ATTACHMENT A

TECHNICAL SPECIFICATIONS

TRANSNUCLEAR WEST, INC.

STANDARDIZED NUHOMS® HORIZONTAL MODULAR STORAGE SYSTEM

CERTIFICATE OF COMPLIANCE NO. 1004

AMENDMENT 3

DOCKET 72-1004

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1.0 INTRODUCTION

This section presents the conditions which a potential user (general licensee) of the standardized NUHOMS® system must comply with, in order to use the system under the general license in accordance with the provisions of 10 CFR 72.210 and 10 CFR 72.212. These conditions have either been proposed by the system vendor, imposed by the NRC staff as a result of the review of the FSAR, or are part of the regulatory requirements expressed in 10 CFR 72.212.

1.1 General Requirements and Conditions

1.1.1 Regulatory Requirements for a General License

Subpart K of 10 CFR Part 72 contains conditions for using the general license to store spent fuel at an independent spent fuel storage installation at power reactor sites authorized to possess and operate nuclear power reactors under 10 CFR Part 50. Technical regulatory requirements for the licensee (user of the standardized NUHOMS® system) are contained in 10 CFR 72.212(b).

Under 10 CFR 72.212(b)(2) requirements, the licensee must perform written evaluations, before use, that establish that: (1) conditions set forth in the Certificate of Compliance have been met; (2) cask storage pads and areas have been designed to adequately support the static load of the stored casks; and (3) the requirements of 10 CFR 72.104 "Criteria for radioactive materials in effluent and direct radiation from an ISFSI or MRS," have been met. In addition, 10 CFR 72.212(b)(3) requires that the licensee review the FSAR and the associated SER, before use of the general license, to determine whether or not the reactor site parameters (including earthquake intensity and tornado missiles), are encompassed by the cask design bases considered in these reports.

The requirements of 10 CFR 72.212(b)(4) provide that, as a holder of a Part 50 license, the user, before use of the general license under Part 72, must determine whether activities related to storage of spent fuel involve any unreviewed safety issues, or changes in technical specifications as provided under 10 CFR 50.59. Under 10 CFR 72.212(b)(5), the general license holder shall also protect the spent fuel against design basis threats and radiological sabotage pursuant to 10 CFR 73.55. Other general license requirements dealing with review of reactor emergency plans, quality assurance program, training, and radiation protection program must also be satisfied pursuant to 10 CFR 72.212(b)(6). Records and procedural requirements for the general license holder are described in 10 CFR 72.212(b)(7), (8), (9) and (10).

Without limiting the requirements identified above, site-specific parameters and analyses, identified in the SER, that will need verification by the system user, are as a minimum, as follows:

1. The temperature of 70° F as the maximum average yearly temperature with solar incidence. The average daily ambient temperature shall be 100° F or less.

2. The temperature extremes of 125° F with incident solar radiation and -40° F with no solar incidence for storage of the DSC inside the HSM.

3. The horizontal and vertical seismic acceleration levels of 0.25g and 0.17g, respectively.

4. The analyzed flood condition of 15 fps water velocity and a height of 50 feet of water (full submergence of the loaded HSM DSC).

5. The potential for fire and explosion should be addressed, based on site-specific considerations.

6. The HSM foundation design criteria are not included in the FSAR. Therefore, the nominal FSAR design or an alternative should be verified for individual sites in accordance with 10 CFR 72.212(b)(2)(ii). Also, in accordance with 10 CFR 72.212(b)(3), the foundation design should be evaluated against actual site parameters to determine whether its failure would cause the standardized NUHOMS® system to exceed the design basis accident conditions.

7. The potential for lightning damage to any electrical system associated with the standardized NUHOMS® system (e.g., thermal performance monitoring) should be addressed, based on site-specific considerations.

8. Any other site parameters or consideration that could decrease the effectiveness of cask systems important to safety.

In accordance with 10 CFR 72.212(b)(2), a record of the written evaluations must be retained by the licensee until spent fuel is no longer stored under the general license issued under 10 CFR 72.210.

