TRANSNUCLEAR WEST

AMENDMENT NO. 6 TO NUHOMS® COC 1004

ADDITION OF 24PHB DSC TO STANDARDIZED NUHOMS® SYSTEM

NON-PROPRIETARY

TRANSNUCLEAR WEST

August 31, 2001
NUH03-01-1721

Mr. Timothy Kobetz
Spent Fuel Project Office, NMSS
U. S. Nuclear Regulatory Commission
11555 Rockville Pike M/S 0-6-F-18
Rockville, MD 20852

Subject: Application for Amendment No.6 of NUHOMS® Certificate of Compliance No. 1004 for Dry Spent Fuel Storage Casks, Revision 0

References:
1. Certificate of Compliance (CofC) No. 1004 Revision 2 and Proposed Amendments 3, 4 and 5 to the CoC.
2. Standard Review Plan for Dry Cask Storage Systems, NUREG-1536, January 1997.
3. Fuel Rod Analysis for Dry Storage of Spent Nuclear Fuel, DPC-NE-2013P, Duke Energy, August 2001.
4. NUH-003, Standardized NUHOMS Horizontal Modular Storage System Final Safety Analysis Report (FSAR), as updated.

Dear Mr. Kobetz:

Transnuclear West Inc. (TN West) herewith submits its Application for Amendment No. 6 of NUHOMS® Certificate of Compliance No. 1004. This application proposes to add high burnup B&W 15x15 spent fuel assemblies to the authorized contents for the Standardized NUHOMS® System. The modified storage system to accommodate the new contents has been designated as the NUHOMS®-24PHB System.

The 24PHB DSC is designed to store 24 intact B&W 15x15 spent fuel assemblies with an assembly average burnup of up to 55,000 MWd/MTU; an initial enrichment of less than or equal to 4.5 weight % U-235;, a maximum decay heat load of 1.3 kW per assembly; and a maximum heat load of 24kW per DSC. The fuel cladding integrity is assured by maintaining the maximum cladding temperature below the limits defined in Reference 3. The 24PHB DSC is designed for storage in the existing Model 102 NUHOMS® Horizontal Storage Module (HSM) and for transfer in the existing Standard, OS 197 or OS 197 NUHOMS® transfer cask (TC).

No design or configuration change is required for the HSM or TC to accommodate the 24PHB canister. The only configuration change for the 24PHB DSC, relative to the existing 24P DSC, is the addition of a test port and plug to the outer top cover plate to allow testing the canister to a "leak tight" condition per ANSI N14.5-1997 criteria.

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

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

Attachment A: Description, Justification and Evaluation of Amendment Changes,
Attachment B: Suggested Changes to Certificate of Compliance (Relative to Reference 1),
Attachment C: FSAR Changes including a new Appendix N of the FSAR.

Appendix N includes a complete evaluation of the NUHOMS®-24PHB DSC and is prepared in a format consistent with the Standard Review Plan for Dry Cask Storage (Reference 2). Where analyses are bounded by the existing FSAR (Reference 4), those sections of the FSAR are referenced. References preceded with an "N" refer to sections in the Appendix while those not preceded with an "N" refer to the existing FSAR.

This submittal includes proprietary information (Attachment C) which may not be used for any purpose other to support your staff's review of the application. In accordance with 10 CFR 2.790, I am providing an affidavit (Enclosure 1) specifically requesting that you withhold this proprietary information from public disclosure.

Duke Energy Corporation (Duke) plans to initiate fabrication of 24PHB DSCs by January 2004 to support loading "high burnup" fuel at its Oconee Nuclear Station in January 2005. Consistent with Duke's needs, a detailed milestone schedule for the review and approval of the Duke's Topical Report (Reference 3) and this amendment application was presented in a NRC meeting on July 20, 2001. TN West respectfully requests that the staff assign appropriate priority for review of this application, consistent with the considerations delineated above and a December 2003 effective date for the amended CoC.

Should you or your staff require additional information to support review of this application, please do not hesitate to contact me at 510-744-6053.

Sincerely,

UB Chopra

U. B. Chopra
Licensing Manager

Docket 72-1004

Enclosures:
1. Affidavit for withholding proprietary information.
2. Ten (10) copies of the Application for Amendment No. 6 to COC 1004 (Proprietary Version).
3. Three (3) copies of the Application for Amendment No. 6 to COC 1004 (Non-Proprietary Version).

AFFIDAVIT PURSUANT
TO 10 CFR 2.790

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

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

The information for which proprietary treatment is sought is contained in the document included in Attachment C of this submittal and as listed below:

- FSAR Appendix N (Proprietary Version).

This document has been appropriately designated as proprietary.

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

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

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

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

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

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

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

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

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

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

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

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

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

Further the deponent sayeth not.

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

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

Subscribed and sworn to me before this 27th day of August, 2001, by Robert M. Grenier.

Notary Public

ATTACHMENT A

Description, Justification, and Evaluation of Amendment Changes

ATTACHMENT A

DESCRIPTION, JUSTIFICATION AND EVALUATION OF AMENDMENT CHANGES

1.0 INTRODUCTION

The purpose of this amendment application is to add high burnup B&W 15x15 spent fuel assemblies to the authorized contents for the Standardized NUHOMS® System. The modified storage system to accommodate the new contents has been designated as the NUHOMS®-24PHB System, The 24PHB DSC is designed to store 24 intact B&W 15x15 spent fuel assemblies with an assembly average burnup of up to 55,000 MWd/MTU; an initial enrichment of less than or equal to 4.5 weight % U-235; a maximum decay heat load of 1.3 kW per assembly; and a maximum heat load of 24kW per DSC. The fuel cladding integrity is assured by maintaining the maximum cladding temperature below the limits defined in DPC-NE-2013P, Duke Energy Topical Report, August 2001 which is being submitted separately.

The 24PHB DSC is designed for storage in the existing Model 102 NUHOMS® Horizontal Storage Module (HSM) and for transfer in the existing Standard, OS 197 or OS 197H NUHOMS® transfer cask (TC). No design or configuration change is required for the HSM or TC to accommodate the 24PHB canister. The only configuration change for the 24PHB DSC, relative to the existing 24P DSC, is the addition of a test port and plug to the outer top cover plate to allow testing the canister to a "leak tight" condition per ANSI N14.5-1997 criteria.

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

2.0 BRIEF DESCRIPTION OF THE CHANGE

2.1 Significant Changes Relative to NUHOMS® CoC 72-1004, Amendment 2 and Amendments 3, 4 and 5

The changes listed below are relative to CoC Amendment 2 including changes proposed by CoC Amendments 3, 4 and 5:

- Revise "Limit/Specification" and "Action" sections of Specification 1.2.1, "Fuel Specification", to add reference to Table 1-1h which specifies the applicable parameters for the high burn up B&W 15x15 class PWR fuel, with or without Burnable Poison Rod Assemblies (BPRAs) allowed to be stored in the NUHOMS®-24PHB DSC.

- Revise the "Bases" section of Specification 1.2.1, "Fuel Specification", to provide the supporting bases for storage of PWR fuel with or without BPRAs in the NUHOMS®-24PHB DSC.

- Add Fuel Qualification Tables 1-2n, 1-2o, and 1-2p for the 24PHB DSC (PWR Fuel with or without BPRAs).

- Add Figures 1-5, and 1-6 to specify the two heat load zoning configurations analyzed for the 24PHB DSC.

- Revise the title and "Applicability" section of Specification 1.2.3a to extend the applicability of this specification to 24PHB DSC.

- Revise the title ,"Applicability" and the "Bases" sections of Specification 1.2.4a to extend the applicability of this specification to 24PHB DSC.

- Add a new Specification 1.2.7a, "24PHB HSM Dose Rates" to reflect the limiting doses rates on the HSM due to the addition of 24PHB DSC.

- Delete Specification 1.2.11, "Transfer Cask Dose Rates" since this is a redundant Technical Specification.

- Add a new Specification 1.2.15b and Figure 1-7 to specify the minimum boron concentration required during loading of the NUHOMS®-24PHB system.

- Add a new Specification 1.2.17a, "Vacuum Drying Duration Limit" to specify a vacuum drying duration limit for NUHOMS®-24PHB DSC.

- Revise Surveillance Specification 1.3.1 to reflect the parameters from the 24PHB blocked vent transient analysis.

- Update Table 1.3.1 to reflect the addition/deletion of Specifications listed above.

2.2 Changes to NUHOMS® FSAR, Revision 5 as Updated

Attachment C of this submittal includes a new FSAR Appendix N which has been prepared in a format consistent with the Standard Review Plan for Dry Cask Storage (NUREG 1536). It provides a complete evaluation of the NUHOMS®-24PHB System. It also documents the changes where applicable to the existing safety analyses provided in the FSAR.

3.0 JUSTIFICATION OF CHANGE

Duke Energy Corporation (Duke) plans to initiate fabrication of 24PHB DSCs by January 2004 to support loading "high burnup" fuel at its Oconee Nuclear Station in January 2005. Consistent with Duke's needs, a detailed milestone schedule for the review and approval of the Duke's Topical Report and this amendment application was presented in a NRC meeting on July 20, 2001. Consistent with the considerations delineated above, a December 2003 effective date for the amended CoC is requested.

4.0 EVALUATION OF CHANGE

TN West has evaluated the NUHOMS®-24PHB system for structural, thermal, shielding and criticality adequacy and has concluded that the storage of the high burn up fuel in the

NUHOMS® 24PHB System has no significant effect on safety. This evaluation is documented in Appendix N of the FSAR (Attachment C).

ATTACHMENT B

Suggested Changes to Technical Specifications of CoC 1004 Amendment No. 2

(Changes proposed by CoC Amendments 3, 4 and 5 have been included as current configuration)

Technical Specifications Included:

- Section 1.2.1
- Section 1.2.3
- Section 1.2.3a
- Section 1.2.4
- Section 1.2.4a
- Section 1.2.7
- Section 1.2.7a
- Section 1.2.11
- Section 1.2.15
- Section 1.2.15a
- Section 1.2.15b
- Section 1.2.17
- Section 1.2.17a
- Section 1.3.1

1.2.1 Fuel Specifications

Limit/Specification:	The characteristics of the spent fuel which is allowed to be stored in the standardized NUHOMS® system are limited by those included in Tables 1-1a, 1-1b, 1-1c, 1-1d, 1-1e, 1-1f, 1-1g, *and 1-1h.*
Applicability:	The specification is applicable to all fuel to be stored in the standardized NUHOMS® system.
Objective:	The specification is prepared to ensure that the peak fuel rod cladding temperatures, maximum surface doses, and nuclear criticality effective neutron multiplication factor are below the design limits. Furthermore, the fuel weight and type ensures that structural conditions in the FSAR bound those of the actual fuel being stored.
Action:	Each spent fuel assembly to be loaded into a DSC shall have the parameters listed in Tables 1-1a, 1-1b, 1-1c, 1-1d, 1-1e, 1-1f, 1-1g, *and 1-1h* verified and documented. Fuel not meeting this specification shall not be stored in the standardized NUHOMS® system.
Surveillance:	Immediately, before insertion of a spent fuel assembly into a DSC, the identity of each fuel assembly shall be independently verified and documented.
Bases:	The specification is based on consideration of the design basis parameters included in the FSAR and limitations imposed as a result of the staff review. Such parameters stem from the type of fuel analyzed, structural limitations, criteria for criticality safety, criteria for heat removal, and criteria for radiological protection. The standardized NUHOMS® system is designed for dry, horizontal storage of irradiated light water reactor (LWR) fuel. The principal design parameters of the fuel to be stored can accommodate standard PWR fuel designs manufactured by Babcock and Wilcox (B&W), Combustion Engineering (CE), and Westinghouse (WE), and standard BWR fuel manufactured by General Electric (GE). The NUHOMS®-24P and 52B systems are limited for use to these standard designs and to equivalent designs by other manufacturers as listed in Chapter 3 of the FSAR. The analyses presented in the FSAR are based on non-consolidated, zircaloy-clad fuel with no known or suspected gross breaches. The NUHOMS®-61BT, 32PT, *and 24PHB* systems are limited for use to these standard designs and to equivalent designs by other manufacturers as listed in Tables 1-1d, 1-1f, *and 1-1h.* The analyses presented in Appendix K, M, *and N* of the FSAR are based on non-consolidated, zircaloy-clad fuel. The physical parameters that define the mechanical and structural design of the HSM and DSC are the fuel assembly dimensions and weight. The calculated stresses given in the FSAR are based on the physical parameters given in Tables 1-1a, 1-1b, 1-1c, 1-1d, 1-1e, 1-1f, ,1-1g, *and 1-1h* and represent the upper bound.

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

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

The design basis PWR BPRA for shielding source terms and thermal decay heat load is the Westinghouse 17x17 Pyrex Burnable Absorber, while the DSC internal pressure analysis is limited by B&W 15x15 BPRAs. In addition, BPRAs with cladding failures were determined to be acceptable for loading into NUHOMS®-24P Long Cavity DSC as evaluated in Appendix J of the FSAR. The acceptability of loading BPRAs, including damaged BPRAs into 32PT-L100 and 32PT-L125 DSC configurations is provided in Appendix M of the FSAR.

The NUHOMS®-24P is designed for unirradiated fuel with an initial fuel enrichment of up to 4.0 wt. % U-235, taking credit for soluble boron in the DSC cavity water during loading operations. Section 1.2.15 defines the requirements for boron concentration in the DSC cavity water for the NUHOMS®-24P design only. In addition, the fuel assemblies qualified for storage in NUHOMS®-24P DSC have an equivalent unirradiated enrichment of less than or equal to 1.45 wt. % U-235. Figure 1-1 defines the required burnup as a function of initial enrichment. The NUHOMS®-52B is designed for unirradiated fuel with an initial enrichment of less than or equal to 4.0 wt. % U-235.

The NUHOMS®-61BT is designed for unirradiated fuel with an initial enrichment of less than or equal to 4.4 wt. % U-235.

The NUHOMS®-32PT is designed for unirradiated fuel with an initial fuel enrichment of up to 5.0 wt. % U-235 as shown in Table 1-1g, taking credit for Poison Rod Assemblies (PRAs), poison plates, and soluble boron in the DSC cavity water during loading operations. Specification 1.2.15a defines the requirements for boron concentration in the DSC cavity water for the NUHOMS®-32PT design only.

The NUHOMS®-24PHB is designed for unirradiated fuel with an initial enrichment of less than or equal to 4.5 wt. % U-235 as shown in Table 1-1h, taking credit for soluble boron in the DSC cavity water during loading *operations. Specification 1.2.15b defines the requirements for boron concentration in the DSC cavity water for the NUHOMS®-24PHB design only.*

The thermal design criterion of the fuel to be stored is that the total maximum heat generation rate per assembly and BPRA be such that the fuel cladding temperature is maintained within established limits during normal and off-normal conditions.

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

The design value average HSM and cask surface dose rates for the 24P and 52B canisters were calculated to be 48.6 mrem/hr and 591.8 mrem/hr respectively based on storing twenty four (24) Babcock and Wilcox 15x15 PWR assemblies (without BPRAs) with 4.0 wt. % U-235 initial enrichment, irradiated to 40,000 MWd/MTU, and having a post irradiation time of five years. To account for BPRAs, the fuel assembly cooling required cooling times are increased to maintain the above dose rate limits.

Table 1-1a
PWR Fuel Specifications for Fuel to be Stored in the Standardized NUHOMS®-24P DSC

Title or Parameter	Specifications
Fuel	Only intact, unconsolidated PWR fuel assemblies (with or without BPRAs) with the following requirements
Physical Parameters (without BPRAs) Maximum Assembly Length (unirradiated) Nominal Cross-Sectional Envelope Maximum Assembly Weight No. of Assemblies per DSC Fuel Cladding	 165.75 in (standard cavity) 171.71 in (long cavity) 8.536 in 1682 lbs ≤ 24 intact assemblies Zircalloy-clad fuel with no known or suspected gross cladding breaches
Physical Parameters (with BPRAs) Maximum Assembly + BPRA Length (unirradiated) With Burnup > 32,000 and ≤ 45,000 MWd/MTU With Burnup ≤ 32,000 MWd/MTU Nominal Cross-Sectional Envelope Maximum Assembly + BPRA Weight No. of Assemblies per DSC No. of BPRAs per DSC Fuel Cladding	 171.71 in (long cavity) 171.96 in (long cavity) 8.536 in 1682 lbs ≤ 24 intact assemblies ≤ 24 BPRAs Zircalloy-clad fuel with no known or suspected gross cladding breaches
Nuclear Parameters Fuel Initial Enrichment Fuel Burnup and Cooling Time BPRA Cooling Time (Minimum)	 ≤ 4.0 wt. % U-235 Per Table 1-2a (without BPRAs) or Per Table 1-2c (with BPRAs) 5 years for B&W Designs 10 years for Westinghouse Designs
Alternate Nuclear Parameters Initial Enrichment Burnup Decay Heat (Fuel + BPRA) Neutron Fuel Source Gamma (Fuel +BPRA) Source	 ≤ 4.0 wt. % U-235 ≤ 40,000 MWd/MTU and Per Figure 1-1 ≤ 1.0 kW per assembly ≤ 2.23 x 10^8 n/sec per assy with spectrum bounded by that in Chapter 7 of SAR ≤ 7.45 x 10^{15} g/sec per assy with spectrum bounded by that in Chapter 7 of SAR

Table 1-1b
BWR Fuel Specifications of Fuel to be Stored in the Standardized NUHOMS®-52B DSC

Title or Parameter	Specifications
Fuel	Only intact, unconsolidated BWR fuel assemblies with the following requirements
Physical Parameters Maximum Assembly Length (unirradiated) Nominal Cross-Sectional Envelope * Maximum Assembly Weight No. of Assemblies per DSC Fuel Cladding	 176.16 in 5.454 in 725 lbs ≤ 52 intact channeled assemblies Zircaloy-clad fuel with no known or suspected gross cladding breaches
Nuclear Parameters Fuel Initial Lattice Enrichment Fuel Burnup and Cooling Time	 ≤ 4.0 wt. % U-235 Per Table 1-2b
Alternate Nuclear Parameters Initial Enrichment Burnup Decay Heat Neutron Source Gamma Source	 ≤ 4.0 wt. % U-235 ≤ 35,000 MWd/MTU ≤ 0.37 kW per assembly ≤ 1.01 x 10^8 n/sec per assy with spectrum bounded by that in Chapter 7 of SAR ≤ 2.63 x 10^{15} g/sec per assy with spectrum bounded by that in Chapter 7 of SAR

*Cross Sectional-Envelope is the outside dimension of the fuel channel.

