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Project M-39

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MEMORANDUM FOR: Leland C. Rouse, Chief
Advanced Fuel and Spent Fuel Licensing Branch
Division of Fuel Cycle and Material Safety

FROM: John P. Roberts
Advanced Fuel and Spent Fuel Licensing Branch
Division of Fuel Cycle and Material Safety

SUBJECT: MEETING WITH NUTECH, INC.

Date and Time: August 27, 1985: 9 a.m.

Location: 5th floor conference room, Willste Building
Silver Spring, MD

Attendees:

NUTECH	NRC
J. Massey	J. Roberts
	J. Schneider
	W. Macnabb (SAIC)

Purpose: A discussion of NUTECH's approach to responding to NRC staff comments on NUTECH's topical report for a concrete module design independent spent fuel storage installation (ISFSI).

Discussion:

NUTECH had prepared preliminary responses (see enclosure), and these were examined and discussed. Issues involving criticality, thermal hydraulic, and shield penetration calculations remain open for further discussion in a full meeting between NRC and NUTECH staff tentatively scheduled for September 10-11, 1985. In the interim, NRC staff will consider NUTECH's preliminary responses. NUTECH will provide responses in finished form at our next meeting and expects to submit a revised topical report on September 30, 1985.

Original signed by
John P. Roberts

John P. Roberts
Advanced Fuel and Spent Fuel
Licensing Branch

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Enclosure: As stated

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DETAILED COMMENTS ON NUTECH
TOPICAL REPORT NUH-001

QUESTION: 1

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

1.1 Introduction

A helium storage atmosphere is specified within the DSC. What measures has NUTECH taken to assure that the helium initially placed in the DSC will not leak to the atmosphere? What is the maximum leakage rate considering the integrity of all welds and diffusion through the canister? What measures are available to monitor the DSC atmosphere composition?

RESPONSE:

NUTECH has included a number of measures in the design, fabrication, and operation of NUHOMS that will ensure the helium initially placed in the DSC will not leak to the atmosphere. Nor will the oxygen in the atmosphere leak or diffuse into the canister. The longitudinal and bottom closure welds of the DSC are designed and will be fabricated in accordance with the ASME Boiler and Pressure Code, an internationally accepted code for the design and fabrication of leak tight containment vessels. The design of the DSC also includes redundant closure welds on the top. All welds that are on the pressure retaining boundaries of the DSC will undergo nondestructive examination. The nondestructive examination includes:

- Ultrasonic or Radiographic Testing	o All shop welds on pressure retaining boundaries o Secondary closure welds
- Dye Penetration	Primary closure weld
- Helium Leak Test	Primary closure weld

In regards to diffusion, discussions with scientists in the helium industry (at Union Carbide and the U.S. Bureau of Mines' Helium Operation) indicate that as a practical matter, helium does not diffuse through steel or stainless steel. The scarcity of references in the

literature seems to bear this out. It was further discussed that significant helium diffusion only occures under condition of a hard vacuum. In the Metals Handbook, R.M. Parke flatly states that "Helium does not diffuse through solid iron".(A)

However, since the reviewers' concern with diffusion is evident, we have calculated the diffusion rate to be on the order of 10^{-8}g-moles/year at nominal design conditions. Diffusivity for this system varies exponentially with temperature so that in an accident condition with elevated temperatures, the diffusion rate could be expected to increase by as much as 3 orders of magnitude. The leakage rate would remain, however, of no practical consequence.

DETAILED COMMENTS ON NUTECH
TOPICAL REPORT NUH-001

QUESTION: 2

1.2.1 General Description of Installation

Even though it may be possible to design a single stand-alone horizontal storage module (HSM) or physical arrangements in other than a 4x2 array, the design contained in the current Topical Report is limited to a 4x2 array (see p. 8.2-13). The Topical Report should clearly state which arrangements(s) are being proposed for review and maintain consisency thoughout the document.

RESPONSE:

The 4x2 array is the only design which is to be reviewed in this Topical Report. The Topical Report will be revised so that consistency is maintained throughout the document.

DETAILED COMMENTS ON NUTECH
TOPICAL REPORT NUH-001

QUESTION: 3

1.2.2 Principal Design Criteria

1. Are there limits on the kind of external atmosphere allowed for this system in order to control corrosion of exposed surfaces or to control growth of algae or other vegetative matter within the HSM? Please discuss.

RESPONSE:

Because of the corrosion resistent properties of components or the coatings used in the NUHOMS system and the hot, drying environment which exists within the HSM no limits on the kind of external atmosphere are required.

All components are either 304 stainless steel or ATSM 36 steel, galvanized or coated with Carbo Zinc 11, an inorganic, zinc based corrosion protection coating.

The interior of the HSM is all cement and steel and is void of any substances which would be conducive to the growth of organic matter. The design has been changed to include a drainage line in the air inlet chamber. The drainage line will drain any water which enters through the air inlets. The air outlets located on the roof of the module are designed to be water tight. Any water entering into the HSM cavity, by this route will evaporate long before any vegetative matter could establish itself and disrupt the air flow through the HSM. The evaporation will be rapid because the module floor is 50 to 70°F above the incoming air temperature due to the heat radiated from the canister. Additionally, as documented in many studies on the environmental effects of radiation, radiation is not conducive to the growth of organic matter.

DETAILED COMMENTS ON NUTECH
TOPICAL REPORT NUH-001

QUESTION: 4

1.2.2 Principal Design Criteria

2. No mention is made of the duty cycle allowed for the materials (particulary concrete). Please provide some discussion of how the duty cycles are addressed or why they do not need to be addressed.

RESPONSE:

According to the ASHRAE Handbook, 1981 Fundamentals, the largest mean daily change of temperature in the United States occurs in Reno, Nevada and is 45°F. Assuming this maximum temperature change occurs every day for 50 years the HSM will experience 18250 thermal cycles. The maximum moment caused by normal operating thermal loads is 2496.2 k. in. This loading is only 62% of the ultimate strength. Consequently, refering to the S-N curve of Figure 6-41 of the handbook of concrete engineering by Fintel, the number of cycles before failure will occur, is approximately 10,000,000. Since only 18250 cycles will occur in the 50 year life, fatigue failure of concrete is not possible. To show the negligible effect of the duty cycle on the fatigue strength of the DSC the fatigue analysis of section 8.2.10 is reevaluated. Overly conservative assumptions are made and daily thermal/pressure cycling is included in this analysis.

Conservatively, the operating pressure loading is assumed to change from zero to operating pressure each day. Secondly, accident pressure loading is assumed to occur once a year for each of the 50 years of design life. One major seismic event is assumed to occur with 100 damaging cycles. For seasonal changes a WT = 175.7°F is used which envelopes the WT from the -40°F to +125°F inlet temparature load cases. As stated previously the largest daily range in the United States is 45°F. The WT is applied each day for the 50 year lifetime to account for daily temperature cycling. All of the five (5) loadings are combined in a histogram. ASME Figure I-9.2 is used to determine the fatigue usage factor for the DSC shell. A value of 0.21 was calculated; therefore, the DSC shell is adequate for cyclic loadings.

DETAILED COMMENTS ON NUTECH
TOPICAL REPORT NUH-001

QUESTION: 5

1.2.4 Safety Features

What are the values for cask movement, cask head and truck transport which were not included in Table 1.2-4?

RESPONSE:

There are no values associated with cask movement, cask head, and truck transport. These are operational capabilities which an system must include in order to be compatible with the NUHOMS system. Table 1.2-4 will be revised to indicate N/A for these parameters. Some of the qualitative parameters which must be met are listed below.

- The combined weight of the cask and the loaded DSC must not exceed the lifting capacity of the sites existing spent fuel handling crane.

The cask must be capable of being rotated by the sites existing crane from a vertical position to a horizontal position.

The cask lid must be removable while the cask is in a horizontal position.

The size of the cask and the combined weight of the cask and the loaded DSC must not be so excessive that they could not be economically transported to the HSM by a tractor-trailer arrangement.

DETAILED COMMENTS ON NUTECH
TOPICAL REPORT NUH-001

QUESTION: 6

3.0 PRINCIPAL DESIGN CRITERIA

3.1.1.1 Physical Characteristics

Based on the data on pages 3.1-1 and 3.1-3 it appears that NUTECH has used the Westinghouse fuel assembly array of 15x15/204. Examination of the data in Table 3.1-2 reveals that several fuel arrays are not enveloped by the 15x15/204 array, namely the B&W 15x15/208, the B&W 17x17/264, the CE 15x15/208, 212 and 216, and the CE 16x16/224 to 236. The text and the tables should be consistent and any fuel assembly types not enveloped by the design criteria case should be deleted or the design modified to envelope all the listed fuel types. In addition to these comments, are there any design parameters of the reference fuel assembly physical characteristics which would lead to a non-conservative analysis?

RESPONSE:

The main structural criteria for choosing an enveloping fuel assembly is the weight distribution per spacer disk Spacer disks are located at fuel assembly grid spacer locations to assure proper support. The weight per spacer disk is obtained by multiplying the distributed weight times the maximum length between spacer disks. Utilizing this procedure, the Westinghouse 15x15/204 envelopes all fuel assemblies in Table 3.1-2 except the combustion engineering 15x15/208, 212, 216. This PWR fuel assembly will be deleted from Table 3.1-2. The following table shows the weight per spacer disk for the various assemblies.

Fuel Assembly	Wt. Per Spacer Disk (kg)
Westinghouse 15x15/204	106.56
Babcock & Wilcox 15x15/208	87.73
B&W 17x17/264	90.41
Combustion Engin. 14x14/176	64.14
C.E. 15x15/208, 212, 216	127.46 (Not Enveloped)
Westinghouse 14x14/179	93.15
Westinghouse 17x17/264	101.66

Some of the aforementioned assemblies weigh more than the Westinghouse 15x15/204, the maximum increase is 83.8 lb per assembly or 583.1 lb per canister. This causes a dead weight increase for a loaded canister from 21236 lb to 21819 lb. This 2.7% increase will cause minimal stress increase on the DSC shell, DSC support assembly, and the HSM. Since significant safety margin exists for these components, the 2.7% increase is negligible.

No other design parameters exist for the referenced fuel assemblies which could lead to non-conservative analyses. Consequently all fuel assemblies in Table 3.1-2 are enveloped except the Combustion Engineering 15x15/207, 212, 216. This array will be deleted from the table.

DETAILED COMMENTS ON NUTECH
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QUESTION: 7

3.1.2.2 Handling and Transfer Equipment

1. If NUTECH intends to assume that the transfer cask will be handled in the reactor building and during loading of the cask onto the transfer vehicle by a single failure-proof crane, it should state that criterion in this section.

RESPONSE:

The transfer cask and irradiated fuel will be handled within the fuel pool and reactor building in accordance with the requirements and criteria already approved in the sites FSAR and in accordance with the sites existing procedures. Information on cask and fuel handling procedures must be provided on a site specific basis. These procedures must show that the transfer cask will be handled in a manner that will not present a safety hazard. That is, the impact energy absorbing properties of the cask in use and the limitations on the drop height are such the resulting impact deceleration is enveloped by the values given in section 8.2 of the Topical Report.

QUESTION: 8

3.1.2.2 Handling and Transfer Equipment

2. In order for the five foot horizontal drop criterion to be valid, NUTECH should specify dimensional criteria for the cask skid and the transfer trailer to show that the maximum conceivable horizontal drop height is five feet of less.

RESPONSE:

The type of cask and transfer trailer is site specific and should be included in the site's FSAR. The Topical Report will be changed to state that the limitation is the enveloping deceleration load (48gs) for which the DSC was designed (section 8.2), not a drop height. The five foot drop criterion previously specified corresponds to the maximum deceleration which the DSC is capable of withstanding in a GE IF-300 cask dropped on an unyielding surface. Different cask, trailers, and ground conditions at various sites may significantly effect any actual drop height limitation. Indeed, drops on actually existing surfaces on reactor sites wll yield much higher allowable drop heights.

DETAILED COMMENTS ON NUTECH
TOPICAL REPORT NUH-001

QUESTION: 9

3.1.2.2 Handling and Transfer Equipment

3. What vertical acceleration factors have been measured or will be assumed for design criteria for the transportation of the GE IF-300 cask loaded with the NUTECH DSC while being transported on the trailer? Show that these accelerations have been enveloped by the five foot drop analysis.

RESPONSE:

The vertical acceleration which the DSC will be subjected to while in transport to and from the HSM will depend on the type of trailer, travel speed of the trailer, and type of road surface. All these factors are site specific.

