NAC-UMS
Docket # 71-9270
TAC # L22452

NAC INTERNATIONAL

RESPONSE TO THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

REQUEST FOR ADDITIONAL INFORMATION

(RAI-1 AUGUST 30, 1999)

NAC UNIVERSAL TRANSPORT SYSTEM (NAC-UMS)

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

MAY 2000

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Section 1.0 General Information

1-1 Remove damaged fuel from the site-specific fuel terminology in Table 1-1, or provide the appropriate analysis to support including damaged fuel.

The site-specific fuel terminology from Table 1-1 includes containerized damaged fuel. Analysis has not been presented for damaged fuel and, therefore, cannot be included as site-specific fuel.

Section 71.7(a) requires complete and accurate information. The inclusion of damaged fuel in this Table implies that analysis for damaged fuel has been included in the SAR.

<u>NAC Response</u>

Table 1-1 is revised to delete the damaged fuel terminology.

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1-2 Specify Zircaloy clad fuel rods in the Standard Fuel description.

Zircaloy clad rods were not specified in the Standard Fuel description. The loading tables are base on Zircaloy clad fuel. Section 71.7(a) requires complete and accurate information.

NAC Response

The definition of Standard Fuel, presented in Table 1-1, is revised to show that the design basis fuel characteristics and analysis are based on Zircaloy clad fuel rods in a standard fuel configuration.

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1-3 Remove the statement on pg. 1-1 of the application to clarify that the transport index for nuclear criticality was based upon both the optimum internal moderation and the optimum external moderation.

In justifying a transport index of zero, NAC's analysis has shown that an infinite number of packages with optimum external <u>and internal</u> moderation would remain subcritical.

Section 71.59 establishes the requirements for determining the transport index based on the evaluation of package arrays for normal and accident conditions. Standard Review Plan (SRP) Section 6.5.6.1 states that the evaluation of accident arrays should consider the most reactive configurations of water inleakage and internal moderation.

<u>NAC Response</u>

Section 1.0 is revised to show that the Transport Index for criticality control is zero based on consideration of an infinite number of packages with optimum internal and external moderation.

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Section 1.2.1.2.1 Cask Body

1-4 Describe the function and configuration of the heat transfer fins in the neutron shield.

There is no description of the fins or heat transfer function in this Section. The description should include dimensions of the fins and material composition. Section 71.43(f) provides that a package must be designed to limit radiation exposure and ensure there is no substantial reduction in the effectiveness of the package. Section 71.33 and SRP Section 1.5.3.2 describe that the application must include a description which contains a sufficient basis for evaluation.

NAC Response

Section 1.2.1.2.1 is revised to incorporate a discussion of the configuration, placement and function of the heat transfer fins.

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Section 1.2.1.2.9 Transportable Storage Canister Cask Cavity Spacers

1-5 Verify the lengths of the Universal Transport Cask spacers for each of the five fuel class canister/spacer configurations. Also, verify that the spacer(s) and Transportable Storage Canister (TSC), or the Greater Than Class C (GTCC) waste canister, will fit into the cavity of the Universal Transport Cask. Change Drawing No. 790-520 and/or the SAR text, as appropriate, to reflect the correct spacer length.

Section 71.7(a) requires that the SAR contain complete and accurate information. This SAR Section indicates that the lengths of the Class 1 and Class 2 pressurized water reactor (PWR) spacers are 7.67 and 16.75 inches, respectively. However, there appears to be an inconsistency with Drawing No. 790-520, which shows the lengths to be 11.25 and 18.25 inches.

NAC Response

The spacer lengths reported in this RAI are taken from Revision 1 of Drawing 790-520. This drawing was revised in June 1999 (Revision UMST-99A) to incorporate a Class 2 canister spacer length of 7.65 inches, and a Class 1 canister spacer length of 16.75 inches.

Section 1.2.1.2.9 is revised to correct a transcription error where the Class 1 and Class 2 spacer lengths are presented in reverse order. The correct spacer lengths are 16.75 inches for the Class 1 canister and 7.65-inches for the Class 2 canister.

Section 1.3.1.1.2 is revised to show the spacer length (16.75-inches) required for the Maine Yankee Greater Than Class C waste. As reported in that section, the Greater Than Class C waste canister has the same external dimensions as the PWR Class 1 canister. Consequently, the spacer for the Class 1 canister is required.

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Section 1.2.3 Contents of Packaging

1-6 Clarify whether unenriched fuel assemblies are intended to be acceptable contents.

The first sentence of the third paragraph from the bottom of page 1.2-14 is ambiguous with respect to the treatment of unenriched fuel assemblies. Section 71.7(a) requires complete and accurate information.

NAC Response

The Section 1.2.3 description of BWR fuel is revised to clarify that the minimum enrichment for BWR fuel is 1.9 wt % ^{235}U and that unenriched fuel assemblies can not be loaded in the transportable storage canister. Provision is made for loading BWR fuel assemblies that have unenriched axial blankets. These fuel assemblies have an enriched central fuel region and unenriched fuel blankets that occupy the outermost 6 inches of the active fuel region. The central fuel region must have a minimum enrichment of 1.9 wt % ^{235}U.

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1-7 Include the minimum initial enrichment in the table (pg. 1.2-15) that describes cask spent fuel contents. Additionally, Tables 1.2-4 and 1.2-5 list only maximum enrichments and also should include minimum enrichments.

The minimum enrichment, in combination with other specifications, is a parameter on which the bounding source term and heat load are determined. As such, it should be included in the SAR. Section 71.33 requires that the application include a description with sufficient detail to identify the package accurately and provide a sufficient basis for evaluation of the package.

NAC Response

The minimum initial enrichment (1.9 wt % ^{235}U) specification is added to Note 1 of Tables 1.2-4 and 1.2-5 and to the table at Item 8 of Section 1.2.3.

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1-8 Include a footnote for clarification and consistency between the table on pg. 1.2-15 which states that the minimum cooling time is 6 years and Table 1.2-6, which lists a minimum cool time of 5 years for high enriched, low burnup fuel.

Section 71.7(a) requires complete and accurate information.

<u>NAC Response</u>

The table shown on page 1.2-15 (Item 8 in Section 1.2.3) is revised to show a limiting minimum cool time of 5 years for PWR fuel to make it consistent with Table 1.2-6.

See the response to RAI 1-7.

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1-9 Justify the shipment of fuels with a burnup greater than 45,000 MWd/MTU. The justification should include the quantitative effects zirconium hydriding may have on the mechanical properties of the cladding (i.e., tensile strength, yield strength, ductility, fracture toughness, etc.) under normal conditions of transport. Consideration should be given to the loads on the cladding associated with both the vibration normally incident to transport and the normal condition free drop from 1 foot. Consideration should also be given to (1) the potential for dissolution of any existing zirconium hydrides during the short-term higher temperatures encountered during transportation, (2) the potential for re-precipitation and/or re-orientation of the hydrides if the temperature of the cladding decreases during transportation, and (3) the impact that zirconium hydrides may have on cladding integrity under normal transport condition loads.

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. The staff needs this information to complete the containment review required by Section 71.51. The amount of hydrogen picked up by the Zircaloy cladding during reactor operation may effect the mechanical properties of high burnup fuel prior to and during transportation. Under normal conditions of transport, the loads associated with vibration and/or a free drop from 1 foot may impart loads that could disrupt the integrity of the fuel cladding and lead to an unanalyzed containment situation.

NAC Response

A report on Maine Yankee fuel with burnups up to 50,000 MWD/MTU (Summary) has been previously provided in conjunction with the NAC response to RAIs received on December 21, 1999, for the NAC-UMS® Universal Storage System (Docket 72-1015). Publicly available DOE-sponsored research studies on high burnup fuel have measured irradiated Zircaloy material properties. These studies show that even at burnups over 50,000 MWD/MTU, Zircaloy cladding has adequate material strength and ductility to maintain the fuel rod integrity throughout all

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NAC Response to RAI 1-9 (Continued)

conditions of storage. The technical details of the DOE-sponsored research studies include hot cell examinations of actual irradiated fuel from the Fort Calhoun (Garde) and Oconee (Newman) reactors.

These reports provide information and data on actual PWR fuel with burnups greater than 50,000 MWD/MTU (to 51,700 MWD/MTU). These reports show that Zircaloy cladding maintains adequate strength, even to the 50,000 MWD/MTU burnup level. More specifically, the material ductility decreases with burnup while the yield and ultimate strengths increase with burnup.

All of the fuel investigated in the studies was initially stored in a wet pool, placed in transport casks and shipped to the hot cell laboratory in the dry condition. The results of the hot cell evaluations do not reflect the postulated dissolution of hydrides, potential re-precipitation or re-orientation of hydrides. The transport condition of the evaluated spent fuel was similar to that which occurs in the transport of the loaded canister, in that the spent fuel was vacuum dried and the cask backfilled with helium, prior to transport.

There are ninety (90) Maine Yankee fuel assemblies that have incurred a burnup between 45,000 and 50,000 MWD/MTU. The high burnup assemblies are similar to the other Maine Yankee fuel planned to be placed in dry storage (i.e., those with a burnup less than 45,000 MWD/MTU, but that have design features that support the high burnup objective).

The Combustion Engineering 14 x 14 high burnup fuel assemblies incorporate a lower (fuel rod) internal pressure and a greater cladding thickness than the UMS design basis fuel. These design differences result in lower cladding stress throughout the CE 14x14 assemblies' reactor and storage life. The greater cladding thickness, together with a larger fuel rod diameter, provides additional margin against regulatory limits. Some of the fuel assemblies have a "low tin" Zircaloy cladding, which results in lower hydrogen pick-up in the cladding and a lower cladding oxide layer thickness than for standard Zircaloy cladding fuel assemblies.

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NAC Response to RAI 1-9 (Continued)

The published reports conclude that there is an increase in the yield and ultimate strengths of the high burnup fuel rod Zircaloy cladding, with an oxide layer thickness less than or equal to 80 microns, while the material ductility decreases. The Fort Calhoun and Oconee fuel rods examined had maximum burnups up to 55,700 and 54,800 MWD/MTU, respectively. Localized burnup of these fuel rods reached over 60,000 MWD/MTU, which encompasses the burnups of the Maine Yankee high burnup fuel. Tables 17, 18 and 19 of the Fort Calhoun report (DOE/ET/34030-11) demonstrate that, for this high burnup fuel, there is a significant increase in the yield and ultimate strengths of the Zircaloy cladding with a corresponding decrease in the material ductility (plastic strain). This is further confirmed in the Oconee fuel examination report (DOE/ET/34212-50) in Table 20. These studies show that the Zircaloy material property changes occur during the early stages of irradiation and do not change significantly during the higher burnup periods. The Fort Calhoun and Maine Yankee fuels are essentially identical and are fabricated by the same supplier (Combustion Engineering). Therefore, it is concluded that the Maine Yankee high burnup fuel (45,000 < Burnup < 50,000 MWD/MTU) Zircaloy cladding ultimate and yield strengths are greater than those of standard burnup fuel assemblies, while maintaining adequate material ductility to perform its design functions.

The Maine Yankee high burnup fuel assemblies were fabricated according to their respective fuel specifications without any discrepancies or deviations that affected cladding. Review of Plant Operating Data did not identify any instances where the fuel had been subjected to any unanalyzed events that could potentially lead to excessive cladding stress.

Review of fuel inspection records and video tapes of the Maine Yankee high burnup fuel assemblies shows that the fuel is essentially identical to Maine Yankee fuel that is burned less than 45,000 MWD/MTU, with no evidence of damage or excessive cladding oxidation.

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NAC Response to RAI 1-9 (Continued)

The supporting data and information demonstrates that the physical and mechanical characteristics of the Maine Yankee high burnup fuel assemblies are essentially identical to those of the Maine Yankee fuel assemblies with burnup less than 45,000 MWD/MTU. Consequently, the Maine Yankee high burnup fuel assemblies can be safely transported in the NAC-UMS® System.

References:

Summary Report on Maine Yankee High Burnup Fuel (Burnup between 45,000 and 50,000 MWD/MTU) (NAC Proprietary).

Garde, M., "Hot Cell Examination of Extended Burnup Fuel from Fort Calhoun," DOE/ET/34030-11, CEND-427A, September 1986.

Newman, L. M., "The Hot Cell Examination of Oconee 1 Fuel Rods after Five Cycles of Irradiation," DOE/ET/34212-50, BAW-1874L, October 1986.

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Section 1.2.3 Contents of Packaging

1-10 Explain the reason for the two different minimum cooling times. In this Section, the minimum cooling time is stated as 6 years, however, in the shielding analyses Chapter the analyses for design basis fuel uses 10 years minimum cooling time.

The different cooling times combined with various fuel burnups will result in different source terms. Section 71.33 requires the identification of the maximum radioactivity of the package constituents.

NAC Response

The minimum cooling times shown in Section 1.2.3 (Item 8) are the shortest cooling times of any of the fuel designs that are evaluated. The values shown are taken from Tables 1.2-6 and 1.2-7 for PWR and BWR fuel, respectively. These tables relate fuel assembly enrichment, burnup and cool time to establish the allowable fuel loading. Note: See RAI 1-8 where the PWR minimum cool time has been changed to 5 years.

Item 8 shows the limiting values for the parameters that are reported; however, these values do not necessarily correspond to any particular fuel design. For example, the PWR design basis fuel for shielding is the Westinghouse 17 x 17 OFA having an enrichment of 3.7 wt% ^{235}U, a burnup of 45,000 MWD/MTU and a cool time of 10 years. These parameters do not appear in Item 8, because none are a limiting value for a parameter that is reported.

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Section 1.2.3 Contents of Packaging

1-11 Provide a table, suitable for inclusion in a Certificate of Compliance, that tabulates the authorized contents of the package. The table should address all requested contents, including variable minimum cooling times, maximum decay heat per assembly, Maine Yankee site-specific fuels (see related RAI 1-12), and the type, form, and maximum quantities of GTCC waste. The contents specified in the table should be consistent with the thermal, shielding, and criticality analyses performed for the package.

Section 71.33(b) requires the application to include a description of the proposed contents in sufficient detail to provide a sufficient basis for evaluation of the package. SRP Section 1.5.2.3 provides guidance that the contents in the package description are sufficient to allow an understanding of exactly what is to be transported.

<u>NAC Response</u>

A draft contents description is provided in Attachment A to these responses. At the time of transport, the canister has already been loaded in accordance with the requirements presented in Chapter 12 of the Safety Analysis Report for the NAC-UMS® Universal Storage System (Docket No. 72-1015). Therefore, any required provisions for preferential loading based on decay heat or on fuel configuration have already been met. Consequently, the limiting cool times for the fuel shown in the transport loading tables are based on the decay heat limit for the transport cask.

Essentially, two sets of limiting cool times are established for the Maine Yankee fuel configurations. The first is for standard fuel assemblies that do not hold additional non-fuel hardware, such as thimble plugs, control components, or replacement rods, that are installed in the Class 1 or Class 2 canister. The second is for fuel assemblies that are changed from the standard configuration. This fuel may also be in the Class 1 or Class 2 canister. For this modified fuel, the bounding cool times are established for fuel that holds CEA control components. Consequently, the cool times for fuel assemblies holding CEAs are used.

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1-12 Provide a table, suitable for inclusion in a Certificate of Compliance, that tabulates the authorized contents of the Maine Yankee site-specific contents. The table may reference the principal characteristics of the standard design basis 14 x 14 fuel assemblies. The table should establish the differences from the standard design basis assemblies, include burnup specifications and the applicability of the variable cooling time loading tables, and define the configurations of the site-specific fuel assemblies that have been shown by analysis to be acceptable.

Section 71.33(b) requires the application to include a description of the proposed contents in sufficient detail to provide a sufficient basis for evaluation of the package. SRP Section 1.5.2.3 provides guidance that the contents in the package description are sufficient to allow an understanding of exactly what is to be transported.

NAC Response

A draft contents description is provided in Attachment A to these responses.

See the NAC Response to RAI 1-11.

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Tables 1.2-6 & -7 Loading Tables for PWR and BWR Fuels

1-13 Provide additional information on the validity of these tables. These tables identify fuels with a wide range of burnup and cool times that could be loaded into the cask. Additionally, a cool time of 5 years is allowed.

For example, this time frame is inconsistent with information provided in Section 1.2.3 and the design basis fuel identified in Chapter 5. Section 71.43(f) provides that a package must be designed to limit radiation exposure and ensure there is no substantial reduction in the effectiveness of the package.

<u>NAC Response</u>

The detailed evaluations that substantiate Tables 1.2-6 and 1.2-7 are provided in Section 5.4-3 for the UMS Universal Transport Cask design basis spent fuel, and in Section 5.5-1 for the Maine Yankee site specific spent fuel.

The Maine Yankee site specific spent fuel evaluation is based on the use of the Combustion Engineering (CE) 14 x 14 fuel assembly in the reactor. The CE 14 x 14 has a smaller fuel mass compared to the UMS Universal Transport Cask design basis fuel. Therefore, it has a lower source term and a shorter minimum cool time.

As described in the response to RAI 1-7, the cask spent fuel contents presented in Item 8 of Section 1.2.3 is revised to show a minimum cool time of 5 years for PWR fuel.

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Section 1.3.1.1.1 Maine Yankee Site-Specific Spent Fuel Configurations

1-14 Provide a description of how these spent fuel assemblies have been determined to be acceptable with respect to maximum activity.

Page 1.3.1-3 of the SAR lists site-specific fuel assembly configurations that meet the criteria for acceptable contents. However, no analysis is presented or referenced. Section 71.33 requires the identification of the maximum radioactivity of the package constituents.

NAC Response

Section 1.3.1.1.1, and other Sections in Chapter 1, are intended to provide only a general description of the UMS Universal Transport Cask contents and contents configurations. As shown on page 1.3.1-4, the shielding source term evaluations for Maine Yankee contents are provided in 5.5.1.1. Section 5.5.1.1 also contains a description of how the contents are bounded by the design basis fuel evaluation.

The shielding evaluation consists of a loading table analysis of the CE14x14 fuel using the methodology presented in Section 5.4.3. Fuel assemblies with inserted non-fuel hardware are addressed explicitly.

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Section 1.3.4 License Drawings

1-15 Include detail E-E from Drawing No. 790-502.

If detail E-E does not show the Cu-SS fins, include an annular cross-sectional view of the cask body, including the fins and their relationship with surrounding components, including the lead and neutron shields. Section 71.33 requires sufficient package detail to provide a basis for evaluation of the package.

NAC Response

Drawing Number 790-502 is revised to incorporate Detail E-E. The heat transfer fins are also shown in Details H-H and M-M on the same drawing. As shown in these details, the stainless steel plate section of the heat transfer fin is welded to the cask body outer shell and to the neutron shield shell. The copper plate section of the fin provides improved heat transfer across the neutron shield annulus. The fins are not connected to, and do not go through, the lead gamma shield or the lead gamma shield annulus.

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Section 1.3.4 License Drawings

1-16 Clarify the term "Out-of-Spec" with respect to boiling water reactor (BWR) fuel tubes. Update the appropriate Sections of the SAR to explain the meaning.

Section 71.7(a) requires that the SAR contain complete and accurate information. The term "Out-of-Spec" is used on Drawing Nos. 790-501 and 790-605, but the meaning of the term is not explained.

NAC Response

The term "Out-of-Spec" refers to the condition of the enlarged size of four of the fuel tubes in the BWR fuel basket. To accommodate BWR fuel channels that may be distorted by small amounts, the four corner fuel assembly storage positions have fuel tubes that are slightly larger than the other fuel tubes in the basket. These four slightly larger fuel tubes were titled, "BWR Fuel Tube, Out-of-Spec Fuel" in Drawing 790-605 to provide a unique identification.

Section 1.2.1.2.8 is revised to clarify the purpose of the "oversized" positions in the UMS BWR fuel basket. The title of Drawing 790-605 is revised to: "BWR Fuel Tube, Over-Sized Fuel, NAC-UMS." Drawing 790-501 is also revised to refer to Over-Sized Fuel.

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Section 1.3.4 License Drawings

1-17 Clarify which paint coating will be applied to the BWR support disks. Revise the appropriate SAR Sections, the Drawings, or both, to reflect the correct choice of paint coating.

Section 71.7(a) requires that the SAR contain complete and accurate information. Drawing No. 790-573 indicates that either a Keeler & Long Hi-Heat Silicone Aluminum No. 3731 or an Ameron Amercoat 878 silicone coating will be applied. However, this specification is inconsistent with the text of Section 2.4.4.2.10 which indicates that an Amercoat PSX 738 coating will be applied.

NAC Response

The coating on the BWR support plates is revised to be electroless nickel (See Drawing 790-573). The evaluation of the chemical and galvanic reactions of the electroless nickel coating on the BWR support disks is provided in Section 3.4.1.2.2 of the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015. A detailed description of the electroless nickel coating process is provided in Section 3.8.3 of that document.

The electroless nickel coating process uses a chemical reducing agent in a hot aqueous solution to deposit nickel on a catalytic surface. The nickel coating combines with oxygen in the air to form a passive oxide layer that effectively eliminates free electrons on the surface that would be available to cathodically react with water to produce hydrogen gas. The coating is plated on the carbon steel disk in accordance with ASTM specification B733-SC3, Type V, Class 1.

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Section 1.3.4 License Drawings

1-18 Revise Drawing No. 790-502, sheet 4, by adding a note on the torque value for Item 26 (threaded inserts) for bolting the secondary lifting trunnions to the cask.

Complete and accurate information should be listed on licensing drawings, per Section 71.7(a).

NAC Response

Item 26 is a commercially purchased Helicoil® threaded insert. The purpose of the insert is to eliminate the possibility of galling between the threads cut in the bolt holes in the top forging of the cask body and the bolts that secure the secondary lifting trunnions. Helicoil® and other vendor inserts are used for this purpose in most packages that have a requirement for large torque values on attachment bolts. The threaded inserts are installed in accordance with the manufacturer's instructions.

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Section 1.3.4 License Drawings

1-19 Revise Drawing No. 790-502, sheets 3 and 4, by noting the filler material grades used for welding the shear rings, trunnions, and rotation pockets to the cask.

The structural analyses of SAR Section 2.5, which demonstrates compliance with 71.45(b)(3), relies on the consideration of specific weld strengths to ensure that, under excessive load, failure of tie-down devices would not impair the ability of the package to meet other 10 CFR Part 71 requirements.

NAC Response

Drawing 790-502 is revised to note the use of AWS E309 filler material for trunnion and rotation pocket welds, and AWS E316L filler material at the shear ring. Filler material grades are selected and used in accordance with the ASME Code, Section II, Part C.

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Section 1.3.4 License Drawings

1-20 Verify thicknesses for the BORAL panels used in the BWR and PWR baskets, as shown in Drawing Nos. 790-575 and 790-581.

NAC's response should indicate the correct panel thickness, the effective thickness of the poison meat, and the volume fraction, boron enrichment, and size distribution of the B_4C particles. A BORAL panel of the thickness (0.075 inch) and minimum poison loading (0.025 g $^{10}B/cm^2$) specified for the PWR baskets would appear to call for an impossibly large volume fraction of B_4C particles in the poison meat. This, and the fact that the BWR-basket panels are thicker (0.135 inch) with a lower poison loading (0.011 g $^{10}B/cm^2$), suggest that the panels in the PWR basket may need to be thicker than shown in Drawing No. 790-581.

Section 71.33(a)(5)(ii) requires an accurate description of neutron absorbers. SRP Section 6.5.1.1 calls for reviewing the dimensions and concentrations of neutron poisons.

NAC Response

NAC has verified that the specified thicknesses of the BORAL sheets for the PWR and BWR fuel tubes are correct as shown on Drawings 790-575 and 790-581. The BWR fuel tube BORAL panel is significantly thicker than the PWR panel (0.135 versus 0.075 inches, respectively) to take advantage of the heat transfer properties of the aluminum in the BORAL panels. The thicker panel in the BWR fuel tube improves heat rejection in the axial direction, thus improving the thermal performance of the BWR basket design.

The vendor of the BORAL product, AAR Manufacturing, Inc., has verified that the BORAL sheets can be fabricated in any thickness from 0.070 to 0.375 inches and that the specified ^{10}B areal densities can be obtained. See the response to RAI 8-3.

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Note: a dual-purpose canister system has previously been approved with significantly higher ^{10}B loading for an equivalent thickness. These loadings were specified as a minimum of 0.0267 g/cm^2 in a 0.075 inch thickness and 0.0372 g/cm^2 in a 0.101 inch thickness.

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Section 1.3.4 License Drawings

1-21 Revise Drawing No. 790-509 to reflect a package identification number with a "-85" suffix.

Section 71.85(c) requires the durable marking, prior to first use, of the package with the assigned package identification number. SRP Section 1.5.2.4 specifies that new transportation packages for spent fuel will be assigned a "-85" suffix.

NAC Response

The Body Nameplate shown in Drawing 790-509 is revised to include the "-85" designation in the package serial number.

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Section 1.3.4 License Drawings

1-22 The application contains 32 fabrication drawings. The drawings show many details and dimensions which are not important to safety. The application should be revised to include only those engineering drawings that are suitable for reference in a Certificate of Compliance. The drawings should show the general arrangement of the package, its principal features, and the design of safety related components. The drawings should contain the following information:

1. dimensions and tolerances important to safety,
2. materials of construction,
3. applicable codes and standard for fabrication of safety related components,
4. size, type, and location of safety related welds and methods of non-destructive examination,
5. torque requirements for closure bolts and other threaded devices,
6. total weight of the package, and
7. location of the tamper indicating features.

Section 71.33 requires sufficient package detail to provide a basis for evaluation of the package. NUREG/CR-5502, "Engineering Drawings for 10 CFR Part 71 Package Approvals," provides additional information for the preparation of drawings.

NAC Response

NAC maintains separate files for License Drawings and Engineering Drawings. The License Drawings have considerably less detail than is presented in the Engineering Drawings, which are used as the basis for fabrication.

Based on a more than 20-year history of cask certification, NAC's License Drawings present a balance between the minimum information that could be presented and the requirements of individual NRC reviewers for information considered important to accurately evaluate the system. NAC is striving to reduce the information presented in the License Drawings;

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NAC Response to RAI 1-22 (Continued)

however, we want to be sure that a reviewer does not have to use the RAI process in order to obtain information historically shown on the License Drawings.

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CHAPTER 2: STRUCTURAL

Section 2.1 Structural Design

2-1 Provide a summary description of the codes and standards and their exceptions applicable to the structural design of the GTCC waste canister and basket.

Complete and accurate information on the GTCC waste basket should also be presented, per Section 71.7(a).

NAC Response

The GTCC waste canister is constructed to the same codes and standards as the canister for spent fuel. The GTCC waste canister has the same external dimensions and is closed and sealed using the same procedures as for the spent fuel canister. The GTCC waste canister basket is fabricated in accordance with Subsection NF of Section III of the ASME Code. It is evaluated in accordance with the load combinations of Regulatory Guide 7.8, and is evaluated against the buckling criteria of NUREG/CR-6322. A brief description of these criteria is added as Section 2.1.1.5. These criteria are similar to those applied to the spent fuel basket, except that the spent fuel basket is designed and fabricated in accordance with Subsection NG of Section III of the ASME Code.

Table 2.1.2-1 is revised to include the code exceptions for the GTCC canister and basket. As noted above, since the waste canister and the spent fuel canister are designed to the same criteria, the ASME Code exceptions are the same for all canisters.

See the response to RAI 2-2.

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Section 2.1.2 Design Criteria

2-2 Update Table 2.1.2-1 to reflect the pertinent ASME Code Exceptions as noted in Table 4-1 of the NAC-UMS Storage SAR for the TSC.

Section 71.7(a) requires complete and accurate information. Drawing Nos. 790-585 and -612 specify a progressive liquid penetrant nondestructive examination, or an ultrasonic examination, of the TSC and GTCC waste canister structural lid-to-shell welds. However, the Drawings are inconsistent with Table 2.1.2-l, which specifies a root and final liquid penetrant examination.

NAC Response

Table 2.1.2-1 is revised to incorporate the ASME Code exceptions previously specified in the NAC-UMS® Storage Safety Analysis Report for the transportable storage canister (TSC) and the spent fuel basket. In addition, the exceptions to the ASME Code for the GTCC waste canister basket are added to the table.

Among other clarifications, the exception to the examination of the structural lid to canister shell weld is revised to include the option of performing an ultrasonic inspection of the weld.

See the Response to RAI 2-1.

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Section 2.1.2.3 Load Combinations

2-3 Define for the cask cavity the maximum normal operating pressure (MNOP) considered in the following SAR text:

1. Pgs. 2.6-4 & 2.6-14, "[T]he loading conditions are: (1) 50 psig internal pressure..."
2. Table 2.6.1.1-2, "[N]ormal Design Pressure--Cask...25 psig..."
3. Pg. 2.10.2-6, "[A] pressure of 150 psig is used to conservatively envelope the normal design pressure of 25 psig for all impact loadings..."
4. Pg. 8.1-4, "[T]he transport cask containment is hydrostatically tested to 85 psig...The containment maximum normal operating pressure (MNOP) is calculated to be 8.5 psig."

Per Section 71.33(b)(5), the MNOP shall be identified.

NAC Response

The pressure used to evaluate the UMS Transport Cask for normal and accident conditions is 150 psig. Various sections of the Safety Analysis Report are revised to clarify the analyzed cask pressure. Tables 2.6.1.1-2 and 2.6.1.1-3 are revised to show an "analyzed" pressure of the Transportable Storage Canister and Transport Cask to be 25 psig and 150 psig, respectively. The Maximum Normal Operating Pressure (MNOP) is specified in Sections 2.6.1.1, 2.6.2.1, 8.1.2.3 and 8.1.2.3 to be 7.3 psig, where the MNOP is as defined by NUREG-1617, Table 4-1.

See the NAC Response to RAIs 2-20, 2-39 and 2-49.

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Section 2.1.2.3 Load Combinations

2-4 Clarify, on Pg. 2.1-9, how for the lead-pour fabrication, -40°F cold test and –20°F ambient thermal stresses are considered in the load combination structural evaluations of the cask inner shell.

Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package performance under the conditions and tests of Sections 71.71 and 71.73.

NAC Response

Section 2.1.2.3 is revised to incorporate a summary of the evaluation of stress associated with the lead pour that occurs in fabrication. The evaluation of stress resulting from lead-pour is presented in Section 2.6.11. The maximum stress in the inner shell during lead solidification is insignificant (only 255 psi). Since the residual stress in the inner shell induced by the shrinkage of the lead after the pouring operation is relieved because of the low creep strength of the lead, this stress is not considered with other loads. There are no other unrelieved stresses that occur in fabrication.

The evaluation of the -40°F cold condition is provided in Section 2.6.2 (Cold). The load combination considered for this case is (1) internal pressure with bolt-preload (2) -40°F ambient with no solar insolation and no decay heat, and (3) gravity.

The -20°F ambient thermal condition is bounded by the "Thermal Cold" condition (-40°F ambient temperature, no solar insolation and maximum decay heat), as described in Section 2.10.2. The "Thermal Cold" condition is combined with internal pressure, bolt pre-load and inertia/impact loads in the evaluation of cask impacts (Sections 2.6.7 and 2.7.1).

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Section 2.10.2 is also revised to clarify the load combinations considered in the analysis. Table 2.1.2-2 is revised to correct several typographical errors and to specify the load combinations that must be evaluated for normal and accident conditions.

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Section 2.1.2.3 Load Combinations

2-5 Clarify, on Pg. 2.1-11, the statement, "[t]he visco-elastic behavior of the lead is considered. . .in the analysis of cask shell components."

The Section 2.7.1.5 discussion on lead slump does not appear to refer to the assumption of visco-elastic material behavior of the lead for the cask shell stress analysis. Complete and accurate information should be provided, per Section 71.7(a).

NAC Response

The subject sentence in Section 2.1.2.4 is revised to delete reference to "visco-elastic behavior," and to refer instead to the low yield strength and the weight of the lead.

The analysis considers the low yield strength of the lead in that it is assumed that the lead does not contribute to the structural strength of the cask. The effect of the weight of the lead is considered in the analysis of the cask shell components.

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Section 2.1.2.3 Load Combinations

2-6 Clarify in Table 2.1.2-1, the code exception to ASME Article NB-6000, Testing, "[T]ransportable storage canister cannot have a hydrostatic or pneumatic test performed..."

The statement on pg. 8.1-7, "[T]he canister is conservatively pressure tested... 1.2 times the 15 psig design pressure..." appears to be in disagreement with the exception taken in SAR Table 2.1.2-1. Complete and accurate information should be provided in the SAR, per Section 71.7(a), for evaluating packaging codes and standards compliance, per Section 71.31(c).

<u>NAC Response</u>

The ASME Code provides that the pressure vessel is pressure tested to confirm its integrity immediately following fabrication and prior to first use. The design of the Transportable Storage Canister is such that closure of the canister (pressure vessel) does not occur until the canister is loaded with the spent fuel or radioactive waste that it confines. Consequently, pressure testing occurs coincident with loading, but prior to completion of closure and initiation of first use.

The intent of the referenced language was to differentiate the order in which use and testing occurs. Table 2.1.2-1 is revised to clarify this exception.

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Section 2.1.2.5.2 Fatigue

2-7 Revise this Section and Sections 2.6.7.6 and 2.7.1.7.2, as appropriate to also consider bolt pre-loads on the cask lid closure bolts for meeting the fatigue evaluation exemption criteria of ASME NB-3222.4(d) and 3232.3(b), including provisions for fatigue strength reduction factors of NB-3232.3(c) for threaded members.

The closure bolt fatigue life needs to be evaluated, and a lower fatigue strength should be considered for the bolt. The effect of repeated use of the closure bolts under normal conditions of transport, per Section 71.71(a), should be included in the evaluation.

<u>NAC Response</u>

The closure bolt fatigue analysis presented in Section 2.7.1.7.2 is deleted and a revised bolt fatigue analysis is presented in 2.6.7.6.1. The fatigue life of the closure bolts is evaluated using ASME Code Section III, Appendix I. The evaluation shows that the Transport Cask lid closure bolts need to be replaced every 944 cycles (installation and removal). Assuming 24 cycles per year, the bolts need to be replaced after approximately 39 years.

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Section 2.2 Weights and Centers of Gravity

2-8 Add a table of calculated weights and centers of gravity of the Universal Transport Cask for transporting the GTCC waste.

Similar to those for the five classes of design basis fuel, complete and accurate weight and center of gravity information on the GTCC waste basket should also be provided, per Section 71.7(a). They are essential for evaluating the applicability of the bounding decelerations determined for the packaging under the free drop tests of Section 71.71(c)(7) and Section 71.73(c)(1).

<u>NAC Response</u>

Table 2.2-3 is added to Section 2.2 to provide the calculated weights and centers of gravity for the Universal Transport Cask transporting GTCC waste.

See the NAC Response to RAI 2-57.

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Section 2.3.1 Summary of Materials

2-9 Correct the gamma shield material to be chemical copper-grade lead instead of chemical lead. Section 71.7(a) requires complete and accurate information.

NAC Response

Section 2.3.1 is revised to specify the gamma shield material as chemical copper lead. This lead is provided in accordance with the 1992 edition of ASTM B29. This specification is called out on NAC-UMS Drawing 790-502.

