NAC-UMS
Docket # 71-9270
TAC # L22452

NAC INTERNATIONAL

RESPONSE TO THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

REQUEST FOR ADDITIONAL INFORMATION

(RAI-2 JUNE 14, 2001)

NAC UNIVERSAL TRANSPORT SYSTEM (NAC-UMS®)

(TAC. NO. L22452, DOCKET NO. 71-9270)

SAR SUBMITTAL – REVISION UMST-01D

NOVEMBER 2001

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

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CHAPTER 1: GENERAL INFORMATION

1-1 Section 1.2.3 Contents of Packaging

Revise Table 1.2-6, "Loading Table for PWR Fuel," and Table 1.2-7, "Loading Table for BWR Fuel," to include the maximum burnup of up to 50,000 MWD/MTU.

In Section 1.2.3 Contents of Packaging, the maximum burnup for both PWR and BWR fuels is indicated to be 50,000 MWD/MTU. Table 1.2-6 and Table 1.2-7 only go up to a burnup of 45,000 MWD/MTU. Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

Item 8 of Section 1.2.3 is revised to say, "Cask general fuel contents. . ." and to show a maximum burnup of 45,000 MWD/MTU for PWR and BWR fuel. Tables 1.2-6 and 1.2-7 are correct as shown.

Item 9 of Section 1.2.3 is revised to say, "Cask site-specific contents may include Maine Yankee fuel with maximum burnup up to 50,000 MWD/MTU and GTCC waste as described in Section 1.3.1.1 based on the site-specific fuel characteristics and preferential loading pattern."

The contents description for Maine Yankee site-specific fuel (Section 1.3.1.1) describes Maine Yankee fuel with burnups up to 50,000 MWD/MTU, but that level of burnup does not apply to the general Contents of Packaging description provided in Item 8 of Section 1.2.3.

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1-2 Provide further justification for the shipment of fuels having burnups greater than 45,000 MWD/MTU.

Section 71.55(d)(2) requires that the geometric form of the package contents of a spent fuel package will not be substantially altered under the conditions specified for normal conditions of transport. This following additional information is needed to demonstrate whether the fuel will maintain its geometry under normal conditions of transport:

1. Since the SAR indicated that transportation of high burnup fuel in the NAC-UMS Transport cask will occur only following storage (i.e., there will be no direct, wet loading of the fuel into the TSC and transportation overpack), evaluate the change in mechanical properties (e.g., yield and tensile strengths, ductility) of the cladding that may occur as a result of 20-years of storage.
2. The response to RAI#1 Question 1-9 mainly addressed the mechanical properties of high burnup fuel spent fuel cladding based on DOE studies. A comparison between the expected stresses and strains on the cladding under normal conditions of transportation (i.e., vibration normally incident to transport and under normal condition free drop from one foot) and the expected mechanical properties of the cladding after 20 years of storage should be provided.

NAC Response

Twenty (or even 50) years of on-site storage will not affect the integrity of spent fuel cladding that is properly stored in a dry storage canister in an inert environment or its ability to withstand the normal conditions of transport. NAC proposes that the fuel specifications for the UMS® system limit the spent fuel contents to have a maximum burnup of 50,000 MWD/MTU for Maine Yankee site specific spent fuel. Already docketed research reports document the characteristics of similar fuel with burnups up to 54,000 MWD/MTU (DOE/ET/34030-11, CEND-4274, "Hot

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Cell Examination of Extended Burnup Fuel from Fort Calhoun" and DOE/ET/34212-50, BAW-18747, "The Hot Cell Examination of Oconee 1 Fuel Rods after Five Cycles of Irradiation"). These reports provide clear evidence that the properties of irradiated higher burnup fuel cladding remain adequate to withstand normal conditions of transport.

During reactor operations, the flux in the reactor is greater than 10^{20} neutrons per sec-cm^2 and the flux level while in storage is in the range of 10^{15} neutrons per sec-cm^2 at the initiation of storage and declining into the future. Even after a 50-year storage timeframe, the cladding would not experience any appreciable additional neutron fluence. Therefore, irradiated fuel, even with higher burnups and additional fluence, will remain structurally sound and maintain its integrity. There would be little change to the cladding material properties (yield strength, ultimate strength, ductility, etc.) of the cladding during a 20-year storage, since there is only a very minor change in the total fluence. Currently, the average discharge fuel burnup from a reactor is greater than 50,000 MWD/MTU; this shows fuel cladding is still structurally sound.

Further, the fuels with burnups exceeding 45,000 MWD/MTU are stored only in the basket periphery so that cladding creep during storage is absolutely minimized. The peak cladding temperatures of this fuel are well below those where research shows cladding creep during storage could become an issue. So, cladding creep after 20 years of storage would result in total creep of less than 0.25%, or so.

Additionally, in preparation for transport, the entire canister loaded with spent fuel will be lifted from the storage cask (with the use of the transfer cask) and placed into the transport cask (a "dry" load of the transport cask). Therefore, the fuel, even after 20 years of storage, would not be cycled through another draining, drying and backfill process, so no new thermal transients from the operations would be experienced.

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Finally, transport stresses on fuel cladding are very low (see Section 3.4 of the SAR). Typical yield strengths of Zircaloy cladding exceed 50 ksi, even with creep strains of 0.25% to 0.5%. Therefore, transport stresses will not threaten the cladding integrity.

It is noted that no particular period of storage is required for a canister before it can be loaded in the UMS® Universal Transport Cask. If all of the fuel in the canister meets the loading table requirements provided in Section 1.2.3, the canister may be closed and immediately placed in the transport cask.

The conclusion is that:

- Research shows that cladding properties at the time of loading are adequate to withstand normal conditions of transport.
- Additional fluence on the cladding during storage is insignificant as compared to the fluence levels in the reactor.
- Cladding creep during storage is not significant, based upon very low peak cladding temperatures.
- Loading operations for transport impose no new thermal transients.
- Actual transport stresses in the cladding are well below cladding yield stress.

Therefore, the mechanical properties of the fuel cladding following up to 20 years in storage remain very similar to those at the time of loading into the canister, and fuel with burnups up to 50,000 MWD/MTU will maintain its structural integrity after 20 years of dry storage and is safe to transport.

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1-3 Justify how all of the acceptance criteria and requirements of ISG-11, Revision 1, are implemented in the NAC-UMS Transport SAR. In the discussion, describe the actions that the licensee will take to assure that the criteria for intact high burnup spent fuel assemblies are met prior to loading the cask with high burnup fuel.

Section 71.55(d)(2) requires that the geometric form of the package contents of a spent fuel package will not be substantially altered under the conditions specified for normal conditions of transport. It is unclear from the SAR whether all of the acceptance criteria of ISG-11, Revision 1, are met.

NAC Response

The ISG-11, Revision 1 (now ISG-15) acceptance criteria and requirements are specified in the loading procedure of the approved Final Safety Analysis Report (FSAR) for the UMS® Universal Storage System (Docket 72-1015), which provides for the loading of the canister with high burnup fuel. Note: This same loading procedure was included in UMS® Transport SAR Section 7.5.1, which is being deleted in response to RAI 7-2. Therefore, at the time of loading the canister into the UMS® transport cask, the criteria and requirements of ISG-15 for high burnup fuel evaluation and loading are satisfied.

To incorporate the defined percentages of failed fuel, the canister and cask cavity internal pressures are recalculated. ISG-11 (now ISG-15) requires a containment evaluation assuming 50% failure of high burnup fuel that has an oxide layer thickness greater than 70 microns. Above this level, assemblies must be placed in damaged fuel cans. The worst case normal conditions containment analysis is, therefore, 12 intact standard assemblies with 3% rod failure, 8 high burnup assemblies classified as intact with 50% failure of the 3% high burnup rods, and 50% failure of rods inside the four damaged fuel cans. This configuration is bounded by the 20% average release fraction applied to a full canister load of high burnup assemblies.

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NAC Response to RAI 1-3 (Continued)

The PWR leak rate calculation is revised based on the 50% assumed failure fraction. The PWR and BWR fuel leak rate evaluations (Sections 4.1.3, 4.2 and 4.3) are revised based on the revised PWR and BWR pressure calculations (see the NAC Response to RAIs 3-10 and 3-11). The BWR allowable leak rate is increased significantly due to a lower normal condition operating pressure. A lower pressure for the PWR and BWR fuel configurations results from a lower cavity gas temperature, which results from an improved fuel conductivity model.

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CHAPTER 2: STRUCTURAL EVALUATION

2-1 Clarify the SAR Section 2.1.1.4 discussions on: (1) the calculated maximum g-load of 54.9 g for the 30-ft free drop and (2) the 55 g end-drop design load for the BWR basket.

(1) SAR Table 2.6.7.5-6 lists a peak side drop deceleration load of 56.2 g, which is greater than the 54.9 g as stated in Section 2.1.1.4. Also, SAR Section 2.6.7.5.8 states that the side drop is bounding for this cask design, which contradicts the Section 2.1.1.4 statement, "...the maximum ...impact load is calculated to be 54.9 g in the oblique-drop orientation." (2) SAR Table 2.6.7.5-6 lists a peak end drop deceleration load of 57.8 g and a design basis deceleration of 60 g.

10 CFR 71.7(a) requires complete and accurate information.

<u>NAC Response</u>

Section 2.1.1.4 is revised to refer to a maximum calculated impact load of 57.8 g in the top end drop orientation. As shown in Table 2.6.7.5-6, the maximum peak acceleration occurs for the top end drop event. The calculated accelerations for the bottom end and side drop events are less. The reference to the BWR basket configuration is deleted as the maximum acceleration applies to either the PWR or BWR configuration.

Section 2.6.7.5.8 is revised to clarify the analysis and parametric studies that have been performed to verify the use of the LS-DYNA program for impact limiter evaluation.

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2-2 Clarify, as appropriate, the underlined typographical or editorial errors:

Table 2.1.2-1, ASME Section III, Subsection NF Code Exception, "NB-2000 ASME Approved Material Supplier."

Article NF-2000 should have been considered. 10 CFR 71.7(a) requires complete and accurate information.

NAC Response

Table 2.1.2-1 is revised to refer to the material supplier exception to Article NF-2000 of the ASME Code for the GTCC waste basket assembly. The table is also revised to refer to NAC-approved suppliers with CMTRs in accordance with Article NF-2000 requirements in lieu of the NB-2000 code requirement for ASME Approved Material Suppliers.

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2-3 Clarify the SAR Section 2.1.2.5.5 discussion on the maximum deceleration of 54.9 g for all impact conditions.

SAR Table 2.6.7.5-6 lists a peak side drop deceleration load of 56.2 g, which is greater than the 54.9 g as stated. 10 CFR 71.7(a) requires complete and accurate information.

NAC Response

Section 2.1.2.5.5 is revised to correct the maximum deceleration from 54.9 g to 57.8 g. As shown in Table 2.6.7.5-6, the peak acceleration of 57.8 g occurs in the top end drop event. The side drop peak acceleration is 52.0 g.

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2-4 With respect to SAR Tables 2.6.7.5-3 and -4, (1) provide necessary references from which the dynamic redwood stress-strain properties are generated and presented, (2) compared to the SAR Table 2.6.7.5-1 tabulation of the static stress-strain curves each defined with nine data points, provide justification of using only three or four data points for a dynamic stress-strain curve, and (3) provide an explanation of the much lower than expected dynamic crush strengths to be considered in the LS-DYNA impact limiter analyses.

(1) Complete and accurate information should be provided for staff review, per 10 CFR 71.7(a), (2) a sufficient number of data points may have to be considered for defining stress-strain material properties, in a LS-DYNA impact limiter finite element model, to properly model potential effects of stress overshooting at small elastic strains and deformation bottoming out at large plastic strains, and (3) the data presented in SAR Tables 2.6.7.5-3 and -4 appear to be inconsistent with a generally observed material behavior that crush strengths by dynamic testing are higher than those by static testing. For instance, the staff notes that, for the redwood parallel-to-grain loading direction, the dynamic stress of about 1,500 psi, at a 40% strain reported in Table 2.6.7.5-3, is much smaller than the static stress of about 6,000 to 8,000 psi, at a 44% strain reported in Table 2.6.7.5-1.

<u>NAC Response</u>

(1) The reference for the dynamic stress-strain properties of redwood is provided separately as proprietary information (Proprietary Calculation EA790-2239, "Reduction of Hot (200°F) and Cold (-40°F) Redwood Stress-Strain Test Data"). The reference for the dynamic stress-strain properties of balsa wood was provided as a proprietary calculation submitted in support of a request for an amendment to Docket # 71-9235 on 11/08/2000. Refer to NAC letter ED20001674, Proprietary Calculation EA790-2233, "Reduction of the Redwood and Balsa Test Data."

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NAC Response to RAI 2-4 (Continued)

(2) Because of the proprietary nature of the redwood and balsa wood data, only four data points are presented in the SAR text. However, for the LS-DYNA analyses, 30 data points are used to accurately represent the stress-strain curve. Refer to Proprietary Calculation EA790-2239, "Reduction of Hot (200°F) and Cold (-40°F) Redwood Stress-Strain Test Data" for complete redwood stress-strain data and Proprietary Calculation EA790-2233, "Reduction of the Redwood and Balsa Test Data" for complete balsa wood stress-strain data.

(3) A complete evaluation of all subcontracted services associated with drop testing led NAC to reperform some of the earlier redwood testing. A new series of parallel-to-grain static and dynamic tests have been performed on redwood samples to determine the characteristic properties (Refer to Proprietary Calculation EA790-2239, "Reduction of Hot (200°F) and Cold (-40°F) Redwood Stress-Strain Test Data"). The results of the new testing program showed that the previous parallel-to-grain redwood properties were incorrectly interpreted. Current analyses use 30 data point stress-strain curves based on the new test data. The following table is a summary of the data used in the LS-DYNA analyses.

Strain (in/in)	Parallel-to-Grain—Stress (psi)			
	Hot Static	Hot 25 ε/sec	Cold Static	Cold 25 ε/sec
0.000	0	0	0	0
0.100	3736	5859	9294	8506
0.400	3685	4996	8531	10734
0.700	10004	15458	22085	19683

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2-5 Provide a SAR description of how the wood grain orientation other than those parallel or perpendicular to the direction of an applied force is considered for stress-strain relationships used in a LS-DYNA analysis of impact limiters.

SAR Figure 2.6.7.5-8 and Drawing 790-210 suggest that some redwood blocks will be subject to forces neither parallel nor perpendicular to grain orientations. 10 CFR 71.7(a) requires complete and accurate information.

NAC Response

SAR Section 2.6.7.5.5 is revised to incorporate the following description of how the wood grain orientation is considered for LS-DYNA. The upper outer (side impact) ring of the impact limiter surrounding the cask consists of 24 equally spaced angular wedges of redwood separated by radial gussets fabricated from steel plates. Since a half-symmetry finite element model is used to represent the impact limiters, only 12 of the redwood wedges are modeled. Out of the 12 modeled redwood wedges, only 3 of the wedges are loaded during a 30-foot side drop. The first wedge of redwood is loaded in the parallel-to-grain direction. The second and third redwood wedges are loaded between the parallel-to-grain and the perpendicular-to-grain directions.

For the wedges of redwood where the impact force is applied between the parallel-to-grain and the perpendicular-to-grain directions, Hankinson's formula (Avallone, E. A., Baumeister III, T., *Marks' Standard Handbook for Mechanical Engineers*, 9th Edition, McGraw-Hill Book Company, New York, 1987, pages 6-127) is used to determine the strength properties of redwood as it varies with the orthotropic axes of the wood grain.

$$N = \frac{PQ}{P\sin^2\theta + Q\cos^2\theta}$$

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NAC Response to RAI 2-5 (Continued)

where:

N = compression strength of wood at an angle to the grain direction,
P = compression strength of wood parallel to the grain direction,
Q = compression strength of wood perpendicular to the grain direction,
θ = angle between the direction of loading and the wood grain direction.

Using Hankinson's formula, stress-strain curves are generated for the second and third wedges of redwood at 15 and 30 degrees from the wood grain direction, respectively. Therefore, in the LS-DYNA input files, unique stress-strain curves are applied to each redwood wedge, depending on the loading angle and strain rate.

The measured, calculated (predicted), and design basis accelerations for the quarter-scale model side drop test are summarized in the following table.