1.1.2 Operating Procedures

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The operating procedures suggested generically in the FSAR should provide the basis for the user's written operating procedure. The following additional procedure requested by NRC staff should be part of the user operating procedures:

> If fuel needs to be removed from the DSC, either at the end of service life or for inspection after an accident, precautions must be taken against the potential for the presence of damaged or oxidized fuel and to prevent radiological exposure to personnel during this operation. This can be achieved with this design by the use of the purge and fill valves which permit a determination of the atmosphere within the DSC before the removal of the inner top cover plate and shield plugs, prior to filling the DSC cavity with water (borated water for the 24P, see FSAR paragraph 5.1.1.9). If the atmosphere within the DSC is helium, then operations should proceed normally with fuel removal either via the transfer cask or in the pool. However, if air is present within the DSC, then appropriate filters should be in place to preclude the uncontrolled release of any potential airborne radioactive particulate from the DSC via the purge-fill valves. This will protect both personnel and the operations area from potential contamination. For the accident case, personnel protection in the form of respirators or supplied air should be considered in accordance with the licensee's Radiation Protection Program.

1.1.3 Quality Assurance

Activities at the ISFSI shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 50, Appendix B, and which is established, maintained, and executed with regard to the ISFSI.

1.1.4 Heavy Loads Requirements

Lifts of the DSC in the TC must be made within the existing heavy loads requirements and procedures of the licensed nuclear power plant. The TC design has been reviewed under 10 CFR Part 72 and found to meet NUREG-0612 and ANSI N14.6. However, an additional safety review (under 10 CFR 50.59) is required to show operational compliance with NUREG-0612 and/or existing plant-specific heavy loads requirements.

1.1.5 Training Module

A training module shall be developed for the existing licensee's training program establishing an ISFSI training and certification program. This module shall include the following:

1. Standardized NUHOMS® Design (overview);
2. ISFSI Facility Design (overview);
3. Certificate of Compliance conditions (overview);
4. Fuel Loading, Transfer Cask Handling, DSC Transfer Procedures; and
5. Off-Normal Event Procedures.

1.1.6 Pre-Operational Testing and Training Exercise

A dry run of the DSC loading, TC handling and DSC insertion into the HSM shall be held. This dry run shall include, but not be limited to, the following:

1. Functional testing of the TC with lifting yokes to ensure that the TC can be safely transported over the entire route required for fuel loading, washdown pit and trailer loading.

2. DSC loading into the TC to verify fit and TC/DSC annulus seal.

3. Testing of TC on transport trailer and transported to ISFSI along a predetermined route and aligned with an HSM.

4. Testing of transfer trailer alignment and docking equipment. Testing of hydraulic ram to insert a DSC loaded with test weights into an HSM and then retrieve it.

5. Loading a mock-up fuel assembly into the DSC.

6. DSC sealing, vacuum drying, and cover gas backfilling operations (using a mock-up DSC).

7. Opening a DSC (using a mock-up DSC).

8. Returning the DSC and TC to the spent fuel pool.

1.1.7 Special Requirements for First System in Place

The heat transfer characteristics of the cask system will be recorded by temperature measurements of the first DSC placed in service. The first DSC shall be loaded with assemblies, constituting a source of approximately 24 kW. The DSC shall be loaded into the HSM, and the thermal performance will be assessed by measuring the air inlet and outlet temperatures for normal airflow. Details for obtaining the measurements are provided in Section 1.2.8, under "Surveillance."

A letter report summarizing the results of the measurements shall be submitted to the NRC for evaluation and assessment of the heat removal characteristics of the cask in place within 30 days of placing the DSC in service, in accordance with 10 CFR 72.4.

