ATTACHMENT 3

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ANUH-01.0150
Revision 3

SAFETY ANALYSIS REPORT
FOR THE
STANDARDIZED ADVANCED NUHOMS®
HORIZONTAL MODULAR STORAGE SYSTEM
FOR IRRADIATED NUCLEAR FUEL

NON-PROPRIETARY
FOR INFORMATION ONLY

By
Transnuclear West Inc., (TN West)
Fremont, CA

Revision 3
July 2001

Executive Summary

Revisions 1, *2 and 3* of this Safety Analysis Report incorporate changes based on the *initial and supplemental* responses to NRC Request for Information.

This Safety Analysis Report provides the generic safety analysis for the standardized Advanced NUHOMS®[1] System for dry storage of light water reactor spent nuclear fuel assemblies. This system provides for the safe dry storage of spent fuel in a passive Independent Spent Fuel Storage Installation (ISFSI) which fully complies with the requirements of 10CFR72 and ANSI 57.9.

This Safety Analysis Report describes the design and forms the basis for generic NRC certification of the standardized Advanced NUHOMS® System and will be used by 10CFR50/10CFR72 general license holders in accordance with 10CFR72 Subparts K and L. It is also suitable for reference in 10CFR72 site specific license applications.

The principal features of the standardized Advanced NUHOMS® System which differ from the previously approved NUHOMS® Systems are:

1. Modification to the C of C No. 1004 HSM (development of Advanced HSM, AHSM) to support qualification for sites with high seismic spectra and/or requirements for a significant reduction in ISFSI dose (e.g., due to congested reactor sites).

2. The AHSM configuration requires a minimum of three AHSMs tied together to limit sliding and uplift during a seismic event.

3. The Dry Shielded Canister used in this application, the 24PT1-DSC, is a modification to the FO-DSC associated with C of C No. 9255 (also used as a transfer cask under Rancho Seco Materials License SNM-2510, Docket No. 72-11) with additional provisions allowing storage of intact and damaged fuel assemblies, along with control components in a single DSC.

The NUHOMS® System provides long-term interim storage for spent fuel assemblies which have been out of the reactor for a sufficient period of time and which comply with the criteria set forth in this Safety Analysis Report. The fuel assemblies are confined in a helium atmosphere by a dry shielded canister. The canister is protected and shielded by a massive reinforced concrete module. Decay heat is removed from the canister and the concrete module by a passive natural draft convection ventilation system.

[1] NUHOMS® is a registered trademark of Transnuclear West Inc.

NUHOMS® System is a totally passive installation that is designed to provide shielding and safe confinement of spent fuel for a range of postulated accident conditions and natural phenomena.

The NUHOMS® System OS197 Cask (C of C No. 1004) is used for transfer operations for the Advanced NUHOMS® System. Evaluations of this cask in this application is limited to those areas where existing analysis (in the aforementioned C of C) is not bounding.

TABLE OF CONTENTS
(continued)

Page

4.2 Summary of Thermal Properties of Materials 4.2-1

4.3 Specifications for Components 4.3-1

4.4 Thermal Evaluation for Normal Conditions of Storage and Transfer 4.4-1

4.4.1 Overview of Thermal Analysis for Normal Conditions of Storage and Transfer 4.4-1
4.4.2 Thermal Model of the 24PT1-DSC Inside the AHSM 4.4-1
4.4.3 Thermal Model of 24PT1-DSC in the Transfer Cask 4.4-7
4.4.4 24PT1-DSC Basket Thermal Model 4.4-8
4.4.5 Test Model 4.4-12
4.4.6 Maximum Temperatures 4.4-12
4.4.7 Minimum Temperatures 4.4-12
4.4.8 Maximum Internal Pressure 4.4-13
4.4.9 Maximum Thermal Stresses 4.4-14
4.4.10 Evaluation of System Performance for Normal Conditions of Storage and Transfer 4.4-14

4.5 Thermal Evaluation for Off-Normal Conditions 4.5-1

4.5.1 Overview of Off-Normal Analysis 4.5-1
4.5.2 Thermal Analysis Results 4.5-1
4.5.3 Maximum Pressure 4.5-1
4.5.4 Evaluation of System Performance for Off-Normal Conditions of Storage and Transfer 4.5-2

4.6 Thermal Evaluation for Accident Conditions 4.6-1

4.6.1 Accident Ambient Conditions 4.6-1
4.6.2 Blockage of AHSM Inlet and Outlet Vents 4.6-1
4.6.3 Transfer Cask Loss of Neutron Shield and Sunshade 4.6-2
4.6.4 Fire Accident Evaluation 4.6-3
4.6.5 Flood Accident 4.6-5
4.6.6 Maximum Pressure 4.6-5
4.6.7 Evaluation of System Performance for Accident Conditions of Storage and Transfer 4.6-5

4.7 Thermal Evaluation for Loading/Unloading Conditions 4.7-1

4.7.1 Vacuum Drying Thermal Analysis 4.7-1
4.7.2 Pressure During Unloading of Cask 4.7-2
4.7.3 Cask Heatup Analysis 4.7-2
4.7.4 Pressure During Loading of Cask 4.7-3

4.8 Supplemental Information 4.8-1

4.8.1 References 4.8-1
4.8.2 Computer Code 4.8-3

TABLE OF CONTENTS
(continued)

Page

5. SHIELDING EVALUATION 5-1

5.1 Discussion and Results 5.1-1

5.2 Source Specification 5.2-1

5.2.1 Gamma Sources 5.2-3

5.2.2 Neutron Source 5.2-4

5.2.3 Evaluation of Effect of Uncertainty in Minimum Initial Enrichment 5.2-4

5.3 Model Specification 5.3-1

5.3.1 Description of the Radial and Axial Shielding Configurations 5.3-1

5.3.2 Shield Regional Densities 5.3-3

5.4 Shielding Evaluation 5.4-1

5.4.1 Computer Programs 5.4-1

5.4.2 Flux-to-Dose-Rate Conversion 5.4-2

5.5 Supplemental Information 5.5-1

5.5.1 References 5.5-1

6. CRITICALITY EVALUATION 6-1

6.1 Discussion and Results 6.1-1

6.2 Spent Fuel Loading 6.2-1

6.2.1 WE 14x14 SC Fuel Assemblies 6.2-1

6.2.2 WE 14x14 MOX Fuel Assemblies 6.2-1

6.3 Model Specification 6.3-1

6.3.1 Description of Criticality Analysis Model 6.3-1

6.3.2 Neutron Absorber Panel Material Efficacy 6.3-6

6.4 Criticality Calculation 6.4-1

6.4.1 Calculational Method 6.4-1

6.4.2 Fuel Loading Optimization 6.4-1

6.4.3 Criticality Results 6.4-4

6.4.4 Evaluation of Effect of Uncertainty in Maximum Initial Enrichment 6.4-5

6.4.5 Effect of Fuel Parameter Tolerances on Reactivity 6.4-5

6.5 Critical Benchmark Experiments 6.5-1

6.5.1 Benchmark Experiments and Applicability 6.5-1

6.5.2 Results of the Benchmark Calculations 6.5-2

6.6 Supplemental Information 6.6-1

6.6.1 References 6.6-1

6.6.2 KENO Input Files 6.6-2

7. CONFINEMENT 7.1-1

7.1 Confinement Boundary 7.1-1

7.1.1 Confinement Vessel 7.1-1

7.1.2 Confinement Penetrations 7.1-2

LIST OF TABLES
(continued)

		Page
Table 4.1-4	Component Minimum and Maximum Temperatures in the Advanced NUHOMS® System (Storage or Transfer Mode) for Off-Normal Conditions	4.1-7
Table 4.1-5	Component Minimum and Maximum Temperatures in the Advanced NUHOMS® System (Storage and Transfer) for Accident Conditions	4.1-8
Table 4.1-6	Limiting Canister Heat Loads for DSC Components	4.1-9
Table 4.4-1	Advanced NUHOMS® System Bulk Air Temperatures	4.4-15
Table 4.4-2	Heat Generations Used in the Thermal Model of the 24PT1-DSC in the AHSM	4.4-16
Table 4.4-3	AHSM Thermal Analysis Results Summary	4.4-17
Table 4.4-4	24PT1-DSC Shell Results, 16 and 14 kW at Hottest Cross Section	4.4-18
Table 4.4-5	24PT1-DSC Maximum Shell Temperatures at 24 kW	4.4-19
Table 4.4-6	24PT1-DSC Basket Temperature Results	4.4-20
Table 4.4-7	Maximum Fuel Cladding Temperature Results, 16 kW	4.4-21
Table 4.4-8	Summary of Cases Considered for Thermal Stress Analysis	4.4-22
Table 4.4-9	Fuel Assembly Characteristics for Pressure Analysis	4.4-23
Table 4.4-10	Control Component Characteristics for Pressure Analysis	4.4-24
Table 4.4-11	24PT1-DSC Cavity Pressure Analysis Summary	4.4-25
Table 4.7-1	Steady State Vacuum Drying Results	4.7-4
Table 4.7-2	Transient Vacuum Drying Results for the Spacer Disc	4.7-5
Table 4.7-3	Summary of Water Heatup Calculation	4.7-6
Table 5.1-1	Advanced NUHOMS® System Shielding Materials	5.1-2
Table 5.1-2	Summary AHSM Dose Rates	5.1-3
Table 5.1-3	Transfer Cask (Loading/Unloading/Transfer Operations) Side Dose Rate Summary	5.1-4
Table 5.1-4	Transfer Cask (Loading/Unloading/Transfer Operations) Top End Dose Rate Summary	5.1-5
Table 5.1-5	Transfer Cask (Loading/Unloading/Transfer Operations) Bottom End Dose Rate Summary	5.1-6
Table 5.2-1	Flux Factor By Fuel Assembly Region	5.2-6
Table 5.2-2	WE 14x14 SC Assembly Materials and Masses	5.2-7
Table 5.2-3	WE 14x14 MOX Assembly Materials and Masses	5.2-8
Table 5.2-4	Thimble Plug Assemblies and Masses	5.2-9
Table 5.2-5	RCCA Materials and Masses	5.2-10
Table 5.2-6	Neutron Source Assembly Materials and Masses	5.2-11
Table 5.2-7	CASK-81 Neutron and Gamma Group Structure	5.2-12
Table 5.2-8	SAS2H Gamma Sources for 45 GWd/MTU, 10-Year Cooled WE 14x14 SC Fuel Per Fuel Assembly	5.2-13
Table 5.2-9	SAS2H Gamma Sources for 40 GWd/MTU, 10-Year Cooled WE 14x14 SC Fuel Per Fuel Assembly	5.2-14

