71- 9270

NAC INTERNATIONAL

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October 21, 2002

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

Attn: Document Control Desk

Subject: Submittal of Responses to Supplemental Thermal Questions on the UMS® Universal Transport Cask Application

Docket No. 71-9270

References:
1 Submittal of Supplemental Responses to the NRC Questions on the UMS® Universal Transport Cask Application, NAC International, September 27, 2002
2. Conference Calls on Final Thermal Questions on the UMS® Universal Transport Cask Application, NRC and NAC International, October 8, 9, & 10, 2002

In accordance with the conference call discussions of Reference 2, NAC International (NAC) herewith submits eight copies of the responses to the supplemental thermal questions related to the review of the NAC-UMS® Universal Transport Package application. Also, provided in this submittal are the SAR Section 5.5.1 2 (Maine Yankee Site Specific GTCC Waste) pages, which were inadvertently omitted from the Reference 1 submittal. Please note that this SAR section was initially submitted in Revision UMST-00A, dated June 29, 2000.

This submittal includes the NRC questions and the NAC responses to those questions presented in the standard NAC response format, followed by the associated NAC-UMS® Safety Analysis Report (SAR) changed pages, which are designated as Revision UMST-02E. Note: The enclosed SAR changed pages are to be inserted as replacement or new additional pages, as applicable, into the existing NAC-UMS® SAR binders. The List of Effective Pages provided in this submittal can be used to ensure that the correct page revisions are incorporated in the SAR binders.

The Revision UMST-02E changed pages have been prepared in accordance with the following conventions:

- Revision indicators (shading and revision bars) are used to highlight changes. Shading indicates a revision from NAC-UMS® SAR, Revision 0, while a revision bar indicates a change in the SAR from a previous revision, subsequent to Revision 0.
- Revision bars also indicate text flow.
- The changed pages for this submittal are designated as Revision UMST-02E to provide a unique identification of the pages and changes.
- All of the pages in the List of Effective Pages are designated Revision UMST-02E and no revision bars are used on those pages.

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NAC is anxious to complete the review and certification process for this UMS® Universal Transport Cask Application in accordance with the published schedule NAC appreciates the continued efforts of the NRC staff to complete this application process.

If you have any comments or questions, please contact me on my direct line at 678-328-1321.

Sincerely,

T C Thompson

Thomas C. Thompson
Director, Licensing
Engineering & Design Services

Enclosures

cc: Paul Plante (MY)
Tom Williamson (MY)
Brian Hansen (APS)
Glenn Michael (APS)
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NAC-UMS
Docket # 71-9270
TAC # L22452

NAC INTERNATIONAL

RESPONSE TO THE

UNITED STATES
NUCLEAR REGULATORY COMMISSION

NAC UNIVERSAL TRANSPORT SYSTEM (NAC-UMS®)

REQUEST FOR SUPPLEMENTAL INFORMATION

SAR SUBMITTAL – REVISION UMST-02E

OCTOBER 21, 2002

NAC INTERNATIONAL RESPONSE
TO
REQUEST FOR SUPPLEMENTAL INFORMATION

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CHAPTER 3: THERMAL EVALUATION

3-1 The Thermal Section of the NAC-UMS Transport Cask SAR needs additional information associated with the Maine Yankee site-specific fuel. A preferential loading discussion regarding its impact on the component temperatures including seals and o-rings allowable temperature limits is needed. Note that a similar discussion is provided for the general fuel in Section 3.4.2.1.

NAC Response

The following text has been added as the last paragraph in Section 3.4.2.1:

The preferential loading does not result in any slot containing fuel with a heat load greater than 0.833 (20/24) kW. As summarized for the above cases, the maximum fuel cladding temperature in Case 1 is bounded by the maximum fuel cladding temperature in Table 3.4-1. Since the total heat for the uniform heat load case bounds the three cases of preferential loading, the component temperatures provided in Table 3.4-1 bound the component temperatures resulting from the preferential loading cases.

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CHAPTER 3: THERMAL EVALUATION

3-2 The Thermal Section of the NAC-UMS Transport Cask SAR needs additional information associated with the Maine Yankee site-specific fuel. Damaged fuel assemblies are discussed on Page 3.6-7, but no mention is made of the effect of compaction of the damaged fuel on seals and o-rings. Rather, only cladding, structural support disks and heat transfer disks are discussed. Additional discussion is needed for the seals and o-rings.

