May 2015

Revision 15A

NAC-STC

NAC Storage Transport Cask

SAFETY ANALYSIS REPORT

WVDP Vitrified Waste
and High Burnup Fuel
Amendment Application

RAI Response Package

Non-Proprietary Version

Docket No. 71-9235

NAC INTERNATIONAL

Atlanta Corporate Headquarters: 3950 East Jones Bridge Road, Norcross, Georgia 30092 USA
Phone 770-447-1144, Fax 770-447-1797, www.nacintl.com

Enclosure 1 – RAI Responses and Supporting Documents

Enclosure 1

RAI Responses and Supporting Documents for

NAC-STC SAR, Revision 15A

Enclosure 1 Sections:

1.1 RAI Responses

1.2 Supporting Documents:

a. DBTT Statistical Derivations, "HBU DBTT Data R08 (MLE and Firth).xlsx", Data Disk 1 of 1
b. Calculation 423-3000, Revision 2
c. Calculation 423-2015, Revision 0
d. Calculation 71160-2126, Revision 3

NAC INTERNATIONAL RESPONSE TO THE

UNITED STATES
NUCLEAR REGULATORY COMMISSION

REQUEST FOR ADDITIONAL INFORMATION

MAY 2015

FOR REVIEW OF THE CERTIFICATE OF COMPLIANCE NO. 9235, STC TRANSPORATION PACKAGE

(CoC NO. 9235 DOCKET NO. 71-9235)

MAY 2015

Table of Contents

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

G-1-1.[1] Revise the application to include a consolidated definition of "Contents" for the Model No. NAC-STC. Verify that all possible contents are clearly defined in the definitions section of the application and that possible contents are consistently described in the application.

Based on the information provided in the application, the possible contents of the Model No. NAC-STC are as follows:

1) Up to 26 pressurized water reactor (PWR) fuel assemblies in the following configurations:
   a) directly loaded fuel (uncanistered) configuration, or
   b) a sealed Transportable Storage Canister containing:
      i) Yankee Class spent fuel,
      ii) Connecticut Yankee spent fuel,
      iii) Greater Than Class C waste, or
      iv) LaCrosse Boiling Water Reactor (LACBWR) spent fuel; or
2) Up to five (5) HLW canisters in a sealed HLW overpack; or
3) Up to 20 PWR fuel assemblies containing high burnup (HBU) fuel up to 60 GWd/MTU with total heat load not exceeding 24 KW.

For example, Section 7.3.2, "Preparation of the NAC-STC Cask for Unloading," of the application describes the possible contents of the package, but does not mention HBU fuel or HLW debris as a potential content of the Model No. NAC-STC.

This information is needed to determine compliance with 10 CFR 71.87(a).

<u>NAC International Response to General Information Evaluation RAI G-1-1:</u>

SAR Section 7.3.2 has been revised to reflect the shipment of uncanisterized HBU PWR fuel assemblies and associated shielded thermal shunts. The number and location, of which, depends on the HBU fuel configuration chosen. The Chapter 1 "Contents (Payload)" definition has been revised to include up to 20 uncanistered HBU PWR fuel assemblies and associated shielded thermal shunts. SAR Section 1.2.1.2.7 has been revised to reflect the four directly loaded fuel basket configurations (i.e., the three high burnup PWR fuel and one low burnup PWR fuel configurations).

[1] In general, the nomenclature used for identifying the RAIs is as follows: Topic-Application's Chapter-Counter. (Topics: G – General Information; Co – Containment; Cr – Criticality; M – Materials; Sh – Shielding; St – Structural; Th – Thermal; and Op – Operations.)

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

Co-1-1. For the WVDP contents and the intact/undamaged HBU PWR fuel provide:

a. the licensing drawings that were not provided as part of this amendment that specify the containment boundary components;
b. a table in the RAI response that ties each licensing drawing containment boundary O-ring item number to each containment boundary O-ring in Table 4.1-1, "NAC-STC Containment Boundary," of the application; and
c. the materials, and corresponding composition and/or specifications for the metallic (WVDP contents and HBU fuel) and Viton (HBU fuel) seals. Revise the NAC-STC drawings to include this information.

There are multiple containment boundary configurations for the NAC-STC. The containment boundary for the WVDP contents was not specified in Section 4.1, "Containment Boundary," of the application, while the containment boundary for the directly loaded intact/undamaged HBU PWR fuel was specified.

Section 7.0, "Operating Procedures," of the application states that metallic O-rings must also be used when loading canistered HLW for transport. The staff is not able to ensure that all containment boundary components have been specified on the licensing drawings, and the containment boundary O-rings and grooves have been specified according to the information provided in Section 4.5.1, "Metallic O-rings," of the application for the WVDP contents. The staff is not able to ensure the containment boundary seals and grooves have been specified according to the information provided in Section 4.5.1, "Metallic O-rings," and Section 4.5.5, "Viton O-rings," of the application for the directly loaded intact/undamaged HBU PWR fuel. For example, the staff could not find the inner lid inner O-ring or groove, or the vent and drain port inner O-rings or grooves on the licensing drawings provided.

The containment boundary and its components, for each of the contents described in this amendment, should be clearly described in the application. See Table 4.1-1 of the application to identify each containment boundary O-ring.

The materials of construction as well as composition of seals and gaskets determine the safe operating temperature range.

This information is needed to determine compliance with 10 CFR 71.33(a)(4) and 71.51(a).

NAC International Response to General Information Evaluation RAI Co-1-1.

a. There are only two containment boundary configurations for the NAC-STC which are currently specified on the license drawings. One configuration uses dual metallic O-rings for the inner lid and the inner lid port covers. The second configuration uses non-metallic O-rings for the inner lid and the inner lid port covers. The innermost O-ring of the inner lid and the inner port covers are part of the containment boundary for their respective transportation configurations.

b. The following containment configurations and their respective O-ring identifications, with the exception of the inner lid item number for the WVDP package, is extracted from the licensing drawings provided in the submittal. License drawing number 630087-501 has been revised to reflect the proper lid configuration for the WVDP contents.

WVDP (Table 4.1-1, Containment Condition: B – Metallic O-ring)

- Dwg. 630087-501-99, Cask Assembly
- Dwg. 423-803-98, Cask Inner lid Assembly
- Dwg. 423-803-6, Metal O-ring, Helicoflex #U2825-(72.251)SOB
- Dwg. 423-803-7, Metal O-ring, Helicoflex #U2825-(73.497)SEB
- Dwg. 423-806-98, Inner Lid Port Coverplate Assembly
- Dwg. 423-806-4, Metal O-ring, Helicoflex #U2412-3875SEB
- Dwg. 423-806-5, Metal O-ring, Helicoflex #U2412-4500SEB

HBU (Table 4.1-1, Containment Condition: B – Non-Metallic O-ring)

- Dwg. 423-800-97, Cask Assembly
- Dwg. 423-803-99, Cask Inner lid Assembly
- Dwg. 423-803-15, Viton O-ring, Parker 0.275 Diameter, VM835-75
- Dwg. 423-803-16, Viton O-ring, Parker 0.275 Diameter, VM835-75
- Dwg. 423-806-99, Inner Lid Port Coverplate Assembly
- Dwg. 423-803-8, Viton O-ring, Parker #2-238, VM835-75
- Dwg. 423-803-9, Viton O-ring, Parker #2-244, VM835-75

c. The drawings have been revised to reflect the requested information.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

Co-1-2. Clarify if the Furon #10061-04-1-0 O-rings are metallic and if it is necessary for the O-rings to be metallic based on their function (see Drawing No. 423-800, Revision 16P, sheet 1 of 3, Item No.17 and Drawing No. 423-501, Revision 0, sheet 1 of 2, Item No. 13).

Section 4.5.2, "Blended Polytetrafluoroethylene (PTFE) O-rings," of the application includes a description of the Furon seal as blended polytetrafluoroethylene material, which is not metallic. Currently the drawings indicate that Part 17 of Drawing No. 800, and part 13 of Drawing No. 501 are metallic FURON #10061-04-0, with Section 4.5 of the application as the reference. This section contains data for different types of seals, all with different maximum operating temperatures.

This information is needed to determine compliance with 10 CFR 71.33(a)(5) and 71.51(a).

NAC International Response to General Information Evaluation RAI Co-1-2:

The original specified Furon "boss seal" is a metallic seal (321 Cres Jacket/302 Cres Crush Ring), silver coated for MS33649-4 ports. The seal was originally specified in support of 10 CFR Part 72 storage functions (i.e., sealing the pressure transducer which would be monitoring the inter lid region of the cask) of the NAC-STC. In the 10 CFR Part 71 transport configuration, it simply functions as a seal for a port plug. The seal provides no containment functions. Furon no longer distributes this component. Therefore, NAC is revising the following drawings to indicate the following description in the bill of materials:

- Dwg. 423-800, Item #17 and Dwg. 630087-501, Item #13
    - "MS33649-4 Boss Seal, metallic, silver coated"

Additionally, as the seals do not provide a containment function, NAC has added the option of using a polymer, VM835-75, boss seal.

With respect to references of PTFE in Section 4.5.2, the information would support future implementation of PTFE seals. Currently none are used in the transport system. This information may be legacy descriptions from earlier cask configurations as the STC has a long licensing history. Any reference to "Furon" is coincidental and does not conflict with the information provided above.

The section supporting the metallic boss seal is Section 4.5.1, page 4.5-10, under "Boss Seals" and includes information provided by the manufacturer "Fluorocarbon" and the seal trade name of "Fit-o-seal". As indicated by the information provided, a metallic boss seal is used for the MS33649 port.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

Co-1-3. Revise the licensing drawings, as necessary, to ensure these include specific information about the Viton compound to be used for the O-rings.

The description of the material should be specific enough to ensure that a compound with a [redacted] rating is used.

This information is needed to determine compliance with 10 CFR 71.71(c)(2).

NAC International Response to General Information Evaluation RAI Co-1-3:

[redacted]

COMPOUND DATA SHEET

Parker O-Ring Division, North America

MATERIAL REPORT

Report Number: 92880
Date: 3/13/2013

Title: Evaluation of Parker Compound VM835-75

Elastomer Type: Fluorocarbon (FKM)

Purpose: To obtain typical test data.

Specification: ASTM D2000 M2HK710 A1-10 B38 E078 Z1 (Shore A Hardness 75 +/-5), Z2 Elongation 125% min, Z3 (Specific Gravity), Z4 (TR-10)

Color: Black

Recommended Temperature Range: -40°F to 400°F

Recommended For: Mineral oil and grease, IRM 901 oil, IRM 902 oil, IRM 903 oil, non-flammable hydraulic fluids, silicone oils and greases, aliphatic hydrocarbons (propane, butane, natural gas), aromatic hydrocarbons (benzene, toluene), chlorinated hydrocarbons (trichloroethylene and carbon tetrachloride), gasoline, high vacuum, ozone, weather, and aging resistance.

Not Recommended For: Glycol based brake fluids, ammonia gas, amines, alkalis, superheated steam, and low molecular weight organic acids (formic and acetic acids).

Additional Approvals: N/A

Parker O-Ring Division
2360 Palumbo Drive
Lexington, Ky 40509
(859) 269-2351

REPORT DATA

Original Physical Properties	Test Method	Spec Limits	Test Results
(Z1) Hardness, Shore A, pts.	ASTM D2240	75 ±5	78
Tensile Strength, PSI (Mpa)	ASTM D412	1450 (10)	3059
(Z2) Ultimate Elongation, %	ASTM D412	125	215
(Z3) Specific Gravity	ASTM D297	±.03	1.8
Fluid Resistance (Basic Requirement)			
IRM 903, 70 hrs @ 302°F			
Volume Change, %	ASTM D471	+10	+2
(A1-10) Heat Age			
70 hrs. @ 482°F			
Hardness Change, pts.	ASTM D573	+10	+3
Tensile Strength Change, %		-25	-22
Ultimate Elongation Change, %		-25	+8
(B38) Compression Set (Plied)			
22 hrs. @ 392°F			
Percent of Original Deflection, Max	ASTM D395 Method B	50	13
(E078) Fluid Resistance			
Service Fluid 101, 70 hrs @ 392°F			
Hardness Change, pts.	ASTM D471	-15 to +5	-8
Tensile Strength Change, %		-40	-6
Ultimate Elongation Change, %		-20	-1
Volume Change, %		0 to +15	+11
(Z4) Low Temperature Resistance			
TR-10, temperature °F , C	ASTM D1329	report	-22 (-30)

"Purchaser use only. Reproduce only in full. Data pertains to items referenced only."
"The recording of false, fictitious, or fruaudulent statements or entries in this report may be punishable as a felony under federal law."

Parker O-Ring Division
2360 Palumbo Drive
Lexington, Ky 40509
(859) 269-2351

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

Cr-1-1. Explain the proposed changes to the spent fuel baskets, including changes to the neutron absorbers and fuel tubes. Revise the drawings and/or criticality analysis in the application, as necessary.

For example, Drawing No. 423-800, appears to indicate that the baskets for direct-load and canister-load packages may use the alternate fuel tubes and the new absorber specifications (see Drawing No. 423-800, Cask Assembly 98). The criticality analysis only considers these changes for the direct-load packages. Therefore, the applicant should modify the drawings to clearly indicate that the proposed fuel tube and neutron absorber changes only apply to the direct-load baskets and packages. Otherwise, the applicant should modify the criticality analysis to address these changes for canister-load packages.

This information is needed to determine compliance with 10 CFR 71.55 and 71.59.

NAC International Response to General Information Evaluation RAI Cr-1-1:

The alternative spent fuel tube configuration is applicable to the 26-assembly directly loaded, non-canister loading of the NAC-STC. Neutron absorber thickness is slightly increased (0.025 inches) with a corresponding reduction in fuel tube outer width, retaining disk opening size and locations.

The Drawing No. 423-870 basket assemblies (invoking the two tube configurations) are not invoked for the canistered system configuration. The cask assembly text on sheet 2 of 423-800, assembly 98, referring to "Canistered Contents" relates to the seal configuration, not basket configuration (no canister/canister spacer configuration for cask cavity contents is invoked on the 423-800 drawing bill of materials). The assembly 98 description is retained from the previous licensed drawing and is not modified as part of this application.

No revision to criticality evaluations are required.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

Cr-1-2. Modify the proposed drawings in the application as follows:

Drawing No.	Request...	Reference to Application
a. 423-875	Include Detail A – A	Section B – B of the drawing includes a reference to Detail A – A. However, that detail label does not appear anywhere else in the drawing.
b. 423-878, Sheet 2 of 2	Modify the neutron absorber width to be 8.2 inches.	The dimension for the neutron absorber plate needs to be consistent on all sheets of the drawing (where it appears) and with the dimension used in the criticality analysis.
c. 423-800	Clarify whether or not the cask assembly No. 423-800-98 should be part of the package in this drawing.	Cask assembly No. 423-800-98 only appears in Drawing No. 423-800.
d. 423-875	Include the dimensions and the tolerances of components that affect the criticality safety design of the package (e.g., the neutron absorber plates and the fuel tubes)	Drawing No. 423-875 was modified to remove the tolerances and change the definitions of the dimensions shown on the drawing.
e. 423-878		New Drawing No. 423-878 lacks tolerances and defines dimensions in the way that is done on the proposed revision to Drawing No. 423-875.

f. Provide tolerances on dimensions of components that affect the criticality safety design of the package. The dimensions and tolerances of the neutron absorber plates and the fuel tubes are important to the criticality safety design of the package. Tolerances should be included to ensure that the criticality analysis is bounding for the package design and that the package is fabricated in a manner that is within the bounds of such analysis.

This information is needed to determine compliance with 10 CFR 71.55 and 71.59.

NAC International Response to General Information Evaluation RAI Cr-1-2:

a. NAC has revised Drawing 423-875 to correctly label Detail A-A.

b. NAC has revised Drawing 423-878 to provide consistency in the neutron sheet width.

c. NAC has not directly implemented the 423-800-98 configuration. There are similar configurations filed under Project Specific drawings, e.g., Connecticut Yankee (414), Yankee Rowe (455), WVDP (630087) and LACBWR (630045). These packages use the 423-843 Assembly drawing; therefore, 423-800-98 should not be part of any package assemblies at this time.

d. Tube tolerances have been shown to have a negligible effect on system criticality. Consistent with MAGNASTOR licensing, the thickness and tolerance of the neutron absorber sheet has been added to the drawing.

e. See d.

f. See d.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-1. Define and characterize the "HLW debris." Incorporate this information in the definitions sections of the SAR as well as any other sections, as appropriate.

Even though the applicant describes "HLW debris" as a potential content, the applicant did not define the term "HLW debris" in the safety analysis report for the NAC-STC. The 10 CFR Part 71 regulations require that the physical and chemical composition of all contents be specified in the application.

This information is needed to determine compliance with 10 CFR 71.33(b)(3).

<u>NAC International Response to General Information Evaluation RAI M-1-1:</u>

The terminology section of Chapter 1 (Table 1-1) is updated to provide the requested information (a HLW Debris Canister line is added).

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-2. Explain and justify that Parker compound V747-75 bounds or has a wider temperature range of performance then Parker compound V0835-75.

These compounds are mentioned in the application as follows:

- Drawing No. 501 specifies Parker 3-908V747-75 as the Viton O-ring.
- Section 3.7.3.2 indicates that the support for the safe operating temperature of the O-ring can be found in Section 4.5 of the application.
- Section 4.5.5 provides the manufacturer's specifications and NAC testing results using ANSI N14.5-1997 procedures for compound V0835-75.

Before the operating temperature range presented in the SAR can be accepted, the applicant needs to provide information demonstrating that these two compounds are the same.

This information is needed to determine compliance with 10 CFR 71.51(1)(2).

<u>NAC International Response to General Information Evaluation RAI M-1-2:</u>

All polymer seals in the NAC-STC package have been updated to Viton, compound VM835-75. Supporting datasheets for the VM835 compound have been added to SAR Section 4.5.2 and demonstrate compliance with the necessary temperature range of -40°F to 400°F.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-3. Provide the list of materials used in the metallic seals or the manufacturers' number for the metallic seals. Revise the NAC-STC drawings to include this information.

The list of changes for Drawing Nos. 423-800 and 423-870 indicates that a Part 29 was added. Nevertheless, this information is not in the Bill of Materials for these drawings.

This information is needed to determine compliance with 10 CFR 71.33(b)(3) and 71.111.

NAC International Response to General Information Evaluation RAI M-1-3:

For metallic seals used in non-containment port boss applications, the drawings have been revised to include size, material, and coating. Information for the metallic inner lid seals, outer lid seal, and the inner port cover seals is included in the data sheets provided in SAR Section 4.5.1. Sizes are listed in the Bill of Materials for each seal's respective drawing.

The list of drawing changes was in error, as there is no Item 29 added to 423-800, and NAC Drawing 423-870 only lists 15 items.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-4. Specify the assembly methods on Drawing No. 423-880 (Shielded Thermal Shunt Assembly, NAC-STC Cask).

Drawing No. 423-880 includes a list of materials, but does not specify assembly methods such as weld types for the Thermal Shunt.

This information is needed to determine compliance with 10 CFR 71.33(a)(5) and 71.111.

NAC International Response to General Information Evaluation RAI M-1-4:

NAC has revised Drawing No. 423-880 (Shielded Thermal Shunt Assembly, NAC-STC Cask) to provide weld details for the shunt assembly.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-5. Provide the following information to support the proposed approach for the high burnup fuel contents, [redacted]

a. Hydrogen (H_2) solubility - An analysis for the H_2 solubility as a function of temperature. [redacted]

b. Precipitation of radial hydrides - Technical support to demonstrate that, [redacted] there would be no change of materials properties [redacted] of the spent nuclear fuel [redacted]

c. An updated Figure 1.1-3, [redacted] to include all available data.

Revise the application to incorporate the responses of Items a. through c.

[redacted]

[2] This is the applicant's assumption for supporting its proposed approach.

[redacted] he applicant should provide a defensible argument supported with quality and sufficient data to justify its position. The current application does not make this argument.

This information is needed to determine compliance with 10 CFR 71.33(b) and 71.55(b)(1).

NAC International Response to General Information Evaluation RAI M-1-5:

[redacted]

[3] The reference provided by the applicant was presented as part of an IAEA Cooperative research program and was only preliminary in nature, therefore, it cannot be used on public licensing documents.
[4] Reference 8 of Section 1.1.1 was co-authored by the reviewer.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

STC
Docket No: 71-9235
CoC No: 9235

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-6. [redacted]

NAC International Response to General Information Evaluation RAI M-1-6:

[redacted]

[redacted]

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-7. For a fuel burn up of 60 GWd/MTU (application's limit), provide the following:

[redacted]

Revise the application to incorporate the responses of Items a. and b.

The analysis provided in the application is not for the burnup limit requested in the application.

This information is needed to determine compliance with 10 CFR 71.35(c) and 10 CFR 71.55(b)(1).

NAC International Response to General Information Evaluation RAI M-1-7:

a. [redacted]

b. [redacted]

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-8. Revise Section 1.1.1, "Licensing Approach (High-Burnup Fuel)," of the application to include a table containing the changes in temperatures (ΔTs) accounting for the temperature drop during the 1-year maximum transport period. [redacted]

The relevant drop in temperature is not only caused by a difference in ambient temperature but also any drop in temperature that might be caused by decay during the time from loading the cask to end of shipment, which is limited to one year. The ΔT in the table on page 1.1-12 only includes the change due to the ambient temperature change. [redacted]

This information is needed to determine compliance with 10 CFR 71.35(c) and 10 CFR 71.55(b)(1).

NAC International Response to General Information Evaluation RAI M-1-8:

[redacted]

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-9. For Section 1.1.1, [redacted] provide the following [redacted]

a. references [redacted]
b. [redacted]

[redacted]

This information is needed to determine compliance with 10 CFR 71.31(a)(2) and (b); 71.33(b); 71.55(b)(1); and 71.55(e)(1).

NAC International Response to General Information Evaluation RAI M-1-9:

[redacted]

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-10. [redacted]

This information is needed to determine compliance with 10 CFR 71.51(a).

NAC International Response to General Information Evaluation RAI M-1-10:

a. [redacted]

b.

c.

d.

[redacted]

STC
Docket No: 71-9235
CoC No: 9235

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-11. Revise the application to:

a. Provide operational experience or plans to provide operational experience confirming that the fuel remains intact after normal transport.

b. Provide consequence analyses (shielding, criticality, confinement, and retrievability):
   i. assuming [redacted] fuel failure or other defensible value for normal conditions of transport, and
   ii. assuming 100% fuel failure or other defensible value for accident conditions of transport.

Based on the staff's knowledge, there is no experimental data available to confirm that the fuel rods do not fail under normal conditions of transport. Therefore, the NRC required defense-in-depth to ensure that the fuel would not be changed after normal conditions of transport. [redacted] The applicant should satisfactorily provide the information requested [redacted]

This information is needed to determine compliance with 10 CFR 71.55(d)(2).

NAC International Response to General Information Evaluation RAI M-1-11:

a. 10 CFR 71.95(b) requires a written report to be submitted to the Commission for instances in which the conditions in the CoC were not followed during shipment. With regards to the high burnup (HBU) PWR fuel scope of this application, HBU PWR fuel is to remain undamaged during normal conditions of transport as presented in the application. In the event that the HBU fuel is no longer intact after a normal shipment, a report would need to be prepared and submitted to the NRC detailing how the fuel-specific functions were not fulfilled. Submittal of this information provides operational experience on the HBU PWR fuel in the STC.

b. Containment

Per the NRC Standard Review Plan, the NAC-STC is evaluated for containment at 3% fuel failure for the normal condition and 100% fuel failure for the accident condition. The NAC-STC, for high burnup fuel shipment, will be configured for leak tight transport. Releasable radionuclide inventory will not impact system performance.

Retrievability

As indicated in SAR Section 7.3.2, the pressure and radionuclide testing is performed at the vent port prior to any additional steps (e.g., cool down and lid removal) being taken. Should unexpected pressure or radionuclide inventories be noted, contingency actions, as needed, can be taken at the receipt facility. Fuel assembly structure (e.g., guide tube/nozzle) is not expected to be deteriorated for high burnup PWR fuel (i.e., no significant stress levels within the guide tube structure that would result in similar postulated concerns as those in the fuel rods). Therefore, removal of the assembly, including the bulk fuel material regardless of hypothetical clad failure, will be possible. A physical mechanism for bulk separation of spent fuel pellets from the clad is not available within the transport cask (EPRI Report 1015050). A postulated release of pellets from the assembly skeleton would further require removal of the material from the tight packed rod matrix of a PWR assembly which will tend to trap loose bulk pellet material prior to exiting the lattice. Overall retrievably of the bulk material is not a concern within the context of high burnup PWR fuel assembly shipment.

Criticality

A number of reports have investigated potential reactivity effects of fuel relocation within a transport cask. ORNL/TM-2012/325 provides a summary of various NRC/ORNL, Presentation (PATRAM), and EPRI reports. The NAC-STC, with PWR high burnup as a payload, was evaluated as fresh fuel content in a flux trap basket. This is a configuration similar to the fresh fuel configuration applied in the reference documents. Per Appendix B of ORNL/TM-2012/235, a credible degraded condition k_{eff} increase is less than 3% Δk_{eff}. As the system is evaluated to a maximum 0.95, the package will remain subcritical. Two primary references to the ORNL report are EPRI Report 1015050, which concludes a reasonable maximum reactivity increase of 0.031, while NRUEG/CR-6835 lists potential reactivity (k_{eff}) increases as high as 0.0703 for a flux trap fresh fuel basket. For the higher NUREG value to be achieved requires complete removal of cladding. Complete removal of the clad from the critical sections of the basket is not a credible scenario. Removal of the "clad removed" scenario yields reactivity impacts similar to those reported in EPRI and ORNL/TM-2013/325.

A postulated normal condition 3% fuel failure percentage will have minimum impact as documented in the referenced reports. Fuel failure of 3% for a 17x17 fuel assembly is representative of 8 fuel rods per assembly. Reactivity impacts of this type of failure were documented in the various reports to be less than 1% Δk_{eff}.

As documented above, the system will remain subcritical when considering credible hypothetical fuel failure scenarios. To accommodate non-credible scenarios and demonstrate substantial reactivity margin, a simplified scoping analysis was performed applying actinide only (AO) burnup credit to the NAC-STC model. As the fuel rod failure concern is associated with high burnup PWR fuel, a 45 GWd/MTU fuel material composition was considered. For simplicity and conservatism, only the lowest burned end-node burnup/composition was applied over the full fuel length (i.e., a 20.8 GWd/MTU material composition). Using the same 26-assembly model applied in the previously submitted analysis, and replacing the fuel material by a depleted composition, (ISG 8, Rev. 3, Table 1 listed AO isotopes) yielded a decrease of ~9% Δk_{eff}. This demonstrates substantial margin to a critical condition exists to accommodate even non-credible fuel rearrangement scenarios.

It should also be noted that the design basis criticality evaluations of the NAC-STC rely on a 26-assembly fuel load, while high burnup PWR fuel configurations are limited to a maximum 20-assembly load. This will provide further increases in reactivity margins.

Shielding
General Discussion:

To obtain any resolvable shielding impact from fuel rearrangement, bulk fuel material must be released from the clad and migrate out of the fuel tube or rearrange within the fuel tube to a significant extent. As documented in NUREG/CR-6835, a postulated movement of the higher neutron source central source region (see SAR Figure 5.8.2-2 for source profile for undamaged rods) to axial ends of the fuel will substantially increase dose rates for top/bottom surface (direction of source shift dependent) with a corresponding decrease in fuel centerline dose rates. As documented in the NAC-STC SAR Section 5.8.2, limiting 2-m licensing dose rates for the exclusive use package are obtained at the cask centerline with dose rates at the end region, where fuel shift would result in dose increase being significantly lower. A source shift would increase cask radial surface dose rates which are reported at a factor of 5 lower than those permitted for exclusive use shipment. Cask top and bottom surface NAC-STC dose rates are well below limits, and margins would not be significantly affected by the ~15% increase documented in the NUREG for this type of scenario. Note that the NUREG employed a 75 GWd/MTU burnup fuel source which is significantly higher in neutron source than proposed for PWR HBU fuel assembly transport in the NAC-STC. Accident condition rearrangement of fuel material will have smaller effects than normal condition changes as the neutron shield primary moderator component (hydrogen bound in the solid NS-4-FR shield) is removed from the analysis. All gaseous material is removed from the model. As the neutron shielding is removed in the model, normal condition axial shift effects associated with the neutron shield to impact limiter gaps are minimized. As documented in the SAR, accident dose rates at the cask radial surface peak sharply at the fuel centerline. Movement of material to the ends of the cask will reduce centerline dose rates. Cask top and bottom accident dose rates are a factor of 5 below limits and provide sufficient margin to have limits not impacted by potential fuel reconfiguration.

Normal Condition:

A 3% fuel failure yields ~8 rods per assembly (264 fuel rod 17x17 lattice). For a maximum 20-assembly PWR HBU payload, this is equivalent to ~160 fuel rods. This is 60% of one fuel assembly. NUREG/CR-6835 investigated full payload failures. At the equivalent of 60% of one assembly, the potential effects, already documented to be acceptable, would be significantly mitigated. Furthermore as discussed in the retrievability/criticality section of this response, the gross release of fuel material would require a credible mechanism to release the bulk pellet material from the clad, which has not been documented to exist, and subsequent release/movement of the bulk material from or within the rod lattice, which would tend to trap material between the rods. Overall, the effect of the postulated failure condition would not result in payloads exceeding regulatory normal condition limits on dose rate.

Accident Condition:

Assuming 100% fuel failure and reconfiguration during accident conditions is not expected to result in dose rates in excess of regulatory limits, as this evaluation sequence removes the critical neutron attenuation components resulting in a neutron dominated dose rate with substantial dose peaking at the cask centerline. As stated in the general discussion, the "neutron shielding removed" model assumption removes/limits the effects of the neutron shield to impact limiter gap seen during normal conditions. Note that NS-4-FR has been documented to be resistant to the fire condition and has been demonstrated to retain its property post fire for the majority of the shield thickness. A redistribution of the higher center line source would result in a smoothing of the radial cask dose rate profile. Axial dose rates (top or bottom) would increase under this assumed condition, but significant margin to limits exist.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-12. Provide and justify the maximum cladding stress expected at the maximum expected cladding temperature of [redacted] for HBU fuel. Revise the application to incorporate this response.

The DBTT is a function of the cladding material and the maximum stress that the cladding experiences at its maximum temperature, during drying or transport. The maximum stress in the cladding should be known to determine the appropriate DBTT value. Section 1.1.1, [redacted]

[redacted]

This information is needed to determine compliance with 10 CFR 71.55(b)(1).

NAC International Response to General Information Evaluation RAI M-1-12:

[redacted]

[redacted]

[redacted]

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

M-1-13. [redacted]

This information is needed to determine compliance with 10 CFR 71.55(d)(2).

NAC International Response to General Information Evaluation RAI M-1-13:

NAC has adequately described the licensing basis and approach in SAR Section 1.1.1. Therefore, the flowchart shown in Figure 1.1-2 has been removed.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

GENERAL INFORMATION EVALUATION

St-1-1. Explain how the applicant ensures that the Model No. NAC-STC package complies with the ambient temperature initial condition of testing per provisions in paragraphs 666 and 679 of IAEA Specific Safety Requirements 6 (SSR-6).

Section 1.1, page 1.1-2, of the application states the following:

> "[T]he NAC-STC is designed to meet 10 CFR 71 and IAEA Safety Series (*sic*) No. SSR-6 licensing requirements for spent fuel transport packages."

IAEA SSR-6 and 10 CFR Part 71 free drop test ambient temperature range for Type B(U) and B(M) packages is different (see table below).

Initial Conditions for Type B(U) and B(M) packages (ambient temperature range)	
10 CFR 71.73(b)	666 and 679
–29°C to +38°C	–40°C to +38°C

The applicant repeats the aforementioned statement in Section 1.1, page 1.1-2, in several citations in the application. The statement is overly broad and it is unclear whether the SSR-6 requirements have been evaluated in detail for the proposed contents. [redacted]

This information is needed to determine compliance with 10 CFR 71.35(a) and 71.73(b).

<u>NAC International Response to General Information Evaluation RAI St-1-1:</u>

On December 30, 1996, NAC submitted an application to the NRC to include Yankee Class fuel as authorized contents. As part of the NRC acceptance review, the NRC noted that the package did not have a "-85" designation. NRC project manager, Marissa Bailey conducted a teleconference with NAC's Tom Thompson to inform NAC that without the "-85" designation only NAC-STC packages whose fabrication have been satisfactorily completed by April 1, 1999 would be authorized for use under the general license of 10 CFR 71.12, which is now 10 CFR 71.17. Since the NAC-STC was originally approved in 1994 it is subject to 10 CFR 71.17(e) and the restrictions of 10 CFR 71.19.

NAC agreed to pursue the "-85" designation with the application at the time. This is documented in NRC letter dated February 14, 1997. Subsequent to these correspondences, NAC submitted supporting information to meet the IAEA standards for obtaining the "-85" designation. This is

documented in the NRC SER for CoC Revision 2 dated March 25, 1999. Later on, NAC obtained the "-96" designation as part of a CoC revision request.

Upon further review, NAC has noted that there is a necessary change due to an omission with the original "-85" designation submission. Specifically, Condition #3 on SAR Revision 17 page 2.7.1.1-3 should say "-40°F instead of "-20°F". This is apparent as the SAR sections referenced in bullet #3 on SAR Revision 17 page 2.7.1.1-2 include SAR Section 3.4.3 for the minimum thermal temperature condition. SAR Section 3.4.3 clearly states the minimum temperature condition is a uniform -40°F temperature distribution. This omission is easily compared to the three conditions listed in SAR Section 2.6.7.1.1 for bullet #3, which are correct.

With regards to the specific reference in this RAI about temperature-dependent crush strengths, SAR Revision 17. Section 2.6.7.4.1.3 on page 2.6.7.4-8 states that the g-loads used for the 1-foot and 30-foot drops are based on a cold crush strength of -40°F for the redwood impact limiters. Likewise, the end of the first full paragraph on SAR Revision 17, page 2.6.7.4-43 states this for the balsa impact limiters. As an example, SAR Revision 17, Section 2.7.1 states these impact loads are applied for the 30-foot free drop in the third bullet on SAR page 2.7.1-1. Therefore, Condition #3 on SAR Revision 17, page 2.7.1.1-3 is revised to say "-40°F instead of "-20°F".

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

St-2-1. With respect to the STC-WVDP design weight of [redacted] with the balsa wood impact limiters (see Sections 2.12.2 and 2.12.6.7.4.1 of the application)[5] configured for a certified NAC-STC design weight of [redacted] reevaluate the package structural performance by:

a. addressing potentially higher than originally calculated g-loads for evaluating the package structural performance for applicable drop orientations, and

b. based on the evaluation requested above, revising appropriate sections of the application (e.g., Section 2.12), to include the applicable g-loads for evaluating individual package components and contents.

For the same free drop distance of 30 ft., a lighter package is known to result in a higher maximum deceleration than the heavier package equipped with the identical impact limiters. On this note, the application does not appear to have performed sufficient evaluation of the applicable free-drop rigid body decelerations for the package structural evaluation.

This information is needed to determine compliance with 10 CFR 71.35(a), 71.71(c)(7) and 71.73(c)(1).

NAC International Response to Structural Evaluation RAI St-2-1:

The lower design basis weight (236,000 lbs) of the STC-WVDP Cask is attributed to the lower contents weight of 45,800 lbs vs. the total fuel basket weight of 67,195 lbs in the licensed design basis weight (260,000 lbs) of the CY-MPC Cask.

The reduction in the contents weight of 21.4 kips also means that the total kinetic energy which must be absorbed is also reduced. Since crush strength curves for balsa wood and redwood monotonically increase with strain, the resulting strain, and crush strength, could be reduced. The bounding condition is the cold condition for which the properties have been evaluated at, -40F. Additionally, for the cold condition, the properties of the balsa wood have been factored by another 10% to account for uncertainties, one of which could be considered to be the reduction of the weight by approximately 10%. If the acceleration of the cask is increased by 11%, the original CY-MPC acceleration of 39.9g increases to 44.2g. The bounding condition for the stresses developed in the cask are those for the top end drop in which the inertial loading due to the contents weight is applied to the lid. The inertial loading applied to the closure lid for the top end drop is the acceleration times the contents weight. A comparison of the inertial loading to the closure lid

[5] Section 2.12.2, "Weights and Centers of Gravity," and Section 2.12.6.7.4.1, "Balsa Impact Limiters."

for the two configurations is shown below. The inertial loading applied to the closure lid associated with the total cask weight configuration of 260,000 pounds is bounding by more than 30% over the loading associated with the STC-WVDP configuration.

Comparison of the Inertial Loading Applied to the Closure Lid for the Top End Drop

	CY-MPC canistered Fuel	STC-WVDP HLW Contents
Total contents weight (lb)	67,195	45,800
End Drop Acceleration (g's)	39.9	44.2
Inertial load applied to the lid (kips)	2,681	2,024

Likewise, for the side drop, this means that the total kinetic energy which must be absorbed is also reduced due to the reduction in the contents weight. While for the end drop, the backed area for the balsa wood (and therefore the cross sectional area for the crushed balsa wood) remains the same, the crush area for the side drop does changes with the crush depth. To evaluate this, the finite element model used for the 260,000 pounds design basis weight was altered to 236,000 pounds, and the resulting acceleration increased from 48.5 g to 51.9g. The inertial loading applied to the cask shells is determined by the product of the contents weight and the acceleration, which is compared in the table below. The inertial loading applied to the cask shells associated with the total cask weight configuration of 260,000 pounds is bounding by more than 35% over the loading associated with the STC-WVDP configuration.

Comparison of the Inertial Loading Applied to the Cask Shells for the Side Drop

	CY-MPC canistered Fuel	STC-WVDP HLW Contents
Total contents weight (lb)	67,195	45,800
End Drop Acceleration (g's)	48.5	51.9
Inertial load applied to the lid (lb.)	3,259	2,377

This confirms that the total inertial loads from the cask contents for the end drop and side drops with lighter weights are bounded by CY-MPC design basis weight of 260,000 pounds. The relevant SAR sections for the STC-WVDP evaluations have been revised to confirm that the evaluations associated with the design basis weight of 260,000 pounds bounds the cask body stresses for STC-WVDP contents.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

St-2-2. Revise, as appropriate, Section 2.12.1.3, "Miscellaneous Structural Failure Modes," statement:

> "[F]or the HLW Overpack, the shell weldment and closure lid are constructed of Type 304/304L stainless steel. The top and bottom spacers are constructed of Type 304 stainless steel..."

to be consistent with Section 2.12.1.1, "Discussion," statement:

> "[F]or transport in the NAC-STC cask there are two spacer assemblies; one spacer is positioned below the HLW Overpack and the second spacer is positioned above the HLW Overpack."

Section 2.12.1.1 notes that the top and the bottom spacers are placed outside the HLW overpack, which appears to be inconsistent with the description in Section 2.12.1.3.

This information is needed to determine compliance with 10 CFR 71.33(a)(5).

NAC International Response to Structural Evaluation RAI St-2-2:

SAR Section 2.12.1.3 will be revised to be consistent with Section 2.12.1.1 as follows:

"For the HLW Overpack, the shell weldment and closure lid are constructed of Type 304/304L stainless steel. There are two spacer assemblies used for transport of the HLW Overpack in the STC Cask. For the two spacer assemblies; one spacer is positioned below the HLW Overpack and the second spacer is positioned above the HLW Overpack. The two spacers are constructed of Type 304 stainless steel. Type 304 stainless steel is an austenitic stainless steel which does not undergo a ductile-to-brittle transition in the temperature range of interest for transport. Therefore, brittle fracture is not a concern."

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

St-2-3. Annotate the finite element analysis (FEA) model depicted in the sketch with sufficient detail to illustrate the following:

a. the element discretization for the closure-lid-to shell weld, and
b. the critical stress evaluation paths in both the closure lid and the adjacent overpack shell.

The sketch presented in Figure 2.12.6.12-3, "HLW Overpack Assembly-End Drop Models," lacks the model attributes information. The information requested should facilitate the safety review of the structural analysis model assumptions and stress results.

This information is needed to determine compliance with 10 CFR 71.33(a)(5).

NAC International Response to Structural Evaluation RAI St-2-3:

a. The plot of the finite element mesh in Figure 2.12.6.12-3, showing closure lid and weld, has been updated to show the location of the lid to shell weld. Further details on the discretization are provided below.

End Drop Model

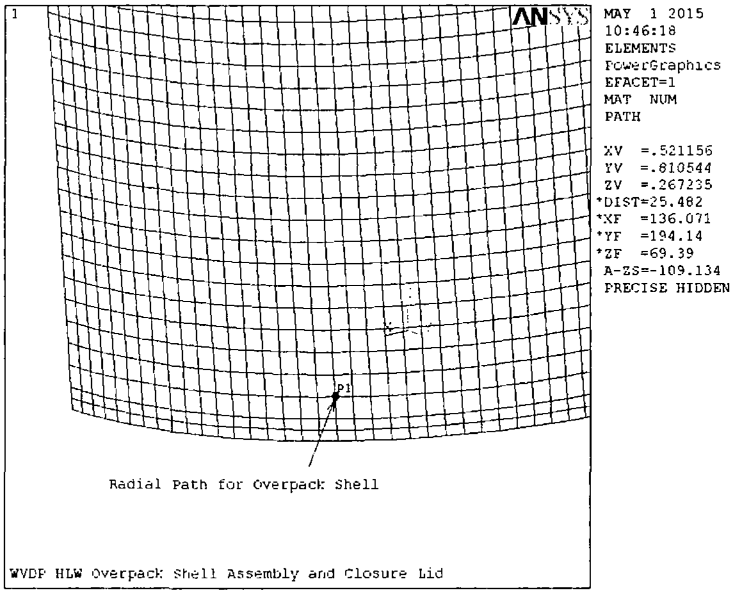
The end drop model contains one element through the thickness of the weld and 70 elements in the circumferential direction which is sufficient, since the stress distribution only varies by a small amount through the weld thickness, and the element density in the circumferential direction is sufficient to track the circumferential stress distribution.

Side Drop Model

The side drop model contains one element through the thickness of the weld and 70 elements in the circumferential direction which is sufficient, since the stress distribution only varies by a small amount through the weld thickness, and the element density in the circumferential direction is sufficient to track the circumferential stress distribution.

b. Stress linearization was not used to evaluate the stresses in the end drop cases. As stated in Section 2.12.6.12.3.1, "Since the peak nodal equivalent stresses calculated were significantly less than the primary membrane stress limit of Sm = 20 ksi for type 304/304L stainless steel at 300 °F, stress linearization was not performed." Note that the peak stress occurs at the center of the lid not at the lid to shell weld location.

For the side drop case, the stress linearization is calculated at the location of maximum equivalent stress in the region outside of the peak stress, which is highly localized. Figure 2.12.6.12-5 has been added to Section 2.12.6.12.2. The figure explicitly shows the path used for stress linearization as shown below.



NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

St-2-4. Regarding Section 2.12.6.12.3.2, “Side Drop Case – 1 Foot Drop,” of the application, provide the following information:

a. sketches depicting the critical section(s) for which the stress linearization post-processing is used for stress evaluation against the allowables; and
b. for the assigned “secondary stress” stress category, justification on why the incurred stress or the internal force components are not part of the global force equilibrium consideration for the resulting secondary stress categorization.

Section 2.12.6.12.3.2 states the following:

> “[T]he evaluation of the side drop case showed a localized peak stress of 51 ksi on the end of the HLW Overpack.”

This statement lacks of details for the model attributes needed for the staff to perform the safety evaluation.

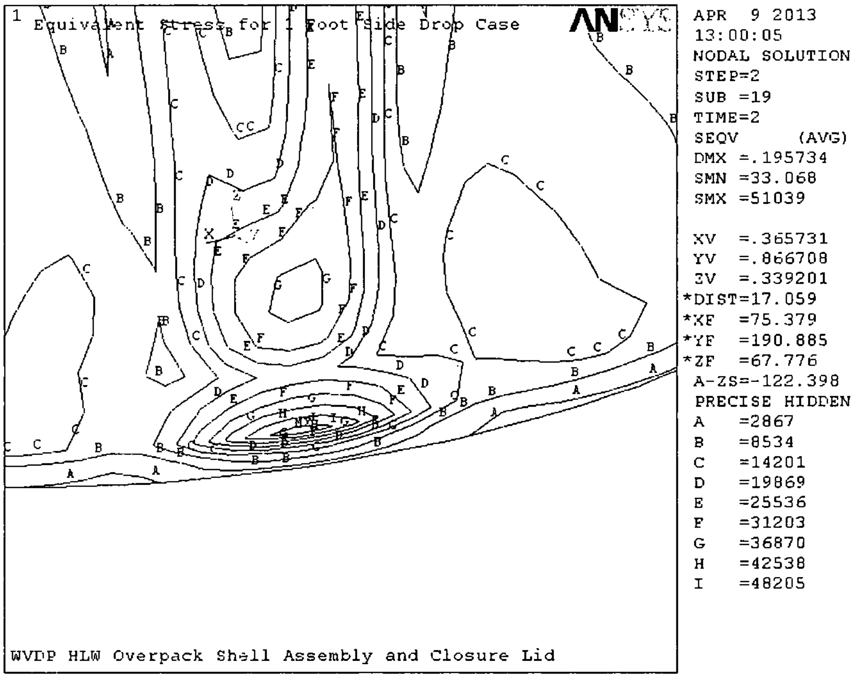
This information is needed to determine compliance with 10 CFR 71.35(a).

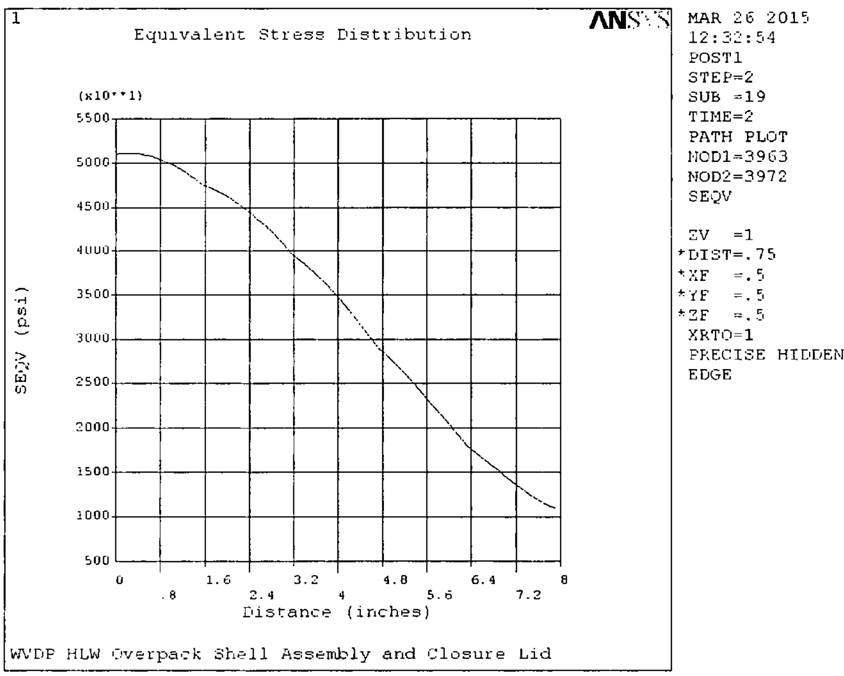
NAC International Response to Structural Evaluation RAI St-2-4:

a. Figure 2.12.6.12-9 has been added to Section 2.12.6.12.2. The figure explicitly shows the path used in the side drop for stress linearization.

b. In accordance with Table 5.6 of the ASME Boiler and Pressure Code, Section VIII, Division 2, the bending stress at the junction of a cylindrical shell and a flat head is classified as a secondary stress.

   The stress reported is a peak stress and is located at the intersection of the Overpack shell and bottom plate. This is a highly localized stress distribution as shown in the figures below. The stress levels in adjacent regions are significantly lower and rapidly drop below the yield strength of 304 stainless steel. Therefore the highly localized stress will redistribute to the adjacent material due to localized yielding of the material. Note that the peak stress level meets the criteria for secondary stress of $3S_m$ which is 60 ksi for SA-240, Type 304 stainless steel.


1
Equivalent Stress for 1 Foot Side Drop Case
ANSYS
APR 9 2013
13:00:05
NODAL SOLUTION
STEP=2
SUB =19
TIME=2
SEQV (AVG)
DMX =.195734
SMN =33.068
SMX =51039
XV =.365731
YV =.866708
ZV =.339201
*DIST=17.059
*XF =75.379
*YF =190.885
*ZF =67.776
A-ZS=-122.398
PRECISE HIDDEN
A =2867
B =8534
C =14201
D =19869
E =25536
F =31203
G =36870
H =42538
I =48205
WVDP HLW Overpack Shell Assembly and Closure Lid


1
Equivalent Stress Distribution
ANSYS
MAR 26 2015
12:32:54
POST1
STEP=2
SUB =19
TIME=2
PATH PLOT
NOD1=3963
NOD2=3972
SEQV
ZV =1
*DIST=.75
*XF =.5
*YF =.5
*ZF =.5
XRTO=1
PRECISE HIDDEN
EDGE
(x10**1)
SEQV (psi)
5500
5000
4500
4000
3500
3000
2500
2000
1500
1000
500
0
.8
1.6
2.4
3.2
4
4.8
5.6
6.4
7.2
8
Distance (inches)
WVDP HLW Overpack Shell Assembly and Closure Lid

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

St-2-5. Considering the STC-HBU design weight of [redacted] with the balsa wood impact limiters configured for a certified NAC-STC design weight of [redacted] reevaluate the package structural performance by

a. addressing potentially higher than originally calculated g-loads for evaluating the package structural performance for applicable drop orientations, and

b. Based on the evaluation requested above, revise:
   i. the appropriate sections of the application (e.g., Section 2.13) for applicable g-loads for evaluating individual package components and contents, including those for the HBU fuel subject to cask free-drop events shown in Section 2.13.6.15; and

[redacted]

This information is needed to determine compliance with 10 CFR 71.35(a), 71.71(c)(7), and 71.73(c)(1).

<u>NAC International Response to Structural Evaluation RAI St-2-5:</u>

SAR Section 2.13.6, the STC-HBU cask weight of 228,050 pounds is an error and has been corrected. As shown in SAR Table 2.13.2-1, the corrected weight of the loaded and ready for transport STC-HBU cask is 242,320 pounds with Balsa impact limiters and 248,620 pounds with Redwood impact limiters. The weight shown for directly loaded fuel with Redwood impact limiter in SAR Table 2.2-1 is 249,520 pounds.

With the Redwood impact limiters, the lower design basis weight (248,620 pounds) of the STC-HBU cask is attributed to the lower contents weight of 54,920 pounds vs. the total fuel basket weight of 55,820 pounds in the licensed design basis weight (250,000 pounds) of the directly loaded fuel cask. The reduction of 900 pounds for the total case weight has an insignificant effect on the performance of the redwood impact limiter.

With the balsa wood impact limiters, the weight shown for the design basis CY-MPC canistered fuel cask is 260,000 lbs. The lower weight of the STC-HBU cask (242,320 pounds) is attributed to the lower contents weight.

The reduction in the contents weight also means that the total kinetic energy which must be absorbed is also reduced. Since crush strength curves for balsa wood and redwood monotonically increase with strain, the resulting strain, and crush strength could be reduced. The bounding condition is the cold condition for which the properties have been evaluated at, -40F. Additionally, for the cold condition, the properties of the balsa wood have been factored by another 10% to account for uncertainties, one of which could be considered to be the reduction of the weight by approximately 10%. If the acceleration of the cask is increased by 8%, the original CY-MPC acceleration of 39.9g increases to 43g. The bounding condition for the stresses developed in the cask are those for the top end drop in which the inertial loading due to the contents weight is applied to the lid. The inertial loading applied to the closure lid for the top end drop is the acceleration times the contents weight. A comparison of the inertial loading to the closure lid for the two configurations is shown below. The inertial loading applied to the closure lid associated with the total cask weight configuration of 260,000 pounds is bounding by more than 13% over the loading associated with the STC-HBU configuration. The comparison of the cask and fuel weight for the CY-MPC canistered fuel cask and HBU fuel is shown in the table below.

Comparison of the Inertial Loading Applied to the Closure Lid for the Top End Drop

	CY-MPC canistered Fuel	STC-HBU Contents
Total contents weight (lb)	67,195	54,920
End Drop Acceleration (g's)	39.9	43
Inertial load applied to the lid (kips)	2,681	2,361

Likewise, for the side drop, this means that the total kinetic energy which must be absorbed is also reduced due to the reduction in the contents weight. While for the end drop, the backed area for the balsa wood (and therefore the cross sectional area for the crushed balsa wood) remains the same, the crush area for the side drop does changes with the crush depth. To evaluate this, the finite element model used for the 260,000 pound design basis weight was altered to 242,320 pounds and the resulting acceleration increased from 48.5 g to 50.7g. The inertial loading applied to the cask shells is determined by the product of the contents weight and the acceleration, which is compared in the table below. The inertial loading applied to the cask shells associated with the total cask weight configuration of 260,000 pounds is bounding by more than 17% over the loading associated with the STC-HBU configuration.

Comparison of the Inertial Loading Applied to the Cask Shells for the Side Drop

	CY-MPC canistered Fuel	STC-WVDP HLW Contents
Total contents weight (lb)	67,195	54,920
Side Drop Acceleration (g's)	48.5	50.7
Inertial load applied to the lid (kips)	3,259	2,784

This confirms that the total inertial loads from the cask contents for the end drop and side drops with lighter weights are bounded by CY-MPC design basis weight of 260,000 pounds. SAR Section 2.13.6.7.4.1 for the STC-HBU evaluation has been revised to confirm that the evaluations associated with the design basis weight of 260,000 pounds bounds the cask body stresses for STC-HBU contents.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

St-2-6. Provide mapping and necessary critical section locator sketches to depict the correlation between component numbers of Table 2.13.6.12.3-1 and section numbers of Table 2.13.6.12.5-1 for evaluating mechanical and thermal stress margins of safety, respectively.

[redacted]

This information is necessary to determine compliance with 10 CFR 71.33(a)(5).

NAC International Response to Structural Evaluation RAI St-2-6:

The component numbers given in Table 2.13.6.12.3-1 and 24 sections used for the STC Cask are described in the table below. The correspondence between the STC section locations and the HBU section numbers is given in the last two columns.

Component Numbers listed in Table 2.13.6.12.3-1 are shown in Figure 2.10.2-33	Component Description	STC Cask Sections listed in Table 2.13.6.12.3-1(1) are shown in Figure 2.10.2-34(2)	HBU Sections listed in Table 2.13.6.12.5-1 are shown in Figure 2.6.7.2-1(2)
1	Bottom Plate	B, D	9, 11
2	Bottom Forging	A, C, E	5, 10, 12
3	Inner Shell Rings	F, P	4, 7
4	Inner Shell	H, J, L, N, P	1, 2, 3
5	Outer Shell	I, K, M, O, Q	6
6	Top Forging	R, S, T	8
7	Inner Lid	W, U	13, 15
8	Outer Lid	X, V	14, 16

Notes: (1) only the maximum stress intensity for each component are listed in Tables 2.13.6.12.3-1 through 2.13.6.12.3-14.
(2) Figures 2.10.2-34 and Figure 2.6.7.2-1 are attached for clarity

Figure 2.10.2-34 ANSYS Finite Element Model - Representative Section Locations

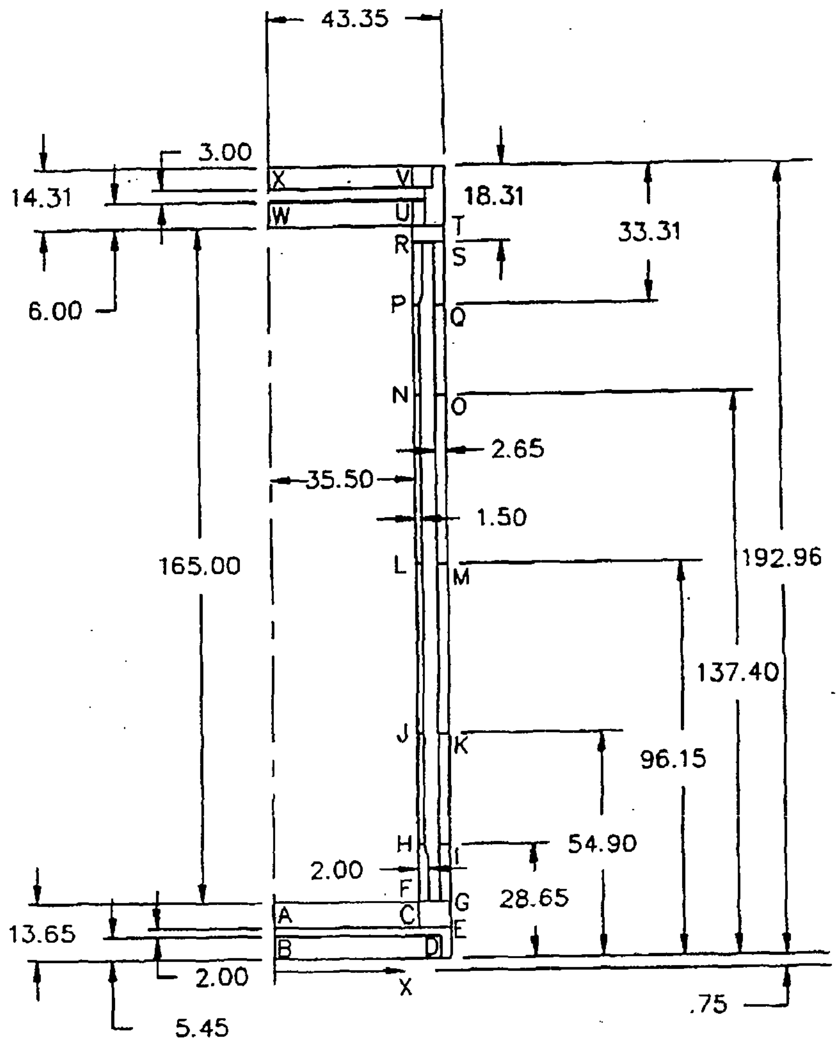
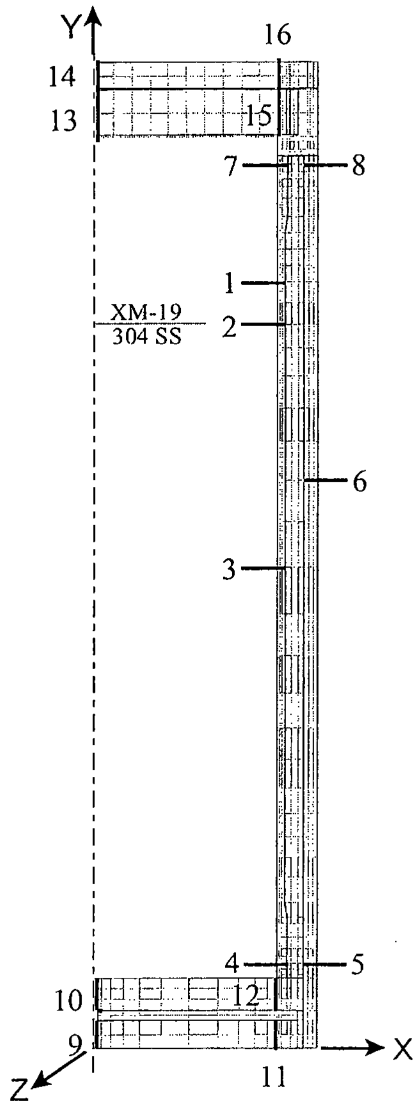


Figure 2.6.7.2-1 Section Locations for Stress Evaluation (Canister Configurations)



NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

St-2-7. Regarding stress margin calculations for the cask body related to HBU fuel transport:

a. provide a synopsis to explain the stress combination approach [redacted]

b. Provide two detailed examples on minimum stress margin calculations for the [redacted]

It is unclear whether the stress combination approach follows those used previously for other NAC-STC configurations. [redacted]

This information is needed to determine compliance with 10 CFR 71.35(a).

<u>NAC International Response to Structural Evaluation RAI St-2-7:</u>

a. The maximum membrane plus bending stress intensity for all of the drop cases is 23.9 ksi for the top corner drop in Table 2.13.6.12.3.3 and occurs in component number 8 (Outer Lid). The maximum stress intensity for the internal pressure case is 2.1 ksi and occurs in component number 7 (Inner Lid). The maximum stress intensity for the thermal cases is 23.9 ksi, and occurs in component number 3 (Inner Shell Rings). These stress intensities are added absolutely, without regard to location in the cask. Therefore, the maximum combined stress reported is:

Maximum combined SI = 23.9 + 2.1 + 23.9 = 49.9 ksi

This is a conservative approach since the combination is without regard to location of the maximum value in the cask. In addition, stress intensities are added directly which produces a higher stress than adding the stress components and recalculating the stress intensity.

b. If the actual locations in the cask are taken into consideration and the stress intensities are added absolutely then the combined stress intensities are:

Component Number	Component Description	Maximum SI ($P_m + P_b$) for Drop Cases(1) (ksi)	Maximum SI ($P_m + P_b$) for Pressure Case(2) (ksi)	Maximum SI ($P_m + P_b$) for Thermal Cases(3) (ksi)	Combined SI(4) (ksi)
1	Bottom Plate	14.5	0.5	5.7	20.7
2	Bottom Forging	15.6	0.9	15.6	40.4
3	Inner Shell Rings	14.8	0.9	23.9	39.6
4	Inner Shell	10.2	1.2	9.0	20.4
5	Outer Shell	6.5	0.1	13.1	19.7
6	Top Forging	17.4	1.8	23.5	42.7
7	Inner Lid	20.9	2.1	1.9	24.9
8	Outer Lid	23.9	1.8	2.1	27.8

Notes: (1) Maximum SI from Tables 2.13.6.12.3-2, 2.13.6.12.3-4, 2.13.6.12.3-6, 2.13.6.12.3-8, 2.13.6.12.3-10, 2.13.6.12.3-12 or 2.13.6.12.3-14
(2) Maximum SI from Table 2.10.4-1
(3) Maximum SI from Tables 2.13.6.12.5-1 or 2.13.6.12.5-2
(4) Combined SI = SI for Drop Cases + SI for Pressure Case + SI for Thermal Cases

As shown in table above, the combined stress for Inner Shell (Component 4) and Top Forging (Component 6) is 20.4 ksi and 42.7 ksi, respectively. Using the same allowable stress of 56.1 ksi, the minimum margin of safety for the Inner shell and Top Forging is +1.75 and +0.31, respectively. Note that the maximum combined stress for the cask components is 42.7 ksi, which is less than the conservatively calculated maximum stress of 49.9 ksi as reported in SAR Section 2.13.6.12.6.

Note that combining stress intensities directly produces higher values of SI than combining stress components and recalculating the stress intensity. As an example, the component stresses for Component 6 are shown below.

Load Case	Reference	S_x (ksi)	S_y (ksi)	S_z (ksi)	S_{xy} (ksi)	S_{yz} (ksi)	S_{zx} (ksi)
Side Drop	Table 2.10.4-71	-2.8	-9.8	7.5	0.5	0.2	1.2
Pressure	Table 2.10.4-1 Section R4	-1.1	0.7	0.2	0.1	0.0	0
Thermal Cold	NAC Calculation 423-2010, Rev. 0	0.8	-10.7	-22.4	-1.1	0.1	1.9
Sum	----	3.1	-19.8	-14.7	-0.5	0.3	3.1

Recalculating the principal stresses and stress intensity using the sum of the stress components gives: S1 = 3.63 ksi, S2 = -15.19 ksi, S3 = -19.84 ksi and SI = 23.47 ksi, which is less than the 42.7 ksi obtained for component 6 from combining stress intensities directly.

In addition, the maximum temperature of component 6 is 256°F. The allowable stress reported in Section 2.13.6.12.6 was conservative based on 400°F. Using 300°F, the allowable stress for Primary + Secondary stress is $3S_m$ = 60 ksi where S_m = 20 ksi for Type 304 SS at 300°F. The allowable stress of 56.1 ksi used in Section 2.13.6.12.6 was conservatively based on 400°F.

Therefore, the procedure discussed in Section 2.13.6.12.6 is conservative.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

St-2-8. Revise the following statements in the application:

Application Reference	Statement	Revise each statement to...
a. Section 2.13.6.13, Accident Condition of Transport	[redacted]	ensure that the appropriate free drop deceleration g-loads are considered in the evaluation
b. Section 2.13.6.14.1, Structural Evaluation of the STC-HBU Basket for Drop Cases.	[redacted]	

The discussion in these sections may need to be clarified, [redacted]

This information is needed to determine compliance with 10 CFR 71.35(a) and 71.73(c)(1).

<u>NAC International Response to Structural Evaluation RAI St-2-8:</u>

The response to RAI St-2-5 shows that when the HBU fuel loaded cask is used with redwood impact limiters, the weight difference of 900 pounds between the HBU cask (248,620 pounds) and the directly loaded fuel cask (249,520 pounds, analyzed cask weight is 250,000 pounds) is less than 0.4% of the analyzed cask weight. The effect on the 30-ft drop deceleration, regardless of drop orientation, is insignificant. When the balsa wood impact limiters are used, the response to St-2-5 confirms that the loading applied to the cask body closure for the end drop and the side drop remain bounded by the loading generated by the loaded configuration transport weight of 260,000 pounds. Section 2.13.6.13 has been revised to provide this information.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

STRUCTURAL EVALUATION

St-2-9. Considering the bounding NAC-HBU package weight of [redacted] revise the fuel rod assessment for the applicable deceleration time history boundary conditions in Section 2.13.6.15.1, "Fuel Rod Assessment for HBU Fuel for 30-ft End Drop," of the application for the two impact limiters alternatives:

a. Balsa Wood impact limiters configured for the package design weight of [redacted] lbs., and
b. Red wood impact limiters configured for the package design weight of [redacted] lbs.

[redacted]

This information is needed to determine compliance with 10 CFR 71.35(a) and 71.73(c)(1).

NAC International Response to Structural Evaluation RAI St-2-9:

With the Red Wood impact limiters, the STC-HBU loaded cask assembly weight is 248,620 pounds, and the weight for the STC cask with the directly loaded fuel in Table 2.2-1 is 249,520 pounds (analyzed cask weight is 250,000 pounds). In determining the deceleration, the total cask transport weight must be used. The difference in total cask weight is attributed to the difference of fuel assembly weights. The fuel assembly weight difference of 900 pounds for the HBU fuel as compared to a total weight of 249,520 pounds for the directly-loaded fuel is considered to have an insignificant effect on the 30-ft drop decelerations.

The minimum NAC-HBU package weight of 228,050 pounds associated with the Balsa wood impact limiters in Section 2.13.6 is an error and has been corrected. The correct minimum weight is 242,320 pounds with the Balsa Wood impact limiters. With Balsa-Wood impact limiters, the peak end drop decelerations for the HBU loaded cask was increased to 43.0 g as discussed in the response to RAI St-2-1.

The fuel response with a peak acceleration of 43.0g was re-evaluated. Results from the 30-ft end drop at weights of 260,000 and 242,320 pounds are summarized in the table below. Section 2.13.6.15.1 has been revised to include the revised results, which show the effect is minimal.

Package Weight (lbs) (Balsa wood Impact Limiters)	Peak Acceleration (g)	Max Stress in Fuel Cladding, (ksi)	Yield Stress (ksi)	Factor of Safety
260,000	39.9	53.7	63.5	1.18
242,320	43.0	56.2	63.5	1.13

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

Th-3-1. Provide the validation of the thermal models used to perform the analysis of the NAC-STC loaded with high burnup fuel. Revise the appropriate sections of the application to incorporate this response.

[redacted]

Cladding exposure to a combination of the following factors may make the cladding prone to experience hydride reorientation:

- temperature,
- burnup,
- rod internal pressure, and
- resultant hoop stress causing the cladding to experience hydride reorientation.

[redacted] Therefore, the applicant needs to provide validation of the thermal models to demonstrate the adequacy of the thermal results.

This information is needed to determine compliance with 10 CFR 71.71(c)(1) and (2), and 71.73 (b) and (c)(4).

NAC International Response to Thermal Evaluation RAI Th-3-1:

[redacted]

" Known as the ductile to brittle transition temperature (DBTT).

[redacted]

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

Th-3-2. Perform thermal analyses for the NAC-STC transportation package loaded with high burnup fuel and provide the calculation packages including:

- realistic maximum and minimum temperatures during normal conditions of transport,
- modeling and application uncertainty of these analyses, and
- temperature variations at any cladding location during a loading and transport event.

Revise Section 3.8 of the application to incorporate this response.

[redacted]

The applicant should perform best estimate calculations in order to obtain realistic maximum and minimum temperatures as well as modeling and application uncertainty of the analysis. NUREG-2152, "Computational Fluid Dynamics Best Practice Guidelines for Dry Cask Applications," March 2013, includes a description of modeling and application uncertainties (and methods) to determine these types of uncertainties.

This information is needed to determine compliance with 10 CFR 71.71(c)(1) and (2).

<u>NAC International Response to Thermal Evaluation RAI Th-3-2:</u>

[redacted]

[redacted]

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

Th-3-3. Clarify and justify [redacted] thermal conductivity of uranium dioxide (UO_2) would degrade thermal conductivity for high burnup fuel. Also, explain how this assumption affects the calculated temperatures. Revise Section 3.8 of the application to incorporate this response.

[redacted] no justification is provided on how this assumption affects the calculated temperatures in order to obtain realistic-calculated high and minimum fuel temperatures.

This information is needed to determine compliance with 10 CFR 71.71(c)(1) and (2), and 71.73 (b) and (c)(4).

NAC International Response to Thermal Evaluation RAI Th-3-3:

[redacted]

[redacted]

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

Th-3-4. Revise all thermal properties for the materials used in the analysis in order to obtain realistic temperatures. Revise Section 3.8 of the application to incorporate this response.

[redacted] Adequate values should be used in order to obtain realistic maximum and minimum temperatures.

This information is needed to determine compliance with 10 CFR 71.71(c)(1) and (2), and 71.73 (b) and (c)(4).

<u>NAC International Response to Thermal Evaluation RAI Th-3-4:</u>

Thermal conductivities used in the thermal models are obtained from established references, which are not biased in either hotter or colder conditions. Conservative (lower) thermal conductivities are used for certain materials, such as the UO_2 and the Neutron Absorbers (MMC), which would result in slightly higher (conservative) temperatures for the steady state hot and cold conditions of transport. As shown by the sensitivity analysis in Section 3.8.4.6 (Cases S1 and S2), these lower thermal conductivities (40% vs 22.5% reduction of thermal conductivities for UO_2 and lowest value vs 20% increase of thermal conductivities of the MMC) have an insignificant effect (< 1°C) on the maximum temperature variation of the fuel cladding for a shipment.

Therefore, the thermal models described in Section 3.8.4.1 are adequate to calculate realistic temperatures as presented in Section 3.8.4.2, including the maximum temperatures for steady state hot and cold transport conditions, as well as the maximum temperature variation for fuel cladding for a fuel shipment.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

Th-3-5. For the thermal model used for analyzing high burnup fuel as an authorized content for the Model No. NAC-STC, explain and justify the following assumptions:

[redacted]

d. Revise Section 3.8 of the application to incorporate this response.

[redacted] The thermal analysis results should be realistic in terms of calculated maximum and minimum temperatures.

This information is needed to determine compliance with 10 CFR 71.71(c)(1) and (2), and 71.73 (b) and (c)(4).

NAC International Response to Thermal Evaluation RAI Th-3-5:

a. [redacted]

b. [redacted]

c. [redacted]

d. The sensitivity analyses discussed in a. and b. above have been incorporated into Section 3.8.4.6.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

Th-3-6. Provide the following information (using the models described in the application to calculate the effective thermal conductivity):

a. grid independent fuel temperatures, and
b. effective thermal conductivities for Item a.

Revise Section 3.8 of the application to incorporate this response.

[redacted]

This information is needed to determine compliance with 10 CFR 71.71(c)(1) and (2), and 71.73 (b) and (c)(4).

<u>NAC International Response to Thermal Evaluation RAI Th-3-6:</u>

A finite element model of a fuel assembly using a finer mesh has been constructed. This model has 133% more elements than the current model shown in Figure 3.8-5. Using the fine-mesh model, thermal conductivity of the fuel assembly is computed. Thermal conductivity of the fuel assembly increased by approximately 10% in the radial direction and 20% in the axial direction. Using the lower fuel conductivities from the current (coarser mesh) model is conservative in determining the maximum temperatures.

To evaluate the effect of fuel conductivities obtained by the finer mesh model on the temperature variation of the fuel cladding, a sensitivity analysis is performed using the thermal models with the higher thermal conductivities for the fuel region (sensitivity analysis Case S5 in Section 3.8.4.6). Analysis results show that the maximum temperature variation remains the same as the temperature variation for the Base Case. Therefore, the mesh used for the fuel model has an insignificant effect on the temperature variation of the fuel cladding for a fuel shipment.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

Th-3-7. Obtain the analysis discretization error for the bounding case [redacted]

Provide all analysis files generated as a result of the GCI calculation.

This information is needed to determine compliance with 10 CFR 71.71(c)(1) and (2).

NAC International Response to Thermal Evaluation RAI Th-3-7:

[redacted]

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

Th-3-8. [redacted]

This information is needed to determine compliance with 10 CFR 71.71(c)(1) and (2), and 71.73(b) and (c)(4).

NAC International Response to Thermal Evaluation RAI Th-3-8:

[redacted]

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

Th-3-9. Explain how the heat transfer coefficient used in the thermal model [redacted]

This information is needed to determine compliance with 10 CFR 71.71(c)(1) and (2).

NAC International Response to Thermal Evaluation RAI Th-3-9:

[redacted]

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

Th-3-10. Provide a detailed calculation, as well as corresponding assumptions, of the NAC-STC loaded with high burnup fuel during fire accident conditions. Revise Section 3.8 of the application to incorporate this response.

[redacted]

This information is needed to determine compliance with 10 CFR 71.73(c)(4).