LIST OF TABLES
(Continued)

		Page
Table 5.2-10	SAS2H Gamma Sources for 35 GWd/MTU, 10-Year Cooled WE 14x14 SC Fuel Per Fuel Assembly	5.2-15
Table 5.2-11	SAS2H Gamma Sources for 25 GWd/MTU, 20-Year Cooled WE 14x14 MOX Fuel Per Fuel Assembly	5.2-16
Table 5.2-12	Gamma Source Term for Thimble Assembly, Eleven Cycles In-core, 10 Year Cooled Per Assembly	5.2-17
Table 5.2-13	Gamma Source Term for RCCA, Eleven Cycles In-core, 10 Year Cooled Per Assembly	5.2-18
Table 5.2-14	Gamma Source Term for Neutron Source Assemblies, Four Cycles In-core, 10 Year Cooled Per Assembly	5.2-19
Table 5.2-15	Design Basis Volumetric Gamma Source Terms	5.2-20
Table 5.2-16	Calculated Total Neutron Sources per Fuel Assembly	5.2-21
Table 5.2-17	Design Basis Volumetric Neutron Source Term	5.2-22
Table 5.2-18	Elemental Composition of LWR Structural Materials	5.2-23
Table 5.2-19	Comparison of Design Basis Shielding Analysis Source Term to Fuel Qualification Table Source Terms	5.2-24
Table 5.3-1	Materials Composition and Atom Number Densities	5.3-5
Table 5.3-2	Density Calculations for Concrete	5.3-6
Table 5.3-3	Materials Composition and Atom Densities During Decontamination and Wet Welding Stage Calculation	5.3-7
Table 5.3-4	ANSI Standard-6.1.1-1977 Flux-to-Dose Factors	5.3-8
Table 5.4-1	Normalized Conservative Burn-Up Shape on WE 14×14 Fuel Assembly	5.4-3
Table 6.1-1	Summary of Limiting Criticality Evaluations for the WE 14x14 SC Fuel Assemblies and the WE MOX Fuel Assemblies	6.1-3
Table 6.2-1	Design Parameters for Criticality Analysis of the WE 14x14 SC Fuel Assemblies	6.2-2
Table 6.2-2	Design Parameters for Criticality Analysis of the WE 14x14 MOX Fuel Assemblies	6.2-3
Table 6.3-1	Geometrical Parameters Used in the Criticality Analysis	6.3-8
Table 6.3-2	Design Parameters for WE 14x14 SC Fuel Assembly Criticality Analysis	6.3-11
Table 6.3-3	Design Parameters for WE 14x14 MOX Fuel Assembly Criticality Analysis	6.3-12
Table 6.4-1	Results for the WE 14x14 SC Fuel Assembly	6.4-6
Table 6.4-2	Results for the WE 14x14 MOX Fuel Assembly	6.4-9
Table 6.4-3	Results for the Damaged Fuel Assemblies	6.4-12
Table 6.4-4	Bounding Criticality Analysis Analyzed for 4.05 weight % ^{235}U	6.4-14
Table 6.4-5	Clad OD Sensitivity Evaluation	6.4-14
Table 6.4-6	*Clad Thickness Sensitivity Evaluation*	*6.4-14a*
Table 6.4-7	*Pellet Diameter Sensitivity Evaluation*	*6.4-14a*
Table 6.5-1	Benchmark Results	6.5-3
Table 6.5-2	USL-1 Results	6.5-6

LIST OF FIGURES
(continued)

		Page
Figure 3.6-11	Concrete Components in the Lower Base Storage Block	3.6-53
Figure 3.6-12	Concrete Components in the Upper Base Storage Block	3.6-54
Figure 3.6-13	Concrete Components in the Top Shield Block	3.6-55
Figure 3.6-14	Symbolic Notations of Force and Moment Capacities (Also for Computed Forces and Moments)	3.6-56
Figure 3.6-15	Components of AHSM Support Structure	3.6-57
Figure 4.4-1	Illustration of Air Flow Paths Through AHSM	4.4-26
Figure 4.4-2	AHSM HEATING7 Model; Cross Section Along x=0.0	4.4-27
Figure 4.4-3	Details 1, 2 and 3 From AHSM HEATING7 Model	4.4-28
Figure 4.4-4	Sections A-A and B-B From AHSM HEATING7 Model	4.4-29
Figure 4.4-5	AHSM HEATING7 Model; 24PT1-DSC and Heat Shield	4.4-30
Figure 4.4-6	Cross Section of 24PT1-DSC Basket Model	4.4-31
Figure 4.4-7	24PT1-DSC Basket Model; Fuel Regions 1 and 2 with Surrounding Regions	4.4-32
Figure 4.4-8	Simplified Axial View of 24PT1-DSC Basket Model	4.4-33
Figure 4.6-1	OS197 Cask and 24PT1-DSC Response to Fire Accident	4.6-6
Figure 4.7-1	Vacuum Drying Transient Response of Spacer Disc	4.7-7
Figure 4.7-2	Results of the Water Heatup Calculations	4.7-8
Figure 5.1-1	Advanced NUHOMS® System Shielding Configuration	5.1-7
Figure 5.1-2	Dry Shielded Canister Shielding Configuration	5.1-8
Figure 5.1-3	Right Elevation Cross Section View of AHSM	5.1-9
Figure 5.1-4	Shielding Configuration of the OS197 Transfer Cask	5.1-10
Figure 5.4-1	DORT R-Z Floor Model	5.4-6
Figure 5.4-2	DORT R-Z Roof Model	5.4-9
Figure 5.4-3	DORT X-Z Midplane Model	5.4-12
Figure 5.4-4	Zone Number by Material in TC Model at Transfer-Storage Stage	5.4-16
Figure 5.4-5	Zone Number by Material in TC Model at Decontamination Stage	5.4-17
Figure 5.4-6	Zone Number by Material in TC Model at Wet Welding Stage	5.4-18
Figure 5.4-7	Zone Number by Material in TC Model at Dry Welding Stage	5.4-19
Figure 5.4-8	AHSM Roof DORT Shielding Analysis Model	5.4-20
Figure 5.4-9	AHSM Floor DORT Shielding Analysis Model	5.4-21
Figure 5.4-10	AHSM Side DORT Shielding Analysis Model	5.4-22
Figure 5.4-11	AHSM Front/Back DORT Shielding Analysis Model	5.4-23
Figure 5.4-12	OS197 DORT Shielding Analysis Model	5.4-24
Figure 6.3-1	KENO Model of the 24PT1-DSC Basket	6.3-13
Figure 6.3-2	Exploded View of KENO Model	6.3-14
Figure 6.3-3	Structure of KENO Model – UNIT 33	6.3-15
Figure 6.3-4	Structure of KENO Model – UNIT 34	6.3-16
Figure 6.3-5	Cross Section of the Design Basis WE 14x14 SC Fuel Assembly	6.3-17

LIST OF FIGURES
(continued)

