71-9235

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March 1, 2006

U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Attn: Document Control Desk

Subject: Responses to Request for Additional Information (RAI) on the NAC-STC Package

Docket No. 71-9235 TAC No. L23885

Reference:

1. Model No. NAC-STC Package, Certificate of Compliance (CoC) No. 9235, Revision 8, U.S. Nuclear Regulatory Commission (NRC), March 31, 2004

2. Safety Analysis Report (SAR) for the NAC Storage Transport Cask, Revision 15, NAC International, March 2004

3. Submittal of a Request for an Amendment of Certificate of Compliance (CoC) No. 9235 for the NAC-STC Cask to Incorporate Vacuum Drying Enhancements for Connecticut Yankee (CY), Licensing Drawing Updates and the "-96" Package Designation, NAC International, August 4, 2005

4. Certificate of Compliance No. 9235 for the Model No. NAC-STC Package – Request for Additional Information, NRC, February 3, 2006

NAC International (NAC) herewith provides responses to the Request for Additional Information (Reference 4) for the review of NAC's request for an amendment of CoC No. 9235 (Reference 3) in which NAC requested incorporation of the Connecticut Yankee (CY) vacuum drying enhancements, various licensing drawing updates and the "-96" package designation.

This submittal includes four copies of this transmittal letter, four copies of the RAI questions with the NAC responses presented in the standard NAC RAI response format, and four copies of the Revision STC-06A changed SAR pages for Reference 2. The changed NAC-STC SAR pages incorporate the requested information. Consistent with NAC administrative practice, this proposed revision is numbered to uniquely identify the applicable changed pages. Revision bars mark the SAR text changes (on Revision STC-06A pages) that are proposed in this submittal. A List of Effective Pages is included for clarification. Upon final approval, the changed pages will be reformatted, assigned the next appropriate revision number, and incorporated into the NAC-STC SAR.

NMSS01

ED20060008

All changes to the NAC-STC SAR have been tabulated in the List of Changes for the NAC-STC SAR, Revision STC-06A (Attachment 1 to this letter).

Approval of this amendment request will bring the NAC-STC SAR up to date. NAC respectfully requests approval of this amendment request by April 30, 2006.

If you have any comments or questions, please contact me on my direct line at 678-328-1274.

Sincerely,

Anthony L. Patko
Director, Licensing
Engineering

Attachment 1 – List of Changes for NAC-STC SAR, Revision STC-06A

Enclosures

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Attachment 1

List of Changes

for

NAC-STC SAR, Revision STC-06A

NAC International

February 2006

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List of Changes for the NAC-STC SAR, Revision STC-06A (based on Revision 05A of the NAC-STC SAR dated August 2005)

Chapter/Page/ Table	Description of Change
Note: The affected Master Table of Contents, Chapter Table of Contents and List of Tables have been revised accordingly to reflect the list of changes detailed below.	
Chapter 1	
Page 1.2-22	Item 8 – changed "10 CFR 71.87(i)(1)" to "10 CFR 71.87(i)"
Chapter 2	
Page 2.6.15.4-2, Table 2.6.15.4-1	Revised "Table" reference column & last two items in "Condition" column
Chapter 3	
Page 3.1-6	Section 3.1.3, 1st paragraph – added new 2nd sentence
Page 3.1-7	2nd full paragraph – added new 4th sentence
Chapter 4	
Pages 4.2-1 & 4.2-2	Section 4.2, old 3rd paragraph – inserted 5 new paragraphs after "licensed inventory as described in Chapter 5." The remainder of the old 3rd paragraph follows this text insertion.
Chapter 7	
Page 7.1-7	Item 15, 1st sentence – changed "an air, nitrogen or helium supply" to "a nitrogen or helium supply"; 3rd sentence – removed "air" in 2 places

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RESPONSE TO THE

UNITED STATES
NUCLEAR REGULATORY COMMISSION

REQUEST FOR ADDITIONAL INFORMATION

(FEBRUARY 3, 2006)

NAC STC PACKAGE

(TAC. No. L23885, Docket No. 71-9235)

FEBRUARY 2006

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NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 1: General Description

1-1 Verify the surface contamination requirements, as described in Item 8 of Section 1.2.3 of the Safety Analysis Report (SAR).

