TRANSNUCLEAR
AN AREVA COMPANY

December 18, 2007
E-25857

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

Subject: Revision 1 to Transnuclear, Inc. (TN) Application for Amendment 2 to the Standardized Advanced NUHOMS® System (Docket No. 72-1029; TAC NO. L24056)

Reference: Letter from Jessica M. Glenny (NRC) to Donis Shaw (TN), "REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF AMENDMENT 2 TO THE STANDARDIZED ADVANCED NUHOMS® SYSTEM (TAC NO. L24056)," October 17, 2007

Gentlemen:

This submittal provides responses to the request for additional information (RAI) forwarded by the referenced letter. Enclosure 1 herein provides each of the NRC staff RAI followed by a TN response. Enclosure 2 provides a list of UFSAR drawings and pages associated with Amendment 2. Enclosure 3 provides Amendment 2 Revision 1 versions of proposed changes to CoC 1029, the associated Technical Specifications, and the Standardized Advanced NUHOMS® System UFSAR, Revision 2.

In the Technical Specifications, the Amendment 2 changes are shown in italics, with revision bars. The Revision 1 changes are shown in light blue color, to distinguish them from the Revision 0 changes. For the UFSAR, Amendment 2 replacement pages and new pages are provided, annotated as Revision 1, with changes indicated by revision bars.

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 410-910-6930.

Sincerely,

Robert Grubb
Senior Vice President - Engineering

NMSS01

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cc: Jessica M. Glenny (NRC SFST) (7 paper copies, provided in a separate mailing)

Enclosures:

1. RAI Responses
2. List of UFSAR Pages associated with Amendment 2
3. Amendment 2 Revision 1 Proposed changes to CoC 1004 (Amendment 1), the associated Technical Specifications, and the UFSAR (Revision 2)

Enclosure 1 to TN E-25857

RAI Responses

Technical Specifications

TS-1 Justify the basis for not lowering the vacuum drying time limits established in proposed TS 3.1.1.b, to account for the conductivity of nitrogen during blow-down.

It is not clear how "conservatisms" referenced in Section 4.4 of the FSAR account for the reduced conductivity of nitrogen instead of air, given the currently approved thermal analyses methodology.

This information is needed to determine compliance with 10 CFR 72.236(d).

Response to TS-1

Use of nitrogen during blowdown of water in the DSC cavity is eliminated. Only helium is now used for this blowdown operation. The operating procedures in Chapter A.8 are revised to reflect this. The Technical Specification 3.1.1.b is revised to eliminate Nitrogen during blowdown.

TS-2 Provide surface dose rate limits for the transfer cask in the proposed TS 5.2.4, "Radiation Protection Program."

The applicant has proposed a modified radiation protection program in TS 5.2.4 that does not include transfer cask dose rate limits. As stated in the guidance provided in NUREG-1745, "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance," the administrative controls are to include a cask loading, unloading and preparation program that establishes criteria that need to be verified to address FSAR commitments and regulatory requirements. The guidance further states that the program requirements are to be completed prior to the cask's removal from the 10 CFR Part 50 structure; for the Standardized Advanced NUHOMS system, this means establishing TS surface dose rate limits for the transfer cask.

10 CFR 72.236(d) requires the provision of radiation shielding features sufficient to meet the requirements of 10 CFR 72.104. The regulation in 10 CFR 72.104(b) requires licensees to establish operational restrictions to meet as low as reasonably achievable (ALARA) objectives for direct radiation levels associated with ISFSI operations. The regulation in 10 CFR 20.1101(b) also requires licensees to use procedures and engineering controls based upon sound radiation protection principles to achieve doses that are ALARA. TS dose rate limits for the transfer cask, therefore, ensure that transfer cask features remain sufficient to enable the licensee to meet these regulatory requirements. Also, TS dose rate limits for the transfer cask provide the licensee with the information necessary to perform a thorough ALARA evaluation and establish appropriate operational restrictions for anticipated cask work to minimize personnel exposure, thus aiding the effectiveness of the licensee's implementation of its 10 CFR Part 50 and Part 20 programs with respect to spent fuel handling (such as ensuring that TS affecting operations in the fuel handling building are met). Additionally, the transfer cask surface dose rate limits also serve as a check against mis-loading of a dry shielded canister (DSC). Staff notes that administrative measures for ensuring proper DSC loading alone may not be sufficient for ensuring proper performance of the cask system such that 10 CFR 72.104(b) will be met. Staff also notes that the licensee's radiation

protection personnel will be making multiple measurements during cask loading operations; therefore, measurements that verify compliance with TS dose rate limits will be among those performed by these personnel and will thus be readily available. Surface dose rate limits should be provided for a transfer cask that contains the 24PT1-DSC and for a transfer cask that contains the 24PT4-DSC. The dose rate limits for both transfer cask configurations will ensure transfer cask features remain sufficient to enable the licensee to meet 10 CFR 72.104(b) and 10 CFR 20.1101(b) requirements for future operations involving both DSCs under this amendment.

This information is needed to determine compliance with 10 CFR 72.104(b) and 72.236(d).

Response to TS-2

The requested surface dose rate limits for the 24PT1 and 24PT4 DSC's are provided in the proposed TS 5.2.4 for the wet welding case.

TS-3 Provide, and justify, the number and locations of the dose rate measurements used to ensure the transfer cask surface dose rates will meet the dose rate limits (see RAI question TS-2) in the proposed TS 5.2.4, "Radiation Protection Program."

Dose rate limits for the surface of the transfer cask, along with the number and locations of dose rate measurements and cask configuration (e.g. prior to DSC closure) when performing measurements (consistent with the shielding analysis and package operations), should be specified as part of the applicant's radiation protection program in proposed TS 5.2.4. Surface dose rate limits should be provided for a transfer cask that contains the 24PT1-DSC and for a transfer cask that contains the 24PT4-DSC. The dose rate limits for both transfer cask configurations will ensure transfer cask features remain sufficient to enable the licensee to meet 10 CFR 72.104(b) and 10 CFR 20.1101(b) requirements for future operations involving both DSCs under this amendment.

This information is needed to determine compliance with 10 CFR 72.104(b) and 72.236(d).

Response to TS-3

The configuration and location of the proposed Transfer Cask surface dose rate limits and their justification for the 24PT1 and 24PT4 DSC's are provided in the proposed Technical Specification 5.2.4.

TS-4 Provide surface dose rate limits for the Advanced Horizontal Storage Module (AHSM) in the proposed TS 5.2.4, "Radiation Protection Program."

The applicant has proposed a radiation protection program (TS 5.2.4) that does not include surface dose rate limits for AHSMs containing the DSCs used with the storage system. AHSM surface dose rates serve as criteria to assure the regulatory limits in 10 CFR 72.104(a) can be met by the licensee as well as assure that the AHSM shielding features continue to meet 10 CFR 72.236(d) for future operations under this amendment. Dose rate limits should be specified for appropriate AHSM surface locations and be supported by the shielding analysis. Measurements of AHSM surface dose rates upon completion of loading provide an immediate indication as to whether or not the regulatory limits of 10 CFR 72.104(a) will be met. Staff notes that an approach to AHSM dose rate limits such as was done for the NUHOMS-HD system is acceptable.

This information is needed to determine compliance with 10 CFR 72.104(a) and 72.236(d).

Response to TS-4

Due to superior shielding capabilities of the AHSM, with the exception of the front and roof birdscreen surfaces, the AHSM surface dose rates are extremely low and do not provide any real indication of meeting 10CFR 72.104(a) limits. Therefore, AHSM front birdscreen surface dose rate limits are selected and added to the proposed TS 5.2.4.

TS-5 Provide the following editorial changes to the last sentence of TS 2.2.a:

a. Change "DAMAGED FUEL ASSEMBLIES" to "DAMAGED FUEL ASSEMBLY".

b. Add "and in Table 2-6" to the end of the sentence after the words "specified in Table 2-5". This change appears necessary for consistency, since Table 2-5 is for intact fuel and Table 2-6 is for damaged fuel and TS 2.2.a refers to both intact and damaged fuel.

This information is needed to determine compliance with 10 CFR 72.11 and 72.236(a).

Response to TS-5

The suggested editorial changes are incorporated in the Technical Specifications.

TS-6 Modify TS 2.2.a and Footnote (2) of TS Tables 2-5 and 2-6 to remove the apparent inconsistency between TS 2.2.a and the footnote of these two TS tables.

The last sentence of TS 2.2.a states that fuel assembly poison rods may be stored in any intact or damaged assembly as long as the total assembly weight is less than that specified in TS Table 2-5 (for intact fuel) and Table 2-6 (for damaged fuel). This statement implies that the fuel assembly poison rods are included in the total assembly weight listed in the referenced tables. However, the Footnote (2) at the base of each table, as proposed in the amendment, indicates that the fuel assembly poison rods are not included in the total assembly weight stated in the tables. Thus, the referenced statement and footnote appear to be inconsistent. The statement and footnote should be modified to be consistent with each other and with the FSAR analyses regarding the maximum allowable weight in each basket assembly slot.

This information is needed to determine compliance with 10 CFR 72.236(a).

Response to TS-6

Technical Specification Tables 2-5 and 2-6 are revised to remove the apparent inconsistency as requested. The equivalent corresponding SAR Tables A.2.1-1 and A.2.1-2 are also revised to be consistent with the technical specification changes.

TS-7 Provide the units for the dimensions specified in TS Figure 4-1.

TS Figure 4-1 defines the minimum ligament width dimensions for the canister flux trap configuration. This information is important for criticality safety purposes, and the dimensions, including the units, should be consistent with the criticality analysis for both the 24PT1-DSC and the 24PT4-DSC. However, the units (e.g., inches) are not specified on the figure in the proposed amendment. Staff notes that the units were specified as inches in the TS for the initial certificate.

This information is needed to confirm compliance with 10 CFR 72.124(b).

Response to TS-7

The units used in TS Figure 4-1 are "inches". The TS Figure 4-1 is revised to add this unit to the dimensions.

TS-8 Modify the definition of intact fuel assembly to state:

Spent Nuclear FUEL ASSEMBLIES without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. *A fuel assembly with missing fuel rods shall not be classified as an intact fuel assembly unless solid filler (or replacement) rods are used to displace an amount of water equal to or greater than that displaced by the original fuel rods.*

(Note: Italics used to indicate where changes in the proposed condition text should be made.)

It is not clear from the definitions of intact fuel assembly and damaged fuel assembly whether assemblies in which missing rods have been replaced can or cannot be loaded as intact fuel. The criticality analysis only supports loading of such assemblies as intact fuel when the replacement rod displaces an amount of water equal to or greater than that displaced by the original rod. The loading of reconstituted fuel is also affected. It is not clear in the currently proposed definition of reconstituted fuel assembly that leaking fuel rods must be replaced by rods (stainless steel or intact fuel rods) that displace an equal, or greater, amount of water as displaced by the original leaking rod. However, TS 2.2.b states that reconstituted fuel assemblies are acceptable for storage in the 24PT4-DSC as intact assemblies. The applicant's criticality analysis only supports this TS loading condition for reconstituted fuel assemblies with replacement rods displacing an amount of water equal to or greater than that displaced by the original leaking fuel rods. If a leaking fuel rod is removed from an assembly, the assembly is considered to be missing a fuel rod; thus, to be classified as intact, a replacement rod must be inserted into the missing rod's lattice location that displaces the same, or greater, amount of water as the rod it has replaced. Therefore, the TS definition of intact fuel should be modified to be consistent with the criticality analysis.

This information is needed to determine compliance with 10 CFR 72.236(a) and (c).

Response to TS-8

The following definition is added for INTACT FUEL ASSEMBLY. *A fuel assembly with missing fuel rods shall not be classified as an intact fuel assembly unless solid filler (or replacement) rods are used to displace an amount of water equal to or greater than that displaced by the original fuel rods in the active fuel region of the fuel assembly.*

This definition is consistent with the assumptions used in the Criticality Analysis of the 24PT1 DSC (Chapter 6, Section 6.3.1 assumption number 6) and the 24PT4 DSC (Appendix A.6, Section A.6.3.1 2nd bullet under major assumptions). The analysis assumes reflective boundary conditions beyond the active fuel regions of the fuel assembly.

TS-9 Include a TS requirement and acceptance criteria for helium leak testing of all canister lid welds for both DSCs used in the Standardized Advanced NUHOMS system.

The TS currently do not require the leak testing of any of the canister lid welds. A weld must satisfy several conditions in order to be exempt from the helium leak test specified in ISG-18. One of those conditions is that the weld cannot have been executed under conditions where the root pass might have been subjected to pressurization from the helium fill in the canister. Note that credit may not be taken for mechanical closure devices (e.g., valve, or quick-disconnect) as these are assumed to permit helium leaks. Such leaks have occurred in the past and resulted in pinhole leaks through the weld. DSC vent and siphon port cover welds do not satisfy this condition (the DSC having been backfilled with helium prior to welding of the covers); therefore, leak testing per ANSI N-14.5 must be performed on these welds and appropriate acceptance criteria (based upon the SAR analyses) established. This testing and the acceptance criteria need to be included in the TS.

In order to be exempt from helium leak testing under the guidance of ISG-18, a weld must meet several conditions. First, the weld must not have been executed with helium backfill under the root pass. Additionally, there must be a minimum of 3 distinct weld passes, or layers. The root and cover passes must be Penetrant Test (PT) examined. The weld deposit thickness for all intermediate layers must be limited based upon critical flaw size (depth) analysis per the method of the ASME Code, Section XI, Subsection IWB-3600. When the intermediate layer weld deposit thickness limit is reached, a PT examination must be performed. Additional intermediate layers are executed in the same manner. If a multi-pass root is employed, a surface-connected flaw size analysis (IWB-3600) must be performed to limit root pass thickness before a PT examination must be performed. Also note that, as stated in Regulatory Guide 1.193, Code Case N-595 (any revision) is not endorsed by the NRC staff and is therefore not permitted as an alternative to the Code requirements.

Affected portions of the SAR and TS should be updated as necessary to comply with the guidance of ISG-18 and Regulatory Guide 1.193.

This information is needed to determine compliance with 10 CFR 72.122(a), 72.236(d) and 72.236(l).

Response to TS-9

A new proposed Technical Specification is added to TS 5.2.4 specifying helium leak testing of the top closure lid welds including vent and siphon port covers is added as requested. Operating procedures in SAR Chapter 8 for the 24PT1 DSC and Appendix A.8 for the 24PT4 DSC are revised to include the same leak testing requirements.

TS-10 Remove reference to use of Code Case N-595-1 provided on TS page 4.3.2.

The NRC has found Code Case N-595 (and any later revisions) unacceptable for use per Regulatory Guide 1.193.

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

Response to TS-10

The references to ASME Code Case N-595-1 are removed and the DSC Shell Assembly Alternates to the ASME Code, Subsection NB in TS 4.3.4 are revised to provide similar text to that used in CoC 1030 for the NUHOMS HD FSAR. This change also resulted in revisions to the UFSAR text in several chapters.

TS-11 Provide wording in the TS to incorporate a hydrogen gas monitoring or mitigation measure to be followed when performing any lid welding or cutting operations.

Hydrogen gas may be evolved during wet loading (or unloading) operations and must be monitored or controlled to preclude the possibility of creating a flammable mixture inside the canister during welding or cutting operations.

This information is needed to determine compliance with 10 CFR 72.236(a).

Response to TS-11

A new Technical Specification 4.2.6 is proposed as requested using the same format and requirements. UFSAR Chapters 8 and A.8 are revised to refer to this technical specification.

TS-12 Provide a TS requirement for controlling the manufacture and testing of all neutron poisons used in a spent fuel canister. The appropriate sections of the SAR, referencing other controlling or proprietary documents, may be incorporated by reference into the TS for brevity.

The TN NUHOMS HD Certificate of Compliance No. 1030, dated January 10, 2007, TS Section 4.3.1, may be used as a model of an appropriate TS requirement.

This information is needed to determine compliance with 10 CFR 72.236(b) and (c).

Response to TS-12

The 24PT1 and 24PT4 DSCs use only Boral as the poison material for criticality control. Requirements for Boral similar to those provided in CoC 1030 TS 4.3.1 are added to a proposed Technical Specification 4.2.3.

TS-13 Provide a TS requirement for the use of copper bearing corrosion resistant steel in the appropriate HSM components where the ISFSI is located near coastal marine environments.

TN has previously provided the following acceptable TS requirement wording to address this issue:

> If an independent spent fuel storage installation site is located in a coastal salt water marine atmosphere, then any load-bearing carbon steel DSC support structure rail components of any associated HSM-H shall be procured with a minimum 0.20 percent copper content for corrosion resistance.

This information is needed to determine compliance with 10 CFR 72.236(b).

Response to TS-13

A new Technical Specification 4.4.3.10 is proposed to incorporate this requirement.

TS-14 Modify the definition of damaged fuel to consider the structural integrity of the entire assembly, not simply the leak-tightness of the cladding and missing rods as with the currently proposed definition contained in the TS.

It is staff's understanding that TN's intent is to limit the loading of damaged spent fuel assemblies to only those assemblies with the type of damage specified in the damaged fuel definition. However, it is not clear from the current definitions that assemblies with other forms of damage, such as damage that impairs the assembly's structural integrity against geometric rearrangement of fuel or gross cladding failure, may not be loaded in the Standardized Advanced NUHOMS system either as intact fuel or at all. Therefore, the TS should be modified to explicitly prohibit loading of assemblies with these other forms of damage either as intact fuel or altogether. This issue can be addressed by modifying the damaged fuel definition to the following:

A Damaged Fuel Assembly is a fuel assembly with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, partial or missing rods, or missing structural components such as grid spacers, an assembly whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, or an assembly that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means are considered Fuel Debris.

The staff further notes that modifications to the TS should be based upon and supported by the appropriate SAR analyses.

This information is required for compliance with 10 CFR 72.236(a~d).

Response to TS-14

The definition of damaged fuel in CoC 1029 is different than the damaged fuel definition used in NUHOMS CoC 1004 and CoC 1030 where end caps are used for damaged fuel compartments. Each damaged fuel assembly stored in a 24PT1 or 24PT4 DSC is placed into a self contained can with an integral bottom and bolted lid that replaces the guide sleeve assembly in the basket as shown on Sheets 7 and 8 of UFSAR Drawing ANUH-01-4001. As such the 24PT4 DSC Failed Fuel Cans are designed to contain rod storage baskets, fuel debris and partial assemblies and are individually removable. The supporting analyses for storage of these damaged fuel components are consistent with the design of a failed fuel can and are provided in UFSAR Chapter A. 6.4.4. Therefore no changes are required to the definition of damaged fuel.

Certificate of Compliance

CoC-1 Modify the proposed CoC Condition No. 10 to have an appropriate heading, and Condition 10.a and 10.c that read as follows:

a. Standardized Advanced NUHOMS® systems that were previously fabricated and put into operation by general licensees in accordance with the original CoC or Amendment 1 *shall* continue to be used under the appropriate CoC or amendment.

c. AHSMs and 24PT1 DSCs fabricated *in accordance with* the original CoC may be *put into use* under Amendment 2 because no design changes are made to the design of the AHSM or 24PT1 DSC under Amendment 2.

(Note: Italics used to indicate where changes in the proposed condition text should be made.)

The applicant has proposed a new Condition No. 10 to the CoC that includes language to allow Standardized Advanced NUHOMS systems fabricated and "put into operation" under earlier amendments to be operated under the proposed amendment. Specifically, the applicant proposes allowance for use of 24PT4-DSCs fabricated in accordance with the Amendment 1 CoC and TS under the Amendment 2 CoC and TS and allowance for use of 24PT1-DSCs fabricated and put into operation in accordance with the initial CoC and TS under the Amendment 2 CoC and TS. Based upon the following, the staff finds the changes requested in this RAI question to the proposed condition to be necessary.

NRC regulations grant a general license to store spent fuel in an independent spent fuel storage installation (ISFSI) at reactor sites to Part 50 licensees, with the conditions of such a general license specified in 10 CFR 72.212. The regulation in 10 CFR 72.212 requires that licensees "perform written evaluations prior to use that establish that: (A) conditions set forth in the Certificate of Compliance have been met" [10 CFR 72.212(b)(2)(i)(A)]. The regulation states the evaluation is to be done "prior to use," meaning before the cask is loaded with spent fuel. The regulation in 10 CFR 72.212 also requires that any changes to the written evaluations be made in accordance with 10 CFR 72.48, which specifies that changes that would result in a change to the "terms, conditions, or specifications incorporated in the CoC" require a CoC amendment [10 CFR 72.48(c)(1)(ii)(B)].

The staff considers the approval of an amendment to a CoC as codifying a new design basis for a cask, requiring an NRC rulemaking before becoming effective. Therefore, each CoC amendment is considered a separate and distinct CoC accompanied, by its own technical specifications and safety evaluation report. A previously loaded cask, or a cask previously "put into operation," is bound by the terms, conditions and technical specifications of the CoC applicable to the cask at the time the licensee loaded the cask. Therefore, a licensee must obtain NRC approval if the licensee wishes to apply any changes of a later CoC amendment to a previously loaded cask, if such changes result in a change to terms or conditions of the CoC under which that cask was loaded.

This information is needed to determine compliance with 72.212(b)(2).

Response to CoC-1

The CoC condition is modified as requested.

Enclosure 2 to TN E-25857

List of UFSAR Pages associated with Amendment 2

Page 1.2-6	Amendment 2 Revision 1
Page 1.2-7	Amendment 2 Revision 1
Page 1.5-1	Amendment 2 Revision 1
Page 2.2-5	Amendment 2 Revision 1
Page 2.3-1	Amendment 2 Revision 1
Page 2.6-1	Amendment 2 Revision 1
Page 3.1-2	Amendment 2 Revision 1
Page 3.1-3	Amendment 2 Revision 1
Page 3.1-6	Amendment 2 Revision 1
Page 3.1-12	Amendment 2 Revision 1
Page 3.1-13	Amendment 2 Revision 1
Page 3.1-14	Amendment 2 Revision 1
Page 3.1-16	Amendment 2 Revision 1
Page 3.1-28	Amendment 2 Revision 1
Page 3.1-28A	Amendment 2 Revision 1
Page 3.4-3	Amendment 2 Revision 1
Page 3.6-58	Amendment 2 Revision 1
Page 3.6-60	Amendment 2 Revision 1
Page 4.2-5	Amendment 2 Revision 0
Page 4.4-25a	Amendment 2 Revision 0
Page 4.8-1	Amendment 2 Revision 1
Page 7.1-1	Amendment 2 Revision 1
Page 7.1-2	Amendment 2 Revision 1
Page 7.4-1	Amendment 2 Revision 1
Page 8.1-4	Amendment 2 Revision 1
Page 8.1-5	Amendment 2 Revision 1
Page 8.1-7	Amendment 2 Revision 1
Page 8.1-10	Amendment 2 Revision 1
Page 8.1-13	Amendment 2 Revision 1
Page 8.2-4	Amendment 2 Revision 1
Page 8.3-2	Amendment 2 Revision 1
Page 9.1-2	Amendment 2 Revision 1
Page 9.2-4	Amendment 2 Revision 0
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Page 10.1-3	Amendment 2 Revision 0
Page 11.3-1	Amendment 2 Revision 1
Page A.2.1-1	Amendment 2 Revision 1
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Page A.2.6-1	Amendment 2 Revision 1
Page A.3.1-2	Amendment 2 Revision 1
Page A.3.1-3	Amendment 2 Revision 1
Page A.3.1-4	Amendment 2 Revision 1
Page A.3.1-6	Amendment 2 Revision 1
Page A.3.1-11	Amendment 2 Revision 1
Page A.3.1-11A	Amendment 2 Revision 1
Page A.3.1-14	Amendment 2 Revision 1
Page A.3.4-1	Amendment 2 Revision 1
Page A.3.7-1	Amendment 2 Revision 1
Page A.4.2-3	Amendment 2 Revision 0
Page A.4.4-23	Amendment 2 Revision 0
Page A.4.11-1	Amendment 2 Revision 1
Page A.7.1-2	Amendment 2 Revision 1
Page A.7.1-3	Amendment 2 Revision 1
Page A.7.4-1	Amendment 2 Revision 1
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Page A.8.1-6	Amendment 2 Revision 1
Page A.8.1-7	Amendment 2 Revision 1
Page A.8.1-8	Amendment 2 Revision 1
Page A.8.1-10	Amendment 2 Revision 1
Page A.8.3-1	Amendment 2 Revision 1
Page A.9.1-1	Amendment 2 Revision 1

<u>Enclosure 3 to TN E-25857</u>

Amendment 2 Revision 1 Proposed changes to CoC 1029 (Amendment 1), the associated Technical Specifications, and the UFSAR (Revision 2)

NRC FORM 651
(10-2004)
10 CFR 72

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS

The U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the Code of Federal Regulations, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72). This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the cask design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified below.

Certificate No.	Effective Date	Expiration Date	Docket No.	Amendment No.	Amendment Effective Date	Package Identification No.
1029	02/05/03	02/05/23	72-1029	~~1~~ 2	~~05/16/05~~ TBD	USA/72-1029

Issued To: (Name/Address)

Transnuclear, Inc.
~~Four Skyline Drive~~
~~Hawthorne, New York 10532~~

7135 Minstrel Way, Suite 300
Columbia, Maryland 21405

Safety Analysis Report Title

Updated

Transnuclear, Inc., "Final Safety Analysis Report for the Standardized Advanced NUHOMS® Horizontal Modular Storage System ~~for Irradiated Nuclear Fuel~~"

CONDITIONS

This certificate is conditioned upon fulfilling the requirements of 10 CFR Part 72, as applicable, the attached Appendix A (Technical Specifications), and the conditions specified below:

1. CASK:

a. Model No. Standardized Advanced NUHOMS®-24PT1 and 24PT4

The two digits refer to the number of fuel assemblies stored in the dry shielded canister (DSC), the character P for pressurized water reactor (PWR) or B for boiling water reactor (BWR) is to designate the type of fuel stored, and T is to designate that the DSC is intended for transportation in a 10 CFR Part 71 approved package

b. Description U

Updated

The Standardized Advanced NUHOMS® System is certified as described in the Final Safety Analysis Report (FSAR) and in the U. S. Nuclear Regulatory Commission's (NRC's) Safety Evaluation Report (SER). The Standardized Advanced NUHOMS® System is a horizontal canister system composed of a steel dry shielded canister (DSC), a reinforced concrete advanced horizontal storage module (AHSM), and a transfer cask (TC). The Standardized Advanced NUHOMS® is similar to the Standardized NUHOMS® except that it has been enhanced to withstand high seismic spectra and to reduce radiological doses. The welded DSC provides confinement and criticality control for the storage and transfer of irradiated fuel. The concrete module provides radiation shielding while allowing cooling of the DSC and fuel by natural convection during storage. The TC is used for transferring the DSC from/to the Spent Fuel Pool Building to/from the AHSM.

CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS
Supplemental Sheet

1.b. Description (continued)

The principal component subassemblies of the DSC are the shell with integral bottom cover plate and shield plug and ram/grapple ring, top shield plug, top cover plate, and basket assembly. The shell length is fuel-specific. The internal basket assembly is composed of guide sleeves, support rods, and spacer disks. This assembly is designed to hold 24 PWR fuel assemblies. The basket assembly aids in the insertion of the fuel assemblies, enhances subcriticality during loading operations, and provides structural support during a hypothetical drop accident. The DSC is designed to slide from the transfer cask into the AHSM and back without undue galling, scratching, gouging, or other damage to the sliding surfaces.