Description	Acceleration (g)		Design Basis Acceleration (g)
	Top Impact Limiter	**Bottom Impact Limiter**	
Side Drop Test Result	190	198	240
LS-DYNA Prediction, WITHOUT redwood grain direction adjustments	218	215	240
LS-DYNA Prediction, WITH redwood grain direction adjustments	193	210	240

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2-6 With respect to SAR Figure 2.6.7.5-3, "LS-DYNA Finite Element Model," describe how the weights associated with the shield and structural lids of the transportable storage canister (TSC) are considered in the finite element analysis.

The SAR should describe the basis for using the relatively thin brick elements to model a large mass lumping effect due to the cask top closures and TSC shield and structural lids. 10 CFR 71.7(a) requires complete and accurate information.

NAC Response

The weight and the CG of the finite element model correspond to that of the full-scale cask design. The cask contents weight, including the mass of the canister lids, is distributed in the brick elements along the length of the cask. In the full-scale cask design, the canister lids are positioned to be in contact with the cask body upper forging, which is a massive ring that is insensitive to the localized loading of the canister lids. For this reason, it is only required that the weight and CG of the finite element model accurately represent the full-scale cask design and the quarter-scale model. This description has been incorporated in Section 2.6.7.5.5 for the analysis of the full-scale impact limiters and in Section 2.10.3.7 for the analysis of the quarter-scale model impact limiters.

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2-7 Considering the stress-strain curves used in the LS-DYNA impact limiter analysis model, compute and plot explicitly the corresponding load-deflection curves for the top impact limiter subject to end, corner, and side deceleration g-loads, which are characteristic of quasi-static tests of impact limiter scale models.

Consistent with the staff practice of requiring quasi-static load testing of impact limiter scale models, such as those reported in SAR Section 2.10.3.3.4, load-deflection curves are needed to aid in evaluating numerical simulation of impact limiter performance. Complete and accurate information should be provided, per Section 71.1(a), for evaluating the package free-drop performance under Sections 71.71(c)(7) and 71.73(c)(1).

NAC Response

The load-deflection curves for the end, corner, and side drops are plotted in the Figures RAI 2.7-1 through RAI 2.7-4, which follow. The corresponding energy dissipated by the impact limiter was computed by determining the area under the force-deflection curve. The following table summarizes the peak load, maximum deflection, and the energy dissipated during quarter-scale drop events.

Description	Drop Configuration			
	End	Corner	Side (Top Limiter)	Side (Bottom Limiter)
Peak Load, lbf	850,777	687,759	376,209	406,470
Maximum Deflection, in.	2.68	4.98	3.20	3.31
Energy Dissipated, in-lb ($\times 10^6$)	1.48	1.46	0.73	0.73

The energy associated with the 30-foot drop for a scale model design weight of 4,063 lb is 1.46×10^6 in-lb. The table shows that the energy dissipated by the impact limiters matches the energy associated with the 30-foot drop.

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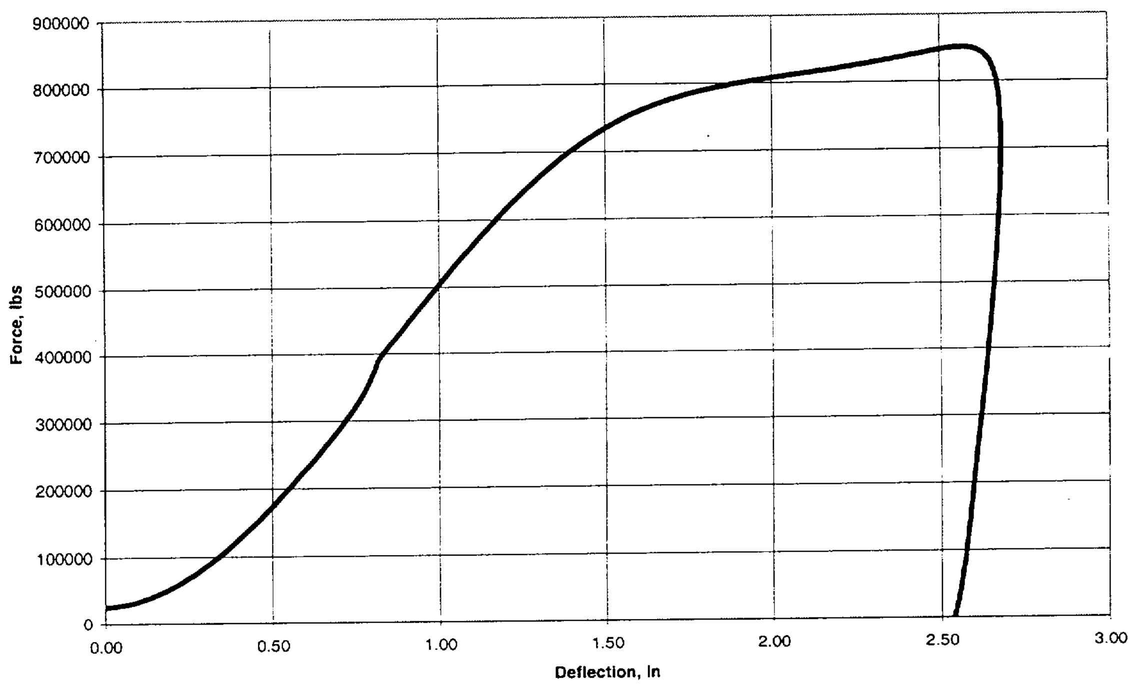
NAC Response to RAI 2-7 (Continued)

Figure RAI 2.7-1 Quarter-Scale Model Impact Limiter Load-Deflection Curve in the End Drop



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<u>NAC Response to RAI 2-7 (Continued)</u>

Figure RAI 2.7-2 Quarter-Scale Model Impact Limiter Load-Deflection Curve in the Corner Drop

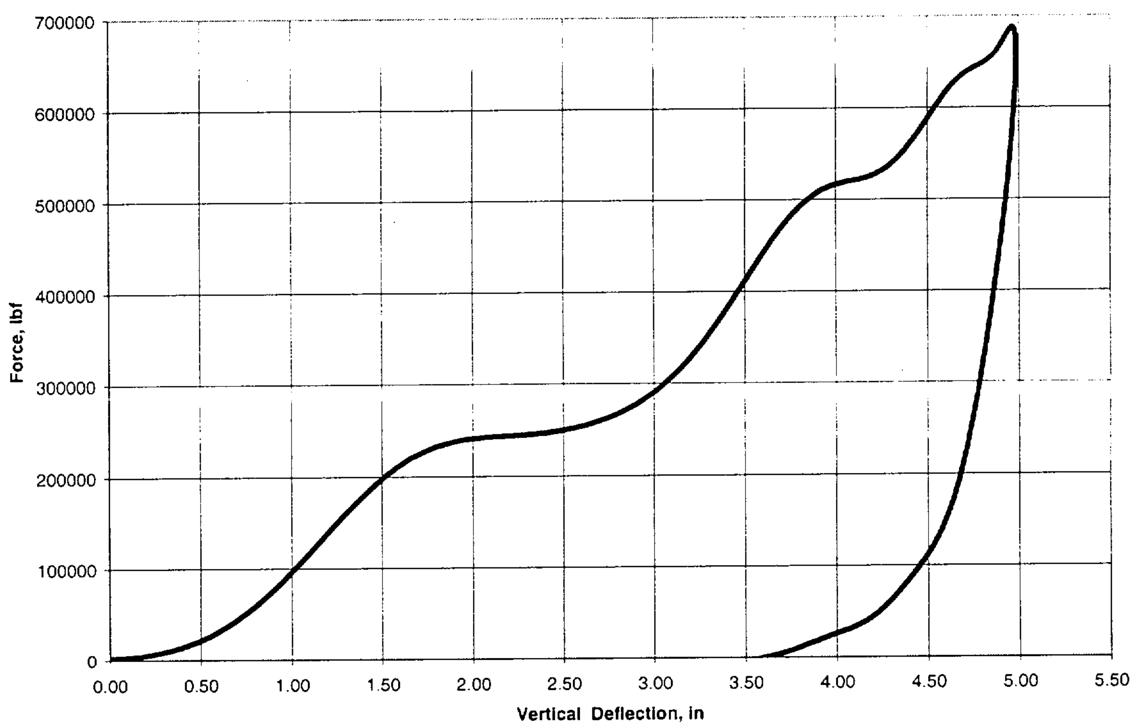

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Figure RAI 2.7-3 Quarter-Scale Model Impact Limiter Load-Deflection Curve in the Side Drop (Top Accelerometer)

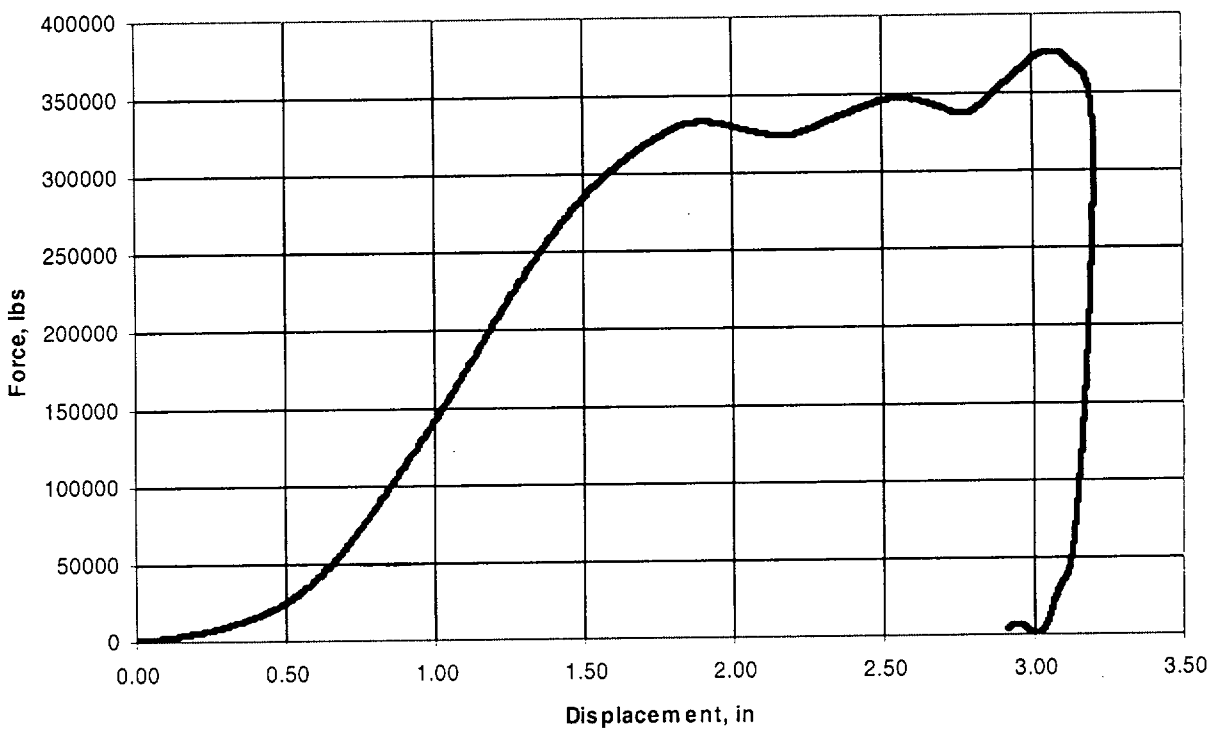

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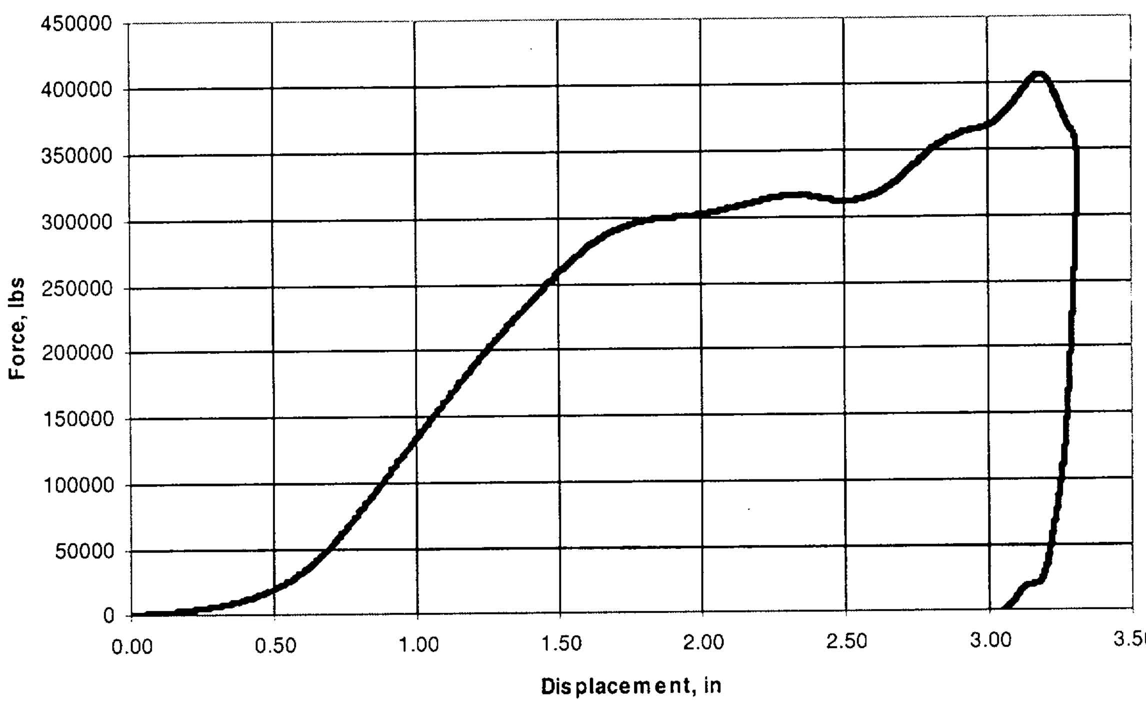
Figure RAI 2.7-4 Quarter-Scale Model Impact Limiter Load-Deflection Curve in the Side Drop (Bottom Accelerometer)



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2-8 Revise the SAR Section 2.6.7.5.8 description to acknowledge, as appropriate, that the NAC-STC quarter-scale drop tests are modeled for parametric studies of cask drop at shallow angles, including a consideration of friction forces between the impact limiters and unyielding surface, to demonstrate that an oblique drop test need not be performed for a NAC-UMS scale model.

The staff notes that the design features specific to the NAC-UMS cask/impact limiter may not have been addressed completely in a similar construction of the NAC-STC quarter-scale model. As such, the staff continues to follow the review practice of requiring the finite element analysis model be benchmarked by drop testing a NAC-UMS specific scale-model. The staff recognized that scale model tests need only be performed for limited drop orientations to provide reasonable assurance that free-drop tests, per 10 CFR 71.71(c)(7) and 71.73(c)(1), can be evaluated appropriately by numerical test simulations for which maximum damage is expected.

NAC Response

Section 2.6.7.5.8 is revised to expand the discussion of the parametric studies performed on the NAC-STC quarter-scale models. Included are descriptions of the effects of varying shallow drop angles and coefficient of friction on the analysis results. The LS-DYNA analysis methodology benchmarking is provided in the NAC proprietary information calculation package EA790-2235, Revision 1, submitted to the NRC on 2/28/2001. Also, the discussion of the NAC-UMS® quarter-scale model analyses and comparison to drop test results has been clarified to refer to Section 2.10.3.7 where the analysis and comparison are presented. The comparison of the NAC-UMS® quarter-scale model finite element analysis results and the quarter-scale side drop test results is also provided in NAC proprietary information calculation package EA790-2234, Revision 1, submitted on 2/28/2001 (NAC Letter ED20010281). Based on the close similarity of the NAC-STC and the NAC-UMS® cask and impact limiter designs, the parametric studies of the NAC-STC shallow angle cask drops and impact limiter friction forces for the NAC-STC bound those for the NAC-UMS®.

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2-9 With respect to SAR Figures 2.6.7.5-5, -6,and -7, provide an evaluation of effects of vibratory response components on the maximum decelerations as presented in Table 2.6.7.5-6.