The properties of the chemical copper grade lead are as specified in Table 1 of ASTM B29-92, "Standard Specification for Refined Lead."

See the response to RAI 2-10.

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Section 2.3.1 Summary of Materials

2-10 Revise the table on pg. 2.3-2 to specify the correct grade of lead that will be used for gamma shielding.

Section 71.7(a) requires complete and accurate information. This Section does not indicate which grade of lead will be used for gamma shielding. In Table 2.3.7-1 and on Drawing No. 790-502, it appears that a "Chemical-Copper Grade" lead will be used.

NAC Response

Section 2.3.1 is revised to specify the gamma shield material as chemical copper grade lead. This lead is provided in accordance with the 1992 edition of ASTM B29. This specification is called out on NAC-UMS Drawing 790-502.

The properties of the chemical copper grade lead are as specified in Table 1 of ASTM B29-92, "Standard Specification for Refined Lead."

See the response to RAI 2-9.

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Section 2.3.7 Shielding Material

2-11 In Table 2.3.7-1, correct the values of the "Tensile Yield Strength." Also, indicate the source of the data in the table.

Section 71.7(a) requires complete and accurate information. It appears that the decimal point was inadvertently misplaced.

NAC Response

Table 2.3.7-1 is revised to correct the error in the 600°F value for Yield Stress and to correct the category title to "Tensile Yield Strength." The references specified in the table are the source for the respective property values. The references are given in Section 2.12.

Chemical copper grade lead is the same material as the previously specified chemical grade lead and, therefore, has the same physical properties. Consequently, the values obtained from the references shown in Table 2.3.7-1 are correct for chemical copper grade lead.

See the response to RAIs 2-9 and 2-10.

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Section 2.4.4 Chemical, Galvanic, or Other Reactions

2-12 Demonstrate that the placement of the Universal Transport Cask into the spent fuel pool, when the cask is wet loaded, will not impact safety during fuel loading operations. Consider the effects of chemical, galvanic, or other reactions between the cask materials, including any coatings, and the spent fuel pool water. The potential for generation of combustible gases should be addressed in this evaluation. Revise the operating procedures to include appropriate controls for detecting the presence, and preventing the ignition, of any combustible gases that may be generated during cask loading operations (See RAI 7-5).

Section 71.43(d) requires that a package be made of materials that assure there will be no significant chemical or galvanic reactions. Reaction of the Universal Transport Cask components or coatings with spent fuel pool water may produce hydrogen or other flammable gases. Since the shield lids of the TSC and GTCC waste containers are welded to their shells during fuel loading operations, there is a source of heat that could lead to their ignition if sufficient flammable gas is present.

NAC Response

Section 2.4.4, and the remainder of the Safety Analysis Report, are revised throughout to delete references to the in-pool loading of the Transport Cask. There is no current operational requirement for direct loading of the transport cask. The Transport Cask is loaded dry with the Transportable Storage Canister and the cask cavity is subsequently evacuated and backfilled with helium. There is no significant potential for chemical, galvanic or other reactions to occur between the stainless steel cask cavity components and the stainless steel canister during loading of the Transportable Storage Canister into the Transport Cask or during subsequent transport.

See the NAC Response to RAI's 2-13, 2-14, 2-15 and 2-16.

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Section 2.4.4 Chemical, Galvanic, or Other Reactions

2-13 Evaluate the potential for the generation of combustible or reactive gases in both the TSC containing Maine Yankee site specific contents and the GTCC waste canister. As part of the evaluation, identify any nonmetallic materials that will be contained in either of the canisters. If particulate material will be placed inside either canister, identify the material of the particles and determine the range of particle sizes.

Section 71.43(d) requires that a package be made of materials that assure there will be no significant chemical or galvanic reactions. Reaction of unusual materials (e.g., non-stainless steel or non-Zircaloy materials, plastics, resins, etc.) with spent fuel pool water or the radiation and moderately high temperatures of the cask environment may produce flammable or combustible gases. Since the shield lids of the TSC and GTCC waste canister are welded to their shells during loading operations, there is a source of heat that could lead to ignition if sufficient amounts of flammable or combustible gas are present.

NAC Response

As noted in the NAC Response to RAI 2-12 and others, reference to the in-pool loading of the canister using the UMS® Universal Transport Cask is deleted. Consequently, at the time of transport, the canister will have previously been loaded, closed and tested in accordance with the detailed requirements presented in the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015. While the GTCC waste packaging is not controlled by the requirements of 10 CFR 72, the GTCC waste canister is closed using essentially the same procedure as is used for closing canisters containing spent fuel as described in the response to RAI DP4-1. No welds are made or repaired while the canister is inside of the Transport Cask.

The Transport Cask is loaded dry with the Transportable Storage Canister and the cask cavity is subsequently evacuated and backfilled with helium. There is no significant potential for chemical, galvanic or other reactions to occur between the stainless steel cask cavity components

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and the stainless steel canister during loading of the Transportable Storage Canister into the Transport Cask or subsequent transport.

The Maine Yankee site specific contents consist of standard fuel assemblies, fuel assemblies having installed thimble plugs, control components, poison rods and hollow rods and Greater Than Class C (GTCC) waste. The thimble plugs and control components are fabricated from Type 304 stainless steel and Inconel. Thimbles have small Inconel parts, while the control component rods (which contain a neutron absorber) are Inconel. The poison rods are fabricated from Zircaloy-4. The hollow rods are also fabricated from Zircaloy-4. The GTCC waste consists of sections of the reactor core shroud. The shroud is fabricated from Type 304 stainless steel.

None of these components exhibit adverse interaction with spent fuel pool water to generate flammable or combustible gases during pool loading or canister closing operations.

The standard spent fuel assemblies, and fuel assemblies with inserted components, are installed in fuel tubes in the basket. The interior of the fuel tube is stainless steel. No adverse reactions occur, in either wet or dry conditions, between the fuel assembly components, or installed components, and the fuel tubes. Consequently, no combustible or flammable gases are formed as a result of contact between the fuel tube surfaces and the installed fuel assemblies.

The GTCC basket is entirely Type 304 stainless steel. Consequently, no adverse reactions occur, in either the wet or dry condition, between the reactor core shroud sections and the GTCC basket components.

During cutting of the core shroud, metal chips from the cutting operation were captured in a stainless steel filter assembly. The filter assembly holds a stainless steel perforated screen having 1/8-inch holes in a 3/16-inch triangular pattern. A 40-mesh stainless steel filter material is used inside of the perforated screen. This filter assembly is entirely Type 304 stainless steel. The

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minimum chip size is expected to be 0.007 x 0.06 x 1.5 inches, with all of the chips being Type 304 stainless steel material. The filter assembly contains no organic material. Since all of the filter materials, including the captured chips, are stainless steel, no adverse reactions that could generate combustible or flammable gases occur.

The shroud cutting operation may generate "swarf" – small size stainless steel particulate material. Swarf is captured using a 120-mesh stainless steel filter assembly at the saw suction discharge. This material is disposed of separately, and is not placed in the GTCC basket.

The Maine Yankee fuel can lid assembly includes an aluminum wiper that precludes the dispersal of gross particulate material. The wiper is a thin, narrow, piece of aluminum that extends around the lid and, thus, represents a very small volume of aluminum. Because Maine Yankee fuel cans are restricted to the corner locations in the PWR basket, only 4 wipers may be present in any canister. Aluminum produces a thin oxide layer that precludes further oxidation of the surface. Although aluminum in PWR pool water has the potential to produce small amounts of combustible gas (hydrogen), the volume of the gas will be insignificant from the four (4) small wipers.

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Section 2.4.4.1 Component Operating Environment

2-14 Revise this Section to include wet-loading of the Universal Transport Cask, as is indicated on pg. 3.4-44, Section 3.4.7, and pg. 7.5-1, Section 7.5.

Sections from Chapters 3 and 7 clearly indicate that the Universal Transport Cask or the transfer cask is immersed in the spent fuel pool. Section 2.4.4.1 indicates that the Universal Transport Cask is dry loaded and is not immersed in the pool. Section 1.2.2 indicates that canister loading is accomplished only by use of the transfer cask. Section 71.7(a) requires complete and accurate information.

NAC Response

Chapters 3 and 7 are revised to delete Sections 3.4.7 and 7.5, which describe the wet-loading (direct loading) provision. There is no current operational requirement for direct loading of the transport cask.

As noted in the NAC Response to RAI 2-12 and others, reference to the in-pool loading of the canister using the UMS® Universal Transport Cask is deleted. Consequently, at the time of transport, the canister will have previously been loaded, closed and tested in accordance with the detailed requirements presented in the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015.

See the response to RAI's 2-12, 2-13, 2-15 and 2-16.

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Section 2.4.4.2.1 Stainless/Nickel Alloy Steels

2-15 In the appropriate Sections, describe how aluminum bronze ferrules and ethylene glycol are used in the Universal Transport Cask, or remove the references to these components if they are not used.

Section 71.7(a) requires complete and accurate information. References are made to aluminum bronze ferrules and ethylene glycol without a description of how these materials are used in the Universal Transport Cask.

NAC Response

Section 2.4.4.2.1 is revised to delete reference to aluminum bronze ferrules and ethylene glycol. These items are not used in the UMS Universal Transport Cask.

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Section 2.4.4.2.3 Nonferrous Materials

2-16 Show that the reaction between the aluminum heat transfer disks of the TSC and the spent fuel pool water is not significant with respect to its impact on safety during fuel loading operations. Revise the operating procedures to include appropriate controls for detecting the presence, and preventing the ignition, of combustible gases during cask loading operations (See RAI 7-5). The potential for generation of combustible gases should be addressed in this evaluation. The evaluation should consider (1) the temperature of the water in the TSC will change during loading and (2) the effects of both irradiation and the contact between the aluminum heat transfer disks and the stainless steel washers, which are used to position the aluminum disks. The evaluation and conclusions should be supported by calculations, experiment, or applicable data gathered from a literature survey.

Section 71.43(d) requires that a package be made of materials that assure there will be no significant chemical or galvanic reactions. Reaction of the aluminum heat transfer disks with spent fuel pool water and/or steel components may produce hydrogen in concentrations close to the lower explosive limit of hydrogen. Since the shield lid of the TSC is welded to the shell during fuel loading operations, there is a source of heat that could lead to ignition if sufficient amounts of flammable gas are present.

NAC Response

As noted in the NAC Response to RAI 2-12 and others, reference to the in-pool loading of the canister using the UMS® Universal Transport Cask is deleted. Consequently, at the time of transport, the canister will have previously been loaded, closed and tested in accordance with the detailed requirements presented in the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015. The detailed evaluation of galvanic reactions, including the potential for the production of explosive gases during loading, is provided in Section 3.4.1 of the UMS® Universal Storage System Safety Analysis Report. As shown in that section, there is little potential for galvanic reactions or corrosion, since compatible materials are used in the design of

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the canister and basket, or the formation of explosive gases. However, as shown in the procedures used in closing the canister, monitoring of combustible gas formation is required. Compensatory actions are required to remove the gases if the amount of combustible gas generated during loading exceeds 60% of the lower flammability limit (i.e., 0.6 x 4.0 = 2.4%).

No galvanic interaction, corrosion or formation of explosive gas occurs during the dry loading of the stainless steel canister in the transport cask. Subsequent to loading the canister, the transport cask cavity is subject to a vacuum and helium backfill operation, which removes air and establishes an inert atmosphere in the transport cask. The use of similar materials (stainless steel) and the presence of the helium atmosphere essentially preclude the formation of explosive gases or materials interaction.

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Section 2.4.4.2.10 Coatings

2-17 Demonstrate that the coating applied to the BWR support disks of the TSC will not impact safety during fuel loading operations. Revise the operating procedures to include appropriate controls for detecting the presence, and preventing the ignition, of combustible gases during cask loading operations (See RAI 7-5). The potential for generation of combustible gases should be addressed in this evaluation. Also, demonstrate that the coating is not reactive and is adherent when it is exposed to PWR and BWR spent fuel pool water, radiation, and temperatures that are expected during fuel loading operations. Describe the process that was used to select the coating. Include a brief discussion of the tests and/or analyses that were conducted to qualify these coatings for use in the radiation and moderately high temperature environment. Indicate the expected impact of flaking or chipping paint on the structural integrity of the BWR support disks. Update all SAR Sections, as appropriate, to include these descriptions.

Section 71.43(d) requires that a package be made of materials that assure there will be no significant chemical or galvanic reactions. A potential reaction of the paint coating with spent fuel pool water and/or steel components may produce hydrogen or other flammable gases, or it may cause difficulty with loading the spent fuel into the cask. Since the shield lid of the TSC is welded to the shell during fuel loading, there is a source of heat that could lead to ignition if sufficient amounts of gas are present.

<u>NAC Response</u>

As noted in the NAC Response to RAI 2-12 and others, reference to the in-pool loading of the canister using the UMS® Universal Transport Cask is deleted. Consequently, at the time of transport, the canister will have previously been loaded, closed and tested in accordance with the detailed requirements presented in the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015.

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Drawing 790-573 is revised to specify an electroless nickel coating on the BWR basket carbon steel support disks. Reference to Hi-Heat Silicone Aluminum No. 3731 and Amercoat 878 silicone coatings for the BWR basket support disks is deleted.

The evaluation of the chemical and galvanic reactions of the electroless nickel coating on the BWR support disks is provided in Section 3.4.1.2.2 of the Safety Analysis Report for the UMS® Universal Storage System. A detailed description of the electroless nickel coating process, is provided in Section 3.8.3 of that document.

In summary, the coating process uses a chemical reducing agent in a hot aqueous solution to deposit nickel on a catalytic surface. The deposited nickel coating is a hard alloy of uniform thickness of 25 μm (0.001 inch), containing from 4% to 12% phosphorus. Adhesion of the nickel coating to properly cleaned carbon steel is excellent with reported bond strength in the range of 40 to 60 ksi. The coating is applied in accordance with ASTM B733-SC3, Type V, Class 1. Following its application, the nickel coating combines with oxygen in the air to form a passive oxide layer that effectively eliminates free electrons on the surface that would be available to cathodically react with water to produce hydrogen gas. Consequently, the production of hydrogen gas in sufficient quantities to facilitate combustion is highly unlikely. Test data for electroless nickel coated steel have been reported to show corrosion rates from 1 to 2 μm per year in water.

As noted in the NAC Response to RAI 2-16, the Transportable Storage Canister loading procedures require the monitoring of explosive gases, and the removal of those gases, if they exceed 60% of the lower flammability limit (i.e., 0.6 x 4.0 = 2.4%).

The process used to select the coating for the BWR support disks considered the service environment and performance requirements for the coating, the products generally available, and

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industry experience with coatings used for a similar purpose. The selection process included these steps:

1. Determine the service and performance requirements for the coating.
2. Evaluate the available coatings against the service and performance requirements.
3. Identify acceptable coatings based on requirements and availability.
4. Obtain current product data sheets and MSDS of acceptable coatings to review against coating requirements.

The service and performance requirements established for the coating of the BWR support disk design are that the coating:

- is applied to carbon steel
- must be submersible for up to a week in clean water
- is rated Service Level 1 or 2 (EPRI TR-106160)
- does not contain Zinc
- has a service temperature of at least 200°F in water and 600°F in a dry helium environment
- has no hydrogen release, or minimal hydrogen release, when submersed in clean water
- has no, or limited, special process required for proper application or curing
- has a service environment in a high radiation field

The BWR support disks will be fabricated to the proper dimensions before each disk is plated. The coating is applied immediately after cleaning and preparation of the disk surface to protect the support disk during the completion of the fabrication process and during storage of the basket. No subsequent cutting, machining or welding of the disk is required.

No plating characteristics that may enhance the performance of the support disks (such as better emissivity) are considered in the structural or thermal analyses. Therefore, no adverse effect on system performance results from incidental scratching or flaking of the plating.

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Section 2.5 Lifting and Tie-Down Standards

2-18 Justify the use of ultimate shear strength for evaluating the structural performance of the lifting and tie-down devices of the package under excessive load.

For consistency, the von Mises failure criterion, which is used in the lifting trunnion and rotation pocket stress analyses, per Section 71.45, should also be considered in the evaluation for excessive load.

NAC Response

Sections 2.5.2.4.1 and 2.5.2.4.2 are revised to consider the von Mises failure criterion for the lifting and tiedown devices. Application of the von Mises criterion shows that failure caused by excessive overload on the rotation trunnions or shear ring will not impair the ability of the package to meet the other requirements of 10 CFR 71.

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Section 2.5.1.1.2 Secondary Lifting Trunnions

2-19 Provide a bolt pre-load analysis to be consistent with the bolt stress distribution assumption depicted in Figure 2.5.1.1-2.

Complete and relevant information on bolt pre-load should be provided to substantiate the stress distribution assumption for the trunnion attachment bolts. The validity of the structural integrity evaluation of the bolted secondary trunnions, as a lifting device per Section 71.45(a), depends on an adequate amount of pre-load in the bolts.

NAC Response

Section 2.5.1.1.2 is revised to demonstrate that the bolts in the compression region depicted in Figure 2.5.2.1-2 maintain an adequate pre-load when torqued to the minimum-specified pre-load value and that the bolts in the tension region maintain a positive margin of safety when torqued to the maximum-specified pre-load value.

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Section 2.6.1 Heat

2-20 Revise the pressures listed in this Section to be consistent with the thermal Section.

The pressures stated on pgs. 2.6-1 and 2.6-2 are inconsistent with Table 3.4-4. Section 71.7(a) requires complete and accurate information.

NAC Response

Section 2.6.1 is revised to incorporate the UMS Universal Transport Cask cavity pressures shown in Table 3.4-4 for PWR fuel and BWR fuel, respectively. The analyzed internal pressure of the cask is revised to 150 psig to reflect the pressure used in the cask model described in Section 2.10.2. Reference to "design pressure" is deleted.

See the NAC Response to RAIs 2-3, 2-39 and 2-49.

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Section 2.6.1.3 Stress Calculations and Comparison to Allowable Stresses

2-21 Clarify, as appropriate, the underlined term below:

Pgs 2.6-4 and 2.6-14 of the SAR, "[T]he stresses throughout the cask body are calculated for...loading conditions for directly loaded fuel."

Complete and accurate information should be provided, per Section 71.7(a).

NAC Response

Sections 2.6.1.3 and 2.6.2.3 are revised to delete the reference to "... directly loaded fuel."

The individual and combined loading conditions specified in these sections are not a function of the contents in the Universal Transport Cask.

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Section 2.6.2 Cold

2-22 Clarify, as appropriate, the meaning of the underlined term below:

Pg. 2.6-15 of the SAR, "[T]hermal hot refers to 100°F solar insolation…"

Complete and accurate information should be provided, per Section 71.7(a).

NAC Response

Section 2.6.2.3 is revised to refer to a -40°F ambient temperature condition. As specified in Section 2.6.2, the evaluated thermal condition considers a -40°F ambient temperature, no solar insolation and no decay heat load, in still air.

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Section 2.6.2 Cold

2-23 Re-evaluate stresses in the cask body, as appropriate, by considering the minimum internal pressure in combination with the minimum decay heat load and the minimum ambient temperature.

The SAR text and Table 2.6.2.1-1 suggest that the pressure and thermal loading conditions considered deviate from those typically associated with the cold condition test of the normal conditions of transport, per Section 71.71(b).

NAC Response

The cold case load combination considered in Section 2.6.2 assumes a -40°F ambient temperature, no solar insolation, no decay heat and an internal pressure of 150 psig. This case is different from the cold case combination loading noted in Table 2.1.2-2 in that the cask minimum pressure is not used.

The cask body stress results obtained considering an internal pressure of 150 psig are bounding with respect to a case considering a minimum or zero internal pressure with all other conditions being the same.

Note that Section 2.6.2 is revised to consider an internal pressure of 150 psig. This pressure is used in accordance with the response to RAI 2-3.

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Section 2.6.5 Vibration

2-24 Submit an evaluation of the fatigue strength of the tie-down system of the package.

The vibration-induced alternating stresses in the tie-down system should be evaluated under the vibration condition of the normal conditions of transport, per Section 71.71(c)(5).

<u>NAC Response</u>

The two cask components of the tie-down system are the shear ring, at the front of the cask, and the rotation pockets at the rear. The shear ring is designed to react the assumed 10g longitudinal load. It is not loaded in the vertical direction, which is the primary vibration direction incident to transport, as discussed in Section 2.6.5.

The analysis presented in Section 2.6.5 shows that the transport cask rotation pockets satisfy the requirements for normal vibration incident to transport in accordance with 10 CFR 71.71(c)(5) and have a Margin of Safety of +3.2.

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Section 2.6.7 Free Drop (1 Foot): Cask Body Analysis

2-25 Clarify the text by defining explicitly the free-drop deceleration g-loads used in the bounding analyses of the cask body.

Complete and accurate information should be provided, per Section 71.7(a).

<u>NAC Response</u>

Section 2.6.7 is revised to show that a 20g deceleration load is used in the normal conditions of transport end and side drop analyses as a bounding g-load. Reference is made to Table 2.6.7.5-3 for the calculated g-load for the end and side drop conditions.

Sections 2.6.7.1 and 2.6.7.2 are similarly revised to show that 20g is used as the bounding g-load for the one-foot end drop and one-foot side drop evaluations, respectively.

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Section 2.6.7 Free Drop (1 Foot): Cask Body Analysis

2-26 In Table 2.6.7.2-1, revise, as appropriate, the stress allowable of 20.0 ksi to be consistent with that of 19.1 ksi of Table 2.6.1.3-1 for shell section 13, and re-evaluate stress margins accordingly.

Shell section 13 is shown to have the lowest stress margin for the cask. The stress allowable in Tables 2.6.7.2-1 and 2.6.1.3-1 are expected to be identical for the shell section under the same ambient temperature of 100°F. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package performance under the Section 71.71 normal conditions of transport.

NAC Response

The allowable stress shown in Table 2.6.7.2-1 for Sections 10 through 13 is revised to 19.1 ksi to be consistent with Table 2.6.1.3-1. The Margins of Safety for these sections are revised accordingly.

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Section 2.6.7 Free Drop (1 Foot): Cask Body Analysis

2-27 With respect to Table 2.6.7.1-2, justify the use of a higher stress allowable of 29.6 ksi for shell Section 22 than that of 19.7 ksi for Section 21 for the same shell material.

Table 2.10.2.2-1 lists an identical temperature of 322.6°F for the two shell sections for stress evaluation. Generally, the same stress allowable should be applicable to the two shell sections in close proximity. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package performance under the Section 71.71 normal conditions of transport.

NAC Response

The allowable stress value of 29.6 ksi used for Section 22 in Table 2.6.7.1-2 is the Primary Membrane plus Primary Bending ($P_m + P_b$) stress limit for the section; this value is correct.

The allowable stresses and margins of safety for Sections 16 through 21, and for Sections 24 through 30, shown in Table 2.6.7.1-2 are not correct. The correct allowable stresses for the Primary Membrane plus Primary Bending ($P_m + P_b$) stress category for sections 16-21 and 24-30 are 29.6 ksi and 28.7 ksi, respectively. Consequently, Table 2.6.7.1-2 is revised to incorporate the $P_m + P_b$ stress allowables for these sections, with the corresponding revisions to the associated margins of safety.

Sections 2.6.7.1 through 2.6.7.3 are also revised throughout as a result of these revisions in the allowable stresses.

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Section 2.6.7.5 Impact Limiters

2-28 Justify the use of a factor of 0.9, on pg. 2.6-64, for the redwood crush stress-strain curve to account for the suggested negative fabrication tolerance for the impact limiters.

A negative fabrication tolerance, as suggested, may not exist. The use of the factor of 0.9 should only be considered for calculating a bounding cask deceleration by the RBCUBED program. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package free-drop performance under Sections 71.71(c)(7) and 71.73(c)(1).

<u>NAC Response</u>

From Section 3.4.2, the average temperature of the wood in the impact limiters is 135°F for the 100°F ambient temperature condition.

The redwood crush strength properties at the maximum normal operating condition temperature reduced by 10% (0.9 factor) for minimum fabrication dimensions, are used in the impact limiter analysis to determine the bounding (largest) deformation of the limiter. Similarly, the redwood crush strength properties at the minimum normal operating condition temperature, increased by 10% (1.1 factor) for maximum fabrication dimensions, are used in the impact limiter analysis to determine the bounding (largest) cask deceleration g-loads.

See the response to RAI 2-29.

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CHAPTER 2: STRUCTURAL

Section 2.6.7.5 Impact Limiters

2-29 Justify the use of a factor of 0.9, on pg. 2.6-65, for the balsa wood crush stress-strain curve, at 152°F, to account for the suggested fabrication tolerance for the impact limiters.

Reference 37 of the SAR, "NAC-STC Safety Analysis Report," considered the same factor of 0.9, but for a higher temperature of 230°F. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package free-drop performance under Sections 71.71(c)(7) and 71.73(c)(1).

NAC Response

See the response to RAI 2-28.

From Section 3.4.2, the average temperature of the wood in the impact limiters is 135°F for the 100°F ambient temperature condition.

The factor of 0.9 accounts for fabrication dimensional tolerances of the wood in the impact limiters and is not related to the wood temperature.

The balsa wood crush strength properties at the maximum normal operating condition temperature (135°F), reduced by 10% (0.9 factor) for minimum fabrication dimensions, are used in the impact limiter analysis to determine the bounding (largest) deformation of the limiter. Similarly, the balsa wood crush strength properties at the minimum normal operating condition temperature, increased by 10% (1.1 factor) for maximum fabrication dimensions, are used in the impact limiter analysis to determine the bounding (largest) cask deceleration g-loads.

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Section 2.6.7.5 Impact Limiters

2-30 Clarify, as appropriate, the underlined typographical or editorial errors:

1. Pg. 2.6-72, "[T]able 2.6.7.5-4 shows that at impact angles of <u>60</u>° and 75°, ...the secondary impact..."

A comparison of the E_{max} and El values in the table suggests that the SAR statement does not appear to be applicable to the case with an impact angle of 60°. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package free-drop performance under Section 71.73(c)(1).

2. Table 2.6.7.5-4, "[E]nergy...absorbed in second limiter...<u>8.2</u>%..."

The percentage value listed does not appear to be consistent with the other data summarized in the table. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package free-drop performance under Section 71.73(c)(1).

<u>NAC Response</u>

Table 2.6.7.5-4 listed the distribution of the energy contributions for different drop orientations of the cask. As a result of the response to RAI 2-31, it is demonstrated that the need for Table 2.6.7.5-4 no longer exists and it has been removed. Section 2.6.7.5 is revised to describe the impact limiter action for the end drop and the side drop. The revised section also presents the technical basis that demonstrates that the side drop is the bounding orientation for lateral accelerations with respect to a shallow drop angle, which could potentially result in larger accelerations. This demonstration did not require the calculations of the contribution of the energies for the different drop angles.

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Section 2.6.7.5 Impact Limiters

2-31 Justify, with test or analytical results, that the free drop at a 75° oblique drop angle would give rise to the bounding deceleration of the trailing impact limiter in a slap-down event.

The SAR should provide the basis for the assumption that a 75° oblique drop would produce the largest deceleration, thus, the most limiting and damaging condition, to the package undergoing a secondary impact. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package free-drop performance under Section 71.73(c)(1).

NAC Response

In evaluating the critical angle for slapdown, the evaluation presented in this response concludes that the angle of drop resulting in the maximum acceleration imposed on the cask corresponds to the side drop event. Consequently, Section 2.6.7.5.6.1 is revised to show that the 75° drop is bounded by the side drop.

The slapdown effect is associated with free drop angles in which the transport cask assumes a near horizontal position (shallow angle). The leading end of the transport cask impacts the ground first followed by a translational and rotational motion. The translational motion is a consequence of the crushing of the limiter at the leading end. The rotational motion is a result of the moment of the cask weight acting at the cask CG about the leading end. The moment results in an angular acceleration and, therefore, an increasing angular velocity as the trailing end of the cask rotates towards the ground. The trailing end of the cask has not impacted the ground during this time and, therefore, does not contribute to the moment about the leading end of the cask. As a result, the trailing end of the cask derives its velocity from two terms; the translational velocity of the leading end of the cask with the addition of the cask's instantaneous angular velocity factored by the length of the cask. The slapdown effect is associated with the increasing velocity

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NAC Response to RAI 2-31 (Continued)

of the trailing end of the cask. Depending on the geometry, the trailing end velocity can be greater than or lesser than the leading end initial impact velocity. If the velocity increase due to the angular rotation contribution were less than the velocity decrease due to the translational deceleration, the trailing end velocity would be less than the initial impact velocity of the leading end. Likewise, with the reverse condition of these two contributions, the trailing end initial impact velocity would be larger than the leading end initial impact velocity, giving rise to an increased deceleration at the trailing end of the cask, which is commonly associated with the slapdown effect.

The ratio of cask length (L) to radius of gyration (r) is identified as the most important geometric parameter affecting slapdown severity [Sandia]. In the absence of friction, a cask with L/r less than two will not experience slapdown effects.

During the time between initial impact of the leading end impact limiter and the initial impact of the trailing end impact limiter, the only forces acting on the package are the crush force and gravity. The crush force is normal to the impact surface and acts vertically upward through the point of rotation. The package weight acts downward through the center of gravity.

The radius of gyration (r) for the NAC-UMS transport package (loaded cask with impact limiters attached) is:

$$r = \sqrt{\frac{I}{m}} = \sqrt{\frac{6.573\times10^{9}}{260{,}000}} = 159.0 \text{ in}$$

where

$I = 6.573\times10^{9}$ lb-in.^{2}, the mass moment of inertia about the point of rotation
$m = 260{,}000$ lb, the cask design weight

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NAC Response to RAI 2-31 (Continued)

The overall length (L) of the package is 273.3 in. The ratio $\frac{L}{r} = \frac{273.3}{159.0} = 1.7 < 2$.

With the L/r ratio being less than 2, in the absence of frictional forces, the shallow angle drop is bounded by the side drop.

The NAC-UMS package configuration has been shown to be similar to the NAC-STC package configuration with respect to the L/r ratio and the impact limiters are identical in construction with identical outer diameters. Comparing the NAC-STC side drop (cask axis 90° from vertical) and oblique drop (cask axis 75° from vertical) test results, no indication of friction effects are observed. The drop impact angles and corresponding accelerations for the upper and lower impact limiters for the NAC-STC package 1/4-scale drop testes are:

Limiter	Drop/impact angle	Maximum acceleration
Upper	90° (side drop)	250 g
Lower	90° (side drop)	200 g
Upper	75° (oblique drop)	225 g
Lower	75° (oblique drop)	200 g

The Sandia analysis indicates that frictional forces, because their lines of action do not pass through the package CG, can have a significant effect on the angular acceleration of the package. This would contribute to the vertical acceleration of both the "leading" and "trailing" impact limiter accelerations. The Sandia work demonstrates that normalized leading end accelerations for the cask, with L/r = 2 for the inelastic spring model, are less than the side drop accelerations. However, for impact angles less than 15° (with respect to the horizontal), the peak trailing end accelerations are predicted to exceed both the peak leading end and side drop accelerations. The report concludes that this acceleration increase is probably due to friction at the trailing end during secondary impact.

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NAC Response to RAI 2-31 (Continued)

Accelerometer data from the NAC-STC oblique drop test are shown in this response as Figures 2.31-1 and 2.31-2 for the accelerometers at the bottom (leading end) and top (trailing end), respectively. Figure 2.31-1 (leading end accelerometer) indicates that from initial impact, t = 0 until t = 0 .02 sec, the leading impact limiter experiences an initial negative acceleration peaking at approximately –60 g and returning to zero near t=0.02 sec. At this time, the trailing end impact limiter (still not in contact with the impact surface) experiences (see Figure 2.31-2) a positive rotational acceleration peaking at less than 50g between t = 0.02 and t = 0.045. The trailing end then impacts and undergoes a sharp negative acceleration peaking at near -225g and returning to 0g at t = 0.052 sec. During this same time period (between t = 0.045 sec and t = 0.05 sec) the leading end acceleration peaks at approximately 65g and returns to 0g at t = 0.052. All significant acceleration excursions of both the leading and trailing impact limiters have ended by t = 0.06 sec and the package reaches equilibrium (at rest) in the horizontal position by t = 0.10 sec.

These test results from the NAC-STC oblique drop (15° with respect to the horizontal) demonstrate that the peak accelerations measured by the upper and lower accelerometers do not differ significantly, indicating that any angular acceleration imparted to the package as a result of friction is insignificantly small. Furthermore, all peak accelerations of either the leading or the trailing impact limiter are less than the peak acceleration of the upper impact limiter in the side drop condition (-250g).

To further confirm the Sandia work, a one-half symmetry LS-DYNA model is used that simulates the NAC-UMS quarter-scale drop test. The model uses 8-node bricks representing the redwood and balsa wood of the impact limiter and 4-node shell elements that model the cask and the steel skin of the impact limiter. Figure 2.31-3 of this response shows the LS-DYNA cask and impact limiter model. The model plot shows that the trunnion cutouts were not modeled. This reduces the complexity of the model and is not considered to interfere with the examination of the drop angle effect.

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NAC Response to RAI 2-31 (Continued)

A piecewise linear plasticity material model is used to represent the stainless steel (Type 304) skin of the impact limiter. LS-DYNA's rigid material model is used to represent the cask body. The weight of the cask is included in the model by increasing the density of the rigid material. The CG is adjusted by the addition of mass elements at the center of the cask's top and bottom lid. The foam material model uses redwood and balsa wood stress-strain data to account for the energy absorption of the redwood and balsa wood. The properties employed for the redwood in the region of overlap between the cask body and the impact limiter correspond to the parallel direction of redwood. The redwood outside the overlap region, which is not backed by the cask body also corresponds to parallel redwood and parallel balsa wood properties. This is considered to be conservative, since it would maximize cask accelerations. To account for the cask weight acting at the CG, the gravitational field corresponding to 386.4 in/sec^2 was also included. The cask has an initial velocity of 527.4 in/sec, which corresponds to the lower edge of the bottom (leading) impact limiter striking the impact plane from an initial height of 30 feet. The impact plane was modeled with the rigid surface available in LS-DYNA. To simulate the friction, the rigid plane used the option in which the impact limiters are not permitted to slide once the contact of the impact limiter surface onto the rigid plane was initiated. Symmetry conditions were used for the nodes in the vertical plane at the plane of symmetry.

A series of analyses were performed in which the drop angle was varied from 70° to 85°, which are the drop angles of interest. The normalized peak leading and trailing end accelerations are plotted in Figure 2.31-4 of this response. The variation is observed to be monotonic for both ends. While the friction employed in the analysis bounds the Sandia work, it indicates that with low L/r values, friction does not significantly contribute to the trailing end acceleration. These results are considered to be consistent with those concluded from the Sandia work.

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NAC Response to RAI 2-31 (Continued)

Based on the Sandia work, the NAC-STC 1/4-scale model oblique drop test data and the NAC-UMS quarter scale model LS-DYNA analysis, it is concluded that the NAC-UMS transport package with L/r < 2 does not undergo significant additional acceleration on the basis of slapdown for the oblique drop condition. Consequently, the side drop orientation of the cask is considered to be the bounding drop.