Table 1-1c
BWR Fuel Specifications of Intact Fuel to be Stored in the Standardized NUHOMS®-61BT DSC

Physical Parameters:	
Fuel Design:	7x7, 8x8, 9x9, or 10x10 BWR fuel assemblies manufactured by General Electric or equivalent reload fuel that are enveloped by the Fuel assembly design characteristics listed in Table 1-1d.
Cladding Material:	Zircaloy
Fuel Damage:	Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact BWR Fuel".
Channels:	Fuel may be stored with or without fuel channels
Maximum Assembly Length	176.2 in
Nominal Assembly Width	5.44 in
Maximum Assembly Weight	705 lbs
Radiological Parameters:	No interpolation of Radiological Parameters is permitted between Groups.
Group 1:	
Maximum Burnup:	27,000 MWd/MTU
Minimum Cooling Time:	5-years
Maximum Lattice Average Initial Enrichment:	See Minimum Boron Loading Below
Minimum Initial Bundle Average Enrichment:	2.0 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 2:	
Maximum Burnup:	35,000 MWd/MTU
Minimum Cooling Time:	8-years
Maximum Lattice Average Initial Enrichment:	See Minimum Boron Loading Below
Minimum Initial Bundle Average Enrichment:	2.65 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 3:	
Maximum Burnup:	37,200 MWd/MTU
Minimum Cooling Time:	6.5-years
Maximum Lattice Average Initial Enrichment:	See Minimum Boron Loading Below
Minimum Initial Bundle Average Enrichment:	3.38 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 4:	
Maximum Burnup:	40,000 MWd/MTU
Minimum Cooling Time:	10-years
Maximum Lattice Average Initial Enrichment:	See Minimum Boron Loading Below
Minimum Initial Bundle Average Enrichment:	3.4 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Minimum Boron Loading	
Lattice Average Enrichment (wt%U-235)	Minimum B-10 Content in Poison Plates
4.4	Type C Basket
4.1	Type B Basket
3.7	Type A Basket

Table 1-1d
BWR Fuel Assembly Design Characteristics[1][2][3] for the NUHOMS®-61BT DSC

Transnuclear, ID	7 x 7-49/0	8 x 8-63/1	8 x 8-62/2	8 x 8 -60/4	8 x 8-60/1	9 x 9-74/2	10x10-92/2
GE Designations	GE2 GE3	GE4	GE-5 GE-Pres GE-Barrier GE8 Type I	GE8 Type II	GE9 GE10	GE11 GE13	GE12
Max Length (in)	176.2	176.2	176.2	176.2	176.2	176.2	176.2
Nominal Width (in) (excluding channels)	5.44	5.44	5.44	5.44	5.44	5.44	5.44
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Number of Fuel Rods	49	63	62	60	60	66 – Full 8 – Partial	78 – Full 14 - Partial
Number of Water Holes	0	1	2	4	1	2	2

(1) Any fuel channel thickness from 0.065 to 0.120 inch is acceptable on any of the fuel designs.
(2) Maximum fuel assembly weight with channel is 705 lb.
(3) Maximum Co-59 content in the Top End Fitting Region is 4.5 gm per assembly.
Maximum Co-59 content in the Plenum Region is 0.9 gm per assembly.
Maximum Co-59 content in the In-Core Region (including the whole fuel channel) is 4.5 gm per assembly.
Maximum Co-59 content in the Bottom Region is 4.1 gm per assembly.

Table 1-1e
PWR Fuel Specifications for Fuel to be Stored in the NUHOMS®-32PT DSC

<u>PHYSICAL PARAMETERS:</u>	
Fuel Assembly Class	B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14 and WE 14x14 assemblies that are enveloped by the fuel assembly design characteristics listed in Table 1-1f.
Fuel Cladding Material	Zircaloy
Fuel Damage	Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact PWR Fuel."
Burnable Poison Rod Assemblies (BPRAs)	Standard BPRA designs for the B&W 15x15 and Westinghouse 17x17 class assemblies as listed in Appendix J of the FSAR.
BPRA Damage	BPRAs with cladding failures are acceptable for loading.
<u>THERMAL/RADIOLOGICAL PARAMETERS:</u>	
Fuel Burnup and Cooling Time without BPRAs	Per Table 1-2d, Table 1-2e, Table 1-2f, Table 1-2g, Table 1-2h, and Figure 1-2 or Figure 1-3 or Figure 1-4.
Fuel Burnup and Cooling Time with BPRAs	Per Table 1-2i, Table 1-2j, Table 1-2k, Table 1-2l, Table 1-2m and Figure 1-2 or Figure 1-3 or Figure 1-4.
Initial Enrichment	Per Table 1-1g.
B&W 15x15 BPRA Burnup and Cooling Time	BPRA Burnup shall not exceed that of a BPRA irradiated in fuel assemblies with a total Burnup of 36,000 MWd/MTU. -Minimum Cooling Time 5 years
WE 17x17 BPRA Burnup and Cooling Time	BPRA Burnup shall not exceed that of a BPRA irradiated in fuel assemblies with a total Burnup of 36,000 MWd/MTU. -Minimum Cooling Time 10 years

Table 1-1f
PWR Fuel Assembly Design Characteristics for the NUHOMS®-32PT DSC

Assembly Class	B&W 15x15	WE 17x17	CE 15x15	WE 15x15	CE 14x14	WE 14x14
DSC Configuration	Max Unirradiated Length (in)					
32PT-S100	165.75	165.75	165.75	165.75	165.75	165.75
32PT-L100	171.71[1]	171.71[1]	171.71	171.71	171.71	171.71
32PT-S125	165.75	165.75	165.75	165.75	165.75	165.75
32PT-L125	171.71[1]	171.71[1]	171.71	171.71	171.71	171.71
Fissile Material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Maximum MTU/assembly[2]	0.475	0.475	0.475	0.475	0.475	0.475
Maximum Number of Fuel Rods	208	264	216	204	176	179
Maximum Number of Guide/ Instrument tubes	17	25	9	21	5	17

(1) Maximum Assembly + BPRA Length (unirradiated)
(2) The maximum MTU/assembly is based on the shielding analysis. The listed value is higher than the actual.

Table 1-1g
Initial Enrichment and Required Number of PRAs (NUHOMS®-32PT DSC)

Assembly Class	Assembly Type	Initial Enrichment, wt. % U-235			
		0 PRAs	4 PRAs	8 PRAs	16 PRAs
WE 17x17[1]	Westinghouse 17x17 LOPAR/Std Westinghouse 17x17 OFA/Vantage 5	3.40	4.00	4.50	5.00
B&W 15x15[1]	B&W 15x15 Mark B	3.30	3.90	NA	5.00
CE 15x15	CE 15x15 Palisades Exxon/ANF 15x15 CE	3.40	Not Evaluated	Not Evaluated	Not Evaluated
WE 15x15	Westinghouse 15x15 Std/ZC Exxon/ANF 15x15 WE	3.40	4.00	4.60	5.00
CE 14x14	CE 14x14 Std/Generic CE 14x14 Fort Calhoun	3.80	4.60	5.00	Not Evaluated
WE 14x14	Westinghouse 14x14 ZCA/ZCB Westinghouse 14x14 OFA Exxon/ANF 14x14 WE	4.00	5.00	Not Evaluated	Not Evaluated

(1) With or without BPRAs. BPRAs shall not be stored in basket locations where a PRA is required.

Table 1-1h

PWR Fuel Specifications for Fuel to be Stored in the Standardized NUHOMS®-24PHB DSC

Title or Parameter	*Specifications*
Fuel	*Only intact, unconsolidated B&W 15x15 class PWR fuel assemblies (with or without BPRAs) with the following requirements*
Physical Parameters (without BPRAs)	
Maximum Assembly Length (unirradiated)	*165.785 in (standard cavity)* *171.96 in (long cavity)*
Nominal Cross-Sectional Envelope	*8.536 in*
Maximum Assembly Weight	*1682 lbs*
No. of Assemblies per DSC	*≤24 intact assemblies*
Fuel Cladding	*Zircalloy-clad fuel with no known or suspected gross cladding breaches*
Reconstituted fuel assemblies	*≤4 assemblies with stainless steel rods (up to 10 rods per assembly) or Zircaloy clad lower enrichment uranium rods (any number of rods per assembly)*
Physical Parameters (with BPRAs)	
Maximum Assembly + BPRA Length (unirradiated)	*171.96 in (long cavity)*
Nominal Cross-Sectional Envelope	*8.536 in*
Maximum Assembly + BPRA Weight	*1682 lbs*
No. of Assemblies per DSC	*≤24 intact assemblies*
No. of BPRAs per DSC	*≤24 BPRAs*
Fuel Cladding	*Zircalloy-clad fuel with no known or suspected gross cladding breaches*
Reconstituted fuel assemblies	*≤4 assemblies with stainless steel rods (up to 10 rods per assembly) or Zircaloy clad lower enrichment uranium rods (any number of rods per assembly)*
Nuclear Parameters	
Fuel Initial Enrichment	*≤4.5 wt. % U-235*
Fuel Burnup and Cooling Time	*Per Table 1-2n (Zone 1) or* *Per Table 1-2o (Zone 2)* *Per Table 1-2p (Zone 3)*
BPRA Cooling Time (Minimum)	*5 years*
Alternate Nuclear Parameters	
Initial Enrichment	*≤4.5 wt. % U-235*
Decay Heat	*Figure 1-5 or 1-6 as applicable*
Neutron and Gamma Source	*Total calculated dose rate shall be less than or equal to 93.7 mrem/hr on the HSM roof surface and 1370.2 mrem/hr on the TC side surface as determined using the "Response Function" provided in Table N.5-15 of the FSAR and the methodology described in Section N.5.2.4.*

Table 1-2a

PWR Fuel Qualification Table for the Standardized NUHOMS®-24P DSC (Fuel Without BPRAs)

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

Burnup (GWd/ MTU)	Initial Enrichment (wt. % U-235)																				
	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
10	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a
15	5	5	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a
20	5	5	5	5	5	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a
25		5	5	5	5	5	5	5	5	a	a	a	a	a	a	a	a	a	a	a	a
28				5	5	5	5	5	5	5	5	5	a	a	a	a	a	a	a	a	a
30						5	5	5	5	5	5	5	5	a	a	a	a	a	a	a	a
32							5	5	5	5	5	5	5	5	5	a	a	a	a	a	a
34								6	5	5	5	5	5	5	5	5	5	a	a	a	a
36									6	6	6	6	5	5	5	5	5	5	5	a	a
38											7	6	6	6	6	6	6	6	5	5	5
40												8	8	8	7	6	6	6	6	6	6
41												9	9	9	8	8	8	8	8	8	8
42													10	9	9	9	9	9	9	8	8
43													10	10	10	10	10	9	9	9	9
44														11	11	11	11	10	10	10	10
45														12	12	11	11	11	11	11	11

Not Acceptable or Not Analyzed

a) Minimum Cooling Time 5 years, and Minimum 2350 ppm soluble boron required in the DSC cavity water during loading or unloading.

Notes:

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 wt. % U-235 must be qualified for storage using the alternate nuclear parameters specified in Table 1-1a. Fuel with an initial enrichment greater than 4.0 wt. % U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage.
- Example: An assembly with an initial enrichment of 3.65 wt. % U-235 and a burnup of 42.5 GWd/MTU is acceptable for storage after a ten-year cooling time as defined at the intersection of 3.6 wt. % U-235 (rounding down) and 43GWd/MTU (rounding up) on the qualification table.

Table 1-2b

BWR Fuel Qualification Table for the Standardized NUHOMS®-52B DSC

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

Burnup (GWd/ MTU)	Initial Enrichment (wt. % U-235)																				
	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
15	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3	3
20	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
30				5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
32					6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5
34						8	8	8	8	8	8	8	8	7	6	6	6	6	6	6	6
35							10	10	10	10	9	8	8	8	8	8	8	8	6	6	6
36							11	11	11	11	11	10	10	10	10	10	10	9	8	8	8
37								13	13	12	12	12	12	11	11	11	11	11	10	10	10
38								15	14	14	14	13	13	13	13	12	12	12	12	12	11
39								18	17	17	16	16	16	15	14	14	14	14	13	13	13
40				Not Acceptable or Not Analyzed					21	21	20	20	19	18	17	17	16	16	16	16	15
42										22	22	22	21	21	20	20	20	19	18	17	17
44										24	24	23	23	23	22	22	21	21	21	20	20
45											25	24	24	23	23	23	22	22	22	21	21

Notes:

- Use burnup and enrichment to lookup required cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 wt. % U-235 must be qualified for storage using the alternate nuclear parameters specified in Table 1-1b. Fuel with an initial enrichment greater than 4.0 wt. % U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage. Fuel with a burnup less than 15 GWd/MTU is acceptable after three years cooling time provided the physical parameters from Table 1-1b have been met.
- Example: An assembly with an initial enrichment of 3.05 wt. % U-235 and a burnup of 34.5 GWd/MTU is acceptable for storage after a nine-year cooling time as defined at the intersection of 3.0 wt. % U-235 (rounding down) and 35GWd/MTU (rounding up) on the qualification table.

Table 1-2c

PWR Fuel Qualification Table for the Standardized NUHOMS®-24P DSC (Fuel With BPRAs)

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

Burnup (GWd/ MTU)	Initial Enrichment (wt. % U-235)																				
	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
10	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a
15	5	5	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a
20	5	5	5	5	5	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a	a
25		5	5	5	5	5	5	5	5	a	a	a	a	a	a	a	a	a	a	a	a
28				5	5	5	5	5	5	5	5	5	a	a	a	a	a	a	a	a	a
30						6	6	6	5	5	5	5	5	a	a	a	a	a	a	a	a
32							6	6	6	6	6	6	5	5	5	a	a	a	a	a	a
34								7	6	6	6	6	6	6	6	6	6	a	a	a	a
36									8	7	7	7	6	6	6	6	6	6	6	a	a
38											8	8	7	7	7	7	6	6	6	6	6
40												9	9	8	8	8	7	7	7	7	6
41												10	9	9	9	9	8	8	8	8	8
42													10	10	9	9	9	9	9	9	9
43													11	11	11	10	10	10	10	9	9
44														12	11	11	11	11	10	10	10
45														13	12	12	12	11	11	11	11

Not Acceptable or Not Analyzed

a) Minimum Cooling Time 5 years, and Minimum 2350 ppm soluble boron required in the DSC cavity water during loading or unloading.

Notes:

- BPRA Burnup shall not exceed that of a BPRA irradiated in fuel assemblies with a total burnup of 36,000 MWd/MTU.
- Minimum cooling time for a BPRA is 5 years for B&W designs and 10 years for Westinghouse designs, regardless of the required assembly cooling time.
- Use burnup and enrichment to lookup minimum fuel assembly cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 wt. % U-235 must be qualified for storage using the alternate nuclear parameters specified in Table 1-1a. Fuel with an initial enrichment greater than 4.0 wt. % U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage.
- Example: An assembly with an initial enrichment of 3.65 wt. % U-235 and a burnup of 42.5 GWd/MTU is acceptable for storage after a ten-year cooling time as defined at the intersection of 3.6 wt. % U-235 (rounding down) and 43 GWd/MTU (rounding up) on the qualification table.

Table 1-2d
PWR Fuel Qualification Table for 1.2 kW per Assembly, Fuel Without BPRAs, for the NUHOMS®-32PT DSC

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

BU (GWd/MTU)	Initial Enrichment wt % U-235																													
	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	4.9	5.0
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
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
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
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
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
36									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	Not Analyzed											5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
41												5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
42												6	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5	5
43													6	6	6	6	6	6	6	6	6	6	6	6	6	5	5	5	5	5	5
44														6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
45														6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 and greater than 5.0 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling
- Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a six-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 1-2e
PWR Fuel Qualification Table for 0.87 kW per Assembly, Fuel Without BPRAs, for the NUHOMS®-32PT DSC

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

BU (GWd/MTU)	Initial Enrichment wt % U-235																														
	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	4.9	5.0
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
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
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
32							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
34								6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	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
38											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	6	6	6	6	6
39											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
40				Not Analyzed								8	8	8	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
41												8	8	8	8	8	8	8	8	8	8	8	8	7	7	7	7	7	7	7	7
42												8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
43													9	9	9	9	9	9	9	8	8	8	8	8	8	8	8	8	8	8	8
44														9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	8	8
45														10	10	10	10	10	10	9	9	9	9	9	9	9	9	9	9	9	9

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 and greater than 5.0 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a eight-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 1-2f

PWR Fuel Qualification Table for 0.7 kW Fuel, Without BPRAs, per Assembly for the NUHOMS®-32PT DSC

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

BU (GWd/ MTU)	Initial Enrichment wt % U-235																														
	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	4.9	5.0
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
28				6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5
30						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
32							7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	6	6	6	6
34								8	8	8	8	8	8	8	8	8	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
36									9	9	9	9	9	9	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
38											10	10	10	10	10	10	10	9	9	9	9	9	9	9	9	9	9	9	9	9	9
39											11	11	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10
40												11	11	11	11	11	11	11	11	11	11	11	11	11	11	10	10	10	10	10	10
41												12	12	12	12	12	12	12	12	12	12	11	11	11	11	11	11	11	11	11	11
42												13	13	13	13	13	13	13	13	12	12	12	12	12	12	12	12	12	12	12	12
43													14	14	14	14	14	14	14	13	13	13	13	13	13	13	13	13	13	13	13
44														15	15	15	15	15	15	15	14	14	14	14	14	14	14	14	14	14	14
45														16	16	16	16	16	16	16	15	15	15	15	15	15	15	15	15	15	15

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 and greater than 5.0 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a thirteen-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 1-2g
PWR Fuel Qualification Table for 0.63 kW per Assembly, Fuel Without BPRAs, for the NUHOMS®-32PT DSC

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

BU (GWd/ MTU)	Initial Enrichment wt % U-235																													
	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	4.9	5.0
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		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
28				7	7	7	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
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
32							8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	7
34								9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9
36									11	11	11	11	11	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10
38											13	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	11	11	11	11
39											14	13	13	13	13	13	13	13	13	13	13	13	13	13	13	13	13	12	12	12	12
40												15	15	14	14	14	14	14	14	14	14	14	14	14	14	14	14	13	13	13	13
41												16	16	16	16	15	15	15	15	15	15	15	15	15	15	15	15	15	15	15	14
42												17	17	17	17	17	17	16	16	16	16	16	16	16	16	16	16	16	16	16	16
43													18	18	18	18	18	18	18	18	18	17	17	17	17	17	17	17	17	17	17
44														20	19	19	19	19	19	19	19	19	19	19	18	18	18	18	18	18	18
45														21	21	21	21	20	20	20	20	20	20	20	20	20	20	20	20	19	19

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 and greater than 5.0 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a sixteen-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 1-2h
PWR Fuel Qualification Table for 0.6 kW per Assembly, Fuel Without BPRAs, for the NUHOMS®-32PT DSC

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

BU (GWd/MTU)	Initial Enrichment wt % U-235																														
	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	4.9	5.0
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		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	5
28				7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	6	6	6	6	6	6	6	6	6
30						8	8	8	8	8	8	8	8	8	8	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
32							9	9	9	9	9	9	9	9	9	9	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
34								10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	9	9	9	9	9	9
36									12	12	12	12	12	12	12	12	12	11	11	11	11	11	11	11	11	11	11	11	11	11	11
38											14	14	14	14	14	14	14	14	14	13	13	13	13	13	13	13	13	13	13	13	13
39											15	15	15	15	15	15	15	15	15	15	15	15	14	14	14	14	14	14	14	14	14
40												17	16	16	16	16	16	16	16	16	16	16	16	16	16	15	15	15	15	15	15
41												18	18	18	18	18	17	17	17	17	17	17	17	17	17	17	17	17	17	17	16
42												19	19	19	19	19	19	19	19	19	18	18	18	18	18	18	18	18	18	18	18
43													21	20	20	20	20	20	20	20	20	20	20	20	20	19	19	19	19	19	19
44														22	22	22	22	21	21	21	21	21	21	21	21	21	21	21	21	20	20
45														23	23	23	23	23	23	23	23	22	22	22	22	22	22	22	22	22	22

