required to provide stiffening for the inner bottom cover plate; the weld that retains the outer bottom cover plate and with it the bottom shield plug is subject to root and final PT examination.*
NB-2130 NB-4121	*Material must be supplied by ASME approved material suppliers.* Material Certification by Certificate Holder	*Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.*
NB-4240 NB-5230	Full penetration welds are required for pressure boundary closure joints Weld examination shall be UT or RT with surface PT	<u>DSC Pressure Boundary Welds:</u> The joint details at the top and bottom end of the DSCs are not full penetration welds and thus do not comply with the requirements of figure NB-4243-1 for Category C flat head closure pressure and containment boundary welds. Volumetric weld inspection (RT or UT) is not practical due to the DSC geometry at the top and bottom closures and due to high radiation at the top closure after fuel loading (ALARA consideration). The inner and outer cover plate closure welds provide redundant closure welds, which are required by the 10CFR72 license. These welds are partial penetration welds that have been designed using a conservative "weld efficiency" factor of 0.6. Breach of the DSC confinement barriers due to an undetected flaw in any single weld layer is implausible due to the requirement for multi-layer welds. The top and bottom outer cover plate to shell welds and the inner bottom cover plate to shell weld receive a root and final PT. The top inner cover plate to shell weld, which is leak tested, has a final PT only.

Table 4.8-1
ASME Code Alternatives for NUHOMS®-24P, 24PHB and 52B DSC Pressure Boundary Components

(continued)

Reference ASME Code Section/Article	*Code Requirement*	*Alternatives, Justification and Compensatory Measures*
NB-7000	Overpressure Protection	No overpressure protection is provided for the NUHOMS® DSCs. The function of the DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The DSC is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature.
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000	The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.
NB-5520	NDE personnel must be qualified to a specific edition of SNT-TC-1A	Permit use of more recent edition of SNT-TC-1A.

Table 4.8-2
ASME Code Alternatives for NUHOMS®-24P, 24PHB, and 52B DSC Basket Assembly

Reference ASME Code Section/Article	*Code Requirement*	*Alternatives, Justification and Compensatory Measures*
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section 4.2.1 may be used for construction, but in no case earlier than 3 years before that specified in Section 4.2.1.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section 4.2.1 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NF-2130	Material must be supplied by ASME approved material suppliers	All DSC Basket Assembly sub-components designated as ASME on the DSC drawings are obtained from TN approved suppliers with Certified Material Test Reports (CMTR's). The DSC basket subcomponents listed below have been designated as non-Code. • Guide Sleeves, Oversleeves, and extraction stops (PWR only) • Neutron Absorber Plates and misc. hardware, such as anti-rotation pin, screws and locknuts, (BWR Only) • Coating for Spacer Discs
NF-4121	Material Certification by Certificate Holder	Material traceability and certification are maintained in accordance with TN's NRC approved QA program
NF -8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.*
NF-5520	*NDE Personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of more recent edition of SNT-TC-1A.*

Table 4.9-1

ASME Code Alternatives for the Standardized NUHOMS® System TCs Except for the OS200 and OS200FC TCs

(Applies to TC structural components only; lead shielding, neutron shielding, and neutron shield jacket of the TC are not addressed by this table)

Reference ASME Code Section/Article	Code Requirement	Exception, Justification, & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality assurance is provided according to 10CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of code editions and addenda*	*Code edition and addenda other than those specified in Section 4.2.1 may be used for construction, but in no case earlier than 3 years before that specified in Section 4.2.1. Materials produced and certified in accordance with ASME Section II material specification from code editions and addenda other than those specified in Section 4.2.1 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable subsection of the Section III edition and addenda used for construction.*
NC-1100	*Requirements for code stamping of components*	*The cask is designed and fabricated to the requirements of Subsection NC, to the maximum extent practical. However, the transfer cask does not have a code stamp. Code stamping is not required by 10CFR72 regulation. Therefore, the fabricator is not required to be ASME Certified.*
NC-2000	*ASME code materials are to be used.*	*The cask bottom ram access cover plate is made of ASTM A240, a non-ASME material. This cover plate is a water tight closure used during fuel loading/unloading operations in the fuel/reactor building only. This is not a pressure boundary component, and its failure does not result in any public safety concerns.*
NC-2130	*Material must be supplied by ASME approved material suppliers.*	*Materials designated as ASME on UFSAR Appendix E drawings are obtained by TN approved suppliers with Certified Material Test Reports (CMTRs). Material is certified to meet all ASME Code criteria but is not eligible for certification or code stamping, if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NC-2130 is not possible.*
NC-4120	*Material certification by certificate holder*	*Material traceability & certification are maintained in accordance with TN's NRC approved QA program.*
NC-4240	*Full penetration welds are required for pressure boundary closure joints.*	*The joint between the ram access penetration forging and the bottom end plate consists of partial penetration welds, while NC-3200 would require full penetration welds. This cover plate is a water tight closure used during fuel loading/unloading operations in the fuel/reactor building only. This is not a pressure boundary component, and its failure does not result in any public safety concerns.*
NC-5250	*Category A and B joints shall be fully radiographed.*	*UFSAR Appendix E drawing NUH-03-8001 permits weld examination of (a) the circumferential and longitudinal welds for the structural shell and (b) the weld between the bottom end plate and the bottom support ring to be done using radiography (RT) or ultrasound (UT) while NC-5250 allows full penetration welds to be examined by RT only. Since the structural shell is not a pressure boundary, this code exception is acceptable.*
NC-6000	*All completed pressure retaining systems shall be pressure tested.*	*With respect to pressure testing requirements, the transfer cask is considered a non pressure retaining component. Therefore, no pressure testing is required. However, the liquid neutron shield cavity, cask bottom neutron shield cavity, and the bottom cover plate assembly are pressure and leak tested.*
NC-7000	*Overpressure protection*	*The transfer cask is considered a non pressure retaining component. Therefore, no overpressure protection is provided for the transfer cask, except that a pressure relief valve is provided for the annular neutron shielding.*
NC-8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The transfer cask nameplate provides the information required by 10CFR72. Code stamping is not required for the transfer cask. QA data packages are prepared in accordance with the requirements of 10CFR72 and TN's NRC approved QA program.*

Table 4.9-1
ASME Code Alternatives for the Standardized NUHOMS® System TCs Except for the OS200 and OS200FC TCs

(concluded)

Reference ASME Code Section/Article	Code Requirement	Exception, Justification, & Compensatory Measures
NC-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A.*	*Permit use of more recent edition of SNT-TC-1A.*

5. OPERATION SYSTEMS

This chapter presents the operating procedures for the standardized NUHOMS® system described in previous chapters and shown on the drawings in Appendix E for the 24P and 52B systems. The operating procedures for the 61BT, 24PT2, 32PT, 24PHB, 24PTH, 61BTH and 32PTH1 systems are described in Appendices K, L, M, N, P, T, and U, respectively. The procedures include preparation of the DSC and fuel loading, closure of the DSC, *transfer* to the ISFSI, DSC transfer into the HSM Model 80 and Model 102, monitoring operations, and DSC retrieval from the HSM Model 80 and Model 102. The operating procedures involving the HSM-H, HSM Model 152, and HSM Model 202 are described in Appendices P, R, and V, respectively. *The operating procedures for loading 61BT and 32PT DSCs when using an OS197L TC are described in Appendix W, Chapter W.8.* The standardized NUHOMS® transfer equipment, and the existing plant systems and equipment are used to accomplish these operations. Procedures are delineated here to describe how these operations are to be performed and are not intended to be limiting. Standard fuel and cask handling operations performed under the plant's 10CFR50 operating license are described in less detail. Existing operational procedures may be revised by the licensee and new ones may be developed according to the requirements of the plant, provided that the limiting conditions of operation specified in Technical Specifications, Functional and Operating Limits of the NUHOMS® CoC (5.6) are not exceeded.

5.1 Operation Description

The following sections outline the typical operating procedures for the standardized NUHOMS® system. These generic NUHOMS® procedures have been developed to minimize the amount of time required to complete the subject operations, to minimize personnel exposure, and to assure that all operations required for DSC loading, closure, transfer, and storage are performed safely. Plant specific ISFSI procedures are to be developed by each licensee in accordance with the requirements of 10CFR72.24 (h) and the guidance of Regulatory Guide 3.61 (5.7). The generic procedures presented here are provided as a guide for the preparation of plant specific procedures and serve to point out how the NUHOMS® system operations are to be accomplished. They are not intended to be limiting, in that the licensee may judge that alternate acceptable means are available to accomplish the same operational objective.

The generic operating procedures presented herein also do not address the use of auxiliary equipment which is optional or represents a level of detail which a licensee may choose to implement based on licensee preference. Examples of such auxiliary items are the Neutron Shield Overflow Tank (used with OS197 or OS197H Cask only), TC/DSC Annulus Pressurization Tank, and the Shield Plug Restraints.

5.1.1 Narrative Description

The following steps describe the recommended generic operating procedures for the standardized NUHOMS® system. Flowcharts of NUHOMS® system loading and retrieval operations are provided in Figure 5.1-1 and Figure 5.1-2, respectively.

5.1.1.1 Preparation of the Transfer Cask and DSC

1. Prior to placement in dry storage, the candidate fuel assemblies are to be visually examined to insure that no known or suspected gross cladding breaches exist. Pinholes and hairline cracks are acceptable. Verification of fuel integrity may also be accomplished using suitable existing plant records. The assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Technical Specification *2.1*. Depending on the length of the authorized fuel assemblies to be loaded, fuel spacers may be placed within the DSC to reduce the fuel assembly/DSC cavity gap in consideration of Part 71 requirements. There are no requirements for fuel spacers under Part 72. Fuel spacers, if used, may be placed below the assembly, above the assembly, or both, and shall be evaluated for any adverse impact.

2. Prior to being placed in service, the transfer cask is to be cleaned or decontaminated as necessary to insure a surface contamination level of less than those specified in Technical Specification *5.2.4.d*.

3. Place the transfer cask in the vertical position in the cask decon area using the cask handling crane and the transfer cask lifting yoke.

4. Place scaffolding around the cask so that the top cover plate and surface of the cask are easily accessible to personnel.

5. Remove the transfer cask top cover plate and examine the cask cavity for any physical damage and ready the cask for service.

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

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

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

9. Fill the DSC cavity with water from the fuel pool or an equivalent source which meets the requirements of Technical Specification *3.2.1*. For PWR fuel, the

CAUTION: Verify that the requirements of Technical Specification *5.3.1.B*, "TC/DSC Transfer Operations at High Ambient Temperatures" are met prior to next step.

3. Using a suitable heavy haul tractor, *transfer* the cask from the plant's fuel/reactor building to the ISFSI along the designated transfer route.

4. Once at the ISFSI, position the *transfer* trailer to within a few feet of the HSM.

5. Check the position of the trailer to ensure the centerline of the HSM and cask approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.

6. Back the cask to within a few inches of the HSM, set the trailer brakes and disengage the tractor. Drive the tractor clear of the trailer. Extend the transfer trailer vertical jacks.

7. Unbolt and remove the cask top cover plate.

8. Connect the skid positioning system hydraulic power unit to the positioning system via the hose connector panel on the trailer, and power it up. Remove the skid tie-down bolts and use the skid positioning system to bring the cask into approximate vertical and horizontal alignment with the HSM. Using optical survey equipment and the alignment marks on the cask and the HSM, adjust the position of the cask until it is properly aligned with the HSM.

9. Using the skid positioning system, fully insert the cask into the HSM access opening docking flange.

10. Secure the cask trunnions to the front wall embedments of the HSM using the cask restraints.

11. After the cask is docked with the HSM, verify the alignment of the transfer cask using the optical survey equipment.

12. Position the hydraulic ram behind the cask in approximate horizontal alignment with the cask and align the ram. Remove either the bottom ram access cover plate or the outer plug of the two-piece temporary shield plug. Power up the ram hydraulic power supply and extend the ram through the bottom cask opening into the DSC grapple ring.

13. Activate the hydraulic cylinder on the ram grapple and engage the grapple arms with the DSC grapple ring.

14. Recheck all alignment marks in accordance with the Technical Specification *5.3.3* limits and ready all systems for DSC transfer.

15. Activate the hydraulic ram to initiate insertion of the DSC into the HSM. Stop the ram when the DSC reaches the support rail stops at the back of the module.

16. Disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.

17. Retract and disengage the hydraulic ram system from the cask and move it clear of the cask. Remove the cask restraints from the HSM.

18. Using the skid positioning system, disengage the cask from the HSM access opening. Insert DSC axial retainer.

19. The trailer may be moved as necessary to install the HSM door. Install the HSM door and secure it in place. *Verify that a loaded HSM meets the dose rate limits of Technical Specification 5.4.2.*

20. Replace the transfer cask top cover plate (optional, may be done later away from the ISFSI). Secure the skid to the trailer, retract the vertical jacks and disconnect the skid positioning system.

21. Tow the trailer and cask to the designated equipment storage area. Return the remaining transfer equipment to the storage area.

22. Close and lock the ISFSI access gate and activate the ISFSI security measures.

5.1.1.7 Monitoring Operations

1. Perform routine security surveillance in accordance with the licensee's ISFSI security plan.

NOTE: *Perform one of the two alternate surveillance activities listed below.*

2a. Perform a daily visual surveillance of the HSM air inlets and outlets (end wall and roof birdscreens) to insure that no debris is obstructing the HSM vents in accordance with Technical Specification *5.2.5.a* requirements.

2b. Perform a temperature measurement of the thermal performance, for each HSM, on a daily basis in accordance with Technical Specification *5.2.5.b* requirements.

5.1.1.8 DSC Retrieval from the HSM

1. Ready the transfer cask, *transfer* trailer, and support skid for service and tow the trailer to the HSM.

5.4 Operation Support System

NUHOMS® is a self contained passive system and requires no effluent processing systems during storage conditions.

5.4.1 Instrumentation and Control System

There are no instrumentation and control systems used during storage conditions, except for potentially the HSM temperature monitoring required by NUHOMS® Technical Specification 5.2.5.b. *(A temperature monitoring system is only required if the licensee decides to use this approach for compliance with Technical Specification 5.2.5.)* The instrumentation and controls necessary during DSC loading, closure and transfer are described in Section 5.1.3.4.

5.4.2 System and Component Spares

Other than spares for the HSM temperature monitoring, there are no instrumentation or control systems used during storage conditions; thus, no other system and component spare parts are required.

7. RADIATION PROTECTION

The analysis presented in this Chapter is specifically applicable to the storage of the NUHOMS®-24P and -52B *DSCs in the HSM Model 80 and Model 102* and transfer in the standardized cask, OS197 or OS197H TCs. Appendices J, K, L, M, N, P, T, and U provide a similar evaluation for the NUHOMS®-24P long cavity, -61BT, -24PT2, 32PT, 24PHB, 24PTH, 61BTH, and 32PTH1 systems, respectively. Appendices R and V provide an evaluation of these various DSCs stored in the HSM Model 152 and HSM Model 202, respectively.

Shielding analysis of the 61BT *and* 32PT when transferred in *an* OS197L TC are provided in Appendix W, *Chapter W.5*.

7.1 Ensuring That Occupational Radiation Exposures Are As Low As Reasonably Achievable (ALARA)

7.1.1 Policy Considerations

The licensee's existing radiation safety and ALARA policies for the plant should be applied to the ISFSI. The ALARA program should follow the general guidelines of Regulatory Guides 1.8, 8.8, 8.10 and 10CFR20. ISFSI personnel should be trained and updated on ALARA practices and dose reduction techniques. Implementation of ISFSI systems and equipment procedures should be reviewed by the licensee to ensure ALARA exposure during all phases of operations, maintenance and surveillance.

7.1.2 Design Considerations

The design of the NUHOMS® DSC and HSM comply with 10CFR72 ALARA requirements. Features of the NUHOMS® system design that are directed toward ensuring ALARA are:

A. Thick concrete walls and roof on the HSM to minimize the on-site and off-site dose contribution from the ISFSI.

B. A thick shield plug on each end of the DSC to reduce the dose to plant workers performing drying and sealing operations, and during transfer and storage of the DSC in the HSM.

C. Use of a heavily shielded transfer cask for DSC handling and transfer operations to ensure that the dose to plant and ISFSI workers is minimized *(Note that use of the OS197L TC involves supplemental shielding, remote crane operations and optical targets as described in Appendix W, Chapter W.8)*.

8.2.5 Accidental Cask Drop

This section addresses the structural integrity of the standardized NUHOMS® on-site transfer cask, the DSC and its internal basket assembly when subjected to postulated cask drop accident conditions.

8.2.5.1 Cause of Accident

A. Cask Handling and Transfer Operation

As described in Section 5.0, all handling operations involving hoisting and movement of the on-site transfer cask and DSC are performed inside the plant's fuel handling building. These include utilizing the crane for placement of the DSC into the cavity of the transfer cask, lifting the transfer cask/DSC into and out of the plant's spent fuel pool, and placement of the transfer cask/DSC onto or off of the *transfer* skid/trailer. An analysis of the plant's lifting devices used for these operations, such as the crane and lifting yoke, is needed to address a postulated drop accident for the transfer cask and its contents. The postulated drop accident scenarios should be consistent with those currently addressed in the plant's 10CFR50 licensing basis for handling of a shipping cask. Such postulated accidents are plant specific and should be addressed by the licensee.

Once the transfer cask is loaded onto the *transfer* skid/trailer and secured, it is pulled to the HSM site by a tractor vehicle. A predetermined route is chosen to minimize the potential hazards that could occur during *transfer*. This movement is performed at very low speeds. System operating procedures and technical specification limits defining the safeguards to be provided ensure that the system design margins are not compromised. As a result, it is highly unlikely that any plausible incidents leading to a transfer cask drop accident could occur. Similarly, at the ISFSI site, the *transfer* skid/trailer is backed-up to, and aligned with, the HSM using hydraulic positioning equipment. The *transfer* cask is then docked with, and secured to, the HSM access opening. The loaded DSC is transferred to or from the HSM using a hydraulic ram system. The hold down mechanisms that secure the transfer cask to the *transfer* skid/trailer remain in place at all times during the DSC *transfer*. As a result, there is no reasonable way during these operations for a cask drop accident to occur.

B. Cask Drop Accident Scenarios

In spite of the incredible nature of any scenario that could lead to a drop accident for the transfer cask, a conservative range of drop scenarios are developed and evaluated. These bounding scenarios assure that the integrity of the DSC and spent fuel cladding is not compromised. Analyses of these scenarios demonstrate that the transfer cask will maintain the structural integrity of the DSC pressure containment boundary. Therefore, there is no potential for a release of radioactive materials to the environment due to a cask drop. The temperatures are limited by the heat up of the total quantity of concrete behind the surface. Therefore, applying a constant heat flux to the HSM concrete and calculating the time dependent temperature distributions through the concrete, the surface temperature of the concrete as a function of time is obtained. Using the calculated HSM surface temperatures, the maximum DSC and fuel cladding temperatures are determined.

A thermal transient analysis of the HSM for the blocked vent condition is performed using a control volume model of the HSM internal air space and the surrounding concrete walls, roof, and floor. The first law of thermodynamics is used to obtain an energy balance equation that is solved using a forward finite differencing scheme with a sufficiently small time step to ensure the accuracy of the solution. The initial conditions for the analysis correspond to the steady state temperatures calculated for the off-normal analysis cases with an ambient temperature of 125°F. The heat source included in the analysis is the 24 kW decay heat rejected from the surface of the DSC. Technical Specification *5.2.5.a* of CoC 1004 *requires surveillance to ensure that HSM vents are not blocked for periods longer than that assumed in the safety analysis* (40 hours). At the end of 40 hours, it is assumed that corrective actions would be completed and natural circulation air flow restored to the HSM. However, to be conservative, the solution is carried out to five days. The change in temperature with time after vent opening blockage for the HSM roof interior surface is shown in Figure 8.2-16.

At the end of the five days, the maximum HSM inside surface is 479°F. The concrete temperature transient results are used to calculate the DSC shell transient temperatures. The same model of the DSC in HSM cases described in Section 8.1.3 is used. All the convection boundary conditions on the interior of HSM are deleted to simulate complete blockage of all inlet and outlet vents. The conductivity of the basket material is conservatively assumed to be that of fuel. The effective density and specific heat of the homogenized basket material and fuel are used. The maximum DSC surface temperature is 640°F at the end of 40 hours in the blocked vent transient. Using the HEATING7 models of the DSC and its internals described in Section 8.1.3, the maximum fuel cladding temperatures for this case are calculated to be 871°F (466°C) for the PWR fuel and 905°F (485°C) for the BWR fuel. The fuel clad temperature at the start of the blocked vent transient for the 125°F extreme ambient temperature case are calculated to be less than 720°F (382°C) for the PWR fuel and 743°F (395°C) for the BWR fuel. The maximum fuel clad temperature as a function of time is expected to vary linearly between these two temperature values. The resulting temperatures are well below the fuel cladding short term temperature limit of 570°C.

These temperatures are below the levels that safety impairing damage would occur to the HSM or DSC. ACI 349 (8.20) imposes a maximum upper limit of 350°F for concrete temperatures for accident or other short term conditions. As shown in Figure 8.2-16, the HSM concrete temperatures exceed 350°F sometime after 40 hours for a blocked vent transient. Hence, the NUHOMS® Technical Specification *5.2.5* requires a daily visual inspection of the HSM air inlets and outlets or daily monitoring of the HSM thermal performance. The short time exposure of the DSC and the spent fuel assemblies to the elevated temperatures will not cause any damage or result in the release of radioactivity. The maximum DSC internal pressure during this event is 12.1 psig for the PWR fuel and 10.0 psig for the BWR fuel (assuming that no fission and fill gas is released).

Table 10-2
Technical Specification Cross Reference Table between Amendment 10 and Amendment 11

Amendment 10 Tech Spec	Amendment 11 Tech Spec	*Location of Amendment 10 Bases*
1.1.1 Reg*ulatory* Requirement of General License	4.*2.1 Horizontal Storage Module,* and 4.3.3 Site Specific Parameters and Analyses	*N/A*
1.1.2 Operating Procedures	5.1 Procedures	*N/A*
1.1.*3* Quality Assurance	Part of CoC	*N/A*
1.1.*4* Heavy Loads	Part of CoC	*N/A*
1.1.5 Training Module	5.2.2 Training Program	*N/A*
1.1.6 Pre-*Operational* Testing and Training Exercise	Part of CoC	*N/A*
1.1.7 Special Requirements for First System in Place	Not in STS	*N/A*
1.1.8 Surveillance Requirement*s* Applicability	3.0 Limiting Condition for Operation (LCO) and Surveillance Requirements (SR) Applicability	*N/A*
1.1.9 Supplement Shielding	4.3.3 Site Specific Parameters and Analyses	*N/A*
1.1.10 HSM-H Storage Configuration	4.3.1 Storage Configuration	*N/A*
1.1.11 Hydrogen Gas Monitoring for 61BTH and 32PTH1 DSCs	*5.2.6 Hydrogen Gas Monitoring for 24P, 52B, 24PHB, 61BT, 32PT, 24PTH, 61BTH and 32PTH1 DSCs*	*N/A*
1.1.12 Codes and Standards	*4.2 Codes and Standards*	*N/A*
1.2.1 Fuel Specifications	*2.1 Fuel to be stored in the standardized NUHOMS® System and 4.1-Canister Criticality control*	*B 10.2*
1.2.2 DSC Vacuum Pressure During Drying	3.1.1 DSC Bulk Water Removal Medium and Vacuum Drying Pressure	*B 10.3.1.1*
1.2.3, 1.2.3a DSC Helium Backfill Pressure for Various DSCs	3.1.2 DSC Helium Backfill Pressure for various DSCs	*B 10.3.1.2*
1.2.4, 1.2.4a DSC Helium Leak Rate of Inner Seal Weld for Various DSCs	5.2.4c	*B 10.5.2.4c*
1.2.5 DSC Dye Penetrant Test of Closure Welds	5.2.4b	*B 10.5.2.4b*
1.2.6 Deleted	N/A	*N/A*
1.2.7, 1.2.7a, 1.2.7b, 1.2.7c, 1.2.7d, 1.2.7e, 1.2.7f, 1.2.7g HSM Dose Rates with Various *Loaded* DSCs	5.4 *HSM or HSM-H Dose Rate Evaluation Program*	*B 10.5.4*
1.2.8, 1.2.8a, 1.2.8b, 1.2.8c HSM Maximum Exit Air Temperature with Various Loaded DSCs	3.1.4 HSM Maximum Air Exit Temperature with *a* Loaded DSC	*B 10.3.1.4*
1.2.9 Transfer Cask Alignment with HSM or HSM-H	5.3.3 Transfer Cask Alignment with HSM or HSM-H	*B 10.5.3.3*
1.2.10, 1.2.13, 1.2.14 and 1.2.14a TC/DSC *Handling/*Lifting Heights and Ambient Temperatures for Various DSCs	5.3.1 TC/DSC Lifting/Handling Height Limits	*B 10.5.3.1*
1.2.11, 1.2.11a through e TC Dose Rates with Various *Loaded* DSCs	*5.2.4e*	*B 10.5.2.4e*
1.2.12 Maximum DSC Removable Surface Contamination	5.2.4d	*B 10.5.2.4d*

Table 10-2
Technical Specification
Cross Reference Table between Amendment 10 and Amendment 11
(concluded)

Amendment 10 Tech Spec	Amendment 11 Tech Spec	*Location of Amendment 10 Bases*
1.2.13 See line above for 1.2.10, which includes 1.2.13	—	—
1.2.14 See line above for 1.2.10, which includes 1.2.14 and 14a	—	—
1.2.15, 1.2.15a, 1.2.15b, 1.2.15c, 1.2.15d Boron Concentration in the DSC Cavity Water for Various DSCs	3.2 Cask Criticality Control	*B 10.3.2*
1.2.16 Provision of TC Seismic Restraint Inside the Spent Fuel Pool Building *as a Function of Horizontal Acceleration and Loaded Cask Weight*	4.3.3 Site Specific Parameters and Analyses	*B 10.4.3.3.7*
1.2.17, 1.2.17a, 1.2.17b, 1.2.17c Vacuum Drying Duration Limits for Various DSCs	Deleted due to use of Helium	*N/A*
1.2.18, 1.2.18a, 1.2.18b Time Limit for Completion of 24PTH, 61BTH Type 2 or 32PTH1 DSC Transfer Operations	3.1.3 Time Limit for Completion of TRANSFER OPERATIONS (24PTH, 61BTH Type 2 or 32PTH1 DSC Only)	*B 10.3.1.3*
1.2.19 61BTH and 32PTH1 DSC Bulkwater Removal Medium	*3.1.1 DSC Bulkwater Removal Medium and Vacuum Drying Pressure*	*B 10.3.1.1*
1.3.1 Visual Inspection of HSM *or HSM-H* Air Inlets and Outlets (Front Wall and Roof Birdscreen)	5.2.5a Daily visual inspection of HSM *or HSM-H* Air Inlets and Outlets (front wall and roof bird screens)	*B 10.5.2.5*
1.3.2 HSM *or HSM-H* Thermal Performance	5.2.5b Daily HSM *or HSM-H* Temperature Measurement	*B 10.5.2.5*
From CoC condition 7, concrete testing for HSM-H	5.5 Concrete testing for HSM-H	*N/A*
From CoC condition 8, HSM-H configuration changes	5.6 HSM-H configuration changes	*N/A*
NRC Request: supplement shielding shall be used with OS197L cask	Included in new Section 4.4.1	*N/A*
NRC Request: modify TN's proposed wording on "Contingency Planning" for abnormal events, eliminate terms contingency planning, abnormal events, high dose rates	Added to Section 5.2.4 "Radiation Protection Program"	*N/A*
NRC Request: include a requirement for user to perform dose assessment ahead of time and augment Part 20 program and address recovery from a potential malfunction of a remote handling device	Added to Section 5.2.4 "Radiation Protection Program" and also modified Appendix W.10 Occupational Exposure Section to include exposure due to recovery operations from a potential malfunction of a remote handling device (Crane failure)	*N/A*
NRC Request: include the requirement of dose assessment for cases when Transfer cask requires use of remote operations.	Added to Section 5.2.4 "Radiation Protection Program"	*N/A*

As discussed on page 10-1, with Amendment 11 to CoC 1004, the Technical Specifications (TS) are being converted to the NUREG-1745 format and the TS bases are being returned to this chapter. The numbering scheme for the TS changed a great deal as the TS were converted to the NUREG-1745 format. Therefore there is not a documented basis for each TS and therefore the numbering scheme of this chapter, as reflected in the table of contents below, is not comprehensive.

TABLE OF CONTENTS

The physical parameters that define the mechanical and structural design of the HSM and DSC are the fuel assembly dimensions and weight. The calculated stresses given in the FSAR are based on the physical parameters given in Tables 1-1a, 1-1b, 1-1c, 1-1d, 1-1e, 1-1f, 1-1g, 1-1i, 1-1j, 1-1l, 1-1m, 1-1t, 1-1u, 1-1aa and 1-1bb, which represent the upper bound.

The design basis fuel assemblies for nuclear criticality safety are Babcock and Wilcox 15x15 fuel assemblies for the NUHOMS®-24P and 24PHB, General Electric 7x7 fuel assemblies for the NUHOMS®-52B and General Electric 10x10 fuel assemblies for the NUHOMS®-61BT and 61BTH designs. The nuclear criticality safety for the NUHOMS®-32PT, NUHOMS®-24PTH and NUHOMS® 32PTH1 designs is based on an evaluation of individual fuel assembly class as listed in Table 1-1e, Table 1-1l and Table 1-1aa, respectively.

The NUHOMS®-24P Long Cavity DSC is designed for use with standard Burnable Poison Rod Assembly (BPRA) designs for the B&W 15x15 and Westinghouse 17x17 fuel types as listed in Appendix J of the FSAR. The NUHOMS®-24PHB Long Cavity DSC is designed for use with standard BPRA designs for the B&W 15x15 fuel types listed in Appendix N of the FSAR.

The design basis PWR BPRA for shielding source terms and thermal decay heat load is the Westinghouse 17x17 Pyrex Burnable Absorber, while the DSC internal pressure analysis is limited by B&W 15x15 BPRAs. In addition, BPRAs with cladding failures were determined to be acceptable for loading into NUHOMS®-24P Long Cavity DSC as evaluated in Appendix J of the FSAR. The acceptability of loading BPRAs, including damaged BPRAs into the long cavity versions of the 32PT and 24PTH DSC configurations is provided in Appendix M and Appendix P respectively of the FSAR.

Control Components (CCs), as listed in Table 1-1e, Table 1-1l and Table 1-1aa are authorized for storage in the NUHOMS®-32PT DSC, NUHOMS®-24PTH DSC and NUHOMS®-32PTH1 DSCs, respectively. For these DSCs, BPRAs are considered as being representative of all CCs, unless specifically excluded. The acceptability of loading CCs into the NUHOMS®-32PT, NUHOMS®-24PTH and NUHOMS®-32PTH1 DSCs is provided in Appendix M, P and U of the FSAR, respectively.

The boron concentration of the spent fuel fool water and the water added to the cavity of the loaded 24P, 32PT, 24PTH or 32PTH1 DSC for criticality control is defined in Technical Specification 3.2. Fixed poison is required for criticality control for the 61BT, 61BTH, 32PT, 24PTH and the 32PTH1 DSCs. Three alternate poison materials are allowed: (a) borated aluminum alloy, or (b) a boron carbide/aluminum metal matrix composite (MMC), or (c) Boral®.

The NUHOMS®-24P is designed for unirradiated fuel with an initial fuel enrichment of up to 4.0 wt. % U-235, taking credit for soluble boron in the DSC cavity water during loading operations. In addition, the fuel assemblies qualified for storage in NUHOMS®-24P DSC have an equivalent unirradiated enrichment of less than or equal to 1.45 wt. % U-235. Figure 1-1 defines the required burnup as a function of initial enrichment. The NUHOMS®-52B is designed for unirradiated fuel with an initial enrichment of less than or equal to 4.0 wt. % U-235.

The NUHOMS®-61BT has three basket configurations, based on the boron content in the poison plates as listed in Table 1-1k. The maximum lattice average enrichment *authorized for Type A, B and C NUHOMS®-61BT DSC is 3.7, 4.1 and 4.4 wt. % U-235 respectively.*

For the 61BT DSC, borated aluminum, MMC, or Boral® shall be supplied in accordance with UFSAR Sections K.9.1.7.1, K.9.1.7.2, K.9.1.7.3, K.9.1.7.5, K.9.1.7.6.5, and K.9.1.7.7.3, with the minimum B10 areal density specified in Table 1-1k. These sections of the UFSAR are hereby incorporated into the NUHOMS® 1004 CoC.

The NUHOMS®-61BTH DSC is designed for unirradiated fuel with a maximum lattice average enrichment of 5.0 wt. % U-235 as shown in Table 1-1t, taking credit for the boron content in the poison plates of the DSC basket, as shown in Table 1-1v for intact fuel and Table 1-1w for damaged fuel. The NUHOMS®-61BTH DSC (similar to 61BT DSC) is designated as Type 1 and Type 2 depending upon the rails used in the basket.

Each 61BTH DSC type is provided with six alternate basket configurations, based on the boron content in the poison plates, as listed in Table 1-1v or Table 1-1w (designated as "A" for the lowest B10 loading to "F" for the highest B10 loading).

For the 61BTH DSC, borated aluminum, MMC, or Boral® shall be supplied in accordance with UFSAR Sections T.9.1.7.1, T.9.1.7.2, T.9.1.7.3, T.9.1.7.5, T.9.1.7.6.5, and T.9.1.7.7.3, with the minimum B10 areal density specified in Table 1-1v or Table 1-1w. These sections of the UFSAR are hereby incorporated into the NUHOMS® 1004 CoC.

The NUHOMS®-32PT is designed for unirradiated fuel with an initial fuel enrichment of up to 5.0 wt. % U-235 as shown in Table 1-1g, taking credit for Poison Rod Assemblies (PRAs), poison plates, and soluble boron in the DSC cavity water during loading operations. The required PRA locations are per Figures 1-5, or 1-6 or 1-7. A 32PT DSC basket may contain 0, 4, 8 or 16 PRAs and is designated a Type A, Type B, Type C or Type D basket, respectively. Each basket type is designed with up to three alternate configurations depending on the configuration of poison plates provided (16, 20 or 24) as shown in Table 1-1g. Table 1-1h specifies the minimum B10 content for poison plates.

For the 32PT DSC, borated aluminum or MMC shall be supplied in accordance with UFSAR Sections M.9.1.7.1, M.9.1.7.2, M.9.1.7.5, M.9.1.7.6.5, and M.9.1.7.7.3, with the minimum B10 areal density specified in Table 1-1h. These sections of the UFSAR are hereby incorporated into the NUHOMS® 1004 CoC.

The NUHOMS®-24PHB is designed for unirradiated fuel with an assembly average initial enrichment of less than or equal to 4.5 wt. % U-235 as shown in Table 1-1i, taking credit for soluble boron in the DSC cavity water during loading operations.

The NUHOMS®-24PTH is designed for unirradiated fuel with an assembly average initial enrichment of less than or equal to 5.0 wt. % U-235, as shown in Table 1-1l, taking credit for soluble boron in the DSC cavity water during loading operations and the boron content in the poison plates of the DSC basket, as shown in Table 1-1p for intact fuel and Table 1-1q for damaged fuel. The 24PTH DSC basket is designated as Type 1, if it is provided with aluminum inserts and Type 2 if it does not contain the aluminum inserts. Each basket type is designed with three alternate configurations, based on the boron content in the poison plates, as listed in Table 1-1r.

For the 24PTH DSC, borated aluminum, MMC, or Boral® shall be supplied in accordance with UFSAR Sections P.9.1.7.1, P.9.1.7.2, P.9.1.7.3, P.9.1.7.5, P.9.1.7.6.5, and P.9.1.7.7.3, with the *minimum B10 areal density specified in Table 1-1r. These sections of the UFSAR are hereby incorporated into the NUHOMS® 1004 CoC.*

The NUHOMS®-32PTH1 is designed for unirradiated fuel with an assembly average initial enrichment of less than or equal to 5.0 wt. % U-235, as shown in Table 1-1aa, taking credit for soluble boron in the DSC cavity water during loading operations and the boron content in the poison plates of the DSC basket, as shown in Table 1-1cc for intact fuel and Table 1-1dd for damaged fuel. The 32PTH1 DSC basket is designated as Type 1 or Type 2, depending upon the rails used in the basket. Each basket type is designed with five alternate configurations, based on the boron content in the poison plates, as listed in Table 1-1ff.

For the 32PTH1 DSC, borated aluminum, MMC, or Boral® shall be supplied in accordance with UFSAR Sections U.9.1.7.1, U.9.1.7.2, U.9.1.7.3, U.9.1.7.5, U.9.1.7.6.5, and U.9.1.7.7.3, with the minimum B10 areal density specified in Table 1-1ff. These sections of the UFSAR are hereby incorporated into the NUHOMS® 1004 CoC.

The thermal design criterion of the fuel to be stored is that the total maximum heat generation rate per assembly and BPRA or control components be such that the fuel cladding temperature is maintained within established limits during normal and off-normal conditions. For the NUHOMS®-24P, 52B and 61BT systems, fuel cladding temperature limits were established based on methodology in PNL-6189 and PNL-4835. For the NUHOMS®-32PT, 24PHB and 24PTH systems, fuel cladding limits are based on ISG-11, Rev. 2 (Reference 3). For the NUHOMS®-61BTH system, NUHOMS®-61BT system with Framatome-ANP 9x9 Version 9x9-2 (FANP9 9x9-2) fuel assemblies, and the NUHOMS®-32PTH1 system, fuel cladding limits are based on ISG-11, Rev. 3 (Reference 4).

The radiological design criterion is that fuel stored in the NUHOMS® system must not increase the average calculated HSM or transfer cask dose rates beyond those calculated for the 24P, 24PHB, 52B, 61BT, 32PT, 24PTH, 61BTH, or 32PTH1 canister full of design basis fuel assemblies with or without BPRAs.

Technical Specification Table 1-1a, Table 1-1b, Table 1-1c, Table 1-1j, Table 1-1e, Table 1-1i, Table 1-1l, Table 1-1t and Table 1-1aa provide the key fuel parameters that require confirmation prior to loading fuel assemblies within a specific standardized DSC model. Each of these Technical Specification Tables lists additional Technical Specification Tables and Figures which provide requirements which also must be met prior to loading.

B 10.4.3.3.7 Seismic Restraints

BASES

For the Provision of TC Seismic Restraint Inside the Spent Fuel Pool Building as a Function of Horizontal Acceleration and Loaded Cask Weight, the basis is the calculation of overturning and restoring moments.

B 10.5.2.4 RADIATION PROTECTION PROGRAM

B 10.5.2.4a ALARA Assessment

BASES

The basis for the ALARA assessment is 10 CFR 72.212 (b)(2)(i)(C).

B 10.5.2.4b DSC Dye Penetrant Test of Closure Welds

BASES

Article NB-5000 Examination, ASME Boiler and Pressure Vessel Code, Section III, Division 1, Sub-Section NB.

B 10.5.2.4c Leak Test

BASES

If the DSC leaked at the maximum acceptable rate of 1.0×10^{-4} *atm · cm*^{3}*/s for a period of 20 years, about 63,100 cc of helium would escape from the DSC. This is about 1% of the* 6.3×10^6 *cm*^{3} *of helium initially introduced in the DSC. This amount of leakage would have a negligible effect on the inert environment of the DSC cavity. (Reference: American National Standards Institute, ANSI N14.5-1987, For Radioactive Materials—Leakage Tests on Packages for Shipment," Appendix B3).*

The 61BT, 32PT, 24PHB, 24PTH, 61BTH and 32PTH1 DSC will maintain an inert atmosphere around the fuel and radiological consequences will be negligible, since it is designed and tested to be leak tight.

B 10.5.2.4d Maximum DSC Removable Surface Contamination

BASES

This non-fixed contamination level is consistent with the requirements of 10 CFR 71.87(i)(1) and 49 CFR 173.443, which regulate the use of spent fuel shipping containers. Consequently, these contamination levels are considered acceptable for exposure to the general environment. This level will also ensure that contamination levels of the inner surfaces of the HSM and potential releases of radioactive material to the environment are minimized.

The use of an inflatable seal in the upper cask liner recess ensures that the TC/DSC annulus area below the seal remains clean during the subsequent fuel loading steps inside the spent fuel pool. Hence, the only area which needs to be checked for contamination is the top 1 foot of the DSC external surface. However, in the unlikely event that contamination is found that exceeds the specified levels, the entire length of the DSC surface needs to be decontaminated.

B 10.5.2.4e Transfer Cask Dose Rates

BASES

These dose rates are based on the shielding analysis for the various DSCs included in the UFSAR Chapter 7 and Appendices J, K, M, N, P, T, U, and W, with some added margin for uncertainty. The cross references to these appendices are shown in the following table.

DSC	Transfer Cask Axial Surface Dose Rate Configuration in UFSAR	Transfer Cask Radial Surface Dose Rate Configuration in UFSAR
24P	Table J.5-3, Shield Plug	Table 7.3-2, TRANSFER
52B	Bounded by 24P	Bounded by 24P
61BT	Inner Cover Welding, Figure K.5-14	Table K.5-2, TRANSFER
32PT	Inner Cover Welding, Figure M.5-27	Table M.5-5, TRANSFER
24PHB	Inner Cover Welding, Figure N.5-14 (neutron); Figure N.5-17 (gamma)	Table N.5-3, TRANSFER
24PTH(1)	Welding, Table P.5-4	Table P.5-3, TRANSFER
24PTH-S-LC	Bounded by 24PTH	Table P.5-5, TRANSFER
61BTH	Welding, Table T.5-5	Table T.5-4, TRANSFER
32PTH1	Welding, Table U.5-3	Table U.5-2, TRANSFER
61BT(2)	Inner Cover Welding, Figure K.5-14	Figure W.5-2 (Maximum)
32PT(2)	Inner Cover Welding, Figure M.5-27	Figure W.5-2 (Maximum)

(1) Does not apply to the 24PTH-S-LC
(2) Applicable only to the OS197L TC

For Visual Inspection of HSM or HSM-H Air Inlets and Outlets (Front Wall and Roof Birdscreen), the concrete temperature could exceed 350 °F in the accident circumstances of complete blockage of all vents. Concrete temperatures over 350 °F in accidents (without the presence of water or steam) can have uncertain impact on concrete strength and durability. A conservative analysis (adiabatic heat case) of complete blockage of all air inlets or outlets indicates that the concrete can reach the accident temperature limit of 350 °F in the time periods specified for HSM. For HSM-H, the time period specified ensures that blockage will not exist for periods longer than that assumed in the safety analysis presented in Appendix P, Appendix T and Appendix U of the FSAR. At the analyzed time limit, the fuel cladding temperature remains well below the accident limit of 1058°F.

For HSM or HSM-H Thermal Performance, the temperature measurement should be of sufficient scope to provide the licensee with a positive means to identify conditions which threaten to approach temperature criteria for proper HSM or HSM-H operation and allow for the correction of off-normal thermal conditions that could lend to exceeding the concrete and fuel clad temperature criteria.

B 10.5.2.6 Hydrogen Gas Monitoring for Specified DSCs

BASES

The basis for this technical specification is the safety concern of keeping the combustible mixture concentration below flammability limits while welding.

B 10.5.3 Cask TRANSFER Controls

B 10.5.3.1 TC/DSC Lifting/Handling Height Limits

BASES

For the TC/DSC Handling Height Outside the Spent Fuel Pool Building, the NRC evaluation of the TC/DSC drop analysis concurred that drops up to 80 inches, of the DSC inside the TC, can be sustained without breaching the confinement boundary, preventing removal of spent fuel assemblies, or causing a criticality accident. This specification ensures that handling height limits will not be exceeded in transit to, or at the storage pad.

Acceptable damage may occur to the TC, DSC, and the fuel stored in the DSC, for drops of height greater than 15 inches. The specification requiring inspection of the DSC and fuel following a drop of 15 inches or greater ensures that the spent fuel will continue to meet the requirements for storage, the DSC will continue to provide confinement, and the TC will continue to provide its design functions of DSC transfer and shielding.

For the TC/DSC Lifting Heights as a Function of Low Temperature and Location, the basis for the low temperature and height limits is ANSI N14.6-1986 paragraph 4.2.6 which requires at least 40°F higher service temperature than nil ductility transition (NDT) temperature for the TC. In the case of the standardized TC, the test temperature is -40°F; therefore, although the NDT temperature is not determined, the material will have the required 40°F margin if the ambient temperature is 0°F or higher. This assumes the material service temperature is equal to the ambient temperature.

The basis for the low temperature limit for the DSC is NUREG/CR-1815. The basis for the handling height limits is the NRC evaluation of the structural integrity of the DSC to drop heights of 80 inches and less.

For the NUHOMS®-24P, 52B and 61BT systems, the basis for the high temperature limit is PNL-6189 (Reference 1) for the fuel clad limit, the manufacturer's specification for neutron shield, and the design basis pressure of the TC internal cavity pressure. For the NUHOMS®-32PT, 24PHB and 24PTH systems, the fuel cladding limits are based on ISG-11, Revision 2 (Reference 3). For the NUHOMS®-61BTH system and the NUHOMS®-61BT system with FANP 9x9-2 fuel assemblies, the fuel cladding limits are based on ISG-11 Revision 3 (Reference 4).

For the TC/DSC Transfer at High Ambient Temperatures (32PTH1 DSC Only), the fuel cladding limits are based on ISG-11 Revision 3 (Reference 4).

B 10.5.3.2 Cask Drop

BASES

The basis for this specification is Section 8.2.5, "Accidental Cask Drop."

B 10.5.3.3 TRANSFER CASK Alignment with HSM or HSM-H

BASES

The basis for the true position alignment tolerance is the clearance between the DSC shell, the transfer cask cavity, the HSM or HSM-H access opening, and the DSC support rails inside the HSM or HSM-H.

B 10.5.3.4 Supplemental Shielding Drop onto OS197L TC

BASES

The bases for the consequences of an accidental drop of the outer trailer shielding onto the inner trailer shielding of the OS197L TC are provided in Section W.11.1.5 of the FSAR. The lifting of the outer shielding, is restricted such that the bottom most part of the body of this shield is less than 4" above the inner shielding.

B 10.5.4 HSM or HSM-H Dose Rate Evaluation Program

BASES

The specified dose rates provide as-low-as-is-reasonably- achievable on-site and off-site doses in accordance with 10 CFR Part 20 and 10 CFR 72.104(a).

These dose rates are based on the shielding analysis for the various DSCs included in the UFSAR Chapter 7 and Appendices J, K, M, N, P, T, U, and W, with some added margin for uncertainty.

The cross references to these appendices are shown in the following table.

DSC	***HSM***	***HSM Door Outer Surface Dose Rate Configuration in UFSAR***	***HSM Front Surface Dose Rate Configuration in UFSAR***	***HSM Front Bird Screen Surface Dose Rate Configuration in UFSAR***
24P	*Standardized HSM*	*Table 7.3-2*	*Table 7.3-2*	*Table 7.3-2*
52B	*Standardized HSM*	*Bounded by 24P*	*Bounded by 24P*	*Bounded by 24P*
61BT	*Standardized HSM*	*Table K.5-2*	*Table K.5-3*	*Table K.5-2*
32PT	*Standardized HSM*	*Table M.5-3*	*Table M.5-4*	*Table M.5-3*
24PHB	*Standardized HSM*	*Table N.5-3*	*Table N.5-4*	*Table N.5-4*
24PTH-S-LC	*Standardized HSM*	*Table P.5-2*	*Table P.5-2*	*Table P.5-2*
61BTH	*Standardized HSM*	*Table T.5-3*	*Table T.5-3*	*Table T.5-3*
24PTH	*HSM-H*	*Table P.5-1*	*Table P.5-1*	*Table P.5-1*
61BTH	*HSM-H*	*Table T.5-1*	*Table T.5-1*	*Table T.5-1*
32PTH1	*HSM-H*	*Table U.5-1*	*Table U.5-1*	*Table U.5-1*

Table K.2-2
Damaged BWR Fuel Assemblies Characteristics

(Concluded)

<u>RADIOLOGICAL PARAMETERS[(1)]:</u>	
Group 4: Maximum Burnup:	40,000 MWd/MTU
Minimum Cooling Time:	10-years
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Bundle Average Enrichment:	3.4 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
<u>ALTERNATE RADIOLOGICAL PARAMETERS:</u>	
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Fuel Burnup, Initial Bundle Average Enrichment, and Cooling Time:	Per Table K.2-11*, except that for a 61BT DSC contained in an OS197L TC, see Tables W.2-4 and W.2-5, and Figure W.2-1*
Maximum Pellet Enrichment:	4.4 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly

(1) Fuel assemblies fully complying with any of the four groups of parameters or alternate radiological parameters are suitable for storage in the NUHOMS®-61BT DSC. No interpolation of Radiological Parameters is permitted between groups 1 to 4.

Table K.3.1-2

ASME Code Alternatives for the NUHOMS®-61BT DSC Confinement Boundary

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section K.2 may be used for construction, but in no case earlier than 3 years before that specified in Section K.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section K.2 may be used, so long the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NB-1100	*Requirements for Code Stamping of Components, Code reports and certificates, etc.*	*Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.*
NB-1132	Attachments with a pressure retaining function, including stiffeners, shall be considered part of the component.	Bottom shield plug and outer bottom cover plate are outside code jurisdiction; these components together are much larger than required to provide stiffening for the inner bottom cover plate; the weld that retains the outer bottom cover plate and with it the bottom shield plug is subject to root and final PT examination.
NB-2130	*Material must be supplied by ASME approved material suppliers.*	*Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.*
NB-4121	*Material Certification by Certificate Holder*	
NB-4243 and NB-5230	Category C weld joints in vessels and similar weld joints in other components shall be full penetration joints. These welds shall be examined by UT or RT and either PT or MT	The joints between the top outer and inner cover plates and containment shell are designed and fabricated per ASME Code Case N-595-1. This includes the inner top cover plate weld around the vent and siphon block. The welds are partial penetration welds and the root and final layer are PT examined. The weld between the vent and siphon block and the shell is made at the fabricator's shop and receives a final PT examination.
NB-6100 and 6200	All completed pressure retaining systems shall be pressure tested	The vent and siphon block is not pressure tested due to the manufacturing sequence. The siphon block weld is helium leak tested when fuel is loaded and then covered with the outer top closure plate.
NB-7000	*Overpressure Protection*	*No overpressure protection is provided for the NUHOMS® DSCs. The function of the DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The DSC is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature.*

Table K.3.1-2

ASME Code Alternatives for the NUHOMS®-61BT DSC Confinement Boundary

Reference ASME Code Section/Article	***Code Requirement***	***Alternatives, Justification & Compensatory Measures***
NB -8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.*
NB-5520	*NDE Personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of more recent edition of SNT-TC-1A.*

Table K.3.1-3
ASME Code Alternatives for the NUHOMS®-61BT DSC Basket

Reference ASME Code Section/Article	*Code Requirement*	*Alternatives, Justification & Compensatory Measures*
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section K.2 may be used for construction, but in no case earlier than 3 years before that specified in Section K.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section K.2 may be used, so long the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NG/NF-1100	*Requirements for Code Stamping of Components, Code reports and certificates, etc.*	*Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.*
NG/NF-2000	*Use of ASME Code Material*	*Some baskets include neutron absorber and aluminum plates that are not ASME Code Class 1 material. They are used for criticality safety and heat transfer, and are only credited in the structural analysis with supporting their own weight and transmitting bearing loads through their thickness.*
NG/NF-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of more recent edition of SNT-TC-1A.*

K.4.7 Thermal Evaluation for Loading/Unloading Conditions

All fuel transfer operations occur when the NUHOMS®-61BT DSC/transfer cask is in the spent fuel pool. The fuel is always submerged in free-flowing pool water permitting heat dissipation. After fuel loading is complete, the Cask/DSC is removed from the pool, drained, dried, backfilled with helium and sealed.