<u>NAC International Response to Thermal Evaluation RAI Th-3-10:</u>

[redacted]

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

THERMAL EVALUATION

Th-3-11. Provide a detailed calculation, as well as corresponding assumptions, of the NAC-STC-WVDP loaded with HLW contents during fire accident conditions. Revise Section 3.7 of the application to incorporate this response.

Page 3.7.5-1 of the application mentions that the analysis results from the fire accident [redacted]

This information is needed to determine compliance with 10 CFR 71.73(c)(4).

NAC International Response to Thermal Evaluation RAI Th-3-11:

The component temperatures for the fire accident presented in Section 3.7.5 are determined using an identical ΔT method as those presented in Section 3.8.5. Based on the sensitivity analysis as discussed in NAC response to RAI Th-3-10, the ΔT method is conservative and, therefore, the component temperatures reported in Table 3.7-6 for the NAC-STC-WVDP loaded with HLW contents during fire accident conditions are conservative.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

CONTAINMENT EVALUATION

Co-4-1. Revise appropriate portions of the application, including Section 4.1, "Containment Boundary," and Table 4.1-1, "NAC-STC Containment Boundary," to identify the NAC-STC containment condition A or B for the WVDP contents.

There appear to be discrepancies within Section 4.0 of the application regarding the containment boundary. For example:

- Section 4.1 of the application does not establish the containment condition (A or B) for the WVDP contents. Nevertheless, Section 4.7.1, "Containment Boundary – STC – WVDP," of the application states the following:

  "The containment boundary of the NAC-STC, including the containment vessel, containment penetrations, seals and welds, and closure *remains as described in Section 4.1*."

- Section 4.1 of the application states the following:

  "...the components of the containment boundary *are described in Table 4.1-1* as a function of the containment condition and contents."

  However, Table 4.1-1 does not include the WVDP contents under the section "Content Condition."

- Section 4.7, "Containment – STC-WVDP," of the application states that the containment boundary is helium leakage tested to ANSI N14.5-1997 leaktight criteria. Section 7.0 of the application states that metallic O-rings <u>must</u> also be used when loading canistered HLW for transport. Therefore, containment condition B with non-metallic seals is not an option. The staff notes, the MPC-LACBWR spent fuel assemblies were identified under Containment Condition B (i.e., metallic containment boundary seals).

This information is needed to determine compliance with 10 CFR 71.33(a)(4).

<u>NAC International Response to Containment Evaluation RAI Co-4-1:</u>

As a response to Co-1-1, Section 4.7.1 was revised to clarify that the applicable boundary for STC-WVDP is the metallic seal Containment Condition B boundary described in Section 4.1. MPC-WVDP HLW overpacks are added to the Section 4.1 Containment Condition B content list and to Table 4.1-1.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

CONTAINMENT EVALUATION

Co-4-2. Revise the following Sections of the application to ensure that the containment requirements will be maintained for normal conditions of transport and hypothetical accident conditions for the directly loaded intact/undamaged HBU PWR fuel and the WVDP contents:

Section 4.2	"Containment Requirements for Normal Conditions of Transport"
Section 4.3	"Containment Requirements for Hypothetical Accident Conditions"
Section 4.7	"Containment – STC-WVDP"

Sections 4.2 and 4.3 of the application do not provide a clear discussion on how the containment boundary and leaktight conditions (i.e., the containment boundary, seal region, and closure bolts do not undergo any inelastic deformation) will be maintained during normal conditions of transport and hypothetical accident conditions for the directly loaded intact/undamaged HBU PWR fuel. If the new content is bound by a previously evaluated NAC-STC structural analysis, or if analysis in other sections of the application shows that the containment boundary and leaktight conditions will be maintained during normal conditions of transport and hypothetical accident conditions that should be summarized in Sections 4.2 and 4.3 of the application.

Section 4.7 of the application states that the containment boundary is helium leakage tested to ANSI N14.5-1997 leaktight criteria. However, the containment requirements for normal conditions of transport and hypothetical accident conditions were not described. See Sections 4.6.2, "Containment Requirements for Normal Conditions of Transport – STC-LACBWR," and 4.6.3, "Containment Requirements for Hypothetical Accident Conditions – STC-LACBWR," of the application where the containment requirements were addressed for the LACBWR contents. In Section 4.7 of the application, it should be clearly stated and justified that the containment boundary and leaktight conditions (i.e., the containment boundary, seal region, and closure bolts do not undergo any inelastic deformation) will be maintained during normal conditions of transport and hypothetical accident conditions. If the HLW overpack and WVDP contents are bounded by a previously evaluated NAC-STC structural analysis during hypothetical accident conditions, this should be briefly described in Section 4.7 of the application.

This information is needed to determine compliance with 10 CFR 71.51(a)(1) and (2).

NAC International Response to Containment Evaluation RAI Co-4-2:

High Burnup Fuel

The NAC-STC is approved for transport of up to 26 PWR fuel assemblies. From the perspective of the containment boundary, the requested increase in fuel assembly burnup has no impact. As the system is licensed for up to 26 PWR assemblies, the underloaded high burnup configuration is bounded. As there is no effect on containment boundary performance, the SAR sections in question (4.2 and 4.3) were not modified in the initial submittal. Section 4.3, as submitted, contains the statement, "The structural integrity of the cask containment during hypothetical accident conditions is demonstrated in Section 2.7." No equivalent statement has previously been included. To address the RAI, Section 4.2 is revised to add, "The structural integrity of the cask containment during normal conditions of transport is demonstrated in Section 2.6", as the final paragraph in Section 4.2.

MPC-WVDP (Section 4.7)

Section 4.7.1 is revised to state that Section 2.6 and 2.7 demonstrate structural integrity of the cask containment during normal conditions of transport and hypothetical accident conditions, respectively.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

CONTAINMENT EVALUATION

Co-4-3. Revise Section 4.7.2, "HLW Overpack and Cask Pressurization," of the application to include the following information:

a. describe any combustible gas generation for the WVDP contents, and
b. show that during a period of one year it does not exceed 5% (by volume) of the free gas volume in any confined region of the package during normal conditions of transport or hypothetical accident conditions.

Section 4.7.2 of the application describes the gas generation (non-alpha particle) and alpha gas generation. Nevertheless, it does not specifically:

- describe the combustible gas generation for the WVDP contents, or
- show that the combustible gas generation during a period of one year does not exceed 5% (by volume) of the free gas volume in any confined region of the package.

The combustible gas generation for all contents in each of the WVDP HLW canisters (i.e., HLW canister, melter-evacuated canister partially filled with glass, and an HLW debris canister) should be specifically addressed in the appropriate discussion or analysis. The free gas volumes in any confined region of the Model No. NAC-STC, HLW canister overpack (HLW overpack), and WVDP HLW canisters (i.e., HLW canister, melter-evacuated canister partially filled with glass, and an HLW debris canister) should also be specifically addressed in any analysis. No credit should be taken for getters, catalysts, or other recombination devices.

This information is needed to determine compliance with 10 CFR 71.43(d).

NAC International Response to Containment Evaluation RAI Co-4-3:

As stated in Section 4.7.2, there is no significant non-alpha particle gas generation in the HLW canisters (or evacuated canisters or debris canisters). As such, there is no combustible gas generation concern. A statement to this effect is added to the section, under "Gas Generation (Non-alpha particle)", in response to this RAI. The response addresses all three types of canisters. Because there is no significant generation of non-alpha particle gas, there is no possibility of filling confined regions of the package with combustible gas to a concentration which exceeds 5% by volume.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

CONTAINMENT EVALUATION

Co-4-4. Revise Section 4.2.2, "Pressurization of Containment Vessel," of the application to include the following information:

a. describe any combustible gas generation for the directly loaded intact/undamaged HBU PWR fuel, and
b. show that during a period of one year it does not exceed 5% (by volume) of the free gas volume in any confined region of the package during normal conditions of transport or hypothetical accident conditions.

Section 4.2.2 of the application has not specifically described the combustible gas generation for the directly loaded intact/undamaged HBU PWR fuel, or specifically shown that the combustible gas generation (e.g., hydrogen, tritium, etc.) during a period of one year does not exceed 5% (by volume) of the free gas volume in any confined region of the package. No credit should be taken for getters, catalysts, or other recombination devices.

This information is needed to determine compliance with 10 CFR 71.43(d).

<u>NAC International Response to Containment Evaluation RAI Co-4-4:</u>

Section 4.2.2.1 is revised to address combustible gas generation and flammability potential of high burnup fuel.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

CONTAINMENT EVALUATION

Co-4-5. Revise Section 4.7, "Containment – STC-WVDP," of the application to address that any results or statements are bounding for:

a. each of the WVDP HLW canisters and its contents, and
b. any potential combinations of contents within the HLW overpack.

Each of the WVDP HLW canisters and its contents (i.e., HLW canister, melter-evacuated canister partially filled with glass, and an HLW debris canister) should address or it should be shown that bounding results have been provided. This applies to any results (but not be limited to) for:

- pressure calculations,
- temperature calculations,
- combustible gas generation, and
- if the containment boundary is maintained during normal conditions of transport and hypothetical accident conditions.

For example:

- Section 1.2.3.3, "WVDP High Level Waste Canisters," of the application states the following:

  "The EC and HLW debris canisters are bounded by the weight and radioactive content of the HLW canisters evaluated."

  The application should briefly state how this statement applies to the examples above (i.e., for pressure calculations, temperature calculations, etc.).

- Section 4.7.2, "HLW Overpack and Cask Pressurization," of the application also states the following:

  "The HLW contents are borosilicate glass. The material was poured into the HLW canisters at a high temperature and allowed to cool within the canister."

  It is not clear if each of the WVDP contents (i.e., HLW Canister, melter-evacuated canister partially filled with glass, and an HLW debris canister) is completely vitrified, or completely contains boroslicate glass for the above statement to be applicable.

- Section 1.2.3.3 of the application also states the following:

  "... a single HLW Overpack will be loaded with (2) melter-evacuated canisters partially filled with glass and one (1) HLW debris canister."

  Because there are five cells in the HLW overpack, it is not clear from the previous sentence if up to two additional HLW canisters will also be inside that single HLW overpack. It should be clear in the application that the results are bounding for the potential combinations of contents within the HLW overpack. This also applies to (but may not be limited to) any results for pressure calculations, temperature calculations, combustible gas generation, and if the containment boundary is maintained during normal conditions of transport and hypothetical accident conditions.

This information is needed to determine compliance with 10 CFR 71.51(a)(1) and (2).

<u>NAC International Response to Containment Evaluation RAI Co-4-5:</u>

As a response to various containment and general RAI's, a text description of the debris canister has been added clarifying content and eliminating it from additional gas generation concerns. A bounding statement as to source/heat generation is added. A reference to Chapter 2 for structural integrity of the boundary during normal and accident conditions is added. Combustible gas generation is addressed for all contents. The combination of the various canisters is addressed within an HLW overpack.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

CONTAINMENT EVALUATION

Co-4-6. Revise Section 4.7.2 of the application to provide clarification and justification for the following statements:

a. "As described in Sections 2.7.11 and 2.11.7.9, the transportable storage canister does not fail in any of the evaluated transport accident conditions defined in 10 CFR 71.73."
b. "No release from the STC confinement boundary is expected to occur."

It is not clear how Sections 2.7.11 and 2.11.7.9 of the application relate to the WVDP contents inside HLW canisters, which are inside a HLW overpack, not a transportable storage canister.

The statement in part b. has not been justified, since the HLW overpack has not been designed to be a separate leaktight containment and evaluated for hypothetical accident conditions.

This information is needed to determine compliance with 10 CFR 71.51(a)(1) and (2).

<u>NAC International Response to Containment Evaluation RAI Co-4-6:</u>

Both sentences are deleted. The text, as intended, should have referred to the HLW overpack, not a transportable storage canister in part "a." and not the STC in part "b." The NAC-STC is the containment boundary and the overpack is not credited as the boundary. Therefore, the sentences are not necessary.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

CONTAINMENT EVALUATION

Co-4-7. Revise Section 4.7, "Containment – STC-WVDP," of the application to include the following for hypothetical accident conditions in which the HLW overpack has not been maintained:

a. supporting calculations assuming that a HLW overpack has not maintained during hypothetical accident conditions; and
b. the containment boundary of the NAC-STC is maintained.

Section 2.12.1.1, "Discussion," of the application states the following:

> "The HLW Overpack is not designed to be a separate leak tight containment for the HLW canisters since the NAC-STC cask body and closure provides the transport containment boundary. Therefore the HLW overpack is only evaluated for normal conditions of transport."

Calculations in Section 4.7 of the application do not address a HLW overpack that has not been maintained during hypothetical accident conditions. This is necessary to ensure the calculations provided in Section 4.7 are bounding for hypothetical accident conditions.

Section 4.7 does not address if there are any consequences to the NAC-STC containment boundary if the HLW overpack is not maintained during hypothetical accident conditions. This is necessary to ensure the NAC-STC containment boundary is maintained during hypothetical accident conditions.

This information is needed to determine compliance with 10 CFR 71.51(a)(2).

<u>NAC International Response to Containment Evaluation RAI Co-4-7:</u>

As stated in Section 4.7, "The HLW Overpack is not a component of the cask containment system." The HLW overpack is not credited with a containment function during any condition within Section 4.7 discussion for normal or accident conditions. As stated, there is no significant gas release or pressure change. Chapter 2 analysis demonstrates there is no structural failure of the containment boundary.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

CONTAINMENT EVALUATION

Co-4-8. Revise Chapter 3, "Thermal Evaluation," of the application to include the maximum temperatures for each containment boundary O-ring (e.g., inner lid inner O-ring, vent port coverplate inner O-ring, drain port coverplate inner O-ring, etc.) as well as specify the O-ring material in:

Table Number	Table Name
3.7-4	"Maximum Component Temperatures – Normal Transport Conditions, Maximum Decay Heat and Maximum Ambient Temperature – STC-WVDP"
3.7-5	"Maximum Component Temperatures – Normal Transport Conditions, Maximum Decay Heat, Minimum Ambient Temperature – STC-WVDP"
3.7-6	"Maximum Temperature of the HLW and Contents, Basket, and HLW Overpack – Hypothetical Accident Condition Fire Transient"
3.8-4	"Maximum Component Temperatures—Normal Transport Conditions, Maximum Decay Heat, Maximum Ambient Temperature, among Three Configurations – the STC-HBU"
3.8-5	"Maximum Component Temperatures - Normal Transport Conditions, Maximum Decay Heat, Minimum Ambient Temperature, among Three Configurations – the STC-HBU"
3.8-6	"Maximum Temperature of the STC-HBU – Hypothetical Fire Accident Condition"

Table 4.1-1 of the application identifies each O-ring and the material for each O-ring for the NAC-STC containment condition for the WVDP contents. Tables 3.7-4, 3.7-5, 3.7-6, 3.8-4, and 3.8-5 of the application do not specify containment boundary O-ring temperatures or materials. In Table 3.8-6 of the application, it is not clear if the temperatures provided are bounding for each of the containment boundary O-rings included in Table 4.1-1 of the application.

This information is needed to determine compliance with 10 CFR 71.51(a)(1) and (2).

NAC International Response to Containment Evaluation RAI Co-4-8:

Tables 3.7-4, 3.7-5, 3.7-6, 3.8-4, and 3.8-5 have been revised to present the maximum (bounding) temperatures for the Inner Lid O-ring and Port Cover Plate O-rings. The material of the O-ring (Viton) is also noted in the Tables (Metallic for STC_WVDP and Viton for STC-HBU).

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

CONTAINMENT EVALUATION

Co-4-9. Revise Section 4.2.2.1, "Containment Pressurization Due to Directly Loaded Fuel," and Section 4.3, "Containment Requirements for Hypothetical Accident Conditions," of the application to consider the average gas temperature, or show that the values used are bounding for directly loaded intact/undamaged HBU PWR fuel in all calculations.

Section 4.2.2.1 of the application states the following:

a. "The pressure with air in the cask cavity, based on a conservative bulk air temperature of 450°F (Section 3.4.4), is 4.3 atm (63.2 psia = 48.5 psig)."

   The average gas temperature for air as the cover gas was not calculated for directly loaded intact/undamaged HBU PWR fuel during normal conditions of transport, and therefore was not considered in Section 4.2.2.1 of the application. Therefore, the use of 450°F and 4.3 atm has not been shown to be accurate or bounding for the directly loaded intact/undamaged HBU PWR fuel.

b. "Based on a bulk average gas temperature of 401°F when helium is the cover gas, the pressure in the cask cavity is 45.1 psig."

   Table 3.8-4 of the application, "Maximum Component Temperatures - Normal Transport Conditions, Maximum Decay Heat, Maximum Ambient Temperature, among Three Configurations – the STC-HBU," stated that the average gas temperature for helium as the cover gas was 438°F for directly loaded intact/undamaged HBU PWR fuel. Therefore, this condition was not considered in Section 4.2.2.1 of the application.

Section 4.3 of the application states the following:

"For directly loaded fuel, assuming a simultaneous occurrence of a fire accident and a 100% rod failure, and on the basis of a bulk average gas temperature of 675K resulting from air in the cavity, the pressure within the cask cavity is calculated to be 5.72 atm."

The average gas temperature for air as the cover gas was not calculated for directly loaded intact/undamaged HBU PWR fuel during hypothetical accident conditions, and therefore was not considered in Section 4.3 of the application. Therefore, the use of 675K and 5.72 atm has not been shown to be accurate or bounding for the directly loaded intact/undamaged HBU PWR fuel.

This information is needed to determine compliance with 10 CFR 71.43(c).

NAC International Response to Containment Evaluation RAI Co-4-9:

The NAC-STC pressure for a payload of directly loaded HBU PWR fuel is addressed in Section 3.8.6 (including the 438°F normal condition gas temperature). Section 4.2.2.1 is revised to include the conclusion that the existing evaluation is bounding for the HBU PWR payload (including a reference to Section 3.8.6). The text added addresses both normal and hypothetical accident conditions.

In regard to the RAI comments on air as cover gas in the context of HBU PWR transport, the HBU Transport will occur with helium gas in the NAC-STC cavity with a leak tight seal configuration. Air analysis is not a requirement for this transport condition. As the partial pressure calculations of the fill gas rely on temperature changes from load to transport conditions, the gas type is inconsequential to the conclusion stated for HBU PWR transport. The gas temperature is lower for HBU PWR transport than that applied in the bounding normal and accident condition pressure calculations; therefore, the pressures listed are conservative for HBU PWR payload.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

CONTAINMENT EVALUATION

Co-4-10. Revise Section 4.1.3.1.2, "Containment System Verification," of the application to include the information requested in items a. and b. Explain the following when a Viton O-ring is tested to meet the leaktight criterion in ANSI N14.5 (i.e., 1.0×10^{-7} reference cubic centimeter per second) for the directly loaded intact/undamaged HBU PWR fuel:

a. Based on physical testing, explain how the leakage rate test to the leaktight criterion can be performed at the maximum temperature expected during loading before the Viton O-ring saturates with helium, and
b. Explain how a leak tester can accurately and repeatedly differentiate permeation from leakage at the leaktight criterion when using helium as a tracer gas on a repeatable basis for this containment design.

Elastomeric O-rings (e.g., Viton) are highly permeable to helium; therefore, it can be difficult to perform a helium leakage rate test on Viton O-rings to meet the leaktight criterion in ANSI N14.5 because:

- High permeability to helium. Leakage rate testing on Viton to the leaktight criterion may require a short leakage rate test time.
- Permeation of helium through Viton increases with temperature. Performing leakage rate tests to the leaktight criterion on a loaded package would further shorten the leakage rate test time and affect the repeatability and accuracy of the test.
- Differentiate helium permeation from leakage. It is not clear how permeation will be accurately differentiated from leakage when leak testing Viton seals with helium to the leaktight criterion on a repeatable basis for this containment design.

The applicant has not demonstrated that the helium leakage rate test for this containment design can be completed successfully before the Viton seals saturate with helium gas repeatedly and accurately. For example, perform a helium leak test of full size Viton O-rings used on the NAC-STC containment design resulting in a leakage rate test duration that includes an appropriate factor of safety, which considers the maximum temperature expected during loading and meets the leaktight criterion in ANSI N14.5.

This information is needed to determine compliance with 10 CFR 71.51(a)(1), 71.85(a), and 71.87(c).

NAC International Response to Containment Evaluation RAI Co-4-10:

NAC understands the concern with helium permeation through Viton O-rings, and its potential impact on the ability to perform the required leaktight testing, especially at higher temperatures, in the area of the O-rings under test with permeation rates increasing by factors > 200% between 25°C and 200°C.

However, at normal test conditions (approximately 25°C) after annual NAC-STC maintenance where the containment boundary Viton O-rings are replaced and verified to be leaktight in accordance with the requirements of Chapter 8, Section 8.2.2.2 for the periodic leakage rate test, the permeation rate will not affect the test performance. At this temperature, the permeation rate of helium through Viton is approximately 2×10^{-9} cm^3/sec and will not be a significant issue during the short testing period which is conservatively estimated to be approximately one hour for the inner lid leakage test and 30 minutes each for the vent and drain port coverplates.

Concerns regarding performance of the maintenance leakage rate test following Viton O-ring seal replacement during NAC-STC fuel loading operations per 7.1.3.1 due to the potential for higher temperatures of the seals under test have also been addressed. The procedures in Section 7.1.3.1 have been revised to address these concerns by specifying that if the inner lid Viton O-ring seal is replaced due to preshipment leakage rate test failure, or due to identification of excessive wear during the visual inspection of the seals, the NAC-STC cavity water level will be partially drained (approximately 50 gallons) by blowdown with helium gas. The inner lid leakage test would then be performed prior to complete cavity water draining and vacuum drying operations. As the lid has just been removed from the spent fuel pool, the cask heat-up due to the fuel decay heat will not be a significant factor. It is expected that the helium maintenance leakage rate test of the inner lid per Section 8.2.2.2 can be completed with the seal temperatures at approximately 35°C, corresponding to a conservative spent fuel pool water temperature. At this temperature, as with the annual periodic leakage test, the permeation of helium will not be significant enough to preclude obtaining acceptable helium leakage test results.

As the conservative helium leakage rate is determined following completion of fabrication, and annually thereafter by performance of the periodic and maintenance leakage rate tests, the higher levels of helium permeation will not change the actual helium leakage rates.

A full scale test of the NAC-STC containment boundary fitted with Viton O-rings will be performed on the first cask fabricated, in accordance with the acceptance test program leakage rate testing defined in Chapter 8, Section 8.1.3.2.2. If an acceptable helium leakage rate test cannot be performed and completed satisfactorily, then the cause of the leakage will be determined. If the issue cannot be satisfactorily resolved, the cask unit will not be accepted for transport operations without further modification, and the cask identification plate will not be installed in accordance with 10 CFR 71.85(c).

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

SHIELDING EVALUATION

Sh-5-1. Explain and clarify why the dose rate in Section 5.7.1-3, and Figure 5.7.1-2 (Location of STC-WVDP Maximum Dose Rates for Hypothetical Accident Conditions) is different from Table 5.7.1-2.

Application's Reference (Hypothetical accident conditions)	Radial dose mrem/hr
Table 5.7.1-2	1.7
Figure 5.7.1-2	2.84

The application should be revised since the actual day of the first loading is no longer April 1, 2014. [7]

This information is needed to determine compliance with 10 CFR 71.51(a) through (c).

NAC International Response to Shielding Evaluation RAI Sh-5-1:

The 1.7 mRem/hr dose rate is retrieved from a surface tally (or detector) that is averaged circumferentially. The 2.84 mRem/hr dose rate is retrieved from a surface tally that is azimuthally split to capture lead slump accident conditions. The 2.84 mRem/hr dose rate is the maximum on the radial surface. Therefore, Table 5.7.1-2 is revised with the correct dose rate.

A shorter decay period applied in the source term evaluation is conservative. No calculation revisions are made for an expected decay period greater than 90 days. Text is added to Section 5.7 to clarify that this evaluation is bounding of later transport dates.

[7] The applicant calculated the maximum activity 90 days post the January 1, 2014 (April 2014) in order to capture the short lived decay products expected at the first loading.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

SHIELDING EVALUATION

Sh-5-2. For different assembly configurations depicted in Drawing No. 423-800, explain why:

a. surface dose rates for packages containing various assembly numbers (i.e., 14, 16, and 20) are identical to each other.

Assembly number	Heat load (kW/assembly)
14	1.7
16	1.5
20	1.2

b. the azimuthal dose rate variation is not considered for finding the location of the maximum dose rates (as it was performed for the WVDP vitrified HLW).

Since these loading patterns are very different from each other, it is not clear why the surface dose rates for all three packages are identical. The staff expects some variations in the surface dose rates for these loading patterns. Nevertheless, the applicant should explain the basis of the calculation for estimating the surface dose rates of the package.

This information is needed to determine compliance with 10 CFR 71.47.

<u>NAC International Response to Shielding Evaluation RAI Sh-5-2:</u>

Loading tables are generated for each loading configuration (i.e., 14, 16 or 20 assemblies) based on meeting the thermal heat load limit for each assembly and a maximum surface dose rate of 200 mRem/hr (a conservative limit imposed by NAC). For a specific load configuration at a burnup and initial enrichment combination, the limiting condition may be either heat load or dose rate. In particular, low enriched, high burnup points in the load table are often dose limited (i.e., heat load at the load table state point below the thermal limit). All three configurations, therefore, contain maximum surface dose rates that are approximately 200 mRem/hr. The results are not identical, although some might appear to be due to the indicated precision of the results. Differences in profiles can be seen in the provided plots. Details of the loading table generation are provided in Section 5.8.5.

Only cell based mesh tallies (or detectors) that include more meshing than is reasonable with surface tallies are used in the STC-HBU evaluations (Section 5.8). The meshing is either cylindrical with azimuthal and axial splitting for cask surfaces, or rectangular with x-y splitting for the 2 meter from convey vehicle surface dose rates. Section 5.8.3.4 contains the discussion of the detector mesh used including the azimuthal (theta) splitting applied. Figures provided (ex., Figure 5.8.7-1) include angle of revolution (azimuthal) distribution of dose rate plots.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

Cr-6-1. Explain how all the requirements in 71.15(c) are met for the proposed WVDP vitrified HLW contents.

The evaluation in Section 6.9.1 of the application only describes how the WVDP vitrified HLW, as authorized contents, will meet the requirements of 10 CFR 71.15(c)(1)(i) and (ii) in order to be considered exempt from fissile material classification. However, all the requirements of 10 CFR 71.15(c) must be met, which include 10 CFR 71.15(c)(2). The evaluation in Section 6.9.1 of the application should clearly describe how the vitrified HLW, as authorized contents, will meet all the requirements of 10 CFR 71.15(c).

This information is needed to determine compliance with 10 CFR 71.15.

<u>NAC International Response to Criticality Evaluation RAI Cr-6-1:</u>

Lead, beryllium, graphite, and hydrogenous material enriched in deuterium are not present in the HLW glass composition. Text is added to Section 6.9.1 to address 10 CFR 71.15(c)(2).

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

CRITICALITY EVALUATION

Cr-6-2. Provide the following information regarding the analyses described in Section 6.4.2.6 of the application:

a. the material properties used for the absorber plates, and
b. the fuel tube dimensions for the three cases using the new absorber thickness listed in Table 6.4.2-4 .

The material properties for the absorber plates should be consistent with the proposed minimum effective areal density for ^{10}B (Boron-10). Moreover, it is not clear if the fuel tube dimension remained constant for each case.

This information is needed to determine compliance with 10 CFR 71.55 and 71.59.

NAC International Response to Criticality Evaluation RAI Cr-6-2:

a. A new table, Table 6.4.2-5, is added to the SAR and contains the sheet density, ^{10}B weight fraction, thickness and resulting ^{10}B areal density. This confirms that all cases apply the minimum 0.015 ^{10}B g/cm^2 areal density.

b. The fuel tube wall dimension for the new cases is listed in Section 6.4.2.6. The absorber thickness values are listed in Table 6.4.2.4. To clarify that the absorber thickness tolerance cases retained the fixed tube wall dimensions, the following statement is added to Section 6.4.2.6: "The tube wall dimensions applied are constant for nominal (0.100 inch) and absorber thickness tolerance cases."

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

OPERATING PROCEDURES EVALUATION

Op-7-1. Revise Section 7.3.2.1, "Preparation for Unloading the NAC-STC Cask (Directly Loaded Fuel Configuration)," (step No. 4) to briefly explain the action(s) to be taken if the analysis of the gas sample indicates possible fuel failure during transport.

In general, step No. 4 in Section 7.3.2.1 of the application describes the steps associated with measurements and analysis of the cavity of the Model No. NAC-STC and the actions to be taken to bring the pressure in the cavity to atmospheric pressure. Nevertheless, the applicant does not describe the action(s) to be taken if fuel failure during transport is determined to be possible (based on the analysis of the gas in the Model No. NAC-STC cavity).

Also, in the event that fuel failure was determined after shipment, the NRC would consider this to be reportable by the licensee in accordance with the requirements of 10 CFR 71.95. This section of the application should include steps to be taken by the licensee such as reporting and determination of corrective actions.

This information is needed to determine compliance with 10 CFR 71.89.

<u>NAC International Response to Operating Procedures Evaluation RAI Op-7-1:</u>

Section 7.3.2.1 has been revised as follows:

f. "If gaseous radioactivity levels in the cavity gas sample indicate that fuel rod cladding failure may have occurred during the transport operation, the Licensee, shall prepare and submit a written report to the USNRC within 60 days in accordance with 10 CFR 71.95 with a copy of the report provided to NAC as Certificate of Compliance Holder. The reports purpose would be to identify a potential non-compliance with the Certificate of Compliance authorized fuel contents, as transport of damaged fuel and/or fuel having failed cladding conditions are not authorized. The report shall include the details specified in 71.95(c) including an assessment of the safety consequences and implications of the event; and a description of any corrective actions planned or taken as a result of the event, including the means employed to repair any defects and actions taken to reduce the probability of similar events occurring in the future."
g. "Connect a venting system to the pressure gauge/gas sampling fixture and discharge to the facility's off-gas system or to HEPA filter system to bring cavity gas pressure to atmospheric pressure, after determining total gaseous radioactivity of cavity gas and verifying that the release of the gaseous radioactivity through the facilities off-gas system will not violate license conditions."

In addition, Section 7.3.3.1 has been revised by the addition of the following note after Step 1:

> "**Note:** If high levels of gaseous radioactivity were measured during initial cavity pressure measurements indicating the potential for failed fuel rod cladding, the unloaded fuel assemblies may require additional inspections of the fuel based on the corrective actions identified in Section 7.3.2.1, Step 4.f."

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

OPERATING PROCEDURES EVALUATION

Co-7-1. Revise Section 7.6 of the application to ensure the operating procedures are focused on preparing the package for transportation and clarify that storage operations are not included in the transportation approval.

In Section 7.6.3 of the application, the current steps 7 and 8 are focused on the operational steps for storage. The NRC does not authorize the movement of spent fuel or GTCC into storage through transportation packaging review. The steps necessary to prepare the components that will be contents in the NAC-STC for transportation should be included, but where there is overlap with storage, clarification should be provided.

This information is needed to determine compliance with 10 CFR 71.43.

<u>NAC International Response to Operating Procedures Evaluation RAI Co-7-1:</u>

Section 7.6.3, Steps 7 and 8 have been revised to read as follows:

"7. The HLW overpack is now prepared for transportation."
"8. Install the VSC lid and move the loaded VSC to the facility for the transfer of the HLW Overpack into the NAC-STC in accordance with the procedures presented in Section 7.1, contingent upon the loaded contents meeting the requirements for the authorized content conditions, as specified in the NAC-STC Certificate of Compliance."

Reference to an Interim Storage Facility has been removed, but the following note has been added:

"Note: The HLW Overpack may be stored at the loading facility in a VSC prior to off-site transport under a separate USDOE authorization."

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

OPERATING PROCEDURES EVALUATION

Co-7-2. Revise Sections 7.4.2, "Leak Testing for Transport After Long-Term Storage," and 7.4.3, "Leak Testing for Transport without Interim Storage," of the application to address the leakage rate testing of the canistered HLW.

The first sentence of each of these Sections of the application does not address the leakage rate testing of the canistered HLW.

This information is needed to determine compliance with 10 CFR 71.51(a)(1) and (2).

<u>NAC International Response to Operating Procedures Evaluation RAI Co-7-2:</u>

Section 7.4.2 has been clarified to state that canister GTCC waste and HLW Overpacks are never stored long-term in the NAC-STC cask, and therefore the section appropriately refers to leak testing after the long-term storage of PWR fuel only. Section 7.4.2 has been revised as follows to clarify the applicability of leakage testing following long-term storage:

"This section summarizes the leak test method used to demonstrate continued containment of PWR spent fuel prior to transport following an extended period of storage in the NAC-STC cask. The containment boundary for this transport condition is defined as Containment Condition A in Section 4.1 and requires the use of metallic O-rings in the containment boundary. In addition to the steel inner lid and port coverplates, the containment boundary is specified as the outer O-rings of the inner lid and of the vent and drain port coverplates and the O-rings of the test port plugs. As specified in the generic loading procedure, the outer lid must be removed to test the inner lid and the vent and drain port coverplates prior to transport. Note that HBU spent fuel assemblies, canistered spent fuel, GTCC Waste canisters, or HLW Overpacks are not loaded into NAC-STC casks for long-term storage, and therefore, Containment Condition A does not apply for these canistered contents."

The first sentence of Section 7.4.3 has been revised to read as follows to incorporate HBU spent fuel assemblies, and HLW Overpack contents into the contents requiring Containment Condition B leak tests following loading without interim storage in the NAC-STC cask:

"This section summarizes the leak tests required to demonstrate containment of directly loaded standard PWR and HBU spent fuel assemblies without interim storage, or for sealed transportable storage canisters containing spent fuel or GTCC waste, or HLW Overpacks containing separate sealed canisters of glassified high level waste."

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

OPERATING PROCEDURES EVALUATION

Co-7-3. Revise Chapter 4, "Containment," and Chapter 7, "Operating Procedures," of the application to address leakage rate testing of the containment boundary and Viton O-rings to the leaktight criterion for the directly loaded intact/undamaged HBU PWR fuel.

The following Sections and Tables of the application need to be modified to ensure the Viton O-rings are tested to the leaktight criterion when used with HBU PWR fuel:

Table 4.1-1	"NAC-STC Containment Boundaries" (Referenced in Chapter 7 of the application)
Section 4.2	"Containment Requirements for Normal Conditions of Transport"
Section 7.1.3.1	"Direct Loading of Fuel (Uncanistered)" step No. 19) c.; step No.21) b. [note (1)]; step No. 21) c.; and step No. 23 [note (1)]
Section 7.4.1	"Containment System Verification Leak Test Procedures"
Section 7.4.3	"Leak Testing for Transport After Loading without Interim Storage"

Section 7.1.3.1 and Section 7.4.1 of the application should be modified to address the leaktight criterion of the Viton seals used with HBU PWR fuel. Section 7.4.3 of the application should be clarified to address the leaktight criterion is for HBU PWR fuel in association with the Viton O-rings. Table 4.1-1 is referenced in Chapter 7 of the application and should be modified to clearly address that the leaktight criterion for the Viton O-rings is required for HBU PWR fuel. Although it was stated in Section 4.2 of the application that a leaktight boundary is required for transport of high burnup PWR fuel, these Sections of the application described above were not modified to ensure the Viton O-rings are tested to the leaktight criterion when used with HBU PWR fuel.

This information is needed to determine compliance with 10 CFR 71.51(a).

<u>NAC International Response to Operating Procedures Evaluation RAI Co-7-3:</u>

Chapters 4 and 7 have been reviewed and revised to incorporate the leaktight criterion for the Viton O-ring seals required for the leaktight condition for the transport of HBU PWR fuel assemblies. Appropriate procedures, tables and test descriptions have been revised to implement the requested changes and a description of the testing required for the HLW overpacks.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

OPERATING PROCEDURES EVALUATION

Co-7-4. Provide Section 7.4, "Leak Test Requirements," and Figure 7.3-1, "Cask Cooldown Piping and Controls Schematic," of the application.

Comparison of Revision 14A and 14B to the NAC-STC Amendment Request.	
Revision 14A	Revision 14B
Page 7-i, states that Section 7.4, "Leak Test Requirements," starts on page 7.4-1 of the application.	Page 7.4-1, includes a small portion of Section 7.4.2, "Leak Testing for Transport After Long-Term Storage," and the beginning of Section 7.4.3, "Leak Testing for Transport After Loading without Interim Storage."
Figure 7.3-1 was not provided.	Figure 7.3-1 was not provided.

Note: The applicant provided Figure 7.3-1 in the consolidated Safety Analysis Report, Revision 17. Nevertheless, Figure 7.3-1 has not been provided in subsequent revisions to the SAR for the Model No. NAC-STC.

The staff needs to ensure that Sections 7.4, "Leak Test Requirements," 7.4.1, "Containment System Verification Leak Test Procedures," and 7.4.2, "Leak Testing for Transport After Long-Term Storage," have not been removed from the application. Because Figure 7.3-1 of the application was not provided in the most recent revisions to the NAC-STC safety analysis report, and it is adjacent to Section 7.4 of the application, it is not clear if Figure 7.3-1 of the application is still part of the application. The staff also needs to ensure Figure 7.3-1 has not been removed from the application.

This information is needed to determine compliance with 10 CFR 71.51(a).

<u>NAC International Response to Operating Procedures Evaluation RAI Co-7-4:</u>

Figure 7.3-1 and SAR Sections 7.4, 7.4.1, and 7.4.2 were unchanged in the original application. As Sections 7.4, 7.4.1, 7.4.2, 7.4.3 and 7.4.4 have now been revised in response to the current RAIs, these sections are included in the revised SAR pages submitted with the RAI responses. Figure 7.3-1 was not revised so this page was not submitted with the application as this page was not a changed SAR page. As noted, the consolidated SAR, Revision 17, contains this information, as it is the latest approved SAR revision. Upon approval of the application, the SAR changed pages will be consolidated and incorporated into SAR Revision 18.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

OPERATING PROCEDURES EVALUATION

Cr-7-1. Make the following changes to ensure consistency of package operations descriptions with respect to the vitrified HLW overpack and to reference the correct regulatory requirements:

a. Section 7.1.2	Add the HLW overpack to the list of contents to which the dry loading option applies.
b. Section 7.3.4, Step 6	Confirm the regulatory reference discussing package labeling. The current test refers to 49 CFR 172.200; however, 49 CFR 172.400 appears to be the relevant regulation for this step.
c. Section 7.3.3.1, Step 3	Modify this step to address the scenario where the next shipment is high burnup fuel but with different configuration requirements (i.e., different number and locations of thermal shunts)

The staff needs to confirm that the package operations are consistent with the package's design basis and complies with the applicable regulatory requirements.

This information is needed to determine compliance with 10 CFR 71.87(a) and (f), and 71.89.

NAC International Response to Operating Procedures Evaluation RAI Cr-7-1:

SAR Section 7.1.2 has been revised to include the HLW overpack. SAR Section 7.3.4, Step 6, has been revised to identify the correct regulation, which is 49 CFR 172.400. SAR Section 7.3.3.1, Step 3, has been revised to include the scenario where the next loading is for HBU fuel assemblies but with a different shielded thermal shunt configuration.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

OPERATING PROCEDURES EVALUATION

Cr-7-2. Include a step in the procedures in Section 7.1.3.2 of the application (beginning on page 7.1-11) to confirm that:

a. the condition of the spent fuel canister, greater than class C (GTCC) canister, or the HLW overpack and the canister/overpack's internals is such that these:
   i. meet the design and CoC requirements of the NAC-STC package,
   ii. account for the effects of any events that these may experience during storage (prior to loading in the NAC-STC package), and

b. the vitrified HLW overpack meets the limits in 10 CFR 71.15 for classifying the contents as fissile exempt.

The description in Section 7.1.3.2 in the application (Package Operations) assumes the canister/overpack and its contents conform to the NAC-STC's design basis and so meet the conditions of the certificate. The package operations should include a step to confirm that the canister/overpack and its contents to be loaded into the package do indeed meet the package's certificate conditions.

Further, the descriptions at the beginning of Section 7.1.3.2 should also state that the GTCC waste and HLW overpacks will be evaluated for the effects of any events (e.g., accidents and natural phenomena) experienced prior to loading. The effects of these events could impact the ability of these contents to meet the transportation package's certificate conditions.

This information is needed to determine compliance with 10 CFR 71.87(b).

<u>NAC International Response to Operating Procedures Evaluation RAI Cr-7-2:</u>

Chapter 7, Section 7.1.3.2 has been revised to incorporate the appropriate steps in the procedure to ensure that the evaluations identified in the introduction of 7.1.3.2 have been appropriately completed prior to loading and transfer of the canister or overpack into the NAC-STC cavity. A note has also been added to require an evaluation to determine that the HLW overpack meets the limits of 10 CFR 71.15 for classification of HLW contents as fissile exempt.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Co-8-1. Revise Section 8.1.10 of the application to remove the following sentence:

> "The overpack serves as the confinement boundary component of the MPC-WVDP System during storage of HLW canister contents in the vertical concrete cask."

In addition, clarify the use of the phrase "confinement boundary" as used in this amendment request for the NAC-STC.

Per the request for revising the CoC Model No. NAC-STC, the applicant is seeking approval for transporting the WVDP waste as authorized contents. Moreover, the staff did not find in the application information stating that the NRC or DOE has approved the use of the HLW Overpack as a confinement boundary. Since the NRC is reviewing this amendment application for transportation use only, the use of the phrase "confinement boundary" should be clarified and addressed in the context of this amendment request.

This information is needed to determine compliance with 10 CFR 71.87(f).

NAC International Response to Acceptance Tests and Maintenance Program RAI Co-8-1:

The HLW Overpack does not serve as a "confinement boundary" during transport. Therefore, Section 8.1.10 is revised to delete the sentence:

> "The overpack serves as the confinement boundary component of the MPC-WVDP System during storage of HLW canister contents in the vertical concrete cask."

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Co-8-2. Revise Chapter 8, "Acceptance Tests and Maintenance Program," of the application to address leakage rate testing of the containment boundary and Viton O-rings to the leaktight criterion for the directly loaded intact/undamaged HBU PWR fuel.

The following Sections of the application should be revised:

- Section 8.1.3, "Leakage Tests"
  - o References a non-leaktight value for the containment criterion, and does not specifically address the leaktight containment criterion for HBU fuel.
  - o States that Viton seals to the non-leaktight criterion can be used under containment condition A. Nevertheless, Table 4.1-1 of the application states that only metallic seals can be used for containment condition A.

- Section 8.1.3.2.2, "Viton O-Ring Testing Acceptance Criteria" – Does not address the leaktight containment criterion for the containment boundary Viton O-rings for HBU fuel.

- Section 8.2.2.3, "Acceptance Criteria" – This section should clearly state the leaktight containment criterion for the containment boundary Viton O-rings for HBU fuel, rather than include the use of the phrase, "Depending on the payload..."

This information is needed to determine compliance with 10 CFR 71.51(a).

<u>NAC International Response to Acceptance Tests and Maintenance Program RAI Co-8-2:</u>

Chapter 8 Sections 8.1.3, 8.1.3.2.2 and 8.2.2.3 have been revised to specifically address the required leaktight criterion of the STC containment boundary provided by Viton O-ring seals for the transport of HBU PWR fuel assemblies, and to define the applicable test method and frequency.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Co-8-3. Revise Sections 7.4, 8.1.3, and 8.2.2 of the application for the WVDP content and the directly loaded intact/undamaged HBU PWR fuel to clarify that written procedures are developed and approved by personnel certified by the American Society of Nondestructive Testing (ASNT) as a Level III examiner for leakage testing.

Section 7.4 of the application provides the initial description of the pre-shipment leakage rate test and also considers if the seals are replaced (maintenance). Subsections include further description of this test. Section 7.4 of the application states that detailed procedures are developed for use at the licensee's facilities (as does Section 7.4.1 of the application).

Sections 8.1.3 and 8.2.2 of the application provide the initial description of the fabrication and periodic leakage rate tests, with further description in the subsections. Sections 8.1.3 and 8.2.2 of the application state that leakage testing of the NAC-STC shall be performed using approved written procedures (as does Section 8.2.2.2 of the application).

Sections 7.4, 8.1.3, and 8.2.2 of the application do not specify that the written procedures are developed and approved by personnel certified by the ASNT as a Level III examiner for leakage testing as indicated by industry standards.

The ANSI/ASNT CP-189-2006, "Standard for Qualification and Certification of Nondestructive Testing Personnel," provides the minimum training, education, and experience requirements for nondestructive testing personnel. This ANSI standard states that a nondestructive testing personnel Level III examiner has the qualifications to develop and approve written instructions for conducting the leak testing.

This information is needed to determine compliance with 10 CFR 71.37, 71.87, and 71.119.

<u>NAC International Response to Acceptance Tests and Maintenance Program RAI Co-8-3:</u>

Chapter 7, Sections 7.4 and Chapter 8, Sections 8.1.3 and 8.2.2 have been revised to incorporate the requirement that leakage testing procedures are required to be developed and approved by a qualified Level III examiner and that personnel performing the leakage testing shall be qualified in accordance with ANSI/ASNT CP-189-2006, "Standard for Qualification and Certification of Nondestructive Testing Personnel."

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

M-8-1. [redacted]

This information is needed to determine compliance with 10 CFR 71.33(a)(5)(ii).

NAC International Response to Acceptance Tests and Maintenance Program RAI M-8-1:

a) Table 8.1-1 is revised to add TALBOR as a material with 75% credit in the same row as Boral.

b) Within the NAC-STC application, TALBOR, while being a Metal Matrix Composite, is treated as a 75% credit tested material. For qualification and testing it is treated the same as Boral. This is the previously licensed configuration of TALBOR.

c) Drawings 423-875 and 423-878 are revised to add TALBOR to the materials required to be at 0.02 g/cm^2.

As noted by the reviewer, TALBOR is a metal matrix composite. But, for the purpose of this application it is treated as a legacy material tested to 75% credit and is, therefore, excluded from the Section 8.1.11 material set.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

M-8-2. [redacted]

This information is needed to determine compliance with 10 CFR 71.33(a)(5)(ii).

<u>NAC International Response to Acceptance Tests and Maintenance Program RAI M-8-2:</u>

Page 8.1-36 has been revised to state the correct units. The correct unit is areal density on both a. and b., and is therefore g/cm^2.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

M-8-3. Define the phrase "similar material" as used in the application. [redacted] Revise the application to incorporate the response to this question.

On page 8.1-37, the applicant uses the phrase "same, or similar, materials..." and it is not clear how the applicant uses this phrase and the specific material properties used to determine that materials are "similar."

This information is needed to determine compliance with 10 CFR 71.33(a)(5)(ii), 71.105(c)(5), and 71.115(b).

NAC International Response to Acceptance Tests and Maintenance Program RAI M-8-3:

Clarification to the statements is presented in the first bullet under the statement. Materials produced by a different supplier (such as Metamic and TALBOR) must have qualification testing performed, as must materials not previously qualified or materials having changes in key process controls (e.g., difference in manufacturing process).

As the qualification section outlines, all key characteristics are tested for, and NAC believes no further clarification is required.

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR ADDITIONAL INFORMATION

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

M-8-4. [redacted]

This information is needed to determine compliance with 10 CFR 71.33(a)(5)(ii).

[8] See Section 8.1.11, "Alternative Neutron Absorber/Poison Tests for NAC-STC Directly Loaded Basket," of the application.
[9] For example, referral to a nonconformance program, increase extent of inspection, etc.

NAC International Response to Acceptance Tests and Maintenance Program RAI M-8-4:

a. The first paragraph of Section 8.1.7 contains the following text: "For the NAC-STC directly loaded basket, a generic neutron absorber test and qualification program suitable for BORAL, metal matrix composite (aluminum based) and borated aluminum is listed in Section 8.1.11." The section is applicable to the directly loaded basket. No mention of the other system configurations is made in the context of Section 8.1.11.

b. The fifth bullet contains the following text: "The sampling plan shall require that each of the first 50 sheets of neutron absorber material from a lot, or a coupon taken therefrom, be tested. Thereafter, coupons shall be taken from 10 randomly selected sheets from each set of 50 sheets." A testing requirement is included in this statement. To clarify the need for testing all coupons the text in this bullet is modified to state, "The sampling plan shall require that each of the first 50 sheets of neutron absorber material from a lot, or a coupon taken therefrom, be tested. Thereafter, coupons shall be taken from 10 randomly selected sheets from each set of 50 sheets. All coupons (100%) taken shall be tested by neutron attenuation." Text of the bullet has also been modified to clarify that inspection in this context is neutron attenuation testing: "... reduced inspection (neutron attenuation testing) is defined as nonconforming, along with other contiguous sheets, and mandates a return to 100% inspection (neutron attenuation testing) for the next 50 sheets."

c. Volume density (g/cm^3) is used in the subject sentence as compared to areal density (g/cm^2), which is used in other sections. The unit is added to the first occurrence of volume density in this section.

d. Page 8.1-35 is revised to require the designation of nonconforming to the set of sheets involved and requires a return to 100% inspection (testing).

e. Neutron absorption capability of Boral is discussed in Section 8.1.11.9. No neutron attenuation testing is required for Boral's 75% credit.

f. The specific approaches for uncertainties and tolerances are test location and equipment related and are typically documented within the neutron attenuation test plan and report. The amount of detail provided in the SAR has been the subject of substantial interaction between NAC and NRC staff. The text, as written, was deemed sufficient by NRC staff for inclusion in the MAGNATOR CoC/Technical Specifications (Certificate 1031) to provide assurance of criticality safety of a storage/transport system. Material testing requirements were directly copied to minimize review.

NAC Calculations and Supporting Documents
Withheld In Their Entirety per 10 CFR 2.390

Enclosure 2

List of Changes

NAC-STC SAR, Revision 15A

List of Changes, NAC-STC SAR, Revision 15A

Note: The List of Effective Pages, Table of Contents, List of Drawings and List of Figures were revised as needed to incorporate the following changes.

Chapter 1

- Page 1-2, modified the text for "Contents (Payload)" terminology in Table 1-1.
- Page 1-3, text flow changes to Table 1-1.
- Page 1-11, added new terminology and definition, "HLW Debris Canister" to Table 1-1.
- Page 1-12, text flow changes to Table 1-1.
- Page 1.1-7, moved the last paragraph on the page up from page 1.1-8, which is an unmarked text flow change.
- Pages 1.1-8 thru 1.1-16, modified and added text throughout Section 1.1.1.
- Pages 1.1-17 thru 1.1-32, previous Section 1.1.2 was moved to Section 1.1.2.1 and modified text throughout.
- Page 1.1-33, references 5 through 7 were added.
- Page 1.1-34 thru 1.1-36, deleted Figure 1.1-2, replaced Figures 1.1-3 and 1.1-4, and deleted Figure 1.1-5.
- Page 1.1-37, added Table 1.1-1.
- Page 1.2-11, modified the first sentence of the first paragraph in Section 1.2.1.2.7.

Chapter 2

- Page 2.7.1.1-3, modified the temperature in "Condition 3" of Item 3 near the top of the page.
- Page 2.12.1-1, modified Section 2.12.1.3 throughout the paragraph.
- Pages 2.12.6-4 thru 2.12.6-5, replaced text in Section 2.12.6.7.4.1.
- Pages 2.12.6-6 thru 2.12.6-11, text flow changes.
- Page 2.12.6-12, replaced Figure 2.12.6.12-3.
- Page 2.12.6-13, text flow changes.
- Page 2.12.6-14, added new Figure 2.12.6.12-5.
- Page 2.12.6-15, renumbered the referenced figure number in the first line of the third paragraph on the page.
- Page 2.12.6-16, renumbered Figure 2.12.6.12-6.
- Page 2.12.6-17, modified the second and third paragraphs of Section 2.12.6.12.3.2.
- Page 2.12.6-18, text flow changes.
- Pages 2.12.6-19 thru 2.12.6-20, added new Figures 2.12.6.12-7, 2.12.6.12-8 and 2.12.6.12-9.
- Pages 2.12.6-21 thru 2.12.6-29, text flow changes.
- Page 2.13.6-1, modified the second and fourth paragraphs of Section 2.13.6.
- Pages 2.13.6-4 thru 2.13.6-5, modified Section 2.13.6.7.4.1.
- Pages 2.13.6-6 thru 2.13.6-26, text flow changes.
- Page 2.13.6-27, modified Section 2.13.6.13.
- Page 2.13.6-48, modified the middle of the fourth paragraph on the page.
- Page 2.13.6-49, text flow changes.
- Page 2.13.6-50, modified the last paragraph on the page.
- Page 2.13.6-51, modified the "Balsa" row on the embedded table.

List of Changes, NAC-STC SAR, Revision 15A (cont'd)

- Page 2.13.6-54, modified the bottom half of Figure 2.13.6.15-3.
- Page 2.13.6-56, replaced Figures 2.13.6.15-6 and 2.13.6.15-7.
- Page 2.13.6-57, modified the first sentence of the last paragraph on the page.

Chapter 3

- Pages 3.7.4-8 thru 3.7.4-9, modified Tables 3.7-4 and 3.7-5 and added a new table note to both tables.
- Page 3.7.5-2, modified Table 3.7-6 and added a new table note.
- Page 3.8.1-1, modified the temperature near the end of the third paragraph of Section 3.8.1.
- Pages 3.8.4-6 thru 3.8.4-9, modified Sections 3.8.4-2 and 3.8.4-6.
- Pages 3.8.4-16 thru 3.8.4-17, modified Tables 3.8-4 and 3.8-5.
- Page 3.8.5-1, added the last sentence to the second paragraph of Section 3.8.5.
- Page 3.8.5-2, deleted the second row and modified the third row (now the second) of Table 3.8-6.

Chapter 4

- Page 4.1-1, added text near the end of the description for the "Containment Condition B" bullet.
- Page 4.1-3, modified the first and third paragraphs on the page.
- Page 4.1-5, added text to the middle of the second column of Table 4.1-1.
- Page 4.1-6, added text to the bottom of the second column of Table 4.1-1.
- Page 4.1-7, added text to the middle of the second column of Table 4.1-1.
- Page 4.1-8, added a new continued page to the end of Table 4.1-1.
- Pages 4.1-9 thru 4.1-10, text flow changes.
- Page 4.2-1, added "(HBU)" to the middle of the fourth paragraph on the page.
- Page 4.2-2, added a new paragraph at the end of Section 4.2; modified the last line on the page in Section 4.2.1.
- Page 4.2-3, modified text near the end of the paragraph in Section 4.2.1 and near the middle of the first paragraph in Section 4.2.2.
- Page 4.2-4, added text in the first paragraph on the page, and added two new paragraphs in the middle of the page.
- Pages 4.2-5 thru 4.2-6, modified text in Sections 4.2.3, 4.2.3.1 and 4.2.3.2.
- Pages 4.2-7 thru 4.2-8, text flow changes.
- Page 4.2-9, modified the last paragraph on the page.
- Page 4.2-10, text flow changes.
- Page 4.2-11, modified the heading for Section 4.2.3.3 and added a second paragraph to the section.
- Pages 4.2-12 thru 4.2-18, text flow changes.
- Page 4.3-1, modified the first paragraph of Section 4.3.
- Pages 4.3-2 thru 4.3-3, text flow changes.
- Page 4.5-14, added new sentence to the end of this paragraph.
- Pages 4.5-18 thru 4.5-19, added new Parker datasheets.

List of Changes, NAC-STC SAR, Revision 15A (cont'd)

- Pages 4.5-20 thru 4.5-35, text flow changes.
- Pages 4.7-1 thru 4.7-2, added text throughout Sections 4.7.1 and 4.7.2.
- Page 4.7-3, deleted the last paragraph on the page.

Chapter 5

- Page 5.7-1, modified Section 5.7 by adding a sentence to the end of the second paragraph.
- Page 5.7.1-5, modified the "Radial" rates in Table 5.7.1-2.

Chapter 6

- Page 6.4.2-7, modified the last paragraph on the page.
- Page 6.4.2-11, added new Table 6.4.2-5.
- Page 6.9.1-1, added text near the end of the paragraph in Section 6.9.1.

Chapter 7

- Page 7-2, modified text to the middle of the last paragraph on the page.
- Page 7.1-2, added text to the middle of the second paragraph in Section 7.1.2.
- Page 7.1-6, modified first paragraph of Section 7.1.3.1.
- Page 7.1-7, modified step 5 of Section 7.1.3.1.
- Pages 7.1-8 thru 7.1-10, modified steps 15, 16, 19c, 21c and footnote (1) of Section 7.1.3.1.
- Pages 7.1-11 thru 7.1-13 modified the second paragraph and step 8 and added the "Note" following step 8, in Section 7.1.3.2, with text flow changes throughout.
- Pages 7.1-14 thru 7.1-15, text flow changes.
- Page 7.2-2, added two notes to step 13 of Section 7.2.1.
- Pages 7.2-3 thru 7.2-5, text flow change.
- Page 7.3-2, added text to the end of the first paragraph at the top of the page, in Section 7.3.2.
- Page 7.3-3, modified steps 4.f and 4.g of Section 7.3.2.1.
- Page 7.3-4, text flow changes.
- Page 7.3-5, added text to steps 1 and 3 of Section 7.3.3.1.
- Pages 7.3-6 thru 7.3-8, text flow changes.
- Page 7.3-9, modified text in step 6 of Section 7.3.4.
- Page 7.3-10, text flow changes.
- Pages 7.4-1 thru 7.4-6, modified Section 7.4 throughout.
- Page 7.6-6, modified steps 7 and 8 as well as the "Note" following step 8, in Section 7.6.3,

Chapter 8

- Pages 8.1-6 thru 8.1-9, modified Section 8.1.3 throughout.
- Pages 8.1-10 thru 22, text flow changes.
- Page 8.1-23, deleted text in the second paragraph of Section 8.1.10.
- Page 8.1-24, text flow changes.

List of Changes, NAC-STC SAR, Revision 15A (cont'd)

- Page 8.1-33, modified the last bullet on the page.
- Page 8.1-34, modified the first sub-bullet on the page.
- Page 8.1-35, modified the last paragraph on the page.
- Page 8.1-36, modified the sixth paragraph on the page.
- Page 8.1-40, modified the third row of Table 8.1-1.
- Pages 8.2-2 thru 8.2-3, modified text in Section 8.2.2 throughout.
- Pages 8.2-4 thru 8.2-7, text flow changes.

Chapter 9

- No changes.

Enclosure 3

List of Drawing Changes

NAC-STC SAR, Revision 15A

List of Drawing Changes, NAC-STC SAR, Revision 15A

Drawing 423-800, Sheets 1 thru 2 of 2, Revision 17P and 17NP

Sheet 1:

1. B.O.M., Item 17, revised Qty for assembly 97 to "A/R", was "1"; Description to "BOSS SEAL MS33649-4, METAL, SILVER COATED", was "FURON #10061-04-1-0"
2. B.O.M., Item 26, revised Description to "PARKER 3-908VM835-75", was "PARKER 3-908V747-75"
3. B.O.M., added item 29 as follows: Qty for assembly 97- "A/R"; Name - "O-RING"; Material - "VITON"; Spec - "COML"; Description - "PARKER 3-904VM835-75".
4. Added Delta note 14 as follows: "METAL OR POLYMER SEAL MAY BE USED. QUANTITY OF ONE."

Sheet 2

5. Zone E1, added identification balloons for items 12, 17 & 29, wording "FOR HBU TRANSPORT" and Delta note 14 symbol.

Drawing 423-803, Sheets 1 thru 2 of 2, Revision 10

Sheet 1:

1. B.O.M., Item 10, revised Description to "PARKER 3-906VM835-75", was "PARKER 3-906V747-75"
2. B.O.M., Item 11, revised Description to "PARKER 3-916VM835-75", was "PARKER 3-916V747-75"
3. B.O.M., Item 12, revised Qty for assembly 99 to "A/R", was "1"; Description to "BOSS SEAL MS33649-16, METAL, SILVER COATED", was "FURON #10061-16-1-0".
4. B.O.M., Item 15, revised Description to "PARKER .275 DIA. VM835-75", was "PARKER .275 DIA. V0835-75"
5. B.O.M., Item 16, revised Description to "PARKER .275 DIA. VM835-75", was "PARKER .275 DIA. V0835-75"
6. B.O.M., added item 19 as follows: Qty for assembly 99 - "A/R"; Name - "0-RING"; Material - "VITON"; Spec - "COML"; Description - "PARKER 3-916VM835-75".
7. Added Delta note 16 as follows: "FOR ASSEMBLY 99, METAL OR POLYMER SEAL MAY BE USED. QUANTITY OF ONE."

Sheet 2

8. Zone F8, added item 19 identification balloon and Delta note 16 symbol to item 12 identification balloon.

Drawing 423-805, Sheets 1 thru 2 of 2, Revision 7

Sheet 1:

1. B.O.M., Item 3, revised Description to "PARKER .275 DIA VM835-75", was "PARKER .275 DIA. VO835-75"

Drawing 423-806, Sheet 1 of 1, Revision 8

1. B.O.M., Item 6, revised Qty for assembly 99 to "A/R", was "1"; Description to "BOSS SEAL MS33649-2, METAL, SILVER COATED", was "FURON #10061-02-1-0".
2. B.O.M., Item 8, revised Description to "PARKER #2-238 VM835-75", was "PARKER 2-238V0835-75"
3. B.O.M., Item 9, revised Description to "PARKER #2-244 VM835-75", was "PARKER 2-244V0835-75"
4. B.O.M., added item 10 as follows: Qty for assembly 99 - "A/R"; Name - "0-RING"; Material - "VITON"; Spec - "COML"; Description - "PARKER #3-902 VM835-75".
5. Zone F2; added item 10 identification balloon and Delta note 6 symbol.
6. Added Delta note 6 as follows: "FOR ASSEMBLY 99, METAL OR POLYMER SEAL MAY BE USED. QUANTITY OF ONE."

Drawing 423-807, Sheet 1 thru 3 of 3, Revision 4

1. B.O.M., Item 5, revised Material to "VITON", was "SEE NOTE 10"; Description to ""PARKER 2-147VM835-75", was "SEE NOTE 10"
2. B.O.M., Item 9, revised Description to "PARKER 3-904VM835-75", was "PARKER 3-904V0835-75"
3. Revised general note 10 to "(DELETED)", was "SEAL DIMENSIONS TO BE Ø103±.005 X 2.675±.003 I.D. SEAL MATERIAL TO BE UNREPROCESSED VIRGIN PTFE BASE MATERIAL WITH PLASTIC FILLER."

Drawing 423-875, Sheets 1 thru 2 of 2, Revision 10

1. General note 1, revised word "BORAL" to "BORAL/TALBOR"
2. Zones A8 & A6, revised view label to "Detail A-A"; was "Detail D-D"
3. Zones B7 & B6, revised dimension to ".075±.005"; was "(.08)"

Drawing 423-878, Sheets 1 thru 2 of 2, Revision 2

Sheet 1:

1. General note 1, revised word "BORAL" to "BORAL/TALBOR"
2. Zone A7, revised dimension to ".100±.006/ITEM 2"; was "(.10) ITEM 2"
3. Zone B7, revised dimension to "(.2)/TYP"; was "(.18)/TYP"; revised dimension to "(.3)/TYP", was "(.32)/TYP"

Sheet 2:

4. Zone E4, revised dimension to "(8.2)"; was "(8.0)"

Drawing 423-880, Sheets 1 thru 2 of 2, Revision 1P

[redacted]

Drawing 630057-501, Sheet 1 thru 2 of 2, Revision 1

Sheet 1:

1. B.O.M., Item 4, revised Drawing No. to "423-803-98", was "423-803-99"
2. B.O.M., Item 5, revised Drawing No. to "423-805-98", was "423-805-99"
3. B.O.M., Item 13, revised Description to "BOSS SEAL MS33649-4, METAL, SILVER COATED", was "FURON # 10061-04-1-0"
4. B.O.M., Item 14, revised Description to "PARKER 3-908VM835-75", was "PARKER 3-908V747-75"

Drawing 630087-510, Sheet 1 of 1, Revision 1

1. Zone B8, revised Note 4 to "...welding of the Closure Lid Assembly", was "...welding of the Closure Lid"
2. Zone B5 and C5, revised Detail C-C graphics
3. B.O.M., Item 3, revised Name to "Closure Lid Assembly", was "Closure Lid"

Drawing 630087-511, Sheet 1 of 1, Revision 1

1. Zone F6, added callout for Detail A-A; Zone C5 add graphics and dimensions for Detail A-A

Drawing 630087-513, Sheets 1 thru 3, Revision

1 Sheet 1:

1. Zone B8, added Delta Note 4 as follows: "Optional welds to aid with placement of Closure Lid Assembly within Overpack Shell. Additional welds, as indicated, permitted to aid fit up."
2. Zone C6, revised callout to "Item 1 – Closure Lid", was "Assembly 99 – Closure Lid"
3. Zone C4, Section A-A, revised graphics to show groove; Zone F4, add groove dimensions
4. Zone C2/C3 thru E2/E3, added graphics, details and balloon callouts for Assembly 99 "Closure Lid Assembly"
5. B.O.M., added Item 2 with Assy 99 Qty. "1", Name "Backing Ring", Material "ASTM A276", and Description "Bar"

6. Title Block, revised sheet count to "1 of 3"
7. B.O.M., Item 1, revised spec to "ASTM A240/A965", was "ASTM A240/A695"

Sheet 2:

8. Added views, details, and dimensions for Item 2 "Backing Ring"

Sheet 3:

9. Added views of Assembly 99 with optional weld details and dimensions

Enclosure 4 – Proposed
CoC Changes

Enclosure 4

Proposed Changes for Certificate of Compliance Revision 13

NAC-STC SAR, Revision 15A

CoC Sections (new)

Page 6 of 15

5.(a)(3) Drawings (Continued)

(ix) For the West Valley Demonstration Project High Level Waste overpack, the shell, overpack closure lid, spacers, transport insert and basket are constructed and assembled in accordance with the following NAC International Drawing Nos.:

630087-501, sheets 1-2, Rev. 1	630087-511, Rev. 1
630087-504, Rev. 0	630087-512, Rev. 0
630087-505, Rev. 0	630087-513, Rev. 1
630087-510, Rev. 1	630087-514, Rev. 0

Page 11 of 15

5.(b)(1) Contents – Type and Form of Material (Continued)

(vi) West Valley Demonstration Project (WVDP) High Level Waste (HLW) canisters containing HLW vitrified in borosilicate glass. An overpack may contain melter-evacuated HLW canisters partially filled with glass or HLW debris. The nominal height of a canister shall be ≤ 118 inches and the nominal width shall be ≤ 24 inches. The heat load shall be ≤ 0.300 kW per HLW canister. The maximum gross weight allowed per canister is 5,500 lbs. The following are the applicable design limits for the HLW:

WVDP HLW Canisters	
Earliest Transport Date:	4/1/2014
Maximum HLW Mass [kg]:	2200
Maximum Ci Content HLW	
Cs-137:	42000
Ba-137m:	40000
Sr-90:	23000
Y-90:	23000
Co-60:	0.2

Page 13 of 15

5.(b)(2) Maximum quantity of material per package (Continued)

(vii) For the contents described in 5.(b)(1)(vi): Up to 5 HLW canisters, including melter-evacuated HLW canisters partially filled with glass or HLW debris, may be transported in the WVDP HLW overpack. For an overpack loaded with less than 5 canisters, a transport insert shall be loaded in all empty basket cell locations.

The NAC-STC content weight shall be ≤ 45,800 lbs. in the WVDP HLW overpack configuration. The HLW overpack heat load shall be ≤ 1.5 kW. Top and bottom spacers are authorized for axially positioning the WVDP HLW overpack with the NAC-STC cask cavity.

<u>CoC Sections (revised)</u>

Page 1 of 15
5.(a)(2) Description

<u>Revised 1st Full Paragraph</u>

A steel, lead and polymer (NS4FR) shielded shipping cask for (a) directly loaded irradiated PWR fuel assemblies, (b) intact, damaged and/or the fuel debris of Yankee Class or Connecticut Yankee irradiated PWR fuel assemblies in a canister, (c) West Valley Demonstration Project (WVDP) High Level Waste (HLW) canisters in a HLW overpack, and (d) non-fissile, solid radioactive materials (referred to hereafter as Greater Than Class C (GTCC) as defined in 10 CFR Part 61) waste in a canister. The cask body is a right circular cylinder with an impact limiter at each end. The package has approximate dimensions as follows:

Page 4 of 15
5.(a)(2) Description (Continued)

<u>New Last Full Paragraph</u>

The WVDP-HLW overpack consists of three (3) principle components. These are the HLW Overpack shell, basket, and closure lid. The HLW Overpack consists of an annular right-circular shell closed at one end by a bottom plate. The shell is constructed of 3/8-inch rolled dual-certified Type 304/304L stainless steel plate. The edges of the rolled plates are joined with full penetration welds. The Type 304/304L stainless steel bottom plate is also attached to the shell by using a full penetration weld. The shell is constructed in accordance with ASME Code Section VIII, Division 2. The nominal inside and outside diameters of the HLW Overpack are 69.8 inches and 70.6 inches, respectively. The overall external length of the HLW Overpack is 126.5 inches, the inside depth is 124.5 inches and the bottom plate is 2.0-inch thick. The closure lid is a 4-inch thick Type 304/304L stainless steel plate or forging. It is joined to the HLW Overpack shell using a partial penetration weld. The basket has 5 cells that allows up to 5 HLW canisters, including melter-evacuated HLW canisters partially filled with glass or HLW debris, to be loaded. Empty cells shall have a transport insert loaded.

5.(a)(3) Drawings

(i) The cask is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-800, sheets 1-3, Rev. 17P & 17NP	423-811, sheets 1-2, Rev. 11
423-802, sheets 1-7, Rev. 21	423-812, Rev. 6
423-803, sheets 1-2, Rev. 10	423-900, Rev. 7
423-804, sheets 1-3, Rev. 9	423-209, Rev. 0
423-805, sheets 1-2, Rev. 7	423-210, Rev. 0
423-806, Rev. 8	423-901, Rev. 2
423-807, sheets 1-3, Rev. 4	

(ii) For the directly loaded configuration, the basket is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-870, Rev. 6	423-874, Rev. 2
423-871, Rev. 5	423-875, sheets 1-2, Rev. 10
423-872, Rev. 6	423-878, sheets 1-2, Rev. 2
423-873, Rev. 2	423-880, Rev. 1P & 0NP

5.(b)(1) Contents

(1) Type and Form of Material

(i) Irradiated PWR fuel assemblies with uranium oxide pellets. Each fuel assembly may have a maximum burnup of 45 GWd/MTU(2). The minimum fuel cool time is defined in the Fuel Cool Time Table, below. The maximum heat load per assembly is 850(2) watts. Prior to irradiation, the fuel assemblies must be within the following dimensions and specifications:

Assembly Type	14x14	15x15	16x16	17x17	17x17 (OFA)	Framatome-Cogema 17x17
Cladding Material	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy	Zirconium Alloy
Maximum Initial Uranium Content (kg/assembly)	407	469	402.5	464	426	464
Maximum Initial Enrichment (wt% ^{235}U)	4.2	4.2	4.2	4.2	4.2	4.5
Minimum Initial Enrichment (wt% ^{235}U)	1.7	1.7	1.7	1.7	1.7	1.7
Assembly Cross- Section (inches)	7.76 to 8.11	8.20 to 8.54	8.10 to 8.14	8.43 to 8.54	8.43	8.425 to 8.518
Number of Fuel Rods per Assembly	176 to 179	204 to 216	236	264	264	264(1)
Fuel Rod OD (inch)	0.422 to 0.440	0.418 to 0.430	0.382	0.374 to 0.379	0.360	0.3714 to 0.3740
Minimum Cladding Thickness (inch)	0.023	0.024	0.025	0.023	0.023	0.0204
Pellet Diameter (inch)	0.344 to 0.377	0.358 to 0.390	0.325	0.3225 to 0.3232	0.3088	0.3224 to 0.3230
Maximum Active Fuel Length (inches)	146	144	137	144	144	144.25

Notes:

(1) - Fuel rod positions may also be occupied by solid poison shim rods or solid zirconium alloy or stainless steel fill rods.

(2) - For 17x17 PWR high burnup (HBU) fuel, a maximum assembly decay heat above 0.85 kW, up to 1.71 kW, and a burnup of 45 to 60 GWd/MTU is allowed provided the loading pattern meets the requirements of Configuration A, B or C, as shown in NAC International Drawing No. 423-800. The maximum time duration from the start of vacuum drying until the cask is placed in the horizontal orientation is limited to 48 hours. HBU fuel shipments are limited to a total duration of 3 months from the time cask loading is complete until the cask arrives at its final destination. The maximum allowed ambient temperature change during the 3-month shipment duration is limited to 75°F.