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 and greater than 5.0 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a nineteen-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 1-2i

PWR Fuel Qualification Table for 1.2 kW per Assembly, Fuel With BPRAs, for the NUHOMS®-32PT DSC

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

BU (GWd/ MTU)	Initial Enrichment wt % U-235																													
	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	4.9	5.0
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
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
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
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
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
36									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
41												6	6	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5	5
42												6	6	6	6	6	6	6	6	5	5	5	5	5	5	5	5	5	5	5	5
43													6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	5	5	5
44														6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
45														6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 and greater than 5.0 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling
- Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a six-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 1-2j
PWR Fuel Qualification Table for 0.87 kW per Assembly, Fuel With BPRAs, for the NUHOMS®-32PT DSC

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

BU (GWd/MTU)	Initial Enrichment wt % U-235																														
	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	4.9	5.0
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
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
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
32							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
34								6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
36									7	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
38											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	6	6
39											7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
40												8	8	8	8	8	7	7	7	7	7	7	7	7	7	7	7	7	7	7	7
41												8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	7	7	7	7
42												9	9	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
43													9	9	9	9	9	9	9	9	9	9	8	8	8	8	8	8	8	8	8
44														9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9
45														10	10	10	10	10	10	10	10	10	9	9	9	9	9	9	9	9	9

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 and greater than 5.0 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a eight-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 1-2k

PWR Fuel Qualification Table for 0.7 kW per Assembly, Fuel With BPRAs, for the NUHOMS®-32PT DSC

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

BU (GWd/ MTU)	Initial Enrichment wt % U-235																													
	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	4.9	5.0
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
28				6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	5	5	5	5	5	5	5
30						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
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
34								8	8	8	8	8	8	8	8	8	8	8	8	7	7	7	7	7	7	7	7	7	7	7	7
36									9	9	9	9	9	9	9	9	9	9	8	8	8	8	8	8	8	8	8	8	8	8	8
38											10	10	10	10	10	10	10	10	10	10	10	10	9	9	9	9	9	9	9	9	9
39											11	11	11	11	11	11	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10
40												12	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11
41												13	13	12	12	12	12	12	12	12	12	12	12	12	12	12	12	11	11	11	11
42												14	14	13	13	13	13	13	13	13	13	13	13	13	12	12	12	12	12	12	12
43													15	14	14	14	14	14	14	14	14	14	14	14	13	13	13	13	13	13	13
44														16	15	15	15	15	15	15	15	15	15	15	15	14	14	14	14	14	14
45														17	17	16	16	16	16	16	16	16	16	16	16	16	15	15	15	15	15

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 and greater than 5.0 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a thirteen-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 1-21
PWR Fuel Qualification Table for 0.63 kW per Assembly, Fuel With BPRAs, for the NUHOMS®-32PT DSC

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

BU (GWd/MTU)	Initial Enrichment wt % U-235																														
	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	4.9	5.0
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		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	6	6
28				7	7	7	7	7	7	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6	6
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
32							8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8
34								10	10	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9
36									11	11	11	11	11	11	11	11	11	11	11	10	10	10	10	10	10	10	10	10	10	10	10
38											13	13	13	13	13	13	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12
39											14	14	14	14	14	14	14	13	13	13	13	13	13	13	13	13	13	13	13	13	13
40												15	15	15	15	15	15	15	14	14	14	14	14	14	14	14	14	14	14	14	14
41												16	16	16	16	16	16	16	16	16	16	16	15	15	15	15	15	15	15	15	15
42												18	18	17	17	17	17	17	17	17	17	17	17	17	16	16	16	16	16	16	16
43													19	19	19	19	18	18	18	18	18	18	18	18	18	18	18	18	17	17	17
44														20	20	20	20	20	20	20	19	19	19	19	19	19	19	19	19	19	19
45														22	21	21	21	21	21	21	21	21	21	20	20	20	20	20	20	20	20

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 and greater than 5.0 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a seventeen-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 1-2m
PWR Fuel Qualification Table for 0.6 kW per Assembly Fuel With BPRAs for the NUHOMS®-32PT DSC (Minimum required years of cooling time after reactor core discharge)

BU (GWd/MTU)	Initial Enrichment wt % U-235																														
	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	4.9	5.0
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		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	6	6
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	6
30						8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	8	7	7	7	7	7	7	7	7	7	7
32							9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	9	8	8	8	8	8	8
34								11	11	11	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10	10
36									12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	12	11	11	11	11	11
38											15	15	14	14	14	14	14	14	14	14	14	14	14	14	14	14	14	14	14	13	13
39											16	16	16	16	16	15	15	15	15	15	15	15	15	15	15	15	15	15	15	15	15
40												17	17	17	17	17	17	17	17	16	16	16	16	16	16	16	16	16	16	16	16
41												19	18	18	18	18	18	18	18	18	18	18	18	17	17	17	17	17	17	17	17
42												20	20	20	20	19	19	19	19	19	19	19	19	19	19	19	19	19	19	19	19
43													21	21	21	21	21	21	21	21	20	20	20	20	20	20	20	20	20	20	20
44														23	22	22	22	22	22	22	22	22	22	22	22	21	21	21	21	21	21
45														24	24	24	24	24	23	23	23	23	23	23	23	23	23	23	23	23	22

- Use burnup and enrichment to lookup minimum cooling time in years. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.
- Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.
- Fuel with an initial enrichment less than 2.0 and greater than 5.0 wt.% U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTU is unacceptable for storage
- Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.
- Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 41.5 GWd/MTU is acceptable for storage after a nineteen-year cooling time as defined by 3.7 wt. % U-235 (rounding down) and 42 GWd/MTU (rounding up) on the qualification table.

Table 1-2n

PWR Fuel Qualification Table for Zone 1 with 0.7 kW per Assembly, Fuel With or Without BPRAs, for the NUHOMS®-24PHB DSC (Minimum required years of cooling time after reactor core discharge)

BU (GWd/MTU)	Maximum Assembly Average Initial U-235 Enrichment (wt %)																									
	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
10	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
15	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
20	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
25		5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
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	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5	5.5
30						6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
32							7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
34								8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
36									9.0	9.0	9.0	9.0	9.0	9.0	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5	8.5
38											10.5	10.5	10.5	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	10.0	9.5	9.5	9.5	9.5
39											11.5	11.0	11.0	11.0	11.0	11.0	11.0	11.0	10.5	10.5	10.5	10.5	10.5	10.5	10.5	10.5
40											12.0	12.0	12.0	12.0	12.0	12.0	11.5	11.5	11.5	11.5	11.5	11.5	11.5	11.0	11.0	11.0
41											13.0	13.0	13.0	13.0	13.0	13.0	12.5	12.5	12.5	12.5	12.5	12.5	12.5	12.0	12.0	12.0
42											14.5	14.5	14.0	14.0	14.0	14.0	13.5	13.5	13.5	13.5	13.5	13.5	13.5	13.0	13.0	13.0
43											15.5	15.5	15.5	15.0	15.0	15.0	15.0	15.0	14.5	14.5	14.5	14.5	14.5	14.5	14.0	14.0
44											17.0	16.5	16.5	16.5	16.5	16.0	16.0	16.0	16.0	16.0	15.5	15.5	15.5	15.5	15.5	15.5
45													18.0	17.5	17.5	17.5	17.5	17.0	17.0	17.0	17.0	17.0	16.5	16.5	16.5	16.5
46													18.8	18.7	18.5	18.5	18.3	18.2	18.1	18.0	17.9	17.8	17.7	17.6	17.5	17.4
47													20.1	20.0	19.9	19.6	19.6	19.5	19.4	19.2	19.1	19.0	18.9	18.8	18.7	18.7
48													21.4	21.3	21.1	21.0	20.8	20.8	20.7	20.5	20.4	20.3	20.2	20.1	20.0	19.9
49													22.7	22.6	22.4	22.3	22.1	22.1	21.9	21.8	21.7	21.6	21.5	21.4	21.3	21.2
50															23.7	23.6	23.5	23.4	23.3	23.2	23.0	22.9	22.8	22.7	22.6	22.5
51															25.0	24.9	24.8	24.6	24.5	24.4	24.3	24.2	24.0	23.9	23.8	23.7
52															26.3	26.2	26.0	25.9	25.8	25.7	25.6	25.4	25.3	25.2	25.2	25.0
53															27.5	27.3	27.2	27.1	27.0	26.9	26.8	26.7	26.5	26.4	26.4	26.2
54															28.8	28.6	28.5	28.3	28.2	28.1	28.0	28.0	27.8	27.7	27.6	27.5
55															29.9	29.8	29.7	29.6	29.5	29.3	29.2	29.1	29.0	28.9	28.8	28.7

- *BU = maximum assembly average burnup*
- *Use burnup and enrichment to lookup minimum cooling time in years. For fuel assemblies reconstituted with up to 10 stainless steel rods only, if the lookup cooling time is less than 9.0 years then a minimum cooling time of 9.0 years shall be used. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.*
- *Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.*
- *Fuel with an initial enrichment greater than 4.5 wt.% U-235 is unacceptable for storage.*
- *Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.*
- *Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 46.5 GWd/MTU is acceptable for storage after a 19.5 years cooling time as defined by 3.7 wt. % U-235 (rounding down) and 47 GWd/MTU (rounding up) on the qualification table.*
- *See Figure 1-5 for a description of zones.*
- For fuel *assemblies* reconstituted with Zircaloy clad uranium-oxide rods use the assembly average enrichment to determine the minimum cooling time.

Table 1-2o
PWR Fuel Qualification Table for Zone 2 with 1.0 kW per Assembly, Fuel With or Without BPRAs, for the NUHOMS®-24PHB DSC
(Minimum required years of cooling time after reactor core discharge)

BU (GWd/MTU)	Maximum Assembly Average Initial U-235 Enrichment (wt %)																									
	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
10	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
15	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
20	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
25		5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
28			5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
30						5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
32							5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
34								5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
36									5.5	5.5	5.5	5.5	5.5	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
38											6.0	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
39											6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	5.5	5.5	5.5	5.5	5.5	5.5	5.5
40											6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
41											6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.0	6.0	6.0	6.0	6.0	6.0	6.0
42											7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
43											7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	6.5	6.5	6.5	6.5
44											7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0
45													8.0	8.0	8.0	8.0	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.4
46													8.2	8.1	8.0	8.0	7.9	7.8	7.8	7.7	7.7	7.6	7.6	7.5	7.5	7.4
47													8.7	8.6	8.5	8.4	8.4	8.3	8.2	8.2	8.1	8.0	8.0	7.9	7.9	7.8
48													9.2	9.1	9.0	9.0	8.9	8.8	8.7	8.6	8.6	8.5	8.5	8.4	8.3	8.3
49													9.8	9.7	9.6	9.5	9.4	9.3	9.2	9.2	9.1	9.0	9.0	8.9	8.8	8.7
50															10.2	10.1	10.0	9.9	9.8	9.7	9.6	9.6	9.5	9.4	9.3	9.3
51															10.9	10.8	10.7	10.6	10.5	10.3	10.3	10.2	10.1	10.0	9.9	9.9
52															11.6	11.5	11.3	11.2	11.1	11.0	10.9	10.8	10.7	10.6	10.5	10.5
53															12.4	12.2	12.1	12.0	11.9	11.8	11.6	11.5	11.4	11.3	11.2	11.1
54															13.2	13.1	13.0	12.8	12.7	12.5	12.4	12.3	12.2	12.1	12.0	11.9
55															14.1	13.9	13.8	13.6	13.5	13.4	13.2	13.1	13.0	12.9	12.8	12.6

- *BU = maximum assembly average burnup*
- *Use burnup and enrichment to lookup minimum cooling time in years. For fuel assemblies reconstituted with up to 10 stainless steel rods only, if the lookup cooling time is less than 9.0 years then a minimum cooling time of 9.0 years shall be used. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.*
- *Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.*
- *Fuel with an initial enrichment greater than 4.5 wt.% U-235 is unacceptable for storage.*
- *Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.*
- *Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 46.5 GWd/MTU is acceptable for storage after a* 8.3 *years cooling time as defined by 3.7 wt. % U-235 (rounding down) and 47 GWd/MTU (rounding up) on the qualification table.*
- *See Figure 1-5 for a description of zones.*
- For *assemblies* fuel reconstituted with Zircaloy clad uranium-oxide rods use the assembly average enrichment to determine the minimum cooling time.

Table 1-2p

PWR Fuel Qualification Table for Zone 3 with 1.3 kW per Assembly, Fuel With or Without BPRAs, for the NUHOMS®-24PHB DSC

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

BU (GWd/MTU)	Maximum Assembly Average Initial U-235 Enrichment (wt %)																									
	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
10	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
15	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
20	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
25		5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
28			5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
30						5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
32							5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
34								5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
36									5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
38											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	5.5
39											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	5.5
40											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	5.5
41											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	5.5
42											6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
43											6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
44											6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
45													6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0	6.0
46													6.1	6.1	6.1	6.1	6.1	6.1	6.1	6.1	6.1	6.1	6.1	6.1	6.1	6.1
47													6.2	6.2	6.2	6.2	6.2	6.2	6.2	6.2	6.2	6.2	6.2	6.2	6.2	6.2
48													6.3	6.3	6.3	6.3	6.3	6.3	6.3	6.3	6.3	6.3	6.3	6.3	6.3	6.3
49													6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
50															6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5	6.5
51															6.7	6.6	6.6	6.6	6.6	6.6	6.6	6.6	6.6	6.6	6.6	6.6
52															7.0	6.9	6.9	6.8	6.8	6.8	6.8	6.8	6.8	6.8	6.8	6.8
53															7.3	7.2	7.2	7.1	7.1	7.0	6.9	6.9	6.9	6.9	6.9	6.9
54															7.7	7.6	7.5	7.4	7.4	7.3	7.3	7.2	7.1	7.1	7.0	7.0
55															8.0	8.0	7.9	7.8	7.7	7.7	7.6	7.5	7.5	7.4	7.3	7.3

- *BU = maximum assembly average burnup*
- *Use burnup and enrichment to lookup minimum cooling time in years. For fuel assemblies reconstituted with up to 10 stainless steel rods only, if the lookup cooling time is less than 9.0 years then a minimum cooling time of 9.0 years shall be used. Licensee is responsible for ensuring that uncertainties in fuel enrichment and burnup are correctly accounted for during fuel qualification.*
- *Round burnup UP to next higher entry, round enrichments DOWN to next lower entry.*
- *Fuel with an initial enrichment greater than 4.5 wt.% U-235 is unacceptable for storage.*
- *Fuel with a burnup less than 10 GWd/MTU is acceptable for storage after 5-years cooling.*
- *Example: An assembly with an initial enrichment of 3.75 wt. % U-235 and a burnup of 46.5 GWd/MTU is acceptable for storage after a 6.2 years cooling time as defined by 3.7 wt. % U-235 (rounding down) and 47 GWd/MTU (rounding up) on the qualification table.*
- *See Figure 1-5* and 1-6 *for a description of zones.*

For fuel *assemblies* reconstituted with Zircaloy clad uranium-oxide rods use the assembly average enrichment to determine the minimum cooling time.

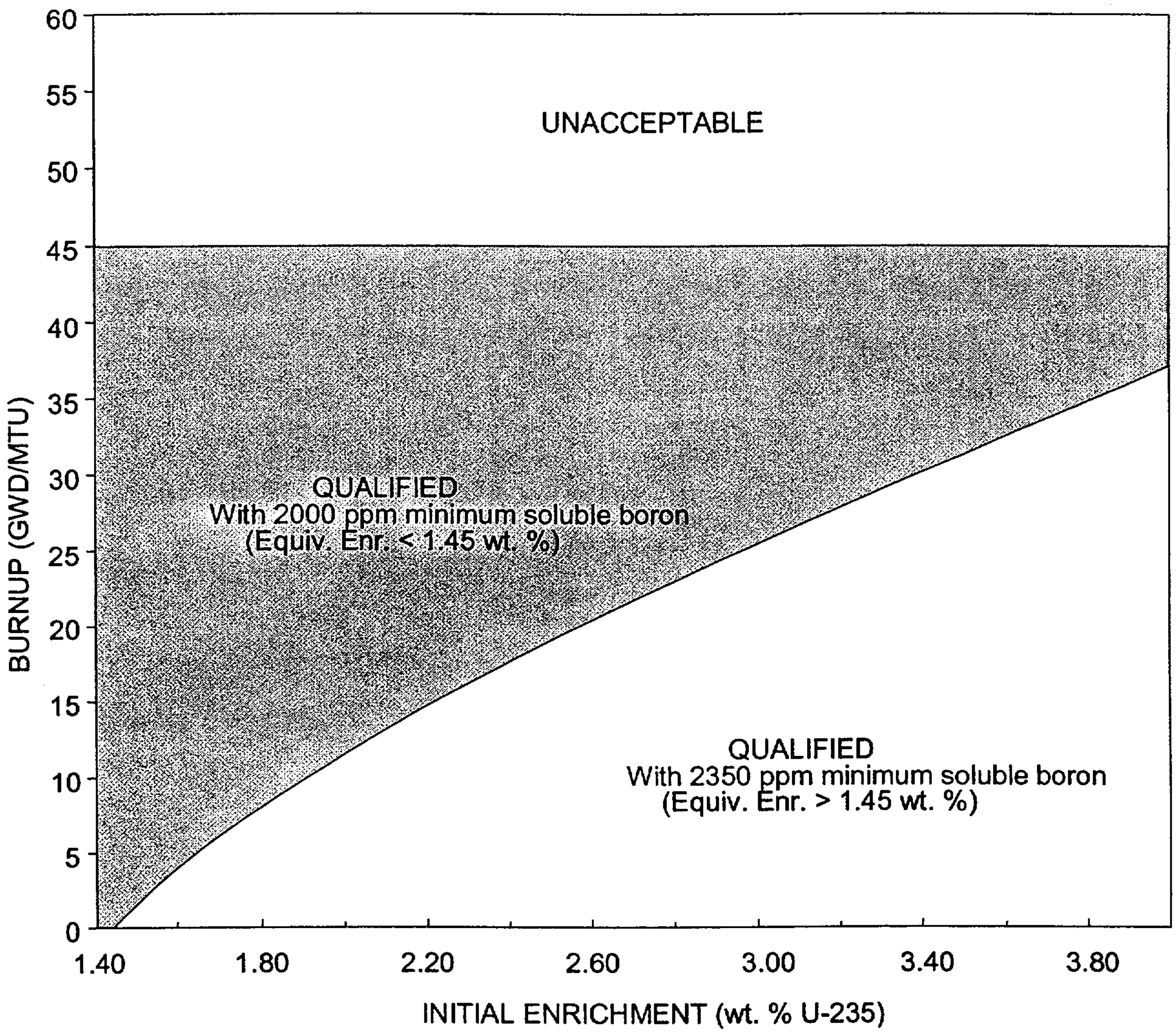


Figure 1-1
PWR Fuel Criticality Acceptance Curve

	0.87	0.87	0.87	0.87	
0.87	0.63	0.63	0.63	0.63	0.87
0.87	0.63	0.63	0.63	0.63	0.87
0.87	0.63	0.63	0.63	0.63	0.87
0.87	0.63	0.63	0.63	0.63	0.87
	0.87	0.87	0.87	0.87	