However, 10 CFR71.31 (d) (1) requires the vertical acceleration design criteria for cask tie downs to be 2g. If this same criteria is assumed for the transportation of the DSC to and from the HSM, then this design criteria is bounded by the 34g vertical acceleration term associated with the drop accident analysis.

DETAILED COMMENTS ON NUTECH
TOPICAL REPORT NUH-001

QUESTION: 10

3.2 Structural and Mechanical Safety Criteria

Pages 3.2-4, 8.2-53 and 8.2-56 show inconsistent maximum internal pressurization levels of 44 and 39.7 psig. Also, associated with these pressures are inconsistent temperature levels of 423° and 413°C. NUTECH should state what the actual design parameters are and use them consistently throughout the Topical Report.

RESPONSE:

New helium filling procedures have been developed. The table below list the internal pressure of the DSC, both under both normal and abnormal conditions (100% rod fill gas release and 25% fission gas release) The Topical Report will be revised to include these new pressures and checked to make sure the numbers used are consistent throughout the text.

Table 3.2

Case	Air Temperature °F	Average Helium Temperature °F	Helium Pressure (Psia)	Partial Pressure[(1)] Fission of Fill Gas (Psia)	Total Pressure (Psia)	Total Pressure (Psig)
1	-40	315	14.2	20.92	35.12	20.42
2	70	389	14.5	22.92	37.42	22.72
3	125	429	15.2	23.99	39.19	24.49
4	Inlets Plugged	502	16.4	25.97	42.37	27.67
5	Complete blockage of inlets and outlets	775	21.05	33.33	54.38	39.68

(1) 100% of rod fill gas and 25% of fission gas released.

QUESTION: 11

3.2.3.1 Seismic Design

1. It is stated on Page 3.2-10 that a damping value of 2 percent of critical damping should be used for a large diameter piping system under a safe shutdown earthquake. This damping value is taken directly from NRC Regulatory Guide 1.61. Shouldn't this damping value be 3 percent?

RESPONSE:

The damping value of 2 percent has been changed to 3 percent.

QUESTION: 12

3.2.3.1 Seismic Design

2. It is stated on page 3.2-10 that the maximum horizontal ground acceleration selected for the design of the NUHOMS is 0.25g and the maximum vertical acceleration component selected is 0.1g. It is further stated in this paragraph that these ground acceleration values correspond to certain recommendations contained in 10 CFR 72.66(2)(ii). There does not appear to be any reference to a vertical ground acceleration level anywhere in 10 CFR 72.66. There is a reference in 10 CFR 72.66(a)(6)(ii) to a standardized ISFSI design earthquake response spectrum anchored at 0.25g. NRC Regulatory Guide 1.60 states that, depending on excitation frequency, that the vertical component of the design response spectra should be either the same as or 2/3 of the corresponding horizontal design spectra. Shouldn't the maximum vertical acceleration component selected for use in designing the NUHOMS be at least 0.17g rather than 0.1g if the maximum horizontal component is 0.25g?

   Additional guidance is given in NUREG-0800, Section 3.7.1 p.3.7.1-4, "To be acceptable the design response spectra should be specified for three mutually orthogonal directions; two horizontal and one vertical. Current practice is to assume that the maximum ground accelerations in the two horizontal directions are equal, while the maximum vertical ground acceleration is 2/3 of the maximum horizontal acceleration."

RESPONSE:

The vertical ground acceleration will be increased to 0.17g. All associated analysis will be revised to incorporate this change.

QUESTION: 13

3.2.3.2 Seismic System Analysis

On page 3.2-11 it is stated that the concrete coefficient of friction of 1.0 is taken from the ACI 349-80 code. However, this is only valid if concrete is placed against hardened concrete and the interface between the concrete surfaces is clean, free of laitance and "intentionally roughened to a full amplitude of approximately 1/4 inch." In the absence of concrete surface preparation specifications, NUTECH should use 0.6 as stated in ACI 318-83 Section 11.7.4.3. This concern arises again in Chapter 8.

RESPONSE:

The concrete coefficient of friction will be changed from 1.0 to 0.6. The only area effected by this change is the sliding of the single HSM under various accident loads. If necessary, a shear key type of connection located at the module walls or a tie-down system, depending on the type of concrete construction (i.e. cast in place vs precast) will be specified.

QUESTION: 14

3.2.5.2 Dry Shielded Canister

1. NUTECH has chosen ASME Level A service limits for normal operating conditions and Level D service limits for accident or faulted conditions. The use of Level D service limits, "permit gross general deformations with some consequent loss of dimensional stability and damage requiring repair, which may require removal of the component from service," (ASME, NCA - 2142.2(b)(4)). If Level D service limits are retained for accident or faulted conditions, NUTECH must state in the structural design criteria, that following an accident (such as the 5 foot drop and drop combined with pressure) which could cause damage to the DSC and DSC internal members, the DSC must be disassembled and inspected for damage.

RESPONSE:

The only accident condition that service level D allowables are used is the drop accident. For this condition the structural design criteria has been changed to state:

> "Following a drop accident the DSC must be opened and fuel assemblies removed and inspected for damage."

QUESTION: 15

3.2.5.2 Dry Shielded Canister

2. If Level D service limits are used, can NUTECH assure that possible deformation within the DSC following an accident will not inhibit the easy retrieval of all fuel assemblies (see Question 8, under 3.3.4 below).

RESPONSE:

For the horizontal drop accident only two components in the DSC basket, the spacer disk and the boral tube, could deform and cause interference with the fuel retrieval operation. To verify that no interference will occur, the maximum elastic deformation of the spacer disk and boral tubes obtained from the STARDYNE analysis reported in section 8.2 will be conservatively added to the maximum possible plastic deformation and then compared to the minimum gap between the fuel assembly and the inside of the boral tube. The maximum elastic deformation of the spacer disk is .0301" at the fuel cell centerline. The maximum boral tube deflection is .0072". An upper limit plastic analysis was performed for the spacer beams between two fuel cells, assuming a three hinged, uniformly loaded beam. The maximum deflection calculated was 0.016". The actual plastic deformation is conservatively enveloped by this value. Since the boral tube stress is far less than yield stress, no permanent deformation occurs. Summing the total calculated deflections yields .0533". The minimum gap is 0.324". Consequently no interference for the horizontal drop accident is present.

For the vertical orientation drop, the permenant deformation in the lead casing is by far smaller than the ½" gap specified between top of fuel & bottom surface of top lead plug. Clearly then, no interference will occur which could inhibit the easy retrieval of all fuel assemblies.

QUESTION: 16

3.3.3.2 Instrumentation

It is stated that there is no need to monitor the DSC internal pressure or temperature. Limiting pressure and temperature values are based on calculations. How does NUTECH assure that the calculational techniques are qualified for use in the design?

RESPONSE:

The code used to calculate the temperatures is a fully benchmarked code and the version which NUTECH used is fully verified and QA'ed. It is the code recommended by the NRC for such analysis and is used in the same manner as used by the NRC in the evaluation of cask internal temperatures.

QUESTION: 17

3.3.4 Nuclear Criticality Safety

1. Were analyses of reactivity as a function of the number of assemblies in the canister performed? During loading of the canister in the pool, does analysis show that an arrangement of fewer than seven assemblies does not have a larger reactivity than the completely loaded canister? Such a situation might be possible due to reflector effects.

RESPONSE:

Reactivity analyses were not performed as a function of the number of fuel assemblies in the canister. NUTECH's design basis for criticality control is to use neutron poison (Boral) in all seven guide sleeves. (Refer to NUTECH response to question 50 section 3.3.4). When the canister is loaded with less than seven fuel assemblies, several physical events take place which prevent the reactivity from being larger than when fully loaded.

Most significantly, a major source of reactivity (a fuel assembly) is removed from the system. To maintain the same reactivity, or a higher reactivity, a certain population of neutrons would have to pass through the vacant cell before striking an adjacent fuel assembly. Neutrons traveling through the cell must now pass through a total of four layers of boral. Since the cell contains only water, neutrons are more efficiently thermalized resulting in their accelerated removal from the population since boron is especially effective for thermal neutrons.

The system's reactivity must decrease since the neutron generation rate will be reduced, and since proportionally more neutrons will expire due to the increased effectiveness of the boral guide sleeves. If reflection is considered, the effect of the guide sleeves is amplified.

Futhermore, the unaccounted for (in the calculations) fission products in the fuel will always assure that the actual keff will be much less than the conservative keff calculated.

QUESTION: 18

3.3.4 Nuclear Criticality Safety

2. Were design clearances and fabrication tolerances considered in the criticality analysis? What are the design clearances that have been provided to ensure insertion and removal of the fuel elements in the guide sleeves? What is the worst case reactivity for a water flooded canister if the design clearances and fabrication tolerances are taken into account?

RESPONSE:

Original criticality analyses did not consider design clearances and fabrication tolerances. The seven fuel assemblies were assumed to be centered in the spacer disk cutouts.

A design clearance of 0.1525" at each edge of the fuel has been provided to assure insertion and removal of the fuel elements in the guide sleeves. Refer to question 3.2.5.2 for further details.

Additional calculations have been performed to determine the worst case reactivity for a water flooded canister if the design clearances are taken into account. The reactivities for cases where fuel assemblies are as far apart as possible and where they are as close together as possible are both lower than that calculated for the nominal case.

Thus, it is concluded that the variations in design tolerancing will not cause the reactivity to go up.

QUESTION: 19

3.3.4 Nuclear Criticality Safety

3. From the hydrogen and oxygen atomic densities in the KENO input it appears that a water density corresponding to 20 degrees Celsius has been assumed. How does this temperature compare to limits of the fuel pool temperature. If the temperatures are different, has analysis shown that the reported KENO results are conservative?

RESPONSE:

Fuel pool temperatures are typically no warmer than 100°-125°F (37.7°-51.7°C). The pool water was conservatively modeled at 20°C because, at lower temperatures, the hydrogen and oxygen densities are higher. Since hydrogen and oxygen are both effective moderators, their higher density results in more effective moderation of neutrons. Since the fission cross section of U-235 is much higher at low energies, more moderation leads to a higher reactivity. The modeling choice is therefore conservative since it results in a higher calculated reactivity than could reasonably be anticipated under actual plant conditions.

Futhermore, the unaccounted for (in the calculations) for fission products in the fuel will always assure that the actual keff will be much less than the conservative keff calculated.

DETAILED COMMENTS ON NUTECH
TOPICAL REPORT NUH-001

QUESTION: 20

3.3.4 Nuclear Criticality Safety

4. What are the errors or biases due to the cell homogenization procedure? If the biases have not been quantitatively evaluated, has it been shown that the biases are conservative?

RESPONSE:

Biases due strictly to the cell homogenization procedure have not been quantitatively evaluated. An overall bias was established by comparison of NUTECH criticality calculations to the benchmark results presented in NUREG/CR-0073.

The overall bias is in the nonconservative direction and has been accounted for as described in Section 3.3.4.3.

The homogenization process used by NUTECH is described in Section 2.1.3 of the SCALE Module CSAS2(1). Bierman, Clayton, and Durst(2) report that fuel regions (pins) do not have to be discretely modeled in KENO IV calculations. The "smeared" modeling process produced more accurate results than discrete modeling.

(1) NUREG/CR-0200
(2) Bierman, Clayton and Durst, "Critical Separation Between Subcritical Clusters of 2.35 Wt% ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons," Battelle Pacific Northwest Laboratories, PNL-2438, October 1977.

QUESTION: 21

3.3.4 Nuclear Criticality Safety

5. The design of the DSC apparently considers that some modules may use boron sleeves, while others may not. What controls are employed to ensure that low burnup assemblies are not inadvertently loaded into unborated canisters?

RESPONSE:

All seven sleeves will be constructed from boral. Any references in the Topical Report to the contrary will be removed or corrected.

QUESTION: 22

3.3.4 Nuclear Criticality Safety

6. What are the manufacturing tolerances of the boron content of the boral guide sleeves? Were the criticality analyses performed with a nominal boron concentration or a minimum concentration? If there is a difference, what is the magnitude of the effect on reactivity?

RESPONSE:

The manufacturing tolerance of the boron content of the Boral guide sleeves is directly related to the tolerance in the core panel thickness. Books & Perkins Product Performance Report 624 indicates a tolerance of ± 4 mils for a 75 mil thick Boral panel (which includes the core and cladding). It is assumed therefore that the boron content per unit area will vary no more than ± 5.3%.

Criticality analyses were performed with a nominal boron concentration.