Reference:
Sandia Report SAND90-2187 TTC-1012 UC-80, *An Analysis of Parameters Affecting Slapdown of Transportation Packages,* Sandia National Laboratories, Albuquerque, NM, 1991.

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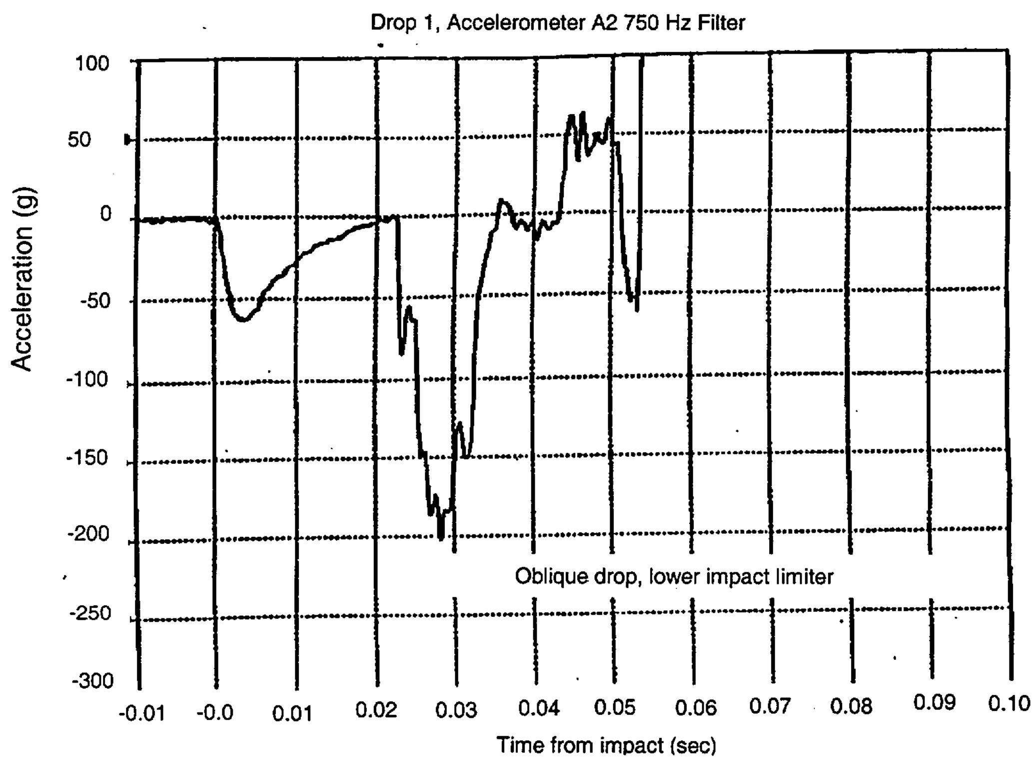
Figure 2.31-1 Accelerometer Time History at the Bottom of the NAC-STC 1/4-Scale Model Cask (75° Oblique Drop)



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<u>NAC Response to RAI 2-31 (Continued)</u>

Figure 2.31-2 Accelerometer Time History at the Top of the NAC-STC 1/4-Scale Model Cask (75° Oblique Drop)

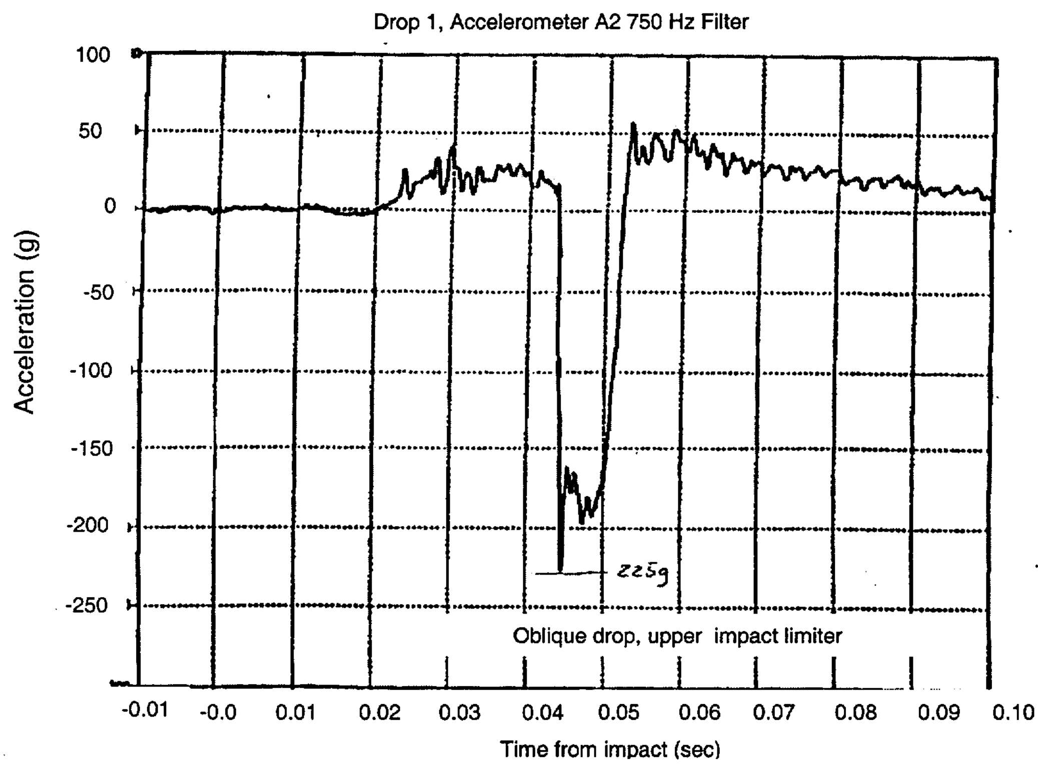

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Figure 2.31-3 Quarter-Scale LS-DYNA Model of the NAC-UMS Cask and Impact Limiters

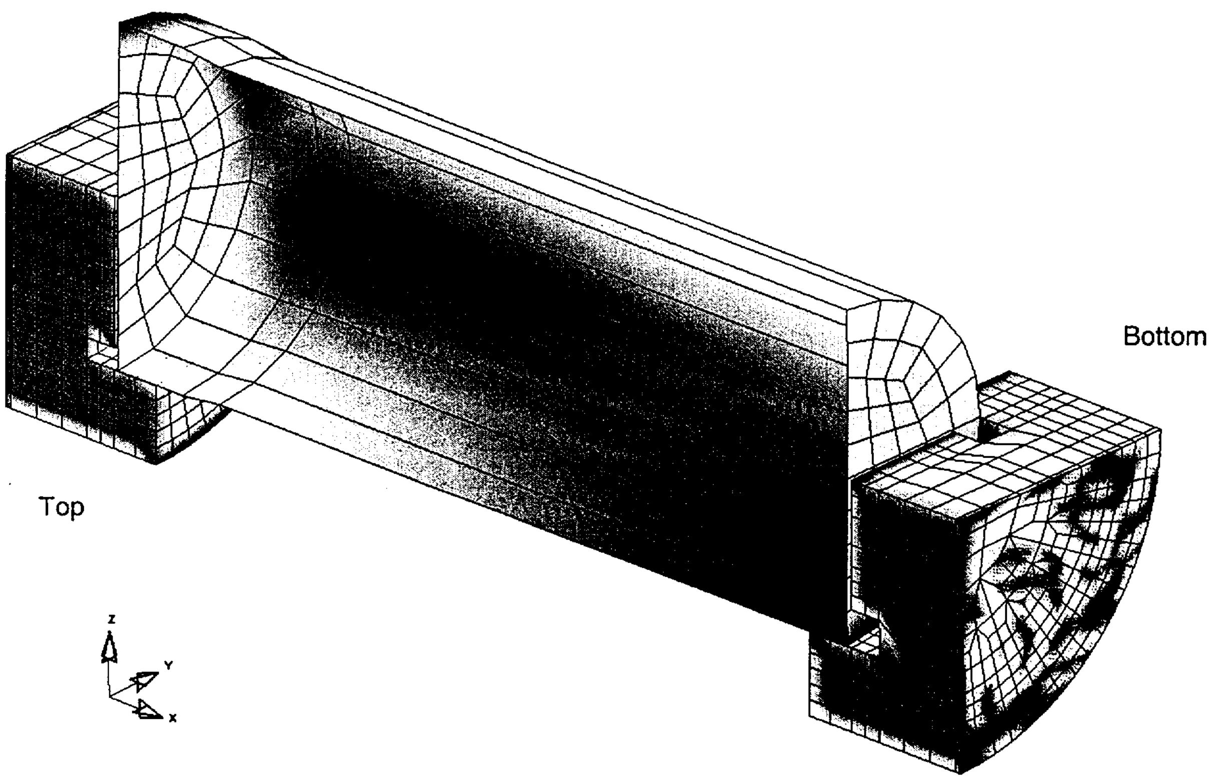

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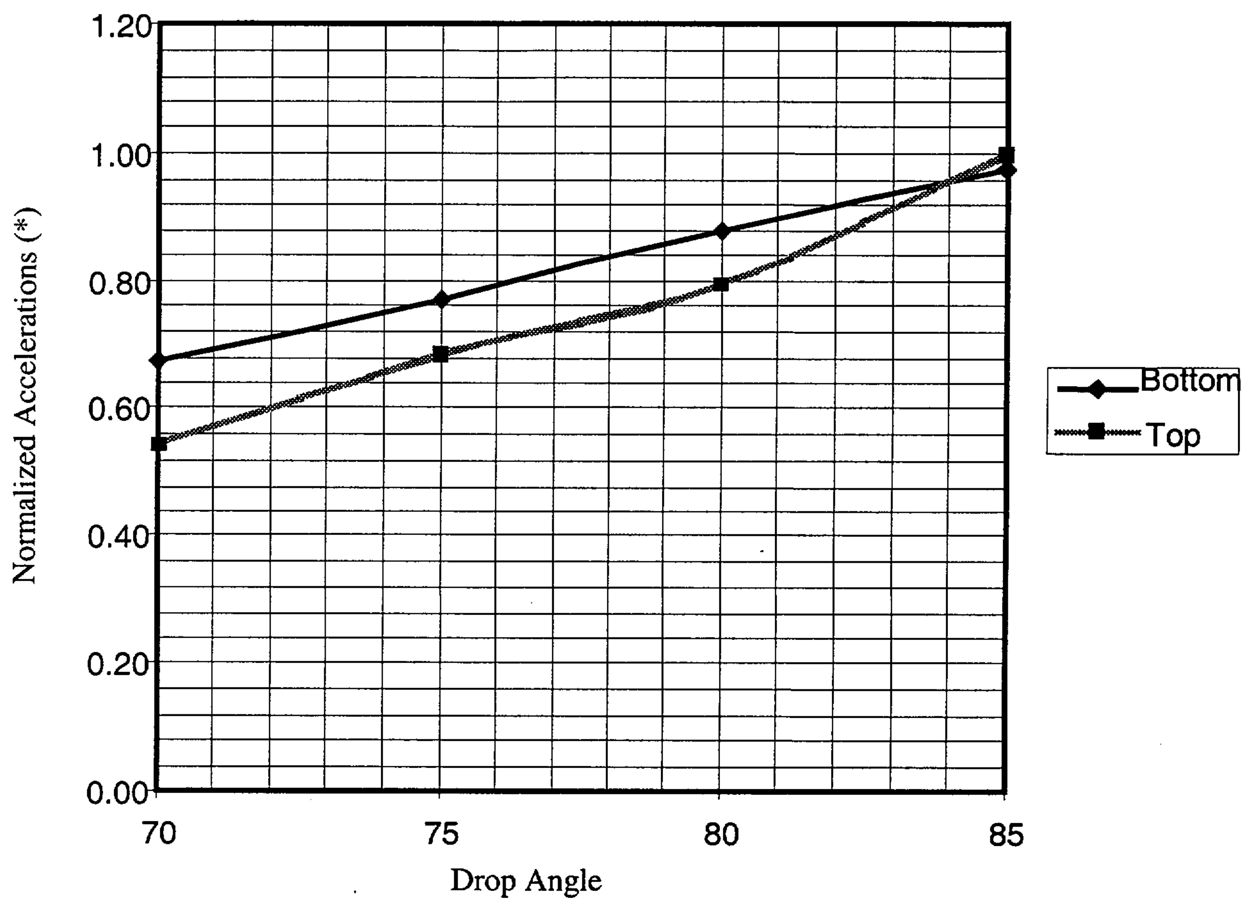
Figure 2.31-4 Normalized Acceleration versus Drop Angle (LS-DYNA for 1/4-Scale NAC-UMS Transport Cask)



(*) Normalized against the 85 degree drop results.

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Section 2.6.7.5 **Impact Limiters**

2-32 Considering the at-temperature impact limiter force-deflection curves in lieu of those associated with Figures 2.6.7.5-11 through 2.6.7.5-17 for bounding temperatures, demonstrate that the impact limiter drop test results can adequately be predicted with those calculated by program RBCUBED.

The impact limiter force-deflection curves displayed in the figures apply to the bounding temperatures. The force-deflection curves for the temperature at which the tests were conducted should be considered for evaluating correlation between the test and calculated results. Complete and accurate information should be provided in the SAR, per Section 71.7(a), for evaluating the package free-drop performance under Sections 71.71(c)(7) and 71.73(c)(1).

<u>NAC Response</u>

Section 2.10.3 is revised to consider test temperature (70°F) material properties in demonstrating that the impact limiter drop test results can be adequately predicted with values calculated using the RBCUBED program.

Crush distance and acceleration values derived from the 70°F RBCUBED analyses, using redwood and balsa wood force-deflection data at 70°F, are compared to measured crush distance and acceleration values from the quarter-scale drop and static tests. The RBCUBED-predicted values, which include dynamic loading effects, are greater than the comparable measured values from the tests; therefore, RBCUBED is demonstrated to provide reasonable and conservative values for use in subsequent structural analyses.

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CHAPTER 2: STRUCTURAL

Section 2.6.7.5 Impact Limiters

2-33 For the 1-foot free drop results summarized in Table 2.6.7.5-1, explain why the calculated deceleration force for the lower impact limiter is larger than that for the upper impact limiter.

For the same impact limiter deformation of 1 inch, the upper impact limiter with a larger backed area than the lower impact limiter should give rise to a higher deceleration force in an end-drop event. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package free-drop performance under Section 71.71(c)(7).

NAC Response

For the 1-foot end and corner drops, the lower (bottom) impact limiter analysis includes the effect of backing afforded by the neutron shield bottom plate, resulting in a larger backed area for the lower impact limiter, even though the bottom end of the cask body has a smaller diameter than the top end. Inclusion of the neutron shield bottom plate in the 1-foot end and corner drops is conservative, since it results in a higher g-load than would result from considering only the diameter of the bottom end of the cask. The approximate radius used for the backed area of the bottom impact limiter is 46.2 inches. The approximate radius of the backed area of the top impact limiter is 43.6 inches. Section 2.6.7 is revised to clarify the difference in the diameters of the two impact limiters.

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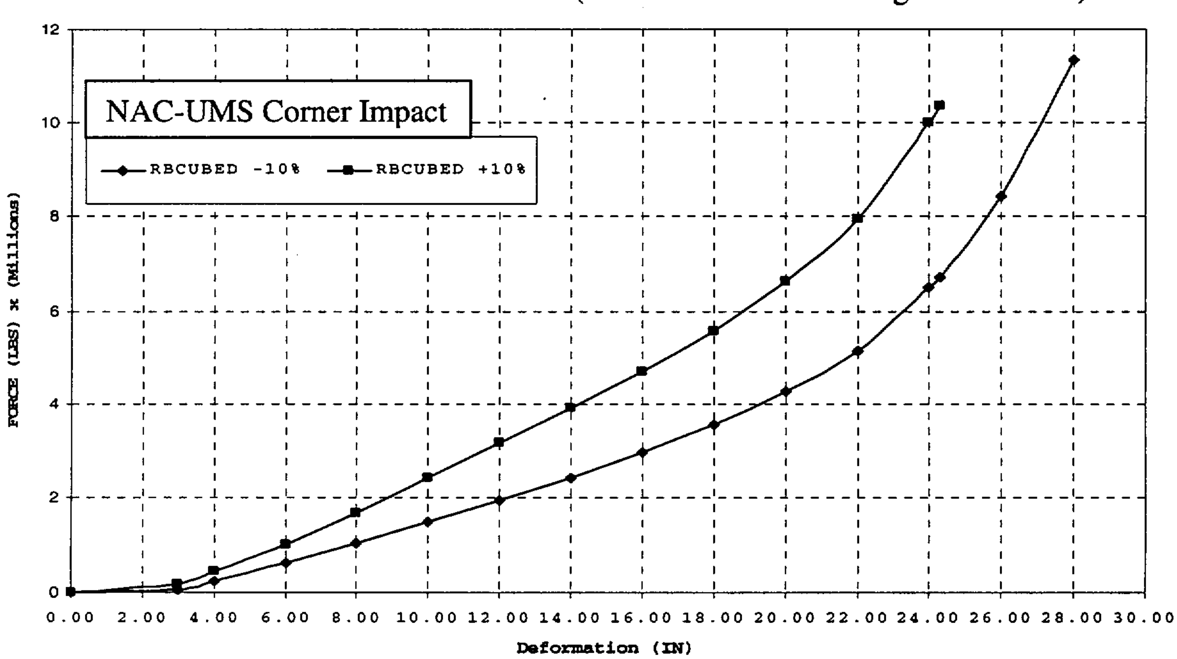
Section 2.6.7.5 Impact Limiters

2-34 For the corner impact force-deformation curves shown in Figures 2.5.7.5-12 and 2.5.7.5-15 for the lower and upper impact limiters, respectively, explain why the curves are markedly different in shape for initial deformations of about 4 inches or less.

For the identical redwood and balsa wood material properties, the RBCUBED calculated force-deformation curves are expected to have a similar shape for the essentially identical design for the upper and lower impact limiters. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package free-drop performance under Sections 71.71(c)(7) and 71.73(c)(1).

<u>NAC Response</u>

The curve in Figure 2.6.7.5-12 is incorrectly plotted for the –10% data at a deformation of 3 inches. The figure is revised with the correct data as shown below, but is now renumbered to Figure 2.6.7.5-13 due to other revisions of text. (Note the correction in figure numbers.)



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Section 2.6.12.12 Canister Buckling Evaluation for 1-Foot End Drop

2-35 Describe how the ANSYS dynamic shell analysis was performed for the maximum stresses used in the buckling evaluation of the TSC.

Complete and accurate information should be submitted for review, per Section 71.7(a).

NAC Response

The stresses used in the buckling evaluation are based on the static analysis for the canister as presented in Section 2.6.12.4 through 2.6.12.9. Section 2.6.12.12 is revised to delete the reference to "dynamic shell analysis," which is an incorrect description of the canister analysis.

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Sections 2.6.13 & 2.6.15 PWR and BWR Basket Analysis—Normal Conditions of Transport

2-36 With respect to Tables 2.6.12.8-1 and 2.6.14.8-1, for the PWR and BWR canisters, respectively, explain why the minimum stress margins and critical cross sections are shown markedly different from each other (0.02 at Section 2 vs. 0.52 at Section 9) for the top corner-drop.

The PWR and BWR canisters are essentially identical in design configurations and loading conditions except that the BWR canister is slightly longer. As such, because of the same stress analysis approach, minimum stress margins of similar order of magnitude are expected for the same canister cross section locations. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package performance under the Section 71.71 normal conditions of transport.

NAC Response

The minimum stress margins and critical cross sections are different because of the geometry of the basket bottom weldment. The PWR bottom weldment is 2 inches high while the BWR bottom weldment is 5 inches high. For the top corner drops, these weldment plates have a lateral component of loading that causes them to bear against the side of the canister shell, right above the location of Section 2 (see Figures 2.6.12.3-1 and 2.6.14.3-1). Since the distance between the bottom weldment plate and the canister bottom plate (Section 2) is shorter for the PWR design, there is a more localized edge contact for the PWR canister and consequently a higher local stress. As shown in Tables 2.6.12.8-1 and 2.6.14.8-1, the radial stress component (SX) at Section 2 is -19.4 ksi and -8.9 ksi for the PWR and BWR canisters, respectively. Therefore, the minimum stress margin for the PWR canister (0.02 at Section 2) is markedly lower than that for the BWR canister (0.52 at Section 9).

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Sections 2.6.13 & 2.6.15 PWR and BWR Basket Analysis—Normal Conditions of Transport

2-37 Submit the support disk modal properties data to demonstrate that dynamic load factors (DLFs) have appropriately been considered in analyzing support disk ligaments.

The cask deceleration may need to be amplified by dynamic effects for defining the deceleration forces for quasi-static analyses of basket support disks. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package performance under the Section 71.71 normal conditions of transport.

NAC Response

The Safety Analysis Report is revised to incorporate Section 2.10.4, which presents an evaluation of the dynamic load factor (DLF) for the PWR and BWR support disks. The 1-foot end drop and 1-foot side drop conditions are selected for the evaluation.

The maximum accelerations and DLFs for the 1-foot end-drop and side-drop orientations for the PWR and BWR support disks.

Fuel Type	Drop Orientation	Maximum Acceleration (g)		DLF
		Input	Response	
PWR	End	17.1	13.45	0.79
PWR	Side	16.4	14.99	0.91
BWR	End	17.1	13.53	0.79
BWR	Side	16.4	15.72	0.96

For all cases, the DLF is less than 1.0 and the maximum response is below the design basis g-load (20g) used in the support disk evaluation for normal conditions of transport. Similar results are expected for the accident conditions and, therefore, no further evaluation is performed.

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Sections 2.6.13 & 2.6.15 PWR and BWR Basket Analysis—Normal Conditions of Transport

2-38 Revise Tables 2.6.15.4-1, -2 and 2.6.15.5-1 of the SAR by also listing stress allowables and corresponding design margins for the support disk.

Complete and accurate information should be presented in the SAR, per Section 71.7(a).

NAC Response

Tables 2.6.15.4-1, 2.6.15.4-2 and 2.6.15.5-1 are revised to include the allowable stress and margin of safety for the reported support disk nodes.

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Section 2.7.1 Free Drop (30-ft)—Cask Body Analysis

2-39 Identify the maximum design internal pressure used in determining cask bounding stresses.

Table 2.7.3.1-4 lists a maximum cask cavity pressure of 75 psig. Pg. 2.10.2-6 cites a pressure of 150 psig. On the basis of the Section 2.7.1 description, however, it is not clear whether a cask internal pressure of 75 psig is considered in the load combination evaluation for the 30-ft cask drops. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package performance under the Section 71.73 hypothetical accident conditions.

NAC Response

Two cask cavity pressure values are applied in the analysis of the UMS Universal Transport Cask. Both pressures are selected to be bounding pressures for purposes of the cask evaluation. A cavity pressure of 150 psig is used in the finite element model analyses of the cask body (model description, Section 2.10.2).

A cavity pressure of 80 psig is used in the evaluation of the cask closure lid bolts (Section 2.6.7.6). The pressure used in the closure bolt analysis (80 psig) represents a bounding pressure for the cask.

Neither pressure is the maximum design internal pressure.

The cask design internal pressure (75 psig) is a pressure conservatively selected to be greater than any calculated cask internal pressure, but less than the bounding pressure used in the analysis (150 psig). It is specified for the purpose of establishing a limit for any subsequent evaluation of

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a package condition. It is not used in any ANSYS or classical analysis of the package.

Table 2.7.3.1-4 is revised to reference the bounding pressures of 150 psig used in the ANSYS finite element analysis and 80 psig used in cask closure lid bolt analysis.

Section 2.7.1 is revised to show that an internal pressure of 150 psig is used in the finite element analysis of the transport cask in the 30-foot drop events.

See the NAC Response to RAIs 2-3, 2-20 and 2-49.

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Section 2.7.1.5 Lead Slump Resulting From a Cask Drop Accident

2-40 Submit an analysis of lead slump resulting from cask drop accidents.

Supporting analyses are necessary to complete the review; the SAR provides only a summary description of lead slump evaluation results. Complete and accurate information should be provided, per Section 71.7(a).

NAC Response

Section 2.7.1.5 is revised to incorporate the lead slump analysis for the end and side impact accident event. As shown in Section 2.7.1.5, the lead slump height in the end impact event is 3.05 inches, and is 0.91 inches in the side impact event. No significant increase in dose rate results from the reduction in shielding represented by the slump condition.

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Section 2.7.1.7 Closure Analysis

2-41 Submit an analysis of deformations and stresses for the protection lip of the cask top forging for the 30-foot free drop conditions.

The SAR uses the NUREG/CR-6007 approach for the closure analysis, which provides that the closure bolts should be protected from direct impact to minimize bolt forces generated by free drops. The deformations of the protection lip should be shown to be less than the design diametric gap of 0.16 inch (78.36" - 78.20" = 0.16") between the closure lid and the protection lip under the free drop hypothetical accident conditions of 10 CFR 71.73(c)(1).

NAC Response

Section 2.7.1.7.1 is revised to include an evaluation of deformation and stresses for the protection lip of the cask top forging.

To ensure that the closure lid bolts are not subjected to any force resulting from the contact of the cask lid and protecting lip of the cask top forging (flange), deflections of the lid edge and the flange during the 30-foot drop impact are examined. The side and top corner drops are considered bounding for the lid/flange interaction. The deflection and stress results for the lid edge and cask flange are obtained from the cask finite element analyses corresponding to these drop conditions (Sections 2.7.1.2 and 2.7.1.3).

The nominal radial gap that exists between the cask lid and flange is (78.36 – 78.20)/2 = 0.08 inch. To determine the amount of change in the nominal gap, the radial deflections at each node on the top outer radius of the lid and the adjacent flange node are obtained from the analyses for both the side and top-corner impacts. The change of radial gap due to the drop events is calculated as:

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Radial gap change = $UX_{flange} - UX_{lid}$

Where:

UX_{flange} is radial deflection of the flange node.
UX_{lid} is the radial deflection of the lid.

Radial gap change is the amount of gap closure (or opening). It is positive when the gap is opening and negative when the gap is closing.

Tables 2.41-1 and 2.41-2 of this response list the calculated deflections for the cask flange and lid relative to the cask centerline for the side and corner drops, respectively. The angular position noted in the table represents the circumferential location of the nodes (0° point of contact for the side drop). For the side drop, the maximum gap closure is 0.0019 inch. For the top-corner drop, the maximum gap closure is 0.0005 inch. Since these gap closures are much less than the nominal radial gap of 0.08 inch, no contact results between the lid and cask flange. Therefore, the cask lid closure bolt will not be subjected to forces due to the deformation of the cask protective lip (flange) during the 30-foot free drops.

The stresses at the cask protective lip (flange) are also reviewed for the 30-foot side and top-corner drops (Sections 2.7.1.2 and 2.7.1.3). The maximum stress at the cask flange (at Section location 36, Figure 2.10.2.2-4) is 22.4 ksi for primary membrane stresses and 24.0 ksi for the primary membrane plus bending stresses. The corresponding margins of safety are 1.14 and 1.61 for membrane and membrane plus bending, respectively.

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Table 2.41-1 Cask Lid and Flange Deflections During 30-foot Side Drop

Lid Node	Lid Radial Deflection (inch)	Flange Node	Flange Radial Deflection (inch)	Angular Position (degree)	Change in Nominal Gap (inch)
801	0.1273	633	0.1262	0	-0.0011
1872	0.1261	1704	0.1242	10	-0.0019
2943	0.1219	2775	0.1201	20	-0.0018
4014	0.1150	3846	0.1136	30	-0.0014
5085	0.1050	4917	0.1041	40	-0.0010
6156	0.0918	5988	0.0913	50	-0.0005
7227	0.0754	7059	0.0754	60	0.0000
8298	0.0548	8130	0.0571	70	0.0022
9369	0.0315	9201	0.0348	80	0.0033
10440	0.0061	10272	0.0084	90	0.0023
11511	-0.0333	11343	-0.0329	105	0.0004
12582	-0.0711	12414	-0.0713	120	-0.0001
13653	-0.1040	13485	-0.1042	135	-0.0003
14724	-0.1293	14556	-0.1297	150	-0.0004
15795	-0.1453	15627	-0.1457	165	-0.0004
16866	-0.1509	16698	-0.1511	180	-0.0002

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Table 2.41-2 Cask Lid and Flange Deflections During 30-foot Top Corner Drop

Lid Node	Lid Radial Deflection (inch)	Flange Node	Flange Radial Deflection (inch)	Angular Position (degree)	Change in Nominal Gap (inch)
801	0.0540	633	0.0537	0	-0.0003
1872	0.0531	1704	0.0527	10	-0.0005
2943	0.0511	2775	0.0506	20	-0.0005
4014	0.0481	3846	0.0478	30	-0.0003
5085	0.0438	4917	0.0437	40	-0.0002
6156	0.0384	5988	0.0384	50	0.0000
7227	0.0311	7059	0.0324	60	0.0013
8298	0.0227	8130	0.0250	70	0.0023
9369	0.0131	9201	0.0157	80	0.0026
10440	0.0027	10272	0.0048	90	0.0021
11511	-0.0135	11343	-0.0122	105	0.0013
12582	-0.0287	12414	-0.0278	120	0.0009
13653	-0.0418	13485	-0.0412	135	0.0006
14724	-0.0520	14556	-0.0515	150	0.0005
15795	-0.0583	15627	-0.0576	165	0.0007
16866	-0.0602	16698	-0.0592	180	0.0010

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Section 2.7.3 Thermal

2-42 The maximum fuel rod cladding temperature tabulated in Table 2.7.3.1-2 is inconsistent with that found in Chapter 3. Revise for consistency. Section 71.7(a) requires complete and accurate information.

NAC Response

Table 2.7.3.1-2 is revised to correct typographical errors and to make it consistent with Table 3.5-2.

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Section 2.7.3 Thermal

2-43 Revise Table 2.7.3.1-3 to the corrected BWR canister internal pressure calculated in Section 3.5-8. Section 71.7(a) requires complete and accurate information.

NAC Response

The calculated internal pressure of the BWR transportable storage canister and UMS Universal Transport Cask are shown on Page 3.5-8 in Section 3.5.4.2.

The transportable storage canister pressure is calculated in Section 3.5.4.2.1 and is 39.96 psig. This value is correct as shown on Page 3.5-8, and in Tables 3.5-3 and 2.7.3.1-3.

The transport cask cavity pressure is calculated in Section 3.5.4.2.2 and is 39.2 psig. This value is correct as shown in Tables 3.5-3 and 2.7.3.1-3, but is not correct as shown on Page 3.5-8. The transport cask cavity pressure calculation shown on Page 3.5-8 incorrectly repeats the calculation of the internal pressure for the canister.

Consequently, Section 3.5.4.2.2, Page 3.5-8, is revised to show the correct calculation of transport cask cavity pressure in the accident case for the BWR fuel contents. The correct cavity pressure is 39.2 psig.

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Section 2.7.8.4 & 2.7.10.4 Fuel Tube Analysis

2-44 Demonstrate the structural integrity of the fuel tube under the "line" load, as exerted by the spacer grid onto the mid-span of the fuel tube, in a cask side drop event.

The uniformly distributed "area" load may not yield bounding results, and loadings based on an equally realistic assumption of line load distribution should also be evaluated. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package under the free-drop condition and test of the Section 71.73(c)(1) hypothetical accident conditions.

<u>NAC Response</u>

Sections 2.7.8.4 (PWR fuel tube analysis) and 2.7.10.4 (BWR fuel tube analysis) are revised to include analysis of the "line" load condition. The weld evaluation for the BORAL cover plate is also revised due to the use of the intermittent weld, rather than the continuous weld, configuration.

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Section 2.7.10 BWR Basket Analysis – Accident Conditions

2-45 Revise Tables 2.7.10.1-23 and -24 by also listing stress allowables and corresponding design margins for the support disk.

Complete and accurate information should be presented in the SAR, per Section 71.7(a).

NAC Response

Tables 2.7.10.1-23 and 2.7.10.1-24 are revised to include the allowable stress and Margins of Safety for the reported support disk nodes.

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Section 2.7.10 BWR Basket Analysis – Accident Conditions

2-46 Clarify, as appropriate, the underlined typographical or editorial errors. Complete and accurate information should be presented, per Section 71.7(a).

Table 2.7.10.1-24, "P_m + P_b Stresses for Support Disk...Thermal Case 2"

Table 2.7.10.1-22, refers to thermal Case 4, in lieu of Case 2, for stress evaluation.

NAC Response

The title of Table 2.7.10.1-24 is revised to refer to Thermal Case 4.

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Section 2.10.1 Computer Program Description

2-47 Describe the revisions made to the RBCUBED program, subsequent to its previous application in July 1992, for the NAC Storable Transport Cask (NAC-STC).

The SAR refers to the November 1996 version of the program. However, it is not clear whether the 1992 version of the program has been modified and appropriately validated for the present Universal Transport Cask application. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the program for meeting the Section 71.101(a) quality assurance requirements.

<u>NAC Response</u>

The RBCUBED program was revalidated in mid-1996 as part of a larger quality assurance program to validate or revalidate all of the analytical software used by NAC. In the course of the RBCUBED revalidation, typographical errors were corrected, and minor editorial changes were made in the RBCUBED Users Manual. No revisions or changes were made to the RBCUBED program. The "version" terminology refers to the Users Manual, not to the RBCUBED program.

Both the NAC-STC and the Universal Transport Cask use the same RBCUBED program for impact limiter analysis.

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Section 2.10.2 Finite Element Model—Universal Transport Cask

2-48 Justify the use of CONTAC52 elements between the stacked annulus plates (Item 33, Drawing No. 790-502) connecting the inner and the outer shells of the cask.

The finite element analysis model should not allow force interaction between the annulus plates because a gap between the plates could potentially result from cask fabrication. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the adequacy of the cask finite element model.

NAC Response

The use of CONTAC52 elements is acceptable so long as an appropriate gap size is defined for the elements. To demonstrate that there is no interaction between the annulus plates, the 1-foot (Section 2.6.7.1) and 30-foot (Section 2.7.1.1) drop impacts are evaluated assuming an initial gap size of 0.01 inch. The finite element model is revised to model this assumed gap using CONTAC52 elements.

The maximum effect on annulus plate stresses due to the gap size between the annulus plates occurs in the bottom end drop. In the bottom end drop, the annulus plates are subjected to the maximum load from the lead in the cask. The analysis results show that the gap between the annulus plates remains open, i.e., there is no interaction between the stacked annulus plates.

The stress tables corresponding to the bottom end drops in Sections 2.6.7.1 and 2.7.1.1 are revised to incorporate the stress results from this modified finite element cask model analysis.

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Section 2.10.2 Finite Element Model—Universal Transport Cask

2-49 Identify appropriately the cask internal pressures considered in evaluating cask structural performance under normal conditions of operation and hypothetical accident conditions.

Pg. 2.10.2-6 states, "A pressure of 150 psig is used to conservatively envelope the normal conditions design pressure of 25 psig for all impact loadings considered." Pgs. 2.6-4 and -14 cite a cask internal pressure of 50 psig for normal conditions of operation. Pg. 2.7-1 discusses the application of the maximum design internal pressure to produce bounding stresses, but provides no pressure value for analyzing hypothetical accident conditions. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package structural performance under the conditions and tests of Sections 71.71 and 71.73.