5.(b)(1)(i) Contents – Type and Form of Material – Irradiated PWR fuel assemblies (Continued)

New Fuel Cool Time Tables

FUEL COOL TIME TABLE
(Configuration A 17x17 PWR HBU)
Minimum Fuel Cool Time in Years

Cobalt [g/kg]	Min Enr. [wt. %]	Burnup [GWd/MTU] 45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60
0.4	2.9	4.0	4.0	4.0	4.3	4.9	5.6	6.3	7.0	7.8	8.7	-	-	-	-	-	-
	3.1	4.0	4.0	4.0	4.0	4.1	4.7	5.3	6.0	6.8	7.5	8.3	9.2	10.1	-	-	-
	3.3	4.0	4.0	4.0	4.0	4.0	4.1	4.5	5.1	5.8	6.5	7.2	8.0	8.9	-	-	-
	3.5	4.0	4.0	4.0	4.0	4.0	4.1	4.2	4.3	4.9	5.5	6.2	7.0	7.8	8.6	9.4	10.3
	3.7	4.0	4.0	4.0	4.0	4.0	4.0	4.2	4.3	4.4	4.7	5.3	6.0	6.7	7.5	8.3	9.1
	3.9	4.0	4.0	4.0	4.0	4.0	4.0	4.1	4.2	4.3	4.4	4.5	5.1	5.8	6.5	7.2	8.0
	4.1	4.0	4.0	4.0	4.0	4.0	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.9	5.6	6.3	7.0
	4.3	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.1	4.2	4.3	4.4	4.5	4.7	4.8	5.4	6.1
	4.5	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.7	4.9	5.2
	4.7	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.1	4.2	4.4	4.4	4.5	4.7	4.8	4.9
	4.9	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.8	4.9
0.8	2.9	6.4	6.9	7.4	8.0	8.6	9.1	9.8	10.4	11.1	11.8	-	-	-	-	-	-
	3.1	5.7	6.2	6.7	7.2	7.8	8.3	8.9	9.5	10.1	10.8	11.5	12.2	12.9	-	-	-
	3.3	5.1	5.6	6.0	6.5	7.0	7.6	8.1	8.7	9.3	9.9	10.6	11.2	11.9	-	-	-
	3.5	4.6	5.0	5.4	5.9	6.4	6.8	7.4	7.9	8.5	9.0	9.7	10.3	11.0	11.6	12.3	13.1
	3.7	4.1	4.5	4.9	5.3	5.7	6.2	6.7	7.2	7.7	8.3	8.8	9.4	10.1	10.7	11.4	12.0
	3.9	4.0	4.0	4.4	4.7	5.2	5.6	6.0	6.5	7.0	7.5	8.1	8.7	9.2	9.8	10.5	11.2
	4.1	4.0	4.0	4.0	4.3	4.6	5.0	5.5	5.9	6.4	6.9	7.4	7.9	8.5	9.0	9.6	10.3
	4.3	4.0	4.0	4.0	4.0	4.2	4.5	4.9	5.4	5.8	6.3	6.7	7.2	7.8	8.3	8.9	9.4
	4.5	4.0	4.0	4.0	4.0	4.0	4.1	4.5	4.9	5.3	5.7	6.1	6.6	7.1	7.6	8.1	8.7
	4.7	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.4	4.8	5.2	5.6	6.0	6.5	7.0	7.5	8.0
	4.9	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.0	4.3	4.7	5.1	5.5	5.9	6.4	6.9	7.4
1.2	2.9	8.9	9.4	9.9	10.4	11.0	11.5	12.0	12.6	13.3	13.8	-	-	-	-	-	-
	3.1	8.3	8.8	9.2	9.7	10.2	10.7	11.3	11.8	12.4	13.0	13.6	14.2	14.9	-	-	-
	3.3	7.7	8.1	8.6	9.0	9.5	10.0	10.5	11.1	11.6	12.1	12.7	13.3	13.9	-	-	-
	3.5	7.2	7.6	8.0	8.4	8.9	9.3	9.8	10.3	10.9	11.4	11.9	12.5	13.1	13.7	14.3	15.0
	3.7	6.7	7.0	7.4	7.9	8.3	8.7	9.2	9.6	10.1	10.7	11.2	11.7	12.3	12.9	13.5	14.0
	3.9	6.2	6.6	6.9	7.3	7.7	8.1	8.6	9.0	9.5	10.0	10.5	11.0	11.5	12.0	12.6	13.2
	4.1	5.8	6.1	6.5	6.8	7.2	7.6	8.0	8.4	8.9	9.3	9.8	10.3	10.8	11.3	11.9	12.4
	4.3	5.4	5.7	6.0	6.4	6.7	7.1	7.5	7.9	8.3	8.8	9.2	9.7	10.1	10.7	11.2	11.7
	4.5	5.0	5.3	5.6	6.0	6.3	6.7	7.0	7.4	7.8	8.2	8.6	9.1	9.5	10.0	10.5	11.0
	4.7	4.6	4.9	5.2	5.6	5.9	6.2	6.6	6.9	7.3	7.7	8.1	8.5	8.9	9.4	9.9	10.3
	4.9	4.3	4.6	4.9	5.2	5.5	5.8	6.2	6.5	6.9	7.2	7.6	8.0	8.4	8.8	9.3	9.7

FUEL COOL TIME TABLE
(Configuration B 17x17 PWR HBU)
Minimum Fuel Cool Time in Years

Cobalt	Min Enr.	Burnup [GWd/MTU]															
[g/kg]	[wt. %]	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60
0.4	2.9	4.3	4.4	5.0	5.7	6.5	7.3	8.2	9.1	10.0	11.0	-	-	-	-	-	-
	3.1	4.2	4.3	4.4	4.8	5.5	6.2	7.0	7.9	8.8	9.7	10.7	11.7	12.7	-	-	-
	3.3	4.1	4.3	4.4	4.5	4.6	5.2	6.0	6.7	7.6	8.4	9.4	10.3	11.3	-	-	-
	3.5	4.1	4.2	4.3	4.4	4.5	4.7	5.0	5.7	6.5	7.3	8.2	9.0	10.0	11.0	11.9	13.0
	3.7	4.0	4.2	4.3	4.4	4.5	4.6	4.8	4.9	5.5	6.3	7.0	7.9	8.8	9.7	10.6	11.6
	3.9	4.0	4.1	4.2	4.3	4.5	4.6	4.7	4.8	5.0	5.3	6.1	6.8	7.6	8.5	9.4	10.3
	4.1	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.8	4.9	5.0	5.2	5.9	6.6	7.4	8.2	9.1
	4.3	4.0	4.0	4.1	4.3	4.4	4.5	4.6	4.7	4.9	5.0	5.2	5.3	5.7	6.4	7.2	8.0
	4.5	4.0	4.0	4.1	4.2	4.3	4.4	4.6	4.7	4.8	5.0	5.1	5.3	5.4	5.6	6.2	7.0
	4.7	4.0	4.0	4.1	4.2	4.3	4.4	4.5	4.6	4.8	4.9	5.0	5.2	5.4	5.6	5.7	6.1
	4.9	4.0	4.0	4.0	4.1	4.2	4.4	4.5	4.6	4.7	4.8	5.0	5.1	5.3	5.5	5.6	5.8
0.8	2.9	7.4	8.0	8.6	9.2	9.9	10.6	11.4	12.1	12.9	13.7	-	-	-	-	-	-
	3.1	6.6	7.2	7.8	8.4	9.0	9.7	10.4	11.1	11.8	12.6	13.4	14.2	15.1	-	-	-
	3.3	5.9	6.5	7.0	7.6	8.1	8.8	9.4	10.1	10.8	11.5	12.3	13.1	13.9	-	-	-
	3.5	5.3	5.8	6.3	6.8	7.4	8.0	8.6	9.2	9.9	10.6	11.3	12.0	12.8	13.6	14.4	15.3
	3.7	4.8	5.2	5.7	6.1	6.7	7.2	7.8	8.4	9.0	9.6	10.3	11.1	11.8	12.5	13.3	14.1
	3.9	4.3	4.7	5.1	5.6	6.0	6.5	7.0	7.6	8.2	8.8	9.4	10.1	10.8	11.5	12.3	13.1
	4.1	4.0	4.2	4.6	5.0	5.4	5.9	6.4	6.9	7.5	8.0	8.6	9.3	9.9	10.6	11.3	12.0
	4.3	4.0	4.0	4.2	4.5	4.9	5.3	5.8	6.3	6.8	7.3	7.9	8.5	9.1	9.7	10.4	11.1
	4.5	4.0	4.0	4.1	4.2	4.4	4.8	5.3	5.7	6.2	6.7	7.2	7.8	8.3	8.9	9.6	10.2
	4.7	4.0	4.0	4.1	4.2	4.3	4.4	4.8	5.2	5.6	6.1	6.6	7.1	7.6	8.2	8.8	9.4
	4.9	4.0	4.0	4.1	4.1	4.3	4.4	4.5	4.7	5.1	5.6	6.0	6.5	7.0	7.5	8.1	8.6
1.2	2.9	9.8	10.4	11.0	11.6	12.1	12.8	13.5	14.1	14.8	15.6	-	-	-	-	-	-
	3.1	9.1	9.6	10.2	10.8	11.3	11.9	12.5	13.2	13.8	14.5	15.3	16.0	16.8	-	-	-
	3.3	8.5	9.0	9.5	10.0	10.6	11.1	11.7	12.3	12.9	13.6	14.2	15.0	15.7	-	-	-
	3.5	7.9	8.3	8.8	9.3	9.8	10.4	10.9	11.5	12.0	12.7	13.4	14.0	14.7	15.4	16.1	16.9
	3.7	7.3	7.8	8.2	8.7	9.1	9.6	10.2	10.7	11.3	11.9	12.5	13.1	13.8	14.4	15.2	15.9
	3.9	6.8	7.2	7.6	8.1	8.5	9.0	9.5	10.0	10.6	11.1	11.7	12.3	12.9	13.5	14.2	14.9
	4.1	6.3	6.7	7.1	7.5	8.0	8.4	8.9	9.4	9.9	10.4	11.0	11.5	12.1	12.7	13.3	14.0
	4.3	5.9	6.3	6.6	7.0	7.4	7.9	8.3	8.8	9.2	9.7	10.2	10.8	11.4	11.9	12.5	13.1
	4.5	5.5	5.9	6.2	6.6	6.9	7.4	7.8	8.2	8.7	9.1	9.6	10.1	10.7	11.2	11.7	12.3
	4.7	5.1	5.5	5.8	6.1	6.5	6.9	7.3	7.7	8.1	8.6	9.0	9.5	10.0	10.5	11.1	11.6
	4.9	4.8	5.1	5.4	5.8	6.1	6.5	6.8	7.2	7.6	8.0	8.5	8.9	9.4	9.9	10.4	10.9

FUEL COOL TIME TABLE
(Configuration C 17x17 PWR HBU)
Minimum Fuel Cool Time in Years

Cobalt	Min Enr.	Burnup [GWd/MTU]															
[g/kg]	[wt. %]	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60
0.4	2.9	7.1	8.0	9.1	10.3	11.6	12.9	14.2	15.6	17.0	18.4	-	-	-	-	-	-
	3.1	6.0	6.9	7.8	8.8	10.0	11.2	12.5	13.8	15.2	16.5	17.9	19.4	20.8	-	-	-
	3.3	5.2	5.8	6.7	7.5	8.5	9.7	10.9	12.1	13.4	14.8	16.1	17.5	18.9	-	-	-
	3.5	5.1	5.3	5.7	6.5	7.3	8.3	9.4	10.6	11.8	13.1	14.4	15.7	17.1	18.4	19.8	21.3
	3.7	5.1	5.3	5.4	5.6	6.3	7.1	8.0	9.1	10.2	11.5	12.7	14.0	15.3	16.7	18.0	19.4
	3.9	5.0	5.2	5.4	5.6	5.8	6.1	6.9	7.8	8.9	10.0	11.2	12.4	13.7	15.0	16.3	17.6
	4.1	5.0	5.1	5.3	5.5	5.7	5.9	6.0	6.8	7.6	8.6	9.7	10.9	12.1	13.4	14.6	15.9
	4.3	4.9	5.1	5.3	5.5	5.6	5.8	6.0	6.2	6.6	7.5	8.5	9.5	10.7	11.8	13.1	14.3
	4.5	4.9	5.0	5.2	5.4	5.6	5.8	5.9	6.1	6.4	6.7	7.3	8.3	9.3	10.4	11.6	12.8
	4.7	4.8	5.0	5.1	5.4	5.5	5.7	5.9	6.1	6.3	6.6	6.8	7.2	8.1	9.1	10.2	11.4
	4.9	4.8	4.9	5.1	5.3	5.5	5.7	5.8	6.0	6.2	6.5	6.7	7.0	7.3	8.0	9.0	10.0
0.8	2.9	10.6	11.4	12.2	13.1	14.0	15.1	16.2	17.4	18.6	19.8	-	-	-	-	-	-
	3.1	9.6	10.4	11.2	11.9	12.8	13.7	14.7	15.8	17.0	18.1	19.4	20.6	21.9	-	-	-
	3.3	8.8	9.4	10.2	10.9	11.7	12.5	13.4	14.4	15.5	16.6	17.7	19.0	20.2	-	-	-
	3.5	8.0	8.6	9.3	10.0	10.7	11.5	12.3	13.2	14.1	15.1	16.2	17.4	18.6	19.8	21.0	22.3
	3.7	7.5	7.8	8.5	9.1	9.8	10.5	11.3	12.0	12.9	13.8	14.8	15.9	17.1	18.2	19.4	20.6
	3.9	7.2	7.4	7.7	8.3	9.0	9.6	10.4	11.1	11.9	12.7	13.6	14.6	15.6	16.7	17.9	19.0
	4.1	6.9	7.1	7.3	7.6	8.2	8.8	9.5	10.2	10.9	11.7	12.5	13.4	14.3	15.4	16.4	17.6
	4.3	6.7	6.8	7.0	7.2	7.5	8.1	8.7	9.4	10.0	10.8	11.5	12.3	13.2	14.0	15.1	16.1
	4.5	6.4	6.6	6.7	6.9	7.1	7.4	8.0	8.6	9.2	9.9	10.6	11.4	12.1	13.0	13.8	14.9
	4.7	6.2	6.3	6.5	6.7	6.8	7.0	7.4	7.9	8.5	9.1	9.8	10.5	11.2	12.0	12.8	13.7
	4.9	5.9	6.1	6.3	6.4	6.6	6.8	6.9	7.3	7.8	8.4	9.0	9.7	10.4	11.1	11.8	12.6
1.2	2.9	12.9	13.6	14.3	15.1	15.9	16.7	17.7	18.7	19.8	20.9	-	-	-	-	-	-
	3.1	12.0	12.7	13.4	14.0	14.8	15.6	16.4	17.3	18.3	19.4	20.5	21.6	22.8	-	-	-
	3.3	11.4	11.8	12.5	13.1	13.8	14.6	15.3	16.1	17.0	17.9	19.0	20.1	21.2	-	-	-
	3.5	10.9	11.2	11.7	12.3	12.9	13.6	14.3	15.1	15.9	16.7	17.7	18.7	19.7	20.8	21.9	23.1
	3.7	10.5	10.8	11.1	11.5	12.1	12.8	13.4	14.1	14.9	15.6	16.5	17.4	18.3	19.4	20.5	21.6
	3.9	10.3	10.4	10.7	11.0	11.4	11.9	12.6	13.3	13.9	14.7	15.4	16.2	17.1	18.0	19.1	20.1
	4.1	10.0	10.1	10.3	10.6	10.8	11.2	11.8	12.4	13.1	13.8	14.5	15.2	16.0	16.9	17.8	18.8
	4.3	9.7	9.9	10.1	10.2	10.5	10.7	11.1	11.7	12.3	12.9	13.6	14.3	15.0	15.8	16.6	17.5
	4.5	9.5	9.7	9.8	10.0	10.1	10.4	10.6	11.0	11.6	12.1	12.8	13.5	14.1	14.9	15.6	16.4
	4.7	9.2	9.4	9.6	9.7	9.9	10.1	10.3	10.5	10.9	11.5	12.0	12.7	13.3	14.0	14.7	15.4
	4.9	9.0	9.2	9.4	9.5	9.7	9.8	10.0	10.2	10.4	10.8	11.4	11.9	12.6	13.2	13.8	14.5

Page 12 of 15

5.(b)(2) Maximum quantity of material per package

(i) For the contents described in Item 5.(b)(1)(i): PWR fuel with a burnup ≤ 45,000 MWd/MTU, 26 PWR fuel assemblies with a maximum total weight of 39,650 lbs. and a maximum decay heat not to exceed 22.1 kW per package. For 17x17 HBU fuel, the positioning of the fuel assemblies and shielded thermal shunts shall meet the requirements as shown in Configurations A, B or C of NAC International Drawing No. 423-800 and a maximum decay heat not to exceed 24 kW per package.

Page 14 of 15

5.(c) Criticality Safety Index (CSI):

(1) CSI=0.0 for contents described in 5.(b)(1)(i), 5.(b)(1)(ii), 5.(b)(1)(iii), 5.(b)(1)(iv) (i.e., Yankee Class and CY Fuel and GTCC Waste), and 5.(b)(1)(vi),

Enclosure 5

SAR Page Changes and LOEP

NAC-STC SAR, Revision 15A

May 2015

Revision 15A

NAC-STC

NAC Storage Transport Cask

SAFETY ANALYSIS REPORT

Non-Proprietary Version

Docket No. 71-9235

NAC INTERNATIONAL

Atlanta Corporate Headquarters: 3950 East Jones Bridge Road, Norcross, Georgia 30092 USA
Phone 770-447-1144, Fax 770-447-1797, www.nacintl.com

List of Effective Pages

List of Effective Pages (continued)

List of Effective Pages (continued)

List of Effective Pages (continued)

List of Effective Pages (continued)

List of Effective Pages (continued)

List of Effective Pages (continued)

List of Effective Pages (continued)

Table of Contents

List of Figures

List of Tables

List of Drawings

Drawing Number	Revision No.	Title
423-800, sheets 1-3	Rev 17P(1)	Cask Assembly – NAC-STC Cask
423-800, sheets 1-3	Rev 17NP(1)	Cask Assembly – NAC-STC Cask
423-802, sheets 1-7	Rev 21	Cask Body – NAC-STC Cask
423-803, sheets 1-2	Rev 10	Lid Assembly – Inner, NAC-STC Cask
423-804, sheets 1-3	Rev 9	Details - Inner Lid, NAC-STC Cask
423-805, sheets 1-2	Rev 7	Lid Assembly – Outer, NAC-STC Cask
423-806	Rev 8	Port Coverplate Assy – Inner Lid, NAC-STC Cask
423-807, sheets 1-3	Rev 4	Assembly, Port Cover, NAC-STC Cask
423-209	Rev 0	Impact Limiter Assy – Upper, NAC-STC Cask
423-210	Rev 0	Impact Limiter Assy – Lower, NAC-STC Cask
423-257	Rev 2	Balsa Impact Limiter, Upper, NAC-STC Cask
423-258	Rev 2	Balsa Impact Limiter, Lower, NAC-STC Cask
423-811, sheets 1-2	Rev 11	Details – NAC-STC Cask
423-812	Rev 6	Nameplates – NAC-STC Cask
423-843	Rev 3	Transport Assembly, Balsa Impact Limiters, NAC-STC
423-859	Rev 0	Attachment Hardware, Balsa Limiters, NAC-STC
423-870	Rev 6	Fuel Basket Assembly, PWR, 26 Element, NAC-STC Cask
423-871	Rev 5	Bottom Weldment, Fuel Basket, PWR, 26 Element, NAC-STC Cask
423-872	Rev 6	Top Weldment, Fuel Basket, PWR, 26 Element, NAC-STC Cask
423-873	Rev 2	Support Disk and Misc. Basket Details, PWR, 26 Element, NAC-STC Cask
423-874	Rev 2	Heat Transfer Disk, Fuel Basket, PWR, 26 Element, NAC-STC Cask
423-875, sheets 1-2	Rev 10	Tube, NAC-STC Cask
423-878, sheets 1-2	Rev 2	Alternate Tube Assembly, NAC-STC Cask
423-880	Rev 1P(1)	Shielded Thermal Shunt Assembly, NAC-STC Cask
423-880	Rev 0NP(1)	Shielded Thermal Shunt Assembly, NAC-STC Cask
423-900	Rev 7	Package Assembly Transportation, NAC-STC Cask
423-901	Rev 2	Transportation Package Concept, NAC-STC Cask
455-800, sheets 1-2	Rev 2	Assembly, Transport Cask, MPC-Yankee
455-801, sheets 1-2	Rev 4	Assembly, Transport Cask, NAC-MPC
455-820, sheets 1-2	Rev 3	Spacers, Transport Cask, MPC-Yankee

(1) Proprietary and Non-proprietary drawing versions are only included in their respective SAR versions.

List of Drawings (continued)

Drawing Number	Revision No.	Title
455-870	Rev 5	Canister Shell, MPC-Yankee
455-871, sheets 1-2	Rev 8	Details, Canister, MPC-Yankee
455-871, sheets 1-3	Rev 7P2	Details, Canister, MPC-Yankee
455-872, sheets 1-2	Rev 12	Assembly, Transportable Storage Canister (TSC), MPC-Yankee
455-872, sheets 1-2	Rev 11P1	Assembly, Transportable Storage Canister (TSC), MPC-Yankee
455-873	Rev 4	Assembly, Drain Tube, Canister, MPC-Yankee
455-881, sheets 1-3	Rev 8	PWR Fuel Tube, MPC-Yankee
455-887, sheets 1-3	Rev 4	Basket Assembly, 24 GTCC Container, MPC-Yankee
455-888, sheets 1-2	Rev 8	Assembly, Transportable Storage Canister (TSC), 24 GTCC Container, MPC-Yankee
455-891, sheets 1-2	Rev 1	Bottom Weldment, Fuel Basket, MPC-Yankee
455-891, sheets 1-3	Rev 2P0	Bottom Weldment, Fuel Basket, MPC-Yankee
455-892, sheets 1-2	Rev 3	Top Weldment, Fuel Basket, MPC-Yankee
455-892, sheets 1-3	Rev 3P0	Top Weldment, Fuel Basket, MPC-Yankee
455-893	Rev 3	Support Disk and Misc. Basket Details, MPC-Yankee
455-894	Rev 2	Heat Transfer Disk, Fuel Basket, MPC-Yankee
455-895, sheets 1-2	Rev 5	Fuel Basket Assembly, MPC-Yankee
455-895, sheets 1-2	Rev 5P0	Fuel Basket Assembly, MPC-Yankee
455-919	Rev 2	Retainer, United Nuclear Test Assy, MPC-Yankee
414-801, sheets 1-2	Rev 2	Cask Assembly, NAC-STC, CY-MPC
414-820	Rev 0	Canister Spacer CY-MPC
414-870	Rev 3	Canister Shell, CY-MPC
414-871, sheets 1-2	Rev 6	Details, Canister CY-MPC
414-872, sheets 1-3	Rev 6	Assembly, Transportable Storage Canister (TSC), CY-MPC
414-873	Rev 2	Drain Tube Assembly, CY-MPC
414-874	Rev 0	Shim, Canister, CY-MPC
414-875	Rev 0	Spacer Shim, Canister, CY-MPC
414-881, sheets 1-2	Rev 4	Fuel Tube, Transportable Storage Canister (TSC), CY-MPC
414-882, sheets 1-2	Rev 4	Oversize Fuel Tube, Transportable Storage Canister (TSC), CY-MPC
414-887, sheets 1-4	Rev 4	Basket Assembly, GTCC, CY-MPC

List of Drawings (Continued)

Drawing Number	Revision No.	Title
MPC-WVDP System Components		
630087-501, sheets 1-2	1	Transport Cask Assembly, MPC-WVDP
630087-504	0	Transport Spacer, Bottom, MPC-WVDP
630087-505	0	Transport Spacer, Top, MPC-WVDP
630087-510	1	HLW Overpack Assembly, MPC-WVDP
630087-511	1	Shell, HLW Overpack, MPC-WVDP
630087-512	0	Basket, HLW Overpack, MPC-WVDP
630087-513, sheets 1-3	1	Closure Lid, HLW Overpack, MPC-WVDP
630087-514	0	Transport Insert, HLW Overpack, MPC-WVDP

THIS PAGE INTENTIONALLY LEFT BLANK

1.0 GENERAL INFORMATION

NAC International (NAC) has designed a Storable Transport Cask (NAC-STC) for spent nuclear fuel. The United States Nuclear Regulatory Commission (NRC) licensed the NAC-STC for the transport of spent nuclear fuel. This Safety Analysis Report (SAR) addresses the ability of the NAC-STC to satisfy the NRC transportation requirements for spent fuel, either directly loaded in the cask (uncanistered) or in a canister, Greater Than Class C (GTCC) waste in a Transportable Storage Canister, and High Level Waste (HLW) contained in welded canisters and loaded in an overpack, as prescribed in 10 CFR 71.

The Transportable Storage Canister and HLW overpack are components of the NAC-MPC dry storage system. The NAC-MPC system is provided in four configurations. The first configuration is designed to store Yankee Class spent fuel and GTCC waste and is referred to as the Yankee-MPC. The second configuration is designed to store Connecticut Yankee spent fuel and GTCC waste and is referred to as the CY-MPC. The third configuration is designed to store Dairyland Power Cooperative (DPC) La Crosse Boiling Water Reactor (LACBWR) spent nuclear fuel and is referred to as the MPC-LACBWR. Refer to Section 1.4, General Information – STC-LACBWR, for the introduction and package description of the STC-LACBWR. The fourth configuration is designed to store West Valley Demonstration Project (WVDP) HLW canisters and is referred to as the MPC-WVDP.

This chapter presents a general introduction to the transport cask and a description of its design features. The terminology used throughout this report is summarized in Table 1-1.

The NAC-STC may be shipped by rail, barge, or heavy-haul vehicle. The NAC-STC is assigned a Transport Index (TI) of 25 based on the shielding evaluation summarized in Section 5.8. This TI bounds all uncanistered (directly loaded) and canistered NAC-STC contents (i.e., the TI of 25 represents the maximum TI for any content). As shown in Chapter 6, the Criticality Safety Index (CSI) is zero for uncanistered (directly loaded) and canistered Yankee-MPC and CY-MPC contents since an infinite number of packages with optimum moderation remain subcritical. The Yankee-MPC and MPC-CY GTCC canisters and MPC-WVDP HLW overpacks have a criticality safety index of zero. The MPC-LACBWR system is limited to one cask in the accident configuration, resulting in a CSI of 100.

The NAC-STC has been designed to satisfy the international requirements of the International Atomic Energy Agency (IAEA), in addition to U.S. requirements prescribed in 10 CFR 71.

Table 1-1 Terminology

Balsa Impact Limiter	A device constructed primarily of balsa wood with limited use of redwood. This device is designed to dissipate energy during normal and accident conditions impact events for packages weighing up to 260,000 lb. The balsa impact limiter is 128 inches in diameter with balsa wood providing primary protection during end and corner impact events and redwood for side impact events.
Cask Model	NAC-STC
NAC-STC Cask	This packaging consists of a spent-fuel storable transport cask body with dual closure lids and energy-absorbing impact limiters.
Packaging	The assembly of components necessary to ensure compliance with the packaging requirements of 10 CFR 71. Within this report, the packaging is denoted as the NAC-STC.
Package	The packaging with its radioactive contents (payload), as presented for transport (10 CFR 71.4). Within this report, the package is denoted as the NAC-STC, the NAC-STC cask or, simply, the cask.
Contents (Payload)	Up to twenty-six (26) pressurized water reactor (PWR) fuel assemblies in the directly loaded fuel (uncanistered) configuration, up to 20 uncanistered high burnup (HBU) PWR fuel assemblies and associated shielded thermal shunts, or a sealed Transportable Storage Canister containing Yankee Class or Connecticut Yankee spent fuel or Greater Than Class C waste, or LaCrosse boiling Water Reactor (LACBWR) spent fuel, or up to five (5) HLW canisters in a sealed HLW overpack.
Containment System	The components of the packaging intended to retain the radioactive material during transport.
Cask Body	
- Multiwall Body	Construction of the cask body, which consists of concentric layers of the inner shell, gamma shielding, outer shell and neutron shielding materials.

Table 1-1 Terminology (continued)

- Neutron Shield	Consists of the stainless steel shell, gussets, and end plates, copper-stainless steel (Cu/SS) fins, and the solid NS-4-FR neutron shielding material.
NS-4-FR	A solid, synthetic polymer developed by BISCO Products, Inc. and supplied by the Japan Atomic Power Company and its licensees. NS-4-FR is a borated hydrogenous material, which results in neutron absorption capabilities similar to borated water.
Lifting Trunnions	Four, high-strength stainless steel components welded to the top forging that are used in pairs for lifting and handling the cask.
Top Forging	
- Interlid Port	A penetration in the top forging that is used as (1) a drain for the interlid region and (2) a pressure test port for the outer lid seal.
- Pressure Port	A penetration in the top forging that houses a pressure transducer, which may be used to monitor the pressure in the interlid region during storage.
- Port Cover Assembly	Includes the port cover body, spacer, retainer, bolts and O-rings.
- Pressure Transducer	An instrument for measurement of pressure in a confined space.
- O-ring	An O-ring seals the interfaces between separate cask components.
- PTFE	Blended Polytetrafluoroethylene (PTFE) is an O-ring material used as a sealing component between metallic surfaces.
- Viton	A fluorocarbon rubber O-ring material used as a sealing component between metallic surfaces.
- Interseal	Refers to the region between pairs of O-rings.

Table 1-1 Terminology (continued)

Bottom Inner Forging	The cup-shaped component that forms the bottom of the NAC-STC cavity.
Bottom Outer Forging	The ring-shaped component that forms the bottom outer region of the NAC-STC.
Bottom Plate	The plate welded to the bottom outer forging, which forms the bottom of the cask. The bottom plate encloses the neutron shielding material in the bottom of the cask.
Rotation Trunnion Recesses	Two stainless steel blocks, each provided with a deep machined groove suitable to accept the rear cask support. These recesses are welded onto the outer shell near the bottom of the cask.
Cask Cavity	The volume of space within the containment boundary.
Transportable Storage Canister (Canister)	The stainless steel cylindrical shell, bottom end plate, shield lid, and structural lid (or single closure lid for MPC-LACBWR) that holds the canistered fuel basket or Greater Than Class C Waste.
Canister Spacer	Structures that position the canister in the NAC-STC cavity during transport. The Yankee-MPC uses two spacers and the MPC-LACBWR uses three spacers constructed of aluminum honeycomb material encased in a shell constructed of 6061-T6 aluminum alloy. The CY-MPC uses a single spacer constructed of concentric rings of stainless steel welded to a stainless steel base plate. The MPC-WVDP uses two spacers constructed of concentric rings of stainless steel welded to a stainless steel base plate.
HLW Canister Overpack (HLW Overpack)	The stainless steel cylindrical shell, bottom end plate, and closure lid that holds the WVDP HLW canisters.

Table 1-1 Terminology (continued)

MPC-WVDP HLW Contents	Up to five (5) welded HLW canisters filled with vitrified, solidified high-level waste, or up to two (2) melter-evacuated canisters that are partially filled with glass, and one (1) HLW debris canister.
HLW Debris Canister	An HLW canister containing borosilicate glass and radioactive waste material within the glass matrix. The canister also contains refractory material (Alfibond 2800) and may contain alumina from melter inserts. The source of the refractory material are melter inserts. Alfibond 2800 is a non-organic insulator composed (99+%) of alumina (Al2O3) and silica (SiO2). No organic material is present in the insulator.
Personnel Barrier	An expanded metal screen with appropriate support structure that is installed between the impact limiters and covers the cask during transport. The expanded metal screen, and its support structure, are aluminum. The personnel barrier precludes incidental contact with the cask surface, which may be at elevated temperature compared to the rail car.
Lattice	A fuel assembly structure that is used to hold up to 204 Intact Fuel Rods or Damaged Fuel Rods from other fuel assemblies. A Lattice is sometimes called a fuel skeleton, cage or structural cage. It is built from the same components as a standard fuel assembly, but some of those components may be modified slightly, such as relaxed grids, to accommodate the distortion that may be present in a Damaged Fuel Rod. The outside dimensions are identical to a standard fuel assembly.

Table 1-1 Terminology (continued)

Failed Rod Storage Canister	A handling container for moving up to 60 individual intact or damaged fuel rods in stainless steel tubes into a CY-MPC Damaged Fuel Can. The steel tubes are held in place by regularly spaced plates welded in an open stainless steel frame. The failed rod storage canister, which is closed at the top end by a bolted closure and at the bottom by a welded plate to capture the fuel rods in the tubes, must be loaded in a CY-MPC Damaged Fuel Can.
Redwood Impact Limiter	A device constructed primarily of redwood with limited use of balsa wood. This device is designed to dissipate energy during normal and accident conditions impact events for packages weighing up to 250,000 lb. The redwood impact limiter is 124 inches in diameter with redwood providing primary protection during end and side impact events and balsa wood for corner impact events.
Structural Damage	Damage to the fuel assembly that does not prevent handling the fuel assembly by normal means. Structural damage is defined as partially torn, abraded, dented or bent grid straps, end fittings or guide tubes. The damaged grid straps or end fittings must continue to provide support to the fuel rods, as designed, and may not be completely torn or missing. Guide tubes cannot be ruptured and must be continuous between the upper and lower end fittings. Fuel assemblies with structural damage are considered to be intact fuel assemblies provided that they do not have failed or damaged fuel rods.

The CY-MPC Failed Rod Storage Canister, shown in Figure 1.2-6, is similar in design to the CY-MPC Reconfigured Fuel Assembly but holds only a maximum of 60 fuel rods classified as failed in stainless steel tubes.

Some of the Connecticut Yankee fuel assemblies will be stored with flow mixers, reactor control cluster assemblies (RCCA) or stainless steel rods installed. Flow mixers are thimble plug assemblies used during reactor operation to maintain equal coolant flow in fuel assemblies that do not contain a reactor control cluster. Reactor control clusters were used to control the reactivity of the Connecticut Yankee reactor during operations and shutdown. Some Connecticut Yankee fuel assemblies may have missing fuel rods and/or may have solid filler rods replacing the missing fuel rods.

A maximum of two assemblies with up to two irradiated stainless steel filler rods per assembly may be loaded in the CY-MPC canister. These assemblies may only be loaded in the two central basket positions.

In the HLW overpack configuration, two stainless steel spacers, one placed below and one above the HLW overpack, are used to position the HLW overpack so that the location of the center of gravity of the packaging is the same as it is for the uncanistered fuel packaging. The HLW overpack shell, bottom, and welded closure lid are fabricated from stainless steel. The basket for the HLW canister configuration is an assembly of five vertical cylindrical cells, each designed to accommodate a welded HLW canister, a melter-evacuated canister (EC) partially filled with glass, or an HLW debris canister.

Two impact limiter designs consisting of a combination of redwood and balsa wood, encased in Type 304 stainless steel, are provided to limit the g-loads acting on the cask during a drop accident load condition. The g-loads are limited by the crush strength of the wood contained in the impact limiters. The predominately balsa wood impact limiter (the balsa impact limiter) is designed for use with all the proposed contents. The predominately redwood impact limiters (the redwood impact limiter) may only be used with directly loaded fuel or the Yankee-MPC configurations.

Any number of NAC-STCs may be shipped at one time, with each cask on its own railcar. The NAC-STC may also be shipped in any number on board ships, barges, or special heavy-haul vehicles.

1.1.1 Licensing Approach (Directly Loaded High-Burnup PWR Fuel)

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

1.1.2 Statistical Derivation of Bounding DBTT Values (High-Burnup Fuel)

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

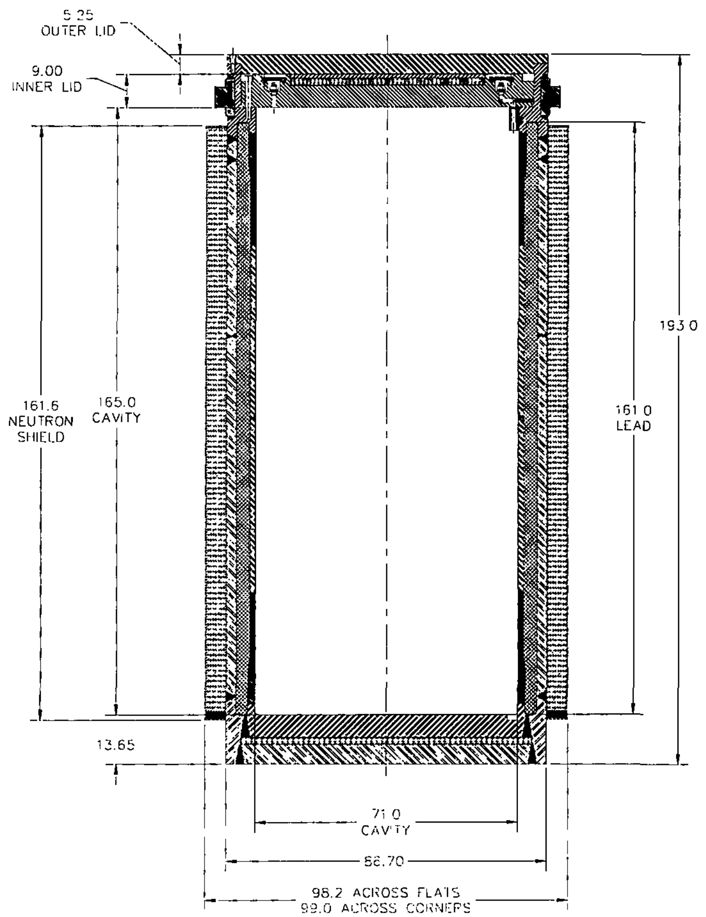
[redacted]

[redacted]

[redacted]

[redacted]

Figure 1.1-1 Major Cask Dimensions



Dimensions in inches

Figure 1.1-2 Deleted

(Deleted)

Figure 1.1-3 Hydrogen Concentration Precipitation-Dissolution Curves

[redacted]

Note: Figure 1.1-3 is from SAR Section 1.1.1 Reference [1].

Figure 1.1-4 STC-HBU Vacuum Drying Cladding Temperatures

[redacted]

Figure 1.1-5 Deleted

(Deleted)

Table 1.1-1 Maximum Variation of Fuel Cladding Temperatures

[redacted]

THIS PAGE INTENTIONALLY LEFT BLANK

1.2.1.1, the lightweight impact limiters must be used in the transport of the CY-MPC canister configurations. Either impact limiter design may be used with the directly loaded fuel or Yankee-MPC configurations, or WVDP-HLW Overpacks. The two impact limiter configurations are similar in size and shape. The redwood impact limiters have a total weight of 17,730 pounds. The total weight of the balsa impact limiters is 11,423 pounds.

1.2.1.2.7 Directly Loaded Fuel Basket

The NAC-STC has four (4) directly loaded fuel basket configurations and three (3) canistered fuel basket configurations. The canistered fuel basket configurations are described in Section 1.2.1.2.8. The directly loaded basket and the canistered baskets are constructed of stainless steel with aluminum heat transfer disks and are of similar design.

The directly loaded fuel basket design is a right-circular cylinder configuration with square fuel tubes laterally supported by a series of support disks, which are retained by square spacer nuts on threaded rods at six locations. The nuts are torqued at installation to provide a solid load path in compression between the support disks. The support disks are 0.5-inch thick, 70.86-inch diameter 17-4 PH stainless steel disks spaced 4.87 inches center-to-center with square holes for the fuel tubes. The top end weldment and the bottom end weldment are fabricated from Type 304 stainless steel, are geometrically similar to the support disks, and are 1.0-inch thick. The threaded rods have a 1-5/8 – 8 UN thread diameter and are fabricated from Type 17-4 PH stainless steel. The nuts are 2.5-inch square bars that are also fabricated from Type 17-4 PH stainless steel. The fuel tubes are fabricated from Type 304 stainless steel and provide support for the encased neutron absorber sheet on each of the four sides. The neutron absorber provides criticality control in the basket. No structural credit is taken for the stainless steel tubes as a contributor to the total structural strength of the basket and support of the fuel assemblies.

The NAC-STC directly loaded fuel basket accommodates 26 PWR fuel assemblies in an aligned configuration in 8.78-inch inside dimension square fuel tubes, which have 0.142-inch thick walls. The alternate aligned configuration has square fuel tubes with an inside dimension of 8.73 inches and a wall thickness of 0.167 inches. The fuel tubes are supported in the basket assembly between the top and bottom weldment plates. The hole in the top weldment is 8.75 inches square. The hole in the bottom weldment is 8.65 inch square. The basket design traps the fuel tube between the top and bottom weldment preventing axial movement of the fuel tube. The minimum width of the support disk webs between the fuel tubes is 1.5 inches, but two webs have a width of 3.3 inches.

Twenty Type 6061-T651 aluminum alloy heat transfer disks, 0.625-inches thick, 70.65 inches in diameter, are supported by the threaded rods and spacer nuts, which also support and locate the stainless steel support disks. These aluminum disks are located at the center of the axial spacing between the stainless steel support disks and are sized to eliminate contact with the cask inner shell and basket threaded rods as a result of differential thermal expansion.

The NAC-STC has been designed to facilitate filling with water and subsequent draining. A 1.0-inch rounded notch is located at the bottom of each fuel tube, ensuring that there will be free flow between the inner tube regions and the disk regions. Water will naturally fill and drain between the basket disks and the cask body. Water will also flow between the disks in the gap between each of the tubes and the disk that surrounds it. Each of the disks also has four 1-inch diameter holes to supplement the flow of water between the disks. Also, to facilitate flow to the drain line, the bottom plate is positioned by supports 1.5 inches above the bottom of the cask. These design features have been provided to ensure that there is a free flow of water in the cask basket that results in even filling and draining of the cask.

1.2.1.2.8 Transportable Storage Canister

There are two transportable storage canister configurations. One configuration is used for Yankee Class spent fuel and GTCC waste (Yankee-MPC). The other is used for Connecticut Yankee spent fuel and GTCC waste (CY-MPC). The canisters differ only in overall length and in the thickness of some components. The transportable storage canister consists of four (4) principle components. These are the canister, canister basket, shield lid and structural lid. The canister consists of an annular right-circular shell closed at one end by a bottom plate. The shell is constructed of 5/8-inch rolled Type 304L stainless steel plate. The edges of the rolled plates are joined with full penetration welds. The Type 304L stainless steel bottom plate is attached to the shell by also using a full penetration weld. The canister shell is constructed in accordance with ASME Code Section III, Subsection NB. The inside and outside diameter of the canister are 69.39 inches and 70.64 inches, respectively.

The overall external length of the Yankee-MPC canister is 122.5 inches, the inside depth is 121.5 inches and the bottom plate is 1.0-inch thick. The overall external length of the CY-MPC canister is 151.75 inches, the inside depth is 150.0 inches and the bottom plate is 1.75-inches thick.

After loading, the canister is closed by a shield lid and structural lid, each welded to the canister shell. The design of the shield lid and structural lid provides redundant confinement seals at the

R-122840

Security-Related Information Figure
Withheld Under 10 CFR 2.390

UNLESS OTHERWISE STATED	GROUP	NAME	DATE
NEXT ASSEMBLY 423-900			
DRAWING TYPE LICENSE			

NAC INTERNATIONAL

CASK ASSEMBLY-
NAC-STC CASK

423 | 800 | N.T.S. | N/A

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

CASK ASSEMBLY
NAC-STC CASK

PROJECT 423		DRAWING 800
SCALE N.T.S.	ESTWT. N/A	SH 2 OF 3

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

CASK ASSEMBLY
NAC-STC CASK

423 | 800
N.T.S. | N/A | 3 of 3

R-1 228406

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LID ASSEMBLY-
INNER,
NAC-STC CASK

GROUP	NAME	DATE
	V. Shtylman	5-15-15
		5-15-15
		5/18/15
		5/15/15
		5/19/15
		5/19/15

PROJECT 423 DRAWING 803 REV 10

SCALE N.T.S. EST. WT. N/A SH 1 of 2

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LID ASSEMBLY–
INNER,
NAC–STC CASK

PROJECT 423		DRAWING 803	REV 10
SCALE N.T.S	CSI N/A	SH 2 OF 2	

1R-228408

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LID ASSEMBLY,
OUTER,
NAC-STC CASK

GROUP	NAME	DATE
		5-15-15
		5-15-15
		5/19/15
		5/15/15
		5/19/15
		5/19/15

NEXT ASSEMBLY 423-800
DRAWING TYPE LICENSE

PROJECT 423 | DRAWING 805 | REV 7
SCALE 1/6 | NOTED | SH 1 OF 2

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

LID ASSEMBLY,
OUTER,
NAC-STC CASK

PROJECT 423
DRAWING 805
REV 7
SCALE 1/6
SHEET 2 OF 2

R-228410

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NEXT ASSEMBLY	423-803
DRAWING TYPE	LICENSE

GROUP	NAME	DATE
	V. Eltzelman	5-15-15
	B. Brown	5-15-15
	[illegible]	5-18-15
	[illegible]	5/15/15
	[illegible]	5/19/15
	Roy V. Bow	5/19/15

NAC INTERNATIONAL

PORT COVERPLATE ASSY.
INNER LID,
NAC-STC CASK

PROJECT	DRAWING	REV
423	806	8

SCALE 1/1 | COLOR NOTED | SH 1 OF 1

R-228412

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NEXT ASSEMBLY	423-800	
DRAWING TYPE	LICENSE	
GROUP	NAME	DATE
	V. Shtylman	5-15-15
		5-15-15
		5-18-15
		5/15/15
		5/19/15
	Ray W. Bau	5/19/15

NAC INTERNATIONAL

ASSEMBLY,
PORT COVER,
NAC-STC CASK

PROJECT	DRAWING	REV
423	807	4
SCALE 2.1	EST WT. 10LBS	SH 1 OF 3

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

ASSEMBLY,
PORT COVER,
NAC-STC CASK

PROJECT 423		DRAWING 807	REV 4
SCALE FULL	EST WT 10LBS	SH 2 OF 3	

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL			
ASSEMBLY, PORT COVER, NAC-STC CASK			
423		807	4
FULL	10LBS	3 of 3	

Security-Related Information Figure
Withheld Under 10 CFR 2.390

UNLESS OTHERWISE STATED

NEXT ASSEMBLY: 423-870

DRAWING TYPE: LICENSE

GROUP	NAME	DATE
		5-15-15
		5-15-15
		5/18/15
		5/15/15
		5/14/15
	Roy W. Bau	5/19/15

NAC INTERNATIONAL

TUBE,
NAC-STC CASK

PROJECT 423

DRAWING 875

REV 10

SH 1 OF 2

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

TUBE,
NAC-STC CASK

PROJECT	DRAWING	REV
423	875	10

SH 2 OF 2

R-228416

Security-Related Information Figure
Withheld Under 10 CFR 2.390

UNLESS OTHERWISE STATED	GROUP	NAME	DATE
NEXT ASSEMBLY 423-870			5-15-15
DRAWING TYPE LICENSE			5/19/15

NAC INTERNATIONAL

ALTERNATE TUBE ASSEMBLY, NAC-STC CASK

PROJECT 423 DRAWING 878

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

ALTERNATE TUBE ASSEMBLY,
NAC-STC CASK

PROJECT 423	DRAWING 878	REV 2	
	SH 2 OF 2		

ℓ-228420

Security-Related Information Figure Withheld Under 10 CFR 2.390

GROUP	NAME	DATE
PREPARER		5-15-15
CHECKER		5-15-15
PROJECT MANAGER		5/18/15
ENGINEERING		5/19/15
LICENSING		5/19/15
QUALITY		5.19.15

NAC INTERNATIONAL

TRANSPORT CASK ASSEMBLY, MPC-WVDP

NEXT ASSEMBLY: 423-843
DRAWING TYPE: LICENSE

PROJECT 630087 | DRAWING 501 | REV 1
SCALE N.T.S. | WEIGHT N/A | SH 1 OF 2

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

TRANSPORT CASK ASSEMBLY,
MPC-WVDP

PROJECT	DRAWING	REV
630087	501	1

R-226005

Security-Related Information Figure
Withheld Under 10 CFR 2.390

GROUP	NAME	DATE
PREPARER	[illegible]	7-18-14
CHECKER	[illegible]	7-18-14
PROJECT MANAGER	[illegible]	7/18/14
ENGINEERING	[illegible]	7-18-14
LICENSING	[illegible]	7/18/14
QUALITY	[illegible] Barrett	7/18/14

NEXT ASSEMBLY: N/A
DRAWING TYPE: LICENSE

NAC INTERNATIONAL

HLW OVERPACK ASSEMBLY, MPC-WVDP

PROJECT 630087 | DRAWING 510 | REV 1
SCALE N.T.S. | WEIGHT N/A | SH 1 OF 1 | 7/18/2014

R-226007

Security-Related Information Figure
Withheld Under 10 CFR 2.390

GROUP	NAME	DATE
PREPARED		7-18-14
CHECKER		7-18-14
PROJECT MANAGER		7/18/14
ENGINEERING		7-18-14
LICENSING		7/18/14
QUALITY		7/18/14

NEXT ASSEMBLY 630087-510

DRAWING TYPE LICENSE

NAC INTERNATIONAL

SHELL,
HLW OVERPACK,
MPC-WVDP

PROJECT 630087 DRAWING 511 REV 1

SCALE N.T.S WEIGHT N/A SH 1 OF 1

R-226009

Security-Related Information Figure
Withheld Under 10 CFR 2.390

GROUP	NAME	DATE
PREPARER	[illegible]	7-24-14
CHECKER	[illegible]	7-24-14
PROJECT MANAGER	[illegible]	7/24/14
ENGINEERING	[illegible]	7-24-14
LICENSING	[illegible]	7/24/14
QUALITY	B Barnett	7/24/14

NEXT ASSEMBLY 630087-510

DRAWING TYPE LICENSE

NAC INTERNATIONAL

CLOSURE LID,
HLW OVERPACK,
MPC-WVDP

PROJECT	DRAWING	REV
630087	513	1

SCALE N T S — WEIGHT N/A — SH 1 OF 3

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

CLOSURE LID,
HLW OVERPACK,
MPC-WVDP

PROJECT 630087		DRAWING 513	REV 1
SCALE N.T.S.	WEIGHT N/A	SH 2 OF 3	

Security-Related Information Figure
Withheld Under 10 CFR 2.390

NAC INTERNATIONAL

CLOSURE LID,
HLW OVERPACK,
MPC-WVDP

PROJECT 630087		DRAWING 513	REV 1
SCALE N.T.S.	WEIGHT N/A	SH 3 OF 3	[illegible]

Table of Contents (continued)

Table of Contents (continued)

Table of Contents (continued)

List of Figures

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Figures (Continued)

List of Tables (Continued)

List of Tables (Continued)

List of Tables (Continued)

2.12 Structural Evaluation – STC-WVDP Cask with HLW Overpack and Spacers

2.12.1 Structural Design – HLW Overpack and Spacers

2.12.1.1 Discussion

For the HLW canisters, the HLW Overpack serves to retain the five HLW canisters. The HLW Overpack has a capacity of up to 5 HLW canisters. 275 HLW canisters are to be shipped in this configuration. In one shipment the HLW Overpack will contain two melter-evacuated (EC) canisters, and a single HLW debris canister contents. All contents of the HLW Overpack are referred to as the HLW contents. The HLW Overpack is designed to incorporate a 5-cell basket for the HLW canisters. The basket is only used to position the HLW canisters during the atmospheric dry loading of the Overpack. No structural credit is assigned to the basket. The Overpack consists of a cylindrical shell with a welded bottom plate and a closure lid. For transport in the NAC-STC cask there are two spacer assemblies; one spacer is positioned below the HLW Overpack and the second spacer is positioned above the HLW Overpack. The spacer assemblies serve to maintain the location of the HLW Overpack within the NAC-STC transport cask. The HLW Overpack is not designed to be a separate leak tight containment for the HLW canisters since the NAC-STC cask body and closure provides the transport containment boundary. Therefore the HLW Overpack is only evaluated for normal conditions of transport.

2.12.1.2 Design Criteria

For transport of the HLW Overpack, the primary containment of the NAC-STC package is not changed. Since the HLW Overpack is not a containment boundary, it is designed to meet ASME Section VIII, Division 2 criteria. The HLW Overpack assembly is evaluated for normal conditions of transport. The top and bottom spacers are designed to ASME Section III, Subsection NF and are evaluated for normal and accident conditions of transport since the function of the spacers is to maintain the location of the HLW Overpack in the middle section of the NAC-STC transport cask.

2.12.1.3 Miscellaneous Structural Failure Modes

For the HLW Overpack, the shell weldment and closure lid are constructed of Type 304/304L stainless steel. There are two spacer assemblies used for transport of the HLW Overpack in the NAC-STC cask. One spacer is positioned below the HLW Overpack and the second spacer is positioned above the HLW Overpack. The two spacers are constructed of Type 304 stainless steel. Type 304 stainless steel is an austenitic stainless steel which does not undergo a ductile-to-brittle transition in the temperature range of interest for transport. Therefore, brittle fracture is not a concern.

THIS PAGE INTENTIONALLY LEFT BLANK

Condition 2. -20°F ambient temperature, with maximum decay heat load, and no solar insolation.

Condition 3. -40°F ambient temperature, with no decay heat load, and no solar insolation.

The cask temperatures calculated for each of the three thermal conditions discussed above are used in the ANSYS structural analyses to determine the values of the temperature-dependent material properties.

Additional heat transfer analyses were performed for the Yankee-MPC canistered fuel configuration as described in Section 2.6.7.1.

4. Impact loads - The impact loads are induced by the impact limiter acting on the cask end during an end drop condition. The impact loads are determined from the energy absorbing characteristics of the impact limiters, as described in Section 2.6.7.4. The impact load is expressed in terms of the design cask weight (loaded or empty), multiplied by appropriate deceleration factors (g's). For details, see Section 2.6.7.4.

The impact limiter load is considered to be uniformly applied over the end surface of the finite element model of the cask. The calculation of impact pressure loads is documented in Section 2.10.2.2.2. The following is a summary of the impact pressures applied to the exterior surface of the impacting end, for the different loading scenarios, with the corresponding design deceleration (g) values.

LOADING CONDITION	IMPACT PRESSURE for 1g	DECELERATION (g)
End impact with basket and fuel	42.35 psi	56.1
End impact with basket, no fuel	35.74 psi	49.4

For the end impact, with basket and fuel, a uniform pressure of 2376 psi ([42.35 psi][56.1 g/1g]) is applied on the exterior surface of the end of the finite element model of the cask. This pressure value is calculated by dividing the total impact load ([56.1g/1g][250,000 lb] = 14.03×10^6 lb) by the impact area (p $\times$ $(43.35)^2$ = 5903.8 in^2), which is the surface area of the end of the cask.

It should be noted that the design weight of the cask is 250,000 pounds, which includes the weight of the empty cask with impact limiters (194,000 lb), plus the weight of the cavity contents (56,000 lb) for the directly loaded fuel configuration. For those load conditions for which the cask contains no fuel, the basket (design weight = 17,000 lb) is still considered to be in the cask, resulting in a weight of 211,000 pounds for the empty cask with basket. The weights of the cavity contents for the Yankee-MPC canistered fuel and the Yankee-MPC canistered GTCC waste configurations are 55,590 pounds and 54,271 pounds, respectively.

5. Inertial body load - The inertial effects, which occur during the end impact, are represented by equivalent static forces, in accordance with D'Alembert's principle. Inertial body load includes the weight of the empty cask (194,000 lb) and the weight of the cavity contents (56,000 lb) for the directly loaded fuel configuration, which envelopes that of the Yankee-MPC canistered fuel or the Yankee-MPC canistered GTCC waste.

   Inertia loads resulting from the weight of the empty cask are imposed by applying an appropriate deceleration factor to the cask mass. The applied decelerations are determined by considering the crush strength and the geometry of the impact limiters, as explained in Section 2.6.7.4.

   The inertial load resulting from the 56,000-pound contents design weight is represented as an equivalent static pressure load uniformly applied on the interior surface of the impacting end of the cask. For the load case with no fuel in the cavity, the basket (design

represented by the Water Spray Condition evaluation in Section 2.6.6. No further evaluation is required.

2.12.6.7 Free Drop (1 Foot)

The free drop scenario outlined by 10 CFR 71.71(c)(7) requires the NAC-STC to be structurally adequate for a 1-foot drop (normal transport conditions) onto a flat, essentially unyielding horizontal surface in the orientation that inflicts the maximum damage to the cask. The following subsections evaluate the cask body, the impact limiters, the closure lid and bolts, the neutron shield shell, and the upper ring components; for the end, side, and corner drop orientations.

2.12.6.7.1 One-Foot End Drop

The NAC-STC cask in the STC-WVDP configuration is bounded by the One-Foot End Drop Condition evaluation of the NAC-STC cask in the Directly Loaded Fuel and Yankee-MPC Canistered Fuel Configurations in Section 2.6.7.1.1 (see discussion in Section 2.12.6). No further evaluation is required.

2.12.6.7.2 One-Foot Side Drop

The NAC-STC cask in the STC-WVDP configuration is bounded by the One-Foot Side Drop Condition evaluation of the NAC-STC cask in the Directly Loaded Fuel and Yankee-MPC Canistered Fuel Configurations in Section 2.6.7.2.1 (see discussion in Section 2.12.6). No further evaluation is required.

2.12.6.7.3 One-Foot Corner Drop

The NAC-STC cask in the STC-WVDP configuration is bounded by the One-Foot Corner Drop Condition evaluation of the NAC-STC cask in the Directly Loaded Fuel and Yankee-MPC Canistered Fuel Configurations in Section 2.6.7.3.1 (see discussion in Section 2.12.6). No further evaluation is required.

2.12.6.7.4 Impact Limiters

Removable transport impact limiters are included in the NAC-STC design to ensure that the design impact loads on the cask are not exceeded for any of the defined impact load conditions. The defined loading conditions include the cask falling 1 foot or 30 feet and landing on its side impacting both impact limiters simultaneously, or landing vertically on one impact limiter at either end. As discussed in Section 2.10.7, the oblique impact events are bounded by the end and/or side drop events, as there is no "slap down" effect exhibited by the cask. Consequently, oblique drops and oblique drop angles are not evaluated.

Two impact limiter configurations have been designed for the NAC-STC cask. The first impact limiter configuration is primarily redwood with a small amount of balsa wood (redwood impact limiter). The wood is encased in a stainless steel shell. The redwood impact limiters may be used only with the directly loaded fuel configuration or the Yankee-MPC canistered fuel and GTCC waste configurations. The second impact limiter configuration is primarily balsa wood with a limited amount of redwood (balsa impact limiter). The wood is also encased in a stainless steel shell. The balsa impact limiters may be used with the directly loaded fuel, Yankee-MPC fuel or GTCC waste, the CY-MPC fuel or GTCC waste, STC-LACBWR fuel or the HLW Overpack configurations.

2.12.6.7.4.1 Balsa Impact Limiters

The lower design basis weight (236,000 lbs) of the STC-WVDP Cask is attributed to the lower contents weight of 45,800 lbs versus the total fuel basket weight of 67,195 lbs in the licensed design basis weight (260,000 lbs) of CY-MPC Cask.

The reduction in the contents weight of 21.4 kips also means that the total kinetic energy which must be absorbed is also reduced. Since crush strength curves for balsa wood and redwood monotonically increase with strain, the resulting strain, and crush strength could be reduced. The bounding condition is the cold condition for which the properties have been evaluated at -40F. Additionally, for the cold condition, the properties of the balsa wood have been factored by another 10% to account for uncertainties, one of which could be considered to be the reduction of the weight by approximately 10%. If the acceleration of the cask is increased by 11%, the original CY-MPC acceleration of 39.9g increases to 44.2g. The bounding condition for the stresses developed in the cask are those for the top end drop in which the inertial loading due to the contents weight is applied to the lid. The inertial loading applied to the closure lid for the top end drop is the acceleration times the contents weight. A comparison of the inertial loading to the closure lid for the two configurations is shown below. The inertial loading applied to the closure lid associated the total cask weight configuration of 260,000 pounds is bounding by more than 30% over the loading associated with the STC-WVDP configuration.

Comparison of the Inertial Loading Applied to the Closure Lid for the Top End Drop

	CY-MPC canistered Fuel	STC-WVDP HLW Contents
Total contents weight (lb)	67,195	45,800
End Drop Acceleration (g's)	39.9	44.2
Inertial load applied to the lid (kips)	2,681	2,024

Likewise, for the side drop, this means that the total kinetic energy which must be absorbed is also reduced due to the reduction in the contents weight. While for the end drop, the backed area

for the balsa wood (and therefore the cross sectional area for the crushed balsa wood) remains the same, the crush area for the side drop does changes with the crush depth. To evaluate this, the finite element model used for the 260,000 pound design basis weight was altered to 236,000 and the resulting acceleration increased from 48.5 g to 51.9g. The inertial loading applied to the cask shells is determined by the product of the contents weight and the acceleration, which is compared in the table below. The inertial loading applied to the cask shells associated the total cask weight configuration of 260,000 pounds is bounding by more than 35% over the loading associated with the STC-WVDP configuration.

Comparison of the Inertial Loading Applied to the Cask Shells for the Side Drop

	CY-MPC canistered Fuel	STC-WVDP HLW Contents
Total contents weight (lb)	67,195	45,800
Side Drop Acceleration (g's)	48.5	51.9
Inertial load applied to the lid (lb.)	3,259	2,377

This confirms that the total inertial loads from the cask contents for the end drop and side drops with lighter weights are bounded by CY-MPC design basis weight of 260,000 pounds.

2.12.6.7.5 Closure Analysis – Normal Conditions of Transport

The cask body evaluation used a contents weight of 56,000 pounds which also bounds the contents weight of the STC-WVDP configuration of 41,825 pounds. Temperatures employed for the allowable also bound the cask temperatures associated with the HLW Overpack heat load of 1.5 kW. The closure analysis of the NAC-STC cask in Section 2.6.7.5 is adequate. No further evaluation is required.

2.12.6.7.6 Neutron Shield Analysis

The canistered contents have an insignificant structural effect on the neutron shield. Temperatures employed for the allowable also bound the cask temperatures associated with the HLW Overpack heat load of 1.5 kW. The NAC-STC cask in the STC-WVDP configuration is bounded by the neutron shield analysis of the NAC-STC cask in Section 2.6.7.6. No further evaluation is required.

2.12.6.7.7 Upper Ring/Outer Shell Intersection Analysis

The cask body evaluation used a contents weight of 56,000 pounds, which also bounds the contents weight of the MPC-WVDP configuration. The NAC-STC cask in the MPC-WVDP configuration is bounded by the upper ring/outer shell intersection analysis of the NAC-STC cask in Section 2.6.7.7. No further evaluation is required.

2.12.6.8 Corner Drop (1 Foot)

According to 10 CFR 71.71(c)(8), this test is not applicable to the NAC-STC because the cask is composed of materials other than fiberboard or wood and the cask weight exceeds 220 pounds (100 kg).

2.12.6.9 Compression

According to 10 CFR 71.71(c)(9), this test is not applicable to the NAC-STC because the package weight is greater than 5,000 kilograms (11,023 lb).

2.12.6.10 Penetration

This condition is defined in 10 CFR 71 as a 40-inch drop of a 13-pound, 1.25-inch diameter penetration cylinder with a hemispherical end onto any exposed surface of the cask. The acceptance criteria is that the drop will not adversely affect the ability of the cask to maintain containment of the contents or to survive a hypothetical accident. The impact limiters, the neutron shield, and the port covers could potentially be damaged by this penetration impact.

The NAC-STC cask in the STC-WVDP configuration is bounded by the Penetration Analyses of the NAC-STC cask in Section 2.6.10. No further evaluation is required.

2.12.6.11 Fabrication Conditions

The NAC-STC cask in the STC-WVDP configuration is bounded by the Fabrication Conditions evaluation of the NAC-STC cask in Section 2.6.11. No further evaluation is required.

2.12.6.12 STC-WVDP Analysis – Normal Transport Conditions

The NAC-STC has four contents configurations – uncanistered (directly loaded fuel) and canistered (fuel or Greater Than Class C [GTCC] waste) and the MPC-LACBWR canister configuration. This section describes the evaluation of the STC-WVDP configuration. The Overpack is evaluated for the Normal Conditions of Transport in this section.

The STC-WVDP configuration consists of an HLW Overpack assembly with bottom and top spacers to properly locate the canister within the NAC-STC cask cavity. The analysis of the supplemental spacers is presented in Section 2.12.6.13. The principal components of the HLW Overpack assembly are the shell assembly, basket and the closure lid. Detailed descriptions of the geometries and materials of construction of the HLW Overpack and basket are provided in Section 1.2.1.2.9.

2.12.6.12.1 HLW Overpack Assembly Analysis Description

The HLW Overpack assembly contains up to 5 HLW canisters positioned in the fuel basket. The HLW Overpack assembly is not considered to be a separate inner container for containment

during transport. The NAC-STC provides the primary containment boundary for transport. A transfer cask serves as the handling component for the Overpack assembly, basket and contents during loading and transfer of the Overpack for storage and/or transport. The HLW Overpack is a right-circular shell fabricated from rolled 3/8-inch thick, Type 304/304L stainless steel plate and closed at the bottom end by a circular 2.0-inch thick, Type 304/304L stainless steel plate that is welded to one end of the shell. The canister is closed at the top end by the installation and welding of the 4-inch thick, Type 304/304L stainless steel closure lid. The HLW Overpack encloses the HLW canisters. The empty HLW Overpack with basket is handled using lifting lugs located on the top end of the HLW Overpack shell. The loaded HLW Overpack is lifted using six hoist rings threaded into the top of the closure lid.

Because the HLW Overpack is not considered to be a secondary containment boundary, the structural design criterion used for the HLW Overpack is ASME Code Section VIII, Division 2. Consistent with this criteria, the structural components of the HLW Overpack are shown to satisfy the allowable stress limits presented in Tables 2.12.6.12-1 (while ASME Section VIII uses the nomenclature of S instead of Sm as in ASME Section III, the two quantities are numerically the same). The allowable stresses used in this analysis are based on a bounding maximum material temperature of 300°F for all locations in the canister. This temperature bounds the maximum temperature in the HLW Overpack as reported in Section 3.7.

Table 2.12.6.12-1 Structural Design Criteria for HLW Overpack

Stress Category	Allowable Equivalent Stress for Normal Conditions of Transport
Primary Membrane	1.0 S_m
Primary Membrane + Primary Bending	1.5 S_m
Range of Primary + Secondary	S_{ps}
Shear	0.6 S_m

The HLW Overpack is analyzed using the ANSYS finite element computer program for the 1-foot drop condition in the end and side impact orientations.

2.12.6.12.2 Finite Element Model Descriptions – HLW Overpack

The loading for the normal operating condition is based on 1-foot drops. Drop orientations considered are the end (top and bottom) and side drops. The operational conditions also contain loads developed from the thermal gradients in the HLW Overpack and due to internal pressure. These are included in the HLW Overpack model analyses as separate analyses. The results are then combined with the maximum stresses due to the drop cases.

End Drop Model

A three-dimensional half-symmetry finite element model of the HLW Overpack is constructed using ANSYS solid elements (SOLID45) for the structural analysis. The model represents a one-half (180°) section of the HLW Overpack.

For the bottom and top end drop cases, the HLW Overpack model was constrained vertically on either the bottom surface of the bottom plate or the top surface of the closure lid respectively. Additional boundary conditions were applied to the cut boundaries of the model to enforce symmetry.

In the end drop orientation, the STC-WVDP contents load is applied to the HLW Overpack assembly ends (bottom plate or closure lid) as an equivalent pressure load.

Figure 2.12.6.12-1 is a plot of the HLW Overpack finite element model. The HLW Overpack bottom plate portion of the model is shown in Figure 2.12.6.12-2. A view of the HLW Overpack closure lid with the weld region is shown in Figure 2.12.6.12-3.

Figure 2.12.6.12-1 HLW Overpack Assembly – End Drop Models

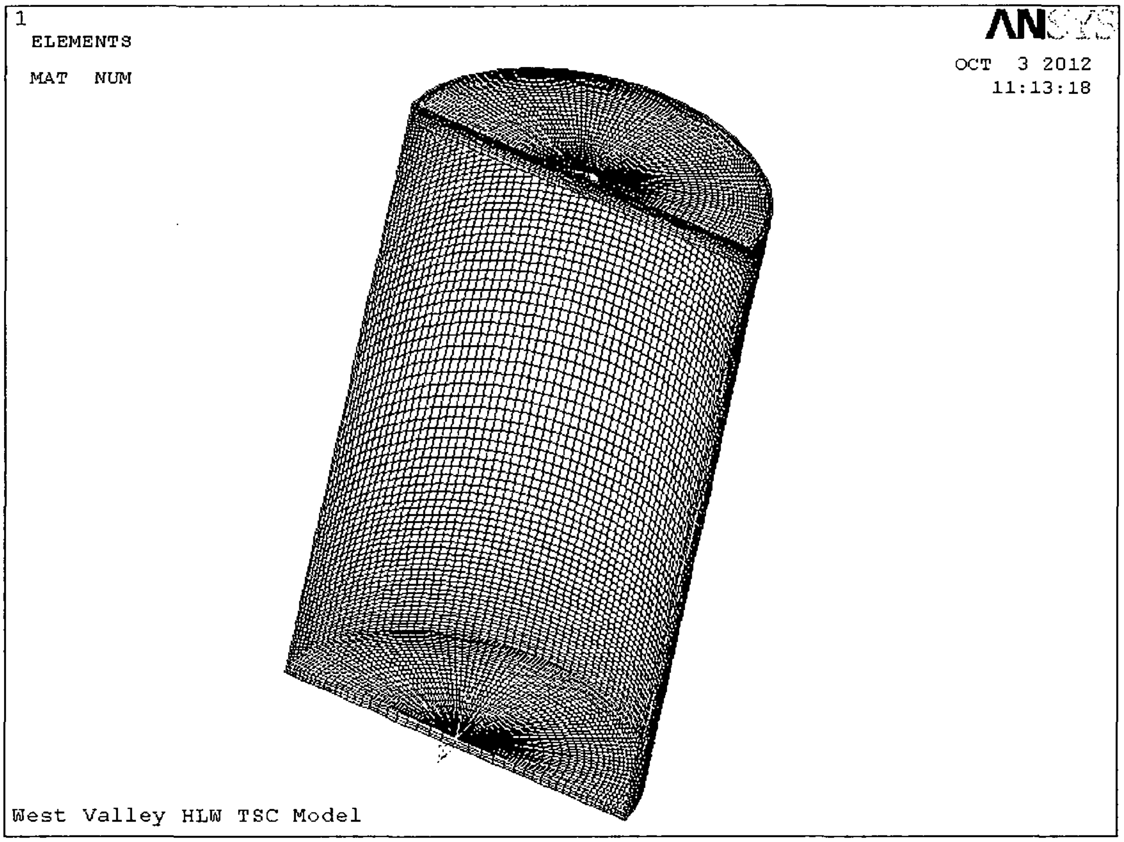


Figure 2.12.6.12-2 HLW Overpack Assembly – End Drop Models

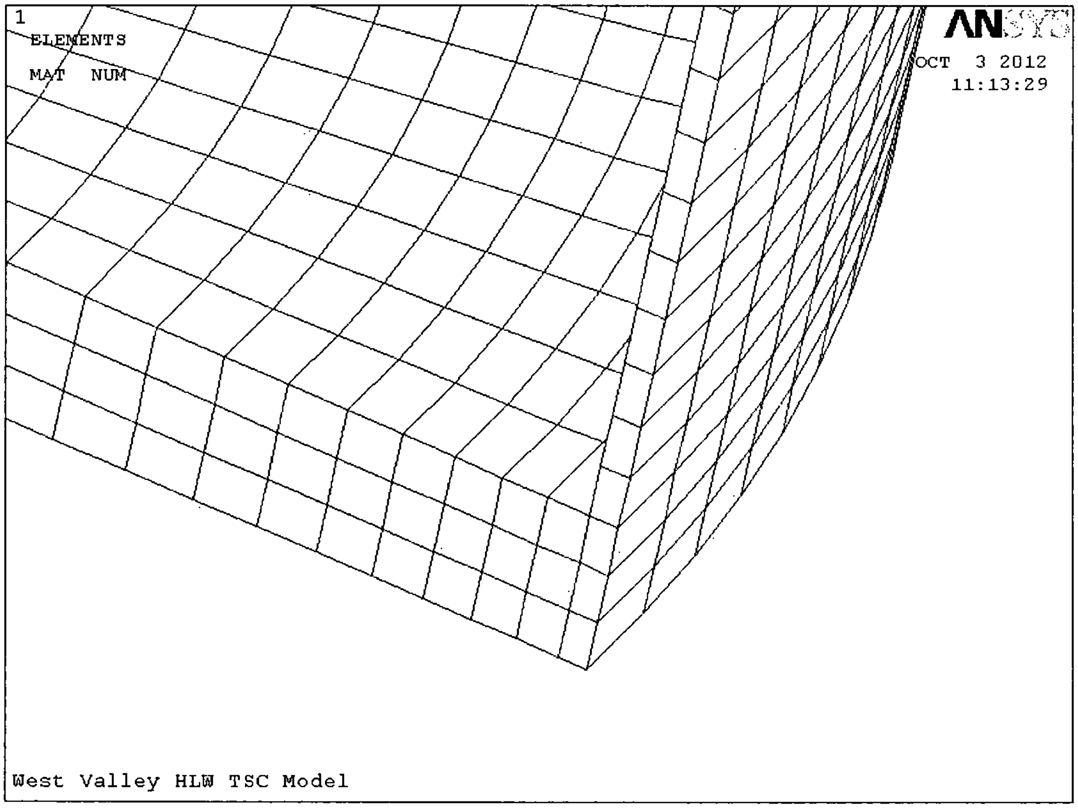
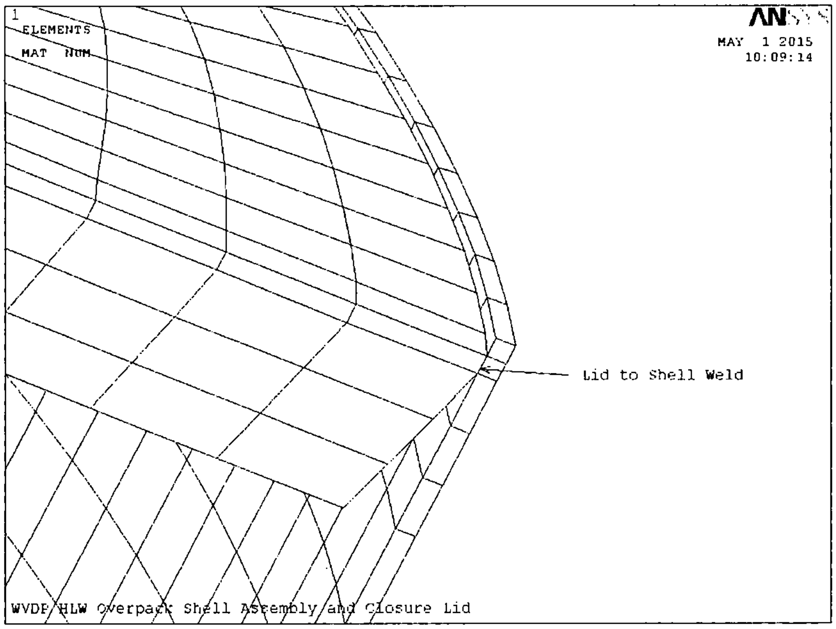


Figure 2.12.6.12-3 HLW Overpack Assembly – End Drop Models



Side Drop Model

For the side drop, the HLW Overpack end drop model was modified to include the inner liner of the NAC-STC cask for a more realistic representation of the interaction between the HLW Overpack Shell assembly and the NAC-STC cask. Interaction between the HLW Overpack assembly and NAC-STC inner liner was accomplished using three-dimensional surface-to-surface contact elements (CONTA170 and CONTA174). All contact elements are assigned a stiffness ratio of 1, unless otherwise noted. Note that the 170/174 contact pair calculates the gap stiffness based on the stiffness of the underlying solid elements. The side drop model is shown in Figure 2.12.6.12-4.