The requirement of surface contamination stated in Item 8 on page 1.2-22 relating to CFR is quoted as 10 CFR 71.87(I)(1). The staff reviewed 10 CFR Part 71 and found that there is no such provision in the regulations. This information should be corrected.

This information is needed to satisfy the requirements of 10 CFR 71.33.

NAC Response

NAC has verified the quoted reference to be 10 CFR 71.87 (i) and has corrected the typographical error by deleting the "(1)" from the end of the quotation.

The corrected Chapter 1 SAR page 1.2-22 is included in Revision STC-06A of the NAC-STC SAR.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 2: Structural

2-1 Clarify entries to Table 2.6.15.4-1 of the SAR to ensure that loading conditions are correctly identified and the stress summary tables are properly referenced.

The loading conditions, "Top End Drop, Pressure + Inertia," for stress intensities P_m and P_m+P_b are inconsistent with those described in SAR Page 2.6.15.4-1, where the bounding conditions are stated to be the top end and bottom end drops without pressure. Also, the minimum margins of safety listed in Table 2.6.15.4-1 are not found in the specific table references listed for each end drop condition.

This information is needed to satisfy the requirements of 10 CFR 71.33.

NAC Response

Table 2.6.15.4-1 has been revised to reference the correct detailed stress table for each load condition presented. The corrected Chapter 2 SAR page 2.6.15.4-2, Table 2.6.15.4-1, is included in Revision STC-06A of the NAC-STC SAR.

The maximum stress intensities reported on Page 2.6.15.4-1 for the top end drop zero internal pressure and bottom end drop zero internal pressure cases are 5.37 ksi and 4.33 ksi, respectively. These two stress intensity values are reported in Table 2.6.15.4-3 - Section 2 and Table 2.6.15.4-5 - Section 5, respectively, in the Stress Intensity column. As stated above, the referenced tables listed in Table 2.6.15.4-1 have been corrected.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 3: Thermal

3-1 Provide the projected maximum number of fuel assemblies, per cask loading, that could have stainless steel rods inserted into the RCCA guide tubes. If the majority of fuel assemblies in any given cask loading could have stainless steel rods inserted, describe, either quantitatively or qualitatively, how the current thermal analysis of the NAC-STC would be affected by the addition of the maximum possible number of stainless steel rods.

The proposed change to allow the insertion of solid stainless steel rods into Connecticut Yankee intact and damaged fuel assembly RCCA guide tubes that do not contain a RCCA could have an impact on the thermal performance of the cask system. The staff needs assurance that the thermal effects of this proposed amendment have been considered.

This information is needed to assure compliance with 10 CFR 71.7.

NAC Response

The 15×15 Connecticut Yankee fuel contains approximately 20 openings for the stainless steel rods and one position for the instrument tube. The thermal modeling conservatively treats the 225 array locations as having fuel rods, including the center position that physically corresponds to an instrument tube location. The homogenization incorporated into the thermal modeling of the fuel provides a conservative thermal response by introducing decay heat into the RCCA guide tube locations. Due to the spacing of the fuel rods and the fact that one fuel rod does not contact and conduct heat directly to the adjacent fuel rod, radiation is the most significant contributor to heat transfer from the fuel assembly. Addition of the stainless steel rod to the RCCA guide tube is internal to the tube and has no effect on the radiation characteristics of the fuel rod comprising the fuel assembly. It is further noted that the stainless steel rods used to displace water in the RCCA dash pot are approximately 21 inches long. Therefore, placement of the stainless steel rods in all 20 possible locations for all fuel assemblies would not affect the current thermal properties used in the thermal analyses, nor the maximum temperatures in the thermal evaluation of the Connecticut Yankee fuel.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 3: Thermal

NAC Response to RAI 3-1 (continued)

Discussion of the fuel assembly fuel model presented in Section 3.1.3 has been revised to include the following statements:

Page 3.1-6 – "The 15×15 assembly may contain stainless steel rods, approximately 21 inches in length, in the RCCA guide tubes to displace water in the guide tubes during the canister drying operations."