NAC Response:

The following text has been added as the third paragraph on page 3.6-8:

The effect of the compaction of the damaged fuel is most significant for the interior of the basket, and this effect is determined to be 10 °F, as shown in the table above. For the cask body closure lid seal, the effect of the damaged fuel is expected to be insignificant, since the transportable storage canister shield and structural lids, representing a thickness of 10 inches of steel, separate the fuel from the cask body closure lid seals. The canister lids act to spread any concentration of heat from the damaged fuel. The port cover seals are even more remote from the damaged fuel than the cask body lid seals and, therefore, are not considered to be affected by the damaged fuel.

NAC INTERNATIONAL RESPONSE
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REQUEST FOR SUPPLEMENTAL INFORMATION

CHAPTER 3: THERMAL EVALUATION

3-3 Evaluate the effect of the Hypothetical Fire Accident on the closure function of the cask lid bolts.

NAC Response

The hypothetical Fire accident is expected to have an insignificant effect on the lid bolts. The impact limiters have been shown to remain attached to the cask during any cask drop events. Therefore, the temperature increase for the lid bolts is not significant due to the fire accident. The sealing function of the closure lid will not be affected (lid bolts will not become loose) due to the temperature increase during the fire accident because the coefficient of thermal expansion of the cask lid (Type 304 stainless steel) is larger than that for the lid bolt (SB-637 Nickel alloy).

As shown in SAR Section 2.7.1.7, a stress evaluation is performed for the closure lid and the bolts for the governing accident condition of the 30-foot drop, which bounds the fire accident condition. The evaluation for the 30-foot drop considered the loadings from the bolt preload, thermal expansion (normal condition of transport), o-ring compression, internal pressure, and cask content and lid self-weight resulting from a 60g top end corner impact. The resulting loads on the lid bolts for the drop accident condition are significantly higher than those for the fire accident condition and the calculated bolt stresses for both conditions are well below the allowable stresses.

Therefore, the hypothetical fire accident condition does not have any adverse effect on the cask lid and closure function of the lid bolts.

NAC INTERNATIONAL RESPONSE
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REQUEST FOR SUPPLEMENTAL INFORMATION

CHAPTER 3: THERMAL EVALUATION

3-4 A sensitivity study was performed in Section 3.4.1.1.1 of the SAR for the 3-D Cask Model. The study analyzed 8 effects that had the general impact of restricting heat flow from the cask, thereby increasing the temperature of the internal canister components (i.e. cladding and disks). However, since the neutron shield is the component closest to its allowable temperature (only 8° F), it appears the combined effect of Cases 1, 2 & 7 plus a reversal of the effects of Cases 3 thru 6 & 8, would all tend to increase the temperature of the neutron shield material well beyond its 300° F temperature limit for the NCT. Please explain.

In addition, please calculate the max neutron shield temp based on the following combination:

Increase effect of Cases 1,3,4 and 8
plus
Decrease effect of Cases 5,6,and 7
plus
Remove the no contact assumption between the fuel assembly & fuel tube and remove it between the fuel tube and the support disk.

Please demonstrate that with the 10% conservatism in the heat transfer coefficient, the neutron shield temperature limit will not be exceeded.

<u>NAC Response:</u>

Using the same methodology as used for the sensitivity table, an additional case was evaluated in which the contact area was increased by 100% (the reversal of the effect of case 3) in conjunction with the increased thickness (of 8%) of the heat transfer disk (the reversal of the effect of case 4). The increased disk thickness represents an increased conductance of the basket, which has a similar effect as the reversal of cases 5, 6 and 8.

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NAC Response to 3-4 continued

The maximum neutron shield temperature was determined to be 293°F for the additional case, which is the same as the maximum temperature reported in Table 3.4-1. This is consistent with the observation that as the contact area is increased the heat is more efficiently dispersed through the 7.5 inches of metal in the cask body shells, which would prevent a concentration of heat in the radial neutron shield. Therefore, the reversal of the effects of cases 3 through 6 and 8 are expected to be negligible on the neutron shield temperature.