The staff recognizes the need for low-pass filtering both the test and calculated results for removing spurious, high frequency, response components which are inherent to the numerical modeling but non-consequential for practical cask design consideration. The staff notes that rigid body response components are generally considered for benchmarking a cask/impact limiter finite element analysis model. However, to determine peak cask decelerations by either testing or analysis, contributions from vibratory response components, which would be associated with predominant cask vibration modes of interest, should also be properly considered. Complete and accurate information should be provided, per Section 71.1(a), for evaluating the package free-drop performance under Sections 71.71(c)(7) and 71.73(c)(1).

NAC Response

To determine the predominant cask vibration modes of interest, a modal analysis of the basket support disks was performed using the ANSYS program that calculates the frequencies for the side drop for the BWR and PWR baskets for the end drops and side drops. To determine the dynamic load factor (DLF), the side drop and end drop are treated as single degree of freedom systems with triangular pulse shape and no damping. Figure RAI 2.9-1 plots the DLF verses the mode frequency times the drop duration for a triangular pulse shape. The following table summarizes the DLF for each orientation.

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	Basket Angle	**Mode (Hz)**	**Drop Duration (sec)**	**Mode × Duration**	**DLF**
PWR	Side 0°	192.5	0.046	8.9	1.05
PWR	Side 45°	182.1	0.046	8.2	1.00
PWR	End	46.8	0.047	2.2	1.00
BWR	End	51.7	0.047	2.4	1.02
BWR	Side 0°	261.0	0.046	12.0	1.00
BWR	Side 45°	56.9	0.046	2.6	1.09

The following table applies the DLF to the peak accelerations for the side, end and corner drops. Because of the long duration of the corner drop, the DLF for the corner is bounded by the DLF calculated for the end drop.

Drop Orientation and Conditions	**Peak Acceleration (g)**	**Peak Acceleration × Maximum DLF (g)**
Side Drop-Cold	52.0	56.7
Top End Drop-Cold	57.8	59.0
Corner Drop (24°)-Cold	36.5	37.2

In all cases, the factored accelerations that include the effects of the cask vibratory response are less than the design accelerations employed in the evaluation of the UMS® cask.

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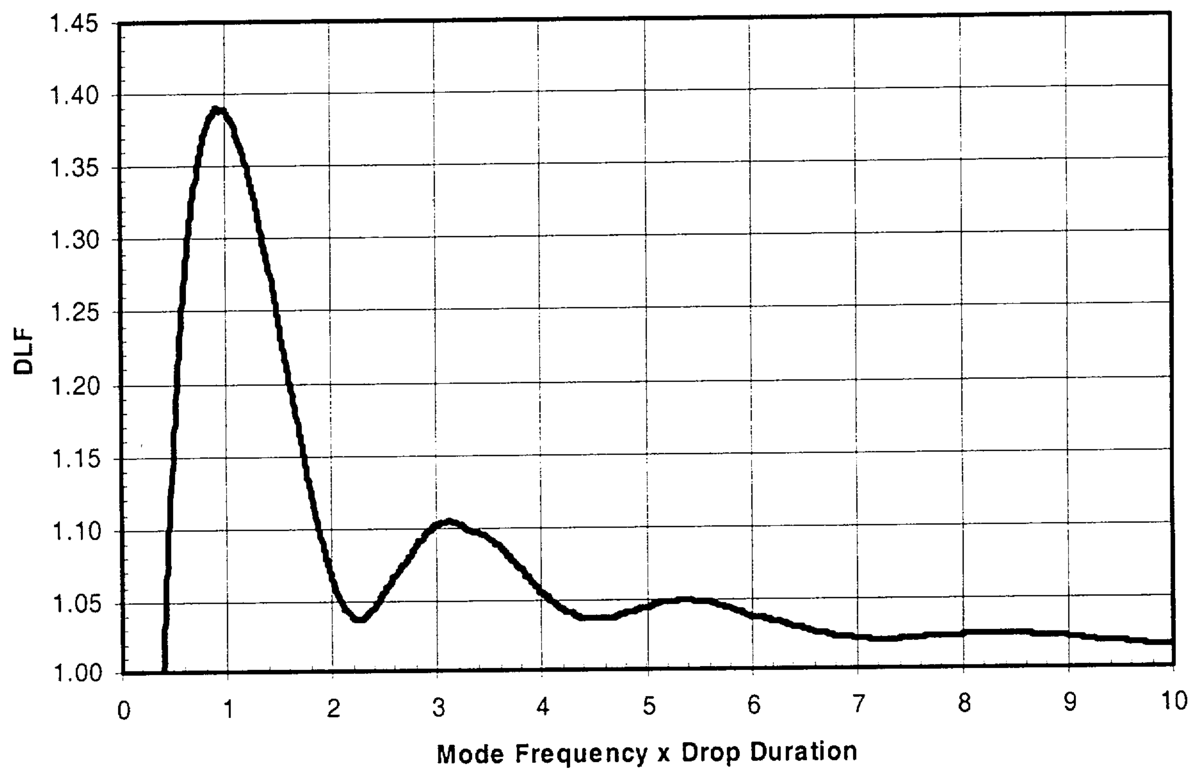
Figure RAI 2.9-1 Dynamic Load Factor for Single Degree of Freedom System – Triangular Pulse Shape, No Damping



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2-10 Clarify the SAR statement on Page 2.10.3-26, "The following table compares the L/r of the UMS and the NAC-STC casks."

The cask length-over-radius-of-gyration data is not reported as stated. 10 CFR 71.7(a) requires complete and accurate information.

NAC Response

Section 2.10.3.5 is revised to include the comparison of the L/r values for the NAC-UMS® Universal Transport cask and the NAC-STC cask. These values were inadvertently omitted from the text.

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2-11 With respect to Figures 2.10.3-8 and –9, explain why the LS-DYNA calculated and low-pass filtered deceleration time histories do not contain similar vibratory response components to those recorded for the side drop test.

The staff notes that both the analysis and test raw data were reduced with the same low-pass filtering parameters. As such, the analysis results are expected to display also the vibratory response components superimposed on the rigid body cask responses. Complete and accurate information should be provided, per Section 71.1(a), for evaluating the package free-drop performance under Sections 71.71(c)(7) and 71.73(c)(1).

NAC Response

The vibratory response noted in the test data is a direct result of the sampling rate at which the data was recorded. The sampling rate used by Sandia National Laboratories during the quarter-scale model side drop test was approximately 250,000 Hz, which results in a data point being recorded every 1/250,000 seconds and reduces the effectiveness of the Butterworth filter. Typical LS-DYNA analyses are performed with sampling rates from 50,000 to 100,000 Hz. For the purposes of this evaluation, the data rate in LS-DYNA was increased to 250,000 Hz. Figures RAI 2.11-1 and RAI 2.11-2 provide the comparison of the analysis and test data with the 250,000 Hz sampling rate. Both curves are filtered at 450 Hz. Figures 2.10.3-8 and –9 have been updated with the new results. The updated curves show similar frequency content in terms of the oscillatory behavior.

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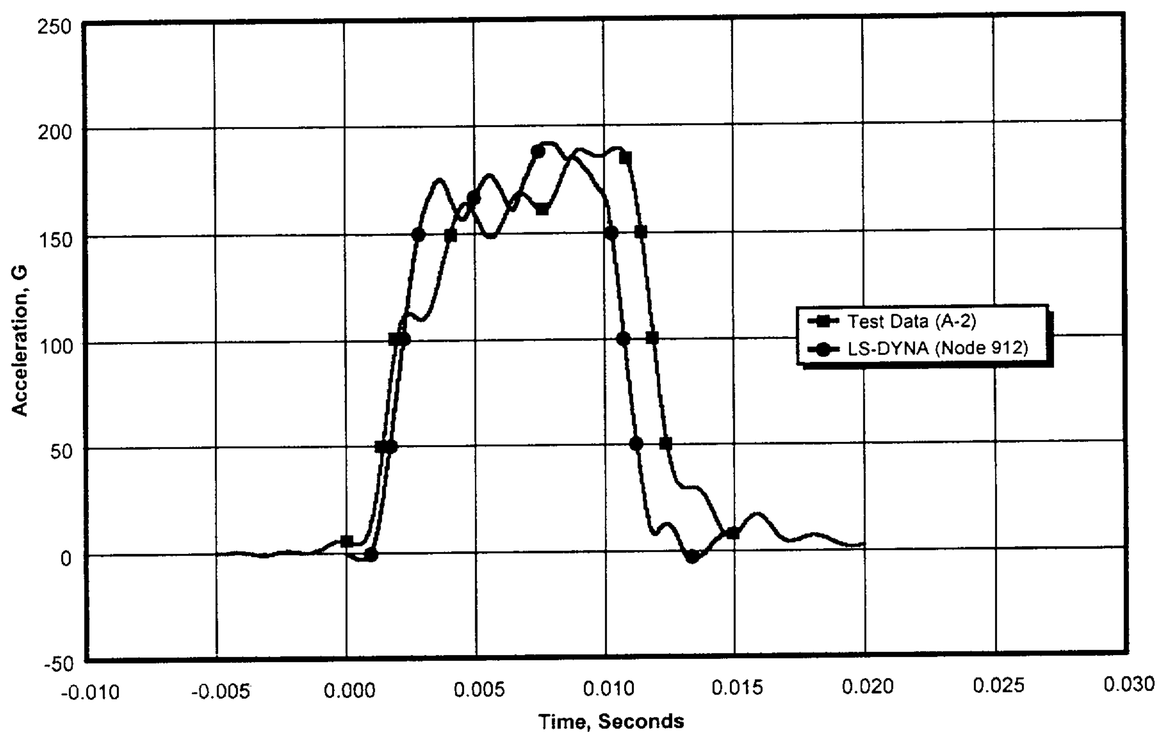
NAC Response to RAI 2-11 (Continued)

Figure RAI 2.11-1 Comparison of Quarter-Scale Model Side Drop (LS-DYNA and Drop Test) Results (Upper Accelerometer)



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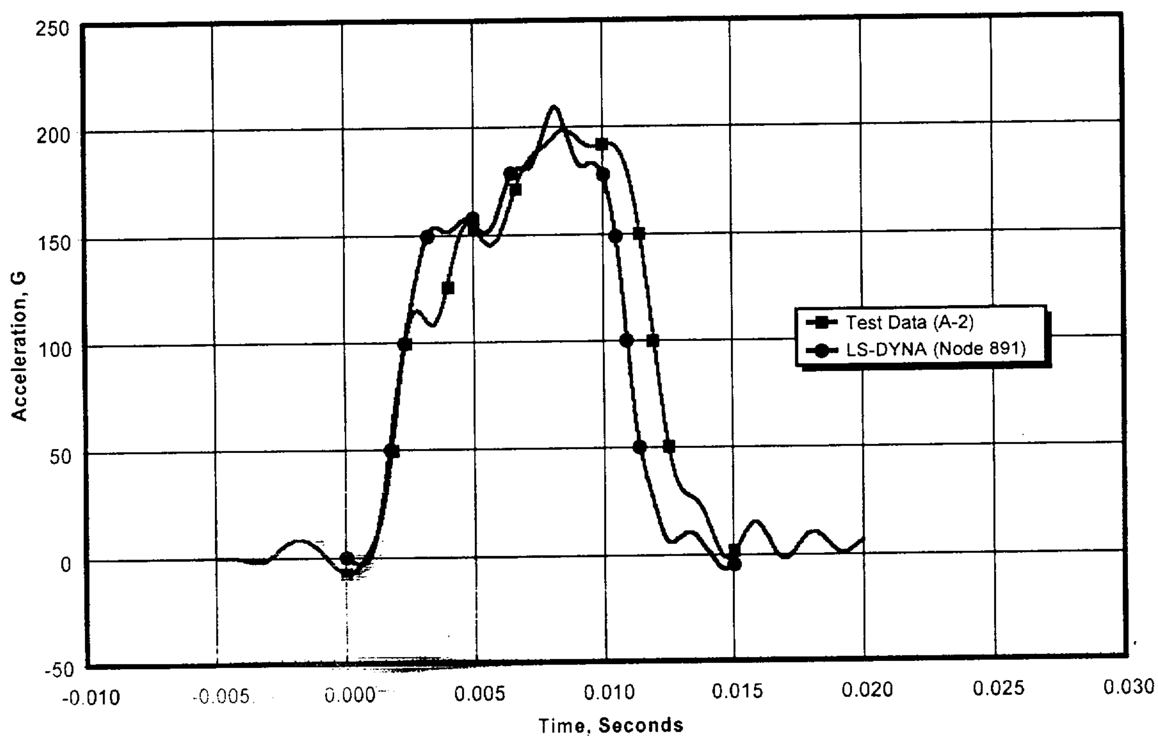
NAC Response to RAI 2-11 (Continued)

Figure RAI 2.11-2 Comparison of Quarter-Scale Model Side Drop (LS-DYNA and Drop Test) Results (Lower Accelerometer)



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CHAPTER 3: THERMAL EVALUATION

3-1 Provide the value used for full solar insolation on the cask surface, and provide justification for the emissivity value of 0.36 and how it will be maintained. State the resulting temperature of the neutron shield material and justify its ability to perform its function if it exceeds its safe operating range during normal conditions of transport. Provide the shielding performance capability at the expected temperature range. Provide assurance that the gap between the end of the copper plate section of the heat transfer fin and the neutron shield shell can be fabricated to be no greater than 0.031 inch and will be no greater than 0.031 inch under Normal Conditions of Transport.

The first paragraph in response to RAI 3-9 states that full solar insolation of 2.3 kW is considered on the cask surface. Page 3.4-6 of the SAR states a value of 1475 Btu/12hr-ft^2. Also, see Figure 3.4-10. The heat energy flux for curved surfaces to be applied as a 12-hour time step function corresponding to the Insolation Data of 71.71(c)(1) is 400W/m^2 or ~1522 Btu/12 hr-ft^2, or 126.85 Btu/hr-ft^2 (0.8809 Btu/hr-in^2). This value applied uniformly over a 24-hour period instead of as a step function results in an energy flux of 63.423 Btu/hr-ft^2 (0.4404 Btu/hr-in^2).

NAC Response

The total solar insolation applied to the exterior surface of the cask per day under Normal Conditions of Transport is 2.33 kW, which is calculated based on the heat flux required by 10 CFR 71.71(a)(1) to be applied over a 12-hour period evaluated in the steady state condition (applied over 24 hours simulating a 12-hour period of solar exposure and a 12-hour period of no solar exposure). Per 71.71(c), insolation for a curved surface is 400 g cal/cm^2 for 12 hours. This converts to the units of BTU/hr-ft^2 by:

$$400 \frac{\text{g cal}}{12 \ \text{hr} \cdot \text{cm}^2} \times \frac{1 \text{Kcal}}{1000 \text{g cal}} \times 1.163 \frac{\text{w} \cdot \text{hr}}{\text{Kcal}} \cdot (2.54 \cdot 12)^2 \frac{\text{cm}^2}{\text{ft}^2} \cdot 3.412 \frac{\text{BTU/hr}}{\text{w}} = 1475 \frac{\text{BTU}}{12 \text{ hr} - \text{ft}^2}$$

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NAC Response to RAI 3-1 (Continued)

The heat flux resulting from insolation on a curved surface is:

$$1475 \frac{\text{BTU}}{12\ \text{hr}-\text{ft}^2} \text{ x } \frac{12\ \text{hr}}{24\ \text{hr}} \text{ x } \frac{1\ \text{ft}^2}{144\ \text{in}^2} = 0.427\ \text{Btu}/\text{hr}-\text{in}^2.$$

These values of solar insolation are consistent with those used in the NRC previously approved transport cask safety analysis reports for NAC and other vendors' casks.

Multiplying this value by the emissivity of the exterior surface of the cask, $\varepsilon = 0.36$, results in a heat flux of 0.154 Btu/hr-in^2. The emissivity of 0.36 for the cask surface is used as the absorptivity of the cask surface.

The total solar insolation applied to the cask surface under Normal Conditions of Transport is:

$$0.154 \frac{\text{Btu}}{\text{Hr}-\text{in}^2} \times 3.14 \times (92.11\ \text{in}) \times 178.56\text{in} \times \frac{1\ \text{kW}}{3412.3\ \text{Btu/Hr}} = 2.33\ \text{kW}$$

where 92.11 in. is the cask surface OD and 178.56 in. is the height of the neutron shield shell.

