February 15, 2000

Mr. Robert M. Grenier
President and Chief Operating Officer
Transnuclear West Inc.
39300 Civic Center Drive, Suite 280
Fremont, CA 94538-2324

SUBJECT: PRELIMINARY CERTIFICATE OF COMPLIANCE AND SAFETY EVALUATION REPORT FOR THE NUHOMS® STORAGE SYSTEM (TAC NO. L22954)

Dear Mr. Grenier:

By letter dated July 26, 1999, as supplemented, Transnuclear West Inc. (TN West) submitted an application to the Nuclear Regulatory Commission, in accordance with 10 CFR Part 72, for the review and approval of Amendment 3 to the NUHOMS® Certificate of Compliance No. 1004.

As a result of our review of your application and its supplements, the staff has prepared a preliminary Certificate of Compliance (CoC) and Safety Evaluation Report (SER) pursuant to the requirements of 10 CFR Part 72. Enclosed is a copy of the preliminary CoC and SER for TN West's review and identification of inaccuracies and omissions. Attached to the CoC is the Technical Specifications with pen and ink changes as discussed by phone with Mr. Robert Grubb of your staff. Please also review this document and provide a revised paper copy and an electronic version with your response. Please respond with any comments by noon on February 28, 2000. Please continue to reference Docket No. 72-1004 and TAC No. L22954 in future correspondence related to this request.

If you have any comment or question concerning this request, please contact me at 301-415-8584.

Sincerely,
ORIGINAL SIGNED BY /s/
Steven Baggett, Project Manager
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No.: 72-1004
Enclosures: 1. Preliminary CoC
2. Preliminary SER

Distribution: w/o encls*

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NRC FORM 651
(3-1999)
10 CFR 72

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

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

Certificate No.	Effective Date	Expiration Date	Docket Number	Amendment No.	Amendment Date	Package Identification No.
1004	1/23/95	1/23/2015	72-1004	3		USA/72-1004

Issued To: (Name/Address)

Transnuclear West Inc.,
39300 Civic Center Drive, Suite 280
Fremont, CA 94538

Safety Analysis Report Title

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

CONDITIONS

1. Casks authorized by this certificate are hereby approved for use by holders of 10 CFR Part 50 licenses for nuclear power reactors at reactor sites under the general license issued pursuant to 10 CFR Part 72.210 subject to the conditions specified by 10 CFR 72.212 and the attached Technical Specifications.

2. The holder of this certificate who desires to change the certificate or Technical Specifications shall submit an application for amendment of the certificate or Technical Specifications.

3. CASK:

a. Model Nos. Standardized NUHOMS-24P and NUHOMS-52B

The two digits refer to the number of fuel assemblies stored in the dry shielded canister (DSC), and the character P for pressurized water reactor (PWR) or B for boiling water reactor (BWR) is to designate the type of fuel stored.

b. Description

The Standardized NUHOMS System is certified as described in the safety analysis report (SAR) and in the NRC's safety evaluation report (SER) accompanying the Certificate of Compliance (CoC). The Standardized NUHOMS System is a horizontal canister system composed of a steel dry shielded canister (DSC), a reinforced concrete horizontal storage module (HSM), and a transfer cask (TC). The welded DSC provides confinement and criticality control for the storage and transfer of irradiated fuel. The concrete module provides radiation shielding while allowing cooling of the DSC and fuel by natural convection during storage. The TC is used for transferring the DSC from/to the Spent Fuel Pool Building to/from the HSM.

The principal component subassemblies of the DSC are the shell with integral bottom cover plate and shield plug and ram/grapple ring, top shield plug, top cover plate, and basket assembly. The shell length is fuel-specific. The internal basket assembly is composed of guide sleeves, support rods, and spacer disks. This assembly is designed to hold 24 PWR fuel assemblies or 52 BWR assemblies. It aids in the insertion of the fuel assemblies, enhances subcriticality during loading operations, and provides structural support during a hypothetical drop accident. The DSC is designed to slide from the transfer cask into the HSM and back without undue galling, scratching, gouging, or other damage to the sliding surfaces.

The HSM is a reinforced concrete unit with penetrations located at the top and bottom of the side walls for air flow. The penetrations are protected from debris intrusions by wire mesh screens during storage operation. The DSC Support Structure, a structural steel frame with rails, is installed within the HSM module to provide for sliding the DSC in and out of the HSM and to support the DSC within the HSM.