		Page
Figure 6.3-6	Cross Section of the Design Basis WE 14x14 MOX Fuel Assembly	6.3-18
Figure 6.3-7	Guidesleeve and Fuel Assembly Cross Section	6.3-19
Figure 6.3-8	Fuel Assemblies Located in the Inner Guidesleeve Corner Closest to the DSC Centerline (Assembly in Case)	6.3-20
Figure 6.3-9	The Fuel Assemblies Moved Radially Outwards from the Center of the DSC (Assembly Out Case)	6.3-21
Figure 6.3-10	The Fuel Assemblies Are Moved Towards the Upper Left Corner of Each Guidesleeve Assembly Upper Left Corner Case	6.3-22
Figure 6.4-1	Rod Pitch Study: Figure Shows Maximum Rod Pitch Allowed	6.4-15
Figure 6.4-2	Single-Ended Rod Shear Example	6.4-16
Figure 6.4-3	Double-Ended Rod Shear Example	6.4-17
Figure 7.1-1	24PT1-DSC Confinement Boundary Welds	7.1-3
Figure 8.1-1	Advanced NUHOMS® System Loading Operations Flow Chart	8.1-12
Figure 8.1-2	Cask Loading Operations	8.1-15
Figure 8.2-1	Advanced NUHOMS® System Retrieval Operations Flow Chart	8.2-7
Figure 10.2-1	Total Annual Exposure from a Single AHSM as a Function of Distance	10.2-15
Figure 10.2-2	Total Annual Exposure from a 2x10 AHSM Array as a Function of Distance	10.2-16
Figure 11.2-1	LS-DYNA AHSM Seismic Stability Model	11.2-34
Figure 11.2-2	Horizontal Time History H1, Set 1 (Taiwan, 1999)	11.2-35
Figure 11.2-3	Horizontal Time History H2 Set 1 (Taiwan, 1999)	11.2-36
Figure 11.2-4	Vertical Time History V, Set 1 (Taiwan 1999)	11.2-37
Figure 11.2-5	Horizontal Time History H1, Set 2 (Tabas 1978)	11.2-38
Figure 11.2-6	Horizontal Time History H2, Set 2 (Tabas 1978)	11.2-39
Figure 11.2-7	Horizontal Time History V, Set 2 (Tabas 1978)	11.2-40
Figure 11.2-8	Horizontal Time History H1, Set 3 (Landers/Lucern 1992)	11.2-41
Figure 11.2-9	Horizontal Time History H2, Set 3 (Landers/Lucern 1992)	11.2-42
Figure 11.2-10	Vertical Time History V, Set 3 (Landers/Lucern 1992)	11.2-43
Figure 11.2-11	AHSM Sliding Response X- Direction	11.2-44
Figure 11.2-12	AHSM Sliding Response Y-Direction	11.2-45
Figure 11.2-13	AHSM Uplift Response Z-Direction	11.2-46
Figure 11.2-14	24PT1-DSC Stability Evaluation	11.2-47
Figure 11.2-15	AHSM Dimension for Missile Impact Stability Analysis	11.2-48
Figure 11.2-16	Analysis Case 2 (TH1-2): AHSM Sliding Response X-Direction	11.2-49
Figure 11.2-17	Analysis Case 3 (TH2-1): AHSM Sliding Response X-Direction	11.2-50
Figure 11.2-18	Analysis Case 4 (TH2-2): AHSM Sliding Response X-Direction	11.2-51
Figure 13.1-1	Project Organization Chart	13.1-3

LIST OF EFFECTIVE PAGES

All pages Revision 1 with the following exceptions:

PAGE	REVISION
Cover Page	3
Executive Summary, page iii	3
Table of Contents, page viii	2
Table of Contents, page xi	3
List of Tables, page xix	3
List of Figures, page xxiii	3
List of Effective Pages, page xxiv	3
2.1-2	3
2.1-2a	3
4.1-6	3
4.1-7	3
4.1-8	3
4.4-1	3
4.4-4	3
4.4-4a	3
4.4-17	3
6.4-5	3
6.4-14	3
6.4-14a	3
6.4-15	3
8.1-1	3
8.1-3	3
12.2-1	2
12.2-2	2
12.3-6	3
12.4-1	3
12.4-1a	DELETED
12.4-3	3
12.4-4	3

2. PRINCIPAL DESIGN CRITERIA

2.1 Spent Fuel to be Stored

The Advanced NUHOMS® System components have currently been designed for the storage of 24 Westinghouse 14x14 (WE 14x14) PWR fuel assemblies. Authorization for storage of this payload only is requested. Additional payloads may be defined in future amendments to this application.

This payload consists of intact and/or damaged WE 14x14 assemblies including stainless steel or zircaloy cladding, UO_2 or mixed-oxide (MOX) fuel pellets, with or without integral control components, and/or damaged fuel.

The thermal and radiological characteristics for the PWR spent fuel were generated using the SCALE computer code package. Spent fuel with various combinations of burnup, decay heat, enrichment, and cooling time can be stored in the 24PT1-DSC as long as the values for decay heat, gamma and neutron sources, including spectra, remain within the design limits specified in Table 2.1-2. Free volume in the 24PT1-DSC cavity and fuel rod fill gas pressures are addressed in Chapter 4. Specific gamma and neutron source spectra are given in Chapter 5.

Although analyses in this SAR are performed only for the design basis fuel, any other intact PWR fuel which falls within the geometric, thermal, and nuclear limits established for the design basis fuel can be stored in the 24PT1-DSC.

2.1.1 Detailed Payload Description

This payload consists of 24 PWR WE 14x14 fuel assemblies, either UO_2 (stainless steel clad) or mixed-oxide (zircaloy clad). Also included in the payload are fuel assemblies utilizing boron coated fuel pellets, Integral Fuel Burnable Absorber (IFBA) assemblies. Each 24PT1-DSC can accommodate a maximum of four damaged WE 14x14 stainless steel clad (SC) fuel assemblies, with the remaining assemblies intact WE 14x14 SC assemblies. One damaged MOX assembly may also be accommodated, with the remaining assemblies (up to 23) intact WE 14x14 SC fuel assemblies. Additionally, each 24PT1-DSC accommodates the storage of integral control components in the WE 14x14 assemblies as described below. The physical characteristics for the PWR fuel assembly types are shown in Table 2.1-1. The fuel to be stored in the 24PT1-DSC is limited to fuel with a maximum initial enrichment of 4.05 weight % U-235. The maximum allowable burnup is given as a function of initial fuel enrichment but does not exceed 45,000 MWd/MTU. The minimum cooling time is ten years *for SC fuel assemblies and 20 years for MOX fuel assemblies.*

Table 2.1-3 presents the thermal and radiological source terms for the Non-Fuel Assembly Hardware (NFAH) which includes Rod Cluster Control Assemblies (RCCAs), Neutron Source Assemblies (NSAs) or Thimble Plug Assemblies (TPAs). These values are consistent with the cumulative exposures and cooling times of the fuel assemblies. The gamma spectra for the RCCAs, NSAs and TPAs are presented in Chapter 5.

The stainless steel clad fuel assembly without RCCAs, NSAs or TPAs is used for stability calculations (wind, tornado, missiles, flood and seismic) due to the lighter weight of the contents and the slightly higher center of gravity.

The 24PT1-DSC may also include two empty fuel slots and/or multiple dummy fuel assemblies. Dummy assemblies have approximately the same weight and center of gravity as a SC fuel assembly. The empty slots must be located at symmetrical locations about the 0-180° and 90-270° axes. The 24PT1-DSC center of gravity, under all loading conditions (including two empty slots and/or dummy fuel assemblies) must be maintained within 0.1" of its design basis (symmetrical loading) radial position to ensure that deviations in the center of gravity are minimal in order to maintain the validity of the structural analyses.

The reduction in weight due to two empty slots will tend to slightly reduce stresses for the 24PT1-DSC, transfer cask and AHSM for the various load conditions. Empty slots and/or multiple dummy fuel assemblies will not have any other impact on the structural analysis based on the controls imposed on the dummy assembly size and cg, and the cg location for a 24PT1-DSC with empty slots.

The effect of two empty slots and/or multiple dummy fuel assemblies on the thermal analysis is bounded by the 24 fuel assembly analyses since these configurations result in heat loads less than the design basis, thus reducing the 24PT1-DSC, cask and AHSM temperatures. The effect of these configurations on the 24PT1-DSC thermal stresses may result in a slight increase in spacer disc thermal stresses if the empty slots and/or dummy assemblies are located on the outer perimeter of the 24PT1-DSC. However, this effect is localized and will not impact the controlling 24PT1-DSC structural analysis load combinations (these load conditions exclude thermal stresses).

The effect of two empty slots and/or multiple dummy fuel assemblies on the shielding analysis is bounded by the 24 fuel assembly analyses since the neutron and gamma source terms are reduced.

The effect of two empty slots and/or multiple dummy fuel assemblies on the criticality analysis is bounded by the analyses for 24 fuel assemblies since the reduction in the source of neutrons has a greater effect than the increased moderation for the undermoderated fuel assemblies. The volume in which the dummy assembly or empty slot is located sees the same number of neutrons entering the region from adjacent assemblies but does not generate additional neutrons since no fuel is present in the volume. The effect of an increase in moderator volume is mitigated by the fact that the moderated neutrons must pass through poison plates before they can interact with fuel in adjacent guidesleeves. This effect can be seen in the results provided in SAR Table 6.4-1 for the Normal Operating Condition, Assembly Position Case. A comparison of the case in which the fuel assemblies are positioned inward towards the centerline versus the case in which the fuel assemblies are positioned radially outward from the center, indicates that the case with fuel outward, which increases moderator between the assemblies, results in a reduction in $k_{eff}+2\sigma$ from 0.8659 to 0.8404. This increase in moderation and reduction in k_{eff} is smaller than that which would result from an empty slot or dummy fuel assembly and therefore bounds the empty slot and dummy assembly configuration described above.

Damaged fuel may include assemblies with known or suspected cladding defects greater than hairline cracks or pinhole leaks, up to and including broken rods, portions of broken rods and rods with missing sections. Damaged fuel assemblies shall be encapsulated in individual failed fuel cans that confine any loose material and gross fuel particles to a known, subcritical volume during normal, off-normal and accident conditions and to facilitate handling and retrievability. The criticality analysis provided in Chapter 6 requires that no more than 14 fuel pins (rods) in each assembly exhibit damage. A visual inspection of assemblies will be performed prior to placement of the fuel in the 24PT1-DSC, which may then be placed in storage or transported anytime thereafter without further fuel inspection.

