72-1029

TRANSNUCLEAR WEST

June 25, 2001
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Mr. Timothy Kobetz
Project Manager, Spent Fuel Project Office
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Subject: Double Sided Pages of Attachments 3 and 4 of Advanced NUHOMS® Storage System Application, Revision 2 (TAC No. L23203)

References: 1. Supplemental Response to Request for Additional Information and Submittal of Revision 2 of the Advanced NUHOMS® Storage System Application (TAC No. L23203), 6/22/01

Dear Mr. Kobetz:

TN West inadvertently submitted one sided prints of Attachments 3 and 4 with Reference 1. Please discard the entire contents of Attachments 3 and 4 submitted with Reference 1 and replace them with the double sided Attachments included herewith. The technical content of these pages is identical to that of Reference 1.

Attachments 1 and 2 of Reference 1 remain unchanged and are therefore not included in this transmittal.

The affidavit provided in Reference 1, Attachment 1 is also applicable to this transmittal.

Please contact Mr. U. B. Chopra (510-744-6053) or me (510-744-6020) if you have any questions regarding this transmittal.

Sincerely,

Robert M. Grenier
President and Chief Operating Officer

Docket 72-1029

NMSS01 Prop

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

Attachments:

1. Not included
2. Not included
3. Revision 2 replacement pages for the proprietary version of the Advanced NUHOMS® Storage System Application (10 copies).
4. Revision 2 replacement pages for the non-proprietary version of the Advanced NUHOMS® Storage System Application (4 copies).

cc: File: SCE-01-0007.01

ATTACHMENT 4

ADVANCED NUHOMS® SAR REVISED PAGES (REVISION 2)

NON-PROPRIETARY VERSION

SEE LIST OF EFFECTED PAGES FOR A COMPLETE LISTING OF PAGES TO BE REPLACED/ADDED

ANUH-01.0150
Revision 2

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

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

Revision 2
June 2001

Executive Summary

Revisions 1 *and 2* of this Safety Analysis Report (ANUH-01.0150) incorporate changes based on the responses *(initial and supplemental)* to NRC Request for Information (TAC No. L23203).

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.

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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 (°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 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 analysis was performed. Maximum 24PT1-DSC heat load for this application is 14 kW. Other heat loads used in analysis provide conservatism and may be used in future amendments. The maximum AHSM heat load for this application is 24kW.

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

Component[4]	Maximum (°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	-40	-40 to 806[1]
WE 14x14 MOX Zirc Cladding	749	-40	-40 to 1058[1]

(1) The derivation of the fuel cladding 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 analysis was performed. Maximum 24PT1-DSC heat load for this application is 14 kW. Other heat loads used in analysis provide conservatism and may be used in future amendments. The maximum AHSM heat load for this application is 24kW.

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 .93*68* 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 Clad OD Tolerances on Reactivity

To evaluate the effect of clad outer diameter (OD) tolerances on reactivity, a sensitivity analysis is performed to evaluate system reactivity as a function of clad outer diameter. The fabrication tolerances for the WE 14x14 SC Fuel Assembly design allow the fuel clad OD to vary from 0.415 to 0.429 inches. 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 reported in Table 6.4-1. This model is revised to include 4.05 wt. % enriched fuel and the revised clad dimensions. The results of the evaluation are presented in Table 6.4-5. The results demonstrate that the calculated changes in reactivity between the various cladding ODs *are within the statistical uncertainty of calculations.*

Table 6.4-7
Bounding Criticality Analysis Analyzed for 4.05 weight % ^{235}U
[PROPRIETARY]

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

K_{eff}	1 sigma	Keff + 2 sigma	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 6.4-9
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)*
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*