The TC is designed and fabricated as a lifting device to meet NUREG-0612 and ANSI N14.6 requirements. It is used for transfer operations within the Spent Fuel Pool Building and for transfer operations to/from the HSM. The TC is a cylindrical vessel with a bottom end closure assembly and a bolted top cover plate. Two upper lifting trunnions are located near the top of the cask for downending/uprighting and lifting of the cask in the Spent Fuel Pool Building. The lower trunnions, located near the base of the cask, serve as the axis of rotation during downending/uprighting operations and as supports during transport to/from the Independent Spent Fuel Storage Installation (ISFSI).

Other fuel transfer and auxiliary equipment are necessary for operation of the Standardized NUHOMS System at an ISFSI. However, this equipment is not certified by this CoC as part of the Standardized NUHOMS System. Such equipment may be site-specific and may include, but is not limited to, special lifting devices, the transfer trailer, and the skid positioning system.

c. Drawings

The drawings for the Standardized NUHOMS System are contained in Appendix E of the SAR.

d. Basic Components

The basic components of the Standardized NUHOMS System that are important to safety are the DSC, HSM, and TC. These components are described in Section 4.2 of the SAR.

4. Not used

5. Notification of fabrication schedules shall be made in accordance with the requirements of 10 CFR 72.232(c).

6. Not used

7. Not used

8. Not used

9. CHANGES, TESTS, AND EXPERIMENTS:

The holder of this Certificate of Compliance (CoC) may:

(1) Make changes in the cask design described in the safety analysis report (SAR),

(2) Make changes in the procedures described in the SAR, or

(3) Conduct tests or experiments not described in the SAR,

without prior Commission approval, unless the proposed change, test or experiment involves a change in the CoC, an unreviewed safety question, a significant increase in occupational exposure, or a significant unreviewed environmental impact.

A proposed change or experiment shall be deemed to involve an unreviewed safety question:

(1) If the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased; or

(2) If a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR may be created; or

(3) If the margin of safety as defined in the basis for any Technical Specification or limit is reduced.

The holder of this CoC shall maintain records of changes in the cask design and of changes in procedures if the changes constitute changes in the cask design or procedures described in the SAR. The holder of this CoC shall also maintain records of tests and experiments it conducts that are not described in the SAR. These records must include a written safety evaluation that provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question. The records of changes in the cask design and of changes in procedures and records of tests or experiments must be maintained until the Commission terminates the certificate.

10. The holder of this CoC shall annually furnish to the Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555, a report containing a brief description of changes, tests, and experiments made under Condition 9, including a summary of the safety evaluation of each. Any such report submitted by a holder of this CoC will be made a part of the public record pertaining to this certificate.

11. All Standardized NUHOMS Systems must be fabricated and used in accordance with CoC No. 1004, Amendment No. 3, except that general licensees may use the Standardized NUHOMS Systems that were fabricated in accordance with the original CoC, or Amendment Nos.1 or 2.

FOR THE NUCLEAR REGULATORY COMMISSION

Proposed

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Signed this day of , 2000

Attachment: Technical Specifications

ATTACHMENT A

TECHNICAL SPECIFICATIONS

TRANSNUCLEAR WEST, INC.

STANDARDIZED NUHOMS® HORIZONTAL MODULAR STORAGE SYSTEM

CERTIFICATE OF COMPLIANCE No. 1004

AMENDMENT ~~2~~ 3

DOCKET 72-1004

Pls Revise page # & Section numbering
It might be time to resubmit
file in its entirety -
Thanks
[illegible]

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 SAR, 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 and 19 kW for BWR fuel, 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/Specifications: 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 and 1-1b.

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1.0 INTRODUCTION

This section presents the conditions which a potential user (licensee) of the standardized NUHOMS® system must comply with, in order to use the system under a general license to be issued according to the provisions of 10 CFR 72.210 and 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 SAR, 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 discusses conditions for issuing a 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).

10 CFR 72.212(b)(2) requires that the licensee 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. It also requires that the licensee review the SAR 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.

As a holder of a Part 50 license, the user, before use of the general Part 72 license, must determine whether activities related to storage of spent fuel involve any unreviewed safety issues, as provided under 10 CFR 50.59. 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. 10 CFR 72.212(b)(7), (8), (9) and (10) deal with record requirements for the general license holder.