Table 4.1-2
Temperature Variation for Extreme Summer Ambient Conditions

Time, Hour	% Daily Range(1)	Normal (°F)	Off-Normal (°F)
1	87	86.6	93.5
2	92	85.6	92.2
3	96	84.8	91.1
4	99	84.2	90.3
5	100	84.0	90.0
6	98	84.4	90.5
7	93	85.4	91.9
8	84	87.2	94.3
9	71	89.8	97.8
10	56	92.8	101.9
11	39	96.2	106.5
12	23	99.4	110.8
13	11	101.8	114.0
14	3	103.4	116.2
15	0	104.0	117.0
16	3	103.4	116.2
17	10	102.0	114.3
18	21	99.8	111.3
19	34	97.2	107.8
20	47	94.6	104.3
21	58	92.4	101.3
22	68	90.4	98.6
23	76	88.8	96.5
24	82	87.6	94.9
Averages		92.7	101.8

(1) Percentage of daily temperature range (see Section 4.1.2 for daily temperature range used for normal and off-normal conditions) below the maximum temperature at a given hour during the day, Reference [4.3], Chapter 26, Table 3.

Table 4.1-3
Component Minimum and Maximum Temperatures in the Advanced NUHOMS® System (Storage or Transfer Mode) for Normal Conditions

Component[3]	*Maximum Storage Mode (F°)*	Maximum[1] *Transfer Mode (F°)*	Minimum[2] (°F)	Allowable Range (°F) Ref
AHSM Concrete	*219*	N/A	0	0 to 300 [4.5]
AHSM Support Steel	*351*	N/A	0	0 to 2,600 [4.6]
AHSM Heat Shield	*258*	N/A	0	0 to 2,600 [4.6]
DSC Shell	*399*	439	0	0 to 800 [4.7]
DSC Top Outer Cover Plate	*294*	337	0	0 to 800 [4.7]
DSC Top Inner Cover Plate	*296*	337	0	0 to 800 [4.7]
DSC Top Shield Plug	*316*	345	0	0 to 700 [4.7]
DSC Bottom Inner Cover Plate	*315*	402	0	0 to 800 [4.7]
DSC Bottom Shield Plug	*313*	400	0	0 to 700 [4.7]
DSC Bottom Outer Cover Plate	*299*	393	0	0 to 800 [4.7]
DSC Spacer Disc	*617*	658	0	0 to 700 [4.7]
DSC Guidesleeve	*618*	658	0	0 to 800 [4.7]
DSC Oversleeve	*618*	658	0	0 to 800 [4.7]
DSC Support Rod/Spacer Sleeve	*479*	522	0	0 to 650 [4.7]
DSC Boral™ Sheet	*618*	658	0	0 to 850 [4.8]
WE 14x14 SS304 Fuel Cladding	*618*	*658*	0	*0 to 806*[4] *Transfer Mode*
				0 to 690 *Storage Mode*
WE 14x14 MOX Zirc Cladding	*618*	*658*	0	*0 to 1058*[4] *Transfer Mode*
				0 to 618 *Storage Mode*

(1) *Temperatures provided are conservatively based on a 14 kW DSC heat load in conjunction with a DSC shell temperature based on a 24 kW transfer cask analysis.*

(2) For the minimum daily averaged temperature condition of 0°F ambient, the resulting component temperatures will approach 0°F if no credit is taken for the decay heat load.

(3) See Table 4.1-6 for the limiting heat loads for which *each component* was *analyzed*. Maximum 24PT1-DSC heat load for this application is 14 kW. Other heat loads used in the analyses provide conservatism and may be used in future amendments. The maximum AHSM heat load for this application is 24kW.

(4) These fuel cladding limits apply to the short term transients such as the transfer operations.

Table 4.1-4
Component Minimum and Maximum Temperatures in the Advanced NUHOMS® System (Storage or Transfer Mode) for Off-Normal Conditions

Component[4]	Maximum[5] (°F)	Minimum[3] (°F)	Allowable Range (°F) Ref
AHSM Concrete	231	-40	-40 to 300 [4.5]
AHSM Support Steel	360	-40	-40 to 2,600 [4.6]
AHSM Heat Shield	270	-40	-40 to 2,600 [4.6]
DSC Shell	443	-40	-40 to 800 [4.7]
DSC Top Outer Cover Plate	350	-40	-40 to 800 [4.7]
DSC Top Inner Cover Plate	350	-40	-40 to 800 [4.7]
DSC Top Shield Plug	358	-40	-40 to 700 [4.7]
DSC Bottom Inner Cover Plate	408	-40	-40 to 800 [4.7]
DSC Bottom Shield Plug	406	-40	-40 to 700 [4.7]
DSC Bottom Outer Cover Plate	400	-40	-40 to 800 [4.7]
DSC Spacer Disc	658[2]	-40	-40 to 700 [4.7]
DSC Guidesleeve	658[2]	-40	-40 to 800 [4.7]
DSC Oversleeve	658[2]	-40	-40 to 800 [4.7]
DSC Support Rod/Spacer Sleeve	522[2]	-40	-40 to 650 [4.7]
DSC Boral™ Sheet	658[2]	-40	-40 to 1000 [4.8]
WE 14x14 SS304 Fuel Cladding	658[2]	-40	-40 to 806[1]
WE 14x14 MOX Zirc Cladding	658[2]	-40	-40 to 1058[1]

(1) The derivation of the fuel cladding *temperature* limits is given in Section 3.5.

(2) The maximum 24PT1-DSC basket temperatures are bounded by the maximum normal case in the cask because of the required sunshade over the cask in the off-normal temperature range.

(3) For the minimum daily averaged temperature condition of –40°F ambient, the resulting component temperatures will approach –40°F if no credit is taken for the decay heat load.

(4) See Table 4.1-6 for the limiting heat loads for which each component was analyzed. Maximum 24PT1-DSC heat load for this application is 14 kW. Other heat loads used in the analyses provide conservatism and may be used in future amendments. The maximum AHSM heat load for this application is 24kW.

(5) *Maximum off-normal temperature is during the transfer mode.*

Table 4.1-5
Component Minimum and Maximum Temperatures in the Advanced NUHOMS® System (Storage and Transfer) for Accident Conditions

Component[4]	Maximum[5] (°F)	Minimum [3] (°F)	Allowable Range (°F) Ref
AHSM Concrete	392 [2]	-40	-40 to 350 [4.5]
AHSM Support Steel	615	-40	-40 to 2,600 [4.6]
AHSM Heat Shield	542	-40	-40 to 2,600 [4.6]
DSC Shell	646	-40	-40 to 800 [4.7]
DSC Top Outer Cover Plate	423	-40	-40 to 800 [4.7]
DSC Top Inner Cover Plate	424	-40	-40 to 800 [4.7]
DSC Top Shield Plug	444	-40	-40 to 700 [4.7]
DSC Bottom Inner Cover Plate	450	-40	-40 to 800 [4.7]
DSC Bottom Shield Plug	448	-40	-40 to 700 [4.7]
DSC Bottom Outer Cover Plate	434	-40	-40 to 800 [4.7]
DSC Spacer Disc	695	-40	-40 to 700 [4.7]
DSC Guidesleeve	696	-40	-40 to 800 [4.7]
DSC Oversleeve	696	-40	-40 to 800 [4.7]
DSC Boral™ Sheet	696	-40	-40 to 1000 [4.8]
DSC Support Rod/Spacer Sleeve	588	-40	-40 to 650 [4.7]
WE 14x14 SS304 Fuel Cladding	749[6]	-40	-40 to 806[1]
WE 14x14 MOX Zirc Cladding	749[6]	-40	-40 to 1058[1]

(1) The derivation of the fuel cladding *temperature* limits is given in Section 3.5.

(2) 392°F is above the 350°F limit given in Reference [4.5] - Testing will be performed to document that concrete compressive strength will be greater than that assumed in structural analyses and that the concrete did not degrade (does not show signs of spalling, cracks and/or loss of cement bond to aggregate) due to the elevated temperature.

(3) For the minimum daily averaged temperature condition of –40°F ambient, the resulting component temperatures will approach –40°F if no credit is taken for the decay heat load.

(4) See Table 4.1-6 for the limiting heat loads for which *each component was analyzed.* Maximum 24PT1-DSC heat load for this application is 14 kW. Other heat loads used in *the analyses* provide conservatism and may be used in future amendments. The maximum AHSM heat load for this application is 24kW.

(5) *The maximum accident temperature is during a storage mode blocked vent condition.*

(6) *Fuel clad temperature based on 16kW heat load while DSC basket temperatures are based on 14kW heat load.*

4.3 Specifications for Components

Allowable temperature ranges for the structural materials used in the design are given in Table 4.1-3, Table 4.1-4, and Table 4.1-5. Because of the passive design of the Advanced NUHOMS® System, there is no need for rupture discs or pressure relief in the safety related components.

4.4 Thermal Evaluation for Normal Conditions of Storage and Transfer

4.4.1 Overview of Thermal Analysis for Normal Conditions of Storage and Transfer

This section of the SAR describes the thermal analysis of the AHSM and 24PT1-DSC. The analytical models of the AHSM, the 24PT1-DSC, and the transfer cask are described and the calculation results are summarized below. The thermophysical properties of the Advanced NUHOMS® System components used in the thermal analysis are listed in Section 4.2. The following evaluations are performed for the Advanced NUHOMS® System:

1. Thermal Analysis of the 24PT1-DSC in the AHSM (Section 4.4.2),
2. Thermal Analysis of the 24PT1-DSC in the Transfer Cask (Section 4.4.3), *and*
3. Thermal Analysis of the 24PT1-DSC basket (Section *4.4.4*).