The AHSM is a reinforced concrete unit with penetrations located at the top and front for air flow. The penetrations are protected from debris intrusions by wire mesh screens during storage operation. The DSC Support Structure, a structural steel frame with rails, is installed within the AHSM module to provide for sliding the DSC in and out of the AHSM and to support the DSC within the AHSM. AHSMs are arranged in arrays to minimize space and maximize self-shielding. Adjacent AHSMs are keyed and tied to provide maximum resistance to environmental conditions including high seismic loads.

The TC is used for transfer operations within the Spent Fuel Pool Building and for transfer operations to/from the AHSM. The TC is a cylindrical vessel with a bottom end closure assembly and a bolted top cover plate. Two upper lifting trunnions are located near the top of the cask for downending/uprighting and lifting of the cask in the Spent Fuel Pool Building. The lower trunnions, located near the base of the cask, serve as the axis of rotation during downending/uprighting operations and as supports during transport to/from the Independent Spent Fuel Storage Installation (ISFSI).

With the exception of the TC, fuel transfer and auxiliary equipment necessary for ISFSI operations are not included as part of the Standardized Advanced NUHOMS® System referenced in this Certificate of Compliance (CoC). Such site-specific equipment may include, but is not limited to, special lifting devices, the transfer trailer, and the skid positioning system

c. Drawings

The drawings for the Standardized Advanced NUHOMS® System are contained in Section 1 and A.1 of the FSAR. u

d. Basic Components

The basic components of ~~the Standardized~~ Standardized Advanced NUHOMS® System that are important to safety are the DSC, AHSM, and TC. These components are described in Section 2.5, A.2.5, and Table 2.5-1 and Table A.2.5-1 of the FSAR. u

2. OPERATING PROCEDURES

Written operating procedures ~~shall~~ be prepared for cask handling, loading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 8 and A.8 of the FSAR. u

3. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Written cask acceptance tests and maintenance program shall be prepared consistent with the technical basis described in Chapter 9 and A.9 of the FSAR. u

4. QUALITY ASSURANCE

Activities in the areas of design, purchase, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important to safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the cask system.

5. HEAVY LOADS REQUIREMENTS

Each lift of a DSC and TC must be made in accordance with the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant-specific safety review (under 10 CFR 50.59 or 10 CFR 72.48, if applicable) is required to show operational compliance with existing plant-specific heavy loads requirements.

6. APPROVED CONTENTS

Contents of the Standardized Advanced NUHOMS® System must meet the fuel specifications description as provided in the Appendix to this certificate.

7. DESIGN FEATURES

Features or characteristics for the site, cask, or ancillary equipment must be in accordance with the Appendix to this certificate.

8. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

A dry run training exercise of the loading, closure, handling, unloading, and transfer of the Standardized Advanced NUHOMS® System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the canister. The dry run may be performed in an alternate step sequence from the actual procedural guidelines in Chapter 8 and A.8 of the FSAR. The dry run shall include but not be limited to the following:

u

Loading Operations

a. Fuel Loading
b. DSC sealing, drying, and backfilling operations
c. TC downending and transport to the ISFSI
d. DSC transfer to the AHSM

NRC FORM 651A
(10-2004)
10 CFR 72

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS
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8. Cont.

Unloading Operations
a. DSC retrieval from AHSM
b. Flooding of DSC
c. Opening of DSC

9. AUTHORIZATION

The Standardized Advanced NUHOMS® System, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, and the attached Appendix A.

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

/RA/

Robert J. Lewis, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Attachments:

1. Appendix A: Technical Specifications

Dated: ~~May 31, 2005~~ TBD

10. SYSTEM FABRICATION AND USE

a. Standardized Advanced NUHOMS® systems that were previously fabricated and put into operation by general licensees in accordance with the original CoC or Amendment 1 shall continue to be used under the appropriate CoC or amendment.
b. Standardized Advanced NUHOMS® system components fabricated in accordance with Amendment 1 may be put into use under Amendment 2 because no changes are made to the Standardized Advanced NUHOMS® system design by Amendment 2.
c. AHSM's and 24PT1 DSC's fabricated in accordance with the original CoC may be put into use under Amendment 2 because no design changes are made to the design of the AHSM or 24PT1 DSC under Amendment 2.

APPENDIX A TO CERTIFICATE OF COMPLIANCE NO. 1029

TECHNICAL SPECIFICATIONS FOR THE ADVANCED NUHOMS® SYSTEM

OPERATING CONTROLS AND LIMITS

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OPERATING CONTROLS AND LIMITS

1.0 Use and Application

1.1 Definitions

-- NOTE --

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required ACTIONS to be taken under designated Conditions within specified Completion Times.
ADVANCED HORIZONTAL STORAGE MODULE (AHSM)	The AHSM is a reinforced concrete structure for storage of a loaded 24PT1-DSC or 24PT4-DSC (DSC) at a spent fuel storage facility
DAMAGED FUEL ASSEMBLY	A DAMAGED FUEL ASSEMBLY is a FUEL ASSEMBLY with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial or missing rods.
DRY SHIELDED CANISTER (DSC)	A 24PT1-DSC or 24PT4-DSC is a welded pressure vessel that provides confinement of INTACT or DAMAGED FUEL ASSEMBLIES in an inert atmosphere.
FAILED FUEL CAN	A FAILED FUEL CAN confines any loose material and gross fuel particles to a known, subcritical volume during normal, off-normal and accident conditions and facilitates handling and retrievability.
FUEL DEBRIS	An intact or partial fuel rod not contained in a FUEL ASSEMBLY grid or an individual intact or partial fuel pellet not contained in a fuel rod. FUEL DEBRIS may be inserted in a ROD STORAGE BASKET.
INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)	The facility within a perimeter fence licensed for storage of spent fuel within AHSMs.

INTACT FUEL ASSEMBLY	Spent Nuclear FUEL ASSEMBLIES without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. *A fuel assembly with missing fuel rods shall not be classified as an intact fuel assembly unless solid filler (or replacement) rods are used to displace an amount of water equal to or greater than that displaced by the original fuel rods in the active fuel region of the fuel assembly.*
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on a DSC while it is being loaded with INTACT or DAMAGED FUEL ASSEMBLIES, and on a TRANSFER CASK while it is being loaded with a DSC containing INTACT or DAMAGED FUEL ASSEMBLIES. LOADING OPERATIONS begin when the first INTACT or DAMAGED FUEL ASSEMBLY is placed in the DSC and end when the TRANSFER CASK is ready for TRANSFER OPERATIONS.
RECONSTITUTED FUEL ASSEMBLY	RECONSTITUTED FUEL ASSEMBLIES include assemblies in which leaking fuel rods are replaced with either stainless steel rods or intact fuel rods prior to return to the reactor. RECONSTITUTED FUEL ASSEMBLIES may contain from one to eight stainless steel rods per assembly.
ROD STORAGE BASKET	A 9x9 array of tubes in a lattice that has approximately the same dimensions as a standard FUEL ASSEMBLY.
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the ISFSI while a DSC containing INTACT or DAMAGED FUEL ASSEMBLIES is located in an AHSM on the storage pad within the ISFSI perimeter.
TRANSFER CASK (TC)	The TRANSFER CASK will consist of a licensed NUHOMS® OS197 or OS197H onsite transfer cask. The TRANSFER CASK will be placed on a transfer trailer for movement of a DSC to the AHSM.
TRANSFER OPERATIONS	TRANSFER OPERATIONS include all licensed activities involving the movement of a TRANSFER CASK loaded with a DSC containing INTACT or DAMAGED FUEL ASSEMBLIES. TRANSFER OPERATIONS begin when the TRANSFER CASK is placed on the transfer trailer following LOADING OPERATIONS and end when the

	DSC is located in an AHSM on the storage pad within the ISFSI perimeter.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on a DSC to unload INTACT or DAMAGED FUEL ASSEMBLIES. UNLOADING OPERATIONS begin when the DSC is removed from the AHSM and end when the last INTACT or DAMAGED FUEL ASSEMBLY has been removed from the DSC.

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors:

EXAMPLE 1.2-1:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Stop . . . OR *A.2* A.2.1 Verify . . . AND *A.2.2* A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Remove . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

Limiting Conditions for Operation (LCOs) specify the lowest functional capability or performance levels of equipment required for safe operation of the facility. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO are not met. Specified with each stated Condition are Required Action(s) and Completion Times(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the facility is in a specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the facility is not within the LCO Applicability.

Once a Condition has been entered, subsequent subsystems, components, or variables expressed in the Condition, discovered to be not within limits, will not result in separate entry into the Condition unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions:

EXAMPLE 1.3-1:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1. AND B.2 Perform Action B.2	12 hours 36 hours

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to complete action B.1 within 12 hours AND complete action B.2 within 36 hours. A total of 12 hours is allowed for completing action B.1 and a total of 36 hours (not 48 hours) is allowed for completing action B.2 from the time that Condition B was entered. If action B.1 is completed within 6 hours, the time allowed for completing action B.2 is the next 30 hours because the total time allowed for completing action B.2 is 36 hours.

EXAMPLES (continued)

EXAMPLE 1.3-2:

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One system not within limit	A.1 Restore system to within limit.	7 days
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1. AND B.2 Perform Action B.2.	12 hours 36 hours

When a system is determined to not meet the LCO, Condition A is entered. If the system is not restored within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the system is restored after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

EXAMPLES (continued)

EXAMPLE 1.3-3:

ACTIONS

-------------------------NOTE-------------------------

Separate Condition entry is allowed for each component.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LCO not met.	A.1 Restore compliance with LCO.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Perform Action B.1. AND B.2 Perform Action B.2.	6 hours 12 hours

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each component, and Completion Times tracked on a per component basis. When a component is determined to not meet the LCO, Condition A is entered and its Completion Time starts. If subsequent components are determined to not meet the LCO, Condition A is entered for each component and separate Completion Times start and are tracked for each component.

IMMEDIATE COMPLETION TIME

When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated Limiting Condition for Operation (LCO). An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "Specified Frequency" is referred to throughout this section and each of the Specifications of Section 3, *"Limiting Condition for Operations (LCO) and* Surveillance Requirement (SR) Applicability.*"* The "Specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With a SR satisfied, *SR 3.0.4* imposes no restriction.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified:

EXAMPLE 1.4-1:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify Pressure within limit.	12 hours

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is determined to not meet the LCO, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the facility is in a condition specified in the Applicability of the LCO, the LCO is not met in accordance with SR 3.0.1.

If the interval as specified by SR 3.0.2 is exceeded while the facility is not in a condition specified in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the specified condition. Failure to do so would result in a violation of SR 3.0.4.

EXAMPLES (continued)

EXAMPLE 1.4-2:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours prior to starting activity AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one-time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time the example activity is to be performed, the Surveillance must be performed prior to starting the activity.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2.

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If the specified activity is canceled or not performed, the measurement of both intervals stops. New intervals start upon preparing to restart the specified activity.

EXAMPLES (continued)

EXAMPLE 1.4-3:

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------NOTE------- Not required to be met until 96 hours after verifying the helium leak rate is within limit. Verify 24PT1-DSC vacuum drying pressure is within limit.	Once after verifying the helium leak rate is within limit.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the vacuum drying pressure not be met immediately following verification of the helium leak rate while in LOADING OPERATIONS, this Note allows 96 hours to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency."

Once the helium leak rate has been verified to be acceptable, 96 hours, plus the extension allowed by SR 3.0.2, would be allowed for completing the Surveillance for the vacuum drying pressure. If the Surveillance was not performed within this 96 hour interval, there would then be a failure to perform the Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

2.0 Functional and Operating Limits

2.1 Fuel To Be Stored In The 24PT1-DSC

The spent nuclear fuel to be stored in each 24PT1-DSC/AHSM at the ISFSI shall meet the following requirements:

a. Fuel shall be INTACT FUEL ASSEMBLIES or DAMAGED FUEL ASSEMBLIES. DAMAGED FUEL ASSEMBLIES shall be placed in screened confinement cans (FAILED FUEL CANS) inside the 24PT1-DSC guidesleeves. DAMAGED FUEL ASSEMBLIES shall be stored in outermost guidesleeves located at the 45, 135, 225 and 315 degree azimuth locations.

b. Fuel types shall be limited to the following:

   UO_2 Westinghouse 14x14 (WE 14x14) Assemblies (with or without IFBA fuel rods), as specified in Table 2-1.

   WE 14x14 Mixed Oxide (MOX) Assemblies, as specified in Table 2-1.

   Fuel burnup and cooling time is to be consistent with the limitations specified in Table 2-4 for UO2 fuel.

   Control Components stored integral to WE 14x14 Assemblies in a 24PT1-DSC, shall be limited to Rod Cluster Control Assemblies (RCCAs), Thimble Plug Assemblies (TPAs), and Neutron Source Assemblies (NSAs). Location of control components within a 24PT1-DSC shall be selected based on criteria which does not change the radial center of gravity by more than 0.1 inches.

c. The maximum heat load for a single FUEL ASSEMBLY, including control components, is 0.583 kW for SC FUEL ASSEMBLIES and 0.294 kW for MOX FUEL ASSEMBLIES. The maximum heat load per 24PT1-DSC, including any integral Control Components, shall not exceed 14 kW when loaded with all SC FUEL ASSEMBLIES and 13.706 kW when loaded with MOX FUEL ASSEMBLIES.

d. Fuel can be stored in the 24PT1-DSC in any of the following configurations:

   1) A maximum of 24 INTACT WE 14x14 MOX or SC FUEL ASSEMBLIES; or

   2) Up to four WE 14x14 SC DAMAGED FUEL ASSEMBLIES, with the balance INTACT WE 14x14 SC FUEL ASSEMBLIES; or

   3) One MOX DAMAGED FUEL ASSEMBLY with the balance INTACT WE 14x14 SC FUEL ASSEMBLIES.

A 24PT1-DSC containing less than 24 FUEL ASSEMBLIES may contain dummy FUEL ASSEMBLIES in FUEL ASSEMBLY slots. The dummy FUEL ASSEMBLIES are unirradiated, stainless steel encased structures that

approximate the weight and center of gravity of a FUEL ASSEMBLY. The effect of dummy assemblies or empty FUEL ASSEMBLY slots on the radial center of gravity of the DSC must meet the requirements of Section 2.1.b.

No more than two empty FUEL ASSEMBLY slots are allowed in each DSC. They must be located at symmetrical locations about the 0-180° and 90-270° axes.

No more than 14 fuel pins in each assembly may exhibit damage. A visual inspection of assemblies will be performed prior to placement of the fuel in the 24PT1-DSC, which may then be placed in storage or transported anytime thereafter without further fuel inspection.

e. Fuel dimensions and weights are provided in Table 2-2.

f. The maximum neutron and gamma source terms are provided in Table 2-3.

2.2 Fuel to Be Stored in the 24PT4-DSC

a. The spent fuel to be stored in the NUHOMS® 24PT4-DSC consists of INTACT (including RECONSTITUTED) Westinghouse-CENP 16x16 (CE 16x16) and/or DAMAGED CE 16x16 FUEL ASSEMBLIES with Zircaloy or ZIRLO™ cladding and UO_2, $(U, Er)O_2$ or $(U, Gd)O_2$ fuel pellets. Assemblies are with or without integral burnable poison rods or integral fuel burnable absorber (IFBA) rods. *Fuel Assembly poison rods installed within the guide tubes for criticality control in the spent fuel pool racks may be stored with any INTACT FUEL ASSEMBLY or DAMAGED FUEL ASSEMBLY as long as the total assembly weight is less than that specified in Table 2-5 and in Table 2-6.*

b. Each 24PT4-DSC can accommodate a maximum of 12 DAMAGED FUEL ASSEMBLIES, with the remaining assemblies being intact.

RECONSTITUTED ASSEMBLIES containing up to eight replacement stainless steel rods in place of DAMAGED FUEL Rods or replacement Zircaloy clad uranium rods (any number per assembly) are acceptable for storage in the 24PT4-DSC as either INTACT or DAMAGED ASSEMBLIES.

DAMAGED FUEL may include assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial and/or missing rods (i.e., extra water holes). DAMAGED FUEL ASSEMBLIES shall be encapsulated in individual FAILED FUEL CANS placed in locations as shown in Figure 2-4.

FUEL DEBRIS and DAMAGED FUEL Rods that have been removed from a DAMAGED FUEL ASSEMBLY and placed in a ROD STORAGE BASKET are also considered as DAMAGED FUEL. A ROD STORAGE BASKET is a 9x9 array of tubes in a lattice that has approximately the same dimensions as a standard FUEL ASSEMBLY. ROD STORAGE BASKETS may also include IFBA and Integral Burnable Poison Rods. Loose FUEL DEBRIS not contained in a ROD STORAGE BASKET may be placed in a FAILED FUEL CAN for storage provided the size of the debris is larger than the FAILED FUEL CAN screen mesh opening. FUEL DEBRIS may be associated with any type of UO_2 fuel provided that the maximum uranium content and enrichment limits are met.

c. The INTACT and/or DAMAGED CE 16x16 FUEL ASSEMBLIES acceptable for storage in 24PT4-DSC are specified in Table 2-5, Table 2-6, and Table 2-7. The fuel to be stored in the 24PT4-DSC is limited to a maximum initial enrichment of 4.85 wt. % ^{235}U. The maximum allowable assembly burnup is given as a function of initial fuel enrichment but does not exceed 60,000 MWd/MTU. The minimum cooling time is 5 years.

d. A 24PT4-DSC containing less than 24 FUEL ASSEMBLIES may contain dummy FUEL ASSEMBLIES in FUEL ASSEMBLY slots, or empty slots. The dummy FUEL ASSEMBLIES are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a FUEL ASSEMBLY.

e. The 24PT4-DSC may store PWR assemblies in any one of the three alternate configurations shown in Figure 2-1 through Figure 2-3 with a maximum heat load of 1.26 kW per assembly and a maximum heat load of 24 kW per DSC. Table 2-9 through Table 2-12 define the FUEL ASSEMBLY cooling time (in years) based on FUEL ASSEMBLY burnup and initial fuel enrichment for the assembly, assuming that no reconstituted fuel with stainless steel rods is present. The fuel qualification tables to be used for reconstituted assemblies with stainless steel rods are provided in Table 2-13 through Table 2-16. These tables ensure that the FUEL ASSEMBLY decay heat load is less than that specified for each table and that the corresponding radiation source term is bounded by that analyzed in Chapter A.5.

f. Two different 24PT4-DSC basket configurations are provided, as shown in Table 2-8. These configurations differ in the boron loading in the Boral® plates. The minimum areal boron –10 (^{10}B) concentrations for the standard (Type A basket) and high (Type B basket) loadings are 0.025 and 0.068 g/cm^2, respectively. Fuel to be stored in the standard ^{10}B loading 24PT4-DSC is limited to an initial ^{235}U enrichment of 4.1 wt. %. Fuel to be stored in the high ^{10}B loading 24PT4-DSC is limited to an initial ^{235}U enrichment of 4.85 wt. %.

g. Up to four DAMAGED FUEL ASSEMBLIES may be stored in a 24PT4-DSC of either ^{10}B loading without impact upon the maximum allowed ^{235}U enrichment and without the use of additional poison rodlets. The DAMAGED ASSEMBLIES shall be stored in FAILED FUEL CANS located at the 45, 135, 225 and 315 degree azimuth locations (Zone A of Figure 2-4).

Five to twelve DAMAGED FUEL ASSEMBLIES may be stored in a 24PT4-DSC of either ^{10}B loading without the use of poison rodlets if the maximum allowed ^{235}U enrichment is reduced for the DAMAGED ASSEMBLIES. The intact assembly enrichment limits remain at their nominal values of 4.1 and 4.85 wt. % for the standard and high ^{10}B loadings, respectively. DAMAGED FUEL to be stored in the standard ^{10}B loading 24PT4-DSC is limited to an initial ^{235}U enrichment of 3.7 wt. %, and DAMAGED FUEL to be stored in the high ^{10}B loading 24PT4-DSC is limited to an initial ^{235}U enrichment of 4.1 wt. %. All DAMAGED ASSEMBLIES shall be stored in FAILED FUEL CANS located in Zones A and B of Figure 2-4.

Five to twelve DAMAGED FUEL ASSEMBLIES may be stored in a 24PT4-DSC of either ^{10}B loading without impact upon the maximum allowed ^{235}U enrichment if poison rodlets are utilized. For the standard ^{10}B loading, a single poison rodlet is inserted into the center guide tube of each INTACT FUEL ASSEMBLY located in Zone C of Figure 2-4. For the high ^{10}B loading, a poison rodlet is inserted into each of the five guide tubes in each INTACT FUEL ASSEMBLY located in Zone C of Figure 2-4. All DAMAGED ASSEMBLIES shall be stored in FAILED FUEL CANS located in Zones A and B of Figure 2-4.

The poison rodlets consist of B_4C (pellets or powder) encased in a 0.75" nominal OD stainless steel tube with a wall thickness of 0.035". The minimum linear B_4C *content is 0.70 g/cm with sufficient length to cover the active fuel length. Each poison rodlet may include a lifting mechanism to allow insertion into the selected SFA guide tube.*

A summary of the storage configurations analyzed is presented in Table 2-8.

2.3 Functional and Operating Limits Violations

If any Functional and Operating Limit of 2.1 *or 2.2* is violated, the following actions shall be completed:

a. The affected FUEL ASSEMBLIES shall be placed in a safe condition.

b. Within 24 hours, notify the NRC Operations Center.

c. Within 30 days, submit a special report which describes the cause of the violation and the actions taken to restore compliance and prevent recurrence.

Table 2-1 Fuel Specifications (24PT1-DSC)

Fuel Type	Maximum Initial Enrichment	Cladding Material	Minimum Cooling Time	Minimum Initial Enrichment	Maximum Burnup
UO_2 WE 14x14 (with or without IFBA fuel rods)	4.05 weight % U-235	Type 304 Stainless Steel	10 years	See Table 2-4 for Enrichment, Burnup, and Cooling Time Limits.	
WE 14x14 MOX	2.84 weight % Fissile Pu - 64 rods 3.10 weight % Fissile Pu - 92 rods 3.31 weight % Fissile Pu - 24 rods	Zircalloy-4	20 years	2.78 weight % Fissile Pu –64 rods 3.05 weight % Fissile Pu –92 rods 3.25 weight % Fissile Pu – 24 rods	25,000 MWd/MTU
Integral Control Components	N/A	N/A	10 years	N/A	N/A

Table 2-2 Fuel Dimension and Weights (24PT1-DSC)

Parameter	WE 14x14 SC(1)	WE 14x14 MOX(1)
Number of Rods	180	180
Number of Guide Tubes/Instrument Tubes	16	16
Cross Section (in)	7.763	7.763
Unirradiated Length (in)	138.5	138.5
Fuel Rod Pitch (in)	0.556	0.556
Fuel Rod O.D. (in)	0.422	0.422
Clad Material	Type 304 SS	Zircaloy-4
Clad Thickness (in)	0.0165	0.0243
Pellet O.D. (in)	0.3835	0.3659
Max. initial ^{235}U Enrichment (%wt)	4.05	Note 2
Theoretical Density (%)	93-95	91
Active Fuel Length (in)	120	119.4
Max. U Content (kg)	375	Note 3
Assembly Weight (lbs)	1210	1150
Max. Assembly Weight incl. NFAH(4) (lbs)	1320	1320

(1) Nominal values shown unless stated otherwise
(2) Mixed-Oxide assemblies with 0.71 weight % U-235 and maximum fissile Pu weight of 2.84 weight % (64 rods), 3.10 weight % (92 rods), and 3.31 weight % (24 rods)
(3) Total weight of Pu is 11.24 kg and the total weight of U is 311.225 kg
(4) Weights of TPAs and NSAs are enveloped by RCCAs

Table 2-3 Maximum Neutron and Gamma Source Terms (24PT1-DSC)

Parameter	WE 14x14 SC	WE 14x14 MOX
Gamma Source (γ/sec/assy)	3.43E+15	9.57E+14
Neutron Source (n/sec/assy)	2.84E+08	4.90E+07

Parameter	RCCAs	TPAs	NSAs
Gamma Source (γ/sec/assy)	7.60E+12	5.04E+12	1.20E+13
Decay heat (Watts)	1.90	1.2	1.66

Table 2-4 Fuel Qualification Table (24PT1-DSC)

(Minimum required years of cooling time after reactor core discharge)

Burnup GWd/MTU	Initial Enrichment (weight % U-235)			
	3.12	3.36	3.76	3.96
45.0	Not Analyzed		15.2	15.2*
43.3			15.2	11.5
40.0		10.9	10.9**	10.9**
36.8		10.9	10.0***	10.0***
35.0 or less	10.0***	10.0***	10.0***	10.0***

Notes
* Cooling time based on 3.76 weight % enrichment is conservatively used.
** Cooling time based on 3.36 weight % enrichment is conservatively used.
*** Cooling time based on shielding analysis source term.

General Notes:
- Use burnup and enrichment to look up minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Example: An assembly with an initial enrichment of 3.90 w/o U-235 and a burnup of 37 GWd/MTU is acceptable for storage after a 10.9 year cooling time as defined at the intersection of 3.76 weight % U-235 (rounding down) and 40 GWd/MTU (rounding up) on the qualification table.

Table 2-5 PWR Fuel Specification of Intact Fuel to be stored in NUHOMS® 24PT4-DSC

Fuel Design:	INTACT CE 16x16 PWR FUEL ASSEMBLY or equivalent reload fuel that is enveloped by the FUEL ASSEMBLY design characteristics as listed in Table 2-7 and the following requirements:
Fuel Damage:	Fuel with known or suspected cladding damage in excess of pinhole leaks or hairline cracks or an assembly with partial and/or missing rods is not authorized to be stored as "INTACT PWR FUEL."
Physical Parameters(1)	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500(2)(3)
No. of Assemblies per DSC	≤24 intact assemblies
Max. U Content (kg)	455.5
Fuel Cladding	Zircaloy-4 or ZIRLO™
RECONSTITUTED FUEL ASSEMBLIES	DAMAGED FUEL Rods replaced by either stainless steel rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly)
Nuclear and Radiological Parameters	
Maximum Initial ^{235}U Enrichment (wt %)	Per Table 2-8 and Figure 2-4
Fuel Burnup and Cooling Time	Per Table 2-9, Table 2-10, Table 2-11 and Table 2-12 For RECONSTITUTED FUEL with stainless steel replacement rods per Table 2-13, Table 2-14, Table 2-15 and Table 2-16
Decay Heat	Per Figure 2-1, Figure 2-2 or Figure 2-3

Notes:

(1) Nominal values shown unless stated otherwise.
(2) Does not include weight of *fuel assembly* Poison Rodlets (25 lbs each) *installed in accordance with Table 2-8.*
(3) Includes weight of fuel assembly Poison Rods installed for 10CFR50 criticality control in spent fuel pool racks.