Site-specific parameters and analyses, identified in the SER, that need verification by the system user, are as follows:

1. The temperature of 70°F as the maximum average yearly temperature with solar incidence (Reference SER Section 2.4.1).

2. The temperature extremes of 125°F with incident solar radiation and -40°F with no solar incidence (Reference SER Section 2.4.1) for storage of the DSC inside the HSM.

3. The horizontal and vertical seismic acceleration levels of 0.25g and 0.17g, respectively (Reference SER Table 2-4).

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) (Reference SER Table 2-4).

5. The potential for fire and explosion should be addressed, based on site-specific considerations (Reference SER Table 2-4).

6. The HSM foundation design criteria are not included in the SAR. Therefore, the nominal SAR design or an alternative should be verified for individual sites in accordance with 10 CFR 72.212(b)(2)(ii).

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 (Reference SER Table 2.4).

According to 10 CFR 72.212(b) 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 SAR were considered appropriate as discussed in Section 11.0 of the SER and should provide the basis for the user's written operating procedure. The following additional procedure requested by the staff in Section 11.3 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 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. 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. (Reference 8). 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.

1.2.

~~10.3.1~~ 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 10.3-1 and 10.3-2, and Figure 10.3-1.

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 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 SAR 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 Table 10.3-1 and Figure 10.3-1 for PWR fuel, or Table 10.3-2 for BWR fuel 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 an 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 SAR 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 system is limited for use to these standard designs and to equivalent designs by other manufacturers as listed in Chapter 3 of the SAR.* The analyses presented in the SAR are based on non-consolidated, zircaloy-clad fuel with no known or suspected gross breaches.

The physical parameters that define the mechanical and structural design of the HSM and the DSC are the fuel assembly dimensions and weight. The calculated stresses are based on the physical parameters given in Tables 10.3-1 and 10.3-2 and represent the upper bound.

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

The NUHOMS®-24P system is designed for use with standard BPRA designs for the B&W 15x15 and Westinghouse 17x17 fuel types as listed in Appendix J of the SAR.

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 the B&W 15x15 BPRAs. In addition, BPRAs with cladding failure were determined to be acceptable for loading into NUHOMS®-24P Long Cavity DSC as evaluated in Appendix J of the SAR.

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 10.3.15 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 10.3-1 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 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 (References 10.5.1, 10.5.2).

The radiological design criterion is that the fuel stored in the NUHOMS® system must not increase the average calculated HSM or transfer cask surface dose rates beyond those calculated for a canister full of design basis fuel assemblies *with or without BPRAs*. The design value average HSM and cask surface dose rates 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.*

Tab 6 1.1a

Table ~~10.3-1~~

PWR Fuel Specifications of 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)* Fuel Design *Maximum* Assembly Length *(Unirradiated)* *Maximum Assembly Width (Unirradiated)* Maximum Assembly Weight No. of Assemblies per DSC Fuel Cladding	 See SAR Table 3.1-1a *165.75 in (Standard cavity)* *171.71 in (Long cavity)* *8.536 in* *1682 lbs* ≤ 24 intact assemblies Zircaloy-clad fuel with no known or suspected gross cladding breaches
Physical Parameters (with BPRAs) *Fuel Design* *Maximum Assembly + BPRA Length (Unirradiated)* *Maximum Assembly Width (Uniirradiated)* *Maximum Assembly Weight* *No. of Assemblies per DSC* *No. of BPRAs per DSC* *Fuel Cladding*	 *See SAR Table 3.1-1a* *171.71 in (Long cavity)* *8.536 in* *1682 lbs* *≤ 24 intact assemblies* *≤ 24 BPRAs* *Zircaloy-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 10.3-3 *(without BPRAs)* *Or* *Per Table 10.3-5 (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 10.3-1 ≤ 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

1-1b

Table ~~10.3-2~~

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 Fuel Design *Maximum* Assembly Length *(Unirradiated)* *Maximum Assembly Width (Unirradiated)* Maximum Assembly Weight (w/fuel channels) No. of Assemblies per DSC Fuel Cladding	 See SAR Table 3.1-2a *176.16 in* *5.454 in* *725* ≤ 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 10.3-4
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

Table ~~10.3-3~~ 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 (w/o 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 10.3.1				
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					Not Acceptable or Not Analyzed							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

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 w/o U-235 must be qualified for storage using the alternate nuclear parameters specified in Table 10.3-1. Fuel with an initial enrichment greater than 4.0 w/o U-235 is unacceptable for storage.
- Fuel with a burnup greater than *45* GWd/MTIHM is unacceptable for storage. Fuel with a burnup less than 15 GWd/MTIHM must be qualified for storage using the alternate nuclear parameters specified in Table 10.3-1.
- Example: An assembly with an initial enrichment of 3.65 w/o U-235 and a burnup of 42.5 GWd/MTIHM is acceptable for storage after a ten-year cooling time as defined at the intersection of 3.6 w/o U-235 (rounding down) and 43 GWd/MTIHM (rounding up) on the qualification table.

