72-1029

TRANSNUCLEAR WEST

June 22, 2001
DCS-TNW0106-13
RMG-01-026

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

Subject: Supplemental Response to Request for Additional Information and Submittal of Revision 2 of the Advanced NUHOMS® Storage System Application (TAC No. L23203)

References:
1. Request for Additional Information Regarding Approval of Advanced NUHOMS® Storage System (TAC No. L23203), March 5, 2001
2. Response to Request for Additional Information and Submittal of Revision 1 of Advanced NUHOMS® Storage System Application (TAC No. L23203), May 18, 2001 (DCS-TNW0105-12)

Dear Mr. Kobetz:

Transnuclear West Inc. herewith submits supplemental responses to specific questions of the Reference 1 RAI to provide the requested clarification sought by your staff in telecons the weeks of 6/11/01 and 6/18/01. The information provided in the supplemental response supercedes the corresponding information related to these specific RAI issues submitted previously in Reference 2. In addition, the affected pages of the proprietary and non-proprietary version of the Advanced NUHOMS® SAR have been updated and are included in this submittal. An affidavit for withholding proprietary information contained in this transmittal is also attached.

Please contact Mr. U. B. Chopra (510-744-6053) or me (510-744-6020) if you require any additional information in support of this submittal.

NMSS01 Prop

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

Sincerely,

Robert M. Grenier
President and Chief Operating Officer

Docket 72-1029

Attachments:

1. Affidavit for withholding proprietary information
2. Supplemental Responses to the RAI (non-proprietary, 14 copies)
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

AFFIDAVIT PURSUANT
TO 10 CFR 2.790

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

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

The information for which proprietary treatment is sought is contained in the documents included in Attachment 3 of this submittal and as listed below:

- Advanced NUHOMS SAR, Revision 2.

These sections of the document have been appropriately designated as proprietary.

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

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

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

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

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

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

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

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

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

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

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

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

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

Further the deponent sayeth not.

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

Robert M Grenier

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

Subscribed and sworn to me before this 22nd day of June, 2001, by Robert M. Grenier.

Shannon L. Christensen

Notary Public

Attachment 2

SUPPLEMENTAL
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
TRANSNUCLEAR WEST INC., TAC NO. L23203
(Supplement to TNW Letter DCS-TNW0105-12, dated 5/18/01)

Note: The responses below include only questions for which a supplemental response was required. These responses include the initial response with the changes comprising the supplemental response identified by revision bars.

Question 3-1

Justify the 392 degrees F maximum concrete temperature for accident conditions given in Table 4.1-5. This temperature exceeds the allowable range of 0 - 350 degrees F stated in the table.

Note 2 in Table 4.1-5 states testing will be performed to document that concrete compression strength will be greater than that assumed in structural analyses. The tests are to be on the exact concrete mix and are to acceptably demonstrate the level of strength reduction which needs to be applied, and to show that the increased temperatures do not cause deterioration of the concrete either with or without load. However, there is no discussion of what type of testing will be performed and why the testing is sufficient to confirm performance of the AHSM. The test details are required for the staff to assess compliance with 10 CFR 72.146 (b).

Response to Question 3-1

As described in the SAR, the following acceptance criteria apply to the concrete materials used in the construction of the AHSM:

- Satisfy ASTM C 33 requirements and other requirements referenced in ACI 349 for aggregates, and

- Have demonstrated a coefficient of thermal expansion (tangent in temperature range of 70°F to 100°F) no greater than 6x10-6 in/in/°F, or be one, or a mixture of the following minerals: limestone, dolomite, marble, basalt, granite, gabbro, or rhyolite.

- If concrete temperatures of general or local areas in normal conditions do not exceed 200°F and in off-normal conditions do not exceed 225°F, in addition to the above list of acceptable aggregates, quartz sands and sandstone sands are also acceptable as a fine aggregate only.

The calculated temperatures within the concrete AHSM demonstrate that the concrete meets ACI 349 and NUREG 1536 temperature criteria for all normal and off-normal cases. There are three areas of the AHSM predicted to experience temperatures in excess of 350°F during a 40 hour duration blocked vent accident that require evaluation for acceptability to ACI 349-97 criteria, as modified by NUREG 1536. The subject areas are as follows:

1. An area of 101" (along the 24PT1-DSC longitudinal axis) x 36" x 2.2" deep on each of the side walls. These areas are centered about the 24PT1-DSC centroid projected horizontally onto the two side walls.
2. An area of 85" (along the 24PT1-DSC longitudinal axis) x 52" x 4" deep immediately above the heat shield opening into the vent system. This area is centered about the 24PT1-DSC centroid projected onto the roof. This area is the top part of the base block with vent openings, which is not required for transferring structural loads.
3. An area 6"x 2"x 1.5"deep located on the top surface of the front vent shielding block under the longitudinal axis of the 24PT1-DSC. Other areas of the shield block that are less than 350°F provide support for the 24PT1-DSC support structure steel.