Should the first user of the system not have fuel capable of producing a 24 kW heat load, or be limited to a lesser heat load, as in the case of BWR fuel, the user may use a lesser load for the process, provided that a calculation of the temperature difference between the inlet and outlet temperatures is performed, using the same methodology and inputs documented in the FSAR, with lesser load as the only exception. The calculation and the measured temperature data shall be reported to the NRC in accordance with 10 CFR 72.4. The calculation and comparison need not be reported to the NRC for DSCs that are subsequently loaded with lesser loads than the initial case. However, for the first or any other user, the process needs to be performed and reported for any higher heat sources, up to 24 kW for PWR fuel stored in the 24P, 19 kW for BWR fuel stored in the 52B and 18.3 kW for BWR fuel stored in the 61BT, which is the maximum allowed under the Certificate of Compliance. The NRC will also accept the use of artificial thermal loads other than spent fuel, to satisfy the above requirement.

1.1.8 Surveillance Requirements Applicability

The specified frequency for each Surveillance Requirement is met if the surveillance is performed within 1.25 times the interval specified in the frequency, as measured from the previous performance.

For frequencies specified as "once," the above interval extension does not apply.

If a required action requires performance of a surveillance or its completion time requires period performance of "once per...," the above frequency extension applies to the repetitive portion, but not to the initial portion of the completion time.

Exceptions to these requirements are stated in the individual specifications.

1.2 Technical Specifications, Functional and Operating Limits

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, and 1-1d.
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, and 1-1d, 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, and Westinghouse, and standard BWR fuel manufactured by General Electric. 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 system is limited for use to these standard designs and to equivalent designs by other manufacturers as listed in Appendix K of the FSAR. The analyses presented in Appendix K 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, and 1-1d, and represent the upper bound.

The design basis fuel assemblies for nuclear criticality safety are Babcock and Wilcox 15x15 fuel assemblies, General Electric 7x7 fuel assemblies and General Electric 10x10 fuel assemblies for the standardized NUHOMS®-24P, NUHOMS®-52B and NUHOMS®-61BT designs, respectively.

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

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

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. Fuel cladding temperature limits were established based on methodology in PNL-6189 and PNL-4835.

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, 52B, or 61BT canister full of design basis assemblies. The calculated surface dose rates for the 61BT DSC are bounding and are determined to be 118 mrem/hr and 1156 mrem/hr for the front surface of the HSM and transfer cask respectively. 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

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) Maximum Assembly Width (unirradiated) 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) Maximum Assembly Width (unirradiated) Maximum Assembly + BPRA Weight No. of Assemblies per DSC No. of BPRAs per DSC Fuel Cladding	171.71 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 FSAR ≤7.45 x 10^{15} g/sec per assy with spectrum bounded by that in Chapter 7 of FSAR

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 1-1b
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) Maximum Assembly Width (unirradiated) 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 FSAR ≤2.63 x 10^{15} g/sec per assy with spectrum bounded by that in Chapter 7 of FSAR

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
Maximum 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)]

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 Width (in) (excluding channels)	5.44	5.44	5.44	5.44	5.44	5.44	5.44
Channel Internal Width (in)	5.278	5.278	5.278	5.278	5.278	5.278	5.278
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 1-2a

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																					
15	5	5																			
20	5	5	5	5	5																
25		5	5	5	5	5	5	5	5												
28				5	5	5	5	5	5	5	5	5									
30						5	5	5	5	5	5	5	5								
32							5	5	5	5	5	5	5	5	5						
34								6	5	5	5	5	5	5	5	5	5				
36									6	6	6	6	5	5	5	5	5	5	5		
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 per Figure 1.1

Not Acceptable or Not Analyzed

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. Fuel with a burnup less than 15 GWd/MTU must be qualified for storage using the alternate nuclear parameters specified in Table 1-1a.**
- **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									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

Not Acceptable or Not Analyzed

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)