Page 3.1.7 – "Effective thermal properties developed with this model bound the thermal response of Connecticut Yankee fuel assemblies using stainless steel rods to displace water in the RCCA guide tubes during canister drying operations."

The revised Chapter 3 SAR pages 3.1-6 and 3.1-7 are included in Revision STC-06A of the NAC-STC SAR.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 4: Containment

4-1 Revise the containment allowable leakage rate calculation in Section 4 of the SAR to list all relevant assumptions regarding the bounding fuel assembly, including the assumed assembly average power, burn time, and cool time for each cycle. Also, explain how each of the relevant fuel assembly assumptions are bounding with respect to containment of radioactive material.

Section 4.2 of the SAR gives basic bounding fuel assembly characteristics such as lattice type, average burnup, enrichment, and cool time, but does not give an assembly average power, burn time, and cool time for each cycle assumed for the source term determination. Also, Section 4.2 states that the source term of the reference fuel assembly bounds the licensed inventory as described in the shielding analysis of Section 5. However, it is not clear that the source term which is bounding for the shielding analysis is also bounding for the containment analysis.

This information is needed in order to determine that the Model. No. NAC-STC meets the containment requirements of 10 CFR Part 71.51(a).

NAC Response

Included in Section 4.5.6 of the SAR is the SAS2H input file used to generate the source term for the containment evaluation. This file lists average power, burn time and cool time for each cycle. The data matches the description provided in Chapter 5, in particular power history parameters in Table 5.2-2, and calculated active cycle length in Table 5.2-3. Key comparison of fuel characteristics, specifically fuel masses that dominate the release source term at a fixed burnup and cool time, are included in Table 5.2-1. A discussion of the input parameters is provided in the following paragraphs to demonstrate that the variables employed are appropriate and provide an overall bounding source description for the containment analysis. The discussion is added to the text in Section 4.2 in Revision STC-06A of the NAC-STC SAR. The additional text highlights the input parameters to the containment source calculation, as well as provides justification for the use of each parameter.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 4: Containment

NAC Response to RAI 4-1 (continued)

The directly loaded fuel is the only payload 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 restricted to 45,000 MWd/MTU as discussed in Chapter 1 and Chapter 5 of the SAR. The minimum cool time for the directly loaded assemblies per Chapter 5, Table 5.4-5, is 5 years for lower burnup materials. At the maximum assembly average burnup of 45,000 MWd/MTU, minimum cool times range from 7 to 15 years depending on initial enrichment and fuel type. Increasing burnup to 60,000 MWd/MTU not only increases fission products, but also generates significant quantities of higher actinides that drive the containment analysis through fine release. Applying a 5-year cool time rather than the minimum cool time indicated in Chapter 5 for higher burnup fuel further serves to increase the sources for the containment evaluation (minimum cool time required for a 60,000 MWd/MTU assembly is in excess of 10 years). A combination of 60,000 MWd/MTU burnup and a 5-year cool-time as basis for the containment source generation presents a very conservative combination of input parameters.

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.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 4: Containment

NAC Response to RAI 4-1 (continued)

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 of the SAR.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 4: Containment

4-2 Revise the containment analysis to explain how the releasable crud activity per cask given in Table 4.2-2 was calculated.

Table 4.2-2 lists a releasable crud activity per cask of 3.71 TBq (100.2 Ci). Based on an assembly surface area of 3.54×10^5 cm^2, a ^{60}Co concentration of 140 $\mu Ci/cm^2$, a crud spallation fraction of 0.15, and 26 assemblies per cask, the releasable crud activity should be 7.15 TBq (193.3Ci).