The proposed elevated temperature testing for the AHSM design mix for the storage block will satisfy the following:

- A minimum of two sets of five 6" x 12" cylinders shall be made from a test batch,
- One set of cylinders will be used as the control (room temperature) set; the second set will be heated to 400 °F at a rate similar to that predicted by the thermal analysis for the high temperature areas. The maximum temperature will be held at a steady state for a time of at least 36 hours to exceed the anticipated effects of the blocked vent case,
- The heated cylinders shall be examined for soundness prior to strength tests. The concrete shall not show signs of spalling, cracks and/or loss of cement bond to aggregate due to the elevated temperatures.
- Each set of cylinders will be broken using standard compressive strength test methods,
- The average test results for the high temperature set, reduced by two standard deviations, shall not be less than 4,500 psi. Computation of the standard deviation shall be consistent with applicable methods within ACI 214. If a proposed design mix does not have sufficient test data to compute a standard deviation, then an equivalent standard deviation shall be computed consistent with trial batch requirements specified within ACI 318.

The above tests are for concrete cylinders heated without load. Since the normal condition compressive stress in the volumes of concrete that exceed 350°F during accident conditions is relatively low, testing cylinders under load is not necessary to simulate service conditions. Relevant published test reports [3-1.1] indicate that testing cylinders under load produces higher compressive strength test results than identical cylinders tested without load. Therefore, heating and testing unrestrained cylinders (without load) is adequate and conservative.

This testing will be re-performed if the concrete mix is changed by the concrete fabricator or as a result of a change in concrete fabricators.

SAR Table 4.1-5, Note 2 has been revised to incorporate a summary of the above discussion.

[3-1.1] M. S. Abrams, Compressive Strength of Concrete at Temperatures to 1600°F, ACI Special Publication SP25 (Paper SP 25-2), American Concrete Institute, Detroit, MI (1971).

Question 4-5

Include a discussion of MOX fuel He production effects on maximum nominal operating pressure and cladding failure.

Section 4.4.8 of the SAR refers to Table 4.4-9 and states that based on the information listed, the UO2 assemblies are bounding for the analysis. However, the SAR does not include a discussion of the MOX rod void volume, a plot of gas generation over time, including He, and cladding strength which would provide a complete characterization of the condition of the contents stored. This information is required by the staff to assess whether the fuel cladding is protected against degradation that could lead to gross ruptures in accordance with 10 CFR 71.122(h)(1).

Response to Question 4-5

A comparison of stainless steel clad fuel (SC) and mixed oxide fuel (MOX) parameters affecting the canister pressure analysis (fill pressure, fuel rod void volume, fission gas generation during operation and during fuel decay, and fuel cladding material strength) have been added to SAR Section 4.4.8 to clarify the basis for performance of the pressure analysis based on SC fuel only.

The MOX fuel fission gas generation is less than 50% of that of the SC fuel due to the lower design basis burnup (45 GWd/MTU for SC fuel and 25 GWd/MTU for MOX fuel) and longer design basis cooling time (10 years for SC fuel and 20 years for MOX fuel). The number of moles of fission gas generated is calculated using the SAS2H/ORIGEN S module from the SCALE 4.4 computer code package. A fuel assembly burnup of 45 GWd/MTU, 3.80 weight % U-235 initial enrichment and 10 year cooling time is used for the SC fuel.

Note that the neutron and gamma source terms used in the shielding analysis documented in Chapter 5 of the SAR are taken from these same output files. Table 12.2-4 contains fuel qualification criteria which requires that for a burnup of 45 GWd/MTU and 3.80 weight % U-235 initial enrichment, the minimum cooling time is 15.2 years. Therefore, the use of 10 years cooling time in the calculation of fission gas generation and also neutron and gamma source term is conservative.

Question 4-6

Clarify that a maximum heat load of 14 kW bounds the allowed contents of the 24PT1-DSC.

Section 1.2.1.1 of the SAR states that the 24PT1-DSC is designed for a maximum heat load of 14 kW. However, the staff found that the maximum heat load, based on allowable contents, is 16 kW. This information is required by the staff to assess compliance with 10 CFR 72.24(d)(1).