Table 2-6 PWR Fuel Specifications of DAMAGED FUEL to be Stored in NUHOMS® 24PT4-DSC

Fuel Design:	DAMAGED CE 16x16 PWR FUEL ASSEMBLY or equivalent reload fuel that is enveloped by the FUEL ASSEMBLY design characteristics as listed in Table 2-7 and the following requirements:
Fuel Damage:	DAMAGED FUEL may include assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial and/or missing rods (i.e., extra water holes). DAMAGED FUEL ASSEMBLIES shall be encapsulated in individual FAILED FUEL CANS and placed in Zones A and/or B as shown in Figure 2-4. FUEL DEBRIS and DAMAGED FUEL Rods that have been removed from a DAMAGED FUEL ASSEMBLY and placed in a ROD STORAGE BASKET are also considered as DAMAGED FUEL. Loose FUEL DEBRIS not contained in a ROD STORAGE BASKET, may also be placed in a FAILED FUEL CAN for storage, provided the size of the debris is larger than the FAILED FUEL CAN screen mesh opening. FUEL DEBRIS may be associated with any type of UO_2 fuel provided that the maximum uranium content and enrichment limits are met.
Physical Parameters(1)	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500(2)(3)
No. of Assemblies per DSC	≤12 DAMAGED ASSEMBLIES, balance INTACT
Max. U Content (kg)	455.5
Fuel Cladding	Zircaloy-4 or ZIRLO™
RECONSTITUTED FUEL ASSEMBLIES	DAMAGED FUEL Rods replaced by either stainless steel rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly)
Nuclear and Radiological Parameters	
Initial ^{235}U Enrichment (wt %)	Per Table 2-8 and Figure 2-4
Fuel Burnup and Cooling Time	Per Table 2-9, Table 2-10, Table 2-11 and Table 2-12 For RECONSTITUTED FUEL with stainless steel replacement rods per Table 2-13, Table 2-14, Table 2-15 and Table 2-16
Decay Heat	Per Figure 2-1, or Figure 2-2 or Figure 2-3

Notes:

(1) Nominal values shown unless stated otherwise.

(2) Does not include weight of *fuel assembly* Poison Rodlets (25 lbs each) *installed in accordance with Table 2-8.*

(3) Includes the weight of fuel assembly Poison Rods installed for 10CFR50 criticality control in spent fuel pool racks.

Table 2-7 PWR Fuel Assembly Design Characteristics (24PT4-DSC)

Assembly Class	CE 16x16(1)
Assembly Length	Table 2-5 or Table 2-6
Max. Initial ^{235}U Enrichment (wt %)	4.85
Fissile Material	UO_2, or $(U, Er)O_2$, or $(U, Gd)O_2$
Number of Rods	236
Fuel Rod Pitch (in)	0.506
Fuel Rod O.D. (in)	0.382
Clad Thickness (in)	0.025
Nominal Pellet O.D., (in)	0.3255(2)
Number of Guide Tubes	5

Notes:

(1) Nominal values shown unless stated otherwise.

(2) Bounds pellets with a nominal OD of 0.325".

Table 2-8 Maximum Fuel Enrichment v/s Neutron Poison Requirements for 24PT4-DSC

Storage Configuration	Maximum No. of DAMAGED FUEL ASSEMBLIES[1]	Maximum ^{235}U Fuel Enrichment (wt %)	DSC Basket, Minimum BORAL® Areal Density (gm/cm^2)	Minimum No. of Poison Rodlets Required[2]
All INTACT FUEL ASSEMBLIES	0	4.1	.025 (Type A Basket)	0
	0	4.85	.068 (Type B Basket)	0
Combination of DAMAGED and INTACT FUEL ASSEMBLIES	4	4.1	.025 (Type A Basket)	0
	4	4.85	.068 (Type B Basket)	0
	12	3.7 (DAMAGED) 4.1 (INTACT)	.025 (Type A Basket)	0
	12	4.1 (DAMAGED) 4.85 (INTACT)	.068 (Type B Basket)	0
	12	4.1	.025 (Type A Basket)	1[2] (Located in center guide tube of each INTACT ASSEMBLY)
	12	4.85	.068 (Type B Basket)	5[2] (Located in all five guide tubes of each INTACT ASSEMBLY)

Notes:

(1) See Figure 2-4 for location of DAMAGED FUEL ASSEMBLIES within the 24PT4-DSC basket (Zones A and/or B only).

(2) Poison rodlets are only required for a specific DSC configuration with a payload of 5-12 DAMAGED ASSEMBLIES in combination with maximum fuel enrichment levels as shown. The poison rodlets are to be located within the guide tubes of the inner Zone C INTACT ASSEMBLIES as shown in Figure 2-4.

Table 2-9 PWR Fuel Qualification Table for 1.26 kW per Assembly for the NUHOMS® 24PT4-DSC (Minimum required years of cooling time after reactor core discharge)

BU (GWd MTU)	Initial Enrichment																													
	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
15	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
20	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
25	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
28	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
30	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
32	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
34	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
36	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
38											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
39											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
40											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
41			Not Analyzed								5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
42											6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
43																5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
44																6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5
45																					5	5	5	5	5	5	5	5	5	5	5
48																					6	6	6	6	6	6	6	6	6	6	6
51																					7	7	7	7	6	6	6	6	6	6	6
54																					7	7	7	7	7	7	7	7	7	7	7
57																					8	8	8	8	8	8	8	8	8	8	8
60																					9	9	9	9	9	9	9	9	9	9	9

Notes:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- This table does not apply to RECONSTITUTED FUEL ASSEMBLIES with stainless steel rods.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 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 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a five-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-10 PWR Fuel Qualification Table for 1.2 kW per Assembly for the NUHOMS® 24PT4-DSC (Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU)	Initial Enrichment																													
	1.8	1.9	2	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
15	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
20	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
25	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
28	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
30	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
32	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
34	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
36	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
38											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
39											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
40											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
41			Not Analyzed								5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
42											6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
43																6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5
44																6	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5
45																					6	6	6	6	6	6	6	6	6	6	6
48																					6	6	6	6	6	6	6	6	6	6	6
51																					7	7	7	7	7	7	7	7	7	7	7
54																					8	8	8	8	8	8	8	8	8	7	7
57																					9	9	9	9	9	9	9	9	8	8	8
60																					10	10	10	10	10	10	10	10	10	10	9

Notes:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- This table does not apply to RECONSTITUTED FUEL ASSEMBLIES with stainless steel rods.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 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 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a five-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-11 PWR Fuel Qualification Table for 1.0 kW per Assembly for the NUHOMS® 24PT4-DSC
(Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU)	Initial Enrichment																														
	1.8	1.9	2	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
15	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
20	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
25	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
28	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
30	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
32	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
34	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
36	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
38											6	6	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5
39											6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	5	5	5
40											6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
41				Not Analyzed							6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
42											7	7	7	7	7	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
43																7	7	7	7	7	7	7	6	6	6	6	6	6	6	6	6
44																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	6
45																					7	7	7	7	7	7	7	7	7	7	7
48																					8	8	8	8	8	8	8	8	8	8	8
51																					9	9	9	9	9	9	9	9	9	9	9
54																					11	11	11	11	11	10	10	10	10	10	10
57																					13	13	13	13	12	12	12	12	12	12	12
60																					15	15	15	15	15	15	15	14	14	14	14

Notes:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- This table does not apply to RECONSTITUTED FUEL ASSEMBLIES with stainless steel rods.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 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 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a six-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-12 PWR Fuel Qualification Table for 0.9 kW per Assembly for the NUHOMS® 24PT4-DSC (Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU)	Initial Enrichment																													
	1.8	1.9	2	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
15	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
20	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
25	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
28	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
30	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
32	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
34	6	6	6	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
36	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	5	5	5
38											6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
39											7	7	7	7	7	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
40											7	7	7	7	7	7	7	7	7	7	7	7	7	7	6	6	6	6	6	6	6
41			Not Analyzed								7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
42											8	8	8	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
43																8	8	8	8	8	7	7	7	7	7	7	7	7	7	7	7
44																8	8	8	8	8	8	8	8	8	8	8	8	8	8	7	7
45																					8	8	8	8	8	8	8	8	8	8	8
48																					10	10	9	9	9	9	9	9	9	9	9
51																					11	11	11	11	11	11	11	11	11	11	11
54																					14	14	14	13	13	13	13	13	13	13	13
57																					17	16	16	16	16	16	16	16	15	15	15
60																					20	19	19	19	19	19	19	18	18	18	18

Notes:

- BU = Assembly average burnup.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- This table does not apply to RECONSTITUTED FUEL ASSEMBLIES with stainless steel rods.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 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 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a seven-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-13 PWR Fuel Qualification Table for 1.26 kW per Assembly for the NUHOMS® 24PT4-DSC, Reconstituted Fuel with Stainless Steel Rods

(Minimum required years of cooling time after reactor core discharge)

BU (GWd/	Initial Enrichment																														
MTU)	1.8	1.9	2	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
15	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
20	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
25	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
28	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
30	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
32	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
34	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
36	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
38											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
39											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
40											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
41				Not Analyzed							7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
42											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
43																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
44																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
45																					7	7	7	7	7	7	7	7	7	7	7
48																					8	8	8	8	8	8	8	8	8	8	8
51																					8	8	8	8	8	8	8	8	8	8	8
54																					9	9	9	9	9	9	9	9	9	9	9
57																					10	10	10	10	10	10	10	10	10	10	10
60																					12	12	12	12	12	12	12	12	12	12	12

Notes:

- BU = Assembly average burnup.
- This table is to be used only for RECONSTITUTED FUEL ASSEMBLIES.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 7-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a seven-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-14 PWR Fuel Qualification Table for 1.2 kW per Assembly for the NUHOMS® 24PT4-DSC, Reconstituted Fuel with Stainless Steel Rods

(Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU)	Initial Enrichment																													
	1.8	1.9	2	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
15	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
20	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
25	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
28	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
30	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
32	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
34	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
36	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
38											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
39											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
40											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
41					Not Analyzed						7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
42											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
43																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
44																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
45																					7	7	7	7	7	7	7	7	7	7	7
48																					8	8	8	8	8	8	8	8	8	8	8
51																					8	8	8	8	8	8	8	8	8	8	8
54																					9	9	9	9	9	9	9	9	9	9	9
57																					10	10	10	10	10	10	10	10	10	10	10
60																					12	12	12	12	12	12	12	12	12	12	12

Notes:

- BU = Assembly average burnup.
- This table is to be used only for RECONSTITUTED FUEL ASSEMBLIES.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 7-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a seven-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 2-15 PWR Fuel Qualification Table for 1.0 kW per Assembly for the NUHOMS® 24PT4-DSC, Reconstituted Fuel with Stainless Steel Rods

(Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU)	Initial Enrichment																													
	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
15	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
20	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
25	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
28	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
30	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
32	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
34	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
36	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
38											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
39											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
40											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
41				Not Analyzed							7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
42											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
43																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
44																7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
45																					7	7	7	7	7	7	7	7	7	7	7
48																					8	8	8	8	8	8	8	8	8	8	8
51																					9	9	9	9	9	9	9	9	9	9	9
54																					11	11	11	11	11	10	10	10	10	10	10
57																					13	13	13	13	12	12	12	12	12	12	12
60																					15	15	15	15	15	15	15	14	14	14	14

Notes:

- BU = Assembly average burnup.
- This table is to be used only for RECONSTITUTED FUEL ASSEMBLIES.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 7-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a seven-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

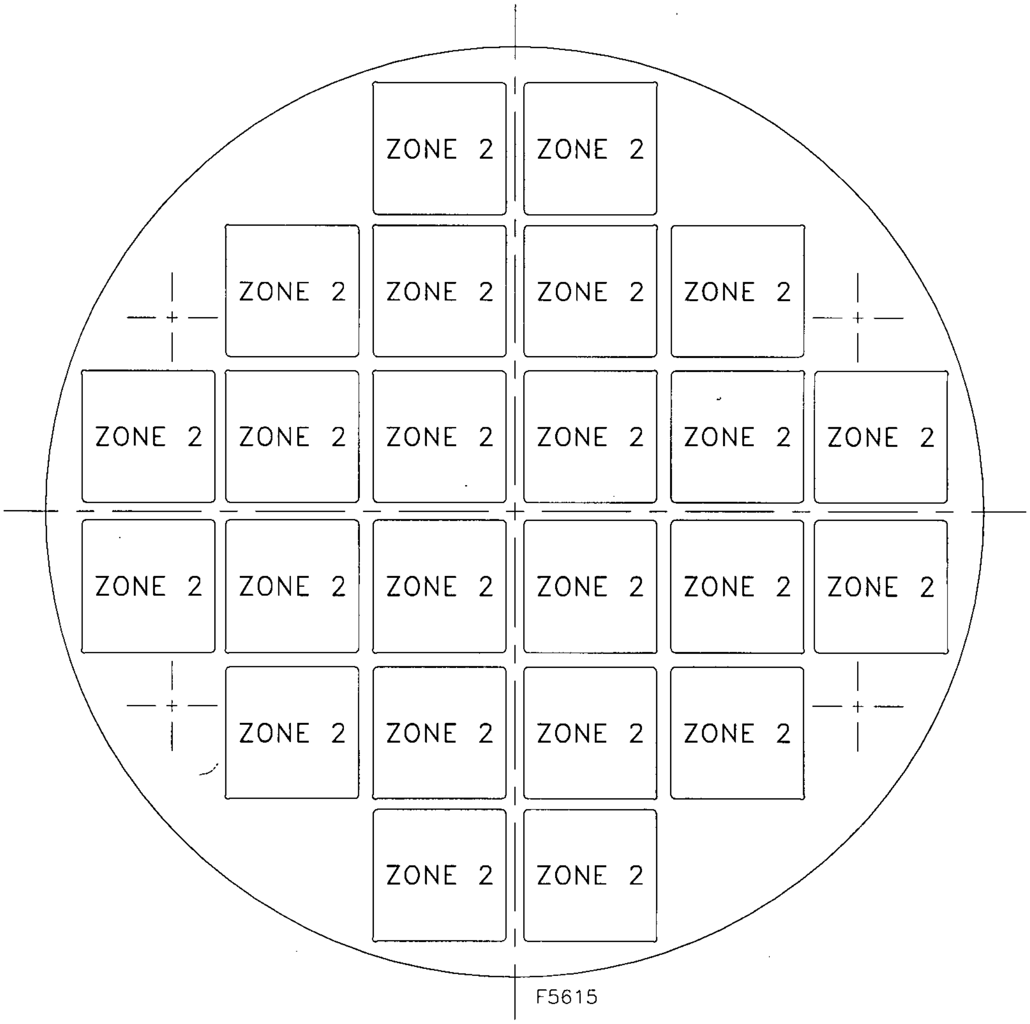
Table 2-16 PWR Fuel Qualification Table for 0.9 kW per Assembly for the NUHOMS® 24PT4-DSC, Reconstituted Fuel with Stainless Steel Rods

(Minimum required years of cooling time after reactor core discharge)

BU (GWd/ MTU)	Initial Enrichment																													
	1.8	1.9	2.0	2.1	2.2	2.3	2.4	2.5	2.6	2.7	2.8	2.9	3.0	3.1	3.2	3.3	3.4	3.5	3.6	3.7	3.8	3.9	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.8
10	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
15	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
20	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
25	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
28	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
30	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
32	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
34	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
36	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
38											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
39											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
40											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
41			Not Analyzed								7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
42											8	8	8	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
43																8	8	8	8	8	7	7	7	7	7	7	7	7	7	7	7
44																8	8	8	8	8	8	8	8	8	8	8	8	8	8	7	7
45																					8	8	8	8	8	8	8	8	8	8	8
48																					10	10	9	9	9	9	9	9	9	9	9
51																					11	11	11	11	11	11	11	11	11	11	11
54																					14	14	14	13	13	13	13	13	13	13	13
57																					17	16	16	16	16	16	16	16	15	15	15
60																					20	19	19	19	19	19	19	18	18	18	18

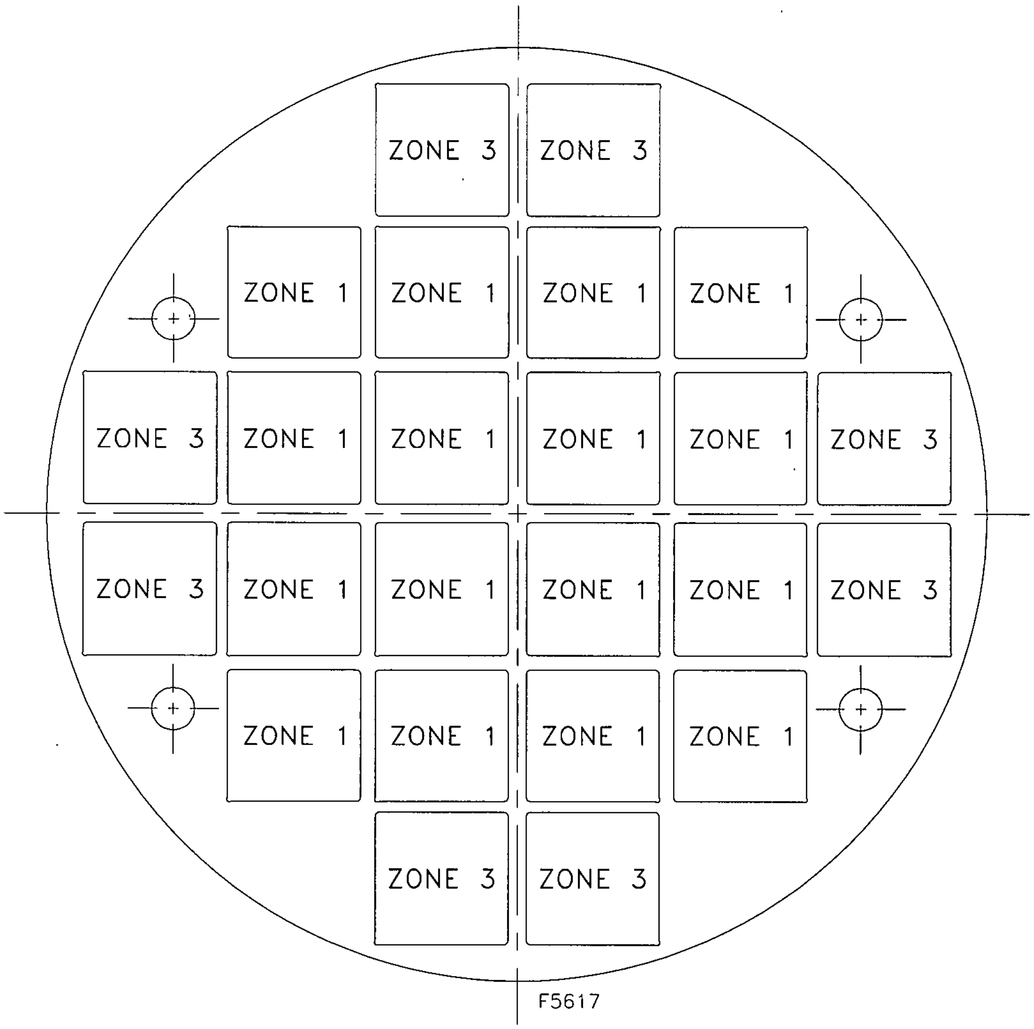
Notes:

- BU = Assembly average burnup.
- This table is to be used only for RECONSTITUTED FUEL ASSEMBLIES.
- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are conservatively applied in determination of actual values for these two parameters.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment either less than 1.8 or greater than 4.85 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 60 GWd/MTU is unacceptable for storage.
- Fuel with a burnup less than 10 GWd/MTU and is acceptable for storage after 7-years cooling.
- Example: An assembly with an initial enrichment of 4.85 wt. % U-235 and a burnup of 47 GWd/MTU is acceptable for storage after a nine-year cooling time as defined by 4.8 wt. % U-235 (rounding down) and 48 GWd/MTU (rounding up) on the qualification table.



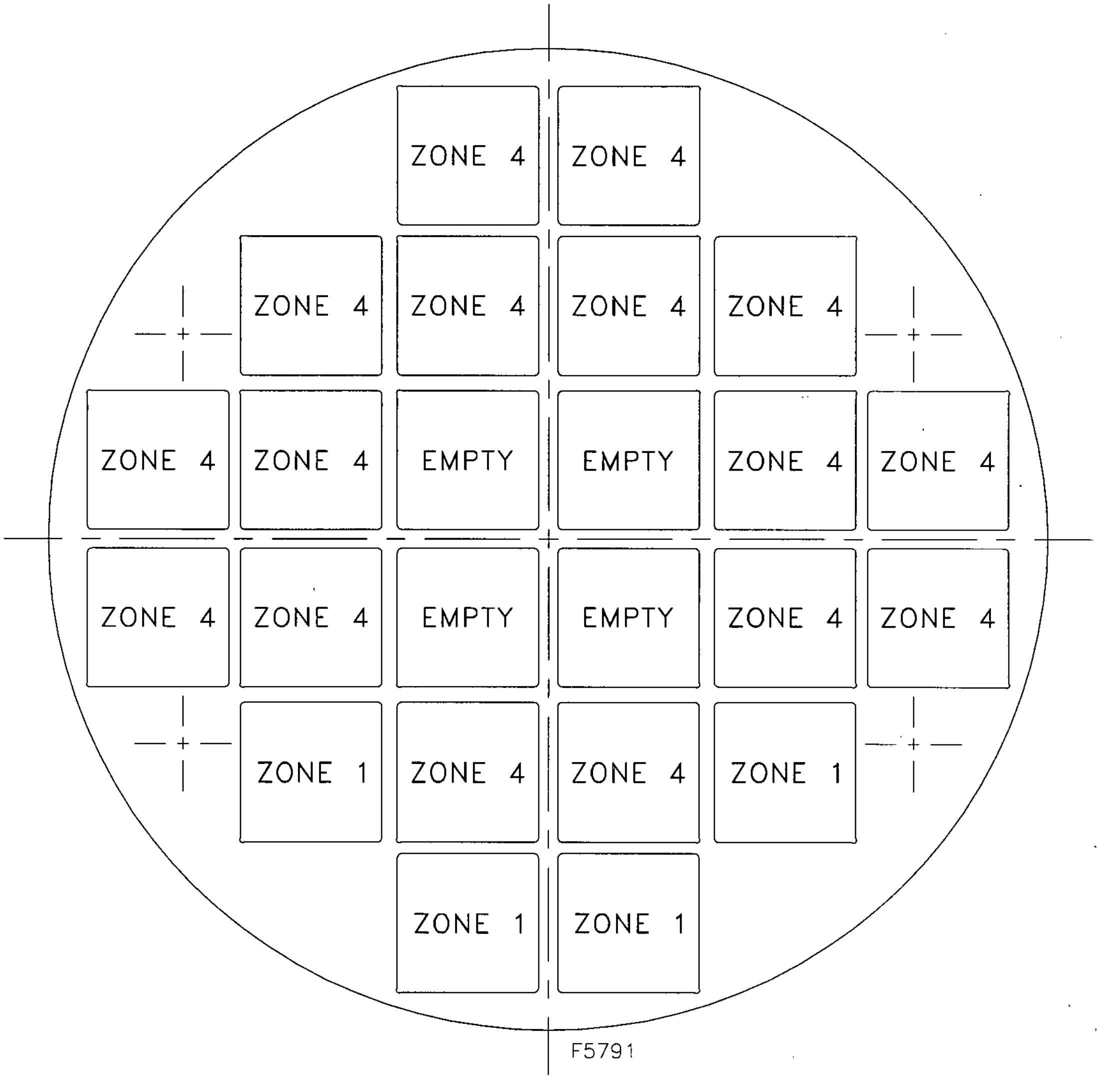
	Zone 1	Zone 2	Zone 3	Zone 4
Maximum Decay Heat (kWatts / FA)	NA	1.0	NA	NA
Maximum Decay Heat per Zone (kWatts)	NA	24.0	NA	NA

Figure 2-1 24PT4-DSC Heat Load Configuration #1, kW/Assembly



	Zone 1	*Zone 2*	*Zone 3*	*Zone 4*
Maximum Decay Heat (kWatts / FA)	**0.9**	**NA**	**1.2**	**NA**
Maximum Decay Heat per Zone (kWatts)	**14.4**	**NA**	**9.6**	**NA**

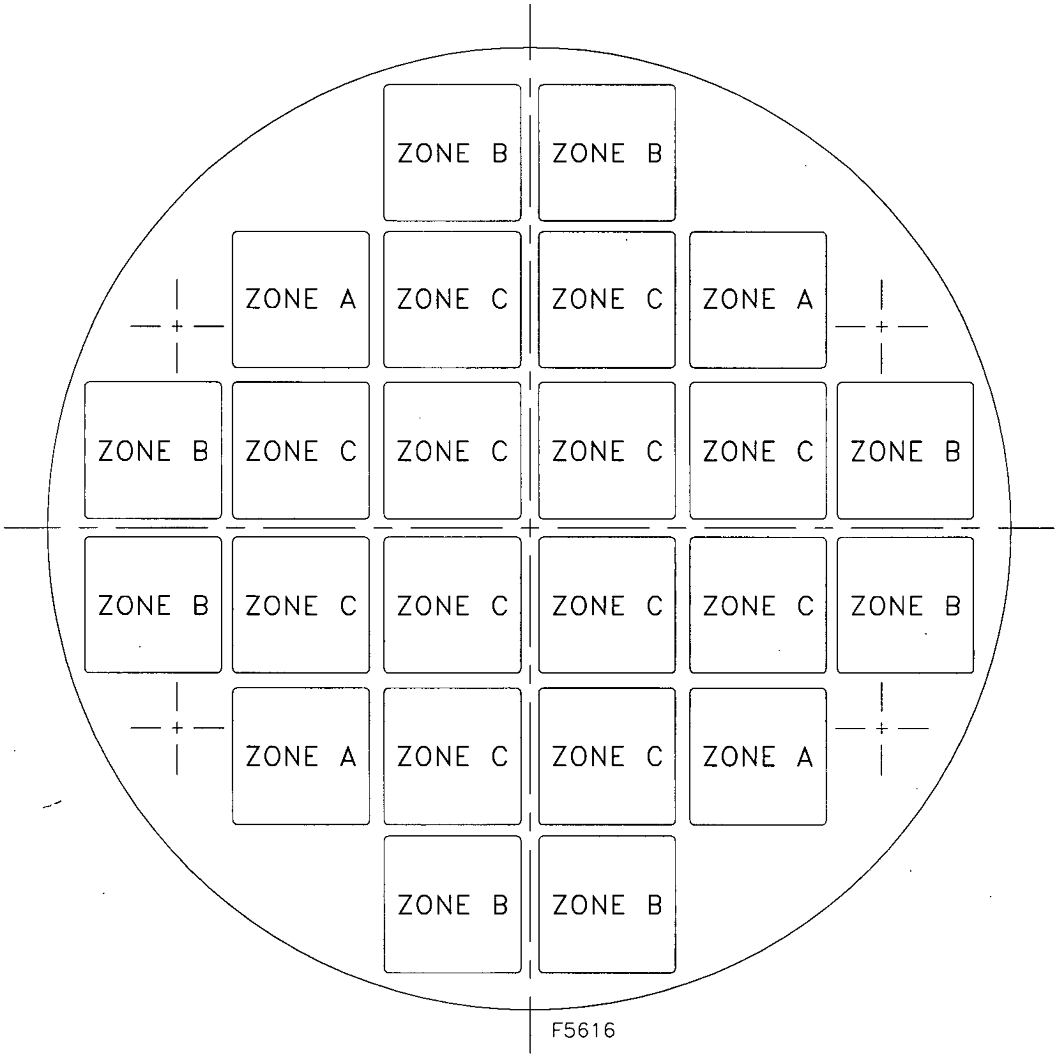
Figure 2-2 24PT4-DSC Heat Load Configuration #2, kW/Assembly



	Zone 1	Zone 2	Zone 3	Zone 4
Maximum Decay Heat (kWatts / FA)	**0.9**	**NA**	**NA**	**1.26**
Maximum Decay Heat per Zone (kWatts)	**3.6**	**NA**	**NA**	**20.16**

Note: FUEL ASSEMBLIES with a heat load of 0.9 kW (Zone 1) may also be placed anywhere in Zone 4.