The emissivity of 0.36 (presented in Table 3.2-2) for stainless steel is from Graph-Part I, Group I-High Alloy Steels, Page A62, Nuclear Systems Materials Handbook. No additional procedures, such as polishing, are required to maintain this value of emissivity over time.

The maximum temperature of the radial neutron shield is 293°F (presented in Table 3.4-1), which is lower than the allowable temperature of 300°F of NS-4-FR; therefore, the neutron shield will maintain functionality.

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NAC Response to RAI 3-1 (Continued)

The end of the copper plate section of the heat transfer fin is in contact with the outer shell and the neutron shield shell when the stainless steel section of the fin is welded to the outer shell and to the neutron shield shell during cask fabrication. The assumption of a uniform gap (1/32-inch) between the end of the copper plate section of the heat transfer fin and the neutron shield shell provides a conservative heat transfer evaluation of the transfer fin configuration, since the copper plate section of the fin is not physically attached to the shells.

The structural integrity of the attachment of the bonded 8-mm thick stainless steel/6-mm thick copper plate heat transfer fin to the neutron shield shell and to the outer shell is assured by welding the stainless steel portion of the heat transfer fin to the shells. The relative positions of the heat transfer fin, the neutron shield shell and the outer shell are fixed.

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CHAPTER 3: THERMAL EVALUATION

3-2 Provide ANSYS thermal analysis input files and justify assumptions made and used in the thermal analysis.

The staff must have assurance that the code has been appropriately used and that assumptions are justified by Section 3.5.7.1 of the SRP.

NAC Response

The ANSYS input files for thermal analyses of the UMS® Transport Cask for Hypothetical Fire Accident and Normal Transport conditions are included on the enclosed CD. There are 64 files on the CD (they are not provided in text form due to their size).

The principal assumptions considered in the thermal models for the analysis for the Normal Conditions of Transport and for the Hypothetical Fire Accident, together with the justification, are:

For Normal Conditions of Transport

1. The impact limiters are not directly included in the model. The insulating effect of the limiters is considered by applying adiabatic boundary conditions to relevant surfaces of the cask body model.
   Justification: It is conservative to apply the adiabatic boundary conditions to the relevant surfaces of the cask body, since it does not allow any heat transfer out of the cask body at those surfaces.

2. Convective heat transfer inside the canister and the cask cavity is not considered.
   Justification: It is conservative to neglect the heat transferred by convection inside the canister and the cask cavity, since only minor convection occurs within the canister. Any convection acts to reduce temperatures in the canister.

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NAC Response to RAI 3-2 (Continued)

3. Since the transport cask is in the horizontal position during transport, the canister shell is considered to be in contact with the cask inner shell and the basket disks are considered to be in contact with the canister shell. A 4-degree angle of contact is considered at both these contact regions.
   Justification: A small angle of contact between the canister shell and the cask inner shell and between the basket disks and the canister shell is expected, since the relatively-thin canister shell will flex elastically to conform to the diameter of the cask inner shell in the horizontal position during transport. A 4-degree angle of contact is equivalent to less than 2.5 inches of contact circumferentially.

For the Hypothetical Fire Accident

1. The thickness of the fireblock insulation is modeled as 0.03 inch (0.125 inch on drawing).
   Justification: It is conservative to use a thinner layer of fireblock in the finite element model, since the thinner layer of fireblock will allow more heat transfer into the cask.

2. The gap between the cask inner shell and the lead is not modeled.
   Justification: The gap size between the inner shell and the lead is small (0.015 inch). The thermal model for the fire accident assumes a full contact between the inner shell and the lead. This assumption is conservative, since it represents a more effective heat transfer path for the heat to transfer into the system during the fire.

3. The NS-4-FR in the neutron shield is considered to be in place before and during the fire. After the fire, the NS-4-FR is replaced with air.
   Justification: The NS-4-FR has an allowable temperature of 300°F, and the surface temperature of the neutron shield reaches 1376°F during the fire accident. This

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NAC Response to RAI 3-2 (Continued)

implies that a significant portion of the NS-4-FR might have become charred and the equivalent conductivity is reduced. Retaining NS-4-FR during the fire allows more heat to go into the cask and is, therefore, conservative. After the fire, the NS-4-FR is replaced with air. This is also conservative, since the air corresponds to additional resistance to the heat being transferred out of the cask and will result in higher cask component temperatures for the post-fire period.

4. A heat transfer coefficient of 0.01222 $\Delta T^{1/3}$ Btu/hr-in.^{2}-°F is considered at the cask surface during the fire.
   Justification: A heat transfer coefficient of 0.01222 Btu/hr-in.^{2}-°F is recommended by the reference: S.D. Wix, Proceedings Volume 2, The 11th International Conference on the Packaging and Transportation of Radioactive Materials (PATRAM'95), pp 672-678 (Convective Effects in a Regulatory and Proposed Fire Model), December, 1995. The heat transfer coefficient of 0.01222 $\Delta T^{1/3}$ Btu/hr-in.^{2}-°F is used in the model, where ΔT is the difference between the fire temperature and the cask surface temperature. Since this heat transfer coefficient value is higher than the recommended value of 0.01222 Btu/hr-in.^{2}-°F, more heat enters the cask during the fire and the resulting analysis is conservative.

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CHAPTER 3: THERMAL EVALUATION

3-3 Clarify whether only contents which have been stored in the UMS Storage system will be transported in the UMS Transport.

The response to RAI 1-11 suggests that only contents which have been stored in the UMS Storage System will be loaded for transport in the UMS transportation cask.

NAC Response

The requirement for transport in the UMS® Transport Cask is that the contents, including site specific fuel, be loaded in a sealed Transportable Storage Canister and meet the specifications of the cask Certificate of Compliance (CoC) for approved contents. Fuel cannot be directly loaded into the transport cask, nor can it be loaded into an otherwise empty canister that is installed in the transport cask. The intent of the response to RAI 1-11 was to state that fuel loaded in a canister for storage must first meet all content specifications for storage. When transported, the fuel (either from storage or loaded for immediate transport) must be in a sealed canister and must meet the CoC limits for transport.

Typically, the canister will be loaded, sealed and moved to a vertical concrete cask for storage for an unspecified time period prior to transportation. It is not technically necessary for a loaded canister to have been actually "stored" in a concrete cask. That is, the canister can be loaded and sealed (while it is in the transfer cask) and then the sealed canister can be loaded in the transport cask. In this case, the fuel in the canister must meet any preferential loading requirements for the contents and must also meet the fuel enrichment, decay heat, cool time and other limits (such as total decay heat) established for transport.

Additional text is added to Section 3.1 to state that the fuel is already in a sealed canister.

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3-4 Describe the method that will be used to determine when a canister can be removed from the storage overpack and placed into the transport cask.

The decommissioned Maine Yankee contents which are stored in the UMS Storage System are the basis for analysis and assumptions (including pool cooling period) which determine the loading and cladding temperature limits in the transportation application. A description of the methodology and factors which will be used to determine when stored contents will meet the requirements of transport will also help the staff to understand compliance in the transition from storage to transport activities. Section 71.33 requires a description in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation.

<u>NAC Response</u>

The determination of acceptability for transport is made on a canister basis for total heat load and on a per assembly basis for cool time. Total canister decay heat is limited to 20 kW for PWR fuel and to 16 kW for BWR fuel.

Each Maine Yankee fuel assembly within a canister must be separately evaluated to verify that its cool time, based on its enrichment and burnup, meets the time limits shown in SAR Tables 5.5.1.1-10 or 5.5.1.1-12, as appropriate to its configuration. When each fuel assembly within a canister meets the required cool time limit(s) and the total canister decay heat meets the allowable limit for the fuel type, the canister may be transported.

The equivalent loading tables for other fuel types are shown in Tables 1.2-6 and 1.2-7 for PWR and BWR fuel, respectively.

The cool times shown in the loading tables for fuel assembly transport take into account the allowable cladding temperature and stress established for the storage condition.

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3-5 Provide the referenced information on the Fiberfrax® Ceramic Paper.

The referenced information should provide sufficient basis for evaluation of the package per Section 71.33.

NAC Response

The product specifications for Fiberfrax® Ceramic Fiber Paper are provided in Attachment A.

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CHAPTER 3: THERMAL EVALUATION

3-6 Provide a justification that the neutron shield can be replaced by air for HAC fire.

The analysis in response to RAI 3-8 indicated a reduction of temperatures in response to removal of the neutron shield. This must be based on the assumption that the neutron shield is replaced by an air gap, which has not been justified. Section 71.41 requires a demonstration of the effects on the package under the tests specified in Section 71.73, and that the requirements of 71.51(a)(2) would then be met.

NAC Response

The original fire transient analysis for the cask was performed based on the assumptions that the radial neutron shield material (NS-4-FR) was in place during the 30-minute fire accident event and was removed immediately after the fire. For NAC's response to RAI 3-8 (UMS® Transport RAI-1), an additional fire accident analysis was performed considering that the NS-4-FR neutron shield material was replaced with air. The stainless steel/copper fins remain in place. The results of the second analysis indicated that the original analysis is the limiting condition, i.e., it maximized the heat input to the cask.

The original analysis and the second analysis (for response to RAI 3-8) represent the two bounding conditions (maximum and minimum conductivity for the neutron shield region, i.e. NS-4-FR or air) for the hypothetical fire accident event. Therefore, no further analysis is required.

Refer to the NAC Response to RAI 3-2, For the Hypothetical Fire Accident, Item 3.

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CHAPTER 3: THERMAL EVALUATION

3-7 Provide information or an alternative operation or design so that components important to safety, including the support disks will not exceed the allowable temperature when air is the canister gas. Include PWR and BWR columns on Table 3.4-3 with the component temperatures when air is the gas present.

The response to RAI 3-5 indicates that the support disks will exceed the allowable temperature when air is the canister gas. Staff must have assurance that the package materials will not exceed specified allowable limits under normal and accident conditions of transport, consistent with the tests specified in Sections 71.71 and 71.73.

NAC Response

The design basis for the UMS® Universal Transport system is that the canister is backfilled with helium. As shown in Chapter 2, there are no design basis normal conditions of transport or accident events that result in loss of containment or loss of the helium atmosphere within the welded closed canister. The canister is leaktight; therefore, the air case is not required, but it was provided for information only based on previous transport cask licensing history. The thermal analysis results associated with air inside the canister are deleted from Chapters 2 and 3.

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CHAPTER 3: THERMAL EVALUATION

3-8 Provide justification for the film coefficient that was chosen and found appropriate for the conditions. Include specific reference page numbers and include assumptions leading to the conclusion.

The heat transfer coefficient is provided in Section 3.2.3. The Standard Handbook for Mechanical Engineers is referenced, but no specific page number is identified, and the reasons that these expressions were chosen have not been explained. Section 3.5.3.1 of the SRP requires verification of the coefficient.

NAC Response

The convection film coefficient applied to the cask surface for normal conditions of transport is obtained from Page 4-88, Equation 4.4.12d of Reference 16 (Baumeister T. and Mark, L.S., Mark's Standard Handbook for Mechanical Engineers, 9th Edition, New York, McGraw-Hill Book Co., 1987). The equation is:

$$h_c = 0.19\,\Delta T^{1/3}\ \text{Btu/hr-ft}^2\text{-°F}$$

where:

$\Delta T =$ temperature difference between the cask surface and the air, °F

This equation is specified for use in modeling natural convection on the surface of a horizontal cylinder with a diameter = D and $D^3\Delta T > 100$, and for air which is at room temperature and atmospheric pressure and is subjected to the gravitational attraction at sea level.

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For the UMS® transport cask, the cask diameter (D) is 7.667 ft and ΔT is greater than 100°F. Consequently, the value of $D^3 \Delta T$ is greater than 45,000, which is significantly larger than 100. Therefore, the equation is appropriate for the application.

Based on the cask surface area in inches, the equation is converted to:

$$h_c = 0.00132\, \Delta T^{1/3} \text{ Btu/hr-in}^2\text{-°F}$$

which is the equation used in SAR Section 3.2.3.

SAR Section 3.2.3 is revised to incorporate the page and equation number from the reference. SAR Section 3.7 is revised to refer to the 9th Edition of the Standard Handbook for Mechanical Engineers.

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3-9 Provide a sensitivity study that considers the uncertainties in the thermal and structural properties of materials and analysis.

Section 3.5.3.4 of the SRP requires that the thermal evaluation appropriately address uncertainties.

NAC Response

Two thermal analyses were performed to evaluate the effect of variations in emissivity and convection heat transfer coefficient on the thermal analysis of the transport cask. The analysis for the normal conditions of transport for PWR fuel configurations is used as the base case (see Section 3.4.1.1.1 for the model description). The first analysis considers a 10% reduction of the emissivity of the transport cask inner and outer shells, the canister shell, and the basket (including support disks, heat transfer disks and fuel tubes). The second analysis considers a 10% reduction in the convection heat transfer coefficient at the transport cask outer surfaces.

A summary of maximum temperatures for the governing components for the base case and for the two sensitivity study cases is shown in the following table. The increase in the maximum fuel cladding and basket temperature is ≤ 6°F for both the reduced emissivity and reduced convection coefficient cases. All component temperatures remain well below their allowable temperatures.

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	Max. Temperature, °F			**Allowable**
	Base Case	$\Delta\varepsilon$ **= -10%**	Δh_c **= -10%**	**Temperature (°F)**
Fuel Cladding	673	678	678	716
Heat Transfer Disk	605	610	610	700
Support Disk	608	613	614	650

The sensitivity study results indicate that the effect on fuel and basket temperature results due to the variations in emissivity and convection coefficient is not significant. The results of the sensitivity analysis are also incorporated in Section 3.4.1.1.1 (Three-Dimensional Cask Model: Cask with PWR Fuel Canister).

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3-10 Provide corrected maximum allowable temperatures for PWR high burnup fuel as indicated in Table 3.4.9 to be consistent with the maximum allowable temperatures in the Safety Analysis Report for the NAC-UMS Storage cask (i.e., the temperatures in the third column of Table 4.5.1.2-2).

Section 71.1 requires that the SAR contain complete and accurate information. As noted in the response to RAI#1 Question 3-2, "The allowable fuel cladding temperature limit for normal conditions of transport is revised to the long term dry fuel storage temperature limit." However, the maximum allowable temperatures for PWR high burnup fuel as indicated in Table 3.4.9 are higher than the maximum allowable temperatures in the Safety Analysis Report for the NAC-UMS Storage cask.

<u>NAC Response</u>

The maximum allowable temperatures for PWR and BWR high burnup fuel are revised based on the increase in BWR rod backfill pressure (RAI 3-16), and to correct for inconsistencies in rod free volumes listed in Tables 3.4-6 and 3.4-9 and used in the canister pressure evaluation. The revised maximum allowable cladding temperatures for PWR and BWR fuel are presented in Table 3.4-15.

The key input into fuel rod cladding stress evaluation is the rod's free volume. The rod's free volume is based on the plenum length, spring volume, pellet to cladding gap, and active fuel length. Minimum rod diameter, maximum cladding thickness and maximum pellet diameter determine the minimum rod backfill volume. While the maximum cladding thickness contributes to the minimum free volume condition for fission gases to accumulate, the thicker cladding also reduces stress levels in the cladding. Evaluating both maximum and minimum cladding thicknesses indicates that the use of the minimum thickness cladding is bounding. Both fuel plenum length and spring volumes are adjusted in the revised cladding allowable temperature calculation for a number of assembly types. Not all of the previously listed active

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fuel and plenum lengths are consistent with fuel rod lengths obtained from the same reference documentation. For example, DOE/RW-0184 lists the B&W 15x15 fuel rod at a maximum active fuel length of 144 inches with an 11.72-inch plenum, and a maximum rod length of 153.68 inches. This data set is clearly not consistent, since the sum of the active fuel and plenum lengths exceeds the total rod length. Plenum length is, therefore, recalculated based on the active fuel and rod length, with adjustments for rod end-caps. This modification reduces the plenum length for the bounding fuel assembly types slightly and, thereby, decreases fuel assembly allowable cladding temperatures. Reference information for the BWR plenum springs indicates a range of spring weights for similar BWR 8x8 rods (EPRI NP-563). As such, the fuel clad allowable temperature calculation is revised to use the maximum spring weight, which produces the minimum fuel free volume (canister pressure calculation uses the minimum spring weight to maximize the pressurized free volume in the fuel rods).