Response to Question 4-6

The maximum heat load is specified in SAR Section 12.2.1.c as a specific limitation to be confirmed by a licensee prior to storage of fuel in the 24PT1-DSC. However, to eliminate the need for further analysis to determine the heat load, Table 12.2-4, Fuel Qualification Table, has been added to the SAR to specify burnup/enrichment/cooling time limits to ensure a heat load of $\leq$ 14 kW per 24PT1-DSC.

In conjunction with this fuel qualification table, the fuel enrichment specification in SAR Tables 12.2-1, 12.2-2 and 12.2-4 have been revised to accommodate an uncertainty in the enrichment specified. Revisions to enrichment values specified in SAR Sections 1.2.3, 2.1.1, Tables 2.1-1 and 2.1-2 have been revised. An analysis of the effects of these uncertainties have been added to the SAR in Sections 5.2.3 and 6.4.4. The effect of these uncertainties on criticality analysis results is addressed in SAR Sections 6.1 and 6.4.3.

Page 6.4-5 of the Revision 1 SAR has been corrected to reflect the change in k_{eff} as an increase from .9368 to .9392 (including 2σ).

Question 6-7

Describe in greater detail in Section 6 how the Upper Subcritical Limit (USL) was determined.

The SAR does not discuss any bias and uncertainty associated with the USL determination, nor does it discuss any uncertainty due to modeling approximations. Note that only biases that increase keff should be applied. This is required for the staff to assess compliance with 10 CFR 72.124.

Response to Question 6-7

SAR Section 6.5.1 has been revised to incorporate additional discussion of the method used for calculation of the Upper Subcritical Limit (USL). The methodology used is based on NUREG/CR-6361, USL method 1.

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 of the SAR. 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 of the SAR. The results demonstrate that the calculated changes in reactivity between the various cladding ODs are within the statistical uncertainty of the calculations.

The allowance for empty fuel assembly slots and dummy fuel assemblies per SAR Section 12.2.1 is bounded by the criticality analyses in Chapter 6. The dummy fuel assemblies are unirradiated, stainless steel encased structures, that approximate the weight and center of gravity of a fuel assembly (these requirements for the dummy fuel assemblies have been added to SAR Section 12.2.1.d). The criticality analysis performed for 24 fuel assemblies bounds the configuration with less than 24 assemblies with some of the fuel assembly slots left open or replaced by a dummy fuel assembly 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 the fuel in the adjacent guidesleeve. This effect can be seen in the results provided in SAR Table 6.4-1 for the Normal Operating Condition, Assembly Position Case . In this analysis 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 and increased moderator between the center assemblies reduces $k_{eff} + 2\sigma$ from 0.8659 to 0.8404. This is a smaller reduction in moderation at the center of the DSC than that resulting from a missing or dummy fuel assembly.

Question 6-13

Revise Section 12.4.0 to include the basket B-10 loading and the flux trap size.

The B-10 loading and flux trap size are design parameters important to criticality safety. This is required by the staff to assess compliance with 10 CFR 72.24(g), 72.26, and 72.44(c).

Response to Question 6-13

SAR Section 12.4 has been revised to incorporate a new Section 12.4.2.3, titled "Canister Neutron Poison" which now specifies the minimum B-10 loading of 0.025 grams/cm^2. A new Section 12.4.2.4, titled "Canister Flux Trap Configuration" has also been added to specify the flux trap size.

ANUH-01.0150
Revision 2

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 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.

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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.

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 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 6.4-4
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 6.4-5
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)
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

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).