This information is needed in order to determine that the Model No. NAC-STC meets the containment requirements of 10 CFR Part 71.51(a).

NAC Response:

As indicated in SAR Section 4.2.3.2, "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." PWR fuel assemblies are evaluated as "5-year cooled" as stated in the second paragraph in SAR Section 4.2.3.2. Decaying the cobalt content from "initial discharge" to "5-year cooled" using the 5.27 year ^{60}Co half life produces the listed curie inventory.

NAC INTERNATIONAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Chapter 8: Operating Procedures *(NAC-STC SAR Chapter 7)*

8-1 Revise Step 15 of Section 7.1.3.1 of the SAR to exclude the use of air as an acceptable gas for draining water from the cask cavity, or provide additional evidence that the use of air during cask draining will not result in damage to the fuel cladding as a result of oxidation of the fuel pellets.

As stated in proposed Interim Staff Guidance No. 22 (ISG-22), the exposure of spent nuclear fuel rods with cladding breaches to an oxidizing atmosphere can lead to oxidation of the fuel pellets, and cause further damage to the fuel cladding as a result of fuel pellet swelling. The staff recommends that applications for spent nuclear fuel storage or transportation design approval demonstrate there will be reasonable assurance that the fuel cladding will be protected. For more information, please refer to Draft ISG-22, "Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-term Cask Loading Operations in LWR or Other Uranium Oxide Based Fuel."

This information is needed to satisfy the requirements of 10 CFR 71.33(b).

NAC Response

This RAI question refers to Chapter 7, Operating Procedures. Section 7.1.3.1, Step 15, has been revised to remove air as an optional gas used for cask draining operations.

The changed Chapter 7 SAR page is included in Revision STC-6A of the NAC-STC SAR.

February 2006

Revision STC-06A

NAC-STC

NAC Storage Transport Cask

SAFETY ANALYSIS REPORT

Docket No. 71-9235

NAC INTERNATIONAL

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additional one inch of stainless steel shielding at the bottom in the axial direction of the cask. A detailed description of the NAC-STC shielding design, as well as the shielding analysis, is provided in Chapter 5.

1.2.1.6 Protrusions

There are no outer protrusions on the NAC-STC other than the four external lifting trunnions that are welded to the top forging near the upper end of the cask. The lifting trunnions are within the envelope protected by the impact limiters. The inner lid and all of the port covers are recessed into the cask body and do not protrude above the cask surface. The outer lid forms a smooth surface with its recessed bolts.

1.2.2 Operational Features

The NAC-STC is designed for ease of operation. The cask is designed to be easily loaded, unloaded, and handled at a nuclear facility. The configuration and surface finish of the cask exterior surfaces have been designed to facilitate and minimize cask decontamination efforts. The inner lid, the outer lid, and the port covers are all one-piece components designed to reduce handling times and to maintain personnel dose rates as low as reasonably achievable (ALARA). Quick-disconnect fittings are provided in the vent and drain ports, the inner lid interseal test port and in the interlid port for improved handling operations. All operational features are shown on the license drawings provided in Section 1.3.2. An operational schematic for the NAC-STC is shown in Figure 1.2-1. Operating procedures are provided in Chapter 7 for both canistered and directly loaded (uncanistered) fuel operations.

1.2.3 Contents of Packaging

Shipments in the NAC-STC shall not exceed the following limits:

1. The maximum content weight of the NAC-STC shall not exceed 67,200 pounds in the CY-MPC configuration and 56,000 pounds in the Yankee-MPC and directly loaded configurations.
2. The limits specified in Tables 1.2-2 through 1.2-4 for the design basis fuels shall not be exceeded.

3. The total decay heat of the cavity contents shall not exceed 22.1 kilowatts for directly loaded fuel, 12.5 kilowatts for Yankee Class canistered fuel and 17 kilowatts for Connecticut Yankee canistered fuel.