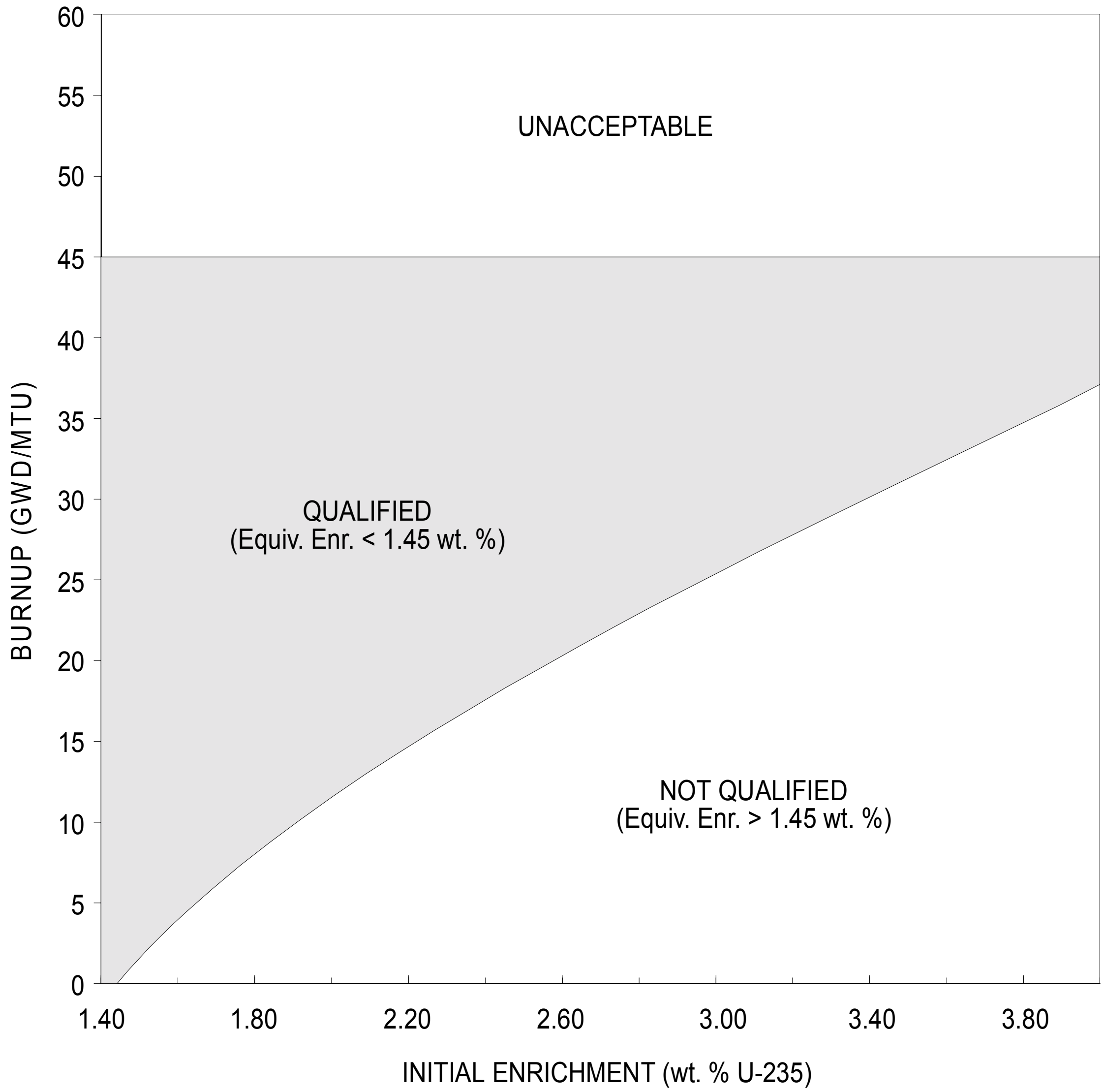
(Minimum required years of cooling time after reactor core discharge)

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																					
15	5	5																			
20	5	5	5	5	5													Not Acceptable per Figure 1.1			
25		5	5	5	5	5	5	5	5												
28				5	5	5	5	5	5	5	5	5									
30						6	6	6	5	5	5	5	5								
32							6	6	6	6	6	6	5	5	5						
34								7	6	6	6	6	6	6	6	6	6				
36									8	7	7	7	6	6	6	6	6	6	6		
38											8	8	7	7	7	7	6	6	6	6	6
40												9	9	8	8	8	7	7	7	7	6
41					Not Acceptable or Not Analyzed							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

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. Fuel with a burnup less than 15 GWd/MTU must be qualified for storage using the alternate nuclear parameters specified in Table 1-1a.**
- **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.**

Figure 1.1



PWR Fuel Criticality Acceptance Curve

1.2.2 DSC Vacuum Pressure During Drying

Limit/Specification:

Vacuum Pressure: ≤3 mm Hg

Time at Pressure: ≥30 minutes following stepped evacuation
Number of Pump-Downs: 2

Applicability: This is applicable to all DSCs.