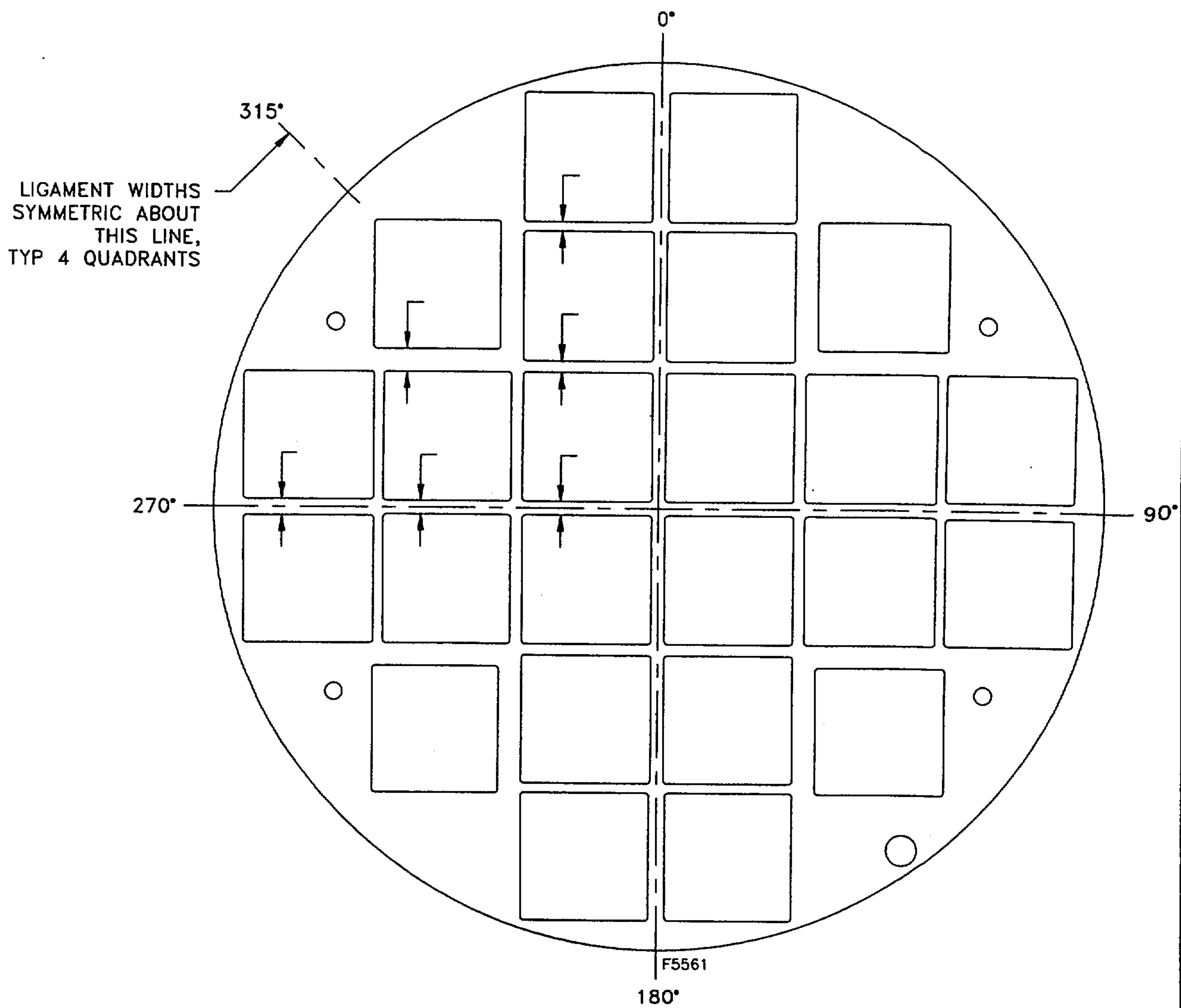

Figure 12.4-1
Minimum Spacer Disc Ligament Widths

Attachment 2

SUPPLEMENTAL
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
TRANSNUCLEAR WEST INC., TAC NO. L23203
(Supplement to TNW Letter DCS-TNW0105-12, dated 5/18/01)

Note: The responses below include only questions for which a supplemental response was required. These responses include the initial response with the changes comprising the supplemental response identified by revision bars.

Question 3-1

Justify the 392 degrees F maximum concrete temperature for accident conditions given in Table 4.1-5. This temperature exceeds the allowable range of 0 - 350 degrees F stated in the table.

Note 2 in Table 4.1-5 states testing will be performed to document that concrete compression strength will be greater than that assumed in structural analyses. The tests are to be on the exact concrete mix and are to acceptably demonstrate the level of strength reduction which needs to be applied, and to show that the increased temperatures do not cause deterioration of the concrete either with or without load. However, there is no discussion of what type of testing will be performed and why the testing is sufficient to confirm performance of the AHSM. The test details are required for the staff to assess compliance with 10 CFR 72.146 (b).

Response to Question 3-1

As described in the SAR, the following acceptance criteria apply to the concrete materials used in the construction of the AHSM:

- Satisfy ASTM C 33 requirements and other requirements referenced in ACI 349 for aggregates, and
- Have demonstrated a coefficient of thermal expansion (tangent in temperature range of 70°F to 100°F) no greater than 6x10-6 in/in/°F, or be one, or a mixture of the following minerals: limestone, dolomite, marble, basalt, granite, gabbro, or rhyolite.
- If concrete temperatures of general or local areas in normal conditions do not exceed 200°F and in off-normal conditions do not exceed 225°F, in addition to the above list of acceptable aggregates, quartz sands and sandstone sands are also acceptable as a fine aggregate only.