4.4.2 Thermal Model of the 24PT1-DSC Inside the AHSM

For normal condition of storage, the Advanced NUHOMS® System components are evaluated for a range of design basis ambient temperatures. The system components are evaluated for the average ambient temperatures given in Table 4.1-1. Ambient temperatures within this range are assumed to occur for a sufficient duration to cause a steady-state temperature distribution in the Advanced NUHOMS® System components. The lifetime average ambient temperature for the 40 year service life is taken as 70°F. The "stress-free" temperature for material properties is also 70°F.

The AHSM is cooled by a natural draft of air entering through the air inlet opening located in the lower front wall of the AHSM, and exiting through the air outlet opening located in the top of the AHSM. Cooler air at the prevailing ambient conditions is drawn into the AHSM. The cooler air flows from the bottom of the AHSM along the outer 24PT1-DSC surface where it is warmed by the decay heat of the spent fuel inside the 24PT1-DSC. The warmed air flows along the ceiling of the AHSM and exits through the air outlet opening. The AHSM vent geometries and flow paths for ventilation air are illustrated in *Figure 4.4-1*.

The AHSM roof and front wall are the primary concrete surfaces conducting heat to the outside environment. For the analytical purpose of calculating maximum temperatures, an AHSM centered in a group of AHSMs, each loaded with a 24PT1-DSC, is assumed. Rows of modules are assumed to exist back to back for this model. For the analytical purpose of calculating maximum concrete temperature gradients, an AHSM alone, with no adjacent modules or rear shield wall, is assumed.

A metal heat shield is placed around the upper half of the 24PT1-DSC to shield the AHSM concrete surfaces above and to the side of the 24PT1-DSC from thermal radiation effects. The location and geometry of the heat shield is shown in Figure 4.4-6 and on the AHSM drawings contained in Chapter 1.The heat shield protects the AHSM surfaces above and to the side of the 24PT1-DSC from direct thermal radiation emanating from the 24PT1-DSC surface and

The HEATING7 analytical model of the AHSM is one-half of a single AHSM unit. Symmetry or an insulated boundary is applied along the vertical centerline of the AHSM model shown in Figure 4.4-2 through Figure 4.4-4. The HEATING7 model includes regions for the concrete, top shield block, base block, and basemat of the AHSM. The soil below the ISFSI basemat is modeled as a seven foot thick region with a constant temperature boundary at the lower edge of this region. Sufficient nodal refinement is used in the AHSM analytical model to obtain accurate temperature distributions through the thickness of the AHSM base and storage units.

The thermal analysis of a typical AHSM is performed for a loaded 24PT1-DSC located in the interior of a multiple module array with a 24PT1-DSC present in the two adjacent AHSMs. The AHSM top and front surfaces are modeled as exposed to the prevailing ambient conditions in this model. The side and back surfaces are modeled as being adiabatic in order to simulate the adjacent modules. An additional model was constructed for the specific purpose of calculating the maximum concrete gradients. For this model, the thermal analysis is performed for a free standing AHSM, with the top and front, as well as the back and side surfaces, exposed to the prevailing ambient conditions.

For summer ambient conditions, a solar heat flux of 72.6 BTU/hr-ft^2 is applied to the top surface for normal summer ambient conditions, which is based on conservative averaging of data in Reference [4.3]. Solar heat loads are conservatively neglected for the AHSM thermal analysis for normal winter ambient conditions. The solar heat loads are listed in Table 4.1-1.

The solar heat load calculated from Reference [4.3] is based on the average heat flux transmitted through horizontal DSA glass. The 74.8 Btu/hr-ft^2 heat flux is adjusted to account for the solar heat gain of the DSA glass by dividing the heat flux obtained by the heat gain factor for the glass, which is 0.87, per Reference [4.3]. To account for the reflective properties of the concrete surface this adjusted heat flux is multiplied by the solar absorptivity of concrete which ranges from 0.65 to 0.80 (Reference [4.3]). Therefore the solar heat flux absorbed by the AHSM concrete roof surface with the maximum absorptivity value of 0.80 (which is conservative) is:

$$74.8 \times 0.80 / 0.87 = 68.8 \text{ Btu/hr-ft}^2$$

The solar insolation value used in the thermal analysis for normal conditions is 72.6 Btu/hr-ft^2 which provides an additional 5% conservatism above the value calculated above.

The 24PT1-DSC cylindrical shell is approximated in the AHSM thermal model as a rectangle, as shown in Figure 4.4-5. The approximation using rectangular regions is necessary since HEATING7 restricts the user to one geometry type in the same analytical model. To improve the approximation, the modeled 24PT1-DSC regions have the same surface area as the outer surface of the 24PT1-DSC cylindrical shell. The analytical model of the AHSM includes regions for the metal heat shield located between the top and sides of the 24PT1-DSC, and the AHSM, as shown in Figure 4.4-5. The surface area of the heat shield was also approximately maintained in the model.

The analytical model also includes regions to model the air gaps between the 24PT1-DSC, heat shield, and the AHSM.

The heat generation in the canister used in the AHSM thermal model is given in Table 4.4-2 for 24, 16, and 14 kW heat load in the canister. The heat generation is distributed over the entire internal cavity volume of the 24PT1-DSC, the model geometry is shown in Figure 4.4-5. Use of the entire 24PT1-DSC cavity volume and exclusion of an axial peaking factor for the AHSM thermal analysis is based on the test data contained in Reference [4.21]. The reference test data for cylindrical casks show that the measured surface temperature profiles are relatively flat over the entire length, indicating that the heat flux is nearly uniform over the surface and axial peaking is not affecting the surface temperature distribution. One reason for the relatively flat temperature profiles is the high thermal conductivity of the 24PT1-DSC shell material. The resulting heat generation is therefore more representative of the manner in which heat is actually rejected to the AHSM air space by the 24PT1-DSC. The active fuel length of 120 inches and the peaking factor of 1.08 are conservatively used in the thermal analysis of the 24PT1-DSC internals presented in Section 4.4.4 for the evaluation of local effects such as the peak fuel clad temperature. The outer surface of the 24PT1-DSC shell dissipates heat to the AHSM through both convection and radiation. The air surrounding the 24PT1-DSC is modeled as a gap filled with gas (air), thus providing a mechanism

for heat transfer from all AHSM interior surfaces and the 24PT1-DSC outer surface. Due to a limitation in the HEATING7 code, conduction in the air gap could not be included, however, this is a minor effect.

Convection heat transfer from the 24PT1-DSC and AHSM surfaces is modeled by defining analytical functions for the bulk air temperatures along the surfaces of the 24PT1-DSC, heat shield, and concrete using the calculated air temperature rises given in Table 4.4-1. A linear rise in the air temperature around the periphery of the 24PT1-DSC is assumed. The maximum temperature of the air from Table 4.4-1 is assumed at the top of the 24PT1-DSC and throughout the top shield block, which is conservative. These temperatures are also used to calculate the heat transfer coefficients along the 24PT1-DSC, heat shield, and concrete surfaces. With the air temperatures and the equations for the heat transfer coefficients described below, the HEATING7 program calculates the temperatures of the 24PT1-DSC exterior surface and the AHSM interior and exterior surfaces.

The majority of the spent fuel assembly decay heat is removed from the 24PT1-DSC outer surface through convection. Since the heat shield and the concrete surrounding the 24PT1-DSC are curved, both sides of the metal heat shield and the AHSM concrete surfaces surrounding the 24PT1-DSC are cooled by air with a heat transfer coefficient of h_{cyl}. Horizontal slab surfaces with convection on their lower surface, such as the AHSM top shield block lower surface, are cooled by natural convection with a heat transfer coefficient of h_{ceil}. Horizontal surfaces with convection on their upper surfaces, such as the AHSM top shield block outer surface, are cooled by natural convection with a heat transfer coefficient of h_{plate}. Vertical, flat surfaces are cooled by natural convection with a heat transfer coefficient of h_{wall}.

Radiation heat transfer is modeled between the 24PT1-DSC outer surface and heat shield, between the 24PT1-DSC outer surface and the basemat and lower shield block, and between the heat shield and the AHSM concrete surfaces. The external surface of the AHSM top shield block is cooled by external air with a heat transfer coefficient of h_{plate}, and by radiation cooling to ambient air. The formulas used for the calculation of the heat transfer coefficients for natural convection are as follows all in BTU/(hr-ft^2°F) [4.22]:

$$h_{cyl} = 0.18\,(\Delta T)^{1/3} \quad (4.4\text{-}3)$$

$$h_{ceil} = 0.12\,(\Delta T/L)^{1/4} \quad (4.4\text{-}4)$$

$$h_{plate} = 0.22\,(\Delta T)^{1/3} \quad (4.4\text{-}5)$$

$$h_{wall} = 0.19\,(\Delta T)^{1/3} \quad (4.4\text{-}6)$$

Where:

ΔT = $T_{surface} - T_{air}$ (°F)

L = conservatively defined characteristic length of surface (ft)

The heat transfer coefficients are updated by HEATING7 following each iteration using the resulting average temperature of the corresponding surface node. A sufficient number of iterations are performed until the temperatures differ by less than 0.1% from the previous temperature calculated in two consecutive iterations indicating that stable convergence is

Table 4.4-2
Heat Generations Used in the Thermal Model of the 24PT1-DSC in the AHSM

Heat Load per Canister (kW)	Heat Generation (Btu/min/in^3)
24	0.00308
16	0.00205
14	0.00180

Table 4.4-3
AHSM Thermal Analysis Results Summary

Operating Condition (°F)	T_{amb} (°F)	$T_{con,max}$ (°F)	$T_{hs,max}$ (°F)	$T_{sup,max}$ (°F)[1]
Normal	0	103	136	268
	70	187	225	328
	104	219	258	351
Off-Normal	-40	56	87	233
	117	231	270	360
40 hr Blocked Vent Transient	-40	243	426	526
	117	392[2]	542	615

(1) The 24PT1-DSC bottom maximum temperature conservatively *represents* the maximum support structure temperature.