1-2b

Table ~~10.3-4~~

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

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

Burnup (GWd/ MTIHM)	Initial Enrichment (w/o 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 w/o U-235 must be qualified for storage using the alternate nuclear parameters specified in Table 10.3-2. Fuel with an initial enrichment greater than 4.0 w/o U-235 is unacceptable for storage.
- Fuel with a burnup greater than 45 GWd/MTIHM is unacceptable for storage. Fuel with a burnup less than 15 GWd/MTIHM is acceptable after three years cooling time provided the physical parameters from Table 10.3-2 have been met.
- Example: An assembly with an initial enrichment of 3.05 w/o U-235 and a burnup of 34.5 GWd/MTIHM is acceptable for storage after a nine-year cooling time as defined at the intersection of 3.0 w/o U-235 (rounding down) and 35 GWd/MTIHM (rounding up) on the qualification table.

1/10/00

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)	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						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												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

Initial Enrichment (w/o U-235)

Upper region: Not Acceptable per Figure 1.1

Lower region: Not Acceptable or Not Analyzed

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 w/o 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 w/o 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 w/o 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 w/o U-235 (rounding down) and 43 GWd/MTU (rounding up) on the qualification table.*

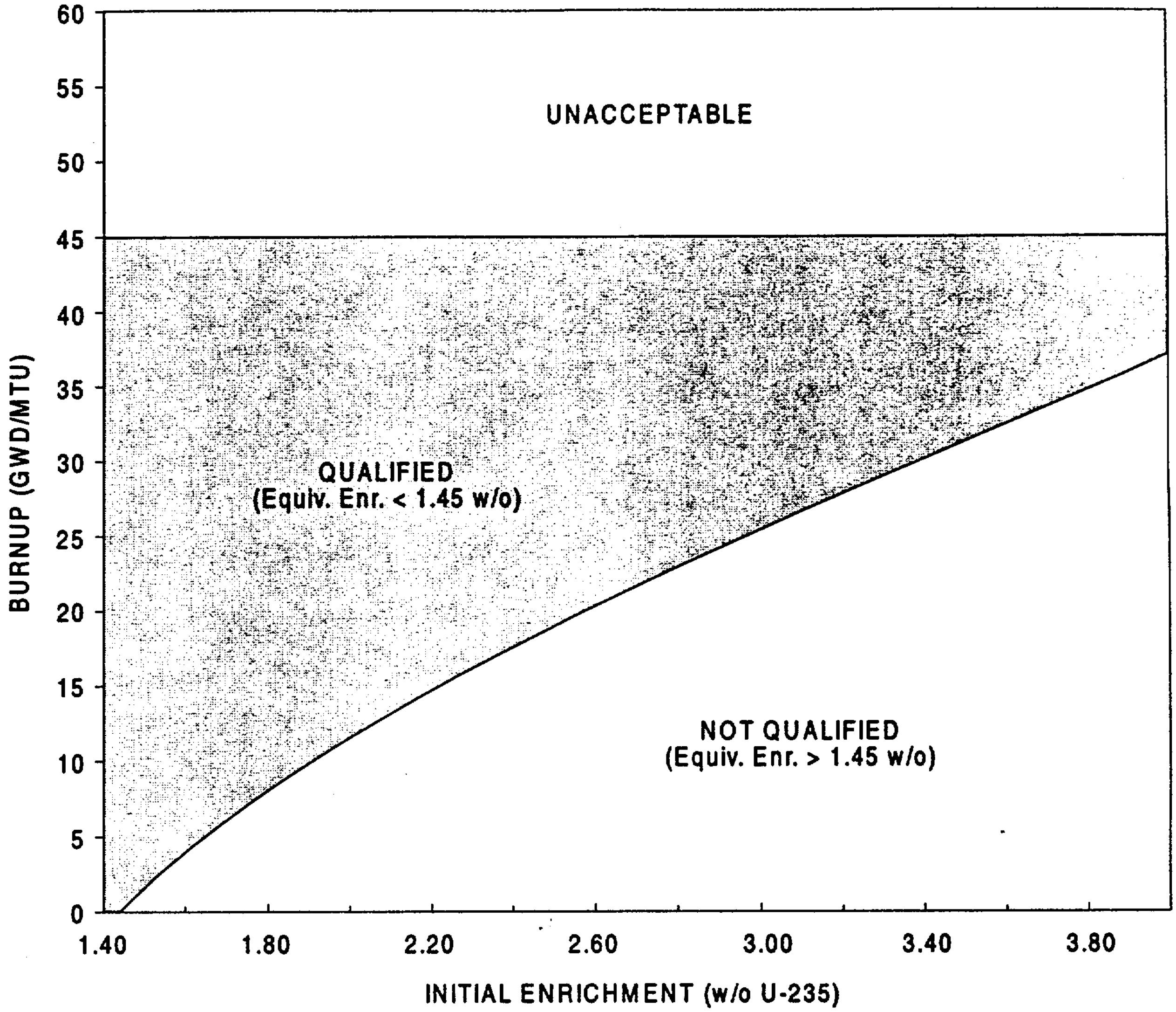

Figure ~~10.3-1~~ 1-1
PWR Fuel Criticality Acceptance Curve

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 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 all DSCs.
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 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 all DSCs.

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).

1.2.5 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 SAR). 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 SAR).

1.2.6 DSC Top End Dose Rates

not used

Limit/Specification:	Dose rates at the following locations shall be limited to levels which are less than or equal to: a. 200 mrem/hr at top shield plug surface at centerline with water in cavity. b. 400 mrem/hr at top cover plate surface at centerline without water in cavity.
Applicability:	This specification is applicable to all DSCs.
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 reasonably achievable during DSC closure operations.
Action:	a. If specified dose rates are exceeded, the following actions should be taken: 1. Confirm that the spent fuel assemblies placed in DSC conform to the fuel specifications of Section 1.2.1 2. Visually inspect placement of top shield plug. Re-install or adjust position of top shield plug if it is not properly seated. 3. Install additional temporary shielding. 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:	Dose rates shall be measured before seal welding the inner top cover plate to the DSC shell and welding the outer top cover plate to the DSC shell.
Basis:	The basis for this limit is the shielding analysis presented in Section 7.0 of the SAR.