NUTECH's scoping studies demonstrated that reactivity is relatively insensitive to small changes in boron concentration, if other parameters are unchanged. The data shows that when the B-10 concentration increases by 36%, the reactivity decreases by 2.7%. Another set of data shows a 1.8% increase in reactivity for a 22% reduction in B-10 concentration.

QUESTION: 23

3.3.4 Nuclear Criticality Safety

7. The dimensions shown in Figure 3.3-2 are inconsistent with the text and the KENO input. It is correct to assume that the Figure is in error?

RESPONSE:

Figure 3.3.2 is in error. The following shall be revised: 0.25" shall read 0.025", 0.75" shall read 0.075".

QUESTION: 24

3.3.4 Nuclear Criticality Safety

8. If Level D service limits are retained for the DSC, how would the multiplication factor be affected, should deformation of the DSC internals occur, when the canister is opened for inspection in the fuel pool? (Refer to Questions in 3.2.5.2 above).

RESPONSE:

The maximum departure from the nominal design condition is described in question #18. No increase in system reactivity is expected as a result of fuel assembly movement. (see question #18).

QUESTION: 25/26

3.3.4.3 Verification Analysis

1. Have other calculations been performed to verify that the implementation of the KENO-IV computer program used for the criticality analysis produces results with expected statistical dispersions?

   If only three benchmark calculations have been calculated for the verification analysis, can a statistically significant determination of the proper and expected operation of the computer code be made.

RESPONSE:

KENO-IV has been fully verified by Boeing Computer Services in accordance with their QA program.

The intent of NUTECH's benchmark calculations using KENO-IV was to verify an adequate analytical technique (i.e., homogenization procedure, appropriate computational flow path and data, etc.) and to establish the bias produced by that technique. No effort was made to determine the proper operation of KENO-IV by executing a "statistically significant" number of calculations. The operation and statistical significance of the KENO program has been verified numerous times by others.

DETAILED COMMENTS ON NUTECH
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QUESTION: 27

3.3.4.3 Verification Analysis

2. What is the justification for the selection of the BNL critical experiments for the verification analysis and in particular the selection of the three experiments that were calculated? Were other series of critical experiments considered? If not, why not?

RESPONSE:

The three BNL experiments were chosen because of their similarity to the NUHOMS system, and in particular to the reactivity control materials used in NUHOMS.

Other series of experiments were not chosen due to their lack of similarity to the NUHOMS design.

QUESTION: 28

3.3.4.3 Verification Analysis

3. Table 3.3-5 shows the results of the NUTECH calculations of the benchmarks as the mean value plus two standard deviations. While this is the accepted conservative value for criticality design calculations, it is not the correct value to determine the computational bias. The mean value should be compared with the experimental results. What are the mean value and standard deviation of the benchmark calculations?

RESPONSE:

The mean value of the benchmark calculations is 0.964. The average standard deviation is 0.00544.

Table 3.3.5 shall be corrected. The reported bias shall be based on the benchmark mean values, rather than the mean plus two standard deviations.

QUESTION: 29

3.3.4.3 Verification Analysis

4. The reported results for the three experiments that were calculated differed from measurements by more than 2.3 to 2.4% (see comment in question 2 above). Is this computational error consistent with errors experienced in calculation of these experiments that have been performed by others? What are the reasons for the reported errors in the present calculations?

RESPONSE:

A reasonable bias of -3.6% was obtained by acceptable methods (See previous question) and applied to the computational results. The NUHOMS design is critically safe as indicated by the resulting K_{eff} which is less than 0.92 after inclusion of all statistical and operational biases.

QUESTION: 30

3.3.7.1 Irradiated Fuel Handling and Storage

The temperature limits on fuel cladding in order to prevent its degradation in storage are based principally on tests in inert atmospheres. What criteria are applied in the design regarding storage atmosphere inside the DSC to prevent degradation and gross rupture per 10CFR72.72(h)?

RESPONSE:

NUHOMS System is designed for the storage of irradiated fuel in a helium atmosphere at temperatures well below those that would cause cladding failure. The criteria applied in the DSC design to assure inert atmosphere are the vaccum level prior to helium backfilling and the closure weld integrity.

QUESTION: 31

3.4 Classification of Structures, Components, and Systems

The safety and quality assurance classifications for each of the components of NUHOMS 0708 important to safety should be provided.

RESPONSE:

The utility must assign the quality assurance classification and requirements to each of these components in accordance with the utilities established quality assurance policies and practices. The quality assurance classification for each of these components should be included in the sites FSAR.

As listed in Table 3.3-1 the following items are those which NUTECH considers to be important to the safe operation of the NUHOMS System:

Cask

Canister

- Basket
- Spacer disk
- Support rods
- Lead plug
- End closer plates and closure welds
- Canister body

Concrete Module

- DSC support assembly
- Concrete Shielding

The exact nomenclature and classification system used by a specific utility at a specific site will depend on the utilities existing way of doing business.

QUESTION: 32

4.0 INSTALLATION DESIGN

4.2.3.1 Dry Shielded Canister

1. It is stated the canister body consists of a 0.5 inch thick rolled and welded stainless steel plate. What is the maximum leakage rate for the longitudinal weld? What inspections and/or tests will be performed to assure that actual leakage is below this rate?

RESPONSE:

The longitudinal weld will be radiographed or ultrasonic tested to assure that the integrity of the weld is equal to the 0.5 inch thick stainless steel plate. Therefore, the leak rate of the longitudinal weld will be equal to that of the parent metal.

Under the NUHOMS system conditions the diffusion will be insignificant (see response to Question 1).

QUESTION: 33

4.2.3.1 Dry Shielded Canister

2. The canister is partially coated with a dry film lubricant to reduce friction during loading. Will this film attract an insulating layer of dust or dirt which could interfere with heat transfer? How much of the canister will be coated with this material?

RESPONSE:

A dry film lubricant has a thin, hard, and dry finish which is bonded to the surface. The lubricant has a film thickness between 0.0003 and 0.0005 inches. The film does not attract dust or dirt and nor does dirt adhere to the finish.

For economic reasons, the dry film lubricant coating has been removed from the DSC design and transferred to the cask liner and T-section rails. By coating these components with a solid film lubricant the coefficient of friction between the sliding surfaces will only be between 0.04 and 0.08, far below the design value of 0.25.

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TOPICAL REPORT NUH-001

QUESTION: 34

5.0 OPERATION SYSTEM

5.1.1.3 Cask Drying Process

1. When backfilling the canister with helium, the valves are closed when the pressure reaches 14.7 psia. What is the temperature of the helium when the valves are closed? How is the temperature measured or inferred? If the helium is not at the final equilibrium temperature, then the pressure will continue to rise as the gas heats up.

RESPONSE:

The procedures have been revised to include hydrostatic testing of the DSC and monitoring the pressure until it reaches equilibrium pressure.

The following procedures will be incorporated in the Topical Report

- o Attach a self priming pump to valve #2 of the siphon line and drain more than 15 gallons from the DSC.
- o Remove the self priming pump.
- o Dry any water from top lead plug and DSC interface and then seal weld the upper stainless steel cladding plate of the top lead plug to the canister body.
- o Connect 0-75 psig pressure gauge to valve #2 on vent tube, and open valves #1 and #2.
- o Connect demineralize water supply to intake side of hydro-pump and connect hose from discharge of hydro-pump to valve #2 of siphon tube.
- o Open valves #1 and #2 on siphon tube.
- o Activate hydro-pump and pressurize the DSC to 50 psig as read on pressure gauge.

RESPONSE: (Continued)

- Once internal pressure of the DSC has reached 50 psig close valves #1 and #2 on siphon tube and disengage hydro-pump.

- Monitor pressure for 10 minutes.

- After 10 minutes examine the primary closure weld (weld between top lead plug and DSC shell) for leaks. Continue monitoring pressure for throughout examination. If a leak is detected or if internal pressure drop is detected release pressure, remove closure weld, and place fuel assemblies back in fuel pool.

- If no leaks are detected disconnect siphon hose from hydro pump discharge and connect to plant's low-level radioactive waste system. Open valves on siphon tubes, allowing pressure to drop to atmospheric pressure.

- Remove pressure gauge from valve #2 on vent tube and connect the sites compressed air supply to valve #2.

- Dry canister in accordance with the existing procedures #53 through # 58.

- Open valve number 2 and allow the premeasured quantity of helium to flow into the DSC cavity.

- Close valve numbers 1 and 2 on helium. Monitor pressure until it is stable.

QUESTION: 35

5.1.1.3 Cask Drying Process

2. What is the differential pressure across the canister wall when the helium sniffer leak test is performed?

RESPONSE:

The differential pressure across the canister wall when helium sniffer leak test is performed is 0.5 atm.

QUESTION: 36

5.2.2.1 Safety Features

What are the features of the spent fuel storage system which are important to safety that provide for safe operation under both normal and abnormal conditions? What, if any, are the limits selected for a commitment to action?

RESPONSE:

Section 5.2.2 of Regulatory Guide 3.48 is intended to describe the operation and safety features associated with the fuel handling system. The safety features associated with the fuel handling system are described in Section 5.2.1.2 of the Topical Report.

The safety features for the DSC and HSM are described in Sections 1.3.1.1 and 1.3.1.2 respectively. Chapter 10 lists the operating controls and limits for the safe operation of NUHOMS and specify commitments to action.

QUESTION: 37

5.3.2 Component/Equipment Spares

In the event that components are damaged during the life of the installation, what provision is made for installation of spare or alternative equipment to provide for continuity of safety under normal and off-normal conditions?

RESPONSE:

As the analysis has shown, the HSM protects the DSC during normal operation or from any credible accident. The only component of the NUHOMS system that can be damaged is the precast shielding blocks located on the roof of the HSM. The consequence of losing one or more shielding blocks is an increase in the skyshine scattered dose in the vicinity of the HSM.

In order to reduce the scattered dose, new shielding blocks could replace the damaged shielding blocks or portable shielding could be placed over the air outlets. Therefore, NUTECH recommends that spare shielding blocks be retained on site. The quantity is site specific and may depend on the number of modules in the installation, frequency of tornadoes, etc. The utility may wish to retain forms for casting shielding or plan to use existing portable shielding.

QUESTION: 38

7.2.1 Characterization of Sources

1. The design basis of the radiation sources for the shielding analysis is fuel assemblies with an average specific power of 37.5 MW(t)/MTHM, 33000 MWd/MTHM and five years post irradiation time. Based on these parameters a design basis neutron and gamma ray source strength have been calculated. Since the burnup parameters are average values, it may be that some combinations of seven fuel assemblies selected for loading into the DSC may have a higher source strength. Is the use of the mean values intended to imply that an arbitrary selection of fuel assemblies from a batch of assemblies satisfying the burnup parameters can be loaded into the DSC? If not, explain how limiting the source strength in any given canister to the design basis source strength will be accomplished in practice.

RESPONSE:

An arbitrary selection of fuel assemblies will not be loaded into the DSC. Selection and insertion into the DSC will be controlled by existing plant records and procedures. Section 5.1.1.1, Item 1 ensures that assemblies will be screened such that they satisfy the radiological requirements specified in Chapters 3 and 10. Furthermore, health physics surveys of the module are also specified after DSC insertion. Any, violation of design bases dose rates will require DSC removal or the use of portable shielding.

QUESTION: 39

7.2.1 Characterization of Sources

2. The flux-to-dose conversion factors shown in Figure 7.2-1 seem to be consistent with the multicollision tissue dose response function at energies greater than 1 MeV, but are a factor of 2 to 10 lower than that response function at lower neutron energies. What is the reference for the flux-to-dose conversion factors that were used?

RESPONSE:

Neutron flux-to-dose conversion factors were obtained from ANSI/ANS 6.1.1-1977 (N666) "American National Standard Neutron and Gamma-Ray Flux-to-Dose-Rate Factors."

Table 7.2-1 will be revised to reflect the correct response function for the 7.1 KeV through 0.41 KeV neutron groups.

QUESTION: 40

7.3.2 Shielding

1. Have verification analyses been performed for the computer codes used in the shielding analysis (ORIGEN, ANISN, DOT-IV, QADMOD-G, and SKYSHINE-II)? Please discuss any shielding benchmark validations that have been performed.

RESPONSE:

Verification analysis has been performed on the subject computer codes by their respective computing vendors. They are, specifically:

ORIGEN2 - Babcock & Wilcox Computer Services
ANISN; DOTIV-Boeing Computer Services
QADMOD-G; SKYSHINE II-UCCEL Corporation

QUESTION: 41

7.3.2 Shielding

2. What was the concrete composition assumed in the shielding analysis? What variation in water content is considered over the service life of the HSM?