A review of the thermal testing of the NAC-LWT cask, which is approximately 1 meter in diameter, shows that the current heat transfer film coefficient used in the UMS thermal evaluation is more than 10% conservative. Using the methodology for the cases in the sensitivity study, with no reduction in film coefficient at cask surface, an additional analysis is performed based on the combination of the following conditions: (1) 10% increase of the emissivity of stainless steel and aluminum; (2) Increased contact area between disks and canister shell (from 2° to 4° in the half-symmetry model); (3) 8% increase in heat transfer disk thickness; (4) Decreased gap between disks and canister shell (5) Decreased gap between canister shell and cask inner shell; (6) 6% reduction of cask radial copper fin thickness; (7) 10% increase of the lead emissivity; and (8) The effective thermal conductivity of the fuel tubes elements in the 3-D model (at the contact region only, i.e. the side of the fuel in contact with the tube) changed to the thermal conductivity of stainless steel (to simulate the contact between the fuel assembly and the fuel tube and between the fuel tube and the disks). The analysis results indicate that the maximum temperature of the neutron shield is 289°F, which is below the allowable temperature of 300°F. This demonstrates that the maximum temperature in the neutron shield will be less than the allowable temperature even when the sensitivities are considered and that the maximum temperature in the neutron shield reported in Table 3.4-1 is conservative.

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NAC Response to 3-4 continued

The following SAR text is inserted at the end of Section 3.4.1.1.1:

The maximum temperature for the radial neutron shield is 293°F as reported in Table 3.4-1. To ensure that sufficient margin exist for the neutron shield, an additional sensitivity study was performed using the model for the nine cases evaluated for the fuel and basket components. The analysis considers a combination of the following conditions: (1) 10% increase of the emissivity of stainless steel and aluminum; (2) Increased contact area between disks and canister shell (from 2° to 4° in the half-symmetry model); (3) 8% increase in heat transfer disk thickness; (4) Decreased gap between disks and canister shell (5) Decreased gap between canister shell and cask inner shell; (6) 6% reduction of cask radial copper fin thickness; (7) 10% increase of the lead emissivity; and (8) The effective thermal conductivity of the fuel tubes elements in the 3-D model (at the contact region only, i.e. the side of the fuel in contact with the tube) changed to the thermal conductivity of stainless steel (to simulate the contact between the fuel assembly and the fuel tube and between the fuel tube and the disks). Note that there is no reduction in the heat transfer coefficient at cask surface for this sensitivity study. The analysis results indicate that the maximum temperature of the neutron shield is 289°F, which is below the allowable temperature of 300°F. This demonstrates that the maximum temperature in the neutron shield will be less than the allowable temperature even when the sensitivities are considered and that the maximum temperature in the neutron shield reported in Table 3.4-1 is conservative.

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CHAPTER 3: THERMAL EVALUATION

3-5 NAC is requested to provide clarification of the preferential loading requirements/configurations for the Maine Yankee (MY) site-specific fuel in the UMS Transport cask. Specifically, SAR Section 3.6.1, Maine Yankee Site-Specific Contents, should be revised to include a clarification of the preferential loading requirements for transport, similar to SAR Section 3.4.2.1. The clarification should address the relationship between the preferential loading of the MY site-specific fuel for storage in the UMS Storage System versus the preferential loading of the MY site-specific fuel for storage in the UMS Transport Cask.

NAC Response:

Fuel clad allowable temperatures limit transport and storage maximum allowable heat loads, either on a total cask basis or on a per assembly basis. UMS safety evaluations set the allowable cladding temperatures by using the Commercial Spent Fuel Management (CSFM) method as indicated in Section 3.4.6 of the SAR. This approach is based on uniform, also referred to as standard, basket loadings and results in a decrease in the maximum allowable clad temperature, and therefore allowable heat load, as cool time and/or burnup increases. Chapter 5 of the SAR employs the varying allowable heat loads in combination with shielding dose rate limits to determine minimum allowable cool times for each fuel assembly type. Minimum allowable cool times are established as a function of minimum initial enrichment and maximum burnup. A generic approach of limiting each assembly in the basket to the CSFM determined maximum allowable assembly heat load is applied to each PWR and BWR assembly category and the CE 14x14 Maine Yankee specific fuel type.

NAC INTERNATIONAL RESPONSE
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NAC Response to 3-5 continued

To provide increased flexibility in loading the MY specific fuel inventory as early as possible after reactor shutdown, an additional evaluation is performed in the storage application. The additional evaluations allow loading of higher heat loads (shorter cooled fuel than the standard, uniform pattern) into the periphery of the basket while lowering the allowable heat loads (longer cooled fuel than the standard, uniform pattern) for the canister interior assemblies, as shown in the Approved Contents and Design Features for the NAC-UMS® System in Chapter 12 Section B 2.1.3 of the Final Safety Analysis Report (FSAR) for the UMS® Universal Storage System, Docket Number 72-1015. The transport thermal evaluation presented in Chapter 3 and the shielding evaluations presented in Chapter 5 of the SAR are based on uniform assembly heat loadings and do not apply to higher heat load peripheral loadings. Not including the peripheral higher heat load patterns requires all canisters loaded under this option to be stored until all fuel assemblies in the canister meet the minimum cool time for transport. While interior fuel assemblies in the high heat load preferential loading configuration may already be at an acceptable transport cool time when placed into storage, the high heat load peripheral assemblies require additional cool time to meet transport cask allowable heat load and dose limits. The standard pattern for transport requires an increase in cool time over the standard pattern requested in storage for all burnup and enrichment combinations.