K.4.7.1 Vacuum Drying Analysis

The loading condition evaluated for determining the maximum fuel cladding temperature is the heatup of the NUHOMS® 61BT DSC before its cavity can be backfilled with helium. This typically occurs during the performance of the vacuum drying operation of the DSC cavity with the cask in the vertical orientation in the fuel handling building, and the annulus between the cask and the DSC full of water. During vacuum drying operation water in the DSC cavity is forced out of the cavity (blowdown operation) before the start of vacuum drying. Helium is used as the medium to remove water and subsequent vacuum drying occurs with a helium environment in the DSC cavity. The vacuum drying of the DSC generally does not reduce the pressure sufficiently to reduce the thermal conductivity of the *water vapor and helium* in the DSC cavity *[4.2] and [*4.1*6]. Due to the use of helium for blowdown and presence of water in the TC/DSC annulus, the results of the vacuum drying analysis are bounded by the off-normal thermal analysis for transfer as summarized in* Table K.4-4.

K.4.7.1.1 Reflooding Evaluation

For unloading operations, the DSC will be filled with the spent fuel pool water through the siphon port. During this filling operation, the DSC vent port is maintained open with effluents

K.4.9 References

4.1 Levy et al., Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy - Clad Fuel Rods in Inert Gas, Pacific Northwest Laboratory, PNL-6189, 1987.

4.2 Rohsenow et al., Handbook of Heat Transfer Fundamentals, McGraw-Hill Publishing, New York 1985.

4.3 American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section II, Part D, 1998 Edition Including 1999 Addenda.

4.4 Scoping Design Analyses for Optimized Shipping Casks Containing 1-, 2-, 3-, 5-, 7-, or 10-Year Old PWR Spent Fuel, J. A. Bucholz, ORNL/CSD/TM-149, TTC-9316, January 1983.

4.5 ANSYS, Inc., ANSYS Engineering Analysis System User's Manual for ANSYS Revision 5.6, Houston, PA.

4.6 Standard Review Plan for Dry Cask Storage Systems, NUREG-1536, Nuclear Regulatory Commission.

4.7 Transnuclear, Inc., TN-68 Dry Storage Cask Final Safety Analysis Report, Revision 0, Hawthorne, NY, 2000 (Docket No. 72-1027).

4.8 Transnuclear West, Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUH-003 Revision 5, NUH003.0103, Fremont, CA, August 2000.

4.9 Johnson et al., Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases, PNL-4835, Pacific Northwest Laboratory, 1983.

4.10 Consolidated Safety Analysis Report for IF-300 Shipping Cask, CoC 9001.

4.11 Calculation, "TN-24P Benchmarking Analysis Using ANSYS," Transnuclear Calculation No. NUH32PT.0408, Revision 0.

4.12 J. M. Greer, et al, "The TN-24 PWR Spent Fuel Storage Cask: Testing and Analyses," PNL Report No. PNL-6054, Pacific Northwest Laboratory, 1987.

4.13 Application for Amendment No. 5 to the NUHOMS® Certificate of Compliance No. 1004 (TAC No. L23343, Docket No. 72-1004); February 24, 2003.

4.14 NUREG/CR-0497, A Handbook of Material Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, MATPRO-version 11, Revision 2, EG&G Idaho, Inc., TREE-1280, August 1981.

4.15 Interim Staff Guidance (ISG) – 11, Revision 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel," NRC Spent Fuel Project Office, dated November 17, 2003.

4.16 David R. Lide, CRC Handbook of Chemistry and Physics, 83rd Edition, 2002-2003, CRC Press.

Table K.6-1
Minimum B10 Content in the Neutron Poison Plates

Maximum Lattice Average Enrichment (wt% U-235)	Minimum B10 Content Boral® or Metamic® (g/cm^2)	Minimum B10 Content Boron-Aluminum Alloy or Boralyn® (g/cm^2)	B10 Content Used in Criticality Evaluation[(1)] (g/cm^2)
3.7	0.025	0.021	0.019
4.1	0.038	0.032	0.029
4.4	0.048	0.040	0.036
For *Damaged* Fuel			
4.0[(2)]	0.048	0.040	0.036

(1) 90% B10 credit for Boron-Aluminum alloy or Boralyn®. 75% B10 credit for Boral® or Metamic®.

(2) Maximum Peak Pellet Enrichment 4.4 wt% U-235.

10. Fill the DSC cavity with water from the fuel pool or an equivalent source which meets the requirements of Technical Specification *3.2.1*. For BWR fuel, demineralized water may be used.

    Note: A TC/DSC annulus pressurization tank filled with water from the fuel pool as described above is connected to the top vent port of the TC via a hose to provide a positive head above the level of water in the TC/DSC annulus. This is an optional arrangement, which provides additional assurance that contaminated water from the fuel pool will not enter the TC/DSC annulus, provided a positive head is maintained at all times.

11. Using the fuel/reactor building main hook and the cask lifting yoke, position the cask lifting yoke above the DSC top shield plug and attach the four designated cable assemblies between the yoke and the DSC top shield plug. Adjust the turnbuckles on the cable assemblies as necessary to level the shield plug. If not already done, test fit the DSC top shield plug onto the DSC.

12. Remove the DSC top shield plug and disconnect it from the yoke. Position the cask lifting yoke above the transfer cask and engage the cask lifting trunnions.

13. Visually inspect the yoke lifting hooks to insure that they are properly positioned and engaged on the cask lifting trunnions.

14. Visually inspect the yoke lifting hooks to insure that they are properly positioned and engaged on the cask lifting trunnions.

15. Connect the vacuum drying system (VDS) or optional liquid pump to the siphon port of the DSC and position the connecting hose such that the hose will not interfere with loading (yoke, fuel, shield plug, rigging, etc.). A rotometer must be installed at a suitable location as part of this connection.

16. Move the scaffolding away from the cask as necessary.

17. Lift the cask just far enough to allow the weight of the cask to be distributed onto the yoke lifting hooks. Reinspect the lifting hooks to insure that they are properly positioned on the cask trunnions.

18. Optionally, secure a sheet of suitable material to the bottom of the transfer cask to minimize the potential for ground-in contamination. This may also be done prior to initial placement of the cask in the decon area.

24. Pressurize the DSC with helium to about 24 psia not to exceed 34 psia.

25. Helium leak test the inner top cover plate weld for leakage in accordance with ANSI N14.5 to a sensitivity of 1×10^{-5} atm$\cdot$cm^3/s. This test is optional.

26. If a leak is found, repair the weld, repressurize the DSC and repeat the helium leak test.

27. Once no leaks are detected, depressurize the DSC cavity by releasing the helium through the VDS to the plant's spent fuel pool or radioactive waste system.

28. Re-evacuate the DSC cavity using the VDS. The cavity pressure should be reduced in steps of approximately 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level, the pump is valved off and the cavity pressure is monitored. When the cavity pressure stabilizes, the pump is valved in to continue the vacuum drying process. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less in accordance with Technical Specification 3.1.1 limits.

29. Open the valve on the vent port and allow helium to flow into the DSC cavity to pressurize the DSC to about 17.2 psia *(2.5 psig)* in accordance with Technical Specification 3.1.2.b limits.

30. Close the valves on the helium source.

31. Remove the Strongback, decontaminate as necessary, and store.

K.8.1.4 DSC Sealing Operations

1. Disconnect the VDS from the DSC. Seal weld the prefabricated plugs over the vent and siphon ports and perform a dye penetrant weld examination in accordance with the Technical Specification 5.2.4.b requirements.

2. Install the automatic welding machine onto the outer top cover plate and place the outer top cover plate with the automatic welding system onto the DSC. Verify proper fit up of the outer top cover plate with the DSC shell.

3. Tack weld the outer top cover plate to the DSC shell. Place the outer top cover plate weld root pass.

4. Helium leak test the inner top cover plate and vent/siphon port plate welds using the leak test port in the outer top cover plate in accordance with Technical Specification 5.2.4.c limits.

5. If a leak is found, remove the outer cover plate root pass, the vent and siphon port plugs and repair the inner cover plate welds. Then install the Strongback and repeat procedure steps from K.8.1.3 step 22.

6. Perform dye penetrant examination of the root pass weld. Weld out the outer top cover plate to the DSC shell and perform dye penetrant examination on the weld surface in accordance with the Technical Specification 5.2.4.b requirements.

7. Seal weld the prefabricated plug over the outer cover plate test port and perform dye penetrant weld examinations *in accordance with the Technical Specification 5.2.4b requirements*.

8. Remove the automatic welding machine from the DSC. Rig the cask top cover plate and lower the cover plate onto the transfer cask.

9. Bolt the cask cover plate into place, tightening the bolts to the required torque in a star pattern.

K.8.1.5 Transfer Cask Downending and *Transfer* to ISFSI

NOTE:

Alternate Procedure for Downending of Transfer Cask: Some plants have limited floor hatch openings above the cask/trailer/skid, which limit crane travel (within the hatch opening) that would be needed in order to downend the TC with the trailer/skid in a stationary position. For these situations, alternate procedures are to be developed on a plant-specific basis, with detailed steps for downending.

1. Drain the neutron shield to an acceptable location.

2. Re-attach the transfer cask lifting yoke to the crane hook, as necessary. Ready the *transfer* trailer and cask support skid for service.

3. Move the scaffolding away from the cask as necessary. Engage the lifting yoke and lift the cask over the cask support skid on the *transfer* trailer.

4. The *transfer* trailer should be positioned so that cask support skid is accessible to the crane with the trailer supported on the vertical jacks.

5. Position the cask lower trunnions onto the transfer trailer support skid pillow blocks.

6. Move the crane forward while simultaneously lowering the cask until the cask upper trunnions are just above the support skid upper trunnion pillow blocks.

7. Inspect the positioning of the cask to ensure that the cask and trunnion pillow blocks are properly aligned.

8. Lower the cask onto the skid until the weight of the cask is distributed to the trunnion pillow blocks.

9. Inspect the trunnions to ensure that they are properly seated onto the skid and install the trunnion tower closure plates (optional for the OS197 TC and the OS197H TC).

10. Fill the neutron shield.

11. Remove the bottom ram access cover plate from the cask. Install the two-piece temporary neutron/gamma shield plug to cover the bottom ram access. Install the ram trunnion support frame on the bottom of the transfer cask. (The temporary shield plug and ram trunnion support frame are not required with integral ram/trailer.)

K.8.1.6 DSC Transfer to the HSM

1. Prior to *transferring* the cask to the ISFSI or prior to positioning the transfer cask at the HSM designated for storage, remove the HSM door, inspect the cavity of the HSM, removing any debris and ready the HSM to receive a DSC. The doors on adjacent HSMs should remain in place.

   Caution: The insides of empty modules have the potential for high dose rates due to adjacent loaded modules. Proper ALARA practices should be followed for operations inside these modules and in the areas outside these modules whenever the door from the empty HSM has been removed.

2. Inspect the HSM air inlet and outlets to ensure that they are clear of debris. Inspect the screens on the air inlet and outlets for damage.

   CAUTION: Verify that the requirements of Technical Specification 5.3.1.B, "TC/DSC Transfer Operations at High Ambient Temperatures" are met prior to next step.

3. Using a suitable heavy haul tractor, *transfer* the cask from the plant's fuel/reactor building to the ISFSI along the designated transfer route.

4. Once at the ISFSI, position the *transfer* trailer to within a few feet of the HSM.

5. Check the position of the trailer to ensure the centerline of the HSM and cask approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.

6. Unbolt and remove the cask top cover plate.

7. Back the cask to within a few inches of the HSM, set the trailer brakes and disengage the tractor. Drive the tractor clear of the trailer. Extend the transfer trailer vertical jacks.

8. Connect the skid positioning system hydraulic power unit to the positioning system via the hose connector panel on the trailer, and power it up. Remove the skid tie-down bolts and use the skid positioning system to bring the cask into approximate vertical and horizontal alignment with the HSM. Using optical survey equipment and the alignment marks on the cask and the HSM, adjust the position of the cask until it is properly aligned with the HSM.

9. Using the skid positioning system, fully insert the cask into the HSM access opening docking collar.

10. Secure the cask trunnions to the front wall embedments of the HSM using the cask restraints.

11. After the cask is docked with the HSM, verify the alignment of the transfer cask using the optical survey equipment.

12. Position the hydraulic ram behind the cask in approximate horizontal alignment with the cask and level the ram. Remove either the bottom ram access cover plate or the outer plug of the two-piece temporary shield plug. Power up the ram hydraulic power supply and extend the ram through the bottom cask opening into the DSC grapple ring.

13. Activate the hydraulic cylinder on the ram grapple and engage the grapple arms with the DSC grapple ring.

14. Recheck all alignment marks in accordance with the Technical Specification 5.3.3 limits and ready all systems for DSC transfer.

15. Activate the hydraulic ram to initiate insertion of the DSC into the HSM. Stop the ram when the DSC reaches the support rail stops at the back of the module.

16. Disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.

17. Retract and disengage the hydraulic ram system from the cask and move it clear of the cask. Remove the cask restraints from the HSM.

18. Using the skid positioning system, disengage the cask from the HSM access opening. Insert the inner tube of the DSC axial retainer.

19. The trailer may be moved as necessary to install the HSM door. Install the HSM door and secure it in place. Verify that the loaded HSM meets the dose rate limits of Technical Specification 5.4.2.

20. Replace the transfer cask top cover plate. Secure the skid to the trailer, retract the vertical jacks and disconnect the skid positioning system.

K.9.1.4 Components

No change.

K.9.1.5 Shielding Integrity

No change.

K.9.1.6 Thermal Acceptance

The analyses to ensure that the NUHOMS®-61BT DSCs are capable of performing their heat transfer function are presented in Section K.4.

K.9.1.7 Poison Acceptance

CAUTION
Sections K.9.1.7.1 through K.9.1.7.3 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1(Note 1) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated aluminum

(b) Boron carbide/aluminum metal matrix composite (MMC)

(c) Boral®

The 61BT DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content of these three types of materials is given in Table K.9-1, Table K.9-2 and Table K.9-3, respectively.

References to metal matrix composites throughout this chapter are not intended to refer to Boral®, which is described later in this section.

K.9.1.7.1 **Borated Aluminum**

See the Caution in Section K.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating *primarily* as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy *(other boron compounds, such as AlB_{12}, can also*

occur). For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section K.9.1.7.5. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings"[9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surface, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

K.9.1.7.2 Boron Carbide / Aluminum Metal Matrix Composites (MMC)

See the Caution in Section K.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

Boron carbide particles for the products considered here shall have an average size of 40 microns or less. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 61BT DSC, MMCs shall pass the qualification testing specified in Section K.9.1.7.6, and shall subsequently be subject to the process controls specified in Section K.9.1.7.7.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section K.9.1.7.5. The specified acceptance testing assures that at any location

in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surfaces, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

K.9.1.7.3 **Boral®**

See the Caution in Section K.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. ***Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns.*** **The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.**

The criticality calculations take credit for 75% of the minimum specified B10 areal density of Boral®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on an approximately 1 cm² area of a coupon taken near one of the corners of the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

Visual inspections shall verify that the Boral® core is not exposed through the face of the sheet at any location.

K.9.1.7.4 Thermal Conductivity Testing

The poison plate material will be qualification tested to verify that the thermal conductivity equals or exceeds the values listed in Section K.4.3. *Acceptance testing shall be performed at room temperature on coupons taken from the rolled or extruded production material. Initial sampling shall be one test per lot, defined by the heat or ingot, and may be reduced if the first five tests meet the specified minimum thermal conductivity.*

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After twenty-five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no *further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.*

Testing *shall* be by ASTM E1225 [9.3], ASTM E1461 [9.4], or equivalent method.

K.9.1.7.4.1 Test Coupons

The poison plates are manufactured in a variety of sizes. Coupons will be removed between every other plate or at the end of the plate so that there is at least one coupon contiguous with each plate. Coupons will generally be the full width of the plate. Thermal conductivity coupons may be removed from the full width coupon. The minimum dimension of the coupon shall be as required for acceptance test specimens; 1 to 2 inches is generally adequate.

K.9.1.7.5 **Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission**

> ***CAUTION***
>
> *Section K.9.1.7.5 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 1) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.*

Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1.2 centimeter diameter. A beam size greater than 1.2 centimeter diameter but no larger than 1.7 centimeter diameter may be used if computations are performed to demonstrate that the calculated $k_{effective}$ of the system is still below the calculated Upper Subcritical Limit (USL) of the system assuming defect areas the same area as the beam.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard.

Alternatively, digital image analysis may be used to compare neutron radioscopic images of the test coupon to images of the standards. The area of image analysis shall be up to 1.1 cm^2. The method shall demonstrate sufficient sensitivity to distinguish between areal density at the specified minimum, and 1% above and below the minimum.

The minimum areal density specified shall be verified for each lot at the 95% probability; 95% confidence level or better. The following illustrates one acceptable method.

(Alternatively show that the material fails in a ductile manner, e.g., by scanning electron microscopy of the fracture surface or by bend testing.)

and

b) testing (ASTM-B311[4]) to verify more than 98% (or 97% for MMCs with integral aluminum cladding) of theoretical density. Testing or examination for exposed interconnected porosity shall be performed by a means to be approved by the Certificate Holder.

K.9.1.7.6.5 **Required Tests and Examinations to Demonstrate B10 Uniformity**

> ***CAUTION***
>
> *Section K.9.1.7.6.5 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 1) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.*

Uniformity of the boron distribution shall be verified either by:

a) **Neutron radioscopy or radiography (ASTM E94[5], E142[6], and E545[7]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or**

b) **Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section K.9.1.7.5, or by chemical analysis for boron carbide content in the composite.**

K.9.1.7.6.6 Approval of Procedures

Qualification procedures shall be subject to approval by the Certificate Holder.

K.9.1.7.7 Specification for Process Controls for Metal Matrix Composites

K.9.1.7.7.1 Applicability and Scope

The applicability of this section is the same as that of Section K.9.1.7.6. It addresses the process controls to ensure that the material delivered for use is equivalent to the qualification test material.

4 ASTM B311, Test Method for Density Determination for Powder Metallurgy (P/M) Materials Containing Less than Two Percent Porosity

5 ASTM E94, Recommended Practice for Radiographic Testing

6 ASTM E142, Controlling Quality of Radiographic Testing

7 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section K.9.1.7.7 is required, depending on the characteristics of the material that could be affected by the process change.

K.9.1.7.7.2 Definition of Key Process Changes

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, or reduce the mechanical strength or ductility of the MMC.

K.9.1.7.7.3 **Identification and Control of Key Process Changes**

CAUTION
Section K.9.1.7.7.3 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications (paragraph 4.1 (Note 1)) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section K.9.1.7.7.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that may be established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a 6000 series aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature, and

Table K.9-3
Specified B10 Areal Density
Boral® and Metamic® (75% credit)

Section K.6 Analysis B10 Content (g/cm^2)	Specified Minimum B10 Content (g/cm^2)
0.019	0.025
0.029	0.038
0.036	0.048
For *Damaged* Fuel	
0.036	0.048

M.2.1 Spent Fuel To Be Stored

There are four design configurations for the NUHOMS®-32PT DSC, two "short" canister configurations (the 32PT-S100 and 32PT-S125), and two "long" canister configurations (the 32PT-L100 and 32PT-L125). The main difference between the –S100/-L100 and –S125/-L125 configuration designs are the thicknesses of shield plugs and DSC cover plates. The basket layout for these two configurations is identical except for the length of the components. Each of the DSC configurations is designed to store 32 intact standard PWR fuel assemblies. The 32PT-L100 and 32PT-L125 are also designed to store 32 intact standard PWR fuel assemblies with or without CCs. The NUHOMS®-32PT DSCs can store intact PWR fuel assemblies and BPRAs with the characteristics described in Table M.2-1. The CCs include burnable poison rod assemblies (BPRAs), thimble plug assemblies (TPAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs), neutron source assemblies (NSAs), and neutron sources. The NUHOMS®-32PT DSC may store PWR fuel assemblies arranged in any of three alternate heat zoning configurations with a maximum decay heat of 1.2 kW per assembly and a maximum heat load of 24 kW per canister. The heat load zoning configurations are shown in Figure M.2-1 through Figure M.2-3. The NUHOMS®-32PT DSC is inerted and backfilled with helium at the time of loading. The maximum fuel assembly weight with a CC is 1682 lb which is the same as the NUHOMS®-24P DSC design.

The maximum fuel cladding temperature limit of 400 °C (752 °F) is applicable to normal conditions of storage and all short term operations from spent fuel pool to ISFSI pad including vacuum drying and helium backfilling of the NUHOMS®-32PT DSC per Interim Staff Guidance (ISG) No. 11, Revision *3* [2.7]. In addition, ISG-11 does not permit thermal cycling of the fuel cladding with temperature differences greater than 65 °C (117 °F) during DSC drying, backfilling and transfer operations.

The maximum fuel cladding temperature limit of 570 °C (1058 °F) is applicable to accidents or off-normal thermal transients [2.7].

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses including shielding, criticality, heat load and confinement. These evaluations are performed in Chapter M.5, M.6, M.4 and M.7. The fuel assembly types considered are listed in Table M.2-2. It was determined that the B&W 15x15 is the enveloping fuel design for the shielding source term calculation because of its total assembly weight and highest initial heavy metal loading. For criticality safety, the B&W 15x15 assembly is the most reactive assembly type for a given enrichment. This assembly is used to determine the most reactive configuration in the DSC. Using this most reactive configuration, criticality analysis for all other fuel assembly classes is performed to determine the maximum enrichment allowed as a function of number of poison rod assemblies (PRAs). For thermal analysis, the WE 14x14 fuel assembly is limiting, since it results in the lowest fuel conductivity. The confinement analyses is based on B&W 15x15 fuel assembly, since it results in the smaller free volume inside the DSC cavity more than a 14x14 fuel assembly.

All four NUHOMS®-32PT DSC design configurations have the same minimum boron content for the poison neutron plates. The minimum boron-10 content for the poison plates is 0.0070

Stress criteria for Level A through Level D service loading conditions are given in Table M.2-16. Local yielding is permitted at the point of contact where the Level D load is applied. If elastic stress limits cannot be met, the plastic system analysis approach and acceptance criteria of Appendix F of ASME Section III are used.

The allowable stress intensity value, S_m, as defined by the Code is based on the temperature calculated for each service load condition or a bounding temperature.

M.2.2.5.1.2 NUHOMS®–32PT DSC Basket Stress Limits

The basket fuel support grid wall thickness is established to meet heat transfer, nuclear criticality, and structural requirements. The basket structure provides sufficient rigidity to maintain a subcritical configuration under the applied loads.

No credit is taken for neutron poison plates in any of the stress or stability analyses.

Normal Conditions

Normal Condition Stress Criteria for Steel Elements

As summarized in Table M.2-17, the normal condition stress criteria for the fuel support structure, and the R90 transition rail cover plates is based on Subsection NG [2.2] of the ASME Code, Section III.

Normal Condition Stress Criteria for Aluminum Transition Rails

The solid aluminum transition rail bodies (R90 and R45) perform their function (support of the fuel grid) by remaining in place. The loads on the rail bodies are primarily bearing from the fuel grid (transmitted through the cover plate on the R90 rails). "Failure" of the transition rail would require that the rail no longer provide support to the fuel grid. Since the solid aluminum rail bodies are constrained between the DSC shell and the fuel support grid, this cannot occur.

Therefore, for deadweight and handling condition loads, stress in the aluminum bodies will be compared to the allowable bearing stress, equal to S_y, from NG-3227.1(a). Values of S_y are taken from Table M.3.3-4 for annealed 6061 aluminum material at temperature (as described in Section M.3.3, these yield stresses are lower bound values).

Normal Condition Stability Criteria

Stability criteria are addressed in two parts:

A. Under axial loads, the DSC shell and transfer cask provide overall/global stability to the 32PT basket structure. Thus, only local stability effects are specifically addressed. For local (panel) stability under axial loads, the allowable stress in the fuel support grid and transition rail panels are taken as the smallest of the following three values:

- The normal condition (Level A) primary membrane stress allowable, P_m,

Table M.2-1
Intact PWR Fuel Assembly Characteristics

PHYSICAL PARAMETERS:	
Fuel Class	Only intact (including reconstituted) B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14 and WE 14x14 class PWR assemblies or equivalent reload fuel manufactured by other vendors that are enveloped by the fuel assembly design characteristics listed in Table M.2-2.
Reconstituted Fuel Assemblies	≤ 32 assemblies per DSC with up to 56 stainless steel rods per assembly or unlimited number of lower enrichment UO_2 rods per assembly.
Fuel Cladding Material	Zircaloy
Fuel Damage	Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact PWR Fuel."
Control Components (CCs)	• Up to 32 CCs are authorized for storage in 32PT DSC. • Authorized CCs include Burnable poison Rod Assemblies (BPRAs), Thimble Plug Assemblies, (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources • Design basis thermal and radiological characteristics for the CCs are listed in Table M.2-2a.
Maximum Assembly plus CC Weight	-1365 lbs for 32PT-S100 & 32PT-L100 DSC System -1682 lbs for 32PT-S125 & 32PT-L125 DSC System
CC Damage	CCs with cladding failures are acceptable for loading.
THERMAL/RADIOLOGICAL PARAMETERS:	
Fuel Burnup and Cooling Time without CCs[1]	Per Table M.2-5, Table M.2-6, Table M.2-7, Table M.2-8, Table M.2-9; and Figure M.2-1 or Figure M.2-2 or Figure M.2-3, *except that for a 32PT DSC contained in an OS197L TC, see Tables W.2-7 and W.2-8, and Figure W.2-2*.
Fuel Burnup and Cooling Time with CCs[1]	Per Table M.2-10, Table M.2-11, Table M.2-12, Table M.2-13, Table M.2-14; and Figure M.2-1 or Figure M.2-2 or Figure M.2-3.
Initial Enrichment	Per Table M.2-3; and Figure M.2-4 or Figure M.2-5 or Figure M.2-6, as applicable.

Table M.3.1-1
Alternatives to the ASME Code for the NUHOMS®-32PT DSC Confinement Boundary

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section M.2 may be used for construction, but in no case earlier than 3 years before that specified in Section M.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section M.2 may be used, so long the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NB-1100	*Requirements for Code Stamping of Components, Code reports and certificates, etc.*	*Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.*
NB-1132	*Attachments with a pressure retaining function, including stiffeners, shall be considered part of the component.*	*Bottom shield plug and outer bottom cover plate are outside code jurisdiction; these components together are much larger than required to provide stiffening for the inner bottom cover plate; the weld that retains the outer bottom cover plate and with it the bottom shield plug is subject to root and final PT examination.*
NB-2130	*Material must be supplied by ASME approved material suppliers.*	*Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.*
NB-4121	*Material Certification by Certificate Holder*	
NB-4243 and NB-5230	Category C weld joints in vessels and similar weld joints in other components shall be full penetration joints. These welds shall be examined by UT or RT and either PT or MT	The joints between the top outer and inner cover plates and containment shell are designed and fabricated per ASME Code Case N-595-2, which provides alternative requirements for the design and examination of spent fuel canister closures. This includes the inner top cover plate weld around the vent & siphon block and the vent and siphon block welds to the shell. The closure welds are partial penetration welds and the root and final layer are subject to PT examination (in lieu of volumetric examination) in accordance with the provisions of ASME Code Case N-595-2. The 32PT closure system employs austenitic stainless steel shell, lid materials, and welds. Because austenitic stainless steels are not subject to brittle failure at the operating temperatures of the DSC, crack propagation is not a concern. Thus, multi-level PT examination provides reasonable assurance that flaws of interest will be identified. The PT examination is done by qualified personnel, in accordance with Section V and the acceptance standards of Section III, Subsection NB-5000. This alternative does not apply to other shell confinement welds, i.e., the longitudinal and circumferential welds applied to the DSC shell, and the inner bottom cover plate-to-shell weld which comply with NB-4243 and NB-5230.

Table M.3.1-1
Alternatives to the ASME Code for the NUHOMS®-32PT DSC Confinement Boundary
(Concluded)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NB-6100 and 6200	All pressure retaining components and completed systems shall be pressure tested. The preferred method shall be hydrostatic test.	The NUHOMS®-32PT DSC is pressure tested in accordance with ASME Code Case N-595-2. The shield plug support ring and the vent and siphon block are not pressure tested due to the manufacturing sequence. The support ring is not a pressure-retaining item and the vent and siphon block weld is helium leak tested after fuel is loaded to the same criteria as the inner top closure plate-to-shell weld (ANSI N14.5-1997 leaktight criteria).
NB-7000	*Overpressure Protection*	*No overpressure protection is provided for the NUHOMS® DSCs. The function of the DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The DSC is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature.*
NB -8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.*
NB-5520	*NDE Personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of more recent edition of SNT-TC-1A.*

Table M.3.1-2
Alternatives to the ASME Code Exceptions for the NUHOMS®-32PT DSC Basket Assembly

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section M.2 may be used for construction, but in no case earlier than 3 years before that specified in the Section M.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section M.2 may be used, so long the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NG-1100	*Requirements for Code Stamping of Components, Code reports and certificates, etc.*	*Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.*
NG-2000	Use of ASME Material	*Some baskets include neutron absorber and aluminum plates that are not ASME Code Class 1 material. They are used for criticality safety and heat transfer, and are only credited in the structural analysis with supporting their own weight and transmitting bearing loads through their thickness.* Material properties in the ASME Code for Type 6061 aluminum are limited to 400°F to preclude the potential for annealing out the hardening properties. Annealed properties (as published by the Aluminum Association and the American Society of Metals) are conservatively assumed for the solid aluminum rails for use above the Code temperature limits.
NG-2130	*Material must be supplied by ASME approved material suppliers.*	*Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.*
NG-4121	*Material Certification by Certificate Holder*	
NG -8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.*
NG-3000/ Section II, Part D, Table 2A	Maximum temperature limit for XM-19 plate material is 800°F	Not compliant with ASME Section II Part D Table 2A material temperature limit for XM-19 steel for the postulated transfer accident case (117°F, loss of sunshade, loss of neutron shield). This is a post-drop accident scenario, where the calculated maximum steady state temperature is 852°F, the expected reduction in material strength is small (less than 1 ksi by extrapolation), and the only primary stresses in the basket grid are deadweight stresses. The recovery actions following the postulated drop accident are as described in Section 8.2.5 of the FSAR.
NG-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of more recent edition of SNT-TC-1A.*

M.8.1 Procedures for Loading the Cask

Process flow diagrams for the NUHOMS® System operation are presented in Figure M.8-1 and Figure M.8-2. The location of the various operations may vary with individual plant requirements. The following steps describe the recommended generic operating procedures for the standardized NUHOMS® System.

M.8.1.1 Preparation of the TC and DSC

1. Prior to placement in dry storage, the candidate intact fuel assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Technical Specification *2.1*. Depending on the length of the fuel assemblies to be loaded, fuel spacers may be placed within the DSC to reduce the fuel assembly/DSC cavity gap in consideration of Part 71 requirements. There are no requirements for fuel spacers under Part 72. Fuel spacers, if used, may be placed below the assembly, above the assembly, or both, and shall be evaluated for any adverse impact.
2. Prior to being placed in service, the TC is to be cleaned or decontaminated as necessary to insure a surface contamination level of less than those specified in Technical Specification *5.2.4.d.*
3. Place the TC in the vertical position in the cask decon area using the cask handling crane and the TC lifting yoke.
4. Place scaffolding around the cask so that the top cover plate and surface of the cask are easily accessible to personnel.
5. Remove the TC top cover plate and examine the cask cavity for any physical damage and ready the cask for service. If loading 32PT-S100 or 32PT-L100 DSC (qualified for 100-ton crane capacity), drain neutron shield water from the TC.
6. Examine the DSC for any physical damage which might have occurred since the receipt inspection was performed. The DSC is to be cleaned and any loose debris removed.
7. Using a crane, lower the DSC into the cask cavity by the internal lifting lugs and rotate the DSC to match the cask and DSC alignment marks.
8. Fill the cask-DSC annulus with clean, demineralized water. Place the inflatable seal into the upper cask liner recess and seal the cask-DSC annulus by pressurizing the seal with compressed air.

8a. Place and verify that the bottom fuel assembly spacers, if required, are present in the fuel cells. Optionally, this step may be performed at any prior time.

9. Fill the DSC cavity with water from the fuel pool or an equivalent source which meets the requirements of Technical Specification *3.2.1*.

   NOTE: A TC/DSC annulus pressurization tank filled with demineralized water as described above is connected to the top vent port of the TC via a hose to provide a positive head above the level of water in the TC/DSC annulus. This is an optional arrangement, which provides additional assurance that contaminated water from the fuel pool will not enter the TC/DSC annulus, provided a positive head is maintained at all times.

10. Place the top shield plug onto the DSC. Examine the top shield plug to ensure a proper fit.

pressure stabilizes, the pump is valved in to complete the vacuum drying process. It may be necessary to repeat some steps, depending on the rate and extent of the pressure increase. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less as specified in Technical Specification *3.1.1*.

19. Open the valve to the vent port and allow the helium to flow into the DSC cavity.

20. Pressurize the DSC with helium to about 24 psia not to exceed 34 psia.

21. Helium leak test the inner top cover plate weld for leakage in accordance with ANSI N14.5 to a sensitivity of 1×10^{-5} atm·cm^3 /sec. This test is optional.

22. If a leak is found, repair the weld, repressurize the DSC and repeat the helium leak test.

23. Once no leaks are detected, depressurize the DSC cavity by releasing the helium through the VDS to the plant's spent fuel pool or radioactive waste system.

24. Re-evacuate the DSC cavity using the VDS. The cavity pressure should be reduced in steps of approximately 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level, the pump is valved off and the cavity pressure is monitored. When the cavity pressure stabilizes, the pump is valved in to continue the vacuum drying process. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less in accordance with Technical Specification *3.1.1* limits.

25. Open the valve on the vent port and allow helium to flow into the DSC cavity to pressurize the DSC to about 17.2 psia *(2.5 psig)* in accordance with Technical Specification *3.1.2.b* limits.

26. Close the valves on the helium source.

27. Decontaminate as necessary, and store.

M.8.1.4 DSC Sealing Operations

1. Disconnect the VDS from the DSC. Seal weld the prefabricated plugs over the vent and siphon ports, inject helium into blind space just prior to completing welding, and perform a dye penetrant weld examination in accordance with the Technical Specification *5.2.4.b* requirements.

2. Open the cask drain port valve and remove the remaining water from the cask/DSC annulus.

3. Install the automatic welding machine onto the outer top cover plate and place the outer top cover plate with the automatic welding system onto the DSC. Verify proper fit up of the outer top cover plate with the DSC shell.

4. Tack weld the outer top cover plate to the DSC shell. Place the outer top cover plate weld root pass.

5. Helium leak test the inner top cover plate and vent/siphon port plate welds using the leak test port in the outer top cover plate in accordance with Technical Specification *5.2.4.c* limits. Verify that the personnel performing the leak test are qualified in accordance with SNT-TC-1A [8.6].

6. If a leak is found, remove the outer cover plate root pass, the vent and siphon port plugs and repair the inner cover plate welds. *Repeat* procedure steps from M.8.1.3 step 18.

7. Perform dye penetrant examination of the root pass weld. Weld out the outer top cover plate to the DSC shell and perform dye penetrant examination on the weld surface in accordance with the Technical Specification *5.2.4.b* requirements.

8. Seal weld the prefabricated plug over the outer cover plate test port and perform dye penetrant weld examinations.

9. Remove the automatic welding machine from the DSC. Rig the cask top cover plate and lower the cover plate onto the TC.

10. Bolt the cask cover plate into place, tightening the bolts to the required torque in a star pattern.

M.8.1.5 TC Downending and *Transfer* to ISFSI

1. If loading 32PT-S100 or 32PT-L100 DSC (qualified for 100-ton crane capacity), drain the neutron shield to an acceptable location.

2. Re-attach the TC lifting yoke to the crane hook, as necessary. Ready the *transfer* trailer and cask support skid for service.

3. Move the scaffolding away from the cask as necessary. Engage the lifting yoke and lift the cask over the cask support skid on the *transfer* trailer.

4. The *transfer* trailer should be positioned so that cask support skid is accessible to the crane with the trailer supported on the vertical jacks.

5. Position the cask lower trunnions onto the transfer trailer support skid pillow blocks.

6. Move the crane forward while simultaneously lowering the cask until the cask upper trunnions are just above the support skid upper trunnion pillow blocks.

7. Inspect the positioning of the cask to insure that the cask and trunnion pillow blocks are properly aligned.

8. Lower the cask onto the skid until the weight of the cask is distributed to the trunnion pillow blocks.

9. Inspect the trunnions to insure that they are properly seated onto the skid and install the trunnion tower closure plates (optional for the OS197 TC and the OS 197H TC).

10. Fill the neutron shield, if it was drained in M.8.1.5 *step 1*.

11. Remove the bottom ram access cover plate from the cask. Install the two-piece temporary neutron/gamma shield plug to cover the bottom ram access. Install the ram trunnion support frame on the bottom of the TC. (The temporary shield plug and ram trunnion support frame are not required with integral ram/trailer.)

M.8.1.6 DSC Transfer to the HSM

1. Prior to *transferring* the cask to the ISFSI, remove the HSM door using a porta-crane, inspect the cavity of the HSM, removing any debris and ready the HSM to receive a DSC. The doors on adjacent HSMs should remain in place.

   Caution: The insides of empty modules have the potential for high dose rates due to adjacent loaded modules. Proper ALARA practices should be follower for operations inside these modules and in the areas outside these modules whenever the door from the empty HSM has been removed.

2. Inspect the HSM air inlet and outlets to ensure that they are clear of debris. Inspect the screens on the air inlet and outlets for damage.

   CAUTION: Verify that the requirements of Technical Specification *5.3.1.B*, "TC/DSC Transfer Operations at High Ambient Temperatures" are met prior to next step.

3. Using a suitable heavy haul tractor, *transfer* the cask from the plant's fuel/reactor building to the ISFSI along the designated transfer route.

4. Once at the ISFSI, position the *transfer* trailer to within a few feet of the HSM.

5. Check the position of the trailer to ensure the centerline of the HSM and cask approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.

6. Unbolt and remove the cask top cover plate.

7. Back the cask to within a few inches of the HSM, set the trailer brakes and disengage the tractor. Drive the tractor clear of the trailer. Extend the transfer trailer vertical jacks.

8. Connect the skid positioning system hydraulic power unit to the positioning system via the hose connector panel on the trailer, and power it up. Remove the skid tie-down bracket fasteners and use the skid positioning system to bring the cask into approximate vertical and horizontal alignment with the HSM. Using optical survey equipment and the alignment marks on the cask and the HSM, adjust the position of the cask until it is properly aligned with the HSM.

9. Using the skid positioning system, fully insert the cask into the HSM access opening docking collar.

10. Secure the cask trunnions to the front wall embedments of the HSM using the cask restraints.

11. After the cask is docked with the HSM, verify the alignment of the TC using the optical survey equipment.

12. Position the hydraulic ram behind the cask in approximate horizontal alignment with the cask and level the ram. Remove either the bottom ram access cover plate or the outer plug of the two-piece temporary shield plug. Power up the ram hydraulic power supply and extend the ram through the bottom cask opening into the DSC grapple ring.

13. Activate the hydraulic cylinder on the ram grapple and engage the grapple arms with the DSC grapple ring.

14. Recheck all alignment marks in accordance with the Technical Specification *5.3.3* limits and ready all systems for DSC transfer.

15. Activate the hydraulic ram to initiate insertion of the DSC into the HSM. Stop the ram when the DSC reaches the support rail stops at the back of the module.

16. Disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.

17. Retract and disengage the hydraulic ram system from the cask and move it clear of the cask. Remove the cask restraints from the HSM.

18. Using the skid positioning system, disengage the cask from the HSM access opening.

19. Install the DSC axial retainer through the HSM door opening.

20. The trailer may be moved as necessary to install the HSM door. Install the HSM door and secure it in place. Door may be welded for security. Verify that the HSM dose rates are compliant with the limits specified in Technical Specification *5.4.2*.

21. Replace the TC top cover plate. Secure the skid to the trailer, retract the vertical jacks and disconnect the skid positioning system.

22. Tow the trailer and cask to the designated equipment storage area. Return the remaining transfer equipment to the storage area.

23. Close and lock the ISFSI access gate and activate the ISFSI security measures.

M.8.1.7 Monitoring Operations

1. Perform routine security surveillance in accordance with the licensee's ISFSI security plan.

   NOTE: *Perform one of two alternative surveillance activities listed below.*

M.9.1.5 Shielding Integrity Tests

No changes associated with this amendment.

M.9.1.6 Thermal Acceptance Tests

The analyses to ensure that the NUHOMS®-32PT DSCs are capable of performing their heat transfer function are presented in Section M.4.

M.9.1.7 Poison Acceptance

CAUTION
Sections M.9.1.7.1 and M.9.1.7.2 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 2) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated aluminum

(b) Boron carbide/aluminum metal matrix composite (MMC)

The 32PT DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content for these materials corresponds to a minimum areal density of 7.0 mg B10/cm^2.

M.9.1.7.1 Borated Aluminum

See the Caution in Section M.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating *primarily* as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy *(other boron compounds, such as AlB_{12}, can also occur)*. For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section M.9.1.7.5. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings"[9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surface, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

M.9.1.7.2 Boron Carbide/Aluminum Metal Matrix Composites (MMC)

See the Caution in Section M.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

Boron carbide particles for the products considered here shall have an average size of 40 microns or less. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 32PT DSC, MMCs shall pass the qualification testing specified in Section M.9.1.7.6, and shall subsequently be subject to the process controls specified in Section M.9.1.7.7.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section M.9.1.7.5. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surfaces, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

M.9.1.7.3 Not Used

M.9.1.7.4 Thermal Conductivity Testing of Poison Plates

The poison plate material shall be qualification tested to verify that the thermal conductivity equals or exceeds the values listed in Section M.4.3. *Acceptance testing shall be performed at room temperature on coupons taken from the rolled or extruded production material. Initial sampling shall be one test per lot, defined by the heat or ingot, and may be reduced if the first five tests meet the specified minimum thermal conductivity.*

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

Testing *shall* be by ASTM E1225 [9.3], ASTM E1461 [9.4], or equivalent method.

M.9.1.7.4.1 Test Coupon and Lot Definitions

Sample taken from the plate material is a test coupon. Test coupons will be removed so that there is at least one coupon contiguous with each plate. These coupons will be used for thermal conductivity testing. The minimum dimension of the coupon shall be as required for the acceptance test procedures.

A lot is defined as all the plates produced from a single cast ingot, or all the plates produced from a single heat.

M.9.1.7.5 Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission

CAUTION

Section M.9.1.7.5 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 2) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1.2 centimeter diameter. A beam size greater than 1.2 centimeter diameter but no larger than 1.7

Thermal damage testing is not required for MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below 842°F, well above the basket temperature under normal conditions of storage or transport[1].

Corrosion testing is not required for full density MMCs consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[2].

M.9.1.7.6.4 Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity

At least three samples, one each from the two ends and middle of the test material production run shall be subject to:

a) room temperature tensile testing (ASTM B557[3]) demonstrating that the material has the following tensile properties:

- Minimum yield strength, 0.2% offset: 1.5 ksi
- Minimum ultimate strength: 5 ksi
- Minimum elongation in 2 inches: 0.5%

(Alternatively show that the material fails in a ductile manner, e.g., by scanning electron microscopy of the fracture surface or by bend testing.)

and

b) testing (ASTM B311[4]) to verify more than 98% (or 97% for MMCs with integral aluminum cladding) of theoretical density. Testing or examination for exposed interconnected porosity shall be performed by a means to be approved by the Certificate Holder.

M.9.1.7.6.5 Required Tests and Examinations to Demonstrate B10 Uniformity

CAUTION

Section M.9.1.7.6.5 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 2) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

1 Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B_4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998.

2 Boralyn testing submitted to the NRC under docket 71-1027, 1998.

3 ASTM B557 Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products.

4 ASTM B311, Test Method for Density Determination for Powder Metallurgy (P/M) Materials Containing Less than Two Percent Porosity

Uniformity of the boron distribution shall be verified either by:

a) **Neutron radioscopy or radiography (ASTM E94[5], E142[6], and E545[7]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or**

b) **Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section M.9.1.7.5, or by chemical analysis for boron carbide content in the composite.**

M.9.1.7.6.6 Approval of Procedures

Qualification procedures shall be subject to approval by the Certificate Holder.

M.9.1.7.7 Specification for Process Controls for Metal Matrix Composites

M.9.1.7.7.1 Applicability and Scope

The applicability of this section is the same as that of Section M.9.1.7.6. It addresses the process controls to ensure that the material delivered for use is equivalent to the qualification test material.

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section M.9.1.7.7 is required, depending on the characteristics of the material that could be affected by the process change.

M.9.1.7.7.2 Definition of Key Process Changes

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, or reduce the mechanical strength or ductility of the MMC.

M.9.1.7.7.3 Identification and Control of Key Process Changes

CAUTION
Section M.9.1.7.7.3 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 2) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

5 ASTM E94, Recommended Practice for Radiographic Testing
6 ASTM E142, Controlling Quality of Radiographic Testing
7 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

N.2.1 Spent Fuel to be Stored

There are two design configurations for the NUHOMS®-24PHB DSC: the 24PHBS and 24PHBL, which are nearly identical to the standard and long cavity 24P DSCs, respectively. Each of the DSC configurations is designed to store 24 intact PWR fuel assemblies, including reconstituted assemblies with characteristics described in Table N.2-1. The 24PHBL DSC is designed to store 24 intact B&W 15x15, WE 17x17, WE 15x15, CE 14x14, and WE 14x14 Class PWR fuel assemblies. BPRAs are allowed only in the B&W 15x15 fuel assembly. Replacement assemblies by other manufacturers are also allowed provided they meet limiting features listed in Table N.2-1.

The NUHOMS®-24PHB DSC may store PWR fuel assemblies arranged in one of two alternate Heat Load Zoning Configurations with a maximum decay heat of 1.3 kW per assembly and a maximum heat load of 24 kW per DSC. The Heat Load Zoning Configurations are shown in Figure N.2-1 and Figure N.2-2. The NUHOMS®-24PHB DSC is vacuum dried and backfilled with helium at the time of loading. The maximum (bounding) fuel assembly weight of 1682 lbs with a BPRA is identical to the NUHOMS®-24P DSC design.

The maximum fuel cladding temperature limit of 400°C (752°F) is applicable to normal conditions of storage and all short term operations from spent fuel pool to ISFSI pad including vacuum drying and helium backfilling of the 24PHB DSC per Interim Staff Guidance (ISG) No. 11, Revision *3* [2.6]. In addition, ISG-11 does not permit thermal cycling of the fuel cladding with temperature differences greater than 65°C (117°F) during DSC drying, backfilling and transfer operations.

The maximum fuel cladding temperature limit of 570°C (1058°F) is applicable to accidents or off-normal thermal transients [2.6].

The information provided in Table N.2-1 is based on the design basis B&W 15x15 fuel which is the bounding fuel assembly. The types of spent fuel considered in Appendix N include the following:

- B&W 15x15 Mark B2, B3, B4, B4Z, BZ, B5, B5Z, B6, B7, B8, B9 and B10 fuel assemblies.
- B&W 15x15 reconstituted fuel assemblies with a maximum of 10 stainless steel rods per assembly or unlimited number of lower enrichment UO_2 rods instead of zircaloy clad enriched UO_2 rods. The stainless steel rods are assumed to have two thirds the irradiation time as the zircaloy rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four.
- The standard BPRA design for the B&W 15x15 class assemblies is described in Appendix J and B&W 15x15 BPRAs with a different material composition (bounding values of 24.46 lbs. of poison, plus 28.17 Kgs of other materials, containing 9.15 grams of Co59) designated Mark-B BPRAs, for the NUHOMS®-24PHB system, are evaluated in this appendix.
- WE 17x17, WE 15x15, CE 14x14 and WE 14x14 fuel assemblies, all with no BPRAs.

Calculations are performed to determine the fuel assembly type which is most limiting for each of the analyses including shielding, criticality, heat load and confinement. Analyses performed

CAUTION: For DSCs with spacer discs coated with aluminum, continuously monitor the hydrogen concentration in the DSC cavity using the tygon tube arrangement described in step 9 during the inner top cover plate or the top shield plug assembly (for the optional 24PHBL "shifted shielding" configuration) welding operations. Verify that the measured hydrogen concentration does not exceed a safety limit of 2.4% (5.4). If this limit is exceeded, stop all welding operations and purge the DSC cavity with 2-3 psig helium (or any other inert medium) via the ¼" tygon tubing to reduce the hydrogen concentration safely below the 2.4% limit. This step is optional for DSCs with spacer discs which have been coated with electroless nickel.

12. Perform dye penetrant weld examination of the inner top cover plate or the top shield plug assembly (for the optional 24PHBL "shifted shielding" configuration) weld in accordance with the Technical Specification *5.2.4.b* requirements.

13. Place the strongback so that it sits on the inner top cover plate and is oriented such that:

    - the DSC siphon and vent ports are accessible
    - the strongback stud holes line up with the TC lid bolt holes.

    Steps 13 and 14 are optional for the optional 24PHBL "shifted shielding" configuration.

24. Helium leak test the inner top cover plate weld for leakage in accordance with ANSI N.14.5 to a sensitivity of 1×10^{-5} atm·cm^3/sec. This test is optional.

30. Remove the Strongback, if used. Decontaminate as necessary, and store.

N.8.1.4 24PHB DSC Sealing Operations

1. Disconnect the VDS from the DSC. Seal weld the prefabricated plugs over the vent and siphon ports, inject helium in blind space and perform a dye penetrant weld examination in accordance with the Technical Specification *5.2.4.b* requirements.

2. Install the welding machine onto the outer top cover plate and place the outer top cover plate with the welding system onto the DSC. Manual welding is also acceptable. Verify proper fit up of the outer top cover plate with the DSC shell.

3. Tack weld the outer top cover plate to the DSC shell. Place the outer top cover plate weld root pass.

4. Helium leak test the inner top cover plate and vent/siphon port plate welds using the leak test port in the outer top cover plate in accordance with Technical Specification *5.2.4.c* limits. Verify that the personnel performing the leak test are qualified in accordance with SNT-TC-1A [8.1]. Alternatively, this can be done with a test head in N.8.1.4 step 1.

5. If a leak is found, remove the outer cover plate root pass, the vent and siphon port plugs and repair the inner cover plate welds. Then install the Strongback and repeat procedure steps from Section 5.1.1.3, step 20.