Objective: To ensure a minimum water content.

Action: If the required vacuum pressure cannot be obtained:

1. Confirm that the vacuum drying system is properly installed.
2. Check and repair, or replace, the vacuum pump.
3. Check and repair the system as necessary.
4. Check and repair the seal weld between the inner top cover plate and the DSC shell.

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

Bases: A stable vacuum pressure of ≤3 mm Hg further ensures that all liquid water has evaporated in the DSC cavity, and that the resulting inventory of oxidizing gases in the DSC is well below the 0.25 volume%.

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 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 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 DSC Helium Leak Rate of Inner Seal Weld

Limit/Specification:

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

Applicability: This specification is applicable to the inner top cover plate seal weld of 61BT 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 $\cdot$ 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 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.

1.2.5 DSC Dye Penetrant Test of Closure Welds

Limit/Specification:

All DSC closure welds except those subjected to full volumetric inspection shall be dye penetrant tested in accordance with the requirements of the ASME Boiler and Pressure Vessel Code Section III, Division 1, Article NB-5000 (Reference 8.3 of FSAR). The liquid penetrant test acceptance standards shall be those described in Subsection NB-5350 of the Code.

Applicability: This is applicable to all DSCs. The welds include inner and outer top and bottom covers, and vent and syphon port covers.

Objective: To ensure that the DSC is adequately sealed in a redundant manner and leak tight.

Action: If the liquid penetrant test indicates that the weld is unacceptable:

1. The weld shall be repaired in accordance with approved ASME procedures.

2. The new weld shall be re-examined in accordance with this specification.

Surveillance: During DSC closure operations. No additional surveillance is required for this operation.

Bases: Article NB-5000 Examination, ASME Boiler and Pressure Vessel Code, Section III, Division 1, Sub-Section NB (Reference 8.3 of FSAR).

1.2.6 Deleted

1.2.7 HSM Dose Rates

Limit/Specification:

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

a. 400 mrem/hr at 3 feet from the HSM surface.

b. Outside of HSM door on center line of DSC 100 mrem/hr.

c. End shield wall exterior 20 mrem/hr.

Applicability: This specification is applicable to all HSMs which contain a loaded 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, Appendix J, and Appendix K 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

1.2.8 HSM Maximum Air Exit Temperature

Limit/Specification:

Following initial DSC transfer to the HSM or the occurrence of accident conditions, the equilibrium air temperature difference between ambient temperature and the vent outlet temperature shall not exceed 100° F for ≥5 year cooled fuel, when fully loaded with 24 kW heat.

Applicability: This specification is applicable to all HSMs stored in the ISFSI. If a DSC is placed in the HSM with a heat load less than 24 kW, the limiting difference between outlet and ambient temperatures shall be determined by a calculation performed by the user using the same methodology and inputs documents in the FSAR and SER.

Objective: The objective of this limit is to ensure that the temperatures of the fuel cladding and the HSM concrete do not exceed the temperatures calculated in Section 8 of the FSAR. That section shows that if the air outlet temperature difference is less than or equal to 100° F (with a thermal heat load of 24 kW), the fuel cladding and concrete will be below the respective temperature limits for normal long-term operation.

Action: If the temperature rise is greater than that specified, then the air inlets and exits should be checked for blockage. If the blockage is cleared and the temperature is still greater than that specified, the DSC and HSM cavity may be inspected using video equipment or other suitable means. If environmental factors can be ruled out as the cause of excessive temperatures, then the fuel bundles are producing heat at a rate higher than the upper limit specified in Section 3 of the FSAR and will require additional measurements and analysis to assess the actual performance of the system. If excessive temperatures cause the system to perform in an unacceptable manner and/or the temperatures cannot be controlled to acceptable limits, then the cask shall be unloaded.