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10.3.7 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 10.3.1 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 10.3.1.

5. Install temporary or permanent shielding to mitigate the dose to acceptable levels in accordance with 10CFR Part 20, 10CFR72.104(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

10CFR72.4 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 of the SAR*. The specified dose rates provide as-low-as-is-reasonably-achievable on-site and off-site doses in accordance with 10CFR Part 20 and 10CFR72.104(a).

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 SAR 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 SAR. 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 SAR 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.

10.3.11 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 10.3.1 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 10.3.1. 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 10CFR72.4 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 *and Appendix J of the SAR.*

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 Design 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 system since that system uses 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.

1.3 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
4. DSC Helium Leak Rate of Inner Seal Weld	L	1.2.4
5. DSC Dye Penetrant Test of Closure Welds	L	1.2.5
6. DSC Top End Dose Rates	L	1.2.6
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	S	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

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 insertion of DSC into HSM
AN As necessary
D Daily (24 hour frequency)

Table 1.3.1

Summary of Surveillance and Monitoring Requirements (Continued)

Surveillance or Monitoring	Period	Reference Section
14. TC/DSC Transfer Operations at High Ambient Temperatures	PL	1.2.14
15. Boron Concentration in DSC Cavity Water (24-P Design 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

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 insertion of DSC into 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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PRELIMINARY SAFETY EVALUATION REPORT

DOCKET NO. 72-1004
MODEL NOS. STANDARDIZED NUHOMS® -24P and -52B
TRANSNUCLEAR WEST, INC.
CERTIFICATE OF COMPLIANCE NO. 1004
AMENDMENT NO. 3

SUMMARY

Currently, the Certificate of Compliance for the NUHOMS®-24P horizontal modular storage system is approved only for storage of spent fuel assemblies. By letter dated July 26, 1999, as supplemented, Transnuclear West Inc. (TN West) submitted an application for Amendment No. 3 to Certificate of Compliance No. 1004. TN West requests approval to add burnable poison rod assemblies (BPRAs) for the Babcock and Wilcox (B&W) 15x15 and Westinghouse 17x17 fuel assembly types to the authorized contents of the NUHOMS®-24P horizontal modular storage system long cavity dry storage canister. The staff performed a detailed safety evaluation of the proposed amendment request and determined that the addition of the BPRAs to the B&W 15x15 and Westinghouse 17x17 fuel assembly types meets the requirements of 10 CFR Part 72.