F5483

Figure 1-2
Heat Load Zoning Configuration 1 *(32PT)*

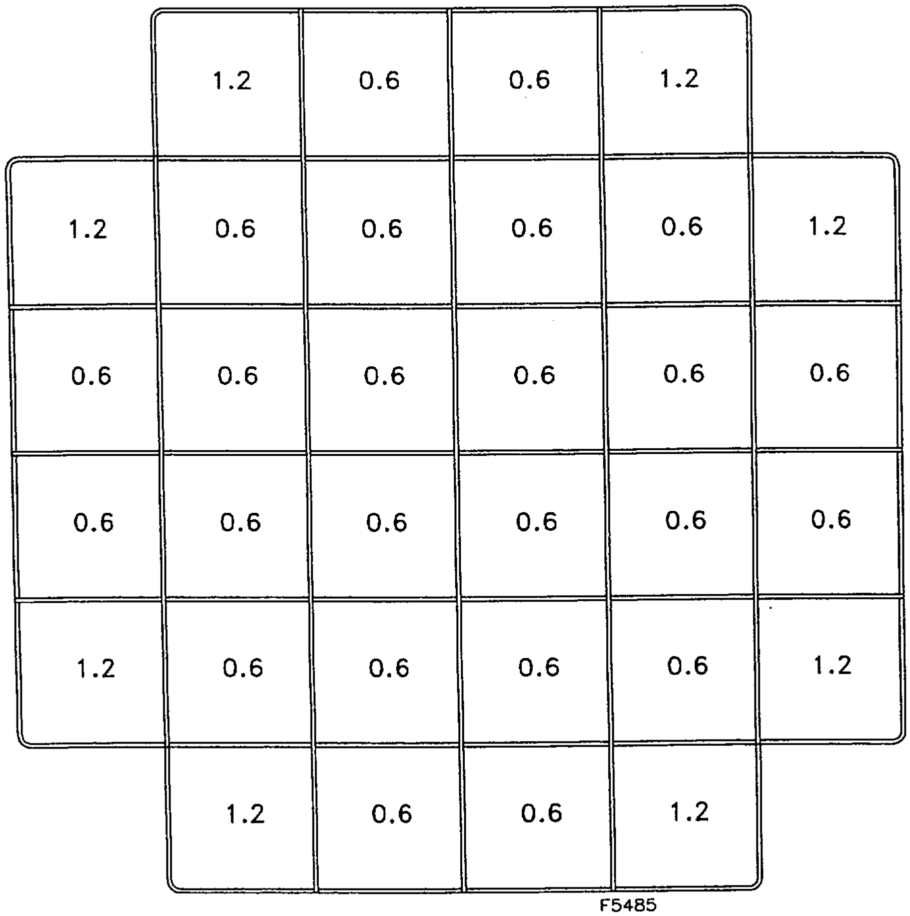


Figure 1-3
<u>Heat Load Zoning Configuration 2 *(32PT)*</u>

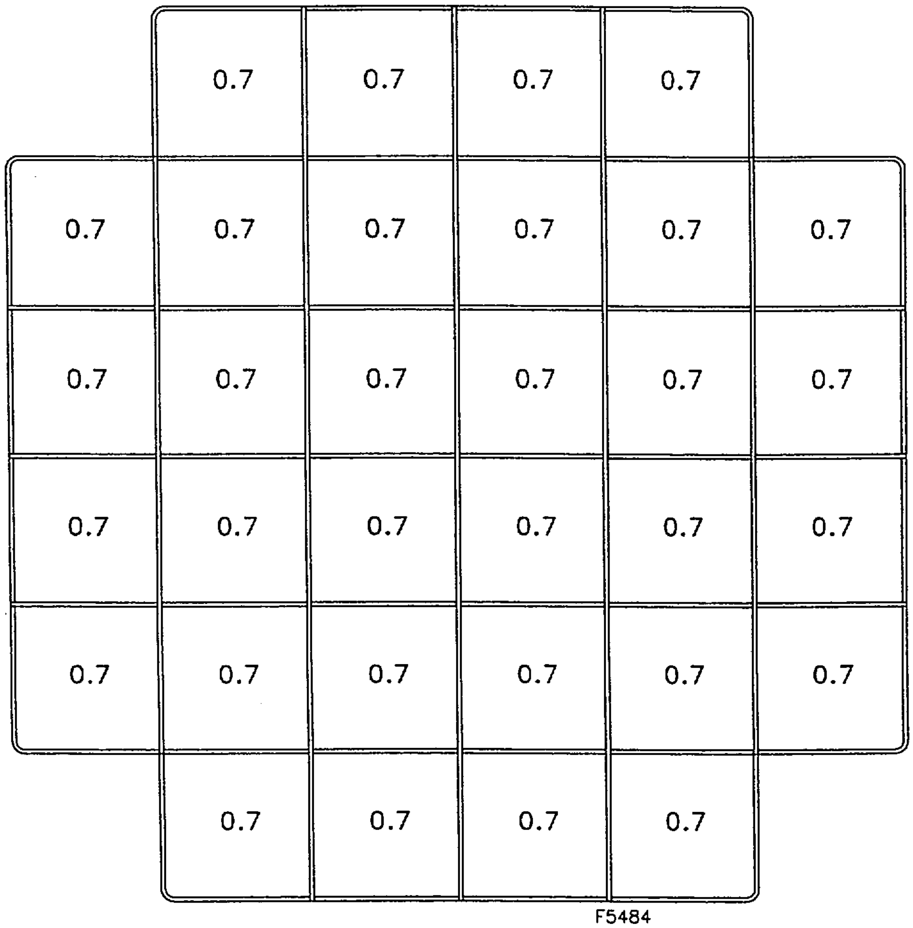
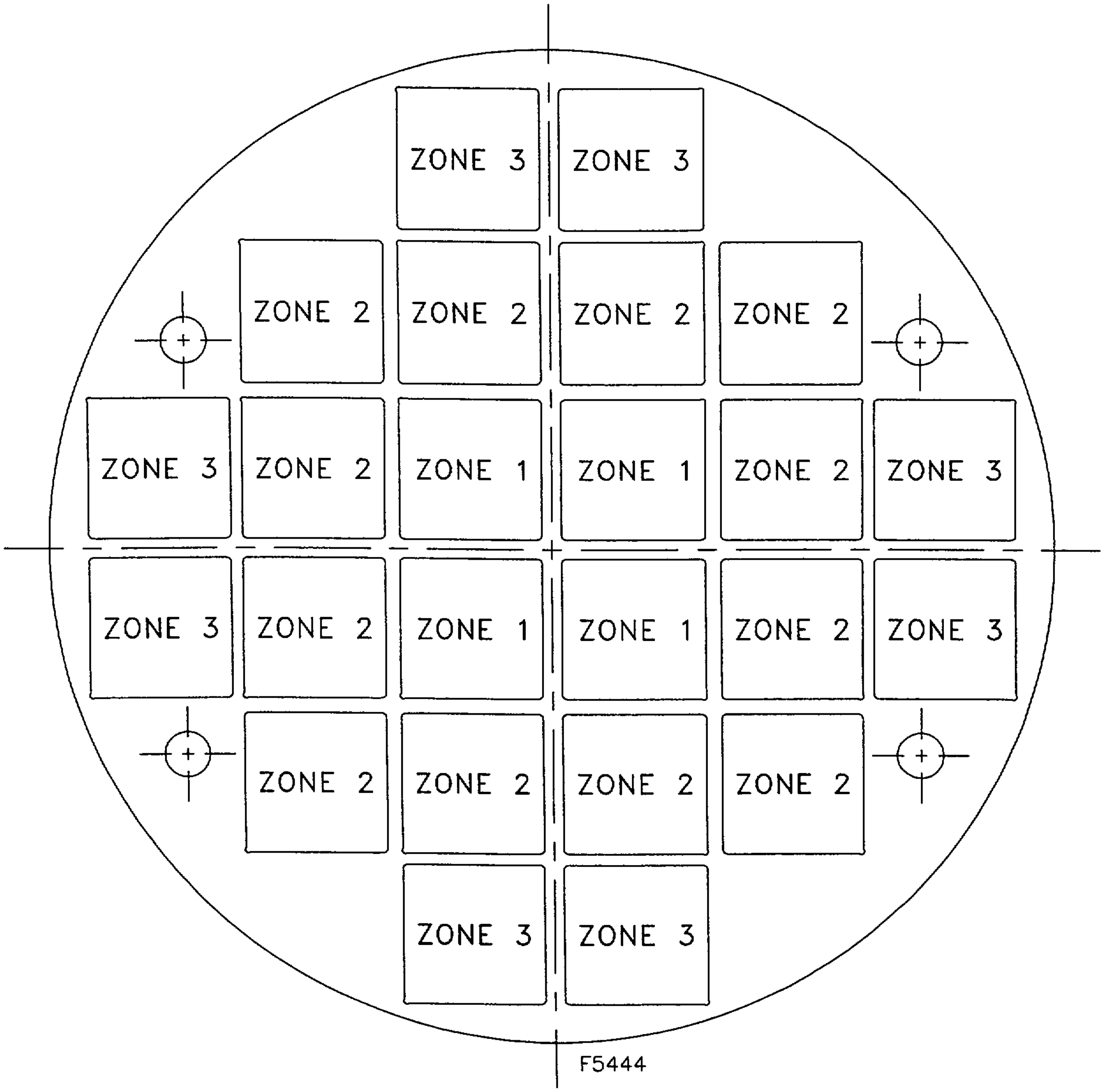
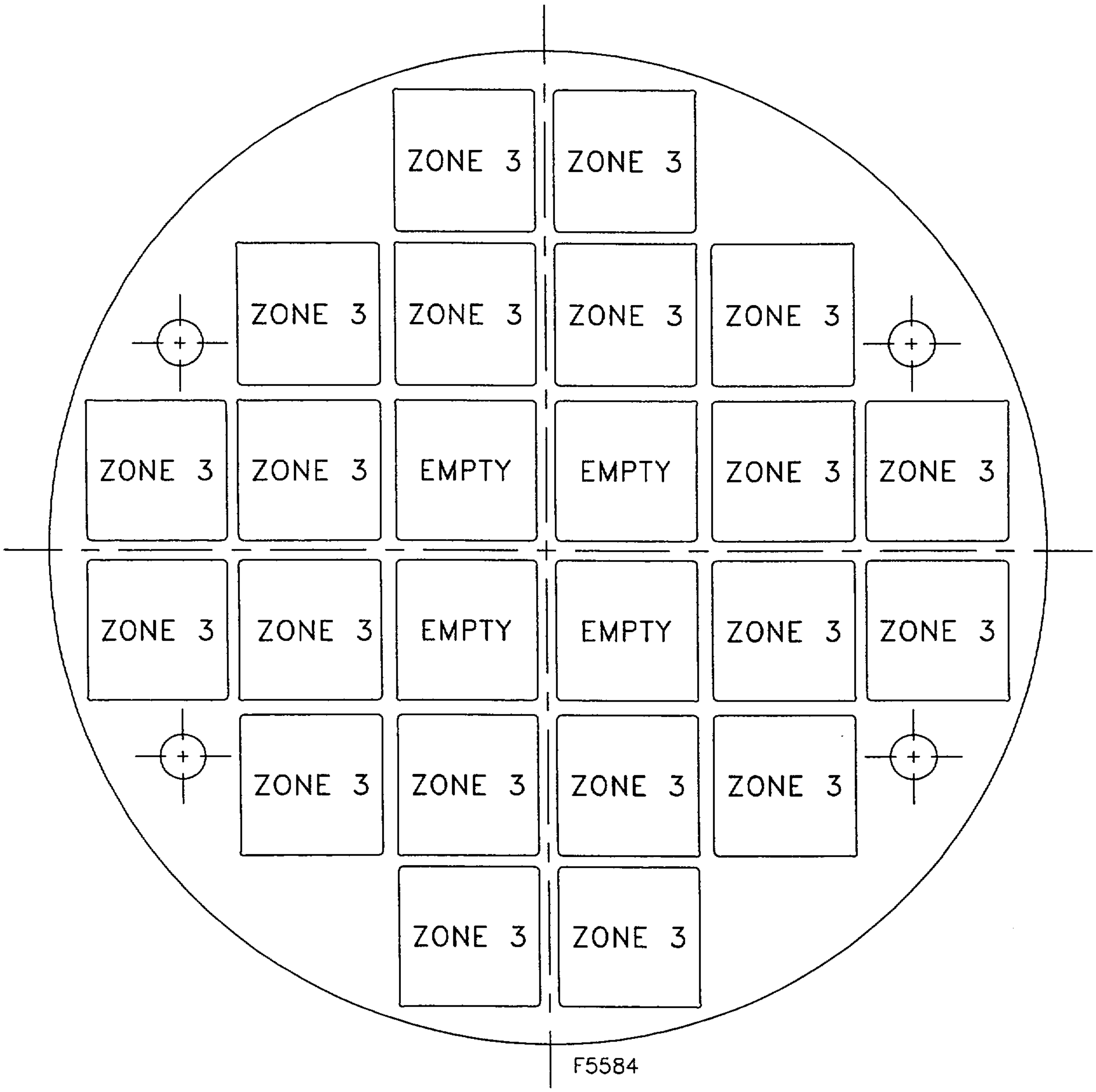


Figure 1-4
Heat Load Zoning Configuration 3 *(32PT)*



	Zone 1	*Zone 2*	*Zone 3*
Maximum Decay Heat (kW / FA)	*0.7*	*1.0*	*1.3*
Maximum Decay Heat per Zone (kW)	*2.8*	*10.8*	*10.4*

Figure 1-5
Heat Load Zoning Configuration for Fuel Assemblies (With or Without BPRAs) Stored in NUHOMS®-24PHB DSC – Configuration 1



	Zone 1	*Zone 2*	*Zone 3*
Maximum Decay Heat (kW / FA)	*NA*	*NA*	*1.3*
Maximum Decay Heat per Zone (kW)	*NA*	*NA*	*24.0*

Figure 1-6
Heat Load Zoning Configuration for Fuel Assemblies (With or Without BPRAs) Stored in NUHOMS®-24PHB DSC – Configuration 2

1.2.3 24P and 52B DSC Helium Backfill Pressure

Limit/Specifications:

Helium 2.5 psig ± 2.5 psig backfill pressure (stable for 30 minutes after filling).

Applicability: This specification is applicable to 24P and 52B DSCs only.

Objective: To ensure that: (1) the atmosphere surrounding the irradiated fuel is a non-oxidizing inert gas; (2) the atmosphere is favorable for the transfer of decay heat.

Action: If the required pressure cannot be obtained:

1. Confirm that the vacuum drying system and helium source are properly installed.
2. Check and repair or replace the pressure gauge.
3. Check and repair or replace the vacuum drying system.
4. Check and repair or replace the helium source.
5. Check and repair the seal weld on DSC top shield plug.

If pressure exceeds the criterion, release a sufficient quantity of helium to lower the DSC cavity pressure.

Surveillance: No maintenance or tests are required during the normal storage. Surveillance of the pressure gauge is required during the helium backfilling operation.

Bases: The value of 2.5 psig was selected to ensure that the pressure within the DSC is within the design limits during any expected normal and off-normal operating conditions.

1.2.3a 61BT, 32PT, *and 24PHB* DSC Helium Backfill Pressure

Limit/Specifications:

Helium 2.5 psig ± 1.0 psig backfill pressure (stable for 30 minutes after filling).

Applicability: This specification is applicable to 61BT, 32PT, *and 24PHB* DSC only.

Objective: To ensure that: (1) the atmosphere surrounding the irradiated fuel is a non-oxidizing inert gas; (2) the atmosphere is favorable for the transfer of decay heat.

Action: If the required pressure cannot be obtained:

1. Confirm that the vacuum drying system and helium source are properly installed.
2. Check and repair or replace the pressure gauge.
3. Check and repair or replace the vacuum drying system.
4. Check and repair or replace the helium source.
5. Check and repair the seal weld on DSC top shield plug.

If pressure exceeds the criterion, release a sufficient quantity of helium to lower the DSC cavity pressure.

Surveillance: No maintenance or tests are required during the normal storage. Surveillance of the pressure gauge is required during the helium backfilling operation.

Bases: The value of 2.5 psig was selected to ensure that the pressure within the DSC is within the design limits during any expected normal and off-normal operating conditions.

1.2.4 24P and 52B DSC Helium Leak Rate of Inner Seal Weld

Limit/Specification:

$\leq 1.0 \times 10^{-4}$ atm · cubic centimeters per second (atm · cm^3/s) at the highest DSC limiting pressure.

Applicability: This specification is applicable to the inner top cover plate seal weld of the 24P and 52B DSCs only.

Objective:

1. To limit the total radioactive gases normally released by each canister to negligible levels. Should fission gases escape the fuel cladding, they will remain confined by the DSC confinement boundary.

2. To retain helium cover gases within the DSC and prevent oxygen from entering the DSC. The helium improves the heat dissipation characteristics of the DSC and prevents any oxidation of fuel cladding.

Action: If the leak rate test of the inner seal weld exceeds 1.0×10^{-4} (atm · cm^3/s):

1. Check and repair the DSC drain and fill port fittings for leaks.

2. Check and repair the inner seal weld.

3. Check and repair the inner top cover plate for any surface indications resulting in leakage.

Surveillance: After the welding operation has been completed, perform a leak test with a helium leak detection device.

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

1.2.4a 61BT, 32PT, *and 24PHB* DSC Helium Leak Rate of Inner Seal Weld

Limit/Specification:

$\leq 1.0 \times 10^{-7}$ atm · cubic centimeters per second (atm · cm^3/s) at the highest DSC limiting pressure.

Applicability: This specification is applicable to the inner top cover plate seal weld of 61BT, 32PT, *and 24PHB* DSC only.

Objective:

1. To demonstrate that the top cover plate to be "leak tight", as defined in "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials," ANSI N14.5 - 1997.
2. To retain helium cover gases within the DSC and prevent oxygen from entering the DSC. The helium improves the heat dissipation characteristics of the DSC and prevents any oxidation of fuel cladding.

Action: If the leak rate test of the inner seal weld exceeds 1.0×10^{-7} (atm · cm^3/s):

1. Check and repair the DSC drain and fill port fittings for leaks.
2. Check and repair the inner seal weld.
3. Check and repair the inner top cover plate for any surface indications resulting in leakage.

Surveillance: After the welding operation has been completed, perform a leak test with a helium leak detection device.

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

1.2.7 *24P, 52B, 61BT, and 32PT HSM Dose Rates*

Limit/Specification:

Dose rates at the following locations shall be limited to levels which are less than or equal to:

a. 600 mrem/hr at 3 feet from the HSM surface.

b. Outside of HSM door on center line of DSC 200 mrem/hr.

c. End shield wall exterior 20 mrem/hr.

Applicability: This specification is applicable to all HSMs which contain a loaded *24P, 52B, 61BT, or 32PT* DSC.

Objective: The dose rate is limited to this value to ensure that the cask (DSC) has not been inadvertently loaded with fuel not meeting the specifications in Section 1.2.1 and to maintain dose rates as-low-as-is-reasonably achievable (ALARA) at locations on the HSMs where surveillance is performed, and to reduce off-site exposures during storage.