6. Perform dye penetrant examination of the root pass weld. Weld out the outer top cover plate to the DSC shell and perform dye penetrant examination on the weld surface in accordance with the Technical Specification *5.2.4.b* requirements.

7. Seal weld the prefabricated plug over the outer cover plate test port and perform dye penetrant weld examinations.

8. Rig the cask top cover plate and lower the cover plate onto the TC.

9. Bolt the cask cover plate into place, tightening the bolts to the required torque in a star pattern.

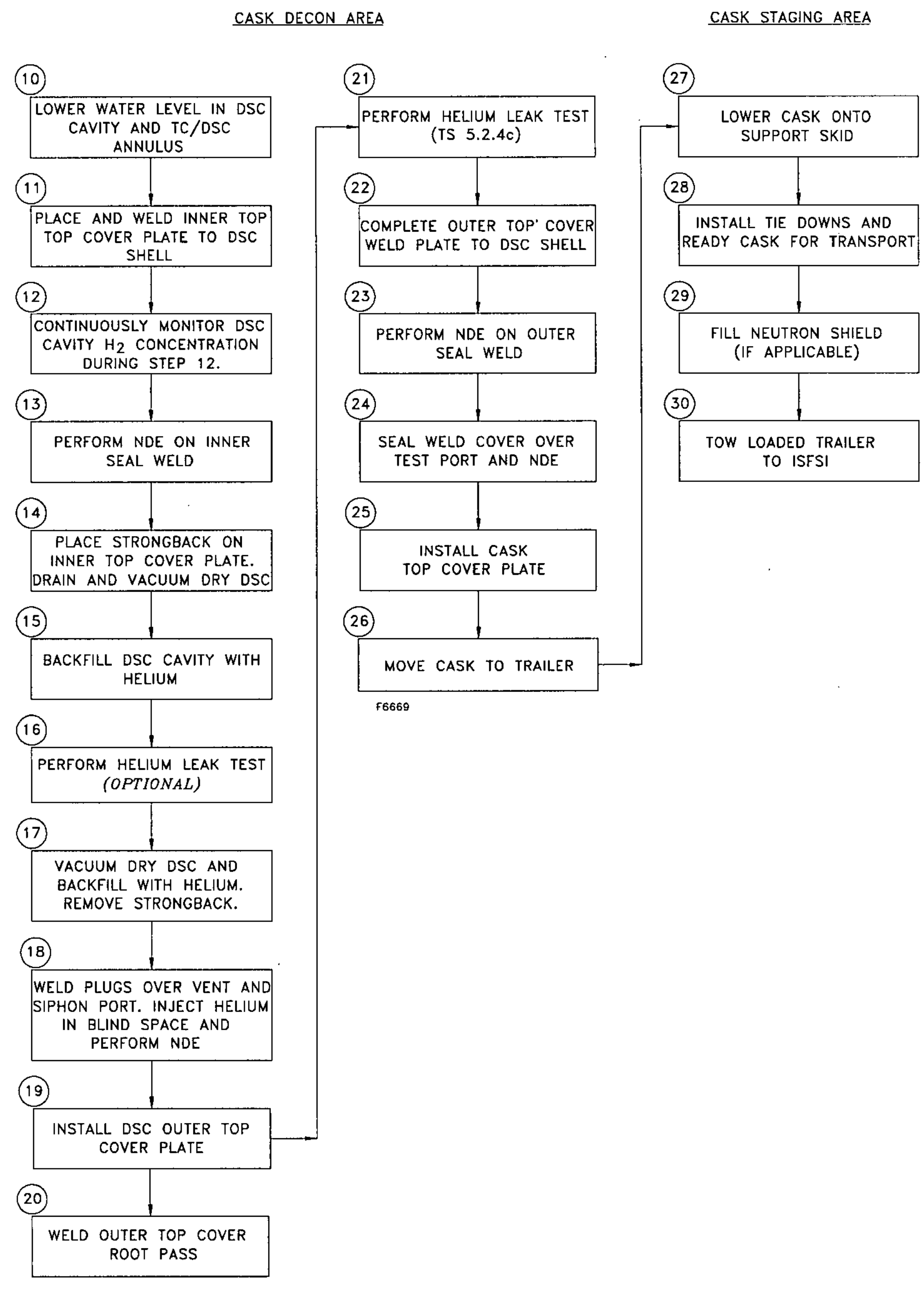

Figure N.8.1-1
NUHOMS® System Loading Operations Flow Chart
(Continued)

A. Under axial loads, the DSC shell and transfer cask provide overall/global stability to the 24PTH basket structure. Thus, only local stability effects are specifically addressed.

For axial compression loads, stability criteria for the fuel compartment tubes are based on NF-3322.1(c)(2) (for austenitic members). Using a span length of 24.0 in, corresponding to the maximum distance between basket straps and a value of K=1.0 (pinned-pinned condition) a slenderness ratio (KL/ r) of 6.42 is calculated.

Application of elastic stability criteria to the fuel tubes is conservative as the low value of KL/ r indicates elastic buckling is not a likely failure mechanism. In addition, buckling is restricted by the adjacent tubes, transition rails, and DSC shell.

In addition, the width to thickness ratio of the tube wall is checked using NF-3322.2(d)(2)(b)(1) to verify that the tubes are fully effective in compression.

B. Under lateral loads, stability of the basket tube structure is demonstrated using hand calculations to evaluate the fuel compartment tubes "ligaments" as columns using the stability criteria of NF-3322.1(c)(2) for stainless steel compression members.

Accident Conditions

<u>Accident Condition Stress Criteria for Steel Elements</u>

As summarized in Table P.2-16 the accident condition (Level D) stress criteria for the fuel support structure and the welded steel transition rails is based on Appendix F of the ASME Code, Section III. Criteria are provided for both linear elastic and elastic-plastic stress analyses.

<u>Accident Condition Criteria for R90 Aluminum Transition Rails</u>

For accident condition loading (i.e., the postulated drops), the R90 aluminum transition rail bodies must support the fuel tubes such that stresses and displacements in the fuel compartment tubes are acceptable. Since, the rail bodies are captured between the fuel compartment tube and the DSC shell, large displacements of the rails are prevented. Thus, no additional checks (of the aluminum) are required for accident/drop loading. Qualification of the fuel tubes demonstrates that the R90 rails perform their intended function.

<u>Accident Condition Stability Criteria</u>

Similar to the normal condition evaluations, stability criteria are addressed in two parts:

A. Accident condition axial stresses in the fuel compartment tubes are evaluated using the equation from F-1334.3(b)(1)[2.2] loads, stability of the basket structure is demonstrated using detailed finite element models and the Collapse Load criteria from F-1341.3 [2.2]. These criteria establish the allowable load as 90% of the Limit Analysis Collapse Load where the Limit Analysis Collapse Load is the maximum load determined using elastic-perfectly plastic material properties with a yield stress equal to the lesser of $2.3S_m$ or $0.7S_u$.

Table P.3.1-1
Alternatives to the ASME Code for the NUHOMS®-24PTH DSC Confinement Boundary
(Part 1 of 2)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section P.2 may be used for construction, but in no case earlier than 3 years before that specified in Section P.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section P.2 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NB-1100	*Requirements for Code Stamping of Components, Code reports and certificates, etc.*	*Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.*
NB-1132	*Attachments with a pressure retaining function, including stiffeners, shall be considered part of the component.*	*Bottom shield plug assembly, outer bottom cover plate, lifting posts, grapple ring, grapple ring support are outside code jurisdiction; these components together are much larger than required to provide stiffening for the inner bottom cover plate; the weld that retains the outer bottom cover plate and with it the bottom shield plug is subject to root and final PT examination.*
NB-2130	*Material must be supplied by ASME approved material suppliers.*	*Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.*
NB-4121	*Material Certification by Certificate Holder*	
NB-4243 and NB-5230	Category C weld joints in vessels and similar weld joints in other components shall be full penetration joints. These welds shall be examined by UT or RT and either PT or MT	The joints between the top outer and inner cover plates (or top forging assembly for the 24PTH-S-LC) and containment shell are designed and fabricated per ASME Code Case N-595-2, which provides alternative requirements for the design and examination of spent fuel canister closures. This includes the inner top cover plate weld around the vent & siphon block and the vent and siphon block welds to the shell. The closure welds are partial penetration welds and the root and final layer are subject to PT examination (in lieu of volumetric examination) in accordance with the provisions of ASME Code Case N-595-2. The 24PTH closure system employs austenitic stainless steel shell, lid materials, and welds. Because austenitic stainless steels are not subject to brittle failure at the operating temperatures of the DSC, crack propagation is not a concern. Thus, multi-level PT examination provides reasonable assurance that flaws of interest will be identified. The PT examination is done by qualified personnel, in accordance with Section V and the acceptance standards of Section III, Subsection NB-5000. This alternative does not apply to other shell confinement welds, i.e., the longitudinal and circumferential welds of the DSC shell, and the inner bottom cover plate-to-shell weld (or bottom forging to shell weld, as applicable) which comply with NB-4243 and NB-5230.

Table P.3.1-1
Alternatives to the ASME Code for the NUHOMS®-24PTH DSC Confinement Boundary
(*Part 2 of 2*)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NB-6100 and 6200	All pressure retaining components and completed systems shall be pressure tested. The preferred method shall be hydrostatic test.	The NUHOMS®-24PTH DSC is pressure tested in accordance with ASME Code Case N-595-2. The shield plug support ring and the vent and siphon block are not pressure tested due to the manufacturing sequence. The support ring is not a pressure-retaining item and the vent and siphon block weld is helium leak tested after fuel is loaded to the same criteria as the inner top closure plate-to-shell weld (ANSI N14.5-1997 leaktight criteria).
NB-7000	*Overpressure Protection*	*No overpressure protection is provided for the NUHOMS® DSCs. The function of the DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The DSC is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature.*
NB -8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.*
NB-5520	*NDE Personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of more recent edition of SNT-TC-1A.*

Table P.3.1-2
Alternatives to the ASME Code for the NUHOMS®-24PTH DSC Basket Assembly

(Part 1 of 2)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section P.2 may be used for construction, but in no case earlier than 3 years before that specified in Section P.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section P.2 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NG-1100	*Requirements for Code Stamping of Components, Code reports and certificates, etc.*	*Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.*
NG-2000	Use of ASME Material	*Some baskets include neutron absorber and aluminum plates that are not ASME Code Class 1 material. They are used for criticality safety and heat transfer, and are only credited in the structural analysis with supporting their own weight and transmitting bearing loads through their thickness.* Material properties in the ASME Code for Type 6061 aluminum are limited to 400°F to preclude the potential for annealing out the hardening properties. Annealed properties (as published by the Aluminum Association and the American Society of Metals) are conservatively assumed for the aluminum transition rails for use above the Code temperature limits.
NG-2130	*Material must be supplied by ASME approved material suppliers.*	*Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.*
NG-4121	*Material Certification by Certificate Holder*	
NG -8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.*
NG-3000/ Section II, Part D, Table 2A	Maximum temperature limit for Type 304 plate material is 800°F	Not compliant with ASME Section II Part D Table 2A material temperature limit for Type 304 steel for the postulated transfer accident case (117°F, loss of sunshade, loss of neutron shield). This is a post-drop accident scenario, where the calculated maximum steady state temperature is 862°F, the expected reduction in material strength is small (less than 1 ksi by extrapolation), and the only primary stresses in the basket grid are deadweight stresses. The recovery actions following the postulated drop accident are as described in Section 8.2.5 of the FSAR.

Table P.3.1-2
Alternatives to the ASME Code for the NUHOMS®-24PTH DSC Basket Assembly

(*Part 2 of 2*)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NG-3352	Table NG 3352-1 lists the permissible welded joints	The fusion (spot) type welds between the stainless steel insert plates (straps) and the stainless steel fuel compartment tube are not permissible welds per Table NG-3352-1. These welds are qualified by testing. The required minimum tested capacity of the welded connection (at each side of the tube) shall be 36 Kips (at room temperature). This value is based on a margin of safety (test-to-design) of 1.6, which is larger than the Code-implied margin of safety for Level D loads. The minimum capacity shall be determined by shear tests of individual specimens made from production material. The tests shall be corrected for temperature differences (test-to-design) and for material properties (actual-to-ASME Code minimum values) to demonstrate that the capacity of the welded connection with ASME minimum properties, tested at design temperatures, will meet the 36 Kips test requirement. The capacity of the welded connection is determined from the test of the weld pattern of a typical insert plate to the tube connection. *The welds will be visually inspected to confirm that they are located over the insert plates, in lieu of the visual acceptance criteria of NG-5260 which are not appropriate for this type of weld. A joint efficiency (quality) factor of 1.0 is utilized for the fuel compartment longitudinal seam welds. Table NG-3352-1 permits a joint efficiency (quality) factor of 0.5 to be used for full penetration weld examined by ASME Section V visual examination (VT). For the 24PTH DSC,the compartment seam weld is thin and the weld will be made in one pass. Both surfaces of weld (inside and outside) will be fully examined b VT and therefore a factor of 2 x 0.5=1.0, will be used in the analysis. This is justified as both surfaces of the single weld pass/layer will be fully examined, and the stainless steel material that comprises the fuel compartment tubes is very ductile.*
NG-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A.*	*Permit use of more recent edition of SNT-TC-1A.*

Table P.4-25
Fuel Cladding Accident Condition Maximum Temperatures

Heat Load Zoning	HLZC 1[(2)] (°F)	HLZC 4 (°F)	HLZC 5 (°F)	Limit (°F)
DSC in HSM, Blocked Vent, 117°F	*891*[(1)]	*<891*[(2)]	*<821*[(3)] */ 821*[(4)]	1058[(5)]
DSC in TC, loss of sun shade, neutron shield water and air circulation with fan, if used, 117°F	914	843	747	

1) Temperature at 38.5 hour and 30.0 hour of blocked vent for louvered top heat shield/finned side heat shield and stainless steel heat shields (top and side) respectively.
2) Temperatures for HLZC 1 bound the temperatures for HLZC 2, 3, and 4.
3) Temperature for storage in HSM-H is bounded by temperature for storage in HSM Model 102.
4) Temperature for storage in HSM Model 102 at 40 hours of blocked vent.
5) ISG-11, Revision *3 [4.19]*

P.8.1 Procedures for Loading the Cask

Process flow diagrams for the NUHOMS® System operations are presented Figure P.8-1 and Figure P.8-2. The location of the various operations may vary with individual plant requirements. The following steps describe the recommended generic operating procedures for the standardized NUHOMS® System.

P.8.1.1 Preparation of the TC and DSC

1. Prior to placement in dry storage, the candidate intact and damaged fuel assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Technical Specification *2.1*.

2. Prior to being placed in service, the TC is to be cleaned or decontaminated as necessary to insure a surface contamination level of less than those specified in Technical Specification *5.2.4.d*.

3. Place the TC in the vertical position in the cask decon area using the cask handling crane and the TC lifting yoke.

4. Place scaffolding around the cask so that the top cover plate and surface of the cask are easily accessible to personnel.

5. Remove the TC top cover plate and examine the cask cavity for any physical damage and ready the cask for service. If required by the plant lifting crane capacity limit, drain the TC neutron shield water to an acceptable location.

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

7. Verify that the DSC basket type (1A, 2A etc.) is appropriate for the specific fuel loading campaign.

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

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

10. If damaged fuel assemblies are included in a specific loading campaign, place the required number of bottom end caps provided (up to a maximum of 12) into the cell locations per Technical Specification *2.1*. Place and verify that the bottom fuel assembly spacers, if required, are present in the fuel cells. Optionally, this step may be performed at any prior time.

11. Fill the DSC cavity with water from the fuel pool or an equivalent source which meets the requirements of Technical Specification *3.2.1*.

P.8.1.4 DSC Sealing Operations

CAUTION: During performance of steps listed in Section P.8.1.4, monitor the cask/DSC annulus water level and replenish as necessary to maintain cooling.

1. Disconnect the VDS from the DSC. Seal weld the prefabricated plugs over the vent and siphon ports, inject helium into blind space just prior to completing welding, and perform a dye penetrant weld examination in accordance with the Technical Specification *5.2.4.b* requirements.

2. Temporary shielding may be installed as necessary to minimize personnel exposure. Install the automatic welding machine onto the outer top cover plate and place the outer top cover plate with the automatic welding system onto the DSC. Verify proper fit up of the outer top cover plate with the DSC shell.

3. Tack weld the outer top cover plate to the DSC shell. Place the outer top cover plate weld root pass.

4. Helium leak test the inner top cover plate and vent/siphon port plate welds using the leak test port in the outer top cover plate in accordance with Technical Specification *5.2.4.c* limits. Verify that the personnel performing the leak test are qualified in accordance with SNT-TC-1A [8.4]. Alternatively this can be done with a test head in P.8.1.4 *step* 1.

5. If a leak is found, remove the outer cover plate root pass, the vent and siphon port plugs and repair the inner cover plate welds. Repeat procedure steps from P.8.1.3 Step 18.

6. Perform dye penetrant examination of the root pass weld. Weld out the outer top cover plate to the DSC shell and perform dye penetrant examination on the weld surface in accordance with the Technical Specification *5.2.4.b* requirements.

7. Seal weld the prefabricated plug over the outer cover plate test port and perform dye penetrant weld examinations.

8. Remove the automatic welding machine from the DSC.

9. Open the cask drain port valve and drain the water from the cask/DSC annulus.

10. Rig the cask top cover plate and lower the cover plate onto the TC.

11. Bolt the cask cover plate into place, tightening the bolts to the required torque in a star pattern.

CAUTION: Monitor the applicable time limits of Technical Specification *3.1.3* until the completion of DSC transfer Step 6 of Section P.8.1.6.

P.8.1.5 TC Downending and Transfer to ISFSI

1. If loading with OS197/OS197H/OS197FC TC, drain the TC neutron shield to an acceptable location as required to meet the plant lifting crane capacity limit.
2. Re-attach the TC lifting yoke to the crane hook, as necessary. Ready the transfer trailer and cask support skid for service.
3. Move the scaffolding away from the cask as necessary. Engage the lifting yoke and lift the cask over the cask support skid on the transfer trailer.
4. The transfer trailer should be positioned so that cask support skid is accessible to the crane with the trailer supported on the vertical jacks.
5. Position the cask lower trunnions onto the transfer trailer support skid pillow blocks.
6. Move the crane forward while simultaneously lowering the cask until the cask upper trunnions are just above the support skid upper trunnion pillow blocks.
7. Inspect the positioning of the cask to insure that the cask and trunnion pillow blocks are properly aligned.
8. Lower the cask onto the skid until the weight of the cask is distributed to the trunnion pillow blocks.
9. Inspect the trunnions to ensure that they are properly seated onto the skid and install the trunnion tower closure plates (optional for the OS197 TC and the OS197H TC).
10. Fill the neutron shield, if it was drained in P.8.1.5 step 1.
11. Remove the bottom ram access cover plate from the cask. Install the two-piece temporary neutron/gamma shield plug to cover the bottom ram access. Install the ram trunnion support frame on the bottom of the TC. (The temporary shield plug and ram trunnion support frame are not required with integral ram/trailer).

P.8.1.6 DSC Transfer to the HSM

1. Prior to transferring the cask to the ISFSI, remove the HSM door using a porta-crane, inspect the cavity of the HSM, removing any debris and ready the HSM to receive a DSC. The doors on adjacent HSMs should remain in place.

   Caution: The insides of empty modules have the potential for high does rates due to adjacent loaded modules. Proper ALARA practices should be followed for operations inside these modules and in the areas outside these modules whenever the door from the empty HSM has been removed.

2. Inspect the HSM air inlet and outlets to ensure that they are clear of debris. Inspect the screens on the air inlet and outlets for damage.

   CAUTION: Verify that the requirements of Technical Specification *5.3.1.B*, "TC/DSC Transfer Operations at High Ambient Temperatures" are met prior to next step.

3. Using a suitable vehicle, transfer the cask from the plant's fuel/reactor building to the ISFSI along the designated transfer route.

4. Once at the ISFSI, position the *transfer* trailer to within a few feet of the HSM.

5. Check the position of the trailer to ensure the centerline of the HSM and cask approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.

6. Unbolt and remove the cask top cover plate.

   CAUTION: Verify that the applicable time limits of Technical Specification *3.1.3* are met.

7. Back the cask to within a few inches of the HSM, set the trailer brakes and disengage the tractor. Extend the transfer trailer vertical jacks.

8. Connect the skid positioning system hydraulic power unit to the positioning system via the hose connector panel on the trailer, and power it up. Remove the skid tie-down bracket fasteners and use the skid positioning system to bring the cask into approximate vertical and horizontal alignment with the HSM. Using optical survey equipment and the alignment marks on the cask and the HSM, adjust the position of the cask until it is properly aligned with the HSM.

9. Using the skid positioning system, fully insert the cask into the HSM access opening docking collar.

10. Secure the cask trunnions to the front wall embedments of the HSM using the cask restraints.

11. After the cask is docked with the HSM, verify the alignment of the TC using the optical survey equipment.

12. Position the hydraulic ram behind the cask in approximate horizontal alignment with the cask and level the ram. Remove either the bottom ram access cover plate or the outer plug of the two-piece temporary shield plug. Power up the ram hydraulic power supply and extend the ram through the bottom cask opening into the DSC grapple ring.

13. Activate the hydraulic cylinder on the ram grapple and engage the grapple arms with the DSC grapple ring.

14. Recheck all alignment marks in accordance with the Technical Specification *5.3.3* limits and ready all systems for DSC transfer.

15. Activate the hydraulic ram to initiate insertion of the DSC into the HSM. Stop the ram when the DSC reaches the support rail stops at the back of the module.

16. Disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.

17. Retract and disengage the hydraulic ram system from the cask and move it clear of the cask. Remove the cask restraints from the HSM.

18. Using the skid positioning system, disengage the cask from the HSM access opening.

19. Install the DSC drop-in retainer through the HSM door opening.

20. The trailer may be moved as necessary to install the HSM door. Install the HSM door and secure it in place. Door may be welded for security. Verify that the HSM dose rates are compliant with the limits specified in Technical Specification *5.4.2.*

21. Replace the TC top cover plate. Secure the skid to the trailer, retract the vertical jacks and disconnect the skid positioning system.

22. Tow the trailer and cask to the designated equipment storage area. Return the remaining transfer equipment to the storage area.

23. Close and lock the ISFSI access gate and activate the ISFSI security measures.

P.8.1.7 <u>Monitoring Operations</u>

1. Perform routine security surveillance in accordance with the licensee's ISFSI security plan.

2. Perform a daily visual surveillance of the HSM air inlets and outlets to insure that no debris is obstructing the HSM vents in accordance with Technical Specification *5.2.5.a* requirements OR perform a temperature measurement of the thermal performance, for each HSM, on a daily basis in accordance with Technical Specification *5.2.5.b* requirements.

P.9.1.4 Component Tests

No change.

P.9.1.5 Shielding Integrity Tests

No change.

P.9.1.6 Thermal Acceptance Tests

The analyses to ensure that the NUHOMS®-24PTH system is capable of performing their heat transfer function are presented in Section P.4.

P.9.1.7 Poison Acceptance

CAUTION
Sections P.9.1.7.1 through P.9.1.7.3 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications (paragraph 4.1 (Note 3)) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated aluminum

(b) Boron carbide/aluminum metal matrix composite (MMC)

(c) Boral®

The 24PTH DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content for these materials is given in Table P.9-1.

References to metal matrix composites throughout this chapter are not intended to refer to Boral®, which is described later in this section.

P.9.1.7.1 **Borated Aluminum**

See the Caution in Section P.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating *primarily* as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy *(other boron compounds, such as AlB_{12}, can also occur).* For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section P.9.1.7.5. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings"[9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surface, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

P.9.1.7.2 Boron Carbide/Aluminum Metal Matrix Composites (MMC)

See the Caution in Section P.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

Boron carbide particles for the products considered here shall have an average size of 40 microns or less. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 24PTH DSC, MMCs shall pass the qualification testing specified in Section P.9.1.7.6, and shall subsequently be subject to the process controls specified in Section P.9.1.7.7.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section P.9.1.7.5. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surfaces, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

P.9.1.7.3 **Boral®**

See the Caution in Section P.9.1.7 before deletion or modification to this section.
This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. ***Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns.*** **The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.**

The criticality calculations take credit for 75% of the minimum specified B10 areal density of Boral®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on an approximately 1 cm^2 area of a coupon taken near one of the corners of the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

Visual inspections shall verify that the Boral® core is not exposed through the face of the sheet at any location.

P.9.1.7.4 Thermal Conductivity Testing of Poison Plates

The poison plate material shall be qualification tested to verify that the thermal conductivity equals or exceeds the values listed in Section P.4.3. *Acceptance testing shall be performed at room temperature on coupons taken from the rolled or extruded production material. Initial sampling shall be one test per lot, defined by the heat or ingot, and may be reduced if the first five tests meet the specified minimum thermal conductivity.*

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

Testing *shall* be by ASTM E1225 [9.3], ASTM E1461 [9.4], or equivalent method.

P.9.1.7.4.1 Test Coupon and Lot Definitions

A sample taken from the plate material is a test coupon. Test coupons will be removed so that there is at least one coupon contiguous with each plate. These coupons will be used for neutron thermal conductivity testing. The minimum dimension of the coupon shall be as required for the acceptance test procedures.

A lot is defined as all the plates produced from a single cast ingot, or all the plates produced from a single heat.

P.9.1.7.5 Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission

CAUTION
Section P.9.1.7.5 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 3) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1.2 centimeter diameter. A beam size greater than 1.2 centimeter diameter but no larger than 1.7 centimeter diameter may be used if computations are performed to demonstrate that the calculated $k_{effective}$ of the system is still below the calculated Upper Subcritical Limit (USL) of the system assuming defect areas the same area as the beam.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard.

Alternatively, digital image analysis may be used to compare neutron radioscopic images of the test coupon to images of the standards. The area of image analysis shall be up to 1.1 cm^2. The method shall demonstrate sufficient sensitivity to distinguish between areal density at the specified minimum, 1% above and below the minimum.

(Alternatively show that the material fails in a ductile manner, e.g., by scanning electron microscopy of the fracture surface or by bend testing.)

and

b) testing (ASTM-B311[4]) to verify more than 98% (or 97% for MMCs with integral aluminum cladding) of theoretical density. Testing or examination for exposed interconnected porosity shall be performed by a means to be approved by the Certificate Holder.

P.9.1.7.6.5 Required Tests and Examinations to Demonstrate B10 Uniformity

CAUTION
Section P.9.1.7.6.5 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note3) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Uniformity of the boron distribution shall be verified either by:

a) Neutron radioscopy or radiography (ASTM E94[5], E142[6], and E545[7]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

b) Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section P.9.1.7.5, or by chemical analysis for boron carbide content in the composite.

P.9.1.7.6.6 Approval of Procedures

Qualification procedures shall be subject to approval by the Certificate Holder.

P.9.1.7.7 Specification for Process Controls for Metal Matrix Composites

P.9.1.7.7.1 Applicability and Scope

The applicability of this section is the same as that of Section P.9.1.7.6. It addresses the process controls to ensure that the material delivered for use is equivalent to the qualification test material.

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4 ASTM B311, Test Method for Density Determination for Powder Metallurgy (P/M) Materials Containing Less than Two Percent Porosity

5 ASTM E94, Recommended Practice for Radiographic Testing

6 ASTM E142, Controlling Quality of Radiographic Testing

7 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section P.9.1.7.7 is required, depending on the characteristics of the material that could be affected by the process change.

P.9.1.7.7.2 Definition of Key Process Changes

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, or reduce the mechanical strength or ductility of the MMC.

P.9.1.7.7.3 Identification and Control of Key Process Changes

CAUTION
Section P.9.1.7.7.3 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specifications 4.1 (Note 3) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section P.9.1.7.7.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that may be established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a 6000 series aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature, and

Table T.3.1-2
ASME Code Alternatives for the NUHOMS®-61BTH DSC Confinement Boundary

(Part 1 of 2)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section T.2 may be used for construction, but in no case earlier than 3 years before that specified in Section T.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section T.2 may be used, so as long the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NB-1100	*Requirements for Code Stamping of Components, Code reports and certificates, etc.*	*Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.*
NB-1132	*Attachments with a pressure retaining function, including stiffeners, shall be considered part of the component.*	*Bottom shield plug and outer bottom cover plate are outside code jurisdiction; these components together are much larger than required to provide stiffening for the inner bottom cover plate; the weld that retains the outer bottom cover plate and with it the bottom shield plug is subject to root and final PT examination.*
NB-2130	*Material must be supplied by ASME approved material suppliers.*	*Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.*
NB-4121	*Material Certification by Certificate Holder*	
NB-4243 and NB-5230	Category C weld joints in vessels and similar weld joints in other components shall be full penetration joints. These welds shall be examined by UT or RT and either PT or MT.	The shell to the outer top cover weld, the shell to the inner top cover/weld, the siphon/vent cover welds and the vent and siphon block welds to the shell are all partial penetration welds. As an alternative to the NDE requirements of NB-5230 for Category C welds, all of these closure welds will be multi-layer welds and receive a root and final PT examination, except for the shell to the outer top cover weld. The shell to the outer top cover weld will be a multi-layer weld and receive multi-level PT examination in accordance with the guidance provided in ISG-15 for NDE. The multi-level PT Examination provides reasonable assurance that flaws of interest will be identified. The PT examination is done by qualified personnel, in accordance with Section V and the acceptance standards of Section III, Subsection NB-5000. All of these welds will be designed to meet the guidance provided in ISG-15 for stress reduction factor.

Table T.3.1-2
ASME Code Alternatives for the NUHOMS®-61BTH DSC Confinement Boundary
(Part 2 of 2)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NB-6100 and 6200	All completed pressure retaining systems shall be pressure tested	The 61BTH is not a complete or "installed" pressure vessel until the top closure is welded following placement of Fuel Assemblies with the DSC. Due to the inaccessibility of the shell and lower end closure welds following fuel loading and top closure welding, as an alternative, the pressure testing of the DSC is performed in two parts. The DSC shell (including all longitudinal and circumferential welds) is pressure tested and examined at the fabrication facility. The shell to the inner top cover closure weld are pressure tested and examined for leakage in accordance with NB-6300 in the field. The siphon/vent cover welds are not pressure tested; these welds and the shell to the inner top cover closure weld are helium leak tested after the pressure test. Per NB-6324 the examination for leakage shall be done at a pressure equal to the greater of the design pressure or three-fourths of the test pressure. As an alternative, if the examination for leakage of these field welds, following the pressure test, is performed using helium leak detection techniques, the examination pressure may be reduced to ≥ 1.5 psig. This is acceptable given the significantly greater sensitivity of the helium leak detection method.
NB-7000	*Overpressure Protection*	*No overpressure protection is provided for the NUHOMS® DSCs. The function of the DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The DSC is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature.*
NB-8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.*
NB-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of more recent edition of SNT-TC-1A*

Table T.3.1-3
ASME Code Alternatives for the NUHOMS®-61BTH DSC Basket

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section T.2 may be used for construction, but in no case earlier than 3 years before that specified in Section T.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section T.2 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NG/NF-1100	*Requirements for Code Stamping of Components, Code reports and certificates, etc.*	*Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.*
NG/NF-2000	Use of ASME Material	*Some baskets include neutron absorber and aluminum plates that are not ASME Code Class 1 material. They are used for criticality safety and heat transfer, and are only credited in the structural analysis with supporting their own weight and transmitting bearing loads through their thickness.* Material properties in the ASME Code for Type 6061 aluminum are limited to 400°F to preclude the potential for annealing out the hardening properties. Annealed properties (as published by the Aluminum Association and the American Society of Metals) are conservatively assumed for the aluminum transition rails for use above the Code temperature limits.
NG/NF-2130	Material must be supplied by ASME approved material suppliers.	Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.
NG/NF-4121	Material Certification by Certificate Holder	
NG-3352	Table NG 3352-1 lists the permissible welded joints and quality factors.	The fuel compartment tubes may be fabricated from sheet with full penetration seam weldments. Per Table NG-3352-1 a joint efficiency (quality) factor of 0.5 is to be used for full penetration weldments examined in accordance with ASME Section V visual examination (VT). A joint efficiency (quality) factor of 1.0 is utilized for the fuel compartment longitudinal seam welds (if present) with VT examination. This is justified because the compartment seam weld is thin and the weldment is made in one pass; and both surfaces of the weldment (inside and outside) receive 100% VT examination. The 0.5 quality factor, applicable to each surface of the weldment, results is a quality factor of 1.0 since both surfaces are 100% examined. In addition,the fuel compartments have no pressure retaining function and the stainless steel material that comprises the fuel compartment tubes is very ductile.
NG/NF-8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.*
NG/NF-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of more recent edition of SNT-TC-1A*

b. Position the lifting yoke and the top shield plug and lower the shield plug into the DSC. Note that separate rigging may be used to install the shield plug prior to engaging the trunnions with the lifting yoke.

CAUTION: Verify that all the lifting height restrictions as a function of temperature specified in Technical Specification 5.3.1.A can be met in the following steps which involve lifting of the transfer cask.

9. Visually verify that the top shield plug is properly seated within the DSC.

10. Position the lifting yoke with the cask trunnions and verify that it is properly engaged.

11. Raise the transfer cask to the pool surface. Prior to raising the top of the cask above the water surface, stop vertical movement.

12. Inspect the top shield plug to verify that it is properly seated within the DSC. If not, lower the cask and reposition the top shield plug and/or remove the shield plug and reposition the hold down ring. Repeat Steps 8 through 12 as necessary.

13. Continue to raise the cask from the pool and spray the exposed portion of the cask with water until the top region of the cask is accessible.

14. Drain any excess water from the top of the DSC shield plug back to the fuel pool. Check the radiation levels at the center of top shield plug and around the perimeter of the cask. Disconnect the top shield plug rigging.

15. Drain a minimum of 50 gallons of water. Optionally approximately 1100 gallons of water (as indicated on the flow meter) may be drained from the DSC back into the fuel pool or other suitable location to meet the weight limit on the crane. Use 1-3 psig of helium to backfill the DSC with an inert gas per ISG-22 [8.2] guidance as water is being removed from the DSC.

16. Lift the cask from the fuel pool. As the cask is raised from the pool, continue to spray the cask with water and decon as directed. Provisions shall be made to assure that air will not enter the DSC cavity. One way to achieve this is by replenishing the helium in the DSC cavity during cask movement from the fuel pool to the decon area in case of malfunction of equipments used for cask movement.

17. Move the cask with loaded DSC to the cask decon area.

17A. Replace the water removed from the DSC cavity in Step 15 with water from the fuel pool or an equivalent source.

18. Install cask seismic restraints if required by Technical Specification 4.3.3 step 7 (required only on plant specific basis).

19. Verify that the transfer cask dose rates are compliant with limits specified in Technical Specification *5.2.4.*

T.8.1.3 DSC Drying and Backfilling

CAUTION: During performance of steps listed in Section T.8.1.3, monitor the TC/DSC annulus water level and replenish as necessary until drained.

1. Check the radiation levels along the perimeter of the cask. The cask exterior surface should be decontaminated as necessary in accordance with the limits specified in Technical Specification 5.2.4.d. Temporary shielding may be installed as necessary to minimize personnel exposure.

2. Place scaffolding around the cask so that any point on the surface of the cask is easily accessible to personnel.

3. Disengage the rigging cables from the top shield plug and remove the eyebolts. Disengage the lifting yoke from the trunnions and position it clear of the cask.

4. Decontaminate the exposed surfaces of the DSC shell perimeter and remove the inflatable TC/DSC annulus seal.

5. Connect the cask drain line to the cask, open the cask cavity drain port and allow water from the annulus to drain out until the water level is approximately 12 inches below the top edge of the DSC shell. Take swipes around the outer surface of the DSC shell and check for smearable contamination in accordance with the Technical Specification 5.2.4.d limits.

CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

6. Drain approximately 1100 gallons of water (as indicated on a flowmeter) from the DSC back into the fuel pool or other suitable location if not drained in T.8.1.2 step 15. Consistent with ISG-22 [8.2] guidance, helium at 1-3 psig is used to backfill the DSC with an inert gas (helium) as water is being removed from the DSC.

7. Monitor TC/DSC annulus water level and replenish as necessary until drained.

8. Install the automatic welding machine onto the inner top cover plate and place the inner top cover plate with the automatic welding machine onto the DSC. Optionally, the inner top cover plate and the automatic welding machine can be placed separately. Verify proper fit-up of the inner top cover plate with the DSC shell.

9. Check radiation levels along the surface of the inner top cover plate. Temporary shielding may be installed as necessary to minimize personnel exposure.

10. Insert approximately ¼ inch tubing of sufficient length and adequate temperature resistance through the vent port such that it terminates just below the DSC top shield plug. Connect the tubing to a hydrogen monitor to allow continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner top cover plate, in compliance with Technical Specification *5.2.6*.

11. Cover the TC/DSC annulus to prevent debris and weld splatter from entering the annulus.

pressure monitored. The cavity pressure will rise as water and other volatiles in the cavity evaporate. When the cavity pressure stabilizes, the pump is valved in to complete the vacuum drying process. It may be necessary to repeat some steps, depending on the rate and extent of the pressure increase. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less as specified in Technical Specification 3.1.1.

Note: The user shall ensure that the vacuum pump is isolated from the DSC cavity when demonstrating compliance with *Technical Specification 3.1.1* requirements. Simply closing the valve between the DSC and the vacuum pump is not sufficient, as a faulty valve allows the vacuum pump to continue to draw a vacuum on the DSC. Turning off the pump, or opening the suction side of the pump to atmosphere are examples of ways to assure that the pump is not continuing to draw a vacuum on the DSC.

CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

22. Open the valve to the vent port and allow the helium to flow into the DSC cavity.
23. Pressurize the DSC with helium (up to 10 psig for Type 1 DSC or 15 psig for Type 2 DSC).
24. Helium leak test the inner top cover plate weld for a leak rate of 1 x 10^{-4} atm·cm^3/sec. This test is optional.
25. If a leak is found, repair the weld, repressurize the DSC and repeat the helium leak test.
26. Once no leaks are detected, depressurize the DSC cavity by releasing the helium through the VDS to the plant's spent fuel pool or radioactive waste system.
27. Re-evacuate the DSC cavity using the VDS. The cavity pressure should be reduced in steps of approximately 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level, the pump is valved off and the cavity pressure is monitored (these levels are optional). When the cavity pressure stabilizes, the pump is valved in to continue the vacuum drying process. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less in accordance with Technical Specification 3.1.1 limits.

    Note: The user shall ensure that the vacuum pump is isolated from the DSC cavity when demonstrating compliance with *Technical Specification 3.1.1* requirements. Simply closing the valve between the DSC and the vacuum pump is not sufficient, as a faulty valve allows the vacuum pump to continue to draw a vacuum on the DSC. Turning off the pump, or opening the suction side of the pump to atmosphere are examples of ways to assure that the pump is not continuing to draw a vacuum on the DSC.
28. Open the valve on the vent port and allow helium to flow into the DSC cavity to pressurize the DSC between 14.5 to 16.0 psig for 61BTH Type 1 and 18.5 to 20.0 psig for 61BTH Type 2 and hold for 10 minutes. Depressurize the DSC cavity by releasing the helium through the VDS to the plant spent fuel pool or radioactive waste system to about 2.5 psig in accordance with Technical Specification 3.1.2.b limits.

CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

29. Close the valves on the helium source.

30. Remove the strongback, if installed in step 14 above, decontaminate as necessary, and store.

T.8.1.4 DSC Sealing Operations

CAUTION: During performance of steps listed in Section T.8.1.4, monitor the cask/DSC annulus water level and replenish as necessary to maintain cooling.

16. Disengage the ram grapple mechanism so that the grapple is retracted away from the DSC grapple ring.

17. Retract and disengage the hydraulic ram system from the cask and move it clear of the cask. Remove the cask restraints from the HSM.

18. Using the skid positioning system, disengage the cask from the HSM access opening. Insert the DSC axial retainer.

19. Install the HSM door using a portable crane and secure it in place. Door may be welded for security. Verify that the HSM dose rates are compliant with the limits specified in Technical Specification 5.4.2.

20. Replace the transfer cask top cover plate. Secure the skid to the trailer, retract the vertical jacks and disconnect the skid positioning system.

21. Tow the trailer and cask to the designated equipment storage area. Return the remaining transfer equipment to the storage area.

22. Close and lock the ISFSI access gate and activate the ISFSI security measures.

23. Ensure the HSM-H maximum air exit temperature requirements of Technical Specification *3.1.4* are met.

T.8.1.7 Monitoring Operations

1. Perform routine security surveillance in accordance with the licensee's ISFSI security plan.

2. Perform **one** of the two alternate daily surveillance activities listed below:

   a. A daily visual surveillance of the HSM air inlets and outlets to insure that no debris is obstructing the HSM vents in accordance with Technical Specification 5.2.5.a requirements.

   b. A temperature measurement of the thermal performance, for each HSM, on a daily basis in accordance with Technical Specification 5.2.5.b requirements.

T.9.1.7 Poison Acceptance

> ***CAUTION***
>
> *Sections T.9.1.7.1 through T.9.1.7.3 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.*

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated aluminum

(b) Boron carbide/aluminum metal matrix composite (MMC)

(c) Boral®

The 61BTH DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content of these three types of materials is given in Table T.9-1, Table T.9-2 and Table T.9-3, respectively.

References to metal matrix composites throughout this chapter are not intended to refer to Boral®, which is described later in this section.

T.9.1.7.1 **Borated Aluminum**

See the Caution in Section T.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating *primarily* as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy *(other boron compounds, such as AlB_{12}, can also occur).* For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section T.9.1.7.5. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings"[9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surface, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

T.9.1.7.2 Boron Carbide / Aluminum Metal Matrix Composites (MMC)

See the Caution in Section T.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

Boron carbide particles for the products considered here shall have an average size of 40 microns or less. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 61BTH DSC, MMCs shall pass the qualification testing specified in Section T.9.1.7.6, and shall subsequently be subject to the process controls specified in Section T.9.1.7.7.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section T.9.1.7.5. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surfaces, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

References to metal matrix composites throughout this chapter are not intended to refer to Boral®, which is described in the following section.

T.9.1.7.3 Boral®

See the Caution in Section T.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. *Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns.* **The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.**

The criticality calculations take credit for 75% of the minimum specified B10 areal density of Boral®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on an approximately 1 cm^2 area of a coupon taken near one of the corners of the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

Visual inspections shall verify that the Boral® core is not exposed through the face of the sheet at any location.

T.9.1.7.4 Thermal Conductivity Testing

All poison plate materials except Boral® will be qualification tested to verify that the thermal conductivity equals or exceeds the values listed in Section T.4.3. *Acceptance testing shall be performed at room temperature on coupons taken from the rolled or extruded production material. Initial sampling shall be one test per lot, defined by the heat or ingot, and may be reduced if the first five tests meet the specified minimum thermal conductivity.*

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

Testing shall conform to ASTM E1225 [9.9], ASTM E1461 [9.10], or equivalent method.

T.9.1.7.5 Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission

CAUTION
Section T.9.1.7.5 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1.2 centimeter diameter. A beam size greater than 1.2 centimeter diameter but no larger than 1.7 centimeter diameter may be used if computations are performed to demonstrate that the calculated $k_{effective}$ of the system is still below the calculated Upper Subcritical Limit (USL) of the system assuming defect areas the same area as the beam.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard.

Alternatively, digital image analysis may be used to compare neutron radioscopic images of the test coupon to images of the standards. The area of image analysis shall be up to 1.1 cm^2. The method shall demonstrate sufficient sensitivity to distinguish between areal density at the specified minimum, and 1% above and below the minimum.

(Alternatively show that the material fails in a ductile manner, e.g., by scanning electron microscopy of the fracture surface or by bend testing.)

and

b) testing (ASTM-B311[4]) to verify more than 98% (or 97% for MMCs with integral aluminum cladding) of theoretical density. Testing or examination for exposed interconnected porosity shall be performed by a means to be approved by the Certificate Holder.

T.9.1.7.6.5 Required Tests and Examinations to Demonstrate B10 Uniformity

CAUTION

Section T.9.1.7.6.5 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Uniformity of the boron distribution shall be verified either by:

a) Neutron radioscopy or radiography (ASTM E94[5], E142[6], and E545[7]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

b) Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section 9.1.7.5, or by chemical analysis for boron carbide content in the composite.

T.9.1.7.6.6 Approval of Procedures

Qualification procedures shall be subject to approval by the Certificate Holder.

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4 ASTM B311, Test Method for Density Determination for Powder Metallurgy (P/M) Materials Containing Less than Two Percent Porosity

5 ASTM E94, Recommended Practice for Radiographic Testing

6 ASTM E142, Controlling Quality of Radiographic Testing

7 ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

T.9.1.7.7 Specification for Process Controls for Metal Matrix Composites

T.9.1.7.7.1 Applicability and Scope

The applicability of this section is the same as that of Section T.9.1.7.6. It addresses the process controls to ensure that the material delivered for use is equivalent to the qualification test material.

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section T.9.1.7.7 is required, depending on the characteristics of the material that could be affected by the process change.

T.9.1.7.7.2 Definition of Key Process Changes

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, or reduce the mechanical strength or ductility of the MMC.

T.9.1.7.7.3 Identification and Control of Key Process Changes

CAUTION
Section T.9.1.7.7.3 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 4) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section T.9.1.7.7.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that may be established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

Normal Conditions

Normal Condition Stress Criteria for Steel Elements

As summarized in Table U.2-15, the normal condition stress criteria for the fuel compartment tubes and the transition rails, is based on Subsection NG of the ASME Code, Section III [2.2].

Normal Condition Stress Criteria for Aluminum Transition Rails

The aluminum transition rail bodies perform their function (support of the fuel compartment tubes) by remaining in place. The loads on the rail bodies are primarily bearing from the fuel compartment tubes. "Failure" of the transition rail would require that the rail no longer provide support to the fuel compartment tubes. Since the aluminum rail bodies are constrained between the DSC shell and the fuel support compartment tubes, this cannot occur.

Therefore, for deadweight and handling condition loads, stress in the aluminum bodies will be compared to the allowable bearing stress, equal to S_y, from NG-3227.1(a). Values of S_y are taken from Table U.3.3-5 for annealed 6061 aluminum material at temperature (as described in Section U.3.3, these yield stresses are lower bound values).

Accident Conditions

Accident Condition Stress Criteria for Steel Elements

As summarized in Table U.2-15 the accident condition (Level D) stress criteria for the fuel support structure and the welded steel transition rails is based on Appendix F of the ASME Code, Section III. Criteria are provided for both linear elastic and elastic-plastic stress analyses.

Accident Condition Criteria for Aluminum Transition Rails

For accident condition loading (i.e., the postulated drops), the aluminum transition rail bodies must support the fuel tubes such that stresses and displacements in the fuel compartment tubes are acceptable. Since, the rail bodies are captured between the fuel compartment tube and the DSC shell, large displacements of the rails are prevented. Thus, no additional checks (of the aluminum) are required for accident/drop loading. Qualification of the fuel tubes demonstrates that the rails perform their intended function.

U.2.2.5.2 NUHOMS® HSM-H Structural Design Criteria

There are no changes to the HSM-H structural design criteria presented in Appendix P.2, except for the modified earthquake loads (EQ) as discussed in Section U.2.2.3 above.

U.2.2.5.3 NUHOMS® OS200 TC Structural Design Criteria

There are no changes to the design criteria presented in Table 3.2-1 Section 3 of the UFSAR for the OS197/OS197H TCs, except for the modified earthquake loads (EQ) of 0.3 g horizontal and 0.25 g vertical ZPA accelerations as discussed in Section U.2.2.3 and the maximum decay heat load of 40.8 kW for the OS200 TC (similar to the OS197FC TC described in Appendix P) *and the edition year of the ASME Code.*

For the high seismic accident scenario with maximum site accelerations of 1.0 g horizontal and 1.0 g vertical, the 75 g accident drop evaluation criteria is considered bounding. *The OS200 TC is designed and fabricated in accordance with the 1998 edition with addenda through 2000 of the ASME Code Section III, Subsection NC and the alternative provisions to the ASME Code as described in Table U.3.1-3.*

U.2.5 Summary of NUHOMS®-32PTH1 DSC and HSM-H Design Criteria

U.2.5.1 32PTH1 DSC Design Criteria

The NUHOMS®-32PTH1 DSC is designed to store intact and/or damaged PWR fuel assemblies with or without Control Components with assembly average burnup, initial enrichment and cooling time as described in Table U.2-1 and Table U.2-3. The maximum decay heat load of the stored fuel is limited to 1.5 kW per fuel assembly for Type 1 DSC and 0.98 kW for a Type 2 DSC. The maximum heat load per canister is limited to 40.8 kW for a Type 1 DSC and 31.2 kW for a Type 2 DSC in order to keep the maximum fuel cladding temperature below the limit [2.5] necessary to ensure cladding integrity. The fuel cladding integrity is assured by the NUHOMS®-32PTH1 DSC and basket design which limit fuel cladding temperature and maintains a nonoxidizing environment in the DSC cavity as described in Section U.4.

The NUHOMS®-32PTH1 DSC (shell and closure) is designed and fabricated as a Class 1 component in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [2.2], and the alternative provisions to the ASME Code as described in Table U.3.1-1.

The NUHOMS®-32PTH1 DSC is designed to maintain a subcritical configuration during loading, handling, storage and accident conditions. A combination of fixed neutron absorbers, soluble boron in the pool and favorable geometry are employed to maintain the upper subcritical limit of 0.9411. The fixed neutron absorbers are in the form of plates made from either borated aluminum alloy or MMC or Boral®. The basket is designed and fabricated in accordance with the rules of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NG, Article NG-3200 [2.2] and the alternative provisions to the ASME Code as described in Table U.3.1-2.

The NUHOMS®-32PTH1 DSC is designed to withstand the effects of severe environmental conditions and natural phenomena such as earthquakes, tornadoes, lightning and floods. Chapter U.11 describes the NUHOMS®-32PTH1 DSC behavior under these accident conditions.

The NUHOMS®-32PTH1 DSC design, fabrication and testing are covered by Transnuclear's Quality Assurance Program, which conforms to the criteria in Subpart G of 10CFR72.

U.2.5.2 HSM-H Design Criteria

There is no change to the HSM-H design criteria presented in Appendix P, Chapter P.2 except for the modified seismic loads as discussed in Section U.2.2.