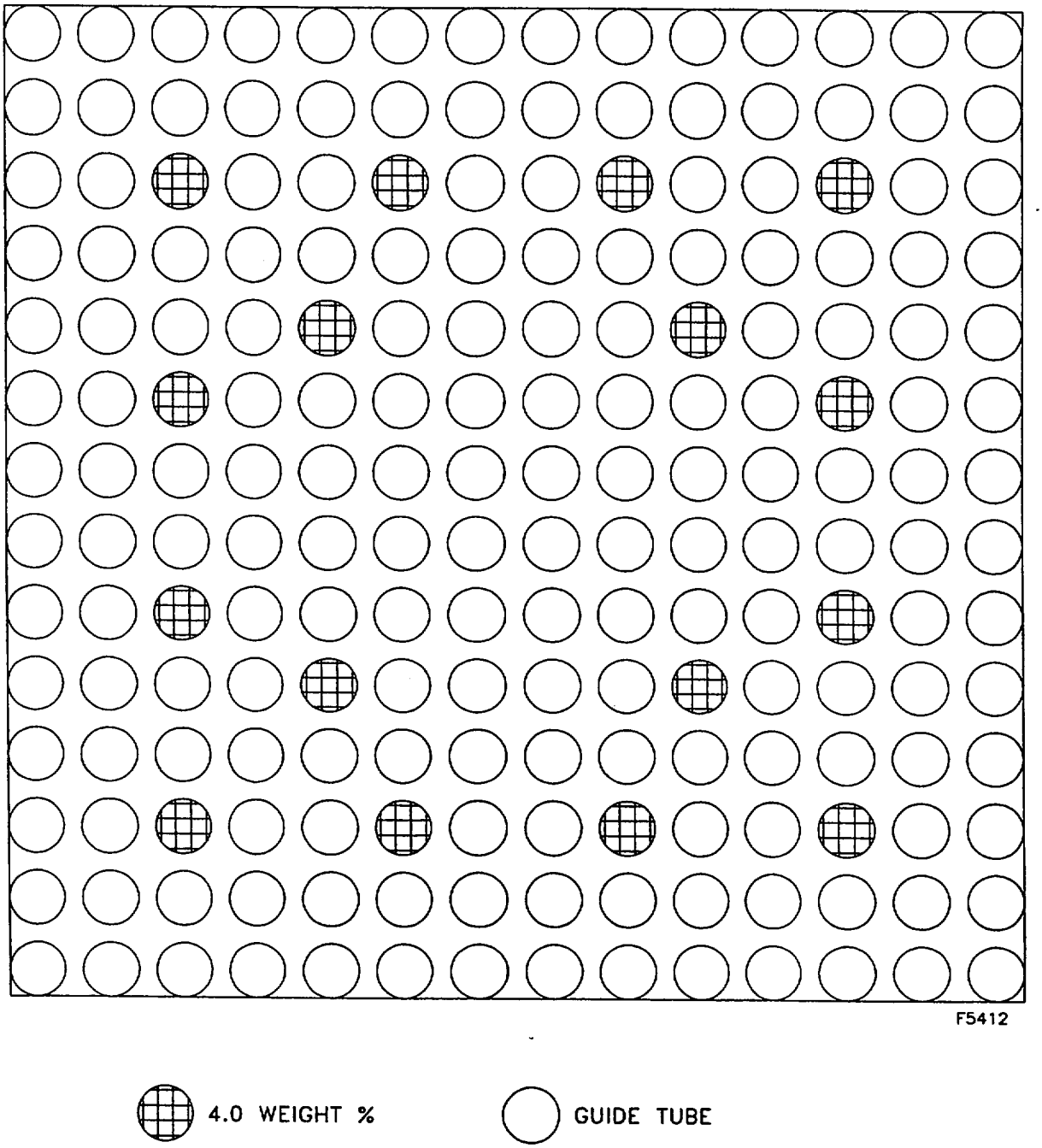

Figure 6.4-1
Rod Pitch Study: Figure Shows Maximum Rod Pitch Allowed

12.2.0 Functional and Operating Limits

12.2.1 Fuel To Be Stored In The 24PT1-DSC

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

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

b. Fuel types shall be limited to the following:

   UO_2 Westinghouse 14x14 (WE 14x14) Assemblies (with or without IFBA fuel rods), as specified in Table 12.2-1.

   WE 14x14 Mixed Oxide (MOX) Assemblies, as specified in Table 12.2-1

   Fuel burnup and cooling time is to be consistent with the limitations specified in Table 12.2-4 for UO2 fuel.

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

c. The maximum heat load for a single fuel assembly, including control components, is 0.583 kW. The maximum heat load per 24PT1-DSC, including any integral Control Components, shall not exceed 14 kW.

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

   1) A maximum of 24 INTACT WE 14x14 MOX or SC fuel assemblies; or

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

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

   A 24PT1-DSC containing less than 24 fuel assemblies may contain dummy fuel assemblies in fuel assembly slots. *The dummy fuel assemblies are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a fuel assembly. The effect of dummy assemblies or empty fuel*

assembly slots on the radial center of gravity of the DSC must meet the requirements of Section 12.2.1.b.

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

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

e. Fuel dimensions and weights are provided in Table 12.2-2.

f. The maximum neutron and gamma source terms are provided in Table 12.2-3.