Surveillance: The temperature rise shall be measured and recorded daily following DSC insertion until equilibrium temperature is reached, 24 hours after insertion, and again on a daily basis after insertion into the HSM or following the occurrence of accident conditions. If the temperature rise is within the specifications or the calculated value for a heat load less than 24 kW, then the HSM and DSC are performing as designed to meet this specification and no further maximum air exit temperature measurements are required. Air temperatures must be measured in such a manner as to obtain representative values of inlet and outlet air temperatures.

Basis: The specified temperature rise is selected to ensure the fuel clad and concrete temperatures are maintained at or below acceptable long-term storage limits.

1.2.9 Transfer Cask Alignment with HSM

Limit/Specification:

The cask must be aligned with respect to the HSM so that the longitudinal centerline of the DSC in the transfer cask is within ±1/8 inch of its true position when the cask is docked with the HSM front access opening.

Applicability: This specification is applicable during the insertion and retrieval of all DSCs.

Objective: To ensure smooth transfer of the DSC from the transfer cask to HSM and back.

Action: If the alignment tolerance is exceeded, the following actions should be taken:

a. Confirm that the transfer system is properly configured.

b. Check and repair the alignment equipment.

c. Confirm the locations of the alignment targets on the transfer cask and HSM.

Surveillance: Before initiating DSC insertion or retrieval, confirm the alignment. Observe the transfer system during DSC insertion or retrieval to ensure that motion or excessive vibration does not occur.

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

1.2.10 DSC Handling Height Outside the Spent Fuel Pool Building

Limit/Specification:

1. The loaded TC/DSC shall not be handled at a height greater than 80 inches outside the spent fuel pool building.

2. In the event of a drop of a loaded TC/DSC from a height greater than 15 inches: (a) fuel in the DSC shall be returned to the reactor spent fuel pool; (b) the DSC shall be removed from service and evaluated for further use; and (c) the TC shall be inspected for damage and evaluated for further use.

Applicability: The specification applies to handling the TC, loaded with the DSC, on route to, and at, the storage pad.

Objective:

1. To preclude a loaded TC/DSC drop from a height greater than 80 inches.

2. To maintain spent fuel integrity, according to the spent fuel specification for storage, continued confinement integrity, and DSC functional capability, after a tip-over or drop of a loaded DSC from a height greater than 15 inches.

Surveillance: In the event of a loaded TC/DSC drop accident, the system will be returned to the reactor fuel handling building, where, after the fuel has been returned to the spent fuel pool, the DSC and TC will be inspected and evaluated for future use.

Basis: The NRC evaluation of the TC/DSC drop analysis concurred that drops up to 80 inches, of the DSC inside the TC, can be sustained without breaching the confinement boundary, preventing removal of spent fuel assemblies, or causing a criticality accident. This specification ensures that handling height limits will not be exceeded in transit to, or at the storage pad. Acceptable damage may occur to the TC, DSC, and the fuel stored in the DSC, for drops of height greater than 15 inches. The specification requiring inspection of the DSC and fuel following a drop of 15 inches or greater ensures that the spent fuel will continue to meet the requirements for storage, the DSC will continue to provide confinement, and the TC will continue to provide its design functions of DSC transfer and shielding.

1.2.11 Transfer Cask Dose Rates

Limit/Specification:	Dose rates from the transfer cask shall be limited to levels which are less than or equal to: a. 200 mrem/hr at 3 feet with water in the DSC cavity. b. 500 mrem/hr at 3 feet without water in the DSC cavity.
Applicability:	This specification is applicable to the transfer cask containing a loaded DSC.
Objective:	The dose rate is limited to this value to ensure that the 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 during DSC transfer operations.
Action:	If specified dose rates are exceeded, place temporary shielding around affected areas of transfer cask and review the plant records of the fuel assemblies which have been placed in DSC to ensure they conform to the fuel specifications of Section 1.2.1. 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 dose rates should be measured as soon as possible after the transfer cask is removed from the spent fuel pool.
Basis:	The basis for this limit is the shielding analysis presented in Section 7.0, Appendix J, and Appendix K of the FSAR.