Boundary conditions were applied to enforce symmetry at the cut boundary of the model. All nodes on the outer surface of the NAC-STC inner liner were fixed in all degrees of freedom. An equivalent acceleration of 16.5 g was applied in the lateral direction in accordance with Table 2.6.7.4.2-3.

For the side drop condition, the loads from the HLW Overpack contents weight are transferred through distributed pressure loads into the HLW Overpack assembly wall. The HLW Overpack assembly wall is backed by the NAC-STC inner shell. Since the HLW Overpack wall and the inner shell have different radii, a gap exists between the two surfaces. This results in the load passing only through regions in which the HLW Overpack shell deflects enough to contact the inner shell. This load pattern is reflected in the side drop analysis.

Figure 2.12.6.12-4 HLW Overpack Assembly and NAC-STC Inner Liner – Side Drop Model

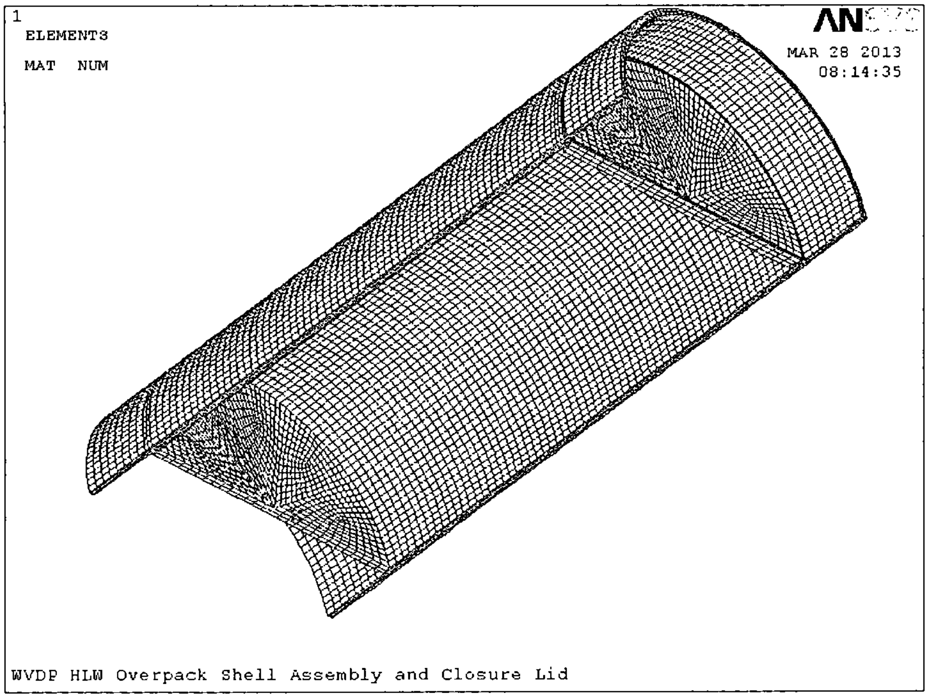
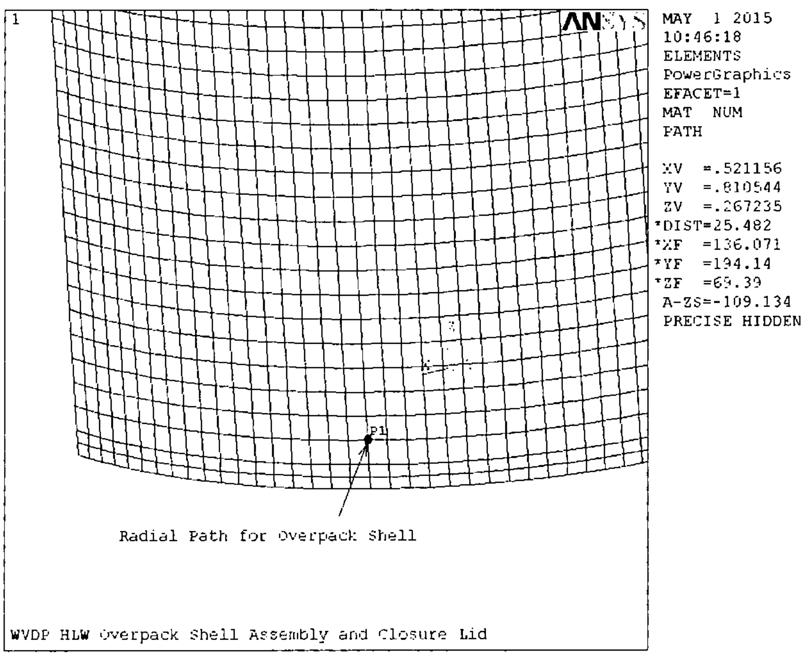


Figure 2.12.6.12-5 HLW Overpack Assembly– Side Drop Model



Internal Pressure Model

The same finite element model used for the end drop cases as shown in Figure 2.12.6.12-1 to Figure 2.12.6.12-3 was also used to evaluate the internal pressure case. The same boundary conditions were used as the bottom end drop but the equivalent pressure on the top of the bottom plate representing the weight of the STC-WVDP contents was deleted and the vertical acceleration was set to zero. Pressure is applied to all internal surfaces of the HLW Overpack to evaluate the pressure induced stresses.

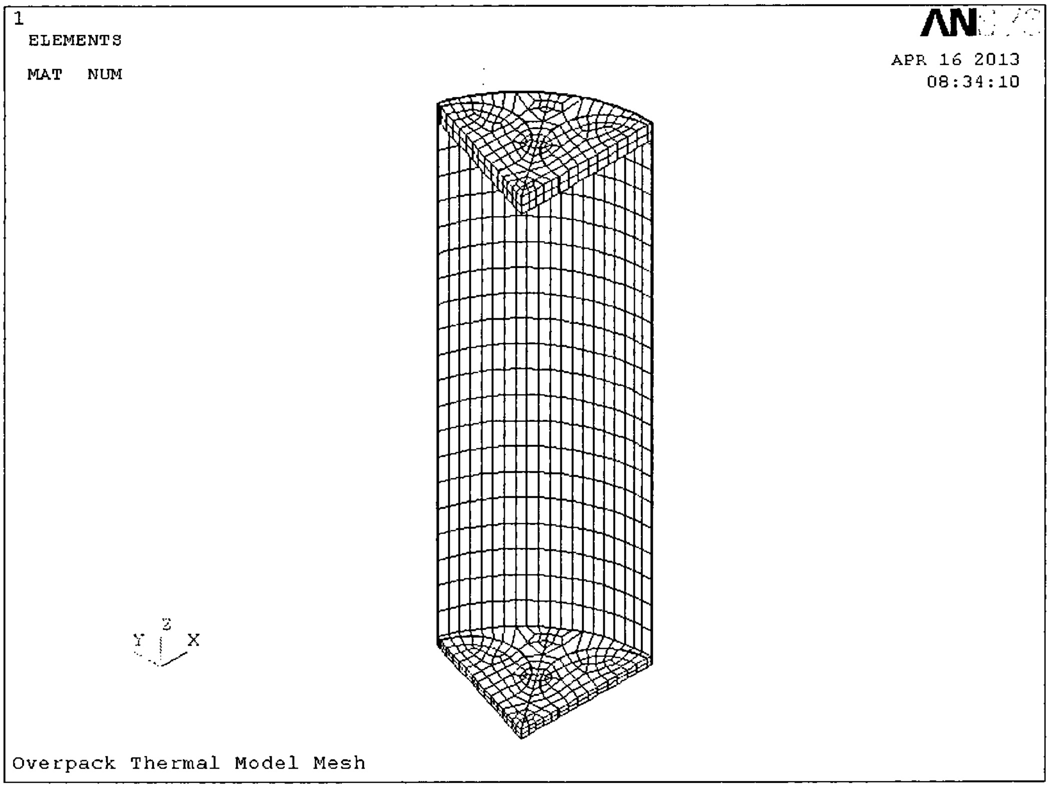
Thermal Stress Model

The thermal stress analyses of the HLW Overpack were performed with temperature distributions corresponding to the hot (100°F ambient with solar heat load) and cold (-40°F ambient) external conditions. The finite element model for the thermally induced stress was extracted from the thermal model discussed In Section 3.7.

The HLW Overpack assembly thermal mesh is shown in Figure 2.12.6.12-6. This mesh was extracted from the overall system thermal model reported in Section 3.7. The thermal system model included the HLW canisters, basket (since it acts as a thermal shield), HLW Overpack assembly and NAC-STC cask assembly. After the HLW Overpack assembly mesh was extracted, the thermal mesh was converted to a structural mesh and structural material properties were defined.

Symmetry boundary conditions were applied to the two cut planes of the model and all nodes on the bottom of the bottom plate were constrained in the vertical direction. Then the temperatures were read in from the thermal solution and used to evaluate the thermally induced stresses.

Figure 2.12.6.12-6 HLW Overpack Assembly – Thermal Model



2.12.6.12.3 Stress Evaluation of HLW Overpack – Normal Condition of Transport

2.12.6.12.3.1 End Drop Cases – 1 Foot Drop

An equivalent acceleration of 20 g was applied in the vertical direction in accordance with Table 2.6.7.4.2-3. Note this is a conservative acceleration since Table 2.6.7.4.2-3 gives a maximum acceleration of 14.5g. The weight of the HLW Overpack contents was applied as an equivalent pressure load on the appropriate interior surface.

The peak nodal equivalent stress is 9.3 ksi (ASME Section VIII uses the equivalent stress, von Mises stress, as opposed to the stress intensity in ASME Section III) for the bottom end drop case and 6.1 ksi for the top end drop case. Since the peak nodal equivalent stresses calculated were significantly less than the primary membrane stress limit of Sm = 20 ksi for type 304/304L stainless steel at 300 °F, stress linearization was not performed. For additional details refer to item 1 in Section 2.12.6.14.

2.12.6.12.3.2 Side Drop Case – 1 Foot Drop

For the side drop case an equivalent acceleration of 16.5 g was applied in the lateral direction in accordance with Table 2.6.7.4.2-3. Simplified analysis for the basket assembly indicated that contact between the basket and HLW Overpack only occurs at two locations. The weight of the HLW Overpack contents was applied as an equivalent pressure load on two regions of the interior surface which correspond to the contact of the HLW canisters with the HLW Overpack. For additional details refer to item 1 in Section 2.12.6.14.

The evaluation of the side drop case showed a localized peak stress of 51 ksi on the end of the HLW Overpack. This stress region is local to the edge and the local stress levels rapidly decay below the yield strength of Type 304/304L stainless steel at 300 °F. The stress distribution in this area is shown in Figure 2.12.6.12-7 and a plot of the stress in the circumferential direction is shown in Figure 2.12.6.12-8. In accordance with the stress criteria listed in Table 5.6 of the ASME Boiler and pressure Code, Section VIII, Division 2, the bending stress at the junction of a cylindrical shell and a flat head is classified as a secondary stress. Therefore, the local stress concentration was classified as a secondary stress. The secondary stress limit for 304/304L stainless steel at 300 °F is 3Sm = 60ksi. This corresponds to a margin of safety of +0.18.

To determine primary stresses, the lower section of the HLW Overpack outside of this region was selected and the stress levels were plotted and listed to determine the highest stress level outside of the peak stress region. After the highest stress in this region was determined, a path was defined from the inner surface to the outer surface. This path is shown in Figure 2.12.6.12-9. At the maximum stress location outside of the peak stress region, the linearized equivalent

stresses were 13.7 ksi for membrane stress and 17.5 for membrane plus bending stress. The margins of safety for membrane stress is +0.46 and for membrane plus bending is +0.71 using Sm = 20 ksi for type 304/304L stainless steel at 300 °F. For additional details refer to item 1 in Section 2.12.6.14.

Figure 2.12.6.12-7 Localized Stress Distribution at Shell to Bottom Plate

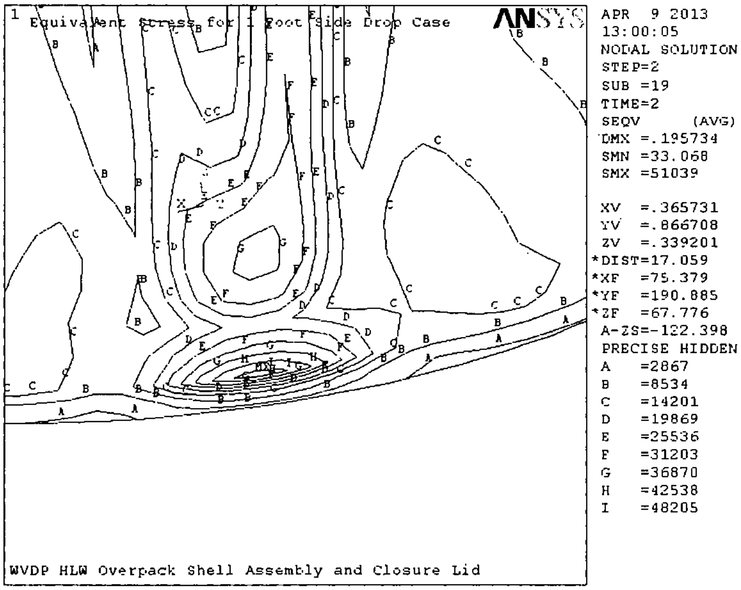
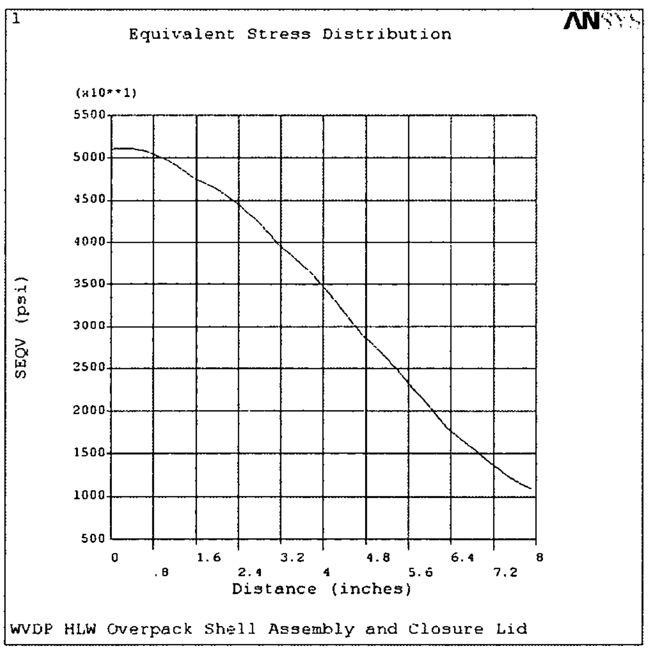
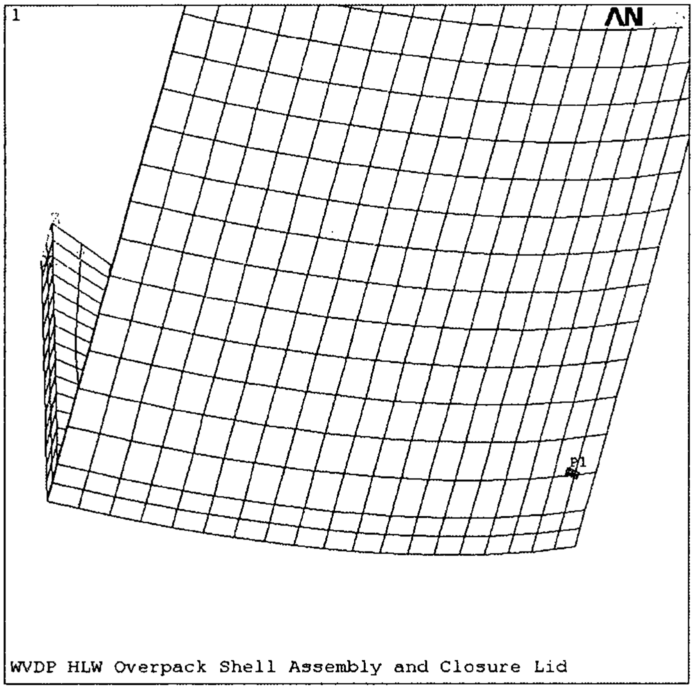


Figure 2.12.6.12-8 Localized Stress Distribution at Shell to Bottom Plate



2.12.6.12-9 Path for Linearized Stress



2.12.6.12.4 Internal Pressure Case

The HLW Overpack was evaluated for an internal pressure of 12 psig which was the maximum pressure calculated for the maximum average gas temperature of 493 °F for the fire transient from Section 3.7. Although this analysis configuration is due to an accident condition it is conservatively compared to normal condition stress limits. Linearization of the equivalent stresses due to internal pressure case gave a maximum membrane stress of 0.39 ksi and maximum membrane plus bending stress of 1.78 ksi. The margins of safety for internal pressure only are +50.3 for membrane stress and +15.9 for membrane plus bending. For additional details refer to item 1 in Section 2.12.6.14.

2.12.6.12.5 Thermal Cases

As discussed in Section 2.12.6.12.2 the thermal mesh was converted to a structural mesh and used to evaluate the thermally induced stresses. The thermal stresses are not directly compared to stress limits since thermal effects are to be combined with other operating loads.

2.12.6.12.5.1 Hot Case – Ambient Temperature of 100 °F

Evaluation and linearization of the equivalent stresses due to the hot thermal case gave a maximum membrane stress of 2.53 ksi and maximum membrane plus bending stress of 2.78 ksi. For additional details refer to item 1 in Section 2.12.6.14.

2.12.6.12.5.2 Cold Case – Ambient Temperature of -40 °F

Evaluation and linearization of the equivalent stresses due to the cold thermal case gave a maximum membrane stress of 3.36 ksi and maximum membrane plus bending stress of 3.76 ksi. For additional details refer to item 1 in Section 2.12.6.14.

2.12.6.12.6 Combined Stress Case

The maximum stress due to the 1 foot drop cases (including the maximum local stress concentration) is 51.0 ksi for the side drop case, the maximum stress for the internal pressure case is 0.48 ksi and the maximum stress for the thermal cases is 3.76 for the cold case. Conservatively adding the maximum equivalent stresses gives a combined stress of 55.2 ksi. The margin of safety for the combined stress is +0.09 using a limit of 3Sm = 60ksi at 300 °F for 304/304L stainless steel. For additional details refer to item 1 in Section 2.12.6.14.

2.12.6.13 Structural Evaluation – NAC-STC-MPC-WVDP Assembly Spacers

2.12.6.13.1 Discussion

Two spacer assemblies are used to maintain the position of the HLW Overpack within the NAC-STC transport cask. The bottom spacer is inserted into the NAC-STC cask prior to loading of the

HLW Overpack. The top spacer is inserted after the HLW Overpack is loaded into the NAC-STC cask. The spacers are sized such that the CG of the HLW Overpack inside of the cask is approximately the same as the Yankee-MPC TSC. The two spacer assemblies are identical except for the overall height. The bottom spacer assembly has a height of 14.7 inches and the top spacer has a height of 23.1 inches. Both spacer assemblies are constructed of Type 304 stainless steel.

2.12.6.13.2 Design Criteria

The spacer assemblies are required to maintain the axial location of the HLW Overpack within the NAC-STC cask. This insures that the CG of the STC-WVDP configuration is not altered during transport. The spacers are not required for criticality control, therefore, the spacers are designed to ASME Section III, Subsection NF and are evaluated for both normal and accident conditions.

Table 2.12.6.13.2-1 Structural Design Criteria for Spacer Assemblies

Stress Category	Allowable Stress Intensity for Normal Conditions of Transport	Allowable Stress Intensity for Accident Conditions of Transport
Primary Membrane	1.0 S_m	Lessor of 2.4 S_m and 0.7 S_u
Primary Membrane + Primary Bending	1.5 S_m	Lessor of 3.6 S_m and 1.0 S_u
Range of Primary + Secondary	3.0 S_m	N/A
Shear	0.6 S_m	0.42 S_u

2.12.6.13.3 Top Spacer Assembly

The top spacer assembly is composed of a flat, circular plate with a series of 6 concentric rings welded to the top of the plate. The top spacer assembly finite element model is shown in Figure 2.12.6.13.3.1-1. Although the spacer operates at a lower temperature than the HLW Overpack the stress limits are conservatively evaluated at 300 °F.

2.12.6.13.3.1 Normal Conditions of Transport – 1 Foot Drop Cases

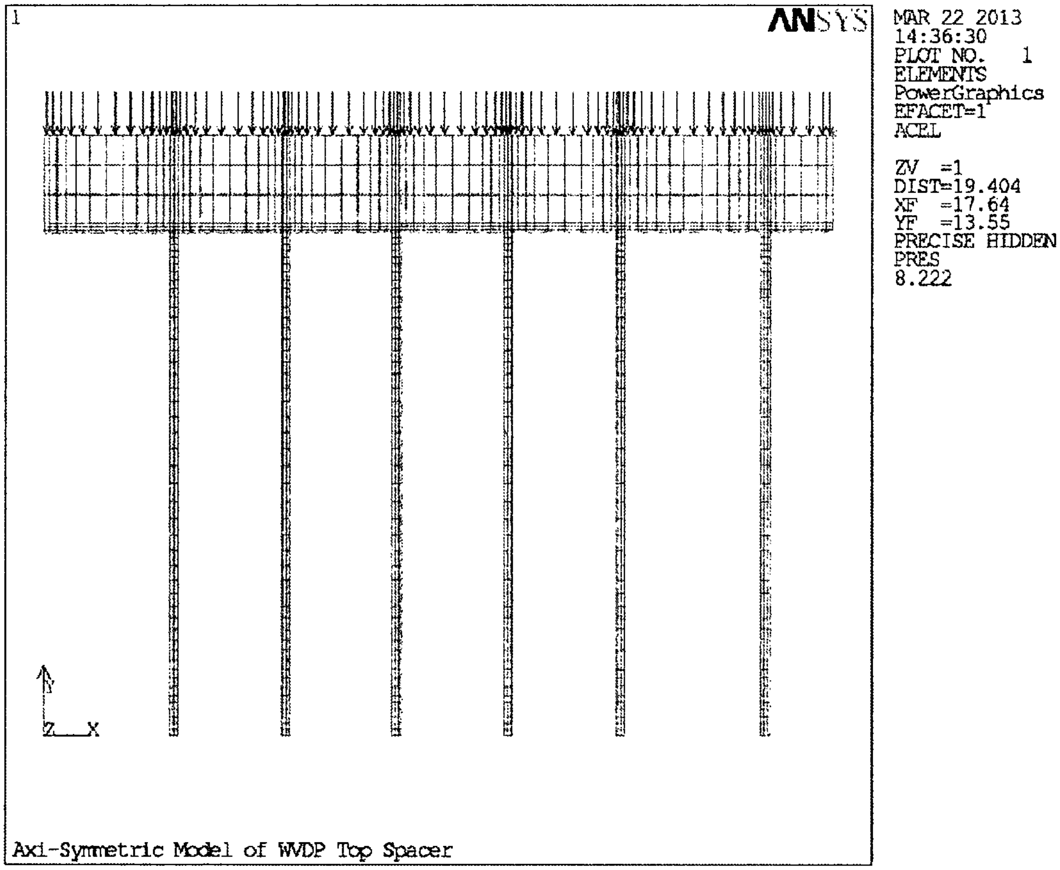
The top spacer is evaluated for the top end drop case and the side drop case. The bottom end drop case is not evaluated since the loading on the top spacer for the bottom end drop is due only to the top spacer's self-weight. Therefore, the top end drop case is the limiting case for the top spacer.

Top End Drop – 1 Foot Drop

For the top end drop evaluation, an axisymmetric finite element model of the top end spacer and the closure lid was constructed using the ANSYS SOLID42 quadrilateral MASS21 and CONTAC52 elements. An axisymmetric model for this load case is appropriate since the applied load on the closure lid is treated as a uniform equivalent pressure load. A small gap (0.001 inch) is modeled between the top of the spacer plate and the bottom of the closure lid. Contact elements are modeled to represent the compressive only loading on the top plate of the spacer. This provides for a realistic interaction between the spacer and closure lid. The finite element model is shown in Figure 2.12.6.13.3.1-1.

The model is constrained in the vertical direction at the bottom end of the rings. An equivalent acceleration of 20 g is applied in the vertical direction. Note that this is a conservative acceleration since calculated acceleration for the end drop case is 14.5 g based in Table 2.6.7.4.2-3. To account for the remaining weight of the HLW Overpack assembly (shell and bottom plate) a concentrated mass is conservatively applied at the outer radius of the closure lid. This element is assigned the weight of the HLW Overpack shell plus the bottom plate. The weight of the HLW Overpack contents is represented by a distributed pressure load on the closure lid as shown in Figure 2.12.6.13.3.1-1.

Figure 2.12.6.13.3.1-1 Top Spacer Model – End Drop Case



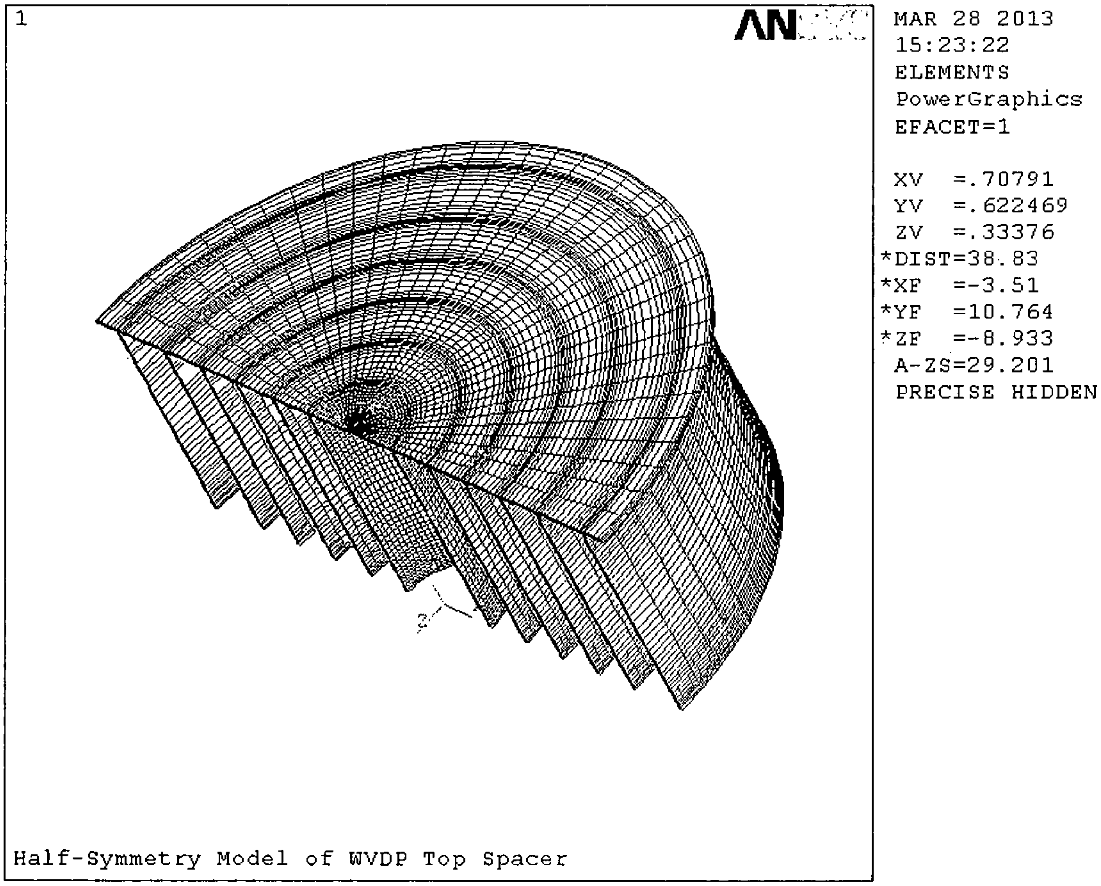
The calculated stresses for the normal condition of transport were linearized at the location of the maximum stress intensity which is at the outer diameter of the outermost support ring. The maximum membrane stress intensity was 5.2 ksi and the maximum membrane plus bending stress intensity was 13.6 ksi. Comparing these to the stress limit for Type 304 stainless steel of Sm = 20 ksi at 300 °F give The margins of safety's for membrane stress and membrane plus bending stress are +2.85 and +1.21 respectively. For additional details refer to item 1 in Section 2.12.6.14.

Side Drop Case – 1 Foot Drop

A separate finite element model was constructed for the side drop case since an axisymmetric model is not appropriate for the lateral loading. A one-half symmetry model was constructed. This model uses ANSYS Solid45 to model the plates and CONTA170/CONTA174 elements to represent the compressive only loading. Note the contact surfaces represent the inner surfaces of the NAC-STC cask. The finite element model is shown in Figure 2.12.6.13.3.1-2.

For this case an equivalent acceleration of 20 g was applied in the lateral direction. Note this is conservative since calculated acceleration for the side drop case is 16.5 g based in Table 2.6.7.4.2-3. No other loads were applied since the spacer is only subject to it's own weight in the side drop case. The model was constrained to enforce symmetry boundary conditions on the cut planes.

Figure 2.12.6.13.3.1-2 Top Spacer Model – Side Drop Case



The calculated stresses were linearized at the maximum stress location for each ring. The maximum membrane stress intensity was 6.47 ksi and the maximum membrane plus bending stress intensity was 22.8 ksi. Using Sm = 20 ksi at 300 °F for stainless steel Type 304, the margins of safety for membrane stress intensity and membrane plus bending stress intensity are +2.09 and +0.32, respectively. For additional details refer to item 1 in Section 2.12.6.14.

2.12.6.13.3.2 Accident Conditions of Transport

The same models were used for evaluating the accident condition of transport as the normal condition of transport. A complete description of the spacer models is given in Section 2.12.6.13.3.1.

Top End Drop – 30 Foot Drop

An equivalent acceleration of 48 g is applied in the vertical direction. Note this is conservative since calculated acceleration for the end drop case is 40.8 g based in Table 2.6.7.4.2-4.

The calculated stresses for the accident condition of transport were linearized at the location of the maximum stress intensity which is at the outer diameter of the outermost support ring. The maximum membrane stress was 12.5 ksi and the maximum membrane plus bending stress was 32.7 ksi. Using Su = 66.2 ksi at 300 °F for stainless steel Type 304, the margins of safety for membrane stress intensity and membrane plus bending stress intensity are +2.71 and +1.02, respectively. For additional details refer to item 1 in Section 2.12.6.14.

Buckling of Spacer Rings

The buckling of the highest loaded spacer ring was checked for accident conditions. The highest loaded ring for the top end drop case was the outermost ring. The average compressive stress in this ring was 11.0 ksi. The critical buckling stress was calculated to be 32.5 ksi which gives a margin of safety of +1.95 for the top spacer rings. For additional details refer to item 1 in Section 2.12.6.14.

Side Drop Case – 30 Foot Drop

The same model was used for evaluating the accident condition of transport as the normal condition of transport. A complete description of the model is given in Section 2.12.6.13.3.1.

For this case an equivalent acceleration of 55 g was applied in the lateral direction. Note this is conservative since calculated acceleration for the side drop case is 49.5 g based in Table 2.6.7.4.2-4.

The calculated stresses were linearized at the maximum stress location for each ring. The maximum membrane stress was 10.29 ksi and the maximum membrane plus bending stress was 46.3 ksi. Using Su = 66.2 ksi at 300 °F for stainless steel Type 304, the margins of safety for

membrane stress intensity and membrane plus bending stress intensity are +3.50 and +0.43, respectively. For additional details refer to item 1 in Section 2.12.6.14.

2.12.6.13.4 Bottom Spacer Assembly

The same model was used for evaluating the accident condition of transport as the normal condition of transport. A complete description of the model is given in Section 2.12.6.13.3.1.

2.12.6.13.4.1 Accident Conditions of Transport

Bottom End Drop

An equivalent acceleration of 48 g is applied in the vertical direction. Note this is conservative since calculated acceleration for the end drop case is 40.8 g based in Table 2.6.7.4.2-4.

The maximum membrane stress was 16.8 ksi and the maximum membrane plus bending stress was 45.6 ksi. Using Sm = 20 ksi at 300 °F for stainless steel Type 304, the margins of safety for membrane stress intensity and membrane plus bending stress intensity are +1.75 and +0.45, respectively. For additional details refer to item 1 in Section 2.12.6.14.

Buckling of Spacer Rings

The buckling of the highest loaded spacer ring was checked for accident conditions. The highest loaded ring for the top end drop case was the outermost ring. The average compressive stress in this ring was 14.5 ksi. The critical buckling stress was calculated to be 32.5 ksi which gives a margin of safety of +1.24 for the top spacer rings. For additional details refer to item 1 in Section 2.12.6.14.

2.12.6.14 References for HLW Overpack

1. NAC Calculation 630087-2025, MPC-WVDP Structural Evaluation for the STC Transport Cask, NAC International.

THIS PAGE INTENTIONALLY LEFT BLANK

2.13.6 STC -HBU Configuration for Normal Conditions of Transport

The 10 CFR 71.71 requires that the NAC-STC be structurally adequate for the following normal conditions of transport: (1) heat, (2) cold, (3) reduced external pressure, (4) increased external pressure, (5) vibration, (6) water spray, (7) free drop, (8) corner drop, (9) compression, and (10) penetration. In the free drop analyses, the cask impact orientation evaluated is the orientation that inflicts the maximum damage to the cask. 10 CFR 71.71 requires that the evaluation of the cask for the normal conditions of transport be at the most unfavorable ambient temperature in the range from -40°F to +100°F. In this section, the NAC-STC is evaluated for structural integrity for the normal conditions of transport in the STC-HBU configuration.

NAC-STC

Two- and three-dimensional structural analyses of the cask discussed in this section and summarized in the tables presented in Section 2.10.4 are for the Yankee-MPC directly loaded fuel in the NAC-STC cask body. The actual contents loading of the directly loaded fuel and the STC-HBU are 55,820 pounds and 54,920 pounds (this is the maximum weight of the three configurations for the STC-HBU fuel contents), respectively, confirming that the HBU fuel loading onto the cask body is bounded by directly loaded fuel in Table 2.2-1. As defined in Section 2.10.2.2.1, the NAC-STC cask body evaluation for the Yankee-MPC used a contents loading for the end drop and the side drop of 56,000 pounds. Since the STC-HBU total contents weight of contents is 54,920 pounds (Table 2.13.2-1), the cask body analyses for the directly loaded fuel are bounding for the one-foot and 30-foot drop evaluations of the STC-HBU contents in the NAC-STC.

[redacted] heat load is 24.0 kW, as compared to the heat load for the STC Directly Loaded Cask, which is 22.1 kW. Therefore, the thermal and stress evaluations for the STC-HBU configuration were performed for normal and accident conditions using finite element analyses. The structural finite element model of the cask body used for the STC-HBU configuration is based on the NAC CY fuel contents model.

STC -HBU Configuration

The STC-HBU configuration has a design weight of 248,620 pounds, including the redwood impact limiters. When the balsa impact limiters are used, the design weight is 242,320 pounds. For either case, the design weight is bounded by the design weight for CY–MPC configuration of 260,000 pounds. The CG of the STC-HBU configuration is located in the NAC-STC cask cavity at the same location as the Directly-Loaded configuration. As discussed previously, the STC-HBU

configuration is bounded by the analysis of the NAC-STC cask body for the CY-MPC configuration.

2.13.6.1 Heat

The NAC-STC cask in the STC-HBU configuration is not bounded by the Heat Condition evaluation of the NAC-STC cask in the Directly Loaded Fuel configuration in Section 2.6.1.1 (refer to the discussion in Section 2.13.6). A re-evaluation of the thermal conditions is performed.

2.13.6.2 Cold

The NAC-STC cask in the STC-HBU configuration is not bounded by the Cold Condition evaluation of the NAC-STC cask in the Directly Loaded Fuel configuration in Section 2.6.2.1 (refer to the discussion in Section 2.13.6). A re-evaluation of the thermal conditions is performed.

2.13.6.3 Reduced External Pressure

The NAC-STC cask in the STC-HBU configuration can be adequately represented by the Reduced External Pressure Condition evaluation in Section 2.6.3. No further evaluation is required.

2.13.6.4 Increased External Pressure

The NAC-STC cask in the STC-HBU configuration can be adequately represented by the Increased External Pressure Condition evaluation in Section 2.6.4. No further evaluation is required.

2.13.6.5 Vibration

Since the contents weight of the STC-HBU configuration is less than the Directly Loaded Fuel configuration, the dynamic response to the vibratory condition is adequately represented by the Vibration Condition evaluation in Section 2.6.5. No further evaluation is required.

2.13.6.6 Water Spray

The NAC-STC cask in the STC-HBU configuration is adequately represented by the Water Spray Condition evaluation in Section 2.6.6. No further evaluation is required.

2.13.6.7 Free Drop (1 Foot)

The free drop scenario outlined by 10 CFR 71.71(c)(7) requires the NAC-STC to be structurally adequate for a 1-foot drop (normal transport conditions) onto a flat, essentially unyielding horizontal surface in the orientation that inflicts the maximum damage to the cask. The

following subsections evaluate the cask body, the impact limiters, the closure lid and bolts, the neutron shield shell, and the upper ring components for the end, side, and corner drop orientations.

2.13.6.7.1 One-Foot End Drop

The NAC-STC cask in the STC-HBU configuration is bounded by the One-Foot End Drop Condition evaluation of the NAC-STC cask in the Directly Loaded Fuel configuration in Section 2.6.7.1.1 (see discussion in Section 2.13.6). Although, the calculated primary stress levels reported in Section 2.10.4 are bounding, the maximum temperatures of some components are not bounded. This affects the allowable stresses for primary stresses and the secondary stresses reported in Section 2.10.4.

2.13.6.7.2 One-Foot Side Drop

The NAC-STC cask in the STC-HBU configuration is bounded by the One-Foot Side Drop Condition evaluation of the NAC-STC cask in the Directly Loaded Fuel configuration in Section 2.6.7.2.1 (see discussion in Section 2.13.6). Although, the calculated primary stress levels reported in Section 2.10.4 are bounding for the STC-HBU configuration, the maximum temperatures of some components are not bounded. This affects the allowable stresses for primary stresses and the secondary stresses reported in Section 2.10.4.

2.13.6.7.3 One-Foot Corner Drop

The NAC-STC cask in the STC-HBU configuration is bounded by the One-Foot Corner Drop Condition evaluation of the NAC-STC cask in the Directly Loaded Fuel configuration in Section 2.6.7.3.1 (see discussion in Section 2.13.6). Although, the calculated primary stress levels reported in Section 2.10.4 are bounding for the STC-HBU configuration, the maximum temperatures of some components are not bounded. This affects the allowable stresses for primary stresses and the secondary stresses reported in Section 2.10.4.

2.13.6.7.4 Impact Limiters

Removable transport impact limiters are included in the NAC-STC design to ensure that the design impact loads on the cask are not exceeded for any of the defined impact load conditions. The defined loading conditions include the cask falling 1 foot or 30 feet and landing on its side impacting both impact limiters simultaneously, or landing vertically on one impact limiter at either end. As discussed in Section 2.10.7, the oblique impact events are bounded by the end and/or side drop events, as there is no "slap down" effect exhibited by the cask. Consequently, oblique drops and oblique drop angles are not evaluated.

Two impact limiter configurations have been designed for the NAC-STC cask. The first impact limiter configuration is primarily redwood with a small amount of balsa wood (redwood impact limiter). The wood is encased in a stainless steel shell. The redwood impact limiters may be used only with the directly loaded fuel configuration or the Yankee-MPC canistered fuel and GTCC waste configurations. The second impact limiter configuration is primarily balsa wood with a limited amount of redwood (balsa impact limiter). The wood is also encased in a stainless steel shell. The balsa impact limiters may be used with the directly loaded fuel, Yankee-MPC fuel or GTCC waste, the CY-MPC fuel or GTCC waste, STC-LACBWR fuel or the STC-HBU configurations.

2.13.6.7.4.1 Impact Limiters

The NAC-STC cask in the STC-HBU configuration includes the balsa impact limiters or the redwood impact limiters.

With the Redwood impact limiters, the lower design basis weight (248,620 lbs) of the STC-HBU Cask is attributed to the lower contents weight of 54,920 lbs versus the total fuel basket weight of 55,820 lbs in the licensed design basis weight (250,000 lbs) of Directly-loaded fuel Cask. The reduction of 900 pounds for the total transport cask weight with contents has an insignificant effect on the performance of the redwood impact limiter.

With the balsa wood impact limiters, the weight for the design basis CY-MPC canistered fuel cask is 260,000 lbs. The lower weight of the STC-HBU Cask (242,320 lbs) is attributed to the lower contents weight. The reduction in the contents weight also means that the total kinetic energy which must be absorbed is also reduced. Since crush strength curves for balsa wood and redwood monotonically increase with strain, the resulting strain, and crush strength could be reduced. The bounding condition is the cold condition for which the properties have been evaluated at -40F. Additionally, for the cold condition, the properties of the balsa wood have been factored by another 10% to account for uncertainties, one of which could be considered to be the reduction of the weight by approximately 10%. If the acceleration of the cask is increased by 8%, the original CY-MPC acceleration of 39.9g increases to 43.0g. The bounding condition for the stresses developed in the cask are those for the top end drop in which the inertial loading due to the contents weight is applied to the lid. The inertial loading applied to the closure lid for the top end drop is the acceleration times the contents weight. A comparison of the inertial loading to the closure lid for the two configurations is shown below. The inertial loading applied to the closure lid associated the total cask weight configuration of 260,000 pounds is bounding by more than 13% over the loading associated with the STC-HBU configuration. The comparison of the cask and fuel weight for the CY-MPC canistered fuel cask and HBU fuel is shown in the table below.

Comparison of the Inertial Loading Applied to the Closure Lid for the Top End Drop

	CY-MPC canistered Fuel	STC-HBU Contents
Total contents weight (lb)	67,195	54,920
End Drop Acceleration (g's)	39.9	43.0
Inertial load applied to the lid (kips)	2,681	2,361

Likewise, for the side drop, this means that the total kinetic energy which must be absorbed is also reduced due to the reduction in the contents weight. While for the end drop, the backed area for the balsa wood (and therefore the cross sectional area for the crushed balsa wood) remains the same, the crush area for the side drop does changes with the crush depth. To evaluate this, the finite element model used for the 260,000 pound design basis weight was altered to 242,320 pounds and the resulting acceleration increased from 48.5g to 50.7g. The inertial loading applied to the cask shells is determined by the product of the contents weight and the acceleration, which is compared in the table below. The inertial loading applied to the cask shells associated the total cask weight configuration of 260,000 pounds is bounding by more than 17% over the loading associated with the STC-HBU configuration.

Comparison of the Inertial Loading Applied to the Cask Shells for the Side Drop

	CY-MPC canistered Fuel	STC-HBU Contents
Total contents weight (lb)	67,195	54,920
Side Drop Acceleration (g's)	48.5	50.7
Inertial load applied to cask (kips)	3,259	2,784

This confirms that the total inertial loads from the cask contents for the end drop and side drops with lighter weights are bounded by CY-MPC design basis weight of 260,000 pounds.

2.13.6.7.5 Closure Analysis – Normal Conditions of Transport

The cask body evaluation used a contents weight of 56,000 pounds which also bounds the contents weight of the STC-HBU configuration of 54,920 pounds. Temperatures employed for the allowables are adjusted for the higher cask temperatures associated with the STC-HBU total heat load of 24.0 kW.

2.13.6.7.6 Neutron Shield Analysis

The canistered contents have an insignificant structural effect on the neutron shield. Temperatures employed for the allowables are adjusted for the higher cask temperatures associated with the STC-HBU total heat load of 24.0 kW.

2.13.6.7.7 Upper Ring/Outer Shell Intersection Analysis

The cask body evaluation used a contents weight of 56,000 pounds, which also bounds the contents weight of the STC-HBU configuration. The NAC-STC cask in the STC-HBU configuration is bounded by the upper ring/outer shell intersection analysis of the NAC-STC cask in Section 2.6.7.7. No further evaluation is required.

2.13.6.8 Corner Drop (1 Foot)

According to 10 CFR 71.71(c)(8), this test is not applicable to the NAC-STC because the cask is composed of materials other than fiberboard or wood and the cask weight exceeds 220 pounds (100 kg).

2.13.6.9 Compression

According to 10 CFR 71.71(c)(9), this test is not applicable to the NAC-STC because the package weight is greater than 5,000 kilograms (11,023 lb).

2.13.6.10 Penetration

This condition is defined in 10 CFR 71 as a 40-inch drop of a 13-pound, 1.25-inch diameter penetration cylinder with a hemispherical end onto any exposed surface of the cask. The acceptance criteria is that the drop will not adversely affect the ability of the cask to maintain containment of the contents or to survive a hypothetical accident. The impact limiters, the neutron shield, and the port covers could potentially be damaged by this penetration impact.

The NAC-STC cask in the STC-HBU configuration is bounded by the Penetration Analyses of the NAC-STC cask in Section 2.6.10. No further evaluation is required.

2.13.6.11 Fabrication Conditions

The NAC-STC cask in the STC-HBU configuration is bounded by the Fabrication Conditions evaluation of the NAC-STC cask in Section 2.6.11. No further evaluation is required.

2.13.6.12 STC-HBU Analysis – Normal Transport Conditions

[redacted] configuration. The STC-HBU is evaluated for the Normal Conditions of Transport in this section.

The principal components of the STC-HBU configuration are the STC shell assembly, basket and the closure lids.

2.13.6.12.1 STC-HBU Assembly Analysis Description

[redacted] basket. The STC-HBU fuel assemblies are directly loaded and do not utilize a separate inner container for containment during transport. The NAC-STC provides the primary containment boundary for transport.

The basket used for the STC-HBU contents is the same basket used for directly loaded PWR fuel with lower burnup. Because the basket is not a secondary containment boundary, the structural design criterion used for the basket is ASME Code Section III, Subsection NG. The stress allowables used in the basket evaluation are defined in Table 2.1.2-2 for noncontainment structures. The basket for the STC-HBU contents is evaluated in Section 2.13.6.14. The allowable stresses used in this analysis are based on a bounding maximum material temperature of 650°F which bounds the maximum temperature in the STC-HBU configuration as reported in Section 3.8.4.2.

The STC-HBU cask and basket stresses are not recomputed for the 1-foot drop condition in the end and side impact orientations since the weight of the STC 26 assembly directly loaded basket is bounding for the STC-HBU configuration. However, the allowable stresses for the cask assembly are reduced based on the maximum cask temperature from the thermal analysis. The allowable stresses for the basket assembly are not changed since the maximum basket temperature used was greater than the maximum calculated temperature of the basket with HBU fuel contents. The thermally induced stresses for the STC-HBU cask are re-evaluated since the maximum temperatures and thermal gradients are higher for the HBU 24 kW contents than the standard 26 fuel assembly 22.1 kW configuration.

2.13.6.12.2 Finite Element Model Descriptions – STC-Cask with HBU contents

The loading for the normal operating condition is based on the 1-foot drop. Drop orientations considered are the end (top and bottom) and side drops. The operational conditions also contain loads developed from the thermal gradients in the cask assembly and cask internal pressure. These are included in the cask model analyses as separate analyses. The results are then combined with the maximum stresses due to the drop cases.

End Drop Model

For the end drop, the same finite element model and stress results as the STC 26 assembly Directly Loaded Fuel are used. Only the maximum temperature representing the HBU contents and allowable stresses was modified.

Plots of the NAC-STC cask finite element model are shown in Figure 2.10.2-1 through Figure 2.10.2-36.

Side Drop Model

For the side drop, the same finite element model and stress results as the STC 26 assembly Directly Loaded Fuel are used. Only the maximum temperature representing the HBU contents and allowable stresses are modified.

Plots of the NAC-STC cask finite element model are shown in Figure 2.10.2-1 through Figure 2.10.2-36.

STC-HBU Cask Internal Pressure Model

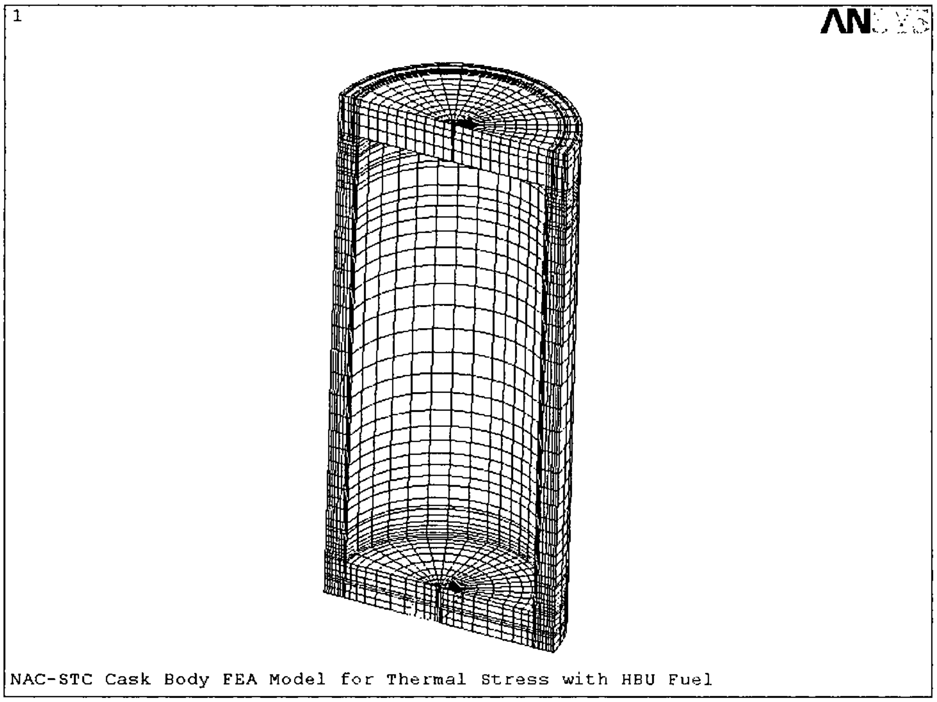
For the internal pressure case, the same finite element model and stress results as the STC 26 assembly Directly Loaded Fuel are used. Only the maximum temperature representing the HBU contents and allowable stresses are modified.

STC-HBU Cask Thermal Stress Model

The thermal stress analyses of the canister are performed with temperature distributions corresponding to the hot (100°F ambient with solar heat load) and cold (-40°F ambient) external conditions. The finite element model for the thermally induced stress is the same as the NAC-STC CY-MPC finite element model. The temperature distribution is extracted from the thermal model and applied to the structural model. This is accomplished by interpolating the temperatures from the thermal model on to the nodes of the structural model.

The STC-HBU cask assembly thermal stress mesh is shown in Figure 2.13.6.12-1. This mesh is the same as the bottom end drop mesh. The thermal system model included the basket, HBU contents and NAC-STC cask assembly.

Figure 2.13.6.12-1 STC-HBU Cask Assembly – Thermal Stress Model



Symmetry boundary conditions were applied to the model symmetry plane and all nodes on the bottom of the bottom plate were constrained in the vertical direction. The nodes on the plane of symmetry were constrained in the horizontal direction. This set of boundary conditions are used to allow free thermal expansion/contraction of the model. The temperatures were interpolated from the thermal solution and used to evaluate the thermally induced stresses. The thermal loading was the only applied load for the thermal stress evaluation.

2.13.6.12.3 Stress Evaluation of STC Cask with HBU Contents – Normal Condition of Transport

2.13.6.12.3.1 End Drop Cases – 1 Foot Drop

The results for Directly Loaded Fuel are taken from primary stresses from Tables 2.10.4-42, 2.10.4-43, 2.10.4-45, 2.10.4-46, 2.10.4-48 and 2.10.4-49 for the top end drop and from Tables 2.10.4-56, 2.10.4-57, 2.10.4-59, 2.10.4-60, 2.10.4-62 and 2.10.4-63 for the bottom end drop. These stress results were used for the HBU configuration since the stresses are bounding. For some components of the HBU configuration the temperatures for the Directly Loaded Fuel are not bounding. The allowable stresses were updated based on the calculated temperatures for the HBU fuel configuration.

Table 2.13.6.12.3-1 Maximum P_m Stress Summary for 1-Foot Top End Drop Hot Condition Case (100 °F Ambient)

Component Number	Stress Intensity[1] (ksi)	Maximum[2] Temperature °F	Allowable Stress (ksi)	Margin of Safety[3]
1	0.2	282	20.0	Large
2	0.8	303	20.0	Large
3	3.7	321	31.2	7.43
4	2.9	376	19.0	5.55
5	2.1	303	20.0	8.52
6	3.9	256	20.0	4.13
7	9.1	268	20.0	1.20
8	3.8	249	45.0	Large

Notes: 1) Taken from Table 2.10.4-42
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

Table 2.13.6.12.3-2 Maximum P_m+P_b Stress Summary for 1-Foot Top End Drop Hot Condition Case (100 °F Ambient)

Component Number	Stress Intensity[1] (ksi)	Maximum[2] Temperature °F	Allowable Stress (ksi)	Margin of Safety
1	0.8	282	30.0	Large
2	0.9	303	30.0	Large
3	5.6	321	46.8	7.36
4	3.0	376	28.5	8.50
5	2.2	303	30.0	Large
6	11.6	256	30.0	1.59
7	20.7	268	30.0	0.45
8	18.9	249	67.5	2.57

Notes: 1) Taken from Table 2.10.4-43
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

Table 2.13.6.12.3-3 Maximum P_m Stress Summary for 1-Foot Top End Drop Cold Condition Case (-40 °F Ambient)

Component Number	Stress Intensity[1] (ksi)	Maximum[2] Temperature °F	Allowable Stress (ksi)	Margin of Safety[3]
1	0.3	162	20.0	Large
2	1.0	187	20.0	Large
3	3.9	208	33.1	7.49
4	2.4	268	20.0	7.33
5	2.2	186	20.0	8.09
6	4.5	138	20.0	3.44
7	9.3	149	20.0	1.15
8	3.7	131	45.0	Large

Notes: 1) Maximum from either Table 2.10.4-45 or Table 2.10.4-48

2) From thermal evaluation in Section 3.8

3) Margin of Safety greater than 10 are reported as large

Table 2.13.6.12.3-4 Maximum P_m+P_b Stress Summary for 1-Foot Top End Drop Cold Condition Case (-40 °F Ambient)

Component Number	Stress Intensity(1) (ksi)	Maximum(2) Temperature °F	Allowable Stress (ksi)	Margin of Safety
1	1.3	162	30.0	Large
2	1.4	187	30.0	Large
3	6.0	208	49.6	7.27
4	2.5	268	30.0	Large
5	2.3	186	30.0	Large
6	11.8	138	30.0	1.54
7	20.0	149	30.0	0.5
8	19.2	131	67.5	2.52

Notes: 1) Maximum from either Table 2.10.4-46 or Table 2.10.4-49

2) From thermal evaluation in Section 3.8

3) Margin of Safety greater than 10 are reported as large

Table 2.13.6.12.3-5 Maximum P_m Stress Summary for 1-Foot Bottom End Drop Hot Condition Case (100 °F Ambient)

Component Number	Stress Intensity[1] (ksi)	Maximum[2] Temperature °F	Allowable Stress (ksi)	Margin of Safety[3]
1	1.9	282	20.0	Large
2	4.2	303	20.0	3.76
3	3.6	321	31.2	7.67
4	2.9	376	19.0	5.55
5	2.1	303	20.0	8.52
6	4.0	256	20.0	4.0
7	7.8	268	20.0	1.56
8	0.5	249	45.0	Large

Notes: 1) Taken from Table 2.10.4-56
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

Table 2.13.6.12.3-6 Maximum P_m+P_b Stress Summary for 1-Foot Bottom End Drop Hot Condition Case (100 °F Ambient)

Component Number	Stress Intensity(1) (ksi)	Maximum(2) Temperature °F	Allowable Stress (ksi)	Margin of Safety
1	5.7	282	30.0	4.26
2	7.8	303	30.0	2.85
3	5.0	321	46.8	8.36
4	3.0	376	28.5	8.50
5	2.3	303	30.0	Large
6	10.5	256	30.0	1.86
7	13.8	268	30.0	1.17
8	2.5	249	67.5	Large

Notes: 1) Taken from Table 2.10.4-57
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

Table 2.13.6.12.3-7 Maximum P_m Stress Summary for 1-Foot Bottom End Drop Cold Condition Case (-40 °F Ambient)

Component Number	Stress Intensity$^{(1)}$ (ksi)	Maximum$^{(2)}$ Temperature °F	Allowable Stress (ksi)	Margin of Safety$^{(3)}$
1	1.9	162	20.0	9.53
2	4.7	187	20.0	3.26
3	4.0	208	33.1	7.28
4	2.4	268	20.0	7.33
5	2.3	186	20.0	7.7
6	4.2	138	20.0	3.76
7	8.0	149	20.0	1.5
8	0.5	131	45.0	Large

Notes: 1) Maximum from either Table 2.10.4-59 or Table 2.10.4-62
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

Table 2.13.6.12.3-8 Maximum P_m+P_b Stress Summary for 1-Foot Bottom End Drop Cold Condition Case (-40 °F Ambient)

Component Number	Stress Intensity[1] (ksi)	Maximum[2] Temperature °F	Allowable Stress (ksi)	Margin of Safety
1	7.3	162	30.0	3.11
2	8.7	187	30.0	2.45
3	5.9	208	49.6	7.41
4	2.4	268	30.0	Large
5	2.5	186	30.0	Large
6	10.7	138	30.0	1.8
7	14.1	149	30.0	1.13
8	2.9	131	67.5	Large

Notes: 1) Maximum from either Table 2.10.4-60 or Table 2.10.4-63

2) From thermal evaluation in Section 3.8

3) Margin of Safety greater than 10 are reported as large

2.13.6.12.3.2 Side Drop Case – 1 Foot Drop

The results for primary stresses (P_m and P_m+P_b) from Tables 2.10.4-70 and 2.10.4-71 for the Yankee Directly Loaded Fuel were used for the HBU contents since these results are bounding. However, the allowable stresses were based on the maximum temperatures for various components for the HBU configuration. The primary plus secondary stresses were updated based on the calculated thermals stresses for the HBU fuel configuration.

Table 2.13.6.12.3-9 Maximum P_m Stress Summary for 1-Foot Side Drop Hot Condition Case (100 °F Ambient)

Component Number	Stress Intensity(1) (ksi)	Maximum(2) Temperature °F	Allowable Stress (ksi)	Margin of Safety(3)
1	4.6	282	20.0	3.35
2	10.9	303	20.0	0.83
3	13.7	321	31.2	1.28
4	9.5	376	19.0	1.00
5	6.0	303	20.0	2.33
6	12.9	256	20.0	0.55
7	8.9	268	20.0	1.25
8	2.8	249	45.0	Large

Notes: 1) Taken from Table 2.10.4-70
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

Table 2.13.6.12.3-10 Maximum P_m+P_b Stress Summary for 1-Foot Side Drop Hot Condition Case (100 °F Ambient)

Component Number	Stress Intensity(1) (ksi)	Maximum(2) Temperature °F	Allowable Stress (ksi)	Margin of Safety
1	5.1	282	30.0	4.88
2	15.6	303	30.0	0.92
3	14.8	321	46.8	2.16
4	10.2	376	28.5	1.79
5	6.5	303	30.0	3.62
6	17.4	256	30.0	0.72
7	12.6	268	30.0	1.38
8	6.8	249	67.5	8.93

Notes: 1) Taken from Table 2.10.4-71
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

2.13.6.12.3.3 Top Corner Drop Case – 1 Foot Drop

The results for primary stresses (P_m and P_m+P_b) from Tables 2.10.4-89 and 2.10.4-90 for the Yankee Directly Loaded Fuel were used for the HBU contents since these results are bounding. However, the allowable stresses were based on the maximum temperatures for various components for the HBU configuration. The primary plus secondary stresses were updated based on the calculated thermals stresses for the HBU fuel configuration.

Table 2.13.6.12.3-11 Maximum P_m Stress Summary for 1-Foot Top Corner Drop Hot Condition Case (100 °F Ambient)

Component Number	Stress Intensity$^{(1)}$ (ksi)	Maximum$^{(2)}$ Temperature °F	Allowable Stress (ksi)	Margin of Safety$^{(3)}$
1	0.4	282	20.0	Large
2	3.0	303	20.0	5.67
3	9.5	321	31.2	2.28
4	8.0	376	19.0	1.38
5	4.8	303	20.0	3.17
6	12.1	256	20.0	0.65
7	11.7	268	20.0	0.71
8	6.4	249	45.0	6.03

Notes: 1) Taken from Table 2.10.4-89
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

Table 2.13.6.12.3-12 Maximum P_m+P_b Stress Summary for 1-Foot Top Corner Side Drop Hot Condition Case (100 °F Ambient)

Component Number	Stress Intensity[1] (ksi)	Maximum[2] Temperature °F	Allowable Stress (ksi)	Margin of Safety
1	0.9	282	30.0	Large
2	3.9	303	30.0	6.69
3	11.3	321	46.8	3.14
4	8.5	376	28.5	2.35
5	4.9	303	30.0	5.12
6	12.3	256	30.0	1.44
7	20.9	268	30.0	0.44
8	23.9	249	67.5	1.82

Notes: 1) Taken from Table 2.10.4-90
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

2.13.6.12.3.4 Bottom Corner Drop Case – 1 Foot Drop

The results for primary stresses (P_m and P_m+P_b) from Tables 2.10.4-103 and 2.10.4-104 for the Yankee Directly Loaded Fuel were used for the HBU contents since these results are bounding. However, the allowable stresses were based on the maximum temperatures for various components for the HBU configuration. The primary plus secondary stresses were updated based on the calculated thermals stresses for the HBU fuel configuration.

Table 2.13.6.12.3-13 Maximum P_m Stress Summary for 1-Foot Bottom Corner Drop Hot Condition Case (100 °F Ambient)

Component Number	Stress Intensity(1) (ksi)	Maximum(2) Temperature °F	Allowable Stress (ksi)	Margin of Safety(3)
1	6.7	282	20.0	1.99
2	8.6	303	20.0	1.33
3	9.4	321	31.2	2.32
4	7.8	376	19.0	1.44
5	4.9	303	20.0	3.08
6	3.2	256	20.0	5.25
7	0.6	268	20.0	Large
8	0.4	249	45.0	Large

Notes: 1) Taken from Table 2.10.4-103
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

Table 2.13.6.12.3-14 Maximum P_m+P_b Stress Summary for 1-Foot Bottom Corner Drop Hot Condition Case (100 °F Ambient)

Component Number	Stress Intensity[1] (ksi)	Maximum[2] Temperature °F	Allowable Stress (ksi)	Margin of Safety
1	14.5	282	30.0	1.07
2	15.6	303	30.0	0.92
3	10.1	321	46.8	3.63
4	8.2	376	28.5	2.48
5	5.0	303	30.0	5.0
6	3.9	256	30.0	6.69
7	1.0	268	30.0	Large
8	0.9	249	67.5	Large

Notes: 1) Taken from Table 2.10.4-104
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

Since the minimum Margin of Safety for the 1-foot drop cases is 0.44 from Table 2.13.6.12.3-12, the STC Cask design is acceptable for Normal Conditions of transport.

2.13.6.12.4 Internal Pressure Case

The STC cask was evaluated for an internal pressure of 50 psig, which bounds the calculated internal pressure of 11.8 psi for Directly Loaded Fuel from Section 3.4.4.1 and the maximum of 38 psig for the CY-MPC configuration in Section 3.4.4.3. Therefore it is conservative to use the 50 psig compared to normal condition stress limits. Linearization of the stresses due to internal pressure case of 50 psig in Table 2.10.4-1 gave a maximum stress intensity of 2.1 ksi at sections U2 and U3, which are located in the inner head. The margins of safety for internal pressure only are +13.3 for membrane plus bending. For additional details refer to Table 2.10.4-1.

2.13.6.12.5 Thermal Cases

As discussed in Section 2.13.6.12.2, the thermal results were interpolated on to the structural mesh and used to evaluate the thermally induced stresses. The thermal stresses are not directly compared to stress limits since thermal effects are to be combined with other operating loads.

2.13.6.12.5.1 Hot Case – Ambient Temperature of 100 °F

Evaluation and linearization of the stress intensities due to the hot thermal case gave a maximum membrane plus bending stress of 17.2 ksi. For additional details refer to Table 2.13.6.12.5-1 below.

Table 2.13.6.12.5-1 Maximum P_m+P_b Stress Summary for Hot Condition Case (100 °F Ambient)

Section Number	Component	Stress Intensity(1) (ksi)
1	Transition Shell	5.6
2	Inner Shell	5.5
3	Inner Shell	5.1
4	Bottom Forging	8.2
5	Outer Shell	13.1
6	Outer Shell	11.5
7	Top Forging	18.6
8	Top Forging	19.1
9	Bottom Plate	5.5
10	Bottom Forging	15.6
11	Bottom Plate	5.7
12	Bottom Forging	9.1
13	Inner Lid	1.8
14	Outer Lid	1.7
15	Inner Lid	1.9
16	Outer Lid	2.1

Notes: 1) Taken from Reference 1

2.13.6.12.5.2 Cold Case – Ambient Temperature of -40 °F

Evaluation and linearization of the stress intensities due to the cold thermal case gave a maximum membrane plus bending stress of 20.4 ksi. For additional details, refer to Table 2.13.6.12.4-2.

Table 2.13.6.12.5-2 Maximum Pm+Pb Stress Summary for Cold Condition Case (-40 °F Ambient)

Section Number	Component	Stress Intensity(1) (ksi)
1	Transition Shell	9.0
2	Inner Shell	9.0
3	Inner Shell	9.0
4	Bottom Forging	14.8
5	Outer Shell	9.6
6	Outer Shell	6.8
7	Top Forging	23.9
8	Top Forging	9.5
9	Bottom Plate	1.1
10	Bottom Forging	13.7
11	Bottom Plate	2.7
12	Bottom Forging	3.8
13	Inner Lid	1.5
14	Outer Lid	1.4
15	Inner Lid	1.4
16	Outer Lid	1.8

Notes: 1) Taken from Reference 1

2.13.6.12.6 Combined Stress Case

The maximum P_m+P_b stress intensity due to the 1 foot drop cases is 23.9 ksi for the top corner drop case, the maximum P_m+P_b stress intensity for the internal pressure case is 2.1 ksi and the maximum linearized stress intensity for the thermal cases is 23.9 ksi for the cold case. Conservatively adding the maximum stress intensities gives a combined stress of 49.9 ksi. The margin of safety for the combined stress is +0.12 using a limit of 3Sm = 56.1 ksi at 400 °F where S_m for 304/304L stainless steel is 18.7 ksi at 400 °F. This is a conservative approach since the maximum temperature of the cask is 376 °F for the hot condition case.

2.13.6.13 Accident Conditions of Transport

The cask body stresses for the accident condition were not recomputed since the STC-HBU contents weight is bounded by the directly loaded (low burnup) contents weight. As discussed in Section 2.13.6.7.4.1, the loading induced by the contents is bounding for the directly loaded configuration for the balsa wood impact limiter. For the configuration with the redwood impact limiters, the weight difference between the STC-HBU and the directly loaded configuration is negligible. For the accident condition, thermally induced stresses are not considered. The temperature change due to the increased heat load has an insignificant effect on the material modulus. Since the stress allowables are affected by the temperature increase, the Critical Margins of Safety are recomputed.

Top End Drop – 30 Foot Drop

The calculated stresses for the accident condition of transport were linearized in Section 2.10.4 for the directly loaded fuel configuration. For hot conditions the maximum membrane stress was 16.4 ksi and the maximum membrane plus bending stress was 37.1 ksi. The margins of safety for membrane stress intensity and membrane plus bending stress intensity are +1.76 and +0.70, respectively. For cold conditions the maximum membrane stress was 16.4 ksi and the maximum membrane plus bending stress was 38.5 ksi. The margins of safety for membrane stress intensity and membrane plus bending stress intensity are +1.89 and +0.76, respectively. For additional details refer to Tables 2.13.6.13-1 through 2.13.6.13-4.

Table 2.13.6.13.1-1 Maximum P_m Stress Summary for 30-Foot Top End Drop Hot Condition Case (100 °F Ambient)

Component Number	Stress Intensity$^{(1)}$ (ksi)	Maximum$^{(2)}$ Temperature °F	Allowable Stress (ksi)	Margin of Safety$^{(3)}$
1	0.7	282	46.8	Large
2	2.7	303	43.1	Large
3	9.7	319	65.5	5.75
4	6.8	375	45.4	5.68
5	6.1	301	46.2	6.57
6	9.7	256	44.5	3.59
7	16.0	268	44.1	1.76
8	8.1	249	94.5	Large

Notes: 1) Taken from Table 2.10.4-115
2) From thermal evaluation in Section 3.8
3) Margin of Safety greater than 10 are reported as large

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

Impact Limiter	Initial Bow (in)	Von-Mises Stress from Model (psi)	Yield Strength (psi)	Factor of Safety
Redwood	0.01	60.4	63.5	1.05
Balsa	0.01	54.5	63.5	1.16

As shown in the preceding table, the maximum stresses in the PWR fuel rods remain below the yield strength in the design basis accident events, confirming that the fuel rods remain elastic and will return to their original configuration after the 30-ft end drop accident condition.