12.2.2 Functional and Operating Limits Violations

If any Functional and Operating Limit of 12.2.1 is violated, the following actions shall be completed:

12.2.2.1 The affected fuel assemblies shall be placed in a safe condition.

12.2.2.2 Within 24 hours, notify the NRC Operations Center.

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

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 Poison

Neutron poison in the configuration shown in the canister drawing provided in Section 1.5.2 with a minimum ^{10}B loading of 0.025 grams/square centimeter is provided for criticality.

12.4.2.4 Canister Flux Trap Configuration

The canister flux trap configuration is defined by the spacer disc ligament width dimensions. *Figure 12.4-1 shows the location and dimensions of the ligaments (the dimensions shown in the one quadrant are applicable to all four quadrants).*

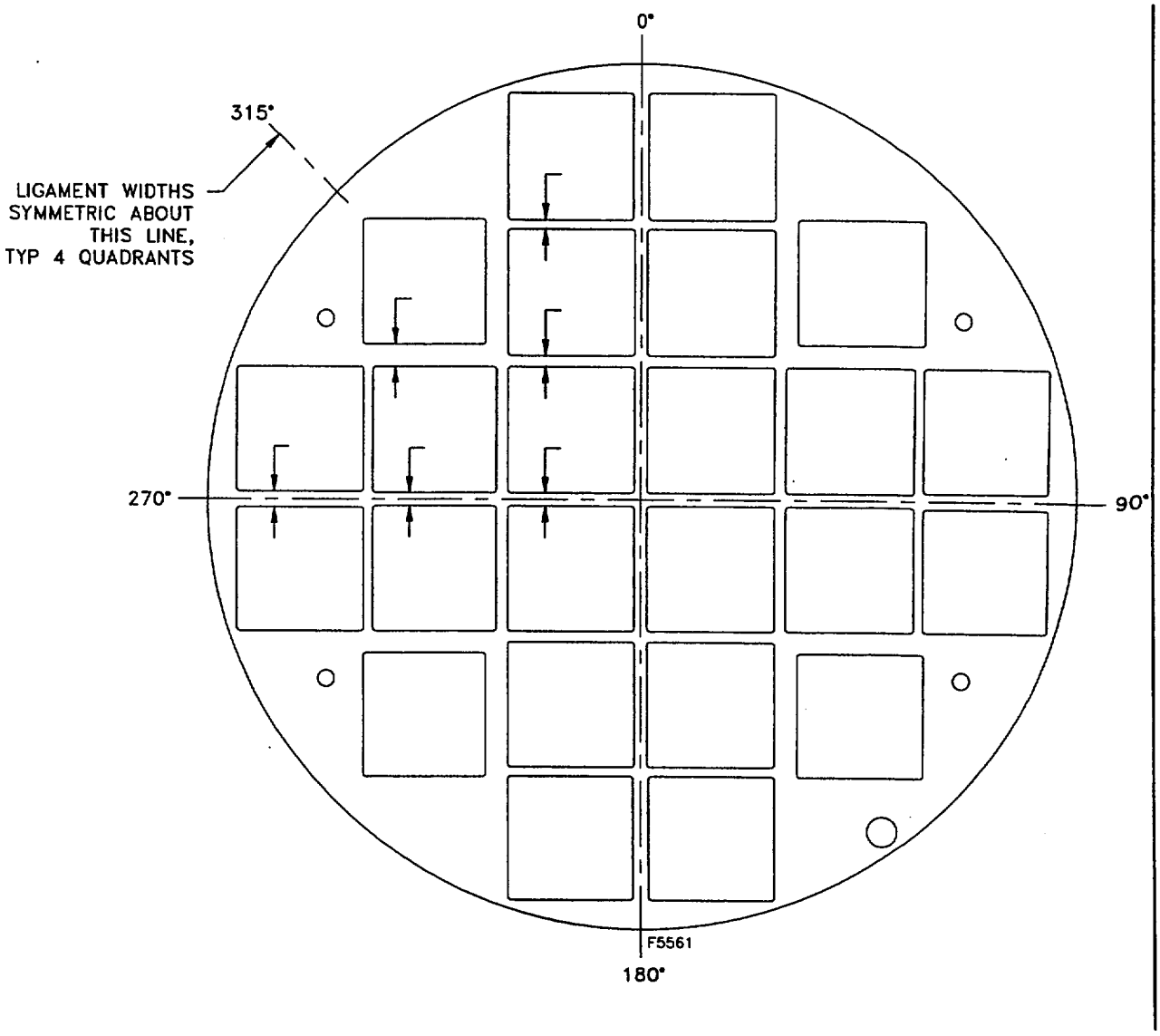

Figure 12.4-1
Minimum Spacer Disc Ligament Widths

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