RESPONSE:

The concrete composition assumed in the shielding analysis is (by weight percent):

Hydrogen	0.5294 %
Oxygen	47.45 %
Sodium	1.629 %
Magnesium	0.2441 %
Aluminum	4.474 %
Silicon	30.02 %
Sulphur	0.1222 %
Potassium	1.833 %
Copper	12.53 %
Iron	1.181 %

The initial water content of the module concrete is expected to be 7.00%. Therefore, since a very conservative value was chosen for the water content, (4.2%, a 40% reduction) no allowance was made for variation in water content over the service life of the HSM.

QUESTION: 42

7.3.2 Shielding

3. What is the relative contribution of primary and secondary gamma rays for the gamma ray surface dose rate on the HSM wall or roof?

RESPONSE:

Secondary gamma-rays account for approximately 6% of the total gamma ray dose on the HSM wall or roof.

QUESTION: 43

7.3.2 Shielding

4. Page 7.3-9 third paragraph defers discussion of the HSM penetration calculation. The deferred discussion is not found later in the report. Please provide a discussion of the penetration calculation.

What was the quadrature order used in the DOT-IV calculation? Justify that the quadrature is adequate to accurately calculate streaming through the HSM air exhaust penetration.

RESPONSE:

a) The words "which is discussed later" will be deleted from the third paragraph on page 7.3-9 to avoid any future confusion.

Only one DOT-IV model was constructed to find the "axial" and "air exhaust" dose rates. Note that in Figure 7.3-7, "DOT-IV Model Axial and HSM Penetration Shielding Analysis," the model includes both axial points of interest as well as the air exhaust outlets.

The words "See Figure 7.3-7" will be added to the end of the first paragraph on page 7.3-14 to emphasize the DOT-IV modelling choice.

b) S_8 is the level of quadrature used in DOT-IV (48 solid angles in each hemispherical space). It was chosen in accordance with the guidelines given in ANSI/ANS-6.4-1977 as well as other technical literature, and sample problems provided by the computer vendor. A comparison between the degree of convergence obtained with S_8 and lower orders of quadrature was made during analysis for NUHOMS. On the basis of those comparisons, the additional computational expense of higher order quadrature did not offer significantly greater accuracy and was therefore unjustifiable.

OPERATION: 44

7.3.2 Shielding

5. The results in Table 7.3-2 indicate that the neutron attenuation through the HSM air exhaust penetration is greater than that for gamma rays. Can this result be explained?

RESPONSE:

The results in Table 7.3-2 do not indicate that the neutron attenuation is greater. The table below shows the resulting attenuation factors.

	Location	Neutron Dose Rate (mrem/hr)	Gamma Dose Rate (mrem/hr)	Gamma to Neutron Ratio
1)	Air Exhaust Outlet	1 ($4.17.10^{-4}$)[1]	11.5 ($6.05.10^{-7}$)	11.5
2)	HSM Roof	0.03 ($1.25.10^{-5}$)	8.2 ($4.32.10^{-7}$)	273
3)	Surface of the DSC (Midplane of fuel)	2.4×10^{3} (-)	1.9×10^{7} (-)	7920

1) Attenuation factor; dose rate in rows 1 & 2 divided by dose rate in row 3.

QUESTION: 45

7.3.2 Shielding

6. Were any measures taken to minimize ray effects in the DOT-IV calculations? Were ray effects evident in the results?

RESPONSE:

It is well known that ray effects associated with two-dimensional discrete ordinates codes are inevitable in most cases. A careful examination of NUTECH's results indicates that the flux oscillations (ray effect) occurs only to a minimum extent.

The results were also checked by examining flux variations from interval to interval. Large gradients in any direction (more than 50%) were not evident.

QUESTION: 46

7.3.2 Shielding

7. Has a verification analysis been performed for the application of the DOT-IV code to this type of shield penetration problem?

RESPONSE:

A verification run for streaming through a pipe chase (a similar penetration to the NUHOMS air outlet) was performed by Boeing Computer Services as part of their verification program.

QUESTION: 47

7.3.2 Shielding

8. Are the "Cask - DSC Annular Gap" dose rates reported in Table 7.3-2 averaged over the cask surface, or are they actually dose rates calculated in the gap?

RESPONSE:

The dose rates in Table 7.3-2 are the actual dose rates in the gap.

QUESTION: 48

7.3.2 Shielding

9. What was the angular quadrature used in the Cask - DSC annular gap calculation? Justify that the angular quadrature was fine enough to adequately model radiation transport through the gap. Were ray effects observed in the calculated fluxes? If the dose rate on the surface of the canister is plotted as a function of distance from the canister centerline, is streaming through the gap observed in the results?

RESPONSE:

The angular quadrature used was S_8. Due to the minor improvement in convergence noted between S_8 and lower order quadratures, an order higher than S_8 appeared unjustifiable in view of the increased computational costs (See the discussion in question 4 of section 7.3.2.

No significant ray effects were noticed.

Streaming through the annular gap is apparent in the results if the surface dose rate at the end of the canister is plotted as a function of distance from the canister centerline.

QUESTION: 49

7.3.2 Shielding

10. Is there any analysis or data to benchmark the validity of the QADMOD/albedo procedure used for estimating gamma ray penetration through the HSM air exhaust?

RESPONSE:

NUTECH is not aware of any published benchmark studies which validate the QADMOD/albedo procedure used for estimating gamma ray penetration through the HSM air exhaust.

QADMOD, a modification of QAD, is an industry-standard point-kernel shielding code which is fully capable of determining incident dose rates on concrete surfaces within the penetration. The albedo method is recommended by ANSI/ANS - 6.4-1977, Section 8.4.4.2, and was applied in strict accordance with reference number 45 therein.

It is therefore NUTECH's opinion that benchmark studies of this particular analytical method are not appropriate.

QUESTION: 50

7.3.2 Shielding

11. Were secondary gamma rays in the concrete considered in the vicinity of the air exhaust penetration?

RESPONSE:

Secondary gamma rays produced in the concrete were not considered in the albedo analysis of the air exhaust penetration. In view of the small neutron flux along the predominant reflecting surfaces, secondary gamma rays were considered to contribute a negligible amount to the overall gamma ray dose rate outside the penetration.

QUESTION: 51

7.3.2 Shielding

12. With reference to the proper axial positioning of the DSC within the HSM both during normal operation and in case of seismic event (refer to Question 6. under 8.1.1.5 and Question 6 uner 8.2.3.2), it should be noted that there could be shielding implications.

RESPONSE:

A seismic retaining assembly has been added to the NUHOMS design. The seismic retaining assembly consists of a stopping block attached to the back end of the DSC support rails and seismic blocking device to be placed between the bottom cover plate and the door prior to closing the door. This retaining assembly will prevent the axial movement of the DSC.

QUESTION: 52

7.4 Estimated On-Site Dose Assessment

1. Figure 7.4-1 indicated that the Total Dose drops slightly below the Scattered Dose at approximately 1000 ft. Is this a plotting error or a problem in the computer code?

RESPONSE:

There is a plotting error in Figure 7.4-1. This figure will be revised.

QUESTION: 53

7.4 Estimated On-Site Dose Assessment

2. The basis for the dose rates in Table 7.4-1 is not evident in the text. Please explain how the dose rate values were obtained.

RESPONSE:

The number of personnel and time span assigned to each of the operational steps in Figure 7.4-1 is based on experience with similar operations and engineering judgement. An average distance from the cask surface was assumed and used to estimate the ambient dose rate. The distances are intended to reflect time-averaged values for the given activity, and are taken from the torso to the cask surface nearest the operator.

The dose rate is obtained using the average distance and dose rate maps constructed from the results of NUTECH shielding calculations. Dose rates are typically conservative and do not assume that portable shielding has been used. NUTECH anticipates, however, that in actual practice on a site specific basis portable shielding and other normal plant health physics procedures will be employed wherever possible to keep total exposure even lower than indicated on Table 7.4-1.

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QUESTION: 54

8.0 ANALYSIS OF DESIGN EVENTS

8.1.1 Normal Operation Structural Analysis

1. It is stated (Page 8.1-1) that "the mechanical properties of materials employed in the structural analysis of the NUHOMS system are presented in Table 8.1-2." This table only contains properties for stainless steel, carbon steel and lead. A number of other materials are used in the NUHOMS system including concrete, steel reinforcing, boral, etc. The properties for each of these materials should be included in Table 8.1-2 to insure that a consistent set of properties are used for all analyses and that appropriate consideration is given to the effect of temperature in each specific analysis.

RESPONSE:

Table 8.1-2 will be revised to include the mechanical properties of concrete, reinforcing steel and boral at various temperatures.

QUESTION: 55

8.1.1 Normal Operation Structural Analysis

2. There is lack of experience on long term storage of spent fuel horizontally. Please address the effects on the fuel of its horizontal storage over long times in the HSM and any related effects on its retrievability.

RESPONSE:

Although there is lack of experience on long term storage of spent fuel horizontally, there is considerable experience on the shipment of the fuel in the horizontal position. A summary of shipping experiences world wide is listed in the document "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases" by Battelle, and is reproduced here.

"Approximately 1000 shipments (2000 tU) of LWR fuel have been transported dry in Europe and from Japan to Europe; these shipments have included 4550 BWR and PWR assemblies. Within the United States there have been 1500 LWR fuel shipments (900 tU; 4100 fuel assemblies). During sea shipments the fuel remains horizontal for up 2 to 3 months, sometimes at cladding temperatures estimated at up to 385°C. Thus, the fuel resides horizontally for significant periods at conditions approaching those in dry storage without evidence of significant damage. If the fuel is not damaged during emplacement and retrieval, horizontal storage does not appear to offer a threat to cladding integrity."

It is worth noting that the environment for shipping fuel, given the transportation shock loadings, vibrations etc., is much harsher than the passive environment under dry storage. Also, on theoretical grounds, studies have shown cladding creep is acceptable with regard to retrievability, at temperatures below 385°C over the time spans associated with fuel heat decay. The Battelle report refers to the document: "Fry Post-Pile Creep: Experimental Procedure, Test Samples and First Results", by Pechs, M, et,al. 1983, where by the studies on the cladding creep is reported. From the discussion above it can reasonably be concluded that the cladding behavior in a horizontal position will not differ significantly than in the vertical direction.

QUESTION: 56

8.1.1.1 Normal Operation Loads

1. What is the basis for the assumptions of 25% fission gas release in the design internal pressure calculations?

RESPONSE:

The selection of a 25% fission gas release fraction is based on an EPRI fuel performance data base. The fraction represents a value substantially higher than that normally seen in LWR fuel operation. The study of fission gas release fraction is of some importance to the nuclear power industry since large release fractions can have a deleterious effect on fuel performance. As part of this study, EPRI has developed a data base of fission gas release measurements for 124 well-characterized fuel rods.* Some 102 of these rods were irradiated in domestic LWRs. These rods encompass a large range of linear heat ratings, burnups, enrichments fuel pellet densities and operating histories. None of these samples exceeds a 25% release fraction. Most release fractions are between 0.33% and 12%.

* Reference: E. Rumble, J. Simpson, S. Lee and A. Woodis, "The EPRI Fuel Performance Data Base; General Description", EPRI NP-1489, October 1980.

QUESTION: 57

8.1.1.1 Normal Operation Loads

2. On page 8.1-5, the normal average canister gas temperature is stated to be 450°F. However, Table 8.1-13 gives an average helium temperature of 429°F. at 125°F. HSM inlet temperature. Which value is correct?

RESPONSE:

At 125°F ambient temperature the helium temperature is 429°F as shown in Table 8.1-13. Page 8.1-5 will be revised.

QUESTION: 58

8.1.1.1 Normal Operations Loads

3. The temperature distribution for the spacer disc (Figure 8.1-2) shows that the temperature at the point of closest approach between the canister and the guide tube is lower than its neighboring points. Is this correct? Please explain.

RESPONSE:

Those locations have incorrect temperatures. The figure and analysis will be revised. However, the error will have a negligible effect on the results of analysis, because of the small temperature difference between the correct temperature value and the temperature shown in the figure, and the maximum temperature of the spacer disk.

QUESTION: 59

8.1.1.1 Normal Operation Loads

4. Please explain how the three steady-state temperature distributions were used to determine the effects of thermal cycling. On page 8.2-51, the heat up rate (100°C temperature change) of the insulated loaded DSC is given as 13 hours. Since this is about one-half of a diurnal temperature cycle, have such cycles been included in the cyclic fatigue analysis? If not, how would their inclusion modify the results?