U.2.5.3 OS200 TC Design Criteria

Same as the OS197/OS197H/OS197FC TC described in Section 3 and Appendix P of the UFSAR with modified seismic loads as described in U.2.2. *The OS200 TC is designed and fabricated in accordance with the 1998 edition with addenda through 2000 of the ASME Code, section III, Subsection NC and the alternatives to the ASME Code as described in Table U.3.1-3.*

Table U.3.1-1
Alternatives to the ASME Code for the NUHOMS® 32PTH1 DSC Confinement Boundary

(Part 1 of 2)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section U.2 may be used for construction, but in no case earlier than 3 years before that specified in Section U.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section U.2 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NB-1100	*Requirements for Code Stamping of Components, Code reports and certificates, etc.*	*Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.*
NB-2130	Material must be supplied by ASME approved material suppliers.	Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.
NB-4121	Material Certification by Certificate Holder	
NB-4243 and NB-5230	Category C weld joints in vessels and similar weld joints in other components shall be full penetration joints. These welds shall be examined by UT or RT and either PT or MT	The shell to the outer top cover weld, the shell to the inner top cover/shield plug weld (including optional design configurations for the inner top cover as described in the 32PTH1 DSC drawings), the siphon/vent cover welds, and the vent and siphon block welds to the shell are all partial penetration welds. As an alternative to the NDE requirements of NB-5230, for Category C welds, all of these closure welds are multi-layer welds and receive a root and final PT examination, except for the shell to the outer top cover weld. The shell to the outer top cover weld will be a multi-layer weld and receive multi-level PT examination in accordance with the guidance provided in ISG-15 for NDE. The multi-level PT examination provides reasonable assurance that flaws of interest will be identified. The PT examination is done by qualified personnel, in accordance with Section V and the acceptance standards of Section III, Subsection NB-5000. All of these welds are designed to meet the guidance provided in ISG-15 for stress reduction factor.
NB-1132	*Attachments with a pressure retaining function, including stiffeners, shall be considered part of the component.*	*Bottom shield plug and outer bottom cover plate are outside code jurisdiction; these components together are much larger than required to provide stiffening for the inner bottom cover plate; the weld that retains the outer bottom cover plate and with it the bottom shield plug is subject to root and final PT examination.*

Table U.3.1-1
Alternatives to the ASME Code for the NUHOMS® 32PTH1 DSC Confinement Boundary
(Part 2 of 2)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NB-6100 and 6200	All pressure retaining components and completed systems shall be pressure tested. The preferred method shall be hydrostatic test.	The NUHOMS® 32PTH1 DSC is not a complete vessel until the top closure is welded following placement of fuel assemblies within the DSC. Due to the inaccessibility of the shell and lower end closure welds following fuel loading and top closure welding, as an alternative, the pressure testing of the DSC is performed in two parts. The DSC shell and inner bottom plate/forging (including all longitudinal and circumferential welds), are pressure tested and examined at the fabrication facility. The shell to the inner top cover/shield plug closure weld (including optional design configurations for the inner top cover as described in the 32PTH1 DSC drawings) is pressure tested and examined for leakage in accordance with NB-6300 in the field. The siphon/vent cover welds are not pressure tested; these welds and the shell to the inner top cover/shield plug closure weld (including Optional design configurations for the inner top cover as described in the 32PTH1 DSC drawings) are helium leak tested after the pressure test. Per NB-6324 the examination for leakage shall be done at a pressure equal to the greater of the design pressure or three-fourths of the test pressure. As an alternative, if the examination for leakage of these field welds, following the pressure test, is performed using helium leak detection techniques, the examination pressure may be reduced to $\geq$1.5 psig. This is acceptable given the significantly greater sensitivity of the helium leak detection method.
NB-7000	*Overpressure Protection*	*No overpressure protection is provided for the NUHOMS® DSCs. The function of the DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The DSC is designed to withstand the maximum possible internal pressure considering 100% fuel rod failure at maximum accident temperature.*
NB-8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.*
NB-5520	*NDE Personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of more recent edition of SNT-TC-1A.*

Table U.3.1-2
Alternatives to the ASME Code for the NUHOMS® 32PTH1 DSC Basket Assembly
(Part 1 of 2)

Reference ASME Code Section/Article	*Code Requirement*	*Alternatives, Justification & Compensatory Measures*
NCA	*All*	*Not compliant with NCA. Quality Assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of Code editions and addenda*	*Code edition and addenda other than those specified in Section U.2 may be used for construction, but in no case earlier than 3 years before that specified in Section U.2.* *Materials produced and certified in accordance with ASME Section II material specification from Code Editions and Addenda other than those specified in Section U.2 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable Subsection of the Section III Edition and Addenda used for construction.*
NG-1100	*Requirements for Code Stamping of Components, Code reports and certificates, etc.*	*Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.*
NG-2000	Use of ASME Material	Some baskets include neutron absorber and aluminum plates that are not ASME Code Class 1 material. They are used for criticality safety and heat transfer, and are only credited in the structural analysis with supporting their own weight and transmitting bearing loads through their thickness. Material properties in the ASME Code for Type 6061 aluminum are limited to 400°F to preclude the potential for annealing out the hardening properties. Annealed properties (as published by the Aluminum Association and the American Society of Metals) are conservatively assumed for the aluminum transition rails for use above the Code temperature limits.
NG-2130	*Material must be supplied by ASME approved material suppliers.*	*Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability and certification are maintained in accordance with TN's NRC approved QA program.*
NG-4121	*Material Certification by Certificate Holder*	
NG-8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The NUHOMS® DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. QA data packages are prepared in accordance with the requirements of TN's approved QA program.*
NG-3000/ Section II, Part D, Table 2A	Maximum temperature limit for Type 304 plate material is 800°F.	Not compliant with ASME Section II Part D Table 2A material temperature limit for Type 304 steel for the postulated transfer accident case (117°F, loss of sunshade, loss of neutron shield) and blocked vent accident (117°F, 40 hr). The calculated maximum steady state temperatures for transfer accident case and blocked vent accident case are less than 1000°F. The only primary stresses in the basket grid are deadweight stresses. The ASME Code allows use of SA240 Type 304 stainless steel to temperatures up to 1000°F, as shown in ASME Code, Section II, Part D, Table 1A. In the temperature range of interest (near 800°F), the S_m values for SA240 Type 304 shown in ASME Code, Section II Part D, Table 2A are identical to the allowable S values for the same material shown in Section B, Part D, Table 1A. The recovery actions following these accident scenarios are as described in the UFSAR.

Table U.3.1-2
Alternatives to the ASME Code for the NUHOMS® 32PTH1 DSC Basket Assembly
(Part 2 of 2)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NG-3352	Table NG 3352-1 lists the permissible welded joints.	The fusion welds between the stainless steel insert plates and the stainless fuel compartment tube are not included in Table NG-3352-1. These welds are qualified by testing. The required minimum tested capacity of the welded connection (at each side of the tube) shall be 45 kips (at room temperature). The capacity shall be demonstrated by qualification and production testing. Testing shall be performed using, or corrected to, the lowest tensile strength of material used in the basket assembly or to minimum specified tensile strength. Testing may be performed on individual welds, or on weld patterns representative of one wall of the tube. ASME Code Section IX does not provide tests for qualification of these type of welds. Therefore, these welds are qualified using Section IX to the degree applicable together with the testing described here. The welds will be visually inspected to confirm that they are located over the insert plates, in lieu of the visual acceptance criteria of NG-5260 which are not appropriate for this type of weld.
		A joint efficiency (quality) factor of 1.0 is utilized for the fuel compartment longitudinal seam welds. Table NG-3352-1 permits a joint efficiency (quality) factor of 0.5 to be used for full penetration weld examined by ASME Section V visual examination (VT). For the 32PTH1 DSC, the compartment seam weld is thin and the weld will be made in one pass. Both surfaces of weld (inside and outside) will be fully examined by VT and therefore a factor of 2 x 0.5=1.0, will be used in the analysis. This is justified as both surfaces of the single weld pass/layer will be fully examined, and the stainless steel material that comprises the fuel compartment tubes is very ductile.
NG-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of more recent edition os SNT-TC-1A*

Table U.3.1-3

Alternatives to the ASME Code to the NUHOMS® OS200 and OS200FC Transfer Casks

(Applies to TC structural components only; lead shielding, neutron shielding, and neutron shield jacket of the TC are not addressed by this table.)

Reference ASME Code Section/Article	***Code Requirement***	***Alternatives, Exception, Justification & Compensatory Measures***
NCA	*All*	*Not compliant with NCA. Quality assurance is provided according to 10 CFR 72 Subpart G in lieu of NCA-4000.*
NCA-1140	*Use of code editions and addenda*	*Code edition and addenda other than those specified in Section U.2 may be used for construction, but in no case earlier than 3 years before that specified in Section U.2. Materials produced and certified in accordance with ASME Section II material specification from code editions and addenda other than those specified in Section U.2 may be used, so long as the materials meet all the requirements of Article 2000 of the applicable subsection of the Section III edition and addenda used for construction.*
NC-1100	*Requirements for code stamping of components*	*The OS200/OS200FC TC is designed and fabricated to the requirements of Subsection NC, to the maximum extent practical. However, the transfer cask does not have a code stamp. Code stamping is not required by 10 CFR 72 regulation. Therefore, the fabricator is not required to be ASME Certified.*
NC-2000	*ASME code materials are to be used.*	*The TC bottom ram access cover plate is made of ASTM A240, a non-ASME material. This cover plate is a water tight closure used during fuel loading/unloading operations in the fuel/reactor building only. This is not a pressure boundary component, and its failure does not result in any public safety concerns.*
NC-2130	*Material must be supplied by ASME approved material suppliers.*	*Materials designated as ASME on the UFSAR Chapter U.1 drawings are obtained by TN approved suppliers with Certified Material Test Reports (CMTRs). Material is certified to meet all ASME code criteria but is not eligible for certification or code stamping, if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NC-2130 is not possible.*
NC-4120	*Material certification by certificate holder*	*Material traceability & certification are maintained in accordance with TN's NRC approved QA program.*
NC-5254	*Category D joints shall be RT or UT examined.*	*The trunnion-to-shell weld is a Category D joint which does not allow adequate UT or RT examination. This weld is not a pressure boundary but serves as lifting point for the TC. During fabrication, this weld is progressive PT examined and then load- tested to three times the design load.* *The weld between the ram access penetration forging and bottom end plate is a Category D joint which does not allow meaningful RT or UT examination. This weld is PT examined root and final layers. This is not a pressure boundary component and its failure does not result in any public safety concerns.*
NC-6000	*All completed pressure retaining systems shall be pressure tested.*	*With respect to pressure testing requirements, the transfer cask is not a pressure retaining component. Therefore, no pressure testing is required. However, the liquid neutron shield cavity, cask bottom neutron shield cavity, and the bottom cover plate assembly are pressure and leak tested.*
NC-7000	*Overpressure protection*	*The TC is not a pressure retaining component. Therefore, no overpressure protection is provided for the transfer cask, except that a pressure relief valve is provided for the annular neutron shielding.*
NC-8000	*Requirements for nameplates, stamping & reports per NCA-8000*	*The TC nameplate provides the information required by 10CFR72. Code stamping is not required for the TC. QA data packages are prepared in accordance with the requirements of 10CFR72 and TN's NRC approved QA program.*
NC-5520	*NDE personnel must be qualified to a specific edition of SNT-TC-1A*	*Permit use of more recent edition of SNT-TC-1A.*

10. Position the lifting yoke with the TC trunnions and verify that it is properly engaged.

11. Raise the TC to the pool surface. Prior to raising the top of the cask above the water surface, stop vertical movement.

12. Inspect the top shield plug to verify that it is properly seated onto the DSC. If not, lower the cask and reposition the top shield plug. Repeat Steps 8 through 12 as necessary.

13. Continue to raise the TC from the pool and spray the exposed portion of the cask with water until the top region of the cask is accessible.

14. Drain any excess water from the top of the DSC shield plug back to the fuel pool.

15. Check the radiation levels at the center of the top shield plug and around the perimeter of the cask. Disconnect the top shield plug rigging.

16. Drain a minimum of 50 gallons of water from the DSC cavity. Optionally, approximately 900 gallons of water (as indicated by the flowmeter) may be drained from the DSC back into the pool or other suitable location to meet the weight limit on the crane. Use 1 to 3 psig of helium to backfill the DSC with helium per ISG-22 [8.5] guidance as water is being removed from the DSC cavity.

17. Lift the TC from the fuel pool. As the cask is raised from the pool, continue to spray the cask with water and decon as directed. Provisions shall be made to assure that air will not enter the DSC cavity. One way to achieve this is by replenishing the helium in the DSC cavity during cask movement from the fuel pool to the decon area in case of malfunction of equipment used for cask movement.

18. Move the TC with loaded DSC to the cask decon area.

18a. Replace the water removed from the DSC cavity in Step 16 with water from the fuel pool or an equivalent source which meets the requirements of Technical Specifications *3.2.1.*

19. If applicable to keep the occupational exposure ALARA, temporary shielding may be installed as necessary to minimize personnel exposure. Install cask seismic restraints if required by Technical Specification 4.3.3 (required only on plant specific basis).

20. Verify that the transfer cask dose rates are compliant with limits specified in Technical Specification *5.2.4.*

U.8.1.3 DSC Drying and Backfilling

CAUTION: During performance of steps listed in Section U.8.1.3, monitor the TC/DSC annulus water level and replenish if necessary until drained.

1. Check the radiation levels along the perimeter of the cask. The cask exterior surface should be decontaminated as necessary in accordance with the limits specified in Technical Specification 5.2.4.d. Temporary shielding may be installed as necessary to minimize personnel exposure.

2. Place scaffolding around the cask so that any point on the surface of the cask is easily accessible to personnel.

3. Disengage the rigging cables from the top shield plug and remove the eyebolts. Disengage the lifting yoke from the trunnions and position it clear of the cask.

4. Decontaminate the exposed surfaces of the DSC shell perimeter and remove the inflatable TC/DSC annulus seal.

5. Connect the cask drain line to the cask, open the cask cavity drain port and allow water from the annulus to drain out until the water level is approximately twelve inches below the top edge of the DSC shell. Take swipes around the outer surface of the DSC shell and check for smearable contamination in accordance with the Technical Specification 5.2.4.d limits.

   CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

6. Drain approximately 900 gallons of water (as indicated on a flowmeter) from the DSC back into the fuel pool or other suitable location, if not drained in *Section U.*8.1.2, *step* 16. Consistent with ISG-22 [8.5] guidance, helium at 1-3 psig is used to backfill the DSC with an inert gas (helium) as water is being removed from the DSC. Only helium may be used to assist in the removal of water.

7. Monitor TC/DSC annular water level and replenish as necessary until drained.

8. Install the automatic welding machine onto the inner top cover plate and place the inner top cover plate with the automatic welding machine onto the DSC. Optionally, the inner top cover plate and the automatic welding machine can be placed separately. Verify proper fit-up of the inner top cover plate with the DSC shell.

9. Check radiation levels along surface of the inner top cover plate. Temporary shielding may be installed as necessary to minimize personnel exposure.

10. Insert a 1/4-inch tubing of sufficient length and adequate temperature resistance through the vent port such that it terminates just below the DSC shield plug. Connect the flexible tubing to a hydrogen monitor to allow continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner cover plate, in compliance with Technical Specification *5.2.6*. Optionally, other methods may be used for continuous monitoring of the hydrogen atmosphere in the DSC cavity during welding of the inner top cover plate, to comply with the Technical Specification.

11. Cover the cask/DSC annulus to prevent debris and weld splatter from entering the annulus.

12. Ready the automatic welding machine and tack weld the inner top cover plate to the DSC shell. Install the inner top cover plate weldment and remove the automatic welding machine.

    CAUTION: Continuously monitor the hydrogen concentration in the DSC cavity using the arrangement or other alternate methods described in Step 10 during the inner top cover plate cutting/welding operations. Verify that the measured hydrogen concentration does not exceed a safety limit of 2.4% [8.2 and 8.3]. If this limit is exceeded, stop all welding operations and purge the DSC cavity with approximately 2-3 psig helium via the tubing to reduce the hydrogen concentration safely below the 2.4% limit.

13. Perform dye penetrant weld examination of the inner top cover plate weld in accordance with the Technical Specification 5.2.4.b requirements.

14. Remove purge lines and connect the VDS to the DSC siphon and vent ports.

15. Install temporary shielding to minimize personnel exposure throughout the subsequent welding operations as required.

16. a. If using blowdown method to remove water, engage helium supply (up to 15 psig) and open the valve on the vent port and allow helium to force the water from the DSC cavity through the siphon port.

    b. If using water pumps to remove water without blowdown, pump water from DSC.

17. Once the water stops flowing from the DSC, close the DSC siphon port and disengage the helium source or turn off the section pump, as applicable.

18. Connect the hose from the vent port and the siphon port to the intake of the vacuum pump. Connect a hose from the discharge side of the VDS to the plant's radioactive waste system or spent fuel pool. Connect the VDS to a helium source.

    Note: Proceed cautiously when evacuating the DSC to avoid freezing consequences.

19. Open the valve on the suction side of the pump, start the VDS and draw a vacuum on the DSC cavity. The cavity pressure should be reduced in steps of approximately 100 mm Hg, 50 mm Hg, 25 mm Hg, 15 mm Hg, 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level (these levels are optional), the pump is valved off and the cavity pressure monitored. The cavity pressure will rise as water and other volatiles in the cavity evaporate. When the cavity pressure stabilizes, the pump is valved in to complete the vacuum drying process. It may be necessary to repeat some steps, depending on the rate and extent of the pressure increase. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less as specified in Technical Specification 3.1.1.

    Note: The user shall ensure that the vacuum pump is isolated from the DSC cavity when demonstrating compliance with *Technical Specification 3.1.1* requirements. Simply closing the valve between the DSC and the vacuum pump is not sufficient, as a faulty valve allows the vacuum pump to continue to draw a vacuum on the DSC. Turning off the pump, or opening the suction side of the pump to atmosphere are examples of ways to assure that the pump is not continuing to draw a vacuum on the DSC.

    CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

20. Open the valve to the vent port and allow the helium to flow into the DSC cavity.

21. Pressurize the DSC with helium up to 15 psig.

22. Helium leak test the inner top cover plate weld for a leak rate of 1×10^{-4} atm·cm^3 /sec. This test is optional.

23. If a leak is found, repair the weld, repressurize the DSC and repeat the helium leak test.

24. Once no leaks are detected, depressurize the DSC cavity by releasing the helium through the VDS to the plant's spent fuel pool or radioactive waste system.

25. Re-evacuate the DSC cavity using the VDS. The cavity pressure should be reduced in steps of approximately 10 mm Hg, 5 mm Hg, and 3 mm Hg. After pumping down to each level, the pump is valved off and the cavity pressure is monitored level (these levels are optional). When the cavity pressure stabilizes, the pump is valved in to continue the vacuum drying process. Vacuum drying is complete when the pressure stabilizes for a minimum of 30 minutes at 3 mm Hg *absolute* or less in accordance with Technical Specification 3.1.1 limits.

    Note: The user shall ensure that the vacuum pump is isolated from the DSC cavity when demonstrating compliance with *Technical Specification 3.1.1* requirements. Simply closing the valve between the DSC and the vacuum pump is not sufficient, as a faulty valve allows the vacuum pump to continue to draw a vacuum on the DSC. Turning off the pump, or opening the suction side of the pump to atmosphere are examples of ways to assure that the pump is not continuing to draw a vacuum on the DSC.

26. Open the valve on the vent port and allow helium to flow into the DSC cavity to pressurize the DSC between *18.5 and 20.0* psig and hold for 10 min. Depressurize the DSC cavity by releasing the helium through the VDS to the plant spent fuel pool or radioactive waste system to about 2.5 psig in accordance with Technical Specification 3.1.2.b limits.

    CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

27. Close the valves on the helium source.

U.8.1.4 DSC Sealing Operations

CAUTION: During performance of steps listed in Section U.8.1.4, monitor the Cask/DSC annulus water level and replenish as necessary to maintain cooling.

1. Disconnect the VDS from the DSC. Seal weld the prefabricated plugs over the vent and siphon ports. Inject helium into blind space just prior to completing welding, and perform a dye penetrant weld examination in accordance with the Technical Specification 5.2.4.b requirements.

2. Temporary shielding may be installed as necessary to minimize personnel exposure. Install the automatic welding machine onto the outer top cover plate and place the outer top cover plate with the automatic welding system onto the DSC. Optionally, outer top cover plate may be installed separately from the welding machine. Verify proper fit up of the outer top cover plate with the DSC shell.

3. Tack weld the outer top cover plate to the DSC shell. Place the outer top cover plate weld root pass.

4. Helium leak test the inner top cover plate and vent/siphon port plate welds using the leak test port in the outer top cover plate in accordance with Technical Specification 5.2.4.c limits. Verify that the personnel performing the leak test are qualified in accordance with SNT-TC-1A [8.4]. Alternatively this can be done with a test head in step 1 of Section U.8.1.4.

23. Close and lock the ISFSI access gate and activate the ISFSI security measures.

24. Ensure the HSM-H maximum air exit temperature requirements of Technical Specification *3.1.4* are met.

U.8.1.7 Monitoring Operations

1. Perform routine security surveillance in accordance with the licensee's ISFSI security plan.

2. Perform **one** of the two alternate daily surveillance activities listed below:

   a. A daily visual surveillance of the HSM air inlets and outlets to insure that no debris is obstructing the HSM vents in accordance with Technical Specification 5.2.5.a requirements.

   b. A temperature measurement of the thermal performance, for each HSM, on a daily basis in accordance with Technical Specification 5.2.5.b requirements.

U.9.1.7 Poison Acceptance

CAUTION
Sections U.9.1.7.1 through U.9.1.7.3 below are incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Borated aluminum

(b) Boron carbide/aluminum metal matrix composite (MMC)

(c) Boral®

The 32PTH1 DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content for these materials is given in Table U.9-1.

References to metal matrix composites throughout this chapter are not intended to refer to Boral®, which is described later in this section.

U.9.1.7.1 **Borated Aluminum**

See the Caution in Section U.9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating *primarily* as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy *(other boron compounds, such as AlB_{12}, can also occur).* For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section U.9.1.7.5. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings"[9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surface, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

U.9.1.7.2 Boron Carbide/Aluminum Metal Matrix Composites (MMC)

See the Caution in Section U.9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. The boron carbide content shall not exceed 40% by volume. The boron carbide content for MMCs with an integral aluminum cladding shall not exceed 50% by volume.

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity. For MMC with an integral cladding, the final density of the core shall be greater than 97% of theoretical density demonstrated by qualification testing, with no more than 0.5 volume % interconnected porosity of the core and cladding as a unit of the final product.

Boron carbide particles for the products considered here shall have an average size of 40 microns or less. No more than 10% of the particles shall be over 60 microns.

Prior to use in the 32PTH1 DSC, MMCs shall pass the qualification testing specified in Section U.9.1.7.6, and shall subsequently be subject to the process controls specified in Section U.9.1.7.7.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section U.9.1.7.5. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings" [9.5]. Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surfaces, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

References to metal matrix composites throughout this chapter are not intended to refer to Boral®, which is described in the following section.

U.9.1.7.3 **Boral®**

See the Caution in Section U.9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. *Before rolling, at least 80% by weight of the B_4C particles in BORAL® shall be smaller than 200 microns.* The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of Boral®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on an approximately 1 cm² area of a coupon taken near one of the corners of the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

Visual inspections shall verify that the Boral® core is not exposed through the face of the sheet at any location.

U.9.1.7.4 Thermal Conductivity Testing

All poison plate materials except Boral® will be qualification tested to verify that the thermal conductivity equals or exceeds the values listed in Section U.4.3. *Acceptance testing shall be performed at room temperature on coupons taken from the rolled or extruded production material. Initial sampling shall be one test per lot, defined by the heat or ingot, and may be reduced if the first five tests meet the specified minimum thermal conductivity.*

If a thermal conductivity test result is below the specified minimum, at least four additional tests shall be performed on the material from that lot. If the mean value of those tests, including the original test, falls below the specified minimum, the associated lot shall be rejected.

After twenty five tests of a single type of material, with the same aluminum alloy matrix, the same boron content, and the same primary boron phase, e.g., B_4C, TiB_2, or AlB_2, if the mean value of all the test results less two standard deviations meets the specified thermal conductivity, no further testing of that material is required. This exemption may also be applied to the same type of material if the matrix of the material changes to a more thermally conductive alloy (e.g., from 6000 to 1000 series aluminum), or if the boron content is reduced without changing the boron phase.

Testing shall conform to ASTM E1225 [9.9], ASTM E1461 [9.10], or equivalent method.

U.9.1.7.5 **Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission**

> ***CAUTION***
>
> *Section U.9.1.7.5 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.*

Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat or from a group of billets from the same heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1.2 centimeter diameter. A beam size greater than 1.2 centimeter diameter but no larger than 1.7 centimeter diameter may be used if computations are performed to demonstrate that the calculated $k_{effective}$ of the system is still below the calculated Upper Subcritical Limit (USL) of the system assuming defect areas the same area as the beam.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard.

Alternatively, digital image analysis may be used to compare neutron radioscopic images of the test coupon to images of the standards. The area of image analysis shall be up to 1.1 cm^2. The method shall demonstrate sufficient sensitivity to distinguish between areal density at the specified minimum, 1% above and below the minimum.

(Alternatively show that the material fails in a ductile manner, e.g., by scanning electron microscopy of the fracture surface or by bend testing.)

and

b) testing (ASTM-B311[4]) to verify more than 98% (or 97% for MMCs with integral aluminum cladding) of theoretical density. Testing or examination for exposed interconnected porosity shall be performed by a means to be approved by the Certificate Holder.

U.9.1.7.6.5 **Required Tests and Examinations to Demonstrate B10 Uniformity**

CAUTION
Section U.9.1.7.6.5 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Uniformity of the boron distribution shall be verified either by:

a) Neutron radioscopy or radiography (ASTM E94[5], E142[6], and E545[7]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

b) Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section U.9.1.7.5, or by chemical analysis for boron carbide content in the composite.

U.9.1.7.6.6 Approval of Procedures

Qualification procedures shall be subject to approval by the Certificate Holder.

U.9.1.7.7 Specification for Process Controls for Metal Matrix Composites

U.9.1.7.7.1 Applicability and Scope

The applicability of this section is the same as that of Section U.9.1.7.6. It addresses the process controls to ensure that the material delivered for use is equivalent to the qualification test material.

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[4] **ASTM B311, Test Method for Density Determination for Powder Metallurgy (P/M) Materials Containing Less than Two Percent Porosity**

[5] **ASTM E94, Recommended Practice for Radiographic Testing**

[6] **ASTM E142, Controlling Quality of Radiographic Testing**

[7] **ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing**

Key processing changes shall be subject to qualification prior to use of the material produced by the revised process. The Certificate Holder shall determine whether a complete or partial re-qualification program per Section U.9.1.7.7 is required, depending on the characteristics of the material that could be affected by the process change.

U.9.1.7.7.2 Definition of Key Process Changes

Key process changes are those which could adversely affect the uniform distribution of the boron carbide in the aluminum, reduce density, or reduce the mechanical strength or ductility of the MMC.

U.9.1.7.7.3 Identification and Control of Key Process Changes

> ***CAUTION***
>
> *Section U.9.1.7.7.3 is incorporated by reference into the NUHOMS® CoC 1004 Technical Specification 4.1 (Note 5) and shall not be deleted or altered in any way without approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.*

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section U.9.1.7.7.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that may be established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

a) Changes in the boron carbide particle size specification that increase the average particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

c) Change in the nominal matrix alloy,

d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

e) For MMCs using a 6000 series aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature, and

W.1 General Description

Appendix W to the NUHOMS® *u*pdated *f*inal *s*afety *a*nalysis *r*eport (UFSAR) addresses the *i*mportant to *s*afety aspects of adding the OS197L TC to the *s*tandardized NUHOMS® system described in the UFSAR. The OS197L TC is added to the UFSAR as an alternative to the OS197 and OS197H TCs. The primary reason for adding the OS197L TC design is to include a *reduced weight* transfer cask that can be used *to transfer specific payloads at* facilities.

The addition of the OS197L TC to the standardized NUHOMS® system results in a change to the technical specifications of the NUHOMS® CoC 1004 [1.3]. The OS197L TC may only be used for the loading and transfer of 61BT or 32PT DSC with a heat load of 13 kW or less. Hence, a general licensee shall NOT implement the contents of Appendix W until the NRC has approved TN's application for Amendment 11 to NUHOMS® CoC 1004.

The format of this *a*ppendix follows the guidance provided in NRC Regulatory Guide 3.61 [1.1]. The analyses presented in this *a*ppendix demonstrate that the OS197L TC system meets all the requirements of 10 CFR 72 [1.2].

Several sections of this *a*ppendix have been identified as "No Change." For these sections, the description or analysis presented in the *identified* sections of the UFSAR for the *s*tandardized NUHOMS® system is also applicable to the OS197L TC. In addition, tables and figures presented in the UFSAR which remain unchanged due to the addition of the OS197L TC to the *s*tandardized NUHOMS® system are not repeated in this *a*ppendix. Table W.1-2 provides a summary of the sections of the main body of the UFSAR applicable to the OS197 TC and addresses the impact of the OS197L TC on these sections.

Note: References to sections or chapters within this *a*ppendix are identified with a prefix W (e.g., Section W.2.3 or Chapter W.2). References to sections or chapters of the UFSAR outside of this *a*ppendix (i.e., main body of the UFSAR) are identified with the applicable UFSAR section, chapter number or *a*ppendix number (e.g., Section 2.3, Chapter 2 or Appendix K). The references used in this *a*ppendix are identified as [X.X] (e.g., [1.1] is Reference 1.1 at the end of Chapter W.1).

OS197 and OS197H TCs in the remainder of this *a*ppendix will be referred to as OS197 TC.

W.1.1 Introduction

As stated in Section 1.3.2.1, the body of this UFSAR is dedicated to *four* on-site transfer cask types: the *standardized* cask, NUHOMS®-OS197, NUHOMS®-OS197H, *and OS200* TCs. The purpose of this *a*ppendix is to provide the safety analysis of the design of a *fifth* type of on-site transfer cask, designated as the NUHOMS® OS197L TC, for use with the standardized NUHOMS® system.

W.1.2 General Description of the NUHOMS® OS197L TC

The OS197L TC on-site transfer cask is designed to accommodate *fuel transfer needs of* plants *where the payload is limited to a maximum of 13.0 kW.* The major differences between the OS197L TC and the OS197 casks are:

- *r*educed cask weight;
  - *no integral* lead shielding (one 2.68" nominal thickness steel shell instead of a combination of a 0.5" nominal thickness steel inner liner, 3.5" nominal thickness lead shield and 1.5" nominal thickness steel structural shell);
- *only authorized for transfer of the NUHOMS®-61BT and 32PT DSCs with a maximum heat load of 13.0 kW;*
- *o*ne-piece solid trunnion configuration for the upper and lower cask trunnions;
- *t*wo-piece neutron shield (inner and outer shell of 1/4" nominal thickness versus an outer shell of 3/16" nominal thickness);
- *a 6" nominal thickness steel decontamination area supplemental shield (see Figure W.1-2) within which the cask is placed for personnel shielding during fuel loading operations;*
- *a cask support skid supplemental shielding (see Figure W.1-3), described in Section W.1.2.1.1, to be used for personnel shielding during transfer operations;*
- *remote crane operations in conjunction with laser/optical targeting and cameras are to be used for handling the OS197L TC when it is not within the decontamination area shielding.*

The OS197L TC key design parameters are compared to the OS197 TC in Table W.1-1.

The OS197L TC, *when used in conjunction with the supplemental shielding provided (see Figures W.1-2 and W.1-3), including the remote cask handling procedures described in Chapter W.8,* provides shielding and protection from potential hazards during the DSC fuel loading/unloading operations and transfer to the *h*orizontal *s*torage *m*odule (HSM). The design and configuration of the OS197L TC is a modified version of the NRC approved OS197 and OS197H TCs described in Section 1.3.2.1 of the UFSAR and is limited to on-site use under 10CFR72. The OS197L TC can be configured to meet a gross weight limit of *77* Te (*85* tons).

The empty weight of the OS197L, with the neutron shield full of water and the stainless steel top cask lid installed is approximately 62,000 lb (31 tons). The nominal loaded weight in the "wet" configuration (water in the DSC, water in the DSC/TC annulus, top cask lid not installed) is approximately 85 tons. The nominal loaded weight in the "dry" configuration (after water in the DSC and DSC/TC annulus has been drained and the top cask lid is installed) is approximately 82 tons. SAR Table W.3-1 has been revised to provide this additional weight information.

Figure W.1-1 provides an overview of the OS197L TC *without the supplemental shielding*. The OS197L TC configuration also requires the use of additional shielding in the decontamination area (see Figure W.1-2) and on the skid/trailer (see Figure W.1-3).

W.1.2.1.1 Transfer Equipment

Transfer **Trailer**: The NUHOMS® OS197L TC *transfer* trailer consists of a heavy industrial trailer with a payload capacity of 136 Te (150 tons), including the skid and loaded cask. The OS197L TC *transfer* trailer is the same as the one shown in Figure 1.3-7 of the UFSAR.

Cask Support Skid: The OS197L TC support skid differs from the OS197 TC support skid shown in UFSAR Figure 1.3-8 as described below:

1. The OS197L TC support skid has permanently mounted 2.5" thick side shielding and accommodates an additional 3" thick side shielding bolted to the permanent shielding when transferring the OS197L TC.
2. The OS197L TC *support skid* also has a 2.5" shielding inner top cover and an additional 3" shielding outer top cover to shield the upper sections of the cask.

The OS197L TC support skid utilized for the standardized NUHOMS® system is illustrated in Figure W.1-3.

Hydraulic Ram: The high capacity hydraulic ram system is similar to the hydraulic ram system described in the UFSAR. The capacity of this ram is increased in order to increase the ram capacity margin (and to accommodate other future DSC designs). There is no change to the maximum ram forces allowed (80 kips) during system operation.

A picture of the OS197L TC system is provided in Figure W.1-4.

W.1.2.2 Operational Features

The primary operations with the OS197L TC (in sequence of occurrence) for the NUHOMS® system are the same as the systems operation described in Section 1.3.3 of the UFSAR except as noted below for operations 8 and 13 (of Section 1.3.3):

Lifting Cask from Pool: The loaded OS197L TC is lifted out of the pool *for placement* (in the vertical position) in a decontamination area shield on the drying pad in the decon pit. During *bare* cask movement from the fuel pool to the decontamination area, remote crane operation*s in conjunction with laser/*optical targeting *and cameras or other similar equipment for confirmation of the cask locations are to* be used to minimize personnel exposure due to the reduced shielding configuration of the OS197L TC during this transit movement. The licensee shall meet the specific radiation protection program requirements associated with the use of OS197L TC as specified in *Technical Specification 5.2.4.a*.

The cask is then placed inside the decontamination area lower shield and the upper shield or bell is then placed on top (see Figure W.1-2).

Placement of Cask on *Transfer* Trailer Skid: The OS197L TC is then lifted onto the cask support skid. The plant's crane is used to downend the cask from a vertical to a horizontal position. *The inner* top *skid* shielding is added to the skid and the cask is also covered with an additional outer top shielding. The outer top additional *skid* shielding is to be installed inside the fuel handling building if the floor loads can accommodate it (if floor loading is a concern, the *outer top trailer* shielding may be placed on the skid outside the fuel handling building). The cask is then secured to the skid and readied for the subsequent *transfer* operations.

The "cask support skid supplemental shielding" described in the SAR drawing (NUH-03-8011-SAR) refers to the temporary shielding required for the OS197L TC during transfer operations. This shielding is also referred to as "shielding, supplemental shielding, trailer area shielding, cask skid supplemental shielding, supplemental OS197L TC shielding, supplemental skid shielding, shield transfer trailer, auxiliary shielding, shielded transfer skid, transfer skid shields, supplementary shielding, supplemental trailer shields, skid shielding, and trailer shielding" in the safety analysis descriptions interchangeably through out this appendix.

W.1.3 Identification of Agents and Contractors

Transnuclear, Inc. (TN) provides the design, analysis, licensing support and quality assurance for the NUHOMS® OS197L TC. Fabrication of the NUHOMS® OS197L TC is done by one or more qualified fabricators under TN's quality assurance program described in Chapter W.13. This program is written to satisfy the requirements of Subpart G of 10CFR72 [1.2] and covers control of design, procurement, fabrication, inspection, testing, operations and corrective action.

TN provides specialized services for the nuclear fuel cycle that support transportation, storage and handling of spent nuclear fuel, radioactive waste and other radioactive materials. TN is the holder of NUHOMS® CoC 1004 [1.3].

W.1.4 Generic Cask Arrays

No change. *The content presented in Sections 1.2.1 and 1.3.4 remains applicable and is not affected by the use of the OS197L transfer cask.*

Table W.1-1
Comparison of Key Parameters of NUHOMS® OS197 Versus OS197L TCs

Characteristic	OS197 TC	OS197L TC	Same? (Yes/No) Note No.
Physical Data			
Outside Diameter	85.50"	80.36"	No (1)
Outside Length	207.22"	207.22"	Yes
Cavity Diameter	68"	68"	Yes
Cavity Length	197.75"	197.75"	Yes
Ram Access Penetration Diameter	22"	22"	Yes
Weight, Empty	106,670 lbs (includes cask top cover plate assembly and neutron shield without water)	57,400 lbs (includes cask top cover plate assembly and neutron shield without water)	No (2)
Cask Materials			
Outer Jacket	3/16" thick plate, ASTM A240, Type 304	1/4" thick plate, ASTM A240, Type 304	No (3)
Neutron Shielding	3" of Water in annulus	3" of Water in annulus	Yes
Structural Shell	1-1/2" thick plate, ASME SA-240 Type 304	2.68" thick plate, ASME SA-240 Type 304	No (4)
Gamma Shielding	3.56" thick, ASTM B29 Chemical Copper Lead	No lead shielding	No (4)
Inner Liner	1/2" thick plate, ASME SA-240 Type 304	No separate inner liner (consists of structural shell)	No (4)
Top Cover Assembly	Consists of 3" thick ASME SA-240, Type 304 structural plate with a thin 1/4" thick shell encapsulating a solid Neutron Absorbing Material (NS-3)	Consists of 3" thick ASME SA-240, Type 304 structural plate with a thin 1/4" thick shell encapsulating a solid Neutron Absorbing Material (NS-3)	Yes
Top Flange	ASME SA-182, Type F304N	ASME SA-182, Type F304N	Yes
Upper Lifting Trunnion	ASME SA-564, Grade 630 steel trunnion with sleeve encapsulating a solid Neutron Absorbing Material (NS-3)	Solid monolithic Trunnion made of ASME SA-182, Type FXM-19	No
Lower Support Trunnion	ASME SA-240, Type F304 steel trunnion with sleeve encapsulating a solid Neutron Absorbing Material (NS-3)	Solid monolithic Trunnion made of ASME SA-182, Type F304	No
Canister Rails	ASTM A240 Nitronic 60	ASTM A240 Nitronic 60	Yes
Bottom End Plate	2" thick, ASME SA-240, Type 304	2" thick, ASME SA-240, Type 304	Yes
Bottom Support Ring	ASME SA-182, Type F304N	ASME SA-182, Type F304N	Yes
Ram Access Penetration Ring	ASME SA-182, Type F304N	ASME SA-182, Type F304N	Yes
Cask Payload			
DSC Type	24P, 52B, 61BT, *61BTH Type 1*, 24PHB, 24PT2, *24PTH*, 32PT	61BT *and* 32PT *Only with modified payloads*	*No*
Heat Load	24 kW	*13* kW	*No*

Notes:

1. The diameter of the OS197L TC is smaller, reflecting the reduced radial shielding. The 2.68" thick SS structural shell replaces the combined thickness of ½" of inner liner, 3.50" of lead, and 1.50" of structural shell, a reduction of approximately 5.5" diametrical.
2. The reduced weight of the OS197L TC reflects the reduced radial shielding.
3. The outer panel of the neutron shield is increased in thickness to stiffen the assembly.
4. The reduced shielding is a result of the lead shielding that is eliminated and the combined inner liner and structural shell.

Table W.1-2
OS197L TC UFSAR Sections Affected

(Part 1 of 3)

Seq	Section/Page	Description	OS197L
1	1.1(3)/1.1-3	Description of TC for *transfer* of DSC	No Change
2	Figures 1.1-2/1.1-6	NUHOMS® System Components including TC	See Section W.1
3	Figures 1.1-3/1.1-7	NUHOMS® System Components including TC	See Section W.1
4	1.2.3/1.2-3	Description of Operating and Handling Systems including TC	Changes addressed in Section W.1.
5	Table 1.2-2/1.2-8	Key Design Parameters for NUHOMS® System	See Section W.1
6	Table 1.2-3/1.2-9	NUHOMS® System Operations Overview	See Section W.8
7	Section 1.3.2.1/1.3-3	Description of On-Site TC	See Section W.1
8	Section 1.3.2.2/1.3-4	Description of Transfer Equipment (Trailer and Skid)	See Section W.1
9	Table 1.3-1/1.3-10	Components, Structures and Equipment for the Standardized NUHOMS® System	See Section W.1
10	Figure 1.3-6/1.3-18	NUHOMS® On-Site TC	See Section W.1
11	Figure 1.3-8/1.3-20	Cask Support Skid for NUHOMS® System	See Section W.1
12	Figure 1.3-10/1.3-22	NUHOMS® System Operational Overview	See Section W.1
13	2.0	Site Characteristics	No Change
14	3.1.2.1/3.1-4	On-Site Transfer Cask	No change to loading conditions See Section W.1 for OS197L description.
15	Table 3.1-7/3.1-13	NUHOMS® Transfer Equipment Criteria	No Change
16	3.2.5.3/3.2-7	On-Site Transfer Cask Load Combinations and Structural Design Criteria	No Change to load combinations or criteria. See Section W.3 for OS197L structural results
17	Table 3.2-1/3.2-11	Summary of NUHOMS® Component Design Loadings	No Change
18	Table 3.2-7/3.2-20	On-Site Transfer Cask Load Combinations and Service Levels	No Change
19	Table 3.2-11/3.2-25	Structural Design Criteria for On-Site Transfer Cask	No Change
20	Table 3.2-12/3.2-26	Structural Design Criteria for Bolts	No Change
21	3.3.5.2/3.3-31	Radiological Protection-Shielding	See Section W.5
22	Table 3.3-1/3.3-36	NUHOMS® System Components Important To Safety	See Table W.2-1
23	3.4.4.1/3.4-2	Classification of Structures, Components, and Systems- Transfer Cask and Yoke	No Change
24	3.4.4.2/3.4-2	Classification of Structures, Components, and Systems- Other Transfer Equipment	No Change
25	Table 3.4-1/3.4-4	NUHOMS® Major Components and Safety Classification	See Table W.2-1
26	4.2.1/4.2-1	Storage Structures – Structural Specifications	No Change
27	4.2.3.3/4.2-9 *and 4.2-10*	Individual Unit Description - On-Site Transfer Cask	See Sections W.1 and W.3 for trunnion load test.
28	Figure 4.2-10/4.2-21	Composite View of NUHOMS® Transfer Cask-24P	*Not Applicable*
29	Figure 4.2-11/4.2-22	Composite View of NUHOMS® Transfer Cask-52B	*Not Applicable*
30	Figure 4.2-12/4.2-23	NUHOMS® On-Site Transfer Cask with BWR Collar	*Not Applicable*
31	Figure 4.2-15a/4.2-26a	NUHOMS® *OS197L* Transfer Cask Lifting Yoke	*Figure title revised to indicate that this is an alternate lifting yoke*

Table W.1-2
OS197L TC UFSAR Sections Affected
(Part 2 of 3)

Seq	Section/Page	Description	OS197L
32	4.5/4.5-1	Transfer Cask and Lifting Hardware Repair and Maintenance	No Change
33	4.7.3.2/4.7-5	Individual Unit Descriptions - Transfer Cask	See Section W.1
34	4.7.3.8/4.7-10	Individual Unit Descriptions – Cask Support Skid	See Section W.1
35	4.9/4.9-1	ASME Code Exceptions List for the Transfer Cask	See Section W.3
36	Table 4.9-1/4.9-3	ASME Code Exceptions List for the Transfer Cask	See Section W.3
37	5.0/5.1-1	Operation Systems	See Section W.8
38	6.0	Waste Confinement and Management	No Change
39	7.1/7.1-1	Radiation Protection-design Considerations	See Section W.5
40	7.3.2.2.F/7.3-6	Transfer Cask Surface Dose Rates	See Section W.5
41	Tables 7.3-2 through 7.3-5/7.3-9 through 7.3-14	Shielding Analysis Results	See Section W.5
42	7.4.1/7.4-1	Operational Dose Assessment	See Section W.5
43	Table 7.4-1/7.4-3	NUHOMS® System Operations – Occupational Dose Calculations	See Section W.5
44	8.0	Analysis of Design Events	See: Section W.3 – Structural Section W.4 – Thermal Section W.11 – Accident
45	9.0	Conduct of Operations	No Change
46	*Table 10-2* *Section B 10.5.3.4*	Operating Controls and Limits	*See Section W.12*
47	11.0	Quality Assurance	No Change
48	Appendix A	Details of Shielding Models of the NUHOMS® System	See Section W.5
49	Appendix B	Details of Heat Transfer Analysis of the NUHOMS® System	No Change
50	Appendix C.1	Deleted	No Change
51	Appendix C.2	Transfer Cask Drop Analysis	See Section W.3
52	Appendix C.3	Transfer Cask Side Drop Analysis	See Section W.3
53	Appendix C.4.1	DSC Fatigue Evaluation	No Change
54	Appendix C.4.2	Transfer Cask Fatigue Evaluation	No Change
55	Appendix C.5	Transfer Cask Structural Analysis NRC Question Resolutions	See Section W.3 for DBT events
56	Appendix C.6	References	No Change
57	Appendix D	Review of Concrete Behavior under Sustained elevated Temperature	No Change
58	Appendix E	Drawings	See Section W.1.5
59	Appendix F	NUHOMS® 24P Topical Report – NRC Questions	No Change
60	Appendix G	Deleted	No Change
61	Appendix H	NUHOMS® 24P – Long Cavity DSC Evaluation for Storing PWR fuel without BPRAs	No Change
62	Appendix I	Deleted	No Change
63	Appendix J	NUHOMS® 24P – Long Cavity DSC Evaluation for Storing PWR fuel with BPRAs	No Change

Table W.1-2
OS197L TC UFSAR Sections Affected
(Part 3 of 3)

Seq	Section/Page	Description	OS197L
64	Appendix K	NUHOMS®-61BT evaluation for storage in HSM and transfer in OS197 TC	No Change
65	Appendix L	NUHOMS®-24PT2 evaluation for storage in HSM and transfer in standardized TC and OS197 TC	No Change
66	Appendix M	NUHOMS®-32PT evaluation for storage in HSM and transfer in OS197/OS197H TC	No Change
67	Appendix N	NUHOMS®-24PHB evaluation for storage in HSM and transfer in OS197/OS197H TC	No Change
68	Appendix P	NUHOMS®-24PTH evaluation for storage in HSM and transfer in OS197/OS197H TC/OS197FC	No Change
69	Appendix R	Evaluation of NUHOMS® HSM Model 152	No Change
70	Appendix T	NUHOMS®-61BTH evaluation for storage in HSM and transfer in OS197/OS197H TC/OS197FC	No Change
71	Appendix U	NUHOMS®-32PTH1 evaluation for storage in HSM and transfer in OS200 or OS200FC	No Change
72	Appendix V	Evaluation of NUHOMS® HSM Model 202	No Change