Figure 2-3 24PT4-DSC Heat Load Configuration #3, kW/Assembly



Notes:

1. Locations identified as Zone A are for placement of up to 4 DAMAGED FUEL ASSEMBLIES.
2. Locations identified as Zone B are for placement of up to 8 additional DAMAGED FUEL ASSEMBLIES (Maximum of 12 DAMAGED FUEL ASSEMBLIES allowed, Zones A and B combined).
3. Locations identified as Zone C are for placement of up to 12 intact FUEL ASSEMBLIES, including 4 empty slots in the center as shown in Figure 2-3.
4. Poison Rodlets are to be located in the guide tubes of intact FUEL ASSEMBLIES placed in Zone C only per Table 2-*8*.

Figure 2-4 Location of FAILED FUEL CANS inside 24PT4-DSC

LCO 3.0.1	LCOs shall be met during specified conditions in the Applicability, except as provided in LCO 3.0.2.
LCO 3.0.2	Upon discovery of a failure to meet an LCO, the Required Actions of the associated Conditions shall be met, except as provided in LCO 3.0.5. If the LCO is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required, unless otherwise stated.
LCO 3.0.3	Not applicable to a spent fuel storage cask.
LCO 3.0.4	When an LCO is not met, entry into a specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS, or that are related to the unloading of a 24PT1-DSC or 24PT4-DSC. Exceptions to this Specification are stated in the individual Specifications. These exceptions allow entry into specified conditions in the Applicability when the associated ACTIONS to be entered allow operation in the specified condition in the Applicability only for a limited period of time.
LCO 3.0.5	Equipment removed from service or not in service in compliance with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate it meets the LCO or that other equipment meets the LCO. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate that the LCO is met.
LCO 3.0.6	Not applicable to a spent fuel storage cask.
LCO 3.0.7	Not applicable to a spent fuel storage cask.

SR 3.0.1	SRs shall be met during the specified conditions in the Applicability for individual LCOs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LCO except as provided in SR 3.0.3. Surveillances do not have to be performed on equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once," the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to this Specification are stated in the individual Specifications.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is less. This delay period is permitted to allow performance of the Surveillance. If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a specified condition in the Applicability of an LCO shall not be made unless the LCO's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into specified conditions in the Applicability that are required to comply with ACTIONS or that are related to the unloading of a DSC.

3.1 DSC Integrity

3.1.1.a 24PT1-DSC Vacuum Drying Time (Duration) and Pressure

LCO 3.1.1.a Duration: Vacuum Drying of the 24PT1-DSC shall be achieved with the following time durations after the start of bulk water removal (blowdown):

Heat Load (kW)	Time Limit
kW ≤ 12	No limit
12 < kW ≤ 13	71 Hours
13 < kW ≤ 14	54 Hours

Pressure: The 24PT1-DSC vacuum drying pressure shall be sustained at or below 3 Torr (3 mm Hg) absolute for a period of at least 30 minutes following stepped evacuation.

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

---------------------------------- NOTE ----------------------------------

This specification is applicable to all 24PT1-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. 24PT1-DSC vacuum drying pressure limit not met within 47 hours for a DSC with heat load greater than 12 kW and ≤ 13 kW or within 30 hours for a DSC with heat load greater than 13 kW and ≤ 14 kW.	A.1 Establish helium pressure of at least 1 atm and no greater than 20 psig in the 24PT1-DSC. <u>OR</u> A.2 Flood the 24PT1-DSC with water submerging all FUEL ASSEMBLIES.	24 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.a.1	Verify that the 24PT1-DSC vacuum pressure is less than, or equal to, 3 Torr (3 mm Hg) absolute for at least 30 minutes, within the specified total time duration based on heat load.	Once per 24PT1-DSC, after an acceptable NDE of the inner top cover plate weld.

3.1.1.b <u>24PT4-DSC Vacuum Drying Time (Duration) and Pressure</u>

LCO 3.1.1.b Duration: No time limits apply for vacuum drying of 24PT4-DSC *because only helium is allowed* for *the* blowdown *of water in the DSC cavity.*

Pressure: The 24PT4-DSC vacuum drying pressure shall be sustained at or below 3 Torr (3 mm Hg) absolute for a period of at least 30 minutes following stepped evacuation.

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

---------------------------------- NOTE ----------------------------------
This specification is applicable to all 24PT4-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Note: Not applicable until SR3.1.1.b.1 is performed. *A. If the required vacuum pressure cannot be obtained.*	*A.1* *A.1.1 Confirm that the vacuum drying system is properly installed. Check and repair the vacuum drying system as necessary.* *<u>OR</u>* *A.1.2 Check and repair the seal weld between the inner top cover plate/top shield plug assembly and the DSC shell.*	*30 days*
	<u>OR</u> *A.2 Establish helium pressure of at least 1.0 atm and no greater than 15 psig in the DSC.*	*30 days*
	<u>OR</u> *A.3 Flood the DSC with spent fuel pool water, submerging all fuel assemblies.*	*30 days*

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.b.1	Verify that the 24PT4-DSC vacuum pressure is less than, or equal to, 3 Torr (3 mm Hg) absolute for at least 30 minutes.	Once per 24PT4-DSC, after an acceptable NDE of the inner top cover plate / *top shield plug assembly* weld.

3.1.2.a <u>24PT1-DSC Helium Backfill Pressure</u>

LCO 3.1.2.a 24PT1-DSC helium backfill pressure shall be 1.5 ± 1.5 psig (stable for 30 minutes after filling).

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

------------------------------------ NOTE ----------------------------------

This specification is applicable to all 24PT1-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Note: Not applicable until SR 3.1.2.a.1 is performed.		
A. The required backfill pressure cannot be obtained or stabilized.	A.1 Establish the 24PT1-DSC helium backfill pressure to within the limit.	24 hours
	<u>*OR*</u>	
	A.2 Flood the 24PT1-DSC with water submerging all fuel assemblies.	24 hours

3.1.2.a 24PT1-DSC Helium Backfill Pressure

LCO 3.1.2.a 24PT1-DSC helium backfill pressure shall be 1.5 ± 1.5 psig (stable for 30 minutes after filling).

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

------------------------------------ NOTE ----------------------------------

This specification is applicable to all 24PT1-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Note: Not applicable until SR 3.1.2.a.1 is performed. A. The required backfill pressure cannot be obtained or stabilized.	A.1 Establish the 24PT1-DSC helium backfill pressure to within the limit. <u>*OR*</u> A.2 Flood the 24PT1-DSC with water submerging all fuel assemblies.	24 hours 24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.2.a.1 Verify that the 24PT1-DSC helium backfill pressure is 1.5 ± 1.5 psig.	Once per 24PT1-DSC, after the completion of TS 3.1.1.a actions.

3.1.2.b 24PT4-DSC Helium Backfill Pressure

LCO 3.1.2.b 24PT4-DSC helium backfill pressure shall be 6.0 + 1.0 / -0.0 psig (stable for 30 minutes after filling).

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

------------------------------------ NOTE ----------------------------------

This specification is applicable to all 24PT4-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Note: Not applicable until SR 3.1.2.b.1 is performed.		
A. The required backfill pressure cannot be obtained or stabilized.	A.1 Establish the 24PT4-DSC helium backfill pressure to within the limit.	24 hours
	<u>OR</u>	
	A.2 Re-initiate vacuum drying.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.2.b.1 Verify that the 24PT4-DSC helium backfill pressure is 6.0 + 1.0 / -0.0 psig.	Once per 24PT4-DSC, after the completion of TS 3.1.1.b actions.

4.0 Design Features

The specifications in this section include the design characteristics of special importance to each of the physical barriers and to maintenance of safety margins in the Advanced NUHOMS® System design. The principal objective of this section is to describe the design envelope that may constrain any physical changes to essential equipment. Included in this section are the site environmental parameters that provide the bases for design, but are not inherently suited for description as LCOs.

4.1 Site

4.1.1 Site Location

Because this *U*FSAR is prepared for a general license, a discussion of a site-specific ISFSI location is not applicable.

4.2 Storage System Features

4.2.1 Storage Capacity

The total storage capacity of the ISFSI is governed by the plant-specific license conditions.

4.2.2 Storage Pad

For sites for which soil-structure interaction is considered important, the licensee is to perform site-specific analysis considering the effects of soil-structure interaction. Amplified seismic spectra at the location of the AHSM center of gravity (CG) is to be developed based on the SSI responses. The AHSM center of gravity is shown in Table 3.2-1. The site-specific spectra at the AHSM CG must be bounded by the spectra presented in Chapter 2.

The storage pad location shall have no potential for liquefaction at the site-specific SSE level earthquake.

Additional requirements for the pad configuration are provided in Section 4.4.2.

4.2.3 Canister Neutron Absorber

For a 24PT1-DSC basket, neutron absorber with a minimum ^{10}B loading of 0.025 grams/square centimeter is provided for criticality control.

For a 24PT4-DSC basket, two alternate neutron absorber specifications are provided for criticality control depending upon the number of DAMAGED ASSEMBLIES and/or the maximum fuel enrichment of the payload as shown in Table 2-8:

- Type A Basket (minimum areal ^{10}B loading of 0.025 gm/cm^2)
- Type B Basket (minimum areal ^{10}B loading of 0.068 gm/cm^2)

<u>Boral®</u>

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. The average size of the boron carbide particles in the finished product is approximately 85 microns before rolling. 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 on the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be treated as nonconforming and 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.

4.2.4 <u>Canister Flux Trap Configuration</u>

The canister flux trap configuration is defined by the spacer disc ligament width dimensions. Figure 4-1 (applicable to 24PT1-DSC and 24PT4-DSC) shows the location and dimensions of the ligaments (the dimensions shown in the one quadrant are applicable to all four quadrants).

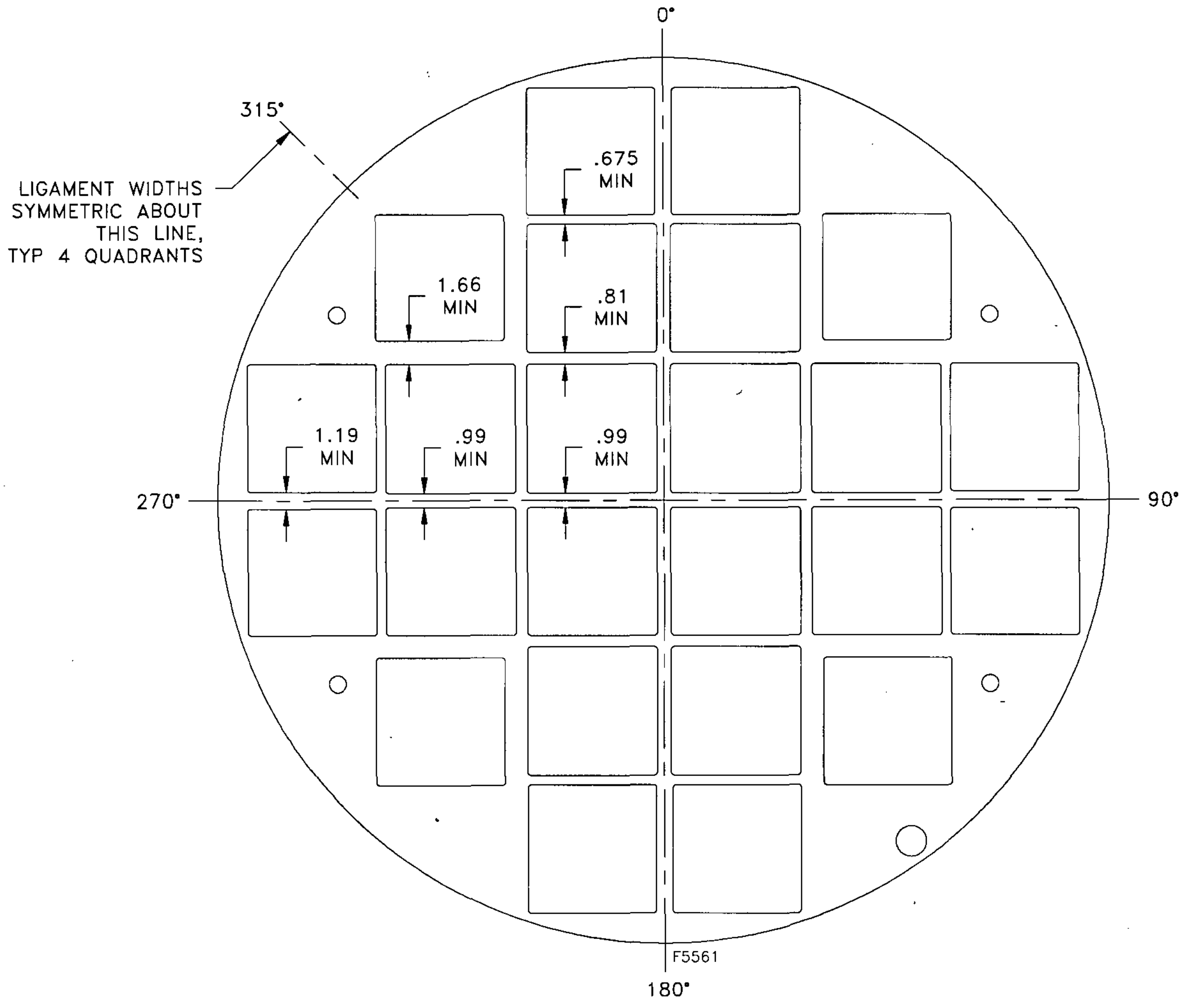
4.2.5 <u>Fuel Spacers</u>

Bottom fuel spacers are required to be located at the bottom of the DSC below each FUEL ASSEMBLY stored in the 24PT1-DSC. Top fuel spacers are required to be located above each INTACT FUEL ASSEMBLY stored in the 24PT1-DSC (the FAILED FUEL CAN design includes an integral top fuel spacer and therefore does not require a top fuel spacer).

No fuel spacers are required for 24PT4-DSC.

4.2.6 <u>Hydrogen Gas Monitoring for 24PT1 and 24PT4</u>

For the 24PT1 and 24PT4 DSCs, while welding the inner top cover plate during loading operations, and while cutting the outer or inner top cover plates during unloading operations, hydrogen monitoring of the space under the shield plug in the DSC cavity is required, to ensure that the combustible mixture concentration remains below the flammability limit.



Note: All ligament width dimensions are in inches.

Figure 4-1 Minimum Spacer Disc Ligament Widths

4.3 Codes and Standards

4.3.1 Advanced Horizontal Storage Module (AHSM)

The reinforced concrete AHSM is designed to meet the requirements of ACI 349-97. Load combinations specified in ANSI 57.9-1984, Section 6.17.3.1 are used for combining normal operating, off-normal, and accident loads for the AHSM.

4.3.2 Dry Shielded Canister, 24PT1-DSC or 24PT4-DSC (DSC)

The DSC is designed fabricated and inspected to the maximum practical extent in accordance with ASME Boiler and Pressure Vessel Code Section III, Division 1, 1992 Edition with Addenda through 1994, Subsections NB, NF, and NG for Class 1 components and supports. In addition, Code Case *N*-499-1 applies to 24PT4-DSC spacer disks. Code Alternatives are discussed in 4.3.4.

4.3.3 Transfer Cask

The TRANSFER CASK (OS197 or OS197H) shall meet the codes and standards that are applicable to its design under Certificate of Compliance C of C 1004.

A solar shield is required for cask TRANSFER OPERATIONS at temperatures exceeding 100°F.

4.3.4 Alternatives to Codes and Standards

ASME Code alternatives for the 24PT1-DSC or 24PT4-DSC (DSC) are listed below:

DSC Shell Assembly Alternatives to ASME Code, Subsection NB

Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
NCA	All	Not compliant with NCA
NB-1100	Requirements for Code Stamping of Components	The DSC shell is designed & fabricated in accordance with the ASME Code, Section III, Subsection NB to the maximum extent practical. However, Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.
NB-2130 NB-4121	Material must be supplied by ASME approved material suppliers Material Certification by Certificate Holder	All materials designated as ASME on the *U*FSAR drawings are obtained from ASME approved MM or MS supplier(s) with ASME CMTR's. 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 & certification are maintained in accordance with TN's NRC approved QA program.

Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
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 plate weld, and the siphon/vent cover welds, 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.*
NB-2531	*Vent & siphon Port Cover; straight beam UT per SA-578 for all plates for vessel*	*SA-578 applies to 3/8" and thicker plate only; allow alternate UT techniques to achieve meaningful UT results.*
NB-6000	*All completed pressure retaining systems shall be pressure tested*	*The 24PT1 is not a complete or "installed" pressure 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, the pressure testing of the DSC is performed in two parts. The DSC shell and shell bottom, including all longitudinal and circumferential welds, is pneumatically tested and examined at the fabrication facility.* *The shell to the inner top cover closure weld is pressure tested and examined for leakage in accordance with NB-6300 following fuel loading.* *The siphon/vent cover welds will not be pressure tested; these welds and the shell to the inner top cover 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, 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 DSC. 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 internal pressure considering 100% fuel rod failure at maximum accident temperature. The DSC is pressure tested to 120% of normal operating design pressure. An overpressure protection report is not prepared for the DSC.

Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000	The DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. In lieu of code stamping, QA Data packages are prepared in accordance with the requirements of 10CFR71, 10CFR72 and TN's approved QA program.

Basket Alternatives to ASME Code, Subsection NG/NF

Reference ASME Code Section/Article	**Code Requirement**	**Alternative, Justification & Compensatory Measures**
NCA	All	Not compliant with NCA
NG/NF-1100	Requirements for Code Stamping of Components	The DSC baskets are designed & fabricated in accordance with the ASME Code, Section III, Subsection NG/NF to the maximum extent practical as described in the *U*FSAR, but 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 be ASME Certified.
NG/NF-2130 NG/NF-4121	Material must be supplied by ASME approved material suppliers Material Certification by Certificate Holder	All materials designated as ASME on the *U*FSAR drawings are obtained from ASME approved MM or MS supplier with ASME CMTR's. 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 NG/NF-2130 is not possible. Material traceability & certification are maintained in accordance with TN's NRC approved QA program.
Table NG-3352-1	Permissible Joint Efficiency Factors	Joint efficiency (quality) factor of 1 is assumed for the guidesleeve longitudinal weld. Table NG-3352-1 permits a quality factor of 0.5 for full penetration weld with visual inspection. Inspection of both faces provides n= (2*0.5)=1. This is justified by this gauge of material (0.12 inch) with visual examination of both surfaces which ensures that any significant deficiencies would be observed and corrected.

Reference ASME Code Section/Article	Code Requirement	Alternative, Justification & Compensatory Measures
NG/NF-8000	Requirements for nameplates, stamping & reports per NCA-8000	The DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the DSC. In lieu of code stamping, QA Data packages are prepared in accordance with the requirements of 10CFR71, 10CFR72 and TN's approved QA program.
N/A	N/A	Oversleeve to guidesleeve welds are non-code welds which meet the requirements of AWS D1.3-98, the Structural Welding Code-Sheet Steel.
NG-3000 / Section II, Part D, Table 2A	Maximum temperature limit for Type 304 plate material is 800°F	For 24PT4-DSC only: The DSC guidesleeves, oversleeves and failed fuel cans do not comply with ASME Code limit of 800°F for Type 304 steel for the postulated blocked vent accident for approximately 25 hours. The maximum predicted temperature of those components for this event is less than 900°F. In accordance with Table I-14.5 of Article NH, the expected reduction in material strength is small (less than 1 ksi) and the calculated stress ratio is very small.

Proposed alternatives to the ASME code, other than the aforementioned ASME Code alternatives may be used when authorized by the Director of the Office of Nuclear Material Safety and Safeguards, or designee. The applicant should demonstrate that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or

2. Compliance with the specified requirements of ASME Code, Section III, 1992 Edition with Addenda through 1994 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Requests for alternatives in accordance with this section should be submitted in accordance with 10CFR 72.4.

4.4 Storage Location Design Features

The following storage location design features and parameters shall be verified by the system user to assure technical agreement with *the U*FSAR.

4.4.1 Storage Configuration

AHSMs are to be tied together in single rows or back to back arrays with not less than 3 modules tied together *in a single row of three (1x3) or for a back to back array in two rows of three (2x3).* Any 2 of the 3 modules *in a single row of 3 (1x3) or 5 of the 6 modules in a back to back array* may be empty (not contain a loaded DSC). Each group of modules not tied together must be separated from other groups by a minimum of 20 feet to accommodate possible sliding during a seismic event. The distance between any module and the edge of the ISFSI pad shall be no less than 10 feet.

4.4.2 Concrete Storage Pad Properties to Limit DSC Gravitational Loadings Due to Postulated Drops

The TC/DSC has been evaluated for drops of up to 80 inches onto a reinforced concrete storage pad. The evaluations are based on the concrete parameters specified in EPRI Report NP-4830, "The Effects of Target Hardness on the Structural Design of Concrete Storage Pads for Spent Fuel Casks," October 1986.

4.4.3 Site Specific Parameters and Analyses

The following parameters and analyses shall be verified by the system user for applicability at their specific site.

1. Tornado maximum wind speeds: 290 mph rotational
70 mph translational

2. Flood levels up to 50 ft. and water velocity of 15 fps.

3. One-hundred year roof snow load of 110 psf.

4. Normal ambient temperatures of 0°F to 104°F.

5. Off-normal ambient temperature range of –40°F without solar insolation to 117°F with full solar insolation.

6. The potential for fires and explosions shall be addressed, based on site-specific considerations.

7. Supplemental Shielding: In cases where engineered features (i.e., berms, shield walls) are used to ensure that the requirements of 10CFR 72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.

8. Seismic restraints shall be provided to prevent overturning of a loaded TC in a vertical orientation in the plant's decontamination area during a seismic event if a certificate holder determines that the horizontal acceleration is 0.40 g or greater. The determination of horizontal acceleration acting at the center of gravity (CG) of the loaded TC must be based on a peak horizontal ground acceleration at the site.

9. The effects of lightning, tsunamis, hurricanes and seiches, based on site-specific conditions shall be shown to be bounded by the design capability of the storage cask system.

10. *If an independent spent fuel storage installation site is located in a coastal salt water marine atmosphere, then any load-bearing carbon steel DSC support structure rail components of any associated AHSM shall be procured with a minimum of 0.20 percent copper content or stainless steel material shall be used for corrosion resistance.*

5.0 Administrative Controls

5.1 Procedures

Each user of the Advanced NUHOMS® System will prepare, review, and approve written procedures for all normal operations, maintenance, and testing at the ISFSI prior to its operation. Written procedures shall be established, implemented, and maintained covering the following activities that are important to safety:

- Organization and management
- Routine ISFSI operations
- Alarms and annunciators
- Emergency operations
- Design control and facility change/modification
- Control of surveillances and tests
- Control of special processes
- Maintenance
- Health physics, including ALARA practices
- Special nuclear material accountability
- Quality assurance, inspection, and audits
- Physical security and safeguards
- Records management
- Reporting
- All programs specified in Section 5.2

5.2 Programs

Each user of the Advanced NUHOMS® System will implement the following programs to ensure the safe operation and maintenance of the ISFSI:

- Safety Review Program
- Training Program
- Radiological Environmental Monitoring Program
- Radiation Protection Program
- AHSM Thermal Monitoring Program

5.2.1 Safety Review Program

Users shall conduct safety reviews in accordance with 10CFR 72.48 to determine whether proposed changes, tests, and experiments require NRC approval before implementation. Changes to the Technical Specification Bases and other licensing basis documents will be conducted in accordance with approved administrative procedures. Changes may be made to Technical Specification Bases and other licensing basis documents without prior NRC approval, provided the changes meet the criteria of 10CFR 72.48.

The safety review process will contain provisions to ensure that the Technical Specification Bases and other licensing basis documents are maintained consistent with the *U*FSAR.

Proposed changes that do not meet the criteria above will be reviewed and approved by the NRC before implementation. Changes to the Technical Specification Bases implemented without prior NRC approval will be provided to the NRC in accordance with 10CFR 72.48.

5.2.2 Training Program

Training modules shall be developed as required by 10CFR 72. Training modules shall require a comprehensive program for the operation and maintenance of the Advanced NUHOMS® System and the INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI). The training modules shall include the following elements, at a minimum:

- Advanced NUHOMS® System design (overview)
- ISFSI Facility design (overview)
- Systems, Structures, and Components Important to Safety (overview)
- Advanced NUHOMS® System *Updated Final* Safety Analysis Report (overview)
- NRC Safety Evaluation Report (overview)
- Certificate of Compliance conditions
- Advanced NUHOMS® System Technical Specifications
- Applicable Regulatory Requirements (e.g., 10CFR 72, Subpart K, 10CFR 20, 10 CFR Part 73)
- Required Instrumentation and Use
- Operating Experience Reviews
- Advanced NUHOMS® System and Maintenance procedures, including:
  Fuel qualification and loading,
  Rigging and handling,
  LOADING OPERATIONS as described in Chapters 8, A.8, and Sections 9.2 and A.9.2 of the *U*FSAR,
  UNLOADING OPERATIONS including reflooding,
  Auxiliary equipment operations and maintenance (i.e., welding operations, vacuum drying, helium backfilling and leak testing, reflooding),
  TRANSFER OPERATIONS including loading and unloading of the Transfer Vehicle,
  ISFSI Surveillance operations,
  Radiation Protection,
  Maintenance,
  Security,
  Off-normal and accident conditions, responses and corrective actions.

5.2.3 <u>Radiological Environmental Monitoring Program</u>

a) A radiological environmental monitoring program will be implemented to ensure that the annual dose equivalent to an individual located outside the ISFSI controlled area does not exceed the annual dose limits specified in 10CFR 72.104(a).

b) Operation of the ISFSI will not create any radioactive materials or result in any credible liquid or gaseous effluent release.

c) In accordance with 10CFR *72.44(d)(3)*, a periodic report will be submitted by the licensee that specifies the quantity of each of the principal radionuclides released to the environment in liquid and gaseous effluents during the previous year of operation.

5.2.4 <u>Radiation Protection Program</u>

The Radiation Protection Program will establish administrative controls to limit personnel exposure to As Low As Reasonably Achievable (ALARA) levels in accordance with 10CFR Part 20 and Part 72.

a. As part of its evaluation pursuant to 10CFR 72.212, the licensee shall perform an analysis to confirm that the limits of 10CFR 20 and 10CFR 72.104 will be satisfied under the actual site conditions and configurations considering the planned number of DSCs to be used and the planned fuel loading conditions.

b. A monitoring program to ensure the annual dose equivalent to any real individual located outside the ISFSI controlled area does not exceed regulatory limits is incorporated as part of the environmental monitoring program in the Radiological Environmental Monitoring Program of Section 5.2.3.

c. On the basis of the analysis performed in TS 5.2.4.a, the licensee shall establish a set of AHSM front birdscreen dose rate limits which are to be applied to 24PT1 and 24PT4 DSCs used at the site.

d. The Transfer Cask dose rate limit may not exceed the following values as calculated for a content of design basis fuel as follows:

The transfer cask side (radial) surface total dose rate shall be less than or equal to 160.0 mrem/hour when loaded with a 24PT1 DSC and 250.0 mrem/hour when loaded with a 24PT4 DSC. The transfer cask side surface dose rate shall be measured with water in the DSC cavity and also in the DSC/transfer cask annulus. The location of the dose rate measurement shall be at approximate centerline of the transfer cask outside surface in the radial direction.