1.0 STRUCTURAL

This section presents the results of the structural design review of the NUHOMS®-24P long cavity dry shielded canister (DSC) for the addition of BPRAs for B&W 15x15 and Westinghouse (WE) 17x17 fuel assembly types to be added as authorized contents. The purpose of this review is to verify that the DSC design is not adversely affected by the addition of BPRAs and that the DSC meets the structural requirements of 10 CFR Part 72.

The impact of the addition of BPRAs in the fuel assembly on the DSC structural design was evaluated by comparing the changes in the structural parameters, which may impact adversely on the DSC design. The significant structural parameters affected by the addition of BPRAs are the weight of the fuel assembly, center of gravity of the mass, temperatures, and internal pressures. Each of these parameters are reviewed below.

Weight of the Fuel Assembly

The application states that the maximum weight of the fuel assemblies with BPRAs is enveloped by the weight used in the NRC approved design basis for the DSC and the NUHOMS® storage system. The staff independently evaluated this as shown below.

PWR Fuel Assembly type	Weight of the Fuel/ (Ref. 1: Table 3.1-3)	Weight of the BPRAs (Ref. 3)	Total Weight of the fuel assembly w/BPRAs
BW 15 x 15	1515 lbs (688 kg)	71 lbs (32 kg)	1588 lbs (720 kg)
WE 17 x 17	1466 lbs (665 kg)	39 lbs (18 kg)	1505 lbs (683 kg)

Appendix H of the applicant's Safety Analysis Report (SAR (Ref. 1)) provides justification for a long cavity canister design and Section H.1.1 refers to the NUHOMS®-24P Topical Report (TR) (Ref. 2) for qualification, including the structural requirements. The TR Section 3 provides information on the maximum fuel assembly weights used for the design of the cask. Table 3.1-1 of the TR and Table 3.1-1 of the SAR specify the maximum assembly weight as 1682 lbs (763 kg).

As noted above, the revised maximum weight of the fuel assembly with BPRAs is 1588 lbs (720 kg) and is less than the original maximum design weight of 1682 lbs (763 kg). Therefore, the staff concludes that the weight of the cask w/BPRAs is enveloped by the approved design basis of the DSC, and additional weight of the BPRAs will not adversely impact the DSC design.

Center of Gravity of the Mass

The applicant stated (Ref. 4) that the center of gravity of a loaded cask with and without BPRAs is located within 1.0 inch and that this is insignificant compared to the cask height of approximately 200 inches. The staff agrees with the assessment and the conclusion that there is no adverse impact on the DSC design.

Temperature Changes

There are no changes in design temperatures due to the addition of BPRAs. Therefore, the DSC design is not affected.

Pressure Changes

The applicant calculated the internal pressure changes in the DSC due to the addition of BPRAs and states that the off-normal pressure increases by 6 percent. The staff reviewed the design basis SAR and concludes that the DSC design has sufficient margins to accommodate the 6 percent increase in stresses due to the internal pressure.

In addition, the governing load combinations with the off-normal pressure are ASME III Level B conditions, for which the SAR limits the stresses to ASME III Level A stress allowables. Since the Level B stress allowables are 10 percent higher than the Level A allowables, the present DSC design has additional margins to accommodate the 6 percent increase in stresses due to off-normal pressure.

Based on the above, the staff concludes that the pressure changes do not adversely affect the DSC structural design.

The staff reviewed applicable sections of the SAR to evaluate the effects, if any, of intimate contact between the BPRAs and the internal hardware. In particular, materials selection and chemical reactions were reviewed. The staff concluded that the materials selected for the BPRAs are inert to the DCS environment and are not expected to be reactive with the cask internal components under all conditions of service during the licensing period.

Based on the review of the statements and representations in the application, as supplemented, the staff concludes that the addition of BPRAs for B&W 15x15 and Westinghouse 17x17 fuel assembly types to the authorized contents of the NUHOMS®-24P will not adversely affect the DSC and the NUHOMS®-24P storage system, and that the structural performance of the NUHOMS® storage system meets the structural requirements of 10 CFR Part 72.