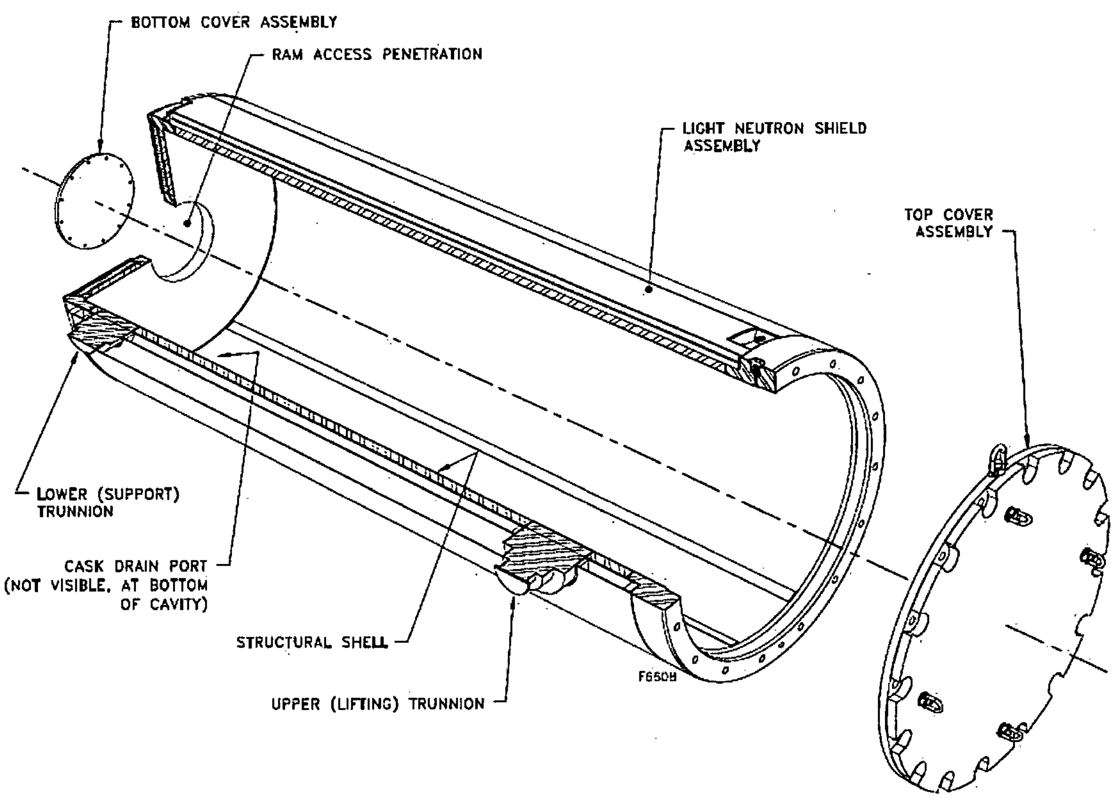


Figure W.1-1
OS197L TC Configuration

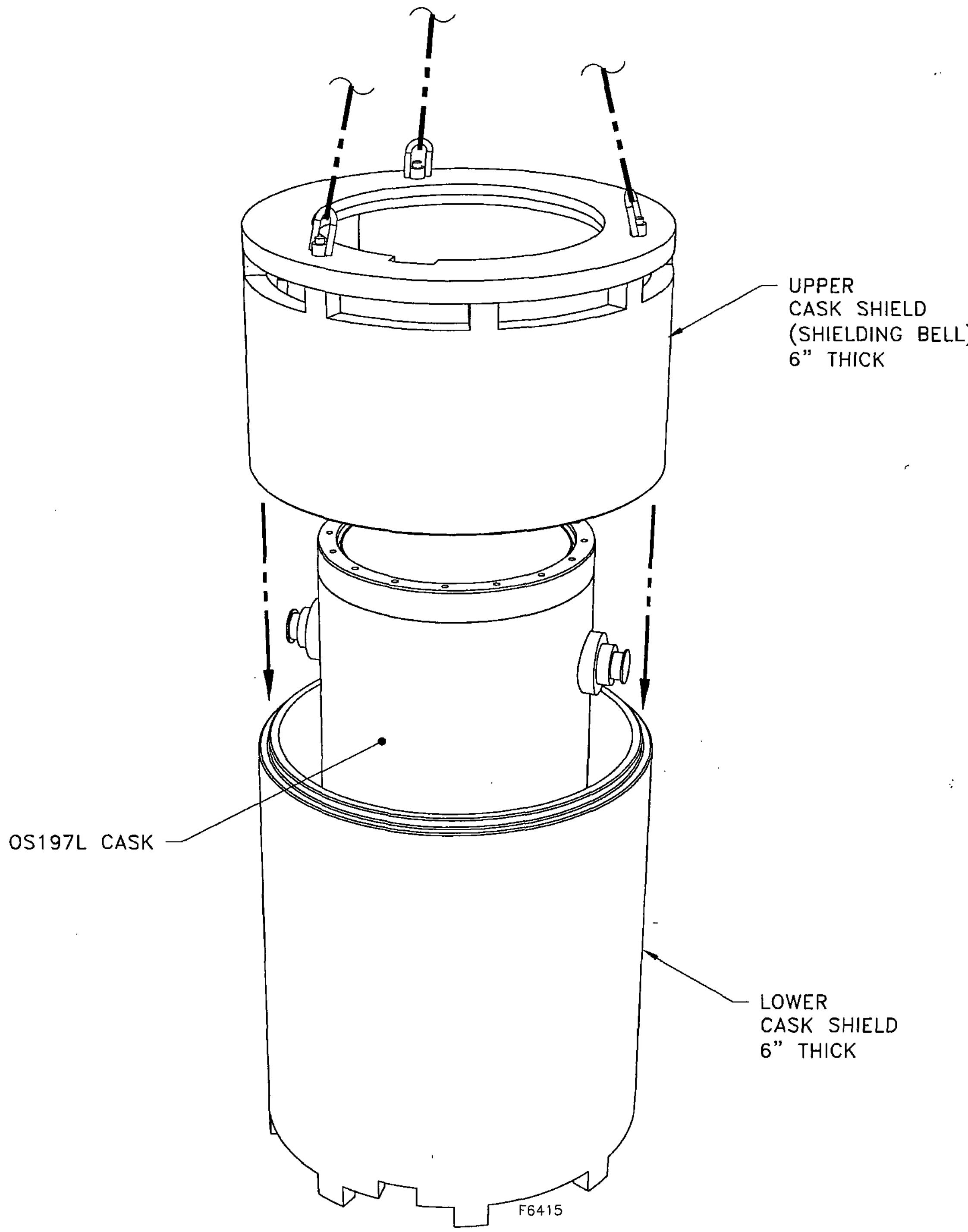


Figure W.1-2
NUHOMS® OS197L TC System Decontamination Area Shielding

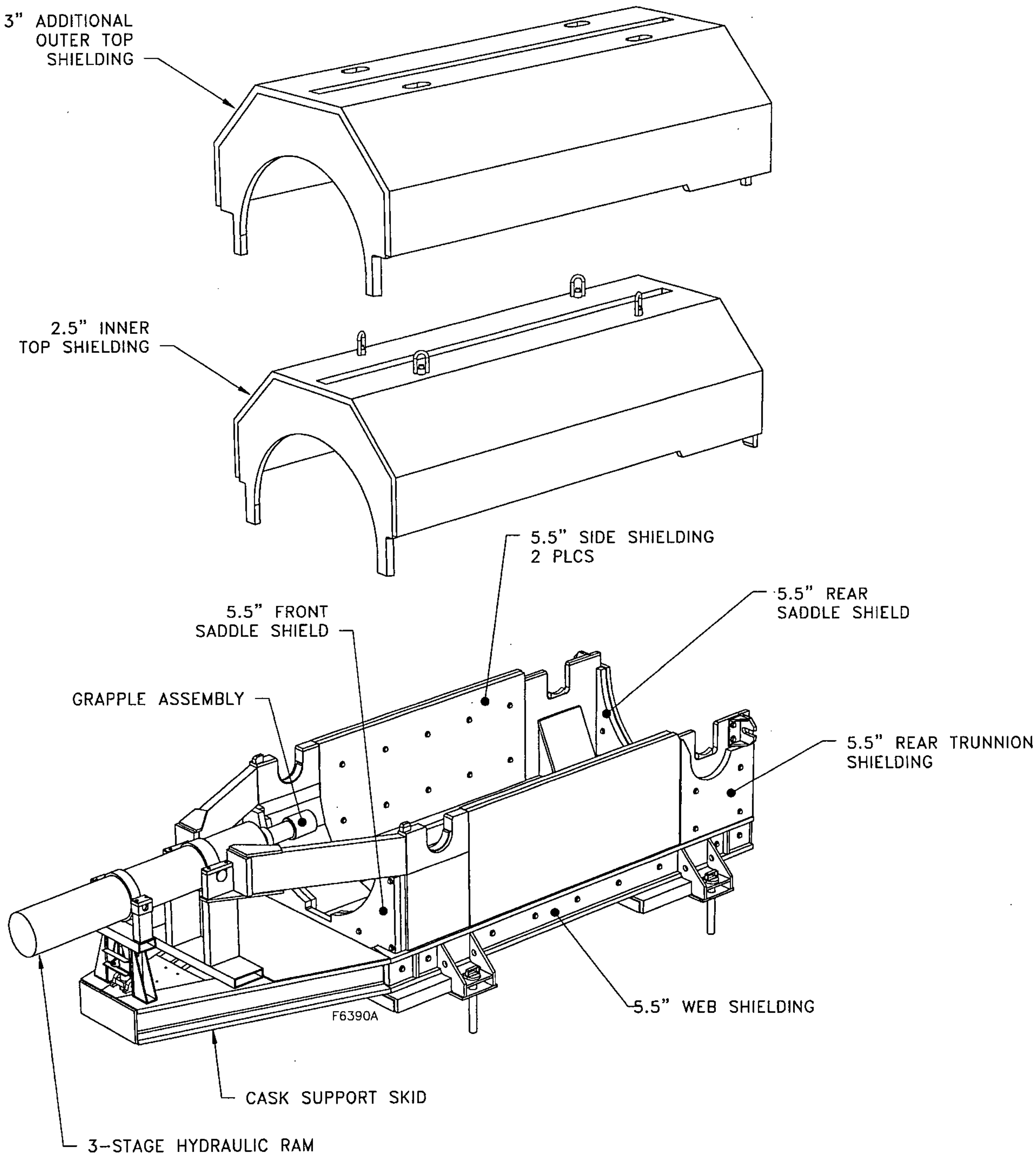


Figure W.1-3
OS197L Transfer Equipment Schematic

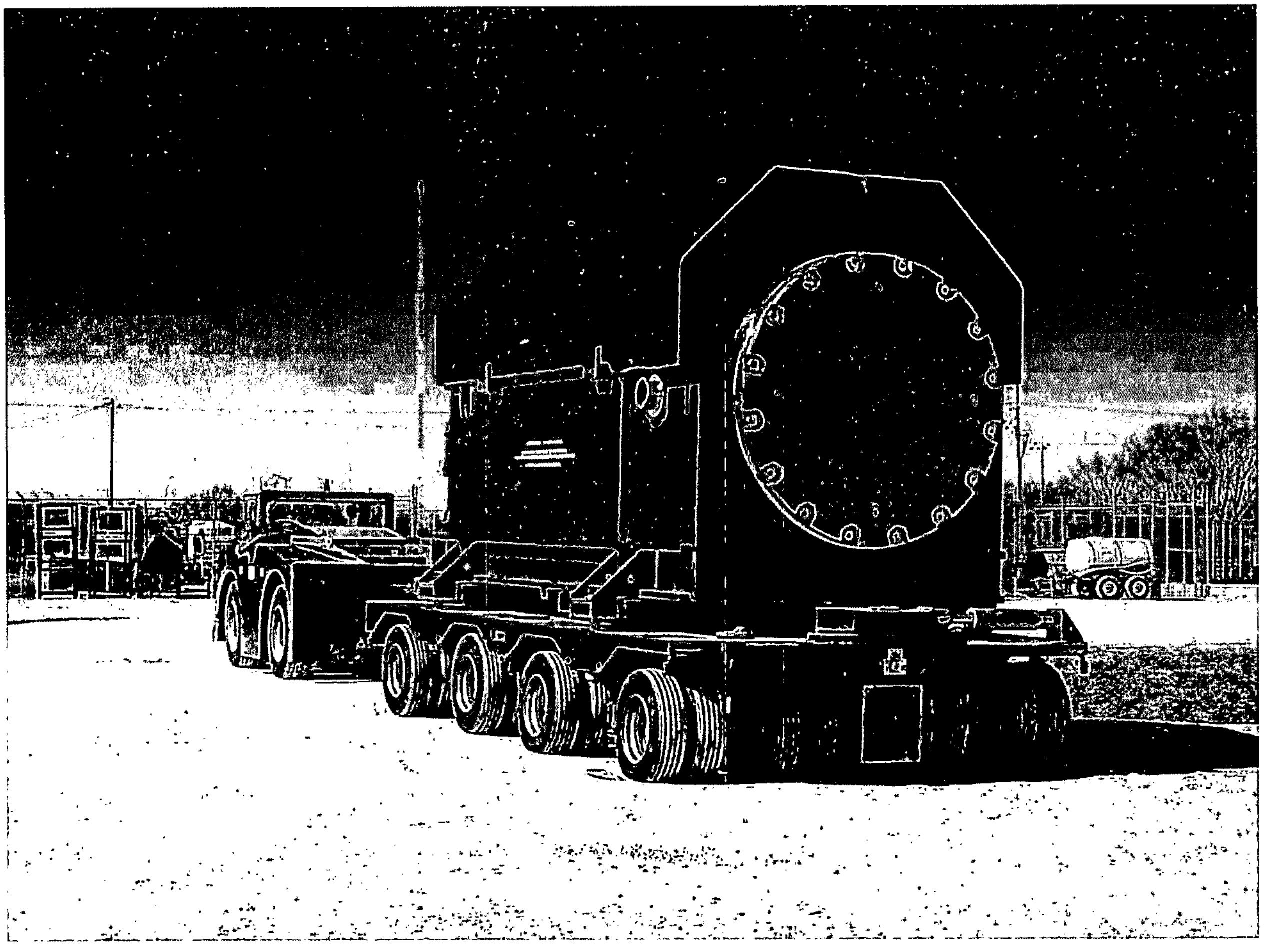


Figure W.1-4
OS197L TC System on Transfer Trailer with Shielding

W.2 Principal Design Criteria

This section provides the principal design criteria for the NUHOMS® OS197L TC System. The principal design criteria for the NUHOMS® OS197L TC are the same as the NUHOMS® OS197 TC as described in Chapter 3. *However, OS197L TC shall only be used for the handling and transfer of NUHOMS®-61BT or 32PT DSC with a heat load of 13 kW or less.* Section W.2.1 presents a general description of the spent fuel to be stored *in 61BT and 32PT DSC when using OS197L TC.* Section W.2.2 provides the design criteria for environmental conditions and natural phenomena. Section *W.2.3* provides a description of the systems which have been designated as important to safety. Section W.2.4 discusses decommissioning considerations. Section W.2.5 summarizes the NUHOMS® OS197L TC design criteria.

W.2.1 Spent Fuel To Be Stored

When using the OS197L TC for loading and transfer activities, the number of PWR and BWR fuel assemblies authorized for storage in the NUHOMS® *32PT and 61BT* DSCs *remain unchanged at up to* a total of 32 and 61, *respectively* with the same *physical* characteristics as those described in *Appendix M.2 and K.2 respectively. However, the thermal and radiological characteristics of the spent fuel authorized for storage have been modified as described in the following sections.*

W.2.1.1 NUHOMS®-61BT DSC Contents

The physical characteristics of the intact and damaged spent fuel assemblies authorized for storage in the NUHOMS®-61BT DSC are as described in Appendix K, Tables K.2-1 and K.2-2, respectively. However, to minimize the dose consequences when using the OS197L TC for the loading and transfer of the 61BT DSC, the contents allowed shall meet the Heat Load Zoning Configuration requirements of Figure W.2-1 and the DSC heat load shall be limited to 12.0 kW or less. The intact and damaged BWR fuel assembly characteristics allowed for storage in the NUHOMS®-61BT when using the OS197L TC are summarized in Tables W.2-2 and Table W.2-3, respectively. Tables W.2-4 and W.2-5 provide the Fuel Qualification Tables for the Zone 1 and Zone 2 fuel assemblies, respectively.

There is no change in the design configuration of the 61BT DSC or the three 61BT DSC basket types (Type A, B or C) relative to that described in Section K.2.1. Finally, the following criteria described in Section K.2.1 also remain unchanged:

- *The maximum fuel cladding temperatures allowed for normal, off-normal, and accident conditions, and*
- *The maximum DSC internal pressures for normal, off-normal, and accident conditions.*

W.2.1.2 NUHOMS®-32PT DSC Contents

The physical characteristics of the intact spent fuel assemblies, with or without control components (CCs), authorized for storage in the NUHOMS®-32PT DSC are as described in Appendix M, Tables M.2-1. However, to minimize the dose consequences when using the

OS197L TC for the loading and transfer of the 32PT DSC, the contents allowed shall meet the Heat Load Zoning Configuration requirements of Figure W.2-2 and the DSC heat load shall be limited to 13.0 kW or less. The intact PWR fuel assembly characteristics allowed for storage in the NUHOMS®-32PT when using the OS197L TC are summarized in Tables W.2-6. Tables W.2-7 and W.2-8 provide the Fuel Qualification Tables for the Zone 1 and Zone 2 fuel assemblies, respectively.

When using the OS197L TC, all four 32PT DSC design configurations (32PT-S100, 32PT-L100, 32PT-S-125 and 32PT-L-125) are allowed. There is no change in these 4 DSC design configurations or in the four DSC basket configurations (Type A, B, C or D) relative to those described in Section M.2.1. The minimum boron-10 content for the poison plates is unchanged at 0.0070 g/cm². The number of reconstituted fuel assemblies allowed also remains unchanged.

Finally, the following criteria described in Section M.2.1 also remain unchanged:

- *The maximum fuel cladding temperatures allowed for normal, off-normal and accident conditions, and*
- *The maximum DSC internal pressures for normal, off-normal and accident conditions.*

W.2.1.3 General Operating Functions

No change. *The content presented in Section 3.1.2 remains applicable and is not affected by the use of the OS197L transfer cask. Additional operational features applicable to the OS197L are presented in Section W.1.2.2.*

W.2.2 Design Criteria for Environmental Conditions and Natural Phenomena

The NUHOMS® OS197L TC is *in general* handled and utilized in the same manner as the existing NUHOMS® OS197 TC System. *Differences in the operation/handling of the OS197L TC include:*

- *Increased use of plant ALARA measures such as remote monitoring devices to keep exposures ALARA due to the high dose rates on the bare OS197L TC during lifts from the fuel pool to the decontamination area and from the decontamination area to the transfer trailer,*
- *Placement of the bare OS197L TC into the decontamination area shield and placement of "shield bell," and*
- *Placement of the bare OS197L TC on the transfer trailer with supplemental shielding.*

The above differences are described in detail in Chapter W.8. The environmental conditions, natural phenomena and design criteria are the same as described for the NUHOMS® OS197 TC in Chapter 3.

W.2.2.1 Tornado Wind and Tornado Missiles

No change. *The tornado wind and tornado missiles criteria presented in Section 3.2.1 remains applicable and are not affected by the use of the OS197L transfer cask.*

W.2.2.2 Water Level (Flood) Design

No change. *The flood design criteria presented in Section 3.2.2 remains applicable and are not affected by the use of the OS197L transfer cask.*

W.2.2.3 Seismic Design

No change. *The seismic design criteria presented in Section 3.2.3 remains applicable and are not affected by the use of the OS197L transfer cask.*

W.2.2.4 Snow and Ice Loading

No change. *The snow and ice loading criteria presented in Section 3.2.4 remains applicable and are not affected by the use of the OS197L transfer cask.*

W.2.2.5 Combined Load Criteria

No change. *The loads, load combinations, and design criteria for the OS197 summarized in Tables 3.2-1, 3.2-7, 3.2-11, and 3.2-12 remain applicable and are not affected by the use of the OS197L transfer cask. Additional design criteria applicable to the OS197L are presented in Section W.3.2.*

W.2.3 Safety Protection Systems

W.2.3.1 General

The discussion presented in Section 3.3.1 is applicable, except for Table 3.3-1 which is replaced with Table W.2-1. Table W.2-1 provides the safety classification of the OS197L TC system components.

W.2.3.2 Protection By Multiple Confinement Barriers and Systems

No change. *The content presented in Section 3.3.2 remains applicable and is not affected by the use of the OS197L transfer cask.*

W.2.3.3 Protection By Equipment and Instrumentation Selection

No change. *The content presented in Section 3.3.3 remains applicable and is not affected by the use of the OS197L transfer cask.*

W.2.3.4 Nuclear Criticality Safety

The content presented in Section 3.3.4 remains applicable and is not affected by the use of the OS197L transfer cask. As referenced in Sections 3.3.4.4 and 3.3.4.6, the criticality evaluations for the 61BT, and 32PT DSCs are contained in Appendices K and M, respectively.

W.2.3.5 Radiological Protection

The bare OS197L TC provides less shielding than the OS197 TC system. The reduced shielding of the bare TC results in higher dose rates on and around the TC when being lifted from the fuel pool to the decontamination area and from the decontamination area to the transfer trailer. To mitigate the effect of these high dose rates on occupational workers, these operations are done remotely as described in Chapter W.8. In addition, when the TC is in the decontamination area and on the transfer trailer, supplemental shielding is used to reduce the dose rates down to those commensurate with the OS197 TC System. Therefore, with the use of remote crane handling operation in combination with the decontamination area and skid shielding features of the OS197L TC, the occupational workers and members of the public are protected against direct radiation and releases of radioactive material.

W.2.3.6 Fire and Explosion Protection

No change. *The fire and explosion protection discussion presented in Section 3.3.6 remains applicable and is not affected by the use of the OS197L transfer cask.*

W.2.4 Decommissioning Considerations

No change. *The content presented in Section 3.5 remains applicable and is not affected by the use of the OS197L transfer cask.*

W.2.5 Summary of NUHOMS® OS197L TC Design Criteria

The principal design criteria for the NUHOMS® OS197L TC are the same as those presented for the NUHOMS® OS197 TC in Chapter 3. The NUHOMS® OS197L TC is designed to handle a *61BT or 32PT* DSC loaded with *BWR* or *PWR* fuel assemblies, *respectively, as described in this appendix.*

Table W.2-1
OS197L TC System Components and Safety Classification

OS197L TC System Components	Safety Classification
Onsite Transfer Cask	
– Structural Shell and Cover Plates	Important to Safety(1)
– Upper and Lower Trunnions	Important to Safety(1)
– Decontamination Area Shield	Not Important to Safety(1)
– *Cask support skid supplemental shielding*	Important to Safety(1)
Transfer Equipment	
– Cask Lifting Yoke	Safety Related(2)
– *Transfer* Trailer/Skid	Not Important to Safety(1)
– Ram Assembly	Not Important to Safety(1)
– Dry Film Lubricant	Not Important to Safety(1)

Notes:

(1) Structures, systems and components "important to safety" are defined in 10CFR 72.3 as those features of the ISFSI whose function is (1) to maintain the conditions required to store spent fuel safety, (2) to prevent damage to the spent fuel container during handling and storage, or (3) to provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

(2) Yoke and rigid or sling lifting members are classified as "Safety Related" in accordance with 10CFR50.

Table W.2-7
Fuel Qualification Table for 0.6 kW PWR FAs in Zone 1 of a NUHOMS®-32PT DSC Contained in an OS197L TC

(Minimum required years of cooling time after reactor core discharge)
(concluded)

Tables W.2-7 and W.2-8 provide a methodology for determination of fuel assemblies qualified for storage in the NUHOMS®-32PT DSC.

- *Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.*
- *For fuel assemblies with CCs, increase the indicated cooling time by 1.5 years.*
- *For fuel assemblies reconstituted with up to 10 stainless steel rods, increase the indicated cooling time by 1.5 years. If more than 10 stainless steel rods are present, increase the indicated cooling time by 6 years.*
- *Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.*
- *Fuel with an initial enrichment less than 1.1 and greater than 5.0 wt.% U-235 is unacceptable for storage.*
- *Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage.*
- *Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.*

Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a nineteen-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

W.3 Structural Evaluation

This section describes the structural evaluation of the NUHOMS® OS197L Transfer Cask (TC). The OS197L TC is a modified version of the OS197/OS197H TCs (henceforth referred *to* as the OS197 TC) designed *to handle a loaded* weight of *up to 125* tons. The OS197L TC may *only* be used for *the loading and* transfer of *61BT or 32PT DSCs with a heat load of 13kW or less as described in Chapter W.2, Figures W.2-1 and W.2-2*. The structural evaluation for the OS197L TC is based on the OS197 TC evaluations documented in *Chapter 8 and in* Appendices K *and* M for payloads associated with the 61BT *and* 32PT respectively. The additional evaluations provided in this section address specific design differences between the OS197L TC and the OS197 TC.

The OS197L TC requires use of supplemental shielding when the transfer cask is in the decontamination area during handling operations and when the transfer cask is placed on the transfer trailer skid. The structural evaluation of the supplemental shielding is summarized in Section W.3.9.

W.3.1 OS197L TC Description

The specific design differences in the OS197L TC relative to OS197 TC are summarized below:

- The 1.5" thick structural shell and the 0.5" thick inner liner (both SA-240 stainless steel) are replaced with a single thicker 2.68" thick shell of the same material. This represents an increase in the TC shell structural capacity relative to the OS197 TC.
- The encapsulated 3.56" thick lead thickness in the OS197 TC is eliminated to achieve the desired weight reduction.
- A neutron shield assembly is provided with the inner and outer shells made from ¼" thick plate material instead of a neutron shield assembly that is integral to the structural shell on the inside and a 3/16" thick outer shell. The neutron shield materials (type 304), total annulus water thickness of 3" and the configuration of the internal stiffening elements remain *essentially* unchanged.
- The two-piece upper trunnions assemblies made from SA-564 Type 630 steel trunnion and welded into a forged Type 304 steel trunnion sleeve with encapsulated NS-3 for the OS197 TC are replaced with one solid trunnion design made from SA-182 Type FXM-19 stainless steel. This modified trunnion design results in a stronger trunnion as it eliminates the SA564, Type 630 to SA 240, Type 304 weld.
- The two-piece lower trunnions made from Type 304 stainless with encapsulated NS-3 are replaced with solid Type 304 forgings.

Specific evaluations are performed to address the modified OS197L TC trunnion configuration. The evaluations also address the effect on local shell stresses. Thermal stresses of the cask are also evaluated. All other structural analyses for the OS197 TC bound the OS197L TC because the cask structural shell capacity of the OS197L TC is higher than that provided by the OS197 and the top and bottom forging assemblies are unchanged.

W.3.2 Design Criteria

The structural design criteria for the OS197L TC are the same as that applicable to the OS197 TC as summarized in Chapter 3. Similar to the OS197 TC, the OS197L TC is designed to meet the stress allowables of the ASME Code [3.2] Subsection NC for Class 2 components. The OS197 TC criteria summarized in Table 3.2-1 (component design loadings, as applicable), Table 3.2-7 (load combinations), Table 3.2-11 (stress criteria) and Table 3.2-12 (bolts design criteria) are applicable to the OS197L TC. The OS197 TC ASME Code exceptions described in Table 4.9-1 *are* also applicable to the OS197L TC.

The test load criteria for the upper trunnions of the OS197L TC are the same as described in Section 4.2.3.3, except that the test load is conservatively equal to 300% of the design load (instead of 150% for the OS197 TC).

The supplemental shielding is designed in accordance with AISC Code, Manual of Steel Construction, Ninth Edition [3.4].

W.3.3 OS197L TC Weight

The *empty and loaded* weights of the OS197L TC *are* presented in Table W.3-1. The total *empty* weight of the cask *(without any payload but* including neutron shield *full of* water *and the cask lid)* is approximately 62,000 lb. This compares with the corresponding weight of 111,250 lb for the OS197 TC.

The OS197L TC weights as described in Table W.3-1 are to be used in conjunction with the payload weights for the various DSCs as described in the applicable sections in Appendices K.3 *and* M.3. Each specific user is to evaluate the total under-the-hook lift weights against plant specific crane capacity limits in accordance with the requirements of 10CFR72.212.

W.3.4 Mechanical Properties of Materials

The materials properties for the OS197L TC are specified in Section 8.1, Table 8.1-3.

W.3.5 General Standards for Casks

The OS197L is fabricated using the same materials as the OS197 TC. Thus, there are no changes to the documentation in Appendices K.3 *and* M.3 relative to chemical and galvanic reactions.

The evaluation of the OS197L TC is based on critical lift weights of 250,000 lb *(125 tons).*

The thermal analysis of the OS197L along with a summary of the effect on pressures and temperatures is described in *Chapter* W.4.

As reported in Section 8.2.5.1C, the g loads for the OS197 TC were determined to be 59 g for the end drop, 49 g for the side drop and 25 g for a corner drop. Based on these accelerations, bounding accelerations of 75g for the horizontal (side) and vertical drops and 25g for the corner drop were used for the OS197 TC drop evaluations. The OS197 TC evaluations are documented in Chapter 8. Using the same methodology as that described in Section 8.2.5.1C for the OS197 TC, the equivalent loads for the OS197L TC are 75 g for an end drop, 61 g for a side drop and 25 g for a corner drop. Therefore, the 75g accident drop evaluation results for the side and end drops and the 25g evaluations for the corner drop performed for the OS197 TC and reported in Section 8.2 remain bounding and are applicable to the OS197L TC. These g-loads are conservative with respect to shell stresses since the thicker OS197L TC shell has a higher load capacity than the OS197 TC shell configuration. Hence, all the cask accident drop results reported in Section 8.2 *(8.2.5.2)*, and Appendices K.3 *(K.3.7.5) and* M.3 *(M.3.7.5)* remain bounding and, thus, are not affected.

As with the OS197 TC and as documented in UFSAR Section 8.2.5.3, complete loss of neutron shield is postulated for the OS197L TC as a consequence of the drop accident event. This event is evaluated in Sections W.4 and W.5 for thermal and shielding effects, respectively. The post accident recovery actions as discussed in Section 8.2.5.4 are applicable to the OS197L TC.

W.3.8 Effect of OS197L Temperatures on DSC Shell and Basket Components

Based on the thermal analysis documented in Chapter W.4, the maximum temperatures applicable to the 61BT and 32PT DSC shell and basket components for transfer in the OS197L TC (normal, off-normal, and accident conditions) are much lower than the corresponding temperatures for transfer in the OS197 TC presented in Appendices K.4 and M.4. Thus, there is no adverse affect on the material properties or the allowables used for the evaluations of 61BT and 32PT DSCs as documented in Appendix K, Chapter K.3 and Appendix M, Chapter M.3 respectively. Furthermore, the accident pressures used in the structural evaluations bound those calculated in Chapter W.4.

W.3.9 Structural Evaluation of OS197L TC Supplemental Shielding Components

The OS197L TC supplemental shielding when the transfer cask is placed in the decontamination area consists of an upper cask shield (shielding bell) and a lower cask shield (shielding sleeve) made from carbon steel plate 6" thick. The supplemental shielding when the transfer cask is mounted on the trailer's skid consists of massive (2.5" thick combined with additional 3" thick) carbon steel plates which are integral (i.e., welded) to the skid or bolted to each other. The supplemental shielding components are evaluated using the stress allowable criteria of AISC Code, Manual of Steel Construction, Ninth Edition, summarized in Table W.3-5. The decontamination area shielding is evaluated for deadweight and seismic loads. The skid-mounted supplemental shielding is evaluated for deadweight and conservatively defined (2g) handling loads. Conservatively evaluated bounding stresses are summarized in Table W.3-6 and Table W.3-7.

The above evaluations assume that the supplemental shielding components are handled using a single failure proof crane when these components are handled inside the fuel/reactor building. If a single failure proof crane is not used, the licensee is to evaluate the accidental drop of these *shielding components under the provisions of 10 CFR 50.59 and 10 CFR 72.212, and evaluate consequences of the accident drops.*

The only component that may be handled outside the fuel/reactor building is the top (outer) skid shielding. The evaluation of accidental drop of this component is provided in Section W.11.1.5.

W.3.10 References

3.1 *Not used.*

3.2 American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1983 Edition with Winter 1985 Addenda.

3.3 American National Standard, "For Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More," ANSI N14.6-1986, American National Standards Institute, Inc. New York, New York (1993).

3.4 AISC Manual of Steel Construction, Ninth Edition, 1989.

Table W.3-1
Summary of OS197L TC Weights

Item	Weight Configuration	
	Wet (lbs)	Dry (lbs)
OS197L Cask Body with Neutron Shield Assembly	52,236	52,236
Neutron Shield Water	4,606	4,606
Top Cask Lid	-	5,147
Water in DSC and DSC/TC Annulus	12,708	-
Bounding Nominal Payload (1)	99,133	102,222
Loaded OS197L TC Total Weight	168,683 (84.3 tons)	164,211 (82.1 tons)

Notes:

(1) Bounding of 32PT and 61BT DSCs

temperatures predicted using this methodology were validated by NRC models that included the effects of non-uniform heat fluxes.

As described in Section W.4.6, a separate model of the *OS197L TC* is used to evaluate the heat transfer within the *TC* using the computed temperatures on the neutron shield as a boundary condition. Based on a 183.85 inch length for the water cavity in the neutron shield, an outside radius of 40.18-inches for the neutron shield shell, and a design decay heat loading of 13 kW for the 32PT DSC, the uniform heat flux applied over the surface area of the shell is computed as:

$$\ddot{q} = \frac{13\text{kW} \cdot 3412.1415 \frac{\text{Btu}/\text{hr}}{\text{kW}}}{\left(2 \cdot \pi \cdot 40.18\text{in} \cdot 183.85\text{in}/144 \frac{\text{in}^2}{\text{ft}^2}\right)} = 137.62 \frac{\text{Btu}}{\text{hr} \cdot \text{ft}^2}\ .$$

A 12-inch long segment at the center of the OS197L TC on the shielded transfer trailer (see *Figure* W.4-*4*) is used to compute the flow field around the TC. Radiation exchange is modeled using the discrete ordinate methodology.

The computational mesh extends *660*-inches in the x-direction and *1,000*-inches in the y-direction. *Figure* W.4-*5* illustrates perspective and plane views of the computational mesh at the centerline. *Figure* W.4-*6* presents enlarged views of the computation mesh illustrating the boundary layer mesh on the cask shell and the inner surface of the shields. A grid sensitivity study, conducted as part of the analysis for a variant of the OS197L TC design, demonstrated that the meshing used for this evaluation was appropriate.

While the computational model is 3-D in its construction, the resulting analysis is effectively 2-D since symmetry conditions are assumed at the axial ends of the model and a uniform heat flux is assumed at the inner surface.

Assumptions Used in CFD Modeling
The general assumptions used in the CFD modeling are:

1. Heat removed through the cask end plugs and by conduction via trunnion contact with the transfer skid is conservatively neglected.

2. The total decay heat is considered to be evenly distributed over the outer surface of the cask's liquid neutron shield shell. The assumption of uniform heat flux is consistent with previous OS197 analysis methodology (See Chapter M.4) and reflects the axial spreading of the decay heat load due to the high axial conductivity of the DSC basket and rails, and the water filled neutron shield.

3. The CFD modeling need only address the geometry of the OS197L TC and its shielded *transfer* skid as it exists between the front and rear trunnion towers. See the justification of this assumption provided below.

4. The outer surfaces of the auxiliary shielding on the transfer skid are assumed to be finished with a 'dark blue' color coating that yields a solar absorptivity of 0.90 or less and an emissivity of 0.85 or greater. Similarly, the inner surface of the auxiliary shielding is assumed to have a similar finish that yields an emissivity of 0.85 or greater.
5. The regulatory insolation [10CFR Part 71] averaged over 24 hours is applied to the outer surfaces of the auxiliary shielding. The thickness of the auxiliary shielding, combined with the thermal mass of the OS197 cask and payload, justifies the use of 24-hour averaged values. The 24-hour average insolation on the roof of the transfer skid is assumed to be 122.9 Btu/hr-ft^2, 30.75 Btu/hr-ft^2 on the vertical surfaces, and 61.45 Btu/hr-ft^2 on the angled portion of the auxiliary shielding. These incident heating values are reduced by 10% to account for the assumed solar absorptivity of 0.90 for the coating used on the shields (see Assumption 4).

6. The ground is conservatively treated as an adiabatic surface.

The analysis for the off-normal ambient condition of 117°F is conducted assuming a 24 hour average steady-state ambient temperature of 107°F. A steady-state analysis at this temperature level has been shown by previous analyses to bound the transient thermal performance achieved using a diurnal cycle for ambient air with a peak temperature of 117°F.

Justification of CFD Model Segmentation

The use of a quasi 2-D thermal model located at the center of the OS197L TC to evaluate the bounding thermal performance of the OS197L TC on its shielded transfer skid is justified for the following reasons:

1. Heat transfer near the center of the OS197L TC is via radiation and convection, while heat transfer near the ends of the cask also involves direct heat transfer through the trunnions to the transfer skid. In addition, the cask's lid and end closures add additional heat transfer area. As such the peak temperatures on the OS197L TC will occur near its center.

2. A flow area clearance of approximately 3.8 inches will exist in the vicinity of the cask trunnions vs. approximately 3.3 inches at the center of the cask.

3. The 113.5 inches between the front and rear trunnion towers spans 62% of the liquid neutron shield length, over which the majority of the heat rejection occurs.

4. The assumption of a uniform heat flux means there will be no significant variation in thermal conditions between the front and rear trunnion towers. Thus, modeling a short segment will yield similar results to modeling the entire length.

5. The airflow through the shielded transfer skid is driven by buoyancy forces which, in turn, are driven by the temperature on the shell of the neutron shield. As such, the convective airflow over the surface of the TC is self-correcting, with the level achieved representing a balance between the local flow resistance and surface heat flux. For this reason the use of a relatively short axial segment to represent the flow regime within the transfer skid enclosure does not entail a risk of over-estimating the prototypic airflow

W.5 Shielding Evaluation

This appendix presents the shielding evaluation of the OS197L TC when used for fuel loading and transfer of 32PT and 61BT DSCs.

The shielding analysis is performed for various configurations of the OS197L TC during loading and transfer operations containing a fully loaded 32PT *or* 61BT DSC with a maximum 13 kW/DSC heat load for PWR and BWR fuel assemblies, respectively. Bounding dose rates for the *OS197L TC* with the 32PT and 61BT DSC payloads are evaluated. The dose rates on and around the bare OS197L TC are dominated by primary gamma sources during normal, off-normal and accident conditions of loading and transfer. The results for normal operations demonstrate that exposures for OS197L TC activities with operational personnel present are bounded by OS197 TC exposures (remote crane operation is used and no personnel are present in the immediate vicinity of the *OS197L* TC while the cask is on the crane hook during normal operations and decontamination area shielding *is used*).

The physical characteristics of the intact and damaged spent fuel assemblies authorized for storage in the NUHOMS®-61BT DSC are as described in Appendix K.2, Tables K.2-1 and K.2-2 respectively. However, to minimize the dose consequences when using the OS197L TC for the loading and transfer of the 61BT DSC, the contents allowed shall meet the heat load zoning configuration requirements of Figure W.2-1 and the DSC heat load shall be limited to 12.0 kW or less. The intact and damaged BWR fuel assembly characteristics allowed for storage in the NUHOMS®-61BT when using the OS197L TC are summarized in Table W.2-2 and Table W.2-3, respectively. Tables W.2-4 and W.2-5 provide the fuel qualification tables *(FQTs)* for the Zone 1 and Zone 2 fuel assemblies, respectively.

The physical characteristics of the intact spent fuel assemblies, with or without control components (CCs), authorized for storage in the NUHOMS®-32PT DSC are as described in Appendix M, Table M.2-1. However, to minimize the dose consequences when using the OS197L TC for the loading and transfer of the 32PT DSC, the contents allowed shall meet the heat load zoning configuration requirements of Figure W.2-2 and the DSC heat load shall be limited to 13.0 kW or less. The intact PWR fuel assembly characteristics allowed for storage in the NUHOMS®-32PT when using the OS197L TC are summarized in Table W.2-6. Tables W.2-7 and W.2-8 provide the *FQT*s for the Zone 1 and Zone 2 fuel assemblies, respectively.

The bounding *dose rates for* normal and accident conditions of transfer are due to radiological sources specified in Section W.5.2. The sources are shown in Table W.5-1 and Table W.5-2. These tables represent radiological sources at burnup and enrichment combinations from 32PT and 61BT DSC transfer FQTs.

The FQTs for the 61BT DSC when using the OS197L TC are presented in Appendix W.2, Table W.2-4 *and* Table W.2-5. *The FQTs for the 32PT DSC when using the OS197L TC are presented in Appendix W.2*, Tables W.2-7 and W.2-8. These tables assure that the total dose rate during normal condition of transfer on the bare cask surface does not exceed the calculated maximum dose rates (shown in Tables W.5-8 *and W.5-11*) as well as assuring that the maximum decay heat load per DSC is not greater than 13 kW/DSC for the 32PT DSC and 12 kW/DSC for the 61BT

DSC. These tables also provide assurance that single fuel assembly decay heat limits shown in Appendix W.2, Figure W.2-1 and *Figure* W.2-2 are not exceeded.

W.5.1 Discussion and Results

A summary of the bounding maximum dose rates on and around the OS197L TC with the 61BT or 32PT DSC during loading and transfer operations during normal and accident conditions are shown in Table W.5-3 and Table W.5-4, respectively. The bounding maximum dose rates *for* various shielding configurations of the OS197L TC with the 61BT or 32PT DSC during the various operational evolutions for normal, off-normal and accident conditions, at various locations are shown in Table W.5-6 through Table W.5-14. A brief description of the various shielding configurations evaluated herein for various loading and transfer operations is provided *in* Figure W.5-1 and in Section W.5.4.10.

A discussion of the method used to determine the bounding source terms for this evaluation is included in Section W.5.2. The shielding material densities are given in Section W.5.3. The *model specification and the* method used to determine the dose rates due to 32 PWR or 61 BWR allowed fuel assemblies in the various OS197L TC design configurations with 32PT DSC and 61BT DSC payload is provided in Section W.5.4. The radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the spent fuel contents. The shielding evaluation is performed with the MCNP5 [5.2] code with the ENDF/B-VI cross section library.

W.5.1.1 Dose Rates Near Shielding Configurations Containing 32PT and 61BT DSC

In general, *the* intensity and shape of *the* radiation field distribution around a shielding configuration is determined by two factors: (1) *the* intensity and spatial distribution of radiological sources, and (2) *the* shielding properties and spatial configuration of shielding materials. Also note that the maximum of gamma and neutron radiation dose rates may occur at different locations. Therefore the maximum of total dose rate is not necessarily equal to the sum of the maximum gamma and maximum neutron dose rates.

The bounding radiological source terms employed herein are shown in Table W.5-1 and Table W.5-2 for the 32PT DSC and 61BT DSC, respectively. Such sources result in bounding dose rates from 32PT and 61BT DSC contents of the cask when compared with other radiological sources at various burnup, enrichment, and cooling time combinations shown in *the FQTs*. Neutron radiation dose rates near the OS197L bare cask containing a 32PT DSC bounding sources are bounding for the neutron radiation dose rates from the cask containing the 61BT DSC bounding sources.

The data presented in Tables W.5-6 through Table W.5-14 are based on MCNP calculated dose rates. There is always a statistical uncertainty in the results obtained using a Monte Carlo method like MCNP.

Because of axial symmetry of the shielding materials distribution on the side of the cask and the DSCs and in the distribution of radiological sources, it is convenient to consider dose rate distribution along the cask side in a cylindrical coordinate system with an axis that coincides with the axis of the cask/DSC. In general, dose rate is dependent on all three coordinates: axial, radial and angular. Dose rate distribution along the side of the cask has an angular symmetry. The 32PT DSC has R45 and R90 solid aluminum rails on its periphery which provide noticeable shielding at the periphery of the DSC where the rails are present. There is no such shielding on the periphery of the 61BT DSC. Therefore, the primary gamma radiation *(PGR)* dose rate on and near *the* side of the cask has a pronounced angular dependence *with the* 32PT DSC. Further, the 32PT DSC can contain stronger radiological sources. As a result, maximum values near the cask surface occur when the cask is loaded with a 32PT DSC. Aluminum does not affect the shape of the neutron radiation dose rate distribution as dramatically as it does the *PGR*.

Two sets of dose rates near the bare cask at various axial and radial distances are calculated for the OS197L TC for both the 32PT and 61BT DSCs at normal and accident conditions. *Normal conditions are modeled with a dry DSC and DSC/TC annulus with a water-filled neutron shield and accident conditions are modeled with a dry neutron shield.* These dose rates are presented in Table W.5-6 through Table W.5-11. For the other shielding configurations described in Section W.5.4.7 *(to determine dose rate distributions during the various operational evolutions with the OS197L TC)*, only bounding dose rates at distances up to 10 meters are presented. They are from a 32PT DSC payload of the cask. The dose rates *for the various operational evolutions* at distances greater than 10 meters *do not have pronounced axial and angular variations and are* presented in Table W.5-6 and *Table* W.5-7 *for the 32PT DSC*, and Table W.5-9 and *Table* W.5-10 *for the* 61BT DSCs.

W.5.1.2 <u>Bounding Dose Rates as a Function of Distance</u>

Dose rates as a function of distance for various shielding configurations of the OS197L TC under normal conditions of transfer are plotted on Figure W.5-2.

Figure W.5-2 displays *five* dose rate versus radial distance *curves* from the side *for 4* shielding configurations: (1) bare cask, *(2)* cask with 2.5" thick steel shell; *(3)* the cask on a trailer platform with 2.5" thick inner top *support skid supplemental* shielding *and without* 3" thick outer top steel shielding; *and (4)* the cask ready for *transfer* (i.e., cask with both inner and outer top trailer shielding installed). Dose rates *shown in Figure W.5-2 are bounding* for the OS197L containing either a 32PT or *a* 61BT DSC.

Note that the support skid supplemental shielding is referred to as trailer shielding or trailer area shielding throughout this chapter.

The bare cask dose rate curve (curve 1) as a function of distance shown in Figure W.5-2 is bounding for all other configurations. It is included to provide a comparative understanding of the shielding effectiveness of the trailer shielding. These dose rates are based on results shown in Table W.5-6.

The dose rate curve (curve 2), as a function of distance corresponding to the "Cask with Additional 2.5" thick Steel Shell Only" shown in Figure W.5-2, is based on an MCNP model that includes a 2.5" thick steel cylindrical shell around the OS197L TC (see description "Pre-Transfer" in figure W.5-1). This configuration is a simplified representation of the OS197L TC prior to the placement of the outer top trailer shielding. These dose rates are based on results shown in Table W.5-12.

*The d*ose rate *curve*s *for the three remaining configuration*s correspond to the cask on the *transfer* trailer platform. Geometry of the modeled shielding configuration *(only the inner top trailer shielding is modeled)* is depicted on Figure W.5-5. The*se* plots display *the* maximum dose rates in *the* horizontal direction from the side of the transfer package. Note that there is no shielding underneath the cask in the computation*al* model, except for the 0.25" thick steel plate *representing* the trailer platform.

The *dose rate curve (curve 3) as a function of distance corresponding to the* *"Cask on Trailer with 2.5" thk. Inner Top Trailer Area Shielding"* *is for the configuration* whe*re* the cask is on *the* trailer platform *where the 2.5" inner top trailer shielding is installed and the* 3" thick steel outer top trailer shielding is not yet installed. The *dose rate* maximum is near the interface between the 2.5" thick inner top trailer shielding over the cask side and the 5.5" thick *side trailer* shielding *(side shielding plates in Figure W.5-5)*. *T*hese dose rates are nearly identical to those *shown in curve 2 because the maximum dose rates occur around the 2.5" thick shielding.*

*The d*ose rate *curve*s *for* the two *remaining configurations* correspond to *calculated* maximums at a vertical elevation between *the ISFI concrete pad* and *the* trunnions.

The *dose rate curve (curve 4) as a function of distance corresponding to the* *"Cask on Trailer with 2.5" thick Inner and 3.0" thick Outer Trailer Top Shielding"* *is* the maximum dose rate at a vertical elevation between the bottom of the 5.5" thick *trailer shielding* along the side of the cask and trunnions level. *These dose rates are based on results shown in Table W.5-13.*

The*se* dose rates are bounding for the cask in the decontamination area for the following reasons. First, these dose rates are for 5.5" thick steel shielding along the side of the cask, while the decontamination area shield thickness is 6.0". Second, the cask is in a horizontal position on *the* trailer platform *with* no shielding underneath the trailer platform modeled when calculating the dose rates. This causes substantial radiation streaming and scattering from the *ISFSI* concrete *pad* surface *which are accounted for in the calculated* dose rates. On the other hand, the cask in the decontamination area *is* in a vertical position and entirely surrounded with a 6" thick steel cylindrical shell.

The *dose rate curve (curve 5) as a function of distance corresponding to the* "*Below Cask Support Skid when Cask is on Trailer with Inner and Outer Trailer Top Shielding*" *is* the maximum dose rate at *the* vertical elevation between the *ISFSI concrete pad* and the cask support skid. *As expected*, this distribution shows fairly high dose rates at short distances. This is because of direct radiation from underneath the cask, where no shielding is present, except for the 0.25" thick steel plate *representing* the trailer platform, and radiation scattered from concrete. The *dose rates* shown in Table W.5-14 *are employed to plot this curve. The dose rate distribution at this elevation is not significantly dependent on the outer top trailer shielding and therefore, the dose rates shown in Table W.5-14 are applicable.*

The dose rate distributions (curve 2 through curve 5) shown in Figure W.5-2 indicate that the maximum dose rates decrease to approximately 100 mrem/hr at a distance of 2 m from the cask transfer trailer.

W.5.1.3 Bounding Dose Rates as a Function of Axial Distance

For the dose rates on top of the cask, two radial regions need to be considered: within and beyond the cask radius. These radial regions are depicted in Figure W.5-3. Dose rates beyond the cask radius are *due* to radiation from the side of the cask and through shielding on its top. Because there is substantially less shielding and sources are stronger (because of the in-core region) on the side, the contribution by radiation from the side is dominating. On the other hand, dose rates just over the ends of the cask at radial distances within the cask radius are contributed *due* to radiation through shielding over the ends of fuel assemblies. It is assumed that the water neutron shielding on the side of the cask is lost during the accident. Therefore, accident and normal condition dose rates on top of the cask are not different when radial distances less than the radius of the cask are under consideration. Hence, the dose rates on the TC axis (r=0) and at r<= TC radius are applicable for both normal and accident conditions at "short" or relatively "closer" axial distances. Here the axial distance is the distance from the cask end when dose rate distribution becomes uniform (i.e., maximum and average dose rates are *approximately the same*). For the axial dose rates, this occurs at distances greater than 10.0 meters. Because shielding properties on the top of the various configurations are identical, dose rates at these axial distances are the same for various shielding configurations when radial distances less than the cask radius are considered.

When the cask is ready for transfer to the ISFSI, there is at least 12.1" of steel and 2.0" of NS3 over the top of the fuel assemblies for the 32PT DSC. The shielding of steel on top of the 61BT DSC is 1.5" thinner. At the same time, primary gamma and neutron radiation sources (in the two energy groups that are the dominant contributors to the *PGR* dose rates, 1.00 to 1.33 and 1.33 to 1.66 MeV) are by a factor of ~3.1 and 4.8 stronger in the central compartments of the 32PT DSC, respectively. It is established in the shielding analysis that such a difference makes the TC dose rates with the 32PT DSC source, bounding. Also, dose rates in *Chapter* M.5, Table M.5-5 are calculated with a model having the geometry depicted in Figure M.5-24. The model uses 2.45" of NS3 but only 9.75" of steel. The model distributes 32PT DSC design basis *(DB)* radiological sources from 1.2 kW/FA assemblies *(conservatively)* in 16 peripheral fuel compartments instead of 8 peripheral compartments at the corners of the DSC basket structural grid.

This makes the total top dose rates predicted with the shielding model of UFSAR Appendix M very conservative. For the OS197L TC, the dose rates on top of the cask depicted on Figure M.5-24 and presented in Table M.5-5 are bounding for the dose rates on top of the OS197L cask containing a 32PT DSC when radial locations bounded by the bare cask radius are under consideration. Therefore, dose rates *on the* top of the reference shielding configuration presented in Table M.5-5 are bounding for dose rates on the top of the OS197L containing a *61BT* or *32PT* DSC *with contents bounded by* source *term*s presented in Table W.5-1 and Table W.5-2, *respectively*.

W.5.2 Source Specification

The radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the fuel. The computational model of the *DB* PWR fuel from Appendix M.5 is directly utilized, as appropriate, to calculate the bounding radiological source *terms* due to assemblies in the 12 central (Zone 1) and 20 peripheral (Zone 2) fuel compartments of the 32PT DSC *shown in Figure W.2-2*. Similarly, the computational model of the *DB* BWR fuel from Appendix T, Chapter T.5, is directly utilized to calculate bounding radiological source *terms* for the assemblies in *the* 48 peripheral (Zone 2) fuel compartments of the 61BT DSC *shown in Figure W.2-1*. The sources for the central 13 fuel compartments (Zone 1) of the DSC are *unchanged and are obtained* directly from Appendix K.5, Table K.5-7.

FQTs for various decay heat *values* are presented in Appendix M*.2*, *Appendix* P*.2 and Appendix U.2 for PWR fuel assemblies, and Appendix* T *for BWR fuel assemblies*. The minimum required cooling times in order not to exceed the desired decay heat limits were determined using the SAS2H\ORIGEN-S models of the corresponding *DB fuel* assemblies. Also, radiological source *term*s were obtained as a "side" product during the calculation for all burnup, enrichment and cooling time (BECT) combinations *evaluated for the OS197L TC. I*mportant observations regarding the radiological source *terms* and their applicability to the OS197L *TC* provide the foundation for the determination of the bounding radiological sources. The*se* observations are summarized below.

- The dose rates along the side of the OS197L *TC* are mainly dominated by assemblies located in peripheral fuel compartments. This is especially true when all the fuel compartments contain the same radiological sources. Because of that, it is desirable to keep "hot" assemblies closer to the cask axis in order to minimize dose rates.
- The *PGR* dose rates are *dominated by the* radiological sources in the 1.0 to 3.0 MeV energy groups.
- ^{106}Rh ($T_{1/2}$=29.8 second) and ^{144}Pr ($T_{1/2}$=17.3 minutes) isotopes are the major contributors to the intensity of the *PGR* source *term* in the 2.0 to 3.0 MeV energy group. *The contribution to the PGR* dose rates by the source*s* in th*is* energy range *is more prominent* at cooling times below 3.0 years. These isotopes are products from ^{106}Ru ($T_{1/2}$=1.015 years) and ^{144}Ce ($T_{1/2}$= 0.7805 years) decay, respectively.
- *The PGR* dose rate is dominated by radiological sources in the 1.00 to 1.66 MeV energy range at cooling times greater than 5.0 years. The contribution is greater than 70%. *The i*ntensity of the *PGR* source in *this* energy range is mainly *due* to *the* ^{60}Co ($T_{1/2}$=5.27 years) *isotope*.
- *B*oth *the* intensity of the *PGR* source in the 1.00 to 1.66 MeV energy range and its fraction in the total *PGR* source intensity *are important for this purpose*. *A*ny of the *BECT* combinations *may be selected* to *yield* the bounding *PGR* source *terms* if the difference in the calculated dose rates falls within a*n* acceptable band *of 10%*. *This* approach is *adequate for* selecting a bounding source for assemblies in a group of fuel compartments *whose contribution* to the total dose rate *is* substantially less than assemblies in other compartments.
- A description of shielding for the bare OS197L cask can be found in various sections of this chapter. Normal and accident conditions dose rate for such a shielding configuration is essentially dominated by *PGR* sources. *Therefore*, one can use, without introducing significant conservatism, the largest neu*t*ron radiation source among considered *BECT* combinations.