RESPONSE:

The heat up rates on page 8.2-51 are for an insulated loaded DSC. These heat up rates were used to determine the temperature of various components when the air inlets and outlets are blocked, an accident condition.

The fatigue analysis due to seasonal thermal cycling is explained on pages 8.2-57 and 8.2-60. This analysis is based on the conservative assumption that the range of alternating thermal stress is from zero to the maximum normal steady state temperature of the DSC (i.e. 228°F at 70°F ambient). This range is greater than the range between the two extreme ambient temperature (132°F at -40°F ambient and 278°F at 125°F ambient) and as such envelops this condition.

QUESTION: 60

8.1.1.1 Normal Operation Loads

5. On page 8.1-5 the density of normal concrete is given as 150 pounds per cubic foot. This value differs from that shown in Tables 8.1-4 and 8.1-4. Please clarify.

RESPONSE:

The unit weight of concrete varies between 140 and 150 lb/ft^3. A 10 lb. variance in the unit weight of concrete is not unusual. The more conservative unit weight of 150 lbs/ft^3 was used in the structural analysis such as dead weight. However, in shielding analysis a unit weight of 145 lbs/ft^3 was assumed. In the site specific application a minimum 145 lb/ft^3 will be specified in the specification.

QUESTION: 61

8.1.1.1 Normal Operation Loads

6. Equations 8.1-9 and 8.1-10 on page 8.1-20 have missing terms. If the equations in the Topical Report were used, the results may be incorrect. Please clarify.

RESPONSE:

Equation 8.1-9 and 8.1-10 on page 8.1-20 are incorrectly typed. The analysis contained missing ν and λ terms. Page 8.1-20 will be revised.

QUESTION: 62

8.1.1.1 Normal Operation Loads

7. The modulus of elasticity is incorrectly taken as 27.0×10^6 psi instead of $26.6 \times 1\text{-}^6$ psi for the design basis temperature of 400°F. Please correct.

RESPONSE:

The Young's Modulus value of 27.0×10^6 will be revised to 26.6×10^6 psi.

QUESTION: 63

8.1.1.1 Normal Operation Loads

8. The temperature distribution within the DSC depends upon the azimuthal orientation of the DSC within the HSM. That is, the analysis assumes that three (3) assemblies are centered on a vertical line passing through the center of the DSC. How will this orientation be assured during loading?

RESPONSE:

It is highly unlikely that any variation in the canister orientation of the DSC within the HSM will cause any significant differences in the internal temperature distribution. However, to assure that the physical orientation is identical to that of the thermal model, markings will be placed on the top cover plate that will indicate the azimuthal orientation of the canister. The DSC will be loaded into the cask in the proper orientation.

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QUESTION: 64

8.1.1.2 Dry Shielded Canister Loads Analysis

1. Thermal variations in the circumferential and axial direction of the canister will result in thermal stresses due to self constraint by the colder canister ends. On page 8.1-22 the statement is made that, "Also, the thermal variation in the circumference and longitudinal directions of the shell are not considered significant since in the unrestrained condition the shell will expand radially and longitudinally to an equilibrium state." NUTECH should show what the overall deflected shape is in the axial direction, and the deflected shapes of a cross section at the end and the center of the DSC. NUTECH should show what the ASME stress intensities are for the following locations on the DSC: (1) top and (2) bottom elements of shell at top cover plate edge (3) top and (4) bottom elements of shell at center section; (5) top and (6) bottom elements of top cover plate at shell edge. Please include effects of discontinuity between shell and top cover plate.

RESPONSE:

The temperature along the length of the DSC does not vary significantly (± 20°F). This temperature change is not as great as the thermal variance that occurs around the circumference of the DSC shell. As shown in Figure 8.1-1, the temperature of the DSC shell varies from 229°F to 285°F. Thermal stresses due to this variance were evaluated by means of finite element analysis, the maximum thermal stress intensities of 11.24 ksi were determined which occurred at the DSC and T-rail interface. This stress is far lower than the thermal stress of the DSC shell reported in Table 8.1-7 of page 8.1-15 of the Topical Report. The stress values reported in Table 8.1-7 is located at the region of the DSC and Spacer Disk interface and is based on the differential thermal expansion of the spacer disk and the DSC shell. Based on the above observation the thermal stresses at other locations requested are envoloped by those reported in Table 3.1-7.

QUESTION: 65

8.1.1.2 Dry Shielded Canister Loads Analysis

2. The drawings provided (ADV001.0204 sheets 1 and 2) do not provide enough dimensions to be able to determine what minimum gaps might exist between the boral baskets and the top lead plug. Please supply actual shop drawings instead of conceptual design drawings. This relates to p. 8.1-22 and 8.1-24.

RESPONSE:

The drawings provided are neither conceptual nor shop drawings. They are design drawings. Clearances will be added to the drawings.

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TOPICAL REPORT NUH-001

QUESTION: 66

8.1.1.3 Dry Shielded Canister Internal Basket Loads Analysis

Please provide information showing where the 0.75 inch irradiation growth of the fuel assembly came from. The drawings do not permit a verification of calculations of this section of the topical report.

RESPONSE:

The irradiation growth of 0.75 inches was based on data obtained from the following report:

Examination of High Burnup fuel at the
H. B. Robinson Reactor End of Cycle 8
Report No. XNNF8317
Exxon Nuclear, Inc.

QUESTION: 67

8.1.1.4 DSC Support Assembly Loads Analysis

The first sentence of the Thermal Analysis section p. 8.1-32 is acceptable if an assembly procedure specifying the torque of the nuts is included.

RESPONSE:

As specified in Table 3, p. 5-214 of the AISC Steel Manual the minimum tension for a 3/4 inch diameter bolt is 28 kips. This corresponds to a tightening torque on the nuts of 4200 in. lb. The amount of friction force created by the tensioning of the bolt is 21 kip. However this force can easily be overcomed by the thermal expansion force created temporarily by the heat up of the rail during normal operating condition.

QUESTION: 68

8.1.1.5 Horizontal Storage Module Loads Analysis

1. A reference in the third paragraph of this section (page 8.1-35) is made to NUREG 0880. Shouldn't this reference to be NUREG 0800?

RESPONSE:

The reference should be NUREG 0800. This page will be revised.

QUESTION: 69

8.1.1.5 Horizontal Storage Module Loads Analysis

2. Table 8.1-10 (Page 8.1-38) includes values for the ultimate moment and shear capacity of the HSM. The value computed for the ultimate moment capacity, 4052.0 in-kips, is given on Page 8.1-49. However, no calculation is presented for the ultimate shear value, 59.2 kips. The details of this calculation should be included.

RESPONSE:

The ultimate shear capacity was based on the following equation from equation 11-28 in American Concrete Institute (ACI) 349-80.

$$V_c = 2 \quad f'_c \quad b \ wd$$

The ultimate shear value calculation will be incorporated into the Topical Report.

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QUESTION: 70

8.1.1.5 Horizontal Storage Module Loads Analysis

3. Figure 8.1-11 (Page 8.1-39) includes moment and force loads that are applied to the reinforced concrete corbel that supports the DSC support Assembly. Calculations should be provided for the stresses in the corbel due to these loads and for the ultimate strength capacity of the corbel.

RESPONSE:

Figure 8.1-11 shows the dead loading and live loading at the centerline of the module walls, not the corbel centerline. Table 8.2-12 show the maximum end loads (at corbel) for each load combination. As stated on page 8.2-63 the corbel, bearing plate and bolts were designed for these load combinations. Additionally, for the corbel design loads were multiplied by 1.7 to envelope all load combinations specified in ACI 349-80.

Design and analysis were performed using the criteria of Section 11.9 of ACI 349-80 and the recommendations of Chapter 5 of Wang & Salmon (Reinforced Concrete Design, Third Edition). The nominal shear strength, Vn, using equation (11-31) of ACI 349-80 yields 29.4 kips. Since the factored shear load is only 17k the corbel is adequate per ACI 349-80. Additional corbel reinforcement and welded plates were added to allow the direct transfer of the horizontal forces to the tension reinforcement per suggested practices in Wang. Therefore, the concrete corbel is designed to take all DSC support assembly load combinations per ACI 349-80.

Table 8.2-12 will be revised to incorporate the capacity of the corbel, bearing pads and the bolts at both normal operating and accident condition.

QUESTION: 71

8.1.1.5 Horizontal Storage Module Loads Analysis

4. Please state your interpretation of ACI 349-80 acceptance criteria for concrete temperatures. An air inlet temperature of 70°F is used to evaluate normal operating temperatures. Please relate this number to meteorological information which could be used to determine site acceptability.

RESPONSE:

ACI 349-80 Appendix A, Section A.4 states for normal operation or any long term period the concrete temperature shall be limited to 150°F except at local areas, where it is limited to 200°F. For accident conditions or periods of short time duration, the temperature may not exceed 350°F except at local areas, where it is limited to 650°F. Section A-4-3 of the code allows for even higher temperatures than those specified if tests are performed to evaluate the loss of strength due to sustained exposure to elevated temperatures. On page 8.1-44 of the Topical Report a number of references are listed which show concrete strength is not effected by temperature below 212°F for both short and long term duration.

Although temperature does impact the mechanical properties of the concrete, the critical factor to consider is the temperature gradient in the concrete and its associated stresses. As shown in Table 8.1-12 the maximum temperature gradient in the concrete occurs at normal ambient temperatures of 70°F, which produces an enveloping thermal stress in the concrete walls & slabs.

QUESTION: 72

8.1.1.5 Horizontal Storage Module Loads Analysis

5. The analysis of local concrete temperatures does not include conduction through the thermal shield support bolts or through the support rail to the corbels. How will consideration of this direct heat transfer path affect local concrete temperatures and satisfaction of ACI 349-80 acceptance criteria? Also, how will concrete thermal conductivity be affected by thermal cycling over the license period, and by the presence of rebar in the concrete?

RESPONSE:

The heat shield support bolt located at the mid-section of the roof slab has been removed, figure 4.2-6 will be revised to incorporate this change.

Furthermore assuming that the heat shield support bolts or corbels do provide a direct heat transfer path, the local concrete temperatures, under normal operating conditions, do not exceed the acceptable criteria established in ACI 349-80. The remaining heat shield support bolts are located at the extreme ends of the heat shield. HEATING6 analysis shows that the temperatures in these areas does not exceed 170°F. Additional heat transfer analysis shows that under normal operating temperatures, the temperature of the DSC support assembly is less than 170°F. Conservatively, assuming that the bolts or corbels are at the same temperature as the base metal and that the local temperature of the concrete is at the same temperature as the bolts or corbels, the local temperature of the concrete is below the criteria in ACI 349-80.

Thermal cycling of the concrete over the license period of the HSM has been discussed in response to question 1.2.2-4.

QUESTION: 73

8.1.1.5 Horizontal Storage Module Loads Analysis

6. If the end of the DSC were to contact the concrete wall, a concrete temperature of the same magnitude as the calculated DSC temperatures would result. What is the clearance between the end of the DSC and the concrete wall and what is the concrete temperature adjacent to the end of the DSC?

RESPONSE:

The clearance between the end of the DSC and the concrete wall is 3 inches.

Additional heat transfer analysis indicates that the average concrete temperature adjacent to the end of the DSC is approximately 70°F lower than the average surface temperature of the DSC end.

The concrete temperature for the end walls of the HSM is not as critical as the side walls and roof slab, since the frame action in the transverse direction of the HSM is considered the main structural resisting system. Also, the air outlet opening at the ends of the HSM roof provides the freedom for the thermal movement of the end walls.

QUESTION: 74

8.1.2 Off-Normal Events

Failure of the supports for the thermal radiation shield (due to seismic events, corrosion, etc.) has not been addressed. Please justify that this event is not feasible, or provide an analysis of the consequences. How would such as failure be detected?

RESPONSE:

The heat shield is designed to withstand all applicable loads and environmental conditions during the lifetime of the horizontal storage module. The heat shield will experience only dead weight and seismic loads. Thermal loads will not be experienced since oversized holes will allow unrestrained thermal expansion. The heat shield will be galvanized to prevent corrosion. The shield and the embedded anchor bolts were analyzed for maximum dead weight plus seismic loadings. Conservatively, the heat shield was treated as a simply supported beam with a uniform loading. A maximum bending stress of 12.43 ksi was calculated. Axial and shear stresses were negligible. The tensile and shear stress of 1.08 ksi and 0.53 ksi were calculated respectively for the anchor bolts by conservative assumptions. A bearing stress of 1.68 ksi was calculated for the anchor bolt/heat shield contact point. Clearly then, these minimal stresses show that the radiation shield is withstanding any postulated loading. Therefore, no failure detection system is necessary.