Preferential loading of the Maine Yankee site-specific fuel assemblies is governed by the standard fuel inventory requirement presented in the Approved Contents and Design Features for the NAC-UMS® System in Chapter 12 of the Final Safety Analysis Report (FSAR) for the UMS® Universal Storage System, Docket Number 72-1015. Loading fuel assemblies for storage with a cool time of less than 7 years requires a preferential loading arrangement with shorter-cooled fuel placed at the canister interior locations. The corresponding thermal evaluation for the transport system is shown in Section 3.4.2.1. Maine Yankee site-specific preferential loading patterns placing high heat load (1.05 or 0.958 kW) fuel in basket peripheral locations, as allowed in Chapter 12 of the UMS® Universal Storage System FSAR, are not applicable for the transport

NAC INTERNATIONAL RESPONSE TO REQUEST FOR SUPPLEMENTAL INFORMATION

NAC Response to 3-5 continued

system. All fuel assemblies loaded in the transport cask must meet the standard configuration transport minimum allowable heat load limits and cool time tables. As such, a transportable storage canister loaded under the Maine Yankee site-specific high heat load preferential loading option will require additional cool time for the peripheral assemblies to meet the transport cask cool time requirements shown in Table 1.3.1-2 (also Table 5.5.1.1-10). This assures that a loaded canister will meet all thermal and shielding limits for transport.

The previous paragraph is added in Section 3.6.1 of the SAR for clarification of the preferential loading requirements for transport.

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CHAPTER 5: SHIELDING EVALUATION

5-1 NAC is requested to provide SAR section 5.5.1.2 pages that are missing from the current version of the SAR.

NAC Response:

SAR Section 5.5.1.2 pages 5.5.1-36 through 5.5.1-41, changed pages for the Master Table of Contents, and an updated List of Effective Pages are included with this response.

EA790-SAR-001

Docket No. 71-9270

UMS®

UNIVERSAL MPC SYSTEM®

SAFETY ANALYSIS REPORT

for the

UMS® Universal Transport Cask

OCTOBER 2002 UMST-02E

VOLUME 1 OF 2

NAC INTERNATIONAL

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2.7-7	Revision UMST-00A
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2.7-10	Revision UMST-00A
2.7-11	Revision 0
2.7-12	Revision 0
2.7-13	Revision 0
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2.7-17	Revision 0
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5.2-16	Revision UMST-01A
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6.1-3	Revision UMST-00A
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6.3-11	Revision 0
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6.3-13	Revision 0
6.3-14	Revision 0
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6.4-35	Revision UMST-01D
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6.4-38	Revision UMST-01D
6.4-39	Revision UMST-01D
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6.5-2	Revision 0
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6.5-6	Revision UMST-99A
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SAFETY ANALYSIS REPORT

for the

UMS® Universal Transport Cask

OCTOBER 2002 UMST-02E

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the corresponding total heat is only 15.7 kW and the heat density is 88% of the 20 kW over 144 inches. The 20kW over 144 inches is considered to be controlling.

There are a number of conservative conditions in this three-dimensional cask model:

1. The fuel assembly is conservatively considered to be located at the center of the fuel tube. (The fuel assembly will be in contact with the fuel tube on its side since the cask is in the horizontal position during transport. The contact will reduce the maximum component temperature.)
2. The fuel tube is conservatively considered to be located at the center of the slots of the support disks. (The fuel tube will be in contact with the support disk since the cask is in the horizontal position during transport. The contact will reduce the maximum component temperature.)
3. Convection heat transfer is conservatively ignored in the model.
4. The gap between the lead and cask inner shell is conservatively considered to 360° around the shell. (A good portion of the lead will be in contact with the inner shell since the cask is in the horizontal position during transport. The contact will reduce the maximum component temperature.)