(2) 392°F is above the 350°F limit given in Reference [4.5] - Testing will be performed to document that concrete compressive strength will be greater than that assumed in structural analyses.

Nomenclature used in table

T_{amb}	Ambient temperature
$T_{con,max}$	Maximum concrete temperature
$T_{hs,max}$	Maximum heat shield temperature
$T_{sup,max}$	Maximum 24PT1-DSC support structure temperature

6.4.3 Criticality Results

This section presents the results of the analyses used to demonstrate the acceptability of storing qualified fuel in the 24PT1-DSC under normal, off-normal, and accident conditions for fuel loading, handling, and storage. Uncertainties are addressed and are applied to the nominal calculated k_{eff} value.

ANS/ANSI-8.1 [6.5] recommends that calculational methods used in determining criticality safety limits for applications outside reactors be validated by comparison with appropriate critical experiments. An Upper Subcritical Limit (USL) provides a high degree of confidence that a given system is subcritical if a criticality calculation based on the system yields a k_{eff} below the USL. In Section 6.5, the minimum USL is determined to be 0.*9401*. The criticality analysis verifies that in normal, off-normal, and accident conditions, $k_{eff} + 2\sigma \leq$ USL. Therefore, the fuel will remain subcritical. Conclusions regarding specific aspects of the methods used or the analyses presented can be drawn from the quantitative results presented in the associated tables.

Reactivity calculations were performed in four sets of parametric studies for the 24PT1-DSC loaded with WE 14x14 SC assemblies, WE 14x14 MOX fuel assemblies, and WE 14x14 SC assemblies containing damaged fuel rods. The results are summarized in Table 6.4-1, Table 6.4-2, and Table 6.4-3, respectively.

The maximum k_{eff} for the 24PT1-DSC with intact WE 14x14 SC fuel assemblies was determined to be 0.8650±0.0014 for HAC conditions with internal moderation at 1.0 g/cc and external moderation at 0.8 g/cc interspersed between an infinite array of packages. The $k_{eff} + 2\sigma$ is less than the USL, 0.8677< 0.*9401*.

The maximum k_{eff} for the 24PT1-DSC with intact WE 14x14 MOX fuel assemblies was determined to be 0.9087±0.0012 for HOC conditions with internal moderation at 1.0 g/cc and external moderation at 0.0001 g/cc in an infinite array of packages. The $k_{eff} + 2\sigma$ is less than the USL, 0.9111< 0.*9401*.

The maximum k_{eff} for the 24PT1-DSC loaded with 24 WE 14x14 SC failed fuel assemblies was determined to be 0.9340 ± 0.0014 for HAC conditions with a rod pitch of 0.652 inches, internal moderation at 1.0 g/cc, and external moderation at 0.8 g/cc in an infinite array of packages. The $k_{eff} + 2\sigma$ is less than the USL, 0.9392< 0.*9401*. This is close to the USL; however, it should be noted that it was conservatively assumed in the modeling of the damaged fuel geometry that the entire 24PT1-DSC was filled with damaged fuel assemblies. There will be significantly more margin to the USL if less than 24 damaged fuel assemblies are loaded in the 24PT1-DSC (a maximum of four damaged fuel assemblies are allowed, see Chapter 12). This analysis, although performed specifically for the WE 14x14 SC damaged fuel assemblies will bound the payload condition of one damaged MOX assembly with no other damaged assemblies in the 24PT1-DSC. The conservatism of the analyzed canister with 24 WE 14x14 SC damaged fuel assemblies bounds the condition of only one MOX damaged fuel assembly at one of the corner locations in the 24PT1-DSC (outermost fuel assembly locations at 45, 135, 225, or 315 degree azimuths). The MOX damaged fuel assembly at this location is also bounded by the 24PT1-DSC analysis for 24 undamaged MOX assemblies. The increase in reactivity of one damaged MOX assembly

damaged MOX assembly at a location with significant neutron leakage (adjacent undamaged, non-MOX assemblies on only 2 sides with the remaining two sides facing the 24PT1-DSC shell without intervening fuel) will have little effect on the low k_{eff} calculated for the WE 14x14 SC intact fuel case.

6.4.4 Evaluation of Effect of Uncertainty in Maximum Initial Enrichment

The maximum initial enrichment used in the criticality analyses for $U0_2$ fuel (4.0%) does not include uncertainties (manufacturing tolerance) in maximum initial enrichment. Also, the maximum initial enrichment for MOX fuel used in the criticality analysis does account for potential uncertainty in Pu maximum initial enrichment.

To address the effect of potential $U0_2$ fuel assembly initial enrichment uncertainty, an evaluation of the effect of an increased initial enrichment from 4.0 weight % to 4.05 weight %, has been performed. The results of this analysis are presented in Table 6.4-4. These results when compared to the equivalent case analyzed for 4.0 weight %, Table 6.4-3, show that the increase in enrichment from 4.0 weight % to 4.05 weight % results in an increase in k_{eff} (incl. 2σ) from .9368 to .9392. The increased k_{eff} remains less than the USL of .9401. Based on these results, storage of fuel of up to 4.05 weight % enrichment is acceptable. Chapter 12 therefore uses a maximum enrichment of 4.05 weight % for $U0_2$ fuel.

6.4.5 Effect of Fuel Parameter Tolerances on Reactivity

To evaluate the effect of fuel parameter tolerances on reactivity, the fuel parameters used in the criticality analyses have been reviewed to identify sensitivity studies needed to evaluate these effects. A review of fuel parameters identified in Tables 6.2-1 and 6.2-2 indicates that all parameters listed with the exception of pellet diameter, clad outer diameter (OD) and clad thickness are enveloped by the criticality analyses performed. To evaluate the effect of tolerances in clad OD, clad thickness and fuel pellet diameter on reactivity, sensitivity analyses are performed to evaluate system reactivity as a function of these parameters. These analyses are performed for the SC fuel assemblies since the damaged SC assembly analysis is the bounding analysis. In addition, changes in the configuration of the stainless steel cladding have a more significant effect on reactivity than the zirconium cladding used in the MOX fuel. The CSAS25 models used to perform this sensitivity analysis are based on the model used to calculate k_{eff} for the fuel assemblies centered in the guidesleeves, the results of which are reported in Table 6.4-1. This model is revised to include 4.05 wt. % enriched fuel and the revised clad and pellet dimensions. The results of the evaluation are presented in Tables 6.4-5, 6.4-6 and 6.4-7. The results demonstrate that the calculated changes in reactivity between the various cladding and pellet dimensions are not significant.

Table 6.4-*4*
Bounding Criticality Analysis Analyzed for 4.05 weight % ^{235}U

Damaged Fuel Assemblies with External Moderator Density Varying for Most Reactive Rod Pitch Case

K_{eff}	+/- 1σ	K_{eff} +2σ	External Moderator (H_20) Density, g/cc
0.9348	0.0013	0.9374	0.0001
0.9364	0.0012	0.9389	0.05
0.9341	0.0012	0.9365	0.10
0.9365	0.0012	0.9389	0.20
0.9348	0.0013	0.9374	0.30
0.9364	0.0013	0.9390	0.40
0.9338	0.0011	0.9360	0.50
0.9362	0.0012	0.9386	0.60
0.9368	0.0012	0.9392	0.70
0.9365	0.0012	0.9389	0.80
0.9345	0.0012	0.9369	0.90
0.9354	0.0011	0.9376	1.00

Table 6.4-*5*
Clad OD Sensitivity Evaluation

Fuel Clad OD: 4.05 wt.% U-235 Fuel Centered In Guide Tube			
K_{eff}	+/- 1σ	K_{eff} +2σ	Clad OD (inches)[1]
0.8645	0.0011	0.8667	0.415
0.8653	0.0013	0.8679	0.418
0.8640	0.0013	0.8666	0.420
0.8625	0.0013	0.8651	0.422
0.8646	0.0012	0.8670	0.424
0.8642	0.0012	0.8666	0.426
0.8631	0.0012	0.8655	0.429

1. Min. clad thickness is used for this evaluation.

Table 6.4-6
Clad Thickness Sensitivity Evaluation

K_{eff}	+/- 1σ	K_{eff} +2σ	Clad OD (inches)	Clad ID
0.8631	0.0012	0.8654	0.429	.3976
0.8599	0.0012	0.8623	0.429	.3970
0.8591	0.0013	0.8617	0.429	.3964
0.8588	0.0011	0.8610	0.429	.3960
0.8575	0.0013	0.8602	0.429	.3954
0.8563	0.0011	0.8584	0.429	.3948
0.8559	0.0012	0.8583	0.429	.3944

Table 6.4-7
Pellet Diameter Sensitivity Evaluation

K_{eff}	+/- 1σ	K_{eff} +2σ	Pellet OD (inches)
0.8626	0.0010	0.8646	0.3831
0.8610	0.0010	0.8631	0.3833
0.8619	0.0012	0.8642	0.3834
0.8630	0.0012	0.8653	0.3836
0.8628	0.0013	0.8655	0.3838
0.8621	0.0013	0.8647	0.3839
0.8615	0.0011	0.8638	0.3841