4. The total weight of the Zircaloy and/or stainless steel clad fuel assemblies shall not exceed 30,600 pounds in the Yankee-MPC, 35,100 pounds in the CY-MPC and 39,000 pounds for directly loaded fuel.

5. The total weight of the Yankee-MPC and CY-MPC GTCC canistered waste shall not exceed 12,340 pounds and 18,742 pounds, respectively.

6. Any number of NAC-STC casks may be shipped at one time on a railcar, a ship, a barge, or a heavy-haul vehicle.

7. Radiation levels shall not exceed the requirements of 10 CFR 71.47, 10 CFR 71.51, and IAEA Safety Standard Series No. TS-R-1, Common Provision B3.

8. Surface contamination levels shall not exceed the requirements of 10 CFR 71.87(i) and IAEA Safety Standard Series No. TS-R-1, Common Provision B4.

1.2.3.1 Design Basis Spent Fuel

The NAC-STC is designed to safely transport spent fuel assemblies in two configurations. The fuel assemblies may be directly loaded into a fuel basket installed in the cask cavity (uncanistered) or sealed in a transportable storage canister (canistered). The design basis fuel assemblies for the uncanistered configuration are the Framatome-Cogema 17×17 and Westinghouse 17×17 or 15×15 PWR fuel assemblies. These assemblies bound smaller array Westinghouse, and similar Babcock & Wilcox, and Combustion Engineering PWR fuel assemblies. The NAC-STC can transport 26 directly loaded PWR fuel assemblies.

The key parameters of the design basis directly loaded spent fuel assemblies are provided in Table 1.2-2.

In the Yankee-MPC configuration, the NAC-STC can transport up to 36 Yankee Class fuel assemblies, including Yankee-MPC Reconfigured Fuel Assemblies and recaged fuel assemblies, including four intact or damaged fuel assemblies or Reconfigured Fuel Assemblies loaded in Yankee-MPC damaged fuel cans (four are required due to shield lid configuration). Any intact spent fuel assembly with fuel rods removed shall have each of the empty fuel rod positions filled

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2.6.15.4 Stress Evaluation of the CY-MPC Canister for 1-Foot End Drop Load Condition

A structural analysis is performed using ANSYS to evaluate the effect of a 1-foot end drop impact for both the bottom and top end orientations of the canister. The ASME Code, Section III, NB requires the stresses arising from operational loads be assessed based on the primary loads. The primary loads for the 1-foot drop are due to the deceleration of the canister and its contents (65,821 pounds - fuel) and the conservatively assumed 20 psig pressure load internal to the canister. The applied deceleration is 20g for both orientations. The inertial load of the canister was addressed by the deceleration factor applied to the canister density, while the fuel and basket weight (49,155 pounds) was represented by a pressure load on the inner end surface of the canister. Displacement constraints are applied to the plane of symmetry and the gap elements attached at the canister end to represent the top or bottom of the transport cask.

To determine the effect of the 20 psig pressure load, the top end and bottom end drop orientations with and without the pressure load were analyzed. The maximum stresses are summarized below:

Drop Orientation	Internal Pressure (psi)	Maximum Stress Intensity (ksi)
Bottom End Drop	0	4.33
Bottom End Drop	20	2.95
Top End Drop	0	5.37
Top End Drop	20	2.88

The bounding cases are the top and bottom end drops without pressure.

The location of the linearized stresses is shown in Figure 2.6.15.3-1. The maximum stresses for P_m and $P_m + P_b$ are tabulated in Tables 2.6.15.4-2 through 2.6.15.4-5. The critical sections for the pressure and the pressure plus the deceleration load, with reference to the section and the appropriate tables, are shown in Table 2.6.15.4-1.