A sensitivity study was performed to assess the effect on maximum temperatures of fuel cladding and basket for the variations of the following parameters, which are considered to be critical in the main heat transfer path:

1. Emissivity of stainless steel (fuel tube, support disks, canister shell, cask shells) and aluminum (heat transfer disk)
2. Convection heat transfer coefficient at transport cask outer surface
3. Contact area between the disks and canister shell
4. Heat transfer disk thickness
5. Gap between the disks and canister shell
6. Gap between the canister shell and the cask inner shell
7. Cask radial neutron shield copper fin thickness
8. Emissivity of copper lead

A total of eight (8) thermal analyses were performed using the thermal model described in this section. The changes in the model and the temperature results for the eight cases are shown in the table following. The analysis results indicate that the increase in the maximum fuel cladding

and basket temperature is ≤ 8°F for each of the cases. Therefore, the effect of variation of all these parameters is not significant. Additionally, an analysis is performed (Case 9) to evaluate the combined effect of Cases 1, 3, 4, 5, 7, and 8. The increase in the maximum fuel cladding and basket temperature for the combined case (no. 9) is ≤ 17°F. The maximum fuel cladding and basket temperatures remain below their allowable temperatures for the combined case. Based on the above discussion on the conservatism in the model and the results of the sensitivity study, it is concluded that the calculated temperatures using the thermal models are conservative and the system has an adequate margin of safety.

Case No.	Description	Maximum Temperature (°F)		
		Fuel Cladding	Support Disks	Heat Transfer Disks
Base	Original analysis	673	608	605
1	10% reduction of the emissivity of stainless steel and aluminum	678	613	610
2	10% reduction of the heat transfer coefficient at cask outer surface	678	614	610
3	Reduced contact area between disks and canister shell (reduced from 2° to 1° in the half-symmetry model)	673	608	605
4	8% reduction in heat transfer disk thickness based on the plate thickness tolerance	680	616	613
5	Increased gap between disks and canister shell based on the tolerance of the diameter of disks and canister shell and the canister shell thickness	676	611	608
6	Increased gap between canister shell and cask inner shell based on the tolerance of the diameter of canister shell and the cask inner shell	675	610	607
7	6% reduction of the cask radial neutron shield copper fin thickness based on the plate thickness tolerance	674	609	606
8	10% reduction of the lead emissivity	673	608	605
9	Combined (1+3+4+5+7+8)	689	625	622

The maximum temperature for the radial neutron shield is 293°F as reported in Table 3.4-1. To ensure that sufficient margin exist for the neutron shield, an additional sensitivity study was performed using the model for the nine cases evaluated for the fuel and basket components. The analysis considers a combination of the following conditions: (1) 10% increase of the emissivity of stainless steel and aluminum; (2) Increased contact area between disks and canister shell (from 2° to 4° in the half-symmetry model); (3) 8% increase in heat transfer disk thickness; (4) Decreased gap between disks and canister shell (5) Decreased gap between canister shell and cask inner shell; (6) 6% reduction of cask radial copper fin thickness; (7) 10% increase of the lead emissivity; and (8) The effective thermal conductivity of the fuel tubes elements in the 3-D model (at the contact region only, i.e. the side of the fuel in contact with the tube) changed to the thermal conductivity of stainless steel (to simulate the contact between the fuel assembly and the fuel tube and between the fuel tube and the disks). Note that there is no reduction in the heat transfer coefficient at cask surface for this sensitivity study. The analysis results indicate that the maximum temperature of the neutron shield is 289°F, which is below the allowable temperature of 300°F. This demonstrates that the maximum temperature in the neutron shield will be less than the allowable temperature even when the sensitivities are considered and that the maximum temperature in the neutron shield reported in Table 3.4-1 is conservative.

3.4.1.1.2 Two-Dimensional Fuel Assembly Model: PWR Fuel

The effective conductivity of the fuel is determined by a detailed two-dimensional finite element thermal model of the PWR 14x14 fuel assembly. Taking advantage of the symmetry of the cross-section of the fuel, the finite element model represents a one-quarter section of the fuel. The model includes the fuel pellets, cladding, gas between the 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. The model is shown in Figure 3.4-3. Thermal analyses of the other PWR fuel assemblies (i.e., 17x17, 16x16, and 15x15) are performed; however, because the PWR 14x14 fuel assembly results in the lowest effective thermal conductivities, only the analysis of that fuel assembly is presented in this section.

ANSYS PLANE 55 conduction elements and LINK31 radiation elements are used in the model, which includes a total of 49 fuel rods (representing a total of 196 fuel rods for the full cross-section). Each fuel rod consists of the pellet, Zircaloy cladding, and a gap between the pellet and clad. The gas in the gap between the pellet and clad, as well as the gas between the fuel rods, is modeled as helium. Radiation elements are defined between rods and from rods to the boundary of the model (inside surface of the fuel tube). Radiation across the gap between the pellet and clad is conservatively ignored. Effective emissivities are determined by using the formula shown in Section 3.4.1.1.1.