*The methodology to calculate the FQT*s for the bounding BWR *assemblies is described in Appendix K, Chapter K.5 and Appendix T, Chapter T.5. The methodology to calculate the FQTs for the bounding* PWR assemblies *is described in Appendix M, Chapter M.5; Appendix P, Chapter P.5; and Appendix U, Chapter U.5*. The term "bounding assemblies" means that radiological and decay heat sources from such assemblies are bounding for qualified BWR and PWR classes of assemblies. However, *these* FQTs provide only minimum required cooling times at various burnup and enrichment combinations *such that the resulting* decay heat *per fuel* assembly *is below a specified maximum. The difference between the reference FQT methodology and that employed for the OS197L TC is that one also needs to ensure that the resulting dose rates are below a specified maximum limit (bare cask maximum dose rate below 10 rem/hr).* Note that radiological sources at various cooling times for each decay heat FQT burnup and enrichment combination are also determined *as part of the FQT methodology. This implies that the results of the SAS2H/ORIGEN-S evaluations employed to determine the FQTs for the bounding BWR and PWR fuel assemblies for a variety of decay heat values and BECT combinations can be utilized for the OS197L TC evaluations.* Additional observations can be made from radiological sources data related to the decay heat FQTs generated in *these* calculations and summarized as follows:

- *For all the burnup and enrichment combinations presented in a decay heat FQT, the bounding neutron radiation source occurs at a combination at the lowest enrichment and at the highest burnup. This statement can be easily verified by simply looking at a set of neutron radiation sources relevant to any FQT presented in the above referenced appendices.*
- *For* a set of burnup and enrichment combinations with cooling times greater than a certain value in an FQT for a given decay heat restriction, *the* bounding *PGR* source occurs at the lowest enrichment, and the *highest* burnup *and the lowest cooling time* combination in that set. *This is illustrated by the following example.*

For the 0.40 kW/PWR FA FQT presented in Appendix W.2, Table W.2-8, set the minimum threshold cooling time at 10.0 years. This ensures that all the entries in the FQT can be included in this example. Since the minimum cooling time is greater than 5 years, the contribution to the total dose rates due to PGR sources are dominated by the source terms in the 1.00 to 1.66 MeV energy range. The burnup and enrichment combinations with the minimum cooling time are selected for further evaluation thereby limiting the search to within those combinations with cooling times equal to 17.5 years (minimum cooling time to ensure that the dose rates are below the limits). For these combinations, the maximum burnup is 24 GWD/MTU and the minimum enrichment is 1.1 wt. % U-235. Therefore, the radiological source associated with 24 GWD/MTU and 1.1 wt.% can be considered as bounding because, (1) it has the strongest intensity in the 1.00 to 1.66 MeV energy range *that contributes the most to the* *PGR* dose rate on the surface of *the* OS197L *TC*, and (2) the intensity of the *PGR* source in the 1.66 to 3.0 MeV energy range is not substantially (greater than 19%) less than the intensity in the same energy groups at 1.2 and 1.3 wt.% enrichments. *This also illustrates that when cooling times are greater than 15 years (approximately 3 half-lives for Co-60), the variation in the source strength with enrichment is not as pronounced as it is at lower cooling times, particularly at low to moderate burnups.*

All the observations summarized herein *are part of the methodology to* determine a bounding radiological source and FQTs for on-site transfer. Since the bare cask dose rate is nearly completely dominated by the *PGR* source, the BECT combination resulting in the bounding radiological source can be identified as that resulting in the highest *PGR* source intensity in the 1.00 to 1.66 MeV range. By placing a restriction on the dose rate at the side of the cask, one can search for a desired combination through successive iterations by considering the sources at cooling times equal to some threshold value. Based on the discussion in the previous section, the bounding combination will be the *highest* burnup and the lowest enrichment at a cooling time equal to the threshold value. The iteration is terminated when the combination resulting in the maximum dose rate less than or equal to the dose rate limit is obtained *for all the entries in the FQT*. Since the intensity of a gamma radiation source in the 1.00 to 1.66 MeV energy range at cooling times greater than 5.0 years is mostly dominated by ^{60}Co ($T_{1/2}$=5.27 years) and the majority of the dose rate on the side of the cask is from assemblies in peripheral fuel compartments, the educated guess for a minimum threshold time can be easily obtained after calculating the maximum dose rate from any radiological source in a set of peripheral fuel compartments at cooling time greater than 5.0 years that is generated as a "side product" when determining a decay heat FQT in the various appendices referenced above.

W.5.2.1 Methodology for Determination of Bounding Radiological Source Terms

The evaluation to obtain the bounding radiological sources can be started with identifying BECT combination(s) in the decay heat FQTs that result in the bare cask normal condition dose rates less than or equal to a predetermined limiting value of 10.0 rem/hr. After that a trial and error process is used to prove that either the calculated BECT(s) combination results in the bounding dose rate or adjust cooling times in the decay heat FQTs in a manner that assures that the radiological sources from the BECT become bounding, as shown in the following steps:

1. Set an MCNP model for the OS197L bare cask containing 32PT and 61BT DSCs. Continue with the next 10 steps starting from the 32PT DSC payload of the cask. A description of the MCNP models is *provided* in Section W.5.4.7.
2. The models should allow one to determine *the* contribution to the maximum dose rate on the side of the cask due to sources in certain groups of fuel compartments. Because of the axial symmetry of *the* canister, it is convenient to group the compartments within certain radial zones *(in this case, two radial zones)*.
3. Obtain *the* maximum of *PGR* dose rate on the side of the cask when all the fuel compartments contain the same radiological sources. Start from radiological sources that result in *the* maximum *PGR* dose rate greater than the desired limit. Determine groups of fuel compartments from which the contribution to the maximum dose rate is not significant in comparison with the sources in peripheral compartments. Quantify their contribution. Such compartments can be designated to hold the "*hotter*" radiological sources.
4. A contribution to the maximum dose rate on side of the cask due to sources in peripheral fuel compartments is also identified by this point. Estimate the intensity of the *PGR* source term strength in the 1.00 to 1.66 MeV range that is needed in order not to exceed the desired dose rate limit. Use an exponential decay formula with a decay constant

relevant to ^{60}Co ($T_{1/2}$=5.27 years) to estimate the change in source term intensity as a function of cooling time.

5. Find a burnup enrichment and cooling time combination in existing decay heat FQTs that have intensity of the PGR source in the 1.00 to 1.66 MeV range that is the closest and bounded by the intensity evaluated in step 4.
6. Include the source terms in the MCNP models set up in step 1. Calculate the maximum dose rate value.
7. If the maximum dose rate obtained in step 6 is greater or substantially lower than the desired limit, repeat steps 4 through 6. Otherwise, proceed with the next step (step 8).
8. An acceptable BECT combination for assemblies in the peripheral fuel compartments generating sources resulting in dose rates on the side of the cask *below* the desired limit at the normal conditions is achieved by this step. Since the dose rate near the OS179L bare cask is nearly entirely dominated by the PGR source, it is conservative to use the absolute bounding neutron radiation source. That source occurs at the lowest enrichment and the highest burnup when considering BECT combinations relevant to decay heat FQTs of the appendices mentioned in the introduction to Section W.5.2.
9. Utilize the neutron and the PGR source obtained so far and repeat steps 6 and 7 until the maximum of the total (due to neutron and gamma radiation) dose rate on the side of the cask is below the desired limit. Then, proceed with the next step (step 10). Note, when dealing with the OS197L TC containing a 61BT DSC, ensure that the maximum dose rate is bounded by that obtained while performing a similar qualification of the radiological sources for the OS197L TC containing a 32PT DSC.
10. The intensity of the PGR source in the 1.00 to 1.66 MeV energy range at the bounding BECT combination for assemblies in peripheral fuel compartments that result in (when combined with the bounding neutron source) dose rates on the side of the cask below the desired limit is established at this stage.
11. Consider the BECT combinations that are intended for assemblies in peripheral fuel compartments. Adjust, if necessary, cooling times for burnup and enrichment combinations that generate greater intensity of the PGR source in the 1.00 to 1.66 MeV energy range than the bounding source obtained by step 10.

A similar sequence of steps is used for the OS197L TC bare cask configuration containing a 61BT DSC.

Finishing step 11 completes an adjustment of cooling times in select decay heat FQTs. The adjusted FQTs are referred to as the transfer FQTs. They assure that both the decay heat limit and a restriction for the maximum dose rate on the side of the OS197L bare cask at normal condition of transfer are satisfied. The decay heat FQTs are adjusted in such a manner that the maximum of the total dose rate on the surface of the OS197L bare cask containing a 32PT DSC is bounding for the cask containing a 61BT DSC at normal conditions of transfer. Therefore, the payload of the cask for the bounding shielding evaluation and the bounding radiological sources are identified. Also, normal conditions of transfer dose rates for the OS197L bare cask shielding configuration are calculated for 32PT and 61BT payloads after completion of step 9.

Note, in general, that there are plenty of options for an arrangement of radiological sources within fuel compartments of the DSCs contained in the cask that would result in the dose rates below the desired limit. This shielding analysis investigates the arrangement of the sources within fuel compartments grouped in two radial zones. It is assumed that loading of the radial zones is uniform, i.e., all the decay heat and radiological sources within each zone are the same.

Numerous iterations employing the MCNP models mentioned in step 1 above determined that the arrangement of fuel assemblies *as shown in Chapter W.2, Figure W.2-1 for the* 61BT DSC and *Chapter W.2, Figure W.2-2 for the* 32PT DSC result in maintaining the maximum dose rate on the side of the bare cask below *the 10 mrem/hr* limit under normal conditions. The fuel assemblies shown as Zone 1 in these figures are limited to a maximum decay heat of 0.60 kW/FA for the 32PT DSC and 0.30 kW/FA for the 61BT DSC. The fuel assemblies shown as Zone 2 in these figures are limited to a maximum decay heat of 0.40 kW/FA for the 32PT DSC and 0.17 kW/FA for the 61BT DSC.

The minimum required cooling times shown in Table W.2-4 ensure that the decay heat from Zone 1 FAs does not exceed the limit of 0.30 kW/FA for the 61BT DSC. The minimum required cooling times shown in Table W.2-5 ensure that the decay heat from Zone 2 FAs does not exceed the limit of 0.17 kW/FA for the 61BT DSC.

The minimum required cooling times shown in Table W.2-7 ensure that the decay heat from Zone 1 FAs does not exceed the limit of 0.60 kW/FA for the 32PT DSC. The minimum required cooling times shown in Table W.2-8 ensure that the decay heat from Zone 2 FAs does not exceed the limit of 0.40 kW/FA for the 32PT DSC.

Note that neutron shielding material on the side of the cask is assumed lost during accident conditions. Since implementation of the methodological steps in the current section results in bounding primary gamma and bounding neutron radiation sources, these are also bounding for accident conditions of transfer.

W.5.2.2 Primary Gamma and Neutron Source Terms

The major contributors to the *PGR* source due to LWR fuel assemblies as well as uncertainties in calculating the intensity of the source are identified in Section P.5.2.1.3, Section T.5.2.1.2 and Section U.5.2.1.3.

W.5.2.2.1 Bounding Radiological Sources for FAs in 12 Zone 1 Fuel Compartments of the 32PT DSC

The radiological source *term*s *are calculated* when preparing the 0.60 kW/FA decay heat FQT in Table *W.2-7*. *These* radiological sources at cooling times greater than or equal to 5.0 years *are employed* for further analysis. Based on the discussion in the introduction to Section W.5.2 it is identified that the bounding source occur*s* at *a burnup of* 20 GWD/MTU, *an enrichment of* 1.1 wt. % *U-235* and *a cooling time of* 5.0 years. As can be observed from intensities, *PGR* source *term* from such a BECT *combination are* bounding because *they* yield the highest intensity in the 1.66 to 3.0 MeV range*,* and intensities in the 0.8 to 1.66 MeV range are *approximately* 3% *lower* than the absolute maximum of the *PGR* source intensity in that energy range. As stated in Section W.5.2, bullet item 4, *the* PGR source in the 1.00 to 1.66 MeV range contribut*es* *significantly* to the *PGR* dose rate at cooling times greater than 5.0 years. The "weight" of the 1.0 to 1.66 MeV energy range in the total intensity of the *PGR* source is the largest at *a burnup of* 25 GWD/MTU, *an enrichment of* 1.1 wt. % *U-235* and *a cooling time of* 6.4 years. The absolute maximum of the total intensity of the *PGR* source, 3.33e+15 gammas per second per assembly, is at *a burnup of* 25 GWD/MTU, *an enrichment of* 1.6 wt. % *U-235* and *a cooling time of* 6.0 years. *The intensities* of the *PGR* source in the most *important* energy groups are not very different for assemblies in the *burnup range of* 20 to 25 GWD/MTU,enrichment *range of* 1.1 to 1.9 wt.% *U-235* and cooling times in the *range of* 5.0 to 6.5 years. *Sensitivity* MCNP runs showed that Zone 1 compartments with *these* sources contribute about 500 mrem/hr to the maximum dose rate on the surface of the *bare OS197L TC* containing a 32PT DSC. Radiological sources from 0.60 kW/FA assemblies are intended for Zone 1 fuel compartments in the 32PT DSC. Therefore, *the choice of the exact* BECT combinations *to determine the* radiological source *terms for Zone 1 locations does not significantly affect the dose rates. Therefore, a* hybrid source is utilized *for Zone 1 locations.* Intensities and spectrum of the *PGR* source in the bottom nozzle, plenum, and top nozzle are due to *fuel assemblies with* a *burnup of* 25 GWD/MTU, *an enrichment of* 1.6 wt. % *U-235* and *a cooling time of* 6.0 years. Intensity of the source in *the* in-core region is also due to *the same* combination but the spectrum is *conservatively* "hardened" by assigning *the* fraction of the total source intensity at 2.0–2.5 MeV energy *range obtained from a fuel assembly with a burnup of* 20 GWD/MTU, *an enrichment of* 1.9 wt.% *U-235* and *a cooling time of* 5.0 years. When used in MCNP calculations, *this* hardening decreases the frequency of the radiological source spectrum sampling in the most *important* energy *range*, 1.00–1.66 MeV, by 0.03% *while increasing the sampling frequency of* the 2.0–2.5 MeV energy *range by* 5 times. Therefore, the "hardened" spectrum for the incore region, the region which essentially dominates dose rates at various radial distances from *the* side of the cask, is bounding.

The combined intensity of the *PGR* "hybrid" source from all the axial exposure regions is 3.34e+15 gammas per second per assembly. As mentioned in Section W.5.2, bullet item 8, the bounding neutron source is due to a BECT combination at the lowest enrichment and the highest burnup *when* considering a set of the BECTs represented in the decay heat FQT. *This* source is due to a *fuel assembly with a burnup of* 45 GWD/MTU, *an enrichment of* 1.1 wt. % *U-235* and *a cooling time of* 27.2 years. Its intensity is 6.86e+8 neutrons per second per FA. The source *terms* are summarized in Table W.5-1. *These source terms are bounding for the 32PT DSC DB fuel assemblies for loading in Zone 1 (Figure W.2-2) with an initial loading of 0.475 MTU.*

The decay heat due to the fuel assembly parameters employed in the PGR source term calculations is 571 watts. The decay heat due to the fuel assembly parameters employed in the neutron source term calculations is 597 watts.

W.5.2.2.2 Bounding Radiological Sources for FAs in Zone 2 Fuel Compartments of the 32PT DSC

The radiological source terms were generated when preparing the 0.40 kW/FA decay heat FQT in Table W.2-8. An iterative process was employed to determine the minimum cooling time of 17.5 years for the Zone 2 fuel assemblies. This ensures that the maximum total dose rate on the side of the OS197L TC containing a 32PT DSC is below the limit of 10 rem/hr.

The fuel assembly with a burnup of 29 GWD/MTU, an enrichment of 5.0 wt. % U-235 and a cooling time of 17.5 years results in source terms with the largest PGR source intensity and the largest fraction of the PGR source intensity in the 1.66 to 2.50 MeV energy range. The fuel assembly with a burnup of 29 GWD/MTU, an enrichment of 1.1 wt. % U-235 and a cooling time of 19.3 years results in source terms with the largest fraction of the PGR source intensity in the 1.00 to 1.66 MeV energy range. Even though the total intensity of the PGR source intensity of this BECT combination is 21% lower than that of the previous combination, the intensity in the 1.00 to 1.66 MeV energy range is 35% higher. Therefore, this combination can be considered as resulting in the bounding PGR source for FAs in the peripheral fuel compartments. To provide more assurance that this combination provides the bounding PGR source, the spectrum in the in-core region is artificially hardened by employing the larger fraction of the total source intensity in the 1.66 to 2.5 MeV energy range from the 29 GWD/MTU, 5.0 wt.% and 17.5 years combination. When used in MCNP calculations, this hardening decreases the frequency of radiological source spectrum sampling in the most important energy range, 1.00 to 1.66 MeV, by 0.002% while doubling the sampling frequency in the 1.66 to 2.5 MeV energy range. Therefore, the "hardened" spectrum for the in-core region, the region which essentially dominates dose rates at various radial distances from side of the cask, is bounding.

The combined intensity of the PGR "hybrid" source from all the axial regions is 1.724e+15 gammas per second per assembly. As mentioned in Section W.5.2, bullet item 7, the bounding neutron source is due to a BECT combination at the lowest enrichment and the highest burnup when considering a set of the BECTs represented in the decay heat FQT. This source is due to a fuel assembly with a burnup of 45 GWD/MTU, an enrichment of 1.1 wt. % U-235 and a cooling time of 47.3 years with an intensity of 3.34e+8 neutrons per second. However, it is slightly, ~1%, lower than the intensity (3.37e+8 neutrons per second) of the fuel assembly at a burnup of 44 GWD/MTU, an enrichment of 1.1 wt. % U-235 and a cooling time of 45.1 years. This difference is within the uncertainty of the SAS2H\ORIGEN-S sequence of the SCALE [5.1] package for calculation of radiological source terms, however the higher source term is employed as bounding.

The source terms for the DB fuel assembly for the 32PT DSC with an initial loading of 0.475 MTU are summarized in Table W.5-1. As stated in Section W.5.2, bullet item 4, the dose rate near the OS197L TC is largely dominated by the PGR source and the dose rate due to that source is mostly contributed to by the 1.00–1.66 MeV energy group. The intensity of the PGR source at 29 GWD/MTU, 1.1 wt. % and 19.3 years combination BECT in this energy group is 4.572E+13 gammas per second per FA. To assure that sources in Table W.5-1 are bounding for assemblies designated as Zone 2 in Figure W.2-1, cooling times for 0.40 kW/FA decay heat FQT presented in Table W.2-8 are adjusted such that the intensity of the PGR source terms in the 1.00–1.66 MeV energy range for all the BECT combinations does not exceed 4.572E+13 gammas per second per FA. Such an adjustment in the cooling times affects combinations *with burnups* less than or equal to 28 GWD/MTU. Adjusted cooling times are shown in Table W.2-8.

In summary, the bounding PGR source for the 20 *Zone 2 locations within* the 32PT DSC is based on *a* 32PT DB FAs with 0.475 *MTU* at *a burnup of* 29 GWD/MTU, *an enrichment of* 1.1 wt. % *U-235, with a cooling time of* 19.3 years *generating* 0.398 kW/FA of decay heat. However, the spectrum in the in-core region is artificially hardened by assigning a larger fraction to the intensity in the 1.66 to 2.5 MeV energy *range (corresponding to a fuel assembly with a burnup of* 29 GWD/MTU, *an enrichment of* 5.0 wt. % *U-235, and a cooling time of* 17.5 years). *The SAS2H input file for the in-core region at a burnup of 29 GWD/MTU and an enrichment of 1.1 wt. % U-235 is included in Section W.5.6.1.*

W.5.2.3 Bounding Radiological Sources for FAs in Fuel Compartments of 61BT DSC

As mentioned in Section W.5.2.1, the dose rates on the side of the bare OS197L TC *do* not exceed *the* limit *of 10 rem/hr* when the *central* fuel compartments (designated as Zone 1 *in* Figure W.2-1) contain 61BT DSC *DB* sources if the *peripheral* fuel compartments *(designated as* Zone 2 *in Figure W.2-1)* contain radiological sources from assemblies generating *less* than 0.17 kW/FA.

The DB gamma radiation source terms *are* based on 27 GWD/MTU burnup, an initial enrichment of 2.00 wt. % and *a* 5 years cooled 61BT *DB* BWR assembly. The source *terms are shown* in Table K.5-7 of the UFSAR and replicated in Table W.5-2. The *DB* neutron source term strength is 1.427e+8 neutrons per second per assembly. It is due to 35 GWd/MTU burnup, an initial enrichment of 2.65 wt. % and *an* 8 years cooled DB *fuel* assembly. *The* MCNP built-in Cm-244 fission spectrum is utilized in the MCNP models *for* the current analysis.

Fuel qualification for 61BT FQT *for Zone 1 assemblies* is provided in *Chapter W.2, Table W.2-4 (identical to that shown in* Appendix K, Table K.2-11). It was determined through multiple iterations that radiological sources in fuel compartments designated as Zone 2 *in* Figure W.2-1 due to 40 GWD/MTU, 1.8 wt.% and 37.0 years cooled *(DB BECT)* assemblies result in dose rates on *the* side of the OS197L *TC* below 10.0 rem/hr when the *Zone 1 fuel compartments contain* 61BT *design basis* radiological source *terms*. On the other hand, the largest cooling time in *the* 61BT DSC FQT based on the decay heat restriction is 16.0 years *and is due to a fuel assembly with a burnup of* 40 GWD/MTU and *an enrichment of* 1.8 wt.% *U-235*. To assure that radiological source from *DB* BECT is bounding for the 48 Zone 2 fuel compartments of *the* 61BT DSC, cooling times in Table *W.2-4* are adjusted. The adjustment is implemented in such a manner that *the* intensity of the *PGR* source in *the* 1.0-1.66 MeV energy *range* is bounded by the corresponding intensity at *DB* BECT, 3.118E+12 *gamma*/(sec*FA). Adjusted cooling times are *shown* in *Chapter* W.2, Table W.2-5.

The bounding radiological source *terms* for *the Zone 2 locations* of *the* 61BT DSC used in the shielding evaluation are shown in Table W.5-2. Note that *the* neutron radiation source corresponds to 35.0 years of cooling, not 37.0 years. Since the dose rate is dominated by the *PGR* source, this adds *additional* conservatism *in* the evaluated dose rates.

The SAS2H input file for the in-core region at a burnup of 40 GWD/MTU and an enrichment of 1.8 wt. % U-235 is included in Section W.5.6.2.

W.5.2.4 Spectral Distributions of Neutron Source Terms and the Main Contributors

The major contributors to the neutron radiation source from LWR FAs as well as uncertainties in calculating the intensity of the source are identified in Section P.5.2.2, Section T.5.2.2 and Section U.5.2.2.

The fixed source spectrum in MCNP is assumed to follow a ^{244}Cm spontaneous fission spectrum for all of the calculations in this chapter. This approach to neutron dose rate calculations is identical to that employed in Appendix P, Section P.5.2 for the MCNP dose rate evaluation of the 24PTH DSC. It is based on the following relationship:

$$p(E) \sim \exp(-E/a)\sinh(bE)^{1/2}$$

The input parameters: a=0.906 MeV and b=3.848 MeV^{-1}, as given in the MCNP manual *Volume I (Appendix H)* [5.2].

W.5.2.5 Axial Peaking

The axial peaking factors for both neutron and gamma sources in PWR fuel utilized herein are directly obtained from those utilized in the 32PT DSC shielding evaluation. These factors are shown in Appendix M, Table M.5-15.

The axial peaking factors for both neutron and gamma sources in BWR fuel are provided *as a function of active fuel height* in Appendix K, Section K.5.2.3. The same peaking factors are used in the MCNP analysis presented herein. These factors are directly applied to MCNP source input for the fuel region. More details about treatment of the peaking factors in the MCNP models are provided in Section T.5.2.3.

W.5.3 Material Densities

The material masses given in Appendix M, Table M.5-6 for the fuel are used to calculate material densities for in-core, plenum, top, and bottom regions of fuel assemblies inside *the* 32PT DSC. The material densities utilized in the MCNP calculations are shown in Appendix M, Table M.5-19.

The material masses given in Appendix T, Table T.5-6 for the fuel are used to calculate material densities for in-core, plenum, top, and bottom regions of fuel assemblies inside *the* 61BT DSC. The material densities utilized in the MCNP calculations are shown in Appendix T, Table T.5-19.

Densities for miscellaneous materials like air, water, aluminum, carbon steel, stainless steel, etc. are obtained from Appendix M, Table M.5-16.

W.5.4 Shielding Evaluation

Dose rate contributions from the bottom, in core, plenum and top regions, as appropriate, from fuel assemblies in 32PT and 61BT DSCs *within* the OS197L *TC* are calculated with the MCNP Code [5.2] at various locations on and around the *various evaluated* shielding configurations.

The following shielding evaluation discussion specifically addresses the NUHOMS® 32PT and 61BT DSC in an OS197L TC using the design-basis source terms described in *Section W.5-2*.

W.5.4.1 Computer Program

MCNP [5.2] is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces. Pointwise (continuous energy) cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation are accounted for in the cross section set. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. Important standard features that make MCNP very versatile and easy to use include a powerful general source, an extensive collection of cross-section data, and an extensive collection of variance reduction techniques that can be employed to track particles through very complex deep penetration problems. MCNP was employed to take advantage of its mesh tallies capabilities in calculating dose rates distributed over the surface of the TC.

W.5.4.2 Spatial Source Distribution

The source components are:

- The neutron sources due to the active fuel region,
- The gamma source due to the active fuel region,
- The gamma source due to the plenum,
- The gamma source due to the top region, and
- The gamma source due to the bottom region.

Axial peaking is accounted for in the active fuel region by inputting an axial shape, as discussed in Section W.5.2.5.

W.5.4.3 Cross-Section Data

The cross-section data used is the continuous energy ENDF/B-VI provided with the MCNP code [5.2]. The cross-section data allows coupled neutron/gamma-ray dose rate evaluation to be made to account for secondary gamma (n,γ) radiation. All of the OS197L TC dose rate calculations account for the dose rate due to secondary gamma radiation.

W.5.4.4 Flux-to-Dose-Rate Conversion and MCNP Tallies

The flux distribution calculated by the MCNP code is converted to dose rates using flux-to-dose rate conversion factors from ANSI/ANS-6.1.1-1977 [5.4] given in Appendix P, Table P.5-19. The same flux-to-dose rate conversion factors have been employed in the 32PT and 61BT shielding analysis with the OS197 TC documented in Appendix M.5 and K.5, respectively.

Dose rates are computed at various distances from the cask in various shielding configurations described in Section W.5.1.2. Mesh tallies calculate neutron, primary and secondary gamma radiation dose rate distributions at various distances from *the* side and ends of the configurations. Cylindrical and rectangular mesh types are used. Locations of mesh nodes are defined either in cylindrical or rectangular Cartesian coordinate systems.

The rectangular mesh tallies are employed to obtain dose rate distributions at various horizontal distances from *the* side of the shielding configurations corresponding to the cask on a trailer platform. The Z-axis of the rectangular coordinate system is along the cask axis. The X-axis is on an imaginary plane through the cask axis and trunnions, perpendicular to the cask axis. The XZ plane is a horizontal plane and the Y axis runs in vertical elevation when the cask is in the transfer position. Rectangular (Z-Y) mesh tallies are used to calculate spatial distributions over 5.5" thick shielding plates on the trailer skid side and down to the ground below *the* trailer. Y=0 corresponds to *the* TC axis. *The surface of the ISFSI concrete pad or* ground level is at Y= -263.91 cm. Y=-140.08 cm corresponds to *the* bottom of the 5.5" thick shield plate along *the* TC side on the trailer and Y=-157.23 cm is related to *the* top of *the* 0.25" thick plate on top of the trailer. There are three distinct Y regions for segmentation: (1) [0 -140.08] cm- along *the* 5.5" thick plate on *the* side of the trailer skid; (2) [-140.08, -157.23] cm, just over the 6.75" wide gap between *the* bottom of the 5.5" thick plate along *the* TC side and the top of *the* 0.25" thick plate on *the* top *of* the trailer platform (see Figure W.5-5); (3) [-157.23, 263.91] cm *slightly* over *the* 42" wide clearance between the top of the trailer platform and the *surface of the ISFSI concrete pad*.

Because the dose rate around the cask is the highest along the cask side, the cylindrical (angular-axial) mesh tallies along the side of the cask between its ends were also employed. The cylindrical (angular-axial) mesh is used for determining the dose rate distribution along the cask side between the ends at various radial distances from the side. The Z axis of *the* cylindrical coordinate system coincides with the cask axis. The axial coordinates of the mesh nodes are measured from *the* bottom end. The axial distance between nodes of the cylindrical mesh is about 30 cm. The angular coordinate is measured in counter-clockwise direction from an imaginary plane through the cask axis and trunnions, which is the XZ plane in the rectangular coordinate system.

A set of mesh tallies with various segmentation is used. The fine segmentation is used for dose rates at close distances and the regions where radiation streaming is expected. A segment size for the fine segmentation is about 30x30 cm but even finer Y segmentation, about 8 cm, is used for the Y region just over the 6.75" wide gap between the bottom of the 5.5" thick plate along the TC side and the top of the 0.25" thick plate on top of the trailer platform. When the distance from the side increases, *the* dose rate distributions become uniform. One can use coarse or no segmentation at all. This is true for distances beyond 50.8 meters.

W.5.4.5 Methodology

The MCNP computer code was utilized to analyze shielding performance of the cask in various shielding configurations. MCNP allows for explicit 3-D modeling of any shielding and source configuration. The methodology used herein is summarized below.

1. Sources are developed for all fuel regions using the source term data described in Section W.5.2. Source regions include the active fuel region, bottom end fitting (including all materials below the active fuel region), plenum, and top end fitting (including all materials above the plenum region).
2. Suitable shielding material densities are calculated for all regions modeled.
3. The 3-D Monte Carlo code MCNP is used to calculate dose rates on and around the OS197L TC loaded with the bounding, from a shielding standpoint, fuel and DSC designs. The MCNP code is selected because of its ability to handle thick, multi-layered shields and account for streaming through the TC/DSC annulus and TC neutron shielding using 3-D geometry.
4. MCNP results are used to calculate occupational and offsite exposures (see Chapter W.10).
5. MCNP models are also generated to determine the effects of off-normal and accident scenarios, such as loss of cask neutron shield and the ~~*support skid*~~supplemental shield*ing* for the OS197L TC.
6. *The relative error in the total dose rate is calculated using the square root of the sum of the squares method as shown below:*

   $\sigma_{(n+\gamma)} = [(\sigma_n * D_n)^2 + (\sigma_\gamma * D_\gamma)^2]^{1/2}/(D_n + D_\gamma)$

   where

   σ_n and σ_γ are the MCNP calculated relative errors in the neutron and gamma dose rates, respectively, and $\sigma_{(n+\gamma)}$ is the relative error in the total dose rate

   D_n and D_γ are the MCNP calculated neutron and gamma dose rates, respectively, and $(D_n + D_\gamma)$ is the calculated total dose rate

W.5.4.6 Assumptions in MCNP Calculations

The following general assumptions are used in the analyses. Some of these assumptions are generic in nature and are similar to those employed to calculate the dose rates with systems in Appendix P, T and U.

W.5.4.6.1 Source Term Assumptions

- The primary neutron source in LWR spent fuel is the spontaneous fission of ^{244}Cm. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables, ^{244}Cm represents more than 85% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein and is assumed to follow the ^{244}Cm fission spectrum provided in Section W.5.2.4.

- Due to *large cooling times from a fuel qualification standpoint,* the BECT combinations with bounding intensity of the *PGR* source in *the* 1.00–1.66 MeV energy *range* provide for the design basis source terms.

- Surface gamma dose rates are calculated for the TC surfaces using the actual photon spectrum applicable for each case.

W.5.4.6.2 OS197L TC Dose Rate Analysis Assumptions

- The 32PT and 61BT DSC models in MCNP include features *like* the basket structure and fuel compartments of the *61BT and 32PT* DSCs *and* solid aluminum peripheral rails for *the* 32PT DSC.

- The borated neutron absorber sheets are modeled as aluminum.

- Axial peaking factors assumed as described in Section W.5.2.5.

- Fuel is homogenized within the fuel assembly perimeter, although the baskets of the DSCs are modeled explicitly.

- Axial dose rates during normal conditions of operation are discussed in Section W.5.1.3. The results with the OS197 TC are bounding to the OS197L TC because there is no change in the axial shielding *design in* the OS197L TC compared to the OS197 TC. *In addition, the axial dose rates calculated with the OS197 TC are due to* stronger radiological sources and *include* conservatism in *the* determination of OS197 TC dose rates *using the 32PT and 61BT DSCs*.

- *All normal condition operations require water to be present in the neutron shielding. All calculations under normal conditions are performed with a water-filled neutron shield. Dose rate results (if any) calculated with a dry neutron shield are therefore conservative.*

- For the design basis accident case, the cask neutron shield (water) along with the *support skid* supplemental shield*ing* is assumed to be lost.

W.5.4.7 Summary of the Calculational MCNP Models

Explicit computational models containing *a* 32PT DSC for various shielding configurations are utilized. *A c*omputational model for the bare cask containing *a* 61BT DSC is also developed. *The* MCNP model of the *61BT* DSC is taken directly from the MCNP model used for the bounding shielding evaluation in Section T.5. *The n*umbering *convention for* cells, surfaces, and material*s* was *modified* to *that* used in MCNP models for *the* 32PT DSC. Dose rates from the bare cask containing *the* 32PT DSC were also calculated. Because dose rates at *radial* distances greater than 10.0 meters are low, dose rates from these two sets can be used as a conservative representation of the radiation field around various shielding configurations described in this chapter. *The radial dose rate results for the bare cask configuration containing the 32PT DSC* *are bounding for the 61BT DSC except at locations near the top of the TC. This is inferred by comparing the dose rates shown in Tables W.5-8 and W.5-11 at axial distances greater than 500 cm. This is due to the differences in the basket design between the 61BT and the 32PT DSCs. However, for all configurations where the maximum dose rates are employed for comparison, the dose rates calculated with the 32PT remain bounding. Where appropriate, the results of shielding calculations with both the 61BT and 32PT DSCs are described.*

The following is a summary of the various MCNP models utilized to obtain the results for the various loading and transfer configurations. All the models employ quarter-symmetry except those which model the OS197L TC in the trailer prior to placement of the top trailer shield where a half-symmetry model is employed. A brief description of the various shielding configurations evaluated herein *is* provided in Figure W.5-1. The basket and source layout geometry for the quarter-symmetry MCNP models *are based on* sketches *shown in* Figure W.5-1*, Figure W.2-1* and Figure W.*2-2*. *The b*are cask model includes models of both 32PT and 61BT DSCs. Bounding radial dose rates for other shielding configurations are calculated using considerations presented at the end of Section W.5.1.1. All the dose rate results shown are radial dose rates unless explicitly specified as axial. For most configurations of loading and transfer the axial *(ends of the TC)* dose rate results from the OS197 calculations are directly applicable. The dose rate results as a function of distance for various shielding configurations of the OS197L TC are also shown in Figure W.5-*2*. The various shielding configuration*s* for which the dose rates are determined are described below.

1. OS197 TC under normal conditions with water in the neutron shield using the MCNP calculational methodology. The results for this case are shown in Table W.5-3 in the row corresponding to the OS197 TC transfer cask configuration *(Case #3-2)*. Dose rates are due to 32PT DSC design basis radiological sources. *The results for this configuration illustrate the conservatism inherent in the results documented for the 32PT DSC in the UFSAR (Case #3-1).*

2. OS197L TC without any supplemental shielding and with water in the neutron shield. The DSC is assumed to be dry. This configuration is expected during the remote handling operations when the cask is lowered into *the transfer trailer from* the decontamination area *cask* shield *after the DSC welding and sealing operations.* The summary results are *shown in* Table W.5-3 in the row corresponding to the OS197L TC *b*are *c*ask transfer cask configuration *(Case #3-3)*. Detailed results for this case are shown in Table W.5-6 *for the 32PT DSC and Table W.5-9 for the 61BT DSC.*

3. OS197L TC without any supplemental shielding and without water in the neutron shield. This configuration conservatively bounds that expected during the remote handling operations when the cask is lowered into the transfer trailer during accident conditions. *The summary results are shown in Table W.5-4 in the row corresponding to the OS197L TC bare cask transfer cask configuration (Case #4-5)*. The radial dose rate results for this case are shown in Table W.5-7 *for the 32PT DSC and Table W.5-10 for the 61BT DSC*. The axial dose rate results for this case are discussed in Section W.5.1.3 and geometric locations for axial dose rate calculations are shown in Figure W.5-3.

4. Configuration is similar to 3, above, except that both the inner and outer liners of the neutron shielding are absent. This configuration provides for the worst case accident for shielding purposes. The results for this case can be inferred from Table W.5-4, using the first note under the table.

5. OS197L TC with 2.5 inches of supplemental shielding and no water in the neutron shield during accident conditions. The maximum of neutron, gamma and total dose rate on *the* side surface of this configuration are bounded by 1466, 540 and 1543 mrem/hr, respectively. *The summary results* are presented in Table W.5-4 *in the row corresponding to the OS197L TC (with supplemental inner top trailer shielding only) transfer cask configuration (Case #4-4)*. The axial dose rates for this configuration are discussed in Section W.5.1.3. Geometric locations for axial dose rate calculations are shown in Figure W.5-3.

6. OS197L TC with *the inner top supplemental trailer shielding (*2.5 inches of shielding in the *top* and 5.5 inches of shielding in the *side of the trailer) only* and water in the neutron shield *(see the "Section B-B" view of Figure W.5-5)*. The *calculational model does not consider any shielding beneath the cask in the horizontal orientation except for the trailer platform (see notes 3 and 4 for Table W.5-3)*. This configuration is expected prior to the installation of the additional 3 inches of the outer top supplemental trailer shielding prior to the commencement of transfer operations. The MCNP model details including geometrical and material descriptions are shown in Figure W.5-*5. The summary results are shown in Table W.5-3 in the row corresponding to the OS197L TC (without the outer top supplemental trailer shielding) transfer cask configuration (Case #3-5).* The results that bound this case are shown in Table W.5-12. The bounding axial dose rate results applicable for this configuration are discussed in Section W.5.1.3.

7. OS197L TC with 5.5 inches of supplemental trailer shielding and with water in the neutron shield. This is the transfer configuration under normal conditions and bounds that during decontamination operations since credit is taken for 5.5 inches of supplemental shielding instead of 6.0 inches for the decontamination area *cask* shield. *The summary results are shown in Table W.5-3 in the row corresponding to the OS197L TC (with decontamination area cask or supplemental trailer shielding) transfer cask configuration (Case #3-4).* The results for this case are shown in Table W.5-13.

8. Configuration is similar to 7, above, except that there is no water in the neutron shield. This represents a loss of neutron shielding accident during transfer operations with the *support skid* supplemental shield*ing* present. The maximum of neutron, gamma and total dose rate on side surface of this configuration are bounded by 727, 134 and 791 mrem/hr, respectively. *The summary results* are presented in Table W.5-4 *in the row corresponding to the OS197L TC (with supplemental inner and outer trailer shielding) transfer cask configuration (Case #4-3).*

The MCNP model to determine the dose rates shown in Table W.5-13 and Table W.5-14 are based on a geometry shown in the "Section B-B" view of Figure W.5-5. Three different dose rate distributions are obtained depending on the location of the dose rate tallies as described in Section W.5.4.4.

Radial dose rates calculated at an elevation near the cask axis provide an estimate of the dose rate distribution representative of the configuration with the inner top trailer shielding installed. These dose rates are also compared to the dose rates from a simplified model (results shown in Table W.5-12) in Figure W.2-1 (curve #2 and curve #3). This comparison indicates that the simplified model is sufficient for this purpose.

Radial dose rates calculated at an elevation below the cask axis and above the bottom of the skid platform provide an estimate of the dose rate distribution representative of the configuration with the inner and outer top trailer shielding installed. These dose rates are shown in Table W.5-13.

Radial dose rates calculated at an elevation between the top of the concrete ISFSI pad and the bottom of the skid platform provide an estimate of the dose rate distribution representative of the configuration with the inner and outer top trailer shielding installed. The dose rate distribution at this elevation is not significantly dependent on the outer top trailer shielding. Therefore, the MCNP model employed for this purpose is appropriate. These dose rates are shown in Table W.5-14.

Note *that* the bounding radial dose rates for the shielding configurations with water *in the neutron shield* are also plotted on Figure W.5-2 and discussed in Section W.5.1.2. *The MCNP neutron input file for the 61BT DSC in the bare cask configuration is included in section W.5.6.3. The MCNP gamma input file for the 32PT DSC in the transfer trailer configuration is included in Section W.5.6.4.*

W.5.4.8 Normal Condition Models

Various MCNP models are developed to perform the shielding evaluation of the 32PT DSC in the OS197L TC. Normal conditions include the 32PT DSC, the OS197L TC with the water filled neutron shield and 5.5 inches of supplemental trailer shielding. Due to the capability of modeling complex geometry in 3-D, several modeling conservatisms, originally employed in the 2-D DORT calculations (Section M.5 for 32PT DSC) are eliminated. However, the conservative estimates for axial *end* dose rates are still employed as discussed in Section W.5.1.3. The resulting MCNP model is also employed to determine the dose rates on and around the OS197 TC for comparison. These results are shown in Table W.5-3 *(Case #3-1)*, see rows marked "UFSAR (Table M.5-5 and Section M.11.2.5.3)" and "OS197 TC." The computational models with the OS197L TC are described in the subsequent sections.

The neutron shield remains filled at all times and therefore any configuration involving an empty neutron shield is considered an accident condition.

W.5.4.8.1 32PT DSC in OS197L TC

Two three-dimensional MCNP models with quarter-symmetry are employed for shielding analyses of the 32PT DSC within an OS197L TC, one model for neutrons and the other for gammas. The z-axis in the MCNP models coincides with the axis of rotation of the cask and the 32PT DSC. The MCNP geometry of the DSC basket structure and the source representation is shown in Figure W.5-4. Note, the lattice cell (2 1 0) and (1 2 0) are loaded with 1.2 kW/FA assemblies and the other cells are loaded with 0.6 kW/FA assemblies when determining "OS197

TC" dose rates presented in Table W.5-3 (Case #3-2) and Table W.5-4 (Case #4-6). Those dose rates are due to 32PT DSC design basis radiological sources that are used in Appendix M, Section M.5 of the UFSAR. Select features within the cask and on its surface are neglected because they produce only localized effects and have minimal impact on operational dose rates. Examples of neglected features include the relief valves, clevises, and eyebolts. The TC trunnions are not explicitly modeled, however, a sensitivity evaluation on the effect of trunnion design is discussed in Section W.5.4.8.2. For the purpose of this evaluation, the trunnions represent areas of increased gamma shielding and are located in regions of relatively low importance for shielding.

In addition, a separate set of 3D MCNP models, similar to the calculational 2D DORT models (Appendix M, Section M.5 of the UFSAR) for the 32PT DSC in the OS197 TC, are also employed for comparison. These models provide a measure of the amount of conservatism present in the DORT models.

Design features relevant to the shielding analysis of the OS197L TC and 32PT DSC are modeled in MCNP. The cask shell is modeled with a thickness of 2.68", the neutron shield inner and outer shells are modeled with thicknesses of 0.26"and 0.19", respectively with a liquid water neutron shield thickness of 3.00". The effect of the two-piece neutron shield on the normal condition dose rates is discussed in Section W.5.4.8.3.

The supplemental trailer shielding is modeled as a half cylinder with full design thickness to determine transfer condition dose rates. The bounding results for this configuration are provided in Table W.5-13. In addition, calculations are performed to determine the dose rates as a function of distance using a transfer configuration model with only the inner top trailer shielding installed. The MCNP model description for this configuration is provided in Figure W.5-5. The bounding results for this configuration are shown in Table W.5-12.

W.5.4.8.2 Solid One-Piece Trunnion Dose Rate Evaluation

Analyses are performed to compare the effect of the solid steel trunnion design to the original trunnion design (multiple pieces) which used NS-3 neutron absorber to reduce neutron dose rates. The result of this analysis indicates that this change does result in an increase in neutron dose rate; however, since the majority of the dose rate contribution is gamma, the overall dose rate is reduced in the solid steel trunnion configuration. A comparison of the dose rates is provided in Table W.5-5.

In summary, the use of a one-piece trunnion reduces the total calculated dose rate by a factor greater than ten, thus providing a beneficial impact on occupational exposures.

W.5.4.8.3 Removable Two-Piece Neutron Shield Dose Rate Evaluation

The two-piece neutron shield provides the same level of shielding as the OS197 TC neutron shield. The water cavity thickness is unchanged. The outer shell of the OS197L TC neutron shield is slightly thicker than that used in the OS197 TC (0.25" versus 0.18"). The addition of the seam between the two halves would reduce gamma dose rate in the vicinity of the seam but would increase neutron dose rate due to less water in the vicinity. As discussed for the trunnion modification above, the cask surface dose rate is dominated by gamma. The presence of steel will result in a net reduction in the total dose rate in the vicinity of the seams.

The seam between the two halves of the neutron shield is 1.5 inches wide and is "filled" with 3 inches of steel instead of water. The calculational MCNP model did not explicitly include the seam between the two halves of the neutron shield. *Instead*, the neutron shield shell is modeled as if it was continuous. This is conservative as the region around the weld seams represents an area of dose *rate* depression due to superior gamma shielding. The justification for such a representation is provided below.

The maximum dose rate at the surface of the OS197L cask with water in the neutron shield, from Table W.5-3 *(Case #3-3)*, is approximately 9,840 mrem/*hr* (320 mrem/hr neutron and 9520 mrem/hr gamma). The maximum dose rate at the surface of the OS197L cask in the vicinity of the seams, from Table W.5-*4 (Case #4-4)*, is approximately *1550* mrem/hr (*1470* mrem/hr neutron and *540* mrem/hr gamma). This dose rate is not calculated using an explicit model of the weld seams but calculated with an equivalent (conservative) model. For this model, the OS197L cask model includes no water in the annulus and neutron shield shell and also includes *a* 2.50" thick steel shell. This is *a conservative* representati*on* of the weld seam region with no water and a thickness of 3.00" of steel. These results are conservative since the thickness of the seam is 3.00" instead of 2.50" employed in the model. These results demonstrate that there is a substantial dose rate reduction in the vicinity of the seams since the dose rate distribution on and around the cask *is* dominated (>95%) by gamma sources.

W.5.4.9 Accident Models

Accident condition models are those where the OS197L cask and its contents (32PT or 61BT DSC with design basis fuel) are modeled with loss of shielding arising out of hypothetical accident conditions. Loss of water in the neutron shielding is the most common consequence of these accidents. The accident condition MCNP models are similar to the normal condition MCNP models except that the *water in the neutron shield* is replaced with air.

The radial dose rates as a function of distance for selected distances are summarized in Table W.5-4. The bounding axial dose rates are discussed in Section W.5.1.3. These accident configurations are described below:

- The first configuration involves the OS197L cask in the supplemental trailer shielding with loss of water in the neutron shield. Dose rate*s* at certain radial distances for this configuration are shown in Table W.5-4 *(Case #4-3)* (see data related to OS197L TC (with supplemental inner & outer *trailer* shielding)).
- The second configuration involves the OS197L cask in the supplemental trailer shielding with*out the* outer top supplemental *trailer* shielding and *without* water in the neutron shield. Dose rate*s* at certain radial distances for this configuration are shown in Table W.5-4 *(Case #4-4)* (see data related to OS197L TC (with supplemental inner *trailer* shielding only)).

- The third configuration involves the OS197L bare cask with loss of supplemental trailer shielding and loss water in the neutron shielding. This configuration can only occur during the handling of a bare cask with loaded fuel and is not likely to occur during transfer operations. *Dose rates at certain radial distances for this configuration are shown in Table W.5-4 (Case #4-5) (see data related to OS197L TC (bare cask)).* Detailed radial dose rate results for this configuration are *obtained from the results* shown in Table W.5-7 *(for 32PT DSC)* and Table W.5-*10 (for 61BT DSC). The maximum dose rates (neutron, gamma, and total) from these tables are reported in Table W.5-4.*

- The fourth configuration involves the loss of inner and outer neutron steel shells (liners) in addition to the loss of supplemental *trailer* shielding and *without* water in the neutron shielding as described above. This configuration has not been evaluated in *an* accident dose rate calculation with any other transfer cask design and is evaluated herein for conservatism. The bounding radial dose rate results for this configuration can be inferred from Table W.5-4, per the first note below the table. Dose rates associated with this configuration have not been used for the evaluation discussed in Chapters W.10 or W.11.

The accident condition dose rate results are compared to those for the 32PT DSC/OS197 TC calculations documented in Appendix M, *Chapter* M.5*, Table M.5-3. Note that these dose rates are also shown in Appendix M, Chapter M.11, Table M.11-2.* These dose rates are also shown in Table W.5-4 *as Case #4-1 for the 32PT DSC (see data related to UFSAR). The accident condition dose rate results are compared to those for the 61BT DSC/OS197 TC calculations documented in Appendix K, Chapter K.5 Table K.5-2. Note that these dose rates are also shown in Appendix K, Chapter K.11, Table K.11-4. These dose rates are also shown in Table W.5-4 as Case #4-2 for the 61BT DSC (see data related to UFSAR).*

W.5.4.10 OS197L TC Models During Fuel Loading and Transfer Operations

MCNP models are developed for the various operational evolutions during fuel loading and transfer using the 32PT or 61BT DSC. For most of the operational sequences, water is always present in the DSC/TC annulus, however, the dose rates are calculated using models that, conservatively, do not credit the presence of water in the annulus. These operational sequences and their modeling are described below.

W.5.4.10.1 TC Loading and Placement in Decontamination Area

This operation involves the loading of fuel in the DSC and the movement of the loaded DSC in the OS197L TC to the decontamination area that houses the 6" supplemental decontamination area *cask* shielding. The decontamination area *cask* shielding is a two-piece shielding structure where the upper portion (shield bell) is placed after the OS197L with the loaded DSC is placed into the lower portion of the shielding. The DSC cavity, DSC/TC annulus and the TC neutron shield are filled with water. The actual lifting and transfer operations are performed using remote crane operation using a laser/optical targeting system and cameras for confirmation of the cask location without the need for personnel in the vicinity of the cask. Should a failure of the crane occur during these operations, procedures will be in place to either repair the crane using proper ALARA practices and resume remote operations, or manually position the load in a safe, shielded location. Therefore, the dose received by operations personnel resulting from this high dose operation will be minimal as these operations are short duration and are performed remotely with no personnel in the vicinity. The applicable bounding dose rate distributions in *the* radial direction for estimating the dose rates for ALARA planning of repair and recovery operations during malfunctions are shown in Tables W.5-7 and W.5-8 *for the 32PT DSC and Tables W.5-10 and W.5-11 for the 61BT DSC*. One can also conservatively apply those dose rates to axial locations *at* radial distances beyond *the* perimeter (as R=TC Radius on Figure W.5-3). Bounding axial dose rates at R< TC radius are discussed in Section W.5.1.3.