The maximum allowed decay heat load for the cask is not affected by this change, since a 5% temperature margin was initially applied to the maximum allowable fuel cladding temperature for conservatism prior to calculating the allowed heat load. This 5% margin is reduced to account for the allowable cladding temperature decrease (associated with the increased fuel rod stresses) without a change in the allowed heat load. The PWR and BWR fuel allowed heat load evaluations maintain margins ranging from 9°C to 15°C and 16°C to 18°C, respectively.

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3-11 Clarify whether the MNOP analysis includes the contribution of helium from actinide decay. If not, revise the MNOP analysis accordingly.

Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

The MNOP analysis did not explicitly account for long-term actinide generated helium or gases (helium) released from BPRAs; instead the analysis applied a fixed gas generation factor of 0.3125 atoms per fission to the fuel material. In response to this RAI and RAIs 3-12, 3-14 and 3-16, the pressure evaluations for both normal and accident conditions have been revised to account for:

- Actinide Decay Produced Helium
  SAS2H isotopics are generated to a cool time of 40 years for fuel burnups ranging up to 60,000 MWD/MTU (conservative) and initial enrichments as low as 1.9 wt % ^{235}U. Xenon, iodine, krypton, tritium (^{3}H), and helium concentrations are extracted from the SAS2H output and summed to form the fuel gas inventory. The helium concentration includes the gas generated from actinide decay. Maximum gas inventories are found at the maximum burnup, lowest enrichment, and longest cool time (i.e., maximum actinide production and decay). For conservatism, a combination of the 60,000 MWD/MTU burnup, 1.9 wt % ^{235}U, 40-year cooled case is employed in the system pressure analysis. Gas inventories for the 1.9 wt % ^{235}U case are 2 to 3% higher than those produced with enrichments around 4 wt. % ^{235}U, which is actually required to reach burnups as high as 60,000 MWD/MTU.

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NAC Response to RAI 3-11 (Continued)

- UMS® Canister Class Specific Pressures

Each of the fuel assembly types expected to be loaded into the UMS® canister/cask is evaluated for maximum normal and accident condition pressures. The results of these evaluations indicate that the B&W 17x17 assemblies produce the maximum PWR canister pressure under normal conditions. Maximum transport cask pressure under normal conditions is produced by the Westinghouse 17x17 standard assembly. Maximum fire accident and 100% fuel failure pressures are produced by the B&W 17x17 fuel assembly. The difference between the canister and the cask pressures produced by B&W 17x17, B&W 15x15, and Westinghouse 17x17 assemblies under normal conditions is less than 0.5 psi and is less than 5 psi under accident conditions. Westinghouse 17x17 and B&W 17x17 fuel assembly-produced canister pressures are less than 1 psi different under accident conditions. The accident condition pressure difference in the cask for the two bounding 17x17 assembly types is approximately 4 psi, due to the larger backfill volume in the cask with a Class 1 canister. GE 7x7 (49 fuel rod) and fuel assemblies produce the bounding BWR system pressures. GE 7x7 fuel assembly canister and cask pressures are bounding, but are similar to those produced by GE 8x8 (63 fuel rod) and GE 9x9 (79 fuel rod) fuel assemblies. Similar fissile material masses in these assemblies produce similar quantities of fission gas, which accounts for approximately 85% of the BWR fuel rod releasable gas inventory at 60,000 MWD/MTU burnup. The difference between GE 7x7, 8x8 and 9x9 payload pressures in the transport cask is less than 0.2 psi at normal operating conditions and less than 3 psi at accident conditions.

- Inclusion of Helium Produced in BPRAs

Pressure calculations are revised to include helium produced by the (n,α) reaction in boron-containing burnable poison rod assemblies and shim rods (CE cores). Due to low (n,α) cross-sections, burnable poison rods containing Erbium or Gadolinium do not produce a significant quantity of helium as the result of neutron absorption. Burnable poison rods typically contain boron in B_4C or B_2O_3 form encased in a solid matrix material, such as borosilicate glass or AlO_2, and are expected to retain a significant portion of the helium

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generated by the neutron absorption within the matrix material. For the pressure analysis, 100% of the modeled boron content is considered to be converted into lithium, and helium with 100% of the generated helium being available for release. Normal conditions pressure calculations include 3% BPRA rod failure and accident conditions consider 100% rod failure.

- Increased BWR Backfill Pressure

  DOE/RW-0184 reports the maximum BWR fuel rod backfill pressure as 60 psig. PNL-4835 indicates a slightly higher BWR fuel rod backfill pressure as 6 atmospheres (88 psig) for the GE 8x8 fuel assembly type. For conservatism, and in direct response to RAI 3-16, a 132 psig backfill pressure is applied to all BWR fuel assembly types in the revised canister and cask pressure evaluation.

  A backfill pressure of 132 psig is also conservatively applied to BWR 8x8 and 9x9 fuel assembly types during the maximum allowable fuel cladding temperature calculations. It is not realistic to apply the backfill pressure of 132 psig to 7x7 BWR fuel assemblies at high burnups; therefore, a backfill pressure of 3 atmospheres (44.1 psig [ORNL/TM-9591]) is applied to the 7x7 assemblies for cladding allowable temperature calculations.

  The calculated maximum pressures for the canister and the cask are summarized in Tables 3.4-7 and 3.4-10 for normal conditions of transport and in Table 3.5-3 for the accident conditions.

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CHAPTER 3: THERMAL EVALUATION

3-12 Provide justification for using the Westinghouse 17x17 fuel assembly as the design basis fuel assembly for the internal pressure calculations. The staff's confirmatory calculations have identified the B&W 15x15 as the bounding fuel assembly for the internal pressure calculations. Provide calculations comparing the two assemblies.

Section 71.1(a) requires complete and accurate information be submitted in the application.

<u>NAC Response</u>

The relevant calculated molar quantities, free volumes, and pressures for the Westinghouse and B&W 15x15 and 17x17 assemblies are:

Fuel Type	WE15x15 (Std)	WE17x17 (Std)	B&W15x15 (Mark B)	B&W17x17 (Mark C)
UMS® Canister Class	1	1	2	2
Rod Backfill (moles)	152	134	149	169
Releasable Fuel Gas (moles)	288	290	298	289
BPRA Gas (moles)	111	133	111	132
Can Backfill (moles)	182	184	194	195
Cask Backfill (moles)	45	45	25	25
Canister Free Volume (liters)	5880	5930	6240	6280
Canister Normal Pressure (psig)	6.1	6.1	6.1	6.2
Canister Accident Pressure (psig)	73.9	74.2	71.0	74.3
Cask + Canister Free Volume (liters)	6950	7000	6830	6860
Cask Normal Pressure (psig)	6.9	6.9	6.6	6.6
Cask Accident Pressure (psig)	64.9	65.2	66.3	69.3

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NAC Response to RAI 3-12 (Continued)

As discussed in the response to RAI 3-11, assembly and canister class specific pressure calculations are performed for the different fuel types designated for UMS® loading. Primarily due to the larger amount of burnable poison modeled, the maximum PWR canister/cask pressures are obtained from the B&W 17x17 and Westinghouse 17x17 standard fuel assemblies. The bounding pressures are calculated based on the design basis fuel assembly, i.e., the B&W 17x17 (Mark C) fuel assembly is bounding for the canister normal condition, the canister accident condition and the cask accident condition, while the WE 17x17 (std) is bounding for the cask normal condition.

The backfill pressure of the B&W 15x15 fuel assembly is revised as a part of the NAC response to RAI 3-10.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CHAPTER 3: THERMAL EVALUATION

3-13 Correct the descriptions in Sections 3.4.4.1 and 3.4.4.2 which imply that the design basis PWR and BWR assemblies have total burnups of 45,000 MWD/MTU and 40,000 MWD/MTU, respectively.

Section 1.2.3 states that the design basis fuel assembly for both the PWR and BWR has a total burnup of 50,000 MWD/MTU. Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

As shown in the response to RAI 1-1, Item 8 of Section 1.2.3 is revised to correct the total burnup for PWR and BWR fuel considered in the design basis to 45,000 MWD/MTU. The loading tables for PWR and BWR fuel having burnups to 45,000 MWD/MTU are provided in Tables 1.2-6 and 1.2-7, respectively. The descriptions provided in Sections 3.4.4.1 and 3.4.4.2 are correct for the general contents fuel considered in the design basis.

The Maine Yankee site-specific fuel is analyzed as a separate case. Maine Yankee site-specific fuel is evaluated to allow loading of Maine Yankee fuel having up to 50,000 MWD/MTU burnup. The loading table for Maine Yankee site-specific fuel is provided in Tables 5.5.1.1-12 and 5.5.1.1-15, depending on its configuration.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CHAPTER 3: THERMAL EVALUATION

3-14 The staff requires the following information: a copy of the SAS2H input file used to calculate the 25 percent correction factor; and a table containing quantities of fission gases, in particular H-3, Kr, I, and Xe, resulting from the SAS2H calculation to justify the use of 0.3125 atoms of gas/fission to derive the quantity of fission gas from both the PWR and BWR design basis spent fuel.

Section 3.4.6.1 states that the number of gas atoms from a single fission is 0.25 based on the reference "Fundamental Aspects of Nuclear Reactors Fuel Elements," by D. R. Olander. Section 3.4.6.1 further states that based on SAS2H runs, this value is increased by 25% to account for decay chains not considered in Olander.

The staff requires this information to independently confirm the applicant's MNOP calculation. Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

The SAS2H-produced gas inventories at 60,000 MWD/MTU burnup, 1.9 wt. % ^{235}U enrichment, and 40 years cool time are directly used in the revised pressure calculation, as described in revised SAR Section 3.4.4.1. The total "per assembly" fuel-generated gas inventories are presented in Tables 3.4-5 and 3.4-8 for PWR and BWR fuel, respectively. Included in Attachment B are SAS2H input files for the Westinghouse 17x17 standard and GE 9x9 (79 fuel rod) assemblies at a burnup of 60,000 MWD/MTU and an initial enrichment of 1.9 wt. % ^{235}U for cool times ranging from 5 to 40 years. At this combination of burnup and cool time, the SAS2H gas inventory is slightly higher (2-3%) than that obtained by the 0.3125 (0.25 x 1.25 – SAR Section 3.4.6.1) atoms/fission factor. The slight underprediction primarily results from increased actinide production and the associated higher alpha decay. At higher initial enrichments (above ~4.2 wt. % PWR and ~3.0 wt. % BWR), which will be required for a fuel assembly to achieve 60,000 MWD/MTU, the gas generation factor used in the fuel rod stress analysis overpredicts the SAS2H calculated inventories at 40 years.

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CHAPTER 3: THERMAL EVALUATION

3-15 Include the gas contribution from BPRAs in the MNOP analysis for the PWR fuel canister. The applicant should treat the presence of BPRAs similar to spent fuel, i.e., assuming 3 percent and 100 percent failure of BPRA rods under normal transport conditions and hypothetical accident conditions, respectively.

The staff requires this information to independently confirm the applicant's MNOP calculation. Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

As discussed in the response to RAI 3-11, BPRA gases are included in the revised analysis. Also, refer to RAI 3-12 response.

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CHAPTER 3: THERMAL EVALUATION

3-16 Justify the backfill pressure of 88.2 psig (102.9 psia) for the GE 9x9 fuel assemblies presented in Table 3.4-5. Staff references have identified a backfill pressure of 132 psig (147 psia) for this fuel assembly type.

Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

DOE/RW-0184 reports the maximum BWR fuel rod backfill pressure as 60 psig. PNL-4835 indicates a slightly higher rod backfill pressure at 6 atmospheres (88 psig) for the GE 8x8 fuel assembly type. For conservatism, a 132 psig backfill pressure is applied to all BWR fuel assembly types in the revised canister and cask pressure evaluation.

Allowable cladding temperatures are recalculated in direct response to the question, based on a conservative 132 psig maximum fuel rod backfill pressure for 8x8 and 9x9 BWR fuel assemblies; 7x7 fuel assemblies remain at 3 atms (44.1 psig) fuel rod backfill pressure (ORNL/TM-9591). The increase in rod backfill pressure slightly reduces the maximum cladding allowable temperatures. The maximum allowable fuel cladding temperatures are summarized in Table 3.4-15. Maximum allowed decay heat (cask heat load) is not impacted by this change, since a 5% temperature margin was initially applied to the calculated maximum allowable temperature, for conservatism, prior to calculating allowed heat load. This 5% margin is reduced (by <1%) to account for the allowable temperature decrease associated with the increased backfill pressure.

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CHAPTER 3: THERMAL EVALUATION

3-17 Justify using the Exxon-ANF 9x9 fuel assembly as the design basis fuel assembly for the internal pressure calculations. Based on the information provided in Tables 1.2-5 and 3.4-6, the GE 9x9 fuel assembly has a larger void space in each fuel pin, therefore it would have greater quantity of fill gas. In addition, the calculation presented in Section 3.4.4.2.1 uses 60 psig (74.7 psia) to calculate the quantity of fill gas which is lower than the rod backfill pressure for both the Exxon-ANF 9x9 and the GE 9x9.

Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

The calculated relevant molar quantities, free volumes, and pressures for the limiting BWR assemblies are:

Fuel Type	**GE7x7 (GE-2a)**	**GE7x7 (GE-3b)**	**GE8x8 (GE-4a)**	**Ex9x9 (JP-4,5)**	**GE9x9 (GE-11)**
Canister Class	4	5	5	5	5
Rod Backfill (moles)	75	69	72	45	54
Releasable Fission Gas (moles)	282	273	267	255	284
BPRA Gas (moles)	0	0	0	0	0
Can Backfill (moles)	194	201	201	206	201
Cask Backfill (moles)	9	8	8	8	8
Canister Free Volume (liters)	6460	6670	6690	6830	6690
Canister Normal Pressure (psig)	3.5	3.4	3.3	3.3	3.3
Canister Accident Pressure (psig)	41.9	40.9	40.4	35.9	40.3
Cask + Canister Free Volume (liters)	6250	6480	6500	6630	6500
Cask Normal Pressure (psig)	3.7	3.5	3.5	3.4	3.5
Cask Accident Pressure (psig)	42.8	40.1	39.6	35.2	39.5

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NAC Response to RAI 3-17 (Continued)

As discussed in RAI 3-11, canister class-specific pressure evaluations are performed for the range of fuel assemblies to be loaded. The maximum pressure for the bounding fuel assembly is provided in revised Sections 2.6.1.1 for normal conditions of transport and 2.7.3.1 for the hypothetical accident conditions.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CHAPTER 3: THERMAL EVALUATION

3-18 Remove the references to damaged fuel in Section 3.6.

Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

Section 3.6 refers to loading fuel assemblies in which damaged fuel rods have been removed and replaced by other solid rods and to fuel assemblies which are classified as damaged. Section 3.6.1 requires that fuel assemblies, classified as damaged, be placed in a Maine Yankee Fuel Can (shown in Drawing 412-501) and provides the thermal evaluation for the damaged fuel configurations. Other chapters of the Safety Analysis Report evaluate the effects of loading these configurations of damaged fuel, including the loading controls placed on damaged fuel assemblies in the Maine Yankee Fuel Can.

Certain fuel in inventory at Maine Yankee is classified as damaged and must be loaded as described in the Maine Yankee site specific sections of the Safety Analysis Report. Consequently, the thermal evaluation for damaged fuel in Section 3.6 is required.

NAC has removed a proposed damaged fuel configuration, which would have allowed the loading of fuel assemblies with up to 24 damaged fuel rods as intact fuel (i.e., not in a Maine Yankee Spent Fuel Can). References to this loading configuration were deleted in a previous submittal.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CHAPTER 4: CONTAINMENT

4-1 Correct the units used for the leak rates in Chapter 4.

The units for leak rates should be consistent with ANSI N14.5. The staff accepts the use of either ref cm^3/sec or cm^3/sec (helium). If a helium leak rate is being used as the acceptance criteria, then the corresponding test pressure and temperature conditions must also be specified. The term "standard conditions" used in the text is easily misinterpreted. Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

The calculated allowable reference (air) leak rates are converted to helium leak rates based on the difference in molecular weight and viscosity. Since the leak tests must be performed with helium, the allowable helium leak rates are also specified.