Table 2.6.15.4-1 Summary of Critical Sections of the Canister for the 1-Foot End Drop Condition

Condition	Stress	Critical Section	Table	Minimum Margin of Safety
Top End Drop Pressure + Inertia	P_m	2	2.6.15.4-2	+ 8.5
Top End Drop Pressure + Inertia	$P_m + P_b$	2	2.6.15.4-3	+ 3.54
Bottom End Drop Pressure + Inertia	P_m	2	2.6.15.4-4	+ 7.21
Bottom End Drop Pressure + Inertia	$P_m + P_b$	5	2.6.15.4-5	+ 4.63

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defined physical contact conditions. 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, canister, cask shells, radial shielding and surface locations.

In the canistered fuel configuration, the NAC-STC is used as a transport cask, while storage of the canister is in a vertical concrete cask. Therefore, the maximum heat generation will occur after loading for direct transport without interim storage, at which time the canister and cask interior can reasonably be assured to contain helium.

A second model determines the effective conductivity of the fuel, which is a detailed two-dimensional thermal model of the fuel assembly. The model includes the fuel pellets, cladding and gas (considered to be helium) 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 for the steady state condition.

In the three-dimensional model, the fuel tube, including the neutron absorber sheet and its cladding, is also modeled using effective conductivity. The effective conductivity is determined by a two-dimensional thermal model. The model includes the fuel tube, the neutron absorber sheet, helium gaps on both sides of the neutron absorber sheet and a helium gap between the stainless steel cladding for neutron absorber and the support disk or heat transfer disk.

Classical analysis is used to calculate temperatures during the hypothetical accident condition fire transient for the various components of the fuel, canister basket, canister, and cask. The methodology is described in Section 3.5.1.1.

The Yankee-MPC canister may contain one or more Reconfigured Fuel Assemblies, Recaged Fuel Assemblies or damaged fuel assemblies in Damaged Fuel Cans.

A two-dimensional finite element model is generated using the ANSYS program to determine the temperature distribution of the Reconfigured Fuel Assembly. The model comprises the fuel rods, fuel tubes, the shell casing (the square tube with the same external dimensions as an intact fuel assembly) and the gas (helium) occupying the gap between fuel rod and tube, the space between fuel tubes and the gap between shell casing and the fuel assembly tube.

Recaged fuel assemblies consist of United Nuclear fuel rods installed in Combustion Engineering fuel assembly lattices. Certain United Nuclear fuel assemblies may be destructively disassembled for the purpose of inspection and testing of individual fuel rods. The United Nuclear fuel rods may be installed in a Combustion Engineering fuel assembly lattice and placed in pool storage. The Combustion Engineering fuel assembly lattice will provide the same grid support structure as did the original United Nuclear fuel assembly lattice, but it will not have the shroud fixture used to preclude water impingement on the fuel rods. No empty fuel rod positions are permitted in the recaged Combustion Engineering fuel assembly lattice. The maximum heat load of the recaged fuel assemblies is bounded by the design basis heat load of 0.347 kW/assembly.

Damaged fuel assemblies are standard fuel assemblies with up to 20 fuel rods having known or suspected cladding defects greater than hairline cracks or pinhole leaks. They include fuel assemblies with structural damage such that the fuel assembly cannot be handled by normal means. Each damaged fuel assembly is placed in a Damaged Fuel Can to confine fuel material and to facilitate handling and retrievability. Damaged Fuel Cans are restricted to the four corner positions of the fuel basket. The thermal analysis of the damaged fuel can configuration is performed using the three-dimensional model modified to account for the assumed redistribution of the fuel in the four corner positions.

3.1.3 Canistered Connecticut Yankee Fuel

The design basis fuel assembly for the Connecticut Yankee canistered fuel thermal evaluation is the B&W or Westinghouse 15×15 assembly, having a burnup of 43,000 MWD/MTU (Zircaloy clad) or 38,000 MWD/MTU (stainless steel clad), a cool time of 10 years and an assembly decay heat of 654 watts. The 15×15 assembly may contain stainless steel rods, approximately 21 inches in length, in the RCCA guide tubes to displace water in the guide tubes during the canister drying operations. This analysis uses a heat load for 26 assemblies of 17 kilowatts, or an individual assembly decay heat limit of 0.654 kilowatts. The thermal analysis for the CY-MPC canistered fuel 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 the fuel basket in contact with the canister, which is, in turn, in contact with the cask inner shell. This simulates "no gap" on one side of the basket and canister, and a "maximum gap" condition at the opposite side of the basket and canister as depicted in Figure 3.1-3. Additionally, Figure 3.1-3