NAC Response

Sections 2.6.1.3 (Page 2.6-4), 2.6.2 and 2.6.2.3 (Page 2.6-14) are revised to delete reference to an internal pressure of 50 psig and to incorporate reference to the analyzed pressure of 150 psig. An internal pressure of 150 psig is used in the cask model described in Section 2.10.2. The 150-psig pressure is also incorporated in Section 2.7.1. Other related sections have been revised to clarify the calculated internal pressure versus the bounding internal pressure used in the analyses.

See the NAC Response to RAIs 2-3, 2-20 and 2-39.

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Section 2.10.3 Confirmatory Testing Program—UMS Impact Limiters and Attachments

2-50 Provide relevant test results and analyses to demonstrate that the RBCUBED program can be used to model the free-drop performance of the UMS Universal Transport Cask.

The SAR description suggests that the confirmatory testing program needs to be clarified for the following inconsistencies in test execution and data reduction:

1. The 1/4-scale cask model should also provide proper simulation of the cask mass moment inertia, in addition to the mass and its center of gravity. The use of weight disks at only the cask top end may not be representative.

2. The measured acceleration time history in Figure 2.10.3-6 suggests significant cask rocking, which is uncharacteristic of a Universal Transport Cask undergoing side-drop response.

3. The end-drop acceleration time history in Figure 2.10.3-1 appears to contain much more spurious components than the similar time history for the 1/4-scale drop test conducted for the NAC-STC.

Complete and accurate information should be provided, per Section 71.7(a), for evaluating the testing program intended for confirming the calculated package performance under the free drops of Sections 71.71(c)(7) and 71.73(c)(1).

<u>NAC Response</u>

1. The mass moment of inertia for the full-size cask is calculated to be 3.99×10^9 lb-in^2, about the centerline of the cask base. Converting the full-size cask mass

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moment of inertia by the scale factor $1/(4^5)$, the quarter-scale model cask should have a mass moment of inertia of 3.90×10^6 lb-in^2 about its base. The calculated mass moment of inertia of the quarter-scale model cask is 3.87×10^6 lb-in^2, a difference of 0.7%. This difference is considered to be acceptable.

Section 2.10.3.2 is revised to clarify that mass moment of inertia was considered in the model design.

2. For the side drop, four accelerometers were mounted on the 2-inch thick shell representing the outer shell of the UMS cask body. The outer diameter of this shell corresponds to a quarter-scale model of the UMS cask. All accelerometers were mounted in the same horizontal plane at the elevation of the center line of the cask body. Two accelerometers were mounted 10 inches from the top of the cask (Accelerometers 1 and 2) and the other two accelerometers were mounted 10 inches from the bottom end of the cask (Accelerometers 3 and 4). The acceleration time history data for the four accelerometers are presented in the response to RAI 2-51. The filtered acceleration is superimposed on the unfiltered acceleration. Along with the acceleration time histories, the energy absorbed by the deceleration was also computed and is shown on a separate figure.

   The energy absorption trace for the bottom accelerometers indicates that the energy due to the side drop was absorbed in the first 10 ms to 15 ms span of time. After absorbing this energy, another significant event occurs which reflects that the energy which was stored in the initial impact is released. This response is also observed in the traces for the bottom accelerometers. The energy absorption by the top accelerometers shows the same type of behavior except to a much more significant extent.

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The bottom accelerometer traces indicate that the total energy can be accounted for. At the top end, the energy trace does not return to the total energy level. The trace beyond the initial impact does not appear to be valid.

The weight of the spacers (steel disks) inside the outer shell of the quarter-scale model is equal to approximately 40% of that of the quarter-scale model steel shells. These spacers are not constrained and are free to move in the direction of the impact. The energy trace for the top accelerometers suggests that the spacers resulted in an additional secondary impact after the initial impact of the cask body. Once the top end initiated rocking action, the bottom end would also respond in a similar fashion. The more extreme response of the secondary impact occurs at the top end. This correlates with the redwood crush strain being larger in the top impact limiter than in the bottom impact limiter. A larger strain results in a larger resisting force and consequently a greater potential to generate a larger secondary impact of the internal spacers.

The acceleration traces for the bottom end of the model are considered to accurately represent the motion of the bottom end of the cask. This is based on the energy absorption. The acceleration traces for the top end of the model are accurate up through the absorption of energy in the initial impact, but the inability to resolve the discrepancy with the energy absorption does not permit the acceleration trace to be used after the initial impact.

3. The method of filtering the acceleration for the end drop was reviewed and was correlated with the balance of energy. This is reflected in the energy absorption traces for all three accelerometers used in the end drop. These figures are presented in the response to RAI 2-51.

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Section 2.10.3 Confirmatory Testing Program—UMS Impact Limiters and Attachments

2-51 Use all measured accelerometer time history traces for data evaluation and correlation with analytical results, and submit those traces and corresponding filtered results for staff review.

Measured accelerometer time histories from all four accelerometers should be considered to ensure that test results are consistently interpreted for data correlation evaluation. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the testing program intended for confirming the calculated package performance under the free drops of Sections 71.71(c)(7) and 71.73(c)(1).

NAC Response

The filtered and unfiltered acceleration time history traces for the 30-foot end and side drop events are provided in Figures 2.51-1 through 2.51-19 of this response. Acceleration data from the three accelerometers used in the 30-foot end drop test and the four accelerometers used in the 30-foot side drop test are presented in the following figures. The filtered data is superimposed on the unfiltered data. The corresponding energy data curves are also provided. The 30-foot corner drop unfiltered and filtered acceleration data are provided in Figures 2.51-15 through 2.51-20.

The end drop and corner drop test configurations used three accelerometers each. Four accelerometers were used in the side drop tests. Data from all accelerometers used in each test are considered in evaluating the test results.

See the NAC Response to RAI 2-50.

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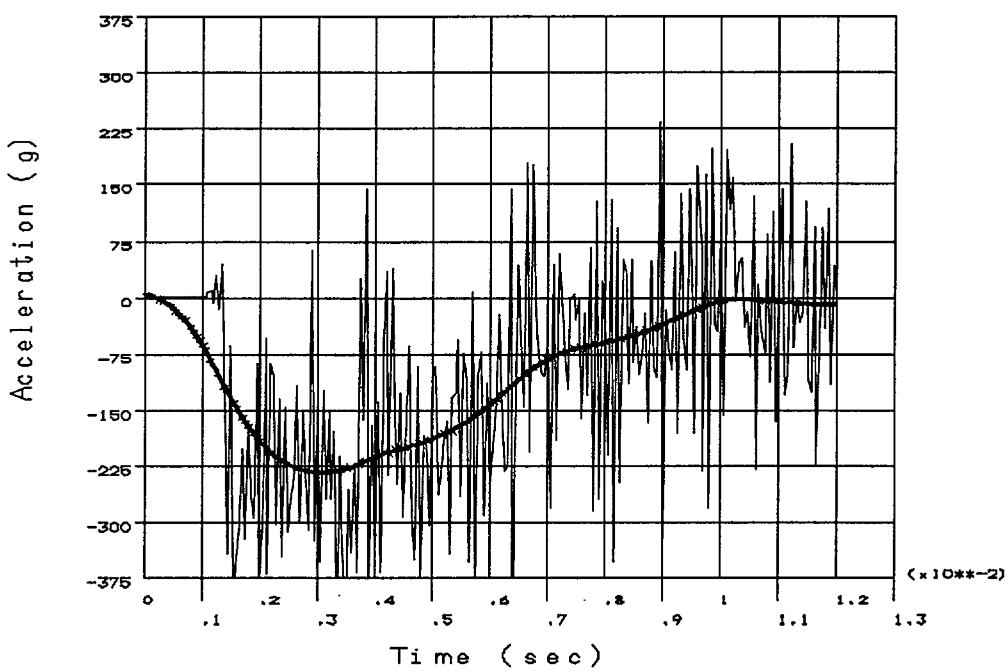
Figure 2.51-1 Upper Impact Limiter Acceleration - 30-ft End Drop



Accelerometer: end1

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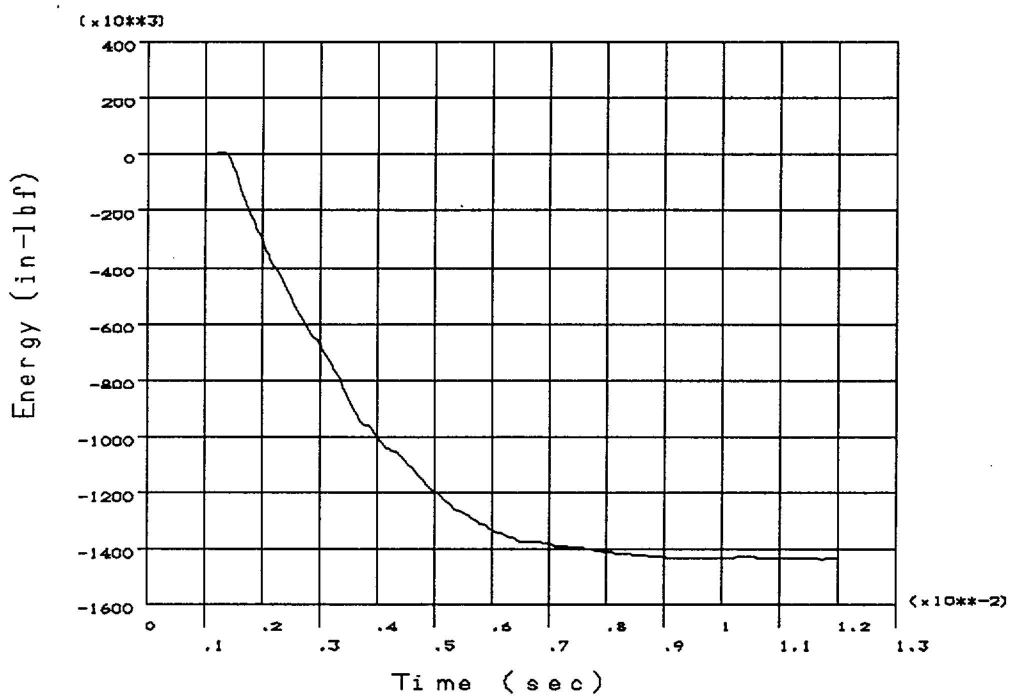
Figure 2.51-2 Upper Impact Limiter Energy - 30-ft End Drop



Accelerometer: end1

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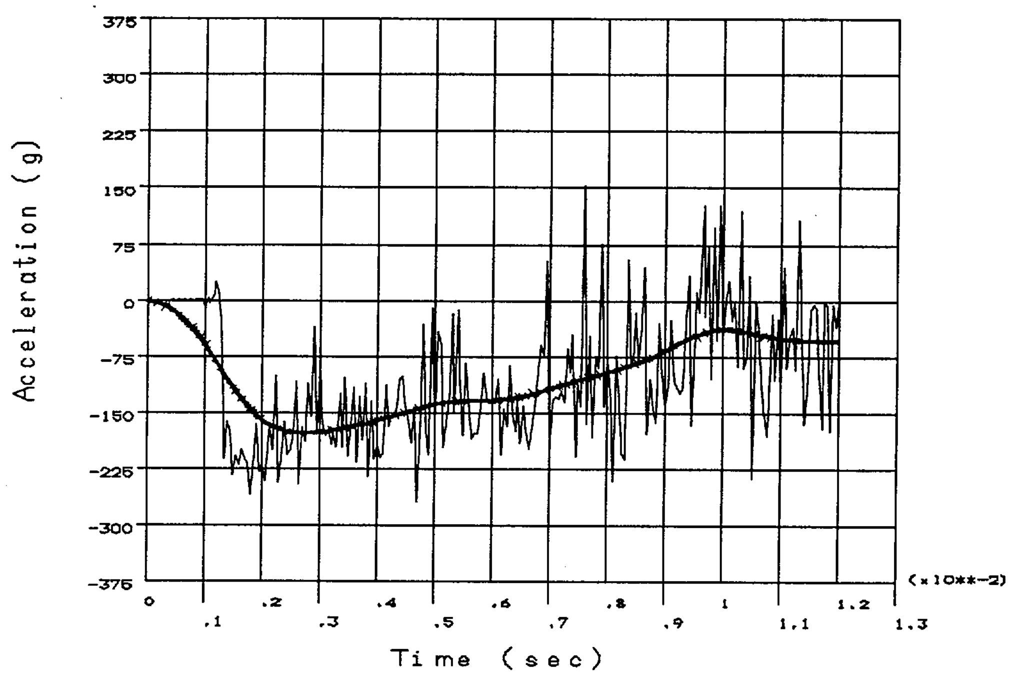
Figure 2.51-3 Upper Impact Limiter Acceleration - 30-ft End Drop



Accelerometer: end2

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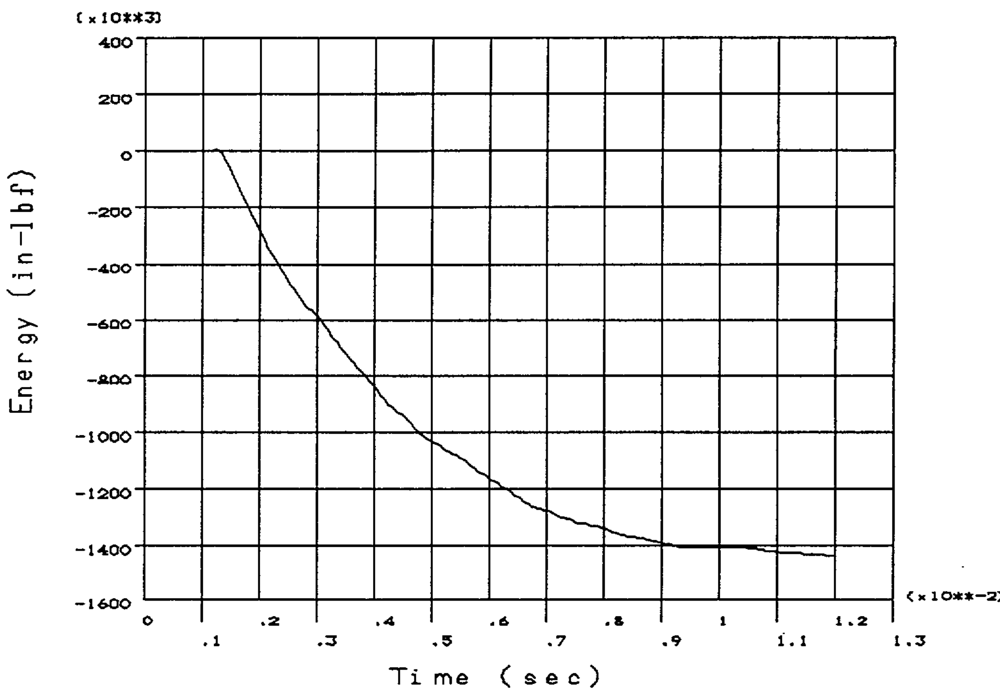
Figure 2.51-4 Upper Impact Limiter Energy - 30-ft End Drop



Accelerometer: end2

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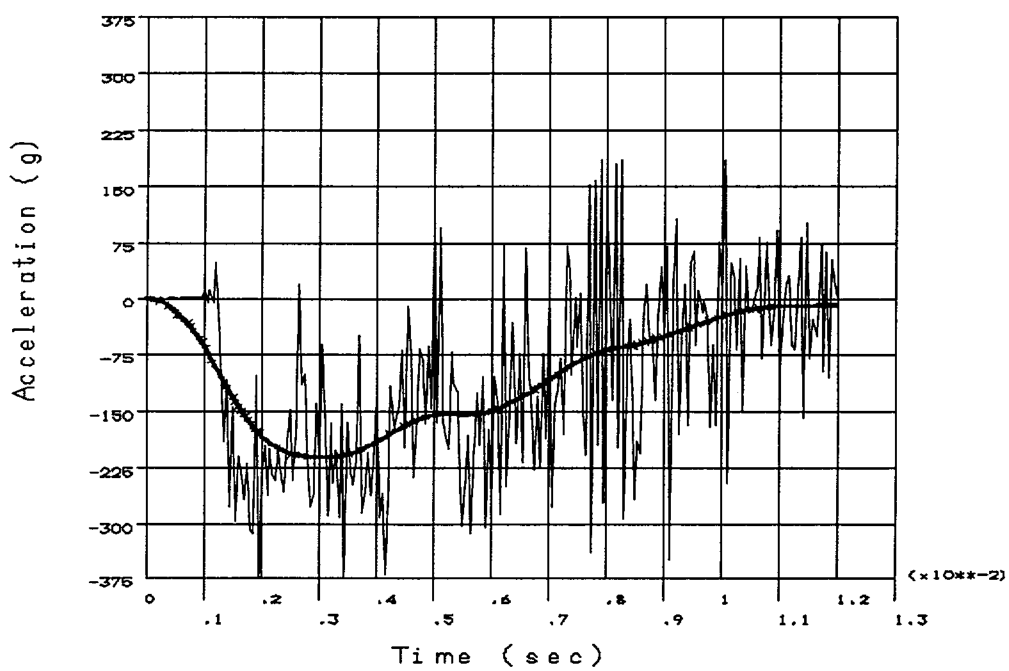
Figure 2.51-5 Lower Impact Limiter Acceleration - 30-ft End Drop



Accelerometer: end3

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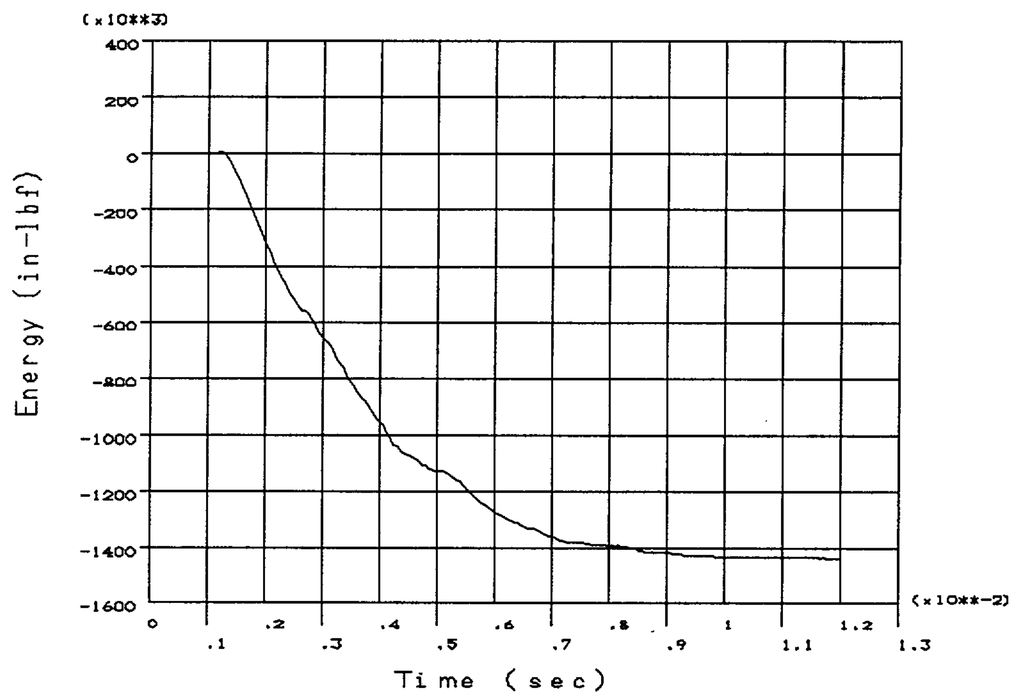
Figure 2.51-6 Lower Impact Limiter Energy - 30-ft End Drop



Accelerometer: end3

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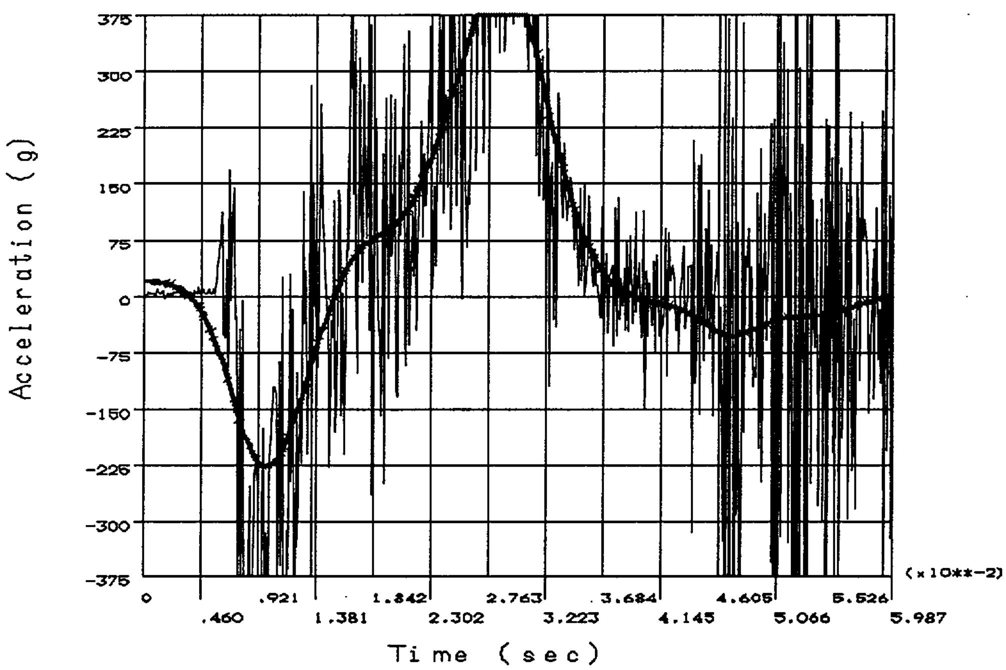
Figure 2.51-7 Upper Impact Limiter Acceleration - 30-ft Side Drop



Accelerometer: side1

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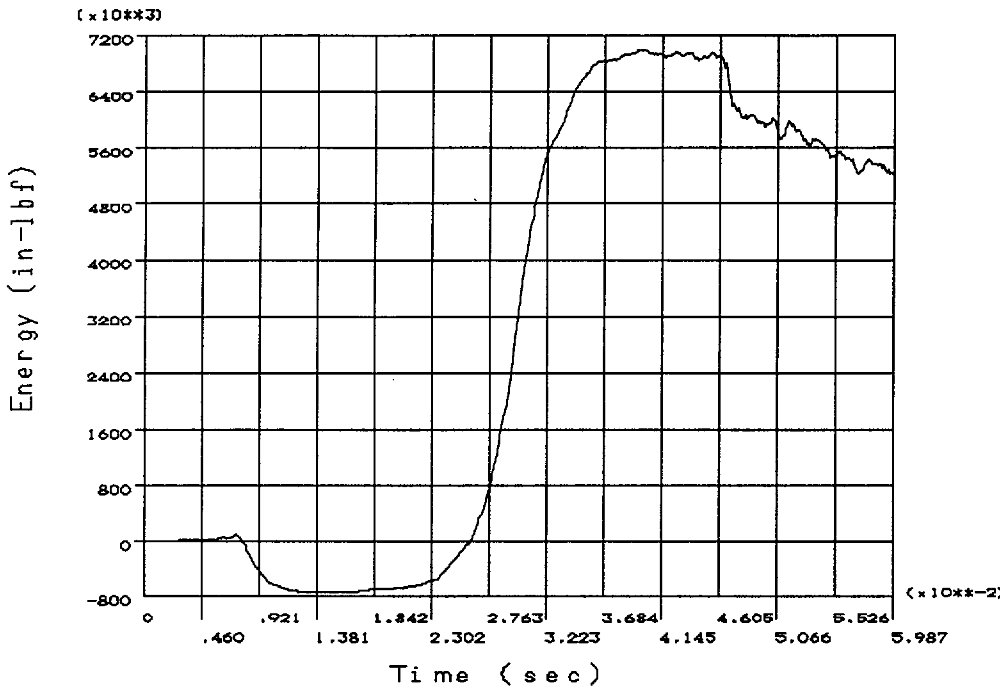
Figure 2.51-8 Upper Impact Limiter Energy - 30-ft Side Drop



Accelerometer: side1

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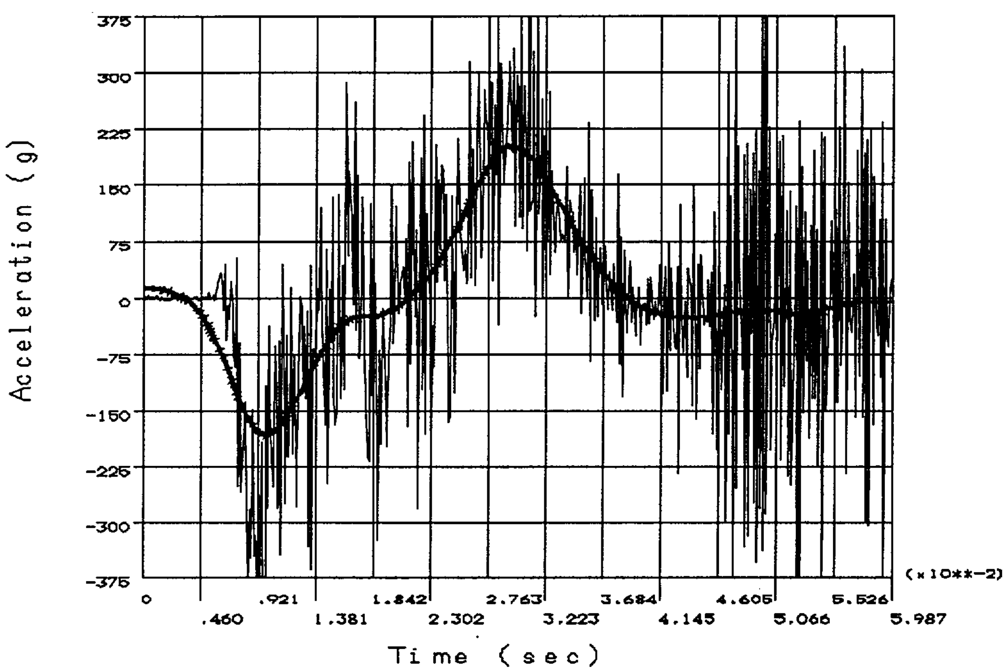
Figure 2.51-9 Upper Impact Limiter Acceleration - 30-ft Side Drop



Accelerometer: side2

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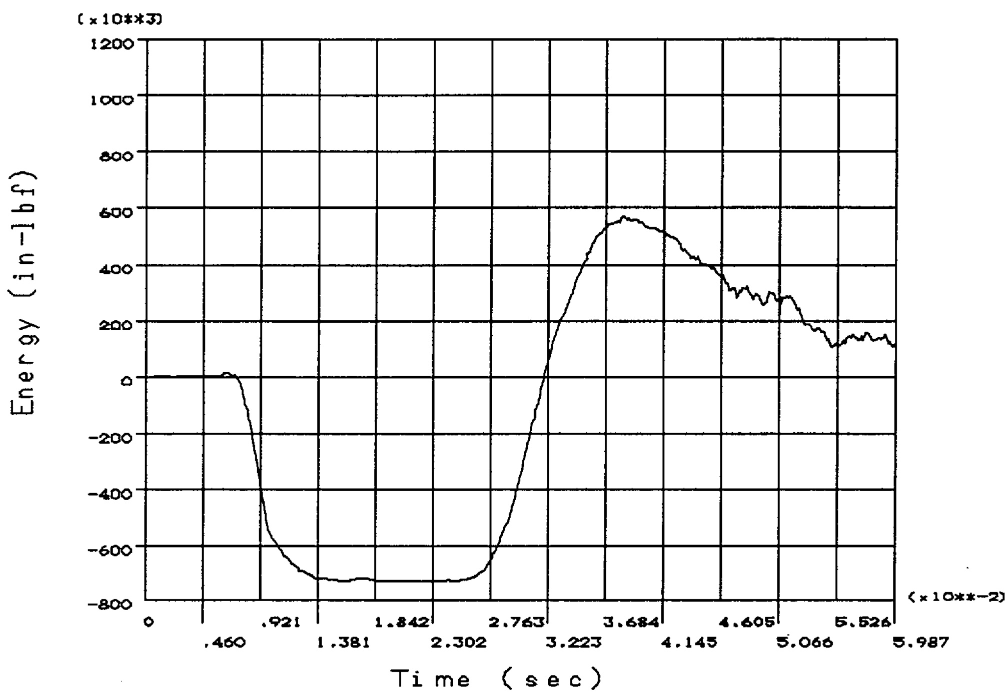
Figure 2.51-10 Upper Impact Limiter Energy - 30-ft Side Drop



Accelerometer: side2

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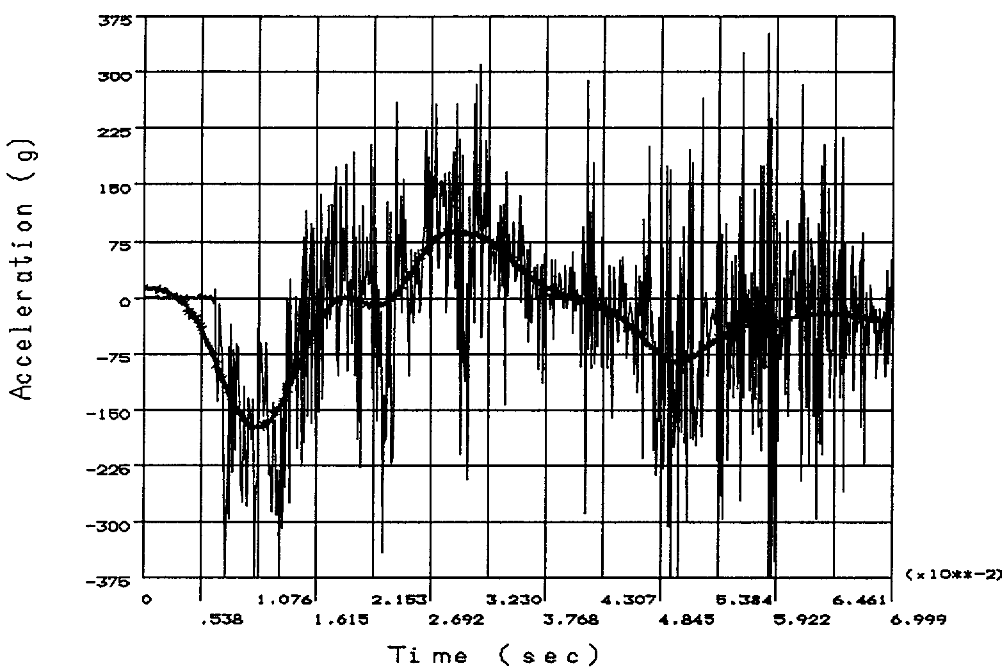
Figure 2.51-11 Lower Impact Limiter Acceleration - 30-ft Side Drop



Accelerometer: side3

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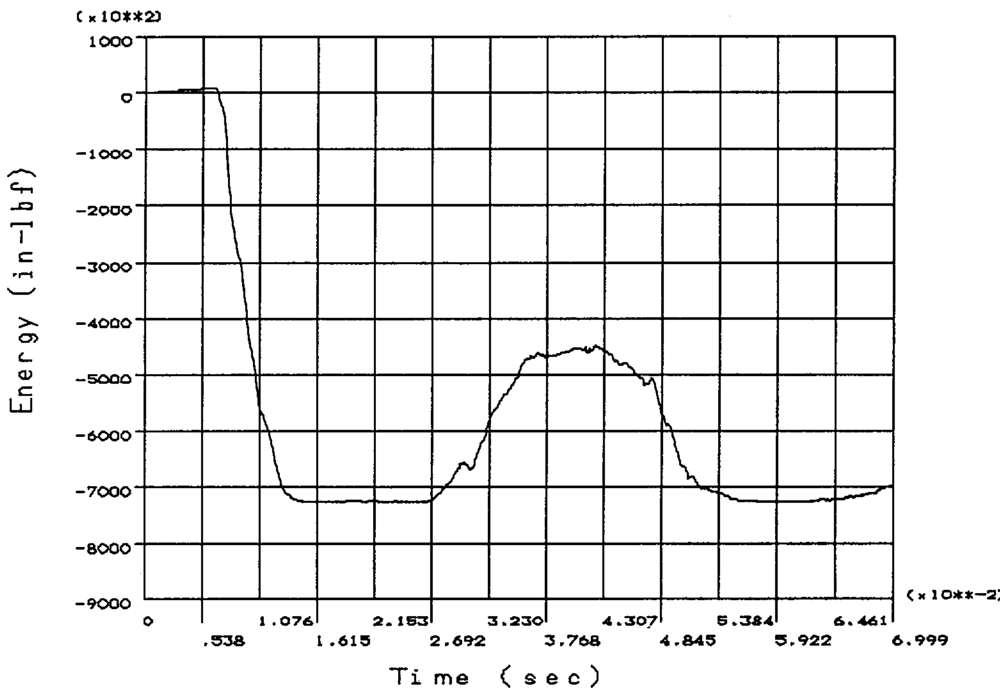
Figure 2.51-12 Lower Impact Limiter Energy - 30-ft Side Drop



Accelerometer: side3

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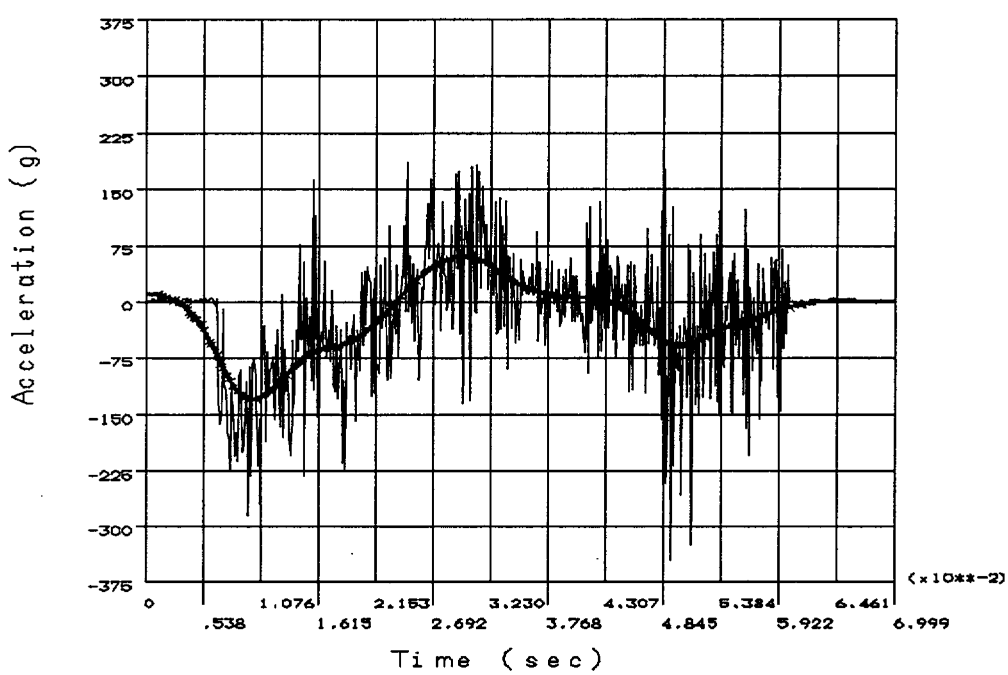
Figure 2.51-13 Lower Impact Limiter Acceleration - 30-ft Side Drop