The effective conductivity for the fuel is determined by using a two-step procedure. Using the fuel assembly model, a uniform temperature is applied to the exterior of the model (see Figure 3.4-3) in conjunction with the volumetric heat generation. From this analysis, the maximum temperature located at the center of the fuel assembly is determined. This maximum temperature occurs at the corner of the model, which represents the center of the entire fuel assembly.

A Sandia National Laboratory Report [10] defines an expression for use in determining the maximum temperature of a square cross section of an isotropic homogeneous fuel with uniform volumetric heat generation. At the boundary of this square cross section, the temperature is constrained to be uniform. The expression for the maximum temperature is given by:

$$T_c = T_e + 0.29468 \frac{Qa^2}{K_{eff}} 1$$

where:

T_c = temperature at center of fuel (°F)
T_e = temperature applied at exterior of fuel (°F)
Q = volumetric heat generation rate (Btu/hr-in^3)
a = half-length of square cross section of fuel (inch)
K_{eff} = effective thermal conductivity for isotropic homogeneous fuel material (Btu/hr-in-°F).

Using the maximum temperature, located at the center of the fuel, from the detailed fuel assembly model, the preceding expression is used to determine the K_{eff} for an isotropic homogeneous representation of the fuel assembly.

Volumetric heat generation based on the design heat load of 20 kW with a peaking factor of 1.1 is applied to the fuel pellets. The temperature at the boundary of the model is constrained to be uniform. The effective conductivity is determined on the basis of the heat generated and the temperature difference from the center of the model to its edge. The temperature-dependent effective properties are established by using different boundary temperatures. The effective conductivity in the axial direction of the fuel assembly is calculated on the basis of a weighted average of the axial cross sectional area.

3.4.1.1.3 <u>Two-Dimensional Fuel Tube Model: PWR Fuel</u>

The effective conductivity of the fuel tube and BORAL plate, which is used in the three-dimensional canister model, is determined by the two-dimensional fuel tube model. As shown in Figure 3.4-4, this model includes the fuel tube, the BORAL plate (including the core matrix sandwiched by aluminum claddings), [redacted] gaps on both sides of the BORAL plate, and a [redacted] gap between the stainless steel cladding for the BORAL plate and the support disk or heat transfer disk. The BORAL plate in the PWR fuel tube is composed of 62.34% B_4C and 37.66% aluminum.

ANSYS PLANE55 conduction elements and LINK31 radiation elements are used to construct the model, which consists of eight layers of conduction elements and six radiation elements that are defined at the [redacted] gaps (two per gap). The thickness of the model (x-direction) is the distance measured from the inside dimension of the fuel tube to the inside dimension of the slot in the support disk (assuming that the fuel tube is located at the center of the disk slot). The tolerance of the BORAL plate core thickness, 0.003 inch, is used as the gap size for both sides of the BORAL plate. The model height is defined to be the same dimension as the model thickness.

A heat flux is applied at the left side of the model and the temperature at the right boundary of the model is constrained. The heat flux is determined on the basis of design heat load of 20 kW with a peaking factor of 1.1. The maximum temperature of the model (at the left boundary where the heat flux is applied) is calculated by using ANSYS. The effective conductivity through the thickness of the tube is determined by using the following equation:

$$q = K_{eff}(A/L)\,\Delta T, \text{ or}$$
$$K_{eff} = qL/(A\,\Delta T).$$

where:

q = heat rate applied to inner surface of fuel tube (Btu/hr)
A = area (in^2)
L = thickness of composite tube model (in)
ΔT = temperature difference across the model (°F)
K_{eff} = effective conductivity (Btu/hr-in-°F).

The temperature-dependent conductivity for heat conduction through the wall (K_{eff}) is determined by varying the temperature constraint at the boundary of the model and then re-solving for the temperature difference. The effective conductivity for heat conduction parallel to the axis of the cask body or in the plane of the tube wall is calculated on the basis of the weighted average of the thickness and conductivity of the individual layers.