defines the two basket orientations analyzed. Note that the nominal diametral gap between the support disk and the canister shell is 0.24 inch (radial gap = 0.12 inch). The nominal diametral gap between the heat transfer disks and the canister shell is 0.52 inch (radial gap = 0.26 inch). In this model, circumferential half-symmetry is used. The model comprises the fuel assemblies, fuel tubes, stainless steel support disks, aluminum heat transfer disks, the canister shell, lids, and bottom plate, the spacer at the bottom of the canister, and the NAC-STC. The NAC-STC portion of the model includes the bottom spacer, the inner and outer shells, top forging, lead gamma shield, inner and outer bottom plates, inner and outer lids, upper and lower neutron shields, radial neutron shield, and the neutron shield shell. The volumetric heat generation rate of the fuel is multiplied by a factor of 1.1 (peaking factor to reflect the power density levels within the fuel). Gas inside the canister and inside the NAC-STC is modeled as helium. Gaps between the support disks and canister shell and between the heat transfer disks and canister shell are adjusted to account for differential expansion based on thermal and defined physical contact conditions. Insolation and ambient temperature conditions are applied to the neutron shield shell when appropriate.

The maximum heat generation occurs immediately after loading for transport, at which time the canister and cask interior can reasonably be assured to contain helium.

In the three-dimensional model, the fuel assemblies are modeled as homogeneous regions with effective thermal properties. A second model is used to determine the effective thermal conductivity of the fuel, which is a detailed two-dimensional thermal model of a single fuel assembly. The model includes the fuel pellets, cladding, and gas (considered to be helium) between the fuel rods and gas occupying the gap between the fuel pellets and cladding. Effective thermal properties developed with this model bound the thermal response of Connecticut Yankee fuel assemblies using stainless steel rods to displace water in the RCCA guide tubes during canister drying operations. Modes of heat transfer modeled include conduction and radiation between individual fuel rods for the steady-state condition. Radiation between the fuel pellets and cladding is conservatively neglected.

In the three-dimensional model, each fuel tube, including the neutron absorber sheet and its cladding, is modeled using effective thermal conductivity properties. The effective conductivity for each side of a fuel tube is determined using a two-dimensional thermal model. The model includes the fuel tube, the neutron absorber sheet, the stainless steel cladding, helium gaps on

both sides of the neutron absorber sheet, and a helium gap between the stainless steel cladding and the support disk or heat transfer disk.

The three-dimensional model is used to calculate fuel, basket, canister, and NAC-STC temperatures for normal transport and hypothetical accident conditions.

The CY-MPC may contain one or more reconfigured fuel assemblies or damaged fuel cans. The thermal analysis for canister configurations containing reconfigured fuel assemblies or damaged fuel cans is bounded by the canister configurations for design basis fuel, as described in Section 3.4.2.4.

3.1.4 Canistered Greater Than Class C Waste

Greater than Class C (GTCC) waste may be transported in the transportable storage canister. GTCC waste comprises reactor core components placed inside of containers having the same geometry as the design basis fuel assemblies, or activated hardware classified as GTCC loaded directly into interior loading positions of the GTCC basket. The GTCC basket has 24 loading positions.

The thermal output of the Yankee GTCC waste is 2.9 kilowatts. Component temperatures for the GTCC basket are conservatively calculated using a thermal resistance model. The calculated temperatures are then used to determine the material properties in the structural analysis of the GTCC basket.

The thermal output of the Connecticut Yankee GTCC waste is 5.0 kilowatts. Component temperatures for the GTCC basket are conservatively calculated using an ANSYS model. The calculated temperatures are then used to determine the material properties in the structural analysis of the GTCC basket.