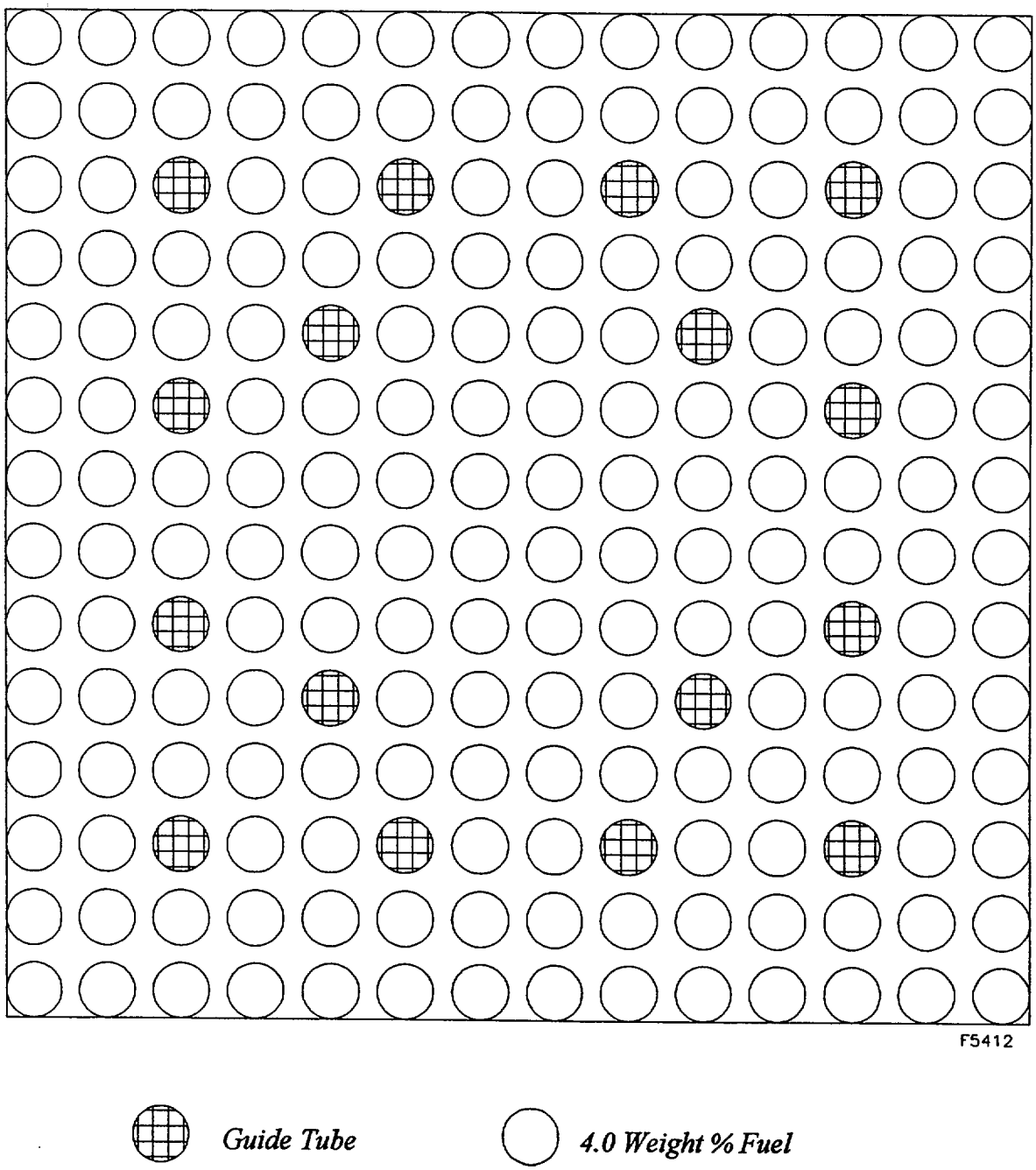

Figure 6.4-1
Rod Pitch Study: Figure Shows Maximum Rod Pitch Allowed

8. OPERATING PROCEDURES

This Chapter presents the operating procedures for the Advanced NUHOMS® System described in previous chapters and shown on the drawings in Chapter 1. The procedures include preparation of the 24PT1-DSC and fuel loading, closure of the DSC, transport to the ISFSI, transfer into the AHSM, monitoring operations, and retrieval from the AHSM. The Advanced NUHOMS® System transfer equipment and the existing plant systems and equipment are used to accomplish these operations. Procedures are delineated here to describe how these operations are to be performed. Standard fuel and cask handling operations performed under the plant's 10CFR 50 operating license are described in less detail. The licensee may revise existing operational procedures and new ones may be developed according to the requirements of the plant, provided that the system requirements specified in Chapter 12 are met.

These generic Advanced NUHOMS® System procedures have been developed to minimize the amount of time required to complete the subject operations, to minimize personnel exposure, and to assure that all operations required for 24PT1-DSC loading, closure, transfer, storage, and unloading are performed safely. Plant specific ISFSI procedures are to be developed by each licensee in accordance with the requirements of 10CFR 20, 10CFR 50 or 10CFR 72, as applicable. The generic procedures presented here are provided as a guide for the preparation of plant specific procedures and serve to demonstrate how the Advanced NUHOMS® System operations are to be accomplished. Procedures should include identification of potential hazards and mitigating actions applicable to site specific configurations. These procedures are not intended to be limiting in that the licensee may decide that alternate acceptable means are available to accomplish the same operational objective.

These guidelines provide operational sequences to ensure that occupational and site dose limits, as determined in Chapter 10, can be maintained.

8.1 Procedures for Loading the 24PT1-DSC and Transfer to the AHSM

The following sections outline the typical operating procedures for loading the 24PT1-DSC into the Transfer Cask and for transferring the loaded canister to the AHSM.

8.1.1 Narrative Description

The following steps describe the recommended generic operating procedures for the Advanced NUHOMS® System. A list of instrumentation used during loading operations is provided in Table 8.1-1. Flowcharts of Advanced NUHOMS® System loading operations are provided in Figure 8.1-1. A pictorial representation of key phases of this process is provided in Figure 8.1-2.

8.1.1.1 Transfer Cask and 24PT1-DSC preparation

1. Prior to placement in dry storage, the candidate fuel assemblies are to be visually examined to insure that no known or suspected gross cladding breaches exist. Pinholes and hairline cracks are acceptable. Visual verification of fuel integrity may also be accomplished in conjunction with existing plant records. The assemblies shall be evaluated (by plant records or other means) to verify that they meet the physical, thermal and radiological criteria specified in Chapter 12. Failed fuel cans, if required for loading damaged fuel assemblies, should be placed into the 24PT1-DSC basket prior to lowering the 24PT1-DSC/TC into the spent fuel pool.
2. Prior to being placed in service, the transfer cask is to be cleaned or decontaminated as necessary to insure an acceptable surface contamination level.
3. Place the transfer cask in the vertical position in the cask decon area using the cask handling crane and the transfer cask lifting yoke (the design basis and criteria for the yoke is addressed in Chapter 1).
4. Place scaffolding around the cask so that the top cover plate and surface of the cask are easily accessible to personnel.
5. Remove the cask top cover plate and examine the cask cavity for any physical damage and ready the cask for service.
6. Examine the 24PT1-DSC for any physical damage which might have occurred since the receipt inspection was performed. The 24PT1-DSC is to be cleaned and any loose debris removed. *Verify that fuel spacers are present in all fuel assembly slots.*
7. Using a crane, lower the 24PT1-DSC into the cask cavity by the internal lifting lugs and rotate the 24PT1-DSC to match the cask alignment marks.
8. Place the top shield plug onto the 24PT1-DSC. Examine the top shield plug to ensure a proper fit.

9. Fill the cask/24PT1-DSC annulus with clean, demineralized water. Place the inflatable seal into the upper cask liner recess and seal the cask/24PT1-DSC annulus by pressurizing the seal with compressed air.

NOTE: A transfer cask/24PT1-DSC annulus pressurization tank filled with demineralized water may be connected to the top vent port of the transfer cask via a hose to provide a positive head above the level of water in the cask/24PT1-DSC annulus. This is an optional arrangement, which provides additional assurance that contaminated water from the fuel pool will not enter the cask/24PT1-DSC annulus, provided a positive head is maintained at all times.

10. Position the cask lifting yoke and engage the cask lifting trunnions and the rigging cables to the 24PT1-DSC top shield plug. Adjust the rigging cables as necessary to obtain even cable tension. Remove the top shield plug and place clear of the cask.

11. Fill the 24PT1-DSC cavity with water from the fuel pool or an equivalent source.

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

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

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

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

16. Prior to the cask being lifted into the fuel pool, the water level in the pool should be adjusted as necessary to accommodate the cask/24PT1-DSC volume. If the water placed in the 24PT1-DSC cavity was obtained from the fuel pool, a level adjustment may not be necessary.

8.1.1.2 24PT1-DSC Fuel Loading

1. Lift the cask/24PT1-DSC and position them over the cask loading area of the spent fuel pool in accordance with the plant's 10CFR 50 cask handling procedures.

2. Lower the cask into the fuel pool until the bottom of the cask is at the height of the fuel pool surface. As the cask is lowered into the pool, spray the exterior surface of the cask with demineralized water to minimize surface adhesion of contaminated particles.