The helium leak rate test units presented in Chapter 4 are considered to be correct and conservative, since the reference leak rate and the helium leak rate are based on the same conditions (i.e., a gas at pressure is leaking to an annulus, which is at vacuum). The specified allowable helium leak rate is conservative with respect to the actual test condition. In the test condition, the helium pressure is expected to be higher than the 1 atmosphere assumed in the reference leak rate calculation due to heatup resulting from the decay heat of the contents, but cannot be less. This results in a higher driving force for the postulated leak, which has a leak path length and diameter established by the reference condition. Consequently, because of the higher pressure, the actual helium gas flow through the postulated leak path would be greater than the value specified in the test criteria.

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NAC Response to RAI 4-1 (Continued)

If the actual pressure at the time of the test were considered, the allowable reference leak rate would be greater, but would not be conservative, since the calculated pressure may not always be achieved.

The increase in helium gas temperature is not a first order effect and can be neglected for small temperature differences. The increase in temperature is relatively small because there is a comparatively short period between the time the lid and coverplates are installed and the time that the leak tests are performed. Consequently, ignoring the increase in test condition pressure and temperature is conservative.

Section 4.2.1.2 is revised to show that the method of determining the allowable helium leak rate is conservative.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CHAPTER 5: SHIELDING

5-1 Section 5.1.1 Fuel Assembly Classification

Provide an evaluation demonstrating that fuels with a burnup to 50,000 MWD/MTU are still bounded by the design basis fuel assemblies for both PWR and BWR fuel.

The design basis fuel for PWR fuel assemblies is 45,000 MWD/MTU and for BWR fuel is 40,000 MWD/MTU. However, in Section 1.2.3, the maximum burnup for both PWR and BWR fuel is listed as 50,000 MWD/MTU. Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

As shown in the response to RAI 1-1, Item 8 of Section 1.2.3 is revised to correct the total burnup to 45,000 MWD/MTU for PWR and BWR fuel considered as the design basis. The loading tables for PWR and BWR fuel having burnups to 45,000 MWD/MTU are provided in Tables 1.2-6 and 1.2-7, respectively. The descriptions provided in Section 5.1.1 are correct for the fuel considered as the design basis.

Maine Yankee site specific fuel is evaluated to allow loading of Maine Yankee fuel having up to a 50,000 MWD/MTU burnup. The loading table for Maine Yankee fuel is provided in Tables 5.5.1.1-12 and 5.5.1.1-15, depending on its configuration.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CHAPTER 5: SHIELDING

5-2 Section 5.1.2 Codes Employed

Explain why the SCALE 4.3 version is being used to determine the source term and one-dimensional and three-dimensional dose rates for the transport cask.

SCALE 4.4 was released in September 1998 and incorporates the many enhancements and corrections made to SCALE in the years since the release of SCALE 4.3. Some of the corrections include changes to the nuclide libraries. This information is necessary to confirm that the application meets the external radiation standards of Section 71.47.

NAC Response

SCALE 4.3 source term and shielding evaluations represent the basis for both the UMS® transport and storage systems. Application of various versions of a software package for identical calculations, such as generating the heat load and source terms for a UMS® fuel assembly at a fixed burnup and initial enrichment, could lead to a number of minor inconsistencies between the Safety Analysis Reports. Both the storage and transport Safety Analysis Reports were developed and initially submitted prior to the release of SCALE 4.4.

SCALE 4.3 presents a validated, and extensively used, set of code sequences for shielding and source term evaluations in spent fuel storage and transport applications. While a number of code corrections were undertaken between SCALE versions 4.3 and 4.4, none of the changes show a significant negative impact on the evaluation presented in the UMS® Universal Transport Cask Safety Analysis Report. NAC code validation for the SCALE 4.4 sequence included an evaluation section comparing the SCALE 4.4 results to those obtained from SCALE 4.3 for identical light water reactor storage and transport system inputs.

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NAC Response to RAI 5-2 (Continued)

For PWR and BWR source term generation, the SCALE 4.4 SAS2H sequences produce a ~1-2% lower decay heat and hardware gamma sources, virtually identical fuel neutron sources and ~4% lower fuel gamma sources. SAS1 (one-dimensional shielding) and SAS4 (three-dimensional shielding), SCALE 4.4 sequences produce results similar to those of the SCALE 4.3 sequences for UMS® inputs. Differences in the results are primarily associated with a higher default density for stainless steel in SCALE 4.4 (7.94 g/cm^3 versus 7.92 g/cm^3 in SCALE 4.3), which noticeably affects gamma cases where the default material density was employed. No significant change has been made to the 27 group libraries between the release of SCALE 4.3 and SCALE 4.4. The only library modification indicated by the code authors is a change to the Rhodium-103 Bondarenko factor.

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CHAPTER 5: SHIELDING

5-3 Section 5.2.2 Neutron Source

Provide an explanation as to how the different flux ratios are determined.

This section indicates that flux ratios of varying values are applied to the hardware regions of the assemblies. This information is necessary to confirm that the application meets the external radiation standards of Section 71.47.

NAC Response

Section 5.2.3 includes a discussion of how the flux ratio is applied to fuel assembly hardware regions that are adjacent to the active fuel region. Section 5.2.3 is revised to incorporate the reference for the flux ratios used. The flux ratios are determined from empirical data presented in, "Characteristics of Spent Fuel High-Level Waste and Other Radioactive Wastes Which May Require Long Term Isolation," DOE/RW-0184, December 1987.

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CHAPTER 5: SHIELDING

5-4 Section 5.2.5 Design Basis Fuel Assemblies

Provide an additional clarification for why the CE 16x16 fuel dose rates are nearly double those of the design basis PWR fuel results. This information is necessary to confirm that the application meets the external radiation standards of Section 71.47.

NAC Response

Dose rates listed in Section 5.2.5 are the result of one-dimensional shielding analyses. The one-dimensional evaluations included the spacer material associated with canister Classes 1, 2, 4, and 5. The spacer configurations are shown in Drawing 790-520. In particular, the shielding models for the Westinghouse fuel assemblies in Class 1 canisters include a 3/8-inch thick stainless steel spacer plate that is not presented for the CE 16x16 Class 3 canister evaluation.

As indicated in the fourth paragraph of Section 5.2, the spacer, in combination with a higher end-fitting source, accounts for the higher CE fuel assembly dose rates.

To provide bounding dose rates, the spacer is not modeled in the three-dimensional shielding analyses. In addition, a shielding analysis is performed (Section 5.4.2.2) with the Westinghouse fuel in the Class 1 canister located at the bottom of the transport cask (i.e., without the spacer included in the shielding model).

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CHAPTER 5: SHIELDING

5-5 Section 5.4.3 Loading Table Analysis

Revise Table 5.4-21, "Loading Table for PWR fuel," and Table 5.4.-22, "Loading Table for BWR fuel," to reflect burnups of up to 50,000 MWD/MTU.

According to the methodology section, minimum cool times were determined for burnups ranging from 30,000 MWD/MTU to 50,000 MWD/MTU. The tables only go up to a burnup of 45,000 MWD/MTU. This information is necessary to confirm that the application meets the external radiation standards of Section 71.47.

NAC Response

As clarified in the response to RAI 1-1, the contents description for Maine Yankee site specific fuel (Section 1.3.1) provides for the transport of Maine Yankee fuel with burnups up to 50,000 MWD/MTU, but that level of burnup does not apply to the general Contents of Packaging description provided in Section 1.2.3. The general contents (non-site specific fuel) is limited to 45,000 MWD/MTU.

The methodology description for the non-site specific spent fuel presented in Section 5.4.3.1 is revised to refer to fuel with burnups between 30,000 and 45,000 MWD/MTU. The description for Maine Yankee (site specific) fuel, presented in Section 5.5.1.1, is revised to clarify that Maine Yankee site specific fuel is evaluated for burnups up to 50,000 MWD/MTU.

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CHAPTER 6: CRITICALITY

6-1 The staff finds that the response to RAI 6-2 is unacceptable because it addresses only W17x17 fuel, which is clearly not the limiting fuel type with regard to poison coverage at the top of the active fuel. Submit a new response that determines and clearly states, for all allowed contents, the maximum overlap of active fuel beyond the tops of the poison panels. This maximum overlap should be explicitly modeled in the applicant's computational models. All assumptions used in determining the maximum overlap should be clearly stated. The staff notes that, for UMS storage, the applicant addressed poison coverage at the bottom of the active fuel by (a) specifying a minimum axial assembly dimension below the active fuel and (b) requiring the use of a bottom spacer below fuel assemblies not shown to meet this dimensional specification. As previously discussed, a similar approach (i.e., specifying a minimum axial assembly dimension above the active fuel and requiring a top spacer for fuel assemblies not meeting this specification) would be acceptable for ensuring that top axial poison coverage is consistent with the analyzed safety basis. Revise the SAR and TS to accurately reflect the response to this RAI.

NAC Response

The particular accident scenario addressed by this analysis is one that exceeds the requirements imposed by regulations and by regulatory review guidance. Specifically, the accident condition may be characterized as incredible because all structural analyses for the design basis conditions show that the water environment assumed in these calculations cannot occur. Both 10 CFR 71 and NUREG-1617 limit consideration of post-accident flooding to credible scenarios, and the SAR has shown that the canister is not breached under any design basis normal or accident condition.

NAC has historically assumed containment flooding for bare (uncanistered) fuel because, in such situations, the accident condition water environment may be classified as credible. NAC elected

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NAC Response RAI 6-1 (Continued)

to continue the use of this accident condition with the UMS® and MPC systems because it was conservative to do so. It is now clear that, with the conservative assumption of a transport cask containment breach, it is overly conservative to assume that the canister is also breached.

At this point in the UMS® transport cask review, however, rather than withdraw this highly conservative calculation which defends an incredible event, NAC has elected to retain the conservative accident condition, to clarify the conservative assumptions, and to improve the fuel design input data.

NAC has been working to obtain formal data from General Electric (GE) to demonstrate the essential conservatisms contained in our August 1999 responses. The noted conservatisms involved fuel rod and fuel pellet movements, based upon fuel rod gas pressures and resistance, unirradiated pellet/clad interference pressures, irradiated pellet/clad interaction after operation, external fuel rod spring characteristics, and plenum spring characteristics. NAC is aware of the ranges of magnitude of these factors, but does not have precise data from the BWR fuel fabricators.

NAC has, as yet, been unsuccessful in obtaining any data from GE in these areas. As a result, NAC has asked consultants to extract as much data as possible from the available literature. While NAC is confident of the conservatism of its August 1999 analysis, the ongoing industry issue of access to detailed fuel data from fuel fabricators cannot be solved soon enough to allow a timely response demonstrating that conservatism. Consequently, NAC has little choice but to take conservative steps to maintain licensing schedules. Therefore, NAC has decided to make a conservative design change for BWR fuel tubes, so that this issue is closed quickly and the licensing review of the UMS® for transport certification does not become needlessly protracted. NAC has decided to add conservatism to its Class 4 and 5 designs by extending the neutron absorber sheet lengths by 3 inches.

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NAC Response RAI 6-1 (Continued)

The evaluation documented in the response for RAI 6-2 (issued in August 1999) bounds all fuel assembly types proposed to be loaded into the UMS®. For that response, top and bottom impact studies for the canister and fuel were performed, which determined the maximum exposure of active fuel, due to axial shifting of the fuel and basket, to be 4.52 inches and 7.625 inches for a top impact of the PWR and BWR systems, respectively.

Additional structural analyses of the top end impact have been performed. The results of these analyses have been incorporated into the method used in determining the exposed fuel dimensions, as described in the following paragraphs. The analysis results show that the previously evaluated exposed fuel heights are still conservative. Fuel assembly minimum intact hardware dimension limits are also provided that ensure that the fuel exposure dimensions used in the criticality analyses are not exceeded.

Evaluation of BWR System Top End Impact

Axial shifting of the contents of the Transportable Storage Canister (TSC) occurs as a result of a top end impact load condition for the transport cask containing a loaded TSC. In this scenario of contents shifting, the fuel assembly and the basket are considered to be shifted upward to contact the canister lid. The distance between the canister lid and the neutron absorber sheets, which are attached to the fuel tubes, and the distance between the top of the fuel assembly and the active fuel region are required to establish the height of active fuel exposed beyond the neutron absorber for any given assembly. Exposure of the active fuel in any specific fuel type occurs if the minimum distance between the top of the assembly and the top of the active fuel region is less than the maximum distance from the canister lid to the top of the neutron absorber sheet. The exposed fuel height evaluation is performed for each BWR fuel assembly type that is proposed to be loaded into the UMS® canister. The calculation is divided into three stages: calculation of the neutron absorber offset, determination of the active fuel offset, and calculation of the fuel exposure.

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NAC Response RAI 6-1 (Continued)

In stage one, the maximum distance between the top of the neutron absorber sheet and the canister lid is determined for each UMS® BWR canister class (Classes 4 and 5). The maximum distance provides the greatest fuel exposure, when considering a shifted fuel assembly. This distance depends on the canister class specific weldment, basket, tube and neutron absorber lengths; the relative location of the neutron absorber on the fuel tube; and tolerances associated with the basket components. The maximum distance for a BWR basket shifted to the canister lid is shown in the following table.

Maximum Distance between Neutron Absorber and Canister Lid

	Class 4 (inch)	Class 5 (inch)
Basket Shifted to Lid	12.76	12.76

In the second stage of the analysis, the minimum distances between the canister lid and the fuel assembly, and between the top of the fuel assembly and the active fuel region are determined for each BWR fuel type. Since the fuel assembly is shifted to contact the canister lid, the distance between the lid and the fuel assembly is always zero. The active fuel shifting condition in the fuel assembly assumes that:

- BWR fuel rods are either tie rods connecting the top and bottom nozzles, or are rods manufactured with an external spring between the top of the fuel rod and the top nozzle tie plate. For this evaluation, all external springs are ignored. Therefore, all BWR fuel rods are allowed to shift axially into contact with the top tie plate.

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NAC Response RAI 6-1 (Continued)

- Within BWR fuel rods, the fuel is assumed to shift upward into the plenum region. Each plenum region contains a plenum spring. Detailed structural analyses have shown that during a 60 g top end impact, the BWR plenum spring will compress and rebound 1.729 inches. The fuel material in the rods is assumed to shift and remain in contact with the compressed plenum spring. A review of plenum length and spring data for various rod designs indicates a minimum of 13% of the BWR plenum space is occupied by a solid height plenum spring. The final height of the plenum spring is calculated from the sum of the solid height of the spring and a spring rebound height of 1.729 inches.
- Detailed structural analyses of the BWR assembly have also shown that during a 60 g top end impact, the lifting bail will deform. The maximum BWR bail deformation was calculated to be 2.371 inches.

Therefore, spacing from the assembly top to the active fuel region is controlled by the top end-fitting height, the fuel rod end-plug height and the distance the active fuel moves into the top plenum. In the case of BWR rods, the end-plug height includes only the portion of the plug below the tie plate when the fuel rod is shifted up. The distance between the top of the fuel assembly and the active fuel region for the bounding (minimum offset) fuel types in each canister class are presented in SAR Table 6.4-19 and are replicated in the following table. Also included in the listing is the Class 5 Exxon/ANF 9x9 assembly, since this assembly represents the maximum reactivity radial lattice geometry.

Fuel Type	Canister Class	Minimum Distance from Top of Fuel Assembly to Active Fuel Region (inches)
Exxon/ANF 9x9	4	8.448
Exxon/ANF 9x9	5	8.458
GE 8x8	5	8.448

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NAC Response RAI 6-1 (Continued)

In the third stage of the evaluation, the maximum active fuel exposure (or minimum coverage) for each fuel type is determined by simply subtracting the active fuel offset from the neutron absorber offset. A positive value indicates active fuel is exposed (i.e., neutron absorber does not cover the entire active fuel region).