1.2.12 Maximum DSC Removable Surface Contamination

Limit/Specification:

2,200 dpm/100 cm^2 for beta-gamma sources
220 dpm/100 cm^2 for alpha sources.

Applicability: This specification is applicable to all DSCs.

Objective: To ensure that release of non-fixed contamination above accepted limits does not occur.

Action: If the required limits are not met:

a. Flush the DSC/transfer cask annulus with demineralized water and repeat surface contamination surveys of the DSC upper surface.

b. If contamination of the DSC cannot be reduced to an acceptable level by this means, direct surface cleaning techniques shall be used following removal of the fuel assemblies from the DSC and removal of the DSC from the transfer cask.

c. Check and replace the DSC/transfer cask annulus seal to ensure proper installation and repeat canister loading process.

Surveillance: Following placement of each loaded DSC/transfer cask into the cask decontamination area, fuel pool water above the top shield plug shall be removed and the top region of the DSC and cask shall be decontaminated. A contamination survey of the upper 1 foot of the DSC and cask shall be taken. In addition, contamination surveys shall be taken on the inside surfaces of the TC after the DSC has been transferred into the HSM. If the above surface contaimination limit is exceeded, the TC shall be decontaminated.

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

1.2.13 TC/DSC Lifting Heights as a Function of Low Temperature and Location

Limit/Specification:

1. No lifts or handling of the TC/DSC at any height are permissible at DSC basket temperatures below -20° F inside the spent fuel pool building.

2. The maximum lift height of the TC/DSC shall be 80 inches if the basket temperature is below 0° F but higher than -20° F inside the spent fuel pool building.

3. No lift height restriction is imposed on the TC/DSC if the basket temperature is higher than 0° F inside the spent fuel pool building.

4. The maximum lift height and handling height for all transfer operations outside the spent fuel pool building shall be 80 inches <u>and</u> the basket temperature may not be lower than 0° F.

Applicability: These temperature and height limits apply to lifting and transfer of all loaded TC/DSCs inside and outside the spent fuel pool building. The requirements of 10 CFR Part 72 apply outside the spent fuel building. The requirements of 10 CFR Part 50 apply inside the spent fuel pool building.

Objective: The low temperature and height limits are imposed to ensure that brittle fracture of the ferritic steels, used in the TC trunnions and shell and in the DSC basket, does not occur during transfer operations.

Action: Confirm the basket temperature before transfer of the TC. If calculation or measurement of this value is available, then the ambient temperature may conservatively be used.

Surveillance: The ambient temperature shall be measured before transfer of the TC/DSC.

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

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

1.2.14 TC/DSC Transfer Operations at High Ambient Temperatures

Limit/Specification:

1. The ambient temperature for transfer operations of a loaded TC/DSC shall not be greater that 100° F (when cask is exposed to direct insolation).

2. For transfer operations when ambient temperatures exceed 100° F up to 125° F, a solar shield shall be used to provide protection against direct solar radiation.

Applicability: This ambient temperature limit applies to all transfer operations of loaded TC/DSCs outside the spent fuel pool building.

Objective: The high temperature limit (100° F) is imposed to ensure that:

1. The fuel cladding temperature limit is not exceeded,

2. The solid neutron shield material temperature limit is not exceeded, and

3. The corresponding TC cavity pressure limit is not exceeded.

Action: Confirm what the ambient temperature is and provide appropriate solar shade if ambient temperature is expected to exceed 100° F.

Surveillance: The ambient temperature shall be measured before transfer of the TC/DSC.

Bases: The basis for the high temperature limit is PNL-6189 (Reference 1) for the fuel clad limit, the manufacturer's specification for neutron shield, and the design basis pressure of the TC internal cavity pressure.

1.2.15 Boron Concentration in the DSC Cavity Water (24-P Designs Only)

Limit/Specification:

The DSC cavity shall be filled only with water having a boron concentration equal to, or greater than 2,000 ppm.