Accelerometer: side4

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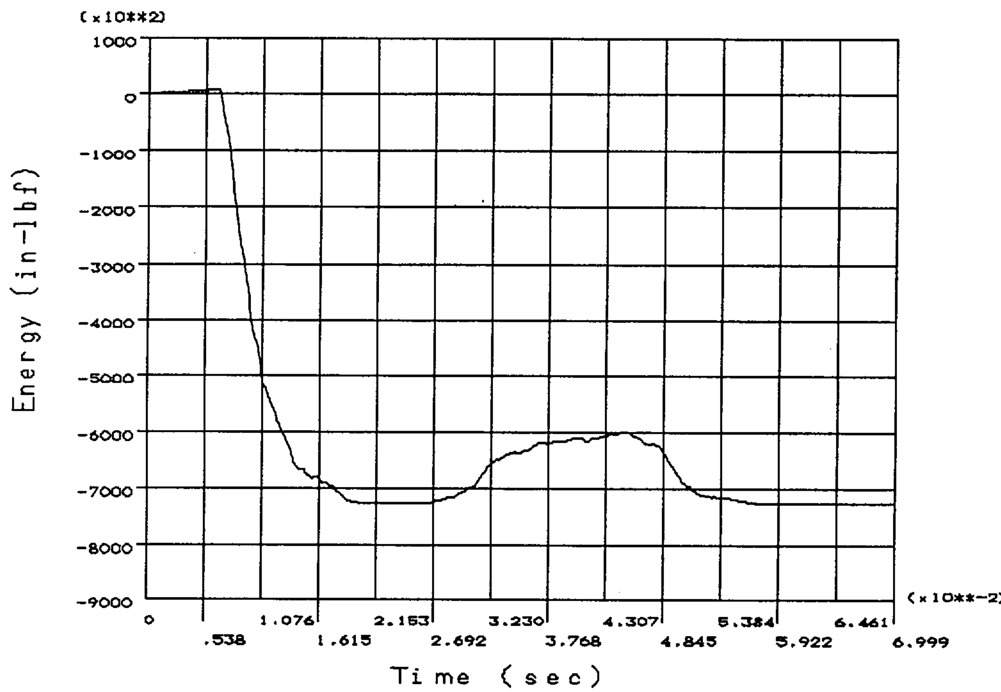
Figure 2.51-14 Lower Impact Limiter Energy - 30-ft Side Drop



Accelerometer: side4

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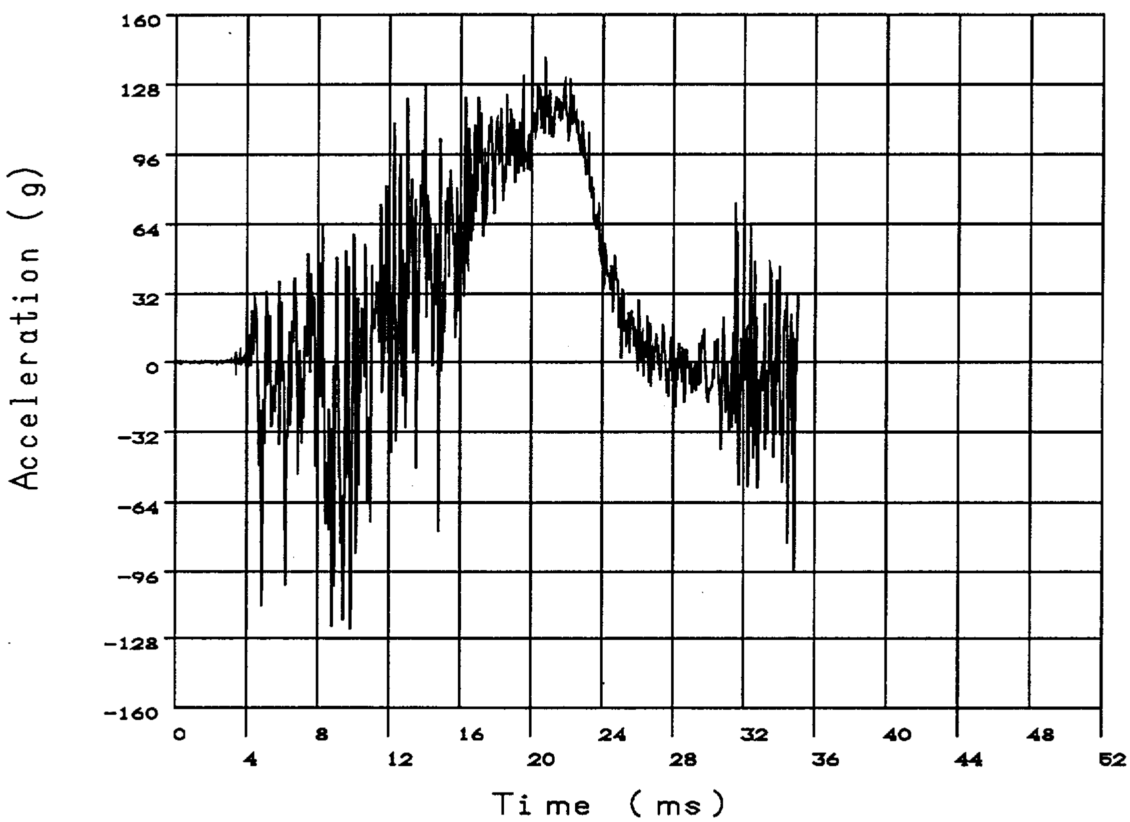
Figure 2.51-15 Corner Drop Unfiltered Acceleration Data



Accelerometer: Corner1

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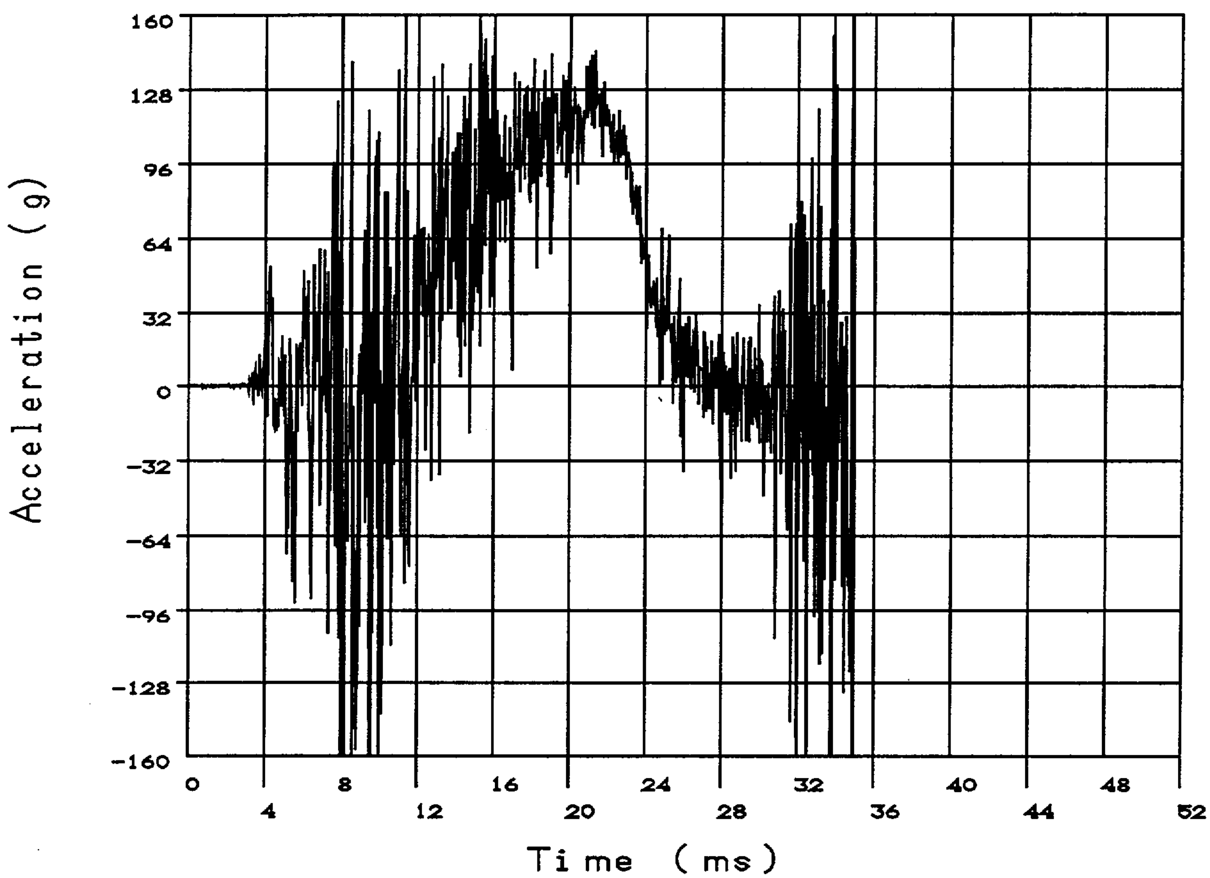
Figure 2.51-16 Corner Drop Unfiltered Acceleration Data



Accelerometer: Corner2

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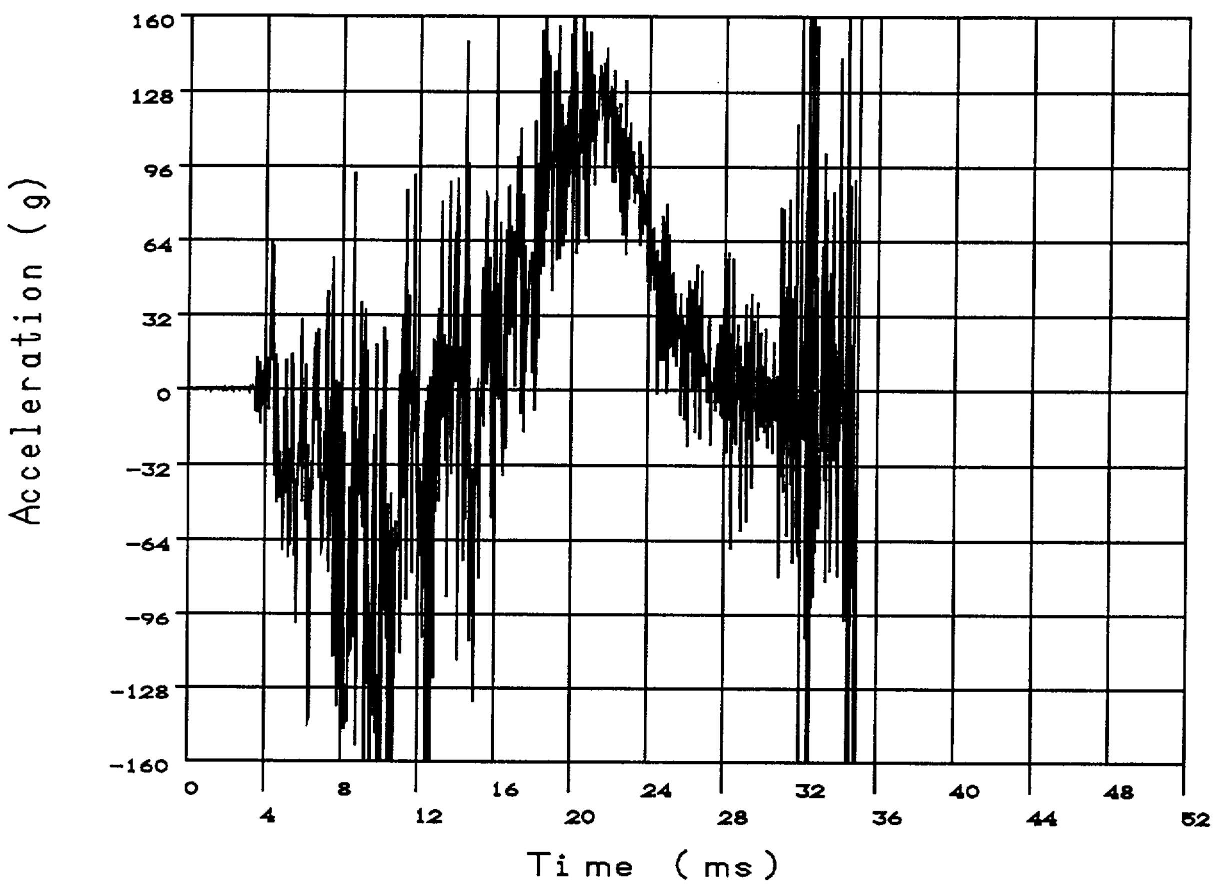
Figure 2.51-17 Corner Drop Unfiltered Acceleration Data



Accelerometer:Corner3

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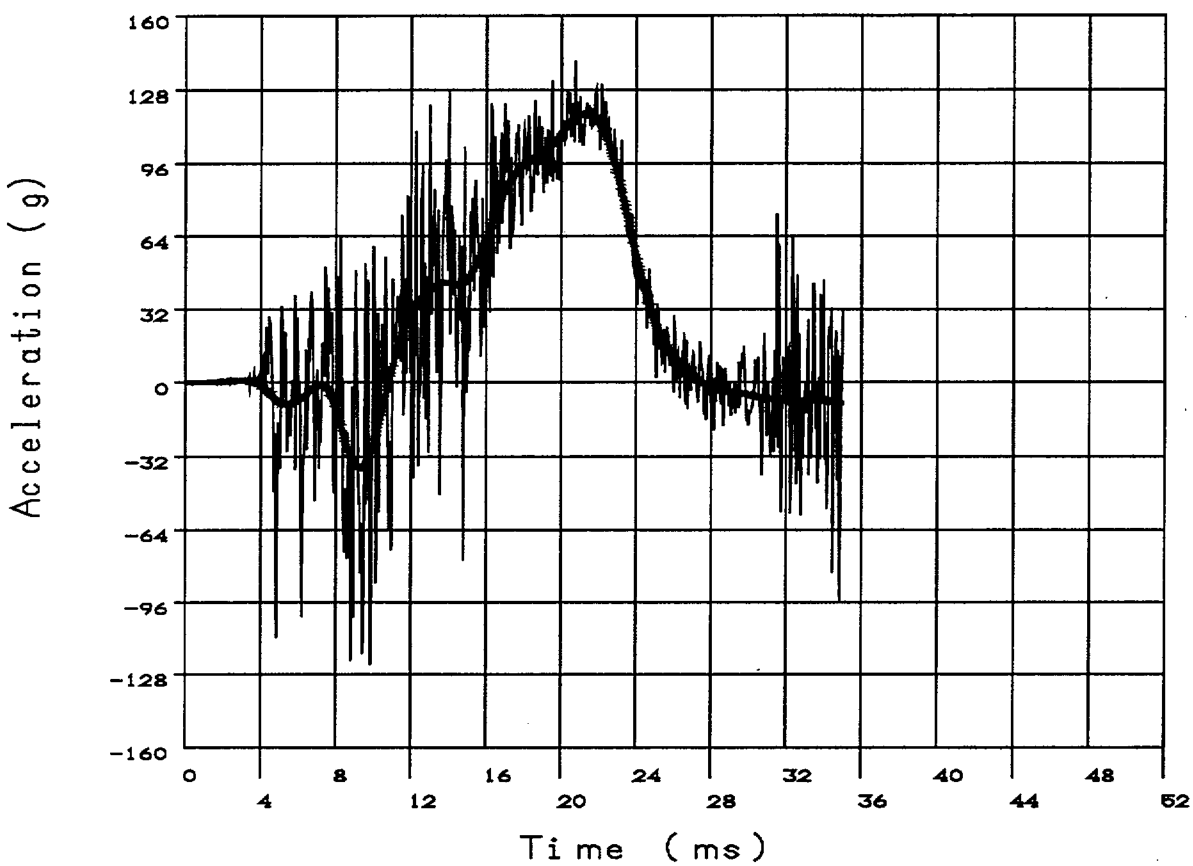
Figure 2.51-18 Corner Drop Filtered Acceleration Data



Accelerometer:Corner1

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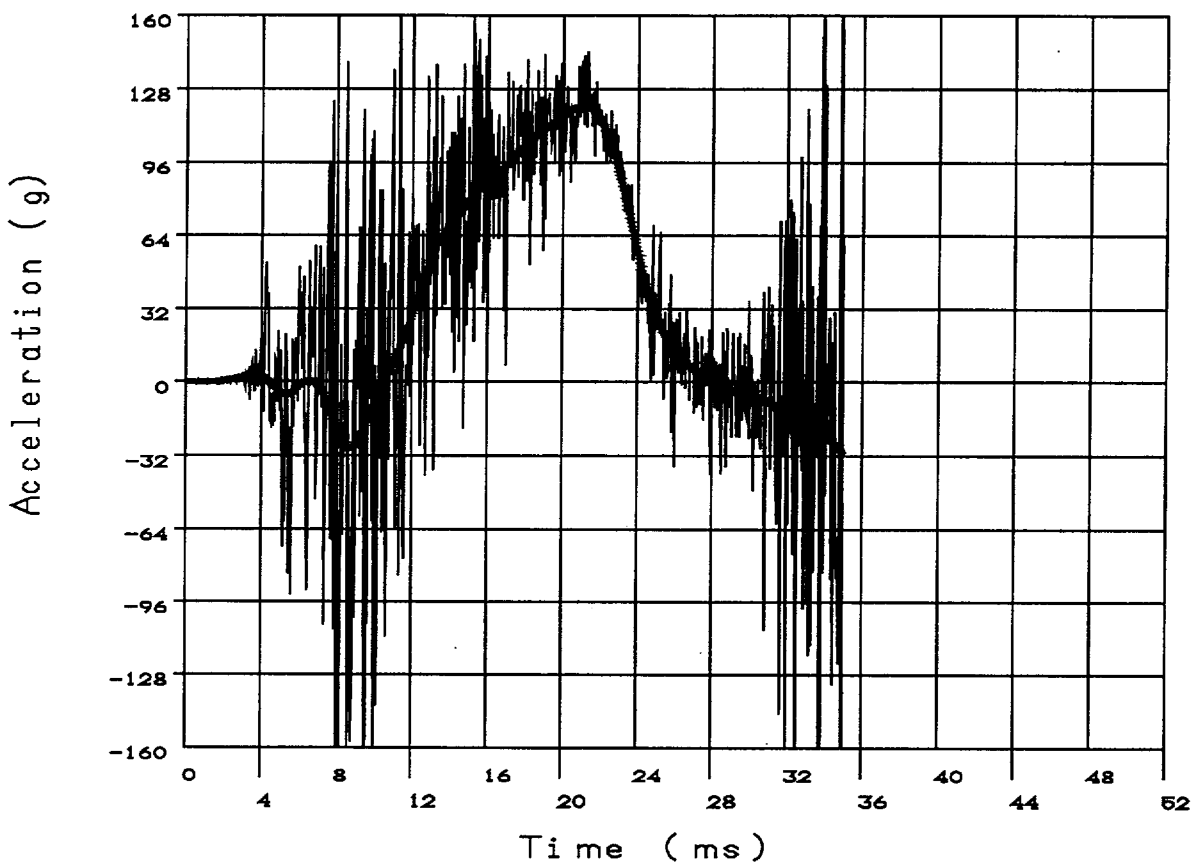
Figure 2.51-19 Corner Drop Filtered Acceleration Data



Accelerometer:Corner2

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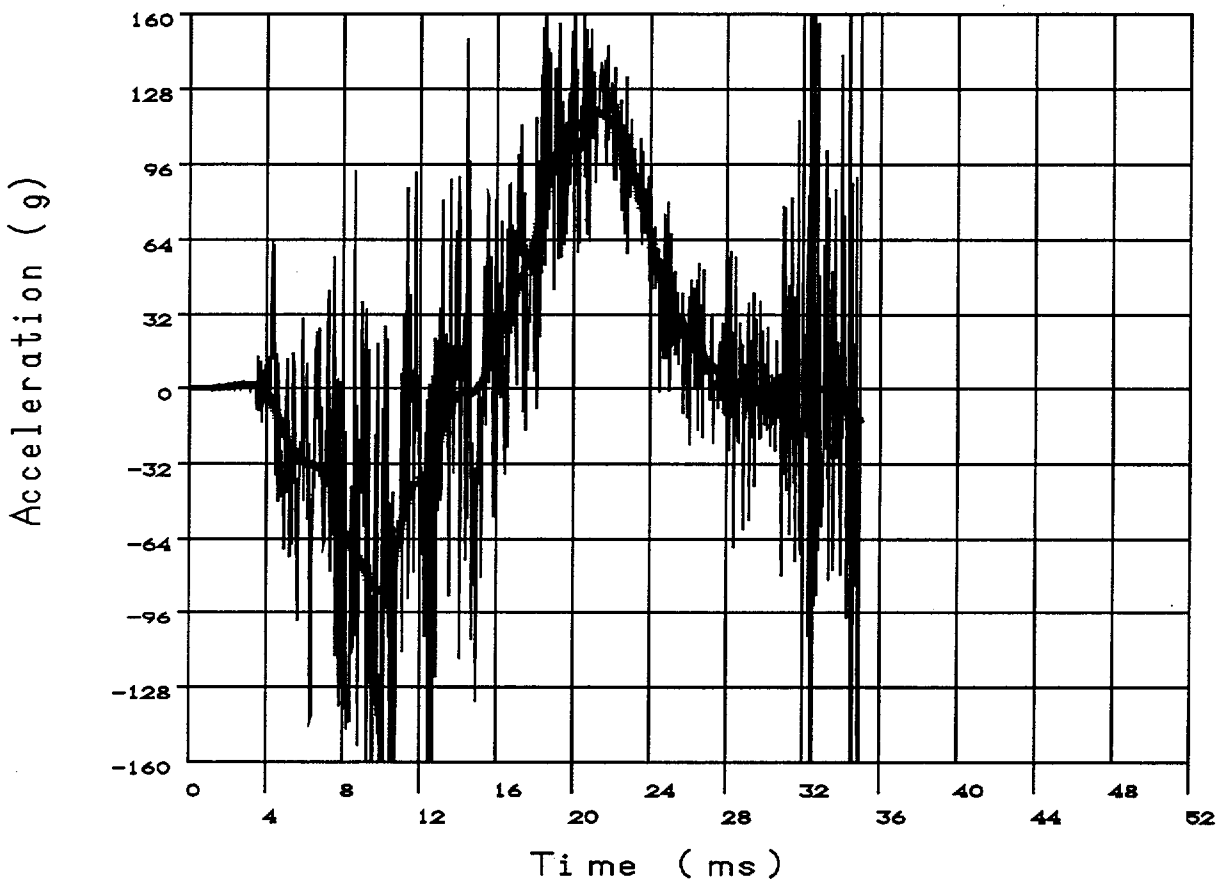
NAC Response to RAI 2-51 (Continued)

Figure 2.51-20 Corner Drop Filtered Acceleration Data



Accelerometer:Corner3

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CHAPTER 2: STRUCTURAL

Section 2.10.3 Confirmatory Testing Program—UMS Impact Limiters and Attachments

2-52 Submit the force-deflection curves for the impact limiter models under the static crush test configurations for the side and oblique drops.

In addition to the end drop static test results presented in Figures 2.10.3-3 and -4, appropriate static test results should be shown to correlate adequately the RBCUBED calculated force-deflection curves of Figures 2.6.7.5-16 and -17, for the oblique and side drops, respectively. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the testing program intended for confirming the calculated package performance under the free drops of Sections 71.71(c)(7) and 71.73(c)(1).

NAC Response

The force-deflection curves for the upper and lower impact limiter side drop at test conditions (70°F) are added to Section 2.10.3.8.1 as Figures 2.10.3-9 and 2.10.3-10, respectively.

Sections 2.10.3.8 and 2.10.3.10 are revised to show the correlation between the results from the quarter-scale tests and the values predicted using the RBCUBED program and to demonstrate that the RBCUBED-derived accelerations are conservative for use in subsequent structural analyses.

Section 2.10.3.9 demonstrates that the accelerations associated with an oblique drop are bounded by the accelerations associated with a side drop, as evaluated in Section 2.10.3.8.

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CHAPTER 2: STRUCTURAL

Section 2.10.3 Confirmatory Testing Program—UMS Impact Limiters and Attachments

2-53 Clarify, as appropriate, the underlined typographical or editorial errors.

Pg. 2.10.3-30, maximum accelerations summary in the data correlation table, "[U]pper impact limiter (peak positive or negative g values)...86...49.57...50.95..."

The referenced SAR tables and figures suggest that some of the listed acceleration peak values are not related to the cask top-corner drop. Complete and accurate information should be presented, per Section 71.7(a).

NAC Response

The calculated full-scale acceleration shown on Page 2.10.3-30 was incorrectly transcribed from the original analysis. However, as required by the response to RAI 2-32, the full-scale acceleration was revised to include RBCUBED results at an impact limiter temperature of 70°F, which is consistent with the test temperature.

The revised full-scale acceleration calculated by RBCUBED at 70°F is 47.8g. The remaining values shown in the table on Page 2.10.3-30 are correct.

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CHAPTER 2: STRUCTURAL

Section 2.11 Site-Specific Contents Structural Evaluation

2-54 Considering sectional (primary membrane and membrane-plus-bending), in lieu of nodal, stresses in the support disk ligaments, re-evaluate normalized stress ratios in Table 2.11.1.1-1 for the Maine Yankee consolidated fuel.

The PWR support disk ligaments are evaluated with sectional stresses for the design basis spent fuel assemblies. When normalized stress ratios are considered in comparing relative structural performance, a consistent evaluation approach should be maintained throughout the SAR, including the Maine Yankee consolidated fuel. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package structural performance under the conditions and tests of Sections 71.71 and 71.73.

NAC Response

The parametric study of the support disk presented in Section 2.11.1.1 is revised to consider the support disk sectional stresses in lieu of nodal stresses. The normalized stress ratios in Table 2.11.1.1-1 are revised based on the sectional stress results. The number of cases evaluated is reduced from 12 to 4, since consolidated fuel is restricted to one of the four corner locations of the basket. As shown in Table 2.11.1.1-1, the stresses in the support disk for this configuration are bounded by the stresses in the support disk for the design basis PWR configuration.

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CHAPTER 2: STRUCTURAL

Section 2.11 Site-Specific Contents Structural Evaluation

2-55 Clarify the statement on Page 2.11.1-1, "[T]his study shows that a consolidated fuel assembly can be located in any position of the UMS PWR basket based on structural loading considerations."

Under a side drop, stresses in the support disk ligaments appear to be governed only by the locally applied equivalent inertia load of the design basis consolidated spent fuel assembly. As a result, because of the relatively large weight of the consolidated fuel lattice, some of the normalized stress ratios for the 12 fuel tube locations are expected to exceed 1.00, the stress ratio for Base Case. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package's structural performance under the conditions and tests of Sections 71.71 and 71.73.

NAC Response

Section 2.11.1.1 is revised to indicate that the consolidated fuel stored in a Maine Yankee fuel basket must be placed in one of the corner positions of the basket.

The stresses in the support disk ligaments during a side drop are governed predominantly by displacement (ovalization) of the disk, rather than the locally applied equivalent inertia load of a fuel assembly.

The pressure on the support disk ligament due to the inertia load (1g) of the UMS System design basis PWR fuel assembly (including the fuel tube) is 12.26 psi. The thickness of the support disk is 0.5 inch. There are three different widths of the ligament: 0.875 inch, 1.0 inch and 1.5 inches, depending on the position within the support disk. The length of the ligament is 9.272 inches. Considering the support disk ligament to be a beam with both ends fixed, subjected to a 20g side impact condition, the maximum bending moment (M) and bending stress (σ) in the ligament are:

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Ligament Height (inch)	M (inch-kip)	S ($inch^3$)	σ (ksi)
0.875	0.8783	0.0638	13.77
1.0	0.8783	0.0833	10.54
1.5	0.8783	0.1875	4.7

In this table, $M = wL^2/12$, S is the Section Modulus, and σ is the bending stress (M / S).

Where:

w = the force per unit length (20g) on the ligament $(0.01226\times0.5)\times20 = 0.1226$ kips/inch)
L = the length of the ligament (9.272 inches)
$S = bt^2/6$, where b is the ligament thickness and t is the ligament width.

As shown in the table, the maximum stress in the support disk ligament due to the locally applied inertia load is 13.77 ksi. This stress is well below the maximum stresses calculated by the three-dimensional canister/basket model for the side drop condition (see Section 2.6.13.6). As shown in Tables 2.6.13.6-3, 2.6.13.6-5, 2.6.13.6-7, 2.6.13.6-9, 2.6.13.6-11, 2.6.13.6-13, 2.6.13.6-15, and 2.6.13.6-17, the maximum P_m+P_b stress in the PWR support disk ligaments is 44.9 ksi, 52.4 ksi, 47.7 ksi and 56.9 ksi for the 0°, 18.22°, 26.28° and 45° basket drop orientations, respectively. Therefore, it is concluded that stresses in the support disk ligaments for a side impact are governed predominantly by the displacement (ovalization) of the disk.

The pressure on the support disk ligament due to equivalent inertia load (1g) of the Maine Yankee consolidated fuel, including the damaged fuel can and the fuel tube, is 17.0 psi. The consolidated fuel is limited to the corner positions of the basket, where the support disk ligament width is 1.5 inches.

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Using the formula above, the maximum bending stress in the ligament is calculated to be 6.5 ksi, an increase of only 1.8 ksi, compared with the maximum stress of 4.7 ksi for the UMS design basis loading as shown in the previous table.

Since the total weight (≈ 35,500 lbs) on the basket for the configuration of 23 Maine Yankee standard fuel assemblies and one consolidated fuel lattice is much less than the total weight of 24 UMS design basis fuel assemblies and fuel tubes (≈ 40,900 lbs), it is concluded that the maximum stress in the support disk for the Maine Yankee consolidated fuel configuration is bounded by the maximum stress in the support disk for the UMS design basis configuration. This is further demonstrated by re-performing the analysis using the PWR support disk model for the governing case (45° basket orientation and thermal condition B) for the side drop condition (Section 2.6.13.6).

See also the NAC Response to RAI 2-56.

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CHAPTER 2: STRUCTURAL

Section 2.11 Site-Specific Contents Structural Evaluation

2-56 Submit a stress summary table on maximum stresses in the support disk for location "Case 6" to demonstrate adequate stress margins for the corner-location preferential loading of the consolidated fuel.

An evaluation of normalized stress ratios, in Table 2.11.1-1, alone may not be sufficient to substantiate the SAR conclusion on maximum stresses in the support disk, and explicit stress margins should be considered for the evaluation. Complete and accurate information should be provided, per Section 71.7(a), for evaluating the package structural performance under the conditions and tests of Sections 71.71 and 71.73.

<u>NAC Response</u>

A support disk analysis is performed for the Maine Yankee fuel configuration consisting of 23 standard fuel assemblies and one consolidated fuel assembly, using the two-dimensional PWR support disk model for the governing case (45° basket orientation and thermal condition B) for the side drop condition (Section 2.6.13.6). The loading condition corresponds to Case 1 of the updated parametric study presented in Table 2.11.1.1-1 and discussed in Section 2.11.1.1 (equivalent to Case 6 in the previous study).

The analysis results of the P_m and P_m+P_b stresses are summarized in Tables 2.11.1.1-2 and 2.11.1.1-3, respectively. The minimum Margins of Safety for the P_m and $P_m + P_b$ stresses are +0.82 and +0.24, respectively.

The minimum Margins of Safety for the corresponding analysis for the UMS System design basis PWR configuration are +0.79 and +0.19 for P_m and $P_m + P_b$ stresses, respectively (See Tables 2.6.13.6-16 and 2.6.13.6-17). This comparison further substantiates the conclusion of the

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parametric study based on the normalized stress ratios using a two-dimensional model (Table 2.11.1.1-1).

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CHAPTER 2: STRUCTURAL

Section 2.11 Site-Specific Contents Structural Evaluation

2-57 Clarify, as appropriate, the underlined typographical or editorial errors.

Pg. 2.11.2-1, "[T]he center of gravity for…GTCC waste canister…identical to the C.G. for the transport cask containing PWR Class 1 fuel (107.99 inches) as shown in Table 2.2-1."

Table 2.2-1 lists the location of C.G. at 106.60 inches from the bottom of the cask body; complete and accurate information should be provided, per Section 71.7(a).

NAC Response

Section 2.11.2.1 is revised to clarify the comparison between the center of gravity (C.G.) for the Transport Cask with the PWR fuel Class 1 canister and the Greater Than Class C (GTCC) waste canister. Table 2.2-3 is added to Section 2.2 to provide the weight and CG information for the GTCC waste configuration.

See the NAC Response to RAI 2-8.

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CHAPTER 3: THERMAL

Section 3.1 Discussion

3-1 Justify the use of a 10-year cooling time as a parameter of the design basis fuel.

The spent fuel contents in Chapter 1 have a minimum cool time of 6 years. The Thermal Section design basis fuel cool time of 10 years does not seem to bound the contents listed in Chapter 1. Section 71.7(a) requires complete and accurate information.

NAC Response

The thermal evaluations are based on a maximum heat load of 20 kW for PWR fuel and 16 kW for BWR fuel. Reference to the design basis burnup and cool time is removed from this section, since a number of combinations of enrichment, burnup, fuel mass and cool time can result in a given heat load.

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CHAPTER 3: THERMAL

Section 3.1 Discussion

3-2 Provide justification for using a normal cladding temperature limit of 1058°F, including calculations from the methodology of the stated reference.

Typically, the cladding temperature limit of 1058°F is only for short-term accident conditions, short-term off-normal conditions, or fuel transfer operations (like vacuum drying or dry transfer) - refer to NUREG-1536. Transportation of spent fuel, which can last for a period of 1 year, may not be a short-term event, and use of this higher temperature limit is not justified. The criticality analysis assumes that the fuel geometry remains intact, and the staff needs this information to complete the Section 71.51(a) and 71.55 determinations.

The staff recognizes that the transportation regulations in 10 CFR Part 71 do not have any specific requirements for ensuring that the spent fuel cladding is maintained below its temperature limit. However, dual-purpose canisters must meet both transportation and storage requirements; and 10 CFR Part 72 requires that cladding be protected during storage such that its degradation would not pose any operational safety problems with respect to its removal from storage (Section 72.122(h)(1)). Also, storage systems must be designed to allow ready retrieval of spent fuel for disposal (Section 72.122(l)). Therefore, for dual-purpose canisters one can readily deduce that cladding temperature limits that are imposed to prevent cladding damage during storage must also be met during transportation, especially if that fuel is to be stored post-transport.

In addition, the guidance in Section 3.5.2.3 of the SRP (NUREG-1617), which is used to determine regulatory adequacy, requires that "...the maximum allowable fuel/cladding temperature is justified. The justification should consider the fuel and clad materials, irradiation conditions (e.g., the absorbed dose, neutron spectrum, and fuel burnup), and the shipping environment including the fill gas. Other necessary considerations include the elapsed time from removal of the spent nuclear fuel from the core to its placement into the transportation packaging, its time duration in the packaging, and its post-transport disposition."

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RAI 3-2 (Continued)

NAC Response

The allowable fuel cladding temperature limit for normal conditions of transport is revised to the long term dry fuel storage temperature limit. The revised maximum allowable decay heats as a function of burnup and cool time are presented in Section 3.4.6. Tables 1.2-6, 1.2-7, 5.4-21 and 5.4-22, showing fuel minimum cool times, are revised accordingly.