NOTE: The introduction of demineralized water may dilute fuel pool boron concentration.

3. Place the cask in the location of the fuel pool designated as the cask loading area.

4. Disengage the lifting yoke from the cask lifting trunnions and move the yoke clear of the cask. Spray the lifting yoke with clean demineralized water if it is raised out of the fuel pool.

5. Prior to insertion of a spent fuel assembly into the 24PT1-DSC, the identity of the assembly shall be verified by two individuals using an underwater video camera. Read and record the fuel assembly identification number from the fuel assembly and check this identification number against the 24PT1-DSC loading plan that indicates which fuel assemblies are acceptable for dry storage.

6. Move a candidate fuel assembly (including NFAH where applicable) from a fuel rack in accordance with the plant's 10CFR 50 fuel handling procedures.

7. Position the fuel assembly for insertion into the selected 24PT1-DSC storage cell and load the fuel assembly. Repeat Steps 5 through 7 for each SFA loaded. After the 24PT1-DSC has been fully loaded, check and record the identity and location of each fuel assembly.

8. After all the SFAs have been placed into the 24PT1-DSC and their identities verified, *fuel spacers are placed above each intact fuel assembly (damaged fuel cans incorporate an integral upper fuel spacer).* Position the lifting yoke and the top shield plug and lower the shield plug onto the 24PT1-DSC.

CAUTION: Verify that all the lifting height restrictions specified in Chapter 12 can be met in the following steps which involve lifting of the transfer cask.

9. Visually verify that the top shield plug is properly seated onto the 24PT1-DSC.

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

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

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

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

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

12.3.1.2 24PT1-DSC Helium Backfill Pressure

LCO 12.3.1.3 24PT1-DSC helium backfill pressure shall be 1.5 ± 1.5 psig (stable for 30 minutes after filling).

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

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

This specification is applicable to all 24PT1-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Note: Not applicable until SR 12.3.1.2.1 is performed.		
A. The required backfill pressure cannot be obtained or stabilized.	A.1 Establish the 24PT1-DSC helium backfill pressure to within the limit.	24 hours
	OR	
	A.2 Flood the DSC with water submerging all fuel assemblies.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR12.3.1.2.1	Verify that the 24PT1-DSC helium backfill pressure is 1.5 ± 1.5 psig.	Once per 24PT1-DSC, after the completion of TS 12.3.1.1 actions.

12.3.1.3 24PT1-DSC Helium Leak Rate of Inner Top Cover Plate Weld and Vent/Siphon Port Cover Welds

LCO 12.3.1.3 The 24PT1-DSC helium leak rate of the inner top cover plate and vent/siphon port cover welds shall be less than or equal to 10^{-7} std-cc/sec.

This leak test shall be performed using helium gas ≥99% purity.

APPLICABILITY: During LOADING OPERATIONS.

ACTIONS

---------- NOTE ----------
This specification is applicable to the inner top cover plate weld and vent/siphon port cover welds of all 24PT1-DSCs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
Note: Not applicable until SR 12.3.1.3.1 is performed.		
A. Helium leak rate not met.	A.1 Establish the 24PT1-DSC leak rate to within the limit.	7 days
B. Required Action A.1 and associated Completion Time not met.	B.1 Determine and complete corrective actions necessary to return the 24PT1-DSC to an analyzed condition.	30 days
	OR	
	B.2 Unload the 24PT1-DSC.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR12.3.1.3.1 Verify that the helium leak rate is within limit.	Once per 24PT1-DSC, after the welding of the root pass(es) for the outer top cover plate.

12.4.0 Design Features

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

12.4.1 Site

12.4.1.1 Site Location

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

12.4.2 Storage System Features

12.4.2.1 Storage Capacity

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

12.4.2.2 Storage Pad

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

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

Additional requirements for the pad configuration are provided in Section 12.4.4.2.

12.4.2.3 Canister Neutron *Absorber*

Neutron *absorber* with a minimum ^{10}B loading of 0.025 grams/square centimeter is provided for criticality *control.*

12.4.2.4 Canister Flux Trap Configuration

The canister *minimum* flux trap *size (minimum spacer disc liganment width) is 0.675".*

12.4.2.5 Fuel Spacers

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

12.4.3 Codes and Standards

12.4.3.1 Advanced Horizontal Storage Module (AHSM)

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

12.4.3.2 Dry Shielded Canister (24PT1-DSC)

The 24PT1-DSC is designed fabricated and inspected to the maximum practical extent in accordance with ASME Boiler and Pressure Vessel Code Section III, Division 1, 1992 Edition with Addenda through 1994, including exceptions allowed by Code Case N-595-1, Subsections NB, NF, and NG for Class 1 components and supports. Code exceptions are discussed in 12.4.3.4.

12.4.3.3 Transfer Cask

The Transfer Cask shall meet the codes and standards that are applicable to its design under Certificate of Compliance C of C *72-1004, OS-197 Transfer Cask.*

A solar shield is required for cask transfer operations at temperatures exceeding 100°F.

12.4.3.4 Exceptions to Codes and Standards

ASME Code exceptions for the 24PT1-DSC are listed below:

DSC Shell Assembly ASME Code Exceptions, Subsection NB

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NCA	All	Not compliant with NCA
NB-1100	Requirements for Code Stamping of Components	The 24PT1-DSC shell is designed & fabricated in accordance with the ASME Code, Section III, Subsection NB to the maximum extent practical. However, Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME "N" or "NPT" stamp, or to be ASME Certified.
NB-2130 NB-4121	Material must be supplied by ASME approved material suppliers Material Certification by Certificate Holder	All materials designated as ASME on the SAR drawings are obtained from ASME approved MM or MS supplier(s) with ASME CMTR's. Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NB-2130 is not possible. Material traceability & certification are maintained in accordance with TNW's NRC approved QA program

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NB-6111	All completed pressure retaining systems shall be pressure tested	The shield plug support ring and vent and siphon block are not pressure tested due to the manufacturing sequence. The support ring is not a pressure-retaining item and the siphon block weld is helium leak tested after fuel is loaded and the inner top closure plate installed.
NB-7000	Overpressure Protection	No overpressure protection is provided for the 24PT1-DSC. The function of the 24PT1-DSC is to contain radioactive materials under normal, off-normal and hypothetical accident conditions postulated to occur during transportation and storage. The 24PT1-DSC is designed to withstand the maximum internal pressure considering 100% fuel rod failure at maximum accident temperature. The 24PT1-DSC is pressure tested to 120% of normal operating design pressure. An overpressure protection report is not prepared for the DSC.
NB-8000	Requirements for nameplates, stamping & reports per NCA-8000	The 24PT1-DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the 24PT1-DSC. *In lieu of code stamping,* QA Data packages are prepared in accordance with the requirements of 10CFR71, 10CFR72 and TNW's approved QA program.

Basket ASME Code Exceptions, Subsection NG/NF

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NCA	All	Not compliant with NCA
NG/NF-1100	Requirements for Code Stamping of Components	The 24PT1-DSC baskets are designed & fabricated in accordance with the ASME Code, Section III, Subsection NG/NF to the maximum extent practical as described in the SAR, but Code Stamping is not required. As Code Stamping is not required, the fabricator is not required to hold an ASME N or NPT stamp or be ASME Certified.
NG/NF-2130 NG/NF-4121	Material must be supplied by ASME approved material suppliers Material Certification by Certificate Holder	All materials designated as ASME on the SAR drawings are obtained from ASME approved MM or MS supplier with ASME CMTR's. Material is certified to meet all ASME Code criteria but is not eligible for certification or Code Stamping if a non-ASME fabricator is used. As the fabricator is not required to be ASME certified, material certification to NG/NF-2130 is not possible. Material traceability & certification are maintained in accordance with TNW's NRC approved QA program.
Table NG-3352-1	Permissible Joint Efficiency Factors	Joint efficiency (quality) factor of 1 is assumed for the guidesleeve longitudinal weld. Table NG-3352-1 permits a quality factor of 0.5 for full penetration weld with visual inspection. Inspection of both faces provides n= (2*0.5)=1. This is justified by this gauge of material (0.12 inch) with visual examination of both surfaces which ensures that any significant deficiencies would be observed and corrected.

Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
NG/NF-8000	Requirements for nameplates, stamping & reports per NCA-8000	The 24PT1-DSC nameplate provides the information required by 10CFR71, 49CFR173 and 10CFR72 as appropriate. Code stamping is not required for the 24PT1-DSC. *In lieu of code stamping,* QA Data packages are prepared in accordance with the requirements of 10CFR71, 10CFR72 and TNW's approved QA program.

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

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

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

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

12.4.4 Storage Location Design Features

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

12.4.4.1 Storage Configuration

AHSMs are to be tied together in single rows or back to back arrays with not less than 3 modules tied together (side by side). Each group of modules not tied together must be separated from other groups by a minimum of 20 feet to accommodate possible sliding during a seismic event. The distance between any module and the edge of the ISFSI pad shall be no less than 10 feet.

12.4.4.2 *Concrete Storage* Pad Properties to Limit 24PT1-DSC Gravitational Loadings Due to Postulated Drops

The TC/24PT1-DSC has been evaluated for drops of up to 80 inches onto a reinforced concrete *storage* pad. The evaluations are based on the concrete parameters specified in EPRI Report NP-4830, "The Effects of Target Hardness on the Structural Design of Concrete Storage Pads for Spent Fuel Casks," October 1986.