Action:

a. If specified dose rates are exceeded, the following actions should be taken:

   1. Ensure that the DSC is properly positioned on the support rails.

   2. Ensure proper installation of the HSM door.

   3. Ensure that the required module spacing is maintained.

   4. Confirm that the spent fuel assemblies contained in the DSC conform to the specifications of Section 1.2.1.

   5. Install temporary or permanent shielding to mitigate the dose to acceptable levels in accordance with 10 CFR Part 20, 10 CFR 72.104(a), and ALARA.

b. Submit a letter report to the NRC within 30 days summarizing the action 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 the administrator of the appropriate NRC regional office.

Surveillance: The HSM and ISFSI shall be checked to verify that this specification has been met after the DSC is placed into storage and the HSM door is closed.

Basis: The basis for this limit is the shielding analysis presented in Section 7.0 and Appendix J, Appendix K and Appendix M of the FSAR. The specified dose rates provide as-low-as-is-reasonably-achievable on-site and off-site doses in accordance with 10 CFR Part 20 and 10 CFR 72.104(a).1.2.11

1.2.7a 24PHB HSM Dose Rates

Limit/Specification:

Dose rates at the following locations shall be limited to levels which are less than or equal to:

a. 500 mrem/hr at 3 feet from the HSM front surface.

b. Outside of HSM door on center line of DSC 20 mrem/hr.

c. End shield wall exterior 300 mrem/hr.

Applicability: *This specification is applicable to all HSMs which contain a loaded 24PHB DSC.*

Objective: *The dose rate is limited to this value to ensure that the cask (DSC) has not been inadvertently loaded with fuel not meeting the specifications in Section 1.2.1 and to maintain dose rates as-low-as-is-reasonably achievable (ALARA) at locations on the HSMs where surveillance is performed, and to reduce off-site exposures during storage.*

Action: *a. If specified dose rates are exceeded, the following actions should be taken:*

1. Ensure that the DSC is properly positioned on the support rails.

2. Ensure proper installation of the HSM door.

3. Ensure that the required module spacing is maintained.

4. Confirm that the spent fuel assemblies contained in the DSC conform to the specifications of Section 1.2.1.

5. Install temporary or permanent shielding to mitigate the dose to acceptable levels in accordance with 10 CFR Part 20, 10 CFR 72.104(a), and ALARA.

b. Submit a letter report to the NRC within 30 days summarizing the action 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 the administrator of the appropriate NRC regional office.

Surveillance: *The HSM and ISFSI shall be checked to verify that this specification has been met after the DSC is placed into storage and the HSM door is closed.*

Basis: *The basis for this limit is the shielding analysis presented in Appendix N of the FSAR. The specified dose rates provide as-low-as-is-reasonably-achievable on-site and off-site doses in accordance with 10 CFR Part 20 and 10 CFR 72.104(a).1.2.11*

1.2.11 *DELETED*

1.2.15 Boron Concentration in the DSC Cavity Water for the 24-P Design Only

Limit/Specification:

The DSC cavity shall be filled only with water having a boron concentration equal to, or greater than:

1) 2,000 ppm for fuel with an equivalent unirradiated enrichment of less than or equal to 1.45 wt. % U-235 per Figure 1-1.
2) 2,350 ppm for fuel with an equivalent unirradiated enrichment of greater than 1.45 wt. % U-235 per Figure 1-1.

Applicability: This limit applies only to the standardized NUHOMS-24P design. No boration in the cavity water is required for the standardized NUHOMS-52B or NUHOMS-61BT system since that system uses fixed absorber plates.

Objective:

1) To ensure a subcritical configuration is maintained in the case of accidental loading of the DSC with unirradiated fuel.
2) To ensure a subcritical configuration is maintained in the case of loading of the DSC with fuel with an equivalent unirradiated enrichment of greater than 1.45 wt. % U-235.

Action: If the boron concentration is below the required weight percentage concentration (gm boron/10^6 gm water), add boron and re-sample, and test the concentration until the boron concentration is shown to be greater than that required.

Surveillance: Written procedures shall be used to independently determine (two samples analyzed by different individuals) the boron concentration in the water used to fill the DSC cavity.

1. Within 4 hours before insertion of the first fuel assembly into the DSC, the dissolved boron concentration in water in the spent fuel pool, and in the water that will be introduced in the DSC cavity, shall be independently determined (two samples chemically analyzed by two individuals).

2. Within 4 hours before flooding the DSC cavity for unloading the fuel assemblies, the dissolved boron concentration in water in the spent pool, and in the water that will be introduced into the DSC cavity, shall be independently determined (two samples analyzed chemically by two individuals).

3. The dissolved boron concentration in the water shall be reconfirmed at intervals not to exceed 48 hours until such time as the DSC is removed from the spent fuel pool or the fuel has been removed from the DSC.

Bases:

1) The required boron concentration is based on the criticality analysis for an accidental misloading of the DSC with unburned fuel, maximum enrichment, and optimum moderation conditions.
2) The required boron concentration is based on the criticality analysis for loading of the DSC with unirradiated fuel, maximum enrichment, and optimum moderation conditions.

1.2.15a Boron Concentration in the DSC Cavity Water for the 32PT Design Only

Limit/Specification:

The DSC cavity shall be filled only with water having a boron concentration equal to, or greater than 2500 ppm.

Applicability: This limit applies only to the standardized NUHOMS®- 32PT design.

Objective: To ensure a subcritical configuration is maintained in the case of loading of the DSC with design basis fuel.

Action: If the boron concentration is below the required weight percentage concentration (gm boron/10^6 gm water), add boron and re-sample, and test the concentration until the boron concentration is shown to be greater than that required.

Surveillance: Written procedures shall be used to independently determine (two samples analyzed by different individuals) the boron concentration in the water used to fill the DSC cavity.

1. Within 4 hours before insertion of the first fuel assembly into the DSC, the dissolved boron concentration in water in the spent fuel pool, and in the water that will be introduced in the DSC cavity, shall be independently determined (two samples chemically analyzed by two individuals).

2. Within 4 hours before flooding the DSC cavity for unloading the fuel assemblies, the dissolved boron concentration in water in the spent pool, and in the water that will be introduced into the DSC cavity, shall be independently determined (two samples analyzed chemically by two individuals).

3. The dissolved boron concentration in the water shall be reconfirmed at intervals not to exceed 48 hours until such time as the DSC is removed from the spent fuel pool or the fuel has been removed from the DSC.

Bases: The required boron concentration is based on the criticality analysis for loading of the DSC with unirradiated fuel, maximum enrichment, and optimum moderation conditions.

1.2.15b Boron Concentration in the DSC Cavity Water for the 24PHB Design Only

Limit/Specification:

- *The DSC cavity shall be filled only with water having a boron concentration equal to, or greater than 2,350 ppm for enrichment of less than or equal to 4.0 wt. % U-235 based on the spent fuel assembly with the maximum initial enrichment in the DSC.*
- *The DSC cavity shall be filled only with water having a minimum boron concentration per Figure 1-7 for initial enrichment of greater than or equal to 4.0 wt. % U-235 based on the spent fuel assembly with the maximum initial enrichment in the DSC.*

Applicability: *This limit applies only to the standardized NUHOMS-24PHB design.*

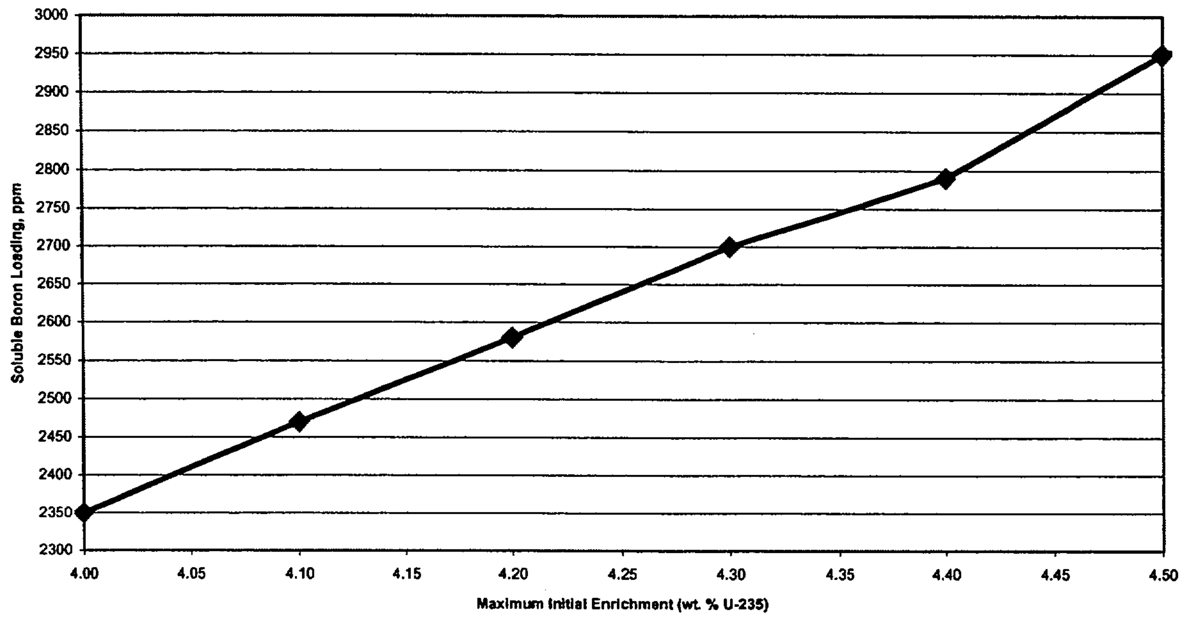
Objective: *To ensure a subcritical configuration is maintained in the case of accidental loading of the DSC with unirradiated fuel.*

Action: *If the boron concentration is below the required weight percentage concentration (gm boron/10^6 gm water), add boron and re-sample, and test the concentration until the boron concentration is shown to be greater than that required.*

Surveillance: *Written procedures shall be used to independently determine (two samples analyzed by different individuals) the boron concentration in the water used to fill the DSC cavity.*

1. *Within 4 hours before insertion of the first fuel assembly into the DSC, the dissolved boron concentration in water in the spent fuel pool, and in the water that will be introduced in the DSC cavity, shall be independently determined (two samples chemically analyzed by two individuals).*
2. *Within 4 hours before flooding the DSC cavity for unloading the fuel assemblies, the dissolved boron concentration in water in the spent pool, and in the water that will be introduced into the DSC cavity, shall be independently determined (two samples analyzed chemically by two individuals).*
3. *The dissolved boron concentration in the water shall be reconfirmed at intervals not to exceed 48 hours until such time as the DSC is removed from the spent fuel pool or the fuel has been removed from the DSC.*

Bases: *The required boron concentration is based on the criticality analysis for loading of the DSC with unirradiated fuel, maximum enrichment, and optimum moderation conditions.*



Linear Interpolation allowed between points

Initial Enrichment	*Boron Loading, ppm*
≤4.0	*2350*
4.1	*2470*
4.2	*2580*
4.3	*2700*
4.4	*2790*
4.5	*2950*

Figure 1-7
Soluble Boron Concentration vs. Fuel Assembly Initial U-235 Enrichment for the 24PHB System

1.2.17 61BT and 32PT DSC Vacuum Drying Duration Limit

Limit/Specifications:

1. Time limit for duration of Vacuum Drying is 96 hrs after completion of 61BT DSC draining.

2. Time limit for duration of Vacuum Drying is 55 hrs after completion of 32PT DSC draining.

Applicability: This specification is applicable to a 61BT DSC with greater than 17.6 kW heat load and a 32PT DSC with a design basis.

Objective: To ensure that 61BT and 32PT DSC basket structure does not exceed 800^0F.

Action:

1. If the DSC vacuum drying pressure limit of Technical Specification 1.2.2 cannot be achieved at 72 hours for 61BT DSC or 31 hours for the 32PT DSC after completion of DSC draining, the DSC must be backfilled with 0.1 atm or greater helium pressure within 24 hours.

2. Determine the cause of failure to achieve the vacuum drying pressure limit as defined in Technical Specification 1.2.2.

3. Initiate vacuum drying after actions in Step 2 are completed or unload the DSC within 30 days.

Surveillance: No maintenance or tests are required during the normal storage. Monitoring of the time duration during the vacuum drying operation is required .

Bases: The time limits of 96 hours for the 61BT DSC and 55 hours for the 32PT DSC were selected to ensure that the temperature of the DSC basket structure is within the design limits during vacuum drying.

1.2.17a 24PHB DSC Vacuum Drying Duration Limit

Limit/Specifications:

The limit for duration of Vacuum Drying is 20 hrs after completion of 24PHB DSC draining.

Applicability: *This specification is applicable to a 24PHB DSC with design basis heat load.*

Objective: *To ensure the fuel cladding temperature in the 24PHB DSC does not exceed 650°F.*

Action:

1. *If the DSC vacuum drying pressure limit of Technical Specification 1.2.2 cannot be achieved at 20 hours after initiation of vacuum drying, the DSC must be backfilled with 0.1 atm or greater helium pressure within 2 hours.*
2. *Determine the cause of failure to achieve the vacuum drying pressure limit as defined in Technical Specification 1.2.2.*
3. *Initiate vacuum drying after actions in Step 2 are completed or unload the DSC within 30 days.*

Surveillance: *No maintenance or tests are required during the normal storage. Monitoring of the time duration during the vacuum drying operation is required.*

Bases: *The time limit of 20 hours for the 24PHB DSC was selected to ensure that the cladding temperature is within acceptable limits during vacuum drying.*

1.3 Surveillance and Monitoring

The NRC staff is requiring the following surveillance frequency for the HSM.

1.3.1 Visual Inspection of HSM Air Inlets and Outlets (Front Wall and Roof Birdscreen)

Limit/Surveillance:	A visual surveillance of the exterior of the air inlets and outlets shall be conducted daily. In addition, a close-up inspection shall be performed to ensure that no materials accumulate between the modules to block the air flow.
Objective:	To ensure that HSM air inlets and outlets are not blocked for more than 40 hours *for HSMs loaded with less than 45,000 MWD/MTU burnup fuel, or 34 hours for HSMs loaded with 24PHB DSC and fuel that is greater than 45,000 MWD/MTU burnup. The time limit is* to prevent exceeding the allowable HSM concrete *and* or the fuel cladding temperature*s*.
Applicability:	This specification is applicable to all HSMs loaded with a DSC loaded with spent fuel.
Action:	If the surveillance shows blockage of air vents (inlets or outlets), they shall be cleared. If the screen is damaged, it shall be replaced.
Basis:	The concrete temperature could exceed 350°F in the accident circumstances of complete blockage of all vents if the period exceeds approximately 40 hours. *The fuel cladding could exceed 400°C (752°F) in the accident circumstances of complete blockage of all vents for more than 34 hours and with high burnup fuel loaded in the 24PHB DSC. Fuel cladding temperatures over 752°F can have uncertain impact on fuel cladding integrity.* Concrete temperatures over 350°F in accidents (without the presence of water or steam) can have uncertain impact on concrete strength and durability. A conservative analysis (adiabatic heat case) of complete blockage of all air inlets or outlets indicates that the concrete can reach the accident temperature limit of 350°F in *the* time period*s specified.*

Table 1.3.1
Summary of Surveillance and Monitoring Requirements

Surveillance or Monitoring	Period	Reference Section
1. Fuel Specification	PL	1.2.1
2. DSC Vacuum Pressure During Drying	L	1.2.2
3. DSC Helium Backfill Pressure	L	1.2.3 or 1.2.3a
4. DSC Helium Leak Rate of Inner Seal Weld	L	1.2.4 or 1.2.4a
5. DSC Dye Penetrant Test of Closure Welds	L	1.2.5
6. Deleted	-	-
7. 24P, 52B, 61BT, and 32PT HSM Dose Rates *7a 24PHB HSM Dose Rates*	L	1.2.7 *or* *1.2.7a*
8. HSM Maximum Air Exit Temperature	24 hrs	1.2.8
9. TC Alignment with HSM	S	1.2.9
10. DSC Handling Height Outside Spent Fuel Pool Building	AN	1.2.10
11. *DELETED*		
12. Maximum DSC Surface Contamination	L	1.2.12
13. TC/DSC Lifting Heights as a Function of Low Temperature and Location	L	1.2.13
14. TC/DSC Transfer Operations at High Ambient Temperatures	L	1.2.14
15. Boron Concentration in DSC Cavity Water (24-P Designs Only) 15a Boron Concentration in DSC Cavity Water (32PT Designs Only) *15b Boron Concentration in DSC Cavity Water (24PHB Designs Only)*	PL PL *PL*	1.2.15 1.2.15a *1.2.15b*
16. Provision of TC Seismic Restraint Inside the Spent Fuel Pool Building as a Function of Horizontal Acceleration and Loaded Cask Weight	PL	1.2.16
17. Visual Inspection of HSM Air Inlets and Outlets	D	1.3.1
18. HSM Thermal Performance	D	1.3.2
19 *61BT and 32PT DSC Vacuum Drying Duration Limit* *19a 24PHB DSC Vacuum Drying Duration Limit*	L	1.2.17 *1.2.17a*

ATTACHMENT C
FSAR Revision 5 Changed Pages

(Revisions indicated relative to FSAR, Revision 5.). Listed below are the affected FSAR pages:

- Page 1.1-2
- Page 1.2-3
- Page 1.2-7
- Page 1.2-10
- Page 1.2-11
- Page 1.3-1
- Page 1.3-7
- Page 1.3-9
- Page 1.3-11
- Page 3.1-1
- Page 3.1-4
- Page 3.2-6
- Page 3.2-8
- Page 3.2-10
- Page 3.2-19
- Page 3.2-20
- Page 3.3-2
- Page 3.3-30
- Page 3.3-31
- Page 3.3-33
- Page 3.4-4
- Page 4.2-1
- Page 4.2-2
- Page 4.2-16
- Page 5.1-1
- Page 7.2-1
- Page 7.3-3
- Page 7.4-1
- Page 7.4-2
- Page 8.1-1
- Page 11.2-1
- Page E.1

Standardized or OS 197 Transfer Cask as the NUHOMS®-24P DSC. The NUHOMS®-24PT2 DSC is similar to the existing DSCs with the following exceptions:

- The inner bottom closure plate employs a full penetration weldment,
- The 24PT2 basket represents a design similar to the 24P basket design except that there are more spacer discs, made of thinner, stronger material,
- The 24PT2 DSC is intended to meet the transportation requirements of 10CFR71 under a future amendment to NUHOMS® CoC 71-9255.

Supporting criticality, thermal, and structural analyses for the 24PT2 basket and shell are included in Appendix L of this FSAR. The specifications of spent fuel assemblies to be stored are the same as those for the 24P design.

The NUHOMS®-24PHB DSC is also included in this SAR. The NUHOMS®-24PHB DSC is designed to store a total of 24 intact B&W 15x15 fuel assemblies with an assembly average burnup of up to 55,000 MWd/MTU and an initial enrichment of up to 4.5 weight % U-235. The NUHOMS®-24PHB DSC is designed for storage in the existing Model 102 NUHOMS® HSM and transfer in an existing Standard, OS 197 or OS 197H transfer cask. The NUHOMS®-24PHB DSC is identical to the existing 24P DSC with one exception: The outer top cover plate of the canister contains a test port and plug to allow testing the canister to a "leak tight" condition per ANSI N14.5-1997 criteria.