2.0 THERMAL

The applicant requested approval to add BPRAs for the B&W 15x15 and Westinghouse 17x17 fuel assembly types to the authorized contents of the NUHOMS®-24P long cavity DSC. The applicant stated that no recalculation of any of the original thermal analysis for the cask was necessary to qualify BPRAs. The staff reviewed the applicant's analysis of decay heat in the NUHOMS®-24P DSC and found the analysis to be adequate. The results are presented below.

Decay Heat

The applicant calculated the maximum heat generation of the BPRA components to be 8 watts from each assembly. The applicant also created a new fuel qualification table to address the addition of the heat generated by the BPRAs. The total decay heat of each spent fuel assembly is taken to be that generated by the fuel plus the decay heat generated by the BPRAs. The criteria for fuel cladding temperature limit remains the same, but the allowable decay heat from the fuel rods in an assembly is reduced by 8 watts to accommodate the BPRAs. Therefore, the applicant concluded that the results from the thermal analysis in Chapter 8 of the SAR (Ref. 1) for normal, off-normal, and accident conditions remain valid for the maximum design basis decay heat of 1 kW per assembly, including the BPRA contribution. The staff reviewed the applicant's analysis and agrees with this conclusion and finds that the NUHOMS®-24P storage system meets the thermal requirements of 10 CFR Part 72.

3.0 CONTAINMENT

The applicant provided an evaluation of the pressure inside the DSC under normal, off-normal, and accident conditions, based on the addition of gasses to the DSC from the BPRAs. The applicant calculated the maximum number of moles of Helium gas that could be generated in each BPRA during reactor operations. The applicant assumed that 30% of the Helium gas produced in the Aluminum Oxide (Al_2O_3) composite (within the BPRA rods) is released into the BPRA rod void volume and is available for release into the DSC cavity in the case of a BPRA rod rupture. The applicant then calculated the total number of moles of Helium gas for 24 B&W BPRAs (with 16 rods each) inside a DSC. The B&W 15x15 BPRAs bound the Westinghouse 17x17 BPRAs for the DSC internal pressure calculations.

For normal and off-normal conditions, the applicant assumed a Helium gas release from 1% and 10% of the BPRA rods, respectively. For hypothetical accident conditions, the applicant assumed a release of Helium gas from 100% of the BPRA rods. The applicant analyzed pressures for 40 and 45 GWd/MTU burnup fuel. A summary of the results of the applicant's analysis is shown below.

Summary of DSC Internal Pressure Evaluation Results for Normal, Off-Normal, and Accident Conditions

Operating Condition	Limiting Case Description	40 GWd/MTU Burnup Pressure (psig)	45 GWd/MTU Burnup Pressure (psig)	Design Basis Pressure (psig)
Normal	DSC in Cask, 100°F	7.1	6.8	10
Off-Normal	DSC in Cask, 100°F	10.6	10.2	41
Accident	Blocked HSM Vents, 125°F	56.1	59.8	60

For the normal and off-normal cases, the 40 GWd/MTU burnup fuel is bounding. For the accident case, the 45 GWd/MTU burnup fuel is bounding. The pressures calculated in all cases are below the design basis pressures for the 24P long cavity DSC.

The staff reviewed the applicant's calculations and determined that the calculations were based on conservative assumptions for the amount of Helium produced, the percentage of Helium released into the void volume (30%), and the number of BPRAs in a single DSC (24). Based on the staff's review, the staff agrees with the applicant's conclusion that the DSC internal pressure with BPRAs for normal, off-normal, and accident conditions does not exceed the design basis pressure values presented in the previous analysis for the NUHOMS® storage system. Based on this analysis, storage of BPRAs in the NUHOMS®-24P long cavity DSC meets the containment requirements of 10 CFR Part 72.

4.0 SHIELDING

The effect on dose rates due to the inclusion of 24 irradiated BPRAs into B&W 15x15 or Westinghouse 17x17 fuel assembly types, loaded into the NUHOMS®-24P, was re-analyzed by the applicant. The applicant used the ORIGEN2 computer code to calculate the source terms for the BPRAs and determined that the bounding BPRA was the B&W 15x15 having burned 36,000 MWd/MTU and 3.3% w/o U-235. The cooling times were then recalculated for a cask containing 24 original design basis fuel assemblies loaded with 24 design basis BPRAs. The calculated increase in surface dose rate resulting from the addition of BPRAs remains within the bounds of the previous analysis. The applicant also provided data showing that few BPRAs are irradiated for more than one cycle. Loading 24 design basis BPRAs in a single cask is not probable, therefore, the calculated dose rates are conservative.