Figure 2.13.6.15-1 ANSYS Model for Fuel Assembly

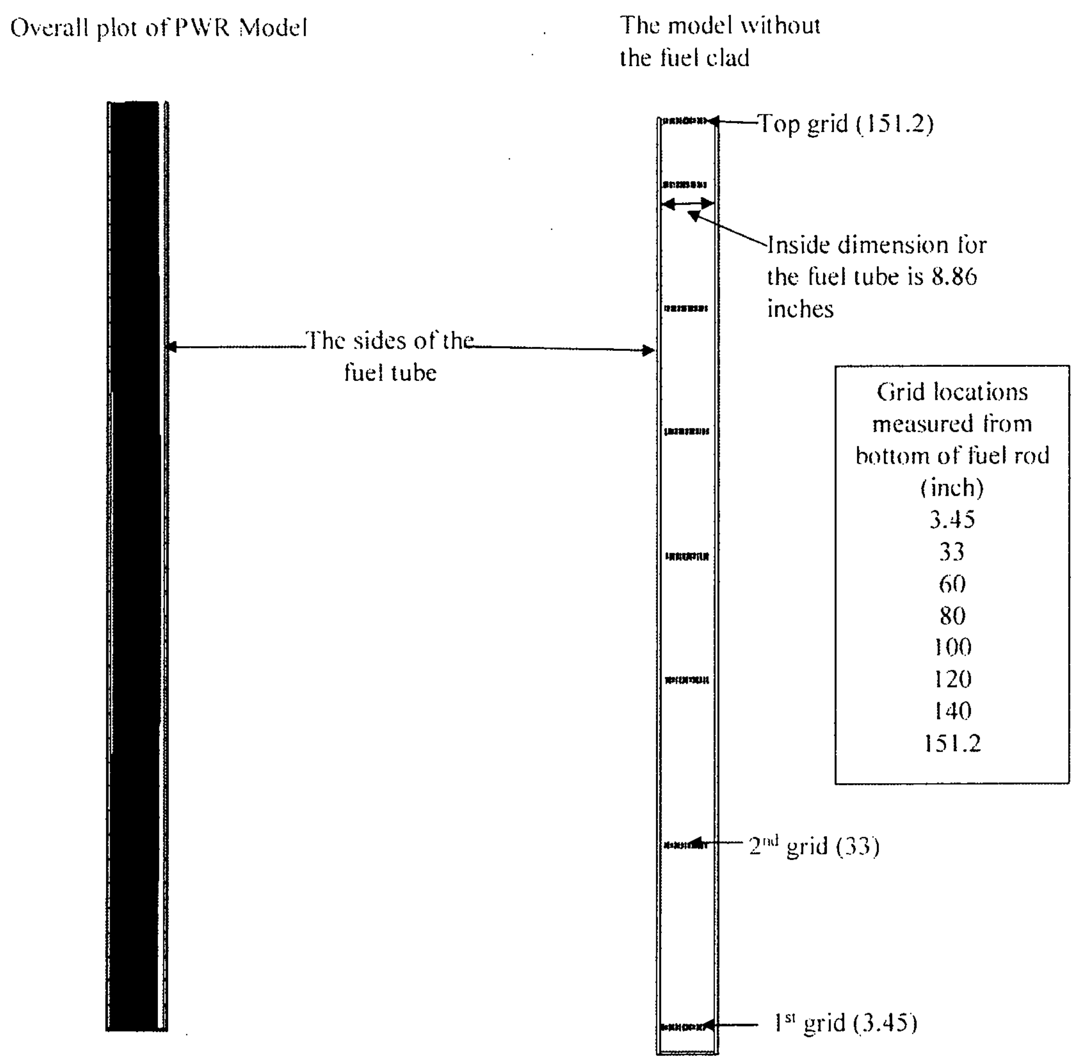


Figure 2.13.6.15-2 LS-DYNA Model for 30-Foot Drop

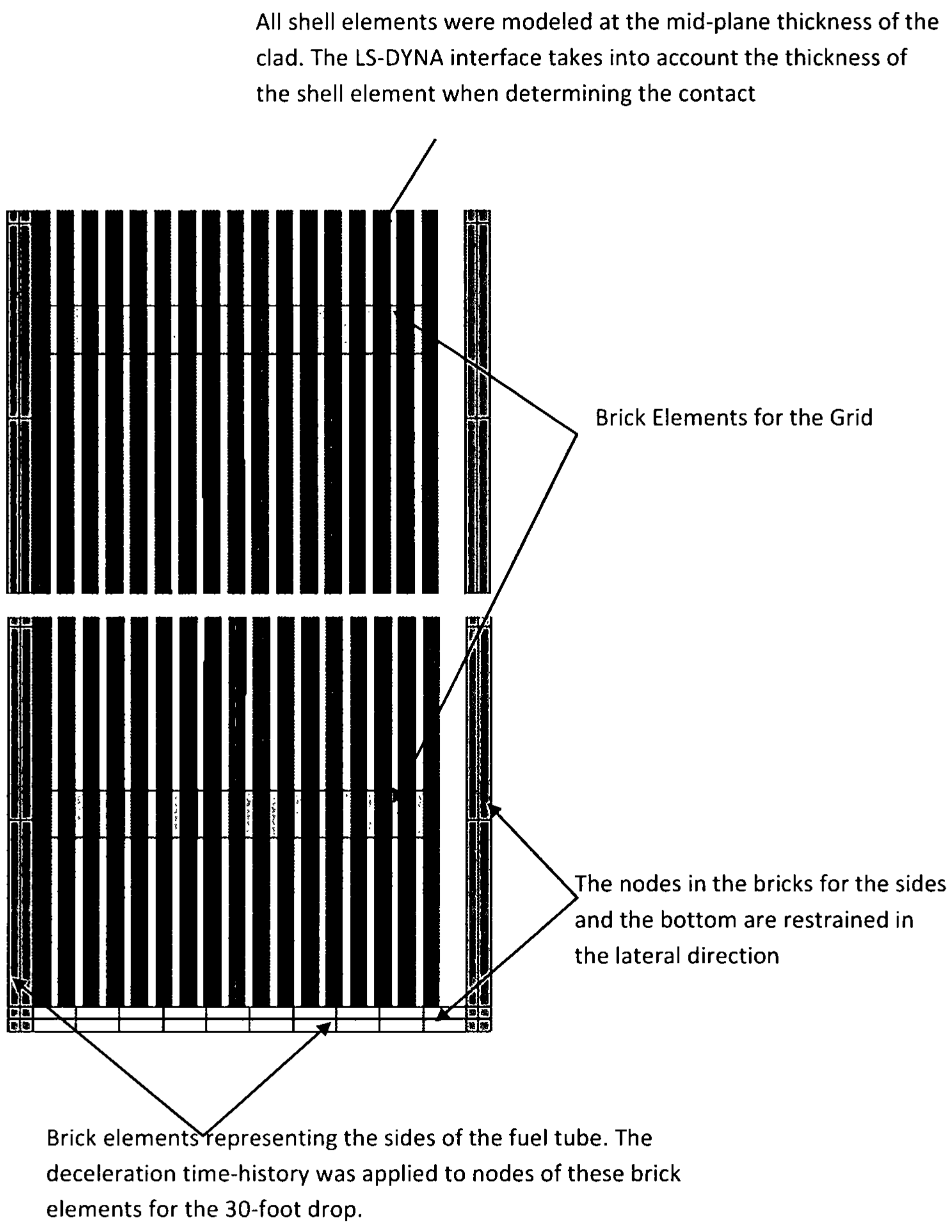
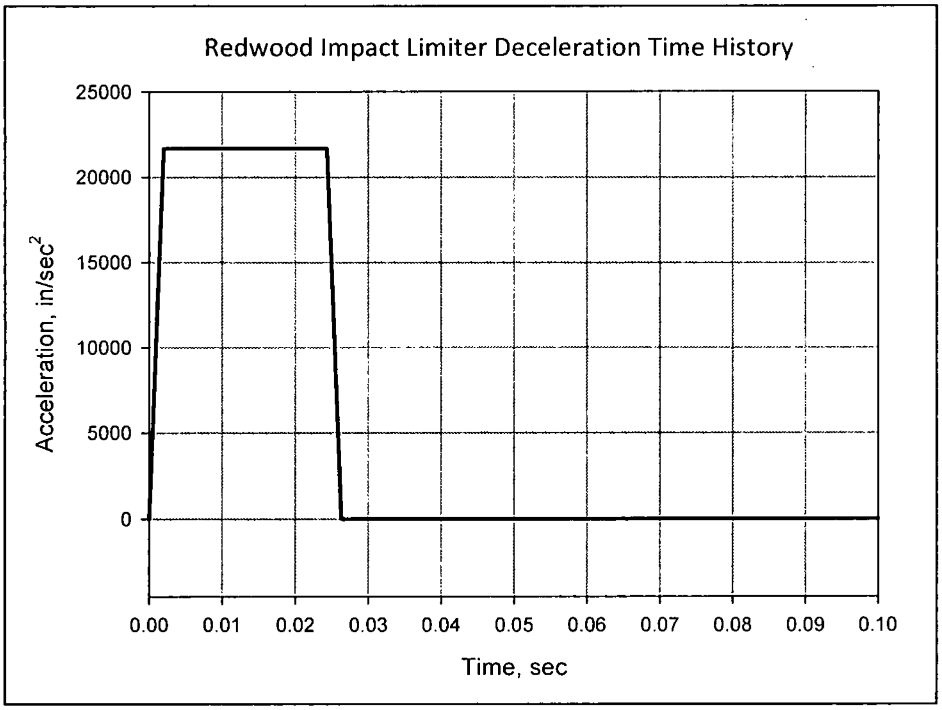


Figure 2.13.6.15-3 Bounding Accelerations from Balsa and Redwood Impact Limiters for the 30-ft End Fuel Rod Drop Evaluation



Balsa Impact Limiter Deceleration Time History

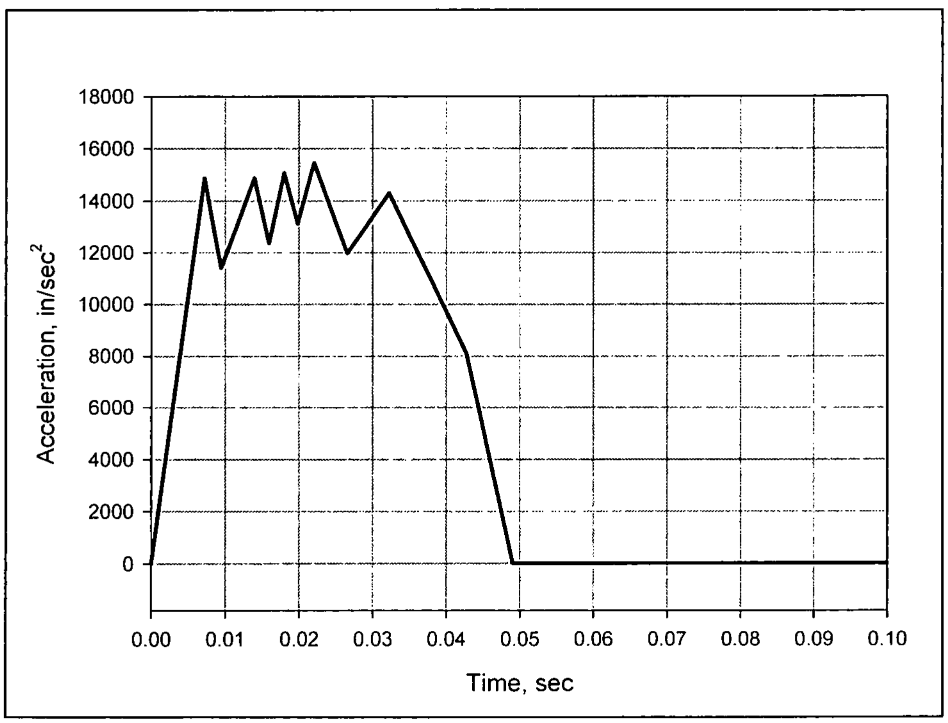


Figure 2.13.6.15-4 Maximum Von-Mises Stress (psi) Time History for the Redwood Impact Limiter Configuration

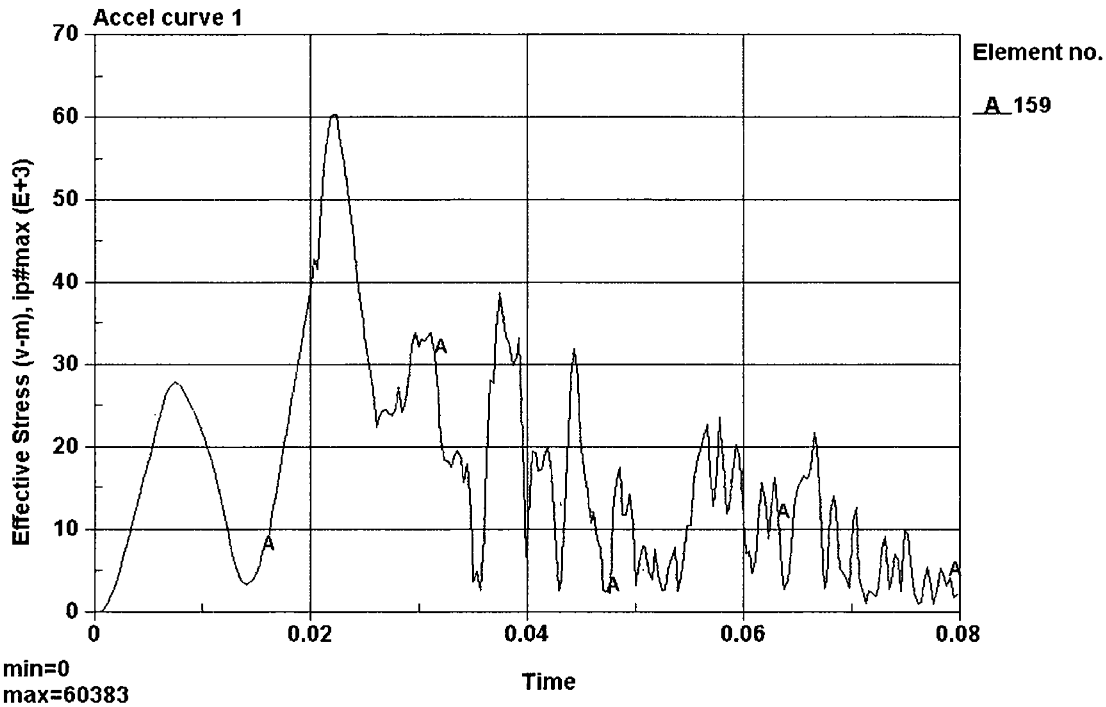


Figure 2.13.6.15-5 Von-Mises Stress (psi) in Fuel Rods at the Time of the Maximum Stress for the Redwood Impact Limiter Configuration

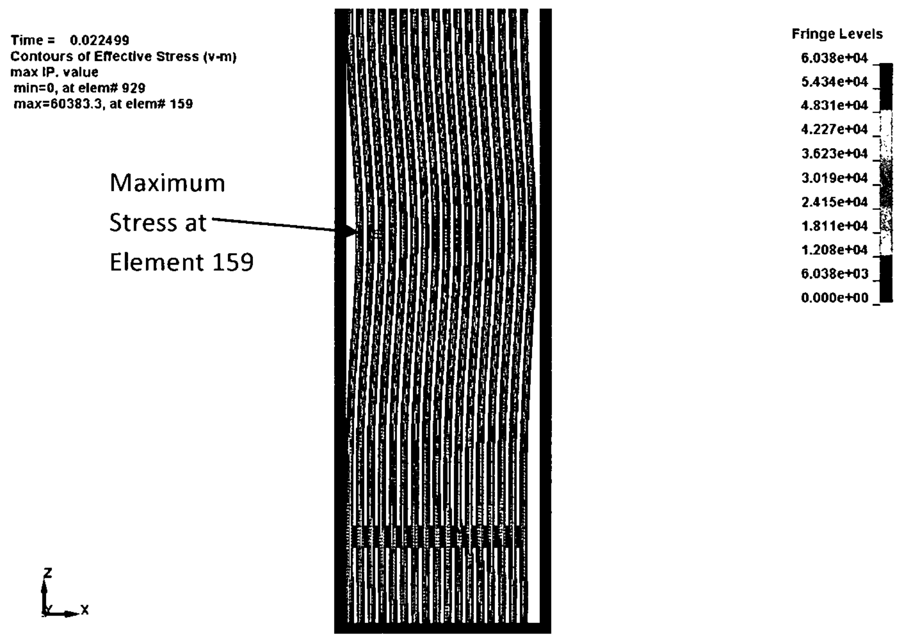


Figure 2.13.6.15-6 Maximum Von-Mises Stress (psi) Time History for the Balsa Impact Limiter Configuration

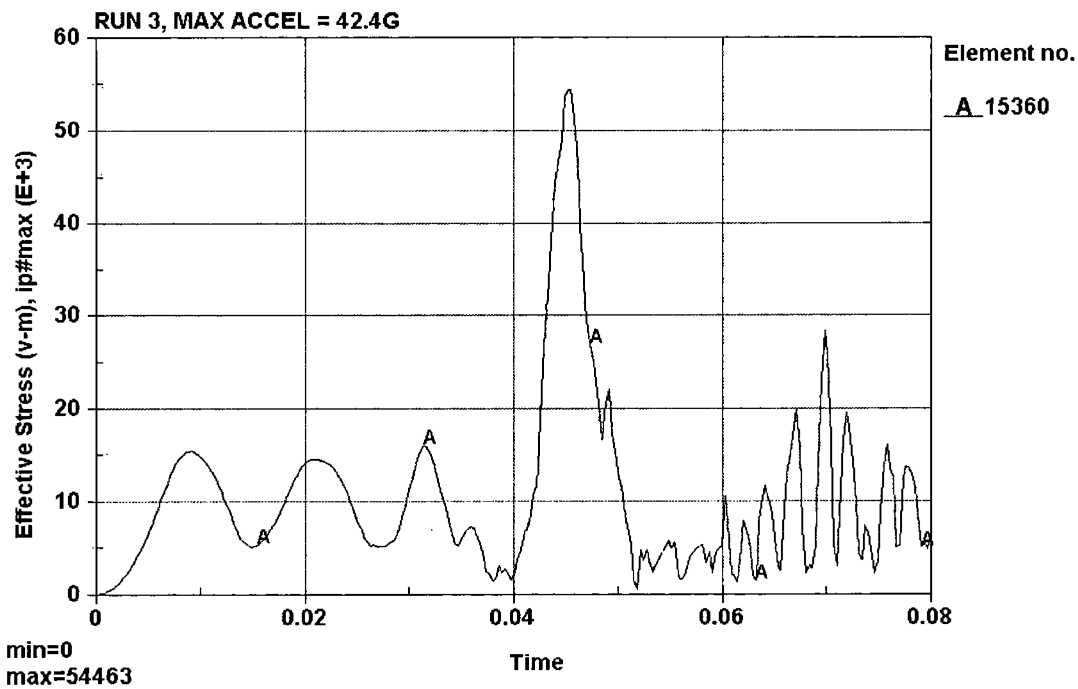
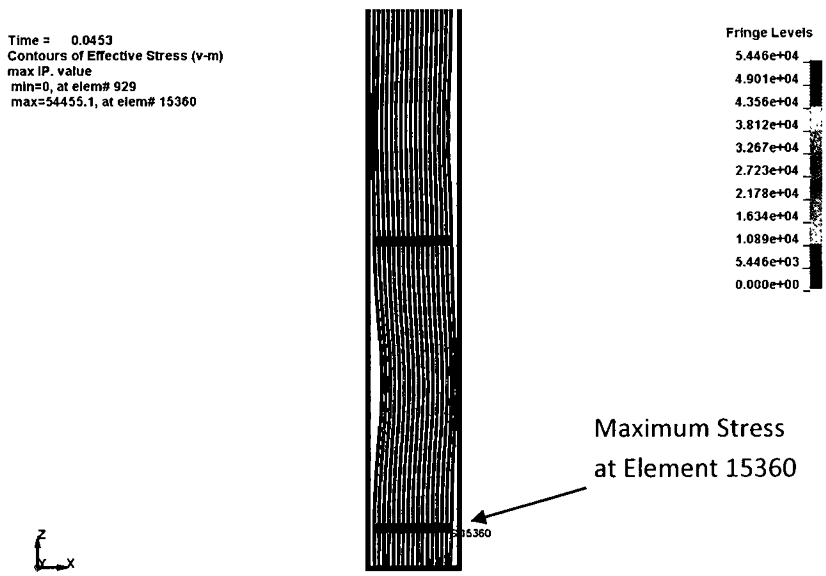


Figure 2.13.6.15-7 Von-Mises Stress (psi) in Fuel Rods at the Time of the Maximum Stress for the Balsa Impact Limiter Configuration



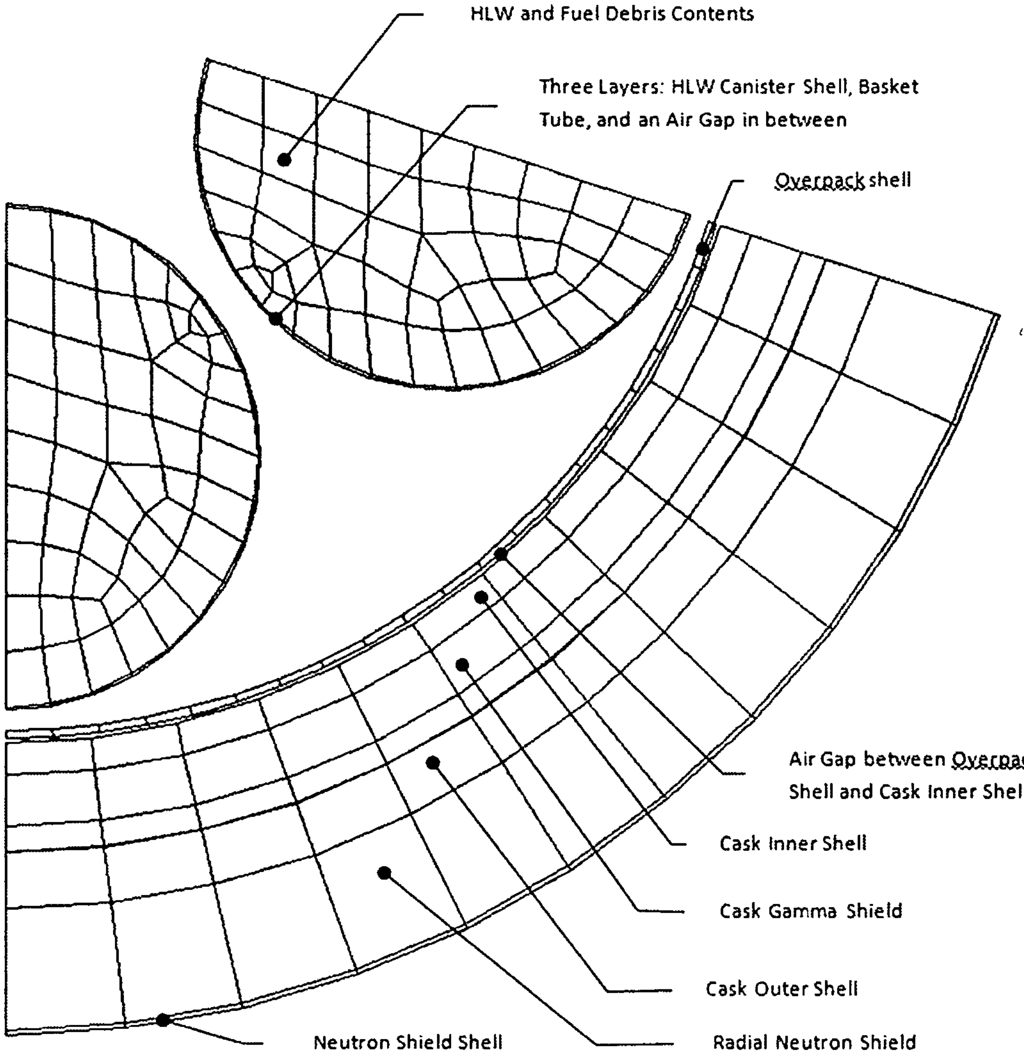
2.13.6.15.2 Fuel Rod Assessment for HBU Fuel for 30-foot Side Drop

[redacted]

[redacted]

This confirms that the PWR fuel rods remain intact for a 60g side drop load condition.

Figure 3.7-2 Three-Dimensional Model for STC-WVDP – Cross-Section



For clarity the air region inside the overpack and outside the basket tubs is not shown.

Table 3.7-4 Maximum Component Temperatures – Normal Transport Conditions, Maximum Decay Heat and Maximum Ambient Temperature – STC-WVDP

	Hot Case 100°F Ambient (°F)
HLW and contents (max.)	206
HLW canister and Basket Tubes (max.)	202
HLW Overpack Shell (max.)	168
HLW Overpack Lid (max.)	156
Average Air Temp. inside the HLW Overpack	184
Cask Inner Shell (max.)	151
Gamma Shield (Lead) (max.)	151
Cask Outer Shell (max.)	147
Radial Neutron Shield (max.)	146
Cask Surface (Neutron Shield Shell) (max.)	145
Average Air Temperature inside the Cask	155
Inner lid and Port Cover Plate O-rings (Metallic) (Max.)	156*

* HLW Overpack Lid maximum temperature conservatively used

Table 3.7-5 Maximum Component Temperatures – Normal Transport Conditions, Maximum Decay Heat, Minimum Ambient Temperature – STC-WVDP

	Cold Case -40°F Ambient (°F)
HLW and contents (max.)	91
HLW canister and Basket Tubes (max.)	87
HLW Overpack Shell (max.)	32
HLW Overpack Lid (max.)	14
Average Air Temp. inside the HLW Overpack	58
Cask Inner Shell (max.)	0
Gamma Shield (Lead) (max.)	-1
Cask Outer Shell (max.)	-5
Radial Neutron Shield (max.)	-6
Cask Surface (Neutron Shield Shell) (max.)	-7
Average Air Temperature inside the Cask	11
Inner lid and Port Cover Plate O-rings (Metallic) (Max.)	14*

* HLW Overpack Lid maximum temperature conservatively used

THIS PAGE INTENTIONALLY LEFT BLANK

3.7.5 Hypothetical Accident Thermal Evaluation – STC-WVDP

The objective of the thermal analysis of the STC-WVDP under hypothetical accident conditions is to demonstrate that the cask containment boundary structural components are maintained within their safe operating temperature ranges.

The maximum decay heat load for the NAC-STC loaded with the HLW contents (STC-WVDP) is 1.5 kW. This total heat load is significantly less than the assumed NAC-STC directly loaded fuel heat load of 22.1 kW. Consequently, when transporting the STC loaded with the HLW and contents, cask component steady-state temperatures are lower than those for NAC-STC directly loaded fuel. The hypothetical fire imposes a large, but short duration, heat load on the STC loaded with the HLW contents. The fire accident causes the cask component temperatures to rise; but because they initially start at a lower temperature (compared to the NAC-STC directly loaded fuel configuration), the maximum post fire accident conditions are also lower. Consequently, the cask component temperatures due to the fire for the 22.1 kW heat load bound the cask component temperatures due to the fire for the 1.5 kW heat load for the STC loaded with the HLW contents.

The maximum temperatures for the NAC-STC loaded with the HLW contents are established by adding a ΔT of 309°F to the STC-WVDP normal condition basket and clad temperatures as listed in Table 3.7-4 (hot case). The ΔT is the fire accident temperature increase of the NAC-STC (directly loaded configuration) cask inner shell surface (460°F, Figure 3.5-2 of the NAC-STC Transport Cask SAR) from the normal condition (151°F, Table 3.7-4). The use of the ΔT in this manner assumes the temperature increase of the inner shell is instantaneously transferred to the canister center. This is conservative since it neglects thermal mass of the overpack, canister, basket and contents. The evaluation results of the NAC-STC cask loaded with the HLW contents for the fire accident analysis are listed in Table 3.7-6. The maximum component temperatures for the cask body are conservatively taken as those shown in Table 3.5-1 of the NAC-STC Transport Cask SAR for directly loaded fuel.

Test Model

NAC International did not create a thermal test model. The methods previously described have been used in prior transport licensing and are sufficient to show that the STC-WVDP meets the criteria set forth in Section 3.5 of the NAC-STC Transport Cask SAR.

Maximum Pressure During Hypothetical Accident Condition of Transport

STC-WVDP cask internal pressure is determined in Section 4.7.

Table 3.7-6 Maximum Temperature of the HLW and Contents, Basket, and HLW Overpack – Hypothetical Accident Condition Fire Transient

Component	**Max. Temperature (°F)**
HLW and contents (maximum)	515
HLW canister and Basket Tubes (maximum)	511
HLW Overpack Shell (maximum)	477
HLW Overpack Bottom Plate (maximum)	487
HLW Overpack Lid (maximum)	465
Inner lid and Port Cover Plate O-rings (Metallic) (Max.)	465*
Average Air Temp. inside the HLW Overpack	493

* HLW Overpack Lid maximum temperature conservatively used

3.8.1 Discussion – STC-HBU and Contents

The loaded STC-HBU basket may be transported in the NAC-STC transport cask. The STC-HBU directly loaded system has three separate configurations as listed below and defined in Figure 3.8-1.

[redacted]

This section demonstrates that the STC-HBU system with the design basis payload of 24 kW, the maximum heat load for the directly loaded STC-HBU fuel thermal evaluation, meets the thermal performance requirements of 10 CFR 71, Sections 71.71 and 71.73, and the requirements of IAEA Safety Standards Series No. SSR-6. The STC cask provides leaktight containment of the spent fuel that it holds. This leaktight level of containment is maintained in all of the design basis normal conditions of transport (Section 3.8.4) and hypothetical accident conditions (Section 3.8.5).

During normal transport and hypothetical accident conditions, the cask must reject the fuel decay heat to the environment without exceeding the operational temperature ranges of the cask seals or other components important to safety. In addition, fuel rod integrity must be maintained for normal transport conditions. This is accomplished by maintaining the fuel in an inert atmosphere and at a sufficiently low temperature that thermally induced fuel rod cladding deterioration is precluded. For the STC-HBU fuel, the maximum fuel rod cladding temperature under normal transport conditions remains below 350°C. The maximum fuel rod cladding temperature under hypothetical fire accident condition remains below 570°C. The thermally induced stresses, in combination with pressure and mechanical load stresses must be maintained below allowable stress levels.

Heat is transferred from the loaded STC-HBU to the environment by passive means only. No forced cooling is used. Heat is transferred from the fuel assemblies to the fuel tubes and through the tubes to the fuel basket support disks and heat transfer disks by conduction and radiation. The aluminum shunts inside fuel tubes also mitigate heat concentration of the basket in both cask

radial direction and cask axial direction. Radiation and conduction are the means by which heat is transferred from the support disks and heat transfer disks to the cask inner shell. From the inner shell, heat is conducted first through the lead gamma shield and through the gap between the lead and the outer shell, and then to the cask outer shell. The outer shell is surrounded by a neutron shield, which conducts the heat to the neutron shield surface, primarily through the copper/stainless steel fins located within the radial neutron shield. The radial neutron shield stainless steel shell is exposed to the environmental ambient temperature. Heat is removed from the cask radial surface by convection and radiation. Because of the impact limiters, essentially no heat is removed through the ends of the cask. The bounding thermal conditions for the analysis required by 10 CFR 71 and IAEA Safety Standards Series No. SSR-6 under normal transport conditions for the STC-HBU are presented in Table 3.8-1. The analyses for these thermal conditions are described in Section 3.8.4.

The design basis fuel assembly for the STC-HBU is the 17×17 PWR fuel assemblies with assembly decay heats [redacted] The thermal analysis for the STC-HBU uses three finite element models evaluated using the ANSYS program.

A three-dimensional model is employed to evaluate the cask in a horizontal position with contact between the basket and the cask inner shell. The uniform gap between the support disk and inner shell, and the uniform gap between the heat transfer disk and the inner shell are computed based on dimensions adjusted for thermal expansions. In this model, circumferential half-symmetry is used. The model is comprised of the fuel assemblies, fuel tubes, aluminum shunts, stainless steel support disks, aluminum heat transfer disks, the NAC-STC inner shell, lead, outer shell, neutron shield and neutron shield shell. The volumetric heat generation rate of the fuel is factored by the peaking factor for high burnup fuels to reflect the power density levels within the fuel. Gas inside the NAC-STC, is modeled as helium. Solar insolation and ambient temperature conditions are applied to the neutron shield shell when appropriate. The model is analyzed to determine the maximum temperatures for the fuel clad, basket including aluminum shunts, cask shells, radial shielding and cask surface.

A second model, which is a detailed two-dimensional thermal model of the fuel assembly, determines the effective orthotropic conductivity of the fuel. The model includes the fuel pellets, cladding and gas between fuel rods and gas occupying the gap between the fuel pellets and cladding. Modes of heat transfer modeled include conduction and radiation between individual fuel rods. Radiation between the fuel pellets and cladding is neglected. Details for this model are described in Section 3.8.4.1.1.2.

[redacted]

applied to the pellets. The temperature at the boundary of the fuel assembly model is constrained to be uniform. The effective conductivity is determined based on the heat generated and the temperature difference between the center and the edge of the model. The temperature-dependent effective properties that follow are established by using different boundary temperatures. The effective conductivity in the axial direction of the fuel assembly is calculated based on the material area ratio.

Temperature (°F)	k_x (Btu/hr-in-°F)	k_y (Btu/hr-in-°F)	k_z (Btu/hr-in-°F)
3	0.0207	0.0207	0.1393
146	0.0239	0.0239	0.1379
288	0.0294	0.0294	0.1326
431	0.0355	0.0355	0.1259
576	0.0425	0.0425	0.1235
721	0.0506	0.0506	0.1248

Where the x and y axes in the above Table are perpendicular to the cask axis and define the plane of the model. The z axes are parallel to the cask axial direction. The temperature associated with each row of properties is the average temperature of the fuel assembly determined by each analysis.

3.8.4.1.1.3 Two-Dimensional Fuel Tube Model for the STC-HBU

The purpose of the two-dimensional fuel tube model is to determine the effective thermal property of the fuel tube, which is used in the three-dimensional transport cask and the basket model.

The model of the fuel tube with neutron absorber is shown in Figure 3.8-6. This model has five layers, which includes the fuel tubing, the neutron absorber, media gaps on both sides of the neutron absorber, and the stainless steel retainer. The media is considered as helium for the model.

Modes of heat transfer modeled include conduction and radiation. Convection is conservatively neglected. ANSYS PLANE55 conduction elements and LINK31 radiation elements are used to construct the model. The model consists of layers of conduction elements and radiation elements that are defined at the helium gaps (two for each gap). The thickness of the model is the distance measured from the inside face of the fuel tubing to the outer surface of the stainless steel retainer. The height of the model is defined as equal to the width of the model.

[redacted] temperature at the right boundary of the model is constrained. The maximum temperature of the model (at the left boundary) and the temperature difference (ΔT) across the model are calculated by ANSYS. The effective conductivities for the fuel tube are determined using the same methodology described in Section 3.4.1.2.3.

3.8.4.1.1.4 Test Model

NAC International did not create a thermal test model. The methods previously described have been used in previous transport licensing and are sufficient to show that the STC-HBU meets the criteria set forth in Section 3.8.4.

3.8.4.2 Maximum Temperatures

This section presents the maximum component temperatures for the STC-HBU. Temperatures are calculated using the model described in Section 3.8.4.1.1.

Using the thermal model described in Section 3.8.4.1.1, temperatures for the major components of the cask body, basket, and fuel cladding are determined for the normal conditions of transport. The STC-HBU cask body maximum allowable component temperatures are shown in Section 3.8.3.2. The maximum temperatures of the major STC-HBU components, the basket components, and fuel rod cladding temperatures, are shown in Tables 3.8-4 and 3.8-5.

[redacted]

[redacted]

[redacted]

3.8.4.3 Minimum Temperatures

The minimum temperatures in the cask occur with no heat load and -40°F ambient temperature, yielding a uniform -40°F temperature distribution throughout the STC-HBU package.

3.8.4.4 Maximum Thermal Stresses

The ANSYS computer code is used to obtain temperatures for use in the structural analyses for the STC-HBU. These temperatures are presented in Tables 3.8-4 and 3.8-5. The thermal stress calculations are performed in Sections 2.13.6.12.3 and 2.13.6.14.2.

3.8.4.5 Summary of the STC-HBU Performance for Normal Conditions of Transport

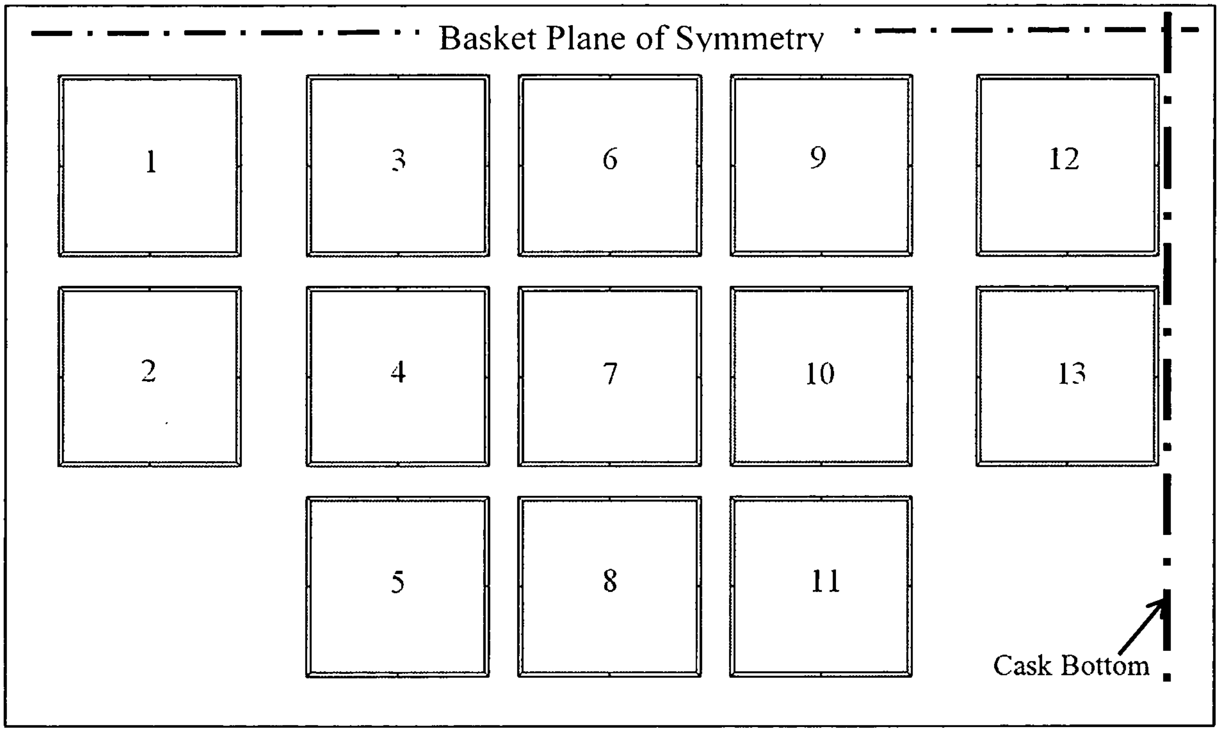
Results of the thermal analysis of the STC-HBU for normal transport conditions are summarized in Tables 3.8-4 and 3.8-5. The maximum fuel rod cladding temperature is maintained below 350°C; temperatures of safety-related cask components are maintained within their safe operating ranges; and thermally-induced stresses in combination with pressure and mechanical load stresses are shown in the structural analysis of Chapter 2 to be less than the allowable stresses for both of the transport configurations. Therefore, the analyses in Section 3.8.4 demonstrate that the STC-HBU can safely transport the high burnup fuels under the normal transport conditions specified in 10 CFR 71.71.

The maximum cask surface temperature of the STC-HBU for the 100°F ambient condition is 264°F, which is 6°F higher than the maximum cask surface temperature of 258°F of the Yankee-MPC with a heat load of 12.5 kW. The increase of 6°F would not significantly alter the natural convection from the cask surface in comparing the STC-HBU and Yankee-MPC thermal performances. By adding this 6°F to the maximum personnel barrier temperature of 140°F for the Yankee-MPC, the maximum temperature of the personnel barrier for the STC-HBU is conservatively estimated to be 146°F, which is well below the allowable temperature of 185°F for exclusive use shipment. Therefore, the STC-HBU can safely transport the design basis fuel under the normal conditions of transport specified in 10 CFR 71.43.

3.8.4.6 Sensitivity Analysis for Maximum Variation of Fuel Cladding Temperature

[redacted]

Figure 3.8-1 Configuration Definition for the STC-HBU



[redacted]

Figure 3.8-6 Two-Dimensional Fuel Tube Model for the STC-HBU

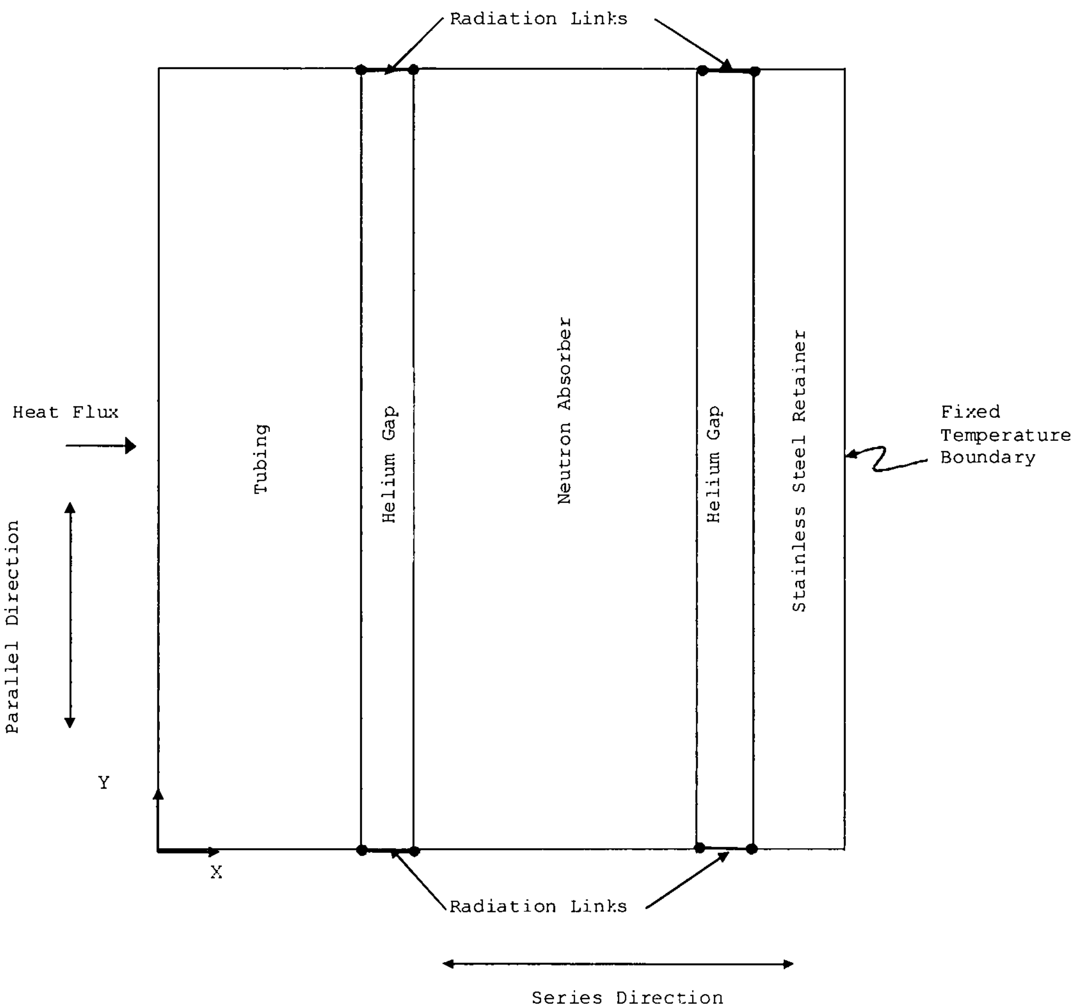


Table 3.8-4 Maximum Component Temperatures—Normal Transport Conditions, Maximum Decay Heat, Maximum Ambient Temperature, among Three Configurations – the STC-HBU

	Hot Case 100°F Ambient (°F)
Fuel (Max.)	638
Support Disk (Max.)	577
Heat Transfer Disk (Max.)	550
Aluminum Shunt (Max.)	499
Radial Neutron Shield (NS-4-FR) (Max.)	295
Lead Shield (Max.)	363
Inner Shell (Max.)	376
Outer Shell (Max.)	303
Average Gas Temperature	438
Cask Surface Temperature (Max.)	264
Inner Lid and Port Cover Plate O-rings (Viton) (Max)	256

Table 3.8-5 Maximum Component Temperatures - Normal Transport Conditions, Maximum Decay Heat, Minimum Ambient Temperature, among Three Configurations – the STC-HBU

	Cold Case -40°F Ambient (°F)
Fuel (Max.)	546
Support Disk (Max.)	476
Heat Transfer Disk (Max.)	447
Aluminum Shunt (Max.)	395
Radial Neutron Shield (NS-4-FR) (Max.)	178
Lead Shield (Max.)	253
Inner Shell (Max.)	268
Outer Shell (Max.)	186
Average Gas Temperature	333
Cask Surface Temperature (Max.)	148
Inner Lid and Port Cover Plate O-rings (Viton) (Max)	136

THIS PAGE INTENTIONALLY LEFT BLANK

3.8.5 Hypothetical Accident Thermal Evaluation – the STC-HBU

The objective of the thermal analysis of the STC-HBU under hypothetical accident conditions is to demonstrate that the cask containment boundary structural components are maintained within their safe operating temperature ranges.

The cask body and the basket design used for the STC-HBU are identical to the cask body and basket for the directly loaded fuel. The heat transfer performance of the cask body and the basket support disks and aluminum heat transfer disk for both configurations would also be the same. The manner in which the fire accident heat is infused into the basket for both designs is therefore the same also. Due to this similarity, the component temperature increase due to the fire for the components of directly loaded fuel heat load of 22.1 kW is used to determine the component temperature increase of the STC-HBU for the fire condition. The fire condition temperature increase (ΔT) for the components of directly loaded fuel heat load of 22.1 kW is obtained based on the analysis results from Table 3.4-1 and Table 3.5-1. The temperature increase of the aluminum disk is used as the temperature increase for the aluminum shunt. By adding this temperature increase (ΔT) to the maximum temperature of the corresponding component for the STC-HBU (from Table 3.8-4), the maximum component temperatures due to the fire for the STC-HBU are obtained and are listed in Table 3.8-6. Note that this method is conservative since it ignores the thermal inertia of the loaded basket.

The analysis results of the STC-HBU under hypothetical accident conditions demonstrate that the cask containment boundary structural components are maintained within their safe operating temperature ranges.

Table 3.8-6 Maximum Temperature of the STC-HBU – Hypothetical Fire Accident Condition

	Max. Temperature (°F)	**Allowable Temperature (°F)***
Inner Lid Bolts	402	-
Inner Lid and Port Cover Plate O-rings (Viton)**	387	400**
Cask Radial Outer Surface	1368	-
Radial Neutron Shield	-	-
Lead Gamma Shield	503	600
Aluminum Disk Interior	695	800
Support Disk Interior	721	800
Aluminum Shunt	644	800
Fuel Rod Cladding, Directly Loaded Fuel	823	1,058

Notes:

* Allowable temperatures for fire accident condition are taken from Table 3.5-1.

** The stated allowable temperature for the Viton seal is for a steady state condition, not the allowable temperature for a temporary condition, which would be larger.

Test Model

NAC International did not create a thermal test model. The methods previously described have been used in prior transport licensing and are sufficient to show that the STC-HBU meets the criteria set forth in Section 3.5.

Table of Contents

Table of Contents (continued)

4.0 CONTAINMENT

4.1 Containment Boundary

The NAC-STC transport containment boundary is designed and analyzed to ensure the containment of the cask contents in accordance with 10 CFR 71.51. The containment boundary is designed, fabricated and inspected in accordance with ASME Code Section III, Subsection NB, with the exception of code stamping.

The containment evaluation of the NAC-STC for transport of the MPC-LACBWR canister is presented in Section 4.6 of this SAR.

The components of the containment boundary are described in Table 4.1-1 as a function of the containment condition and the contents. The containment conditions are:

- Containment Condition A: The containment boundary for the transport of directly loaded (i.e., no canister) intact PWR spent fuel assemblies following extended storage of the cask at an ISFSI licensed in accordance with 10 CFR 72.

- Containment Condition B: The containment boundary for the transport of: (1) directly loaded intact/undamaged PWR spent fuel assemblies loaded immediately prior to transport using either metallic or nonmetallic O-rings; or (2) canistered Yankee Class, Connecticut Yankee or DPC LACBWR spent fuel assemblies, Reconfigured Fuel Assemblies, Damaged Fuel Can or GTCC waste, or (3) MPC-WVDP HLW overpacks loaded into the NAC-STC immediately prior to transport using metallic O-rings.

The NAC-STC containment boundary is designed to permit leak testing of the cask containment boundary penetrations prior to transport to confirm the containment requirement of the contents. The leak test criteria, minimum test sensitivity and leak test methods and locations for each containment condition are described in Table 4.1-1.

4.1.1 Containment Vessel

The primary containment vessel for the NAC-STC consists of a 71.0-inch inside diameter, 1.5-inch thick inner shell, two 1.5-inch to 2.0-inch thick transition sections, a 6.2-inch thick bottom inner forging, and a 7.85-inch thick top forging. The containment vessel components, except for the transition sections, are fabricated from ASME Boiler and Pressure Vessel Code, Type 304 stainless steel nuclear pressure vessel material. The two transition sections are ASME Boiler and Pressure Vessel Code, Type XM-19 stainless steel nuclear pressure vessel material.

The weld examination requirements for the cask body are defined in Table 4.1-2 and are shown on the drawings in Section 1.3.2.

4.1.2 Containment Penetrations

The physical penetrations in the NAC-STC containment vessel are the inner lid and the vent and drain ports in the inner lid. The penetrations are designed to ensure sealing of the containment boundary and to ensure that the leakage from the boundary does not exceed 1×10^{-7} ref cm^3/sec using metallic seals, or either 1×10^{-7} ref cm^3/sec or 7.5×10^{-5} ref cm^3/sec using nonmetallic O-rings. The quick-disconnect fittings installed in the vent and drain openings and in the interseal test port in the inner lid are not considered to be part of the containment boundary.

4.1.3 Seals and Welds

4.1.3.1 Seals

The O-rings of the inner lid, the vent port coverplate and the drain port coverplate are the seals that provide primary containment, as described in Section 4.1 and as shown in Table 4.1-1. Section 4.5 contains the specifications that describe the PTFE O-rings of the interlid and pressure port covers and the metallic or nonmetallic O-rings used in the containment boundary and outer lid. Also included in Section 4.5 are the manufacturer's technical data bulletins for the expansion foam and the Fiberfrax Ceramic Fiber Paper. Leak testing of the cask is performed prior to acceptance from the manufacturer. Leak testing is also performed following fuel loading for either immediate transport or for transport following a storage period. Technical information for Viton O-rings is provided in Section 4.5.5.

4.1.3.1.1 Containment System Fabrication Verification

Upon completion of fabrication, a Containment System Fabrication Verification shall be performed on the cask containment boundary as described in Section 8.1.3. These leak tests verify that the leakage rate of containment components does not exceed the maximum allowable leakage rate of 1×10^{-7} ref cm^3/sec for metallic O-rings or the total (cumulative) leakage rate of containment components does not exceed 7.5×10^{-5} ref cm^3/sec for Viton O-rings. Viton O-rings may also be tested to leaktight conditions for HBU PWR spent fuel assembly contents by confirming the maximum allowable leakage rate of 1×10^{-7} ref cm^3/sec (leaktight testing applies to individual seals). The allowable leak test shall conform to the O-ring design, since the inner and outer lids, and the vent and drain port coverplates must be fabricated using the O-ring groove appropriate to the O-ring design. Metallic O-rings and nonmetallic O-rings cannot be used interchangeably.

4.1.3.1.2 Containment System Verification

The Containment System Verification shall be performed on the NAC-STC package containment boundary seals and components prior to each shipment for packages assembled with metallic O-rings in accordance with the leak test acceptance criteria defined in the procedures in Chapter 8. For cask transport immediately after loading, the leak test shall be performed in accordance with the procedures and acceptance criteria described in Section 7.4.1. For cask shipments following storage, the verification leak test shall be performed in accordance with the procedures and acceptance criteria described in Section 7.4.2.

Whenever a containment component is replaced, the containment component shall be leak tested following replacement using the Maintenance Leakage Rate Test (Section 8.2.2.2). This test will verify that the replacement component has been properly installed and that the leakage rate meets acceptance criteria. For packaging using Viton O-rings, the Periodic Leakage Rate Test shall be performed annually, or when a Viton O-ring is replaced, in accordance with the acceptance criteria in Section 8.2.2.2.

4.1.3.2 Welds

The NAC-STC containment vessel is assembled by welding. A list of containment vessel welds, the examinations and tests performed on the welds, and the applicable ASME Code acceptance criteria are provided in Table 4.1-2. The acceptance tests for the NAC-STC are provided in Section 8.1.

4.1.4 Closure

The primary closure assembly for the NAC-STC containment for transport consists of the inner lid, bolts, and O-rings. The inner lid is recessed and bolted into the top forging of the cask body. The 9.0-inch thick, 79.00-inch diameter inner lid is made of SA-336, Type 304 stainless steel. The inner lid is retained by 42 inner lid bolts that are 1 1/2 - 8 UNC socket head cap screws fabricated from SB-637, Grade N07718 nickel alloy steel bolting material. The initial torque for installation of the inner lid bolts is specified in Table 7-1.

The vent port and the drain port are recessed into the inner lid. The vent and drain port coverplates are secured by four 1/2 - 13 UNC bolts fabricated from SA-193, Grade B6, Type 410 stainless steel. Each coverplate is sealed to the inner lid by a metallic or nonmetallic O-ring, with a second, concentric O-ring of the same material, providing an annulus to test the seal.

A secondary closure is provided by the outer lid which provides puncture protection to the primary closure assembly. The 5.25-inch thick, 86.7-inch diameter outer lid is made of SA-705, Type 630, H1150, 17-4 PH stainless steel. The outer lid is retained by 36 outer lid bolts that are 1 - 8 UNC socket head cap screws fabricated from SA-564, Type 630, H1150, 17-4 PH stainless steel. The initial torque for installation of the outer lid bolts is specified in Table 7-1. The bottom surface of the outer lid is sealed to the top forging of the cask body by a metallic or nonmetallic O-ring.

Port covers protect the interlid and pressure ports, which are located in the top forging and access the region between the inner and outer lids. For transport operations, solid port covers with no penetrations are installed in the interlid and pressure ports. These port covers are secured by three 3/8 - 16 UNC bolts, fabricated from SA-193, Grade B6, Type 410 stainless steel material. Each cover is sealed to the cask body by two "piston-type" (bore seal) PTFE O-rings, with a test port located between the O-rings.

Table 4.1-1 NAC-STC Containment Boundaries

Containment Condition	Content Condition	Containment Components	Allowable Test Leakage Rate/Sensitivity	Test Location/Method	Remarks
A Using metallic O-rings	Up to 26 directly loaded intact PWR spent fuel assemblies following storage operations per 10 CFR 72 having burnups of ≤ 45,000 MWd/MTU	• Inner shell • Upper and lower shell rings (transition sections) • Bottom inner forging • Top forging • Inner lid • Inner lid outer metallic O-ring • Inner lid interseal test port threaded plug with metallic O-ring • Vent port coverplate • Vent port coverplate outer metallic O-ring • Vent port coverplate interseal port threaded plug with metallic O-ring • Drain port coverplate • Drain port coverplate outer metallic O-ring • Drain port coverplate interseal test port plug with metallic O-ring	• Allowable leakage rate is $\leq 2 \times 10^{-7}$ cm^3/sec (helium) (i.e., leaktight) • Minimum test sensitivity is $\leq 1 \times 10^{-7}$ cm^3/sec (helium)	• Evacuated envelope method (envelope provided by outer lid with test performed through the interlid port) with helium in interseal regions between lid and vent and drain coverplate O-rings	These series of leak tests are performed on the NAC-STC containment boundary following directly loaded fuel storage operations. The outer O-rings are the designated boundary as access to the cask cavity to verify helium backfill conditions is not planned. Testing is performed in accordance with ANSI N14.5 requirements immediately prior to transport.

Table 4.1-1 NAC-STC Containment Boundaries (Continued)

Containment Condition	Content Condition	Containment Components	Allowable Test Leakage Rate/Sensitivity	Test Location/Method	Remarks
B Using metallic O-rings	Up to 26 directly loaded intact/undamaged PWR spent fuel assemblies for immediate transport, or canistered Yankee Class or Connecticut Yankee or MPC-LACBWR spent fuel assemblies, Reconfigured Fuel Assemblies, Damaged Fuel Cans or GTCC waste and MPC-WVDP HLW overpack	• Inner shell • Upper and lower shell rings (transition sections) • Bottom inner forging • Top forging • Inner lid • Inner lid inner O-ring • Vent port coverplate • Vent port coverplate inner O-ring • Drain port coverplate • Drain port coverplate inner O-ring	• Allowable leakage rate is $\leq 2 \times 10^{-7}$ cm^3/s (helium) (i.e., leaktight) for each component test • Minimum test sensitivity is $\leq 1 \times 10^{-7}$ cm^3/s (helium)	• Vent port inner O-ring. • Drain port inner O-ring. • Inner lid inner O-ring. • Tests performed at the interseal test ports with helium in the cask using the evacuated envelope method (envelope provided by the inner and outer O-rings).	Inner O-rings form the containment boundary. Testing is performed in accordance with ANSI N14.5 requirements.

Table 4.1-1 NAC-STC Containment Boundaries (Continued)

Containment Condition	Content Condition	Containment Components	Allowable Test Leakage Rate/Sensitivity	Test Location/Method	Remarks
B Using either non-metallic (e.g., Viton) O-rings or metallic seals	Up to 26 directly loaded intact/undamaged PWR spent fuel assemblies having burnups of ≤ 45,000 MWd/MTU for immediate transport.	• Inner shell • Upper and lower shell rings (transition sections) • Bottom inner forging • Top forging • Inner lid • Inner lid inner O-ring • Vent port coverplate • Vent port coverplate inner O-ring • Drain port coverplate • Drain port coverplate inner O-ring	Either • Allowable leakage rate is $\leq 9.3 \times 10^{-5}$ cm^3/s (helium) for the total of the three leakage tests • Minimum test sensitivity is $\leq 4.7 \times 10^{-5}$ cm^3/s (helium) for nonmetallic seals; Or, • Allowable leakage rate is $\leq 2.0 \times 10^{-7}$ cm^3/s (helium) (leaktight) • Minimum test sensitivity is $\leq 1.0 \times 10^{-7}$ cm^3/s (helium) for metallic seals.	• Vent port inner O-ring. • Drain port inner O-ring. • Inner lid inner O-ring. • Tests performed at the interseal test ports with helium in the cask using the evacuated envelope method (envelope provided by the inner and outer O-rings).	Inner O-rings form the containment boundary. Testing is performed in accordance with ANSI N14.5 requirements.

Table 4.1-1 NAC-STC Containment Boundaries (Continued)

Containment Condition	Content Condition	Containment Components	Allowable Test Leakage Rate/Sensitivity	Test Location/Method	Remarks
B Using non-metallic (e.g., Viton) O-rings	Up to 20 directly loaded intact/undamaged HBU PWR spent fuel assemblies with burnups of ≤ 60,000 MWd/MTU for immediate transport.	• Inner shell • Upper and lower shell rings (transition sections) • Bottom inner forging • Top forging • Inner lid • Inner lid inner O-ring • Vent port coverplate • Vent port coverplate inner O-ring • Drain port coverplate • Drain port coverplate inner O-ring	• Allowable leakage rate is $\leq 2 \times 10^{-7}$ cm^3/sec (helium) (i.e., leaktight) • Minimum test sensitivity is $\leq 1 \times 10^{-7}$ cm^3/sec (helium)	• Vent port inner O-ring. • Drain port inner O-ring. • Inner lid inner O-ring. • Tests performed at the interseal test ports with helium in the cask using the evacuated envelope method (envelope provided by the inner and outer O-rings).	Inner O-rings form the containment boundary. Testing is performed in accordance with ANSI N14.5 requirements.

Table 4.1-2 NAC-STC Containment Boundary Welds, Examinations and Tests

Primary Containment Boundary				
Weld Location	**Weld Type**	**ASME Code Category**	**Inspections/Tests**	**ASME Acceptance Criteria**
Inner shell longitudinal and inner shell rings longitudinal	Full Penetration Double Groove	A	VT on Tack Welds VT Final Pass PT Final Pass RT Final Weld Hydrostatic Test Post Hydrostatic Test – PT Helium Leak Test	NB-4424 and NB-4427 NB-5350 NB-4424 and NB-4427 NB-5320 NB-6000 NB-5350 Section V, Art. 10 and ANSI N14.5
Inner shell circumferential	Full Penetration Double Groove	B	VT on Tack Welds VT Final Pass PT Final Pass RT Final Weld Hydrostatic Test Post Hydrostatic Test – PT Helium Leakage Test	NB-4424 and NB-4427 NB-5350 NB-4424 and NB-4427 NB-5320 NB-6000 NB-5350 Section V, Art. 10 and ANSI N14.5
Inner shell to top and bottom inner shell rings circumferential	Full Penetration Double Groove	B	VT on Tack Welds VT Final Pass PT Final Pass RT Final Weld Hydrostatic Test Post Hydrostatic Test – PT Helium Leakage Test	NB-4424 and NB-4427 NB-5350 NB-4424 and NB-4427 NB-5320 NB-6000 NB-5350 Section V, Art. 10; and ANSI N14.5

Table 4.1-2 NAC-STC Containment Boundary Welds, Examinations and Tests (Continued)

Primary Containment Boundary				
Weld Location	**Weld Type**	**ASME Code Category**	**Inspections/Tests**	**ASME Acceptance Criteria**
Top inner shell ring to upper forging, bottom inner shell ring to bottom inner forging, shell-to-shell longitudinal and shell-to-shell circumferential	Full Penetration Double Groove	C	VT on Tack Welds VT Final Pass PT Final Pass RT Final Weld Hydrostatic Test Post Hydrostatic Test – PT Helium Leak Test	NB-4424 and NB-4427 NB-5350 NB-4424 and NB-4427 NB-5320 NB-6000 NB-5350 ANSI N14.5

4.2 Containment Requirements for Normal Conditions of Transport

The NAC-STC has been designed to safely transport spent fuel assemblies in either of two configurations. The spent fuel assemblies may be sealed in a transportable storage canister (canistered), or loaded directly into a fuel basket installed in the cask cavity.

In the canistered configuration, the NAC-STC can transport Yankee Class or Connecticut Yankee spent fuel and GTCC waste. The NAC-STC in this configuration is designed and tested to leaktight conditions as defined by ANSI N14.5-1997 and, therefore, meets the requirements of 10 CFR 71.51 for containment of radioactive materials.

For directly loaded fuel, a reference 17×17 fuel assembly having a burnup of 60,000 MWd/MTU, an enrichment of 3.5 wt % ^{235}U and a cool time of 5 years is used to establish the source term for the containment analysis. The reference fuel assembly is also used in the shielding analysis and is described in Section 5.1.2.

The directly loaded fuel is the only payload having the option of employing the nonmetallic seal not tested to leaktight conditions and is, therefore, the only payload addressed in the containment evaluation. Maximum burnup of directly loaded fuel is limited to 60,000 MWd/MTU as discussed in Chapter 1 and Chapter 5. As limited information is available on failure rates in transport and release fractions of radionuclides from high burnup fuel (>45 GWd/MTU), a leaktight boundary is required for transport of high burnup (HBU) PWR fuel. The A_2 and release evaluation using high burnup isotope content is conservative for lower burnup materials. A minimum 5-year cool time is employed in this analysis. While cool times down to 4.9 years are permitted for the HBU material at 60 GWd/MTU, in the 14-assembly configuration, the containment analysis uses a full 26-assembly assumption and a lower enrichment than that allowing 4.9 years (3.5 versus 4.9 wt%). While lower burnups allow shorter cool time (down to 4 years) the radionuclide/gas source for the modeled assemblies (60 GWd/MTU, 3.5 wt% initial enrichment and 5-year cool time) bounds those of the lower burnup combinations.

As shown in Chapter 5, Table 5.2-1, the reference 17×17 fuel assembly contains a slightly lower (<1% difference) fuel mass than the 15×15 reference assembly, but significantly higher mass/source than the 14×14 and 16×16 reference assemblies. Combined with the highest in-core power per assembly (see Table 5.2-2), maximum source terms are calculated for the 17×17 reference assembly.

A minimum enrichment of 3.5 wt % was chosen as a reasonable lower enrichment band for the 60,000 MWd/MTU fuel. This initial enrichment is significantly lower than typical fuel discharged at burnup levels in excess of 50,000 MWd/MTU and bounds lower burnup reduced enrichment combinations.

The power history, including power per assembly and downtime at the end of the cycle, for each reference assembly evaluated is included in SAR Table 5.2-2. For the 17×17 reference fuel assembly type, a power level of 18.55 MW per assembly and 60 days of downtime between cycles is applied. The power level is based on a review of power plant thermal output divided by the number of assemblies in the core (dependent on fuel type selected). The downtime between cycles is an estimate based on overall cycle length, including ramp-up and down in power. A conservative three-power cycle history is applied to generate the 60,000 MWd/MTU burnup. This power history produces a set of three approximately 18-month-long cycles producing a combined time of 54 months between BOL (beginning of life) and discharge. A 54-month fuel life is considered bounding for generating a 60,000 MWd/MTU burnup fuel assembly.

The complete SAS2H model, including power level and cool time, for the generation of containment source terms is included in Section 4.5.6.

For direct loading for immediate transport or transport after interim storage using metallic O-rings in the containment boundary, the containment boundary is tested to a leaktight condition as defined in ANSI N14.5-1997. For direct loading for immediate transport using Viton O-rings, the containment boundary is tested to either leaktight condition (high burnup fuel) or for standard burnup fuel ($\leq$ 45 GWd/MTU assembly average) to 7.5×10^{-5} ref cm^3/sec, or 9.3×10^{-5} cm^3/sec (helium).

The structural integrity of the cask containment during normal conditions of transport is demonstrated in Section 2.6.

4.2.1 Containment of Radioactive Material

The NAC-STC uses one of two O-ring configurations based on the loading condition. For directly loading of fuel for transport without interim storage, the O-rings may be either metallic or Viton. For direct fuel loading for storage, the O-rings must be metallic. For loading of canistered fuel or GTCC waste, or HLW overpacks for transport, the O-rings must also be

metallic. For configurations using metallic O-rings, the containment boundary is designed and tested to leaktight conditions as defined by ANSI N14.5-1997. For direct fuel loading for transport of standard burnup fuel without interim storage using Viton O-rings, the allowable leak rate is calculated using the methodology of NUREG/CR-6487. For high burnup (HBU) fuel the Viton O-ring seals are tested to leaktight conditions. Consequently, the cask meets the requirements of 10 CFR 71.51 for directly loaded and for canistered fuel or GTCC waste, and HLW overpacks.

4.2.2 Pressurization of Containment Vessel

The maximum normal operating pressure in the cask during normal transport conditions is conservatively based on 100% failure of the fuel rods, using the methodology presented in Section 3.4.4. The cask cavity under normal transport conditions is backfilled to one atmosphere with 99.9% pure helium gas. To determine the limiting temperature conditions, it has been assumed that the helium gas could possibly be replaced by air. Therefore, the normal operating pressure is determined for both gas conditions. From Section 3.4.4, the free gas volume, fuel fill gas volume, and fuel fission gas volumes for the two spent fuel configurations are presented below. The GTCC waste and HLW do not release any gas. The Reconfigured Fuel Assemblies and Damaged Fuel Cans contain failed fuel. The initial charge gas and any significant fission product gases have already been released from the Reconfigured Fuel Assemblies and from fuel in the Damaged Fuel Cans.

Regulatory Guide 1.25 suggests that 10% of the tritium and 30% of the krypton-85 should be assumed to be released from each failed fuel rod. It is conservatively assumed that 30% of both tritium and krypton-85 escape each failed fuel rod. Other radiologically important gaseous nuclides are present only in negligible amounts after the minimum cooling period for the design basis directly loaded and canistered fuels. The postulated release of other radionuclides, including volatiles, fines, particulates and crud, does not contribute to an increase in internal pressure.

4.2.2.1 Containment Pressurization Due to Directly Loaded Fuel

An increase in pressure within the containment boundary results from an increase in the cask cavity temperature and the postulated failure of 100 percent of the fuel rods in normal conditions of transport (MNOP).

The pressure with air in the cask cavity, based on a conservative bulk air temperature of 450°F (Section 3.4.4), is 4.3 atm (63.2 psia = 48.5 psig). For standard burnup fuel (≤45 GWd/MTU assembly average) based on a bulk average gas temperature of 401°F when helium is the cover gas, the pressure in the cask cavity is:

$$P_2 = (4.3)\left(\frac{478}{505}\right) = 4.07\text{atm} = 59.8\text{psia} = 45.1\text{psig}$$

This is less than the containment boundary design pressure of 75 psig.

High burnup fuel as content was determined in Section 3.8.6 to produce a slightly higher average gas temperature under normal conditions of 438°F. This is still bounded by the 450°F temperature used in the Section 3.4.4 pressure evaluations. Normal and hypothetical accident condition pressures are addressed in Section 3.8.6 for high burnup fuel.

To address flammability concerns within the context of PWR fuel transport, the hydrogen (in particular tritium) content was extracted from a high burnup (60 GWd/MTU assembly average) SAS2H output. Tritium quantity generated was determined to be larger for higher burnup and lower enrichment. A conservative 1.7 wt%, 60 GWd/MTU, was applied and yielded 0.0429 grams of tritium per assembly at reactor discharge (lower quantity at transport due to decay to helium). For 26 assemblies (conservative) tritium quantity is thereby < 1.2 grams (0.4 mole). At standard conditions, 1 mole of gas occupies ~22.4 liters. As the cavity volume of the system is documented as greater than 7000 liters (see Table 4.2-4), no flammability hazard exists (volume fraction <0.1%).

4.2.2.2 Canister and Cask Pressurization Due to Yankee Class Fuel

The maximum normal operating pressure (MNOP) during transport conditions in the transportable storage canister is calculated in Section 3.4.4, and found to be 3.23 atmospheres, or 32.8 psig. This pressure is conservatively calculated at 450°F, compared to the calculated maximum normal conditions of transport bulk gas temperature of 442°F, and conservatively assumes the rupture of 100% of the fuel rods. The MNOP is below the design pressure of 55 psig. The GTCC waste, Damaged Fuel can contents and Reconfigured Fuel Assemblies classified as failed do not release gases to the canister cavity due to rupture of fuel rods. Consequently, there is no increase in canister internal pressure due to these contents.

Since the canister does not fail in any of the evaluated normal transport or accident conditions, this pressure increase occurs within the canister. There is no pressure increase in the cask cavity except that due to the increase in cavity temperature. As the cask cavity is backfilled to 0 psig, a hypothetical canister failure would result in a containment vessel pressure lower than the canister pressure. Hypothetical canister failure would, therefore, result in significantly lower system pressure than the containment boundary design pressure.

4.2.2.3 Canister and Cask Pressurization Due to Connecticut Yankee Fuel

The MNOP during transport conditions in the transportable storage canister is calculated in Section 3.4.4 and found to be 3.9 atmospheres, or 42.3 psig. This pressure is conservatively calculated at 450°F, compared to the calculated maximum normal conditions of transport bulk gas temperature of 402°F, and conservatively assumes the rupture of 100% of the fuel rods. The MNOP is below the design pressure of 55 psig. As described above, the GTCC waste, Damaged Fuel Can contents and Reconfigured Fuel Assemblies do not release gases to the canister cavity due to failures. Consequently, there is no increase in canister internal pressure due to these contents.

Since the canister does not fail in any of the evaluated normal transport or accident conditions, this pressure increase occurs within the canister. There is no pressure increase in the cask cavity except that due to the increase in cavity temperature. As the cask cavity is backfilled to 0 psig, a hypothetical canister failure would result in a containment vessel pressure lower than the canister pressure. Hypothetical canister failure would, therefore, result in significantly lower system pressure than the containment boundary design pressure.

4.2.3 Containment Criterion for Normal Conditions of Transport

The NAC-STC is designed and tested to meet the containment criteria of 10 CFR 71.51. The 10 CFR 71 limit for the release of radioactive material under normal conditions of transport is 10^{-6} A_2 per hour. The containment criteria are met for the metallic O-ring configuration by testing the NAC-STC to leaktight conditions, as defined by ANSI N14.5-1997, of 1×10^{-7} ref cm^3/sec. This corresponds to a helium leakage rate of 2×10^{-7} cm^3/sec and a test sensitivity of 1×10^{-7} cm^3/sec (helium). The containment criteria are met for the Viton O-ring configuration by testing the NAC-STC to either a leaktight condition (1×10^{-7} cm^3/sec) , which is equivalent to $\leq$ 2.0×10^{-7} cm^3/sec (helium) at a test sensitivity of $\leq 1.0\times10^{-7}$ cm^3/sec (helium), for the directly

loaded shipment of HBU PWR fuel assemblies, or to the cumulative directly loaded standard burnup assembly calculated leakage rate of 7.5×10^{-5} ref cm^3/sec (air), which is equivalent to 9.3×10^{-5} cm^3/sec (helium) at a test sensitivity of 4.7×10^{-5} cm^3/sec (helium) for directly loaded standard PWR fuel assembly (e.g., burnup of ≤45 GWd/MTU). The calculation of the allowable leak rate for the Viton O-ring configuration for standard PWR fuel assemblies is provided in Section 4.2.3.2.

4.2.3.1 Permissible Release Rate for the NAC-STC with Metallic O-rings

Metallic O-rings are required to be used in the containment boundary when the cask is directly loaded with PWR spent fuel for long-term storage with subsequent transport, and when the cask is loaded with a transportable storage canister or HLW overpack. Metallic O-rings may also be used in the containment boundary when the cask is directly loaded with standard PWR fuel assemblies (e.g., burnup of ≤ 45 GWd/MTU) for transport without interim storage. For the metallic O-ring configuration, the containment boundary is tested to leaktight conditions as defined by ANSI N14.5-1997 and, therefore, meets the requirements of 10 CFR 71.51 for containment of the radioactive contents.

Since the cask containment boundary is tested to demonstrate a leaktight condition, an allowable release rate, based on gases, fines, volatiles and particulates that are available for release from the directly loaded spent fuel or GTCC waste in the transportable storage canister, or HLW overpack is not calculated.

4.2.3.2 Permissible Release Rate for the NAC-STC with Viton O-rings

Viton O-rings may be used in the containment boundary when the cask is directly loaded with PWR spent fuel for transport without interim storage. For the Viton O-ring configurations, the containment boundary is tested to 7.5×10^{-5} ref cm^3/sec for standard PWR fuel assemblies with burnups of ≤ 45,000 MWd/MTU; and for high burnup (HBU) fuel assemblies with burnup > 45 GWd/MTU, a leaktight criterion is applied and no leakage analysis is required. As described in this section, this leak rate meets the requirements of 10 CFR 71.51 for containment of PWR spent fuel with burnups of ≤45 GWd/MTU.

The 10 CFR 71.51 limit for the release of radioactive material under normal conditions of transport is 10^{-6} A_2/hr. In this analysis, A_2 for a mixed gas is determined by using the method described in 10 CFR 71, Appendix A. The release fractions for the various radionuclides

transported in the NAC-STC are obtained from NUREG/CR-6487 and summarized in Table 4.2-1. The curie content for gases, volatiles, fines and particulates for the directly loaded 5-year cooled PWR reference fuel assembly is provided in Tables 4.2-6 through 4.2-8.

In addition to the radionuclides produced by the fuel material, fuel assemblies develop a coating of impurities deposited by cooling water during power generation. This coating is known as crud. Crud contains mostly non-radioactive elements but also contains a significant amount of ^{60}Co. NUREG/CR-6487 lists the maximum ^{60}Co concentrations on spent fuel assemblies to be 140 $\mu Ci/cm^2$ for PWR assemblies at initial discharge. The surface area of the reference 17×17 PWR assembly is calculated to be 3.54×10^5 cm^2, based on the assembly characteristics provided in Table 5.2-2.