The calculated temperatures within the concrete AHSM demonstrate that the concrete meets ACI 349 and NUREG 1536 temperature criteria for all normal and off-normal cases. There are three areas of the AHSM predicted to experience temperatures in excess of 350°F during a 40 hour duration blocked vent accident that require evaluation for acceptability to ACI 349-97 criteria, as modified by NUREG 1536. The subject areas are as follows:

1. An area of 101" (along the 24PT1-DSC longitudinal axis) x 36" x 2.2" deep on each of the side walls. These areas are centered about the 24PT1-DSC centroid projected horizontally onto the two side walls.
2. An area of 85" (along the 24PT1-DSC longitudinal axis) x 52" x 4" deep immediately above the heat shield opening into the vent system. This area is centered about the 24PT1-DSC centroid projected onto the roof. This area is the top part of the base block with vent openings, which is not required for transferring structural loads.
3. An area 6"x 2"x 1.5"deep located on the top surface of the front vent shielding block under the longitudinal axis of the 24PT1-DSC. Other areas of the shield block that are less than 350°F provide support for the 24PT1-DSC support structure steel.

The proposed elevated temperature testing for the AHSM design mix for the storage block will satisfy the following:

- A minimum of two sets of five 6" x 12" cylinders shall be made from a test batch,
- One set of cylinders will be used as the control (room temperature) set; the second set will be heated to 400 °F at a rate similar to that predicted by the thermal analysis for the high temperature areas. The maximum temperature will be held at a steady state for a time of at least 36 hours to exceed the anticipated effects of the blocked vent case,
- The heated cylinders shall be examined for soundness prior to strength tests. The concrete shall not show signs of spalling, cracks and/or loss of cement bond to aggregate due to the elevated temperatures.
- Each set of cylinders will be broken using standard compressive strength test methods,
- The average test results for the high temperature set, reduced by two standard deviations, shall not be less than 4,500 psi. Computation of the standard deviation shall be consistent with applicable methods within ACI 214. If a proposed design mix does not have sufficient test data to compute a standard deviation, then an equivalent standard deviation shall be computed consistent with trial batch requirements specified within ACI 318.

The above tests are for concrete cylinders heated without load. Since the normal condition compressive stress in the volumes of concrete that exceed 350°F during accident conditions is relatively low, testing cylinders under load is not necessary to simulate service conditions. Relevant published test reports [3-1.1] indicate that testing cylinders under load produces higher compressive strength test results than identical cylinders tested without load. Therefore, heating and testing unrestrained cylinders (without load) is adequate and conservative.

This testing will be re-performed if the concrete mix is changed by the concrete fabricator or as a result of a change in concrete fabricators.

SAR Table 4.1-5, Note 2 has been revised to incorporate a summary of the above discussion.

[3-1.1] M. S. Abrams, Compressive Strength of Concrete at Temperatures to 1600°F, ACI Special Publication SP25 (Paper SP 25-2), American Concrete Institute, Detroit, MI (1971).

Question 4-5

Include a discussion of MOX fuel He production effects on maximum nominal operating pressure and cladding failure.

Section 4.4.8 of the SAR refers to Table 4.4-9 and states that based on the information listed, the UO2 assemblies are bounding for the analysis. However, the SAR does not include a discussion of the MOX rod void volume, a plot of gas generation over time, including He, and cladding strength which would provide a complete characterization of the condition of the contents stored. This information is required by the staff to assess whether the fuel cladding is protected against degradation that could lead to gross ruptures in accordance with 10 CFR 71.122(h)(1).

Response to Question 4-5

A comparison of stainless steel clad fuel (SC) and mixed oxide fuel (MOX) parameters affecting the canister pressure analysis (fill pressure, fuel rod void volume, fission gas generation during operation and during fuel decay, and fuel cladding material strength) have been added to SAR Section 4.4.8 to clarify the basis for performance of the pressure analysis based on SC fuel only.