Supporting criticality, thermal, shielding and structural analyses for the 24PHB DSC, as well as specifications of spent fuel assemblies to be stored, are included in Appendix N of this FSAR.

The remainder of this chapter provides a general overview of the standardized NUHOMS® system and summarizes the contents of this SAR.

1.1 Introduction

Due to the unavailability of nuclear fuel reprocessing or a permanent geologic repository in the United States (U.S.), long-term storage of spent fuel assemblies (SFAs) has become necessary. To date, storage systems have, to a large extent, relied on the plant's spent fuel pools. However, as existing pools have begun to approach their capacity (with high-density storage racks), out-of-pool dry storage system designs have emerged. NUHOMS® is a proven system for dry storage which has been in use at reactor sites since March of 1989.

Figure 1.1-1, Figure 1.1-2 and Figure 1.1-3 show the primary components and arrangement of an ISFSI utilizing the NUHOMS® system. The SFAs are loaded into the DSC (which is placed inside the transfer cask) in the fuel pool at the reactor site. The and roof. Air enters near the bottom of the HSM, circulates and rises around the DSC and exits through shielded openings near the top of the HSM. The cross-sectional areas of the air inlet and outlet openings, and the interior flow paths are designed to optimize ventilation air flow in the HSM for decay heat removal including worst case extreme summer ambient conditions. The thermal performance features of the NUHOMS® system are described in Chapters 4 and 8.

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

The ambient temperatures selected for the design of the NUHOMS® system range from -40°F to 125°F, with a lifetime average ambient temperature of 70°F. The extreme ambient temperatures of -40°F, *117°F* and 125°F are expected to last for a short period of time, i.e., on the order of hours. The minimum and maximum average ambient temperatures of 0°F and 100°F are expected to last for longer periods of time, i.e., on the order of days.

1.2.3 Operating and Fuel Handling Systems

Some handling equipment and support systems within the plant needed to implement the NUHOMS® system are covered by the licensee's 10CFR50 operating license. The on-site transfer cask is designed to satisfy a range of plant specific conditions and requirements. The general operations for a typical NUHOMS® system installation are summarized in Table 1.2-3. A more detailed procedure for this sequence of operations is provided in Section 5.1 *and Appendix N*. The majority of the fuel handling operations involving the DSC and transfer cask (i.e. fuel loading, draining and drying, transport trailer loading etc.) utilize procedures similar to those already in place at reactor sites for SFA shipment. The remaining operations (canister sealing, cask-HSM alignment and DSC transfer) are unique to the NUHOMS® system.

1.2.4 Safety Features

The principal safety features of a NUHOMS® ISFSI include the high integrity containment for the confinement of spent fuel materials, the axial shielding provided by the DSC, and the extensive biological shielding and protection against extreme natural phenomena provided by the massive reinforced concrete HSM. The shielding materials incorporated into the DSC and HSM designs reduce the gamma and neutron flux emanating from the SFAs so

Table 1.2-2
Key Design Parameters for the Standardized NUHOMS® System

Category	Criteria or Parameter	Value	
		PWR [4]	BWR
Fuel Assembly[1] Criteria:	Initial Uranium Content (kg/assembly)	475	198
	Initial Enrichment (U-235 equivalent)	4.0%	4.0%
	Fuel Burnup (MWD/MTU)	45,000	45,000
	Gamma Radiation Source (photons/sec/assembly)	4.48E15 (10 year cooled)[2] 7.45E15 (5 year cooled)	1.55E15 (10 year cooled)[2] 2.63E15 (5 year cooled)
	Neutron Radiation Source (neutron/sec./assembly)	1.55E8 (10 year cooled)[2] 2.23E8 (5 year cooled)	8.40E7 (10 year cooled)[2] 1.01E8 (5 year cooled)
	Decay Heat Power (kW/assembly)	1.00	0.37
	Minimum BPRA Cooling Time (years)	5 for B&W Designs 10 for Westinghouse Designs	N/A
Dry Shielded Canister:	Fuel Assemblies per DSC	24	52
	Size [3]:		
	Overall Length	4.72m (186.0 in.)	4.97m (196.0 in.)
	Outside Diameter	1.71m (67.2 in.)	1.71m (67.2 in.)
	Shell Thickness	16mm (0.625 in.)	16mm (0.625 in.)
	Heat Rejection (kW)	24.0	19.2
	Internal Atmosphere	Helium	Helium

(1) Enveloping design basis fuel plus BPRAs.
(2) 10 year cooled fuel data provided for information only.
(3) These are nominal dimensions. See Appendix E drawings for actual dimensions.
(4) *Appendix N contains the key design parameters for the PWR fuel assemblies and the 24PHB DSC.*

Table 1.2-3
NUHOMS® System Operations Overview[(1) (2)]

1. Clean and load the DSC into the transfer cask

2. Fill the DSC and cask with water and install the cask/DSC annulus seal

3. Place the transfer cask containing the DSC in the fuel pool

4. Load the spent fuel assemblies into the DSC

5. Place the top shield plug on the DSC

6. Remove the loaded cask from the fuel pool and place it in the decon area

7. Lower the water level in the DSC cavity below the shield plug

8. Place and weld the inner top cover plate to the DSC shell and perform NDE

9. Drain the water from the cask/DSC annulus (may be delayed until after completion of step 11 or 16)

10. Drain the water from the DSC

11. Evacuate and dry the DSC

12. Backfill the DSC with helium

13. Perform a helium leak test on the seal weld

14. Seal weld the siphon and vent port plugs and perform NDE

15. Fit-up the outer top cover plate with the DSC shell

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

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

---

(1) See Section 5.1 for more detailed system operation description.
(2) *See Appendix N for the operations overview of the NUHOMS®-24PHB system.*

1.3 General Systems Description

The components, structures and equipment which make up the NUHOMS® system are listed in Table 1.3-1. The following subsections briefly describe the design features and operation of these NUHOMS® system elements.

1.3.1 Storage Systems Descriptions

1.3.1.1 Dry Shielded Canister

The principal design features of the NUHOMS® DSC are listed in Table 1.3-1 and shown in Figure 1.3-1, Figure 1.3-2 and Figure 1.3-3. Table 1.2-2 lists the capacity, dimensions and design parameters for the NUHOMS® DSC. The cylindrical shell, and the top and bottom cover plate assemblies form the pressure retaining containment boundary for the spent fuel. The DSC is equipped with two shield plugs so that occupational doses at the ends are minimized for drying, sealing, and handling operations.

The DSC has double, redundant seal welds which join the shell and the top and bottom cover plate assemblies to form the containment boundary. The bottom end assembly containment boundary welds are made during fabrication of the DSC. The top end assembly containment boundary welds are made after fuel loading. Both top plug penetrations (siphon and vent ports) are redundantly sealed after DSC drying operations are complete. This assures that no single failure of the DSC top or bottom end assemblies will breach the DSC containment boundary. Furthermore, there are no credible accidents which could breach the containment boundary of the DSC as documented by this SAR.

The internal basket assembly contains a storage position for each fuel assembly. The criticality analysis performed for the NUHOMS®-24P, *24PHB* and 24PT2 DSC for PWR fuel accounts for fuel burnup or takes credit for soluble boron and demonstrates that fixed borated neutron absorbing material is not required in the basket assembly for criticality control. The Boral™ of the 24PT2 DSC is modeled only as unborated aluminum. Fixed neutron absorbing material is used for the NUHOMS®-52B DSC for channeled BWR fuel. Subcriticality during wet loading, drying, sealing, transfer, and storage operations is maintained through the geometric separation of the fuel assemblies by the DSC basket assembly and the neutron absorbing capability of the DSC materials of construction.

Structural support for the PWR fuel and basket guide sleeves or BWR fuel and channels in the lateral direction is provided by circular spacer disk plates *in the 24P, 24PHB or 52B DSCs*. Axial support for the NUHOMS®-24P *and 24PHB* DSC basket assembl*ies* is provided by four support rods which are welded to the spacer discs. Axial support for the NUHOMS®24PT2 DSC basket assembly is provided by four preloaded support rods and spacer sleeves. Axial support for the NUHOMS®52B DSC basket assembly is provided by six preloaded support rods and spacer sleeves. For the 24P, *24PHB,* 24PT2 and 52B DSCs, reduced, the helium lines removed, and the siphon and vent port penetrations seal welded closed.

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

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

Placement of Cask on Transport Trailer Skid: The transfer cask is then lifted onto the cask support skid. The plant's crane is used to downend the cask from a vertical to a horizontal position. The cask is then secured to the skid and readied for the subsequent transport operations.

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

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

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

Storage: After the DSC is inside the HSM, the hydraulic ram is disengaged from the DSC and withdrawn through the cask. The transfer trailer is pulled away, the HSM shielded access door installed and the DSC axial retainer inserted. The DSC is now in safe storage within the HSM.

Retrieval: For retrieval, the transfer cask is positioned and the DSC is transferred from the HSM to the cask. The hydraulic ram is used to pull the DSC into the cask. All transfer operations are performed in the same manner as previously described. Once back in the

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

Dry Shielded Canister

Internal Basket Assembly:

Guide Sleeves	(24 for 24P, *24PHB* & 24PT2)
Oversleeves	(24P, *24PHB* & 24PT2)
Fixed Neutron Absorbers	(88 for 52B; 72 for 24PT2)
Spacer Disks	(8 for 24P *& 24PHB*; 9 for 52B; 26 for 24PT2)
Support Rods	(4 for 24P *& 24PHB*; 6 for 52B; 4 for 24PT2)
Spacer Sleeves	(52B & 24PT2)

Cylindrical Shell

Shield Plugs (top and bottom)

Inner and Outer Cover Plates (top and bottom)

Siphon and Vent Port

Grapple Ring

Horizontal Storage Module

Reinforced Concrete Walls, Roof, and Floor

DSC Support Structure

DSC Axial Retainer

Cask Docking Collar and Cask Restraint Eyes

Heat Shield

Shielded Access Door

Ventilation Air Openings (four inlets, four outlets)

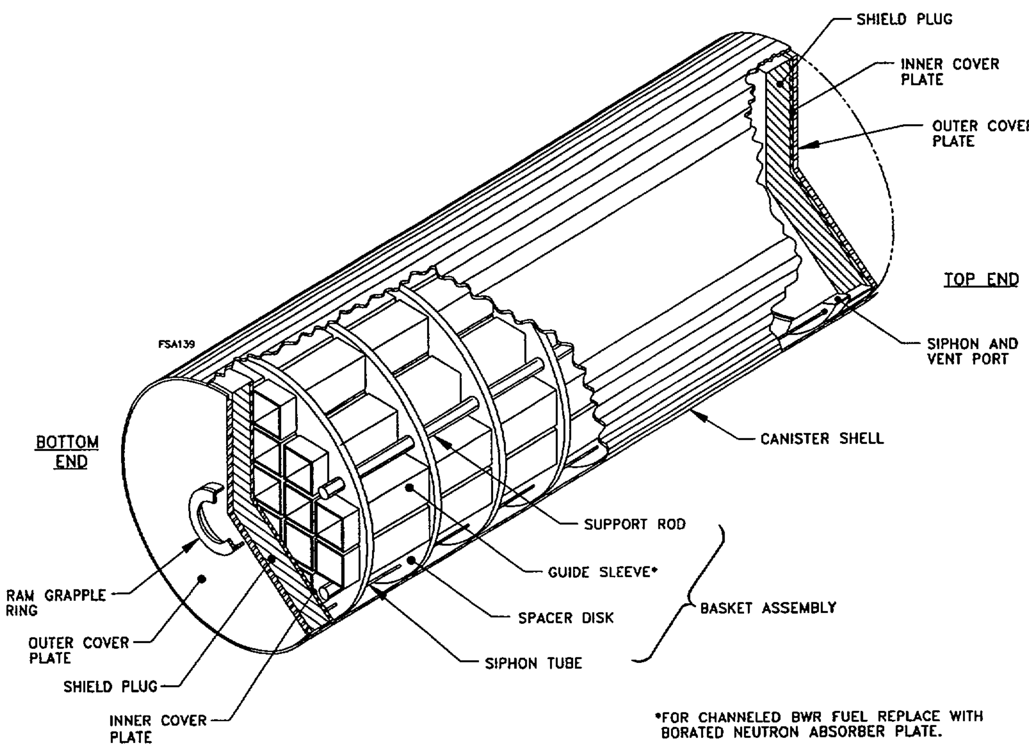


Figure 1.3-1
NUHOMS® Dry Shielded Canister Assembly Components

Note: Appendix N.1.5 shows the outer top cover plate and the test port plug details for the 24PHB DSC

3. PRINCIPAL DESIGN CRITERIA

3.1 Purpose of Installation

The NUHOMS® system provides an ISFSI for horizontal, dry storage (in a helium atmosphere) of SFAs in a high integrity stainless steel DSC which is placed inside a massive reinforced concrete HSM. The function of the DSCs and HSMs is to provide for the safe, controlled, long-term storage of SFAs.

The standardized NUHOMS® system can be utilized to store a wide range of the various light water reactor fuel assembly types which presently reside in spent fuel pools. This SAR addresses the most common types of both PWR and BWR spent fuel. The following subsection provides a description of the spent fuel assemblies which are acceptable for storage using the standardized NUHOMS® system.

The storage capacity of a single standardized NUHOMS® DSC and HSM is 24 PWR fuel assemblies or 52 BWR fuel assemblies. Multiple HSMs can be grouped together to form arrays which provide the needed storage capacity consistent with available site space and reactor fuel discharge rates.

3.1.1 Material to be Stored

The inventory of PWR fuel types which currently resides in spent fuel pools in the U.S. is shown in Figure 3.1-1. B&W 15x15 fuel is selected as the enveloping fuel design for a wide range of PWR fuel types as it is the most reactive and has the most limiting physical characteristics. Table 3.1-1 lists the principal design parameters for the B&W 15x15 fuel selected as the design basis for the standardized NUHOMS®-24P system documented in this SAR. Table 3.1-1a lists the PWR fuel assembly designs (with or without BPRAs) that have currently been demonstrated to be suitable for storage in the standardized NUHOMS®-24P system provided they meet the requirements of 72-1004 CoC Technical Specifications. Similarly, the inventory of BWR fuel types residing in spent fuel pools in the U.S. is shown in Figure 3.1-2. GE 7x7 fuel is selected as the enveloping fuel design for a wide range of BWR fuel types. Table 3.1-2 lists the principal design parameters for the GE 7x7 fuel selected as the design basis for the standardized NUHOMS®-52B system documented in this SAR. Table 3.1-2a lists the BWR fuel designs which have currently been demonstrated to be suitable for storage in the standardized NUHOMS®-52B system provided they meet the requirements of 72-1004 CoC Technical Specifications. *Appendix N lists the principal design parameters for the NUHOMS®-24PHB system.*

The following acceptance criteria is established for BWR and PWR fuels other than the SAR design basis fuels.

concrete walls and slabs which act as biological radiation shields. The storage operation of the HSMs and DSCs is totally passive. No active systems are required.

3.1.2.1 Handling and Transfer Equipment

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

On-Site Transfer Cask: The on-site transfer cask used for the NUHOMS® system has certain basic features. The DSC is transferred from the plant's fuel pool to the HSM inside the transfer cask. The cask provides neutron and gamma shielding adequate for biological protection at the outer surface of the cask. The cask is capable of rotation, from the vertical to the horizontal position on the support skid. The cask has a top cover plate which is fitted with a lifting eye allowing removal when the cask is oriented horizontally. The cask is capable of rejecting the design basis decay heat load to the atmosphere assuming the most severe ambient conditions postulated to occur during normal, off-normal and accident conditions. For the NUHOMS®-24P, *24PHB* or -24PT2 DSC, the standardized transfer cask has a cylindrical cavity of 1.73m (68 inches) diameter and 4.75m (186.75 inches) in length and a maximum dry payload capacity of 36,000 Kg (80,000 pounds). For the NUHOMS®-52B, the standardized transfer cask is fitted with an extension collar to accommodate the longer BWR DSC and fuel. Alternatively, the OS197 and OS197H transfer casks with a full length cavity of 5.0m (196.75 inches) may be used for the NUHOMS®-24P, *24PHB,* (with cask spacer), NUHOMS®-52B DSCs, NUHOMS®-24PT2 (with cask spacer) DSCs. The OS197 and the OS197H casks can carry a maximum dry payload of 40,800 kg (90,000 lb) and 52,600 kg (116,000 lb), respectively. The cask and the associated lifting yoke are designed and operated such that the consequences of a postulated drop satisfy the current 10CFR50 licensing bases for the vast majority of plants.

The NUHOMS® transfer cask is designed to meet the requirements of 10CFR72 (3.6) for normal, off-normal and accident conditions. The NUHOMS® transfer cask is designed for the following conditions:

A.	Seismic	Reg. Guide 1.60 (3.11) and 1.61 (3.12)
B.	Operational Handling Loads	ANSI/ANS-57.9-1984 (3.36)
C.	Accidental Drop Loads	ANSI/ANS-57.9-1984

in Section 6.17.3.1 of ANSI 57.9-1984 are used for combining normal operating, off-normal, and accident loads for the HSM. All seven load combinations specified are considered and the governing combinations are selected for detailed design and analysis. The resulting HSM load combinations and the appropriate load factors are presented in Table 3.2-5. The effects of duty cycle on the HSM are considered and found to have negligible effect on the design. The corresponding structural design criteria for the DSC support structure are summarized in Table 3.2-8 and *Table 3.2-10.* The HSM load combination results with 24P and 52B DSCs are presented in Section 8.2.10. The HSM load combination results with the 24PT2 DSC are included in Appendix L. *The HSM load combination results with the 24PHB DSC are included in Appendix N.*

3.2.5.2 Dry Shielded Canister

With the exceptions noted in Table 4.8-1 and Table 4.8-2, the DSC is designed by analysis to meet the stress intensity allowables of the ASME Boiler and Pressure Vessel Code (1983 Edition with Winter 1985 Addenda) (3.14) Section III, Division 1, Subsection NB for Class 1 components (for the DSC pressure boundary components), and Subsection NF for Class 1 plate and shell supports (for the internal basket assembly components). The DSC is conservatively designed by utilizing linear elastic or non-linear elastic-plastic analysis methods.

The load combinations considered for the DSC normal, off-normal and postulated accident loadings are shown in *Table 3.2-6 for the 24P, 24PT2 and 52B DSCs and in Appendix N for the 24PHB DSC.* ASME Code Service Levels A and B allowables are conservatively used for normal and off-normal operating conditions. Service Levels C and D allowables are used for accident conditions such as a postulated cask drop accident. 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 effects of fatigue on the DSC due to thermal and pressure cycling are addressed in Section 8.2-10.

The DSC pressure boundary components which include the DSC shell and cover plates are designed using the stress criteria of the ASME B&PV Code Subsection NB. The shell longitudinal and circumferential welds are full penetration welds fabricated and inspected in accordance with Subsection NB. The cover plates to shell welds are partial penetration welds, designed using a "joint efficiency" factor of 0.6 on the ASME Code Subsection NB criteria. Table 3.2-9a summarizes the stress design criteria for the pressure boundary components of the DSC. In addition to stress criteria, buckling of the DSC shell is evaluated using the ASME Code Subsection NB (for Service Levels A,B, C) and ASME Code Appendix F (for Service Level D) stability criteria.