The maximum permissible leak rate from the cask under normal conditions of transport is determined from the 10 CFR 71 limit of 10^{-6} A_2/hr.

$$R_N = L_N C_N \leq A_2 \text{ x } \cdot 1 \text{ x } 10^{-6}\,\text{hr}^{-1} \text{ or}$$
$$R_N = L_N C_N \leq A_2 \text{ x } \cdot 2.78 \text{ x } 10^{-10}\,\text{sec}^{-1}$$

where:

L_N	=	Volumetric gas leakage rate [cm^3/s]
C_N	=	Curies per unit volume (termed "activity density") of the radioactive material that passes through the leak path [Ci/cm^3]
R_N	=	Release rate for normal transport conditions [Ci/sec]

<u>Activity Density of Radioactive Material (C_N)</u>

The total inventories of fission product gases, volatiles and fines are shown in Table 4.2-6 through Table 4.2-8. These inventories are calculated by using the source terms produced by the SAS2H sequence, the release fractions and the postulated crud (^{60}Co). The ^{60}Co content is decayed 5 years from discharge.

$$C_n = C_{Crud} + C_{Volaties} + C_{FissionGas} + C_{Fines}$$

$$C_{Crud} = \frac{f_C M_T}{V} = \frac{f_C S_C N_A (N_R S_{AR})}{V}$$

where:

C_{crud}	=	Activity density inside containment vessel resulting from crud spallation [Ci/cm^3]
M_T	=	Total crud activity inventory [Ci]
f_c	=	Crud spallation factor
V	=	Free volume inside containment vessel [cm3]
S_C	=	Crud surface activity [Ci/cm^2]
N_R	=	Number of fuel rods per assembly
N_A	=	Number of assemblies
S_{AR}	=	Surface area per rod [cm^2]

and,

$$C_{fines} = \frac{f_F W_R A_R N_R N_A f_B}{V}$$

where:

C_{fine}	=	Activity concentration inside containment vessel resulting from fines released from cladding breaches [Ci/cm^3]
f_F	=	Fraction of fuel rod mass released as fines resulting from cladding breach
f_B	=	Fraction of fuel rods that develop cladding breach
W_R	=	Mass of the fuel in fuel rod [g]
N_R	=	Number of fuel rods per assembly
N_A	=	Number of assemblies
A_R	=	Specific activity of fines emitted from cladding breach in fuel rod [Ci/g]
V	=	Containment vessel void volume [cm^3]

and,

$$C_{vg} = C_{vol} + C_{gas} = \frac{N_R N_A f_B W_R (A_V f_V + A_G f_G)}{V}$$

where:

C_{vg}	=	Releasable activity concentration inside the containment vessel resulting from gases and volatiles released from cladding breaches [Ci/cm^3]
C_{vol}	=	Releasable activity concentration inside the containment vessel resulting from volatiles released from cladding breaches [Ci/cm^3]
C_{gas}	=	Releasable activity concentration inside the containment vessel resulting from gases released from cladding breaches [Ci/cm^3]
W_R	=	Mass of the fuel in a fuel rod [g]

N_R	=	Number fuel rods per assembly
N_A	=	Number of assemblies
f_B	=	Fraction of rods that develop cladding breaches
A_V	=	Specific activity of volatiles in fuel rod [Ci/g]
f_V	=	Fraction of volatiles in fuel rod released if rod develops cladding breach
A_G	=	Specific activity of gas in fuel rod [Ci/g]
f_G	=	Fraction of gas that would escape from fuel rod that develops cladding breach
V	=	Void volume inside containment vessel [cm^3]

Activity Values for Radionuclides

A_2 values used in this analysis (based on 10 CFR 71 Appendix A) are listed in Tables 4.2-6 through 4.2-8 for all radionuclides produced by the SAS2H analysis (plus ^{60}Co). The mixture A_2 value is shown in Table 4.2-2. For those isotopes for which no specific A_2 values are given in 10 CFR 71 Appendix A, the generic values listed in Table A.2 of Appendix A are applied. A_2 values for mixed isotopes are calculated from the following:

$$A_2 = \frac{1}{\sum \frac{F_i}{A_2^i}}$$

where:

$$F_i = \frac{S_i}{S_n}$$

and

F_i = Fraction of isotope i with respect to the entire mixture
S_i = Activity of isotope i [Ci]
S_n = Total group activity [Ci]

Mixture A_2 values are determined for gas, volatile, fine and crud mixtures and are then combined for a total cask mixture A_2 value. Table 4.2-2 provides the source term and A_2 values per group for directly loaded PWR fuel release rate calculations.

Maximum Allowable Leak Rate for Viton O-rings

On the basis of the methodology described, the maximum allowable volumetric leak rate for PWR fuel with burnups of ≤ 45 GWd/MTU directly loaded for immediate transport in normal conditions of transport is calculated to be 7.2×10^{-5} cm^3/sec (Table 4.2-3).

Correlation of Allowable Leak Rates to Air Standard

The volumetric gas leak rate, L, is independent of cask pressure and temperature. The maximum allowable release must be correlated with air standard leak rates, which depend on gas temperatures, pressures, and leakage path length and diameter. This correlation requires calculation of the capillary opening diameter through which the flow occurs. Depending on pressure and condition of the flow, a combination of continuum and molecular flow occurs.

Continuum flow and molecular flow equations are obtained from NUREG/CR-6487, Section 2, which are adjusted to upstream flow rate in accordance with NUREG/CR-6487 and ANSI N14.5-1997. The continuum volumetric flow rate of the gas (cm^3/sec), L_c, is given by:

$$L_c = \frac{2.48 \times 10^6 D^4}{a\mu}(P_u - P_d) * \frac{P_a}{P_u} = F_c * (P_u - P_d) * \frac{P_a}{P_u}$$

where:

F_c	=	Coefficient for continuum flow [cm^3/atm-s]
D	=	Capillary diameter [cm]
a	=	Capillary length [cm]
μ	=	Fluid viscosity [cP]
P_u	=	Upstream pressure [atm] – pressure inside containment
P_d	=	Downstream pressure [atm] – pressure outside containment
P_a	=	Average pressure $(P_u+P_d)/2$ [atm]

and, the molecular volumetric flow rate of the gas (cm^3/sec), L_m, is given by:

$$L_m = \frac{3.81 \times 10^3 D^3 \sqrt{\frac{T}{M}}}{aP_a}(P_u - P_d) * \frac{P_a}{P_u} = F_m * (P_u - P_d) * \frac{P_a}{P_u}$$

where:

L_m	=	Volumetric flow rate of gas at P_a [cm^3/sec]
F_m	=	Coefficient for molecular flow [cm^3/atm-s]
D	=	Capillary diameter [cm]
T	=	Gas temperature [K]
M	=	Gas molecular weight [g/mole]
P_a	=	Average pressure $(P_u+P_d)/2$ [atm]
P_u	=	Upstream pressure [atm]
P_d	=	Downstream pressure [atm].
a	=	Capillary diameter [cm]

For this analysis, the gas temperature used for molecular flow analysis is identical to the upstream temperature. Pressure and temperature at normal operating conditions are summarized in Table 4.2-4. Based on the pressure, temperature and allowable leakage rate (L_N), the capillary diameter of the leak is determined. The calculated capillary diameter is then used to determine the air standard leak rate and helium test leak rate. Air standard condition leak rates are determined for air leaking from 1 atmosphere to 0.01 atmosphere at a temperature of 298K. The test gas is helium leaking from 1 atmosphere (0 psig) to a vacuum. Table 4.2-3 provides the reference and test leak rates. The minimum sensitivity for these tests is one-half the air reference leak rate. Key containment analysis parameters are summarized in Table 4.2-5.

This analysis is conservative, since a higher upstream pressure, which could result from a higher average gas temperature based on decay heat, results in a higher allowable leak rate assuming that the leak path length and the leak path diameter (calculated based on the reference air condition) are held constant. Since the test condition pressure cannot be less than 1 atmosphere and the average gas temperature does not have a first order effect on calculated leak rate, the helium test condition is conservative with respect to the allowable reference leak rate.

4.2.3.3 Permissible Release Rate for Canistered Fuel and GTCC Waste, and HLW Overpacks

The transportable storage canister welded closure is leak tested at final assembly to leaktight conditions, 1×10^{-7} ref cm^3/ sec, as defined by ANSI N14.5-1997. To meet this requirement, the allowable leak rate is 2×10^{-7} cm^3/sec (helium). The leak test sensitivity applied in testing the canister at the time it is closed is 1×10^{-7} cm^3/sec (helium), or less. Consequently, the canister provides adequate containment for the spent fuel or GTCC waste. For the Yankee-MPC configuration, the allowable leak rate for the canister is specified as 8×10^{-8} cm^3/sec (helium), with a corresponding test sensitivity of 4×10^{-8} cm^3/sec (helium). This specified test condition is conservative with respect to the leaktight condition specified by ANSI N14.5-1997.

HLW overpacks contain up to five (5) canisters of glassified HLW, and the closure lid weld is not leakage tested prior to storage. The glassified HLW welded canister contents will not generate releasable radioactive material, and therefore, no confinement credit is required or applied to HLW overpack. The containment of the HLW overpack contents is provided by the Condition B leaktight containment boundary provided with metallic seals.

Correlation to Air Standard Conditions

The air standard leak rate is 1×10^{-7} ref cm^3/ sec, the leak tight condition as defined by Section 2.1 of ANSI N14.5-1997. Leak testing of the NAC-STC cask and the transportable storage canister is performed using helium gas. The NAC-STC cask leak test is performed using an allowable leak rate of 2×10^{-7} cm^3/sec (helium) with a detection sensitivity of 1×10^{-7} cm^3/sec (helium).

Table 4.2-1 Release Fractions: Normal and Accident Conditions

Radionuclide Origin	Fraction: Normal Conditions	Fraction: Accident Conditions
Volatiles releasable	2.00E-04	2.00E-04
Fission gas releasable	0.3	0.3
Rod mass released	3.00E-05	3.00E-05
Crud spallation factor	0.15	1.0
Fraction of fuel that fails	0.03	1.0

Table 4.2-2 Allowable Release Rate Source and A_2 Values for Directly Loaded PWR Fuel: Normal Conditions

Reference 17×17 PWR Fuel[1]	Crud	Gas	Volatiles	Fines	Total
Total Activity per Assembly (TBq)	N/C[2]	1.79E+02	6.51E+03	9.99E+03	1.67E+04
Releasable Activity per Cask (TBq)	3.71E+00	4.19E+01	1.02E+00	2.34E-01	4.69E+01
Cask Volumetric Activity (TBq /cm^3)	4.98E-07	5.63E-06	1.37E-07	3.14E-08	6.30E-06
A_2 Value (TBq)	0.40	10.49	0.42	0.018	11.33
Fraction of Activity	0.079	0.894	0.022	0.005	1.000
Fraction of Activity / A_2 (1/ TBq)	0.1977	0.0852	0.0512	0.2810	0.6152
Mixture A_2 Value (TBq)					1.63

1. Based on 3% rod failure.
2. Not explicitly calculated.

Table 4.2-3 Leak Rate and Leak Test Sensitivity: Normal Conditions

Contents	O-rings	Vol. Activity (TBq/cm^3)	Leak Rate (cm^3/sec) Allowable (L)	Air Reference (L_R)	Test Sensitivity
Directly Loaded Reference 17×17 PWR Fuel	Viton	6.3E-06	7.2E-05	7.5E-05[1]	3.8E-05[1]

1. The corresponding helium test leak rates and leak test sensitivities for the directly loaded PWR fuel configuration are 9.3×10^{-5} cm^3/sec and 4.7×10^{-5} cm^3/sec, respectively, at standard conditions.

Table 4.2-4 Cask Free Volumes and Pressures: Normal and Accident Conditions

Contents	Directly Loaded Reference PWR Fuel	
Cask Operating Condition	Normal	Accident[1]
Free Gas Volume (liters)[2]	7440	7540
Pressure (atm)[2]	1.80	5.72
Average Gas Temperature (K)	505.0	675.0

1. The accident condition for this analysis is 100% rod failure in combination with a fire accident that raises the cask temperature. This hypothetical dual-failure accident conservatively maximizes both available releasable material and cask pressure.
2. Bounding values were chosen for free volume (minimum) and pressure (maximum). This conservatively minimizes free volume and capillary diameter.

Table 4.2-5 Containment Parameters for Nonmetallic O-rings in Normal Conditions of Transport

Contents	Containment Free Volume (cm^3)	Capillary Length (cm)	Capillary Diameter (cm)	Upstream Pressure (atm)	Gas Temperature (K)
Reference 17×17 PWR Fuel	7.44E+6	0.597	8.6E-4	1.80	505

Note: Based on 3% of the fuel rods failing in normal transport conditions.

Table 4.2-6 17×17 Reference Fuel SAS2H Output and Group A_2 Values – Gases

Isotope	Activity/Assembly (TBq)	Fraction of Source	Isotope A_2 Value (TBq)	Isotope Fraction/A_2 (1/TBq)	Group A_2
H3	1.12E+01	6.24E-02	4E+01	1.559E-03	
I129	1.01E-03	5.64E-06	1E+60	5.638E-66	
KR85	1.68E+02	9.38E-01	1E+01	9.376E-02	
Total	1.79E+02			9.53E-02	10.491

Table 4.2-7 17×17 Reference Fuel SAS2H Output and Group A_2 Values – Volatiles

Isotope	Activity/Assembly (TBq)	Fraction of Source	Isotope A_2 Value (TBq)	Isotope Fraction/A_2 (1/TBq)	Group A_2
CS134	1.23E+03	1.89E-01	7E-01	2.695E-01	
CS135	1.24E-02	1.90E-06	1E+00	1.903E-06	
CS137	2.95E+03	4.52E-01	6E-01	7.538E-01	
RU106	5.59E+02	8.58E-02	2E-01	4.290E-01	
SR90	1.78E+03	2.73E-01	3E-01	9.110E-01	
Total	6.51E+03			2.36E+00	0.423

Table 4.2-8 17×17 Reference Fuel SAS2H Output and Group A_2 Values – Fines

Isotope	Activity/Assembly (TBq)	Fraction of Source	Isotope A_2 Value (TBq)	Isotope Fraction/A_2 (1/TBq)	Group A_2
AG108	4.40E-05	4.41E-09	2E-02	2.20E-07	
AG108M	5.03E-04	5.04E-08	7E-01	7.20E-08	
AG109M	4.51E-06	4.52E-10	2E-02	2.26E-08	
AG110	1.78E-02	1.78E-06	2E-02	8.89E-05	
AG110M	1.31E+00	1.31E-04	4E-01	3.27E-04	
AM241	2.58E+01	2.58E-03	1E-03	2.58E+00	
AM242	2.47E-01	2.47E-05	2E-02	1.24E-03	
AM242M	2.48E-01	2.48E-05	1E-03	2.48E-02	
AM243	1.55E+00	1.55E-04	1E-03	1.55E-01	
BA137M	2.78E+03	2.78E-01	2E-02	1.39E+01	
BI212	1.85E-03	1.86E-07	6E-01	3.09E-07	
BK249	2.18E-04	2.18E-08	3E-01	7.26E-08	
C 14	3.63E-06	3.63E-10	3E+00	1.21E-10	
CD109	4.51E-06	4.52E-10	2E+00	2.26E-10	
CD113M	1.02E+00	1.02E-04	5E-01	2.04E-04	
CE144	2.61E+02	2.61E-02	2E-01	1.31E-01	
CF249	3.12E-05	3.12E-09	8E-04	3.90E-06	
CF250	1.25E-04	1.25E-08	2E-03	6.26E-06	
CF251	1.55E-06	1.55E-10	7E-04	2.21E-07	
CF252	1.92E-04	1.92E-08	3E-03	6.41E-06	
CM242	1.01E+00	1.01E-04	1E-02	1.01E-02	
CM243	1.18E+00	1.19E-04	1E-03	1.19E-01	
CM244	2.87E+02	2.88E-02	2E-03	1.44E+01	
CM245	3.52E-02	3.52E-06	9E-04	3.91E-03	
CM246	2.19E-02	2.20E-06	9E-04	2.44E-03	
CM248	1.46E-06	1.46E-10	3E-04	4.87E-07	
EU152	8.73E-02	8.74E-06	1E+00	8.74E-06	
EU154	3.02E+02	3.02E-02	6E-01	5.03E-02	
EU155	1.44E+02	1.44E-02	3E+00	4.79E-03	
GD153	4.11E-03	4.11E-07	9E+00	4.57E-08	
HO166M	6.22E-05	6.22E-09	9E-05	6.91E-05	
NB 93M	9.07E-03	9.08E-07	3E+01	3.03E-08	
NB 94	2.98E-06	2.99E-10	7E-01	4.27E-10	
NB 95	1.59E-04	1.59E-08	1E+00	1.59E-08	
NP235	7.03E-05	7.04E-09	4E+01	1.76E-10	
NP236	2.08E-06	2.08E-10	2E-02	1.04E-08	
NP237	1.04E-02	1.04E-06	2E-03	5.22E-04	
NP238	1.11E-03	1.12E-07	2E-02	5.58E-06	
NP239	1.55E+00	1.55E-04	4E-01	3.87E-04	

Table 4.2-8 17×17 Reference Fuel SAS2H Output and Group A_2 Values – Fines (Continued)

Isotope	Activity/Assembly (TBq)	Fraction of Source	Isotope A_2 Value (TBq)	Isotope Fraction/A_2 (1/TBq)	Group A_2
PA233	1.04E-02	1.04E-06	7E-01	1.49E-06	
PA234	6.88E-06	6.89E-10	2E-02	3.45E-08	
PA234M	5.29E-03	5.30E-07	2E-02	2.65E-05	
PB212	1.85E-03	1.86E-07	2E-01	9.28E-07	
PD107	4.70E-03	4.70E-07	1E+60	4.70E-67	
PM145	4.88E-03	4.89E-07	1E+01	4.89E-08	
PM146	8.77E-02	8.78E-06	2E-02	4.39E-04	
PM147	8.03E+02	8.04E-02	2E+00	4.02E-02	
PO212	1.19E-03	1.19E-07	9E-05	1.32E-03	
PO216	1.85E-03	1.86E-07	9E-05	2.06E-03	
PR144	2.61E+02	2.61E-02	2E-02	1.31E+00	
PR144M	3.65E+00	3.65E-04	2E-02	1.83E-02	
PU236	2.23E-02	2.23E-06	3E-03	7.43E-04	
PU238	1.42E+02	1.42E-02	1E-03	1.42E+01	
PU239	5.70E+00	5.70E-04	1E-03	5.70E-01	
PU240	1.19E+01	1.19E-03	1E-03	1.19E+00	
PU241	2.52E+03	2.53E-01	6E-02	4.21E+00	
PU242	9.32E-02	9.34E-06	1E-03	9.34E-03	
RA224	1.85E-03	1.86E-07	2E-02	9.28E-06	
RH102	2.08E-02	2.08E-06	5E-01	4.16E-06	
RH106	5.59E+02	5.59E-02	2E-02	2.80E+00	
RN220	1.85E-03	1.86E-07	9E-05	2.06E-03	
SB125	6.70E+01	6.71E-03	1E+00	6.71E-03	
SB126	2.74E-03	2.74E-07	4E-01	6.85E-07	
SB126M	1.96E-02	1.96E-06	2E-02	9.80E-05	
SE 79	1.99E-03	2.00E-07	2E+00	9.98E-08	
SM145	1.24E-03	1.24E-07	1E+01	1.24E-08	
SM151	8.62E+00	8.63E-04	1E+01	8.63E-05	
SN119M	2.43E-02	2.44E-06	3E+01	8.13E-08	
SN121	6.07E-02	6.08E-06	2E-02	3.04E-04	
SN121M	7.81E-02	7.82E-06	9E-01	8.68E-06	
SN123	6.81E-04	6.82E-08	6E-01	1.14E-07	
SN126	1.96E-02	1.96E-06	4E-01	4.90E-06	
TC 99	3.70E-01	3.70E-05	9E-01	4.12E-05	
TE123M	1.48E-05	1.48E-09	1E+00	1.48E-09	
TE125M	1.64E+01	1.64E-03	9E-01	1.82E-03	
TE127	2.98E-03	2.99E-07	7E-01	4.27E-07	
TE127M	3.05E-03	3.05E-07	5E-01	6.10E-07	
TH228	1.85E-03	1.85E-07	1E-03	1.85E-04	

Table 4.2-8 17×17 Reference Fuel SAS2H Output and Group A_2 Values – Fines (Continued)

Isotope	Activity/Assembly (TBq)	Fraction of Source	Isotope A_2 Value (TBq)	Isotope Fraction/A_2 (1/TBq)	Group A_2
TH231	8.58E-05	8.59E-09	2E-02	4.30E-07	
TH234	5.29E-03	5.30E-07	3E-01	1.77E-06	
TL208	6.66E-04	6.67E-08	2E-02	3.33E-06	
TM171	1.90E-05	1.90E-09	4E+01	4.75E-11	
U232	2.70E-03	2.70E-07	9E-05	3.00E-03	
U233	3.63E-07	3.64E-11	6E-03	6.06E-09	
U234	2.56E-03	2.56E-07	6E-03	4.27E-05	
U235	8.58E-05	8.59E-09	1E+60	8.59E-69	
U236	5.37E-03	5.37E-07	6E-03	8.95E-05	
U237	6.03E-02	6.04E-06	2E-02	3.02E-04	
U238	5.29E-03	5.30E-07	1E+60	5.30E-67	
Y 90	1.78E+03	1.78E-01	3E-01	5.94E-01	
Y 91	6.62E-06	6.63E-10	6E-01	1.11E-09	
ZR 93	3.33E-02	3.33E-06	1E+60	3.33E-66	
ZR 95	6.99E-05	7.00E-09	8E-01	8.75E-09	
Total	9.99E+03			5.64E+01	0.018

4.3 Containment Requirements for Hypothetical Accident Conditions

The NAC-STC has been designed to safely transport 26 design basis directly loaded PWR fuel assemblies, up to 20 HBU PWR fuel assemblies, or canistered spent fuel or GTCC waste in either the Yankee-MPC or Connecticut Yankee-MPC configurations, and HLW overpacks. The structural integrity of the cask containment during hypothetical accident conditions is demonstrated in Section 2.7. Therefore, the cask containment is maintained under hypothetical accident conditions. As described in Section 2.7.11, the transportable storage canister does not fail in any of the evaluated transport accident conditions defined in 10 CFR 71.73. Consequently, its leaktight condition is maintained in the hypothetical accident conditions. As described in Section 4.1, metallic O-rings are required to be used for the direct loading of fuel for long-term storage and subsequent transport and for the transport of transportable storage canisters and HLW overpacks. Either metallic or Viton O-rings may be used for directly loaded standard burnup PWR fuel assemblies ($\leq$ 45 GWd/MTU) for transport without interim storage. Viton O-rings are required to be used for directly loaded HBU PWR fuel assemblies (> 45 GWd/MTU) for transport without interim storage.

With the exception of high burnup fuel (defined to be fuel with an assembly average burnup of greater than 45 GWd/MTU) for direct loading for transport without interim storage using Viton O-rings, the containment boundary requirement under hypothetical accident conditions is met by ensuring that a leak rate limit of 5.9×10^{-3} ref·cm^3/sec is not exceeded. High burnup PWR fuel employs a leaktight limit of 1×10^{-7} ref·cm^3/sec. Calculation related to the non-leaktight limit is provided in Section 4.3.3.

For directly loaded fuel, assuming a simultaneous occurrence of a fire accident and a 100% rod failure, and on the basis of a bulk average gas temperature of 675K resulting from air in the cavity, the pressure within the cask cavity is calculated to be 5.72 atm. The hypothetical presence of air in the cask provides an upper bound on the gas temperature. This pressure represents the maximum possible cask internal pressure. While fuel could fail within the transportable storage canister, no release from the canister occurs.

4.3.1 Fission Gas Products

The calculated amounts of fission gases contained by the design basis directly loaded and canistered PWR fuel assemblies for both normal and hypothetical accident conditions are reported in Section 4.2.2. The accident conditions assume a 100% fuel rod failure with 30% of the available tritium and 30% of the available krypton-85 being released to the cask cavity or to

the canister. These gases contribute to an increase in the cask cavity pressure due to the postulated failure of the directly loaded, intact/undamaged fuel and to an increase in the canister pressure due to the postulated failure of the canistered fuel.

Other released radionuclides, including crud, volatiles, fines and particulates, are not assumed to contribute to an increase in internal pressure of either transport configuration. The GTCC waste does not contain any gaseous products and does not have a failure mode in the hypothetical accident conditions. Consequently, there is no increase in pressure due to the GTCC contents.

The release of material from the postulated failure of the intact/undamaged fuel assemblies bounds the possible release of material from the Reconfigured Fuel Assemblies and Damaged Fuel Cans since the allowable contents of these components is less than or equal to that of an intact fuel assembly.

4.3.2 Containment of Radioactive Material

For directly loaded fuel intended for transport without interim storage using metallic O-rings, the containment boundary is tested to a leaktight condition as defined in ANSI N14.5-1997. As shown in Section 2.7 for the NAC-STC cask and in Section 2.7.11 for the transportable storage canister, the containment boundary of the cask and canister do not fail during the hypothetical accident events. Consequently, leaktight containment is maintained by both the cask and the canister in the hypothetical accident events. For the Viton O-ring configuration for direct loading, the containment criteria is either leaktight (for high burnup fuel) or the allowable leak rate in the hypothetical accident condition is 5.9×10^{-3} cm^3/sec, as shown in Section 4.3.3. The radionuclide activities for the reference PWR fuel assembly are provided in Section 4.2.3.

4.3.3 Calculation of Allowable Leak Rate for Non Leaktight Directly Loaded Fuel with Viton O-rings

The allowable leak rates under hypothetical accident conditions for the non leaktight configuration are calculated by using the method described in Section 4.2.1.1 for normal conditions of transport. The total inventories of fission product gases, volatiles, fines and crud are calculated by using the source terms generated by SAS2H, using the release fractions. Using the A_2 values from 10 CFR 71, Appendix A (Table 4.3-1), the mixture A_2 values are determined for gas, volatile, fine and crud mixtures. Finally, the maximum allowable release rates are calculated by using the hypothetical accident conditions allowable release limit:

$$R_A = L_A C_A \leq A_2 \cdot \text{per week}$$

or

$$R_A = L_A C_A \leq 1.65\text{x}10^{-6} A_2 \text{ per sec}$$

where:

L_A	=	Volumetric gas leakage rate [cm^3/sec]
C_A	=	Curies per unit volume (termed "activity density") of the radioactive material that passes through the leak path [Ci/cm^3]
R_A	=	Release rate for accident transport conditions

The assumptions applied to the calculations for the hypothetical accident conditions are that 100% of the fuel rods fail and that 100% of the assumed crud is released. The gas, volatile, fine and crud mixture A_2 is not affected by the change in the magnitude of releasable material. However, the combined A_2 changes based on the change in activity fraction in each group.

The calculated maximum permissible release rate for the reference directly loaded PWR fuel under hypothetical accident conditions using Viton O-rings is tabulated in Table 4.3-2.

Correlation of Allowable Leak Rates to Air Standard

The maximum allowable leak rate for the hypothetical accident conditions is correlated with the standard leak rate by using the methodology described in Section 4.2.1.2. The results for the reference PWR fuel loaded for transport without interim storage, using Viton O-rings, are shown in Table 4.3-2.

4.3.4 Containment Criterion for Accident Conditions

The containment criteria of 10 CFR 71 limits the release rate in accident conditions to A_2 per week. The NAC-STC cask using metallic O-rings is designed and tested to leaktight conditions as defined in Section 2.1 of ANSI N14.5-1997. The allowable leak rate calculated for the Viton O-ring configuration in the hypothetical accident conditions is much greater than that allowed under the normal conditions of transport. Consequently, the cask meets the regulatory containment criterion for the hypothetical accident conditions in either the metallic O-ring or nonmetallic O-ring configuration.

Table 4.3-1 Allowable Release Rate Source and A_2 Values for Directly Loaded PWR Fuel: Accident Conditions Using Nonmetallic O-rings

17×17 Reference	Crud	Gas	Volatiles	Fines	Total
Total Activity per Assembly (TBq)	N/C[1]	1.79E+02	6.51E+03	9.99E+03	1.67E+04
Releasable Activity per Cask (TBq)	2.47E+01	1.40E+03	3.39E+01	7.79E+00	1.46E+03
Cask Volumetric Activity (TBq/cm^3)	3.28E-06	1.85E-04	4.49E-06	1.03E-06	1.94E-04
A_2 Value (TBq)	0.40	10.49	0.42	0.02	11.33
Fraction of Activity	0.017	0.955	0.023	0.005	1.000
Fraction of Activity / A_2 (1/TBq)	0.0422	0.0910	0.0547	0.3000	0.4879
		Mixture A_2 Value (TBq)			2.05

1. Not explicitly calculated.

Table 4.3-2 Standard Leak Rate for the Accident Condition

Contents	O-rings	Vol. Activity (TBq/cm^3)	Leak Rate (cm^3/sec) Allowable (L)	Leak Rate (cm^3/sec) Air Reference (L_R)
Directly Loaded Reference 17×17 PWR Fuel	Viton	1.9E-04	1.7E-02	5.9E-03

Table 4.3-3 Containment Parameters for Non-Metallic O-rings in the Accident Condition

Contents	Containment Free Volume (cm^3)	Capillary Length (cm)	Capillary Diameter (cm)	Upstream Pressure (atm)	Gas Temperature (K)
Reference 17×17 PWR Fuel	7.54E+6	0.597	2.6E-3	5.72	675

Note: 100 % of the fuel rods are postulated to fail in the accident condition.

How to Specify O-Rings

Denotes Metallic O-Ring | Tubing OD (Thirty-Seconds) | Wall Thickness (Thousandths)

U 2 3 12 — 03 625 SE A

Metallic O-Ring OD (Inches) (Thousandths)

Materials

1—Alloy 718	7—Stainless Steel 304
2—Stainless Steel 321	8—Stainless Steel 316
3—Aluminum	9—Stainless Steel 347
4—Copper	X—As Specified
5—Alloy 600	
6—Alloy X-750	

Type

- SE—Self-energized on ID
- PF—Pressure filled
- NP—Not self-energized, not pressure filled
- SO—Self-energized on OD
- SX—Self-energized as spec.

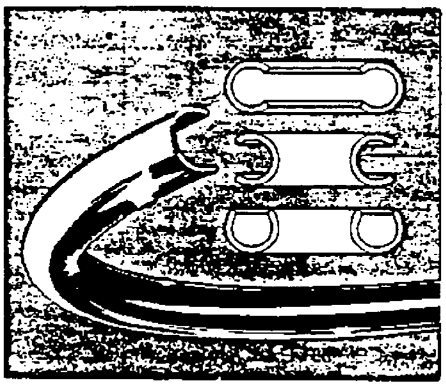
Example:

U2312-03625SEA

The above example, U2312-03625SEA, indicates a type 321 stainless steel O-Ring, $\frac{3}{32}$" (2.38 mm) tube size, .012 (0.30 mm) wall thickness, 3.625" (92.08 mm) OD, self-energized (ID) and .001-.002" (0.03/0.05 mm) silver coating.

Coatings

A—Silver	.001/.002 (0.03/0.05)	N—None	
B—Silver	.002/.003 (0.05/0.08)	P—Lead	.001/.002 (0.03/0.05)
D—Teflon*	.001/.003 (0.03/0.08)	R—Indium	.001/.002 (0.03/0.05)
E—Teflon	.003/.004 (0.08/0.10)	T—Nickel	.001/.002 (0.03/0.05)
L—Copper	.001/.002 (0.03/0.05)	V—Gold	.0005/.001 (0.02/0.03)
X—As Specified			



Fluorocarbon Metallic C-Rings

Fluorocarbon Metallic C-Rings (designated MCR) are designed for static sealing on machinery or equipment and are available for internal pressure, external pressure, or axial pressure ID/OD applications. Because C-Rings are designed with an open side on the pressure side of the installation, the seal is self-energizing. Fluorocarbon C-Rings are offered in round or irregular shapes in a broad range of sizes from .126" (3.2 mm) OD x .032" (0.81 mm) free height to over 300" (7620 mm) OD x 2" (50.80 mm) free height. They are available in a wide variety of metal alloys and metallic or Teflon coatings. Sealing application temperature range is from cryogenic to 3,000° F. (1650° C.); pressure tolerances are from 10^{-10} torr to 100,000 psi (6,804 atm). Where customer requirements are large, the C-Ring provides the lowest unit price of any high performance seal on the market.

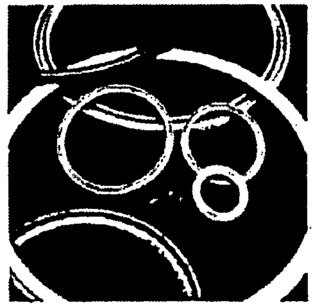
*Teflon is DuPont's Registered Trademark.

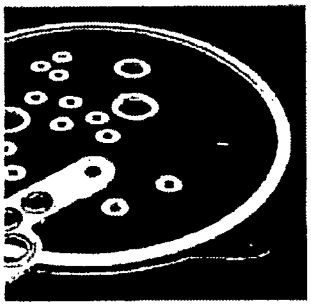
Helicoflex

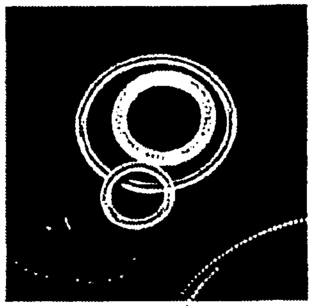
Components Division
P.O. Box 9889
Columbia, South Carolina 29290
Telephone (803) 783-1880
FAX (803) 783-4279

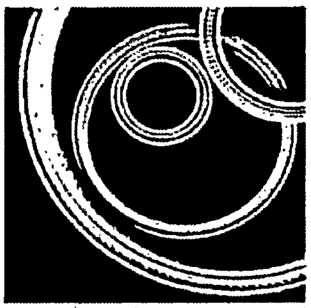
4.5.2 Blended Polytetrafluoroethylene (PTFE) O-rings

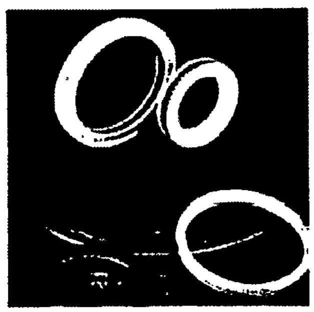
This section contains applicable technical data from a typical manufacturer of blended polytetrafluoroethylene (PTFE) O-rings. The PTFE O-rings used in the NAC-STC port covers are manufactured from virgin (unreprocessed) polytetrafluoroethylene base material filled with plastic. One product that satisfies the design requirements is the Fluoroloy K O-ring manufactured by the Furon Company, which has an operating temperature range of -450°F to +650°F. NAC has completed supplemental O-ring testing and has determined that the operating range of the PTFE O-rings can be extended to 735 °F. A description of tests performed and the results are contained in Certified Test Report D9-3362-1, Applied Technical Services, Inc., February 8, 1989. Another product that satisfies the design requirements is Parker Compound VM835-75. The compound's recommended operating temperature range is -40°F to 400 °F.

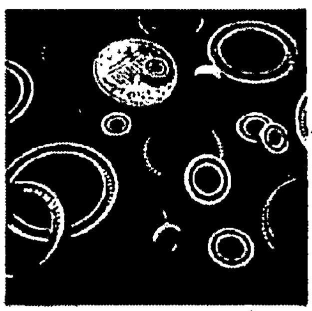












FLUOROCARBON
Dedicated To Excellence
MECHANICAL SEAL DIVISION
4412 Corporate Center Drive
P.O. Box 520
Los Alamitos, CA 90720
(213) 594-0941 (714) 995-1818
FAX (714) 761-1270
TWX 910-341-7672

COMPOUND DATA SHEET

Parker O-Ring Division, North America

MATERIAL REPORT

Report Number: 92880
Date: 3/13/2013

Title: Evaluation of Parker Compound VM835-75

Elastomer Type: Fluorocarbon (FKM)

Purpose: To obtain typical test data.

Specification: ASTM D2000 M2HK710 A1-10 B38 E078 Z1 (Shore A Hardness 75 +/-5), Z2 Elongation 125% min, Z3 (Specific Gravity), Z4 (TR-10)

Color: Black

Recommended Temperature Range: -40°F to 400°F

Recommended For: Mineral oil and grease, IRM 901 oil, IRM 902 oil, IRM 903 oil, non-flammable hydraulic fluids, silicone oils and greases, aliphatic hydrocarbons (propane, butane, natural gas), aromatic hydrocarbons (benzene, toluene), chlorinated hydrocarbons (trichloroethylene and carbon tetrachloride), gasoline, high vacuum, ozone, weather, and aging resistance.

Not Recommended For: Glycol based brake fluids, ammonia gas, amines, alkalis, superheated steam, and low molecular weight organic acids (formic and acetic acids).

Additional Approvals: N/A

Parker O-Ring Division
2360 Palumbo Drive
Lexington, Ky 40509
(859) 269-2351

REPORT DATA

Original Physical Properties	Test Method	Spec Limits	Test Results
(Z1) Hardness, Shore A, pts.	ASTM D2240	75 ±5	78
Tensile Strength, PSI (Mpa)	ASTM D412	1450 (10)	3059
(Z2) Ultimate Elongation, %	ASTM D412	125	215
(Z3) Specific Gravity	ASTM D297	±.03	1.8
Fluid Resistance (Basic Requirement)			
IRM 903, 70 hrs @ 302°F			
Volume Change, %	ASTM D471	+10	+2
(A1-10) Heat Age			
70 hrs. @ 482°F			
Hardness Change, pts.	ASTM D573	+10	+3
Tensile Strength Change, %		-25	-22
Ultimate Elongation Change, %		-25	+8
(B38) Compression Set (Plied)			
22 hrs. @ 392°F			
Percent of Original Deflection, Max	ASTM D395 Method B	50	13
(E078) Fluid Resistance			
Service Fluid 101, 70 hrs @ 392°F			
Hardness Change, pts.	ASTM D471	-15 to +5	-8
Tensile Strength Change, %		-40	-6
Ultimate Elongation Change, %		-20	-1
Volume Change, %		0 to +15	+11
(Z4) Low Temperature Resistance			
TR-10, temperature °F, C	ASTM D1329	report	-22 (-30)

"Purchaser use only. Reproduce only in full. Data pertains to items referenced only."
"The recording of false, fictitious, or fruaudulent statements or entries in this report may be punishable as a felony under federal law."

Parker O-Ring Division
2360 Palumbo Drive
Lexington, Ky 40509
(859) 269-2351

2 of 2

4.5.3 Expansion Foam

This section contains the manufacturer's technical bulletin for the material used to allow for the expansion of the neutron shield as the cask heats up.

Foamega® Brand Cellular Silicones

New Products...New Versatility...New Applications

Foamega® Brand Cellular Silicones, a unique family of products, possess a wide range of physical properties that meet the challenging and demanding applications presented by industry.

Design engineers are recognizing that, because of their light weight, low compressibility and resistance to compression set, they are excellent for gasketing, sealing, sound and vibration damping and thermal insulation.

Foamega offers many exceptional properties and characteristics:

- Withstands extremes in chemical and climatic environments.
- Maintains flexibility and compressibility over a temperature range of −65°F to +450°F.
- Is inherently inert and stable, non-degradable, and ozone and UV-resistant.
- Certain grades are UL-recognized for flame resistance and produce no toxic by-products upon forced combustion.
- Minimal water absorption, maximum 5 percent.
- Easy application to all surfaces, using silicone bonding adhesives; also available with either acrylic or silicone pressure-sensitive adhesives.
- Non-corrosive to metal surfaces.
- Available 1/16" to 1" thick (no laminations) in continuous lengths, 36" wide. Continuous length allows for the most efficient utilization of material during conversion or fabrication into pre-cut shapes or gasket stripping.

Typical Physical Properties

Product	Density Lbs/Ft.³ (Typical)	Compression Deflection 25% @ 25°C Psi	Compression Set 50% 22 Hr. 212°F % (Max.)	Tensile Strength Psi (Min.)	Elongation % (Min.)	Color
Low Density/Flame Retardant FOAM						
HT-803	14	3-7	10	25	100	White
General Purpose SPONGE						
HT-800 Medium	22	8-14	15	35	80	Red Grey
HT-820 Firm	26	12-20	15	40	80	Black
Low Compression Set SPONGE						
HT-810 Medium	22	8-14	5	35	80	Red Grey
HT-840 Soft	16	3-7	5	25	80	Black
Low Compression Set/Flame Retardant SPONGE						
HT-860 Medium	22	8-14	5	25	80	Red Grey
HT-870 Soft	16	3-7	5	25	80	Blue

Specifications

Product	AMS 3195	AMS 3196	MIL-R-6130 Type II, Gr. B&C	MIL-R-46089	Boeing BMS 1-23	Boeing BMS 1-60	Boeing BMS 1-68C	Douglas DMS 1857	Douglas DMS 1880	Lockheed LAC 33-364	Lockheed STM-33-471	General Dynamics F-0091	UL 1439	UL 94V1	UL 94V0
HT-803							X								
HT-800	X		X		X			X			X	X	X	X	X
HT-820		X						X			X	X			
HT-810	X		X	X								X			
HT-840												X			
HT-860	X		X			X			X	X			X	X	X
HT-870						X			X	X					

Tolerances

Size	Tolerance
1/16"	+1/32—1/64
1/8"	±1/32
3/16"	±1/32
1/4"	+3/64—1/32
3/8"	±1/16
1/2"	±1/8
5/8"	±1/8
3/4"	±5/32
1"	±13/64

. . . the High-Performance Silicone with 10 Ways to Improve Product Performance

1 Insulates in two ways—against electrical current and against heat and cold. Foamega has good dielectric properties that make it ideal for applications requiring electrical insulation, *plus* thermal insulation and one or more of Foamega's other attributes—in computers and other electronic equipment, in microwave ovens and other appliances, in lighting fixtures and in numerous other applications. Its low thermal conductivity gives Foamega an advantage when it is used as gasketing around metal window and door frames and between metal and glazing.

2 Handles heat and cold in a range from −65°F to +450°F. While other materials tend to become dry and brittle and to disintegrate when subjected to heat, Foamega retains its form and density. Foamega HT-803 and HT-850 are UL-listed for fire resistance, are self-extinguishing in a maximum of 10-15 seconds under forced combustion, and produce no toxic by-products. They are quite possibly the best materials for use as fire barriers in automobiles, as bulkhead seals and fire stops in aircraft, as a backup to upholstery and slipcover materials used in aircraft, automobiles, hotels and motels, hospitals and other institutions. In applications requiring heat-resistance, non-combustibility and freedom from dangerous fumes and gases, Foamega proves its value in saving lives and property.

3 Stands up to pressure. Low compression set at high and low temperatures is one of the reasons Foamega makes such excellent gasketing for engine exhaust manifolds, cooling systems and other high-temperature, high-pressure applications.

4 Cushions vibration and stops the wear and tear it can cause. Vibrational damping in aircraft, in automobile steering systems, in air conditioners and other appliances, in laboratory instruments and in hundreds of other applications is an important Foamega contribution. Wherever vibrations can cause a potential hazard, an interference with operation of other systems or merely an annoyance, look to Foamega for a most-efficient solution.

5 Quiets noise. Foamega foam and sponge are not only sound-absorptive by nature, they also provide one of the best means for stopping the distracting, irritating noises of hard surfaces contacting hard surfaces. Automobile dash panels, for example, often hide a potential for squeaks and rattles that need not exist—with Foamega "treatment."

6 Dams water damage. Foamega does not absorb water and provides an impermeable barrier to water incursion when used as washers, "O" rings and other forms of gasketing. Being non-conductive and non-corrosive also, it is perfect for water seals around electrical and electronic components. For watercraft, it is highly effective in making hatches and portholes watertight. For automobiles, it provides a moisture barrier around windshields and other glass. Sidelight: Foamega is also impervious to fungus, insect infestation and rodent damage.

7 Resists chemicals. Foamega is inherently inert and stable and is non-degradable in most chemicals. It resists ozone and UV radiation (Bisco silicone products are used for nuclear shielding in hundreds of nuclear power plants.) For these reasons it is able to perform in environments that rule out use of many other materials.

8 Keeps out weather in all its forms—heat, cold, wind, rain and snow. Foamega's low level of thermal conductivity, its non-water absorptive qualities, and its ability to form a tight seal recommend its use for gasketing building components, lighting fixtures and other items requiring a tight weather seal.

9 Stays flexible at temperatures as low as −100°F and as high as 600°F.

10 Installs easily by means of silicone adhesive bonding. Additionally, all Foamega products come with either acrylic (350°F) or silicone (450°F) pressure-sensitive adhesive pre-applied.

Foamega—a complex of benefits

When you use Foamega for any of the above reasons, you get a combination of advantages—not merely the one you seek. For example: use Foamega for thermal insulation between glass and metal and you also get excellent vibration damping that can prevent fracturing of glass. Use it as a weather barrier on electronics exposed to the elements and you gain insulation values as well.

Few, if any, other materials offer the multiple advantages of Foamega.

How can Foamega help you?

Only a limited number of Foamega's applications are mentioned here. Hundreds more are possible, depending on your needs and imagination. Agencies like NASA and DOT are presently evaluating Foamega for fireblocking and related applications.

Bisco Products stands ready to help you—with evaluating your present products in terms of Foamega capabilities, and with custom design and engineering services.

We welcome your inquiry.

bisco
One of the Brand Companies

bisco products, inc.
1420 renaissance drive
park ridge, illinois 60068
(312) 298-1200
telex 282482

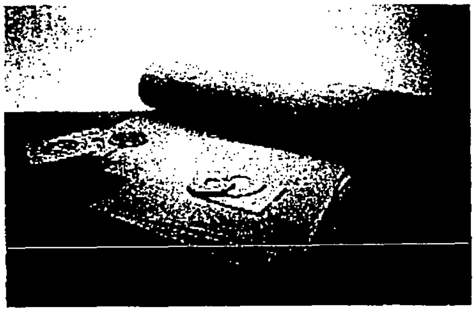
The properties listed herein are typical values and should not be used for writing specifications.

4.5.4 Fiberfrax Ceramic Fiber Paper

This section contains the manufacturer's technical data for the material used to preclude a lead melt during fabrication welding or a fire accident.

UNIFRAX Product Information Sheet

Fiberfrax® Ceramic Fiber Paper



Introduction

The Fiberfrax® ceramic fiber paper product line is a unique family of products which is manufactured by forming alumino-silicate fibers in a nonwoven matrix. The ceramic fibers are randomly orientated during manufacture, then held in place with a latex binder system. A specialized paper-making process is statistically controlled to form uniform, lightweight, flexible sheets.

Unifrax Corporation has been producing Fiberfrax papers for over 25 years and is the largest ceramic fiber producer worldwide with in-house paper-making capabilities.

By blending different fibers, binders, and additives while varying the manufacturing process, Unifrax Corporation now produces a variety of Fiberfrax paper products for a wide range of applications.

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

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

Product Line Advantages

Fiberfrax ceramic fiber papers offer our customers many unique problem-solving advantages which include:

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

General Uses of Fiberfrax Papers

Fiberfrax papers are used to solve a wide variety of heat-related problems, and are used as:

- Highly efficient refractory backup
- Dependable fire protection
- Thermal insulation
- Hot gas filtration media
- Molten metal splash and spark protection
- High-temperature gasket, separator, or parting agent

Typical Markets/Applications

Based on the uses listed in the preceding text, Fiberfrax papers solve a range of application problems in the industries listed below:

Aerospace
- Heat shields
- Nose cone ablative shields
- Igniter line protection
- Oxygen generators

Appliance
- Self-cleaning ovens
- Woodburning stoves
- Electrical heaters
- Mobile home appliance insulation

Ceramic and Glass
- Ware separator
- Metal clad brick gaskets
- Glass tank refractory backup

Petrochemical
- Transfer line protection
- Welding
- Brazing protection

Automotive
- Muffler insulation
- Heat shielding

Steel and Nonferrous
- Investment casting mold wrapping
- Ladle refractory backup
- Thermocouple tube protection
- Heat treating parting agent
- Foundry gasketing
- Ladle shroud wrap

Refer to the product Material Safety Data Sheet (MSDS) for recommended work practices and other product safety information

UNIFRAX

Product Range

Product Segmentation

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

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

Utility Grade

440 Paper

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

Rollboard

The lower density, binder chemistry, and bulk ceramic fiber grade used to manufacture Fiberfrax Rollboard paper result in a product with lower cost, higher flexibility, and reduced smoke and odor during burnout. Rollboard paper is best suited for wrapping intricate shapes or molds and as a standard grade single use product in disposable applications.

Standard Grade

110 Paper

110 paper is a clay-filled, sheeted ceramic fiber paper which is denser and more rigid than other standard grade products. The rigidity is maintained even after burnout of the organic bonding agents. The good dielectric strength, compression resistance, and die cutting characteristics of 110 paper are advantageous in many high-temperature gasketing applications.

550 Paper

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

970 Paper

970 paper is made from high-purity Fiberfrax washed fiber. During the manufacture of this product, a large portion of the unfiberized particles in the bulk fiber are removed prior to paper lay-up. The washing of the fiber gives great uniformity to the paper's structure while reducing weight and improving the thermal performance; in addition, this product is preferred in automatic die stamping operations where unfiberized particles in the paper can lead to excessive die wear.

Premium Grade

880 Paper

880 paper is made from a higher alumina content, shorter, smaller diameter fiber and laid up at higher densities. These product parameters lead to reduced shrinkage, higher strength, an increased operating temperature range and better chemical resistivity. This product is used in applications where the service life of standard ceramic fiber papers is reduced.

HSA Paper

HSA paper is made from high surface area (HSA) fibers that contain a low percentage of unfiberized material. Use of this fiber results in a paper with lighter weight and extremely low thermal conductivity, making it the choice of the aerospace industry. It is also used when uniform pore structure and a low content of unfiberized material are required in applications such as glass contact or gas filtration.

Inorganic Papers

Fiberfrax papers are available without the organic binder system. These products are completely free of organics and used when higher fired strength is required or in processes and applications where even small amounts of organic burnout is unacceptable. Two temperature grades and several thicknesses and widths are available.

- **972-H** is heat treated during the manufacturing process to remove organic binders. As manufactured, 972-H paper remains soft and flexible allowing it to conform to most shapes or contours.
- **882-H** has higher temperature stability and higher density than 972-H Paper. The fiber geometry and product density lead to the maximum burn strength of an unbindered paper.

Certifications/Approvals

Fiberfrax papers have been independently tested for conformance to a wide variety of industry standards. For example, several Fiberfrax papers are listed as "Recognized Components" with Underwriters Laboratories, Inc.; conform to U.S. Coast Guard requirements for incombustible materials; and are tested in accordance with ASTM methods. For details of existing approvals and test procedures, contact the Unifrax Application Engineering Group at 716/278-3899.

Additional Capabilities

Unifrax has several manufacturing capabilities which can enhance the performance of Fiberfrax papers in a wide variety of applications. Utilizing precision high-speed slitters, Unifrax can slit paper materials down to one-inch (1") widths for installation speed and convenience. Material can be laminated, foil faced or adhesive backed to tailor the material form to specific application requirements.

UNIFRAX

Form C-1423
Effective 11/00

Printed in USA
Page 2 of 4

Fiberfrax Ceramic Fiber Papers
Typical Product Properties

Paper Grade		440*	Roll Board	110	550	970	880	HSA	972-H	882-H	HSA** (OF)
Physical Properties											
Color		Gray	Off-White	Tan	White	White	White	White	White	White	White
Temperature Grade	°F	1600	2300	2300	2300	2300	2600	2300	2300	2600	2300
	°C	870	1260	1260	1260	1260	1427	1260	1260	1427	1260
Recommended Operating Temp.	°F	1300	2000	1900	2000	2000	2100	2000	2000	2100	2000
	°C	704	1100	1040	1100	1100	1150	1100	1100	1150	1100
Melting Point	°F	1800	3200	2800	3260	3260	3500	3100	3260	3500	3100
	°C	982	1760	1538	1793	1793	1927	1704	1793	1927	1704
Compression (PSI % Deformation)											
10%		5	1	1	4	1.3	—	3	—	—	—
25%		34	5	6	26	5.8	—	16	—	—	—
50%		489	32	35	167	22	—	44	—	—	—
Strength											
Tensile (PSI) (as manufactured)		86	58	147	102	94	136	55	—	—	—
Burst (PSI) (as manufactured)		45	22	19	248	25	—	37	—	—	—

Notes About Chart

*The 440 paper contains a fire retardant smoke suppressant.

**The HSA "OF" designation signifies materials made without the use of organic binders.

- "H" designation references the heat treating process used to remove organics.
- The recommended operating temperature of Fiberfrax insulation is determined by a maximum irreversible linear change criteria, not product melting point.

The test data shown are average results of tests conducted under standard procedures and are subject to variation. Results should not be used for specification purposes.

UNIFRAX

Form C-1423
Effective 11/00

Printed in USA
Page 3 of 4

Fiberfrax Ceramic Fiber Papers
Typical Product Parameters

Paper Grade	440*	Roll Board	110	550	970	880	HSA	972-H	882-H	HSA** (OF)
Physical Properties										
Density (pcf)	13	10	18	12	10	18	10	12	16	7
Fiber Index (% Wt)	n/a	40	n/a	50	70	45	100	70	45	100
LOI (incl. binder)	9.5	3.0	8.5	6.5	7.0	8.0	3.0	0.1	0.1	0.1
Chemistry (% Wt)										
Al_2O_3	32-35	47-52	45-50	47-52	47-52	58-60	47-52	47-52	58-60	47-52
SiO_2	42-46	48-53	40-44	48-53	48-53	40-42	47-52	48-53	40-42	47-52
Na_2O_3	<2	<0.5	<1.5	<0.5	<0.5	<0.3	<0.5	<0.5	<0.3	<0.5
Fe_2O_3	<2	<0.5	<1.1	<0.5	<0.5	<0.1	<0.05	<0.5	<0.1	<0.05
Thickness inches (mm)***										
A = 1/32 (0.8)					X			X		
F = 1/16 (1.6)	X		X	X	X	X	X	X	X	
J = 1/8 (3.2)	X	X	X	X	X	X		X	X	X
K = 1/4 (6.35)				X						
Roll Sizes (std)	25#, Mill	Mill	Sheet	25#, Mill	10#, 25#, Mill	25#, Mill	Sheets	25#	10#, 25#	500sf
Width (std. inches)	24, 48	18, 24	42x48	24, 48	12, 24, 48	12, 24, 48	42x48	12, 24	12, 24	51

Availability
Nonstandard widths available upon request.

Notes About Chart
*The 440 paper contains a fire retardant smoke suppressant.

**The HSA "OF" designation signifies materials made without the use of organic binders.

***Measured under 4 PSF.

"H" designation references the heat-treating process used to remove organics.

For additional information about product performance or to identify the recommended product for your application, please contact the Unifrax Application Engineering Group at 716-278-3899.

Data are average results of tests conducted under standard procedures and are subject to variation. Results should not be used for specification purposes.

UNIFRAX

Form C-1423
Effective 11/00

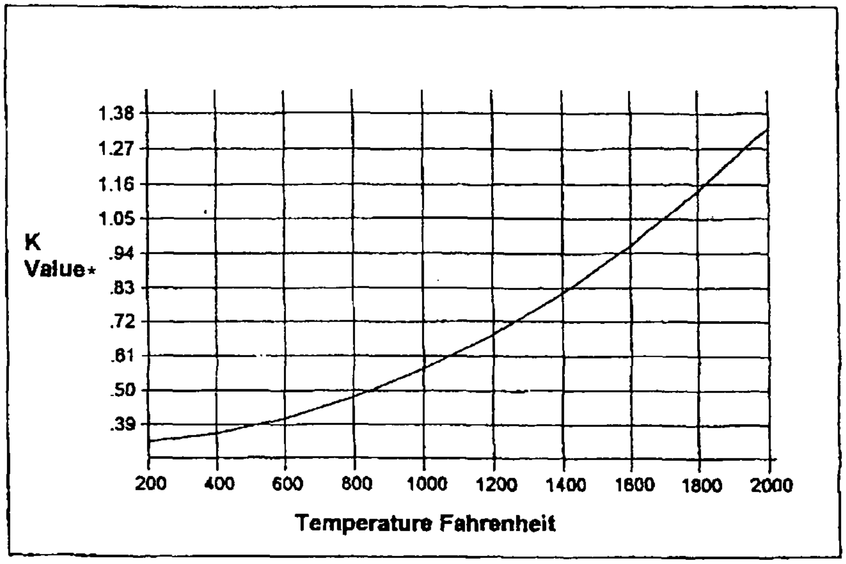
Printed in USA
Page 4 of 4

The test data shown are average results of tests conducted under standard procedures and are subject to variation. Results should not be used for specification purposes.

Unifrax Corporation
Corporate Headquarters
2351 Whirlpool Street
Niagara Falls, New York 14305-2413
Telephone: 716-278-3800
Telefax: 716-278-3900
Internet: www.unifrax.com
Email: info@unifrax.com

Unifrax Heatflow Calculation

970 Paper



Calculated Graph Points for 970 Paper

Deg F	200	400	600	800	1000	1200	1400	1600	1800
Deg C	93	204	316	427	538	649	760	871	982
K Val	00.334	00.362	00.410	00.480	00.570	00.682	00.815	00.969	01.144

*** Units for K Value are Btu/hr/in/ft²/ F**

4.5.5 Viton O-rings

This appendix provides a description of the leak testing performed using the Viton O-rings at temperatures exceeding the manufacturer's elevated temperature limit. In addition, it also contains the O-ring manufacturer's material report on the Viton material.

NAC, with the aid of an independent laboratory, performed leak testing in excess of 550°F to demonstrate the capability of Viton to perform at the elevated temperature and to determine the leak rate of the alternate port cover design at the elevated temperature. It was determined that the alternate port cover O-ring maintains its sealing capability at a temperature of 575°F after prolonged heating above 400°F. Testing was done in accordance with NAC Specifications. Two fixtures were put into a thermal test chamber. All the fittings attached to the test assemblies were checked and confirmed leaktight. The assemblies were heated in a manner that conservatively approximates the fire-transient analysis and one fixture was held at a temperature above 550°F for more than 4 hours, 37 minutes. The region inside the port cover was evacuated to below 2 psia, backfilled with helium at 0 psig, evacuated and backfilled again and then leak checked. The leak test procedure emulates the testing of the O-ring with one atmosphere of pressure acting on the O-ring during the test. The data pertinent to the test is:

	Test Assembly 16	Test Assembly 64	Fire-Transient
Time Above 400°F	~6:32 hours	~5:52 hours	4:37 hours
Time Above 550°F	~5:05 hours	~4:25 hours	0 hours
Maximum Seal Temperature	~575°F	~575°F	547°F

The test temperature of 550°F was selected because it approximates the maximum calculated O-ring temperature in the fire-transient analysis. The duration was selected because it is the calculated duration that the O-ring is above the manufacturer's maximum recommended O-ring temperature of 400°F. This result in a conservative test due to the slower heat-up rate of the oven compared to the heat-up rate of the port cover in the fire-transient analysis.

Each test assembly was leak checked after the temperature test, while at a temperature of approximately 575°F. The measured leak rate for each of the assemblies was less than 4.0×10^{-8} atm-cc/sec. In conclusion, the Viton O-rings can provide a leaktight seal, in accordance with ANSI N14.5-1997, at an elevated temperature.

Sep-17-99 03:35P P.01

Parker Seals

Software Version: 2.0 9/17/99

Customer Identification

Company:	NAC International
Contact:	George Carver
Project Name:	
Address:	
City:	**Zip Code:**
State:	
Telephone No.:	770-447-1797 fax
Date/Time:	9-17-1999 15:27

Ordering Specifications

Application:	O-ring Only
Compound Number:	V0835-75
Size:	

Compound Information

Search Parameter

Material Selection Method:	Compound Search
Contained Media:	
Desired Temperature Range	
High:	
Low:	

Selected Material Information

Durometer (Shore A):	75
Polymer:	Fluorocarbon GLT - LOWTEMP COMPOUND.
Temperature	
Normal High:	400 °F
Extended High:	400 °F
Normal Low:	-40 °F
Color:	Black
Static Application Only:	No
Military Spec.:	MIL-R-83485
AMS NAS Spec.:	None
SAE/ASTM Spec.:	None

Seal Size Information

Sizing Selection Method: Known: O-ring P/N. Search for: O-ring dimensions.

Sep-17-99 03:35P P.02

Compound Data Sheet
O-Ring Division United States

MATERIAL REPORT

REPORT NUMBER: KJ0835
DATE: 10/10/89

TITLE: Test of Parker Compound V0835-75 to MIL-R-83485, Type I.

PURPOSE: To determine if V0835-75 meets MIL-R-83485, Type I.

CONCLUSION: V0835-75 meets the above specification.

Parker O-Ring Division
2360 Palumbo Drive
Lexington, Kentucky 40509
(606) 269-2351

Sep-17-99 03:35P

P.03

REPORT DATA

Report Number: KJ0835

ORIGINAL	MIL-R-83485 TYPE 1, O-RINGS & COMPRESSION SEALS	V0835-75 ACTUAL VALUES
Specific Gravity	As determined	1.75
Hardness points	75 ± 5	78
Tensile Strength, psi. min.	1600	1708
Elongation, % min.	120	180
Temperature Retraction, 10% (TR-10), °F, max.	-20	-22
AFTER AIR AGING, 70 HRS.@ 75° + 5°F. Compression Set		
% of original deflection, max.	25	-- (14)
AFTER AGING, 70 HRS. @ 75°F IN TT-S-735, TYPE III		
Hardness Change, pts.	+5	77 (-1)
Tensile Strength decrease, %, max.	30	1662 (-3)
Elongation decrease, %, max.	20	165 (-8)
Volume change, %, max.	1 to 10	-- (+2)
AFTER AIR AGING, 70 HRS. @ 528° ± 5°F		
Hardness change, pts.	+5	78 (0)
Tensile Strength decrease, % max.	35	1136 (-33)
Elongation decrease, %, max.	10	235 (+31)
Weight loss, %, max.	12	-- (-7)
AFTER AIR AGING, 166 HRS @ 347° ± 5°F. COMPRESSION SET		
% of original deflection, max.	25	-- (15)
18 hrs. cooling		-- (24)
AFTER AIR AGING, 22 HRS @ 392° ± 5°F. COMPRESSION SET		
% of original deflection, max.	20	-- (11)

Sep-17-99 03:35P P.04

AFTER AGING, 70 HRS. @ 347°MIL-R-83485 ±5°F in AMS-3021	MIL-R-83485 TYPE 1, O-RINGS % COMPRESSION SEALS	V0835-75 ACTUAL VALUES
Hardness change, pts	+0, -15	73
Tensile Strength decrease, %, max.	35	1406 (-18)
Elongation decrease, %, max.	20	171 (-5)
Volume change, %	1 to 20	-- (+15)
Compression set, % of original deflection, max.	10	-- 7
18 hr. cooling		-- 9

4.5.6 Sample SAS2H Input File

This section provides a sample SAS2H input file employed in the containment analysis of the directly loaded 17×17 fuel at 60,000 MWD/MTU and 3.5 wt % ^{235}U.

Sample File

```
=SAS2H       PARM=(HALT09,SKIPSHIPDATA)
Class 1 - aa17b - STC Hybrid17 (Rev 0) - 3.5 w/o U235, 60000 MWD/MTU, 5 - 16 years cool time
27GROUPNDF4 LATTICECELL
UO2          1 0.943 900 92235 3.5 92238 96.5 END
ZIRCALLOY  2 1.0 620 END
H2O          3 DEN=0.725 1.0 580 END
ARBM-BORMOD 0.725 1 1 0 0 5000 100 3 550.0E-6 580 END
ZIRCALLOY  4 1.0 580 END
H2O          5 DEN=0.725 0.9772 580 END
ZIRCALLOY  5 0.0228 580 END
END COMP
SQUAREPITCH 1.2598 0.8192 1 3 0.9500 2 0.8360 0 END
NPIN=264 FUEL=365.760 NCYC=3 NLIB=3 PRIN=6 LIGH=5
INPL=1 NUMH=24 NUMI=1 MXTUBE=4 ORTU=0.6025 SRTU=0.5644 END
POWER=18.5535 BURN=499.7636 DOWN=60 END
POWER=18.5535 BURN=499.7636 DOWN=60 END
POWER=18.5535 BURN=499.7636 DOWN=1461 END
FE 0.6738 CR 0.1900 NI 0.1150 MN 0.0200 CO 0.0012
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71  E
1$$  1 1T
COOLING 5 - 16 YEARS AND FISSION PRODUCT GAMMA REBIN
3$$  21 0 1 28 A33 22 E
54$$  A8 1 E T
35$$  0 T
56$$  0 9 A13 -2 5 3 E
57**  4.0 E T
COOLING 5 - 16 YEARS AND FISSION PRODUCT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 6.0 7.0 8.0 9.0 10.0 12.0 14.0 16.0
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F.00000001
81$$  2 51 26 1 E
82$$  F6
83**  1.40e+7  1.20e+7  1.00e+7  8.00e+6   6.50e+6  5.00e+6
      4.00e+6  3.00e+6  2.50e+6  2.00e+6   1.66e+6  1.44e+6
      1.22e+6  1.00e+6  0.80e+6  0.60e+6   0.40e+6  0.30e+6
      0.20e+6  0.10e+6  0.05e+6  0.02e+6   0.01e+6
84**  1.46e+7  1.36e+7  1.25e+7  1.125e+7  1.00e+7
      8.25e+6  7.00e+6  6.07e+6  4.72e+6   3.68e+6
      2.87e+6  1.74e+6  0.64e+6  0.39e+6   0.11e+6
      6.74e+4  2.48e+4  9.12e+3  2.95e+3   9.61e+2
      3.54e+2  1.66e+2  4.81e+1  1.60e+1   4.00e+0
      1.50e+0  5.50e-1  7.09e-2  1.00e-5   T
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
FISSION PRODUCT GAMMA SPECTRA IN AEA GROUPS
56$$ F0  T
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71  E
1$$  1 1T
COOLING 5 - 16 YEARS AND ACTINIDE GAMMA REBIN
3$$  21 0 1 28 A33 22 E
```

Sample Input File (Continued)

```
54$$  A8 1 E T
35$$  0 T
56$$  0 9 A13 -2 5 3 E
57**  4.0 E T
COOLING 5 - 16 YEARS AND ACTINIDE GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 6.0 7.0 8.0 9.0 10.0 12.0 14.0 16.0
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F.00000001
81$$  2 51 26 1 E
82$$  F5
83**  1.40e+7  1.20e+7  1.00e+7  8.00e+6   6.50e+6  5.00e+6
      4.00e+6  3.00e+6  2.50e+6  2.00e+6   1.66e+6  1.44e+6
      1.22e+6  1.00e+6  0.80e+6  0.60e+6   0.40e+6  0.30e+6
      0.20e+6  0.10e+6  0.05e+6  0.02e+6   0.01e+6
84**  1.46e+7  1.36e+7  1.25e+7  1.125e+7  1.00e+7
      8.25e+6  7.00e+6  6.07e+6  4.72e+6   3.68e+6
      2.87e+6  1.74e+6  0.64e+6  0.39e+6   0.11e+6
      6.74e+4  2.48e+4  9.12e+3  2.95e+3   9.61e+2
      3.54e+2  1.66e+2  4.81e+1  1.60e+1   4.00e+0
      1.50e+0  5.50e-1  7.09e-2  1.00e-5   T
ACTINIDE GAMMA SPECTRA IN AEA GROUPS
ACTINIDE GAMMA SPECTRA IN AEA GROUPS
ACTINIDE GAMMA SPECTRA IN AEA GROUPS
ACTINIDE GAMMA SPECTRA IN AEA GROUPS
ACTINIDE GAMMA SPECTRA IN AEA GROUPS
ACTINIDE GAMMA SPECTRA IN AEA GROUPS
ACTINIDE GAMMA SPECTRA IN AEA GROUPS
ACTINIDE GAMMA SPECTRA IN AEA GROUPS
ACTINIDE GAMMA SPECTRA IN AEA GROUPS
56$$ F0  T
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71  E
1$$  1 1T
COOLING 5 - 16 YEARS AND LIGHT ELEMENT GAMMA REBIN
3$$  21 0 1 28 A33 22 E
54$$  A8 1 E T
35$$  0 T
56$$  0 9 A13 -2 5 3 E
57**  4.0 E T
COOLING 5 - 16 YEARS AND LIGHT ELEMENT GAMMA REBIN
SINGLE REACTOR ASSEMBLY
60** 5.0 6.0 7.0 8.0 9.0 10.0 12.0 14.0 16.0
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F.00000001
81$$  2 51 26 1 E
82$$  F4
83**  1.40e+7  1.20e+7  1.00e+7  8.00e+6   6.50e+6  5.00e+6
      4.00e+6  3.00e+6  2.50e+6  2.00e+6   1.66e+6  1.44e+6
      1.22e+6  1.00e+6  0.80e+6  0.60e+6   0.40e+6  0.30e+6
      0.20e+6  0.10e+6  0.05e+6  0.02e+6   0.01e+6
84**  1.46e+7  1.36e+7  1.25e+7  1.125e+7  1.00e+7
      8.25e+6  7.00e+6  6.07e+6  4.72e+6   3.68e+6
      2.87e+6  1.74e+6  0.64e+6  0.39e+6   0.11e+6
      6.74e+4  2.48e+4  9.12e+3  2.95e+3   9.61e+2
      3.54e+2  1.66e+2  4.81e+1  1.60e+1   4.00e+0
      1.50e+0  5.50e-1  7.09e-2  1.00e-5   T
LIGHT ELEMENT AEA GROUP STRUCTURE
LIGHT ELEMENT AEA GROUP STRUCTURE
LIGHT ELEMENT AEA GROUP STRUCTURE
LIGHT ELEMENT AEA GROUP STRUCTURE
LIGHT ELEMENT AEA GROUP STRUCTURE
LIGHT ELEMENT AEA GROUP STRUCTURE
LIGHT ELEMENT AEA GROUP STRUCTURE
LIGHT ELEMENT AEA GROUP STRUCTURE
LIGHT ELEMENT AEA GROUP STRUCTURE
56$$ F0  T
END
```

THIS PAGE INTENTIONALLY LEFT BLANK

4.7 Containment – STC-WVDP

The transport containment boundary of the NAC-STC Transportation Cask is designed and analyzed to ensure the containment of the cask contents in accordance with 10 CFR 71 (71.43 and 71.51). The containment boundary is helium leakage tested to ANSI N14.5-1997 leaktight criteria and is designed, fabricated, and inspected in accordance with ASME Code, Section III, Subsection NB. The cask is designed to facilitate leakage testing of the containment boundary penetrations (i.e., inner lid and inner lid port cover plates) prior to transport to confirm the containment boundary.

The HLW Overpack is not a component of the cask containment system.

4.7.1 Containment Boundary – STC-WVDP

The applicable containment boundary for the STC-WVDP is the metallic seal Containment Condition B described in Section 4.1. The containment boundary of the NAC-STC, including the containment vessel, containment penetrations, seals and welds, and closure remains as described in Section 4.1.

The structural integrity of the cask containment during normal conditions of transport is demonstrated in Section 2.6 and for hypothetical accident conditions in Section 2.7.

4.7.2 HLW Overpack and Cask Pressurization

The HLW contents are borosilicate glass. The material was poured into the HLW canisters (HLW canisters and evacuated canisters) at a high temperature and allowed to cool within the canister. The HLW debris canister contains shards of vitrified glass and refractory material.

Pressurization of the HLW canister was evaluated during waste form qualification. Considered in this process was gas generated by the glass, alpha/helium generation within the glass, and changes in temperature of the canister. Evacuated canisters and HLW canisters contain borosilicate glass containing radioactive waste. The HLW debris canister contains borosilicate glass and radioactive waste (within the glass matrix), refractory material (high temperature stable material composed of alumina and silica) and potentially small amounts of alumina from melter inserts. All three types of canisters contain completely vitrified waste material.

The contents of HLW canisters contain the maximum radionuclide quantities, generate the maximum heat, and have the potential for maximum gas generation (alpha generation). As documented in the following section debris canisters do not contain material that will increase pressure. The HLW canister content bounds that of the evacuated and debris canisters. Containment/pressure related conditions allow any combination of canisters to be loaded into the NAC-STC.

Gas generation (Non-alpha particle)

Literature review and testing of the HLW documented that no significant amount of gas will accumulate inside the canister at temperatures under 500°C. As the refractory material and alumina do not generate gases (high temperature stable Al_2O_3 and SiO_2), this conclusion also applies to the HLW debris canister. The maximum accident condition temperature of the HLW is under 300°C in the NAC-STC, with normal condition temperatures significantly lower. As no significant non-alpha gas is generated by the canisters (HLW, evacuated, or debris), there are no combustible gas generation concerns with the transport of this payload.

Alpha Gas Generation/Pressurization

The HLW canister qualification report estimated a maximum helium generation equivalent to 0.16 psi over 100 years (pressure rise in canister). Transport prior to this time frame would proportionally reduce the pressure increment. Distributing the pressure rise over the larger free volume of the HLW Overpack and cask also reduces the container calculated pressure rise. This small pressure increase is insignificant to system performance. Thus, it is not applied in other evaluations.

Temperature Effects

As the HLW Overpack is not backfilled, it contains room temperature air at atmospheric pressure when it is sealed. The STC is backfilled with helium to 1 atm (0 psig). The initial temperature of both the HLW Overpack and the STC backfill is room temperature (60°F). The maximum hypothetical accident condition temperature for the HLW contents is 515°F. The increase in pressure within the STC and HLW Overpack is found using the ideal gas law (Equation 4.7-1), which simplifies to Equation 4.7-2 as the free volume and moles of gas remain constant. Conservatively rounding the maximum temperature to 520°F and applying it to the HLW Overpack and STC cask backfill gas yields a maximum HLW Overpack and cask pressure of 1.89 atm (13.1 psig). This pressure is well under the NAC-STC containment boundary design pressure of 75 psig. Similarly, the maximum temperature under normal conditions with 100°F ambient temperature is 206°F. Rounding the maximum normal condition temperature to 210°F produces a maximum normal condition pressure of 1.29 atm (4.25 psig).

Equation 4.7-1 Ideal Gas Law

$$PV = nRT$$

Equation 4.7-2 Isometric Ideal Gas Law (Constant Volume)

$$P_2 = \frac{P_1 T_2}{T_1}$$

THIS PAGE INTENTIONALLY LEFT BLANK

5.7 Shielding Evaluation – STC-WVDP

This section provides the shielding evaluation of the STC-WVDP. The STC-WVDP is designed to transport HLW Overpacks containing up to 5 High Level Waste (HLW) canisters, HLW evacuated canisters, or HLW debris canisters. The analysis of the HLW canisters is performed using the ORIGENS module of the SCALE 6 package for source terms and MCNP5 version 1.6 for shielding.

The STC-WVDP is assigned a nominal Transport Index of 0.1 (TI = 0.1) based on the requirement of 10 CFR 71.4 and the analysis results presented in Section 5.7.4.3. The maximum dose rate at 1 meter from the STC-WVDP in normal conditions of transport is 0.09 mrem/hour. Analyses are based on a transport date of April 1, 2014. Dose rates for transport dates later than the evaluated are bounded by this evaluation.

The shielding evaluation for the STC-WVDP demonstrates compliance with 10 CFR 71 limits.

The overpack containing HLW canisters is placed in the NAC-STC cavity between bottom and top spacers. In addition to the shielding provided by the cask body and lids, radial and axial shielding is provided by the overpack 3/8-inch shell, the 4 inches of stainless steel from the overpack closure lid, 2 inches of steel from the overpack bottom, and 1 inch steel from the basket bottom.