The transfer cask dose rate limits are to maintain dose rates as-low-as-reasonably-achievable during DSC transfer operations. These dose rate limits are based on the shielding analysis for the wet welding case included in the UFSAR Chapter 5 and Chapter A.5 with some added margin for uncertainty.

If the measured dose rates exceed above values, place temporary shielding around the affected areas of the transfer cask and review plant records of the fuel assemblies which have been placed in the DSC to ensure that they conform to the fuel specification of Technical Specification 2.1 for the 24PT1 DSC or Technical Specification 2.2 for the 24PT4 DSC. Submit a letter report to the NRC within 30 days summarizing actions taken and the results of the surveillance, investigation and findings. The report must be submitted using instructions in 10 CFR 72.4 with a copy sent to administrator of the appropriate NRC regional office.

e. Following placement of each loaded TRANSFER CASK into the cask decontamination area and prior to transfer to the ISFSI, the DSC smearable surface contamination levels on the outer surface of the DSC shall be less than 2,200 dpm/100 cm^2 from beta and gamma emitting sources, and less than 220 dpm/100 cm^2 from alpha emitting sources.

The contamination limits specified above are based on the allowed removable external radioactive contamination specified in 49 CFR 173.443 (as referenced in 10 CFR 71.87(i)) the system provides significant additional protection for the DSC surface than the transportation configuration). The AHSM will protect the DSC from direct exposure to the elements and will therefore limit potential releases of removable contamination. The probability of any removable contamination being entrapped in the AHSM air flow path released outside the AHSM is considered extremely small.

f. Not withstanding the limits established in TS 5.2.4.c, the dose rate limit on the AHSM front birdscreen sufrace may not exceed the following values as calculated for a content of design basis fuel as follows:

The AHSM front birdscreen surface total dose rate shall be less than or equal to 50.0 mrem/hour when loaded with a 24PT1 DSC and 55.0 mrem/hour when loaded with a 24PT4 DSC. The AHSM front birdscreen dose rate shall be measured at the approximate centerline of the front birdscreen surface.

The AHSM dose rate limits are to maintain dose rates as-low-as-reasonably-achievable during DSC loading operations and to limit off-site exposures during storage operations. These dose rate limits are based on the shielding analysis for the storage case included in the UFSAR Chapter 5 and Chapter A.5 with some added margin for uncertainty.

If the measured dose rates exceed the limits of TS 5.2.4.c or above values whichever are lower, the licensee shall take the following actions:

1. *Notify the U.S. Nuclear Regulatory Commission (Director of the Office of Nuclear Material Safety and Safeguards) within 30 days,*
2. *Administratively verify that the correct fuel was loaded,*
3. *Ensure proper installation of the AHSM door,*
4. *Ensure that the DSC is properly positioned on the support rails, and*
5. *Perform an analysis to determine that placement of the as-loaded DSC at the ISFSI will not cause the ISFSI to exceed the radiation exposure limits of 10 CFR Part 20 and 72 and/or provide additional shielding to assure exposure limits are not exceeded.*

g. *Following completion of the seal weld of the DSC inner top cover plate/top shield plug assembly (including vent and siphon port cover) , this weld shall be leak tested with a helium leak detection device. The leak testing is performed to the criteria listed below:*

DSC Model	***Leak Test Criterion***	***Reference***
24PT1 or 24PT4	$\leq 1x10^{-7} atm.cm^3/sec$	*"Leak-Tight" as defined in ANSI N14.5-1997*

If the leakage rate of the inner seal weld exceeds the specified criterion, check and repair (a) the inner seal welds (b) the inner top cover for any surface indications resulting in leakage.

5.2.5 AHSM Thermal Monitoring Program

This program provides guidance for temperature measurements that are used to monitor the thermal performance of each AHSM. The intent of the program is to prevent conditions that could lead to exceeding the concrete and fuel clad temperature criteria.

a) AHSM Air Temperature Difference *Verification*

Following initial DSC transfer to the AHSM, the air temperature difference between ambient temperature and the roof vent temperature will be measured 24 hours (plus or minus 8 hours) after DSC insertion into the AHSM and again 5 to 7 days after insertion into the AHSM and prior to removing the AHSM door to perform the DSC retainer adjustment. If the air temperature differential is greater than 100°F, the air inlets and exits should be checked for blockage. If after removing any blockage found, the temperature differential is still greater than 100°F, corrective actions and analysis of existing conditions will be performed in accordance with the site corrective action program to confirm that conditions adversely affecting the concrete or fuel cladding do not exist.

The specified air temperature rise ensures the fuel clad and concrete temperatures are maintained at or below acceptable long-term storage limits. If the temperature rise is within the ≤100°F, then the AHSM and DSC are performing as designed and no further temperature measurements are required.

b) AHSM Concrete Temperature

The temperature measurement will be a direct measurement of the AHSM concrete temperature, or other means that would identify and allow for the correction of off-normal thermal conditions that could lead to exceeding the concrete and fuel clad temperature criteria. A temperature measurement of the thermal performance for each AHSM will be taken on a daily basis for the 24PT1-DSC with a 40 hour blocked vent time limit and twice a day for the 24PT4-DSC with a 25 hour blocked vent time limit. *Optionally, measurement of AHSM concrete temperatures with the 24PT1-DSC may also be taken twice a day.*

If the AHSM Concrete temperature *monitoring limits specified in Table 5-1 for maximum temperature and temperature rise* of the AHSM *concrete are exceeded*, based on this surveillance, then it is possible that some type of an inlet and or outlet vent blockage has occurred. Visual inspection of the vents will be initiated and appropriate corrective actions will be taken to avoid exceeding the concrete and cladding temperature limits.

The *maximum concrete temperature* values are obtained from a review of a transient thermal analysis of the AHSM with *a 14 kW heat load for the 24PT1-DSC and* a 24 kW heat load *for the 24PT4-DSC* to ensure that the rapid heatup is detected in time to initiate corrective action prior to exceeding concrete or DSC basket material temperature limits for the respective AHSM DSC payloads.

The *maximum concrete* temperature limits are chosen based on the expected concrete temperature for the 24 kW blocked vent scenarios to ensure that the associated fuel clad temperature is not exceeded.

The AHSM Thermal Monitoring Program provides a positive means to identify conditions that could approach the temperature criteria for proper AHSM operation and allow for the correction of off-normal thermal conditions that could lead to exceeding the concrete and fuel clad temperature criteria.

Monitoring of the AHSM concrete temperature is initiated following successful completion of the AHSM Air Temperature Difference Verification per Section 5.2.5.a.

c) *Visual Inspection of* AHSM Air Vents *(bird screens)*

Since the AHSMs are located outdoors, there is a possibility that the AHSM air inlet and outlet openings could become blocked by debris. Although the ISFSI security fence and AHSM bird screens reduce the probability of AHSM air vent blockage, the ISFSI *U*FSAR postulates and analyzes the effects of air vent blockage.

The AHSM design and accident analyses demonstrate the ability of the ISFSI to function safely if obstructions in the air inlets or outlets impair airflow through the AHSM for extended periods. This specification ensures that blockage will not exist for periods longer than assumed in the analyses.

Visual inspections *on a daily basis for 24PT1-DSC and twice a day for the 24PT4-DSC* of the AHSM air vents *(bird screens)* with the *24PT1-DSC or* 24PT4-DSC will be performed only if the temperature monitoring system data is unavailable or if the temperature limits specified in Section 5.2.5.*b* are exceeded to ensure that AHSM air vents are not blocked for more than *40* hours *for the 24PT1-DSC or 25 hours for the 24PT4-DSC.*

5.3 Lifting Controls

5.3.1 Cask Lifting Heights

The lifting height of a loaded TC/DSC, is limited as a function of location and temperature as follows:

a) The maximum lift height of the TC/DSC inside the Fuel Handling Building shall be 80 inches if the ambient temperature is below 0°F but higher than -80°F.

b) No lift height restriction other than 10CFR50 administrative controls, is imposed on the TC/DSC during LOADING OPERATIONS provided that a single-failure-proof crane is used and if the ambient temperature is higher than 0°F.

c) The maximum lift height and handling height for all TRANSFER OPERATIONS shall be 80 inches if the ambient temperature is greater than 0°F.

These restrictions ensure that any DSC drop as a function of location or low temperature is within the bounds of the accident analysis. If the ambient temperature is outside of the specification limits, LOADING and TRANSFER OPERATIONS will be terminated.

5.3.2 Cask Drop

Inspection Requirement

The DSC will be inspected for damage after any TRANSFER CASK drop of fifteen inches or greater.

Background

TC/DSC handling and loading activities are controlled under the 10CFR 50 license until a loaded TC/DSC is placed on the transporter, at which time fuel handling activities are controlled under the 10CFR 72 license. Although the probability of dropping a loaded TC/DSC while en route from the Fuel Handling Building to the ISFSI is small, the potential exists to drop the cask 15 inches or more.

Safety Analysis

The analysis of bounding drop scenarios shows that the TRANSFER CASK will maintain the structural integrity of the DSC confinement boundary from an analyzed drop height of 80 inches. The 80-inch drop height envelopes the maximum vertical height of the TRANSFER CASK when secured to the transfer trailer while en route to the ISFSI.

Although analyses performed for cask drop accidents at various orientations indicate much greater resistance to damage, requiring the inspection of the DSC after a drop of 15 inches or greater ensures that:

1. The DSC will continue to provide confinement
2. The TRANSFER CASK can continue to perform its design function regarding DSC transfer and shielding.

Table 5-1
AHSM Tempearture Monitoring Limits for the 24PT1-DSC and 24PT4-DSC

	Maximum Concrete Temperature (°F)	***Maximum Concrete Temperature Rise (°F) (in 24 hours)[1]***	***Maximum Concrete Temperature Rise (°F) (in 12 hours)[1][2]***
24PT1-DSC	*175*	*8*	*3*
24PT4-DSC	*200*	*N/A*	*5*

1. *Monitoring of the maximum temperature rise is initiated following successful completion of the AHSM Air Temperature Difference Verification per Section 5.2.5.a.*

2. *Measurement of AHSM concrete temperatures for the 24PT1-DSC over a 12 hour period is optional.*

pool area. The operations are similar to those for a shipping cask which are performed by plant personnel using existing procedures.

24PT1-DSC Preparation: The internals and externals of the 24PT1-DSC are inspected and cleaned if necessary. This ensures that the 24PT1-DSC will meet plant cleanliness requirements for placement in the spent fuel pool.

Place 24PT1-DSC in Transfer Cask: The empty 24PT1-DSC is inserted into the transfer cask.

Fill Transfer Cask/24PT1-DSC Annulus with Water and Seal: The transfer cask/24PT1-DSC annulus is filled with uncontaminated water and is then sealed prior to placement in the pool. This prevents contamination of the 24PT1-DSC outer surface and the transfer cask inner surface by the pool water.

Fill 24PT1-DSC Cavity with Water: The 24PT1-DSC cavity is filled with pool or demineralized water to prevent an in-rush of water as the transfer cask is lowered into the pool.

Lift Transfer Cask and Place in Fuel Pool: The transfer cask, with the water-filled 24PT1-DSC inside, is then lowered into the fuel pool. The transfer cask liquid neutron shield, if provided, may be left unfilled to meet hook weight limitations.

Spent Fuel Loading: Spent fuel assemblies are placed into the 24PT1-DSC. This operation is identical to that presently used at plants for shipping cask loading.

Top Shield Plug Placement: This operation consists of placing the top shield plug into the 24PT1-DSC using the plant's crane or other suitable lifting device.

Lifting Transfer Cask from Pool: The loaded transfer cask is lifted out of the pool and placed (in the vertical position) on the drying pad in the decon pit. This operation is similar to that used for shipping cask handling operations. The transfer cask liquid neutron shield, if left unfilled for weight reduction, shall be filled.

Inner Top Cover Plate Sealing: Using a pump, the water contained in the space above the top shield plug is drained. The inner top cover plate is placed onto the top shield plug and is welded to the shell. This weld provides the inner seal for the 24PT1-DSC.

Vacuum Drying and Backfilling: The initial blowdown of the 24PT1-DSC is accomplished by pressurizing the vent port with nitrogen or helium. The remaining liquid water in the cavity is forced out of the siphon tube and routed back to the fuel pool or to the plant's liquid radwaste processing system via appropriate size flexible hose or pipe, as appropriate. The 24PT1-DSC is then evacuated to remove the residual liquid water and water vapor in the cavity. When the system pressure has stabilized, the 24PT1-DSC is backfilled with helium and re-evacuated. After the second evacuation, the 24PT1-DSC is again backfilled with helium.

Pressure Test: Perform a pressure test of inner top cover/shield plug weld by backfilling the DSC cavity with helium.

After the pressure test, remove the helium lines then the vent and siphon cover plates are installed and welded to the inner top cover/shield plug.

Leak Test: Perform a leak test of the inner top cover/shield plug to the DSC shell weld and siphon/vent cover welds using a temporary test head or any other alternative means.

Outer Top Cover Plate Sealing: After helium backfilling, the 24PT1-DSC outer top cover plate/vent and siphon cover plates are installed by using a partial penetration weld between the outer top cover plate and the shell. The outer cover plate or shell weld and inner cover plate welds provide redundant seals at the upper end of the 24PT1-DSC.

Transfer Cask/24PT1-DSC Annulus Draining and Transfer Cask Top Cover Plate Placement: The transfer cask/24PT1-DSC annulus is drained. Demineralized water is flushed through the transfer cask/24PT1-DSC annulus, as required, to remove any contamination left on the 24PT1-DSC exterior. The transfer cask top cover plate is installed, using the plant's crane or other suitable lifting device, and bolted closed.

Place Loaded Transfer Cask on Transfer Skid/Trailer: The transfer cask is lifted onto the transfer cask support skid and downended onto the transfer trailer from the vertical to horizontal position. The transfer cask is secured to the skid.

Move Loaded Transfer Cask to AHSM: Once loaded and secured, the transfer trailer is towed to the ISFSI along a predetermined route on a prepared road surface. Upon entering the ISFSI the cask is positioned and aligned with the designated AHSM into which the 24PT1-DSC is to be transferred.

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

Insertion of 24PT1-DSC into AHSM: After final alignment of the transfer cask, AHSM, and hydraulic ram, the 24PT1-DSC is pushed into the AHSM by the hydraulic ram.

AHSM Closure: Install 24PT1-DSC seismic restraint and install AHSM door. Seismic restraint readjustment will be performed after the 24PT1-DSC and AHSM rails have reached thermal equilibrium (approximately one week).

1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

1.2.2.3.1 Criticality Prevention

Criticality is controlled by utilizing the fixed borated neutron absorbing material, Boral™, in the 24PT1-DSC basket. During storage, with the cavity dry and sealed from the environment, criticality control measures within the installation are not necessary because water cannot enter the canister during storage.

1.5 Supplemental Data

1.5.1 References

[1.1] 10CFR 72, Rules and Regulations, Title 10, Code of Federal Regulations - Energy, U.S. Nuclear Regulatory Commission, Washington, D.C., "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

[1.2] U.S. Nuclear Regulatory Commission, Regulatory Guide 3.61, Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask, February 1989.

[1.3] U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG 1536, U.S. NRC (January 1997).

[1.4] Nuclear Regulatory Commission, Safety Evaluation Report of Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, December 1994, USNRC Docket No. 72-1004, File NUH003.0103.02.

[1.5] TN, Safety Analysis Report for the NUHOMS® MP187 Multi-Purpose Cask, NUH-005, Revision 17, July 2003, USNRC Docket No. 71-9255.

[1.6] TN, Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 9, February 2006, File NUH003.0103, USNRC Docket No. 72-1004.

[1.7] Rancho Seco Independent Spent Fuel Storage Installation, Final Safety Analysis Report, Revision 0, November 2000, USNRC Docket No. 72-11.

[1.8] 10CFR 71, Rules and Regulations, Title 10, Code of Federal Regulations - Energy, U.S. Nuclear Regulatory Commission, Washington, D.C., "Packaging and Transportation of Radioactive Material."

[1.9] NRC Certificate of Compliance 72-1004, NUHOMS® General License Spent Fuel Storage System, Amendment No. 9, April 2007.

1.5.2 Drawings

- 24PT1-DSC: NUH-05-4010, Rev. 4 (PROPRIETARY)

- AHSM: NUH-03-4011, Rev. 4 (PROPRIETARY)

2.2.7 Combined Load Criteria

2.2.7.1 Advanced Horizontal Storage Module

The reinforced concrete AHSM is designed to meet the requirements of ACI 349-97 [2.6]. The alternate temperature criteria of NUREG-1536 will be utilized as discussed in Chapters 3 and 11. The ultimate strength method of analysis is utilized with the appropriate strength reduction factors as described in Chapter 3. The load combinations specified in Section 6.17.3.1 of ANSI 57.9-1984 [2.10] are used for combining normal operating, off-normal, and accident loads for the AHSM. All seven load combinations specified are considered and the governing combinations and the appropriate load factors are presented in Chapter 3. The resulting AHSM load combinations and load factors are presented in Chapter 3. The effects of duty cycle on the AHSM are considered and found to have negligible effect on the design. The corresponding structural design evaluation for the 24PT1-DSC support structure is presented in Chapter 3.

2.2.7.2 24PT1-DSC

The 24PT1-DSC is designed by analysis to meet the stress intensity allowables of the ASME Boiler and Pressure Vessel Code (1992 Edition with 1994 Addenda) Section III, Division I, Subsection NB with the alternatives to the ASME Code as specified in Table 3.1-4, NG and NF for Class 1 components and supports [2.7]. The 24PT1-DSC is conservatively designed by utilizing linear elastic or non-linear elastic-plastic analysis methods. The load combinations considered for the 24PT1-DSC normal, off-normal and postulated accident loadings are described in Chapter 3. ASME Code Service Level A and B allowables are used for normal and off-normal operating conditions. Service Level C and D allowables are used for accident conditions such as a postulated cask drop accident. Using these acceptance criteria ensures that in the event of a design basis drop accident, the 24PT1-DSC confinement boundary is not breached. The maximum shear stress theory is used to calculate principal stresses. Normal operational stresses are combined with the appropriate off-normal and accident stresses. It is assumed that only one postulated accident condition occurs at any one time. The accident analyses are documented in Chapter 11. The structural evaluation for the 24PT1-DSC is documented in Chapter 3.

2.2.8 Burial Under Debris

Debris resulting from natural phenomena or accidents that may affect system performance are to be determined by the licensee. Such debris can result from floods, wind storms, or land slides. The principal effect is typically on thermal performance. See Chapter 11 for a generic evaluation of AHSM burial.

2.2.9 Thermal Conditions

The Advanced NUHOMS® System component temperatures and thermal gradients are affected by the following thermal conditions:

- Fuel Loading
- Decay Heat

2.3 Safety Protection Systems

2.3.1 General

The Advanced NUHOMS® System is designed to provide long term storage of spent fuel. The canister materials are selected such that degradation is not expected during the storage period. The 24PT1-DSC cylindrical shell, and the top and bottom cover plate assemblies form the pressure retaining confinement boundary for the spent fuel. The 24PT1-DSC is equipped with two shield plugs to minimize occupational doses at the ends during drying, sealing, and handling operations. The 24PT1-DSC top closure has redundant welds which join the shell and the top cover plate assemblies to form the confinement boundary. The 24PT1-DSC shell and bottom end assembly confinement boundary weld is made during fabrication of the 24PT1-DSC, in accordance with the alternatives to the ASME Code described in Table 3.1-14.

The Advanced NUHOMS® System is designed for safe and secure, long-term confinement and dry storage of SFAs. The key elements of the Advanced NUHOMS® System and their operation which require special design consideration are:

A. Minimizing the contamination of the 24PT1-DSC exterior by fuel pool water.

B. The 24PT1-DSC top end, double closure welds form dual pressure retaining confinement boundaries and maintain a helium atmosphere.

C. Minimizing personnel radiation exposure during 24PT1-DSC loading, closure, and transfer operations.

D. The coating materials used in the design of the 24PT1-DSC are chosen to minimize hydrogen generation.

E. Design of the AHSM and 24PT1-DSC for postulated accidents.

F. Design of the AHSM passive ventilation system for effective decay heat removal to ensure the integrity of the fuel cladding. The AHSM is designed with no active safety systems.

G. Design of the 24PT1-DSC to ensure subcriticality.

2.3.2 Protection by Multiple Confinement Barriers and Systems

2.3.2.1 Confinement Barriers and Systems

The radioactive material which the Advanced NUHOMS® System ISFSI confines is the spent fuel assemblies and the associated contaminated or activated materials.

2.6 Supplemental Information

2.6.1 References

[2.1] U.S. Government, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation (ISFSI)," Title 10 Code of Federal Regulations, Part 72, Office of the Federal Register, Washington, D.C.

[2.2] U.S. Atomic Energy Commission, "Design Basis Tornado for Nuclear Power Plants," Regulatory Guide 1.76 (1974).

[2.3] U.S. Nuclear Regulatory Commission, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Revision 2, (1981).

[2.4] U.S. Atomic Energy Commission, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Regulatory Guide 1.60, Revision 1 (1973).

[2.5] U.S. Atomic Energy Commission, "Damping Values for Seismic Design of Nuclear Power Plants," Regulatory Guide 1.61 (1973).

[2.6] American Concrete Institute, Code Requirements for Nuclear Safety Related Concrete Structures, ACI 349-97.

[2.7] American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1992 Edition with 1994 Addenda.

[2.8] American Society of Civil Engineers, ASCE 7-95, Minimum Design Loads for Buildings and Other Structures, (formerly ANSI A58.1).

[2.9] Not Used.

[2.10] American National Standards Institute, American Nuclear Society, ANSI/ANS 57.9-1984, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).

[2.11] NRC NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, July 1980.

[2.12] American National Standards Institute, ANSI N14.6-1993, American National Standard for Special Lifting Device for Shipping Containers Weighing 10,000 lbs. or More for Nuclear Materials.

[2.13] American Concrete Institute, "Building Code Requirements for Reinforced Concrete," ACI-318, 1989 (92).

[2.14] TN, Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 9, February 2006, File NUH003.0103, USNRC Docket No. 72-1004.

3.1.1.1 General Description of the 24PT1-DSC

The principal characteristics of the 24PT1-DSC are described in Section 1.2.1 and shown in Figure 1.1-2. The drawings in Section 1.5.2 provide the principal dimensions and design parameters of the 24PT1-DSC.

For purposes of the structural analysis, the 24PT1-DSC is divided into the 24PT1-DSC shell assembly and the internal basket assembly. The 24PT1-DSC shell assembly, shown in Figure 3.1-1, includes the pressure retaining confinement boundary for the spent fuel, and consists primarily of a cylindrical shell, and the top and bottom cover plate assemblies. The 24PT1-DSC pressure boundary (shown in Figure 3.1-1) consists of the cylindrical shell, the inner bottom cover plate, the inner and outer top cover plates, and the associated welds. The outer top cover plate provides a redundant pressure-retaining boundary.

The remaining 24PT1-DSC shell assembly components include the outer bottom cover plate, the solid shield plugs (one at each end of the 24PT1-DSC assembly), the grapple ring assembly, support ring, and the lifting lugs. The shield plugs provide biological shielding during fuel loading operations and storage of a loaded 24PT1-DSC. The grapple ring assembly is welded to the outer bottom cover plate for the purpose of inserting/extracting the 24PT1-DSC from the Advanced Horizontal Storage Module (AHSM). The support ring, welded to the cylindrical shell, supports the top shield plug. Four lifting lugs are welded to the inside of the cylindrical shell and the support ring and are used to lift the unloaded 24PT1-DSC into the transfer cask prior to fuel loading operations.

All pressure boundary components are constructed of Type 316 stainless steel. Non-pressure boundary components welded to the pressure boundary components are also constructed of Type 316 stainless steel. The shield plugs are constructed of A36 carbon steel. The shield plugs are constrained by, but not mechanically fastened to, the stainless steel 24PT1-DSC shell assembly components, allowing free thermal expansion of the dissimilar materials.

The 24PT1-DSC cylindrical shell and bottom end assembly (which includes the inner and outer bottom cover plates, the bottom shield plug and the grapple ring assembly), and the internal basket assembly, are shop-fabricated (and assembled) components. The top shield plug and inner and outer top cover plates are installed at the plant after the spent fuel assemblies have been loaded into the 24PT1-DSC internal basket.

The 24PT1-DSC shell assembly is designed, fabricated, examined and tested in accordance with the requirements of Subsection NB of the ASME Code including the alternatives to the ASME Code specified in Table 3.1-14. The circumferential and longitudinal shell plate weld seams are multi-layer full penetration butt welds. The butt weld joints are fully radiographed and inspected according to the requirements of NB-5000 of the ASME Boiler and Pressure Vessel Code. The full penetration inner bottom cover plate to shell weld is inspected to the same Code standards, using either radiographic or ultrasonic inspection methods.

The 24PT1-DSC top closure is designed, fabricated and inspected using the alternatives to the ASME Code specified in Table 3.1-14. The top cover plates are sealed by separate, redundant closure welds. The inner top cover plate is welded to the 24PT1-DSC shell to form the inner pressure boundary at the top end of the 24PT1-DSC, as shown in Figure 3.1-2. The secondary, outer, pressure boundary is provided by the outer top cover plate. All closure welds are multiple-layer welds. This effectively eliminates any pinhole leaks which may occur in a single-pass weld, since the chance of pinholes being in alignment on successive weld passes is negligibly small. Also, both welds are examined by multi-level liquid penetrant to effectively eliminate through wall leaks.

The top end assembly of the 24PT1-DSC design incorporates a vent siphon block, with two small-diameter tubing penetrations into the 24PT1-DSC cavity for draining and filling operations. One penetration, the vent port, is terminated at the bottom of the shield plug assembly. The other port is attached to a siphon tube, which continues to the bottom of the 24PT1-DSC cavity. Both ports include dog-leg type, offsets to prevent radiation streaming. The vent and siphon ports terminate in normally closed quick-connect fittings. Both ports are used to remove water from the 24PT1-DSC during the drying and sealing operations.

During fabrication, the 24PT1 DSC shell and bottom assemblies are leak tested to an acceptance criterion of 1×10^{-7} ref. cm^3/sec as defined in ANSI N14.5 [3-10]. The welds between the DSC shell and the inner top cover including the siphon/vent covers are also leak tested to an acceptance criteria of 1×10^{-7} ref. cm^3/sec after the fuel assemblies are loaded into the canister and the pressure test satisfactorily completed.

The stringent design and fabrication requirements described above ensure that the pressure retaining confinement function is maintained for the design life of the 24PT1-DSC. Pressure monitoring instrumentation is not used since penetration of the pressure boundary would be required. The penetration itself would then become a potential leakage path and, by its presence, compromise the leaktightness of the 24PT1-DSC design.

During draining, backfilling, and leak testing, a "Strongback Device" may be installed to minimize deformation of the inner top cover plate during blowdown. The strongback is bolted to the top flange of the transfer cask and provides support to the inner cover plate during those operations that may involve significant pressurization of the 24PT1-DSC cavity.

Transfer of the 24PT1-DSC from the transfer cask into the AHSM is performed using a hydraulic ram that applies a load to the outer bottom cover plate, at the center of the DSC. During insertion of the 24PT1-DSC into the AHSM, the load is shared by the outer bottom cover plate, the bottom shield plug, and the inner bottom cover plate.