The 24P DSC basket components include the spacer discs, support rods, guide sleeves oversleeves, and their associated welds. The 52B DSC basket components include the spacer discs, support rod and spacer sleeve assemblies, neutron absorber plates (poison plates), poison plate support bars and connecting hardware. Table 3.2-9b summarizes the addressed in Appendix L. *The effects of handling the NUHOMS®-24PHB DSC in the Standard, OS-197 or OS-197H transfer cask are addressed in Appendix N.*

Table 3.2-1
Summary of NUHOMS® Component Design Loadings
(continued)

Component	Design Load Type	SAR Section Reference	Design Parameters	Applicable Codes
	Accident Condition Temperatures	8.2.7.2	Same as off-normal conditions with HSM vents blocked for 40 hours	ANSI 57.9-1984
	Normal Handling Loads	8.1.1.1	For concrete component evaluation 80,000 lb.(DSC HSM insertion) 60,000 lb (DSC HSM extraction)	ANSI 57.9-1984
	Off-normal Handling Loads	8.1.1.4	For concrete component evaluation 80,000 lb (DSC HSM insertion) 80,000 lb (DSC HSM extraction)	ANSI 57.9-1984
	Live Load	8.1.1.5	Design load: 200 psf (includes snow and ice loads)	ANSI 57.9-1984
	Fire and Explosions	3.3.6	Enveloped by other design basis events	10CFR72.122(c)
Dry Shielded Canister (1):	---	---	---	ASME Code, Section III, Subsection NB, Class 1 Component
	Flood	3.2.2	Maximum water height: 50 ft.	10CFR72.122(b)
	Seismic	3.2.2	Horizontal ground acc.: 0.25g Vertical ground acc.: 0.17g	NRC Reg. Guides 1.60 & 1.61
	Dead Load	8.1.1.2	Weight of loaded 24P & 52B DSC: 80,000 lb. enveloping. Weight of loaded 24PT2 DSC: 85,000 lb. enveloping.	ANSI 57.9-1984
	Normal and Off-Normal Pressure	8.1.1.2	Enveloping internal pressure of ≤10.0 psig	10CFR72.122(h)
	Test Pressure	8.1.1.2	Enveloping internal pressure of 12 psig applied w/o DSC outer top cover plate	10CFR72.122(h)

(1) See Appendix N for the NUHOMS®-24PHB DSC design loadings.

Table 3.2-6
DSC Load Combinations and Service Levels (10)

Load Case		Normal Operating Conditions								Off-Normal Conditions				Accident Conditions								
		1	2	3	4	5	6	7	8	1	2	3	4	1	2	3	4	5	6	7	8	9
Dead Weight Load	Vertical/Horizontal DSC, Empty	X	X																			
	Vertical, DSC w/Fuel + Water			X										X								
	Vertical, DSC w/Fuel				X	X(5)							X		X(9)							
	Horizontal, DSC w/Fuel		X				X	X	X	X	X	X			X(9)	X	X	X	X	X	X	X
Thermal Load	Inside HSM: 0° to 100°F							X	X									X	X	X	X	X
	Inside Cask: 0° to 100°F (1)		X	X	X	X	X						X	X	X							
	Inside HSM: -40° to 125°F										X	X				X						
	Inside Cask -40°to 125°F								X													
	Inside HSM: Blocked Vents; 125°F																X					
External Pressure			X	X	X								X	X						X		
Internal Pressure Load	Hydrostatic Pressure		X(6)	X										X								
	Normal Pressure (4)					X	X	X	X									X				
	Off-Normal Pressure (4)									X	X	X	X(7)									
	Accident Pressure (3)														X	X	X		X		X	X
Test Pressure					X																	
Lifting/ Handling Loads	Lifting Loads	X																				
	Normal DSC Transfer						X		X													
	Off-Normal DSC Transfer									X		X										
	Accident DSC Transfer																				X	X
Cask Drop Load (end, side, or corner drop)															X							
Seismic Load														X(8)				X	X			
Flooding Load																				X		
ASME Code Service Level		A	A	A	A	A	A	A	A	B	B	B	B	C	D	C	D	C	C	C	C	D
Analysis Load Cases in Chapter 8, Table 8.2-24		NO-3 NO-4	FL-1 FL-2 FL-3	FL-4 FL-5 FL-6	DD-1 DD-2 DD-3 DD-4 DD-5	TL-1 TL-2 TL-3 TL-4	TR-1 to TR-8 LD-1 LD-2 LD-3	HSM-2	UL-1 UL-2 UL-3	LD-4 LD-5 LD-6 LD-7	HSM-1 HSM-3	UL-4 UL-5 UL-6	RF-1	FL-7	TR-9 TR-10 TR-11	HSM-4	HSM-5 HSM-6	HSM-7 HSM-8	HSM-8a	HSM-9 HSM-10	UL-7	UL-8

NOTES:

1. At temperatures over 100°F, a sunshade is required over the Transfer Cask. Temperatures for the 125°F with shade are enveloped by the 100°F without sun shade case.

2. The stress limits of ASME Code NB-3226 apply

3. Accident pressure for Service Level C condition is applied to inner top and bottom cover plates. Accident pressures on the inner and outer top and bottom cover plates are evaluated for Service Level D allowables.

4. 10 psig is conservatively used for Normal and Off-normal pressure

5. DSC inside cask is laydown to horizontal for load cases TL-3, TL-4

6. Internal hydrostatic pressure. Applies to FL-3, FL-4

7. Reflood pressure is 20 psig

8. Fuel deck seismic loads are assumed enveloped by handling loads

9. Both horizontal and vertical drop cases are considered

10. *See Appendix N for the NUHOMS®-24PHB DSC load combinations and service levels used*

minimizes the likelihood of contaminating the DSC exterior surface. The combination of the above operations assures that the DSC surface loose contamination levels are within those required for shipping cask externals (see Section 3.3.7.1). Compliance with these contamination limits is assured by taking surface swipes of the upper end of the DSC while resting in the cask in the decon area.

Once inside the DSC, the SFAs are confined by the DSC shell and by multiple barriers at each end of the DSC. As listed in Table 3.3-2, the fuel cladding is the first barrier for confinement of radioactive materials. The fuel cladding is protected by maintaining the cladding temperatures during storage below those levels which may cause degradation of the cladding. In addition, the SFAs are stored in an inert atmosphere to prevent degradation of the fuel, specifically cladding rupture due to oxidation and its resulting volumetric expansion of the fuel. Thus, a helium atmosphere for the DSC is incorporated in the design to protect the fuel cladding integrity by inhibiting the ingress of oxygen into the DSC cavity.

Helium is known to leak through valves, mechanical seals, and escape through very small passages because of its small atomic diameter and because it is an inert element and exists in a monatomic species. Negligible leakage rates can be achieved with careful design of vessel closures. Helium will not, to any practical extent, diffuse through stainless steel (3.33). For this reason the DSC has been designed as a redundant weld-sealed containment pressure vessel with no mechanical or electrical penetrations. *The NUHOMS®-24PHB DSC is designed and tested to meet the leak tight criteria discussed in Appendix N.*

The DSC itself has a series of barriers to ensure the confinement of radioactive materials. The DSC cylindrical shell is fabricated from rolled ASME stainless steel plate which is joined with full penetration 100% radiographed welds. All top and bottom end closure welds are multiple-layer welds. This effectively eliminates a pinhole leak which might occur in a single layer weld, since the chance of pinholes being in alignment on successive weld layers is not credible. Furthermore, the DSC cover plates are sealed by separate, redundant closure welds. All the DSC pressure boundary welds are inspected according to the appropriate articles of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB except as noted in the Table 4.8-1. This criteria insures that the 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 DSC design. The DSC shell and welded cover plates provide total confinement of radioactive materials. Once the DSC is sealed, there are no credible events which could fail the DSC cylindrical shell or the double closure plates which form the DSC containment pressure boundary. This is discussed further in Chapter 8.

3.3.4.2.5 Safety Criteria Compliance (NUHOMS®-52B)

The calculated worst-case k_{eff} value for a fully loaded NUHOMS®-52B DSC flooded with pure unborated water is 0.919. This calculated maximum k_{eff} includes consideration of all calculational, geometrical, and material uncertainties and biases at a 95/95 tolerance level as required by ANSI/ANS 57.2-1983 to demonstrate criticality safety.

Additionally, there are no credible off-normal conditions which could increase reactivity beyond the normal conditions considered above.

The analyses presented in this SAR section demonstrate that the ANSI/ANS 57.2-1983 criteria limiting k_{eff} to 0.95 is satisfied under all postulated conditions for the NUHOMS®-52B.

3.3.4.3 NUHOMS®-24PT2 DSC Criticality Safety

The NUHOMS®-24PT2 DSC criticality analyses are described in Appendix L.

3.3.4.4 *NUHOMS®-24PHB DSC Criticality Safety*

The NUHOMS®-24PHB DSC criticality analyses are described in Appendix N.

3.3.5 Radiological Protection

The NUHOMS® ISFSI is designed to maintain on-site and off-site doses ALARA during transfer operations and long-term storage conditions. ISFSI operating procedures, shielding design, and access controls provide the necessary radiological protection to assure radiological exposures to station personnel and the public are ALARA. Further details on on-site and off-site doses resulting from NUHOMS® ISFSI operations and the ISFSI ALARA evaluation are provided in Chapter 7.

3.3.5.1 Access Control

The NUHOMS® ISFSI is located within the owner controlled area of the plant. A separate protected area consisting of a double fenced, double gated, lighted area is generally installed around the ISFSI. Access is controlled by locked gates, and guards are stationed when the gates are open. The licensee's Security Plan should describe the remote sensing devices which are employed to detect unauthorized access to the facility.

In addition to the controlled access, a method of providing a security tamper seal may be implemented after insertion of a loaded DSC. The form to use could be, but is not limited to, one of the following:

Tack welding an HSM access door

Fully welding an HSM access door

Tack welding 2 or more closure bolts on the HSM access door

Tamper seals

Existing HSM closure bolt torquing

The HSM access door weighs approximately three tons and requires heavy equipment for removal. This ensures that there is ample time to respond to an unauthorized entry into the ISFSI before access can be gained to any radiological material.

3.3.5.2 Shielding

For the NUHOMS® system, shielding is provided by the HSM, transfer cask, and shield plugs of the DSC. The NUHOMS® standardized HSM is designed to minimize the surface dose to limit occupational exposure and the dose at the ISFSI fence. Experience has confirmed that the dose rates for the HSM are extremely low. A shield wall may be removed for a period of time as part of facility installation or expansion. However, if applicable, compensatory measures shall be taken for radiation shielding. The NUHOMS® transfer cask and the DSC top shield plug are designed to limit the surface dose rates (gamma and neutron) ALARA. Temporary neutron shielding may be placed on the DSC shield plug and top cover plate during closure operations. Similarly, additional temporary shielding may be used to further reduce surface doses. Radiation zone maps of the HSM, cask, DSC surfaces and the area around these components are provided in Chapter 7. Appendix L provides the results with the NUHOMS®-24PT2 DSC. *Appendix N provides the results with the NUHOMS®-24PHB DSC.*

3.3.5.3 Radiological Alarm Systems

There are no radiological alarms required for the NUHOMS® ISFSI.

3.3.6 Fire and Explosion Protection

The NUHOMS® HSM and DSC contain no flammable material and the concrete and steel used for their fabrication can withstand any credible fire hazard. There is no fixed fire suppression system within the boundaries of the ISFSI. The facility should be located such that the plant fire brigade can respond to any fire emergency using portable fire suppression equipment.

ISFSI initiated explosions are not considered credible since no explosive materials are present in the fission product or cover gases. Externally initiated explosions are considered to be bounded by the design basis tornado generated missile load analysis presented in Section 8.2.2. Licensees are required by 10CFR72 Subpart K to confirm that no conditions exist near the ISFSI that would result in pressures due to off-site explosions which would exceed those postulated herein for tornado missile or wind effects. An

All spent fuel handling outside the plant's fuel pool is performed with the fuel assemblies contained in the DSC. Subcriticality during all phases of handling and storage is discussed in Section 3.3.4. The criterion for a safe configuration is an effective mean plus two-sigma neutron multiplication factor (k_{eff}) of 0.95. Section 3.3 calculations show that the expected k_{eff} value is below this limit.

Lift height restrictions are imposed on the TC and DSC with regard to their location and load temperatures. These restrictions are provided in Section 10.3.13.

3.3.7.1.1 Cladding Temperature Limits

Maximum allowable cladding temperature limits are determined for both BWR and PWR design basis fuel according to the methodology presented in Reference 3.21. The maximum allowable average cladding temperature for long term storage is based on the end of life hoop stress in the cladding and the cladding temperature at the beginning of dry storage. The method is estimated to calculate a storage temperature limit that will result in a probability of cladding breach of less than 0.5% in the peak rod during storage. Using this methodology produces cladding temperature limit of 381°C for design basis PWR fuel and 394°C for the design basis BWR fuel cooled for five years or more. Appendix L addresses the cladding temperature limits for the PWR fuel in the NUHOMS®-24PT2 DSC. *Appendix N addresses the cladding temperature limits for the PWR fuel in the NUHOMS®-24PHB DSC.* Since the damage mechanism in this methodology is thermal creep, the temperature limits are based on an average long term ambient temperature during storage of 70°F.

381°C (718°F) and 394°C (741°F) are the cladding temperature limits calculated for design basis 5-year cooled PWR and BWR fuel, respectively. Three steps were taken to extend the same methodology to the range of cooling times in the Fuel Qualification Table shown in 72-1004 CoC technical specifications. First, the same thermal computer models used to perform the design basis cladding temperature calculation were run parametrically to determine cladding temperature vs. heat input for the PWR and BWR baskets. Second, the methodology of Reference 3.21 was used to develop a relationship between the maximum cladding temperature limit vs. cooling times beyond 5 years. This relationship is shown as a function of fuel burnup in Figure 3.3-17 for PWR fuel and in Figure 3.3-18 for BWR fuel. Third, these two functions were combined to obtain maximum heat input vs. cooling time. In this way, each cell of the Fuel Qualification Table has its own unique cladding temperature limit based on the same methodology as was used for the design basis fuel assemblies.

Higher cladding temperatures may be sustained for brief periods without affecting cladding integrity, however. During short term conditions such as DSC drying, transfer of the DSC to and from the HSM, and off-normal and accident temperature excursions, the maximum fuel cladding temperature is limited to 570°C (1,058°F) or less. This value is based on the results of experiments which have shown that Zircaloy clad rods subjected

Table 3.4-1
NUHOMS® Major Components and Safety Classification

Component	10CFR72 Classification
Dry Storage Canister (DSC)	Important to Safety(1)
Guide Sleeves (24P *and 24PHB*)	
Oversleeves (24P *and 24PHB*)	
Oversleeves (24PT2)	
Spacer Disks	
Support Rods	
Spacer Sleeves (52B & 24PT2 only)	
Support Bars (52B only)	
Neutron Absorbing Plates (52B only)	
Shield Plugs (3)	
DSC Shell	
Cover Plates	
Grapple Ring and Grapple Support	
Siphon and Vent Block	
Siphon and Vent Port Cover Plates	
DSC Support Ring	
Weld Filler Metal	
Horizontal Storage Module (HSM)	Important to Safety(1)
Reinforced Concrete	
DSC Support Structure	
ISFSI Basemat and Approach Slabs	Not Important to Safety
Transfer Equipment	
On-site Transfer Cask	Important to Safety(1)
Cask Lifting Yoke	Safety Related(2)
Transport Trailer/Skid	Not Important to Safety
Ram Assembly	Not Important to Safety
Dry Film Lubricant	Not Important to Safety
Auxiliary Equipment	Not Important to Safety
Vacuum Drying System	
Automatic Welding System	

(1) Structures, systems and components "important to safety" are defined in 10CFR72.3 as those features of the ISFSI whose function is (1) to maintain the conditions required to store spent fuel safely, (2) to prevent damage to the spent fuel container during handling and storage, or (3) to provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

(2) Yoke and rigid or sling lifting members are classified as "Safety Related" in accordance with 10CFR50.

(3) For 24P Long Cavity, *24PHBL* and 24PT2L DSC.

4.2 Storage Structures

4.2.1 Structural Specifications

The design bases for the NUHOMS® ISFSI are described in Chapter 3. Fabrication and construction specifications will be utilized in accordance with 10CFR72 (4.1) and industry codes and standards. The codes and standards used for fabrication and construction the NUHOMS® components, equipment, and structures are identified throughout the SAR. They are summarized as follows:

Component, Equipment, Structure	Code of Construction
DSC	ASME Code, Section III, Division 1, 1983 Edition with Winter 1985 Addenda (4.5) Subsection NB, Subsection NF, and Appendix F with exceptions as noted in Section 4.8 of this SAR.
Transfer Cask	ASME Code, Section III, Division 1, 1983 Edition with Winter 1985 Addenda (4.5) Subsection NC as applicable for non-pressure retaining vessels, with exceptions as noted in Section 4.9 of this SAR.
HSM	ACI-318-83 Code (4.10)
DSC Supports	AISC Specification, 1990, Ninth Edition (4.11)
Transfer Equipment	AISC, ANSI, AWS and/or other applicable Standards

The ASME Code boundaries for the *24P, 24PT2, 24PHB and 52B* DSC*s* and transfer cask are identified on the corresponding Appendix E drawings.

4.2.2 Installation Layout

The specific layout of the ISFSI will be developed by the licensee in accordance with the requirements of 10CFR72. Layouts for typical NUHOMS® ISFSIs are shown in Figures 1.3-11 through 1.3-13. The functional features of the NUHOMS® storage structures are shown on the Appendix E drawings. Radioactive particulate matter and gaseous fission products are confined within the DSC as discussed in Sections 1.2 and 1.3.

4.2.3 Individual Unit Description

4.2.3.1 Dry Shielded Canister

The following description is applicable to the 24P, *24PHB* and 52B DSC designs. *Any differences in the 24PHB DSC configuration relative to the 24P DSC are described in Appendix N.* The 24PT2 DSC design description is included in Appendix L. The DSC is a high integrity stainless steel welded pressure vessel that provides confinement of radioactive materials, encapsulates the fuel in a helium atmosphere, and, when placed in the transfer cask, provides biological shielding during DSC closure and transfer operations. With the exceptions noted in Section 4.8, the DSC shell assembly and associated subcomponents conform to the requirements of the ASME B&PV Code Section III, Division 1, Subsection NB while the DSC basket assembly conforms to Subsection NF. The NUHOMS® DSC design is illustrated in Figures 1.3-1 through 1.3-3. Drawings for the standardized DSC *and 24PHB DSC* are contained in Appendix E *and Appendix N respectively.*

The DSC cylindrical shell is fabricated from rolled and butt-welded stainless steel plate material as shown in Figure 4.2-1. Stainless steel cover plates and thick carbon steel or lead encased in steel shielding material form the DSC top and bottom end assemblies. The cover plates are double seal welded to the DSC shell to form the containment pressure boundary.

The DSC shell, and top and bottom end assemblies enclose a non-pressure retaining basket assembly which serves as the structural support for the SFAs as shown in Figure 4.2-2 and Figure 4.2-3. The primary components of the basket assembly are the spacer discs, which maintain cross sectional spacing of (and provide lateral support to) the fuel assemblies within the DSC, and the support rods, which hold the spacer discs in place and maintain longitudinal separation of the spacer discs during a postulated cask drop accident.