Fuel Type	Canister Class	Neutron Absorber Offset (inch)	Fuel Offset (inch)	Fuel Exposure (inch)
Exxon/ANF 9x9	4	12.76	8.448	4.312
Exxon/ANF 9x9	5	12.76	8.458	4.302
GE 8x8	5	12.76	8.448	4.312

As shown in the preceding table, the maximum lengths of exposed BWR fuel result for the Exxon/ANF 9x9 and GE 8x8 BWR fuel assemblies at 4.312 inches. For further conservatism in the criticality analysis, the active fuel exposure length is increased to 7.625 inches for BWR fuel assemblies. The significant difference between the maximum evaluated fuel exposure of 7.625 inches and the maximum calculated exposure of 4.312 inches is primarily the result of an additional 3 inches of axial neutron absorber added to the design after the criticality evaluations were completed. The remaining 0.313 inches are accounted for by modifications to the spring and bail dimensions resulting from the detailed structural evaluation of the assembly and basket configuration in response to the top end impact.

As previously shown, the maximum length of exposed fuel is not obtained from the BWR fuel assembly defined as having the maximum reactivity in Section 6.4 of the SAR. The maximum reactivity BWR assembly documented in the SAR is the UMS® Class 5 Exxon/ANF 9x9 (79 fuel rod) assembly for BWR canisters. Therefore, rather than analyzing each fuel type with its specific exposed fuel height, the evaluated exposed fuel height of 7.625 inches (BWR) is applied to the Class 5 Exxon/ANF 9x9 BWR assembly.

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NAC Response RAI 6-1 (Continued)

The BWR criticality model shown in SAR Figure 6.6.3-3 is based on a 7.625 inch exposed fuel height. As such a number of differences are apparent between the model shown in the SAR and a model based on the description listed above. To minimize confusion a matrix containing key differences is constructed.

	Model Figure 6.6.3-2	Model Based On Revised Design / Analysis
Fuel Assembly Location	Shifted to Lid	Shifted To Lid
Basket Location	Bottom of Canister	Top of Canister
Additional Neutron Absorber	0.0	3.0 inches
External Spring	Full Credit (No Shift)	No Credit (Rod Shifted to Tie Plate)
Plenum Spring	50% Compression	Full Compression Plus Rebound
Plenum Spring Rebound	0.0 inches	1.729 inches
Bail Deformation	4.7 inches	2.371 inches
Limiting Assembly Adjustment[1]	0.009 inches	0.01 inches
Resulting Exposed Fuel Height	7.625 inches	4.312 inches

(1) The maximum reactivity assembly is the Class 5 Exxon/ANF assembly, while the maximum exposed fuel height is obtained from the Exxon/ANF Class 4 assembly. To apply the maximum exposed fuel height to the maximum reactivity assembly an adjustment is made decreasing the neutron absorber length.

NAC INTERNATIONAL RESPONSE
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NAC Response RAI 6-1 (Continued)

Evaluation of PWR System Top End Impact

Axial shifting of the contents of the TSC occurs as a result of a top end impact load condition for the transport cask containing a loaded TSC. In this scenario of contents shifting, the basket is assumed to remain in contact with the canister baseplate, while the fuel assembly is shifted up to contact the canister lid. The distance between the canister lid and the neutron absorber sheets, which are attached to the fuel tubes, and the distance between the top of the fuel assembly and the active fuel region are required to establish the height of active fuel exposed beyond the neutron absorber for any given assembly. Exposure of the active fuel in any specific fuel type occurs if the minimum distance between the top of the assembly and the top of the active fuel region is less than the maximum distance from the canister lid to the top of the neutron absorber sheet. The exposed fuel height evaluation is performed for each PWR fuel assembly type that is proposed to be loaded into the UMS® canister. The calculation is divided into three stages: calculation of the neutron absorber offset, determination of the active fuel offset, and calculation of the fuel exposure.

In stage one, the maximum distance between the top of the neutron absorber sheet and the canister lid is determined for each UMS® PWR canister class (Classes 1, 2, and 3). This maximum distance depends on the canister class specific weldment, basket, tube and neutron absorber lengths; the relative location of the neutron absorber on the fuel tube; and tolerances associated with the basket components. This maximum distance provides the greatest fuel exposure, when considering a shifted fuel assembly. The maximum distances for a basket at the bottom of the canister and for a basket shifted to the canister top are shown in the following table (Note: shifted basket dimensions are provided to show the conservatism in retaining the basket in contact with the canister bottom plate).

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NAC Response RAI 6-1 (Continued)

Maximum Distance between Neutron Absorber and Canister Lid

	Class 1 (inch)	Class 2 (inch)	Class 3 (inch)
Basket Shifted to Lid	8.89	10.99	7.38
Basket at Canister Bottom	10.19	12.29	8.69

In the second stage of the analysis, the minimum distances between the canister lid and the fuel assembly, and between the top of the fuel assembly and the active fuel region are determined for each PWR fuel type. Since the fuel assembly is shifted to contact the canister lid, the distance between the lid and the fuel assembly is always zero (no credit is taken for any offset produced by the PWR leaf springs). The active fuel shifting condition in the fuel assembly assumes that:

- In PWR fuel assemblies, where a space exists between the fuel rod end-cap and the end-fitting, the fuel rods are shifted within the grid until contact is made with the top end-fitting (zero gap).
- Within the PWR fuel rods, the fuel is assumed to shift upward into the plenum region. Each plenum region contains a spring, which will fully compress during an upper end impact. The fuel material in the rods is assumed to shift and remain in contact with the fully compressed (solid height) plenum spring. A review of plenum length and spring data for various rod designs indicates a minimum of 31% of the PWR plenum space is occupied by a solid height plenum spring.
- Detailed structural analyses of the PWR assembly have shown that during a 60 g top end impact, no significant damage to the top end-fitting (i.e., no height reduction) occurs.

Therefore, spacing from the assembly top to the active fuel region is controlled by the end-fitting height, the fuel rod end-plug height and the distance the active fuel moves into the top plenum. The distance between the top of the fuel assembly and the active fuel region for the bounding

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NAC Response RAI 6-1 (Continued)

(minimum offset) fuel types in each canister class are presented in SAR Table 6.4-19 and are replicated in the following table. Also included in the listing is the Westinghouse 17x17 OFA assembly, since this assembly represents the maximum reactivity radial lattice geometry.

Fuel Type	Canister Class	Minimum Distance from Top of Fuel Assembly to Active Fuel Region (inches)
Westinghouse 15x15	1	5.718
Westinghouse 17x17 OFA	1	6.296
B&W 15x15	2	10.289
CE 16x16 (SYS 80)	3	13.176

In the third stage of the evaluation, the maximum active fuel exposure (or minimum coverage) for each fuel type is determined by simply subtracting the active fuel offset from the neutron absorber offset. A positive value indicates active fuel is exposed (i.e., neutron absorber does not cover the entire active fuel region).

Fuel Type	Canister Class	Neutron Absorber Offset (inch)	Fuel Offset (inch)	Fuel Exposure (inch)
Westinghouse 15x15	1	10.19	5.718	4.472
Westinghouse 17x17 OFA	1	10.19	6.296	3.894
B&W 15x15	2	12.29	10.289	2.001
CE 16x16 (SYS 80)	3	8.69	13.176	-4.486

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NAC Response RAI 6-1 (Continued)

As shown in the preceding table, the maximum lengths of exposed PWR fuel result for the Westinghouse 15x15 PWR fuel assembly at 4.472 inches. For further conservatism in the criticality analysis, the active fuel exposure length is increased to 4.52 inches for PWR fuel assemblies.

As previously shown, the maximum length of exposed fuel is not obtained from the PWR fuel assembly defined as having the maximum reactivity in Section 6.4 of the SAR. The maximum reactivity PWR assembly documented in the SAR is the Westinghouse 17x17 OFA assembly for PWR canisters. Therefore, rather than analyzing each fuel type with its specific exposed fuel height, the evaluated exposed fuel height of 4.52 inches is applied to the Westinghouse 17x17 OFA PWR assembly.

To model the 4.52-inch exposed fuel length for the Westinghouse 17x17 OFA PWR fuel assembly criticality evaluation, a number of modifications are made to the nominal (unshifted) fuel and basket model.

- Fuel assemblies are shifted to the canister lid.
- Fuel rods are shifted to the top end-fitting.
- The active fuel is moved to the midpoint of the plenum.
- The top end-fitting height is reduced by 1 inch.
- The PWR neutron absorber sheet is reduced by 0.815 inch.

These model modifications produce a fuel exposure identical to that obtained by shifting the fuel in the rod against the solid height plenum spring, by axial movement of the fuel rods and assembly, and a 0.626-inch reduction in neutron absorber (BORAL) height. This neutron absorber height reduction is determined based on modeling the 4.52-inch fuel exposure versus the actual 3.894-inch fuel exposure of the Westinghouse 17x17 OFA fuel assembly.

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NAC Response RAI 6-1 (Continued)

Minimum Fuel Assembly Intact Hardware Dimension Limits

Based on limiting the exposed height of active fuel to 4.52 inches for the PWR fuel assemblies and to 7.625 inches for the BWR fuel assemblies, intact fuel assembly hardware limits are defined to assure compliance with the safety basis of the analysis. These limits consider zero PWR top end-fitting deformation, 2.37 inches of BWR top end-fitting (lifting bail) deformation and a BWR plenum spring rebound height of 1.729 inches. The limits for each UMS® canister class containing BWR fuel are calculated by subtracting the sum of the height of exposed fuel, 7.625 inches, and the plenum spring rebound height, 1.729 inches, from the sum of the lifting bail deformation, 2.371 inches, and the distance between the canister lid and the top of the neutron absorber. The resulting limits are provided in the following table. The limits for each

UMS® canister class containing PWR fuel are calculated by subtracting the height of exposed fuel, 4.52 inches, from the distance between the canister lid and the top of the neutron absorber. These limits are also provided in the following table.

UMS® Fuel Class	1	2	3	4	5
Distance between Lid and Neutron Absorber	10.19	12.29	8.69	12.76	12.76
Evaluated Height of Exposed Fuel	4.52	4.52	4.52	7.625	7.625
Top End Fitting Deformation	0	0	0	2.371	2.371
Plenum Spring Rebound Height	Not Analyzed			1.729	1.729
Minimum Intact Hardware Dimension Limit	5.67	7.77	4.17	5.777	5.777

* All units are in inches.

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NAC Response RAI 6-1 (Continued)

Compliance with these limits will ensure that the exposed fuel heights evaluated and found to result in a subcritical system will not be exceeded. For PWR fuel, the minimum intact assembly hardware dimension above the active fuel shall be calculated by summing the top end-fitting height, the top end-cap height, and the solid height of the plenum spring. For BWR fuel, the minimum axial assembly dimension above the active fuel shall be calculated by summing the intact top nozzle height, the portion of the top end-cap height below the tie plate when the fuel rod is shifted up, and the solid height of the plenum spring. Tolerances on these components shall be conservatively considered when calculating the subject dimension.

Evaluation of Bottom End Axial Fuel Shifting

Similar to the top end evaluation, a bounding hypothetical axial fuel-shifting condition is considered in which all of the fuel rods are shifted to the bottom of each assembly. For PWR fuel assemblies with a lower plenum, the fuel within every rod is assumed to shift downward to contact a fully compressed plenum spring. Each fuel assembly is assumed to remain in contact with the canister bottom plate. The basket dimensions used assume conservative tolerances, and the basket is conservatively assumed to be shifted upward to contact the canister shield lid. This bounding axial shifting scenario results in the maximum distance from the canister bottom plate to the lower end of the neutron absorber. For all UMS® PWR canister classes, this distance is limited to 5.22 inches. For all UMS® BWR canister classes, the distance is limited to 8.19 inches.

However, all PWR and BWR fuel assembly types have rod bottom end caps, tie plates and/or components of the bottom end-fitting/nozzle that will not deform to a total height of less than 0.7 inches. Consequently, the top end impact axial fuel shifting condition, which considers exposed fuel lengths of 4.52 inches for PWR fuel and 7.625 inches for BWR fuel, bounds the bottom end impact axial fuel shifting condition.

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CHAPTER 7: OPERATING PROCEDURES

7-1 Correct the units used for the leak rates in Chapter 7.

The units for leak rates should be consistent with ANSI N14.5. The staff accepts the use of either ref cm^3/sec or cm^3/sec (helium). If a helium leak rate is being used as the acceptance criteria then the corresponding test pressure and temperature conditions must also be specified. Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

The calculated allowable reference (air) leak rates are converted to helium leak rates based on the difference in molecular weight and viscosity. Since the leak rate tests must be performed with helium, the allowable helium leak rates are specified in the procedure (Section 7.1.3).

The units presented for the helium leak rate tests in Section 7.1.3 and Table 7-2 are considered to be correct and conservative, since the reference leak rate and the helium leak rate are based on the same conditions (i.e., a gas at pressure is leaking to an annulus, which is at vacuum). The specified allowable helium leak rate is conservative with respect to the actual test condition because: 1) In the test condition, the helium pressure is expected to be higher than the 1 atmosphere assumed in the reference leak rate calculation due to the decay heat of the contents, but cannot be less. This results in a higher driving force for the postulated leak; and 2) The change in average gas temperature is a higher order effect and can be neglected for small temperature differences. If the pressure at the time of the test were considered, the allowable reference leak rate would be greater, but might not always be conservative, since the calculated pressure may not always be achieved. Consequently, ignoring the test condition pressure and temperature is conservative.

See the response to RAI 4-1.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CHAPTER 7: OPERATING PROCEDURES

7-2 Remove the steps for loading the Transportable Storage Canister while in the Transfer Cask from Section 7.5.1.

The application should address only those areas necessary for transportation. Procedures regarding the storage aspects of a dual-purpose canister must be submitted within a 10 CFR Part 72 application and should be incorporated by reference only within a 10 CFR Part 71 application.

NAC Response

Section 7.5 is deleted. While the RAI question refers only to Section 7.5.1, the reason for removing Section 7.5.1 is also applicable to Sections 7.5.2 and 7.5.3, which described how GTCC waste is loaded in the transportable storage canister and how the canister is opened, respectively. Neither of these sections is directly relevant to transport; therefore, these sections are removed.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CHAPTER 8 – ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8-1 Correct the units used for leak rates in this Chapter.

The units for leak rates should be consistent with ANSI N14.5. The staff accepts the use of either ref cm^3/sec or cm^3/sec (helium). If a helium leak rate is being used as the acceptance criteria then the corresponding test pressure and temperature conditions must also be specified. Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

Section 8.1.3 is revised to delete the reference to standard ("std") cm^3/sec for the helium leak rate.

The calculated allowable reference (air) leak rate specified in the acceptance criteria is converted to a helium leak rate based on the difference in molecular weight and viscosity. Since the leak tests must be performed with helium, the allowable helium leak rates are specified in the acceptance criteria (Section 8.1.3).

The helium leak rate test units of "cm^3/sec" presented in Section 8.1.3 are considered to be correct, since the acceptance tests are expected to be performed at the pressure difference (essentially one atmosphere) and at approximately the temperature (298K) used in the calculation of the reference air leak rate. Use of a higher pressure differential for the leak rate test results in a conservative leak test due to the higher driving pressure differential. Since the test temperature does not have a first order effect, any likely test temperature differential does not have a significant effect.

See the response to RAI 4-1.

NAC INTERNATIONAL RESPONSE
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CHAPTER 8 – ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8-2 Revise the SAR to include qualifications of the individual(s) performing the leak tests at all stages of the NAC-UMS lifetime (i.e., fabrication, pre-shipment, periodic, and maintenance).

Section 71.1(a) requires complete and accurate information be submitted in the application.

<u>NAC Response</u>

As described in Table 4.1-1 of Chapter 4, leak testing is performed in accordance with the requirements of ANSI N14.5-1997. Consequently, tests are conducted in accordance with Section 8.0, "Conduct of Tests," in ANSI N14.5-1997. Section 8.5 of ANSI N14.5-1997 requires that tests be conducted by trained and qualified personnel in accordance with written procedures and that the results of the tests be documented.

NAC may elect to either subcontract the leak testing activity to a company or individual qualified in accordance with the American Society for Nondestructive Testing, Level II or III inspector, or have appropriately trained company personnel perform the testing. In either case, testing will be performed in accordance with ANSI N14.5-1997. Training of NAC personnel will be in accordance with SNT-TC-1A, "Recommended Practices, Nondestructive Testing, Personnel Qualification and Certification," to assure that those personnel are qualified as Level II or Level III inspectors.

Sections 8.1.3 and 8.2.2 are revised to require compliance with Section 8.5 of ANSI N14.5-1997.

NAC INTERNATIONAL RESPONSE
TO
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CHAPTER 8 – ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8-3 Revise Table 8.2-1 to specifically list the vent and drain port coverplates.