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 these systems use fixed absorber plates.

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 an accidental misloading of the DSC with unburned fuel, maximum enrichment, and optimum moderation conditions.

1.2.16 Provision of TC Seismic Restraint Inside the Spent Fuel Pool Building as a Function of Horizontal Acceleration and Loaded Cask Weight

Limit/Specification:

Seismic restraints shall be provided to prevent overturning of a loaded TC during a seismic event if a certificate holder determines that the horizontal acceleration is 0.40 g or greater and the fully loaded TC weight is less than 190 kips. The determination of horizontal acceleration acting at the center of gravity (CG) of the loaded TC must be based on a peak horizontal ground acceleration at the site, but shall not exceed 0.25 g.

Applicability: This condition applies to all TCs which are subject to horizontal accelerations of 0.40 g or greater.

Objective: To prevent overturning of a loaded TC inside the spent fuel pool building.

Action: Determine what the horizontal acceleration is for the TC and determine if the cask weight is less than 190 kips.

Surveillance: Determine need for TC restraint before any operations inside the spent fuel pool building.

Bases: Calculation of overturning and restoring moments.

1.2.17 61BT DSC Vacuum Drying Duration Limit

Limit/Specifications:

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

Applicability: This specification is only applicable to a 61BT DSC with greater than 17.6 kw heat load.

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

Action:

1. If the DSC vacuum drying pressure limit of Technical Specification 1.2.2 cannot be achieved at 72 hours 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 limit of 96 hours was selected to ensure that the temperature within the DSC is within the design 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 to prevent exceeding the allowable HSM concrete temperature or the fuel cladding temperature.

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. 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 a time period of approximately 40 hours.

1.3.2 HSM Thermal Performance

Surveillance: Verify a temperature measurement of the thermal performance, for each HSM, on a daily basis. The temperature measurement could be any parameter such as (1) a direct measurement of the HSM temperatures, (2) a direct measurement of the DSC temperatures, (3) a comparison of the inlet and outlet temperature difference to predicted temperature differences for each individual HSM, or (4) other means that would identify and allow for the correction of off-normal thermal conditions that could lead to exceeding the concrete and fuel clad temperature criteria. If air temperatures are measured, they must be measured in such a manner as to obtain representative values of inlet and outlet air temperatures. Also due to the proximity of adjacent HSM modules, care must be exercised to ensure that measured air temperatures reflect only the thermal performance of an individual module, and not the combined performance of adjacent modules.

Action: If the temperature measurement shows a significant unexplained difference, so as to indicate the approach of materials to the concrete or fuel clad temperature criteria, take appropriate action to determine the cause and return the canister to normal operation. If the measurement or other evidence suggests that the concrete accident temperature criteria (350° F) has been exceeded for more than 24 hours, the HSM must be removed from service unless the licensee can provide test results in accordance with ACI-349, appendix A.4.3, demonstrating that the structural strength of the HSM has an adequate margin of safety.

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

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. HSM Dose Rates	L	1.2.7
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. Transfer Cask Dose Rates	L	1.2.11
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
13. TC/DSC Lifting Heights as a Function of Low Temperature and Location	L	1.2.13

<u>Legend</u>

PL Prior to loading
L During loading and prior to movement to HSM pad
24 hrs Time following DSC insertion into HSM
S Prior to movement of DSC to or from HSM
AN As necessary
D Daily (24 hour frequency)

14. TC/DSC Transfer Operations at High Ambient Temperatures	L	1.2.14
15. Boron Concentration in DSC Cavity Water (24-P Designs Only)	PL	1.2.15
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 Vacuum Drying Limits	L	1.2.17

Legend

PL Prior to loading
L During loading and prior to movement to HSM pad
24 hrs Time following DSC insertion into HSM
S Prior to movement of DSC to or from HSM
AN As necessary
D Daily (24 hour frequency)

References

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

2. Johnson, A.B., Jr., and E.R. Gilbert, "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases," PNL-4835, September 1983.