Frictional loads during 24PT1-DSC transfer are reduced by application of a dry film lubricant to the hardened nitronic surface on the support rails of the AHSM and the transfer cask. The lubricant chosen for this application is a tightly adhering inorganic lubricant with an inorganic binder. The dry film lubricant provides a thin, clean, dry, layer of lubricating solids that is intended to reduce wear, and prevent galling in metals. It is applied as a thin sprayed coating, similar to paint, using a carefully controlled process. The lubricant is not affected by water and is designed to be highly resistant to aggressive chemicals. This product is designed for radiation service and has a low coefficient of sliding friction for stainless steel.

3.1.2.1 24PT1-DSC Design Criteria

3.1.2.1.1 Stress Criteria

The 24PT1-DSC is designed utilizing linear elastic and non-linear elastic-plastic analytical methods. ASME Code Service Level A and B allowables are used for normal and off-normal operating conditions, respectively. Service Level C and D allowables are used for accident conditions.

The 24PT1-DSC shell is designed by analysis to meet the criteria of the ASME Boiler and Pressure Vessel Code Section III, Division l, Subsection NB, 1992 Edition through 1994 Addenda, supplemented by the alternatives to the ASME Code specified in Table 3.1-14 and ISG-4 [3.8]. Stress criteria for pressure boundary components are summarized in Table 3.1-2 of this section. Stress criteria for (partial penetration) pressure boundary top closure welds are summarized in Table 3.1-3.

The major internal basket components, spacer discs and guidesleeve assemblies, are designed to the criteria of ASME B&PV Code, Subsection NG as summarized in Table 3.1-4. The support rods and spacer sleeves are designed to the criteria of ASME B&PV Code, Subsection NF. The Boral™ neutron absorbing material is non-Code and is not considered a load-carrying component.

3.1.2.1.2 Stability Criteria

Stability of the 24PT1-DSC shell assembly is addressed for those load conditions in which the 24PT1-DSC is under external hydrostatic pressure (e.g., vacuum drying and external flood load cases) and/or axial compression, (e.g., loading the shell due to the shield plug's deadweight). Stability criteria are from ASME Section III, NB-3133.3 and NB-3133.6.

For the basket assembly, global stability is provided by the 24PT1-DSC shell, which provides continuous lateral support at each spacer disc. Local stability of individual basket components (spacer discs, support rod/spacer sleeve assemblies, guidesleeves assemblies) is addressed as described below.

Stability of the spacer discs is demonstrated using an eigenvalue buckling analysis and the criteria of ASME Section III, Appendix F. With the spacer disc loaded by the 75g-side drop, the eigenvalue analysis determines the margin to buckling (i.e., the multiple of the 75g load that would result in stability failure). In accordance with Appendix F, where the allowable load is 2/3 of the calculated stability load, an eigenvalue of greater than 1.5 demonstrates acceptable qualification of the discs.

Stability of the guidesleeves is assessed by evaluation of both overall guidesleeve stability and single panel stability. Overall stability is evaluated by considering the guidesleeve as a column under axial loads, laterally supported at the spacer discs, and applying the column stability criteria of NF-3322.1 and NF-1334.3. Panel stability is evaluated using equations from Roark [3.19] for plates under in-plane loading.

Compressive loads in the support rod assembly are carried by the spacer sleeves. Stability of the spacer sleeves is addressed using the criteria for combined axial compression and bending loads from Subsection NF and Appendix F for linear supports.

3.1.2.2.4.2 Fire and Explosion Overpressure

Overpressure due to externally initiated fires and explosions are assumed to be bounded by the design basis tornado wind pressure.

3.1.2.2.5 Design Load Combinations

In accordance with ANSI 57.9, [3.9] the design basis concrete loads are multiplied by load factors and combined in load combinations to simulate the most adverse load conditions considering credible variations in magnitude and direction. The nominal ultimate concrete strength is reduced by the strength reduction factors provided in Table 3.1-9 to obtain the design strength of concrete.

The load combinations specified in [3.9] are used for combining normal operating, off-normal and accident loads for the AHSM. All seven-load combinations specified are considered and governing combinations are selected for detailed design and analysis. The resulting AHSM load combinations and the appropriate load factors are presented in Table 3.1-10. The corresponding structural design criteria for the 24PT1-DSC support structure are summarized in Table 3.1-11 and Table 3.1-12.

The overturning and sliding load combinations and factors of safety for the tornado and flood loading cases are provided in Table 3.1-13. For the Design Earthquake, the analysis presented in Chapter 11 shows that an AHSM array may slide up to a maximum of 44 inches without any significant tipping. This guarantees the safe retrieval of all stored 24PT1-DSCs.

3.1.2.3 Alternatives to the ASME Code for the 24PT1-DSC

This section documents and justifies alternatives to the ASME Code Section III, Division 1 requirements. The 24PT1-DSC is a not a Code-stamped vessel and, as such, the services of an Authorized Nuclear Inspector (ANI) are not required. The design of the 24PT1-DSC and internal basket components is specified to meet the technical provisions of the ASME B&PV Code.

The following sections of the ASME Code apply to the technical requirements for the design, fabrication, testing, and inspection of the 24PT1-DSC's:

- Section II for materials.
- Section III for materials, design, fabrication, testing, inspection, and over pressure protection.
- Section V for non-destructive examination.
- Section IX for welder and welding procedure qualifications.

3.1.2.3.1 Code Alternatives

Alternatives to the ASME Code can be broken down into three basic areas. These are:

- Code General Requirements
- Technical design
- Component fabrication, inspection, and examination

Although each of these areas are interrelated, the exceptions come under different authorities.

3.1.2.3.2 Code General Requirements

The 24PT1-DSCs will typically be procured under the technical requirements of the Code without requiring the use of an Authorized Inspector or the application of an N-stamp. Hence, many of the administrative items that would allow the 24PT1-DSC to be stamped are not typically in place. This includes such things as a Design Specification certified by a professional engineer, a formal overpressure protection report; and requiring design and fabrication work to be done by firm(s) holding an N-stamp. These items have no affect on the functionality of the component, it does affect its ability to comply with the requirements of the ASME Code. The qualifications of the firms and personnel, procedures used to develop the design reports and fabrication specifications, and the lack of an N-stamp vendor are all exceptions to the requirements of Subsection NCA. Hence, the 24PT1-DSC is not compliant with Subsection NCA.

3.1.2.3.3 Technical Compliance

Technical compliance is compliance with Code design rules and materials specification, processes, joint configurations, etc. The evaluation and design performed for the 24PT1-DSC components are based on compliance with Section III of the ASME Code, NRC Regulatory Guides and NUREG. Table 3.1-14 provides a discussion of alternatives to the Code provisions for materials, fabrication, examination, and testing. The alternatives to the design portions of the Code are typically the result of not meeting Subsection NCA of the Code. The top closure weld details for the completed pressure boundary are discussed in Table 3.1-14. Similarly, the internal basket is designed to the rules of Subsection NF and/or NG with the exceptions listed in Table 3.1-15.

3.1.2.3.4 Fabrication, Inspection, and Examination of Components

There are no specific exceptions taken to the Code inspection requirements, except a non-ASME Code certified fabricator is permitted to build the 24PT1-DSCs. Neither an Authorized Nuclear Inspector (ANI) nor Code certified shop is required by the procurement documents to fabricate or inspect the 24PT1-DSC. Therefore, the role of the Certificate Holder is not required to be met for the fabrication and inspection process. Fabrication and closure welding alternatives are provided in Table 3.1-14 and Table 3.1-15.

Table 3.1-1
Codes and Standards for the Fabrication and Construction of Principal Components

Component, Equipment, Structure	Code of Construction
24PT1-DSC	ASME Code, Section III, 1992 Edition through 1994 Addenda, including alternatives to the ASME Code specified in Table 3.1-14.
AHSM	- ACI 318-89 (92) - AWS D1.1-98 - AWS D1.6-99 - ACI 349-97 - AISC Ninth Edition - Load Combinations from ANSI 57.9-1984

Table 3.1-3
Stress Criteria for Partial Penetration Pressure Boundary Welds

Service Level	Allowable Primary Stress	Primary and Secondary	Notes
Level A	0.8 S_m	0.8 (3.0 S_m)	Note 1
Level B	0.8 S_m	0.8 (3.0 S_m)	Note 1
Level C	greater of 0.8 (1.2 S_m) or 0.8 S_y	N/A	Note 1
Level D Elastic	lesser of 0.8 (2.4 S_m) or 0.8 (0.7 S_u)	N/A	Note 1
Level D Plastic	greater of 0.8 (0.7 S_u) or 0.8 (S_y + 1/3 (S_u-S_y))	N/A	Note 1

Note:

1. These limits are based on ISG-4.

Table 3.1-14
ASME Code Exceptions for the 24PT1-DSC (NB)

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NCA	All	Not compliant with NCA
NB-1100	Requirements for Code Stamping of Components	The 24PT1-DSC shell is designed & fabricated in accordance with the ASME Code, Section III, Subsection NB to the maximum extent practical. However, Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.
NB-2130 NB-4121	Material must be supplied by ASME approved material suppliers Material Certification by Certificate Holder	All materials designated as ASME on the SAR drawings are obtained from ASME approved MM or MS supplier(s) with ASME CMTR's. 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 & certification are maintained in accordance with TNW's NRC approved QA program
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 plate weld, and the siphon/vent cover welds, 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.
NB-2531	Vent & siphon Port Cover; straight beam UT per SA-578 for all plates for vessel	SA-578 applies to 3/8" and thicker plate only; allow alternate UT techniques to achieve meaningful UT results.

Table 3.1-14
ASME Code Exceptions for the 24PT1-DSC (NB)
Concluded

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NB- 6000	All completed pressure retaining systems shall be pressure tested	The 24PT1 is not a complete or "installed" pressure 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, the pressure testing of the DSC is performed in two parts. The DSC shell and shell bottom, including all longitudinal and circumferential welds, is pneumatically tested and examined at the fabrication facility. The shell to the inner top cover closure weld is pressure tested and examined for leakage in accordance with NB-6300 following fuel loading. The siphon/vent cover welds will not be pressure tested; these welds and the shell to the inner top cover 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, 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 24PT1-DSC. The function of the 24PT1-DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The 24PT1-DSC is designed to withstand the maximum internal pressure considering 100% fuel rod failure at maximum accident temperature. The 24PT1-DSC is pressure tested to 120% of normal operating design pressure. An overpressure protection report is not prepared for the DSC.
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000	The 24PT1-DSC nameplate provides the information required by 10CFR 71, 49CFR173 and 10CFR 72 as appropriate. Code stamping is not required for the 24PT1-DSC. QA Data packages are prepared in accordance with the requirements of 10CFR 71, 10CFR 72 and TNW's approved QA program.

chlorides, low temperature and short time of exposure to the corrosive environment eliminates the possibility of stress corrosion cracking in the guidesleeve welds. The predicted maximum guidesleeves temperature in the pool is 150°F.

3.4.1.2 Behavior of Boral™ in Borated Water

Boral™ is a proven neutron poison used extensively in spent fuel storage racks. The short term exposure of the material to borated water in the spent fuel storage canisters will have significantly less effect on the Boral™ than that experienced in spent fuel pools.

3.4.1.3 Electroless Nickel Plated Carbon Steel

The carbon steel top shield plug and the spacer discs are plated with electroless nickel. This coating is similar to the coating used on the Standard NUHOMS® 52B canister (C of C 72-1004) [3.15]. This coating has been evaluated for potential galvanic reactions in Transnuclear West's response to NRC Bulletin 96-04 [3.28]. The reported corrosion rates are insignificant in PWR pools, and will result in a negligible rate of reaction for the 24PT1-DSC systems.

3.4.1.4 Hydrogen Generation

During the initial passivation state, small amounts of hydrogen gas may be generated in the 24PT1-DSC. The passivation stage may occur prior to submersion into the spent fuel pool. Any amounts of hydrogen generated in the canister will be insignificant and will not result in a flammable gas mixture within the canister [3.28].

3.4.1.5 Effect of Galvanic Reactions on the Performance of the System

There are no significant galvanic reactions that could reduce the overall integrity of the canister, or its contents, during storage.

There are no reactions that would cause binding of the mechanical surfaces or the fuel to guidesleeves due to galvanic or chemical reactions.

There is no significant degradation of any safety component caused directly by the effects of the reactions, or by the effects of the reactions combined with the effects of long-term exposure of the materials to neutron or gamma radiation, high temperatures, or other possible conditions.

The canister and fuel cladding thermal properties are provided in Chapter 4. The emissivity of the basket components ranges from 0.10 for Boral™ sheets to 0.4 for stainless steel. If the stainless steel is oxidized, this value would increase, improving heat transfer. Therefore, the passivation reactions would improve the thermal properties of the 24PT1-DSC materials.

3.4.2 Positive Closure

Positive closure is provided by the redundant closure welds for the inner and outer top cover plates, the vent and siphon cover welds, and the leak-tight 24PT1-DSC shell assembly.

3.6.3 References

[3.1] CFR Title 10, Part 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

[3.2] NRC Regulatory Guide 1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants, Revision 1, 1973.

[3.3] NRC Regulatory Guide 1.61, Damping Values for Seismic Design of Nuclear Power Plants, October 1973.

[3.4] NRC Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants, April 1974.

[3.5] NRC Regulatory Guide 3.61, Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask, February 1989.

[3.6] NRC NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Revision 2, July 1981.

[3.7] NRC NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, January 1997.

[3.8] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-4, Cask Closure Weld Inspections, Revision 2.

[3.9] American National Standards Institute, American Nuclear Society, ANSI/ANS 57.9-1984, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type).

[3.10] ANSI N14-5–1997, Leakage Tests on Packages for Shipment," February 1998.

[3.11] American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III, 1992 Edition with Addenda through 1994.

[3.12] American Concrete Institute, Code Requirements for Nuclear Safety Related Concrete Structures, ACI 349-97.

[3.13] American Institute of Steel Construction, AISC Manual of Steel Construction, Ninth Edition.

[3.14] American Society of Civil Engineers, ASCE 7-95, Minimum Design Loads for Buildings and Other Structures, (formerly ANSI A58.1).

[3.15] TN, Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 9, February 2006, USNRC Docket Number 72-1004.

[3.16] TN, Safety Analysis Report for the NUHOMS® MP187 Multi-Purpose Cask, NUH-005, Revision 17 July 2003, USNRC Docket Number 71-9255.

[3.31] Rancho Seco Independent Spent Fuel Storage Installation, Final Safety Analysis Report, Revision 0, November 2000, USNRC Docket Number 72-11.

[3.32] Johnson, A. B. and E. R. Gilbert, Technical Basis for Storage of Zircalloy-Clad Spent Fuel, September 1983, Pacific Northwest Laboratory, PNL Document PNL-4835.

[3.33] U.S. Nuclear Regulatory Commission, Interim Staff Guidance (ISG)-15, "Materials Evaluation," January 10, 2001.

[3.34] UCID – 21246, "Dynamic Impact Effects on Spent Fuel Assemblies", October 20, 1987.

[3.35] "Consolidated Safety Analysis Report for IF-300 Shipping Cask", NEDO-10084, Vectra Technologies, Inc., Revision 4, March, 1995.

[3.36] Not used.

[3.37] Holman, W.R., Langland, R. T., "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick," NUREG/CR-1815, August 1981.

[3.38] AWS D1.1-98, Structural Welding Code-Steel.

[3.39] AWS D1.6-99, Structural Welding Code-Stainless Steel.

[3.40] Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988.

4.8 Supplemental Information

4.8.1 References

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

[4.2] M. Cunningham, E. Gilbert and A. Johnson, Jr., M. A. McKinnon, "Evaluation of Expected Behavior of LWR Stainless Steel-Clad Fuel in Long-Term Dry Storage", April 1996, Electric Power Research Institute, EPRI Document TR-106440.

[4.3] ASHRAE Handbook 1981 Fundamentals, 4th Printing, 1983.

[4.4] "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages," Office of Civilian Radioactive Waste Management, DOE/RW-0472, Revision 2, September 1998.

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[4.6] Bolz, R. E., G. L. Tuve, CRC Handbook of Tables for Applied Engineering Science, 2nd Edition, 1973.

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7. CONFINEMENT

7.1 Confinement Boundary

The 24PT1-DSC is a high integrity austenitic stainless steel welded vessel that provides confinement of radioactive materials, encapsulates the fuel in a helium atmosphere and provides biological shielding during 24PT1-DSC closure and transfer and storage operations. The 24PT1-DSC is designed to maintain confinement of radioactive material within the limits of 10CFR 72.104(a), 10CFR 72.106(b) and 10CFR 20 under normal, off-normal, and credible accident conditions. Chapter 3 concludes that the design including the helium atmosphere within the 24PT1-DSC will adequately protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures during storage. The design ensures that fuel degradation during storage will not pose operational safety problems with respect to removal of the fuel from storage.

The cylindrical shell, and the inner top and bottom cover plates form the confinement boundary for the spent fuel. The vent and siphon cover plate welds and the vent and siphon block weld are also included in the confinement boundary. The outer top and bottom cover plate's function as redundantly welded barriers for confining radioactive material within the 24PT1-DSC. The dimensions and material descriptions for the confinement boundary assemblies and the redundantly welded barriers are discussed in Chapter 1. The components important to safety are identified in Chapter 2.

7.1.1 Confinement Vessel

The cylindrical shell and inner shell to bottom cover plate welds are made during fabrication of the 24PT1-DSC and are fully compliant to ASME Section III [7.1], Subsection NB. The vent and siphon block weld is also made during fabrication. The inner top cover plate weld is made after fuel loading. Both top plug penetrations (siphon and vent ports) and cover plates are welded after drying operations are complete.

Stringent design and fabrication requirements ensure that the confinement function of the 24PT1-DSC is maintained. The shell and inner bottom cover plate are pressure tested in accordance with the ASME Code, Section III, Subarticle NB-6300. This pressure test is performed after installation of the inner bottom cover plate and may be performed concurrently with the leak test, provided the requirements of NB 6300 are met.

Following the pressure test, a leak test of the shell assembly, including the inner bottom cover plate, is performed in accordance with ANSI N14.5 [7.2] and the ASME Code, Section V, Article 10. These test are typically performed at the fabricator. The acceptance criteria for the test is "leaktight" as defined in ANSI N14.5 1997 [7.2].

The process involved in leak testing the 24PT1 DSC involves temporarily sealing the shell from the top end. The gas filled envelope and evacuated envelope testing methodologies have the required nominal test sensitivity for leaktight construction and are used for leak testing. A helium mass spectrometer is used to detect any leakage as defined in ANSI N14.5 [7.2].

During final drying and sealing operations of the 24PT1-DSC, the top closure confinement welds are applied to confine radioactive materials within the cavity. The inner top cover plate is welded to the shell using automated welding equipment. Once the 24PT1-DSC has been vacuum dried, a pressure test is performed by backfilling the DSC Cavity with helium. Following satisfactory completion of the siphon port pressure test the vent and penetrations are welded, and the outer top cover plate is lowered onto the 24PT1-DSC. The outer top cover plate is also welded in place using automated welding equipment. The outer top cover plate to shell weld acts as a redundant barrier for confining radioactive material within the 24PT1-DSC throughout its service life.

Leak testing of the 24PT1-DSC inner top cover plate to shell weld, vent and siphon cover plate and vent/siphon block to shell welds is performed using a test head prior to placing the outer top cover plate or by pulling a vacuum between the inner and outer top cover plates through a test port in the outer top cover plate and monitoring for helium in accordance with the requirements of ANSI N14.5 [7.2] to demonstrate that these welds are leaktight.

7.1.2 Confinement Penetrations

All penetrations in the 24PT1-DSC confinement boundary are welded closed. The 24PT1-DSC is designed and tested to be "leaktight" as described above.

7.1.3 Seals and Welds

The austenitic stainless steel welds made during fabrication of the 24PT1-DSC that affect the confinement boundary include the weld applied to the inner bottom cover plate and the circumferential and longitudinal seam welds applied to the shell. These welds are inspected (radiographic or ultrasonic inspection, and liquid penetrant inspection) according to the requirements of Subsection NB of the ASME Code. The vent and siphon block weld is also made during fabrication and is liquid penetrant inspected in accordance with Subsection NB of the ASME Code.

The welds applied to the vent and siphon port covers and the inner and outer top cover plates (including test plug) during closure operations, define the confinement boundary at the top end of the 24PT1-DSC. These welds are applied using a multiple-layer technique with multi-level PT in accordance with the alternatives to the ASME Code as specified in Table 3.1-14. This effectively eliminates any pinhole leak which might occur in a single-pass weld, since the chance of pinholes being in alignment on successive weld passes is negligibly small. Figure 7.1-1 provides a graphic representation of the confinement boundary welds.

7.1.4 Closure

The 24PT1-DSC is closed entirely by welding and thus, no closure devices are utilized for confinement.

7.4 Supplemental Data

7.4.1 Confinement Monitoring Capability

The Advanced NUHOMS® System is a self-contained passive system that does not produce routine, solid, liquid or gaseous effluents. Effluent processing systems, or monitoring for airborne or liquid radioactivity, are not required to protect personnel or the environment during storage conditions. Since the 24PT1-DSC is closed entirely by welding, a closure monitoring system is not required as discussed in NRC ISG-5 [7.3].

7.4.2 References

[7.1] American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III, 1992 Edition with Addenda through 1994.

[7.2] NRC Spent Fuel Project Office, Interim Staff guidance, ISG-18, Revision 0.

[7.3] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-5, Revision 1, Confinement Evaluation.

[7.4] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-15, Revision 0.

12. Inspect the top shield plug to reverify that it is properly seated onto the 24PT1-DSC. If not, lower the cask and reposition the top shield plug. Repeat Steps 11 and 12 as necessary.

13. Continue to raise the cask from the pool and spray the exposed portion of the cask with demineralized water.

14. Drain any excess water from the top of the 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.

16. Lift the cask from the fuel pool. As the cask is raised from the pool, continue to spray the cask with demineralized water.

17. Move the cask with loaded 24PT1-DSC to the cask decon area.

8.1.1.3 24PT1-DSC Drying and Backfilling

1. Verify that the transfer cask dose rates are compliant with limits specified in Technical Specification 5.2.4. Temporary shielding may be installed as necessary to minimize personnel exposure. Liquid neutron shield, if left unfilled for weight reduction, shall be filled.

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, remove the eyebolts. Disengage the lifting yoke from the trunnions and move it clear of the cask.

4. Decontaminate the exposed surfaces of the 24PT1-DSC shell perimeter and remove the inflatable cask/24PT1-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 24PT1-DSC shell. Take swipes around the outer surface of the shell and check for smearable contamination in accordance with Chapter 12 limits.

6. Install the automated welding machine onto the inner top cover plate and place the inner top cover plate with the automated welding machine onto the 24PT1-DSC. Verify proper fit-up of the inner top cover plate with the shell.

7. Check radiation levels along the surface of the inner top cover plate. Temporary shielding may be installed as necessary to minimize personnel exposure.

8. Connect the vacuum drying system (VDS) to the 24PT1-DSC and use the liquid pump to drain approximately 60 gallons to the fuel pool. This will lower the water level about four inches below the bottom of the shield plug.

9. Disconnect the VDS from the 24PT1-DSC.

10. Cover the cask/24PT1-DSC annulus to prevent debris and weld splatter from entering the annulus.

11. Continuous hydrogen monitoring during the welding of the inner cover plate is required [[8.1] and Technical Specification 4.2.6]. Connect a hydrogen monitor to the vent port using tygon tubing or a quick disconnect stem fitting to allow continuous monitoring of the hydrogen atmosphere in the 24PT1-DSC cavity during welding of the inner cover plate. The 24PT1-DSC internal pressure is to be maintained at atmospheric pressure during welding of the inner top cover plate.

12. Ready the automated welding machine and tack weld the inner top cover plate to the 24PT1-DSC shell. Verify that the measured hydrogen concentration does not exceed a safety limit of 2.4% [8.1]. If this limit is exceeded, stop all welding operations and purge the 24PT1-DSC cavity with 2-3 psig helium (or other inert medium) via vent port to reduce hydrogen concentration safely below the 2.4% limit. Complete the inner top cover plate weldment and remove the automated welding machine.

13. Perform dye penetrant weld examination of the inner top cover plate weld.

14. If required to limit elastic deformation of inner top cover plate, place the strongback so that it sits on the inner top cover plate and is oriented such that:

    - The siphon and vent ports are accessible.

    - The strongback stud holes line up with the cask lid bolt holes.

15. Lubricate the studs and, using a crossing pattern, adjust the strongback studs to snug tight ensuring approximately even pressure on the cover plate.

16. Connect the VDS to the siphon and vent ports.

17. Install temporary shielding to minimize personnel exposure throughout the subsequent welding operations as required.

18. Engage the nitrogen or helium supply and open the valve on the vent port and allow compressed gas to force the water from the 24PT1-DSC cavity through the siphon port.

28. Open the valve on the vent port and allow helium to flow into the cavity to pressurize the 24PT1-DSC in accordance with the limits specified in Chapter 12.

29. Close the valves on the helium source.

NOTE: If during drying and backfilling the system is inadvertently vented, re-evacuation and backfilling with helium will be required.

8.1.1.4 24PT1-DSC Sealing Operations

1. Disconnect the VDS from the 24PT1-DSC. Seal weld the prefabricated covers over the vent and siphon ports and perform a dye penetrant weld examination.

2. Install the automated welding machine onto the outer top cover plate and place the outer top cover plate with the automated welding system onto the 24PT1-DSC. Verify proper fit up of the outer top cover plate.

3. Tack weld the outer top cover plate to the 24PT1-DSC shell. Place the outer top cover plate weld root pass.

4. Perform dye penetrant examination of the root pass weld. Perform a leak test of the inner top cover to DSC shell weld and siphon/vent cover welds in accordance with the requirements of ANSI N14.5 and Technical Specification 5.2.4 to demonstrate the covers are leaktight. Verify that the personnel performing the leak test are qualified in accordance with SNT-TC-1A. Note: This test may be performed using a test head following Step 1 as an alternate.

5. If a leak is found, remove the outer top cover plate root pass, the vent and siphon port plugs and repair the inner top cover plate welds. Repeat steps from 8.1.1.3-21.

6. Weld out the outer top cover plate to the shell and perform dye penetrant examination on the weld surface.

7. Open the cask drain port valve and remove the remaining water from the cask/24PT1-DSC annulus.

8. Remove the automated welding machine from the 24PT1-DSC.

9. Rig the cask top cover plate and lower the cover plate onto the cask.

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

11. Check the radiation levels along the perimeter of the cask.

8.1.1.5 Transfer Cask Downending and Transport to ISFSI

1. Verify liquid neutron shield, if used, is filled. Re-attach the transfer cask lifting yoke to the crane hook, as necessary. Ready the transfer trailer and cask support skid for service.

20. Install the AHSM door using a portable crane and secure it in place. Verify that the AHSM dose rates are compliant with the limits specified in Technical Specification 5.2.4.

21. Replace the transfer cask 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 ISFSI security measures.

24. Adjust the seismic restraint on the 24PT1-DSC one week following initial placement.

8.1.1.7 Monitoring Operations

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

2. Perform a temperature measurement for each AHSM on a daily basis in accordance with Technical Specification 5.2.5 requirements. Temperature monitoring is provided to alert operators to a possible blocked vent condition.