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CHAPTER 3: THERMAL

Section 3.1 Discussion

3-3 Justify in the Thermal Section why the 16 kW and 20 kW limits bound the fuels to be transported. The justification should include considerations of the variable cooling times and enrichments.

This information is necessary to verify that the thermal analysis is bounding for the contents to be transported and the conditions specified in Sections 71.71 and 71.73.

NAC Response

The total heat load limits of 16 kW for BWR fuel and 20 kW for PWR fuel establish the basis by which the spent fuel parameters, such as enrichment and cooling time, are determined. Since the total heat load establishes the basis by which the limiting values of fuel enrichment, burnup and cooling time are determined, the heat load limit for each fuel type is inherently bounding.

The evaluation of the BWR and PWR fuel, based on the respective 16 kW and 20 kW total heat loads, is provided in Section 5.4.3 for the UMS design basis spent fuel, and in Section 5.5.1 for the Maine Yankee spent fuel. These evaluations result in the development of loading tables such as those presented in Tables 1.2-6 and 1.2-7 and Tables 5.4-21 and 5.4-22. Loading of the UMS System in accordance with these tables ensures a total heat load equal to, or less than, the limiting values.

The total heat loads and the results of the evaluations provided in Sections 5.4.3 and 5.5.1.1, are applied in Chapter 3 primarily for the purpose of determining the corresponding temperatures of key components and evaluating system heat rejection. Consequently, Chapter 3 does not present an additional discussion of the spent fuel parameters and loading limits.

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CHAPTER 3: THERMAL

Section 3.2.3 Convective Properties

3-4 Provide verification of the properties and title of Table 3.2-4 as necessary to reflect the properties of Chemical Copper-grade Lead as opposed to chemical lead. Section 71.7(a) requires complete and accurate information.

NAC Response

The lead, originally specified as Chemical Lead provided in accordance with ASTM B29-79, can no longer be procured to that specification, but is now available as Chemical Copper grade lead provided in accordance with ASTM B29-92. This specification is called out on NAC-UMS® Drawing 790-502. The title of Table 3.2-4 is revised to reflect Chemical Copper lead.

Comparison of the two ASTM lead specifications shows that the chemical and physical properties of the lead are identical. The properties of the Chemical Copper grade lead are as specified in Table 1 of ASTM B29-92, "Standard Specification for Refined Lead."

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CHAPTER 3: THERMAL

Section 3.3.2 Safe Operating Ranges

3-5 Justify why the support disks are not within the safe operating temperature range and are not designated in Section 3.3.2 as a component that must be maintained within the safe operating temperature range.

For example, Table 3.4-1 lists the support disk interior temperature to be 686°F. Table 3.4-3 lists the support disk allowable range as -40°F to 650°F. Section 71.71(a) requires a determination of the effects on the design of the conditions and tests associated with the normal conditions of transport.

NAC Response

Section 3.3.2 is revised to include the PWR and BWR support disks as components that must be maintained within the safe operating temperature range.

Table 3.4-1 shows the calculated component temperatures with the design basis fuel loading in the PWR and BWR system configurations. For each configuration, component temperatures are provided for two cases. The first case assumes that the canister contains air. The second case assumes that the canister contains helium. The second case, with a helium atmosphere, is the design basis normal condition for the canister for both the PWR and BWR configurations. As indicated in Section 3.1, the results for the air case are conservatively presented for use in the structural evaluations in Section 2.0.

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CHAPTER 3: THERMAL

Section 3.3.2 Safe Operating Ranges

3-6 Correct typographical errors as needed for consistency between the table in this Section, and the reference included on pg. 4.5-7.

Pg. 4.5-7 lists the temperature range of the EPDM O-rings to be -65°F to 300°F. Section 71.7(a) requires complete and accurate information.

NAC Response

Section 3.3.2 is revised to correct the low temperature limit for the EPDM O-rings to be -65°F in accordance with the EPDM O-ring technical data presented in Section 4.5.2.

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CHAPTER 3: THERMAL

Section 3.3.2 Safe Operating Ranges

3-7 Justify the ability of the neutron shield to perform its function when it exceeds its safe operating range during normal conditions of transport. Provide the shielding performance capability at the expected temperature range.

At maximum PWR/BWR fuel decay and maximum ambient temperature, the temperature of the radial neutron shield exceeds the upper limit of the safe temperature range. Also, Section 3.5.2.3 of the SRP (NUREG-1617) states that "...the temperature range of the thermal and structural properties for each package material exceed the specified and predicted temperature limits for the material." Section 71.71(a) requires a determination of the effects on the design of the conditions and tests associated with the normal conditions of transport.

NAC Response

The thermal analysis using the three-dimensional cask model for a cask containing the hotter PWR fuel (Section 3.4.1.1.1) has been revised. Based on the results of the revised analysis, the maximum temperature for the neutron shield (NS-4-FR) material is 293°F. This temperature is below the allowable temperature of 300°F.

The revised analysis recalculates the effective thermal conductivity of the neutron shield region and it applies the heat flux simulating the solar insolation to the cask side surface based on a cosine distribution.

The effective thermal conductivity of the neutron shield region is recalculated to remove a specific conservatism. The previous calculation conservatively assumed a gap of 0.125 inch between the end of the copper plate section of the heat transfer fin and the neutron shield

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shell; as fabricated, the copper is effectively in contact with the neutron shield shell. The effective thermal conductivity of the neutron shield region is re-calculated with the 0.125 (1/8) inch gap size reduced to a value of 0.031 inch.

The heat flux simulating the solar insolation is applied to the cask side surface based on a cosine distribution (in lieu of a uniform distribution). This is justified, since the cask surface is subjected to maximum insolation at the top and minimum (zero) insolation at the bottom while it is in the horizontal (normal transport) position.

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CHAPTER 3: THERMAL

Section 3.3.2 Safe Operating Ranges

3-8 Justify the assumption that retaining the radial neutron shield during a 30-minute fire transient, and removing it afterward is the most conservative approach for determining component temperatures. Provide support via calculations.

The staff agrees that in a longer term fire transient, the approach may be conservative. However, over a short period of time, the shield may act more as a barrier as the shield itself is heated. Section 71.41(a) requires an evaluation of the effects on the package of the tests specified in Section 71.73.

NAC Response

A thermal transient sensitivity analysis was performed using the finite element model described in Section 3.5.1. The analysis was performed with the radial neutron shield (NS-4-FR) removed during the 30-minute fire. The analysis results indicate a slight reduction in the maximum temperature for all components. The maximum reduction occurs at the lead (5°F) and the inner shell (4°F). Therefore, it is conservative to perform the thermal transient analysis with the radial neutron shield in place during the 30-minute fire and remove it at the end of the fire event.

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CHAPTER 3: THERMAL

Section 3.4.1 Thermal Models

3-9 Justify neglecting the personnel barrier in the models that arrive at component temperatures during the normal conditions of transport.

Section 3.5.4 of the Standard Review Plan suggests that the personnel barrier should be considered when determining the package temperatures for normal conditions of transport but should be neglected during the hypothetical accident.

NAC Response

The governing thermal condition for the normal conditions of transport is 38°C (100°F) still air with insolation in accordance with 10 CFR 71.71(c)(1). The personnel barrier design for the UMS® Transport Cask is expected to reduce the solar insolation by approximately 0.93 kW (40% of the total insolation of 2.3 kW). At the cask surface, heat is rejected mainly by natural convection and radiation. The convection heat transfer is not expected to be affected by the existence of the personnel barrier. The thermal radiation heat rejection path is changed, since instead of radiating directly to the ambient, a portion of the heat is radiated from the cask surface to the personnel barrier, and then from the personnel barrier to the surrounding air. The personnel barrier structure also serves as a heat transfer fin to reject heat to the surrounding air primarily by convection.

Therefore, the existence of the personnel barrier is expected to have an insignificant effect on the component temperatures of the cask during normal conditions of transport. The thermal models described in Section 3.4.1 are adequate to predict the maximum component temperatures.

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Section 3.4.1.4 Personnel Barrier Thermal Model

3-10 Include thermal radiation in the analysis of the accessible surface temperature.

In calculating the temperature of the personnel barrier, it is not conservative to neglect thermal radiation from the cask to the barrier. Section 3.5.4 of the Standard Review Plan suggests that the model consist of a heat balance at the surface of the package (at the personnel barrier) between the content decay heat and the convective and radiative heat losses to the environment at 100°F. Section 71.33 requires a description in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation.

NAC Response

Section 3.4.1.4 is revised to include thermal radiation from the cask surface to the personnel barrier. The thermal model is revised as shown in Figure 3.4-11.

See the response to RAI 3-11.

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Section 3.4.1.4 Personnel Barrier Thermal Model

3-11 Specify the distance between the personnel barrier and the cask at the centerline of the model. Provide a figure similar to Figure 3.4-11 that includes temperatures at key points. Further, justify averaging the temperatures along the top of the personnel barrier to come up with a reported value which is only 3°F below the 122°F limit that the analysis in this Section is intended to meet.

The staff could not find a distance between the personnel barrier and the cask at the centerline. The method of averaging the temperatures does not seem justified if the personnel barrier is positioned close to the cask. It is not clear if the hottest region of the barrier would exceed the compliance criterion. Section 71.33 requires a description in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation.

NAC Response

The distance between the personnel barrier and the cask at the centerline of the model (10 inches) is specified in Figure 3.4-11. Figure 3.4-12 is added to show the calculated temperatures at key points on the personnel barrier. The maximum temperature of the personnel barrier occurs at the top centerline and is calculated to be 153°F. This temperature is below the accessible package surface allowable temperature of 185°F specified in 10 CFR 71.43 for exclusive use shipments.

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Section 3.4.4.1 Maximum Internal Pressure for PWR Fuel Canister and Cask

3-12 Review the normal conditions of transport and hypothetical accident conditions pressure calculations for appropriate term usage.

For example, pg. 3.4-28, should indicate 5,968 ℓ/canister. Another example, pg. 3.4-31, V_{UTC} Free Gas Volume calc, should be 1,096.25 ℓ/cask, as was appropriately used in the next equation determining the molar quantity of gas in the cask. These errors occur in both the PWR and BWR pressure calculations. Section 71.7(a) requires complete and accurate information.

NAC Response

Sections 3.4.4.1 and 3.4.4.2 have been reviewed and revised as necessary to incorporate appropriate term usage.

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Table 3.4.1 **Summary Table of Temperatures – Maximum Component Temperatures - Normal Condition of Transport, Maximum Decay Heat, Maximum Ambient Temperature**

3-13 Correct typographical errors as needed for consistency of temperatures between Tables 3.4-1, 2.6.1.1-1 and 3.4-3.

The guidance provided by the SRP suggests that the summary tables of temperatures of package components in the Thermal Section of the SAR must be consistent with the temperatures presented in the General Information Section and the Structural Evaluation Sections of the SAR for the normal conditions of transport and hypothetical accident conditions. Section 71.7(a) requires complete and accurate information.

NAC Response

Tables 2.6.1.1-1 and 3.4-1 are revised to correct several transcription errors in the component temperatures which are reported in these tables. In addition, Table 2.6.1.1-1 is revised to incorporate the same components as presented in Table 3.4-1 so that these tables are directly comparable.

Table 3.4-3 is revised to incorporate the component temperature limits for the stainless steel PWR support disk and the carbon steel BWR support disk. See the response to RAI 3-5.

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Section 3.4.6 Evaluation of Package Performance for Normal Conditions of Transport

3-14 Submit an analysis that justifies mixed loadings (i.e., spent fuel with longer cooling times and with lower temperature limits being loaded into a canister with shorter cooled spent fuel) are bounded. Indicate the effect the results have on the loading tables.

Since this transportation application includes spent fuel that is substantially different from the counterpart storage application (e.g., burnup of 50 vs 45 GWD/MTU) such a comparison is warranted.

The current acceptable standard for establishing spent fuel cladding temperature limits is PNL-6189 which determines a range of temperature limits depending on variations in fuel design, burnup level, cooling time, and storage cask design. Since multiple temperature limits are likely to exist for a given dual-purpose canister design, it would be insightful to understand the possible limitations on loading the canister with fuel assemblies with different cladding temperature limits to ensure that the lowest cladding temperature limit would not be exceeded. Section 71.33 requires a description in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation.

NAC Response

As noted in the NAC Response to RAI 2-12 and others, reference to the in-pool loading of the canister using the UMS® Universal Transport Cask is deleted. Consequently, at the time of transport, the canister will have previously been loaded, closed and tested in accordance with the detailed requirements presented in the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015. The Storage System Safety Analysis Report (Section B2.12, Chapter 12) contains detailed, specific requirements and limits related to the spent fuel loaded into the canister, including preferential loading requirements for spent fuel with variable cool times in the same canister.

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NAC Response to RAI 3-14 (Continued)

Loading of the fuel assemblies designated for a given canister must be administratively controlled to ensure that the dry storage fuel cladding temperature limits are not exceeded for any fuel assembly, unless all of the designated fuel assemblies have a cooling time of 7 years or more. Canisters containing fuel assemblies, all of which have a cooling time of 7 years, or more, do not require preferential loading because analyses have shown that the fuel cladding temperature limits will always be met for those canisters.

For the transport cask, three thermal analyses (3 Cases) have been performed using the three-dimensional cask model for the cask containing PWR fuel (Section 3.4.1.1.1). Where referenced, basket locations refer to those shown in Figure 3.14-1 of this response. For this analysis, heat load per assembly is based on Table 3.4-10: 5-year cooled fuel: 0.833 (20/24) kW; 6-year cooled fuel: 0.8125 (19.5/24) kW; 7-year cooled fuel: 0.742 (17.8/24) kW and 15-year cooled fuel: 0.704 (16.9/24) kW.

Case 1 considers the loading pattern having 5-year cooled fuel in the center of the basket (Locations 1 and 7), 7-year cooled fuel in the periphery positions (Location 3, 5, 6, 9, 11, and 12) and 6-year cooled fuel in the intermediate positions (Locations 2, 4, 8, and 10). The heat load for each fuel assembly is determined based on the maximum allowable heat load as shown in Table 3.4-10. The allowable temperatures are obtained from Table 3.4-9. The calculated maximum temperature at each fuel position, and the comparison with allowable temperature, are:

Basket Location	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**
Fuel Cool Time (Years)	5	6	7	6	7	7	5	6	7	6	7	7
T_{max} (°F)	654	613	544	628	573	571	636	594	524	575	518	485
$T_{allowable}$ (°F)	694	680	644	680	644	644	694	680	644	680	644	644

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Case 2 considers the loading pattern having 5-year cooled fuel in the center of the basket (Locations 1 and 7) and 7-year cooled fuel in all other basket positions (Locations 2 to 6 and 8 to 12). The calculated maximum temperature at each position and the corresponding allowable temperature are:

Basket Location	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**
Fuel Cool Time (Years)	5	7	7	7	7	7	5	7	7	7	7	7
T_{max} (°F)	642	595	536	609	564	562	624	576	516	557	510	478
$T_{allowable}$ (°F)	694	644	644	644	644	644	694	644	644	644	644	644

Case 3 considers the loading pattern having 15-year cooled fuel in the center of the basket (Locations 1 and 7) and 7-year cooled fuel in all other basket positions (Locations 2 to 6 and 8 to 12). The calculated maximum temperature at each position and the corresponding allowable temperature are:

Basket Location	**1**	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**
Fuel Cool Time (Years)	15	7	7	7	7	7	15	7	7	7	7	7
T_{max} (°F)	610	583	528	596	555	553	593	564	508	546	502	471
$T_{allowable}$ (°F)	624	644	644	644	644	644	624	644	644	644	644	644

Cases 1 and 2 bound all possible loading configurations for fuel assemblies with a cool time of 7 years or less, which require preferential loading. These results show that the maximum fuel cladding temperature for each assembly will not exceed its allowable temperature, if fuel loading is administratively controlled such that fuel with the shortest cooling time is placed in the center positions of the basket and fuel with the longest cooling time is placed in the periphery positions.

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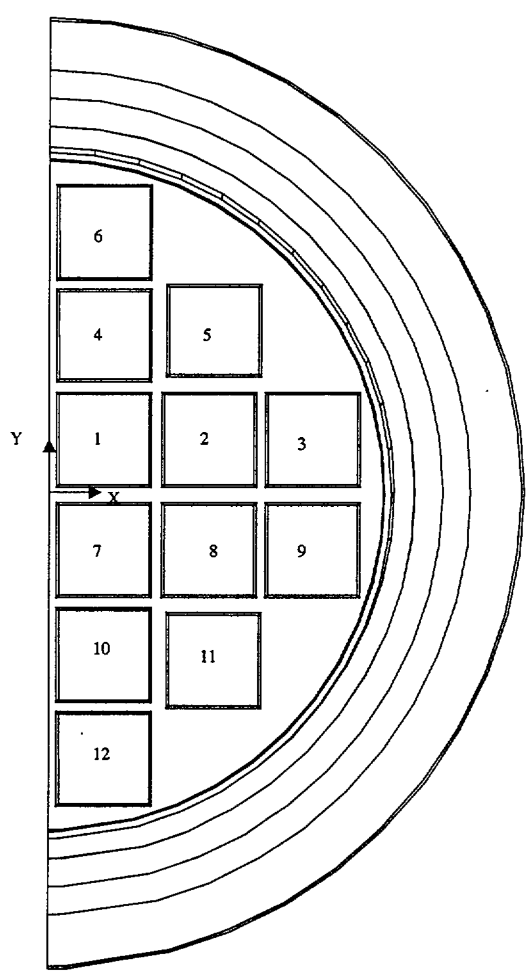
Case 3 represents a bounding configuration for canisters containing fuel assemblies, all of which have a cooling time of 7 years or more. The analysis results show that no preferential loading is required.

Figure 3.14-1 Basket Fuel Loading Location (PWR Fuel)



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Section 3.4.7 Thermal Evaluation – Direct Loading of the Universal Transport Cask

3-15 Demonstrate that the toughness of ASTM A693 17-4 PH stainless steel, used for the PWR support disks, will be sufficient to withstand the hypothetical accident conditions after being exposed to temperatures near the prescribed short-term limit of 800°F during transfer operations. Clarify the temperature limits specified in Tables 3.4-3 and 3.4-9, and provide justifications for using those values.

As required in Section 71.43(f), the package must be designed, constructed, and prepared for transport so there will be no significant decrease in packaging effectiveness under normal conditions of transport. ASME Boiler and Pressure Vessel Code, Section 11, Part D, Table TM-1 indicates that this material may have reduced toughness at room temperature after being exposed to temperatures above 650°F. Also, the value of the temperature limit for the 17-4 PH stainless steel support disks in Table 3.4-3 is inconsistent with the value in Table 3.4-9.

NAC Response

The in-pool direct loading of the UMS® Transport Cask (i.e., the loading of fuel into a Transportable Storage Canister that is positioned within the transport cask) was the only condition in which the temperature of the 17-4PH PWR support disks (739°F) approached the material's 800°F short-term temperature limit. As noted in the response to RAI's 2-12, 2-13, 2-14 and 2-16, the analysis and procedures associated with in-pool direct loading of the UMS® Transport Cask are deleted from the Safety Analysis Report. There is no current operational requirement for direct loading of the transport cask. Consequently, Section 3.4.7, "Thermal Evaluation – Direct Loading of the Universal Transport Cask," is removed from the Safety Analysis Report, which removes Figures 3.4-12 through 3.4-16 and Tables 3.4-5 through 3.4-10. Section 3.4.6, "Evaluation of Package Performance in Normal Conditions of Transport," is

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NAC Response to RAI 3-15 (Continued)

renumbered as Section 3.4.7. A new Section 3.4.6, "Maximum Allowable Cladding Temperature and Canister Heat Load," is inserted, including new Figures 3.4-12 through 3.4-15 and new Tables 3.4-5 through 3.4-10.

The maximum calculated temperature of the support disk in the transfer condition is 686°F in air, as shown in Table 4.1-4 of the Safety Analysis Report for the UMS® Universal Storage System (Docket 72-1015). This temperature is well below the 800°F short-term temperature limit. The fracture toughness of the ASTM A693 17-4PH stainless steel is addressed in detail in the NAC Response to RAI 4-2 for the UMS® Universal Storage System (RAI-1, October 30, 1998). As shown in the Response, 17-4PH has adequate structural toughness for this application.

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Section 3.5.1.1 Analytical Models

3-16 Clarify the method used to determine component temperatures during the hypothetical accident condition fire. Provide sufficient supporting calculation(s) to allow staff verifications of your results. Also, the explanation in the paragraph at the top of pg. 3.5-2 should be consistent with Footnote "b" of Tables 3.5-1 and 3.5-2.

Section 71.41(a) requires, in part, that the effects on a package of the hypothetical accident conditions must be evaluated by specific test or another acceptable method.

Clarification and supporting calculations are needed to allow the staff to independently verify your results.

NAC Response

The fuel basket and fuel are not modeled explicitly in the model described in Section 3.5.1.1. The maximum temperatures of the basket components and the fuel cladding are calculated by adding the maximum temperature difference between the cask inner shell and the component of interest from the normal condition results to the peak temperature of the inner shell in the hypothetical accident condition. Normal condition results are taken from Table 3.4-1 (Air Cases for PWR and BWR fuel).

The peak temperature of the inner shell in the hypothetical accident condition is 479°F for PWR fuel (Table 3.5-1). The maximum component temperatures PWR fuel are:

Component	Temperature Difference in Normal Conditions (°F)	Calculated Component Accident Temperature (°F)
Heat Transfer Disk	315 (683-368)	794 (315 + 479)
Support Disks	318 (686-368)	797 (318 + 479)
Fuel Cladding	440 (808-368)	919 (440 + 479)
Canister Gas (Avg)	124 (492-368)	603 (124 + 479)

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The peak temperature of the inner shell in the hypothetical accident condition is 462°F for BWR (Table 3.5-2). The maximum component temperatures for BWR fuel are:

Component	Temperature Difference in Normal Conditions (°F)	Calculated Component Accident Temperature (°F)
Heat Transfer Disk	272 (608-336)	734 (272 + 462)
Support Disks	274 (610-336)	736 (274 + 462)
Fuel Cladding	334 (670-336)	796 (334 + 462)
Canister Gas (Avg)	96 (432-336)	558 (96 + 462)

The description of the model provided in Section 3.5.1.1 is revised to provide a description of the method of calculating the component temperature in the hypothetical accident condition.

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Section 3.5.4.2.2 Maximum Internal Pressure for Cask with BWR Fuel Canister (100% Fuel Rod Failure)

3-17 Revise the pressure calculation of this Section, and enter the resulting pressure in Table 3.5-3 and consistently in other Chapters.

The cask volume as reported on pg. 3.4-43 (6895.73 liters/cask) should be used, not the canister volume. The staff recognizes that this error results in a conservative pressure. Section 71.7(a) requires complete and accurate information.

NAC Response

Section 3.5.4.2.2 and Table 3.5-3 are revised to incorporate the use of the correct cask volume.

See the response to RAI 3-12.

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Section 3.6.1.1 Maine Yankee Site-Specific Spent Fuel

3-18 Consolidated Fuel

Justify the use of the 17xl7 model used to obtain the effective conductivity of the consolidated fuel lattice with stainless steel rods at the perimeter. Re-assess the maximum fuel cladding temperature for all affected cases if needed.

The lattice model contains 41.5% stainless steel rods, while the actual case being analyzed is only 30% stainless steel dummy rods. This does not seem to be a conservative representation of the effective conductivity of the consolidated lattice. Section 71.33 requires a description in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation.

NAC Response

A sensitivity study analysis was performed using the three-dimensional thermal model (see Figure 3.6.1.1-1) and a modified version of the Maine Yankee consolidated fuel assembly thermal model. For this study, the Maine Yankee consolidated fuel assembly thermal model was modified such that only the outer layer of rods are modeled as being solid stainless steel, while the remainder are modeled as fuel rods. All five cases (i.e., "Base Case," and Cases 2 through 5) presented in Section 3.6.1.1 were re-analyzed using the effective thermal conductivities generated by the modified consolidated fuel model.

The modified model represents a fuel assembly with 64 total solid stainless steel rods in the outer layer. Therefore, there are approximately 22% (64/289) of the locations modeled as stainless steel while the remainder are modeled as regular fuel rods. The original model had 41.5% of the rods modeled as stainless steel (120/289).

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NAC Response to RAI 3-18 (Continued)

The effective thermal conductivities calculated using the modified model were compared to those calculated using the model presented in Section 3.6.1.1 (i.e., the outer two layers of rods modeled as solid stainless steel rods). This comparison shows that the modified model generates an effective thermal conductivity in the transverse direction of the fuel assembly that is approximately 20% to 22% greater over a temperature range of 400°F to 700°F, than the original model. Additionally, the effective thermal conductivity in the axial direction is approximately 16% to 19% less than the original model over the same temperature range.

The results of the analysis indicate that the maximum fuel cladding temperature for each of the five cases is within 0.1°F of the results using the original Maine Yankee consolidated fuel assembly thermal model. Therefore, it is concluded that the thermal analysis presented in Section 3.6.1.1 adequately describes the thermal performance of the UMS Transport Cask with Maine Yankee site specific contents.

Section 3.6.1.1 is revised to include a description and the results of the sensitivity study.

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CHAPTER 3: THERMAL

Section 3.6.1.1 Maine Yankee Site-Specific Spent Fuel

3-19 Standard Fuel assemblies that have been repaired by removing damaged fuel rods and replacing them with stainless steel dummy rods, solid zirconium rods, or 1.95 wt% enriched fuel rods.

Justify that cladding temperatures remain acceptable for replacement rods with low enrichment and variable burnup into a cask population of possibly cooler fuel, with lower temperature limits.

Section 71.33 requires a description in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation.

NAC Response

Section 3.6.1.1 is revised to clarify that the low enriched fuel rods do not affect the over all heat load and the maximum fuel cladding temperature of the modified fuel assembly. The 1.95 wt % ^{235}U enriched fuel rods replaced other fuel rods in the fuel assembly after the first or second burnup cycles were completed. Therefore, these replacement fuel rods were burned a minimum of one cycle less than the remainder of the fuel rods in the host fuel assembly, producing a proportionally lower per fuel rod heat load. The heat load (on a per rod basis) of the fuel rods in a standard assembly, bounds the heat load of the 1.95 wt % ^{235}U enriched fuel rods. Consequently, the maximum fuel cladding temperature of the standard fuel assembly bounds that of the modified fuel assembly.

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CHAPTER 4: CONTAINMENT

Chapter 4.0 Containment

4-1 Demonstrate that the design basis PWR and BWR fuels, as described in Section 4.0 of the containment analyses, bound the contents as described in Section 1.2.3. Revise Sections 4.0 and 1.2.3, as appropriate, to include all relevant points of this analysis. If the design basis fuels as described in Section 4.0 do not bound the contents as described in Section 1.2.3, recalculate the leakage rates for normal and hypothetical accident conditions.

Section 71.33 requires a description of the package in sufficient detail to provide an adequate basis for its review. As indicated in Section 4.0, the design basis BWR and PWR fuels are cooled 10 years at 50,000 MWD/MTU. However, Section 1.2.3 indicates that spent fuel contents have a maximum burnup of 50,000 MWd/MTU and a minimum cool time of 6 years.

<u>NAC Response</u>

Introductory Section 4.0 is revised to show that the containment evaluation is made based on PWR and BWR fuel having a cool time of 5 years, a minimum enrichment of 1.9 wt % ^{235}U and a burnup of 50,000 MWD/MTU.

The use of these parameters, specifically the 5-year cooling time, results in a containment analysis that bounds all of the PWR and BWR fuels, including the design basis fuel presented in Table 1.2-6 (Section 1.2.3). The minimum cool time used in Table 1.2-6 is also 5 years. The minimum cool time shown in Table 1.2-7 is 6 years. The 5-year cool time assumed for the BWR fuel conservatively bounds the BWR fuel considered for transport.

The change in cool time also results in changes to the fuel release source terms shown in Tables 4.2-2 (PWR fuel) and 4.2-3 (BWR fuel) for normal conditions. The corresponding tables for accident condition releases are also revised (Tables 4.3-1 and 4.3-2 for PWR and BWR fuel, respectively).

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CHAPTER 4: CONTAINMENT

Chapter 4.0 Containment

4-2 Demonstrate that the design basis PWR and BWR fuels, as described in Section 4.0 of the containment analyses, bound all of the contents, including GTCC waste, that are described in Section 1.3.l. This can be accomplished by comparing the source terms and leakage rates of the contents in Section 1.3.1 to the source terms and leakage rates of the contents of the design basis fuel. If the design basis PWR and BWR fuels described in Section 4.0 do not bound the contents described in Section 1.3.1, recalculate the leakage rates for normal and hypothetical accident conditions. Revise Sections 4.0 and 1.2.3, as appropriate, to include all relevant points of this analysis.

Section 71.33 requires a description of the package in sufficient detail to provide an adequate basis for its review. There is insufficient technical basis to show that, from a containment analysis perspective, the contents described in Section 1.2.3 are bounded by design basis PWR and BWR fuels.

NAC Response

As described in the response to RAI 4-1, a bounding containment analysis is performed by using a 50,000 MWD/MTU burnup, a 1.9 wt % ^{235}U minimum enrichment, and a cool time of 5 years. The use of these parameters results in a containment analysis that bounds the remaining fuel that is considered for transport.

Containment calculations for the GTCC waste are not performed since the majority of the GTCC waste activity is due to activation of cobalt within the stainless steel plates. This activity is not released under any normal condition of transport or hypothetical accident condition.

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CHAPTER 4: CONTAINMENT

Chapter 4.0 Containment

4-3 Update the SAR to reflect the correct value of the minimum enrichment for BWR fuel.

Section 71.7(a) requires complete and accurate information. Two different values for the minimum enrichment for BWR fuel (e.g., 1.9% and 3.25%) are specified in this Section.

NAC Response

Section 4.0 is revised to correct the lower assembly average enrichment limit for BWR fuel to 1.9 wt % ^{235}U.

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CHAPTER 4: CONTAINMENT

Section 4.1 Containment Boundary

4-4 Revise the SAR to indicate that all leak tests will be performed in accordance with ANSI Nl4.5-1997. Describe how the confinement analyses, as presented in Chapter 4, comply with the ANSI N14.5-1997 standard.

Section 71.31(c) requires identification of all established codes and standards applicable to the containment design. Currently, there is no reference to the ANSI N14.5-1997 standard in the Containment Section. This change is requested to update the SAR to current staff guidance that is specified in Interim Staff Guidance No. 11.

NAC Response

Sections 4.1.3.1.2 and 4.1.3.1.3 are revised to specify that leak testing for post-fabrication acceptance, annual maintenance and prior to transport are performed in accordance with ANSI N14.5-1997.

The allowable leak rate analyses presented in Sections 4.2 and 4.3 for normal transport and accident conditions, respectively, utilize the methodology of ANSI N14.5-1997. These Sections are revised to incorporate more specific reference to the ANSI Standard as appropriate. As noted in Sections 4.2 and 4.3, the concentration of nuclides considered in the analyses is based on guidance contained in NUREG-6487.

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CHAPTER 4: CONTAINMENT

Section 4.1 Containment Boundary

4-5 Justify why the cask lid bolts are considered part of the containment boundary.

Section 71.7(a) requires complete and accurate information. It is unclear why the cask lid bolts are considered to be part of the containment boundary.

NAC Response

Section 4.1 is revised to delete identification of the lid bolts, vent port coverplate bolts and the drain port coverplate bolts as components of the cask containment boundary. These bolts are outside of the cask containment boundary.

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CHAPTER 4: CONTAINMENT

Section 4.1.3.1 Seals

4-6 In a table, clarify the leak test criteria (i.e., air standard leak rate) and the test frequency for the containment system fabrication verification, periodic, maintenance, and pre-shipment leak tests. Also, in this table, indicate the corresponding replacement schedule for the o-rings. If there is a different allowable leakage rate for different contents (e.g., PWR fuel, BWR fuel, GTCC waste, or site-specific fuel), also note this in the table. Revise the appropriate Sections of Chapters 7 and 8 to clearly indicate the leak rate criteria for each type of test that will be performed and the seal replacement schedule.

Section 71.7(a) requires complete and accurate information. It is unclear what criteria are applied to each of the containment system fabrication verification, periodic, maintenance and pre-shipment leak tests. As an example of an area that needs clarification, Section 4.1.3.1.1 refers to the containment system fabrication verification test as described in Section 8.1.3, and Section 4.1.3.1.2 refers to the leakage test procedures as described in Section 7.1.3. However, Section 8.1.3 does not clearly describe any of the containment system leak tests, and Section 7.1.3 contains incorrect leak rates.

NAC Response

Table 4.1-1 is added to Chapter 4 and Table 7-2 is added to Chapter 7 to indicate the allowable leak test rates for post-fabrication, annual maintenance and prior to shipment. These tables also indicate the requirements for o-ring replacement. Other sections are revised as necessary to clarify the leak test requirements.

As noted in the tables, the allowable leak rate is based on the bounding GE 9 x 9 BWR fuel assembly containment requirement. This leak rate is conservatively applied to PWR fuel and GTCC waste transport.

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NAC Response to RAI 4-6 (Continued)

No containment analysis is performed for the GTCC waste since the majority of the GTCC waste activity is due to activation of cobalt within the stainless steel plates. This activity is not released under any normal condition of transport or hypothetical accident condition.

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CHAPTER 4: CONTAINMENT

Section 4.2.1 Containment of Radioactive Material

4-7 Include the values of the following parameters used in the containment analysis leakage rate calculations for normal conditions of transport: fraction of rods that develop breaches (f_b), crud surface activity (S_c), free volume inside the containment vessel (V), capillary length (a), the hole diameter (D), and the gas temperature (T) and upstream pressure (P_u) that were used to calculate the hole diameter. Include values for both PWR and BWR fuels, as appropriate.

Section 71.7(a) requires complete and accurate information. The values for these parameters are not presented. The staff needs the information to perform confirmatory analysis and to verify that bounding values were used for the containment analysis.

NAC Response

Tables 4.2-7 and 4.3-4 for the normal conditions of transport and accident conditions, respectively, have been added. These tables report principal parameters used in the leak rate calculation. The capillary (hole) diameter is determined using equations B-3 and B-4 of ANSI N14.5-1997. The gas temperature and gas pressure are those conservatively assumed to exist at the test condition. The actual upstream pressure may be higher than the assumed 1 atmosphere (0.0 psig) due to heat up of the helium. Consequently, the driving pressure for the postulated leak is higher than the assumed value.

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CHAPTER 4: CONTAINMENT

Section 4.2.3 Containment Criteria

4-8 Include the calculations that were used to determine the leak rates and sensitivities for the PWR and BWR baskets under helium leak testing conditions. Also, list the value of the parameters used as inputs to the calculations.

Section 71.7(a) requires that the SAR contain complete and accurate information. Only the leak rate and sensitivities are included as a note to Table 4.2-4. There is not enough information to verify that these leakage rates are bounding for test conditions. The staff needs the information to perform confirmatory analysis and to verify that bounding values were used for the containment analysis.