W.5.4.10.2 Cask Decontamination

The DSC and the OS197L TC are placed inside the decontamination area shield. The top shield plug is assumed to be in place and the DSC/TC annulus and the neutron shielding are filled with water. This is identical to the decontamination operation documented in Appendix M, *Chapter* M.5 for the 32PT DSC/OS197 TC in the axial direction. *This is also identical to the decontamination operation documented in Appendix K, Chapter K.5 for the 61BT DSC/OS197 TC in the axial direction.* Therefore, the axial dose rate results from Appendix M, Figure M.5-26 *or Appendix K, Figure K.5-13* can be conservatively applied. *The top end dose rate for the 32PT DSC with the OS197 TC is slightly less than 7000 mrem/hr at the DSC axis and is less than 10,500 mrem/hr at the DSC periphery and DSC/TC annulus. The top end dose rate for the 61BT DSC with the OS197 TC is slightly less than 7000 mrem/hr at the DSC axis and DSC periphery and is less than 11,000 mrem/hr at the DSC/TC annulus.*

For the OS197L TC, the qualification of fuel to be loaded within the 32PT or the 61BT DSC is limited to fuel that will result in significantly lower dose rates than calculated with the OS197 TC. Further, the results in Table W.5-8 and Table W.5-11 show that the maximum radial dose rates near the axial ends (approximately 500 cm from the cask bottom) are less than 3000 mrem/hr and are clearly bounded by the 10,000 mrem/hr dose rate results from the Appendix K and Appendix M calculations. Therefore, the axial dose rates calculated with the OS197 TC remain bounding for the OS197L TC.

Cask decontamination operations are described in SAR Section W.8.1.3.

The only additional step that is not evaluated is the operation involving the "inspection" of the upper and lower openings of the decontamination area *cask* shielding for blockage. The maximum radial surface dose rates for normal conditions at the axial locations of these openings are conservatively utilized to determine the dose rates for this operation. The radial dose rates as a function of axial height for *the* bare OS197L cask (with and without water in the neutron shield) *are* shown in Table W.5-8 *for 32PT DSC and Table W.5-11 for 61BT DSC* and *are* utilized to determine the dose rates at the upper and lower openings of the decontamination area shield.

The decontamination area cask shielding completely covers the DSC cavity. The upper and lower openings of the decontamination area cask shielding are at axial locations of approximately 10 inches from the top and bottom of the TC. A separate MCNP model is not developed to determine the actual dose rate distribution for the purpose of decontamination, in particular, the dose rate distribution near the openings of the decontamination area cask shielding. The dose rates at these locations initially increase with radial distance (due to contribution from the middle regions). Therefore, the 1m dose rate results shown in Table W.5-8 and W.5-11 are employed to determine the maximum dose rates near these openings. From Table W.5-11, the maximum dose rate is approximately 3000 mrem/hr. Note that this is based on *a very conservative estimate of a bare cask configuration. With the decontamination area cask shielding in place, the maximum dose rate from Table W.5-3 (Case #3-4) is 61 mrem/hr. To calculate an average dose rate for the purpose of occupational exposure, a simplified calculation is performed. A scaling factor to determine the effect of the presence of the decontamination area cask shielding on the dose rates in the vicinity of the openings is estimated. Considering the fact that the maximum dose rate reduces by approximately a factor of 150 due to the presence of the decontamination area cask shielding, a scaling factor of 4.5 (volumetric scaling) is considered conservative. Therefore, it is estimated that the average dose rate is less than 700 mrem/hr.*

W.5.4.10.3 Welding Operations

The 32PT DSC and the OS197L TC are still inside the decontamination *area cask* shield*ing* area. The 32PT DSC top shield plug and inner top cover plate are assumed to be in place for inner top cover welding operation. The DSC cavity is *assumed to be* dry *(for modeling simplicity)* and the DSC/TC annulus and the neutron shielding are filled with water. Temporary shielding consisting of three inches of NS-3 and one inch of steel is assumed to cover the 32PT DSC inner top cover plate. In addition, the DSC outer top cover plate is not present. This is identical to the inner top cover welding operation documented in Appendix M, *Chapter* M.5 for the 32PT DSC/ OS197 TC in the axial direction. *This is also identical to the decontamination operation documented in Appendix K, Chapter K.5 for the 61BT DSC/OS197 TC in the axial direction.* Therefore, the axial dose rate results from Appendix M, Figure M.5-27 *or Appendix K, Figure K.5-14* can be conservatively applied. *The maximum top end dose rate at the DSC/TC annulus for both the 32PT DSC and the 61BT DSC is approximately 5000 mrem/hr. Since these dose rates are also calculated using design basis source terms, they can be utilized conservatively for the OS197L TC.*

W.5.4.10.4 TC Placement in the Transfer Trailer

Following welding and sealing of the DSC, the TC is lifted and placed horizontally on the transfer trailer containing the supplemental trailer shielding. The supplemental trailer shielding consist*s* of a 5.5" *side* shielding attached to the trailer where the TC is lowered. Subsequently, the 2.5" "inner" top *trailer* shielding encloses the TC and prepares the system for transfer. An MCNP model, employing half-symmetry, representing this configuration is utilized to determine the dose rates as a function of distance. This model also considers the absence of shielding *beneath the cask except for the 0.25" thick plate representing the trailer platform as* shown *in Figure W.5-5. The vents in the trailer shielding are not modeled explicitly; however, the details in the MCNP model are sufficient to obtain a conservative calculation of the radial dose rates.* The MCNP model details including geometrical and material descriptions are shown in Figure W.5-*5*. *Additional description is provided in Section W.5.4.7.* The bounding radial dose rate results for this configuration are shown in Table W.5-12 and also plotted on Figure W.5-2 (*described as curve 3 in Section W.5.1.2*). The dose rates below the cask support skid are expected to be maximized, in particular at closer distances, because of the absence of shielding between the trailer deck and trailer shield. The results demonstrate that this effect diminishes with distance and at distances greater than 2m the dose rate peaking is minimized. The maximum radial neutron, gamma and total dose rate for this configuration below the cask support skid are bounded by *1466*, *540*, and *1543* mrem/hr, respectively. The bounding dose rates *for* this configuration when *the* neutron shield is filled with water are plotted on Figure W.5-2 (*described as curve 5 in Section W.5.1.2*). The maximum dose rate at 100 m prior to the installation of the outer *top* trailer shielding is about 0.1 mrem/hr, thereby ensuring that off-site doses are not significantly affected during the placement of the *outer* top trailer shielding.

It is expected that the actual lifting and transfer operations are performed using remote crane operation using cameras for confirmation of the cask location without the need for personnel in the vicinity of the cask. Should a failure of the crane occur during these operations, procedures will be in place to either repair the crane using proper ALARA practices and resume remote operations, or manually position the load in a safe, shielded, location. Therefore, the dose received by operations personnel resulting from this high dose operation will be minimal as these operations are short duration and are performed remotely with no personnel in the vicinity. The applicable dose rate distributions for estimating the dose rates for ALARA planning of repair and recovery operations during malfunctions are shown in Table W.5-*6* and Table W.5-8 in the radial direction *for the 32PT DSC and in Table W.5-9 and Table W.5-11 in the radial direction for the 61BT DSC*. Geometric locations for axial dose rate calculations are shown in Figure W.5-3. The dose rate distribution for accidental configurations during these operations bound those during decontamination. The results of these calculations are shown in Table W.5-14. As discussed previously, the maximum radial neutron, gamma and total dose rate for this configuration with empty neutron shield below the cask support skid are bounded by *1466*, *540* and *1543* mrem/hr, respectively.

W.5.4.10.5 Cask Transfer to ISFSI Operations

These operations are performed outside the fuel building when the DSC is actually transferred to the HSM. For this purpose, the neutron shielding is filled with water and the cask lid is in place. The additional 3" of "outer" top shielding is also in place. The loss of neutron shielding accident or the loss of "outer" shielding accidents are bounded by the dose rates calculated for the "pre-transfer" operations described in Section W.5.4.10.4. The dose rates are calculated assuming that the OS197L cask is completely enclosed by supplemental trailer shielding. The results for this configuration *with* water in the neutron shield are shown in Table W.5-13. Dose rates at certain radial distances from the same shielding configuration but without water in the neutron shield are provided in Table W.5-*4 (Case #4-3)*, see data relevant to *OS197L TC (with Supplemental Inner & Outer Trailer Shielding)*. Applicable bounding axial dose rates are discussed in Section W.5.1.3. Dose rates shown in Table W.5-*6 and* Table W.5-*9 can be conservatively applied to determine the axial dose rates for this configuration with water in the neutron shield at radial distances beyond the edge of the transfer trailer. Dose rates shown in* Table W.5-7 *and* Table W.5-10 can be conservatively applied *to determine* the axial dose rates for this configuration without water in the neutron shield *at radial distances beyond the edge of the transfer trailer*.

The dose rates at the axial ends of the OS197L TC within the radius of the TC are bounded by those for the transfer operation documented in Appendix M, Chapter M.5 for the 32PT DSC/OS197 TC. This is also identical to the transfer configuration documented in Appendix K, Chapter K.5 for the 61BT DSC/OS197 TC in the axial direction. Therefore, the axial dose rate results from Appendix M, Table M.5-5 or Appendix K, Table K.5-4 can be conservatively applied. The maximum top end dose rate within the radius of the TC for the 32PT DSC is 107 mrem/hr and that for the 61BT DSC is 132 mrem/hr. The maximum bottom end dose rate within the radius of the TC for the 32PT DSC is 1710 mrem/hr and that for the 61BT DSC is 2540 mrem/hr. Since these dose rates are also calculated using design basis source terms, they can be utilized conservatively for the OS197L TC.

W.5.5 References

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

5.2 A General Monte Carlo N-Particle Transport Code, Version 5, *Volume I: Overview and Theory, LA-UR-03-1987 and* Volume II: User's Guide, LA-CP-03-0245, *2005*.

5.3 CASK-81 - 22 Neutron, 18 Gamma-Ray Group, P3, Cross Sections for Shipping Cask Analysis," DLC-23, Oak Ridge National Laboratory, RSIC Data Library Collection, June 1987.

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

W.5.6 List of Input Files

The list of input files begins on the next page.

Proprietary information on pages W.5-30 through W.5-76 withheld pursuant to 10 CFR 2.390

Table W.5-1
Bounding Radiological Source for *Fuel* Assemblies in *the* 32PT DSC *within the* OS197L *TC*

PGR Spectrum			Normalized Spectrum and Total Strengths per Assembly for *PGR* in Each Axial Region. Total Neutron Source Strength per Assembly.							
			12 Zone 1 Fuel Compartments				20 Zone 2 Fuel Compartments			
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**	**Bottom Nozzle**	**In-core**	**Plenum**	**Top Nozzle**
0.00	to	0.05	2.072e-2	2.880e-1	3.299e-2	2.073e-2	2.172e-2	3.254e-1	2.435e-2	2.178e-2
0.05	to	0.10	2.601e-3	5.674e-2	2.613e-3	2.601e-3	2.608e-3	6.637e-2	2.610e-3	2.608e-3
0.10	to	0.20	6.275e-4	4.338e-2	1.615e-3	6.277e-4	6.295e-4	4.203e-2	8.209e-4	6.296e-4
0.20	to	0.30	3.118e-5	1.242e-2	9.036e-5	3.120e-5	3.140e-5	1.259e-2	4.302e-5	3.142e-5
0.30	to	0.40	4.086e-5	8.460e-3	2.765e-4	4.088e-5	4.092e-5	8.216e-3	8.637e-5	4.092e-5
0.40	to	0.60	2.584e-6	8.636e-2	5.267e-3	2.584e-6	2.584e-6	8.855e-3	1.018e-3	2.584e-6
0.60	to	0.80	7.701e-5	4.027e-1	2.882e-3	8.684e-5	4.293e-4	5.043e-1	1.414e-3	4.845e-4
0.80	to	1.00	2.502e-3	3.746e-2	4.090e-4	2.239e-3	4.472e-4	5.700e-3	8.895e-4	5.004e-4
1.00	to	1.33	7.590e-1	5.019e-2	7.438e-1	7.592e-1	7.596e-1	2.190e-2	7.555e-1	7.594e-1
1.33	to	1.66	2.144e-1	1.367e-2	2.100e-1	2.144e-1	2.145e-1	4.628e-3	2.133e-1	2.145e-1
1.66	to	2.00	1.997e-12	1.228e-4	6.120e-12	1.988e-12	1.392E-15	2.825e-5	2.258E-11	1.572E-15
2.00	to	2.50	5.087e-6	4.640e-4	4.985e-6	5.088e-6	5.091e-6	1.519e-6	5.063e-6	5.090e-6
2.50	to	3.00	7.887e-9	9.451e-6	7.729e-9	7.890e-9	7.894e-9	6.128e-8	7.850e-9	7.892e-9
3.00	to	4.00	9.082e-24	1.192e-6	1.848e-23	1.020e-23	5.792e-24	1.381e-8	1.217e-23	6.279e-24
4.00	to	5.00	0.0	2.393e-9	0.0	0.0	0.0	4.529e-9	0.0	0.0
5.00	to	6.50	0.0	9.604e-10	0.0	0.0	0.0	1.817e-9	0.0	0.0
6.50	to	8.00	0.0	1.884e-10	0.0	0.0	0.0	3.565e-10	0.0	0.0
8.00	to	10.0	0.0	4.000e-11	0.0	0.0	0.0	7.569e-11	0.0	0.0
Total Gamma, g/(sec*FA)			**1.2862e+13**	**3.3097e+15**	**5.0346e+12**	**7.9104e+12**	**2.9087e+12**	**1.7183e+15**	**1.1213e+12**	**1.7895e+12**
Total Neutrons, n/(sec*FA)			**6.86e+8**				**3.37e+8**			

Table W.5-2
Bounding Radiological Source for Fuel Assemblies in the 61BT DSC within the OS197L TC

Spectrum and Total Strengths per Assembly for PGR in Each Axial Region. Total Neutron Source Strength per Assembly.											
48 Peripheral Fuel Compartments							(1)13 Inner Fuel Compartments				
PGR Spectrum, Total Source Strength per Assembly											
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	In-core	Plenum	Top Nozzle	PGR Spectrum, E_{ave}, MeV	Bottom Nozzle	In-core	Plenum	Top Nozzle
0.00	to	0.05	2.8811e+9	2.1727e+14	8.1557e+8	2.0220e+9	0.01	1.113e+11	4.864e+14	3.892e+10	8.060e+10
0.05	to	0.10	2.6683e+8	4.6909e+13	7.9633e+7	1.9827e+8	0.025	3.023e+10	1.220e+14	6.659e+10	2.145e+10
0.10	to	0.20	6.4317e+7	2.6703e+13	2.1269e+7	4.8066e+7	0.0375	1.342e+10	1.270e+14	1.825e+10	9.680e+9
0.20	to	0.30	3.1990e+6	8.3024e+12	1.0856e+6	2.4071e+6	0.0575	1.190e+10	9.809e+13	3.894e+9	8.759e+9
0.30	to	0.40	4.2356e+6	5.6236e+12	1.7628e+6	3.1695e+6	0.085	4.683e+ 9	6.835e+13	1.556e+9	3.447e+9
0.40	to	0.60	1.7739e+6	3.9877e+12	1.1541e+7	1.6655e+6	0.125	1.866e+9	6.506e+13	9.486e+8	1.369e+9
0.60	to	0.80	1.0805e+7	3.2911e+14	5.9619e+6	5.0905e+7	0.225	1.614e+9	5.372e+13	5.482e+9	1.118e+9
0.80	to	1.00	1.3025e+7	1.6177e+12	1.0281e+6	5.0851e+7	0.375	6.267e+9	3.555e+13	3.160e+10	4.193e+9
1.00	to	1.33	7.7394e+10	2.6000e+12	2.3057e+10	5.7562e+10	0.575	7.853e+9	8.468e+14	4.055e+10	5.240e+9
1.33	to	1.66	2.1856e+10	3.1550e+11	6.5113e+9	1.6256e+10	0.85	2.222e+10	1.923e+14	6.666e+9	6.970e+9
1.66	to	2.00	2.2526e+0	1.4279e+10	1.8180e+1	2.3428e+0	1.25	4.000e+12	6.670e+13	1.152e+12	2.947e+12
2.00	to	2.50	5.1867e+5	7.4338e+8	1.5452e+5	3.8577e+5	1.75	2.511e+2	1.395e+12	8.200e+1	1.896e+2
2.50	to	3.00	8.0426e+2	6.0989e+7	2.3960e+2	5.9817e+2	2.25	2.120e+7	6.823e+11	6.103e+6	1.562e+7
3.00	to	4.00	2.2255e-12	9.3055e+6	9.4410e-16	1.4295e-11	2.75	6.560e+4	2.634e+10	1.889e+4	4.833e+4
4.00	to	5.00	0.0	3.1409e+6	0.0	0.0	3.5	9.733e-15	3.386e+9	1.944e-18	2.559e-14
5.00	to	6.50	0.0	1.2604e+6	0.0	0.0	5.0	0.0	4.127e+6	0.0	0.0
6.50	to	8.00	0.0	2.4723e+5	0.0	0.0	7.0	0.0	4.759e+5	0.0	0.0
8.00	to	10.0	0.0	5.2487e+4	0.0	0.0	9.5	0.0	5.468e+4	0.0	0.0
Total Gamma, g/(sec*FA)			**1.0250e+11**	**6.4245e+14**	**3.0506e+10**	**7.6196e+10**	**Total PGR**	**4.211e+12**	**2.160e+15**	**1.366e+12**	**3.090e+12**
Total Neutrons, n/(sec*FA)			9.86e+7				**Total Neutrons**	**1.427e+8**			

(1) This is a 61BT design basis radiological source from Table K.5-7 from the UFSAR. It is replicated here for convenience.

Table W.5-3
Summary of OS197L TC Normal Condition Bounding Dose Rates

Case #	Transfer Cask Configuration	Dose Rate(5) Component	Dose Rates (mrem/hr) at Different Distances from Side Surface Normal Condition–Water in Neutron Shield			
			On Side Surface	4.57 meters (15')	100 meters	609.9 meters (2000')
3-1	UFSAR (Table M.5-5 and Section M.11.2.5.3)	Neutron	261	Not calc.	Not calc.	Not calc.
		Gamma	784	Not calc.	Not calc.	Not calc.
		Total	950	Not calc.	Not calc.	0.01
3-2	OS197 TC(1) *Results are directly shown in this table.*	Neutron	102	7.20	0.006	7.09e-6
		Gamma	248	20.3	0.03	5.29e-5
		Total	346	25.9	0.03	5.67e-5
3-3	OS197L TC bare cask *(Maximum from Table W.5-6 and Table W.5-9)*	Neutron	323	22..4	0.022	1.51e-5
		Gamma	9,521	824	1.41	1.45e-3
		Total	9,835	845	1.42	1.46e-3
3-4	OS197L TC with decontamination area cask or supplemental trailer shielding(2,4) *Table W.5-13*	Neutron	27.9	4.00	0.02	1.32e-5
		Gamma	39.5	25.0	0.08	1.83e-4
		Total	60.6	29.0	0.10	1.96e-4
3-5	OS197L TC without the outer top supplemental trailer shielding(3,4) *Table W.5-12*	Neutron	58.6	4.30	0.02	8.20e-5
		Gamma	336	36.7	0.17	6.32e-4
		Total	394	40.8	0.20	7.14e-4

(1) Dose rates are due to 32PT DSC design basis radiological sources. *These are calculated to compare against those shown for the OS197 TC in the UFSAR, shown in this table as Case #3-1.*

(2) The dose rates are also applicable to the cask on trailer at vertical elevations above the trailer support skid. These are dose rates a person could potentially be exposed when doing manual operations at altitudes above the trunnions level, for example, traversing the crane bridge above the OS197L. Use data in *Table W.5-14* for dose rates below the trailer support skid.

(3) The dose rates are applicable for radial locations over the 2.5" thick inner top supplemental trailer shielding *prior to the installation of the outer top trailer shielding. These dose rates do not reflect those at radial distances from the side of the trailer.* Use data in *Table W.5-14* for dose rates below the trailer support skid.

(4) Table W.5-14 dose rates below the trailer support skid do not account for shielding from the trailer gear boxes, wheels assembly that may provide substantial shielding at certain locations near the trailer platform. Therefore, they represent a conservative estimate.

(5) Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

Table W.5-4
Summary of OS197L TC Accident Condition Bounding Dose Rates

Case #	Transfer Cask Configuration	Dose Rate[5] Component	Dose Rates (mrem/hr) at Different Distances from Side Surface Accident Condition–No Water in Neutron Shield: On Side Surface	4.57 meters (15')	100 meters	609.9 meters (2000')
4-1	UFSAR (*Table M.5-3*, Table M.11-2)	Neutron	3,780	Not calc.	Not calc.	Not calc.
		Gamma	1,070	Not calc.	Not calc.	Not calc.
		Total	4,640	Not calc.	Not calc.	< 0.02
4-2	*UFSAR (Table K.5-2, Table K.11-4)*	Neutron	*3,700*	*Not calc.*	*Not calc.*	*Not calc.*
		Gamma	*4,820*	*Not calc.*	*Not calc.*	*Not calc.*
		Total	*8,520*	*Not calc.*	*Not calc.*	*< 0.02*
4-3	OS197L TC[1, 3] (with supplemental inner & outer trailer shielding) *Results are directly shown in this table.*	Neutron	727	79.0	0.32	2.31e-4
		Gamma	134	40.0	0.14	3.04e-4
		Total	791	104	0.46	5.34e-4
4-4	OS197L TC[1, 4] (with supplemental inner trailer shielding only) *Results are directly shown in this table.*	Neutron	1466	87.0	0.48	7.36e-4
		Gamma	540	59.0	0.29	1.07e-3
		Total	1543	129	0.77	1.81e-3
4-5	OS197L TC[1] (bare cask) *(Maximum from Table W.5-7 and Table W.5-10)*	Neutron	4,194	175	0.20	4.97e-5
		Gamma	15,305	1,332	2.30	2.43e-3
		Total	18,210	1,438	2.48	2.46e-3
4-6	OS197 TC[2] *Results are directly shown in this table.*	Neutron	1,282	66.0	0.07	1.87e-5
		Gamma	291	30.0	0.04	5.14e-5
		Total	1573	84.0	0.10	6.48e-5

(1) 0.19" thick neutron shield shell(s) are credited in the calculations. To obtain a rough and conservative estimates for dose rates without the neutron shield shells one can scale the dose rates by the factor of exp(ln(2)*0.19/0.85)=1.17, where 0.85" is a half layer thickness of steel for Co-60 radiation.

(2) Dose rates are due to 32PT DSC design basis radiological sources. Those sources would result in nearly 87 rem/hr maximum dose rate on the side of the bare OS197L cask without water in the neutron shield. *These are calculated to compare against those shown for the OS197 TC, shown in this table as Case #4-1.*

(3) The dose rates are also applicable to the cask on the trailer at vertical elevations above the trailer support skid.

(4) The dose rates are applicable for radial locations *around the* 2.5" thick inner top supplemental trailer shielding *prior to the installation of the outer top supplemental trailer shielding. These dose rates do not reflect those at radial distances from the side of the trailer.*

(5) Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

Table W.5-5
Dose Rate Results for Two Trunnion Designs (mrem/hr)

Trunnion Type	Neutron Dose Rate	Gamma Dose Rate	Total Dose Rate
Original *u*pper	0.2*0*	621	**621**
Solid *s*teel *u*pper	51.1	.14	**51.2**
Original *l*ower	1.0*0*	1702	**1703**
Solid *s*teel *l*ower	79.5	1.3*0*	**80.8**

Note that the dose rates presented are due to 32PT DSC design basis sources *and are conservative for this purpose.*

Table W.5-6
Bounding Radial Dose Rates for the Bare OS197L TC with 32PT DSC
(Normal Condition, Water in Neutron Shield)

Distance from TC Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	323	0.010	9,521	0.004	9,835	0.004
1	136	0.003	4,022	0.002	4,158	0.002
2	71.8	0.003	2,306	0.002	2,378	0.002
3	42.7	0.003	1,463	0.002	1,506	0.003
4.57 (15')	22.4	0.005	824	0.003	845	0.007
10	5.63	0.005	217	0.003	223	0.010
50.8 (2000")	0.16	0.010	7.52	0.010	7.67	0.010
100	0.02	0.020	0.98	0.010	1.01	0.010
200	2.30E-03	0.070	0.11	0.020	0.11	0.020
300	5.02E-04	0.130	0.02	0.030	0.02	0.030
609.6 (2000')	1.51E-05	0.290	6.37E-04	0.060	6.45E-04	0.060

Note: Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

Table W.5-7
Bounding Radial Dose Rates for the Bare OS197L TC with 32PT DSC
(Accident Condition, No Water in Neutron Shield)

Distance from TC Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	2,904	0.003	15,305	0.010	18,210	0.008
1	968	0.002	6,522	0.004	7,491	0.003
2	470	0.002	3,755	0.004	4,225	0.004
3	271	0.002	2,378	0.004	2,649	0.010
4.57 (15')	141	0.003	1,333	0.010	1,430	0.011
10	34.6	0.003	351	0.010	386	0.010
50.8 (2000")	0.99	0.010	12.2	0.030	13.2	0.030
100	0.14	0.010	1.56	0.020	1.70	0.020
200	1.22E-02	0.020	0.19	0.080	0.21	0.076
300	2.02E-03	0.030	0.04	0.040	0.04	0.040
609.6 (2000')	3.85E-05	0.120	9.53E-04	0.100	9.88E-04	0.100

Note: Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

Table W.5-8
Bounding Radial Dose Rates for the OS197L TC with 32PT DSC as a Function of Axial Height

Axial Distance from TC Bottom, (cm)	Bare Cask Dose Rates (mrem/hour) Normal Condition with Water in Neutron Shield		Bare Cask Dose Rates (mrem/hour) Accident Condition without Water in Neutron Shield	
	Surface	1 M from Surface	Surface	1 M from Surface
2.5	73	568	91	1,039
16.5	60	747	108	1,336
39.5	252	1,134	539	1,982
62.5	3,142	1,649	4,904	2,874
85.5	5,148	2,196	8,393	3,797
108.5	6,824	2,740	11,563	4,722
131.5	8,745	3,202	15,251	5,558
154.5	9,505	3,554	17,077	6,207
177.5	9,809	3,825	17,787	6,704
200.5	9,809	3,978	18,046	7,030
223.5	9,742	4,095	18,123	7,264
246.5	9,835	4,140	18,128	7,375
269.5	9,811	4,158	18,210	7,416
292.5	9,782	4,129	18,180	7,371
315.5	9,781	4,069	18,184	7,223
338.5	9,828	3,961	17,976	7,002
361.5	9,723	3,794	17,608	6,611
384.5	9,130	3,525	16,100	6,082
407.5	7,832	3,201	13,408	5,463
430.5	5,607	2,809	9,559	4,750
453.5	5,900	2,399	8,991	4,028
476.5	5,610	1,984	8,380	3,284
499.5	3,807	1,574	5,761	2,584
522.5	911	1,208	1,498	1,994
545.5	50	860	140	1,448
568.5	38	560	63	959
Maximum	9,835	4,158	18,210	7,416
Average	6,183	2,695	10,926	4,696

Note: Bottom of the TC is at 0 cm. Top of the TC is at 500 cm.

Table W.5-9
Bounding Radial Dose Rates for the Bare OS197L TC with 61BT DSC
(Normal Condition, Water in Neutron Shield)

Distance from TC Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	*311*	0.010	*8,817*	0.010	*9,129*	0.010
1	*116*	<0.01	*3,566*	<0.01	*3,682*	0.010
2	*56.5*	<0.01	*2,030*	<0.01	*2,083*	0.010
3	*32.6*	<0.01	*1,326*	<0.01	*1,357*	0.010
4.57 (15')	*14.3*	0.006	*621*	0.010	*641*	0.013
10	*4.16*	0.010	*222*	0.010	*226*	0.010
50.8 (2000")	*0.12*	0.020	*8.00*	0.020	*8.12*	0.020
100	*0.02*	0.040	*1.41*	0.020	*1.42*	0.020
200	*2.05E-03*	0.120	*0.18*	0.050	*0.18*	0.050
300	*3.98E-04*	0.090	*0.04*	0.080	*0.04*	0.078
609.6 (2000')	*1.50E-05*	0.280	*1.45E-03*	0.140	*1.46E-03*	0.140

Note: Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

Table W.5-10
Bounding Radial Dose Rates for the Bare OS197L TC with 61BT DSC
(Accident Condition, No Water in Neutron Shield)

Distance from TC Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	*4,194*	0.004	*14,816*	0.010	*18,141*	0.008
1	*1279*	0.003	*6,080*	0.003	*7,359*	0.003
2	*596*	0.003	*3,444*	0.003	*4,038*	0.003
3	*339*	0.003	*2,228*	0.004	*2,559*	0.010
4.57 (15')	*175*	0.005	*1,304*	0.010	*1,438*	0.010
10	*42.7*	0.005	*366*	0.010	*408*	0.010
50.8 (2000")	*1.26*	0.010	*13.4*	0.020	*14.7*	0.020
100	*0.20*	0.010	*2.30*	0.030	*2.48*	0.030
200	*1.68E-02*	0.030	*0.28*	0.050	*0.29*	0.050
300	*2.89E-03*	0.040	*0.06*	0.080	*0.07*	0.080
609.6 (2000')	*4.97E-05*	0.140	*2.43E-03*	0.190	*2.46E-03*	0.190

Note: Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

Table W.5-11
Bounding Radial Dose Rates for the OS197L TC with 61BT DSC as a Function of Axial Height

Axial Distance from TC Bottom, (cm)	Bare Cask Dose Rates (mrem/hour) Normal Condition with Water in Neutron Shield		Bare Cask Dose Rates (mrem/hour) Accident Condition without Water in Neutron Shield	
	Surface	1 M from Surface	Surface	1 M from Surface
2.5	29	379	72	835
16.5	26	489	84	1,053
39.5	68	753	260	1,561
62.5	736	1,120	1,527	2,273
85.5	2,330	1,589	4,478	3,160
108.5	5,403	2,129	9,988	4,169
131.5	7,589	2,625	14,231	5,146
154.5	8,458	3,039	16,226	5,925
177.5	8,843	3,334	17,329	6,490
200.5	9,044	3,542	17,866	6,919
223.5	9,129	3,657	18,141	7,136
246.5	9,103	3,682	18,021	7,207
269.5	8,994	3,666	17,899	7,138
292.5	8,747	3,600	17,086	6,994
315.5	8,346	3,488	16,219	6,729
338.5	8,077	3,337	15,453	6,361
361.5	7,676	3,168	14,318	5,951
384.5	6,701	2,976	12,220	5,482
407.5	5,345	2,885	9,535	5,124
430.5	3,360	2,931	5,848	4,978
453.5	3,134	3,141	6,503	4,994
476.5	3,393	3,404	4,917	5,179
499.5	3,405	3,414	4,862	5,119
522.5	3,378	3,386	4,741	4,979
545.5	2,687	2,696	3,778	3,989
568.5	1,612	1,617	2,234	2,392
Maximum	9,129	3,682	18,141	7,207
Average	5,216	2,694	9,763	4,896

Note: Bottom of the TC is at 0 cm. Top of the TC is at 500 cm.

Table W.5-12
OS197L TC Radial Dose Rates over 2.5″ Inner Top Trailer Area Shielding
(Normal Condition, Water in Neutron Shield)

Distance from Inner Top Trailer Area Shielding Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	58.6	0.010	336	0.010	394	0.010
1	22.2	0.010	156	0.004	178	0.004
2	12.0	0.010	94.4	0.004	106	0.004
3	7.60	0.010	62.9	0.005	70.2	0.005
4.57 (15')	4.30	0.010	36.7	0.010	40.8	0.010
10	1.30	0.010	11.0	0.010	12.2	0.010

Note: Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.

Table W.5-13
OS197L TC Radial Dose Rates with 5.5" Trailer Area Shielding
(Normal Condition, Water in Neutron Shield)

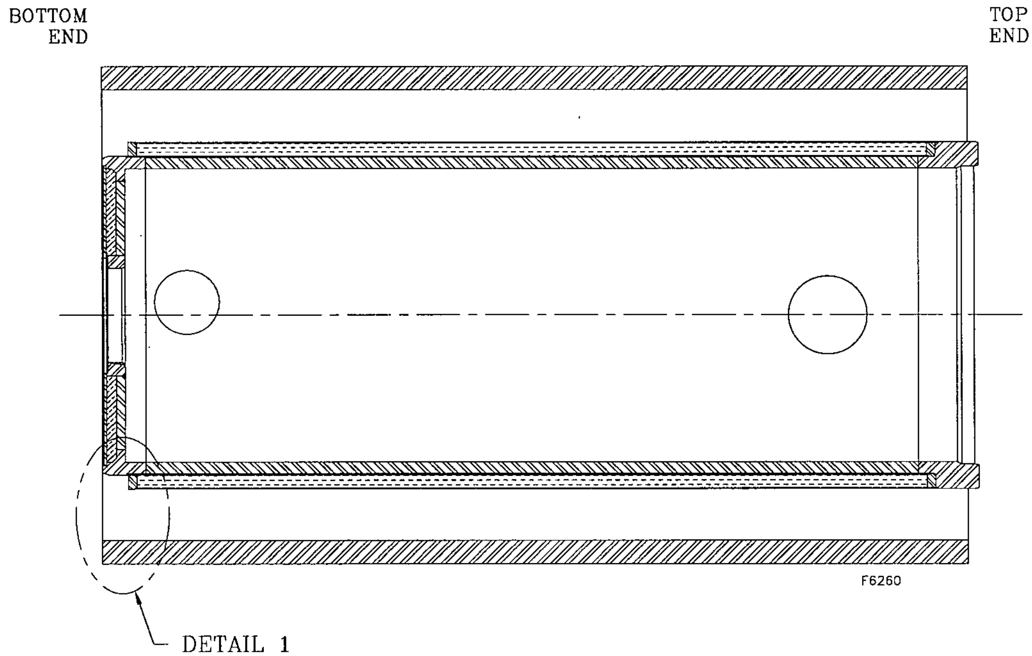
Distance from Outer Top Trailer Area Shielding Side, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	27.9	0.010	39.5	0.004	60.6	0.010
1	17.4	0.010	81.9	0.010	97.3	0.009
2	10.6	0.010	52.6	0.010	63.0	0.009
3	6.90	0.010	39.3	0.010	46.2	0.010
4.57 (15')	4.00	0.020	25.0	0.010	29.0	0.010
10	1.10	0.030	7.10	0.020	8.20	0.020

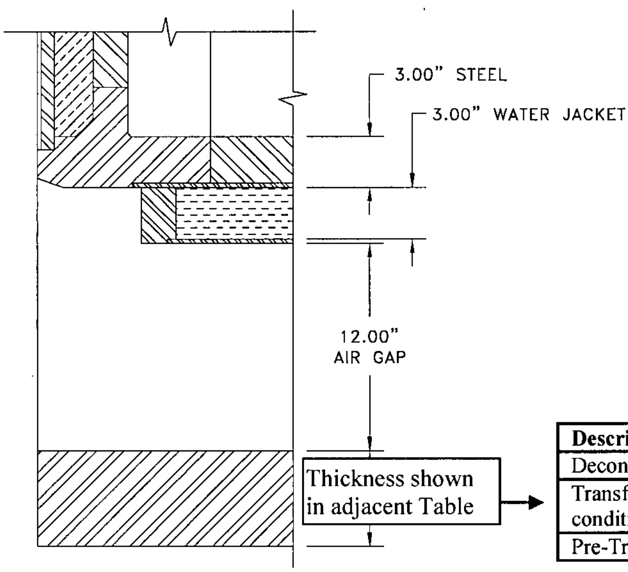
Note: Dose rates presented are bounding for OS197 L TC radial dose rates above cask support skid. There is no shielding underneath in the computation model used for determination of the dose rates presented in the tables except for only 0.25" thick steel plate on top of the trailer platform. Geometry of the model is depicted on sketches of Figure W.5-5. Contribution of scattered radiation is pronounced at short (less than 2 meters) radial distances. The presented maximum values of the dose rate are determined at vertical elevations above the cask support skid and they account for radiation scattered from concrete at grade level. *Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.*

Table W.5-14
OS197L TC Radial Dose Rates below Cask Support Skid
(Prior to Installation of Outer Top Trailer Area Shielding,
Water in Neutron Shielding)

Distance in horizontal direction from side of 5.5" thick Side Shielding Plates, m	Neutron Radiation		Gamma Radiation		Total Radiation	
	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error	Dose Rate, mrem/hr	Relative Error
0	56.8	0.010	1878	0.006	1934.3	0.01
1	20.1	0.020	475	0.010	494.0	0.01
2	8.50	0.010	79.3	0.010	87.7	0.01
3	5.50	0.010	31.2	0.010	36.3	0.01
4.57 (15')	3.20	0.020	14.2	0.010	17.3	0.01
10	1.00	0.030	4.50	0.020	5.50	0.02

Note: Gamma and neutron dose rate peaks do not always occur at the same location; therefore, the total dose rate is not always the sum of the gamma plus neutron dose rate.





Description	Steel Thickness
Decontamination	6.0", conservatively used 5.5"
Transfer Trailer (normal condition for transfer)	5.5"
Pre-Transfer	2.5" (simple model)

DETAIL 1

Figure W.5-1
OS197L TC MCNP Shielding Configuration Description

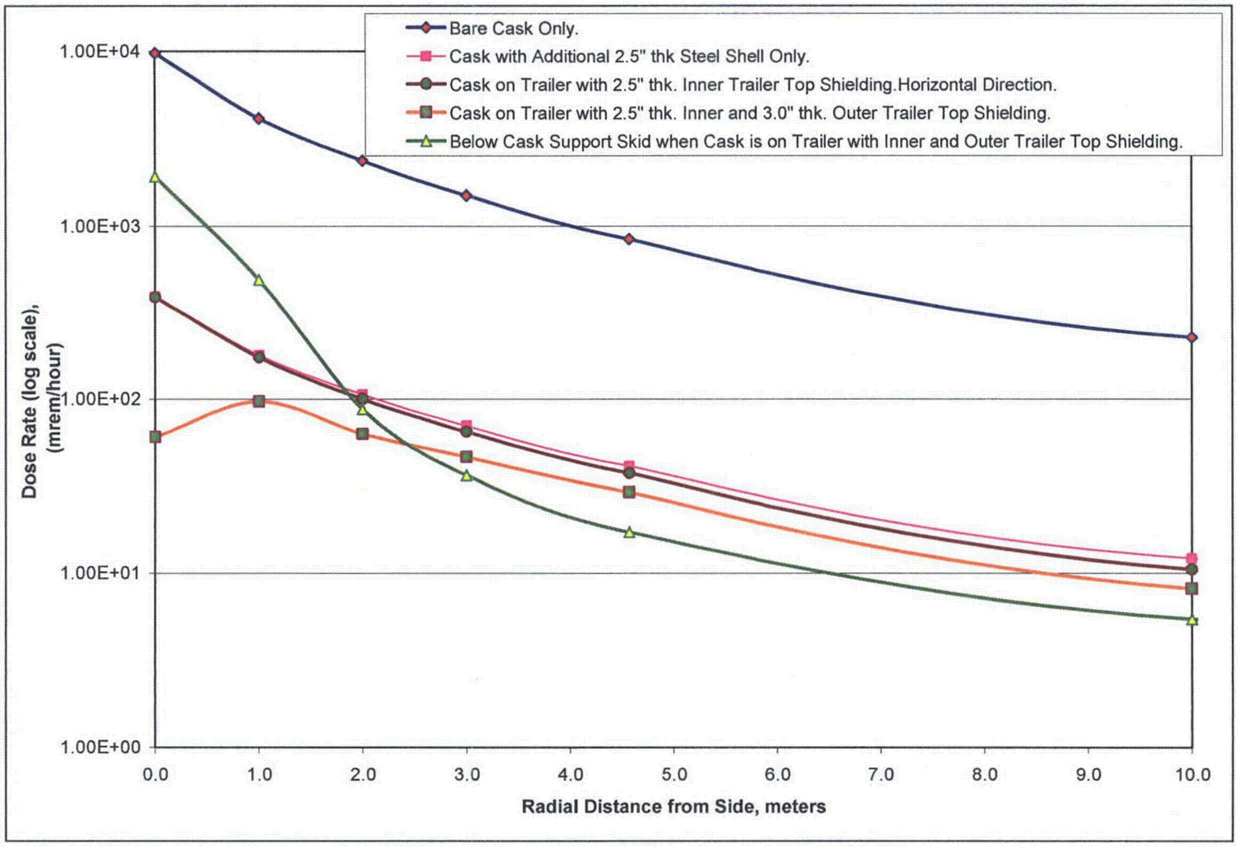


Figure W.5-2
Bounding Radial Dose Rate Results for the Various OS197L TC Shielding Configurations

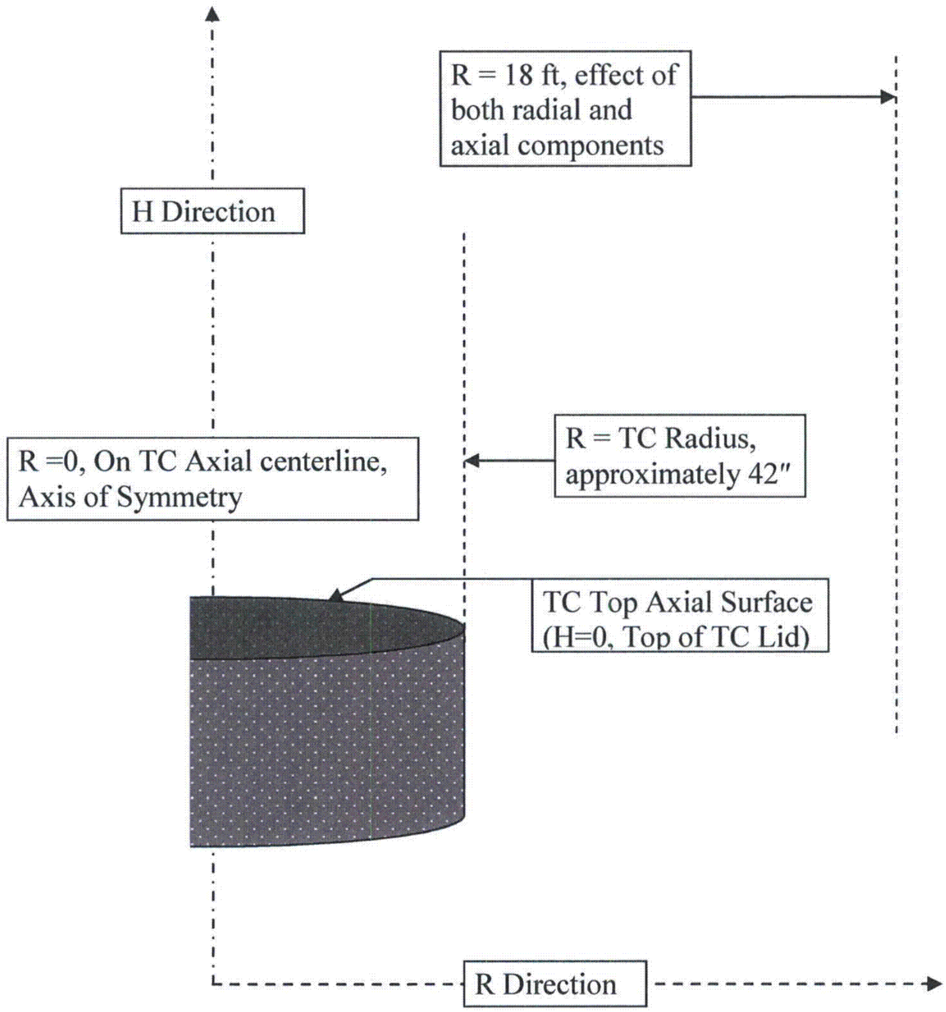


Figure W.5-3
Geometrical Layout for OS197L Axial Dose Rate Calculations

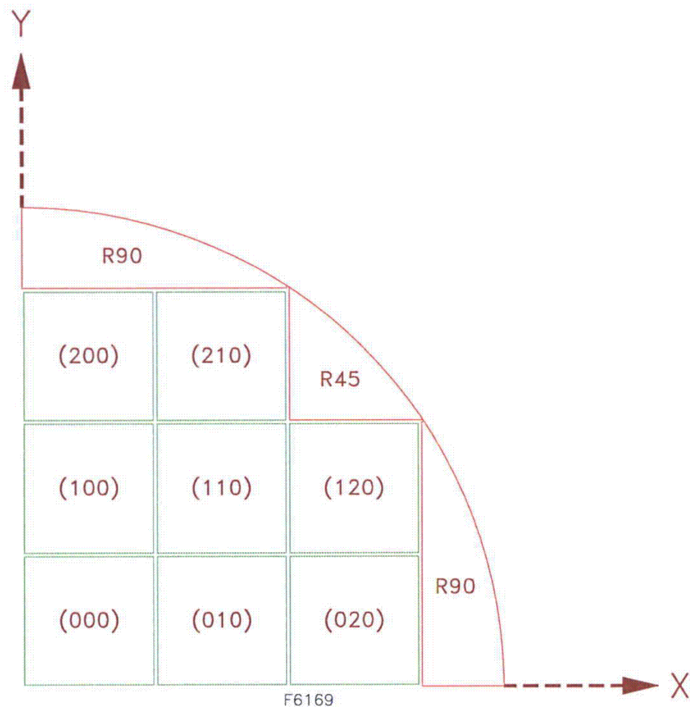


Figure W.5-4
MCNP Geometry of the 32PT DSC Basket Structure and Source Region

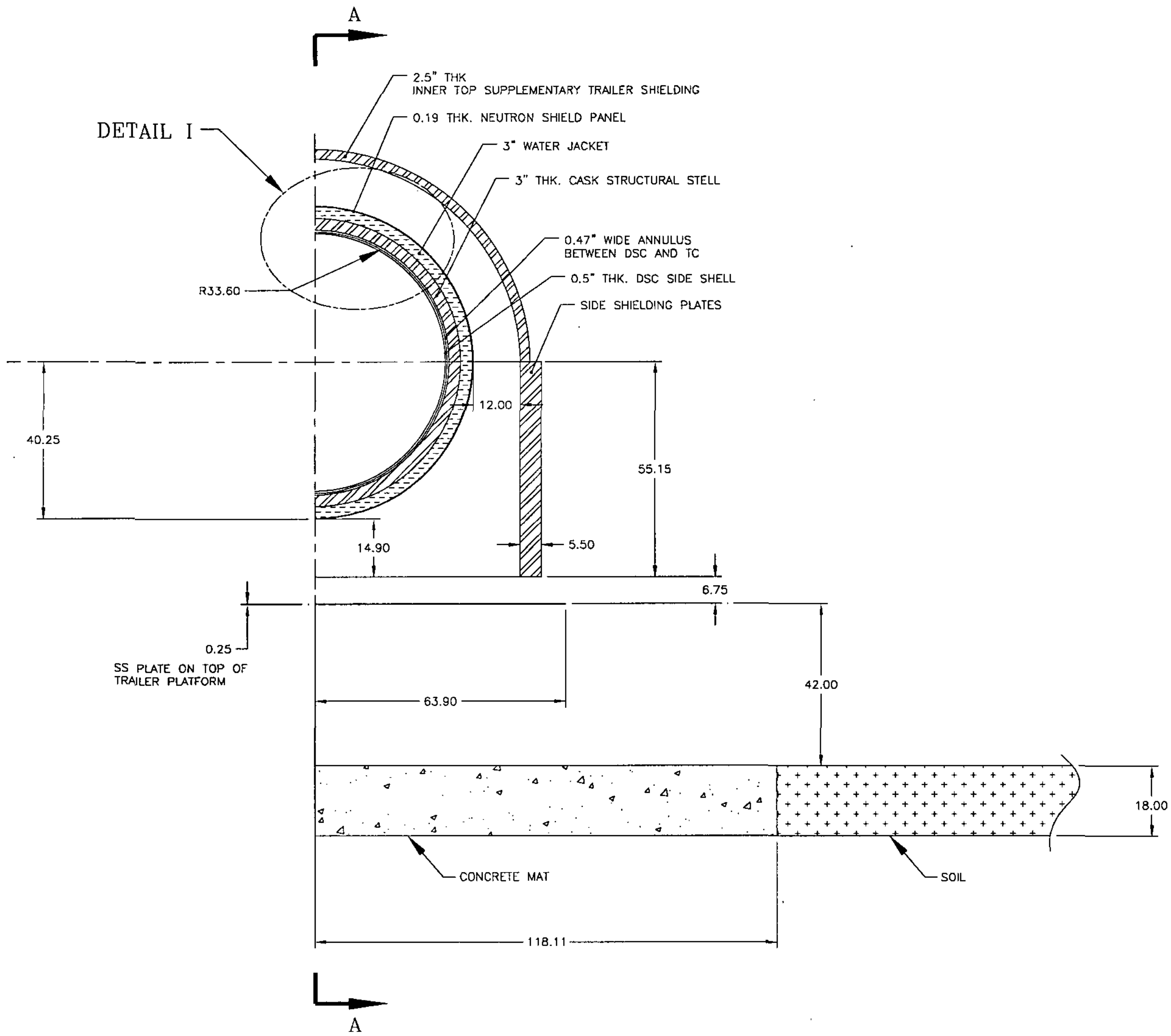


Figure W.5-5
Description of the Trailer Area Shielding Calculational Model

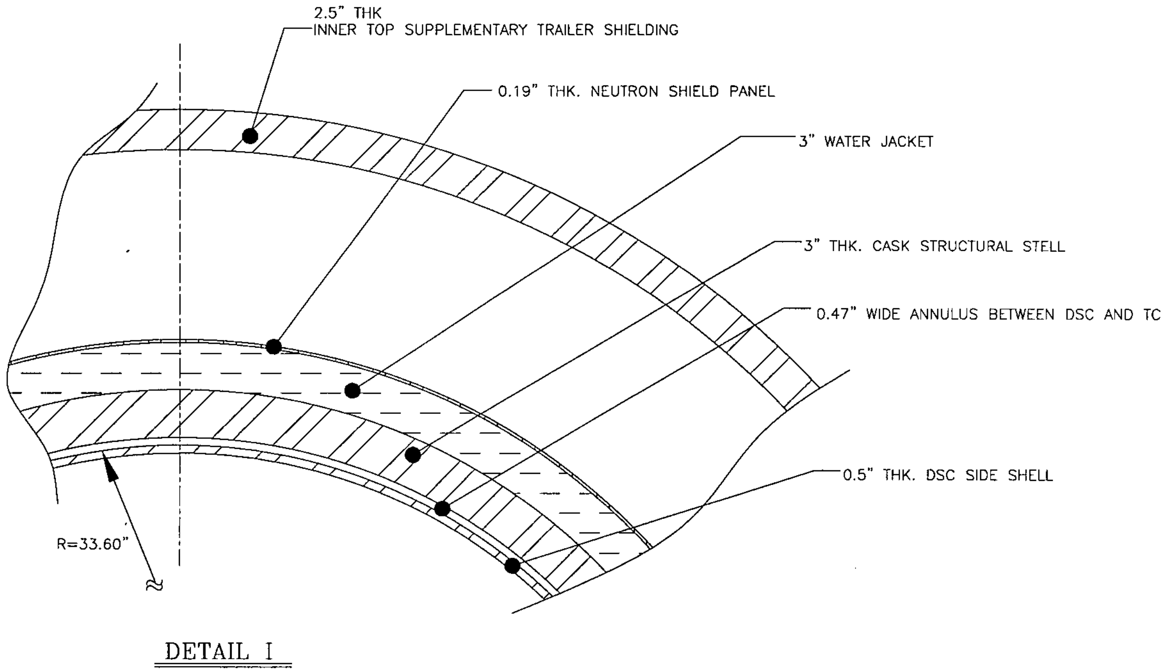


Figure W.5-5
Description of the Trailer Area Shielding Calculational Model
(continued)