QUESTION: 75

8.1.2.2 Blockage of the Horizontal Storage Module Inlet - Cause of Event and Detection

1. Air flow through the HSM with both inlets blocked appears to be a random process with zero mean value. Please provide experimental confirmation based on scaled or appropriate data to demonstrate the assumed magnitude of flow for this case.

RESPONSE:

The air flow is not a zero mean, random process. The 7 kw of heat produced by the fuel will heat the air and cause a significant driving force to push the air out of the HSM. As the air is forced out, the pressure in the HSM will decrease and air from the outside will be drawn in. Exact flow paths will depend on existing meter-logical conditions, but because of the constant heat source, the air will flow.

As part of the demonstration program, extensive testing will be performed on the first prototype module to be built at Carolina Power and Light. Evaluation of the airflow through the HSM when both air inlets are blocked will be included in the electrical heater testing program.

QUESTION: 76

8.1.2.2 Blockage of the Horizontal Storage Module Inlet - Cause of Event and Detection

2. Please clarify the pressure given on page 8.1-60. The values are not consistent. According to the ASME text booklet "SI Units in Heat Transf^r" 1st Ed., 14.504 psi - 1 bar. Also, refer to subsequent units conversions between psi and bar.

RESPONSE:

The pressure given on page 8.1-60 will be revised.

QUESTION: 77

8.1.3.1 Thermal-Hydraulics of the Horizontal Storage Module Principles of HSM Cooling System

1. Reference 8.44 referred to here is not given in the reference list.

RESPONSE:

Reference 8.44 was inadvertently omitted. Reference 8.44 is

I.E. Idel'Chik, Handbook of Hydraulic Resistance, Coefficients of Local Resistance and of Friction, the U.S. Atomic Energy Commission and the National Science Foundation, Washington, D.C., 1960.

This reference will be included in the revision of the Topical Report.

QUESTION: 78

8.1.3.1 Thermal-Hydraulics of the Horizontal Storage Module Principles of HSM Cooling System

2. The thermal analysis is based on an assumed axial peaking factor of 1.08. This is an additional limiting design parameter which is not cited in Table 3.1-1. If it is not intended to impose this limit on stored fuel, please justify its conservatism for proposed fuel types, burnup histories, etc.

RESPONSE:

In most heat transfer analysis of decay heat during dry storage the heat generation rate of a fuel assembly is assumed to be uniformly distributed along the active fuel length. However, in NUTECH's analysis the axial peaking factor of 1.08 was conservatively applied to the uniform heat generation rate in order to determine the worst case temperature distribution in a two dimensional cross section of the DSC and HSM. Even with the conservatism the maximum fuel cladding temperature was calculated to be 338°C, under the worst ambient temperature of 125°F. This maximum cladding temperature is 62°C less than the accepted limiting fuel rod temperature of 400°C.

As stated in Section 10.3.1.1, the decay power per fuel assembly is limited to 1 kw.

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QUESTION: 79

8.1.3.1 Thermal-Hydraulics of the Horizontal Storage Module Principles of HSM Cooling System

3. Equations 8.1-37 through 8.1-40 are applicable within restricted domains of L^3 ΔT. Please show that these restrictions are satisfied.

RESPONSE:

Equations 8.1-37, 8.1-39, and 8.1-40 are valid for flow in the turbulent range. As the table below indicates the flow is in the turbulent range. Gr:Pr falls between 10^8 and 10^{12}. The ΔT is based on the average steady state temperature of the particular surface and the average air temperature in the region.

Gr:Pr for HSM
for Various Ambient Temperatures

Flow Region	Ambient Temperature -40°F	70°F	125°F
Canister Circumference	2.8×10^{10}	2.53×10^{10}	2.38×10^{10}
Floor	2.3×10^{9}	2.68×10^{9}	2.92×10^{9}
Wall	1.83×10^{10}	2.0×10^{10}	2.29×10^{10}

8.2-138 is valid over any L^3 Δt domain.

QUESTION: 80

8.1.3.1 Thermal-Hydraulics of the Horizontal Storage Module Principles of HSM Cooling System

4. What levels of solar radiation were considered in the heat transfer analysis? Of particular concern is the levels associated with the hottest days (Tair=125°F).

RESPONSE:

The following equivalent solar heat flux were included in the HSM thermal model.

Ambient Temperature (°F)	Solar Heat Flux ($BTU/hr.ft^2$)
-40	23.5 (1)
70	93.8 (2)
125	187.6 (3)

(1) 1/12 of the half day solar heat gain for 40 degree north latitude on December 21.

(2) 1/12 of the half day solar heat gain for 40 degree north latitude on June 21.

(3) Twice the solar heat flux for 70°F case.

QUESTION: 81

8.1.3.1 Thermal-Hydraulics of the Horizontal Storage Module Principles of HSM Cooling System

5. Page 8.1-69 gives the maximum concrete ceiling temperature as 244°F, whereas Table 8.1-12 gives 249°F. This analysis does not consider direct conduction through the supports to the concrete. Is the severe summer condition considered to be an off-normal or transient condition? If so, what is the limiting ambient condition to be used for determination of whether acceptance criteria on concrete temperature are met? The value of 70°F is quite low for a limiting ambient condition.

RESPONSE:

The maximum concrete ceiling temperature is 249°F page 8.1-69 will be revised to read 249°F.

The question of "Direct Conduction through the Supports" is responded in Section 8.1.1.5, question 5.

The severe summer condition is consid red to be a transient condition, since the time required for the concrete temperature to reach steady state is by far greater than the short period associated with extreme 125°F ambient temperature.

The selection of 70°F as the limiting ambient temperature is based on the fact that the thermal gradient through concrete thickness is greater for the 70°F ambient temperature than the other ambient extremes. Comparison of cases 1 through 3 of Table 8.1-12 will indicate this fact.

Since the concrete thermal stress is maximized by the application of maximum thermal gradient, the selection of 70°F ambient condition is considered appropriate.

Furthermore, because of the long time constants associated with the thermal masses of the HSM and DSC, they will, over a daily bases, be at temperatures which are more representative of the 24 hour average air temperature than any ± 30 to 40°F daily temperature swings. Thus, again, 70°F is a good value of the normal operating ambient temperature.

QUESTION: 82

8.1.3.3 Thermal-Hydraulic Analysis of the Canister Inside the Transfer Cask Model

1. In equation 8.1-40, the emissivity of the stainless steel transfer cask exterior is taken as 0.80, whereas a value of 0.587 was used for stainless steel surfaces within the DSC (page 8.1-74) and transfer cask (Equation 8.1-43). Please justify this use of different values for the emissivity.

RESPONSE:

The thermal-hydraulic analysis of the canister inside the transfer cask assumed that the DSC was seated within the cavity of the GE IF-300 shipping cask. The emissivity value, 0.80, for the stainless steel exterior of the transfer cask, corresponds to the effective emissivity for the finned surface of the GE IF-300 shipping cask, 0.83 rounded down. The effective emissivity is reported on page 6-41 of the GE IF-300 Shipping Cask Consolidated Safety Analysis Report.

QUESTION: 83

8.1.3.3 Thermal-Hydraulic Analysis of the Canister Inside the Transfer Cask Model

2. What is the sensitivity of the calculated results to variation of input parameters, e.g., emissivity, axial peaking factor, apparent thermal conductivity, decay heat, etc.? How is the margin of safety to limiting conditions established?

RESPONSE:

The purpose of the reported analysis is to show that the safe operation of the system is not jeopardized under various operating scenarios. The margin of safety was established through conservatisms assumed in the calculations and the extreme conditions which were assumed.

QUESTION: 84

8.1.5 Design Basis Internal Pressure Loads

Please describe any hydrostatic testing to be performed to assure integrity of the DSC. Has NUTECH considered pressurizing the loaded DSC with helium as a test of DSC integrity? In such a test how would the test pressure relate to the design pressure and the accident pressure discussed in Section 8.2.9?

RESPONSE:

As stated in response to Question 5.1.1.3 the DSC will be hydrostatically tested in accordance with the requirements of ASME, Section III, Subarticle NB-6220. The test will be conducted at a pressure of 1.25 the accident pressure, discussed in Section 8.2.9. The hydrostatic test should be more than adequate in assuring the DSC integrity.

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QUESTION: 85

8.2.2.2 Accident Analysis

1. The coefficient of friction between a single, unanchored module is stated to be 1.0 on Page 8.2-7. The ACI 349-80 code states in Sections 11.7.5 and 11.7.9 that the coefficient of friction shall be 1.0 if concrete is placed against hardened concrete and the interface between the concrete surfaces is clean, free of laitance and "intentionally roughened to a full amplitude of approximately 1/4 inch." The ACI 318-83 code states in Section 11.7.4.3 that the coefficient of friction shall be 0.6L for the case wherein concrete is placed against hardened concrete not intentionally roughened (:L: is this relation 1.0 for normal weight concrete). shouldn't coefficient of friction of no greater than 0.6 be used between the NUHOMS system and the supporting concrete pad, especially since no specific mention is made of the need for roughening the surface of the concrete pad under the NUHOMS system? (See question on Section 3.2.3.2).

RESPONSE:

As stated in response to question 3.2.3.2, the coefficient of friction will be changed from 1.0 to 0.6. The only area effected by this change is the sliding of the single HSM under various accident loads. A shear key type of connection located at the module walls or a tie-down system will be specified.

QUESTION: 86

8.2.2.2 Accident Analysis

2. It is noted on Page 8.2-13 that a single HSM will require tie downs for resistance against overturning and sliding under the action of a massive missile impact (i.e., a 3,976 lb automobile). It is further noted on this page that the NUHOMS 0708 concept, which has 8 modules either poured in place or pre-cast, will not slide under the postulated impact. Since it is understood that actual, site specific configurations for the NUHOMS system may have any number of modules, wouldn't it be appropriate to either require tie downs, to permanently anchor the NUHOMS system to its supporting concrete pad or to set the HMS in a key-way embedded into the pad rather than rely on the use of friction to restrain the HMS from sliding on the pad and potentially exposing contaminated surfaces?

RESPONSE:

As stated in response to question 3.2.3.2, a shear key type of connection located at the module walls on a tie-down system will be specified.

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QUESTION: 87

8.2.3.2 Accident Analysis

1. The values of E on pages 8.2-14 and 8.2-16 are incorrectly taken at room temperature. Please use the value of E at the maximum calculated temperature.

RESPONSE:

The values of E on pages 8.2-14 and 8.2-16 of the report will be changed from 27.0 E5 to 26.6 E6.

QUESTION: 88

8.2.3.2 Accident Analysis

2. The calculations in this section assume the maximum vertical earthquake acceleration to be 0.1g. As noted in Question 3.2.3.2, this acceleration should be increased to at least 0.17g. Each of the calculations in Section 8.2.3.2 that are based on the maximum vertical acceleration should be redone to reflect this increased acceleration and each reference to a vertical acceleration of 0.1g should be changed to 0.17g.

RESPONSE:

As stated in response to Question 3.2.3.1-2, the vertical ground acceleration will be increased to 0.17. All associated analysis will be revised to incorporate this change.

QUESTION: 89

8.2.3.2 Accident Analysis

3. The term V_w in Equation 8.2-20 (Page 8.2-19) only includes the volume of one wall of the HSM whereas the term V_r includes the entire weight of the roof. Furthermore, the value for I (74,088 in ^{4}) given on Page 8.2-20 represents the moment of inertia of only one wall. Shouldn't the calculation for mass presented in Equation 8.2-20 include only one-half of the mass of the roof? Also, shouldn't this calculation include the effect of the DSC and its support?

RESPONSE:

As stated on page 8.2-19 the mass was taken as the entire top slab plus one half of the side walls. Therefore, the term V_w is the volume of one half of two walls or one total wall. The mass of the lower half portion of both walls is assumed to be reacted at ground level. This approach of lumping mass based on tributary area is commonly practiced throughout the industry. Additionally the moment of inertia is based on one wall. Equation 8.2-21, however, contains a factor of 2 to account for the stiffness of both walls. Hence, the module frequency calculations are correct as shown on pages 8.2-19 and 8.2-20.