   The basis for temperature monitoring limits to be used as a function of thermocouple location are provided in Section 4.4.2.

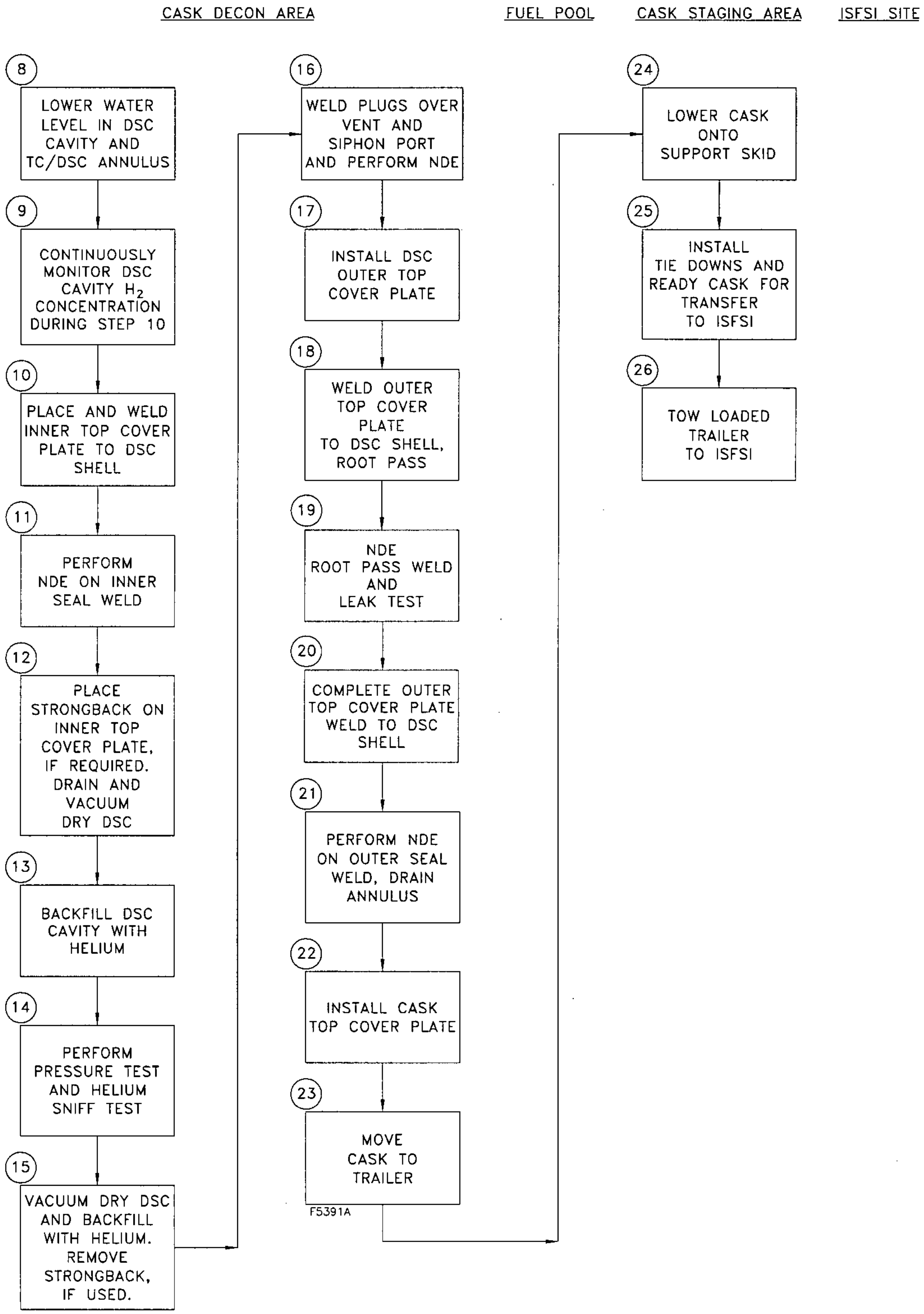

Figure 8.1-1
Advanced NUHOMS® System Loading Operations Flow Chart (continued)

2. Place an exhaust hood or tent over the 24PT1-DSC, if necessary. The exhaust should be filtered or routed to the site radwaste system.

3. Drill a hole through the top cover plate to expose the siphon port quick connect.

4. Drill a second hole through the top cover plate to expose the vent port quick connect.

   CAUTION:

   (a) The water fill rate must be regulated during this reflooding operation to ensure that the DSC vent pressure does not exceed 20.0 psig.

   (b) Per Technical Specification 4.2.6, provide for continuous hydrogen monitoring of the DSC cavity atmosphere during all subsequent cutting operations to ensure that a safety limit of 2.4% hydrogen concentration is not exceeded. Purge with 2-3 psig helium (or any other inert medium) as necessary to maintain the hydrogen concentration safely below this limit.

5. Obtain a sample of the 24PT1-DSC atmosphere (confirm acceptable hydrogen concentration). Fill the 24PT1-DSC with water from the fuel pool through the siphon port with the vent port open and routed to the plant's off-gas system.

6. Place welding blankets around the transfer cask and scaffolding.

7. Using plasma arc-gouging, a mechanical cutting system or other suitable means, remove the seal weld from the outer top cover plate and 24PT1-DSC shell. A fire watch should be placed on the scaffolding with the welder, as appropriate. The exhaust system should be operating at all times.

8. The material or waste from the cutting or grinding process should be treated and handled in accordance with the plant's low level waste procedures unless determined otherwise.

9. Remove the top of the tent, if necessary.

10. Remove the exhaust hood, if necessary.

11. Remove the outer top cover plate.

12. Reinstall tent and temporary shielding, as required. Remove the seal weld from the inner top cover plate to the shell in the same manner as the outer cover plate. Remove the inner top cover plate. Remove any remaining excess material on the inside shell surface by grinding.

13. Clean the transfer cask surface of dirt and any debris which may be on the transfer cask surface as a result of the weld removal operation. Other procedures which

[8.2] SNT-TC-1A, "American Society of Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

Base materials and welds are examined in accordance with the applicable codes (e.g., ASME, ASTM, AWS). All welding is performed using qualified processes and qualified personnel according to the applicable code requirements (e.g., ASME, AWS). NDE requirements for welds are specified on the drawings provided in Chapter 1. Weld-related NDE is performed in accordance with written and approved procedures. NDE personnel are qualified in accordance with SNT-TC-1A [9.3].

The confinement welds on the 24PT1-DSC are designed, fabricated, tested and inspected in accordance with ASME B&PV Code Subsection NB [9.2] including the alternatives to the ASME Code specified in Table 3.1-4 and discussed in Chapter 3 and 7.

24PT1-DSC non-confinement welds are inspected to the NDE acceptance criteria of ASME B&PV Code Subsection NG or NF, based on the applicable code for the components welded.

24PT1-DSC lifting lugs are provided for empty DSC handling operations only. These operations are performed away from plant safety related equipment; therefore these lifting lugs are not subject to load testing requirements of ANSI N14.6 [9.4] for heavy loads.

9.1.3 Leak Tests and Hydrostatic Pressure Tests

The DSC confinement boundary except inner top cover to the DSC shell weld is pressure tested at the fabricator's shop in accordance with ASME Article NB-6300.

The inner top cover to the DSC shell weld is pressure tested after the fuel assemblies are loaded in the canister. This test is in accordance with the alternatives to the ASME code specified in Table 3.1-14.

DSC confinement welds in the DSC shell and bottom are leak tested at the fabricator's shop to an acceptance criterion of 1×10^{-7} ref cm^3/s, i.e., "leaktight" as defined in ANSI N14.5 [9.5]. Personnel performing the leak test are qualified in accordance with SNT-TC-1A [9.3].

The weld between the DSC shell and inner top cover and siphon vent cover welds are also leak tested to an acceptance criteria of 1×10^{-7} ref cm^3/s after the fuel assemblies are loaded in the canister.

9.1.4 Components

Components that comprise the Advanced NUHOMS® System and which perform an important-to-safety function are described in Chapter 2. The Advanced NUHOMS® System important-to-safety components do not include any active components requiring testing or any additional testing beyond that described in Sections 9.1.1, 9.1.2 and 9.1.3 above.

9.1.4.1 Valves, Rupture Discs, and Fluid Transport Devices

There are no valves performing a function important to safety. The Advanced NUHOMS® System design does not utilize valves, rupture discs, or fluid transport devices.

9.1.4.2 Gaskets

There are no gaskets performing a function important to safety. The Advanced NUHOMS® System design does not utilize gasket devices.

9.4 Supplemental Information

9.4.1 References

[9.1] Electric Power Research Institute, "NUHOMS® Modular Spent-Fuel Storage System: Performance Testing," EPRI Report NP-6941, September 1990.

[9.2] ASME Boiler and Pressure Vessel Code, Section III, 1992 Edition with 1994 Addenda.

[9.3] SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1984.

[9.4] ANSI N14.6, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials," New York, 1996.

[9.5] ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials", February 1998.

[9.6] ASTM E94, "Guide for Radiographic Testing", 1993.

[9.7] ASTM E142, "Method for Controlling Quality of Radiographic Testing", 1992.

[9.8] ASTM E545, "Method for Determining Image Quality in Direct Thermal Neutron Radiographic Examination", 1991.

[9.9] Nuclear Regulatory Commission, Safety Evaluation Report of Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, December 1994, USNRC Docket Number 72-1004, File NUH003.0103.02.

[9.10] Rancho Seco Nuclear Plant ISFSI FSAR, Revision 0, November 2000, USNRC Docket Number 72-11.

[9.11] American Concrete Institute, Code Requirements for Nuclear Safety Related Concrete Structures and Commentary, ACI 349-97 and ACI 349R-97, American Concrete Institute, Detroit, Michigan.

[9.12] Hahn, G. J., "Statistical Intervals for a Normal Population, Part I," Journal of Quality Technology, Vol. 2, No. 3, July 1970.

[9.13] Hahn, G. J., "Statistical Intervals for a Normal Population, Part II," Journal of Quality Technology, Vol. 2, No. 4, October 1970.

[9.14] Owen, D. B., "A Survey of Properties and Applications of the Noncentral t-Distribution," Technometrics, Vol. 10, No. 3, August 1968.

[9.15] "SCR607, Factors for One-Sided Tolerance Limits and for Variables Sampling Plans," U.S. Department of Energy, Sandia Corporation, March 1963.

11.3 Supplemental Information

11.3.1 References

[11.1] American National Standards Institute, American Nuclear Society, ANSI/ANS-57.9-1984, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type), 1984.

[11.2] NRC Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," Revision 1, 1973.

[11.3] LS-DYNA Version 950(C), User's Manual, May 1999, Livermore Software Technology Corporation.

[11.4] NRC NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Revision 2, July 1981.

[11.5] NRC Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," October 1973.

[11.6] Swanson Analysis Systems Inc., ANSYS Engineering Analysis System User's Manual, Version 5.3, Pittsburgh, PA.

[11.7] CFR Title 10, Part 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

[11.8] Not used.

[11.9] NRC Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants, April 1974.

[11.10] American Concrete Institute, Code Requirements for Nuclear Safety Related Concrete Structures and Commentary, ACI 349-97 and ACI 349R-97, American Concrete Institute, Detroit, MI.

[11.11] J. Roark and W. C. Young, Formulas for Stress and Strain, Sixth Edition, McGraw-Hill, New York, N.Y., (1989).

[11.12] "Fluid Mechanics," Raymond C. Binder, 4th Edition, Prentice-Hall, Inc.

[11.13] American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, 1992 Edition with 1994 Addenda.

[11.14] American Society of Civil Engineers, ASCE 7-95, Minimum Design Loads for Buildings and Other Structures, (formerly ANSI A58.1).

[11.15] American Society of Civil Engineers, ASCE Manual No. 58, Structural Analysis and Design of Nuclear Plant Facilities, 1980.

A.2 PRINCIPAL DESIGN CRITERIA

A.2.1 Spent Fuel to be Stored

Sections of this Chapter have been identified as "No change" due to the addition of 24PT4-DSC to the Advanced NUHOMS® system. For these sections, the description or analysis presented in the corresponding sections of the FSAR for the Advanced NUHOMS® system with 24PT1-DSC is also applicable to the system with 24PT4-DSC.

The NUHOMS® 24PT4-DSC is designed to accommodate the storage of 24 Westinghouse-CENP 16x16 (CE 16x16) PWR fuel assemblies.

This payload consists of intact (including reconstituted) and/or damaged CE 16x16 fuel assemblies with Zircaloy/ZIRLO™ cladding and UO_2, $(U,Er)O_2$ or $(U,Gd)O_2$ fuel pellets. Assemblies may be with, or without, integral burnable poison rods. The thermal and radiological characteristics for the PWR spent fuel were determined using the SCALE computer code package. Spent fuel with various combinations of burnup, enrichment, and cooling time can be stored in the 24PT4-DSC provided the values for decay heat, gamma and neutron sources, remain within the design limits specified in Table A.2.1-1 and Table A.2.1-2.

A.2.1.1 Detailed Payload Description

The spent fuel to be stored in the NUHOMS® 24PT4-DSC consists of intact (including reconstituted) Westinghouse-CENP 16x16 (CE 16x16) and/or damaged CE 16x16 fuel assemblies with Zircaloy or ZIRLO™ cladding and UO_2 or $(U,Er)O_2$ or $(U,Gd)O_2$ fuel pellets. Assemblies are with or without Integral Fuel Burnable Absorber (IFBA) rods or integral burnable poison rods.

Each 24PT4-DSC can accommodate a maximum of 12 damaged fuel assemblies, with the remaining assemblies intact.

Reconstituted assemblies containing up to eight replacement stainless steel rods in place of damaged fuel rods (these rods must displace an amount of water equal to or greater than that displaced by the original fuel rods in the active fuel region of the fuel assembly) or replacement Zircaloy clad uranium rods (any number per assembly) are acceptable for storage in the 24PT4-DSC as either intact or damaged assemblies.

Damaged fuel may include assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial and/or missing rods (i.e., extra water holes). Damaged fuel assemblies shall be encapsulated in individual Failed Fuel Cans in locations as shown in Figure A.2.1-4.

Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a Rod Storage Basket are also considered as damaged fuel. A Rod Storage Basket is a 9x9 array of tubes in a lattice that has approximately the same dimensions as a standard fuel assembly. Rod storage baskets may also include IFBA and integral burnable poison rods. Fuel debris not contained in a Rod Storage Basket may also be placed in a Failed Fuel Can for storage provided the size of the debris is larger than the Failed Fuel Can screen mesh opening. Fuel

Table A.2.1-1
PWR Fuel Specification of Intact Fuel to be Stored in NUHOMS® 24PT4-DSC

Fuel Design:	Intact CE 16x16 PWR fuel assembly or equivalent reload fuel that is enveloped by the fuel assembly design characteristics as listed in Table A.2.1-3 and the following requirements:
Fuel Damage:	Fuel with known or suspected cladding damage in excess of pinhole leaks or hairline cracks or an assembly with partial and/or missing rods is not authorized to be stored as "intact PWR Fuel."
Physical Parameters$^{(1)}$	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500$^{(2)\ (3)}$
Max. U Content (kg)	455.5
No. of Assemblies per DSC	≤ 24 intact assemblies
Fuel Cladding	Zircaloy-4 or ZIRLO™
Reconstituted Fuel Assemblies	Damaged fuel rods replaced by either stainless rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly).
Nuclear and Radiological Parameters	
Maximum Initial ^{235}U Enrichment (wt %)	Per Table A.2.1-4 and Figure A.2.1-4
Fuel Assembly Average Burnup and Cooling Time	Per Tables A.2.1-5, A.2.1-6, A.2.1-7, A.2.1-8. For Reconstituted Fuel with stainless steel replacement rods, per Tables A.2.1-9, A.2.1-10, A.2.1-11, A.2.1-12.
Decay Heat	Per Figure A.2.1-1, A.2.1-2 or A.2.1-3.

Notes:

(1) Nominal values shown unless stated otherwise.

(2) Does not include weight of Poison Rodlets (25 lbs each) installed in accordance with Table A.2.1-4.

(3) Includes the weight of fuel assembly Poison Rods installed for 10CFR50 criticality control in spent fuel pool racks.

Table A.2.1-2
PWR Fuel Specifications of Damaged Fuel to be Stored in NUHOMS® 24PT4-DSC

Fuel Design:	Damaged CE 16x16 PWR fuel assembly or equivalent reload fuel that is enveloped by the fuel assembly design characteristics as listed in Table A.2.1-3 and the following requirements:
Fuel Damage:	Damaged fuel may include assemblies with known or suspected cladding defects greater than pinhole leaks or hairline cracks or an assembly with partial and/or missing rods (i.e., extra water holes). Damaged fuel assemblies shall be encapsulated in individual Failed Fuel Cans and placed in Zones A and/or B as shown in Figure A.2.1-4. Fuel debris and damaged fuel rods that have been removed from a damaged fuel assembly and placed in a Rod Storage Basket are also considered as damaged fuel. Loose fuel debris, not contained in a Rod Storage Basket may also be placed in a Failed Fuel Can for storage, provided the size of the debris is larger than the Failed Fuel Can screen mesh opening. Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met.
Physical Parameters[(1)]	
Unirradiated Length (in)	176.8
Cross Section (in)	8.290
Assembly Weight (lbs)	1500[(2) (3)]
Max. U Content (kg)	455.5
No. of Assemblies per DSC	≤ 12 damaged assemblies, balance intact.
Fuel Cladding	Zircaloy-4 or ZIRLO™
Reconstituted Fuel Assemblies	Damaged fuel rods replaced by either stainless rods (up to 8 rods per assembly) or Zircaloy clad uranium rods (any number of rods per assembly).
Nuclear and Radiological Parameters	
Initial ^{235}U Enrichment (wt %)	Per Table A.2.1-4 and Figure A.2.1-4.
Fuel Assembly Average Burnup and Cooling Time	Per Tables A.2.1-5, A.2.1-6, A.2.1-7, A.2.1-8. For Reconstituted Fuel with stainless steel replacement rods, per Tables A.2.1-9, A.2.1-10, A.2.1-11, A.2.1-12.
Decay Heat	Per Figure A.2.1-1, or A.2.1-2 or A.2.1-3.

Notes:

(1) Nominal values shown unless stated otherwise.

(2) Does not include weight of Poison Rodlets (25 lbs each) installed in accordance with Table A.2.1-4.

(3) Includes the weight of fuel assembly Poison Rods installed for 10CFR50 criticality control in spent fuel pool racks.

A.2.2.3.1 Input Criteria

No change.

A.2.2.3.2 Seismic-System Analyses

No change.

A.2.2.4 Snow and Ice Loadings

No change.

A.2.2.5 Tsunami

No change.

A.2.2.6 Lightning

No change.

A.2.2.7 Combined Load Criteria

A.2.2.7.1 Advanced Horizontal Storage Module

No change.

A.2.2.7.2 24PT4-DSC

The 24PT4-DSC is designed by analysis to meet the stress intensity allowables of the ASME Boiler and Pressure Vessel Code (1992 Edition with 1994 Addenda) Section III, Division I, Subsections NB with the alternatives to the ASME Code specified in Table A.3.1-5, NG and NF for Class 1 components and supports [A2.1] as described in Tables A.3.1-5 and A.3.1-6. The 24PT4-DSC is conservatively designed by utilizing linear elastic or non-linear elastic-plastic analysis methods. The load combinations considered for the 24PT4-DSC normal, off-normal and postulated accident loadings are described in Chapter A.3. ASME Code Service Level A and B allowables are used for normal and off-normal operating conditions. Service Level C and D allowables are used for accident conditions such as a postulated cask drop accident. Use of these acceptance criteria ensures that in the event of a design basis drop accident, the 24PT4-DSC confinement boundary is not breached. The maximum shear stress theory is used to calculate principal stresses. Normal operational stresses are combined with the appropriate off-normal and accident stresses. It is assumed that only one postulated accident condition occurs at any one time. The accident analyses are documented in Chapter A.11. The structural evaluation for the 24PT4-DSC is documented in Chapter A.3.

A.2.2.8 Burial Under Debris

No change.

assures that the 24PT4-DSC surface loose contamination levels are within those required for shipping cask externals. Compliance with these contamination limits is assured by taking surface swipes of the upper end of the 24PT4-DSC before transferring the cask from the fuel building.

Once inside the 24PT4-DSC, the SFAs are confined by the 24PT4-DSC shell and the top and bottom shield plug assemblies. The fuel cladding integrity is ensured by maintaining the storage cladding temperatures below levels which are known to cause degradation of the cladding. In addition, the SFAs are stored in an inert atmosphere to prevent degradation of the cladding, specifically cladding rupture due to oxidation and its resulting volumetric expansion of the fuel. Thus, a helium atmosphere for the 24PT4-DSC is incorporated in the design to protect the fuel cladding integrity by inhibiting the ingress of oxygen into the cavity.

Helium is known to leak through valves, mechanical seals, and escape through very small passages because it has a small atomic diameter, is an inert element, and exists as a monatomic species. Helium will not, to any practical extent, diffuse through stainless steel. For this reason the 24PT4-DSC has been designed as a welded confinement pressure vessel with no mechanical or electrical penetrations. See Chapter A.7 for a detailed discussion of the confinement boundary design.

The 24PT4-DSC itself has a series of barriers to ensure the confinement of radioactive materials. The cylindrical shell is fabricated from rolled ASME stainless steel plate which is joined with full penetration welds that are 100% inspected by non-destructive examination. All top end closure welds are multiple-layer welds. This effectively eliminates any pinhole leaks which might occur in a single pass weld, since the chance of pinholes being in alignment on successive weld passes is not credible. Furthermore, the top shield plug assembly and outer cover plate are sealed by separate, redundant closure welds. Pressure boundary welds and welders are qualified in accordance with Section IX of the ASME B&PV Code and inspected according to the appropriate articles of Section III, Division 1, Subsection NB (including the alternatives to the ASME code specified in Table A.3.1-5). These criteria insure that the as-deposited weld filler metal is as sound as the parent metal of the pressure vessel.

Pressure monitoring instrumentation is not used since penetration of the pressure boundary would be required. The penetration itself would then become a potential leakage path and by its presence compromise the integrity of the 24PT4-DSC design. The shell and welded cover plates provide total confinement of radioactive materials. Once the 24PT4-DSC is sealed, there are no credible events, as discussed in Chapter A.11, which could fail the cylindrical shell or the closure plates which form the confinement boundary.

A.2.3.2.2 <u>24PT4-DSC Cooling</u>

No change.

A.2.6 Supplemental Information

A.2.6.1 References

[A2.1] American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1992 Edition with 1994 Addenda.

[A2.2] Interim Staff Guidance No. 11, Revision 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel."

[A2.3] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," 1997.

All pressure boundary components are constructed of Type 316 stainless steel. Non-pressure boundary components welded to the pressure boundary components are also constructed of Type 316 stainless steel. The lead shield plugs are made of ASTM B29 lead encased in stainless steel.

The 24PT4-DSC cylindrical shell and bottom end assembly including the bottom shield plug assembly, outer bottom cover plate, and the grapple ring assembly, and the internal basket assembly, are shop-fabricated components. The top shield plug assembly and the outer top cover plate are shop-fabricated and tested for fit-up but installed at the plant after the spent fuel assemblies have been loaded into the 24PT4-DSC internal basket.

The 24PT4-DSC shell assembly is designed, fabricated, examined and tested in accordance with the requirements of Subsection NB of the ASME Code including the alternatives to the ASME Code specified in Table A.3.1-5 [A3.2] for closure welds. The circumferential and longitudinal shell plate weld seams are full penetration butt welds. The butt weld joints are fully radiographed and inspected according to the requirements of NB-5000 of the ASME Boiler and Pressure Vessel Code.

The 24PT4-DSC top closure is designed, fabricated and inspected to meet ASME Code with the alternatives specified in Table A.3.1-5. The inner top cover plate of the top shield plug assembly is welded to the 24PT4-DSC shell to complete the pressure boundary as shown in Figure A.3.1-2. The outer top cover plate is sealed by a separate, redundant closure weld. All closure welds are multiple-layer welds and are examined by multi-level liquid penetrant methods to effectively eliminate leaks through welds.

The top end assembly of the 24PT4-DSC design incorporates a vent/siphon block, with two small-diameter penetrations into the 24PT4-DSC cavity for draining and filling operations. The vent port is terminated at the bottom of the shield plug assembly. The other port is attached to a siphon tube, which continues to the bottom of the 24PT4-DSC cavity. The ports include dog-leg type offsets to minimize radiation streaming. The vent and siphon ports terminate in normally closed quick-connect fittings.

During fabrication, leak tests of the 24PT4-DSC shell assembly are performed in accordance with ANSI N14.5-1997 [A3.4] to demonstrate that the shell assembly, including the bottom closure, is leak tight. The top closure weld between the inner top cover plate and DSC shell, including the vent and siphon covers are also leak tested per ANSI N14.5-1997 after fuel load, draining and drying operations and the pressure test are completed.

The stringent design and fabrication requirements described above ensure that the pressure retaining confinement function is maintained for the design life of the 24PT4-DSC. Pressure monitoring instrumentation is not used since penetration of the pressure boundary would be required. The penetration itself would then become a potential leakage path and, by its presence, compromise the leaktightness of the 24PT4-DSC design.

Transfer of the 24PT4-DSC from the TC into the AHSM is performed using a hydraulic ram that applies a load to the outer bottom cover plate, at the center of the 24PT4-DSC. During insertion of the 24PT4-DSC into the AHSM, the load is shared by the outer bottom cover plate and the inner bottom cover plate.

Frictional loads during 24PT4-DSC transfer are reduced by application of a dry film lubricant to the hardened nitronic surface of the AHSM support rails and the TC. The lubricant chosen for this application is a tightly adhering inorganic lubricant with an inorganic binder. The dry film lubricant provides a thin, clean, dry, layer of lubricating solids that is intended to reduce wear, and prevent galling in metals. It is applied as a thin sprayed coating, similar to paint, using a carefully controlled process. The lubricant is not affected by water and is designed to be highly resistant to aggressive chemicals. This product is designed for radiation service and has a low coefficient of sliding friction for stainless steel.

The internal basket assembly, shown in Figure A.3.1-3, provides structural support for and geometric separation of the SFAs. The basket assembly consists of 24 stainless steel guidesleeve assemblies, 28 carbon steel spacer discs, and four-support rod/spacer sleeve assemblies. The support rods and spacer sleeves are fabricated of precipitation hardened martensitic stainless steel.

The spacer disc details, shown in Figure A.3.1-4, identify the twenty-four cutouts for the SFAs and the four support rods. The spacer discs maintain cross-sectional spacing and support for the fuel assemblies and the guidesleeves when the 24PT4-DSC is in the horizontal position. When the 24PT4-DSC is in the vertical position, the spacer discs are held in place by the support rods and spacer sleeves; the rod assemblies maintain longitudinal separation between discs during all normal operating and postulated accident conditions. Fuel weight is transferred to the top or bottom cover plates by direct bearing.

Damaged fuel assemblies are stored in Failed Fuel Cans. The Failed Fuel Can is provided with a welded bottom closure and a removable top closure. Slots are provided in the Failed Fuel Can to allow independent removal of the can and the enclosed fuel assembly. Failed Fuel Cans are provided with screens at the bottom and top to contain fuel debris and allow fill/drainage of water from the Failed Fuel Can.