The MOX fuel fission gas generation is less than 50% of that of the SC fuel due to the lower design basis burnup (45 GWd/MTU for SC fuel and 25 GWd/MTU for MOX fuel) and longer design basis cooling time (10 years for SC fuel and 20 years for MOX fuel). The number of moles of fission gas generated is calculated using the SAS2H/ORIGEN S module from the SCALE 4.4 computer code package. A fuel assembly burnup of 45 GWd/MTU, 3.80 weight % U-235 initial enrichment and 10 year cooling time is used for the SC fuel.

Note that the neutron and gamma source terms used in the shielding analysis documented in Chapter 5 of the SAR are taken from these same output files. Table 12.2-4 contains fuel qualification criteria which requires that for a burnup of 45 GWd/MTU and 3.80 weight % U-235 initial enrichment, the minimum cooling time is 15.2 years. Therefore, the use of 10 years cooling time in the calculation of fission gas generation and also neutron and gamma source term is conservative.

Question 4-6

Clarify that a maximum heat load of 14 kW bounds the allowed contents of the 24PT1-DSC.

Section 1.2.1.1 of the SAR states that the 24PT1-DSC is designed for a maximum heat load of 14 kW. However, the staff found that the maximum heat load, based on allowable contents, is 16 kW. This information is required by the staff to assess compliance with 10 CFR 72.24(d)(1).

Response to Question 4-6

The maximum heat load is specified in SAR Section 12.2.1.c as a specific limitation to be confirmed by a licensee prior to storage of fuel in the 24PT1-DSC. However, to eliminate the need for further analysis to determine the heat load, Table 12.2-4, Fuel Qualification Table, has been added to the SAR to specify burnup/enrichment/cooling time limits to ensure a heat load of ≤ 14 kW per 24PT1-DSC.

In conjunction with this fuel qualification table, the fuel enrichment specification in SAR Tables 12.2-1, 12.2-2 and 12.2-4 have been revised to accommodate an uncertainty in the enrichment specified. Revisions to enrichment values specified in SAR Sections 1.2.3, 2.1.1, Tables 2.1-1 and 2.1-2 have been revised. An analysis of the effects of these uncertainties have been added to the SAR in Sections 5.2.3 and 6.4.4. The effect of these uncertainties on criticality analysis results is addressed in SAR Sections 6.1 and 6.4.3.

Page 6.4-5 of the Revision 1 SAR has been corrected to reflect the change in k_{eff} as an increase from .9368 to .9392 (including 2σ).

Question 6-7

Describe in greater detail in Section 6 how the Upper Subcritical Limit (USL) was determined.

The SAR does not discuss any bias and uncertainty associated with the USL determination, nor does it discuss any uncertainty due to modeling approximations. Note that only biases that increase keff should be applied. This is required for the staff to assess compliance with 10 CFR 72.124.

Response to Question 6-7

SAR Section 6.5.1 has been revised to incorporate additional discussion of the method used for calculation of the Upper Subcritical Limit (USL). The methodology used is based on NUREG/CR-6361, USL method 1.

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 of the SAR. 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 of the SAR. The results demonstrate that the calculated changes in reactivity between the various cladding ODs are within the statistical uncertainty of the calculations.

The allowance for empty fuel assembly slots and dummy fuel assemblies per SAR Section 12.2.1 is bounded by the criticality analyses in Chapter 6. The dummy fuel assemblies are unirradiated, stainless steel encased structures, that approximate the weight and center of gravity of a fuel assembly (these requirements for the dummy fuel assemblies have been added to SAR Section 12.2.1.d). The criticality analysis performed for 24 fuel assemblies bounds the configuration with less than 24 assemblies with some of the fuel assembly slots left open or replaced by a dummy fuel assembly 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 the fuel in the adjacent guidesleeve. This effect can be seen in the results provided in SAR Table 6.4-1 for the Normal Operating Condition, Assembly Position Case . In this analysis 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 and increased moderator between the center assemblies reduces $k_{eff} + 2\sigma$ from 0.8659 to 0.8404. This is a smaller reduction in moderation at the center of the DSC than that resulting from a missing or dummy fuel assembly.

Question 6-13

Revise Section 12.4.0 to include the basket B-10 loading and the flux trap size.

The B-10 loading and flux trap size are design parameters important to criticality safety. This is required by the staff to assess compliance with 10 CFR 72.24(g), 72.26, and 72.44(c).

Response to Question 6-13

SAR Section 12.4 has been revised to incorporate a new Section 12.4.2.3, titled "Canister Neutron Poison" which now specifies the minimum B-10 loading of 0.025 grams/cm^2. A new Section 12.4.2.4, titled "Canister Flux Trap Configuration" has also been added to specify the flux trap size.