3.4.1.2 Analytical Models: Cask with BWR Fuel Canister

The finite element ANSYS models used in the thermal analysis of the cask transporting BWR fuel are similar to those used in the thermal analysis of the cask with PWR fuel canister discussed in previous sections. A three-dimensional model is employed to evaluate the cask in a horizontal position with the basket in contact with the canister, which, in turn, is in contact with the cask inner shell. The fuel regions and the fuel tubes with BORAL plates are modeled by using effective conductivities. A detailed two-dimensional thermal model of the fuel assembly is used to determine the effective conductivity of the fuel. A two-dimensional thermal model of the fuel tube is used to calculate the effective conductivities of the fuel tube wall and BORAL plate. Another two-dimensional thermal model for the fuel tube is used to calculate the effective conductivity of the fuel tube wall with no BORAL plate present. These four ANSYS thermal models are described in the following sections.

3.4.1.2.1 Three-Dimensional Cask Model: Cask with BWR Fuel Canister

The three dimensional Universal Transport Cask model is a half-symmetry finite element model constructed by using ANSYS Revision 5.5. The model considers the fuel assemblies, fuel tubes, stainless steel support disks, aluminum heat transfer disks, canister shell, lids and bottom plate, spacers at the bottom of the canister, cask inner shell, lead, outer shell, neutron shield, and neutron shield shell. The ANSYS model is shown in Figure 3.4-5. As shown in the figure, the internal cavity of the canister contains the active fuel region: the top and bottom fittings of the

fuel assemblies, fuel tubes enclosing the top and bottom fittings, and the first stainless steel support.

For the BWR configuration, the gas inside the canister and the cask cavity is modeled as helium because the cavity will be backfilled with helium prior to transport. Conduction and radiation are modeled by using ANSYS "SOLID70" and "LINK31" elements, respectively. The principal gaps applied to the model are shown in Figure 3.1-2 and are described in Section 3.2.2.3. In establishing these gaps, the differential thermal expansion between the components is considered.

Because the canister is in horizontal position during transport, the elements for the canister shell are shifted downwards to simulate contact with the inner shell of the cask. Similarly, the support disks and the heat transfer disks are shifted downward to simulate contact with the canister shell. As shown in Figure 3.1-2, a 2-degree contact is considered for the gaps between the canister shell and the cask inner shell and between the support disk and the canister shell. This contact is simulated by using appropriate conductivity (100 Btu/hr-inch-°F) for elements at the contact locations. The aluminum heat transfer disks are assumed to have only a line contact with the canister shell because the heat transfer disks are not subjected to any loads other than their own weight.

To account for differential expansion, gaps within the model are adjusted on the basis of temperature and defined physical contact conditions. Solar insolance and ambient temperature conditions are applied to the neutron shield shell when appropriate. Insolance is used at the exterior surface of the cask and is based on the amount of insolation required by 10 CFR 71 to be applied over a 12-hr period evaluated in the steady state (applied over 24 hr simulating 12-hr period of solar exposure and 12-hr period of no solar exposure). The heat flux resulting from insolation on a curved surface is calculated as follows:

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

Multiplying this value by the emissivity of the cask surface, $\in = 0.36$, gives a heat flux resulting from insolance on curved surfaces of 0.154 Btu/hr-in^2. Using the same method and a heat flux of

2,950 Btu/12 hr-ft^2 (0.853 Btu/hr-in^2), gives a heat flux resulting from insolance on flat surfaces of 0.307 Btu/hr-in^2.

The model is analyzed to determine the maximum temperatures for the basket, canister, cask shells, radial shielding, and surface conditions under normal conditions of transport. All material properties are shown in Tables 3.2-1 through 3.2-13.

The fuel regions (inside tubes) are modeled as homogeneous regions with effective conductivities determined by the two dimensional fuel model as described in Section 3.4.1.2.2. All sides of the BWR fuel tubes do not contain the BORAL plate. Therefore, two different two-dimensional BWR fuel tube models are analyzed to establish the effective conductivities used in the three dimensional analysis of the cask with BWR fuel. The models consist of the BORAL plate (where applicable), including gas gaps on both sides of the BORAL sheet (where applicable), and the gap between the stainless steel cladding for the BORAL and the support disks and heat transfer disks. These models are discussed in Section 3.4.1.2.3.

The radial neutron shield of the transport cask for the BWR configuration is identical to PWR configuration. The modeling of the radial neutron shield is described in Section 3.4.1.1.

In the model, radiation heat transfer is considered from the top of the fuel region to the bottom surface of the canister shield lid, from the bottom of the fuel region to the top surface of the canister bottom plate, and from the exterior surfaces of the fuel tubes to the inner surface of the canister shell. This radiation is modeled by using LINK31 radiation elements. Radiation across gaps in the model is described in Sections 3.2.2.3 and 3.2.2.4.