The thermal output of the Yankee and Connecticut Yankee GTCC waste (2.9 kW and 5 kW, respectively) is significantly less than the design basis thermal load for directly loaded fuel and for canistered fuel. Consequently, the temperatures produced in the cask by the design basis fuel bound the temperature effects on the NAC-STC cask body and the canister due to the GTCC waste.

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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 source term of the reference fuel assembly bounds the licensed inventory as described in Chapter 5.

The directly loaded fuel is the only payload 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 restricted to 45,000 MWd/MTU as discussed in Chapter 1 and Chapter 5. The minimum cool time for the directly loaded assemblies per Chapter 5, Table 5.4-5, is 5 years for lower burnup materials. At the maximum assembly average burnup of 45,000 MWd/MTU, minimum cool times range from 7 to 15 years depending on initial enrichment and fuel type. Increasing burnup to 60,000 MWd/MTU not only increases fission products, but also generates significant quantities of higher actinides that drive the containment analysis through fine release. Applying a 5-year cool time rather than the minimum cool time indicated in Chapter 5 for higher burnup fuel further serves to increase the sources for the containment evaluation (minimum cool time required for a 60,000 MWd/MTU assembly is in excess of 10 years). A combination of 60,000 MWd/MTU burnup and a 5-year cool-time as basis for the containment source generation presents a very conservative combination of input parameters.

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 7.5×10^{-5} ref cm^3/sec, or 9.3×10^{-5} cm^3/sec (helium).

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 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

without interim storage using Viton O-rings, the allowable leak rate is calculated using the methodology of NUREG/CR-6487. Consequently, the cask meets the requirements of 10 CFR 71.51 for directly loaded and for canistered fuel or GTCC waste.

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 does 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% 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). 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.

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 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). The calculation of the allowable leak rate for the Viton O-ring configuration 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 for long-term storage with subsequent transport, and when the cask is loaded with a transportable storage canister. Metallic O-rings may also be used in the containment boundary when the cask is directly loaded 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, 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 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. As described in this section, this leak rate meets the requirements of 10 CFR 71.51 for containment of the radioactive contents.

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]

Activity Density of Radioactive Material (C_N)

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.

$$C_n = C_{Crud} + C_{Volatiles} + C_{FissionGas} + C_{Fines}$$

$$C_{Crud} = \frac{f_C M_T}{V} = \frac{f_C S_C N_A (N_R S_{AR})}{V}$$

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 [cm$_3$]
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 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:

$$L_c = \frac{2.48x10^6 D^4}{a\mu}(P_u - P_d) * \frac{P_a}{P_u} = F_c * (P_u - P_d) * \frac{P_a}{P_u}$$

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:

$$L_m = \frac{3.81x10^3 D^3 \sqrt{\frac{T}{M}}}{aP_a}(P_u - P_d) * \frac{P_a}{P_u} = F_m * (P_u - P_d) * \frac{P_a}{P_u}$$

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

The transportable storage canister welded closure is leak tested at final assembly to leaktight conditions, 1 x 10^{-7} ref cm^3/ sec, as defined by ANSI N14.5-1997. To meet this requirement, the allowable leak rate is 2 x 10^{-7} cm^3/sec (helium). The leak test sensitivity applied in testing the canister at the time it is closed is 1 x 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 x 10^{-8} cm^3/sec (helium), with a corresponding test sensitivity of 4 x 10^{-8} cm^3/sec (helium). This specified test condition is conservative with respect to the leaktight condition specified by ANSI N14.5-1997.

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

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Chapter 7

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 nitrogen or 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 nitrogen or helium supply off and disconnect the nitrogen or helium supply line from the vent port. Then, disconnect the drain line from the drain port quick-disconnect.
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.
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. For new replacement Viton O-rings, use a leak detector connected to the interseal test port to verify the total 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).

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).

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.

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(1) For new Viton O-rings, 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.