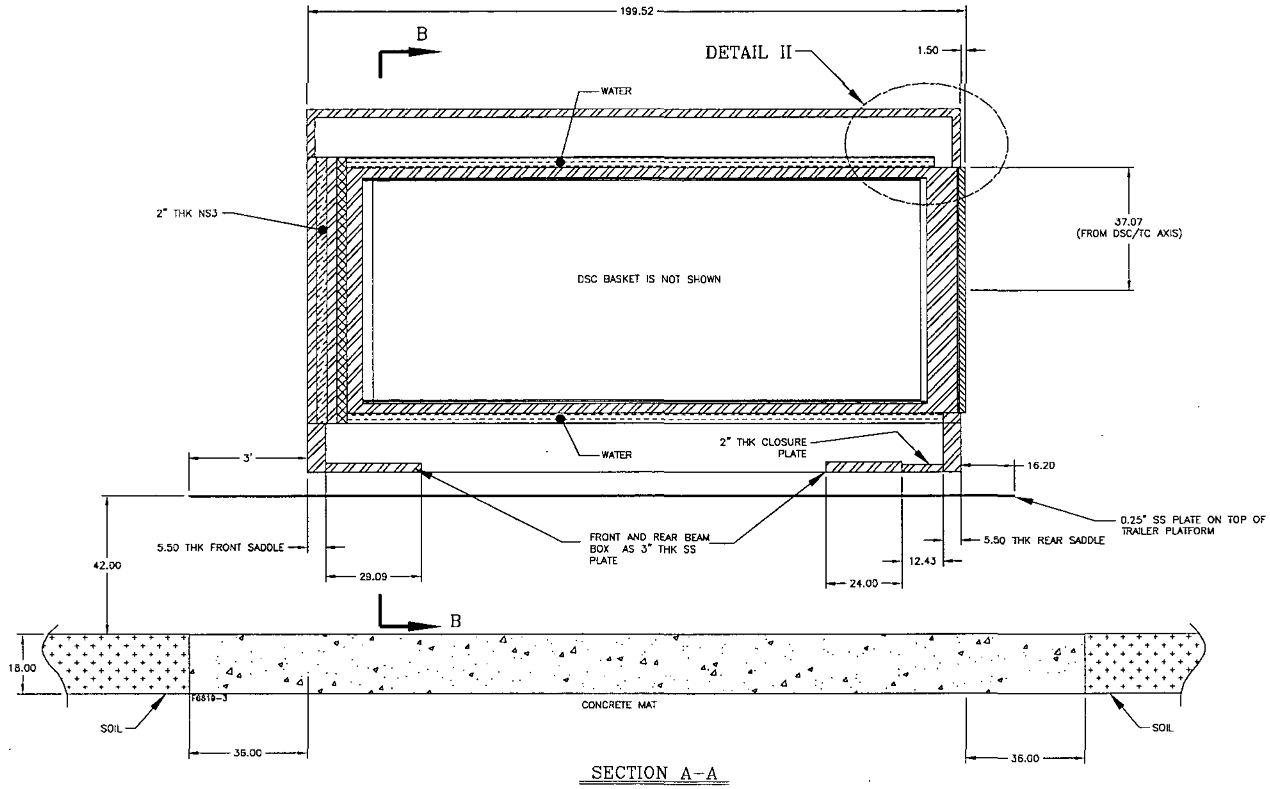
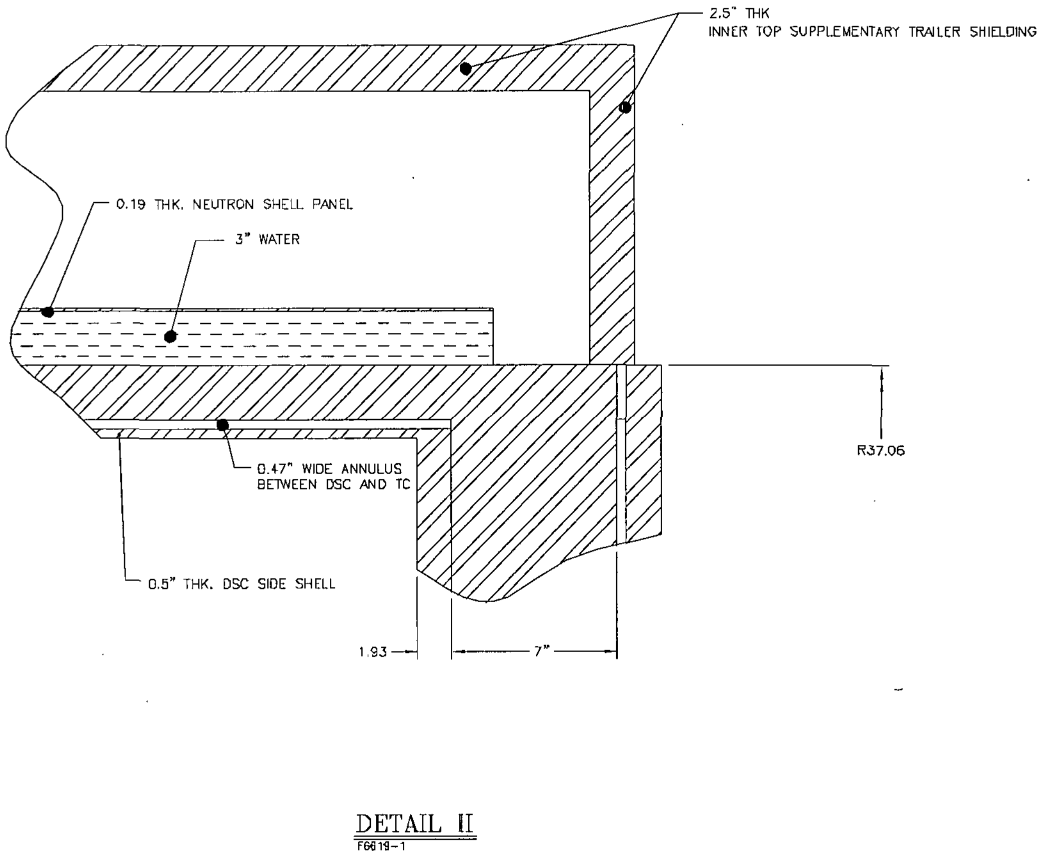


Figure W.5-5
Description of the Trailer Area Shielding Calculational Model
(continued)



From Section A-A

Figure W.5-5
Description of the Trailer Area Shielding Calculational Model
(continued)

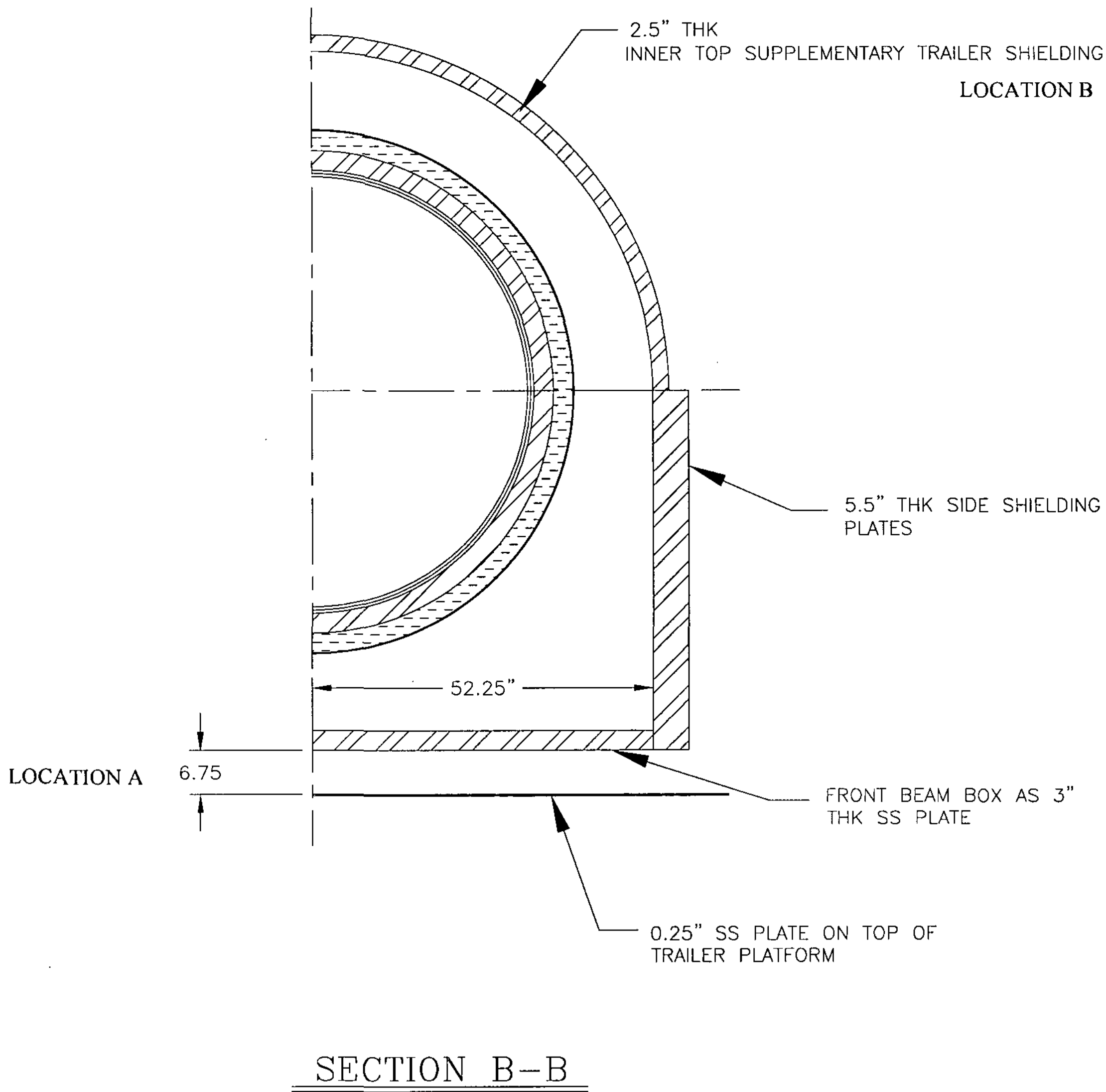


Figure W.5-5
Description of the Trailer Area Shielding Calculational Model
(continued)

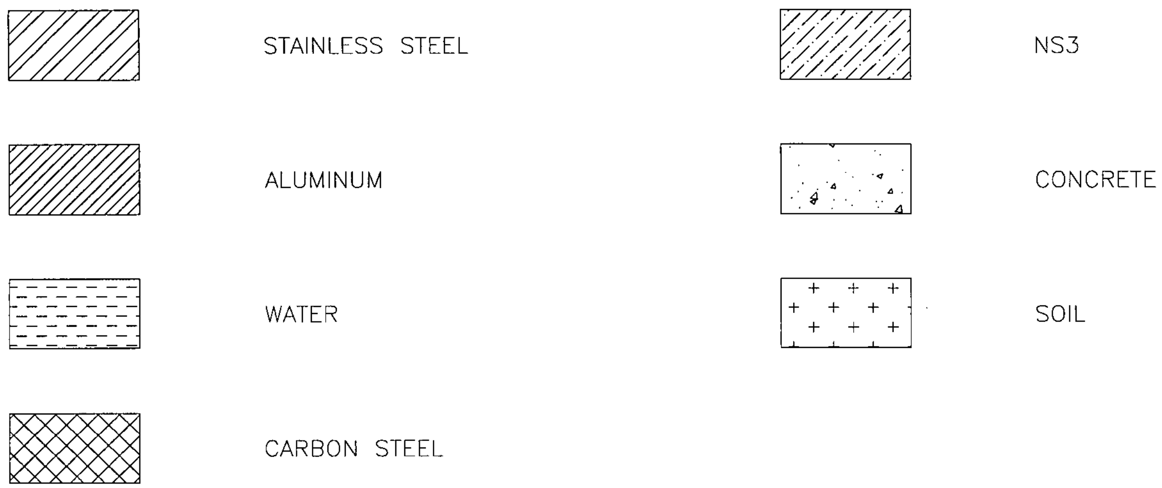


Figure W.5-5
Description of the Trailer Area Shielding Calculational Model
(concluded)

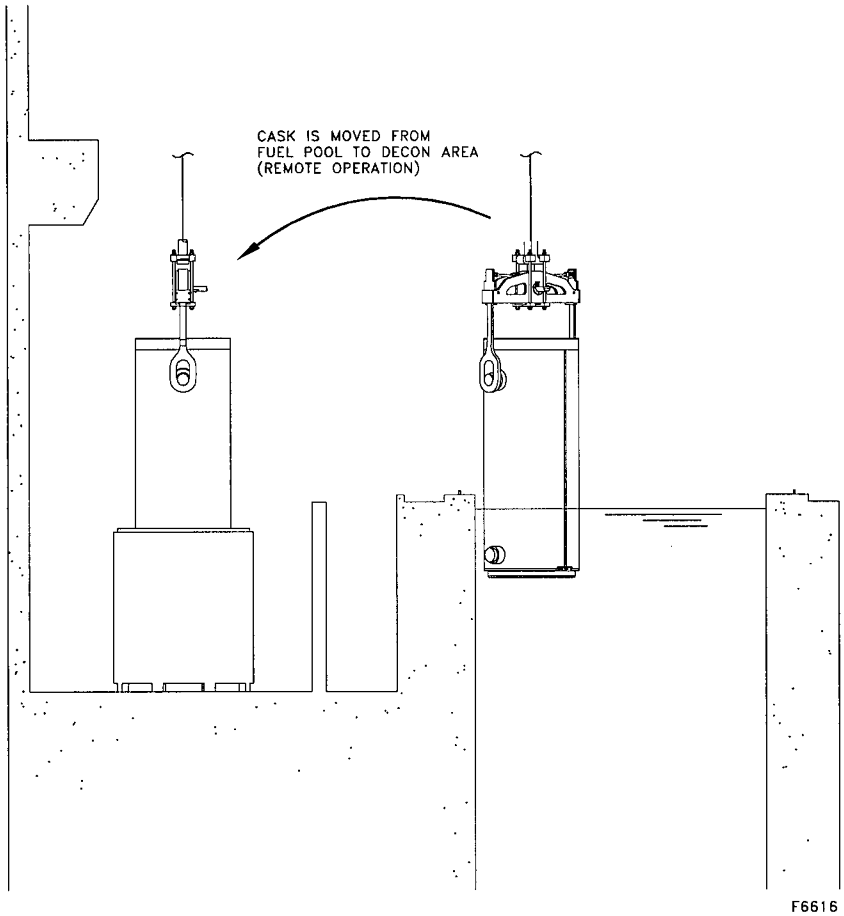
The cask is then lifted until the cask top just breaks the surface of the fuel pool.

CAUTION: Prior to performing the next step of lifting OS197L TC from the pool, the licensee shall meet the specific radiation protection program requirements of applicable Amendment 11 Technical Specification associated with the use of OS197L TC and remote monitoring devices. The licensee shall develop appropriate measures to keep the doses ALARA during recovery from a potential malfunction of these devices, *such as cameras for monitoring, targeting devices, remote controls, etc.*

CAUTION: The dose rates during the movement of a bare loaded TC from the spent fuel pool to the decontamination area supplemental shielding may be as high as 9.8 rem/hr.

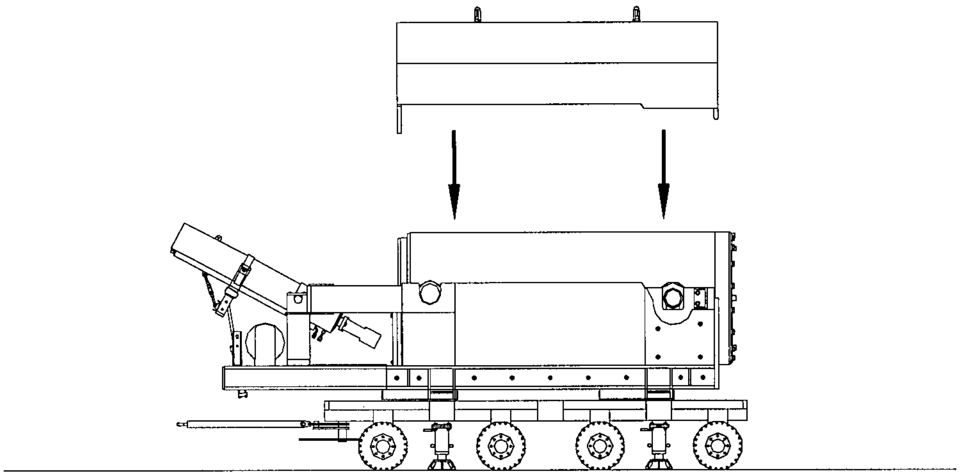
*T*he OS197L TC *is next* lifted from the fuel pool to the decontamination area. The *TC* itself has significantly reduced shielding. However, the OS197L TC *system* utilize*s supplementary* shielding and *additional operational* measures to achieve *acceptable* shielding capacity. As described in the applicable Amendment 11 Technical Specification, the OS197L TC system consists of the bare cask and the upper and lower cask shielding utilized in the decontamination area and the additional shielding provided on the cask support skid. The bare cask is in this reduced shielding configuration ONLY during the movement of the cask from the fuel pool to the decontamination area and from the decontamination area to the transfer trailer. Both of these operations are of short time duration (i.e., minutes).

During bare cask movement from the fuel pool to the decontamination area and from the decontamination area to the trailer, remote crane operation and an optical targeting system with remote camera monitoring *is* used to minimize personnel exposure to the reduced shielding configuration. This remote operation is shown in Sequence *4*.



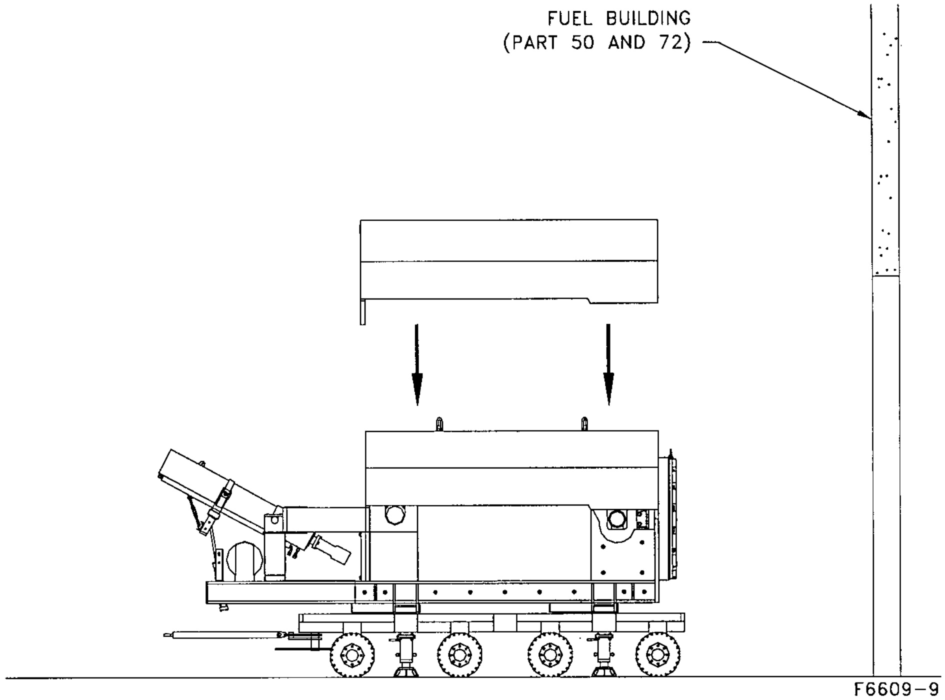
4 FROM FUEL POOL TO DECONTAMINATION AREA, OS197L IS PLACED IN SHIELDING SLEEVE (PART OF OS197L CASK SYSTEM)

In the decontamination area, the bare cask is placed in a shielding sleeve (lower cask shield) which provides shielding below the trunnions. An upper cask shield (shielding bell) is then placed on top of the shielding sleeve to shield the upper section of the cask. The shielding sleeve and shield bell are nominally 6" thick carbon steel. Placement of the cask in the shielding sleeve and placement of the shielding bell on the cask is performed using remote crane operation and an optical targeting system with remote camera monitoring. The OS197L TC system configuration of the cask, shielding sleeve and bell is shown as Sequences *4* and *5*.



(8) INSTALLATION OF SUPPORT SKID INNER TOP SHIELDING

As previously indicated, if the fuel building weight limits preclude placement of the outer top skid shielding inside the fuel building, the OS197L TC may be moved out of the fuel building and the outer top *skid* shielding installed outside *the fuel building.* This is shown in Sequence *9*.



9 INSTALLATION OF PART 72 SUPPORT SKID OUTER TOP SHIELDING

CAUTION: Visually monitor the outer top trailer skid vents and the openings around the cask ends for any sign of steaming which may indicate leakage of water from the cask neutron shield. If steaming is determined to be due to leakage of neutron shield water and not due to any rain or snow or other ambient conditions, then Licensee must take appropriate corrective actions including terminating the transfer operation and returning the loaded cask to the fuel handling building for further assessment.

The transfer trailer, with loaded OS197L TC including the supplemental shielding, is then moved to the ISFSI and the cask docked with the HSM. The DSC is then inserted into the HSM using the same methods as the OS197 TC. This is shown in Sequence *10*.

5. Remove the TC top cover plate and examine the cask cavity for any physical damage and ready the cask for service.

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

7. *Record the DSC serial number which is located on the grapple ring. Verify the correct DSC type, basket type and poison material types against the DSC serial number. Verify that the DSC is appropriate for the specific fuel loading campaign per the criteria specified in Technical Specification 2.1.*

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

9. *If damaged fuel assemblies are included in a specific loading campaign, place the required number of bottom end caps provided into the cell locations per Technical Specification 2.1. Optionally, this step may be performed at any prior time.*

10. *Fill the cask/DSC annulus with clean, demineralized water. Place the inflatable seal into the upper cask liner recess and seal the cask-DSC annulus by pressurizing the seal with compressed air.*

    For the 32PT DSC, fill the DSC cavity with water from the fuel pool, or an equivalent source which meets the requirements of Technical Specification 3.2.1.

11. *For the 61BT DSC, fill the DSC cavity with water from the fuel pool, an equivalent source, or demineralized water. (Note: this step may be accomplished in the fuel pool).*

    NOTE: A TC/DSC annulus pressurization tank filled with demineralized water as described above is connected to the top vent port of the TC via a hose to provide a positive head above the level of water in the TC/DSC annulus. This is an optional arrangement, which provides additional assurance that contaminated water from the fuel pool will not enter the TC/DSC annulus, provided a positive head is maintained at all times.

12. *If not done previously, place the top shield plug onto the DSC and examine the top shield plug to ensure a proper fit. The top shield plug, once fitted is removed and disconnected from the yoke.*

13. *Position the cask lifting yoke above the transfer cask and engage the cask lifting trunnions and the rigging cables to the DSC top shield plug. Adjust the rigging cables as necessary to obtain even cable tension.*

14. *Visually inspect the yoke lifting hooks to insure that they are properly positioned and engaged on the cask lifting trunnions.*

15. Provide for later connection to the vacuum drying system (VDS) or an optional water *draining/pumping device to the siphon port of the DSC and position any connecting hose such that the hose will not interfere with loading (yoke, fuel, shield plug, rigging, etc.). A*

flowmeter or other suitable means for measuring the amount of water removed must be installed at a suitable location as part of this water removal system.

16. *Move the scaffolding away from the cask as necessary.*

17. *Lift the cask just far enough to allow the weight of the cask to be distributed onto the yoke lifting hooks. Reinspect the lifting hooks to insure that they are properly positioned on the cask trunnions.*

18. *a. Optionally, secure a sheet of suitable material to the bottom of the TC to minimize the potential for ground-in contamination. This may also be done prior to initial placement of the cask in the decon area.*

    b. Fill the TC liquid neutron shield. This step may be completed at any time prior to immersion of the TC/DSC into the pool.

19. *Prior to the cask being lowered into the fuel pool, the water level in the pool should be adjusted as necessary to accommodate the cask/DSC volume. If the water placed in the DSC cavity was obtained from the fuel pool, a level adjustment may not be necessary.*

W.8.1.2 DSC Fuel Loading

Note: The licensee shall verify that the lifting device used for handling the OS197L TC meets the requirements of the sites lifting program. Licensee shall use remote operations and optical targeting system and other mitigating ALARA practices when handling the bare OS197L TC when loaded with fuel as required by the sites ALARA program and the Radiation Protection Program requirements of Technical Specification 5.2.4a.

1. *Lift the cask/DSC and position it over the cask loading area of the spent fuel pool in accordance with the plant's 10CFR50 cask handling procedures.*

2. *Lower the cask into the fuel pool until the bottom of the cask is at the height of the fuel pool surface. As the cask and yoke are lowered into the pool, spray the exterior surfaces with demineralized water.*

3. *Place the cask in the designated location of the fuel pool.*

4. *Disengage the lifting yoke from the cask lifting trunnions and move the yoke clear of the cask. Spray the lifting yoke with clean demineralized water if it is raised out of the fuel pool.*

5. *The potential for fuel misloading is essentially eliminated through the implementation of procedural and administrative controls. The controls instituted to ensure that damaged and/or intact fuel assemblies and control components (CCs), if applicable, are placed into a known cell location within a DSC, will typically consist of the following:*

1. *Check the radiation levels along the perimeter of the cask/shields. The cask exterior surface should be decontaminated by providing spray mechanisms on the inside of the shield bell or other methods, using good ALARA practices. Water mixed with a commercial decontamination agent may be used for decontamination. Install additional temporary shielding as necessary to minimize personnel exposure.*

2. *Place scaffolding around the cask so that any point on the surface of the cask is easily accessible to personnel.*

3. *Decontaminate the exposed surfaces of the DSC shell perimeter above the TC/DSC annulus seal. Remove the inflatable TC/DSC annulus seal.*

4. *Connect the cask drain line to the cask, open the cask cavity drain port and allow water from the annulus to drain out until the water level is approximately twelve inches below the top edge of the DSC shell. Take swipes around the outer surface of the DSC shell and check for smearable contamination in accordance with Technical Specification 5.2.4 d, taking corrective actions in accordance with that technical specification, if required, potentially involving removal of the fuel assemblies, removal of the DSC from the TC, and decontamination of the entire length of the DSC outer surface.*

   CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

5. *Drain approximately number of gallons of water shown in table below (as indicated on a flowmeter) from the DSC back into the fuel pool or other suitable location. Consistent with ISG-22 [8.5] guidance and Technical Specification 3.1.1, helium at 1-3 psig is used to backfill the DSC with an inert gas (helium) as water is being removed from the DSC. Only helium may be used to assist in the removal of water.*

DSC	*Gallons of Water*
32PT	*750*
61BT	*1100*

   Alternatively, if a slow helium purge is used while monitoring for hydrogen, less than these amounts of water may be drained, because this approach will prevent buildup of flammable gas to a flammability limit.

6. *Disconnect hose from DSC siphon port.*

7. *Install the automatic welding machine onto the inner top cover plate and place the inner top cover plate with the automatic welding machine onto the DSC. Optionally, the inner top cover plate and the automatic welding machine can be placed separately. Verify proper fit-up of the inner top cover plate with the DSC shell*

8. *Check radiation levels along surface of the inner top cover plate. Temporary shielding may be installed as necessary to minimize personnel exposure.*

CAUTION: Per Technical Specification 4.4.6, during transfer operation of a loaded OS197L TC, every hour, visually monitor the outer top trailer shield vents and the opening around the cask ends for any sign of steaming which may indicate leakage of water from the cask neutron shield. If steaming is determined to be due to leakage of neutron shield water and not due to any rain or snow or other ambient conditions, then Licensee shall take appropriate corrective actions including terminating the transfer operation and returning the loaded cask to the fuel handling building for further assessment.

The following step may be performed outside if the fuel building weight limits preclude placement of the outer top skid shielding inside the fuel building (See Technical Specification 4.4.5 for restrictions). CAUTION: Verify that the requirements of Technical Specification 5.3.1b are met prior to next step.

12. *Install the outer top skid shielding. During installation, the bottom most part of the body of the outer top shield shall not be hoisted by the crane more than 4 inches above the top horizontal plate of the inner top shield.*

13. *Verify that the TC radial dose rates measured at the surface of the decontamination area shielding are compliant with limits specified in Technical Specification 5.2.4e. The methodology for determining the TC radial surface dose rates shall be in accordance with TS 5.2.4e.*

W.8.1.6 *<u>DSC Transfer to the HSM</u>*

CAUTION: Per Technical Specification 4.4.6, during transfer operation of a loaded OS197L TC, every hour, visually monitor the outer top trailer shield vents and the opening around the cask ends for any sign of steaming which may indicate leakage of water from the cask neutron shield. If steaming is determined to be due to leakage of neutron shield water and not due to any rain or snow or other ambient conditions, then Licensee shall take appropriate corrective actions including terminating the transfer operation and returning the loaded cask to the fuel handling building for further assessment.

CAUTION: During the actual movement of the transfer cask on the transfer trailer to the ISFSI, the gap between the transfer trailer deck and bottom of the skid shall be monitored (visual inspection) to assure no significant blockage of airflow. Although blockage is improbable as over 60 feet of gap would require sealing, personnel shall maintain a visual scan of the trailer.

1. *Prior to transferring the cask to the ISFSI or prior to positioning the transfer cask at the HSM designated for storage, remove the HSM door using a porta-crane, inspect the cavity of the HSM, removing any debris and ready the HSM to receive a DSC. The doors on adjacent HSMs should remain in place.*

CAUTION: Very high dose rates in the empty HSM are expected if adjacent to a loaded HSM. Proper ALARA practices should be followed during these operations.

2. *Inspect the HSM air inlet and outlets to ensure that they are clear of debris. Inspect the screens on the air inlet and outlets for damage.*

   CAUTION: Verify that the requirements of Technical Specification 5.3.1b are met prior to next step.

3. *Using a suitable vehicle, transfer the cask from the plant's fuel/reactor building to the ISFSI along the designated transfer route.*

4. *Once at the ISFSI, position the transport trailer to within several inches of the HSM.*

5. *Check the position of the trailer to ensure the centerline of the HSM and cask approximately coincide. If the trailer is not properly oriented, reposition the trailer, as necessary.*

6. *If not already installed, install the alignment targets, including the cast top centerline target through the skid shielding.*

7. *Using crane, unbolt and remove the cask top cover plate.*

8. *Back the cask to within a few inches of the HSM, set the trailer brakes and disengage the tractor. Drive the tractor clear of the trailer. Extend the transfer trailer vertical jacks.*

9. *Remove the skid tie-down bracket fasteners and use the skid positioning system to bring the cask into approximate vertical and horizontal alignment with the HSM. Using optical survey equipment and the alignment marks on the cask and the HSM, adjust the position of the cask until it is properly aligned with the HSM.*

10. *Using the skid positioning system, fully insert the cask into the HSM access opening docking collar.*

11. *Secure the cask trunnions to the front wall embedments of the HSM using the cask restraints.*

12. *After the cask is docked with the HSM, verify the alignment of the TC using the optical survey equipment.*

13. *Position the hydraulic ram behind the cask in approximate horizontal alignment with the cask and level the ram. Remove either the bottom ram access cover plate or the outer plug of the two-piece temporary shield plug if installed. Power up the ram hydraulic power supply and extend the ram through the bottom cask opening into the DSC grapple ring.*

14. *Activate the hydraulic cylinder on the ram grapple and engage the grapple arms with the DSC grapple ring.*

15. *Recheck all alignment marks in accordance with the Technical Specification 5.3.3 limits and ready all systems for DSC transfer.*

W.8.3 Identification of Subjects for Safety Analysis

There is no change relative to Section 5.1.3 regarding criticality control, chemical safety, operational shutdown modes and maintenance techniques.

In addition to the typical instrumentation listed in Table 5.1-1 of Section 5.1.3, the use of OS197L TC may require optical targets and instruments to implement specific remote crane operations described in Section W.8.1 above.

W.8.4 Fuel Handling Systems

W.8.4.1 Spent Fuel Handling and Transfer

No change. The general description of the spent fuel handling and transfer components presented in Section 5.2.1 remains applicable. However, there are differences in design features and operations as noted below.

W.8.4.1.1 Functional Description

The OS197L TC and the associated transfer equipment are described in sections W.1.2 and W.1.2.1.1. The difference in design features of these components relative to the OS197 TC described in Section 5.2.1.1 are discussed in Appendix W.1.

Note: There are significant differences in operation steps when using the OS197L TC. Hence, the illustration provided in Figure 5.2-1 is not applicable when using the OS197L TC.

W.8.4.1.2 Safety Features

No change. The safety features described in Section 5.2.1.2 remain applicable.

W.8.4.2 Spent Fuel Storage

Descriptions of the operations used for the transfer and retrieval of the DSC from the HSM are presented in Section W.8.1. There is no change to the safety features discussed in Section 5.2.1.2.

W.8.5 Other Operating Systems

No change is needed to the information provided in SAR Section 5.3 except that the accident analysis is provided in Section W.11 for the OS197L system.

W.8.6 Operation Support System

No change to the information provided in Section 5.4 except for the optical targets and facilities required for remote crane operations described in Section W.8.

W.9 Acceptance Criteria and Maintenance Program

The use of the OS197L includes the acceptance criteria, and maintenance program requirements described elsewhere in the UFSAR for the OS197 TC, with the follow clarifications and additional considerations unique to the OS197L.

As described in Section W.3.2, the test load criteria for the upper trunnions of the OS197L are the same as those for the OS197 TC, except that the test load is conservatively equal to 300% of the design load instead of 150% for the OS197 TC in Section 4.2.3.3.

Consistent with Section 4.3.9 the transfer cask is designed to require only minimal maintenance, limited to periodic inspection of critical components and replacement of damaged or nonfunctioning components. A discussion of these requirements is provided in Section 4.5. Section 4.5 of the UFSAR describes routine and annual visual/functional inspections and the repair or replacement actions to be taken if any of the inspections fail. For the OS197L the routine inspections should include the supplemental shielding (visual inspections for damage) and the mechanical and hydraulic/pneumatic components (visual and functional inspection) of the lifting yokes or other hardware used for remote operations when using the OS197L TC.

In addition, the applicable pre-operational testing requirements described in Section 9.2, and the training requirements in Section 9.3 are applicable when the OS197L TC is used, with the following additional requirements:

1. *The pre-operational testing and operation programs for licensees using the OS197L system should specifically include the unique operational features of the OS197L system based on normal planned operating procedures and based on procedures for recovery from a crane malfunction and failure of other remote operations equipment such as the optical targeting system, cameras, etc. The unique operational features should include as a minimum: 1) operations with the yoke to engage and disengage the TC trunnions during operations at the cask handling area and during upending/downending, 2) use of supplemental shielding in the cask handling area and on the transfer trailer/skid, and 3) operations including remote targeting or any other hardware needed for remote handling. These unique operational features for normal and recovery conditions should be included in dry run or practice exercises prior to each loading campaign, even if done at the same site with the same personnel.*

2. *The training programs for licensees using the OS197L system should specifically include the unique operational features of the OS197L system based on normal planned operating procedures and based on procedures for recovery from a crane malfunction and failure of other remote operations equipment such as the optical targeting system, cameras, etc.*

The OS197L payloads are the 61BT and 32PT DSCs. The acceptance tests and maintenance programs for these DSCs as described in Chapters K.9 and M.9 of the UFSAR are not changed by the use of the OS197L TC.

W.10 Radiation Protection

As discussed in Section W.5, use of the OS197L TC does not significantly affect personnel doses (during closure operations, handling, or storage) or site boundary dose rates during normal operations with supplemental shielding as described in Section W.1. The OS197L TC is used only for loading/unloading and transfer operations, and the storage conditions are unchanged. Therefore, the personnel doses, occupational exposures and site bounding dose rates documented for each DSC/HSM storage configuration in Chapter K.10 *and* Chapter M.10 remain unchanged and are applicable to operations using the OS197L TC.

The use of the OS197L TC is not expected to have any significant impact on personnel doses during normal operation since the operations for placement and removal of bare OS197L TC from the fuel pool into the decontamination area shielding sleeve, placement and removal of the decontamination area shielding bell, engagement of the yoke to the cask trunnions, movement of the cask to the trailer, lowering of the cask onto the trailer and placement of the *skid* shielding on the cask will be performed remotely using cameras and laser/target positioning.

Additional occupational exposures due to operations unique to the OS197L TC are evaluated in the following sections:

W.10.1 Recovery/Repair Operations due to Remote Handling Device Malfunction

The OS197L TC uses remote handling devices for movement of the TC during loading and transfer operations. In the event of a failure of *either* the remote handling device *or a malfunction of the crane,* an evaluation has been performed to assess the additional occupational exposure during recovery operations. This evaluation was performed using the dose rates from the OS197L TC when loaded with a NUHOMS® 32PT DSC.

For this evaluation, the crane is postulated to fail as the OS197L TC is being lowered onto either the decontamination area or the transfer trailer.

Dose rates around various shielding configurations are presented in Chapter W.5. The dose rate distribution utilized in the occupational exposure calculations during a crane malfunction can be estimated using equations provided below.

The bounding dose rate at a radial distance, r, from the side of a bare cask can be approximated by the expression:

$$D(\vec{r}) = \sum_{i=0}^{3} a_i \cdot \exp(-b_i \cdot r) \qquad \textit{Eq. W.10-1}$$

where the values for a_i and b_i are given in Table W.10-1 for $0 \le r \le 10$ *and* $10 < r \le 200$. *This equation represents dose rates that are in a very good agreement with data in Table W.5-3, Table W.5-6 and Table W.5-9 (OS197L TC Bare Cask).*

Table W.10-1 Parameters of Eq. W.10-1

Radial Distance (meters)	*Parameters ([a] = mrem/hr and [b] = m⁻¹)*							
	a_0	b_0	a_1	b_1	a_2	b_2	a_3	b_3
$0 \le r \le 10$	1.9339	0.0	2696.5243	2.7571	5562.0454	0.7258	1574.6363	0.1965
$10 < r \le 200$	0.0085	0.0	11.0493	0.0208	569.6182	0.0963	0	0

Similarly, the maximum dose rate within the outer radius of the cask, R, at a distance, H (see Figure W.5-3), above the cask top in the axial direction can be approximated by the expressions:

$$D(H) = (25.2 - 107) * H + 107, \text{ for } r < R \text{ and } 0 \le H \le 1 \text{ meter} \qquad \text{Eq. W.10-2a}$$

$$D(H) = 26 / H, \text{ for } r < R \text{ and } 1 < H < 10 \text{ meters} \qquad \text{Eq. W.10-2b}$$

To provide an example of how exposure to a worker can be calculated, consider a bare cask hang-up due to a crane malfunction which requires a worker to perform some manual operations on the crane so that the cask can be lowered and laid on its side on a trailer. It is assumed that this occurs in a large room and at a distance from the walls and ceiling sufficient to ignore backscatter, so no backscatter contributions are included in this example. However, the significance of backscatter contributions should always be considered when calculating this type of occupational exposure. Factors such as a small room geometry (defined as having dimensions less than twice the length of the cask) or operations performed close to the wall or ceiling may require the use of backscatter correction factors.

For this scenario, a worker must perform the following steps:

1. *Use a ladder to reach the elevation of a crane bridge,*
2. *Use a walkway to reach the crane bridge,*
3. *Crawl on the crane bridge itself to reach a location just above the cask top,*
4. *Move to the cask top and perform necessary repair operation on the crane or cask,*
5. *Return to the crane bridge,*
6. *Remain in place on the crane bridge while the cask is lowered to the trailer and tilted on its side,*
7. *Crawl back across the crane bridge to the walkway,*
8. *Use the walkway to reach the ladder, and*
9. *Descend the ladder.*

Figure W.10-1 depicts scenario as seen in a plan view.

It is assumed that the elevation of the crane bridge, the length of the walkway, and the distance from the walkway to the cask are all 15.24 m (50 ft). It is also assumed that the top of the cask is 1 m below the crane bridge when the hang-up occurs and that the trailer is 0.91 m (3 ft) above the ground. When applying this type of calculation to a specific site, it is important to use the actual distances expected to be encountered during the operation.

*the cask. Assuming the cask it tilted over a period of 0.5 hrs, the worker is exposed to **101.33** mrem.*

Note: The following sequence of steps 7 through 10 are the reverse of step 4 through 1 above. Speed and completion times remain unchanged.

*7. Operations are performed at a radial distance of 11.19 m from the cask, which corresponds to a dose rate of 203 mrem/hr. The worker is exposed to **33.78** mrem.*

*8. As the worker crawls back across the crane bridge, the radial distance to the cask increases from 11.19 m to 18.91 m. Conservatively, it is assumed that the worker is exposed to the maximum predicted dose rate of 203 mrem/hr at a radial distance of 11.19 m. The worker is exposed to **3.20** mrem.*

*9. The walkway is now at a radial distance of 18.91 m from the cask where the predicted dose rate is 100 mrem/hr. The worker is exposed to **0.31** mrem.*

*10. As the worker descends the ladder, the radial distance to the cask decreases from 18.91, passing at a minimum at 15.24 m. Conservatively it is assumed that the worker is exposed to the maximum predicted dose rate of 139 mrem/hr at a radial distance of 15.24 m. The worker is exposed to **1.37** mrem.*

*The summation of the total exposure received by a worker from the above 10 steps results in a total exposure of **186.96** mrem. If more than one worker is involved in conducting the above described repair/recovery operations, the exposure must be scaled accordingly.*

W.10.2 Inspection of Decontamination Shield Openings for Blockage

In addition to the operations that are performed using remote handling equipment, there are additional minor operational steps that are necessitated due to the OS197L TC. For example, an operational step is needed during decontamination operations to ensure that the "openings" in the decontamination area shield are not blocked. The bottom openings are located at the bottom radial location of the decontamination area shield and the top openings are located at the top radial location of the decontamination area shielding bell. *A discussion of the dose rate results associated with cask decontamination is included in Section W.5.4.10.2 of the SAR.*

The bare cask radial surface dose rates at these axial locations are conservatively employed to determine the dose rate fields. *The maximum dose rate near the top opening using the bare cask radial dose rate results from Table W.5-8 and Table W.5-11 is approximately 3000 mrem/hr (at a distance greater than 500 cm from the bottom of the OS197L TC). The maximum radial dose rate at the side of the OS197L TC during decontamination as displayed in Figure W.5-2 (results corresponding to curve 4) is less than 100 mrem/hr.* The average dose rate at the top opening location is *conservatively estimated to be* less than 700 mrem/hr *(Section W.5.4.10.2).*

Since this operation is of the order of a minute or less, the total contribution to the occupational exposure is less than *12* mrem. Workers are expected to follow the appropriate ALARA practices while performing this step, particularly at the top of the OS197L TC.

W.11 Accident Analyses

This section describes the postulated accident events that could occur during fuel loading, draining, drying, welding and transfer of the DSC using a NUHOMS® OS197L TC. Sections which do not affect the evaluations presented in Appendices K.11 *and* M.11 *for the NUHOMS®-61BT and 32PT* DSC designs are identified as "No change." Detailed analyses of the events are provided in other sections and are referenced herein. *The cask support skid supplemental shielding is referred to as trailer shielding throughout this chapter.*

W.11.1 Postulated Accidents

Only those accidents affecting the OS197L TC are addressed in this section. There is no change to accident evaluations affecting other NUHOMS® components.

W.11.1.1 OS197L TC Missile Impact Analysis

This event is described in Section 8.2.2.4. The OS197L TC uses a 2.68" steel shell in lieu of a 1.5" steel shell with a nominal 3.5" lead annulus and a 0.5" inner liner for OS197 TC. The missile impact analyses for the OS197 TC are therefore bounding for the OS197L TC.

W.11.1.2 Earthquake

This event is described in Section 8.2.3.D. The OS197L TC configuration (cg location, cask length, trunnion location and bottom forging configuration) does not significantly differ from that of the OS197 TC. The OS197L TC remains stable when subjected to the design basis earthquake.

W.11.1.2.1 OS197L TC in a Vertical Configuration during Vacuum Drying and Welding Operations

The bottom forging on which the cask is resting during vertical cask operations is the same size and configuration as the OS197 TC. The OS197L TC cg location is not significantly altered by the change in the cask shell configuration. The addition of the decontamination area shield will provide a larger diameter, more stable shell, outside the cask envelope, thereby potentially enhancing the OS197L TC seismic capacity.

W.11.1.2.2 OS197L TC in a Horizontal Configuration during Transfer Operations

The cask seismic stresses for the OS197L TC are bounded by the OS197 TC stresses due to the similar configurations of the cask ends (top and bottom forgings and covers) and larger thickness structural shell.

The trailer with the OS197L TC, with the additional shielding, remains stable for the design basis seismic accelerations.

W.11.1.3 OS197L TC Accidental Cask Drop

This event is described in Sections 8.2.5.2.B, D and E.

See Section W.3.7 for a discussion of the OS197L TC drop accident. This drop accident is bounded by the results for the OS197 TC drop accident discussed in Section 8.2.

W.11.1.4 Loss of Neutron Shield

This event is described in Section 8.2.5.3.

For the accident condition (the unlikely cask drop scenario), a complete loss of the OS197L TC neutron shield is postulated similar to the OS197 TC evaluation described in Section 8.2.5.3. The analysis conservatively assumes that all the trailer shielding is lost. However, the trailer shield is fabricated using two sets of plate shields (the inside shield is 2.5" thick, the outside shield is 3" thick) which may be damaged in a drop but are unlikely to separate completely from the skid and cask. A comparison of the dose rate results for *various OS197L shielding configurations including complete loss of trailer shielding* (shown as "Bare Cask" configuration) is shown in the table below.

Assuming the non-mechanistic drop scenario occurs and the trailer shields and the cask are dislodged completely from the trailer and skid, recovery actions are required to manipulate the shields or providing supplemental shielding to reduce dose rates to a reasonable value until a long term recovery plan is in place. *As discussed above, the dose rates for this accident case are calculated assuming a complete loss of trailer shielding.*

OS197L TC Accident Condition Dose Rates

(From results shown in Table W.5-4)

Transfer Cask Configuration	Dose Rate Component	Dose Rates at Different Distances from Side Surface – Accident Condition No Neutron Shield			
		On Side Surface	4.57 meters (15')	100 meters	609.9 meters (2000')
		Dose Rate, mrem/hr	Dose Rate, mrem/hr	Dose Rate, mrem/hr	Dose Rate, mrem/hr
OS197 L TC (with supplemental inner and outer trailer shielding)	Neutron	*727*	*79.0*	*0.32*	*2.31e-4*
	Gamma	*134*	*40.0*	*0.14*	*3.04e-4*
	Total	*791*	*104*	*0.46*	*5.34e-4*
OS197 L TC (with supplemental inner and without outer trailer shielding)	Neutron	*1466*	*87.0*	*0.48*	*7.36e-4*
	Gamma	*540*	*59.0*	*0.29*	*1.07e-3*
	Total	*1543*	*129*	*0.77*	*1.81e-3*
OS197L TC (Bare Cask)	Neutron	*4,194*	*175*	*0.20*	*4.97e-5*
	Gamma	*15,305*	*1,332*	*2.30*	*2.43e-3*
	Total	*18,210*	*1,438*	*2.48*	*2.46e-3*

The dose rates on the ends of the OS197L TC will be the same as the OS197 TC since the top and bottom forging and cover plate configurations have not been modified. *Therefore*, the dose rates at the controlled area boundary, assuming a 100 meter boundary, would be approximately *2.48* mrem/hr during the timeframe that the cask trailer shield is dislodged from the cask and until the trailer shield is repositioned. The 8 hours of recovery period assumed is appropriate because the repositioning of the trailer shields will be performed using lifting hardware pre-positioned prior to transfer operations. This will facilitate quick positioning using a crane to minimize the need for personnel to approach the cask. *The dose rates at the controlled area* *boundary after repositioning of the inner and outer trailer shields are 0.77 and 0.46 mrem/hour respectively.*

The *total* dose *due to an 8-hour exposure* at the controlled area boundary (100 meters) is *calculated to be slightly less than 20.0 mrem. Note that this is lower than the 42 mrem exposure reported in UFSAR Section M.11.2.5.3 for the 32PT DSC with the OS197 TC. In summary,* the total dose at the 100 meter *controlled area* boundary still remains very low and below the regulatory limit of 5,000 mrem.

The total dose due to an 8-hour exposure to off site individuals at the site boundary (2000 ft) is calculated to be 0.02 mrem. Note that this is lower than the 0.09 mrem exposure reported in UFSAR Section M.11.2.5.3 for the 32PT DSC and 0.13 mrem exposure reported in UFSAR Section K.11.2.5.3 for the 61BT DSC with the OS197 TC.

The thermal evaluation for this accident condition is included in Chapter W.4.

W.11.1.5 Accidental Drop of Top Trailer Shielding

Placement of the inner and outer shields on the skid inside the fuel/reactor building is to be performed in accordance with the plant's heavy loads procedures. If a single failure proof crane is not used, the licensee is to evaluate the accidental drop of the shields under the provisions of 10 CFR 50.59 and 10 CFR 72.212 and evaluate consequences of this drop accident.

In the case when fuel/reactor building floor loads limit placement of both the inner and outer trailer shields inside the fuel/reactor building, the outer top trailer shield may be placed outside the fuel/reactor building. This condition is evaluated for accidental drop of the outer top trailer shield onto the already mounted inner trailer shield.

The stresses in the inner shield are evaluated in accordance with Subsection NF stress criteria for accident (Level D) conditions. For Level D loads, the Subsection NF stress criteria for accident loads use the Appendix F stress limits. Based on a conservative elastic analysis model used in the stress evaluation and using conservation of energy principles, the maximum drop height which will meet the level D stress limits is on the order of 4 inches. Thus, the movement of the outer shield over the skid is to be controlled such that the maximum drop height does not exceed 4 inches.