NAC Response

The principal parameters used to establish the allowable leak rate are shown in Tables 4.2-7 and 4.3-4 provided in the response to RAI 4-7. In addition, the values for fines, particulates, volatiles and gases are revised to account for the 5-year cool time for the PWR and BWR fuel. The methodology of calculation is that specified by ANSI N14.5-1997 and presented in Section 4.2.1. The method is summarized in Section 4.2.1.2.

While both the PWR and BWR fuel allowable leak rates are determined, the BWR fuel allowable leak rate is conservatively selected since it bounds that of the PWR fuel.

Consistent with the requirements of ANSI N14.5-1997, Paragraph 8.4, the sensitivity is set at one half of the allowable leak rate. This sensitivity is the minimum leak rate that the leak detector must be capable of detecting in the test conditions.

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CHAPTER 4: CONTAINMENT

Section 4.5.3 SAS2H Output and Group A_2 Values for B&W 15x15 and GE 9x9 Assemblies

4-9 Include the SAS2H input files for PWR and BWR design basis fuel source terms.

Section 71.7(a) requires complete and accurate information. The input files for PWR and BWR design basis fuel source terms are not presented. The staff needs the input files to verify that the design basis fuels represent the bounding source terms for the containment analysis.

NAC Response

The SAS2H input files for the bounding PWR and BWR fuel assemblies are shown in Figures 4.5-1 and 4.5-2. The bounding fuel assembly parameters are a maximum design burnup of 50,000 MWD/MTU, a minimum enrichment of 1.9 wt % ^{235}U, and a 5-year cool time for both fuel types.

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CHAPTER 4: CONTAINMENT

Section 4.5.3 SAS2H Output and Group A_2 Values for B&W 15x15 and GE 9x9 Assemblies

4-10 In Table 4.5.3-5, correct the A_2 values, and recalculate the leakage rates, for transport casks containing BWR fuel.

Section 71.7(a) requires complete and accurate information. The incorrect A_2 values were used to calculate the Group A_2 for volatiles.

NAC Response

Table 4.5.3-5 is revised to correct a transcription error in which the isotopes were entered in alphabetical order without regard to the associated values in the table.

The calculated A_2 value for the volatiles group is correct as shown.

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CHAPTER 5: SHIELDING

Section 5.1.3 Results of Analysis

5-1 For Tables 5.1-1, 5.1-2, 5.1-3, 5.1-4, "PWR and BWR Maximum total dose rate summaries for normal and accident conditions": provide a figure identifying the locations of the maximum dose rate values. The staff needs the information to confirm that the application meets the external radiation standards of Section 71.47.

<u>NAC Response</u>

Section 5.1 is revised to incorporate Figures 5.1-1 and 5.1-2, which show the locations of the maximum dose rates relative to the cask body and transporter for the normal conditions of transport and hypothetical accident conditions, respectively. Because of the shielding provided by the canister shield and structural lids, the top surface cask dose rate is significantly lower than the cask side dose rate. Consequently, as shown in the figures, the maximum top dose rate is (radially) to the side of the cask.

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CHAPTER 5: SHIELDING

Section 5.2 Source Specification

5-2 Provide a list of resources that confirm justifying the Co-59 impurity level of 1.2 g/kg in the Type 304 stainless steel hardware.

Section 71.33 requires the identification of the maximum radioactivity of the package constituents. This information is necessary to confirm the validity of your results.

NAC Response

The value of 1.2 g/kg for cobalt impurity in stainless steel and Inconel materials used by NAC is based on a review of published literature. Further confirmation of the conservatism has been achieved via phone interviews with utility and fuel manufacturer representatives.

All of the information or data received for fuel assembly hardware materials indicate that the 1.2 g/kg cobalt impurity value is sufficiently conservative for fuel assembly hardware manufactured in the last 10 to 15 years.

A value of cobalt impurity of 4.7 g/kg is identified for Inconel-718 only, and is from a 30-year old literature source based on limited data points. The cobalt impurity value for stainless steel in that same source is 0.8 g/kg, which is only two-thirds of the 1.2 g/kg value used by NAC.

Discussions with utility representatives have indicated that detailed dose measurements of loaded spent fuel storage casks exhibit no indication of peaking or otherwise unexpected dose rates that

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NAC Response to RAI 5-2 (Continued)

would result from ^{59}Co source terms higher than those considered by cask designers. Thus, it is reasonable to conclude that the source terms utilized by cask designers are adequate to determine the dose rates on the external surfaces of a loaded spent fuel storage cask.

For the purpose of comparison, if it is assumed that the 4.7 g/kg cobalt impurity value is realistic for fuel more than 10 years old, the cobalt activation resulting from this older fuel would be a factor of 4 higher than that used by NAC. The half-life of ^{60}Co is about 5 years. Consequently, if the 4.7 g/kg cobalt impurity value is assumed for fuel assemblies that are more than 10 years old, the quantity of cobalt impurity remaining after 15 years of cooling for these assemblies is approximately the same as that in a fuel assembly with an assumed cobalt impurity of 1.2 g/kg and an assumed cool time of 5 years. In other words, the additional cooling time of older fuel will offset the potentially higher cobalt source term.

Therefore, based on available information, the 1.2 g/kg cobalt impurity value utilized by NAC is bounding for the fuel assembly hardware utilized by utilities within the last 10 to 15 years. Current analytical methodologies such as those utilized by NAC have been demonstrated to be reasonable in EPRI TR-104329, and by independent dose measurements taken by utilities.

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CHAPTER 5: SHIELDING

Section 5.2 Source Specification

5-3 For Tables 5.2-10 and 5.2-11, "One-Dimensional Dose Rate Results Relative to PWR and BWR, respectively, Design Basis Fuel": For the normal conditions, radial dose rates are for surface and 2.4 meters. Provide an explanation of why a distance of 2.4 meters was used. The staff needs the information to confirm that the application meets the external radiation standards of Section 71.47.

NAC Response

In accordance with 10 CFR 71.47, dose rate limits are applied to the external surface of the package and at 2 meters from the vertical planes projected from the outer edge of a standard width rail car. The 2.4 meters is the distance from the package surface to the projected edge of the rail car (0.4 meters) + the 2 meter distance from the projected edge of the rail car.

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CHAPTER 5: SHIELDING

Section 5.3 Model Specifications

5-4 Provide figures that clearly distinguish the regions and materials for the following figures, and provide the names of the materials listed in the legend. Section 71.33 requires a description in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation.

Figure 5.3-14	PICTURE Representation of PWR Top Model - Normal Conditions - Showing Trunnion Recesses and Lid Vent
Figure 5.3-15	PICTURE Representation of PWR Bottom Model – Normal Conditions - Slice Through Lower Rotation Pockets
Figure 5.3-17	PICTURE Representation of PWR Top Model - Accident Conditions
Figure 5.3-18	PICTURE Representation of BWR Fuel Region and Heat Transfer Models at Fuel Axial Midplane

NAC Response

Figures 5.3-14, 5.3-15, 5.3-17, and 5.3-18 are the result of the SCALE PICTURE code representation of the transport cask model MARS geometry. The MARS geometry is documented in the input files presented in Section 5.5.3.2.

The figures are intended solely as an aid to the analyst to verify that the proper translation of the model sketches in Figure 5.3-10 through Figure 5.3-13 was made to the MARS geometry. As such, they duplicate the information presented in the model sketches and sample input files.

Due to the large number of materials and computer code restrictions, the resolution of the material regions is limited to the configuration as shown. Material numbers in the figure legends

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NAC Response to RAI 5-4 (Continued)

correspond to the material identifications shown in the input files included in Section 5.5.3.2. A note is added to each figure (Figures 5.3-14, 5.3-15, 5.3-17, and 5.3-18) to indicate the location of the material description of the materials listed in the figure legend.

The sketches shown in Figures 5.3-10 through 5.3-13 provide additional illustrations of the materials and dimensions of the models.

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CHAPTER 5: SHIELDING

Section 5.3.1.1 One-Dimensional Radial Model

5-5 The paragraph references several figures, however, Figure numbers have been omitted from the text. Section 71.7(a) requires complete and accurate information.

NAC Response

The first paragraph of Section 5.3.1.1 is revised to refer to Figures 5.3-3, 5.3-4 and 5.3-5. Figure numbers 5.3-3 and 5.3-4 were inadvertently omitted from the original text.

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CHAPTER 5: SHIELDING

Section 5.4.2.1 Normal Conditions of Transport

5-6 Provide additional explanation of the results depicted in the graph in Figure 5.4-8.

The effect of the heat transfer fins on the dose rate is shown in Figure 5.4-8. Justify the assumption that the localized peak in neutron dose rate is offset by a localized depression in the gamma contribution. Also, explain the apparent lack of symmetry to the projected dose rates in the graph. In addition, revise Figure 5.3.3 to show the heat transfer fins.

Section 71.33 requires a description in sufficient detail to identify the package accurately and provide a sufficient basis for evaluation.

NAC Response

Figure 5.4-8 is intended to show the increase in neutron dose at cooling fin locations due to neutron streaming, and a decrease in gamma dose due primarily to shielding of the gamma radiation by the metal fins. The dose rate plot is revised to include a profile of the total dose rate to show that there is no significant radiation effect due to the cooling fins. The peaks in the total dose profile (at 0°, 90°, 180° and 270°) are due to the geometry of the canister basket, which places the fuel closer to the canister shell at these locations. The irregularity of the profile is due to the large number of small detector surfaces used in the evaluation.

Figure 5.3-3 is a representation of the one-dimensional model used in the calculation, which did not include the cooling fins. Consequently, the fins are not shown in the model sketch. The fins are included in the three-dimensional model (Figure 5.3-18).

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CHAPTER 5: SHIELDING

Section 5.4.3 Loading Table Analysis

5-7 For Table 5.4-22, "Loading Table for BWR Fuel": clarify if low enrichment fuel (1.9%-2.3%) with 45<Burnup<50 GWD/MTU is expected, or clearly specify that it has not been evaluated.

The high burnup, low enriched fuel requires a cooling time greater than 40 years. It does, however, appear to be included in the loading table. This is a bounding fuel and must be evaluated if it is intended to be placed in the Universal Transport Cask. Section 71.7(a) requires complete and accurate information.

NAC Response

Table 5.4-22 is revised to delete the column showing required cool time for fuel having a burnup between 45,000 MWD/MTU and 50,000 MWD/MTU. This prevents loading of BWR fuel having a burnup in that range in the UMS Transport Cask.

Based on the projected discharge data provided in DOE/RW-0184, BWR fuel assemblies with burnup above 45,000 MWD/MTU are expected to have an initial enrichment above 3.0 wt % ^{235}U. Consequently, no BWR fuel assemblies are expected to be excluded by the limitation in burnup.

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CHAPTER 6: CRITICALITY

Section 6.1 Discussion and Results

6-1 Clarify the conflicting statements on the maximum initial enrichment for BWR fuel contents.

The maximum initial enrichment of BWR fuel is stated at various places in the SAR to be either 3.75 or 4.0 wt% ^{235}U. A maximum enrichment of 4.0 wt% ^{235}U appears in Sections 1.2.3, 6.3.2, 6.3.3, 6.4.1.3.2, and 6.4.3.1, whereas 3.75 wt% ^{235}U appears in Sections 6.1 and 6.3.4.1 and in the CSAS computational inputs shown in Figures 6.6.2-3 and 6.6.2-4.

Section 71.33(b) requires an accurate description of the package contents. SRP Section 6.5.2 identifies initial enrichment as an important specification for spent fuel contents.

NAC Response

The maximum initial enrichment of BWR fuel is 4.0 wt % ^{235}U. Sections 6.1 and 6.3.4.1 and the computational inputs/outputs presented in Figures 6.6.2-3 and 6.6.2-4 are revised to show the 4.0 wt % ^{235}U maximum initial enrichment.

See the Response to RAI 6-3.

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CHAPTER 6: CRITICALITY

Section 6.2 Package Fuel Loading

6-2 For each of the five fuel/canister classes, describe how axial poison coverage is ensured under the hypothetical accident conditions of transport.

The staff's preliminary calculations indicate that, following an end-drop accident, the active fuel tips can protrude significantly beyond the ends of the UMS poison panels for certain fuel designs with especially short dimensions of the top hardware. Therefore, in showing that all allowed fuel contents are within the analyzed safety basis, NAC should describe in detail how axial poison coverage is determined. The determination should consider the extent to which a top or bottom end-drop accident could axially dislocate the fuel pins within the assemblies and the active fuel within the fuel pins. Conclusions regarding the potential for end-drop accidents to produce hardware damage and material dislocations in the fuel assemblies should also be reflected in the description of modeling assumptions in Section 6.3.2.

Section 71.55 requires evaluation of the most reactive credible configurations. SRP Section 6.5.1.1 specifies the review of features that locate the fuel relative to neutron absorbing material. SRP Section 6.5.3.1 specifies considering off-nominal relative positionings of fuel and basket components.

NAC Response

Chapter 6 is revised to incorporate the evaluation of the criticality effects of the hypothetical protrusion of active fuel beyond the ends of the neutron poison panels. The analysis shows that there is no significant change in reactivity for PWR type fuel. The change in k_{eff} for BWR fuel is calculated to be 0.0249. The resulting BWR system reactivity, including bias and uncertainty, k_s, is 0.9497. The evaluation was performed using the same assumptions as the original analysis, including maximum fuel enrichment, optimum water moderation in and around the fuel rod and 75% reduction in neutron poison material.

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NAC Response to RAI 6-2 (Continued)

Evaluation of Top End Axial Fuel Shifting

Axial shifting of the contents of the Transportable Storage Canister is considered as a result of a top end impact in transport. A bounding hypothetical fuel-shifting condition is considered in which all of the fuel rods are shifted to the top of each assembly. The fuel within every fuel rod is assumed to shift into the lower half of the top fuel rod plenum (which is not a credible configuration), and each assembly is shifted up until it is in contact with the lid of the canister. The basket is conservatively toleranced for this condition and is assumed to remain in contact with the canister bottom plate, which, under a 60g top end impact, is not a credible configuration. Nonetheless, the assumption was conservatively retained.

Analysis of the PWR fuel configuration is performed using the design basis WE 17x17 OFA assembly. The transport cask top end impact does not deform the PWR top nozzle. However, the top nozzle is conservatively modeled as being deformed 1.0 inch. The reduction in top nozzle height combined with the hypothetical fuel-shifting results in a maximum of 4.520 inches of active fuel protruding beyond the neutron poison panels. The analysis of this condition, presented in Section 6.4.5, shows that the hypothetical shifting of the PWR fuel in the top end impact does not significantly affect the reactivity of the UMS system.

Analysis of the BWR fuel configuration shows that for both classes of BWR fuel, the lifting bail and the corner posts at the top of each assembly may deform in a 60g end impact. The deformation is modeled using a 4.7-inch reduction in top nozzle height. The reduction in top nozzle height combined with the hypothetical fuel-shifting assumed for the BWR fuel case, results in a maximum of 7.625 inches of active fuel protruding above the top of the neutron poison panels. Analysis of this configuration is performed using the design basis EX\ANF 9x9 fuel. The analysis of this condition, presented in Section 6.4.5, shows that the system reactivity, k_{eff}, increases by 0.0249. This increase in reactivity is included in the revised

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NAC Response to RAI 6-2 (Continued)

analysis results reported in Table 6.1-1. The resulting BWR system maximum reactivity, including bias and uncertainty, k_s, of 0.9497 is less than the subcritical limit of 0.95.

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. Each fuel assembly and the basket are assumed to remain in contact with the canister bottom plate. Conservative basket tolerances are used for this condition.

The maximum distance from the canister bottom plate to the lower end of the neutron poison panels occurs when the conservatively toleranced basket components are shifted up towards the canister lid, which is not a credible configuration. For the PWR configuration, this distance is limited to 5.30 inches. For the BWR configuration, the distance is limited to 8.2 inches. However, both the PWR and BWR fuel types have rod end caps, tie plates and/or components of the bottom nozzle that deform to less than 0.78 inches in the bottom impact condition. Consequently, the top end impact condition, which exposes 4.52 inches of PWR fuel and 7.625 inches of BWR fuel, bounds the bottom end impact condition.

Based on these analyses, the basket neutron poison material coverage is adequate for all of the PWR and BWR fuel in either the top end or bottom end impact hypothetical accident conditions. The hypothetical accident conditions result in no significant change in the reactivity of the PWR configuration and a small increase in reactivity of the BWR configuration. The reactivity of the BWR configuration remains less than the subcritical limit of 0.95.

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CHAPTER 6: CRITICALITY

Section 6.3.4.1 Fuel Region

6-3 Correct the inconsistency in the tabulation of fuel material densities.

The second material in the left column is UO_2 with 3.75 wt% ^{235}U, yet the atom densities listed in the right column correspond to UO_2 with 4.0 wt% ^{235}U. The latter material is inconsistent with the CSAS inputs and outputs shown in Figures 6.6.2-3 and 6.6.2-4.

Section 71.33(b) requires an accurate description of the package contents. SRP Section 6.5.3.2 specifies verification of appropriate mass densities and atom densities for all modeled materials of the packaging and contents.

NAC Response

Section 6.3.4.1, and Figures 6.6.2-3 and 6.6.2-4, are revised to correct the material description to 4.0 wt % ^{235}U, which corresponds to the stated density.

See the Response to RAI 6-1.

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CHAPTER 6: CRITICALITY

Section 6.3.4.2 Cask Material

6-4 Either correct the "BORAL core" compositions used in the CSAS models or provide a calculation showing that the as-modeled compositions do not lead to significant errors in the calculated values of k_{eff}.

The staff has confirmed that the "BORAL core" material densities used in NAC's CSAS models do lead to the appropriate areal densities of ^{10}B. However, the model's densities of ^{11}B, C, and Al are significantly off. For example, ^{10}B comprises 15.4% instead of 20% of the boron atoms, and the stoichiometric ratio of boron to carbon atoms for B_4C is 38 (tabulated) or 3.8 (CSAS output) instead of 4.

Section 71.33(b) requires an accurate description of the package materials used for neutron absorption. SRP Section 6.5.3.2 specifies verification of appropriate mass densities and atom densities for all modeled materials of the packaging and contents.

NAC Response

The ^{10}B density in the neutron poison sheets is reduced to 75% of its minimum specified loading in accordance with Section 6.5.3.2 of the Standard Review Plan for Transportation Packages for Spent Nuclear Fuel (NUREG-1617). This section states in part that "... no more than 75% of the specified minimum neutron poison concentration should generally be considered in the criticality evaluation."

Reducing the ^{10}B component by 75% reduces the ^{10}B ratio in boron from 20% to 15% and reduces the stoichiometric ratio of boron to carbon atoms from 4.0 to 3.8. The ^{11}B, C and Al components of the sheet are not considered to be affected by the 75% requirement as they do not represent neutron poison components of the sheet.

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CHAPTER 6: CRITICALITY

Section 6.4.3.1 Summary of Maximum Criticality Values

6-5 Correct "75% of nominal..." to "75% of minimum ^{10}B loading."

The staff notes that other Sections of the SAR refer to the minimum ^{10}B loading.

SRP Section 6.5.3.2 states that no more than 75% of the specified minimum neutron poison concentration of the packaging should be considered in the criticality evaluation. Any deviations from the SRP guidance should be justified.

NAC Response

Section 6.4.3.1 is revised to refer to "75% of the minimum ^{10}B loading in the BORAL."

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CHAPTER 6: CRITICALITY

Table 6.5-3 Most Reactive Configuration System Parameters

6-6 Correct the enrichment entry in this table from 4.0 to 4.2 wt% ^{235}U.

Section 71.33(b) requires accurate description of the package contents. SRP Section 6.5.2 identifies initial enrichment as an important specification for spent fuel contents.

NAC Response

Table 6.5-3 is revised to correct the value of the enrichment used in the most reactive system from 4.0 to 4.2 wt % ^{235}U.

See the Response to RAI 6-7.

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CHAPTER 6: CRITICALITY

Appendix 6.6.1.1 Criticality Evaluation of Maine Yankee Site-Specific Spent Fuel

6-7 Describe the applicability of the critical benchmark experiments to the site-specific Maine Yankee contents.

The current discussion in Section 6.5.1.2 covers the applicability of experiments to the standard fuel contents only. A similar discussion for Maine Yankee contents does not appear in this appendix for site-specific spent fuel. The requested description should compare the parameters *H/U volume ratio* and the *average group causing fission* for the most-reactive site-specific contents to those for the experiments.

Section 71.41 requires that compliance be demonstrated by methods acceptable to the Commission. SRP Section 6.5.7.1 states that the applicant should justify that the benchmark experiments are applicable to the package design and contents.

NAC Response

Table 6.5-3 is revised to show a comparison of the most reactive system configuration parameters to the range of applicability of the critical benchmark parameters as shown:

Parameter	Benchmark Minimum Value	Benchmark Maximum Value	UMS Design Basis PWR Fuel Most Reactive Configuration	Maine Yankee Fuel Most Reactive Configuration
Enrichment (wt % ^{235}U)	2.35	4.74	4.2	4.2
Rod pitch (cm)	1.26	2.54	1.26	1.50
H/U volume ratio	1.6	11.5	1.9	2.6
^{10}B areal density (g/cm^2)	0.00	0.45	0.025	0.025
Average energy group causing fission	21.7	24.2	22.3	22.5
Flux gap thickness (cm)	0.64	5.16	2.2 to 3.8	2.22 to 3.8

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NAC Response to RAI 6-7 (Continued)

As shown in the table, the evaluated parameters have values that are approximately in the mid-range of the benchmark parameters. Therefore, the critical benchmark evaluation provided in Section 6.5.4 is applicable to the Maine Yankee fuel.

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CHAPTER 7: OPERATING PROCEDURES

Chapter 7.0 Operating Procedures

7-1 In Table 7-1, correct the torque values and corresponding tolerances in accordance with the License Drawings for the following components:

Component	Torque Value (Ch. 7)	Torque Value on License Drawing
Coverplates	300±20 ft-lbs	300±50 in-lb (Dwg. No.790-500)
		300±20 in-lb (Dwg No. 790-503)
Secondary Trunnion Bolts	500±10 ft-lb	500±50 ft-lb (Dwg. No. 790-500)

Section 71.7(a) requires complete and accurate information. Update the SAR Drawings or Chapter 7, as appropriate, to reflect the correct torque values.

NAC Response

The torque value and tolerance for the coverplate bolts is revised on Drawing 790-503 and in Table 7-1 to 300 ± 50 in-lbs as shown on Drawing 790-500.

The torque value and tolerance for the secondary trunnion bolts are correct as shown on Drawing 790-500. Table 7-1 is revised to the value shown on Drawing 790-500.

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CHAPTER 7: OPERATING PROCEDURES

Chapter 7.0 Operating Procedures

7-2 In Table 7-1, clarify which component the "test plug" is associated with, and add a more descriptive entry for this component to the table.

Section 71.7(a) requires complete and accurate information. There are three test ports on the Universal Transport Cask used for helium leak testing. One is used to test the cask lid o-ring, while the other two are used to test the o-rings of the two port coverplates. The description of the test plug in Table 7-1 does not distinguish which test plug is torqued to 30±3 ft-lb.

NAC Response

The "test plug" refers to Item 13 on Drawing 790-503 and to Item 3 on Drawing 790-504. Item 13 provides a closure to the test port for the lid o-ring annulus. Item 3 provides a closure for the test port for the o-ring annulus in the vent and drain port coverplates.

As an aid to clarifying the purpose of the component, Table 7-1 is revised to include two items. The first is "Lid o-ring test port plug" with a specified torque of 30 ± 5 in-lbs. The second is "Vent/Drain o-ring test port plug" with a specified torque of 125 ± 5 in-lbs. Drawing 790-503 is revised to specify a torque value of 125 ± 5 in-lbs for Item 13. Drawing 790-504 is revised to specify a torque value of 30 ± 5 in-lbs for Item 3.

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CHAPTER 7: OPERATING PROCEDURES

Chapter 7.0 Operating Procedures

7-3 Describe procedures for conducting the helium leak tests of the lid and port coverplates. The procedure should also include a discussion of the corrective actions if the components or configurations fail to meet the test criteria.

Section 71.7(a) requires complete and accurate information, and Section 71.87 requires that the gaskets be property installed and secured and free of defects. The application does not contain enough of a description to ensure the adequate conduct of the helium leak test to assure that the o-rings are property installed.

NAC Response

The operating procedures do not require helium leak testing of the lid and port coverplate o-rings prior to transport operations, unless a containment boundary component (o-ring set or port coverplate) is replaced. Helium leak testing of the o-rings is also required post-fabrication and during annual maintenance. The performance of the helium leak test is described in Sections 8.2.2 and 8.1.3.

During routine preparation for shipment, a form and fit pressure drop test is performed to demonstrate proper installation. In these tests, the annulus between the o-rings in the closure lid and in the vent and drain port coverplates is pressurized to 15 psig and held for 10 minutes to demonstrate a leak rate of not more than 1 x 10^{-3} cm^3/sec. No loss of pressure is permitted during the test period. The procedure requires that the o-rings in the closure lid and in the port coverplates be inspected for damage and replaced if necessary. The procedure also requires that the lid and the port coverplates be stored in such a way that the o-rings and o-ring grooves are protected.

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NAC Response to RAI 7-3 (Continued)

The replacement of the o-rings and cleaning of the o-ring grooves and sealing surfaces are the only available actions to correct a failed leak test addressed in the handling procedure. If these actions do not result in a successful leak test, then additional maintenance action, such as polishing the sealing surfaces, or replacement of a port coverplate, may be required.

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CHAPTER 7: OPERATING PROCEDURES

Chapter 7.0 Operating Procedures

7-4 Include procedures for the preparation of the Universal Transport Cask for wet loading, and describe the basis for wet loading the TSC or the GTCC waste canister into the cask.

Section 71.87(f) requires that a verification must be made to ensure that the package has been loaded and closed appropriately. Section 7.5.1 discusses loading fuel into the TSC while it is installed in the Universal Transport Cask. This is referred to as wet loading. However, there are no procedures for preparing the Universal Transport Cask for the wet loading operation, and there is no discussion of the purpose for wet loading of the fuel in the SAR.

<u>NAC Response</u>

Section 7.5 is revised throughout to delete reference to wet loading of the UMS® Universal Transport Cask. There is no current operational requirement for direct loading of the transport cask.

Section 7.5 presents the generic procedures that are applicable to loading and unloading the Transportable Storage Canister using the transfer cask. These procedures are taken from the Safety Analysis Report for the NAC-UMS® Universal Storage System, Docket 72-1015, and are provided for reference.

As described in Section 7.5.2, the generic procedure provided in Section 7.5.1 is applicable to the loading of GTCC waste.

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CHAPTER 7: OPERATING PROCEDURES

Chapter 7.0 Operating Procedures

7-5 Revise the operating procedures to include appropriate controls for detecting and preventing the ignition of any combustible gases during welding, grinding, or cutting operations associated with closure of the TSC or the GTCC waste canister.

Section 71.43(d) requires that a package be made of materials that assure there will be no significant chemical or galvanic reactions. Potential reactions between the TSC or Universal Transport Cask paint coatings and/or other components (such as the aluminum heat transfer disks) with spent fuel pool water may produce hydrogen or other flammable gases. Since the shield lids of the TSC and GTCC waste canister are welded to their shells during fuel loading, there is a source of heat that could lead to ignition if sufficient amounts of gas are present.

NAC Response

Section 7.5 is revised throughout to delete reference to wet loading of the UMS® Universal Transport Cask. There is no current operational requirement for direct loading of the transport cask.

Section 7.5 presents the generic procedures that are applicable to loading and unloading the Transportable Storage Canister using the transfer cask. These procedures are taken from the Safety Analysis Report for the UMS® Universal Storage System, Docket 72-1015, and are provided for reference. The reference generic loading and unloading procedures (Sections 7.5.1, 7.5.2 and 7.5.3) describe the method used to detect and control combustible gases during canister opening and closing operations.

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CHAPTER 7: OPERATING PROCEDURES

Section 7.3 Receiving Universal Transport Cask and Unloading Transportable Storage Canister From Universal Transport Cask

7-6 Revise the SAR to include operating procedures for unloading the contents of the TSC and GTCC waste canisters. Section 71.7(a) requires complete and accurate information.

NAC Response

Section 7.5 is revised throughout to delete reference to wet loading of the UMS® Universal Transport Cask. There is no current operational requirement for direct loading of the transport cask.

Section 7.5 presents the generic procedures that are applicable to loading and unloading the Transportable Storage Canister using the transfer cask. These procedures are taken from the Safety Analysis Report for the UMS® Universal Storage System, Docket 72-1015, and are provided for reference. The reference generic unloading procedure (Section 7.5.3) describes the unloading of the Transportable Storage Canister.

The procedure provided in Section 7.5.3 is suitable for the unloading of GTCC waste. However, the procedure contains a number of steps directed at the controlled cool down of the canister, basket and spent fuel. These steps would not be applicable to the unloading of GTCC waste, and would not be included in a site specific procedure.

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CHAPTER 7: OPERATING PROCEDURES

Section 7.5.1 Loading and Closing the Transportable Storage Canister Containing Spent Fuel

7-7 Specify the minimum helium gas purity that will be used to vacuum dry and backfill the cask cavity, TSC, and the GTCC waste canister. Perform a calculation to show that this minimum helium gas purity assures that the spent fuel will not undergo significant degradation during transport.

Section 71.55(d)(2) requires that the geometric form of the package contents of a fissile materials package will not be substantially altered under the test conditions specified for normal conditions of transport. The SAR does not specify the minimum helium gas purity that is used to vacuum dry and backfill the cask cavity, TSC, and GTCC waste canister. PNL-6365, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," recommends that the amount of oxidizing gases in a storage cask be limited to 1 gram-mole. This 1 gram-mole limit reduces the amount of oxidants below levels where any cladding degradation is expected.

NAC Response

The minimum helium purity level of 99.9% specified in Section 7.5.1 maintains the maximum quantity of oxidizing contaminants to less than one mole per cask or canister. The calculated impurities in the canister volumes due to the helium purity are:

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NAC Response to RAI 7-7 (Continued)

Canister Class	**Class 1**	**Class 2**	**Class 3**	**Class 4**	**Class 5**
Canister Free Volume (Liters)	5868	6312	6933	6650	6829
Gas in Canister Free Volume (Moles)	244.1	262.6	291	276.6	284.1
Canister Potential Impurity (Moles)	**0.24**	**0.26**	**0.29**	**0.28**	**0.28**
Cavity Spacer Volume (Liters)	1140	613	173	532	254
Cask Total Free Volume (Liters)	7008	6925	7106	7182	7083
Gas in Cask Free Volume (Moles)	292	288.1	295.6	298.7	294.7
Cask Potential Impurity (Moles)	**0.29**	**0.29**	**0.30**	**0.30**	**0.29**

The evaluation of the Transport Cask cavity is based on the assumed loss of confinement of the Transportable Storage Canister even though there are no design basis accident events that could result in loss of canister confinement.

The number of moles, conservatively calculated at room temperature (293K), is (1 x Liters)/(0.0821 x 293) = 0.0416L, where L is the canister or cask free volume in liters. The displacement of the spacers is conservatively ignored.

As shown in the table, the potential quantity of residual impurities is less than 1 mole for all of the potential transport configurations.

The number of moles of residual gas due to the vacuum excursion to 3 torr is (291 x 3)/760 = 1.1 moles for the worse case canister configuration, and is (298.7 x 3)/760 = 1.2 moles for the worse case cask configuration. As noted in PNL-6365, only a fraction of this residual gas is made up of oxidizing gases. In addition, there are two evacuation cycles to the 3 torr vacuum pressure level with an intermediate helium backfill. This procedure further reduces the quantity of oxidizing gases that could remain in the cask or canister. Consequently, there is high confidence that there is less than 1 mole of oxidizing gases remaining in the cask or canister after closure.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CHAPTER 7: OPERATING PROCEDURES

Section 7.5.1 Loading and Closing the Transportable Storage Canister Containing Spent Fuel

7-8 Revise the operating procedure to require a TSC vacuum hold time of 30 minutes, or provide a justification for using 20 minutes.

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 test conditions specified for normal conditions of transport. The procedure in the SAR specifies a hold time of 20 minutes. However, PNL-6365, "Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel," recommends that a constant pressure of 3 Torr should be maintained for 30 minutes without the aid of a vacuum pump to assure that oxidizing gases are sufficiently removed from the cask to prevent gross degradation of the cladding due to excessive fuel oxidation.

<u>NAC Response</u>

As noted in the response to RAIs 7-4, 7-5 and 7-6 the operating procedures are revised to delete wet loading of the Transport Cask. Consequently, the Transportable Storage Canister is already closed when presented for loading in the Transport Cask.

The operating procedure in Section 7.5.1 specifies a vacuum hold time of 30 minutes for the Transportable Storage Canister, which is consistent with the procedure provided in the Safety Analysis Report for the NAC-UMS® Storage System, Docket Number 72-1015.

Note that NAC's cask operating experience and technical evaluation confirm that holding a vacuum of 10 mm Hg for 20 minutes with a pressure rise less than 5 mm Hg does ensure that all free water is removed from the canister.

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CHAPTER 7: OPERATING PROCEDURES

Section 7.5.1 Loading and Closing the Transportable Storage Canister Containing Spent Fuel

7-9 Revise the allowable leakage rate and sensitivity to reflect the correct values as specified in Chapter 4. In the procedure, also include the allowable leakage rates for canisters containing both PWR and BWR fuels.

Section 71.7(a) requires complete and accurate information. Incorrect values for the leakage rate and sensitivity are specified in the operating procedure.

NAC Response

The procedure presented in Section 7.5.1 is taken from Section 8.1.1 of the UMS® Universal Storage System Safety Analysis Report, Docket No. 72-1015. As shown in Section 7.5.1, the Transportable Storage Canister is tested to a leak tight condition during closing. The leak tight test condition is applied to a canister holding PWR or BWR spent fuel or GTCC waste.

The allowable leak rate for the UMS® Universal Transport Cask for containment periodic verification testing is based on the highly conservative assumption that the spent fuel is not confined by the canister. The Transport Cask test leak rates are provided in Table 7-2.

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CHAPTER 8: ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Section 8.1.3 Leak Tests

8-1 Revise the SAR to include the following details:

1. a reference to your commitment to use ANSI N14.5-1997 as the accepted standard for leak testing;
2. the allowable leakage rates for the Universal Transport Cask to be consistent with the containment analysis of Chapter 4 for both PWR and BWR fuels;
3. descriptions of the methods used to assure that the test plugs of the Universal Transport Cask lid and the two port coverplates are adequately sealed; and
4. description of the helium leak test acceptance or rejection criteria.

Section 71.7(a) requires complete and accurate information. There is not enough information in Chapter 8 to conclude that the helium leak testing will be performed in accordance with the specifications of ANSI Nl4.5-1997.

NAC Response

Section 8.1.3 is revised to incorporate a description of the acceptance leak testing of the containment weldment and the containment boundary port coverplate and lid o-rings. As described in the revised Section 8.1.3, leak tests are performed in accordance with the methodologies and requirements of ANSI N14.5-1997. As shown in Sections 4.2.3 and 4.3.3, the calculated minimum allowable leak rate is 3.3×10^{-5} std cm^3/sec (helium) for the BWR fuel in normal conditions of transport. This allowable leak rate is applied to the containment boundary o-rings, and is incorporated in the cask handling procedure, using a leak test sensitivity of 1.6×10^{-5} std cm^3/sec (helium). A leak tight test criterion is applied to the containment boundary weldment. As shown in revised Section 8.1.3, the containment weldment is tested to 2×10^{-7} std cm^3/sec (helium) using a detector sensitivity of 1×10^{-7} std cm^3/sec, or better. The test acceptance criterion for each test condition is provided in Section 8.1.3.