The effects of the DSC and the support assembly were omitted since they act below the halfway point on the module walls and will not have a significant effect on the results. The mass of the DSC and support assembly per 12 inch section of the HSM is 3.9 $\frac{\text{lb. s}^2}{\text{in}}$. If this mass were conservatively lumped at the HSM roof the frequency would be 35.9 Hz. This conservative value is still beyond the control point A of 33 Hz shown in Tables 1 and 2 of NRC Regulatory Guide 1.60. Therefore, the maximum amplification factor of 1.0 is still valid for the analysis.

QUESTION: 90

8.2.3.2 Accident Analysis

4. Shouldn't the parameter V_u in Table 8.2-6 (page 8.2-21) actually be labeled "M_u" as in Table 8.2-3?

RESPONSE:

The term V_u in Table 8.2-6 will be replaced by the correct term M_u in the next revision of the Topical Report.

QUESTION: 91

8.2.3.2 Accident Analysis

5. No mention is made in the sub-sections entitled "DSC Stress Analysis" (Page 8.2-16) and "Horizontal Seismic" (Page 8.2-24) of combining horizontal seismic loads from two directions. Seismic loads from two horizontal directions and the vertical direction should be combined unless a statement can be made that loads from a particular direction can be considered to be negligible (i.e., as was done on Page 8.2-20).

RESPONSE:

The DSC is retained axially for a seismic event as described in the response to Question 7.3.2 (Number 12). Consequently, an axial membrane stress will be experienced due to the 0.5G horizontal loading. Assuming an equal stress distribution throughout the DSC yields a stress of 0.19 ksi. Compared to the local bending stress of 9.58 ksi calculated by a horizontal loading perpendicular to the DSC, this axial stress is negligible. Additionally, no increase in combined stress will occur when summed with the other horizontal stress using the SRSS method. Therefore, stress due to an axial horizontal acceleration along the DSC shell is minimal and can be considered negligible. A statement of this effect will be added on page 8.2.24 of the Topical Report.

For the support assembly seismic evaluation the stress due to horizontal acceleration in the axial direction can be calculated directly by factoring the results from the friction load case. The maximum bending and shear stress in the W8 x 28 are 0.56 ksi and 0.68, respectively. The maximum bending and shear stress in the W4 x 33.5 are 0.76 ksi and 0.34 ksi. These values will be combined by the SRSS method with the other horizontal stress and then added absolutely with the vertical seismic stress. This revised value will be included in the subsequent load combination reported in Table 8.2-11. The seismic load combinations in Table 8.2-11 will be revised to reflect the results listed below.

Component	Load Combination	Calculated Stress (ksi)		
		Axial	Bending	Shear
W 8 x 28	$DW_S + DW_C + SSE$	0.40	8.19	5.08
WT4 x 33.5	$DW_S + DW_C + SSE$	0.77	17.37	5.01

QUESTION: 92

8.2.3.2 Accident Analysis

6. As a follow-on to the above question, it is noted that no mention is made of the possibility of the cask sliding longitudinally along the rails when loaded seismically. If such a possibility exists, could some part of the HSM structure be impacted by the DSC (e.g., the door), resulting in damage to either the HSM or the grapple on the end of the DSC? If the DSC cannot slide, shouldn't some sort of load be applied to the rails, and consequently to the DSC support assembly?

RESPONSE:

As described in the response to question 7.3.2 (number 12) a seismic retaining assembly will be installed prior to closing the HSM door. This assembly will prevent longitudinal sliding of the DSC along the rails during a seismic event. Therefore, no possibility of impact damage is present.

As described in the answer to question 8.2.3.2 (number 5) the DSC support assembly will experience seismic axial loads along the WT 4x33.5 rails. Consequently, the Topical Report will be revised to include stresses caused by this loading. See answer to question 8.2.3.2 (number 5, for revised stresses due to this additional horizontal seismic loading.)

QUESTION: 93

8.2.4.2 Accident Analysis

1. Reference to Table 3.2-2 for flood loadings is incorrect (Page 8.2-24). Also reference to Table 3.2 is incorrect (Page 8.2-28).

RESPONSE:

The reference to Table 3.2-2 on page 8.2-24 and Table 3.2 on page 8.2-28 will be changed to Table 3.2-3.

QUESTION: 94

8.2.4.2 Accident Analysis

2. A flood level which caused the bottom portion of the DSC to be in contact with water while the upper portion remained in air could cause considerable thermal stress in the canister. Have these loadings been included with the pressure loadings?

RESPONSE:

A flood level precisely high enough to block air flow but is not considered a credible situation and therefore the analysis has not been performed.

QUESTION: 95

8.2.5.2 Accident Analysis

1. The discussion on page 8.2-29 implies that the decelerations used by NUTECH were obtained by multiplying actual drop-test data (obtained by ORNL) by one-sixth to reflect a 5 foot drop height instead of a 30 foot drop height. Examination of the GE Consolidated Safety Analysis Report of the IF-300 shipping cask reveals that NUTECH used decelerations which the Stearns-Rogers Company calculated using a dynamic analysis program, "Dyrec". Since the Stearns-Rogers results envelope the ORNL results, this is a conservative approach. However NUTECH should revise the discussion to reflect their actual process. Also NUTECH should reference the data by appendix.

RESPONSE:

The discussion on page 8.2-29 will be revised to reflect the actual process. Additionally, data will be referenced by appendix.

QUESTION: 96

8.2.5.2 Accident Analysis

2. A second related question concerning drop decelerations has to do with the data in Appendix V2 of the GE document NEDO-10084-3. Why did NUTECH not use the deceleration for top end vertical drop as reported in this appendix? It is more conservative than the Stearns-Rogers data.

RESPONSE:

GE document NEDO-10084-3, Appendix V2, September 1984, was reviewed for possible higher vertical drop deceleration. This document presents the 30 ft drop impact analysis for the fuel rods and not the reevaluation of the impact time history. As a matter of fact, the time history, Figure 5-1, page V2-47, presented in this document is identical to the time history developed by Stearn-Roger and presented in Appendix V-1 of the GE Safety Analysis Report.

Additionally, as stated in response to previous questions the maximum deceleration forces which the DSC is the 48g vertical and 34g horizontal. These are to be considered as the governing limits and not the drop height of the cask.

QUESTION: 97

8.2.5.2 Accident Analysis

3. NUTECH has not provided a discussion on the possibility of the cask dropping vertically through 5 feet and then tipping over and slapping down. Logically, if the cask could experience a 5 foot vertical drop, then it would almost certainly also experience a tip-over and slap down. In fact, the slap-down case would be more severe than the 5 foot vertical drop because the centers of gravity of the shipping cask and the DSC both exceed 5 feet in the vertical orientation. Please discuss the reason for not including the slap-down case.

RESPONSE:

As explained in response to question 2 of Section 3.1.2.2, the limiting drop deceleration for the DSC is 34g in the horizontal orientation and 48g in the vertical orientation. The users of NUHOMS Systems need to ensure the DSC is not subject to a drop which could cause deceleration values in excess of these values during handling operations.

Nevertheless as shown in Figure 1 (Page 8.45) the effective height of the drop during a slap down is the difference in the cask center of gravity while standing on its corner and in the horizontal position which is less than 5 feet, and as such the deceleration associated by the slap down is enveloped by the deceleration of the 5 ft horizontal drop.

QUESTION: 98

8.2.5.2 Accident Analysis

4. In order for the statement of page 8.2-31, "The liner eliminates any secondary impact forces generated by the impact of the two surfaces" to be evaluated, NUTECH should specify what the liner material is and what the dimensions of the liner are. Without these data it is not possible to evaluate what the attenuation or amplification characteristics of the liner might be.

RESPONSE:

The liner material is stainless steel 304. The dimensions of the liner will be dependent on the cask dimensions and fabrication tolerances of the DSC.

QUESTIONS: 99

8.2.5.2 Accident Analysis

5. It appears that the effect of the 2 inch diameter rods was not considered for the horizontal drop case. NUTECH should discuss this omission, or recalculate the stresses in the spacer disc.

RESPONSE:

The weight of the rods in comparison to the weight of the seven fuel assemblies and spacer disk are negligible. Their additional weight would only have a minimal impact on the stresses within the spacer disk. The total weight of the 4x26 inch segments of the 2 inch diameter rods is 96 pounds while the total weight of the 26 inch segment of the 7 fuel assemblies and spacer disk is 1818.1 pounds. Additionally, the rod locations are in an area away from the fuel assemblies, areas considered to be critical stress locations. Because of their small incremental weight and location, the 2 inch diameter rods were not included in the stress analysis of the spacer disk.

QUESTION: 100

8.2.8 Dry Storage Canister Leakage

The statement that the DSC is designed for no leakage under any conditions cannot be substantiated since very small leakage rates cannot be detected or measured. Helium is difficult to contain, so it would be expected that some decrease in Helium concentration would occur over the lifetime of the canister. Please provide an estimate based on experimental data or analysis of the leakage rate or concentration versus time.

RESPONSE:

Please see the answer to Question 1, Section 1.1.

QUESTION: 101

8.2.9.1 Accident Analysis

1. What is the basis for using a value of 25% fission gas release fraction as opposed to the 30% upper bound cited from reference 8.42 (should read 8.43) and also quoted on page 8.2-54 in Section 8.2.8?

RESPONSE:

As stated in response to question 8.1.1.1, an EPRI study of 124 fuel rods with a large range of linear heat ratings, burnups, enrichment, fuel pellet densities, and operating histories has shown only 0.33% to 12% of the fission gas is released. Futhermore, the EPRI document is much more conclusive than reference 8.43. Also, the 30% upper bound cited on page 8.2-54 is used for radiological assessment and not for pressure calculations. Therefore, the 25% fission gas release exceeds the maximum documented fission gas release of 12%.

The 30% release fraction of reference 8.43 is not an "upper bound" of actual release fractions but is the authors' selected conservative value for radiological assessment.

QUESTION: 103

8.2.9.1 Accident Analysis

3. As previously mentioned in comment 4 for Section 8.1.1.1, the effect of diurnal temperature variation on DSC fatigue should be considered in addition to the 50 seasonal cycles.

RESPONSE:

DSC fatigue analysis due to diurnal temperature variation was included in the response to question 2 of Section 1.2.2.

QUESTION: 104

8.2.9.1 Accident Analysis

4. What is the maximum gas inventory within one fuel assembly including fill and fission gases? How was this maximum inventory established? Do criteria need to be established on maximum gas inventory of fuel to be stored? Please discuss.

RESPONSE:

The fill gas for one fuel assembly was assumed to be 9669.6 in^3 at 72°F and 1 atm while the maximum fission gas contained in one fuel assembly was 25071 in^3 at 72°F and 1 atm. The volume of fill gas based on a free volume of a fuel assembly being filled with helium at 500 psi and 72°F. The fission gas inventory is based on a fuel assembly burnup of 35,000 MWD/MTM and a fission gas production rate of 1 atom per 4 fission.

Because of the high fill gas pressure, 500 psi, that was assumed no criteria needs to be established on the maximum gas inventory of the fuel to be stored.

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QUESTION: 105

8.2.10 Load Combinations

1. The load combinations presented in Table 8.2-10 (Page 8.2-61) do not consider the effects of either wind, tornado or flood on the HSM. Shouldn't these loads be considered as required by ACI 349-80. Section 9.2.1? Also shouldn't the material properties be evaluated at the worst thermal case examined, i.e., ambient T=125°F with inlets plugged? (See page 8.1-70).

RESPONSE:

As stated on page 8.2-60 many of the general event combinations, as shown in Table 3.2-5 are enveloped by the load combinations shown in Table 8.2-10. Hence the effects of wind, tornado and flood were considered, but were enveloped by the cases listed in Table 8.2-10. Results of these accident analyses presented in Chapter 8 verify this statement.

The material properties for the concrete are taken at the temperature of the individual load combination. For example, material properties for load combinations 2, 4, and 10 are taken at the thermal operating temperature since these combinations are for operating conditions. Load case 7, however, contains thermal accident loads. Therefore, material properties for this combination are taken at the HSM temperature when air inlets are blocked and ambient temperature is 125°F.

QUESTION: 106

8.2.10 Load Combinations

2. Please provide an analysis of the HSM corbels, bearing plates, bolts and reinforcing steel for the load combination cases #7 and 10. The temperature of the corbels should be specified and the material properties for temperature should be used.

RESPONSE:

As stated in page 8.1-32 of the Topical Report, slotted holes are used in the wideflange support beams for the purpose of allowing thermal movement, and as such there is no direct thermal force transferred to the corbels. Furthermore as stated in the response to question 3 of Section 8.1.1.5 a multiplier of 1.7 was used for various load combinations presented in Table 8.2-12. The use of 1.7 multiplier conservatively envelopes all the load combination cases specified by ACI 349-80.