The staff confirmed the applicant's conclusion by reviewing the submitted calculations. Additionally, an independent review was conducted by generating source terms for the B&W 15x15 BPRA using SAS2H from the SCALE 4.4 suite of computer codes. The staff's analysis agreed with the applicant's conclusions. The BPRAs are limited to maximum burnups and minimum cooling times as shown in Table 1-2c of the revised Technical Specifications. Based on the confirmatory review of the information provided by the applicant, the staff has reasonable assurance that the NUHOMS®-24P will continue to meet the shielding requirements of 10 CFR Part 72.

5.0 CRITICALITY

The applicant performed analyses to determine the effect of including BPRAs with the B&W 15x15 Mark B and Westinghouse 17x17 (Standard and OFA) PWR fuel assemblies in the NUHOMS®-24P DSC. For criticality control, the casks are filled with borated water during loading and unloading, and the BPRA rods displace the borated water. Calculations were performed with the CSAS26 code and 44-group neutron cross section in the SCALE suite of codes.

The applicant modeled an infinite array of fuel assemblies with guide sleeves in a typical basket pitch and found that for water densities from 0 to 0.85 g/cc, the fuel assemblies are less reactive with BPRAs inserted than without. For the water densities from 0.85 to 1 g/cc, k_{eff} increases slightly when BPRAs are inserted. For this upper range of water densities, the applicant modeled an infinitely long cask reflected by 12 inches of borated water surrounded by an infinite layer of fresh water. In both models described above, the BPRAs were represented by filling the inside of the fuel assembly guide and instrument tubes with $^{11}B_4C$. This simulated the presence of the BPRAs without taking credit for any unburned ^{10}B.

The results of the applicant's analyses show that for water densities from 0 to 0.85 g/cc, the addition of BPRAs causes the reactivity of the fuel assemblies to be less than the previously approved fuel assemblies without BPRAs. For water densities from 0.85 to 1 g/cc, the results of the cask model calculations when the BPRAs are included gave a peak k_{eff} of 0.9203 which is less than the maximum k_{eff} in the previous application. Since the cask model had used nominal mechanical tolerances and fuel assembly positions in the basket, the applicant also performed a calculation with the tolerances and positions at the values which maximized k_{eff}. This calculation was at the point of maximum k_{eff} for the nominal cask model and gave a k_{eff} value below 0.95 when adjusted for bias and uncertainty.

The staff reviewed the applicant's analyses and methods and performed independent calculations. The independent calculations agree with the trends in the analysis. Since the applicant modeled the fuel rods as dry in the pellet-to-clad gap rather than being flooded, the staff's calculations included cases to assess this assumption. Staff results found that reactivity is maximized when the gap is dry.

Based on its review of the statements and representations in the application and on its independent calculations, the staff has reasonable assurance that the proposed amendment meets the criticality safety requirements of 10 CFR Part 72.

CONCLUSIONS

The staff performed a detailed safety evaluation of the proposed Certificate of Compliance amendment request and found that the addition of the BPRAs to the B&W 15x15 and Westinghouse 17x17 fuel assembly types does not reduce the safety margin. In addition, the staff has determined that the storage of BPRAs in the NUHOMS®-24P storage system does not pose any increased risk to public health and safety. The remaining areas of review addressed in NUREG 1536,"Standard Review Plan for Dry Cask Storage Systems," January 1997, are not affected by the applicant's amendment request. Based on the statements and representations contained in the applicant's SAR and the conditions in the Certificate of Compliance, the staff concludes that the addition of BPRAs to the B&W 15x15 and Westinghouse 17x17 fuel assembly types to the authorized contents of the NUHOMS®-24P storage system meets the requirements of 10 CFR Part 72.

References

1. Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel, Revision 4A, NUH003.0103, June 1996.
2. Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS 24-P, NUH002.0103, Revision 2A, April 1991.
3. Nonfuel Assembly Database, Notz. K.J., and Moore, R.S., DOE, Chattanooga, TN.
4 Revision 1 of the Application for Amendment No. 3 of NUHOMS® CoC No. 1004 for Dry Spent Fuel Storage Casks, November 29, 1999.