The HLW basket is constructed from a base plate, 5 loading tubes with support plates and a top shield disk. The basket height is 119.75 inches and has a diameter sized to fit inside the overpack, which has an outer diameter of 70.56 inches.

THIS PAGE INTENTIONALLY LEFT BLANK

Table 5.7.1-1 Summary of STC-WVDP Normal Condition Maximum Dose Rates – Maximum Source

Tally	Source	Surface	
		mrem/hr	FSD
Radial	Neutron	0.487	2.3%
	Gamma	0.006	3.9%
	Total	0.493	2.2%
Top Axial	Neutron	0.171	0.7%
	Gamma	0.002	1.9%
	Total	0.173	0.6%
Bottom Axial	Neutron	0.156	1.5%
	Gamma	0.010	4.2%
	Total	0.166	1.4%

Table 5.7.1-2 Summary of STC-WVDP Accident Condition Maximum Dose Rates – Maximum Source

Tally	Source	1 meter[3]	
		mrem/hr	FSD
Radial	Neutron	1.528	2.0%
	Gamma	1.309	4.2%
	Total	2.837	2.2%
Top Axial	Neutron	0.223	1.3%
	Gamma	0.003	2.6%
	Total	0.226	1.3%
Bottom Axial	Neutron	0.351	0.6%
	Gamma	0.010	6.3%
	Total	0.361	0.6%

[3] Measured from the surface of the cask.

THIS PAGE INTENTIONALLY LEFT BLANK

having aluminum as the base material and boron (in elemental boron or boron carbide form) as the primary neutron absorber. The three types are borated aluminum, metal matrix composite, and BORAL. BORAL is credited at 75% effectiveness. This efficiency factor converted the 0.02 g ^{10}B/cm^2 specified content to an effective content of 0.015 g ^{10}B/cm^2. While TALBOR, a MMC, was credited at 75% in previous sections the generic absorber definition in Chapter 8 applies 90% effectiveness to borated aluminum and MMC. 90% may be credited if the specified fabrication and qualification testing and material requirements are met. An effective areal density of 0.015 g ^{10}B/cm^{2}converts to an as build requirement of 0.017 g ^{10}B/cm^{2}when adjusted for 90% effectiveness.

MMCs and borated aluminum materials typically don't permit the high boron content (wt. %) that BORAL allows for mechanical and manufacturing reasons. To allow maximum flexibility in absorber choice an optional 0.100 inch absorber thickness, versus 0.075 inch in the baseline analysis, is added to the system. To retain a similar tube OD with absorber installed the tube ID is reduced by 0.05 inches (two times the 0.1-0.075 inch difference in absorber thickness)

Criticality evaluations for absorber thickness modifications are based on the maximum reactivity case hypothetical accident condition documented in the Section 6.4.2.5. This case models the bounding AFAM fuel assembly at 4.5 wt. % ^{235}U, and interior moderator density of 1 g/cm^3, an exterior moderator density of 0.00001 g/cm^3, a wet pellet-clad gap, and 100% geometric tolerances applied to the fuel rods. The k_s for this case is 0.9479. The absorber thickness is increased to 0.100 inch with the tube changed to a 8.83 inch nominal OD (producing a 8.734 nominal ID based on a 0.048 wall thickness versus the original evaluation 8.78 nominal ID). Tolerance evaluations for the absorber are then added. The tube wall dimensions applied are constant for nominal (0.100 inch) and absorber thickness tolerance cases. Absorber plate thickness and the needed ^{10}B input description for each absorber thickness evaluated are listed in Table 6.4.2-5. All three thicknesses evaluated produce a 0.015 g^{10}B/cm^2 areal density. The results of the absorber study are shown in Table 6.4.2-4. None of these increases was statistically significant ($\Delta k/\sigma \leq 2$).

Table 6.4.2-1 Criticality Results for Normal Conditions of Direct Fuel Loading

Cask Pitch	H_2O Inside	H_2O Outside	Neutron Shield	^{10}B	k_{eff}	σ	k_s
250 cm	1.0	1.0	Yes	75 %	0.91291	0.00086	0.92698
270 cm	1.0	1.0	Yes	75 %	0.91137	0.00085	0.92543
300 cm	1.0	1.0	Yes	75 %	0.91086	0.00087	0.92493
250 cm	0.8	0.8	Yes	75 %	0.84595	0.00083	0.86001
270 cm	0.8	0.8	Yes	75 %	0.84564	0.00083	0.85970
300 cm	0.8	0.8	Yes	75 %	0.84631	0.00083	0.86037
250 cm	0.6	0.6	Yes	75 %	0.76900	0.00114	0.78319
270 cm	0.6	0.6	Yes	75 %	0.76642	0.00110	0.78059
300 cm	0.6	0.6	Yes	75 %	0.76671	0.00117	0.78092
250 cm	0.4	0.4	Yes	75 %	0.67331	0.00106	0.68746
270 cm	0.4	0.4	Yes	75 %	0.67276	0.00104	0.68691
300 cm	0.4	0.4	Yes	75 %	0.67441	0.00110	0.68858
250 cm	0.2	0.2	Yes	75 %	0.55708	0.00121	0.57131
270 cm	0.2	0.2	Yes	75 %	0.55593	0.00120	0.57015
300 cm	0.2	0.2	Yes	75 %	0.55529	0.00110	0.56946
250 cm	0.1	0.1	Yes	75 %	0.49153	0.00123	0.50577
270 cm	0.1	0.1	Yes	75 %	0.49294	0.00130	0.50722
300 cm	0.1	0.1	Yes	75 %	0.49293	0.00134	0.50723

Table 6.4.2-4 Directly Loaded Basket – Neutron Absorber Thickness Study

Absorber Thickness	k_{eff}	σ	k_s	$k_{eff}+2\sigma$	Δk	Δk/σ
0.0075 inch base case	0.93388	0.00083	0.94794	0.93554	—	—
0.100 inch Ab. plus tube redimension (Nominal)	0.93408	0.00084	0.94814	0.93576	0.00022	0.3
0.094 in Ab. (Min Ab. Tolerance)	0.93431	0.00086	0.94838	0.93603	0.00049	0.6
0.106 in Ab. (Max Ab. Tolernace)	0.93549	0.00086	0.94956	0.93721	0.00167	1.9

Table 6.4.2-5 Directly Loaded Basket – Neutron Absorber Thickness Study – Material Description

	Minimum Absorber Thickness	**Nominal Absorber Thickness**	**Maximum Absorber Thickness**
Density (g/cm^3)	2.71	2.71	2.71
^{10}B Weight Fraction	0.023179	0.021791	0.020561
Sheet Thickness (cm)	0.2386	0.2538	0.2690
^{10}B Areal Density (g/cm^2)	0.014988	0.014988	0.014989

Note: Material modeled for this study is a Al-B sheet.

THIS PAGE INTENTIONALLY LEFT BLANK

6.9.1 Discussion and Results – WVDP HLW Canisters

West Valley HLW canisters include high level waste glass log canisters, evacuated canisters, and high level waste debris canisters. The HLW canisters meet the fissile material exemption. In order to meet the fissile material exemption in 10 CFR 71.15 (c) per subsection (1)(i), there must be less than 1 gram of fissile material per 2 kilograms of solid non-fissile material commingled together. Also, per 10 CFR 71.15 (c) subsection (1)(ii), no more than 180 grams of fissile material are allowed within 360kg of contiguous non-fissile material. The maximum amount of fissile material in an HLW canister is 420 grams, requiring at minimum 840 kilograms of non-fissile material to meet both requirements. The process of producing the HLW glass, waste and debris generates a relatively uniform composition assuring compliance with 10 CFR 71.15(c)(1)(ii). As the minimum HLW canister weight exceeds 1800 kg, HLW canisters meet this specification. Additionally, 10 CFR 71.15(c)(2) stipulates that lead, beryllium, graphite, and hydrogenous material enriched in deuterium must not be included in determining the required mass of solid non-fissile material. Those materials are not present in the glass composition. Therefore, 10 CFR 71.15(c)(2) requirements are met. The HLW debris and evacuated canisters may be under-loaded and contain remnants of the glass material with other non-fissile debris (i.e., insulation) and meet the exemption requirements. The CSI for this system in this configuration is 0.

THIS PAGE INTENTIONALLY LEFT BLANK

Table of Contents

List of Figures

List of Tables

7.0 OPERATING PROCEDURES

This chapter provides an outline of the operating procedures and tests that are performed to ensure proper function of the NAC-STC during transport operations. The operating procedures provided in this chapter are the minimum generic requirements for loading, unloading, preparation for transport, and for inspection and testing of the cask. Bolt torque values are provided in Table 7-1. Each licensee and cask user will develop, prepare and approve site specific procedures, based on the approved detailed operating procedures provided by NAC, to assure that cask handling and shipping activities are performed in accordance with the package Certificate of Compliance and the applicable Nuclear Regulatory Commission and Department of Transportation regulations governing the packaging and transport of radioactive materials.

These procedures assume that the unloaded NAC-STC arrives at a site already configured for use at the site. If this is not the case, then additional operations would be specified in the site specific procedures to configure the cask for the intended use.

The operating procedures in this chapter have been written assuming direct loading or unloading of fuel in the basket in the NAC-STC in a spent fuel pool, or dry loading and unloading of a sealed canister in the reactor cask receiving area, fuel building or other suitable location identified by the user. With minor modifications, site-specific procedures can be written to accommodate the dry direct loading or unloading of fuel from the cask in a hot cell.

Procedures are also provided for the preparation for shipment of an NAC-STC cask that has been loaded and stored at an Independent Spent Fuel Storage Installation (ISFSI) in accordance with the ISFSI license and the 10 CFR 72 requirements.

It is the responsibility of the cask user to prepare site-specific handling procedures in accordance with the Certificate of Compliance, these generic procedures, and the licensee's Quality Assurance program. User approved operating procedures ensure that critical steps are not overlooked, that the packaging is handled in accordance with its Certificate of Compliance and Safety Analysis Report.

The user will verify by fuel accounting, historical data, and inspection records, that the fuel assemblies to be loaded are in compliance with the content conditions of the Certificate of Compliance. In the directly loaded configuration, fuel assemblies or fuel rods with known or suspected cladding defects that exceed pin holes and hairline cracks are not to be loaded into the

NAC-STC. In the canistered configuration, damaged (failed) fuel will be separately containerized (canned) and sealed in the canister prior to transport.

The user shall verify that the NAC-STC transport cask has the correct O-ring configuration for the intended use. The transport cask may be configured with either metallic O-rings or with non-metallic Viton O-rings. The O-rings may not be used interchangeably, since each O-ring type requires a different lid O-ring groove configuration. Consequently, the inner lid, vent and drain port coverplates and outer lid are machined with a square O-ring groove to accept metallic O-rings or are machined with a truncated triangular (dove-tail) groove to accept Viton O-rings.

Viton O-rings may be used only when directly loading spent fuel for transport without interim storage. Metallic O-rings must be used when directly loading spent fuel for an extended period of storage and may be used when directly loading standard PWR spent fuel assemblies having burnups of ≤ 45 GWd/MTU for transport without interim storage. Metallic O-rings must also be used when loading canistered fuel, Greater Than Class C (GTCC) waste, or canistered High Level Waste (HLW) for transport. Viton O-rings tested to leaktight criteria are required for the containment boundary for the directly loaded HBU fuel assemblies. The metallic and nonmetallic O-rings have different limits of allowable leak rate as specified in the procedures.

7.1 Outline of Procedures for Receipt and Loading the Cask

The following receipt and loading procedures are based on an acceptable cask receipt inspection for first time loading with spent fuel. For casks previously loaded and transported, the receiving inspections will require performance of radiation and removable contamination surveys of the empty cask and vehicle in accordance with 10 CFR 71 and 49 CFR 173 in the U.S. Similar requirements are contained in IAEA SSR-6.

7.1.1 Receiving Inspection

1. Perform radiation and removable contamination surveys in accordance with 49 CFR 173.441 and 173.443 requirements.
2. Move the transport vehicle with the cask to the cask receiving area.
3. Secure the transport vehicle. Remove the personnel barrier hold down bolts from both sides of the personnel barrier. Using the lifting sling, lift the personnel barrier off of the cask and store it in a designated area.
4. Visually inspect the NAC-STC while secured to the transport vehicle in the horizontal orientation for any signs of damage.
5. Attach slings to the top impact limiter lifting points, remove impact limiter lock wires, impact limiter jam nuts, impact limiter nuts and retaining rods. Remove impact limiter and store upright. Repeat operation for the bottom impact limiter.
6. Release the tiedown assembly from the front support by removing the front tiedown bolts and lock washers.
7. Attach a sling to the tiedown assembly lifting eyes and remove the tiedown assembly from the transport vehicle.
8. Attach the cask lifting yoke to a crane hook with the appropriate load rating. Engage the two yoke arms with the lifting trunnions at the top (front) end of the cask. Rotate/lift the cask to the vertical orientation and raise the cask off of the blocks of the rear support structure of the transport vehicle. Place the cask in the vertical orientation in a decontamination area or other suitable location identified by the user. Disengage the cask lifting yoke from the lifting trunnions.

7.1.2 Preparation of Cask for Loading

The loading procedures are based on the assumption that the cask is being prepared for first time fuel loading following fabrication, or that the scheduled annual maintenance required by the

Certificate of Compliance has been successfully completed within the previous 12 months. If the cask has been used previously, at the start of this procedure, the cask is assumed to be externally decontaminated, empty of fuel contents, and sitting in the decontamination area, or in another location convenient for preparing the cask.

There are two (2) loading options for the NAC-STC. Each requires different preparation steps. The first is direct loading of fuel assemblies into a fuel basket installed in the cask, which is typically performed under water in the spent fuel pool cask loading area. The second is dry loading of a welded transportable storage canister that is already loaded with spent fuel assemblies, Reconfigured Fuel Assemblies, damaged fuel in damaged fuel cans, HLW overpack or with Greater Than Class C (GTCC) waste. Dry loading of the canister into the cask is performed in the cask receiving area, or another convenient location established by the user, using a transfer cask system. This section presents the generic procedures used to prepare the cask for loading for either wet direct fuel loading or dry canister loading.

7.1.2.1 Preparation for Direct Fuel Loading (Uncanistered)

This procedure presents the steps necessary to prepare the cask for under water direct loading of fuel into a basket contained in the NAC-STC cask. This procedure may be modified to accommodate the dry direct loading of fuel in a hot cell.

1. Install appropriate work platforms/scaffolding to allow access to the top of the cask.
2. Detorque in reverse torquing sequence and remove the outer lid bolts. Install the two outer lid alignment pins.
3. Install lifting eyes in the outer lid lifting holes and attach the outer lid lifting sling to the outer lid and overhead crane. Remove the outer lid and place it aside in a temporary storage area. When setting the outer lid down, protect the O-ring and the O-ring groove of the lid from damage. Remove the outer lid alignment pins. Decontaminate the surface of the inner lid and top forging as required. At a convenient time, if a metallic O-ring is used, remove and replace the metallic O-ring in the outer lid. If a Viton O-ring is used, inspect the O-ring and replace as necessary.
4. Detorque drain and vent coverplate bolts and remove the drain port and the vent port coverplates from the inner lid. Store in temporary storage area.

9. Lower auxiliary hook to above inner lid and engage lid lifting sling to auxiliary crane hook.
10. Slowly lift and remove the inner lid. The inner lid alignment pins will guide the inner lid until it clears the top forging.
11. Store the inner lid in a temporary storage area. When storing the inner lid, ensure that the O-rings and O-ring grooves of the lid are protected from damage. Decontaminate the inner lid, as necessary.
12. Visually examine the internal cavity to ensure that the cavity is free of damage and foreign materials.
13. Install the appropriate bottom spacer(s) for the canister to be loaded. (Note: The MPC-LACBWR canister requires the Yankee-MPC bottom spacer and the MPC-LACBWR supplemental bottom spacer. The MPC-WVDP Overpacks require the use of the MPC-WVDP bottom and top transport spacers.) Attach the spacer lift fixture to the spacer(s). Using a suitable crane, lower the spacer(s) into the cask cavity and remove the lift fixture.
14. Install the adapter ring and torque the three bolts to 100 ± 20 ft.-lb.
15. Install the transfer cask adapter plate on top of the NAC-STC cask.

7.1.3 Loading the NAC-STC Cask

There are three loading options for the NAC-STC cask, with each requiring different steps. The first is direct loading of fuel assemblies for transport without interim storage and the second option is for transport after a period of interim storage. These loading configurations are assumed to be performed under water in the spent fuel pool cask loading area. The third option is dry loading into the cask of a sealed transportable storage canister or HLW overpack that already contains spent fuel, GTCC waste, or canistered HLW contents. Dry loading of the canister into the cask is performed in the cask receiving area, adjacent to a storage pad at an Interim Storage Facility, or other convenient location established by the user, using a transfer cask. This section presents the generic loading procedures for these options. In all cases, the fuel assemblies to be directly loaded, or those contained within the sealed canister, canistered GTCC waste, or HLW contents must conform to the content conditions of the NAC-STC Certificate of Compliance (COC).

7.1.3.1 Direct Loading of Fuel (Uncanistered)

The NAC-STC may be closed with either metallic or nonmetallic O-rings in the containment boundary and outer lid. Metallic O-rings are required: 1) when directly loading spent fuel for an extended period of storage; and 2) when loading canistered fuel, GTCC waste, or canistered HLW (for transport). Metallic O-rings or Viton O-rings may be used when directly loading standard PWR spent fuel for transport without interim storage. Viton O-rings are required to be used when directly loading HBU PWR spent fuel with burnups of > 45 GWd/MTU for transport without interim storage. However, the metallic and non-metallic O-rings may not be used interchangeably, as the O-ring grooves in the lids and port covers are different for each O-ring type. As specified in the appropriate steps of this procedure, the two types of O-rings have different allowable leak rates so the lid and O-ring configurations to be used must be confirmed and the associated leak test requirements identified.

1. Using approved fuel identification and handling procedures and fuel handling equipment, engage the fuel handling tool to the top of the fuel assembly, lift it from the storage rack location, transfer it to above the cask, and carefully lower it into the designated location in the fuel basket. Be careful not to contact any of the sealing surfaces on the top forging, or to come in contact with the inner lid guide pins during fuel assembly movement.

   Note: 1. Each fuel assembly shall contain the standard number of fuel rods for an assembly of that type. Dummy rods of equivalent water displacement must be substituted for removed fuel rods.

   2. Perform an independent verification that the spent fuel assemblies loaded in the fuel basket are in full compliance with the content conditions of the NAC-STC Certification of Compliance (CoC) No. 9235.

   3. Following loading of HBU fuel assemblies, perform an independent verification that the HBU fuel contents comply with the designated maximum heat load per assembly specified for the loading configuration and that the correct number and loading positions for the fuel assemblies and shielded thermal shunts correspond to the applicable loading configuration identified on NAC License Drawing No. 423-800.
2. Record in the cask loading report the fuel identification number and basket position where the fuel assembly was placed.
3. Repeat steps 1 and 2 until the basket is fully loaded or until all desired fuel assemblies have been loaded. If the cask is going to be partially loaded, the fuel assemblies should

be loaded, if possible, in a fully symmetric pattern to ensure that the center of gravity of the cask remains aligned as close as possible to the longitudinal axis of the cask.

4. Attach the inner lid lifting sling to an auxiliary crane hook and lift the inner lid. For the Viton O-ring assembly, inspect the O-ring and replace if damaged. For the metallic O-ring assembly, remove the inner lid O-rings, clean the groove surfaces, and install new metallic O-rings. Inspect new O-rings for damage prior to installation. Secure the metallic O-rings in the groove by the use of the O-ring clips and screws. Similarly, replace the metallic O-rings in the vent and drain port coverplates, or inspect the Viton O-rings and replace if required.
5. After replacing the inner lid O-rings, as required, lift the inner lid and place it on the cask using the inner lid alignment pins to assist in proper lid seating and orientation. Visually verify proper lid position.
6. Disconnect the lid lifting device from the auxiliary crane hook and remove crane hook from area.
7. Attach the lifting yoke to the crane hook, lower the lifting yoke into the lifting position over the cask lifting trunnions, and engage the lifting arms to the lifting trunnions. Slowly lift the cask out of the pool until the top of the cask is slightly above the pool water level.

   Note: As an alternative method, the cask and inner lid may be handled simultaneously. In the event that this method is chosen, instead of performing steps 5, 6 and 7, attach the lifting yoke to a crane hook and the inner lid lifting eyes to the lift yoke. Lower the lid and engage to the cask using the lid alignment pins. Engage lifting arms to lifting trunnions. Slowly lift the cask out of the pool until the top of the cask is slightly above the pool water level.
8. Attach a drain line to the quick-disconnect in the interlid port (located in the top forging) and allow the water to drain from the interlid region. Once drained, disconnect the drain line.
9. Install at least 10 inner lid bolts equally spaced on the bolt circle to hand tight.
10. Continue raising the cask from the pool while spraying the external cask surfaces with clean water to minimize surface contamination levels.
11. Move the cask to the cask decontamination area, lower the cask to the floor and disengage the lift yoke (or lift beam and inner lid lifting slings if the alternate method of handling the inner lid was used). Remove the lift yoke and crane from the area.
12. Connect a vent line to the vent port quick-disconnect. Direct the free end of the vent line to a radioactive waste handling system capable of handling liquids and gas.

13. Remove the inner lid alignment pins and install the remaining inner lid bolts and torque all of the bolts to the torque value specified in Table 7-1. The bolt torquing sequence is shown on the inner lid.
14. Connect a drain line to the drain port quick-disconnect (located in the inner lid). Remove the vent line from the vent port quick-disconnect.
15. Drain the cask cavity by connecting a helium supply to the vent port quick-disconnect (located in the inner lid). Purge the water from the cask by pressurizing to 35 to 40 psig and hold until all water is removed (observed when no water is coming from the drain line). Turn the helium supply off and disconnect the helium supply line from the vent port. Then, disconnect the drain line from the drain port quick-disconnect.

    Note: In cases where the inner lid Viton O-ring requires replacement due to failure of the preshipment leakage rate test, blowdown approximately 50 gallons from the cask cavity using helium. Perform the helium leakage test specified in Step 19c. on the replaced inner lid Viton O-ring. Once maintenance leakage rate test results are acceptable, blowdown the remaining volume of water from the cavity per Step 15.
16. Connect a vacuum pump to the cask cavity via the vent and drain port quick-disconnects in the inner lid. Evacuate the cask cavity until a pressure of 4 mbar is reached. Continue pumping for a minimum of 1 hour after reaching 4 mbar. Valve off vacuum pump from system and using a calibrated vacuum gauge (minimum gauge readability of 2.5 mbar), observe for a pressure rise. If a pressure rise (ΔP) of more than 12 mbar in ten minutes is observed, continue pumping until the pressure does not rise more than 12 mbar in ten minutes. Repeat dryness test until cavity dryness has been verified ($\Delta P < 12$ mbar in 10 minutes). Record test results in the cask loading report.

    Note: The maximum vacuum time for HBU PWR spent fuel assemblies from the start of cavity draining operations through completion of helium backfill in Step 15 through completion of helium backfill per Step 17 shall not exceed 40 hours. If maximum vacuum drying time is reached, the cask shall be reflooded and cooled down in accordance with the cooldown procedures of Section 7.3.2.1, Steps 4 through 7. When the cask and fuel contents are cooled, restart draining and vacuum drying operations per Step 15 above. The maximum vacuum drying time is limited to $\leq$ 40 hours for each drying cycle.

    Note: The total time from the start of draining operations through placement of the loaded NAC-STC in a horizontal position on the transport vehicle in Section 7.2.1, Step 2 is 48 hours.

17. Without allowing air to re-enter the cask cavity, turn off and isolate the vacuum pump. Connect a supply of helium (99.9% minimum purity) to the vent port quick-disconnect and backfill the cask cavity to 0 psig helium pressure.
18. Install the drain and vent port coverplates using new metallic O-rings or inspected Viton O-rings. Torque the bolts to the value indicated in Table 7-1.
19. Perform inner lid O-ring leakage testing as follows:
19a. For the metallic O-ring assembly, connect the leak detector vacuum pump to the inner lid interseal test port and evacuate the air between the O-rings to <1 mbar. Hold the vacuum on the interseal for the metallic O-ring assembly region. Using the helium leak detector, verify that any detectable leak rate for metallic O-rings is $\leq 2 \times 10^{-7}$ cm^3/sec (helium). The test sensitivity shall be $\leq 1 \times 10^{-7}$ cm^3/sec (helium).
19b. For Viton O-rings, perform the preshipment leakage rate test to confirm no detected leakage to a test sensitivity of 1×10^{-3} ref cm^3/sec by pressurizing the O-ring annulus to 15 (+2, -0) psig and isolating for a minimum of 15 minutes. There shall be no loss in pressure during the test period.
19c. Following the replacement of inner lid Viton O-rings required due to excessive wear of the O-rings or failure of the preshipment leakage rate test, the inner lid maintenance leakage rate test shall be performed following Step 14 by partially blowing down the NAC-STC of 50 gallons of cavity water with helium gas. Connect the helium Mass Spectrometer Leak Detector (MSLD) to the interseal test port to verify the total cumulative leakage rate is $\leq 9.3 \times 10^{-5}$ cm^3/sec (helium) [(1)] with a minimum test sensitivity of 4.7×10^{-5} cm^3/sec (helium) for standard PWR spent fuel assemblies. For HBU spent fuel assembly contents, connect the MSLD to the interseal test port to verify the new Viton O-rings leakage rate is $\leq 2.0 \times 10^{-7}$ cm^3/sec (helium) [(1)] with a minimum test sensitivity of 1.0×10^{-7} cm^3/sec (helium). After completion of the leakage rate test of the inner lid O-ring seals, the cask preparation procedures will restart at Step 15 except that Step 19b. does not need to be performed as the maintenance leakage rate test has been completed.
20. Install the test port plug for the inner lid interseal test port using a new metallic O-ring and torque the plug to the value specified in Table 7-1.
21a. For the metallic O-ring, connect a vacuum pump to the vent port coverplate interseal test port and evacuate the air between the O-rings to <1 mbar. Hold the vacuum on the interseal for the metallic O-ring assembly region. Using the helium leak detector, verify that any detectable leak rate for metallic O-rings is $\leq 2 \times 10^{-7}$ cm^3/sec (helium). The test sensitivity shall be $\leq 1 \times 10^{-7}$ cm^3/sec (helium).
21b. For Viton O-rings[(1)], perform the preshipment leakage rate test to confirm no detected leakage to a test sensitivity of 1×10^{-3} ref cm^3/sec by pressurizing the O-ring annulus to

15 (+2, -0) psig and isolating for a minimum of 15 minutes. There shall be no loss in pressure during the test period.

21c. For new replacement Viton O-rings, use a leak detector connected to the interseal test port to verify the leakage rate is $\leq 9.3 \times 10^{-5}$ cm^3/sec (helium) [1] with a minimum test sensitivity of 4.7×10^{-5} cm^3/sec (helium) for standard PWR spent fuel assemblies and $\leq 2.0 \times 10^{-7}$ cm^3/sec (helium) [1] with a minimum test sensitivity of 1.0×10^{-7} cm^3/sec (helium) for HBU spent fuel assemblies (> 45 MWd/MTU).

22. Install the test port plug for the vent port coverplate using a new metallic O-ring and torque the plugs to the value specified in Table 7-1.
23. Repeat Steps 21 and 22 for the drain port coverplate.[1]
24. Drain residual water from the pressure port, ensuring that the pressure port is clear to also allow water to drain from the interlid region.
25. Install the transport pressure port cover on the pressure port. Torque the port cover bolts to the value specified in Table 7-1.
26. Perform a functional leak test on the pressure port cover by removing the O-ring test plug and using a test fixture, pressurize the annulus between the pressure port cover O-rings to 15 psig and isolate. During a 10-minute test period, there shall be no loss in pressure during the test period.
27. Install the pressure port cover interseal test port plug and O-ring and torque the plug to the value specified in Table 7-1.
28. For the metallic outer lid O-ring assembly, remove the O-ring, clean the O-ring seating surface and groove, and install a new metallic O-ring. For Viton O-ring assemblies, inspect the O-ring and replace if damaged.
29. Install outer lid and align vent pins.

---

(1) For new Viton O-rings on a package containing directly loaded standard PWR spent fuel assemblies, the combined leakage rates are for the inner O-ring of the lid, inner O-ring of the vent port cover plate and inner O-ring of the drain port cover plate, which are part of the containment boundary. The combined measured leakage rate from all three Viton O-rings must be less than or equal to $\leq 9.3 \times 10^{-5}$ cm^3/sec (helium) in accordance with 10 CFR 71.51. For new Viton O-rings on a package containing up to 20 directly loaded HBU fuel assemblies (> 45 GWd/MTU) the leakage rate for each tested component shall be $\leq 2.0 \times 10^{-7}$ cm^3/sec (helium) [1] with a minimum test sensitivity of 1.0×10^{-7} cm^3/sec (helium).

30. Attach the outer lid lifting device to the outer lid and overhead crane. Install the outer lid using the alignment pins to assist in proper seating. Remove the outer lid alignment pins. Install the outer lid bolts and torque to the value specified in Table 7-1. The bolt torquing sequence is shown on the outer lid.
31. Attach a supply of air or helium to the interlid port quick-disconnect. Backfill the interlid volume to 15 psig air or helium and hold for 10 minutes. There shall be no pressure loss during the test period. Disconnect air or helium supply.
32. Install the interlid port cover using new metallic O-rings. Torque the interlid port cover bolts to the value specified in Table 7-1.
33. Remove the test plug from the interlid port cover and, using the O-ring test fixture, pressurize the O-ring annulus to 15 psig with air or helium. Isolate the annulus and hold for 10 minutes. No loss of pressure is permitted during the test period.
34. Remove the air or helium supply and vent the annulus pressure. Replace the metallic O-ring on the interlid port cover test plug, install the test plug and torque it to the value specified in Table 7-1.
35. Perform final external decontamination and perform survey to verify acceptable level of removable contamination to ensure compliance with 49 CFR 173.443. Perform final radiation survey. Record the survey results.
36. Perform final visual inspection to verify assembly of the NAC-STC in accordance with the Certificate of Compliance. Verify that the loading documentation has been appropriately completed and signed off.

7.1.3.2 Loading Canistered Fuel, Canistered GTCC Waste, or HLW Overpacks

Canistered fuel, canistered GTCC waste, or HLW Overpacks are loaded into the NAC-STC using a transfer cask. This procedure assumes that the canister, or overpack, has been previously loaded, drained, vacuum dried, backfilled with helium and welded closed, as applicable. The canister, or overpack, may have been retrieved from dry storage, or it may have been loaded and sealed immediately prior to loading in the NAC-STC.

Canisters containing spent nuclear fuel that are to be retrieved from storage for off-site transport will be evaluated to ensure that the specific canister stored in the storage overpack, which may have been subject to 10 CFR 72 normal, off-normal, accident and natural phenomena events, retain their ability to satisfy functional and performance requirements of the NAC-STC packaging certified content conditions. Similarly, GTCC Waste canisters and HLW Overpacks will be evaluated to ensure that the specific canister or overpack, which may have been exposed to off-normal, accident and/or natural phenomena events during storage operations prior to

loading for transport, retain their ability to satisfy functional and performance requirements of the NAC-STC packaging certified content conditions.

Canisters containing spent nuclear fuel experiencing only normal or off-normal events during storage, and canistered GTCC waste and HLW Overpacks need only be evaluated for potential corrosion at the welds and any damage caused by removal from the storage cask.

In addition to the evaluation done for normal/off-normal storage, canisters containing spent nuclear fuel that have experienced accident or natural phenomena events must be evaluated for potential degradation of the fuel, basket and neutron absorbers. This evaluation will be performed for each canister as part of the preparation for loading for off-site transport using: 1) the annual inspection and surveillance records and off-normal and accident event reports that are maintained by the licensee for each loaded NAC-MPC system in compliance with 10 CFR 72 requirements; and 2) in the case of storage accidents and natural phenomena events, any necessary examinations at the time of transfer to ensure the condition of the canister and contents.

Dry storage systems that have been maintained within an Aging Management Program will include system specific review and assessment of this information record as part of the off-site transport evaluation to ensure the NAC-STC packaging certified content conditions are validated. Maximum assembly average burnup for fuel assemblies retrieved from dry storage for off-site transport is limited to 45,000 MWd/MTU. System loading into the NAC-STC will be observed by operations staff noting any system interferences that occur during canister retrieval from the storage overpack and placement of the canister into the transport overpack. The cause of the interference and potential damage caused by the interference will be determined prior to shipment. Noted interferences will be made part of the canister evaluation record to the extent required to validate NAC-STC packaging content conditions are satisfied when the spent fuel canister is placed within the NAC-STC containment boundary for off-site transport.

This procedure assumes that the sealed canister, or HLW Overpack, conforms to the design basis of the NAC-STC with appropriate spacer configuration and that the canister is already in the transfer cask.

1. Attach the transfer cask yoke to the cask handling crane hook.
2. Engage the transfer cask yoke to the trunnions of the transfer cask.

3. Raise the transfer cask over the NAC-STC cask and lower it until it rests on the transfer cask adapter plate. Remove and store the transfer cask lifting yoke. Remove the transfer cask shield door stops.
4. Attach the two (2) canister 3-legged lifting sling sets to the hoist rings in the canister lid. Attach the opposite end of the slings to the crane hook.
   Note: Alternative canister lifting systems may be utilized.
5. Attach the hydraulic system to the operating cylinders on the transfer cask adapter plate.
6. Using the crane, raise the canister just enough ($\leq$ 1 inch) to take the canister weight off of the transfer cask bottom shield doors.
7. Open the transfer cask shield doors.
8. Lower the canister or HLW overpack into the NAC-STC cask. Exercise caution to avoid contact with the interior cavity wall.
   Note: Prior to loading into the NAC-STC cavity the condition of the spent fuel canister, greater than class C (GTCC) waste canister, or the HLW overpack, and the canister/overpack internals shall be evaluated to verify the canisters/overpacks:
   a. Meet the design requirements and CoC content conditions of the NAC-STC package;
   b. Account for the effects of any accident or natural phenomena events that the canisters or overpacks may have been exposed to during storage operations prior to loading in the NAC-STC package, and,
   c. The vitrified HLW overpack meets the limits in 10 CFR 71.15 for classifying the contents as fissile exempt.
9. Disconnect and remove the canister lifting sling from the crane hook and lower it onto the top of the canister.
10. Close the transfer cask shield doors and install the door stops.
11. Retrieve the transfer cask lifting yoke and engage the transfer cask trunnions. Lift the transfer cask from the transfer cask adapter plate. Store the transfer cask and transfer cask lifting yoke in the designated locations.
12. After removal of the lift slings, install the NAC-MPC canister top spacer, as required (for the loading of Yankee-MPC, MPC-LACBWR and HLW Overpacks only).
13. Retrieve the cask adapter plate lifting sling and attach it to the transfer cask adapter plate.
14. Remove the transfer cask adapter plate and store it in the designated location. Using the appropriate lifting sling, remove the adapter ring and bolts. Install the inner lid alignment pins.

15. Remove the inner lid O-rings and clean inner lid O-ring groove surfaces. Replace the metallic O-rings on the inner lid, carefully inspecting the new O-rings for damage prior to installation. Secure the O-rings in the groove using the O-ring clips and screws.
16. Attach the inner lid lifting slings to an auxiliary crane hook, lift the inner lid and place it on the cask using the inner lid alignment pins to assist in proper lid seating and orientation. Visually verify proper lid position.
17. Disconnect the lid lifting device from the crane hook and remove it from the inner lid.
18. Install at least 10 inner lid bolts equally spaced on the bolt circle to hand tight. Remove the inner lid alignment pins.
19. Install the remaining inner lid bolts and torque all of the bolts to the torque value specified in Table 7-1. The bolt torquing sequence is shown on the inner lid.
20. Remove the metallic O-rings in the drain port coverplate, and clean and inspect the O-ring groove. Install new metallic O-rings and install the coverplate. Torque the coverplate bolts to the value specified in Table 7-1.
21. Connect the vacuum pump to the cask vent port and evacuate the cask cavity to a stable vacuum pressure of less than, or equal to, 4 mbar (approximately 3 mm of Hg) and backfill the cask cavity with helium (99.9% minimum purity) to 0 psig without allowing air to re-enter the cask. Disconnect the vacuum pump and helium supply from the vent port.
22. Remove the metallic O-rings in the vent port coverplate and clean and inspect the O-ring groove. Install new metallic O-rings in the vent port coverplate and install the coverplate. Torque the coverplate bolts to the value specified in Table 7-1.
23. Connect the leak detector to the inner lid interseal test port and evacuate the air between the metallic O-rings until a pressure of <1 mbar is reached. Using the helium leak detector, verify that any detectable leak rate is $\leq 2 \times 10^{-7}$ cm^3/sec (helium). The test sensitivity shall be $\leq 1 \times 10^{-7}$ cm^3/sec (helium).
24. Install the test port plug for the inner lid interseal test port using a new metallic O-ring and torque the plug to the value specified in Table 7-1.
25. Connect the leak detector to the vent port coverplate interseal test port. Evacuate the interseal volume until a pressure of <1 mbar is reached. Using the helium leak detector, verify that any detectable leak rate is $\leq 2 \times 10^{-7}$ cm^3/sec (helium). The test sensitivity shall be $\leq 1 \times 10^{-7}$ cm^3/sec (helium).
26. Install the test port plug for the vent port coverplate using a new metallic O-ring and torque the plug to the value specified in Table 7-1.
27. Repeat Steps 25 and 26 for the drain port coverplate test port.

28. Remove the outer lid metallic O-ring. Clean the outer lid O-ring seating surface and groove. Install a new metallic outer lid O-ring. Install the outer lid alignment pins.
29. Attach the outer lid lifting device to the outer lid and cask handling crane. Install the outer lid using the alignment pins to assist in proper seating. Remove the outer lid alignment pins. Install the outer lid bolts and torque to the value specified in Table 7-1. The bolt torquing sequence is shown on the outer lid.
30. Attach a supply of air, nitrogen, or helium to the interlid port quick-disconnect and backfill the interlid volume to 15 psig air, nitrogen, or helium and hold for 10 minutes. No loss of pressure is permitted during the 10-minute test period. Disconnect air, nitrogen, or helium supply.
31. Install the transport interlid port cover in the interlid port using new O-rings. Torque the interlid port cover bolts to the value specified in Table 7-1.
32. Remove the O-ring test plug from the interlid port cover and, using the O-ring test fixture, pressurize the O-ring annulus to 15 psig with air, nitrogen, or helium. Isolate the annulus and hold for 10 minutes. No loss of pressure is permitted during the test period.
33. Vent the annulus pressure, remove the air, nitrogen, or helium supply, replace the metallic O-ring on the interlid port cover test plug and install the test plug. Torque the plug to the value specified in Table 7-1.
34. Perform final external decontamination and perform survey to verify acceptable level of removable contamination to ensure compliance with 49 CFR 173.443. Perform final radiation survey. Record the survey results in the cask loading report.
35. Perform final visual inspection to verify assembly of the NAC-STC in accordance with the CoC. Verify that the loading procedure and checklist are appropriately completed and signed off.

THIS PAGE INTENTIONALLY LEFT BLANK

7.2 Preparation for Transport

Perform the procedures of either Section 7.2.1 or 7.2.2, whichever is appropriate. Section 7.2.1 addresses preparation for transport without interim storage after loading the cask either with directly loaded fuel or with a previously loaded canister. Section 7.2.2 addresses transport following long-term storage of directly loaded fuel. Transport following long-term storage requires the verification of containment by leak testing the containment boundary formed by the outer O-rings of the inner lid and port covers and the O-ring test ports.

7.2.1 Preparation for Transport (Immediately After Loading)

1. Engage the lift beam to the cask lifting trunnions and move the cask to the cask loading area.
2. Load the cask onto the transport vehicle by gently lowering the rotation trunnion recesses into the rear support. Rotate the cask to horizontal by moving the overhead crane in the direction of the front support. Maintain the crane cables vertical over the lifting trunnions.
3. Using a lifting sling, place the tiedown assembly over the cask upper forging between the top neutron shield plate and front trunnions. Install the front tiedown bolts and lock washers to each side of the front support.
4. Complete a Health Physics removable contamination survey of the cask to ensure compliance with 49 CFR 173.443. Complete a Health Physics radiation survey of the entire package to ensure compliance with 49 CFR 173.441.
5. Using the designated lifting slings and a crane of appropriate capacity, install the top impact limiter. Install the impact limiter retaining rods into each hole and torque to the value specified in Table 7-1. Install the impact limiter attachment nuts and torque to the value specified in Table 7-1. Install the impact limiter jam nuts and torque to the value specified in Table 7-1. Install the impact limiter lock wires. Repeat the operation for the bottom impact limiter installation.

   Note: Balsa impact limiters shall be used for transport of the Connecticut Yankee fuel and GTCC waste canisters, MPC-LACBWR canisters, and HLW canisters loaded in HLW Overpacks. The balsa impact limiters may also be used for transport of directly loaded fuel and for Yankee-MPC fuel or GTCC waste canisters. Redwood impact limiters may only be used for transport of directly loaded fuel and for Yankee-MPC fuel or GTCC waste canisters.

6. Install security seals through holes provided in the upper impact limiter and one of the lifting trunnions; and through holes provided in all three bolts in the interlid port cover and the pressure port cover. Record the security seal identification numbers in the cask loading report.
7. Install the personnel barrier/enclosure and torque all attachment bolts to the prescribed torque value. Install padlocks on all personnel barrier/enclosure accesses.
8. Complete a Health Physics radiation survey of the entire package to ensure compliance with 49 CFR 173.441.
9. Complete a Health Physics removable contamination survey of the transport vehicle to ensure compliance with 49 CFR 173.443.
10. Determine the transport index (TI) corresponding to the maximum dose rate at 1 meter from the cask. Record on the shipping documents.
11. Determine the appropriate Criticality Safety Index (CSI) assigned to the package contents in accordance with the CoC, and indicate the correct CSI on the fissile material labels applied to the package.
12. Apply placards to the transport vehicle in accordance with 49 CFR 172.500 and provide special instructions to the carrier/shipper for an Exclusive Use Shipment.
13. Complete the shipping documentation in accordance with 49 CFR Subchapter C.

    Note: The allowable time for transport for HBU PWR spent fuel assemblies from completion of loading operations to arrival at the receiving facility is restricted to ≤ 3 months, as specified in the CoC.

    Note: The licensee will select a shipping timeframe in order to ensure that the maximum ambient temperature change for 3-month is ≤ 75°F (based on 3-day average) as specified in the CoC.

7.2.2 Preparation for Transport (After Long-Term Storage)

This procedure applies to the transport of directly loaded fuel that has been in storage in the NAC-STC. Canistered fuel, canistered GTCC waste, and HLW Overpacks may not be loaded in the NAC-STC for storage. Canistered fuel, GTCC waste canisters, and HLW canisters loaded in HLW Overpacks may have been loaded just prior to shipment or may have been in interim storage in a separate storage overpack.

Prior to placing a directly loaded cask in long-term storage, the cask cavity is backfilled with 1.0 atmosphere (absolute) of helium (99.9% minimum purity) as the normal coolant for the spent fuel and to provide an inert atmosphere to prevent possible oxidation of the fuel. The inner lid interseal volume between the two inner lid metallic gaskets and the interseal volume between the O-rings in the vent and drain port covers are backfilled with 15 psig of helium (99.9% minimum

purity). The interlid volume is pressurized to 100 psig and that pressure is monitored for pressure loss by a pressure transducer installed in the cask upper forging, and closed by a specially equipped port cover filled with a pressure feed-through tube (License Drawing No. 423-807). This overpressure system ensures that in the off-normal event of any leakage of the inner lid or port cover O-rings, the leakage path will be clean helium into the cavity. If, during the storage period, no significant pressure loss is observed in the pressure monitoring volume or system (normally recorded at a minimum of once every 24 hours during storage), it can be concluded that at the end of the storage period, the cask cavity remains backfilled with helium gas.

Prior to preparing the cask for transport, the pressure transducer wiring has been disconnected.

1. Move cask from extended storage location to a designated work area.
2. Evacuate a sample bottle using a vacuum pump and remove the interlid pressure port cover. Isolate the sample bottle and connect it to the interlid port quick-disconnect and fill it with interlid region atmosphere.

   Note: The interlid pressure may be as high as 100 psig. Use caution in collecting the gas sample.
3. Isolate the sample bottle and disconnect it from the interlid port quick-disconnect.
4. Bring the sample bottle to the appropriate facility and analyze the contents of the sample bottle.
5. If krypton-85 is present in the sample bottle, additional radiological precautions may be imposed by Health Physics personnel prior to proceeding with the removal of the outer lid. A determination shall also be made as to whether replacement of the inner lid seals is required. If the gas sample is acceptable, proceed with normal operations.
6. Attach valved venting hose to interlid port quick-disconnect and open valve to vent interlid region.
7. Remove the outer lid bolts and install the outer lid alignment pins and outer lid lifting eye bolts.
8. Attach the outer lid lifting device to the outer lid lifting eye bolts and overhead crane. Remove the outer lid and place it aside in a temporary storage area. Protect the O-ring and O-ring groove of the lid from damage. Remove the outer lid alignment pins.
9. Verify the torque of the inner lid bolts and vent and drain port coverplate bolts by torquing the bolts in accordance with the bolt torque sequence to the values specified in Table 7-1.
10. Remove the drain port coverplate port plug. Connect the leak detector vacuum pump to the drain port coverplate test port and evacuate the helium between the metallic O-rings to a pressure of <1 mbar. Without allowing air to re-enter the interseal region,

backfill the drain port coverplate interseal region with helium (99.9% minimum purity) to a pressure of 0 psig.

11. Install the drain port coverplate test plug using a new O-ring and torque to the value specified in Table 7-1.
12. Repeat steps 10 and 11 for the vent port coverplate test plug.
13. Remove the inner lid interseal test port plug and connect a vacuum pump to the inner lid interseal test port quick-disconnect. Evacuate the inner lid interseal volume until a pressure of <1 mbar.
14. Without allowing air to re-enter the interseal volume, backfill the interseal volume with helium (99.9% minimum purity) to 0 psig. Disconnect helium supply.
15. Install the inner lid interseal test port plug with a new metallic O-ring and torque the plug to the value specified in Table 7-1.
16. Clean the outer lid O-ring seating surface and groove surface. Install a new metallic O-ring in the outer lid. Reinstall the outer lid alignment pins.
17. Attach the outer lid lifting device to the outer lid lifting eye bolts and the overhead crane. Install the outer lid and visually verify proper seating. Remove the alignment pins and lifting eye bolts, and install the outer lid bolts and torque to the value specified in Table 7-1. The bolt torquing sequence is shown on the outer lid.
18. Perform an evacuated envelope leakage test on the outer O-rings of the vent and drain port coverplates, the outer O-ring of the inner lid, and the interseal test ports by connecting a vacuum pump and a helium mass spectrometer leak detector connected to the interlid port quick-disconnect. Evacuate the interlid region to a vacuum of <1 mbar.
19. Using the helium leak detector, verify that the leakage rate into the evacuated envelope is $\leq 2 \times 10^{-7}$ cm^3/sec (helium) with a minimum leak test sensitivity of $\leq 1 \times 10^{-7}$ cm^3/sec.
20. Upon completion of the leak test, backfill the interlid region with helium (99.9% minimum purity) to 0 psig and disconnect the helium supply and leak test equipment.
21. Install the transport interlid port cover using new O-rings and torque the port cover bolts to the value specified in Table 7-1.
22. Remove the interseal port plug, attach the test fixture to the interlid port interseal test hole and perform a functional leak test on the interlid port cover O-rings by pressurizing the O-ring annulus to 15 psig and isolating for a minimum of 10 minutes. There shall be no loss in pressure during the test period. Record completion of an acceptable leakage test on the cask loading report. Upon completion of the test,

equalize interseal region pressure with ambient and disconnect the test fixture. Install the interseal port plug and torque to the value specified in Table 7-1.

23. Using the lift yoke, load the cask on the transport vehicle.
24. Using a lifting sling, place the tiedown assembly over the cask upper forging between the top neutron shield plate and front trunnions. Install the front tiedown bolts and lock washers to each side of the front support.
25. Complete a Health Physics removable contamination survey of the entire package to ensure compliance with 49 CFR 173.443.
26. Using the designated lifting slings and a crane of appropriate capacity, install the top impact limiter. Install the impact limiter retaining rods into each hole and torque to the value specified in Table 7-1. Install the impact limiter attachment nuts and torque to the value specified in Table 7-1. Install the impact limiter jam nuts and torque to the value specified in Table 7-1. Install the impact limiter lock wires. Repeat the operation for the bottom impact limiter installation.
27. Install security seals through holes provided in the upper impact limiter and one of the lifting trunnions; and through holes provided in all three bolts in the interlid port cover and the pressure port cover.
28. Install personnel barrier/enclosure and torque all attachment bolts to the prescribed torque value. Install padlocks on all personnel barrier/enclosure accesses.
29. Complete radiation and contamination surveys to ensure compliance with 49 CFR 173.441 and 173.443 requirements.
30. Determine the transport index (TI) corresponding to the maximum dose rate at 1 meter from the cask. Record on the shipping documents.
31. Determine the appropriate Criticality Safety Index (CSI) assigned to the package contents in accordance with the CoC, and indicate the correct CSI on the fissile material labels applied to the package.
32. Apply placards to the transport vehicle in accordance with 49 CFR 172.500.
33. Complete the shipping documentation in accordance with 49 CFR Subchapter C and provide special instructions to the carrier/shipper for an Exclusive Use Shipment.

THIS PAGE INTENTIONALLY LEFT BLANK

7.3 Outline of Procedures for Unloading the Cask

This section presents the procedures to be followed for unloading the cask following transport of directly loaded fuel, canistered fuel, canistered GTCC waste, or HLW canisters loaded in overpacks.

7.3.1 Receiving Inspection

1. Perform radiation and removable contamination surveys in accordance with 10 CFR 20.1906, 49 CFR 173.441 and 173.443 requirements.
2. Remove the personnel barrier/enclosure and complete radiation and removable contamination surveys at the cask surfaces.
3. Visually inspect the NAC-STC while secured to the transport vehicle in the horizontal orientation for any signs of damage and record any damage. Verify that the tamper-indicating seals are in place and verify their numbers.
4. Secure the transport vehicle. Attach slings to the top impact limiter lifting points, remove impact limiter lock wires, jam nuts, attachment nuts and retaining rods, and remove the impact limiter. Store the impact limiter upright. Repeat the operation to remove the bottom impact limiter. Complete radiation and removable contamination surveys for exposed cask surfaces.
5. Release the tiedown assembly from the front support by removing the front tiedown bolts and lock washers.
6. Attach a sling to the tiedown assembly lifting eyes and remove the tiedown assembly from the transport vehicle.
7. Attach the cask lifting yoke to a crane hook with the appropriate load rating. Engage the two yoke arms with the lifting trunnions at the top end of the cask. Rotate/lift the cask to the vertical orientation and raise the cask off of the rear support structure of the transport vehicle. Place the cask in the vertical orientation in a decontamination area or other location identified by the user.
8. Wash any road dust and dirt off of the cask and decontaminate cask exterior, as required by contamination survey results.

7.3.2 Preparation of the NAC-STC Cask for Unloading

The NAC-STC may contain fuel directly loaded into a basket within the cask, or a sealed transportable storage canister containing spent fuel assemblies, Reconfigured Fuel Assemblies,

Recaged Fuel Assemblies, four fuel assemblies or fuel debris loaded in Damaged Fuel Cans, canistered GTCC waste, or HLW canisters loaded in an HLW Overpack. Directly loaded fuel includes the shipment of uncanisterized high burnup (HBU) PWR fuel assemblies and associated shielded thermal shunts. The number and location of fuel assemblies and shunts depends on the HBU fuel configuration chosen.

Unloading of fuel from the directly loaded cask basket typically takes place under water in the spent fuel pool cask loading area. Canistered fuel and waste from unloading operations are performed dry using a transfer cask. Canister unloading will take place in the cask receiving area, or other location identified by the user.

7.3.2.1 Preparation for Unloading the NAC-STC Cask (Directly Loaded Fuel Configuration)

1. Verify that excessive pressure does not exist in the interlid region by removing the interlid port cover and attaching a pressure test fixture to the interlid port quick-disconnect that will allow the monitoring of the cask interlid region for any pressure buildup that may have occurred during transport. If a positive pressure exists, connect a vent/drain line to the interlid quick-disconnect and vent the pressure to the off-gas system.
2. Remove the outer lid bolts and install the outer lid alignment pins and outer lid lifting eye bolts.
3. Attach the outer lid lifting device to the outer lid lifting eye bolts and the overhead crane. Remove the outer lid and place it aside in a temporary storage area. Protect the O-ring and the O-ring groove of the lid from damage.
4. Access the NAC-STC cavity as follows:
   a. Remove the port cover plates from the drain and vent ports in the inner lid with caution.
   b. Attach a pressure gauge/gas sampling fixture with an evacuated stainless steel sample bottle with valve to the vent port.
   c. Measure and record cavity pressure for any pressure buildup that may have occurred during transport on the cask unloading report.
   d. Open gas sample bottle valve and obtain a cavity gas sample, isolate sample bottle, and disconnect for analysis of cavity gaseous radioactivity to determine if spent fuel cladding failures occurred during transport.
   e. Record the final cavity gaseous radioactivity levels on the cask unloading report.

f. If gaseous radioactivity levels in the cavity gas sample indicate that fuel rod cladding failure may have occurred during the transport operation, the Licensee, shall prepare and submit a written report to the USNRC within 60 days in accordance with 10 CFR 71.95 with a copy of the report provided to NAC as Certificate of Compliance Holder. The reports purpose would be to identify a potential non-compliance with the Certificate of Compliance authorized fuel contents conditions, as transport of damaged fuel and/or fuel having failed cladding are not authorized. The report shall include the details specified in 71.95(c) including an assessment of the safety consequences and implications of the event; and a description of any corrective actions planned or taken as a result of the event, including the means employed to repair any defects and actions taken to reduce the probability of similar events occurring in the future.

g. Connect a venting system to the pressure gauge/gas sampling fixture and discharge to the facility's off-gas system or to HEPA filter system to bring cavity gas pressure to atmospheric pressure, after determining total gaseous radioactivity of cavity gas and verifying that the release of the gaseous radioactivity through the facilities off-gas system will not violate license conditions.

h. Disconnect the pressure gauge/gas sampling fixture from the vent port.

5. Connect the cask cooldown system to the drain and vent quick-disconnects. The cask cooldown piping and controls schematic is shown in Figure 7.3-1.
6. To facilitate cooldown and to minimize thermal effects to the cask and its contents, <u>slowly</u> (8 – 10 gpm) fill the cask cavity with clean demineralized water (cavity is full when water flows out of the vent port drain line). Circulate water through the cask until the water leaving the vent port drain line is within 50°F of the average spent fuel pool water temperature.
7. Disconnect the fill line from the drain port quick-disconnect in the inner lid (Note: Leave a short drain line attached to the vent port quick-disconnect for continuous venting).
8. Loosen and remove all but 10, approximately equally spaced, inner lid bolts. Leave the 10 remaining inner lid bolts hand tight. Install the inner lid alignment pins at locations marked on the inner lid and the lid lifting eyebolts.
9. Remove the interlid port cover from the top forging. Disengage the vent line from the vent port quick-disconnect.
10. Attach the lifting yoke to a crane hook and engage the yoke arms with the lifting trunnions. Lift the cask and move it over to the cask loading area in the pool.

11. Spray the external surface of the cask with clean demineralized water to minimize external decontamination efforts. Slowly lower the cask into the pool. Just prior to submerging the top forging of the cask, complete the unthreading of the 10 remaining inner lid bolts and remove them.

    Note: Use caution when removing these bolts as pressure may rise slightly in the cask during the time since completion of Step 9.

12. Continue lowering the cask until it rests in the cask loading area on the pool floor.
13. Disconnect the lifting yoke from the lifting trunnions and move the yoke so that it will not interfere with fuel movements.
14. Using the inner lid lifting device attached to an auxiliary crane hook, remove the inner lid from the cask.

    Note: If the alternate method of handling the cask is being used, slowly raise the lift yoke and the inner lid using the lid alignment pins to guide movement. Move the lift yoke and the inner lid out of the area so that it will not interfere with fuel movements.

15. Place the inner lid aside ensuring that the O-rings and O-ring grooves are protected from damage. Decontaminate, as necessary, and clean all sealing surfaces.

7.3.2.2 Preparation for Unloading the NAC-STC Cask (Canistered Configuration)

1. Verify that excessive pressure does not exist in the interlid region by removing the interlid port cover and attaching a pressure test fixture to the interlid port quick-disconnect that will allow the monitoring of the cask interlid region for any pressure buildup that may have occurred during transport. If a positive pressure exists, connect a vent line to the interlid quick-disconnect and vent the pressure to the off-gas system.
2. Remove the outer lid bolts and install the outer lid alignment pins and outer lid lifting eye bolts.
3. Attach the outer lid lifting device to the outer lid lifting eye bolts and the overhead crane. Remove the outer lid and place it aside in a temporary storage area. Protect the O-ring and the O-ring groove of the lid from damage. Remove the outer lid alignment pins.
4. Remove the port coverplates from the drain and vent ports in the inner lid with caution. Attach a pressure test fixture to the vent port that will allow the monitoring of the cask cavity for any pressure buildup that may have occurred during transport. If a positive pressure exists, vent the pressure to the off-gas system.

5. Loosen and remove all inner lid bolts. Install the inner lid alignment pins at locations marked on the inner lid and the lid lifting hoist rings.
6. Using the inner lid lifting slings, attached to a suitable crane, remove the inner lid from the cask. Remove the inner lid alignment pins.
7. Place the inner lid aside ensuring that the O-rings and O-ring grooves are protected from damage. Decontaminate, as necessary, and clean all sealing surfaces.
8. If present, remove the top spacer from the NAC-STC cask cavity (Yankee-MPC canisters, MPC-LACBWR canisters, or MPC-WVDP-HLW overpacks only).
9. Install the adapter ring on the NAC-STC and torque the three captive bolts to the torque specified in Table 7-1.
10. Install the transfer cask adapter plate on the top surface of the cask and remove the handling slings.

7.3.3 Unloading the NAC-STC Cask

The NAC-STC may contain either fuel directly loaded in the cask basket, or a welded transportable storage canister. The procedures for unloading the directly loaded fuel or canisters are presented in the following.

7.3.3.1 Unloading Directly Loaded (Uncanistered) Fuel

1. Using approved fuel identification and handling procedures, withdraw one fuel assembly from the basket and deposit it in the proper storage rack location. Be careful not to contact any of the sealing surfaces on the top forging or the inner lid alignment pins.

   Note: If high levels of gaseous radioactivity were measured during initial cavity pressure measurements indicating the potential for failed fuel rod cladding, the unloaded fuel assemblies may require additional inspections of the fuel based on the corrective actions identified in Section 7.3.2.1, Step 4.f.
2. Record and document the fuel movement from the cask to the fuel rack.
3. Repeat steps 1 and 2 until all fuel assemblies have been removed from the cask. If HBU fuel assemblies were transported and a different HBU fuel configuration is intended for the next loading, remove any installed shielded thermal shunts that are not part of that configuration. Similarly, if HBU fuel assemblies were transported and standard fuel assemblies are intended for the next loading, remove all the installed shielded thermal shunts from their fuel basket locations.

4. Attach the inner lid lifting slings to a crane hook, lift the inner lid and place it on the cask using the alignment pins to assist in proper seating. Visually verify proper lid position.

   Note: O-ring seals on the lids, port coverplates and test plugs do not require replacement for an empty packaging shipment.

5. Disconnect the lid-lifting sling from the crane hook.
6. Attach the lifting yoke to the crane hook, lower to lifting position and engage lifting arms to lifting trunnions. Slowly lift the cask out of the pool until the top of the cask is slightly above the pool water level.

   Note: As an alternative method, the cask and inner lid may be handled simultaneously. In the event that this method is chosen, instead of performing steps 4, 5 and 6, attach the lifting yoke to a crane hook and the inner lid to the lift yoke. Lower the lid and engage to the cask using the lid alignment pins. Engage lifting arms to lifting trunnions. Slowly lift the cask out of the pool until the top of the cask is slightly above the pool water level.

7. Attach a drain line to the quick-disconnect in the interlid port (located in the top forging) and allow the water to drain from the interlid region.
8. Install at least four inner lid bolts approximately equally spaced on the bolt circle to hand tight. Remove the inner lid alignment pins.
9. Move the NAC-STC cask to the cask decontamination area and disengage the lift yoke or lift beam and inner lid lifting slings if the alternate method of handling the inner lid was used. Remove the inner lid lifting eye bolts.
10. Move the cask lifting equipment away from the cask work area.
11. Install the remaining inner lid bolts and torque all of the inner lid bolts to the value specified in Table 7-1 in accordance with the bolt torquing sequence shown on the inner lid.
12. Disconnect the drain line from the quick-disconnect in the interlid port.
13. Connect a drain line to the drain port quick-disconnect and a regulated air fill line to the vent port quick-disconnect.
14. Purge the water from the cask by pressurizing to 35 to 40 psig and hold until all water is removed (observed when no water is coming from the drain line). Adjust final internal cavity pressure to 0 psig.
15. Remove the lines from the drain and the vent port quick-disconnects.
16. Install the port coverplates over the vent and drain ports in the inner lid. Torque the coverplate bolts to the value specified in Table 7-1.

17. Decontaminate the surfaces of the inner lid and the inner surfaces of the top forging.
18. Install the outer lid alignment pins. Using the outer lid lifting device, install the outer lid using the alignment pins to assist in proper seating. Remove the lid lifting device, lid lifting eyebolts, and the outer lid alignment pins.
19. Install the outer lid bolts and torque them to the value specified in Table 7-1, using the bolt torquing sequence shown on the outer lid.
20. Install the interlid port cover and torque the bolts to the value specified in Table 7-1.

7.3.3.2 Unloading Canistered Fuel, Canistered GTCC Waste, or HLW Contained in HLW Overpacks

Canistered fuel and GTCC waste, and HLW contents loaded in an HLW Overpack are unloaded from the NAC-STC using a transfer cask. The transfer cask could be used to transfer the loaded canister to a work station where the canister could be opened, or to transfer it to another storage or disposal overpack.

1. Install the lift hoist rings in the canister lid.

   Note: The canister lid may be thermally hot.
2. Attach the canister lifting sling to the hoist rings in the canister lid. Position the sling so that the free end of the sling can be engaged by the cask handling crane hook.
3. Attach the transfer cask lifting yoke to the cask handling crane hook. Engage the yoke to the lifting trunnions of the transfer cask.
4. Lift the transfer cask and move it over the NAC-STC cask. Lower the transfer cask to engage the transfer cask adapter plate. Once the transfer cask is fully seated, remove the transfer cask lifting yoke and store it in the designated location.
5. Remove the shield door stops, connect the hydraulic operating system, and open the transfer cask bottom doors.
6. Using tag lines, lift the canister lifting slings through the transfer cask and attach them to the crane hook.

   Note: Alternative canister handling systems may be used.
7. Raise the canister into the transfer cask just far enough to allow the transfer cask bottom doors to close. Use caution to minimize the contact between the canister and the cavity walls of the NAC-STC and of the transfer cask.
8. Close the bottom doors and install the door stops.

9. Carefully lower the canister until it rests on the transfer cask bottom doors. Disengage the canister lifting sling from the crane hook.
10. Retrieve the transfer cask lifting yoke and attach it to the transfer cask trunnions. Lift the transfer cask from the NAC-STC cask and move it to its intended destination.
11. Attach the transfer cask adapter plate-lifting slings and disconnect the hydraulic operating system.
12. Using the crane, lift the transfer cask adapter plate from the top of the cask. Move the transfer cask adapter plate to the designated storage location.
13. Detorque the three bolts and remove the adapter ring.
14. At the option of the user, at this point the spacer(s) may be removed from the cask for decontamination and subsequent packaging for shipment.

Note: O-ring seals on the lids, port coverplates and test plugs do not require replacement for an empty packaging shipment.

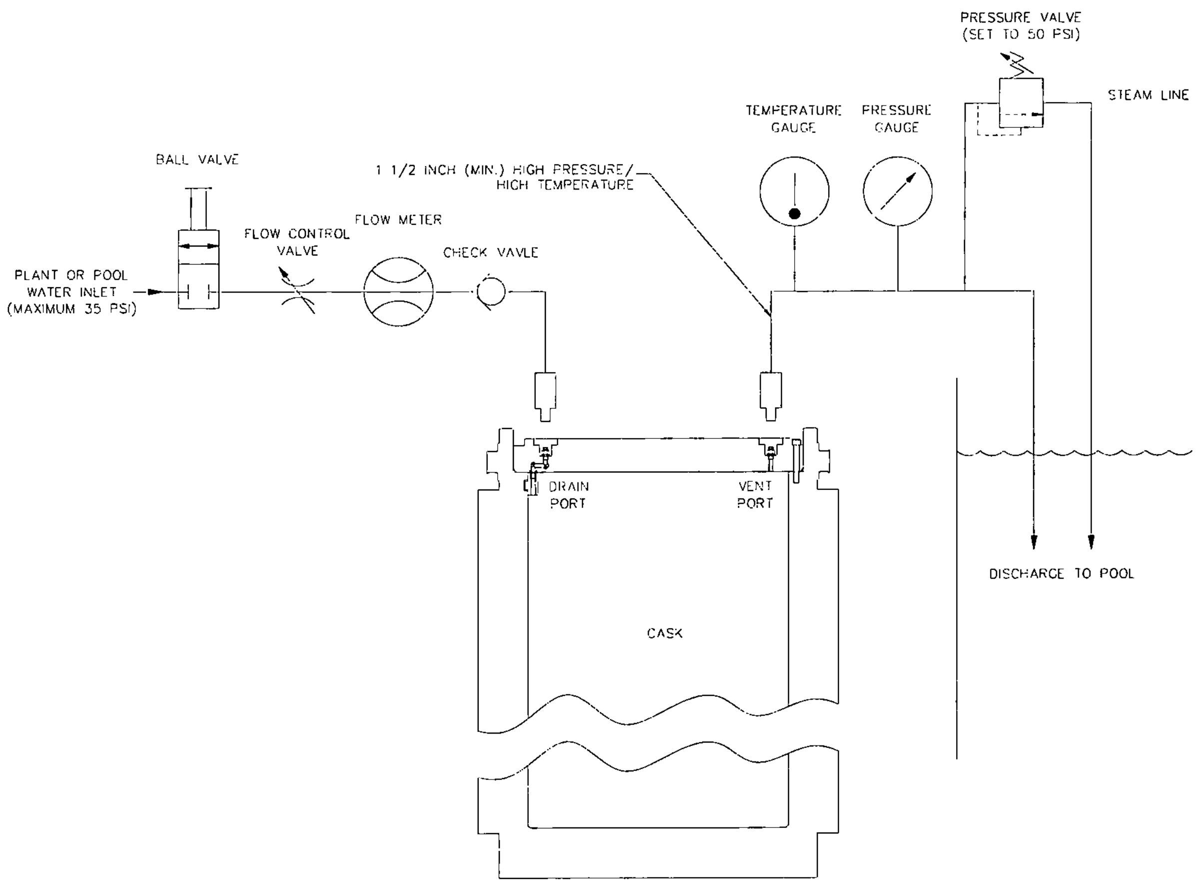
15. Install the inner lid alignment pins.
16. Attach the inner lid lifting fixture to the inner lid and engage the lifting fixture to the auxiliary crane. Install the inner lid in the NAC-STC using the alignment pins to assist in proper seating.
17. Disconnect the lifting fixture and remove the guide pins.
18. Install and torque the inner lid bolts to the values specified in Table 7-1 using the bolt torquing sequence shown on the inner lid.
19. Install the port coverplates over the vent and drain ports in the inner lid. Torque the coverplate bolts to the values specified in Table 7-1.
20. Decontaminate the surfaces of the inner lid and the inner surfaces of the top forging.
21. Install the outer lid alignment pins. Using the outer lid lifting device, install the outer lid using the alignment pins to assist in proper seating. Remove the lid lifting device, lid lifting eyebolts, and the outer lid alignment pins.
22. Install the outer lid bolts and torque them to the value specified in Table 7-1 using the bolt torquing sequence shown on the outer lid.
23. Install the interlid port cover and torque the bolts to the value specified in Table 7-1.

7.3.4 Preparation of Empty Cask for Transport

1. Decontaminate all surfaces of the cask to acceptable release limits as defined in 49 CFR 173.

2. Attach the lifting yoke to a crane hook and engage the yoke arms with the lifting trunnions. Lift the cask onto the transport vehicle and lower to the horizontal position.
3. Using a lifting sling, place the tiedown assembly over the cask upper forging between the top neutron shield plate and front trunnions. Install the front tiedown bolts and lock washers to each side of the front support. Torque each of the tiedown bolts.
4. Initiate Health Physics radiation and removable contamination surveys to ensure compliance with 49 CFR 173.441 and 49 CFR 173.443.
5. Using the designated lifting slings and a crane of appropriate capacity, install the top impact limiter. Install the impact limiter retaining rods into each hole and torque to the value specified in Table 7-1. Install the impact limiter attachment nuts and torque to the value specified in Table 7-1. Install the impact limiter jam nuts and torque to the value specified in Table 7-1. Install the impact limiter lock wires. Repeat the operation for the bottom impact limiter installation.
6. Apply labels to the package in accordance with 49 CFR 172.400.
7. Install the personnel barrier/enclosure and torque all attachment bolts to the prescribed torque value. Install padlocks on all personnel barrier/enclosure accesses.
8. Complete the Health Physics radiation and removable contamination surveys to ensure compliance with 49 CFR 173 requirements.
9. Complete the shipping documents.
10. Apply placards, if required, to the transport vehicle in accordance with 49 CFR 172.500.

Figure 7.3-1 Cask Cooldown Piping and Controls Schematic



7.4 Leak Test Requirements

This section provides the leak testing procedures used to perform the Containment System Verification Leak Tests for the NAC-STC containment boundary O-ring seals. These tests are required following cask loading operations for transport without interim storage for casks provided with metallic seals and after long-term storage in preparation for transport. For transport of uncanistered spent fuel without interim storage, casks provided with Viton O-rings are leak tested to confirm that the containment system is properly assembled for shipment. The preshipment leakage rate test confirms that there is no detected leakage from any seal to a minimum sensitivity of 1×10^{-3} ref·cm^3/sec. Detailed procedures, describing the equipment and the leak test system used to perform the leak tests, are developed for use at the licensee's facilities. Leakage test procedures shall be prepared by qualified personnel, and approved by personnel qualified in accordance with the requirements of SNT-TC-1A as a Level III NDE (Leak Testing). The containment boundary conditions, required leak tests and leak test acceptance criteria are provided in Table 4.1-1.

The transport cask may be configured with either metallic O-rings or with Viton O-rings. The two types of O-rings may not be used interchangeably, since each O-ring type requires a different O-ring groove configuration. Consequently, the inner lid, vent and drain port coverplates and outer lid are machined with a square O-ring groove to accept metallic O-rings or are machined with a truncated triangular (dove-tail) groove to accept Viton O-rings.

Viton O-rings may be used only when directly loading spent fuel, including HBU fuel assemblies, for transport without interim storage. Metallic O-rings must be used when directly loading spent fuel for an extended period of storage and for canistered spent fuel, GTCC waste and HLW contents. Metallic O-rings may be used when directly loading standard PWR spent fuel with burnups of $\leq$ 45 GWd/MTU for transport without interim storage. The metallic and nonmetallic O-rings have different allowable leak rates, as specified in the procedures.

7.4.1 Periodic and Maintenance Leakage Test Procedures

As described in Chapter 4, the NAC-STC primary containment boundary is designed and tested to assure that there is no leakage under any of the normal conditions of transport or accident conditions that exceeds the allowable value determined in accordance with 10 CFR 71.51. This leakage rate is verified prior to transport by the performance of leak tests on the containment

boundary sealed with metallic O-rings to ensure that the leakage rate is less than 2×10^{-7} cm^3/sec (helium). For NAC-STC intended to transport standard PWR low burnup fuel assemblies and sealed with Viton O-rings, the periodic leakage rate test is performed annually or after replacement of a Viton O-ring, and the cumulative leak rate is less than 9.3×10^{-5} cm^3/sec (helium). For NAC-STC casks intended to transport HBU PWR fuel assemblies and sealed with Viton O-rings, the periodic leakage rate test is performed annually or after replacement of a Viton O-ring, and the leak rate shall be $\leq 2.0 \times 10^{-7}$ cm^3/sec (helium) for each tested component. Helium leakage test procedures shall be prepared by qualified personnel, and approved by personnel qualified in accordance with the requirements of SNT-TC-1A as a Level III NDE (Leak Testing). Leak tests shall be performed by personnel qualified for helium leakage testing in accordance with the requirements of ANSI/ASNT CP-189-2006, "Standard for Qualification and Certification of Nondestructive Testing Personnel."

As described in Section 4.1, the containment boundary is defined differently for transport after long-term storage than for loading for transport without interim storage.

The leak test requirements and acceptance criteria performed after long-term storage in preparation for transport and performed following cask loading operations for transport without interim storage are described in Sections 7.4.2 and 7.4.3, respectively. The generic procedures used to perform leak testing are incorporated in the NAC-STC loading procedures in Section 7.2. Detailed helium procedures describing the equipment and the leak test system used to perform the leak tests are developed and prepared by qualified personnel and approved by personnel qualified in accordance with the requirements of SNT-TC-1A as a Level III NDE (Leak Testing). As noted in Section 7.1, the GTCC Waste canister or HLW overpack will have been loaded, closed and sealed prior to loading into the NAC-STC. The transportable storage canister for spent fuel is a separate inner container for the transport of damaged fuel. GTCC Waste canisters and HLW overpacks are not qualified as separate inner containers for confinement/containment purposes.

Section 7.4.4 provides the procedural guidance on corrective actions to be taken in the event a leak test does not meet the acceptance criteria.