Table 8.2-12 will be revised to incorporate the capacity of the corbel, bearing pads and the bolts at both the operating and accident temperatures.

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QUESTION: 107

8.2.10 Load Combinations

3. Note 7 in Table 8.2-10 on page 8.2-61 implies that the material properties for the thermal accident load are taken at 300°F, however Table 8.1-2 shows that the maximum concrete temperature for the inside roof is 321°F, and the side wall is 344°F. Please clarify.

RESPONSE:

Note 7 in Table 8.2-10 on page 8.2-61 will be changed to 350°F to conservatively bend all accident temperature throughout the HSM.

QUESTION: 108

8.2.10 Load Combinations

4. Table 8.2-11 reports DSC support assembly stresses for various load combinations. The note (3) indicates allowable stresses for the SA 36 structural steel at 200°F. NUTECH should show what the temperature of the DSC is where it and the Tee support assembly rails are in contact for the worst thermal case examined, i.e., ambient T=125°F with inlets plugged (see page 8.1-70).

RESPONSE:

As stated on page 3.2-12, only one accident is considered to occur at one time. Therefore, the HSM and DSC are considered to be at temperatures associated with the normal operating conditions, 70°F ambient temperature during a seismic event or when friction loads are applied. The jammed DSC event is considered to be only credible during the HSM loading operation when the HSM is at ambient temperature 70°F. The allowable stresses for note 3 will be changed to 250°F. This bounds the conservative assumption that the temperature of the DSC support assembly is equal to the exterior surface temperature of the DSC, 230°F.

QUESTION: 109

10.0 OPERATING CONTROLS AND LIMITS

10.1 Proposed Operating Controls and Limits

The fuel stored should also have an axial power peaking of equal or less than 1.08 for decay heat, since this parameter was used in the thermal design.

RESPONSE:

As stated in response to Question 8.1.3.1-2, the axial peaking power factor was intended to be a conservatism in the thermal analysis not a limiting parameter on the fuel that may be stored in NUHOMS. The only thermal parameter for fuel is the heat output per fuel assembly must be limited to 1.0 kw, as indicated in Section 10.3.1.1.

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QUESTION: 110

10.2.2.2 Technical Conditions and Characteristics

1. Condition 3 regarding the DSC helium leak rate of the primary weld should apply to all welds on the canister including the axial seam weld. Leak rate as a function of temperature also needs to be addressed.

RESPONSE:

Please see the answer to Question 1, Section 1.1.

QUESTION: 111

10.2.2.2 Technical Conditions and Characteristics

2. Why is the limiting heat load of 1 kw/assembly not included in the list of seven technical conditions and characteristics?

RESPONSE:

The heat load of 1 kw per assembly is considered to be a functional limit as discussed in Sections 10.2.1 and 10.3.1.

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QUESTION: 112

10.2.3 Surveillance Requirements

1. What provisions have been made for inspection of internal air passages for blockage due to buildup of organic matter, insect activity, etc.?

RESPONSE:

Visual inspection of the internal air passages will be conducted from outside of the HSM. Any regions of the internal air passages that are not visible from outside are in an area which is not conducive to the growth of organic matter due to radiation and temperature levels.

See response to Question 1.2.2.1 for additional information.

QUESTIONS: 113

10.2.3 Surveillance Requirements

2. What provisions have been made for drainage of any water which enters the HSM through the inlets or outlets? Buildup of water could also cause partial or complete blockage of inlet flow in the absence of drains. What inspections are performed to assure that water does not accumulate in the bottom of the HSM?

RESPONSE:

Two 1 inch diameter pipes will be installed in the front wall of the HSM. In the event that any water should enter through the air inlet the pipes will allow for water to drain from the air inlet chamber. This will eliminate the possibility of water building up and causing a partial or complete blockage of the inlet flow and the necessity of inspecting passage ways.

The air outlets are designed to be water tight and therefore water will not be allowed to flow into the HSM.

QUESTION: 114

10.2.3 Surveillance Requirements

3. Surveillance is performed every 48 hours to assure that air is flowing through the module. How will acceptability of the flow rate be determined? How will the flow rate be measured, or how will it be determined that the flow criteria are satisfied.

RESPONSE:

Routine surveillance or inspection of the air inlets conducted on a weekly basis and within 48 hours after an unusual events.

QUESTION: 115

10.3.1.1 Fuel Specifications

Please clarify which parameters are for an assembly and which are for the canister, e.g., neutron source, weight.

RESPONSE:

The following list clarifies which parameters are for the assembly and which are for the canister. This table will be incorporated into the revised Topical Report.

Burnup	$\leq$ 33,000 MWd/Mt
Post irradiation time	$\geq$ 5 years
Initial enrichment	$\leq$ 3.5 % ^{235}U
Weight per distance between any adjacent spaces	$\leq$ 665 kg
Decay power per assembly	$\leq$ 1 kw
Neutron source per canister	$9.98x10^8$n/sec
Gamma source for canister	$7.76x19^{15}$ photons/sec

QUESTION: 116

10.3.2.1 DSC Vacuum Pressure During Drying

What is the basis for the one hour time at pressure criteria for the drying operation? Has the possibility of a waterlogged rod been considered?

RESPONSE:

Based on engineering judgement one hour time at the specified pressure should be sufficient.

A water logged fuel assembly has not been considered. Prior to inserting the fuel assemblies into the DSC, a fuel assembly will be visually examined for structural and mechanical integrity.

Additionally, at the temperature and vacuum pressure which the fuel rods will be exposed to during the drying operations, any water in the fuel rods should evaporate and be suctioned from the DSC cavity. If any water does not come out under the vacuum conditions it is not likely to come out with 1 Atm of He in the canister.

QUESTION: 117

10.3.2.2 DSC Helium Backfill Pressure

It appears that a hold time at this pressure will need to be specified to assure that thermal equilibrium has been reached.

RESPONSE:

As discussed in the response to Question 5.1.1.3-1, the backfilling procedures have been changed to include monitoring the pressure.

QUESTION: 118

10.3.2.3 DSC Helium Leakage Rate of Primary Weld

1. Calculation of 631 cm^3 of He appears to correspond to 2 atmosphere of pressure rather than 1.5 atmosphere.

RESPONSE:

631 cm^3 of He does correspond to 2 atmosphere and 473 cm^3 corresponds to 1.5 atmosphere. 631 cm^3 will be revised to 473 cm^3. This change has no impact on the leak rate specification of the primary weld.

QUESTION: 119

10.3.2.3 DSC Helium Leakage Rate of Primary Weld

2. The leak rate corresponds to the detection level of the helium sniffer. However, the calculation assumes a single leak at this level. In reality, leakage may occur at several locations along the many feet of weld and by diffusion through the metal matrix of the caniser itself. Has this type of leakage been considered? Please address.

RESPONSE:

Please see the answer to Question 1, Section 1.1.

QUESTION: 120

10.3.2.7 Maximum Air Exit Temperature

1. If the temperature rise is greater than 100°F and small fans are provided as stated in Item 5, the passive nature of the cooling mechanism will be lost. Since the design is based on achieving passive cooling, the addition of fans should either be included in the design or eliminated as a corrective action.

RESPONSE:

The addition of fans will be eliminated as a corrective action in the report. If the temperature rise is greater than 100°F, the DSC will be removed from the HSM or additional information and analysis will be provided to show that the existing condition does not represent an unsafe situation.

QUESTION: 121

10.3.2.7 Maximum Air Exit Temperature

2. The cooling air temperature rise is to be checked twenty-four hours after placement of the canister in the HSM. What features of the design preclude the need for later measurements to assure that the 100°F temperature rise criterion is satisfied. If such assurance cannot be provided, then more frequent measurements may be necessary.

RESPONSE:

There are no moving components that could significantly alter the air flow or cause an increase in the temperature change.

QUESTION: 122

10.3.3.1 Surveillance of the HSM Air Inlets

1. How does this surveillance assure that there is no blockage of the outlets or internal blockage? Does surveillance include any measurement of flow or temperature rise of the cooling air?

RESPONSE:

There is no reason to inspect any other HSM air pathways other than the air inlets. Bird screen on the air inlets and outlets will prevent any debris which may significantly alter the air flow through the HSM. There are no lose or moving parts within the HSM that could fail and block the air flow. Additionally, the non-visible portions of the air pathways are not conducive to the growth of organic matter. Therefore, inspection of the air inlets and outlets, the only points where air flow could be restricted, will assure that there is no blockage of the outlets or intervals.

Surveillance does not include measurement of air flow or temperature rise of the cooling air.

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QUESTION: 123

10.3.3.1 Surveillance of the HSM Air Inlets

2. Page 10.3-17 states that for "normal operations, inspection of air inlets once per week will assure that any local obstructions can be removed". If the HSMs are inspected once per week, how can NUTECH assume that an air blockage would be no longer than [illegible] hours? This question also relates to Section 8.2.7.2.

RESPONSE:

Analysis in Chapter 8 showed that no temperature limits are exceeded if all inlets of the HSM are completely blocked. The complete blockage of all inlets and outlets is not expected to occur except during an accident condition. As stated in Section 10.3.3.1, the air inlets will be inspected within 24 hours of an abnormal event.

QUESTION: 124

DETAILS OF HEAT TRANSFER ANALYSIS OF THE NUHOMS SYSTEM

1. On page B.4, it is assumed that the gas behaves the same as the gas in a horizontal annulus. However, on page B.5, a formula said to be applicable for a vertical annulus is used to determine apparent thermal conductivity. Please explain this discrepancy. The gas "annulus" is horizontal for drying and vertical for storage.

RESPONSE:

Reference to the vertical annulus on page B-5 is incorrect. The words vertical on page B-5 and B-6 will be changed to horizontal. The equation listed in Appendix B is used to determine the thermal conductivity of gas in a horizontal annulus.

When the cask is in the vertical position, the most sever temperature conditions will be encountered during vacuum dry of the DSC. As stated on page B.6, the apparent thermal conductivity of the gas in the thermal model was approximated by dividing the apparent thermal conductivity of air in a horizontal position by 1/760.

QUESTION: 125

DETAILS OF HEAT TRANSFER ANALYSIS OF THE NUHOMS SYSTEM

2. The length of the fuel assembly L is given on page B.5, but does not appear to be used in any of the equations. Is this correct? Why is the number cites?

RESPONSE:

Length of the fuel assembly, L, is not required and will be removed in the revised Topical Report.

QUESTION: 126

DETAILS OF HEAT TRANSFER ANALYSIS OF THE NUHOMS SYSTEM

3. Table 8.1-13 gives the average helium temperature for air inlet temperatures of -40, 70 and 125°F as 315, 389 and 429°F. Table B-2 gives apparent helium conductivity at three values of delta T for average gas temperatures of 80, 160, 440 and 620°F. A value of 1.6 was determined for apparent thermal conductivity by averaging the values in Table B-2. In light of the range of helium temperatures for which analyses were performed, it does not seem appropriate to include the values at 620°F, and particularly at 80°F in arriving at the average value. Justify that 1.6 is a representative value for apparent thermal conductivity.

RESPONSE:

The average value of the apparent helium conductivity for the two temperatures values of 160 and 440°F is 1.55. This 0.05 difference in average value will have an insignificant difference in the temperature.

QUESTION: 127

DETAILS OF HEAT TRANSFER ANALYSIS OF THE NUHOMS SYSTEM

4. For the calculation of h_{eff} on page B.13, the assumption of delta-T = 300°F is stated, but does not appear to be needed.

RESPONSE:

The assumption of delta-T = 300°F is necessary. This assumption is used later in the derivation on page B-15.

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QUESTION: 128

DETAILS OF HEAT TRANSFER ANALYSIS OF THE NUHOMS SYSTEM

5. Contrary to the statement on page B.16, the two-step approach to using the Wooten-Epstein Formula (WEF) appears to be closer to the actual application. The second step is essentially the manner in which the result is used.

RESPONSE:

The first step is deriving the boral sleeve temperature considering the DSC wall temperature and a pseudo assembly of 7 rods. The second step goes from the boral guide sleeve temperature to the central rod temperature using a single assembly.

QUESTION: 129

DETAILS OF HEAT TRANSFER ANALYSIS OF THE NUHOMS SYSTEM

6. Corrected values for K_{fuel} for the alternative WEF approach should be provided.

RESPONSE:

The correct value for K_{fuel} using the WEF on a single 15x15 fuel assembly at the range of expected temperatures is 5.23 k^{He}.