A.3.1.1.2 General Description of the AHSM

No change.

A.3.1.2 24PT4-DSC and AHSM Design Criteria

No change.

A.3.1.2.1 24PT4-DSC Design Criteria

A.3.1.2.1.1 Stress Criteria

No change.

The 24PT4-DSC is designed utilizing linear elastic and non-linear elastic-plastic analytical methods. ASME Code Service Level A and B allowables are used for normal and off-normal operating conditions, respectively. Service Level C and D allowables are used for accident conditions.

The 24PT4-DSC shell is designed by analysis to meet the criteria of the ASME Boiler and Pressure Vessel Code Section III, Division 1, Subsection NB, 1992 Edition through 1994

Addenda [A3.2], supplemented by the alternatives to the ASME Code specified in Table A.3.1-5, ISG-15 [A3.3] and ISG-18 [A3.18] . Stress criteria for pressure boundary components are summarized in Table 3.1-2. Stress criteria for (partial penetration) pressure boundary top closure welds are summarized in Table A.3.1-2. The major internal basket components, spacer discs and guide sleeve assemblies, are designed to the criteria of ASME B&PV Code, Subsection NG as summarized in Table 3.1-4, supplemented by Code Case N-499-1 (for the spacer discs). The support rods and spacer sleeves are designed to the criteria of ASME B&PV Code, Subsection NF. The Boral® neutron absorbing material is non-Code and is not considered a load-carrying component.

A.3.1.2.1.2 Stability Criteria

No change.

A.3.1.2.1.3 Loads and Load Combinations

The load combinations for the 24PT4-DSC are summarized in Table A.3.1-3.

A.3.1.2.1.3.1 Deadweight

No change.

A.3.1.2.1.3.2 Internal and External Pressure

Internal pressure loads for the 24PT4-DSC are developed as described in Chapter A.4. The bounding normal, off-normal, and accident pressures used for the structural analyses of the 24PT4-DSC are given in Table A.3.1-4.

Load cases which include external pressures for the 24PT4-DSC are the same as those given in Table 3.1-7.

The internal basket components, such as the support rod assemblies, spacer discs, and guidesleeves, are not affected by pressure loads.

A.3.1.2.1.3.3 Thermal Loads

No change.

A.3.1.2.1.3.4 DSC Transfer/Handling Loads

No change.

A.3.1.2.1.3.5 Cask Drop

No change.

A.3.1.2.1.3.6 Seismic Loads

No change.

Table A.3.1-1
Codes and Standards for the Design, Fabrication and Construction of 24PT4-DSC Principal Components

Component, Equipment, Structure	Code of Construction
24PT4-DSC	ASME Code, Section III, 1992 Edition through 1994 Addenda, supplemented by the alternatives to the ASME Code specified in Table A.3.1-5, Code Case N-499-1, ISG-15 and ISG-18

Table A.3.1-5
Alternatives to the ASME Code for the 24PT4-DSC (NB)

Reference ASME Code Section/Article	Code Requirement	Alternatives, Justification & Compensatory Measures
NCA	All	Not compliant with NCA
NB-1100	Requirements for Code Stamping of Components	The 24PT4-DSC shell is designed & fabricated in accordance with the ASME Code, Section III, Subsection NB to the maximum extent practical. However, Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.
NB-2130 NB-4121	Material must be supplied by ASME approved material suppliers Material Certification by Certificate Holder	All materials designated as ASME on the FSAR drawings are obtained from ASME approved MO or MO supplier(s) with ASME CMTR's. 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 & certification are maintained in accordance with TN's NRC approved QA program.
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 plate weld, and the siphon/vent cover welds, 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 A.3.1-5
Alternatives to the ASME Code for the 24PT4-DSC (NB)
Concluded

NB-2531	Vent & siphon Port Cover; straight beam UT per SA-578 for all plates for vessel	SA-578 applies to 3/8" and thicker plate only; allow alternate UT techniques to achieve meaningful UT results.
NB- 6000	All completed pressure retaining systems shall be pressure tested	The 24PT4 is not a complete or "installed" pressure vessel until the top closure is welded following placement of Spent Fuel Assemblies within the DSC. Due to the inaccessibility of the shell and lower end closure welds following fuel loading and top closure welding, the pressure testing of the DSC is performed in two parts. The DSC shell and shell bottom, including all longitudinal and circumferential welds, is pneumatically tested and examined at the fabrication facility. The shell to the inner top cover plate closure weld is pressure tested and examined for leakage in accordance with NB-6300 following fuel loading, draining, and drying operations. The siphon/vent cover welds will not be pressure tested; these welds and the shell to the inner top cover plate 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, 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 24PT4-DSC. The function of the 24PT4-DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The 24PT4-DSC is designed to withstand the maximum internal pressure considering 100% fuel rod failure at maximum accident temperature. The 24PT4-DSC is pressure tested to 120% of normal operating design pressure. An overpressure protection report is not prepared for the 24PT4-DSC.
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000	The 24PT4-DSC nameplate provides the information required by 10CFR 71, 49CFR 173 and 10CFR 72 as appropriate. Code stamping is not required for the 24PT4-DSC. QA data packages are prepared in accordance with the requirements of 10CFR 71, 10CFR 72 and TN's approved QA program.

Figure Withheld Under 2.390

Figure A.3.1-2
Advanced NUHOMS® System 24PT4-DSC Pressure Boundary Location

A.3.4 General Standards for 24PT4-DSC and AHSM

A.3.4.1 Chemical and Galvanic Reactions

The review for chemical and galvanic reactions presented in Section 3.4.1 for the 24PT1-DSC is applicable to the 24PT4-DSC, since fuel loading, unloading, handling and storage processes and environments are similar for both the 24PT1 and the 24PT4-DSCs. The following applies specifically to the 24PT4-DSC:

- Materials used for the 24PT4-DSC are shown in the Parts List of Drawing ANUH-01-4001 in Section A.1.5.2.
- From a chemical and galvanic reaction standpoint, the only differences between the 24PT4-DSC and the 24PT1-DSC designs are the shell assembly top and bottom ends which include stainless steel-encased and sealed lead in the shield plugs. The lead is not exposed to the external environment and is thus not subject to any chemical reactions. Both the 24PT1-DSC spacer discs and the 24PT4-DSC spacer discs are fabricated from Carbon Steel and plated with electroless nickel.

A.3.4.1.1 Behavior of Austenitic Stainless Steel in Borated Water

No change.

A.3.4.1.2 Behavior of Boral® in Borated Water

No change.

A.3.4.1.3 Electroless Nickel Plated Carbon Steel

No change.

A.3.4.1.4 Hydrogen Generation

No change.

A.3.4.1.5 Effect of Galvanic Reactions on the Performance of the System

No change.

A.3.4.2 Positive Closure

Positive closure is provided by the redundant closure welds consisting of the inner top cover plate of the shield plug assembly-to-shell weld, the vent and siphon cover plate welds, the outer top cover plate-to-shell weld, and the leaktight 24PT4-DSC shell assembly.

A.3.7 References

[A3.1] Nuclear Regulatory Commission, Safety Evaluation Report of Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, December 1994, USNRC Docket Number 72-1004.

[A3.2] American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section II, Section III, 1992 Edition with Addenda through 1994 with Code Case N-499-1.

[A3.3] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-15, Materials Evaluation.

[A3.4] ANSI N14.5-1997, "American National Standard for Radioactive Materials, Leakage Tests on Packages for Shipment", February 1998.

[A3.5] Holman, W.R., Langland, R. T., "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick," NUREG/CR-1815, September 1981.

[A3.6] Teitz, T. E., "Determination of the Mechanical Properties of a High Purity Lead and a 0.058% Copper-lead Alloy," WADC Technical Report 57-695, ASTIA Document No. 151165, Stanford Research Institute, Menlo Park, CA, April 1958.

[A3.7] American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III, Division 1, Code Case N-499-1, Use of SA-533 Grade B, Class 1 Plate and SA-508 Class 3 Forgings and their Weldments for Limited Elevated Temperature Service, Section III, Division 1; Approval Date: December 12, 1994, Reaffirmed October 2, 2000, Expires October 2, 2003.

[A3.8] Transnuclear, Inc., Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 9, February 2006, USNRC Docket Number 72-1004.

[A3.9] Swanson Analysis Systems Inc., ANSYS Engineering Analysis System User's Manual, Versions 5.3, 5.6.2, and 8.1, Swanson Analysis Systems, Inc., Pittsburgh, PA.

[A3.10] Levy, Chin, Simonen, Beyer, Gilbert and Johnson, Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircalloy-Clad Fuel Rods in Inert Gas, May 1987, Pacific Northwest Laboratory, PNL Document PNL-6189.

[A3.11] Johnson, A. B. and E. R. Gilbert, Technical Basis for Storage of Zircalloy-Clad Spent Fuel in Inert Gas, September 1983, Pacific Northwest Laboratory, PNL Document PNL-4835.

[A3.12] "Consolidated Safety Analysis Report for IF-300 Shipping Cask", NEDO-10084, Vectra Technologies, Inc., Revision 4, March, 1995.

A.4.11 Supplemental Information

A.4.11.1 References

[A4.1] "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages," Office of Civilian Radioactive Waste Management, DOE/RW-0472, Revision 2, September 1998.

[A4.2] NRC NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, January 1997.

[A4.3] Bolz, R. E., G. L. Tuve, CRC Handbook of Tables for Applied Engineering Science, 2nd Edition, 1973.

[A4.4] American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III, 1992 Edition with Addenda through 1994 with Code Case N-499-1.

[A4.5] "Standard Specification for BORAL® Composite Sheet," Specification Number BPS-9000-04, AR Advanced Structures, Livonia, Michigan. PROPRIETARY.

[A4.6] E. A. Brandes (Editor), Smithells Metals Reference Book, 6th Ed., Butterworths, London, UK, 1983.

[A4.7] Roshenow, W. M., J. P. Hartnett, and Y. I. Cho, Handbook of Heat Transfer, 3rd Edition, 1998.

[A4.8] Incropera, F. P., D. P. DeWitt, Fundamentals of Heat and Mass Transfer, 4th Edition, Wiley, 1996.

[A4.9] Bucholz, J. A., Scoping Design Analysis for Optimized Shipping Casks Containing 1-, 2-, 3-, 5-, 7-, or 10-Year old PWR Spent Fuel, Oak Ridge National Laboratory, January 1983, ORNL/CSD/TM-149.

[A4.10] Siegel, R. and J. R. Howell, Thermal Radiation Heat Transfer, 2nd Edition, Hemisphere, 1981.

[A4.11] MATPRO – Version 11: A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, EG&G Idaho, Idaho Falls, February 1979, NUREG-CR/0497.

[A4.12] Transnuclear, Inc., Updated Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 9, February 2006, NRC Docket No. 72-1004.

[A4.13] Title 10, Code of Federal Regulations, Part 71 (10CFR71), Packaging and Transportation of Radioactive Materials, U.S. Nuclear Regulatory Commission, 1997.

the required test sensitivity to demonstrate leaktight construction and are used for leak testing. A helium mass spectrometer is used to detect any leakage as defined in ANSI N14.5 [A7.2].

During final drying and sealing operations of the 24PT4-DSC, the top closure confinement welds are completed to confine radioactive materials within the cavity. The inner top cover plate is welded to the shell using automated welding equipment. Once the 24PT4-DSC has been vacuum dried, a pressure test is performed by backfilling the DSC cavity with helium. Following satisfactory completion of the pressure test the vent and siphon port penetrations are welded, and the outer top cover plate is lowered onto the 24PT4-DSC. The outer top cover plate is welded in place using automated welding equipment. The outer top cover plate and associated closure weld to the shell acts as a redundant barrier for confining radioactive material within the 24PT4-DSC throughout its service life.

Leak testing of the 24PT4 DSC inner top cover plate to shell weld, vent and siphon cover plate and vent/siphon block to shell welds is performed using a test head prior to placing the outer top cover plate or by pulling a vacuum between the inner and outer top cover plates through a test port in the outer top cover plate and monitoring for helium in accordance with the requirements of ANSI N14.5 [A7.2] to demonstrate that these welds are leaktight.

A.7.1.2 Confinement Penetrations

All penetrations in the 24PT4-DSC confinement boundary are welded closed.

A.7.1.3 Seals and Welds

The austenitic stainless steel welds made during fabrication of the 24PT4-DSC that affect the confinement boundary include the weld applied to the inner bottom cover plate and the circumferential and longitudinal seam welds applied to the shell. These welds are examined (radiographic or ultrasonic and liquid penetrant) according to the requirements of Subsection NB of the ASME Code. The vent and siphon block-to-shell weld is also made during fabrication and is liquid penetrant examined in accordance with Subsection NB of the ASME Code.

The welds of the vent and siphon port covers, and the inner top cover plate to shell, completed during closure operations, and the vent and siphon block to shell weld define the confinement boundary at the top end of the 24PT4-DSC. These welds are applied using multiple-layer techniques with multi-level PT in accordance with the alternatives to the ASME Code specified in Table A.3.1-5. This effectively eliminates any pinhole leaks which might occur in a single-pass weld, since the chance of pinholes being in alignment on successive weld passes is negligibly small. Figure A.7.1-1 provides a graphic representation of the confinement boundary welds.

A.7.1.4 Closure

The 24PT4-DSC is entirely closed by welding and thus, no closure devices are utilized for confinement.

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A.7.4 Supplemental Data

A.7.4.1 Confinement Monitoring Capability

No change.

A.7.4.2 References

[A7.1] American Society of Mechanical Engineers, Boiler & Pressure Vessel Code, Section III, 1992 Edition with Addenda through 1994.

[A7.2] ANSI N14.5-1997, "American National Standard for Radioactive Materials, Leakage Tests on Packages for Shipment", February 1998.

[A7.3] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-5, Revision 1, Confinement Evaluation.

[A7.4] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-15, Revision 0, Materials Evaluation.

[A7.5] NRC Spent Fuel Project Office, Interim Staff Guidance, ISG-18, The Design Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation, dated May 2, 2003.

10. Connect the top shield plug assembly to the yoke. Position the lifting yoke and the top shield plug assembly and lower the top shield plug assembly onto the 24PT4-DSC.

CAUTION: Verify that all the lifting height restrictions specified in the Technical Specifications can be met in the following steps which involve lifting of the transfer cask.

11. Visually verify that the top shield plug assembly is properly seated onto the 24PT4-DSC.

12. Position the lifting yoke and verify that it is properly engaged with the cask trunnions.

13. Raise the transfer cask to the pool surface. Prior to raising the top of the cask above the water surface, stop vertical movement.

14. Inspect the top shield plug assembly to re-verify that it is properly seated onto the 24PT4-DSC. If not, lower the cask and reposition the top shield plug assembly. Repeat Steps 11 to 14 as necessary.

15. Continue to raise the cask from the pool and spray the exposed portion of the cask with water.

16. Drain any excess water from the top of the top shield plug assembly back to the fuel pool.

17. Deleted.

18. If required, and not done previously, drain neutron shield as required to maintain total weight within plant crane limits, while lifting cask from pool. Lift the cask from the fuel pool. As the cask is raised from the pool, continue to spray the cask with demineralized water. Disconnect annulus pressurization tank from cask.

19. Move the cask with loaded 24PT4-DSC to the cask decon area.

A.8.1.1.3 <u>24PT4-DSC Drying and Backfilling</u>

1. Verify that the transfer cask dose rates are compliant with limits specified in Technical Specification 5.2.4. Temporary shielding may be installed as necessary to minimize personnel exposure. Liquid neutron shield, if left unfilled or drained during lift from pool for weight reduction, shall be filled.

2. Place scaffolding around the cask so that any point on the surface of the cask is accessible to personnel.

3. Disengage the rigging cables from the top shield plug assembly. Eyebolts may be removed now or later. Disengage the lifting yoke from the trunnions and move it clear of the cask.

4. Decontaminate the exposed surfaces of the 24PT4-DSC shell perimeter and remove the inflatable cask/24PT4-DSC annulus seal. A neutron shield tank overflow hose may, as an option, be connected to the cask neutron shield.

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 24PT4-DSC shell. Take swipes around the outer surface of the shell and check for smearable contamination in accordance with Technical Specification limits.

6. Check radiation levels along the surface of the top shield plug assembly. Temporary shielding may be installed as necessary to minimize personnel exposure. Boiling within the 24PT4-DSC may result in an increase in dose rates.

7. Install the automated welding machine onto the top shield plug assembly.

8. Connect the vacuum drying system (VDS) to the 24PT4-DSC and use the liquid pump to remove approximately 60 gallons to the fuel pool. This will lower the water level below the bottom of the top shield plug assembly.

9. Disconnect the VDS from the 24PT4-DSC.

10. Cover the cask/24PT4-DSC annulus to prevent debris and weld splatter from entering the annulus.

11. Continuous hydrogen monitoring during the welding of the top shield plug assembly to the shell is required [[A8.1] and Technical Specification 4.2.6]. Connect a hydrogen monitor to the vent port using tygon tubing or a quick disconnect stem fitting to allow continuous monitoring of the hydrogen atmosphere in the 24PT4-DSC cavity during welding of the top shield plug assembly. The 24PT4-DSC internal pressure is to be maintained at atmospheric pressure during welding of the top shield plug assembly.

12. Not used.

13. Ready the automated welding machine and tack weld the top shield plug assembly to the 24PT4-DSC shell. Verify that the measured hydrogen concentration does not exceed a safety limit of 2.4% [A8.1]. If this limit is exceeded, stop all welding operations and purge the 24PT4-DSC cavity with 2-3 psig helium (or other inert medium) via the vent port to reduce hydrogen concentration safely below the 2.4% limit. Complete the top shield plug assembly weldment and remove the automated welding machine.

14. Perform dye penetrant weld examination of the top shield plug assembly weld.

15. Not Used.

16. Install temporary shielding to minimize personnel exposure throughout the subsequent draining/drying and welding operations as required.

NOTE: Do not use strongback during blowdown with 24PT4-DSC. Only helium is allowed for blowdown.

17. Engage the helium supply and open the valve on the vent port and allow compressed gas to force the water from the 24PT4-DSC cavity through the siphon port.

18. Once the water stops flowing from the 24PT4-DSC, close the siphon port and disengage the gas source.

19. 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 or other appropriate filtration system. Connect the VDS to a helium source.

20. Open the valve on the suction side of the pump, start the VDS and draw a vacuum on the 24PT4-DSC cavity. The cavity pressure should be reduced in steps to approximately 100 torr, 50 torr, 25 torr, 15 torr, 10 torr, 5 torr, and 3 torr. This staged drawdown will verify no ice blockage of the evacuation path. After pumping down to each level, 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 torr or less as specified in the Technical Specifications.

21. Open the valve to the vent port and fill the 24PT4-DSC cavity with helium.

NOTE: Helium gas introduced into the 24PT4-DSC shall be welding grade (>99% purity) and Important to Safety.

22. Pressurize the 24PT4-DSC with helium in accordance with Technical Specifications requirements.

23. Pressurize the 24PTH DSC cavity to 24 psig and perform a pressure test of the inner top cover plate to the requirements of NB-6300.

24. At Licensee discretion, perform a helium sniff test on the top shield plug assembly and vent/siphon block. If a leak is found, repair the weld in accordance with the Code of Construction.

25. Once no leaks are detected, depressurize the 24PT4-DSC cavity by releasing the helium through the VDS to the plant's spent fuel pool or radioactive waste system.

26. Re-evacuate the 24PT4-DSC cavity using the VDS. The cavity pressure should be reduced in steps to approximately 10 torr, 5 torr, and 3 torr. 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 torr or less in accordance with the Technical Specifications requirements.

NOTE: No time limits for vacuum drying apply since helium is present in the DSC prior to initiating vacuum drying operations per this step.

27. Open the valve on the vent port and allow helium to flow into the cavity to pressurize the 24PT4-DSC in accordance with the limits specified in the Technical Specifications.

28. Close the valves on the helium source.

NOTE: If during drying and backfilling the system is inadvertently vented, re-evacuation and backfilling with helium will be required.

A.8.1.1.4 <u>24PT4-DSC Sealing Operations</u>

1. Disconnect the VDS from the 24PT4-DSC. Seal weld the prefabricated covers over the vent and siphon ports and perform a dye penetrant weld examination.

2. Install the automated welding machine onto the outer top cover plate and place the outer top cover plate with the automated welding system onto the 24PT4-DSC. Verify proper fit up of the outer top cover plate.

3. Tack weld the outer top cover plate to the 24PT4-DSC shell. Place the outer top cover plate weld root pass. Perform dye penetrant examination of the root pass weld.

4. Perform a leak test of the inner top cover plate to DSC shell and vent and siphon cover plate welds in accordance with the requirements of ANSI N14.5 and Technical Specification 5.2.4 to demonstrate the covers are leaktight. Verify

personnel performing test are qualified in accordance with SNT-TC-1A. Note: This test may be performed using a test head following Step 1 as an alternate.

5. If a leak is found, remove the outer top cover plate root pass, the vent and siphon port plugs and repair the inner top cover plate welds. Repeat steps from 8.1.1.3-21.

6. Weld out the outer top cover plate to the shell and perform dye penetrant examination on the weld surface.

7. Open the cask drain port valve and remove the remaining water from the TC/24PT4-DSC annulus.

8. Remove the automated welding machine from the 24PT4-DSC.

9. Rig the cask top cover plate and lower the cover plate onto the cask.

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

11. Check the radiation levels along the perimeter of the cask.

A.8.1.1.5 <u>Transfer Cask Downending and Transport to ISFSI</u>

No change.

A.8.1.1.6 <u>24PT4-DSC Transfer to the AHSM</u>

No change. If a neutron shield overflow system is used, monitor to maintain water inventory in cask.

A.8.1.1.7 <u>Monitoring Operations</u>

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

2. Perform a temperature measurement for each AHSM on a daily basis in accordance with Technical Specifications. Temperature monitoring is provided to alert operators to a possible blocked vent condition.

   The basis for temperature monitoring limits to be used as a function of thermocouple location is provided in Section A.4.4.2.4.

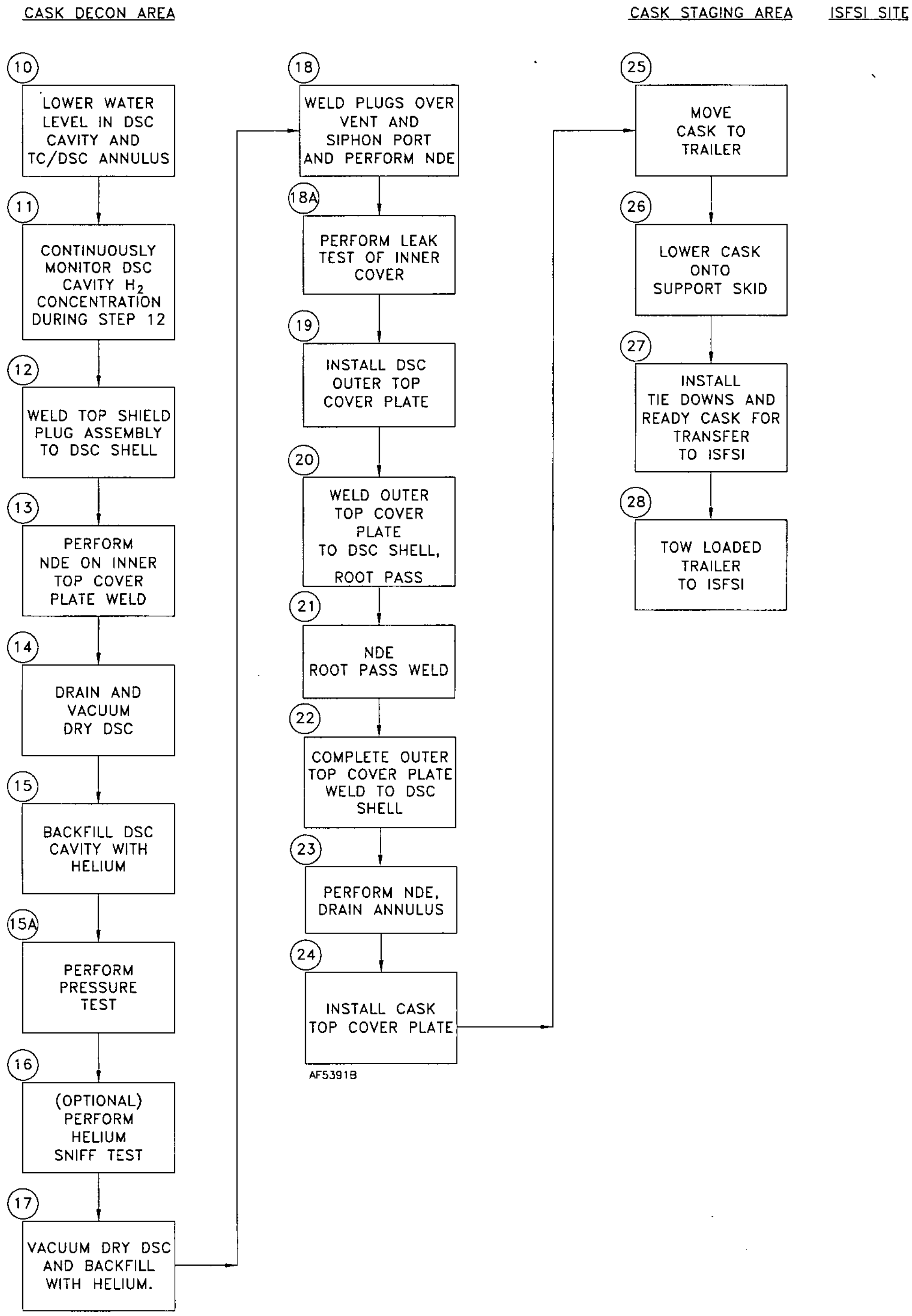

Figure A.8.1-1
Advanced NUHOMS® System Loading Operations Flow Chart
(continued)

A.8.3 Supplemental Information

A.8.3.1 Other Operating Systems

No change.

A.8.3.1.1 Component/Equipment Spares

No change.

A.8.3.2 Operation Support System

No change.

A.8.3.2.1 Instrumentation and Control System

No change.

A.8.3.2.2 System and Component Spares

No change.

A.8.3.3 Control Room and/or Control Areas

No change.

A.8.3.4 Analytical Sampling

No change.

A.8.3.5 References

[A8.1] U.S. Nuclear Regulatory Commission, Office of the Nuclear Material Safety and Safeguards, "Safety Evaluation of VECTRA Technologies' Response to Nuclear Regulatory Commission Bulletin 96-04 for NUHOMS® -24P and NUHOMS®-7P Dry Spent Fuel Storage System," November 1997 (Dockets 72-1004, 72-3, 72-4, 72-8, and 72-14).

[A8.2] SNT-TC-1A, "American Society of Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

A.9 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Sections of this Chapter have been identified as "No change" due to the addition of 24PT4-DSC to the Advanced NUHOMS® system. For these sections, the description or analysis presented in the corresponding sections of the FSAR for the Advanced NUHOMS® system with 24PT1-DSC is also applicable to the system with 24PT4-DSC.

A.9.1 Acceptance Criteria

No change.

A.9.1.1 Visual Inspection

No change.

A.9.1.2 Structural

No change.

A.9.1.3 Leak Tests and Hydrostatic Pressure Tests

No change

A.9.1.4 Components

No change.

A.9.1.4.1 Valves, Rupture Discs, and Fluid Transport Devices

No change.

A.9.1.4.2 Gaskets

No change.