7.4.2 Leak Testing for Transport After Long-Term Storage

This section summarizes the leak test methods used to demonstrate continued containment of PWR spent fuel prior to transport following an extended period of storage in the NAC-STC cask.

The containment boundary for this transport condition is defined as Containment Condition A in Section 4.1 and requires the use of metallic O-rings in the containment boundary. In addition to the steel inner lid and port coverplates, the containment boundary is specified as the outer O-rings of the inner lid and of the vent and drain port coverplates and the O-rings of the test port plugs. As specified in the generic loading procedure, the outer lid must be removed to test the inner lid and the vent and drain port coverplates prior to transport. Note that HBU spent fuel assemblies, canistered spent fuel, GTCC Waste canisters, or HLW Overpacks are not loaded into NAC-STC casks for long-term storage, and therefore, Containment Condition A does not apply for these canistered contents.

To conduct the leak test, the inner seal regions (annulus between the O-rings) of the inner lid and the vent and drain port coverplates are evacuated to less than one millibar, and backfilled to 0 psig with 99.9% pure helium, and the test port plugs are reinstalled. The outer lid is reinstalled using a new metallic O-ring. The interlid region (between the inner and outer lids) is evacuated to a vacuum of 1 millibar, or less. After the vacuum condition is reached, a helium leak detector is used to sample the interlid region for helium leakage past the inner lid outer O-ring, the vent and drain port coverplate outer O-rings, and O-ring test port plugs. The allowable leak rate is $\leq 2 \times 10^{-7}$ cm^3/sec (helium) with a minimum test sensitivity of $\leq 1 \times 10^{-7}$ cm^3/sec (helium). This test method conforms to A5.4 (evacuated envelope) of Appendix A of ANSI N14.5-1997. If helium leakage is detected exceeding the criteria, corrective action is taken as described in Section 7.4.4.

The outer lid and pressure port are tested using a pressure drop method to confirm the installation of the outer lid and pressure port O-rings. The interlid region is pressurized using the interlid port to 15 psig with air or helium, and the pressure is held for 10 minutes. No loss of pressure is permitted during the test period. Following the test, the interlid region pressure is reduced to 0 psig. The interlid port cover is installed and the annulus between the O-rings of the port cover is tested using the same method. This test confirms the installation of the interlid port cover O-rings and conforms to test method A.5.1 (gas pressure drop) of Appendix A of ANSI N14.5-1997.

7.4.3 Leak Testing for Transport After Loading without Interim Storage

This section summarizes the leak tests required to demonstrate containment of directly loaded standard PWR and HBU spent fuel without interim storage, or for sealed transportable storage

canisters containing spent fuel or GTCC waste, or HLW Overpacks containing separate sealed canisters of glassified high level waste. The containment boundary for these transport conditions is defined as Containment Condition B in Section 4.1. In addition to the steel inner lid and port coverplates, the containment boundary is specified as the inner O-rings of the inner lid and of the vent and drain port coverplates. The inner lid O-ring and vent and drain port coverplate O-rings are leak tested using the evacuated envelope method (test description A5.4 of Appendix A of ANSI N14.5-1997) with a vacuum in the annulus between the O-rings.

The containment boundary O-rings for standard PWR spent fuel assemblies with burnups ≤ 45 GWd/MTU and directly loaded for transport without interim storage may be either metallic or Viton. The containment boundary seals for HBU PWR spent fuel assemblies with burnups > 45 GWd/MTU and directly loaded for transport are required to be Viton O-rings. The containment boundary O-rings for canistered spent fuel assemblies or GTCC waste, or HLW Overpacks are required to be metallic O-rings. The leak detector is used to detect helium in the annulus between the O-rings. The allowable leakage rate for each metallic O-ring for NAC-STC casks configured with metallic seals defined as the containment boundary is $\leq 2 \times 10^{-7}$ cm^3/sec (helium) with a minimum test sensitivity of $\leq 1 \times 10^{-7}$ cm^3/sec (helium). For NAC-STC casks intended to transport standard PWR spent fuel assemblies and provided with Viton O-rings, the allowable cumulative leakage rate for all components defined as the containment boundary is $\leq 9.3 \times 10^{-5}$ cm^3/sec (helium) with a minimum test sensitivity of $\leq 4.7 \times 10^{-5}$ cm^3/sec (helium). The allowable leakage rate for NAC-STC casks intended to transport HBU spent fuel assemblies, which require a leak tight containment boundary and are provided with Viton O-rings, shall be $\leq 2 \times 10^{-7}$ cm^3/sec (helium) with a minimum test sensitivity of $\leq 1 \times 10^{-7}$ cm^3/sec (helium).

The preshipment leakage test prior to transport of a NAC-STC cask utilizing reusable Viton O-rings that have not been replaced is performed to a minimum test sensitivity of 1×10^{-3} ref·cm^3/sec. The higher sensitivity maintenance leakage rate test for the Viton O-rings (e.g., $\leq 9.3 \times 10^{-5}$ cm^3/sec (helium) for standard directly loaded PWR fuel assemblies, and $\leq 2 \times 10^{-7}$ cm^3/sec (helium) for directly loaded HBU spent fuel assemblies) shall be performed during annual maintenance testing after replacement of the Viton O-rings or when the Viton O-rings or other containment components are replaced between annual maintenance periods during operations.

For NAC-STC casks provided with metallic O-rings as the containment boundary, the metallic seals are replaced for each loaded transport, and the maintenance leakage rate test is performed

to an acceptance criteria of $\leq 2 \times 10^{-7}$ cm^3/sec (helium) with a minimum test sensitivity of $\leq 1 \times 10^{-7}$ cm^3/sec (helium). The series of leakage tests described confirms that the allowable leak rates are satisfied for the types of O-rings used in the containment boundary for Containment Condition B. Section 7.4.4 provides the procedural guidance on corrective actions to be taken in the event a leak test does not meet the acceptance criteria.

Following completion of NAC-STC leakage testing of the containment boundaries, the outer lid and pressure port are tested using a pressure drop method to confirm the installation of the outer lid and pressure port O-rings. The interlid region is pressurized using the interlid port to 15 psig with air and the pressure is held for a minimum of 10 minutes. No loss of pressure is permitted during the test period. Following the test, the interlid region pressure is reduced to 0 psig. The interlid port cover is installed and the annulus between the O-rings of the port cover is tested using the same method. This test confirms the installation of the interlid port cover O-rings. These components provide an additional barrier against the release of radioactive material, but are not part of the containment boundary.

7.4.4 Corrective Action

If a specific containment component containing an O-ring fails to meet the leak test acceptance criteria for that component, the component is removed and the O-ring is removed from the groove. The O-ring groove is cleaned and visually inspected to ensure proper cleanliness and surface condition. A new O-ring of the appropriate material, size and identification (i.e., metallic or Viton) is installed. The removed component is reinstalled and the bolts torqued to the appropriate torque value. The component is then retested in accordance with the applicable original test procedure and acceptance criteria.

The replacement of the inner lid O-ring(s), either immediately after loading or after extended storage for the directly loaded spent fuel configurations, will require the NAC-STC cask to be returned to the spent fuel pool for inner lid removal and inner lid O-ring replacement. The cask unloading procedures (Section 7.3.3) will be utilized to prepare the cask for placement in the spent fuel pool, including cask cool down operations, to allow inner lid removal for seal replacement. At cask storage facilities having appropriate dry transfer or hot cell facilities, the inner lid O-ring can be replaced without placement of the cask in a fuel pool for shielding purposes. Prior to removal of the inner lid, a gas sample should be taken at the vent port to verify the condition in the cavity environment. If there are indications that fuel has failed during the

storage period, care should be exercised in both flooding the cask and in removing the inner lid (refer to procedures in Section 7.3.2.1, Steps 4.f. and .g.).

For NAC-STC casks being loaded with canister spent fuel, GTCC Waste canisters or HLW Overpacks, the containment boundary seals are metallic seals and the inner lid metallic seals may be replaced without returning the cask to the pool as the canister or overpack confines and shields the spent fuel, GTCC waste, or HLW.

6. Cut the weld joining the vent port cover to the shield lid (MPC-Yankee and CY-MPC canisters) or the inner vent port cover to closure lid weld (MPC-LACBWR).
7. Remove the vent port cover.
8. Sample the canister cavity cover gas and vent any pressure in the canister to a radioactive waste handling system.
9. Cut the weld joining the drain port cover to the shield lid (Yankee-MPC and CY-MPC canisters) or the inner drain port cover to the closure lid (MPC-LACBWR).
10. Attach a nitrogen gas line to the drain port quick-disconnect and a discharge line from the vent port quick-disconnect to an off-gas handling system.
11. Continue to flow nitrogen through the line until there is no evidence of fission gas activity in the discharge line (or 10 minutes minimum).
12. Attach a source of clean water with a minimum temperature of 70°F and a maximum supply pressure of 15 psig to the drain port quick-disconnect. Replace the vent port quick-disconnect with a straight-through fitting fitted with a Viton O-ring, and attach the discharge water line. Slowly start the flow of clean water to establish a flow rate of 5 (+ 3, - 0) gpm.
13. Continue to flow water through the canister until the exit water temperature stabilizes to a temperature below 200°F.
14. Stop the flow of water and remove the cool down system from the vent and drain ports.
15. Set up the weld cutting equipment to cut the shield lid weld (Yankee-MPC and CY-MPC canisters) or the closure ring and closure lid welds (MPC-LACBWR).
16. Remove approximately 50 gallons of water from the Yankee-MPC or the MPC-LACBWR canister, or approximately 65 gallons of water from the CY-MPC canister.
17. Using a hydrogen gas detector, check the vent port for hydrogen gas. Purge the hydrogen gas if the concentration of hydrogen gas exceeds 2.4%.
18. Cut the shield lid weld (Yankee-MPC and CY-MPC) or the closure ring and closure lid welds (MPC-LACBWR). Remove the closure ring (MPC-LACBWR). Attach the shield lid or closure lid (MPC-LACBWR) lifting slings.
19. Attach the clean water line to the transfer cask.
20. Retrieve the transfer cask lifting yoke and engage the transfer cask lifting trunnions.
21. Move the transfer cask over the pool and lower the bottom of the transfer cask to the surface. Start the flow of clean water to the transfer cask annulus. Continue to lower the transfer cask, as the annulus fills with clean water, until the top of the transfer cask is about 4 inches above the pool surface. Hold the cask in this position until clean water fills the top of the transfer cask.

22. Lower the transfer cask to the bottom of the cask loading area and remove the lifting yoke.
23. Attach the shield lid (Yankee-MPC and CY-MPC) or closure lid (MPC-LACBWR) lifting slings to the crane hook.
24. Slowly lift the shield lid (Yankee-MPC and CY-MPC) or closure lid (MPC-LACBWR) and remove the lid from the canister.
25. Visually inspect the spent fuel or GTCC waste.

At this point, the spent fuel or GTCC waste can be removed from the canister.

7.6.3 Loading and Closing the HLW Overpack

This procedure assumes that the HLW Overpack and the MPC-WVDP Vertical Storage Cask (VSC) are positioned in a suitable work station with a suitable crane available.

1. Visually inspect the HLW Overpack and VSC to ensure that each is clean and free of debris.
2. Rig, lift and place the empty HLW Overpack into the VSC.
3. Using suitable site transport equipment lift and move the VSC into the dry transfer and loading facility.
4. Load the previously designated HLW canisters into the Overpack.

   Note: Each HLW Overpack has five (5) canister cell locations. Empty cells not loaded with a HLW canister will have a transport insert installed.
5. Install the closure lid into the top of the loaded overpack.
6. Weld the closure lid in place and verify the adequacy of welds with direct and/or remote visual weld examinations. Record results of examinations as required.
7. The HLW overpack is now prepared for transportation.
8. Install the VSC lid and move the loaded VSC to the facility for the transfer of the HLW Overpack into the NAC-STC in accordance with the procedures presented in Section 7.1, contingent upon the loaded contents meeting the requirements for the authorized content conditions, as specified in the NAC-STC Certificate of Compliance.

   Note: The HLW Overpack may be stored at the loading facility in a VSC prior to off-site transport under a separate U.S. DOE authorization.

Table of Contents

List of Figures

List of Tables

will be applied in a vertical direction and equally distributed between the two rotation trunnion recesses by the use of hydraulic rams combined with a load-spreading beam.

Following completion of the rotation trunnion recesses load test, all accessible trunnion recess welds and load bearing surfaces shall be visually inspected for permanent deformation, galling or cracking. Inspections utilizing liquid penetrant examination shall be performed in accordance with the ASME Code, Section V, Article 6. Liquid penetrant acceptance standards shall be as indicated in paragraph NF-5350 of the ASME Code, Section III, Division 1.

Any evidence of permanent deformation, cracking, galling of the load bearing surfaces or unacceptable dye penetrant results shall be cause for rejection of the rotation trunnion recesses or related welds.

8.1.2.3 Hydrostatic Testing

A hydrostatic test shall be performed on the NAC-STC cask containment boundary, prior to final acceptance of the cask, in accordance with the ASME Code, Section III, Division I, Article NB-6200. The hydrostatic test pressure shall be at least 76 psig, which is 150 percent of the Maximum Normal Operating Pressure. This test shall be performed in accordance with approved written procedures. All pressure retaining components, appurtenances, and completed systems shall be pressure tested.

The vent port will be used for the test connection. Only the vent port quick-disconnect will be installed during the testing. The hydrostatic test will be performed with the inner lid and the drain port coverplate installed and torqued.

The hydrostatic test system components, although not part of the cask containment boundary, will be visually inspected prior to the start of the hydrostatic test. Leakage from the valves or connections will be corrected prior to the start of the hydrostatic test.

The test pressure gauge installed on the cask will have an upper limit of approximately twice that of the test pressure. The hydrostatic test pressure shall be maintained for a minimum of 30 minutes, during which time a visual inspection is made to detect any evidence of a leak. Any evidence of a leak during the minimum hold period will be cause for rejection.

After completion of the hydrostatic test, the cask containment boundary will be dried and prepared for visual and/or dye penetrant inspections as appropriate. The components of the cask containment boundary shall be visually inspected. All accessible welds within the cavity shall be liquid penetrant inspected. Any evidence of cracking or permanent deformation is cause for rejection of the affected component.

8.1.2.4 Pneumatic Bubble Testing of the Neutron Shield Tank

A pneumatic bubble test of the neutron shield tank will be performed in accordance with Section V, Article 10, Appendix I, of the ASME Code following final closure welding of the bottom closure plates. The pneumatic test pressure shall be 12.5 + 1.5/-0 psig, which is 125 percent of the relief valve set pressure. The test shall be performed in accordance with approved written procedures.

During the test, the two relief valves on the neutron shield tank will be removed. One of the relief valves threaded connections will be used for connection of the air pressure line and test pressure gauge. The other relief valve connection will be plugged with a threaded plug.

Following introduction of pressurized air into the neutron shield, a 15-minute minimum soak time will be required. Following completion of the soak time, approved soap bubble solution will be applied to all fin to shell, shell to end plate, and end plate to outer shell welds. The acceptance criteria for the bubble test will be no air leak from any tested weld as indicated by continuous bubbling of the solution. If an air leak is indicated, the weld shall be repaired in accordance with approved weld repair procedures and the pneumatic bubble test shall be repeated until no unacceptable air leak is observed.

8.1.3 Leakage Tests

Fabrication leakage rate testing is performed on both the NAC-STC transport cask containment boundary weldment (with the inner lid and inner lid vent and drain port coverplates installed) during fabrication prior to lead pouring, and on the cask containment closures (with the inner lid and inner lid vent and drain port coverplates installed) upon completion of cask body fabrication (i.e., following lead pouring and final cask assembly) to demonstrate that the containment boundary, as fabricated, will provide an appropriate containment capability.

The leakage rate testing of the NAC-STC containment boundary and closures will be performed in accordance with the requirements and standards contained in ANSI N14.5-1997 and the ASME Code, Section V, Article 10. Helium leakage test procedures shall be developed and

approved by personnel qualified in accordance with the requirements of SNT-TC-1A as a Level III NDE (Leak Testing) examiner. The leakage tests shall confirm that the leakage rate meets the containment criteria established in Chapter 4 (e.g., leaktight for the NAC-STC containment boundary weldment; leaktight for Containment Condition A containment boundary closures with metallic seals for spent fuel transport following long-term storage; leaktight for Containment Condition B closures with metallic seals for directly loaded PWR spent fuel assemblies with burnups of $\leq$ 45 GWd/MTU, and canistered spent fuel and GTCC waste, and HLW overpacks; leaktight for Containment Condition B closures with Viton O-ring seals for HBU spent fuel assemblies; or a cumulative leakage rate $\leq 9.3 \times 10^{-5}$ cm^3/sec (helium) for containment boundary components with Viton O-ring seals for standard, directly loaded PWR spent fuel assemblies). Leak tests shall be performed by personnel qualified for helium leakage testing in accordance with the requirements of ANSI/ASNT CP-189-2006, "Standard for Qualification and Certification of Nondestructive Testing Personnel."

8.1.3.1 Containment Boundary Weldment Fabrication Leakage Test

Following the satisfactory completion of hydrostatic pressure testing of the NAC-STC containment boundary weldment per Section 8.1.2.3, the containment vessel cavity is drained and cleaned. Per Paragraph 7.3 of ANSI N14.5-1997, a helium fabrication leakage rate test of the containment boundary weldment will be performed in accordance with the requirements of Section V, Article 10 of the ASME Code. The containment boundary weldment shall be leakage tested to demonstrate a leak rate of less than, or equal to, 2×10^{-7} cm^3/sec (helium) with a minimum test sensitivity of 1×10^{-7} cm^3/sec (helium) to verify the containment boundary weldment, including containment welds and base materials, is leaktight as defined in ANSI N14.5-1997.

If a leak exceeding the leakage rate acceptance criteria is detected, the affected weld or area of base metal shall be rejected. Rejected welds or areas of base metal shall be repaired in accordance with the requirements of Article NB-4450 of the ASME Code. The repaired weld or base metal area shall be reexamined using the same procedure and acceptance criteria as specified for the original weld examination. The helium fabrication leakage rate test of the containment boundary weldment shall then be re-performed in accordance with the original test requirements and acceptance criteria prior to final acceptance.

8.1.3.2 Final Fabrication Leakage Rate Testing

Upon completion of cask body fabrication, a helium fabrication leakage rate test is performed on the removable containment boundary closure components and their respective metallic seals or

Viton O-ring seals (i.e., cask inner lid, lid bolts, inner lid vent and drain port coverplates and bolts) in accordance with Paragraph 7.3 of ANSI N14.5-1997.

Final containment fabrication leakage rate testing is performed with the cask assembled in accordance with the cask assembly drawing, except that the vent or drain quick-disconnect is not installed (Note: The test is repeated to ensure that both the vent and drain port coverplates are individually leakage tested to the applicable criteria). This ensures that when the cask cavity is backfilled with helium, helium is present on the containment side of the vent or drain port coverplate containment O-ring. Leakage rate tests are performed on the cask lid and the lid port coverplate and their respective O-ring seals. The test is performed using a helium mass spectrometer leak detection system by establishing a vacuum of ≤ 0.1 torr in the seal or O-ring annulus of the cask lid and, separately, of the vent and drain port coverplate using the applicable test port. The cask containment boundary is backfilled with a known concentration of high purity helium gas. The acceptance criteria for the containment fabrication leakage rate testing of the removable closures are provided in Section 8.1.3.2.1 and 8.1.3.2.2, depending on the seal material and type (e.g., metallic seals or Viton O-rings). A leakage rate that exceeds the allowable leakage rate is cause for rejection of the component and seal being tested. Seal replacement or other corrective actions shall be taken to repair any detected leaks. The component and replaced seal shall then be retested and re-inspected in accordance with the original test requirements and acceptance criteria prior to final acceptance. After successful completion of the leakage tests, quick-disconnects are installed in the inner lid vent and drain port openings and torqued.

8.1.3.2.1 Metallic Seal Testing Acceptance Criteria

The fabrication leakage rate testing of the containment boundary closures using metallic seals consists of a series of leak tests. The acceptance criteria for each metallic seal is a detected leakage rate of $\leq 2 \times 10^{-7}$ cm^3/sec (helium) at a test system sensitivity of $< 1 \times 10^{-7}$ cm^3/sec (helium) or better.

8.1.3.2.2 Viton O-Ring Testing Acceptance Criteria

The fabrication leakage rate testing of the containment boundary closures using Viton O-rings consists of a series of leak tests. The acceptance criteria for the Viton O-ring seals is a cumulative (sum of the three individual leakage tests results) detected leakage rate of $\leq 9.3 \times 10^{-5}$ cm^3/sec (helium) at a test system sensitivity of $< 4.7 \times 10^{-5}$ cm^3/sec (helium) or better for standard directly loaded PWR spent fuel assemblies, or leak tight leakage rate of $\leq$ 2.0×10^{-7}

cm^3/sec (helium) at a test system sensitivity of < 1.0×10^{-7} cm^3/sec (helium) or better for directly loaded HBU PWR spent fuel assemblies.

8.1.4 Component Tests

Tests performed on individual components are designed to ensure that the component meets the design requirements for correct and proper operation of the cask system.

Acceptance criteria are established based on the functions and design requirements of the component being tested.

8.1.4.1 Valves

There are no valves that are part of the NAC-STC containment boundary for transport. Quick-disconnects are installed in the vent, drain and interseal test port openings in the inner lid to provide access to the cavity, and in the interlid port to provide access to the interlid region. These fittings serve as valves when the mating parts are connected, and are used to connect ancillary equipment to the cask cavity for filling, draining, drying, backfilling, gas sampling, and leak testing operations. Upon removal of the external fitting, the valve in the quick-disconnect closes automatically. The design and selection of the quick-disconnects is based on similar equipment and procedures used with other NRC-approved storage and transport casks. For transport, the quick-disconnects are sealed inside the transport containment boundary using a bolted coverplate fitted with two O-ring seals.

There are no rupture disks on the NAC-STC.

Two self-actuating pressure relief valves are installed on the external shell of the neutron shield to provide for venting of vapor from the shielding material during transport thermal accident conditions. These valves have stainless steel bodies and an operating pressure range of zero to 200 psig with an adjustable cracking pressure within this range. The cracking pressure is set at 10 psig. These relief valves do not provide a safety function, but have been designed to minimize recovery efforts in the unlikely event of a neutron shield overpressure condition.

8.1.4.2 Gaskets

As described in Section 8.1.3, the containment boundary of the NAC-STC may use either metallic O-rings or non-metallic Viton O-rings. The two O-ring types require different O-ring

groove designs and, therefore, may not be used interchangeably and must be used with the inner lid, vent and drain port coverplates and outer lid having the appropriate O-ring groove machined in the component. Metallic O-rings must be used for direct loading of the NAC-STC with fuel for extended storage and for loading of a transportable storage canister (for transport). For direct loading of fuel for immediate transport, either metallic or non-metallic O-rings may be used.

The outer lid, inner lid, drain port coverplate, vent port coverplate, interlid port cover, pressure port cover, and interseal test plug gaskets are O-rings. For transport after an extended period of storage, the containment boundary is formed by the outer metallic O-ring of the inner lid, the outer metallic O-rings on the vent and drain port coverplates, and the interseal test plug metallic O-rings for the inner lid, the vent port coverplate and the drain port coverplate. The inner metallic O-rings of the inner lid, vent port coverplate and drain port coverplate, the metallic O-ring of the outer lid, and the PTFE O-rings of the interlid and pressure port covers provide a secondary closure to the cask contents. For immediate transport, the containment boundary is formed by the inner O-rings of the inner lid and vent and drain port coverplates. A second boundary is formed by the O-rings of the outer lid and interseal and pressure port covers.

The O-ring replacement schedule depends upon the O-ring material. The metallic O-ring(s) of any component shall be replaced prior to reinstallation of the component. Viton O-rings are inspected prior to each use and replaced as necessary. The PTFE O-rings of the interlid and pressure ports will be visually inspected prior to each use, and replaced if necessary. The PTFE O-rings shall be replaced at least once every two years during cask transport operations, or prior to transport if they have been installed longer than two years (i.e., after extended storage).

The containment boundary O-ring shall be tested and maintained in accordance with the Maintenance Program Schedule of Table 8.2-1 and the leak test criteria of Section 8.2.2.

8.1.4.3 Miscellaneous

The removable transport impact limiters consist of redwood and balsa wood. License drawings and the supporting analyses specify the crush strengths of the redwood and balsa wood to be 6240 psi ± 620 psi and 1550 psi ± 150 psi respectively. For manufacturing purposes, verification of the impact limiter material is accomplished by verifying the densities of the wood. Three samples from each redwood board are to be tested for density, and the average density of the samples shall be 23.5 ± 3.5 pounds/cubic foot. Each 15-degree and 30-degree pie shaped section of the impact limiter shall have a density of 22.3 ± 1.2 pounds/cubic foot in accordance with the License Drawings. The moisture content for any single redwood board must be greater than 5

percent, but less than 15 percent. The average moisture content for a lot of redwood used in impact limiter construction must not be greater than 12 percent.

Following final closure welding of the transport impact limiter stainless steel shell, a leak test of the shell welds shall be performed to verify weld integrity. The test shall be performed by evacuating the impact limiter to 75 mbar and performing a 30-minute test to determine if there is any increase in the impact limiter pressure. Any detected leak shall not exceed 1 x 10^{-2} cm^3/sec. If a leak exceeding this value is detected, the cause of the leak shall be determined, and the weld repaired and retested.

8.1.5 Tests for Shielding Integrity

8.1.5.1 Gamma Shield Test

The gamma scan test shall be conducted by continuous scanning or probing over 100 percent of all accessible cask surfaces using a 3-inch detector and a ^{60}Co source. The source strength shall be of an intensity sufficient to produce a count rate that equals or exceeds three times the background count rate on the external surfaces of the cask. The count rate shall be maintained for greater than one minute prior to the start of scanning. The detector scan path spacing (cask exterior surface) will be a maximum of 2.5 inches and the scanning speed will be 4.5 feet per minute or less. The source scan path spacing (cask interior surface) will be on a 2-inch grid pattern (when using a 3-inch detector). Flat surfaces, such as the cask bottom and closure lids, will use a 2.5 inch spacing for both the detector and source scan paths (when using a 3-inch detector).

The acceptance criteria for the shield test will be that the shield effectiveness of the cask body and lids shall be equal to or greater than the shield effectiveness of a lead and steel mock-up. The steel thickness of the mockup shall be equivalent to the minimum steel thickness specified on the License Drawings and the lead thickness shall be equivalent to the minimum lead thickness specified in the License Drawings less 3 percent. The shielding mock-up will be produced using the same fabrication techniques as those approved for the cask.

Measured count rates that exceed those established by the test mock-up shall cause the component to be rejected. The rejected areas/components shall be evaluated to determine the corrective action to be taken. Any repaired areas shall be retested prior to acceptance.

An additional gamma shield effectiveness test shall be performed on each cask following first fuel loading. The neutron and gamma shield effectiveness test procedures and acceptance criteria are described in Section 8.1.5.4.

8.1.5.2 Neutron Shielding Test

The neutron shielding of the NAC-STC is provided by a solid layer of NS-4-FR, which is a hard polymer material. A 5.5-inch layer of NS-4-FR is located in the annulus formed by the outer shell and the 0.236-inch (6 mm) thick neutron shield shell. The neutron shield is divided in sections by the copper/stainless steel fins. A 2-inch thick layer of NS-4-FR is also installed in the cask inner lid and in the cask bottom.

The installation of NS-4-FR material in the fabrication of the cask is a special process and, as such, procedures will be prepared and qualified to ensure that the mix ratios, mixing method, degassing, pouring, and curing of the material is properly performed. The NS-4-FR raw material is provided in the form of a 3-part mixing kit. The material content of the raw material is tested and certified at the time of kit preparation. The neutron shielding material is installed into the annulus between the outer shell and the neutron shield shell by pouring it with the cask in an inverted vertical position. Prior to installation, samples from each mix of the actual material being poured into the annulus are wet density tested to ensure that the material is properly mixed. Mixes that do not meet the wet density acceptance criteria are rejected. Procedures used for installation of the material are validated prior to use by destructive examination of a full scale mock-up of the neutron shield cavity. Qualification of the installation procedure verifies material homogeneous properties and minimizes the potential deleterious voids.

8.1.5.3 Neutron Shielding Material Testing

The neutron shield properties of NS-4-FR are provided in Chapters 1 and 3. Each lot (mixed batch) of neutron shield material shall be tested to verify that the material composition (aluminum and hydrogen), boron concentration, and neutron shield density meet the requirements specified in Chapters 1 and 3 and the License Drawings. Testing shall be performed by qualified laboratories in accordance with written and approved procedures. Material composition, boron concentration, and density data for each lot of neutron shield material shall become part of the quality record documentation package.

Dimensional inspection of the cavities containing the neutron shielding material shall ensure that the required thickness specified in the License Drawings is incorporated into the cask.

The installation of the neutron shielding material shall be performed in accordance with written, approved, and qualified procedures. The procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps or unacceptable voids from occurring in the material. Procedures shall be qualified by the use of mock-ups to ensure that the NS-4-FR installation does not result in the creation of unacceptable voids. Samples of each lot of neutron shield material shall be maintained as part of the quality record documentation package.

8.1.5.4 Neutron and Gamma Shield Effectiveness Tests

Following first fuel loading, a neutron and gamma shield effectiveness test shall be performed for each cask prior to transport. The test shall be performed with the cask loaded with fuel, drained, vacuum dried and backfilled with helium. The purpose of the test is to document the effectiveness of the neutron and gamma shielding materials. The test shall be performed in accordance with detailed, approved written test procedures.

Calibrated neutron and gamma dose rate meters shall be used to measure the neutron and gamma dose rate at contact with the outer shell of the neutron shield and at 2.3 meters from the surface (equivalent to 2 meters from the sides of the railcar). Dose measurement points shall be established on the external surface of the shell at 30° intervals and at five points along the height of the shield (a total of 60 measuring points). In addition, neutron and gamma dose rate measurements shall be made of the trunnion areas above the neutron shield, at four points below the neutron shield, and at the edges and center of the cask top (outer lid) and cask bottom surfaces. Dose rates at the top and bottom of the cask shall be measured with the transport impact limiters installed. The dose rates measured at contact and at 2.3 meters shall be recorded on the test data sheet, along with the total power of the loaded fuel assemblies; date, time and location of test; identification and calibration of instrumentation; and identification of test engineer and operators.

To allow an evaluation of the measured dose rates to be completed, the burnup and cool time for the actual fuel assemblies loaded into the cask will be determined and recorded. From this fuel history data, the total actual neutron and gamma source terms will be estimated using ORIGEN or similar calculations.

If the measured dose rates exceed the applicable regulatory limits, the licensee shall notify the NRC. Appropriate corrective measures will be taken, including fuel unloading and correction of the shielding deficiency. Following corrective actions, the test will be re-performed to the original acceptance criteria prior to final acceptance.

8.1.6 Thermal Test

Prior to acceptance at the factory, a thermal test shall be performed on each fabricated packaging to confirm and verify that the fabricated and assembled cask possesses the heat rejection capabilities predicted by the thermal analyses. The thermal test shall be performed in accordance with approved written procedures.

8.1.6.1 Thermal Test Set-up

The thermal test set-up is shown in Figure 8.1-1(a). As depicted, the thermal test shall be performed with the cask positioned horizontally on a test frame. The transport impact limiter or equivalent insulating material shall be installed on each end of the cask to simulate the transport configuration. The cask will be located in a covered building in a still environment. The cask shall be assembled with the basket installed. A thermal test lid with connections for thermocouple leads and electric heater power cables shall be installed in place of the inner lid. The outer lid will not be installed for the test. The thermal test lid will be provided with an O-ring seal capable of containing the containment cavity helium atmosphere.

Electric heaters shall be installed in each fuel tube. The electric heaters will have an active length of between 120 and 150 inches and be capable of generating a minimum of 22 kilowatts (kw). The heaters will be supported in the basket so as to not be in contact with the wall of the fuel tube. The power supplied to the heater will be recorded throughout the test duration.

Calibrated test thermocouples, with an accuracy of ±2°F, will be installed on the cask basket, inner shell, and outer neutron shield shell surfaces. The location of the test thermocouples are shown in Figure 8.1-1. The specific location of the thermocouples are as follows:

- TC1 - basket top steel weldment
- TC2 - steel disk at cask basket midpoint
- TC3 - aluminum disk at cask basket midpoint
- TC4 - basket bottom steel weldment

TC5; TC6; TC7; and TC8 - located at 90° intervals on the inner shell surface at cavity midpoint

TC9 - top of inner shell surface at 30-40 inches from top of cavity

TC10 - bottom of inner shell surface at 30 to 40 inches from base of the cavity

TC11; TC12; TC13; and TC14 - located at 90° intervals on the neutron shield shell surface (at fin tip) at cask midpoint

TC15 - top of neutron shield shell surface (at fin tip) at 30-40 inches from top of neutron shell

TC16 - bottom of neutron shield shell surface (at fin tip) at 30-40 inches from bottom of neutron shield shell

TC17 - top of upper forging

TC18 - outer shell surface at centerline of cask bottom face

TC19 - inner fuel tube wall surface near the center of the cask basket

TC20 - ambient temperature of testing area

The output of the test thermocouples will be recorded throughout the test by a strip chart recorder.

8.1.6.2 Test Procedure

With the cask assembled and instrumented as described above, the cask cavity is evacuated and backfilled to 1.0 atmosphere absolute (14.6 psia) with helium. Power will be applied to the heaters to simulate the cask contents. After initiation of power to the heaters, the temperatures of all thermocouples and heater power levels will be monitored and recorded on data sheets at 60 minute intervals. Power will be maintained to the electrical heaters until the cask has reached thermal equilibrium.

For the purpose of the test, thermal equilibrium is defined as being achieved when over two consecutive hours:

$\Delta t_{TC13} \leq 2°F/hr$, and

$\Delta t_{TC3} \leq 2°F/hr$

Based upon the thermal heat-up evaluation, thermal equilibrium should be achieved in approximately five days.

After verification of thermal equilibrium, final temperature measurements will be recorded for all test thermocouples. The final power readings for the electric heaters will also be recorded. The strip chart will be marked to indicate the time of the final cask measurements. The printout of the strip chart recorder and the completed test data sheets will be incorporated into an approved final thermal test report. The test will be determined to be acceptable if the acceptance criteria of Section 8.1.6.3 are met.

If the acceptance criteria are not met, the cask will not be accepted until appropriate corrective actions are completed. Upon completion of corrective actions, the cask shall be retested to the original test requirements and acceptance criteria.

8.1.6.3 Acceptance Criteria

The purpose of the thermal test is to confirm the heat rejection capabilities of the as-built cask are acceptable and correspond to the temperatures calculated by thermal analyses for the directly loaded (uncanistered) configuration presented in Chapter 3.0 of this application.

Package heat dissipation acceptance testing assures: 1) maximum material temperatures do not exceed material allowables; and that 2) measured temperature gradients are less than the thermal gradients calculated in the package thermal analyses.

The thermal acceptance test is accepted when the following criteria are met:

1) When corrected for physical test boundary conditions and heat load, the following measured temperatures are not exceeded:

TC No.	Location	Temperature °F
TC1	Top Basket Steel Weldment	435
TC3	Aluminum Disk Center	485
TC2	Steel Support Disk Center	495
TC4	Basket Bottom Steel Weldment	475
TC5-TC8	Cask Inner Shell	330
TC11-TC14	Neutron Shield Shell	240
TC17	Cask Top Forging	200
TC18	Cask Bottom	330
TC19	Tube Wall	540

2) The measured temperature gradient across the central steel disk from TC2 to the average of TC5, TC6; TC7 and TC8 is less than 200□F;
3) The measured temperature gradient across the central aluminum disk from TC3 to the average of TC5; TC6; TC7 and TC8 is less than 190□F; and
4) The measured temperature gradient across the cask body as measured by thermocouple pairs TC5-TC13; TC6-TC14; TC7-TC11; and TC8-TC12 are less than 90°F.

8.1.7 Neutron Absorber Tests for NAC-STC Directly-Loaded Fuel Basket and for Yankee-MPC and CY-MPC Canistered Fuel Baskets

Two alternate neutron poison materials, BORAL and TalBor, have been qualified by NAC for use in the NAC-STC directly loaded, the Yankee-MPC and the CY-MPC fuel baskets. For the NAC-STC directly loaded basket, a generic neutron absorber test and qualification program suitable for BORAL, metal matrix composite (aluminum based) and borated aluminum is listed in Section 8.1.11. This program is designed to demonstrate structural, thermal, and nuclear requirements are met without specification of a particular manufacturer or material.

BORAL is manufactured by Ceradyne Corporation, Chicoutimi (Quebec), Canada under a Quality Assurance/Quality Control program in conformance with the requirements of 10 CFR 50, Appendix B. The manufacturing process consists of several steps: the first step is the mixing of the aluminum and boron-carbide powders that form the core of the finished material, with the amount of each powder a function of the desired ^{10}B areal density. The methods used to control the weight and blend of the powders are patented and proprietary processes of AAR Advanced Structures (AAR) (subsequently Ceradyne). The mixture of powders is placed in an aluminum box with walls approximately one inch thick. The top lid is welded in place. This "ingot" is heated for several hours and then is hot-rolled to produce the sheet of design thickness. The rolling process densifies and bonds the powder mixture. The aluminum box walls become the cladding for the $Al\text{-}B_4C$ core.

TalBor is manufactured by Talon Composites, Inc. (TalBor was formerly called Boralyn, and was produced by Alyn Corporation. Alyn Corporation went out of business and Talon Composites acquired the major production equipment and the patent rights for Boralyn. TalBor is essentially identical to Boralyn.) TalBor is manufactured and controlled using a Quality Assurance program that is compliant with the applicable requirements of 10 CFR 50, Appendix, B. TalBor is a metal matrix composite (MMC). The aluminum and B_4C powders are mixed to

the specified ^{10}B areal density and the powder mixture is vacuum sintered and hot pressed to achieve a fully dense billet. The billet is extruded, then cut and rolled to the design thickness.

After manufacturing, test samples from each batch of neutron absorber (poison) sheets shall be tested to verify the presence, proper distribution, and minimum weight percent of ^{10}B. Neutron transmission testing or augmented wet chemistry testing may be used. The tests shall be performed in accordance with approved written procedures.

8.1.7.1 Neutron Absorber Material Sampling Plan

The neutron absorber sampling plan is selected to demonstrate a 95/95 (95% probability and 95% confidence level) statistical confidence level in the neutron absorber sheet material compliance with the specification. In addition to the specified sampling plan, each sheet of material is visually and dimensionally inspected using at least 6 measurements (along the edges near each corner and the longitudinal centerline) on each sheet. No rejected neutron absorber sheet is used. The sampling plan is supported by written and approved procedures.

The sampling plan requires that a coupon sample be taken from each sheet of the first set of 100 sheets of absorber material. Thereafter, coupon samples are taken from 20 randomly selected sheets from each set of 100 sheets. This 1 in 5 sampling plan continues until there is a change in lot or batch of constituent materials of the sheet (i.e., boron carbide powder, aluminum powder, or aluminum extrusion), or a process change, at which time the sampling process is reinitiated as previously described. The sheet samples are indelibly marked and recorded for identification. This identification is used to document neutron absorber test results, which become part of the quality record documentation package.

8.1.7.2 Wet Chemistry Test Performance

An approved facility with chemical analysis capability shall be selected to perform the wet chemistry tests. The tests will ensure the presence of boron and enable the calculation of the ^{10}B areal density. Acceptability of the uniformity of boron distribution is based on the manufacturer's material qualification tests.

The most common method of verifying the acceptability of neutron absorber material is the wet chemistry method—a chemical analysis where the aluminum is separated from a sample with known thickness and volume. The remaining boron-carbide material is weighed and the areal density of ^{10}B is computed. A statistical conclusion about the BORAL or TalBor sheet from

which the sample was taken and that batch of sheets may then be drawn based on the test results and the established manufacturing processes previously noted.

8.1.7.3 Neutron Absorption Transmission Test Performance

An approved facility with a neutron source and neutron detection capability shall be selected to perform the described tests, if the neutron absorption transmission test method is used. The tests will assure that the neutron absorption capacity of the material tested is equal to, or higher than, the given reference value and will verify the uniformity of boron distribution. The principle of measurement of neutron absorption is that the presence of boron results in a reduction of neutron flux between the thermalized neutron source and the neutron detector—depending on the material thickness and boron content.

Typical test equipment will consist of thermal neutron source equipment, a neutron detector and a counting instrument. The test equipment is calibrated using a known standard, whose ^{10}B content has been checked and verified by an independent method such as chemical analysis. This calibration process shall be repeated daily (every 24 hours) while tests are being performed.

8.1.7.4 Acceptance Criteria

The neutron transmission test results shall be considered acceptable if the minimum ^{10}B areal density is determined to be equal to, or greater than, that specified on the fuel tube drawings. Any specimen not meeting the acceptance criteria shall be rejected and all of the sheets from that batch shall be similarly rejected unless coupons from each individual absorber plate are tested and confirmed to meet or exceed the specified areal density.

8.1.8 Neutron Absorber Tests for MPC-LACBWR Canistered Fuel Basket

Neutron absorber material (commercially available as BORAL®), in the form of sheets consisting of boron-carbide evenly dispersed within a matrix of aluminum and clad with aluminum, is used in the NAC-MPC transportable storage canister fuel baskets. The manufacturing process consists of several steps – the first being the mixing of the aluminum and boron-carbide powders that form the core of the finished material, with the amount of each powder a function of the desired ^{10}B areal density. The methods used to control the weight and to blend the powders were patented and proprietary processes of AAR and, subsequently, of Ceradyne Corporation of Chicoutimi (Quebec), Canada.

After manufacturing, test samples from each batch of BORAL® neutron absorber (poison) sheets shall be tested using wet chemistry or neutron absorption techniques to verify the presence, proper distribution, and minimum weight percent of ^{10}B. The tests shall be performed in accordance with approved written procedures.

8.1.8.1 Neutron Absorber Material Sampling Plan

The neutron absorber sampling plan is selected to demonstrate a 95/95 (95% probability and 95% confidence level) statistical confidence level in the neutron absorber sheet material compliance with the specification. In addition to the specified sampling plan, each sheet of material is visually and dimensionally inspected using at least six measurements (along the edges near each corner and the longitudinal centerline) on each sheet. No rejected neutron absorber sheet is used. The sampling plan is supported by written and approved procedures.
The sampling plan requires that a coupon sample be taken from each sheet of the first set of 50 sheets of absorber material. Thereafter, coupon samples are taken from 10 randomly selected sheets from each set of 50 sheets. This 1 in 5 sampling plan continues until there is a change in lot or batch of constituent materials of the sheet (i.e., boron carbide powder, aluminum powder, or aluminum extrusion), or a process change, at which time the sampling process is reinitiated as previously described. The sheet samples are indelibly marked and recorded for identification. This identification is used to document neutron absorber test results, which become part of the quality record documentation package.

8.1.8.2 Wet Chemistry Test Performance

An approved facility with chemical analysis capability shall be selected to perform the wet chemistry tests. The tests will ensure the presence of boron and enable the calculation of the ^{10}B areal density. Acceptability of the uniformity of boron distribution is based on the manufacturer's material qualification tests.

The most common method of verifying the acceptability of neutron absorber material is the wet chemistry method - a chemical analysis where the aluminum is separated from a sample with known thickness and volume. The remaining boron-carbide material is weighed and the areal density of ^{10}B is computed. A statistical conclusion about the BORAL® sheet from which the sample was taken and that batch of BORAL® sheets may then be drawn based on the test results and the established manufacturing processes previously noted.

8.1.8.3 Neutron Absorption Test Performance

An approved facility with a neutron source and neutron detection capability shall be selected to perform the described tests, if the neutron absorption test method is used. The tests will assure that the neutron absorption capacity of the material tested is equal to, or higher than, the given reference value and will verify the uniformity of boron distribution. The principle of measurement of neutron absorption is that the presence of boron results in a reduction of neutron flux between the thermalized neutron source and the neutron detector—depending on the material thickness and boron content.

Typical test equipment will consist of thermal neutron source equipment, a neutron detector and a counting instrument. The test equipment is calibrated using standards whose ^{10}B content has been checked and verified by an independent method such as chemical analysis. The highest permissible counting rate is determined from the neutron counting rates of the reference sheet(s), which should be ground to the minimum allowable plate thickness. This calibration process shall be repeated daily (every 24 hours) while tests are being performed.

8.1.8.4 Acceptance Criteria

The wet chemistry test results shall be considered acceptable if the ^{10}B areal density is determined to be equal to, or greater than, that specified on the fuel tube drawings. The neutron absorption test shall be considered acceptable if the neutron count determined for each test specimen is less than or equal to the highest permissible neutron count rate determined from the BORAL standard, which is based on the ^{10}B areal density specified on the fuel tube drawings. Any specimen not meeting the acceptance criteria for either test method shall be considered to be nonconforming material and shall be evaluated within the NAC International QA Program. Nonconforming material shall be assigned one of the following dispositions: "use-as-is," "rework" or "reject." Only material that is determined to meet all applicable conditions of the license will be accepted.

8.1.9 Transportable Storage Canister

The transportable storage canister is constructed of Type 304L (Yankee-MPC and CY-MPC) or 304/304L (MPC-LACBWR) stainless steel and is fabricated by welding. If circumferential welds are required to join two shell sections, the seam welds shall not be aligned within 45° circumferentially. The welded cylinder is closed at the bottom by a circular plate welded to the

shell wall. The top of the cylinder is closed by two field-installed circular plates, welded to the canister shell wall following fuel loading.

The transportable storage canister is a welded closed component. The canister serves as the confinement boundary component of the NAC-MPC System during storage of spent fuel in the vertical concrete cask.

The finished surfaces of all canister welds are visually examined in accordance with ASME Code Section V, Article 9, to verify that the components are assembled in accordance with the License Drawings and that the components are free of nicks, gouges, and other damage. The acceptance criteria for the visually examined welds for the Yankee-MPC and the CY-MPC canisters is in accordance with ASME Code Section VIII, Division 1, UW-35 and UW-36 and Section III, Subsection NB, NB-4424 and NB-4427. The acceptance criteria for the visually examined welds of the MPC-LACBWR canister are in accordance with ASME Code, Section III, Subsection NF, NF-5360.

The seam and girth welds in the transportable storage canister shell are full-penetration welds that are radiographic (RT) examined in accordance with ASME Code Section V, Article 2. The acceptance criteria for the RT-examined welds is that specified in ASME Code Section III, Subsection NB, Article NB-5320. The canister shell to bottom plate weld is a full-penetration double-bevel weld with an inside fillet weld that is ultrasonic examined in accordance with ASME Code Section V, Article 5, with acceptance criteria as specified in ASME Code Section III, Subsection NB, Article NB-5330. The final surfaces of the seam and girth welds in the canister and the canister shell to bottom plate weld are also liquid penetrant examined in accordance with ASME Code Section V, Article 6, with the acceptance criteria being that specified in ASME Code Section III, Subsection NB, Article NB-5350.

Field installed partial-penetration groove welds attach the shield and structural lids (Yankee-MPC and CY-MPC) or the closure lid (MPC-LACBWR) to the canister shell, and the vent and the drain port covers to the shield lid (Yankee-MPC and CY-MPC) or the inner and outer vent and drain port covers to the closure lid (MPC-LACBWR) after the canister is loaded. The closure ring for the MPC-LACBWR canister is welded to both the canister shell and closure lid by partial penetration welds. For the Yankee-MPC and CY-MPC canister, the root and final surfaces of the shield lid weld are liquid penetrant examined. For the MPC-LACBWR canister, the closure lid to canister shell weld is progressively liquid penetrant examined at the root, mid-plane and final surfaces. The structural lid to shell weld for the Yankee-MPC and CY-MPC

canisters is progressively liquid penetrant examined at the root, every 3/8-inch weld layer and final surface. Canister vent and drain port cover welds are liquid penetrant examined at the root and final surfaces unless the welds are completed in a single pass. Welds completed in a single pass require only final surface liquid penetrant examination.

All liquid penetrant examinations are completed in accordance with ASME Code, Section V, Article 6. Acceptance criteria for all liquid penetrant examinations are as specified in ASME Code Section III, Division 1, Subsection NB, Article NB-5350.

The Yankee-MPC and CY-MPC canister shield lid welds are helium leakage tested in accordance with ASME Code Section V, Article 10, Appendix V, using a minimum leak rate test sensitivity of 1×10^{-7} cm^3/sec (helium). The MPC-LACBWR canister closure lid to canister shell weld is hydrostatically tested following completion of the weld.

The fabricator of the transportable storage canister will establish a written weld inspection plan in accordance with an approved quality assurance program. The weld inspection plan will include visual, liquid penetrant, ultrasonic, and radiographic examination. In addition, the weld inspection plan will identify the welds to be examined, the sequence of the examinations, the type of examination method to be used, and the criteria for acceptance of the weld in accordance with the applicable sections of the ASME Code.

8.1.10 HLW Overpack and Basket

The HLW Overpack is constructed of Type 304/304L stainless steel and is fabricated by welding. If circumferential welds are required to join two shell sections, the seam welds shall not be aligned within 45° circumferentially. The welded cylinder is closed at the bottom by a circular plate welded to the shell wall. The top of the cylinder is closed by a field-installed circular plate, welded to the canister shell wall following HLW canister loading.

The HLW Overpack is a welded closed component.

The finished surfaces of all HLW Overpack welds are visually examined in accordance with ASME Code Section V, Articles 1 and 9, to verify that the components are assembled in accordance with the License Drawings and that the components are free of nicks, gouges, and other damage. The acceptance criteria for the visually examined welds for the HLW overpack are in accordance with ASME Code Section VIII, Division 2, Section 7.5.2.2.

The seam, girth, and shell to bottom plate welds in the HLW overpack shell are full-penetration welds that are dye penetrant (PT) examined in accordance with ASME Code Section V, Articles 1 and 6. The acceptance criteria for the PT examined welds are those specified in ASME Code Section VIII, Division 2, Section 7.5.7.2.

Field installed partial-penetration groove welds attach the closure lid to the HLW Overpack shell after HLW canister loading. The closure lid to canister shell weld is visually examined at the final surface. Visual examinations are completed in accordance with ASME Code, Section V, Articles 1 and 9. Acceptance criteria for all visual examinations are as specified in ASME Code Section VIII, Division 2, Section 7.5.2.2.

The HLW Overpack basket is fabricated from Type 304 stainless steel and is fabricated by welding. If circumferential welds are required to join two shell sections, the seam welds shall not be aligned within 45° circumferentially. The five (5) welded HLW cylinder cells are closed at the bottom by a plate welded to the cell wall where accessible. The top of the HLW cylinder cell is open to allow vertical dry loading of a HLW canister.

The finished surfaces of all HLW overpack basket assembly welds are visually examined in accordance with ASME Code Section V, Articles 1 and 9 to verify that the components are assembled in accordance with the License Drawings and that the components are free of nicks, gouges, and other damage. The acceptance criteria for the visually examined welds for the HLW overpack basket are in accordance with ASME Code Section III, Subsection NF, NF-5360. Liquid penetrant (PT) examination will be performed on all HLW Overpack basket welds in accordance with ASME Code, Section V, Articles 1 and 6. Acceptance criteria for the PT examined welds shall be in accordance with ASME Code, Section III, Subsection NF, NF-5350.

The fabricator of the HLW Overpack and basket assemblies will establish a written weld inspection plan in accordance with an approved quality assurance program. The weld inspection plan will include visual and liquid penetrant examination. In addition, the weld inspection plan will identify the welds to be examined, the sequence of the examinations, the type of examination method to be used, and the criteria for acceptance of the weld in accordance with the applicable sections of the ASME Code.

Testing by Neutron Attenuation

Acceptance testing shall be performed to ensure that neutron absorber material properties for sheets in a given production run are in compliance with the materials requirements for the NAC-STC Directly Loaded fuel baskets and that the process is operating in a satisfactory manner.

Statistical tests will be run to augment findings relating to isotopic content, impurity content or uniformity of the ^{10}B distribution.

- Determination of neutron absorber material acceptance shall be performed by neutron attenuation testing. Neutron attenuation testing of the final product, or the coupons, shall compare the results with those for calibrated standards, which may be composed of homogeneous or heterogeneous materials. The heterogeneous standard will be calibrated to a recognized standard (e.g., homogeneous material such as ZrB_2 plate material or a NIST-produced standard) or by attenuation of a thermal neutron beam correlated to the known cross-section of ^{10}B at the beam energies. These tests shall include a statistical sample of finished product or test coupons taken from each lot of material to verify the presence, uniform distribution and the minimum areal density of ^{10}B.
- The ^{10}B areal density is measured using a collimated thermal neutron beam of up to 2.54 cm in diameter, with a tolerance of 10 percent.
- Based on the required minimum effective areal density of ^{10}B – 0.015 g/cm^2 and the credit taken for the ^{10}B for the criticality analyses, i.e., 90% for borated aluminum alloys and for borated metal matrix composites, a required minimum areal density for the as-manufactured neutron absorber sheets is established.
- Test locations/coupons shall be well distributed throughout the lot of material, particularly in the areas most likely to contain variances in thickness, and shall not contain unacceptable defects that could inhibit accurate physical and test measurements.
- The sampling plan shall require that each of the first 50 sheets of neutron absorber material from a lot, or a coupon taken therefrom, be tested. Thereafter, coupons shall be taken from 10 randomly selected sheets from each set of 50 sheets. All coupons (100%) taken shall be tested by neutron attenuation. This 1 in 5 sampling plan shall continue until there is a change in lot or batch of constituent materials of the sheet (i.e., boron carbide powder or aluminum powder) or a process change. A measured value less than the required minimum areal density of ^{10}B during the reduced inspection (neutron attenuation testing) is defined as nonconforming, along with other contiguous sheets, and mandates a return to 100% inspection (neutron attenuation testing) for the next 50 sheets. The coupons are indelibly marked and recorded for identification. This identification

will be used to document the neutron absorber material test results, which become part of the quality record documentation package.

- The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level (also expressed as 95/95 level) or better. The following illustrates one acceptable method:
  - The acceptance criterion for individual plates is determined from a statistical analysis of the test results for that lot. The minimum ^{10}B areal densities determined by neutron attenuation are converted to volume density (g/cm^3), i.e., the neutron attenuation measurement divided by the maximum thickness of the coupon. The lower tolerance limit of ^{10}B volume density is then determined - defined as the mean value of ^{10}B volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor for a normal distribution with 95% probability and 95% confidence.
  - Finally, the minimum specified value of ^{10}B areal density is divided by the lower tolerance limit of ^{10}B volume density to arrive at the minimum plate thickness that provides the specified ^{10}B areal density.
  - Any plate that is thinner than this minimum or the minimum design thickness, whichever is greater, shall be treated as nonconforming, with the following exception. Local depressions are acceptable, as long as they total no more than 0.5% of the area on any given plate and the thickness at their location is not less than 90% of the minimum design thickness.
- All neutron absorber material acceptance verification will be conducted in accordance with the NAC International Quality Assurance Program. The neutron absorber material supplier shall control manufacturing in accordance with the key process controls via a documented quality assurance system (approved by NAC or NAC's approved fabricator), and the designer shall verify conformance by reviewing the manufacturing records.
- Nonconforming material shall be evaluated within the NAC International Quality Assurance Program and shall be assigned one of the following dispositions: "Use-As-Is," "Rework/Repair" or "Reject." Only material that is determined to meet all applicable conditions of the license will be accepted.

8.1.11.9 Boral Neutron Absorber Tests

The Boral neutron absorbing material is an aluminum matrix material formed from aluminum and boron-carbide. The mixing of the aluminum and boron-carbide powder forming the neutron

absorber material is controlled to assure the required ^{10}B areal density. The constituents of the neutron absorber material shall be verified by chemical testing and by dimensional measurement to ensure the quality of the finished plate or sheet. The results of all neutron absorber material tests and inspections, including the results of wet chemistry coupon testing, are documented and become part of the quality records documentation package for the fuel tube and basket assembly.

The manufacturing process of Boral consists of several steps. The initial step is the mixing of the aluminum and boron carbide powders that form the core of the finished material. The amount of each powder is a function of the desired ^{10}B areal density. The methods used to control the weight and blend the powders are proprietary processes of the manufacturer.

After manufacturing, test samples from each Boral batch of neutron absorber sheets shall be tested using wet chemistry techniques to verify the presence and minimum weight percent of ^{10}B. The tests shall be performed in accordance with approved written procedures.

The Boral neutron absorber sampling plan is selected to demonstrate a 95/95 statistical confidence level in the neutron absorber sheet material in compliance with the specification. In addition to the specified sampling plan, each sheet of material is visually and dimensionally inspected using at least six measurements on each sheet. The sampling plan is supported by written and approved procedures.

The sampling plan requires that a coupon sample be taken from each of the first 100 sheets of Boral neutron absorber material and tested. Thereafter, coupon samples are taken and sampled from 20 randomly selected sheets from each set of 100 sheets. This 1 in 5 sampling plan continues until there is a change in lot or batch of constituent materials of the sheet (i.e., boron carbide powder, aluminum powder, or aluminum extrusion) or a process change. If either of these circumstances occurs, the sampling plan reverts back to a coupon sample being taken from each of the first 100 sheets of absorber material, followed by the 20 randomly selected sheets from each set of 100 sheets. A measured value less than the required minimum areal density of ^{10}B during the reduced inspection is defined as nonconforming, along with other contiguous sheets, and mandates a return to 100% inspection for the next 100 sheets. The sheet samples are indelibly marked and recorded for identification. This identification is used to document neutron absorber test results, which become part of the quality record documentation package.

Wet Chemistry Testing

Wet chemistry testing of the test coupons obtained from the sampling plan is used to verify the ^{10}B content of the neutron absorber material. Wet chemistry testing is applied because it provides an accurate and practical direct measurement of the boron and B_4C content of metal materials.

An approved facility with chemical analysis capability, which could include the neutron absorber vendor's facility, shall be selected to perform the wet chemistry tests. Personnel performing the testing shall be trained and qualified in the process and in the test procedure.

Wet chemistry testing is performed by dissolving the aluminum in the matrix, including the powder and cladding, in a strong acid, leaving the B_4C material. A comparison of the amount of B_4C material remaining to the amount required to meet the ^{10}B content specification is made using a mass-balance calculation based on sample size.

A statistical conclusion about the neutron absorber sheet from which the sample was taken and that batch of neutron absorber sheets may then be drawn based on the test results and the controlled manufacturing processes.

The adequacy of the wet chemistry method is based on its use to qualify the standards employed in neutron blackness testing. The neutron absorption performance of a test material is validated based on its performance compared to a standard. The material properties of the standard are demonstrated by wet chemistry testing. Consequently, the specified test regimen provides adequate assurance that the neutron absorber sheet thus qualified is acceptable.

The wet chemistry test results shall be considered acceptable if the ^{10}B areal density is determined to be equal to, or greater than, 0.02 g/cm^2 (0.015 g/cm^2 adjusted for 75% efficiency). Failure of any coupon wet chemistry test shall result in 100% sampling, as described in the sampling plan, until compliance with the acceptance criteria is demonstrated.

Yield Strength Testing

Yield strength qualification testing of the neutron absorber shall conform to ASTM Test Method B 557/B 557M, E8 or E21. For Boral, a laminated absorber, yield strength credited in the structural analysis was limited to the outer aluminum cover sheets. Therefore, only the cover sheet must be shown to meet the required strength.

"Use-As-Is," "Rework/Repair" or "Reject." Only material that is determined to meet all applicable conditions of the license will be accepted.

8.1.11.11 Additional Material Specifications

Boron carbide particles for MMCs shall have an average size in the range of 10-40 microns and no more than 10% of the particles shall be over 60 microns. The material shall have negligible interconnected porosity exposed at the surface or edges.

Open porosity for borated aluminum and borated MMC neutron absorber material must be no greater than 0.5% unless qualification tests are performed to ensure that blisters are not produced under submerging and subsequent vacuum drying conditions.

Chemical composition of the boron carbide powder must meet the requirements of Table 1 of ASTM C 750-03, Type 3. Additional chemical requirements, applicable to a particular absorber material, may be placed on the boron carbide powder as a result of the "key manufacturing process controls" invoked by Section 8.1.11.10. Additional requirements may include, but are not limited to, upper limits on fluorine and chlorine content.

Table 8.1-1 Neutron Absorber Material Minimum ^{10}B Loading (NAC-STC Directly Loaded Basket)

Neutron Absorber Type	Required Minimum <u>Effective</u> Areal Density (^{10}B g/cm^2)	% Credit Used in Criticality Analyses	Required Minimum <u>Actual</u> Areal Density (^{10}B g/cm^2)
Borated Aluminum Alloy	0.015	90	0.0167
Borated MMC	0.015	90	0.0167
Boral/TALBOR	0.015	75	0.02

Table 8.1-2 Mechanical Properties of Neutron Absorber (NAC-STC Directly Loaded Basket)

Property (units)	Values at Temperature (°F)
	70
Ultimate Tensile Strength, S_u (ksi) [a]	13.1
Yield Strength, S_y (ksi) [a]	5.0

[a] Equal to aluminum alloy 1100-O properties

8.2 Maintenance Program

To ensure that the NAC-STC packaging is in compliance with the requirements of the regulations, the Certificate of Compliance, and this application, a cask Maintenance Program for the NAC-STC shall be established. The cask Maintenance Program shall specify the inspections, tests, and replacement of components to be performed, and the frequency and schedule for these activities. This chapter describes the overall requirements of the Maintenance Program and establishes the frequency and schedule for the maintenance activities. The detailed, written inspection, test, component replacement, and repair procedures shall be included in the NAC-STC Operations Manual. The NAC-STC Operations Manual will be issued to Users of the packaging and will be prepared and issued prior to first use of the cask in each configuration.

There are no maintenance requirements for the welded canister containing either fuel or GTCC waste.

8.2.1 Structural and Pressure Tests of the Cask

The four lifting trunnions and the two rotation trunnion recesses shall be visually inspected prior to each shipment. The visual inspections shall be performed in accordance with approved written procedures, and inspection results shall be evaluated against established acceptance criteria.

Evidence of cracking on the load bearing surfaces shall be cause for rejection of the affected trunnion until an approved repair has been completed, and the surfaces re-inspected and accepted. Such repairs shall be implemented and documented in accordance with an approved QA program.

The lifting trunnions are also inspected annually in accordance with Paragraph 6.3.1(b) of ANSI N14.6. All accessible trunnion welds and accessible welds that are part of the load path are visually inspected for permanent deformation, galling, or cracking. Liquid penetrant examinations of welds and load bearing surfaces are performed in accordance with the ASME Code, Section V, Article 6 [3]. Liquid penetrant acceptance standards are those of Paragraph NF-5350 of the ASME Code, Section III, Division 1.

During periods of nonuse of the transport cask, the inspection of the trunnions may be omitted provided that the trunnions are inspected in accordance with this section prior to the next use.

8.2.2 Leak Tests

Leak tests are performed in accordance with the methodologies and requirements of ANSI N14.5-1997, using helium leakage test procedures developed and approved by personnel qualified in accordance with the requirements of SNT-TC-1A as a Level III NDE (Leak Testing) examiner. Leak tests shall be performed by personnel qualified for helium leakage testing in accordance with the requirements of ANSI/ASNT CP-189-2006, "Standard for Qualification and Certification of Nondestructive Testing Personnel."

8.2.2.1 Fabrication Leakage Rate Test

The fabrication leakage rate test is performed on each NAC-STC cask at the fabricator's facility in accordance with Section 8.1.3.

8.2.2.2 Periodic and Maintenance Leakage Rate Tests

The periodic or maintenance leakage rate test shall be performed on each cask after the third use (prior to fourth cask loading sequence), every twelve months thereafter, and whenever a replaceable containment component is installed to verify the containment capability. Metallic seals used for the containment boundary seals shall be replaced during each cask loading operation and the seals leak tested in accordance with the maintenance leakage rate test requirements. Viton O-rings shall be inspected prior to each use and replaced as necessary. Viton O-ring performance shall be demonstrated by preshipment leakage rate testing prior to each shipment.

The periodic and maintenance leakage rate tests shall be performed using helium leakage test procedures developed and approved by personnel qualified in accordance with the requirements of SNT-TC-1A as a Level III NDE (Leak Testing) examiner and in accordance with the test requirements and acceptance criteria established in Section 8.1.3 for the fabrication leakage rate tests.

During periods when the cask is not in use for transport, the periodic leakage rate test need not be performed on an annual basis, but shall be re-performed prior to returning the cask to service and use as a transport package.

8.2.2.3 Acceptance Criteria

8.2.2.3.1 Metallic Seal Testing Acceptance Criteria

The periodic or maintenance leakage testing of the containment boundary closures using metallic seals consists of a series of leak tests. The acceptance criteria for each metallic seal is a detected leakage rate of $\leq 2 \times 10^{-7}$ cm^3/sec (helium) at a test system sensitivity of $< 1 \times 10^{-7}$ cm^3/sec (helium) or better.

Unacceptable leakage test results shall be cause for rejection of the component tested. Corrective actions, including repair or replacement of the seals and/or closure component, shall be taken and documented as appropriate. The leakage test shall be repeated and accepted prior to returning the cask to service.

8.2.2.3.2 Viton O-Ring Testing Acceptance Criteria

The periodic or maintenance leakage testing of the containment boundary closures using Viton O-rings consists of a series of leak tests. The acceptance criteria for the Viton O-ring seals is a cumulative (sum of the three individual leakage tests results) detected leakage rate of $\leq 9.3 \times 10^{-5}$ cm^3/sec (helium) at a test system sensitivity of $< 4.7 \times 10^{-5}$ cm^3/sec (helium) or better for standard directly loaded PWR spent fuel assemblies; or leak tight leakage rate of $\leq 2.0 \times 10^{-7}$ cm^3/sec (helium) at a test system sensitivity of $< 1.0 \times 10^{-7}$ cm^3/sec (helium) or better for directly loaded HBU PWR spent fuel assemblies.

Unacceptable leakage rate test results shall be cause for rejection of the component tested. Corrective actions, including repair or replacement of the O-rings and/or closure component, shall be taken and documented as appropriate. The leak test shall be repeated and accepted prior to returning the cask to service.

8.2.3 Subsystems Maintenance

There are no subsystems maintenance requirements on the NAC-STC.

8.2.4 Valves, Rupture Disks and Gaskets on the Containment Vessel

There are no valves on the NAC-STC packaging providing a containment function. Four quick-disconnects, one each on the vent, drain, inner lid interseal test and interlid ports, are provided for ease of cask operation.

The quick-disconnect shall be inspected during each cask loading and unloading operation for proper performance and function. As necessary, the subject quick-disconnect shall be replaced. The quick-disconnects shall be replaced every two years during transport operations, and following fuel unloading after extended storage.

There are no rupture disks on the NAC-STC containment vessel.

All O-rings on the NAC-STC shall be visually inspected for damage during each cask operation. All metallic O-rings shall be replaced during each cask loading sequence. PTFE O-rings shall be replaced if damage is noted during the visual inspection and every two years during transport operations. Viton O-rings shall be replaced annually and as required, based on leak testing results and inspections during operations.

8.2.5 Shielding

The gamma and neutron shields of the NAC-STC packaging do not degrade with time or usage. The radiation surveys performed by licensees prior to transport and upon receipt of the loaded cask provide a continuing validation of the shield effectiveness of the NAC-STC.

8.2.6 Periodic Thermal Test

A periodic thermal test program will be established for each operational NAC-STC packaging. During use of the packaging for transport operations, the periodic thermal test will be performed every five years, or prior to the next use if the period exceeds five years. For NAC-STC packagings utilized for extended storage operations exceeding five years, the periodic thermal test will be performed prior to transport. The periodic thermal test shall be performed in accordance with written, approved procedures.

8.2.6.1 Periodic Thermal Test Set-Up

For periodic thermal test performance, the cask will be in a vertical orientation, loaded with spent fuel, and at thermal equilibrium. For the periodic thermal test, thermal equilibrium is defined as a temperature change of ≤ 3°F/hr at a single centerline fin tip location.

The decay heat load and fuel cycle history of the fuel assemblies loaded in the cask will be known and recorded on the test data sheets. Thermocouples and/or a surface pyrometer calibrated to an accuracy of ± 2°F will be used for temperature measurements during the test.

8.2.6.2 Periodic Thermal Test Procedure

With the cask in a vertical orientation located in a cask preparation area or on a storage pad, a temperature measurement will be taken at a marked cask neutron shield shell centerline fin tip and recorded. Repeat temperature measurement at the marked fin tip location until thermal equilibrium criteria of a temperature change of ≤ 3°F/hr is met. Upon verification of thermal equilibrium, the test temperature measurements will be performed as follows:

- eight (8) fin tip locations will be marked and identified at 45° intervals in the top one-third of the neutron shield shell;
- eight (8) fin tip locations will be marked and identified at 45° intervals in the central one-third of the neutron shield shell;
- eight (8) fin tip locations will be marked and identified at 45° intervals in the bottom one-third of the neutron shield shell;
- four (4) upper forging surface locations will be marked and identified at 90° intervals above the neutron shield; and
- four (4) bottom forging surface locations will be marked and identified at 90° intervals below the neutron shield.

Temperature measurements will be taken and recorded at the twenty-four (24) marked fin tip locations and recorded on the test data sheets. Temperature measurements of the top forging and bottom forging will also be taken and recorded.

The test results will be reviewed and evaluated to verify the acceptance criteria of 8.2.6.3 are met. The results of the test will be documented in an approved test report and maintained in the packagings maintenance program records. If the acceptance criteria are not met, the cask will be tagged as non-conforming until corrective actions are taken. Upon completion of the corrective actions, the cask shall be retested to the original periodic thermal test requirements and acceptance criteria.

8.2.6.3 Periodic Thermal Test Acceptance Criteria

The relationship between the temperature of both ends of the cask relative to the average mid-plane temperature approaches unity as the cask heat load decreases from the design bases. The results of the periodic thermal test will be accepted if the test criteria of a), b), c), and d) below are met:

a) The temperature ratio for the outside surface of the top forging with respect to the average mid-plane neutron shield shell surface temperature meets the test criteria:

Design	Test
$\frac{T_{Top}}{T_{mid-plane}} = \frac{170}{243} = 0.7$	$1.0 \geq \frac{T_{Top}}{T_{mid-plane}} \geq 0.7$

b) The ratio of temperatures of the outside surface of the bottom forging with respect to average mid-plane neutron shield shell surface temperature meets the test criteria:

Design	Test
$\frac{T_{Bottom}}{T_{mid-plane}} = \frac{280}{243} = 1.2$	$1.0 \leq \frac{T_{Bottom}}{T_{mid-plane}} \leq 1.2$

c) Measured temperatures at the top, bottom, and package mid-plane will be equal to or less than design values when corrected for heat load and ambient temperature in accordance with the following relationship:

$$T_{Actual} \leq \frac{Q_{Decay}(T_{Design} - 100)}{Q_{Design}} + T_{Test\ Ambient}$$

d) The individual variations of fin tip temperatures around each zone (upper, middle, and lower) do not exceed 20°F from the zone average.

8.2.7 Miscellaneous

The transport impact limiters shall be visually inspected prior to each shipment. The limiters shall be visually inspected for gross damage or cracking to the stainless steel shells in accordance with approved written procedures and established acceptance criteria. Impact limiters not meeting the established acceptance criteria shall be rejected until repairs are performed and the component reinspected and accepted.

The cask cavity shall be visually inspected prior to each fuel loading. Evidence of gross scoring of the cavity surface, or build-up of other foreign matter in the cask cavity that could block the cavity drainage paths shall be cause for rejection of the cask for use until approved maintenance and/or repair activities have been acceptably completed. The basket assembly for the directly loaded (uncanistered) or canistered configuration shall be visually inspected for deformation of the basket disks or tubes. Evidence of damage shall be cause for rejection of the basket until approved repair activities have been completed, and the basket has been re-inspected and approved for use.

The overall condition of the cask, including the fit and function of all removable components, shall be visually inspected and documented during each cask use. Components or cask conditions which are not in compliance with the Certificate of Compliance shall cause the cask to be rejected for transport use until repairs and/or replacement of the cask or component are performed, and the component reinspected and accepted.

The results of the visual inspections, leak tests, shielding and radiological contamination surveys; fuel identification information for the package contents; date, time, and location of the cask loading operations; and remarks regarding replaced components shall be included in the cask loading report for each loaded cask transport. The requirements of the cask loading report shall be detailed in the NAC-STC Operations Manual.

8.2.8 Maintenance Program Schedule

Table 8.2-1 presents the overall maintenance program schedule for the NAC-STC.

Table 8.2-1 Maintenance Program Schedule

Task	Frequency
Cavity Visual Inspection	Prior to Fuel Loading
Basket Visual Inspection	Prior to Fuel Loading
O-ring Visual Inspection	Prior to Fuel Loading
Outer Lid, Inner Lid and Port Coverplate Bolt Visual Inspection	Prior to installation during each use
Cask Visual and Proper Function Inspections	Prior to each Shipment
Lifting and Rotation Trunnion Visual Inspection Liquid Penetrant Inspection of surfaces and accessible welds	 Prior to each Shipment Annually during use
Maintenance Periodic Leak Rate Test of Inner Lid and Port Coverplate O-rings	For Viton O-rings, annually or when replaced. For metallic O-rings, prior to each shipment
Preshipment Leak Rate Test	Prior to shipment for casks with Viton O-rings
Transport Impact Limiter Visual Inspection	Prior to each shipment
Quick-disconnect Inspection for Proper Function	During each Cask Loading/Unloading Operation
Quick-disconnect Replacement	Every two years during transport operations
Metallic O-ring Replacement	Prior to installation for a loaded transport
Viton O-ring Replacement	Annually, or more often, based on inspection or leak test results
Inner and Outer Lid Bolt Replacement	Every 240 bolting cycles (Every 20 years at 12 cycles per year)
PTFE O-ring Replacement	Every two years during transport operations or as required by inspection
Periodic Leakage Rate Test	Performed within 12 months prior to each shipment for Viton O-rings. No testing needed for out-of-service packaging.
Periodic Thermal Test	Every five years during transport operations, or prior to transport following extended storage periods exceeding five years.