The PWR NUHOMS-24P fuel basket assembly consists of 24 stainless steel guide sleeves, eight carbon steel spacer discs and four Type XM-19 stainless steel support rods. The inner guidesleeves in the assembly are equipped with stainless steel oversleeves placed at both ends of the basket assembly between the two top and bottom spacer discs. No connection exists between the spacer discs and the guidesleeves. Guidesleeve stops fabricated from stainless steel plate strips and plug welded to the sides of the guidesleeves prevent removal of the guidesleeves from the basket if a fuel assembly

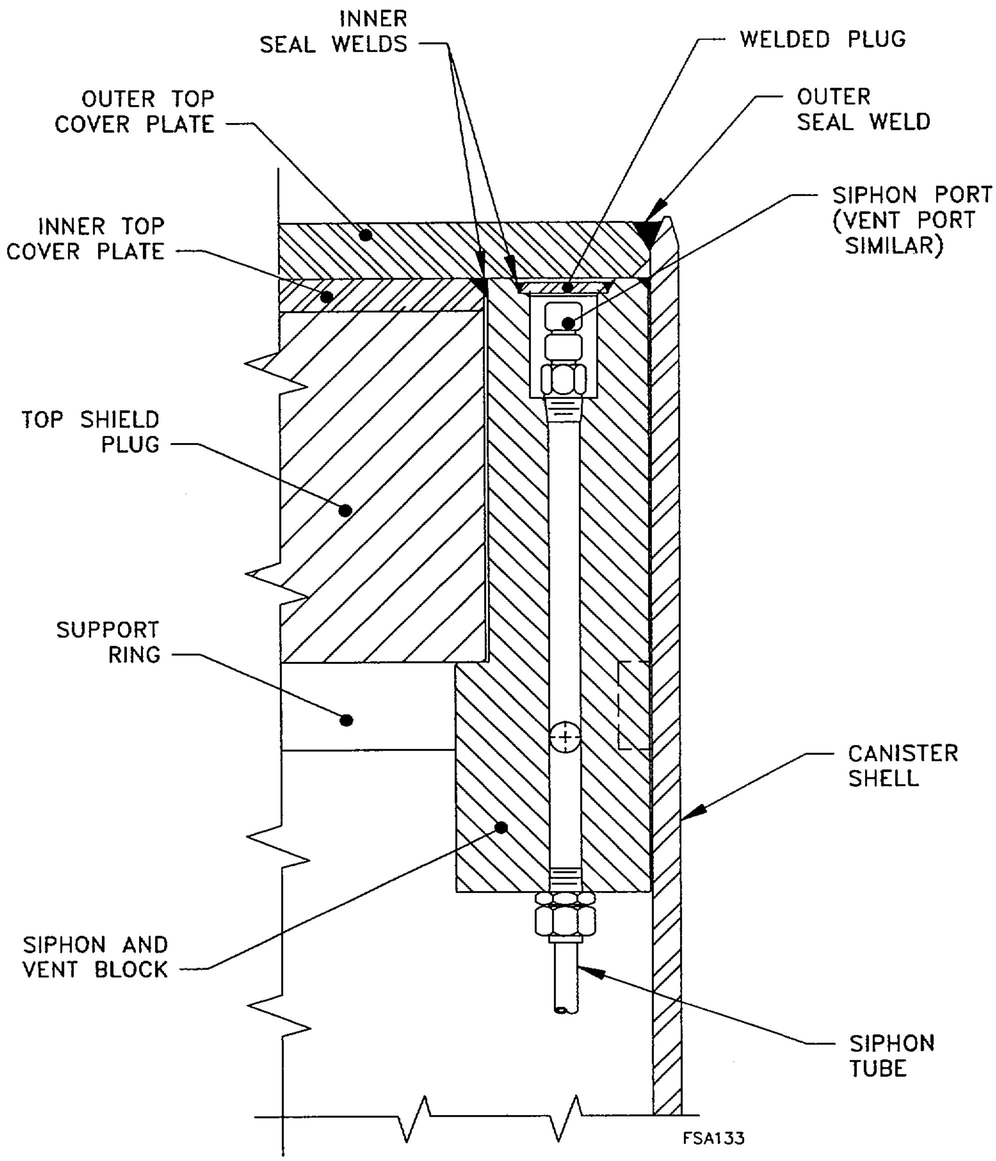


Figure 4.2-5
DSC Siphon and Vent Port Closure Welds

Note: For 24PHB DSC Outer Top Cover Plate and Test Port Plug Weld Closure Details, See Appendix N

5. OPERATION SYSTEMS

This Chapter presents the operating procedures for the standardized NUHOMS® system described in previous chapters and shown on the drawings in Appendix E for the 24P and 52B systems. The 24PT2 system operating procedures are described in Appendix L. *The 24PHB system operating procedures are described in Appendix N.* The procedures include preparation of the DSC and fuel loading, closure of the DSC, transport to the ISFSI, DSC transfer into the HSM, monitoring operations, and DSC retrieval from the HSM. The standardized NUHOMS® transfer equipment, and the existing plant systems and equipment are used to accomplish these operations. Procedures are delineated here to describe how these operations are to be performed and are not intended to be limiting. Standard fuel and cask handling operations performed under the plant's 10CFR50 operating license are described in less detail. Existing operational procedures may be revised by the licensee and new ones may be developed according to the requirements of the plant, provided that the limiting conditions of operation specified in Technical Specifications, Functional and Operating Limits of the NUHOMS® COC (5.6) are not exceeded.

5.1 Operation Description

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

The generic operating procedures presented herein also do not address the use of auxiliary equipment which is optional or represents a level of detail which a licensee may choose to implement based on licensee preference. Examples of such auxiliary items are the Neutron Shield Overflow Tank (used with OS 197 Cask only), TC/DSC Annulus Pressurization Tank, and the Shield Plug Restraints.

7.2 Radiation Sources

7.2.1 Characterization of Sources

This section describes the design basis radiation source strengths and source geometries used for the standardized NUHOMS® 24P and 52B system shielding design calculations. Appendix L describes the NUHOMS®-24PT2 system. *Appendix N describes the NUHOMS®-24PHB system.*

The neutron and gamma radiation sources include the design basis PWR and BWR spent fuel, activated portions of the fuel assembly, and secondary gammas. All sources, except secondary gammas, are considered physically bound in the source region. Secondary gammas are produced by neutrons passing through shielding regions.

The design basis PWR spent fuel for the NUHOMS®-24P system has been subjected to an average fuel burnup of 40,000 MWD/MTU. The maximum initial enrichment is 4.0 weight percent U-235 and a post-irradiation cooling time equivalent to five years is assumed. Similarly, the design basis BWR spent fuel for the NUHOMS®-52B system has been subjected to an average fuel burnup of 35,000 MWD/MTU with a maximum initial enrichment of 4.0 weight percent U-235 and a cooling time of five years. Spent fuel assemblies which meet these criteria are bounded by the source strengths used in this analysis.

Neutron sources are based on spontaneous fission contributions from six nuclides (predominantly Cm-242, Cm-244, and Cm-246 isotopes), and (α,n) reactions due almost entirely to eight alpha emitters, (predominantly Pu-238, Cm-242, and Cm-244). The fission spectrum used in shielding calculations is a weighted combination of the principal contributors. The total neutron source strength for PWR fuel is 2.23E8 neutrons per second per assembly. Similarly, the total neutron source strength for BWR fuel is 1.01E8 neutrons per second per fuel assembly. The neutron energy spectrum and flux-to-dose conversion factors are presented in Table 7.2-1 and Table 7.2-2.

Gamma radiation sources include 70 principal fission product nuclides within the spent fuel, and several activation products and actinide elements present in the spent fuel and fuel assemblies. The gamma energy spectrum includes contributions from each source isotope as determined by ORIGEN calculations for the design basis spent fuel. The total gamma source strength for PWR fuel is 5.81E15 MeV/s/MTHM. Similarly the total gamma source strength for BWR fuel is 4.86E15 MeV/s/MTHM. The gamma energy spectrum and flux-to-dose conversion factors are presented in Table 7.2-3 and Table 7.2-

the axial direction to that of the DSC shield plugs. Figure 1.3-6 shows the physical arrangement of the transfer cask top and bottom end assembly.

Additional portable shielding during DSC handling, transport and transfer operations may be utilized by the licensee, if desired. Section 7.4 conservatively provides an assessment of design basis on-site doses without the use of portable shielding.

7.3.2.2 Shielding Analysis

This section describes the radiation shielding analytical methods and assumptions used in calculating NUHOMS® 24P and 52B system dose rates during the handling and storage operations. Appendix L describes the same for the NUHOMS®-24PT2 system. *Appendix N describes the same for the NUHOMS®-24PHB system.* The dose rates of interest are calculated at the locations listed in Table 7.3-2 for 5 year cooled design basis PWR fuel. Table 7.3-3 shows the dose rates for 10 year cooled PWR fuel which are included for information only. *Figure 7.3-3* shows these locations on the HSM, DSC and transfer cask. The computer codes used for analysis are described below, each with a brief description of the input parameters generic to its use. Descriptions of the individual analytical models used in the analysis are also provided. Consistent with the relative design basis source strengths, the shielding analysis results for the NUHOMS®-24P envelop those of the NUHOMS®-52B systems, except on the bottom of the DSC. The bottom of the NUHOMS®-52B canister has 0.5" less steel shielding. The effect of this difference on the dose rates at the bottom surface of the transfer cask and the HSM with and without the door are provided in Tables 7.3-4 and 7.3-5 for 5 and 10 year cooled BWR fuel, respectively.

A. Computer Codes ANISN (7.1), a one-dimensional, discrete ordinates transport computer code, is used to obtain neutron and gamma dose rates at the outer HSM walls, and at the outside surface of the loaded transfer cask in the radial direction. ANISN is also used to obtain the axial neutron dose rates at the shield plugs of the DSC, the transfer cask, and outside the HSM access door. The CASK cross section library, which contains 22 neutron energy groups and 18 gamma energy groups, is applied in an S_8P_3 approximation for cylindrical or an $S_{16}P_3$ approximation for slab geometry, respectively (7.7). Calculated radiation fluxes are multiplied by flux-to-dose conversion factors (Table 7.2-1, Table 7.2-2, Table 7.2-3,and Table 7.2-4) to obtain final dose rates. The ANISN calculations use the coupled neutron and gamma libraries. Therefore, dose rates from both primary and secondary gammas are calculated in each run.

QAD-CGGP (7.2), a three-dimensional point-kernel code, is used for the axial gamma shielding analysis of the HSM access door, the DSC and cask end assemblies, the DSC-cask annular gap, and the HSM air vent penetrations. Mass attenuation and buildup

7.4 Estimated On-Site Collective Dose Assessment

7.4.1 Operational Dose Assessment

This SAR section establishes the anticipated cumulative dose exposure to site personnel during the fuel handling and transfer activities associated with utilizing one NUHOMS® HSM for storage of one DSC. Chapter 5 describes in detail the NUHOMS® operational procedures, a number of which involve potential radiation exposure to personnel.

A summary of the operational procedures which result in radiation exposure to personnel is given in Table 7.4-1. The cumulative dose can be calculated by estimating the number of individuals performing each task and the amount of time associated with the operation. The resulting man-hour figures can then be multiplied by appropriate dose rates near the transfer cask surface, the exposed DSC top surface, or the HSM front wall. Dose rates can be obtained from the Section 7.3 results of dose rate versus distance from the cask side, DSC top end (with and without the top cover plate and cask lid in place) and HSM front wall for the 24P and 52B DSCs. Similar results with the NUHOMS®-24PT2 DSCs are provided in Appendix L. *Similar results with the NUHOMS®-24PHB DSCs are provided in Appendix N.*

Every operational aspect of the NUHOMS® system, from canister loading through drying, sealing, transport, and transfer is designed to assure that exposure to occupational personnel is as low as reasonably achievable (ALARA). In addition, many engineered design features are incorporated into the NUHOMS® system which minimize occupational exposure to plant personnel during placement of fuel in dry storage as well as off-site dose to the nearest neighbor during long-term storage. The resulting dose at the ISFSI site boundary is to be within the limits specified by 10CFR72 and 40CFR190.

Based on the experience for an operating NUHOMS® system, the occupational dose for placing a canister of spent fuel into dry storage for the operational steps listed in Table 7.4-1 is less than 1.2(1) man-rem. With the use of effective procedures and experienced ISFSI personnel, the total accumulated dose can be reduced further below one man-rem per canister.

---

(1) The expected small additional occupational dose when loading PWR fuel with BPRAs into a NUHOMS® Long Cavity DSC is presented in Appendix J of the SAR. *Similar evaluation is presented in Appendix N for NUHOMS®24PHB system.*

7.4.2 Site Dose Assessment

Dose rate maps are constructed from the shielding analysis described in the previous sections. Direct neutron and gamma flux, as well as the air-scattered radiation from the module surfaces are considered. Figure 7.4-2 and Figure 7.4-3 provide a dose rate map in the general vicinity of a 2x10 array and two 1x10 arrays containing ten year cooled fuel. Ten year fuel is shown since it is a physical impossibility for a utility to have a facility full of five year fuel. In fact, given the average age of fuel in U.S. storage pools, and the most probable NUHOMS® loading schedules, filled NUHOMS® ISFSIs should have substantially older fuel than indicated in the Figures.

The surface radiation sources used for the direct and air scattered dose calculations are shown in Figure 7.4-5 and Figure 7.4-6. The energy distribution of the neutron and gamma fluxes is taken from the applicable calculation as described in the previous sections. Air-scattered dose rates are determined with the computer code Micro SKYSHINE (7.4); direct dose rates are calculated using the computer code MICROSHIELD (7.11). No credit is taken for shielding by nearby structures or terrain. Initial loading of all HSMs with the ten year cooled fuel is assumed. Dose rates for the PWR DSC are provided since these values bound the BWR DSC dose rates.

The ISFSI is generally surrounded by a large open area for operational and security purposes. Access to the storage modules is restricted such that during storage, no access is allowed except for security and surveillance inspection purposes. There are generally no work areas close to the ISFSI. Additional dose to plant workers due to exposure from the ISFSI is negligible. Inspection of the HSM air vents can be maintained ALARA by keeping inspection personnel back from the HSM front wall a distance which permits adequate inspection. *Appendix N provides the evaluation for the NUHOMS® -24PHB system.*

Since the site dose for an ISFSI is highly site specific, each licensee should perform a dose analysis in accordance with 10CFR72.212. The analysis should consider existing plant conditions, the site specific arrangement of the ISFSI, the characteristics of the spent fuel to be placed in dry storage, and relevant empirical data as appropriate. The on-site dose analysis should demonstrate compliance with the 10CFR20.105 limits for normal conditions and 10CFR72.106 and 10CFR100 for accident conditions.

8. ANALYSIS OF DESIGN EVENTS

In previous chapters of this SAR, the features of the standardized NUHOMS® system which are important to safety have been identified and discussed. The purpose of this chapter is to present the engineering analyses for normal and off-normal operating conditions, and to establish and qualify the system for a range of credible and hypothetical accidents. As stated in Chapter 1, the analyses presented in this section are applicable to the standard length 24P and 52B canisters. An evaluation of the long cavity 24P canister, for the same design criteria, is provided in Appendix H and Appendix L includes the same for the NUHOMS®-24PT2 DSC. *Appendix N includes the same for the NUHOMS®-24PHB DSC.* Evaluations for other canisters and modules may be included as additional appendices at a later time.

In accordance with NRC Regulatory Guide 3.48 (8.1), the design events identified by ANSI/ANS 57.9-1984, (8.2) form the basis for the accident analyses performed for the standardized NUHOMS® system. Four categories of design events are defined. Design event Types I and II cover normal and off-normal events and are addressed in Section 8.1. Design event Types III and IV cover a range of postulated accident events and are addressed in Section 8.2. These events provide a means of establishing that the NUHOMS® system design satisfies the applicable operational and safety acceptance criteria as delineated herein.

It is important to note that, given the generic nature of this SAR, the majority of the analyses presented throughout this chapter are based on bounding conservative assumptions and methodologies, with the objective of establishing upper bound values for the responses of the primary components and structures of the standardized NUHOMS® system for the design basis events. Because of the conservative approach adopted herein, the reported temperatures and stresses in this chapter envelope the actual temperatures or states of stress for the various operating and postulated accident conditions. More rigorous and detailed analyses and/or more realistic assumptions and loading conditions would result in temperatures and states of stress which are significantly lower than the reported values.

8.1 Normal and Off-Normal Operations

Normal operating design conditions consist of a set of events that occur regularly, or frequently, in the course of normal operation of the NUHOMS® system. These normal operating conditions are addressed in Section 8.1.1. Off-normal operating design conditions are events that could occur with moderate frequency, possibly once during any calendar year of operation. These off-normal operating conditions are addressed in Section 8.1.2. The thermal-hydraulic, structural, and radiological analyses associated with these events are presented in the sections which follow.

11.2 "Important-to-Safety" and "Safety Related" NUHOMS® System Components

TN West will apply the TN West Quality Assurance Program to those NUHOMS® components for which TN West has responsibility and which are "important to safety" and "safety related" as delineated in Section 3.4. These include the DSC with closure weld filler metal, the HSM, and the transfer cask. The lifting yoke is classified as "safety related".

Each item is first identified as "important to safety," "safety related" or "not important to safety." Items that are considered "important to safety" are further categorized using a graded quality approach. When the graded quality approach is used, a list shall be developed for each "important to safety" item which includes an assigned quality category consistent with the item's importance to safety. Quality categories shall be determined based on the guidance from Regulatory Guide 7.10:

Category A items are critical to safe operation. These items include structures, components, and systems whose failure or malfunction could result directly in a condition adversely affecting (1) safe spent fuel storage, (2) integrity of the spent fuel, or (3) public health and safety. This would include conditions as loss of primary containment with subsequent release of radioactive material, loss of shielding or an unsafe geometry compromising criticality control.

Category B items have a major impact on safety. These items include structures, components, and systems whose failure or malfunction could indirectly result in a condition adversely affecting (1) safe spent fuel storage, (2) integrity of the spent fuel, or (3) public health and safety. An unsafe operation could result only if a primary event occurs in conjunction with a secondary event or other failure or environmental occurrence.

Category C items have a minor impact on safety. These items include structures, components, and systems whose failure or malfunction would not significantly reduce the packaging effectiveness and would be unlikely to create a condition adversely affecting (1) safe spent fuel storage, (2) integrity of the spent fuel, or (3) public health and safety.

The Quality Assurance Program as described in paragraph 11.3 is applicable to each "important to safety" graded category and is limited as follows: For "safety related" items the program is applied as described in Category A items. Appendix L provides clarification for the procurement of Category A items for the NUHOMS®-24PT2 DSC. *Appendix N provides clarification for the procurement of Category A items for the NUHOMS®-24PHB DSC.*

Category A

This appendix contains the following items:

The drawings for the NUHOMS®-24PT2S and –24PT2L DSCs are contained in Appendix L. *The drawings for the NUHOMS®-24PHB DSCs are contained in Appendix E and N.*

TABLE OF CONTENTS

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

LIST OF TABLES

LIST OF TABLES
(Continued)

LIST OF TABLES
(Continued)

LIST OF FIGURES

LIST OF FIGURES
(Continued)