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NAC Response to RAI 8-1 (Continued)

The cask lid and coverplate test plugs are outside of the containment boundary of the cask and are not tested to ensure that they provide a level of containment. However, the test plug is a solid, threaded stainless steel piece that is torqued in place. An o-ring seals the test plug to the lid or coverplate. Consequently, the test plug arrangement provides a substantial mechanical seal.

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CHAPTER 8: ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Section 8.1.3 Leak Tests

8-2 Clarify the reference made to the fabrication leak tests.

Section 71.7(a) requires complete and accurate information. A reference is made to the fabrication leak tests of Section 8.1.3. However, Section 8.1.3 does not describe the leak tests acceptance criteria applied to the containment system fabrication verification, periodic, maintenance or the pre-shipment leak tests.

NAC Response

Section 8.1.3 is revised to describe the leak tests that are performed on the containment boundary weldment and on the assembled cask to qualify the cask containment boundary o-rings. As shown in the revised Section 8.1.3, the containment weldment is tested to demonstrate a leak tight condition and the o-rings are tested to demonstrate a leak rate of 3.3×10^{-7} std cm^3/sec (helium), or less. As described in Section 8.1.3, leak testing is performed in accordance with the requirements of ANSI N14.5-1997.

Section 8.3.2 is revised to incorporate the leak test requirement in the description of the major chronological steps in the fabrication of the transport cask body.

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CHAPTER 8: ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Section 8.1.7 Neutron Absorber Verification Tests

8-3 Describe the reference BORAL standard plates and the testing that is required for qualifying them as reference standards.

The description of the BORAL standard plates used for calibration should address the size distributions of B_4C particles, the absolute magnitude and uniformity of the ^{10}B areal densities, and how these characteristics are determined. To bound any effects of neutron channeling on the calibration, the ^{10}B isotopic composition (natural or enriched) should be no higher and the B_4C particles should be no larger in the standards than in the panels. If neutron transmission or radiography measurements are used for qualifying the reference BORAL standard plates, the calibration standards used for those measurements should likewise be described.

Section 71.123 and 71.125 require the establishment of a program and measures to identify, perform, and control the testing required to demonstrate that the packaging components will perform satisfactorily in service. SAR Section 6.5.3.2 calls for ensuring that the neutron absorbers are property controlled during fabrication to meet their specified properties.

NAC Response

Section 8.1.7 is revised to incorporate wet chemical analysis as the method used to verify ^{10}B content in the BORAL plate.

This method is generally accepted throughout the industry and is the method chosen by EPRI for determination of Boron content in neutron absorber materials. It provides the most accurate and practical direct measurement method for ^{10}B, Boron and B_4C content of metal materials and is considered an industry standard for this application.

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NAC Response to RAI 8-3 (Continued)

In wet chemistry analysis, a strong acid is used to dissolve all non-B_4C material in the BORAL, including the aluminum powder in the matrix and the solid aluminum cladding. A mass-balance analysis is then applied to the remaining non-soluble B_4C to verify the amount of B_4C in the tested sample. The B_4C quantity is compared to the chemical specification used to make the tested sample.

The wet chemistry determination of the minimum ^{10}B content of the BORAL (specified on the applicable fuel tube drawings) will be performed in accordance with written test procedures approved by the supplier, AAR Manufacturing, Inc., and by NAC International.

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CHAPTER 8: ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Section 8.1.7 Neutron Absorber Verification Tests

8-4 Characterize (a) the uncertainty in the neutron luminance measurement and (b) the sensitivity of the measured luminance to ^{10}B content over the two ranges of areal density, 0.011 and 0.025 g/cm^2, for the BWR and PWR BORAL panels, respectively.

Section 71.123 and 71.125 require the establishment of a program and measures to identify, perform, and control the testing required to demonstrate that the packaging components will perform satisfactorily in service. SAR Section 6.5.3.2 calls for ensuring that the neutron absorbers are properly controlled during fabrication to meet their specified properties.

NAC Response

Section 8.1.7 is revised to delete radiographic testing of the BORAL and replace it with the wet chemistry test method. Wet chemistry testing is the method used in the commercial manufacture and testing of the BORAL by the vendor, AAR Manufacturing, Inc. See the response to RAI 8-3.

BORAL can be manufactured in any thickness from 0.070 to 0.375 inches. The maximum ^{10}B areal density for a 0.075-inch thickness is approximately 0.025 gm/cm^2. For the 0.135-inch thick sheet used in the BWR fuel tube, the maximum ^{10}B loading is approximately 0.045 gm/cm^2. Consequently, the ^{10}B loading specified for the UMS basket designs are readily achievable.

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CHAPTER 8: ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Section 8.2.1 Structural and Pressure Tests

8-5 Address the annual testing provisions, per your commitments to meet ANSI N14.6.

The SAR discusses only visual inspection of trunnions prior to each shipment. Sections 6.3 and 6.5 of ANSI N14.6, however, provides that a liquid penetrant or magnetic particle examination shall be performed for the trunnions, if a load testing is omitted.

NAC Response

Section 8.2.1 is revised to require an annual inspection and liquid penetrant examination of trunnion welds and load bearing surfaces using the same criteria as is used in the acceptance inspection. Based on accessibility of the trunnions and trunnion welds and implementation of the non-destructive testing option defined in ANSI N14.6, annual load testing of the trunnions is not performed.

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CHAPTER 8: ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Section 8.3.2 Cask Body Fabrication

8-6 Include a more clear description of the leak test criteria (i.e., air standard leak rate) and the test frequency for the containment system fabrication verification, periodic, maintenance, and pre-shipment leak tests.

Section 71.7(a) requires complete and accurate information. Section 8.1.3 does not clearly describe the criteria applied to each of the containment system fabrication verification, periodic, maintenance, and pre-shipment leak tests.

NAC Response

The description of the leak tests performed for fabrication acceptance are provided in Section 8.1.3. The leak tests include leak testing of the containment weldment to demonstrate leak tightness and leak testing of the assembled transport cask containment boundary (o-rings in the lid and vent and drain ports) to demonstrate a leak rate of 3.3 x 10^{-5} std cm^3/sec (helium) or less.

The reference to Section 8.3.2 for fabrication leak test requirements is deleted. Section 8.3.2 is revised to incorporate the containment weldment and containment boundary leak testing requirement in the list of major chronological steps in the fabrication of the transport cask body.

The periodic (annual) maintenance leak test requirement is presented in Section 8.2.2. It is identical to the containment boundary o-ring leak test described in Section 8.1.3 and is performed annually when the cask is in continuous use. As noted in Section 8.2.2, leak testing need not be performed if the cask is not being used, but that the cask must be leak tested prior to the next use of the cask. This leak test is also performed after the third use of the new cask, and when an o-ring set, or port coverplate, is replaced during operations.

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NAC Response to RAI 8-6 (Continued)

The pre-shipment leak test is a pressure drop test to demonstrate form and fit as described in ANSI N14.5 Paragraph 7.6.4. The pressure drop test is performed for each loaded (Section 7.1.3) and unloaded (empty) cask transport (Section 7.3.2). The pressure drop test is used provided that the o-rings are not replaced. As noted above, if an o-ring set or port coverplate is replaced, then a leak test in accordance with Section 8.1.3 is performed to demonstrate containment for that o-ring set.

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CHAPTER 8: ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Section 8.3.3.2 Lead Pour Operations

8-7 Specify the correct grade of lead that will be poured into the cask annulus.

Section 71.7(a) requires complete and accurate information. This Section does not indicate which grade of lead will be poured into the cask annulus. Table 2.3.7-1 and Drawing No. 790-502 indicate that "Chemical-Copper Grade" lead will be used.

NAC Response

Sections 8.0, 8.3.1, 8.3.2 and 8.3.3.2 are revised to refer to chemical copper lead.

The lead, originally specified as Chemical Lead provided in accordance with ASTM B29-79, can no longer be procured to that specification, but is now available as Chemical Copper lead provided in accordance with ASTM B29-92. This specification is called out on NAC-UMS® Drawing 790-502.

Comparison of the two ASTM lead specifications shows that the chemical and physical properties of the lead are identical. The properties of the Chemical Copper lead are as specified in Table 1 of ASTM B29-92, "Standard Specification for Refined Lead."

Reference number 10 (Section 8.3.1) is revised to refer to the 1992 edition of ASTM B29 for refined lead.

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DUAL-PURPOSE CANISTER

Dual-Purpose Canister

DP4-1 Specify, and provide the technical basis and calculations for, the allowable leakage rate and sensitivity of the TSC and the GTCC waste canister. Revise Sections 7 and 8 as appropriate to include the allowable leakage rate and sensitivity for the TSC and GTCC waste canister.

Dual-purpose canisters must meet both transportation and storage requirements for use under the respective regulations. Sections 72.24(d), 72.104(a), and 72.106(b) require that the storage system will reasonably maintain confinement of radioactive materials under Section Part 72 normal, off-normal, and credible accident conditions. Since the TSC and the GTCC waste canister may be used for storage following transport, an analysis of the TSC and GTCC waste canister leakage rates and sensitivities is needed to assure that the canisters perform their intended function during storage.

NAC Response

As noted in the NAC Response to RAI 2-12 and others, reference to the in-pool loading of the canister using the UMS® Universal Transport Cask is deleted. Consequently, at the time of transport, the canister will have previously been loaded, closed and tested in accordance with the detailed requirements presented in the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015. Among the requirements for canister closure is that the canister be leak tested to demonstrate leak tight confinement. To demonstrate a leak tight condition, the canister is subject to a helium leak test of 2×10^{-7} std cm^3/sec, using a detector sensitivity of 1×10^{-7} cm^3/sec. The leak test conforms to the definition of leak tight provided in Section 2.1 of ANSI N14.5-1997 and to the evacuated envelope test configuration described in A.5.4 of Table A.1 of the ANSI Standard. The leak tight test requirement applies regardless of canister contents.

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NAC Response to RAI DP4-1 (Continued)

While the GTCC waste packaging is not controlled by the requirements of 10 CFR 72, the GTCC waste canister is closed using essentially the same procedure as is used for closing canisters containing spent fuel. The procedure differs only in that certain Limiting Conditions of Operation (LCOs) specified for spent fuel loading, which are intended to limit the fuel cladding temperatures, are not applicable to the loading of GTCC waste. Otherwise, a written procedure is used for handling, loading and closing the GTCC waste canister that conforms to the spent fuel procedure for similar tasks. Among other requirements, the GTCC waste procedure requires air over water pressure testing, vacuum drying, helium backfill, leak tight leak testing, and closure weld inspection.

The containment analysis for the transport cask does not take credit for the presence of the canister, even though there are no normal or accident conditions that result in the breech of the canister while it is in the transport cask. Since credit for canister containment is not taken, the analysis assumes that the gases, volatiles, fines and particulates are released to the cask interior.

As shown in Section 4.2 of the UMS® Universal Transport Cask Safety Analysis Report, the transport cask containment boundary is helium leak tested to 3.3 x 10^{-5} std cm^3/sec, based on the postulated release from GE 9 x 9 BWR fuel in normal conditions of transport. The release from this BWR fuel bounds the release from all other canister contents.

The detailed methodology and analysis for transport cask containment conforms to ANSI N14.5-1997 and is presented in Chapter 4 of the Transport Cask Safety Analysis Report. As noted in Chapter 4, the GTCC waste does not contain a source of releasable gases and its principal radioactive source term consists of activated metal that is not releasable.

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DUAL-PURPOSE CANISTER

Dual-Purpose Canister

DP4-2 Include a discussion of the welding requirements of the TSC and the GTCC waste canister.

Dual-purpose canisters must meet both transportation and storage requirements for use under the respective regulations. Section 72.122(l) requires that the storage system be designed to allow ready retrieval of spent fuel for further processing or disposal. Since the TSC and the GTCC waste canister are integral components of the Universal Transport Cask, a description of the welding requirements for the canisters is needed to assure that the canister can be removed from the Universal Transport Cask.

<u>NAC Response</u>

As noted in the NAC Response to RAI 2-12 and others, reference to the in-pool loading of the canister using the UMS® Universal Transport Cask is deleted. Consequently, at the time of transport, the canister will have previously been loaded, closed and tested in accordance with the detailed requirements presented in the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015. While GTCC waste packaging is not controlled by the requirements of 10 CFR 72, the GTCC waste canister is closed using essentially the same procedure as is used for closing canisters containing spent fuel as described in the response to RAI DP4-1. No welds are made while the canister is inside of the Transport Cask.

All of the welds made on the spent fuel and GTCC waste canisters, including shop and field installed welds, are fully specified in the License Drawings for these components provided in Section 1.3.4. The specification of the welds includes the appropriate reference to inspection and acceptance of those welds. See the NAC Response to RAI DP8-2.

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DUAL-PURPOSE CANISTER

Dual-Purpose Canister

DP7-1 For the TSC and the GTCC waste canister, demonstrate that the 3/8-inch maximum welding pass thickness and associated nondestructive examination methods will provide reasonable assurance that the confinement system maintains structural integrity for the situation where the canister is placed into a storage overpack for long-term storage. The discussion should address the points of Interim Staff Guidance No. 4. Update the appropriate Sections (e.g., Sections 4.1.3.2, etc.) to address the welding procedures and acceptance criteria.

Dual-purpose canisters must meet both transportation and storage requirements for use under the respective regulations. Section 72.122(l) requires that the storage system must be designed to allow ready retrieval of spent fuel for further processing or disposal. Since the TSC and GTCC waste containers are integral components of the Universal Transport Cask, a description of the welding requirements for the canister is needed to assure that the canister can be removed from the Universal Transport Cask.

NAC Response

As noted in the NAC Response to RAI 2-12 and others, reference to the in-pool loading of the canister using the UMS® Universal Transport Cask is deleted. Consequently, at the time of transport, the canister will have previously been loaded, closed and tested in accordance with the detailed requirements presented in the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015. While GTCC waste packaging is not controlled by the requirements of 10 CFR 72, the GTCC waste canister is closed using essentially the same procedure as is used for closing canisters containing spent fuel as described in the response to RAIs DP4-1 and DP4-2. No welds are made while the canister is inside of the Transport Cask.

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NAC Response to RAI DP7-1 (Continued)

The evaluation of the 3/8 inch maximum weld pass thickness is presented in Section 3.4.4.1.11 ("Canister Closure Weld Evaluation") of the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015.

The closure weld for the canister is a 0.9-inch groove weld between the structural lid and the canister shell. The evaluation of this weld incorporates a 0.8 stress reduction factor in accordance with NRC Interim Staff Guidance (ISG) No. 4, Revision 1. The use of this factor is in accordance with ISG No. 4, since the strength of the weld material (E308) is greater than that of the base material (Type 304 or 304L stainless steel).

The stresses for the canister closure weld are evaluated using sectional stresses as permitted by Subsection NB of the ASME Code. The factored allowables, incorporating the 0.8 stress reduction factor, and the resulting controlling Margins of Safety are:

Stress Category	Analysis Stress Intensity (ksi)	0.8 x Allowable Stress (ksi)	Margin of Safety
P_m	1.78	13.36	6.51
$P_m + P_b$	2.46	20.04	7.15
P + Q	4.13	40.08	8.70

This evaluation confirms that the canister closure weld is acceptable.

Critical Flaw Size for the Canister Closure Weld

The closure weld for the canister comprises multiple weld beads using a compatible weld material for Type 304L stainless steel. An allowable (critical) flaw evaluation has been performed to determine the critical flaw size in the weld region. The result of the flaw evaluation is used to define the minimum flaw size, which must be identifiable in the nondestructive

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<u>NAC Response to RAI DP7-1 (Continued)</u>

examination of the weld. Due to the inherent toughness associated with Type 304L stainless steel, a limit load analysis is used in conjunction with a J-integral/tearing modulus approach.

The safety factor used in this evaluation is that defined in Section XI of the ASME Code.

The stress component used in the evaluation for the critical flaw size is the radial stress component in the weld region of the structural lid. The maximum reported radial tensile stress is 0.9 ksi.

To perform the flaw evaluation, a 10 ksi stress is conservatively assumed, resulting in a significantly larger actual safety factor than the required safety factor of 3. Using a 10 ksi stress as the basis for the evaluation of the structural lid weld, the critical flaw size is 0.52 inch for a flaw that extends 360 degrees around the circumference of the structural lid weld. The maximum resultant tensile stress reported in the weld is 1.8 ksi, which is also enveloped by the value of 10 ksi used in the critical flaw evaluation for stresses in the radial direction.

The 360-degree flaw employed for the circumferential direction is bounding with respect to any partial flaw in the weld, which could occur in the radial and horizontal directions. Therefore, using a minimum detectable flaw size of 0.375 inch is acceptable, since it is less than the very conservatively determined 0.52-inch critical flaw size.

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DUAL-PURPOSE CANISTER

Dual-Purpose Canister

DP8-1 Include the allowable leakage rate and sensitivity for the TSC and the GTCC waste canister.

Dual-purpose canisters must meet both transportation and storage requirements for use under the respective regulations. Sections 72.24(d), 72.104(a), and 72.106(b) require that the storage system will reasonable maintain confinement of radioactive materials under Section Part 72 normal, off-normal, and credible accident conditions. Since the TSC and the GTCC waste canister may be used for storage following transport, an analysis of their leakage rates and sensitivities is needed to describe the canisters intended functions during storage.

NAC Response

As noted in the NAC Response to RAI 2-12, DP4-1 and others, reference to the in-pool loading of the canister using the UMS® Universal Transport Cask is deleted. Consequently, at the time of transport, the canister will have previously been loaded, closed and tested in accordance with the detailed requirements presented in the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015. Among the requirements for canister closure is that the canister be leak tested to demonstrate leak tight confinement. To demonstrate a leak tight condition, the canister is subject to a helium leak test of 2×10^{-7} std cm^3/sec, using a detector sensitivity of 1×10^{-7} cm^3/sec. The leak test conforms to the definition of leak tight provided in Section 2.1 of ANSI N14.5-1997 and to the evacuated envelope test configuration described in A.5.4 of Table A.1 of the ANSI Standard. The leak tight test requirement applies regardless of canister contents.

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NAC Response to RAI DP8-1 (Continued)

The containment analysis for the transport cask does not take credit for the integrity of the canister, even though there are no normal or accident conditions that result in the breech of the canister while it is in the transport cask. Since credit for canister containment is not taken, the analysis assumes that the gases, volatiles, fines and particulates are released to the cask interior.

As shown in Section 4.2 of the UMS® Universal Transport Cask Safety Analysis Report, the transport cask containment boundary is helium leak tested to 3.3×10^{-5} std cm^3/sec, based on the postulated release from GE 9 x 9 BWR fuel in normal conditions of transport. The release from this BWR fuel bounds the release from all other canister contents.

The detailed methodology and analysis for transport cask containment conforms to ANSI N14.5-1997 and is presented in Chapter 4 of the Transport Cask Safety Analysis Report. As noted in Chapter 4, the GTCC waste does not contain a source of releasable gases and its principal radioactive source term consists of activated metal that is not releasable.

See the NAC Response to RAI DP4-1.

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DUAL-PURPOSE CANISTER

Dual-Purpose Canister

DP8-2 Include the acceptance or rejection criteria and describe the repair methods for welding the TSC and GTCC waste canister lids onto the shells.

Dual-purpose canisters must meet both transportation and storage requirements for use under the respective regulations. Section 72.122(l) requires that the storage system must be designed to allow ready retrieval of spent fuel for further processing or disposal. Since the TSC and GTCC waste canister are integral components of the Universal Transport Cask, a description of the welding requirements for the canisters is needed to assure that the canisters can be removed from the Universal Transport Cask.

<u>NAC Response</u>

As noted in the NAC Response to RAI 2-12, DP4-2 and others, reference to the in-pool loading of the canister using the UMS® Universal Transport Cask is deleted. Consequently, at the time of transport, the canister will have previously been loaded, closed and tested in accordance with the detailed requirements presented in the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015. While the GTCC waste packaging is not controlled by the requirements of 10 CFR 72, the GTCC waste canister is closed using essentially the same procedure as is used for closing canisters containing spent fuel as described in the response to RAI DP4-1. No welds are made or repaired while the canister is inside of the Transport Cask.

All of the welds made on the spent fuel and GTCC waste canisters, including shop and field installed welds, are fully specified in the License Drawings for these components provided in Section 1.3.4. The specification of the welds includes the appropriate

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reference to inspection and acceptance of those welds. See the NAC Response to RAI DP4-2.

If weld defects are identified which exceed the specified acceptance criteria for the weld, the weld defect is manually removed by grinding and the weld repaired. The weld repair shall comply with the requirements of NB-4450 of the ASME Code, Section III, Subsection NB (1995 Edition).

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Attachment for RAIs 1-11 and 1-12

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Draft Certificate of Compliance Contents Descriptions

5(b) Contents

(1) Type and form of material

(i) Irradiated PWR fuel assemblies with uranium oxide pellets having the same configuration as when originally fabricated. The maximum heat load per assembly is 833.3 watts. Prior to irradiation, the fuel assemblies must be within the following dimensions and specifications:

Assembly Type	**14 x 14**	**15 x 15**	**16 x 16**	**17 x 17**
Cladding Material	Zirc-4	Zirc-4	Zirc-4	Zirc-4
Maximum Initial Uranium Content (MTU/assembly)	0.415	0.481	0.442	0.468
Maximum Initial Enrichment (wt% ^{235}U)	4.2	4.2	4.2	4.2
Minimum Initial Enrichment (wt% ^{235}U)	1.9	1.9	1.9	1.9
Assembly Cross-Section (inches)	7.76 to 8.11	8.20 to 8.54	8.10	8.43 to 8.54
Number of Fuel Rods per Assembly	176 to 179	204 to 216	236	264
Fuel Rod OD (inch)	0.400 to 0.438	0.417 to 0.430	0.382	0.360 to 0.0379
Minimum Cladding Thickness (inch)	0.022	0.026	0.025	0.022
Pellet Diameter (inch)	0.345 to 0.380	0.358 to 0.369	0.325	0.303 to 0.324
Maximum Active Fuel Length (inch)	145.2	144	150	144
Maximum Assembly Length (inch)	160.2	159.8	178.3	165.8
Maximum Assembly Weight (lb)	1,302	1,515	1,430	1,505

Unenriched fuel may not be loaded.

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5(b) Contents

(1) Type and form of material (Continued)

Prior to transport, the PWR fuel assemblies placed in the Transportable Storage Canister must meet the cool times specified for each fuel assembly based on its enrichment and burnup:

Enrichment	Burnup ≤ 30 GWD/MTU Minimum Cool Time [years]				30 < Burnup ≤ 35 GWD/MTU Minimum Cool Time [years]			
wt % ^{235}U (E)	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	8	8	7	8	10	11	9	10
2.1 ≤ E < 2.3	7	8	6	7	10	10	8	10
2.3 ≤ E < 2.5	7	7	6	7	9	10	8	9
2.5 ≤ E < 2.7	7	7	6	7	9	9	7	8
2.7 ≤ E < 2.9	7	7	6	7	8	9	7	8
2.9 ≤ E < 3.1	7	7	6	6	8	8	7	8
3.1 ≤ E < 3.3	6	7	6	6	8	8	7	7
3.3 ≤ E < 3.5	6	6	6	6	7	8	6	7
3.5 ≤ E < 3.7	6	6	6	6	7	7	6	7
3.7 ≤ E ≤ 4.2	6	6	6	6	7	7	6	7
Enrichment	35 < Burnup ≤ 40 GWD/MTU Minimum Cool Time [years]				40 < Burnup ≤ 45 GWD/MTU Minimum Cool Time [years]			
wt % ^{235}U (E)	14 x 14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	15	15	13	15	20	21	20	20
2.1 ≤ E < 2.3	13	14	12	13	19	19	18	19
2.3 ≤ E < 2.5	12	13	11	12	17	19	17	17
2.5 ≤ E < 2.7	12	12	10	11	16	18	15	17
2.7 ≤ E < 2.9	11	11	9	11	15	18	14	17
2.9 ≤ E < 3.1	10	10	9	10	14	18	13	15
3.1 ≤ E < 3.3	10	10	9	10	13	17	13	15
3.3 ≤ E < 3.5	9	10	8	9	12	17	13	15
3.5 ≤ E < 3.7	9	10	8	9	11	17	12	15
3.7 ≤ E ≤ 4.2	8	10	8	9	11	15	12	14

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5(b) Contents

(1) Type and form of material (Continued)

(ii) Irradiated BWR fuel assemblies with uranium oxide pellets having the same configuration as when originally fabricated, which may include the channel. The maximum heat load per assembly is 285.7 watts. Prior to irradiation, the fuel assemblies must be within the following dimensions and specifications:

Assembly Type	**7 x 7**	**8 x 8**	**9 x 9**
Cladding Material	Zirc-4	Zirc-4	Zirc-4
Maximum Initial Uranium Content (MTU/assembly)	0.199	0.188	0.198
Maximum Initial Enrichment (wt% ^{235}U)	4.0	4.0	4.0
Minimum Initial Enrichment (wt% ^{235}U)	1.9	1.9	1.9
Assembly Cross-Section (inches)	5.51	5.51	5.51
Number of Fuel Rods per Assembly	48 to 49	60 to 63	74 to 79
Fuel Rod OD (inch)	0.563 to 0.570	0.483 to 0.493	0.424 to 0.441
Minimum Cladding Thickness (inch)	0.032	0.032	0.028
Pellet Diameter (inch)	0.487 to 0.490	0.405 to 0.416	0.357 to 0.376
Maximum Active Fuel Length (inch)	144	150	150
Maximum Assembly Length (inch)	175.9	176.1	176.1
Maximum Assembly Weight (lb)	683	681	646
Maximum Channel thickness (inch)	0.120	0.120	0.120

BWR fuel assemblies may be channeled with Zircaloy channels or unchanneled. Assemblies with stainless steel channels may not be loaded. Unenriched fuel may not be loaded.

Prior to transport, the BWR fuel assemblies held in the Transportable Storage Canister must meet the cool times specified for each fuel assembly based on its enrichment and burnup:

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(1) Type and form of material (Continued)

Enrichment wt % ^{235}U (E)	Burnup ≤ 30 GWD/MTU Minimum Cool Time [years]			30 < Burnup ≤ 35 GWD/MTU Minimum Cool Time [years]		
	7 x 7	8 x 8	9 x 9	7 x 7	8 x 8	9 x 9
1.9 ≤ E < 2.1	8	8	8	15	13	14
2.1 ≤ E < 2.3	8	7	7	13	12	12
2.3 ≤ E < 2.5	7	7	7	11	10	11
2.5 ≤ E < 2.7	7	6	7	10	9	9
2.7 ≤ E < 2.9	6	6	6	9	8	9
2.9 ≤ E < 3.1	6	6	6	8	8	8
3.1 ≤ E < 3.3	6	6	6	8	7	7
3.3 ≤ E < 3.5	6	6	6	7	7	7
3.5 ≤ E < 3.7	6	6	6	7	7	7
3.7 ≤ E ≤ 4.0	6	6	6	7	7	7
Enrichment wt % ^{235}U (E)	**35 < Burnup ≤ 40 GWD/MTU Minimum Cool Time [years]**			**40 < Burnup ≤ 45 GWD/MTU Minimum Cool Time [years]**		
	7 x 7	**8 x 8**	**9 x 9**	**7x 7**	**8 x 8**	**9 x 9**
1.9 ≤ E < 2.1	25	23	24	35	33	34
2.1 ≤ E < 2.3	22	20	21	32	30	31
2.3 ≤ E < 2.5	20	18	19	29	28	29
2.5 ≤ E < 2.7	17	16	17	27	25	26
2.7 ≤ E < 2.9	15	14	14	24	23	24
2.9 ≤ E < 3.1	13	12	13	22	20	21
3.1 ≤ E < 3.3	12	11	11	20	18	19
3.3 ≤ E < 3.5	11	10	10	18	16	17
3.5 ≤ E < 3.7	10	9	10	16	14	15
3.7 ≤ E ≤ 4.0	10	9	10	15	13	14

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5(b) Contents

(1) Type and form of material

(iii) Irradiated Maine Yankee PWR Uranium Oxide Fuel

(a) Irradiated intact Maine Yankee PWR 14 x 14 standard fuel assemblies with uranium oxide pellets having a configuration as when originally fabricated. The average heat load per assembly is 833 watts. Prior to irradiation, the fuel assemblies must be within the following dimensions and specifications.

Maine Yankee PWR Assembly	**14 x 14**
Cladding Material	Zirc-4
Maximum Initial Uranium Content (MTU/assembly)	0.415
Maximum Assembly Average Initial Enrichment (wt% ^{235}U)	4.2
Maximum Individual Fuel Rod Initial Enrichment (wt% ^{235}U)	4.203
Minimum Initial Enrichment (wt% ^{235}U)	1.9
Assembly Cross-Section (inches)	8.11
Number of Fuel Rods per Assembly	160 to 176
Fuel Rod OD (inch)	0.44
Minimum Cladding Thickness (inch)	0.024
Pellet Diameter (inch)	0.380
Maximum Active Fuel Length (inch)	145.2
Maximum Assembly Length (inch)	157.2
Maximum Assembly Weight (lb)	1,602

Intact fuel includes spent fuel assemblies with fuel rods having variable enrichments with a maximum fuel rod enrichment up to 4.21 wt % ^{235}U and that also have a maximum planar average enrichment up to 3.99 wt % ^{235}U. It includes fuel assemblies with annular axial end blankets. The axial end blanket enrichment may be up to 2.6 wt % ^{235}U.

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Intact fuel also includes fuel assemblies with one or more grid spacers missing or damaged such that the unsupported length of the fuel rods does not exceed 60 inches. End fitting damage including damaged or missing hold-down springs is allowed, as long as the assembly can be handled safely by normal means.

Prior to transport, the fuel assemblies must meet the indicated minimum cool times:

Minimum Initial Enrichment [wt %]	Loading Table for 14x14 Fuel Standard Configuration Maine Yankee Fuel Minimum Cool Time (Years) Burnup (B) (GWD/MTU)				
	B≤30	<30 B ≤ 35	<35 B ≤ 40	<40 B ≤ 45	<45 B ≤ 50
1.9 ≤ E < 2.1	6	8	11	18	27
2.1 ≤ E < 2.3	6	7	10	15	24
2.3 ≤ E < 2.5	6	7	9	14	22
2.5 ≤ E < 2.7	6	7	9	12	19
2.7 ≤ E < 2.9	6	6	8	11	17
2.9 ≤ E < 3.1	5	6	8	10	15
3.1 ≤ E < 3.3	5	6	7	10	15
3.3 ≤ E < 3.5	5	6	7	9	15
3.5 ≤ E < 3.7	5	6	7	9	14
3.7 ≤ E ≤ 4.2	5	6	7	9	14

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(b) Irradiated intact Maine Yankee PWR fuel with uranium oxide pellets, having a modified configuration including spent fuel installed in a Maine Yankee fuel can. Modifications include the removal, replacement, or insertion of fuel and non-fuel components, except fuel assemblies with fuel rods replaced with stainless steel rods. The authorized modified fuel assembly configurations are:

1. Fuel assemblies with inserted Control Element Assemblies (CEA) or inserted In-Core Instrument (ICI) Thimbles.
2. Fuel assemblies with fuel rods replaced with stainless steel or Zircaloy rods or with Uranium oxide rods nominally enriched up to 1.95 wt % ^{235}U.
3. Fuel assemblies with solid filler rods or burnable poison rods occupying up to 16 of 176 fuel rod positions.
4. Fuel assemblies with fuel rods replaced by solid stainless steel or Zircaloy rods.
5. Fuel assemblies with burnable poison rods replaced by hollow Zircaloy rods.

Prior to transport, the fuel assemblies must meet the indicated minimum cool times:

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Minimum Initial Enrichment [wt %]	Loading Table for 14x14 Fuel with Installed Non-Fuel Material or Modified Configuration Minimum Cool Time (Years)					Loading Table for 14x14 Fuel with Installed Control Element Assemblies in the Class 2 Canister Minimum Cool Time (Years)				
	Burnup (GWD/MTU)					Burnup (GWD/MTU)				
	B ≤30	<30 B ≤ 35	<35 B ≤ 40	<40 B ≤ 45	<45 B ≤ 50	B ≤30	<30 B ≤ 35	<35 B ≤ 40	<40 B ≤ 45	<45 B ≤ 50
1.9 ≤ E < 2.1	6	8	11	18	27	7	9	14	21	29
2.1 ≤ E < 2.3	6	7	10	16	24	7	9	13	19	27
2.3 ≤ E < 2.5	6	7	9	14	22	6	8	12	18	25
2.5 ≤ E < 2.7	6	7	9	13	19	6	8	10	16	23
2.7 ≤ E < 2.9	6	6	8	11	17	6	7	10	14	21
2.9 ≤ E < 3.1	5	6	8	10	15	6	7	9	13	19
3.1 ≤ E < 3.3	5	6	7	10	15	6	7	8	12	18
3.3 ≤ E < 3.5	5	6	7	9	15	6	7	8	11	17
3.5 ≤ E < 3.7	5	6	7	9	14	6	6	8	10	15
3.7 ≤ E ≤ 4.2	5	6	7	9	14	6	6	7	10	15

(c) Irradiated Maine Yankee PWR fuel with uranium oxide pellets, installed in a Class 1 UMS® Transportable Storage Canister, modified by the replacement of fuel rods with stainless steel rods.

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Assembly Number	Burnup (GWD/MTU)	Enrichment (wt %)	Cool Time (years)
N420	45	3.3	10
N842	35	3.3	6
N868	40	3.3	7
R032	45	3.5	9
R439	50	3.5	14
R444	50	3.5	19

(d) Irradiated Maine Yankee PWR fuel with uranium oxide pellets, having a modified configuration, which require prior loading in a Maine Yankee fuel can. Modifications include fuel that is consolidated. The allowable contents in a Maine Yankee fuel can are:

1 Consolidated fuel lattice structure with a 17 x 17 array formed by grids and top and bottom end fittings connected by four solid stainless steel rods. Maximum contents are up to 289 fuel rods with a total weight ≤ 2,100 pounds.
2 An intact Maine Yankee fuel assembly conforming to 5(b)(1)(iii)(a).

The required cool time prior to transport of spent fuel installed in a Maine Yankee fuel can shall be the same as the fuel would have in its original configuration in accordance with the cool times shown in 5(b)(1)(iii)(a).

(e) Irradiated Maine Yankee Greater Than Class C (GTCC) waste. The GTCC waste consists primarily of activated stainless steel reactor baffle core sections installed in a GTCC Waste Canister. The GTCC Waste Canister may contain shoring used to position GTCC waste sections, but shall not contain any significant fissionable material.

1 The total weight of the GTCC waste and shoring (if used) ≤ 20,000 pounds.
2 The total weight of the loaded GTCC Waste Canister ≤ 70,460 pounds.