Section 71.1(a) requires complete and accurate information be submitted in the application.

NAC Response

Table 8.2-1 is revised to separately list the vent and drain port coverplates.

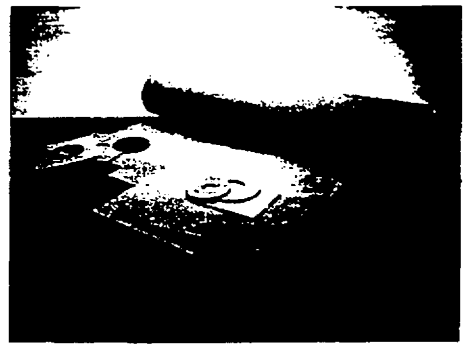
NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Attachment A
for the Response to RAI 3-5

Product Specifications for
Fiberfrax® Ceramic Fiber Paper

UNIFRAX Product Specifications

Fiberfrax® Ceramic Fiber Paper



Introduction

Fiberfrax® ceramic fiber papers are a unique class of products which consists primarily of an alumino-silicate fiber in a non-woven matrix with a latex binder system. The ceramic fibers are randomly orientated forming uniform, flexible, lightweight sheets in a specialized paper-making process which is statistically controlled.

Unifrax Corporation has been producing Fiberfrax papers for over 25 years and still remains the only basic ceramic fiber producer worldwide with in-house paper-making facilities.

By blending different fibers, binders, and additions while varying the manufacturing process, Unifrax Corporation now produces a variety of Fiberfrax paper products as diverse as the applications for which they are used.

Product Line Advantages

Fiberfrax ceramic fiber papers offer industrial engineers many unique problem-solving advantages which include:

- High temperature stability
- Low thermal conductivity
- Low heat storage
- Weight reduction
- Resiliency
- Thermal shock resistance
- High heat reflectance
- Good dielectric strength
- Excellent corrosion resistance
- Easy to wrap, shape, or cut

General Uses of Fiberfrax Papers

Fiberfrax papers are used to solve all types of heat related problems, and are used as:

- Highly efficient refractory backup
- Dependable fire protection
- Thermal insulation
- Hot gas filtration media
- Refractory tube fabrication
- High temperature gasket, separator, or parting agent

Typical Markets/Applications

Aerospace
Heat shields, nose cone ablative shields, igniter line protection, and oxygen generators.

Appliance
Self-cleaning ovens, woodburning stoves, electrical heaters, mobile home appliance insulation.

Ceramic and Glass
Ware separator, metal clad brick gaskets, glass tank refractory backup.

Petrochemical
Transfer line protection, welding, and brazing protection.

Automotive
Muffler insulation, heat shielding.

Steel and Nonferrous
Investment casting mold wrapping, ladle refractory backup, tube couple protection.

UNIFRAX

Product Range

Product Segmentation

Fiberfrax® ceramic fiber papers are differentiated by thickness, density, fiber index, and chemistry. They are often segmented into three groups:

- Utility grades, which include 440 and Rollboard paper, are the most cost effective where performance characteristics are less critical.
- Standard grades: 550, 970, 880, and 110 paper are used where reliability and consistency are important.
- Premium grades: 882-H, 972-H, and HSA paper are used when organic outgassing cannot be tolerated and performance is critical.

440 Paper

440 paper is a low cost, high strength composite paper made from a combination of ceramic fiber, inert fillers, and reinforcing fiberglass. The fiberglass gives added strength to the 440 paper in environments between 450 and 1300°F. This product is further formulated with a fire retardant smoke suppressant reducing the effects of the organic binder burnout.

Rollboard

The lower density, binders, and ceramic fiber purity characteristics of Fiberfrax Rollboard paper lead to lower cost, higher flexibility, and reduced smoke and odor during burnout. Rollboard paper is best suited for wrapping intricate shapes or molds along with single use disposable type applications.

550 Paper

550 paper is made from unwashed high purity ceramic fiber. Its higher density and binders give performance properties ideal for most refractory-type applications.

110 Paper

110 paper is a clay-filled, sheeted ceramic fiber paper which is both rigid yet flexible. The rigidity is maintained even after burn-out of the organic bonding agents. The good dielectric strength, compression restraints, and die cutting characteristics of 110 paper are advantageous in many high temperature gasketing applications.

970 Paper

970 paper is made from high purity Fiberfrax washed fiber wherein a large portion of the unfiberized particles are removed prior to lay-up. The washing of the fiber gives a great uniformity to the structure while reducing weight and improving the thermal resistance. This product is also preferred in automatic die stamping operations where shot can lead to excess die wear.

880 Paper

880 paper is made from higher alumina, shorter, smaller diameter fibers and layed up at significantly higher densities. These factors lead to high temperature, slow shrinkage, higher strength, and better chemical resistivity. This product is used in environments where standard ceramic fiber papers will not hold up.

HSA Paper

HSA paper is made from unique high surface area (HSA) fibers that contain no unfiberized material. The results are lighter weight and extremely low thermal conductivity, making it the choice of the aerospace industry. It is also used when uniform pore structure and/or shot cannot be tolerated, such as in glass contact or gas filtration. This product is available with or without an organic binder.

Inorganic Papers

Fiberfrax papers are available without the organic binder system. These products are completely free of organics and used when higher fired strength is required or in processes and applications where even small amounts of organic burnout is unacceptable. However, since these products rely on ceramic bonding rather than thermal plastic binders, repeated fluxing will result in dusting and loss of strength. Two grades and several thicknesses and widths are available. (In addition, HSA Paper is available with or without organic binder):

- **882-H** has higher temperature stability (2600°F) and higher density leading to the maximum burn strength of an unbindered paper.
- **972-H**, like 882-H paper, is heat treated to remove organic and to anneal the fibers for strength. 972-H paper remains soft and flexible allowing it to conform to most shapes or contours.

Typical Chemical/Physical Properties

Fiberfrax papers exhibit excellent chemical stability resisting attack from most corrosive agents. Exceptions are hydrofluoric and phosphoric acids and concentrated alkalies. If Fiberfrax papers are wet by water or steam, all thermal and physical properties are completely restored upon drying. No water of hydration is present in most Fiberfrax paper grades. Fiberfrax papers have good dielectric strengths.

Fiberfrax papers, with the exception of the inorganic series, will generate small amounts of smoke and trace element out-gassing during the initial exposure to temperatures above 450°F

Certifications/Approvals

Most Fiberfrax papers are recognized under the component program at Underwriters Laboratories, Inc., conform to U.S. Coast Guard requirements for incombustible materials, and are tested in accordance with ASTM methods whenever applicable. For details of approvals and test methods, contact your local sales outlet.

Certification of compliance to specifications is available upon request

UNIFRAX

Form C-1423
Effective 6/96

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Fiberfrax® Ceramic Fiber Papers Property Comparison
(Typical Values)

Paper Grade	440 (FR)*	Roll Board	550	110	970	972-H** (970-H)	880	882-H (880-H)	HSA	HSA (OF)
Physical Properties										
Color	Gray	Off-white	White	Tan	White	White	White	White	White	White
Use limit (°F)	1300	2300	2300	2300	2300	2300	2600	2600	2300	2300
Soft point (°F)	1800	3200	3260	2800	3260	3260	3500	3500	3100	3100
Density (PCF)	13	10	12	18	10	12	18	16	10	7.0
Fiber index (% Wt)	n/a	40	50	n/a	70	70	45	45	100	100
Chemistry (% Wt)										
Al_2O_3	33.0	43.0	49.2	46.4	49.2	49.2	54.8	54.8	46.0	46.0
SiO_2	45.0	54.0	50.5	44.4	50.5	50.5	44.0	44.0	52.0	52.0
Na_2O_3	2	0.2	0.2	1.5	0.2	0.2	0.2	0.2	0.2	0.2
Fe_2O_3	2	0.8	0.06	1.0	0.06	0.06	0.06	0.06	trace	trace
Others	18	2.0	0.04	6.7	0.04	0.04	0.04	0.04	<2	<2
LOI (incl. binder)	9.5	3.0	6.0	8.5	5.0	0.1	8.0	0.1	3.0	0.1
Thermal Conductivity (Btu in/hr ft²°F)										
@ 500°F	0.40	0.42	0.40	0.55	0.40	0.40	0.40	0.40	0.3	0.3
@ 1000°F	0.79	0.91	0.79	0.78	0.78	0.76	0.76	0.77	0.47	0.47
@ 2000°F	1.91	2.55	1.29	1.58	1.26	1.19	1.18	1.25	0.82	0.82
Compression (PSI % Deformation)										
10%	5	1	1	4	1.3	—	3	—	—	—
25%	34	5	6	26	5.8	—	16	—	—	—
50%	489	32	35	167	22	—	44	—	—	—
Strength										
Tensile (GM/in)	6000	4000	4800	12000	5100	(7000)	6500	(7200)	—	—
Burned (GM/in)	150	105	260	630	264	400	322	524	—	—
Burst (PSI)	45	22	19	248	25	—	37	—	—	—
Thickness (Inches)***										
A = 1/32	n/a	n/a	n/a	n/a	0.038	0.037	n/a	n/a	n/a	n/a
F = 1/16	0.070	n/a	0.070	0.065	0.070	0.067	0.070	0.064	0.080	n/a
J = 1/8	0.130	0.100	0.130	0.125	0.130	0.122	0.130	0.125	—	0.130
K = 1/4	—	n/a	0.250	—	n/a	n/a	n/a	n/a	—	—
Roll Sizes (STD)	25#, Mill	Mills	25#, Mill	Sheet	10#, 25#, Mill	25#	25#, Mill	10#, 25#	Sheets	500SF
Width (STD)	24, 48	18, 24	24, 48	42 x 48	12, 24, 48	12, 24	12, 24, 48	12, 24	42 x 48	51

Availability

Nonstandard widths available upon request.

Notes About Chart

- *The 440(FR) designation references the addition of a fire retardant smoke suppressant.
- **972-H – 882-H papers were formerly designated as 970-H and 880-H respectively.
- ***Measured under 4 PSF.
- "H" designation references the heat treating process used to remove organics.
- "OF" designation signifies materials made without the use of organic binders.
- The continuous use limit of Fiberfrax insulation is determined by a maximum allowable linear change criteria, not product melting point.

UNIFRAX

Form C-1423
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NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

Attachment B
for the Response to RAI 3-14

SAS2H Input Files for Westinghouse 17x17 Standard Fuel Assembly
And
SAS2H Input Files for GE 9x9 (79 Fuel Rod) Assembly

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

SAS2H Input Files for Westinghouse 17x17 Standard Fuel Assembly–60,000 MWD/MTU Burnup, 1.9 wt. % ^{235}U Enrichment

```
=SAS2H      PARM=(HALT09,SKIPSHIPDATA)
Class 1 - we17a - WE17 (Std) - 1.9 w/o U235, 60000 MWD/MTU, Max 40 years cool time
27GROUPNDF4 LATTICECELL
UO2        1 0.950 900 92235 1.9 92238 98.1 END
ZIRCALLOY  2 1.0 620 END
H2O        3 DEN=0.725 1.0 580 END
ARBM-BORMOD 0.725 1 1 0 0 5000 100 3 550.0E-6 580 END
ZIRCALLOY  4 1.0 580 END
H2O        5 DEN=0.725 0.9751 580 END
ZIRCALLOY  5 0.0249 580 END
END COMP
SQUAREPITCH 1.2598 0.8192 1 3 0.9500 2 0.8357 0 END
NPIN=264 FUEL=365.760 NCYC=3 NLIB=3 PRIN=6 LIGH=5
INPL=1 NUMH=24 NUMI=1 MXTUBE=4 ORTU=0.6121 SRTU=0.5740 END
POWER=18.5535 BURN=503.4734 DOWN=60 END
POWER=18.5535 BURN=503.4734 DOWN=60 END
POWER=18.5535 BURN=503.4734 DOWN=1461 END
FE 0.6738 CR 0.1900 NI 0.1150 MN 0.0200 CO 0.0012
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71  E
1$$  1 1T
COOLING To 40 YEARS AND FISSION PRODUCT GAMMA REBIN
3$$  21 0 1 28 A33 22 E
54$$  A8 1 E T
35$$  0 T
56$$  0 6 A13 -2 5 3 E
57**  4.0 E T
COOLING To 40 YEARS AND FISSION PRODUCT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 10.0 15.0 20.0 35.0 40.0
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F.00000001
81$$  2 51 26 1 E
82$$  F6
83**  1.40e+7  1.20e+7  1.00e+7  8.00e+6   6.50e+6  5.00e+6
      4.00e+6  3.00e+6  2.50e+6  2.00e+6   1.66e+6  1.44e+6
      1.22e+6  1.00e+6  0.80e+6  0.60e+6   0.40e+6  0.30e+6
      0.20e+6  0.10e+6  0.05e+6  0.02e+6   0.01e+6
84**  1.46e+7  1.36e+7  1.25e+7  1.125e+7  1.00e+7
      8.25e+6  7.00e+6  6.07e+6  4.72e+6   3.68e+6
      2.87e+6  1.74e+6  0.64e+6  0.39e+6   0.11e+6
      6.74e+4  2.48e+4  9.12e+3  2.95e+3   9.61e+2
      3.54e+2  1.66e+2  4.81e+1  1.60e+1   4.00e+0
      1.50e+0  5.50e-1  7.09e-2  1.00e-5   T
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
56$$ F0  T
```

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

SAS2H Input Files for GE 9x9 (79 Fuel Rod) Assembly– 60,000 MWD/MTU Burnup, 1.9 wt. % ^{235}U Enrichment

```
=SAS2H       PARM=(HALT15,SKIPSHIPDATA)
Class 5 - ge09b - GE9 (GE-11 ) - 1.9 w/o U235, 60000 MWD/MTU, Max 40 years cool time
27GROUPNDF4 LATTICECELL
UO2         1 0.950 900 92235 1.9 92238 98.1 END
ZIRCALLOY 2 1.0 620. END
H2O 3 DEN=0.446 1.0 562 END
H2O 4 DEN=0.743 1.0 553 END
ZIRCALLOY 5 1.0 553 END
H2O 6 DEN=0.446 1.0 562 END
END COMP
SQUAREPITCH 1.4400 0.9550 1 3 1.1200 2 0.9780 0 END
NPIN=79 FUEL=381.000 NCYC=3 NLIB=5 PRIN=6 LIGH=5
INPL=2 NUMZ=7 END
4 0.4890 5 0.5600 6 1.1490 500 7.3119 6 7.6730 5 7.8987 4 8.5982
POWER=4.7250 BURN=837.6619 DOWN=60 END
POWER=4.7250 BURN=837.6619 DOWN=60 END
POWER=4.7250 BURN=837.6619 DOWN=1461 END
FE 0.6738 CR 0.1900 NI 0.1150 MN 0.0200 CO 0.0012
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71  E
1$$  1 1T
COOLING To 40 YEARS AND FISSION PRODUCT GAMMA REBIN
3$$  21 0 1 28 A33 22 E
54$$  A8 1 E T
35$$  0 T
56$$  0 6 A13 -2 5 3 E
57**  4.0 E T
COOLING To 40 YEARS AND FISSION PRODUCT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 10.0 15.0 20.0 35.0 40.0
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F.00000001
81$$  2 51 26 1 E
82$$  F6
83**  1.40e+7  1.20e+7  1.00e+7  8.00e+6   6.50e+6  5.00e+6
      4.00e+6  3.00e+6  2.50e+6  2.00e+6   1.66e+6  1.44e+6
      1.22e+6  1.00e+6  0.80e+6  0.60e+6   0.40e+6  0.30e+6
      0.20e+6  0.10e+6  0.05e+6  0.02e+6   0.01e+6
84**  1.46e+7  1.36e+7  1.25e+7  1.125e+7  1.00e+7
      8.25e+6  7.00e+6  6.07e+6  4.72e+6   3.68e+6
      2.87e+6  1.74e+6  0.64e+6  0.39e+6   0.11e+6
      6.74e+4  2.48e+4  9.12e+3  2.95e+3   9.61e+2
      3.54e+2  1.66e+2  4.81e+1  1.60e+1   4.00e+0
      1.50e+0  5.50e-1  7.09e-2  1.00e-5   T
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
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FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
56$$ F0  T
END
```