Radiation at the neutron shield shell surface to ambient is combined with the convection effect by using the method described in Section 3.2.2.2. The convection heat transfer coefficient is calculated on the basis of the formula shown in Section 3.2.3. Effective emissivities are used for all radiation calculations, with the form factor taken to be unity. Effective emissivity is computed by using the following formula [9] based on corresponding material emissivities:

$$\varepsilon_{eff} = 1 / (1/\varepsilon_1 + 1/\varepsilon_2 - 1)$$

Solar insolance is applied to the neutron shield shell surface for the "Hot" condition (ambient temperature = 100°F). A value of 0.154 Btu/hr-inch^2 is used as the heat flux at the neutron shield shell surface on the basis of the 1,475 Btu/hr-ft^2 heat flux for a curved surface. Calculation of the heat flux resulting from insolation on a curved surface is discussed earlier in this section.

Volumetric heat generation (Btu/hr-inch^3) is applied to the active fuel region on the basis of a total heat load of 16 kW, a shortest active fuel rod length of 144 inches, and an axial power with a peaking factor of 1.22 as shown in Figure 3.4-6.

3.4.1.2.2 Two-Dimensional Fuel Assembly Model: BWR Fuel

The effective conductivity of the fuel is determined by a detailed two-dimensional finite element thermal model of the BWR 9x9 fuel assembly. Taking advantage of the symmetry of the cross-section of the fuel, the finite element model represents a one-quarter section of the fuel. The model includes the fuel pellets, cladding, gas between the 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. The model is shown in Figure 3.4-7. Thermal analyses of the other BWR fuel assemblies (i.e., 7x7 and 8x8) are performed; however, because the BWR 9x9 fuel assembly results in the lowest effective thermal conductivities, only the analysis of that fuel assembly is presented in this section.

ANSYS PLANE55 conduction elements and LINK31 radiation elements are used in the model, which includes a total of 20.25 fuel rods (representing a total of 81 fuel rods for the full cross-section). Each fuel rod consists of the pellet, Zircaloy cladding, and a gap between the pellet and clad. The gas in the gap between the pellet and clad, as well as the gas between the fuel rods, is modeled as helium. Radiation elements are defined between rods and from rods to the boundary of the model (inside surface of the fuel tube). Radiation effect at the gaps between the pellet and clad is conservatively ignored. Effective emissivities are determined by using the formula shown in Section 3.4.1.1.1.

The effective conductivity for the fuel is determined by using a two-step procedure. Using the fuel assembly model, a uniform temperature is applied to the exterior of the model (see Figure 3.4-7) in conjunction with the volumetric heat generation. From this analysis, the maximum

temperature located at the center of the fuel assembly is determined. This maximum temperature occurs at the corner of the model, which represents the center of the entire fuel assembly.

A Sandia National Laboratory Report [10] defines an expression for use in determining the maximum temperature of a square cross section of an isotropic homogeneous fuel with uniform volumetric heat generation. At the boundary of this square cross section, the temperature is constrained to be uniform. The expression for the maximum temperature is given by:

$$T_c = T_e + 0.29468 \frac{Q a^2}{K_{eff}}$$

where:

T_c = temperature at center of fuel (°F)
T_e = temperature applied at exterior of fuel (°F)
Q = volumetric heat generation rate (Btu/hr-in^3)
a = half-length of square cross section of fuel (inch)
K_{eff} = effective thermal conductivity for isotropic homogeneous fuel material (Btu/hr-in-°F).

Using the maximum temperature, located at the center of the fuel, from the detailed fuel assembly model, the preceding expression is used to determine the K_{eff} for an isotropic homogeneous representation of the fuel assembly.

Volumetric heat generation based on the design heat load of 16 kW with a peaking factor of 1.22 is applied to the fuel pellets. The temperature at the boundary of the model is constrained to be uniform. The effective conductivity is determined on the basis of the heat generated and the temperature difference from the center of the model to its edge. The temperature-dependent effective properties are established by using different boundary temperatures. The effective conductivity in the axial direction of the fuel assembly is calculated on the basis of the material area ratio.

3.4.1.2.3 Two-Dimensional Fuel Tube Models: BWR Fuel

The fuel tubes in the BWR fuel basket differ from those in the PWR fuel basket in that not all sides of the fuel tubes contain BORAL. Therefore, two effective conductivity models are

necessary—one fuel tube model with the BORAL plate (a total of 10 layers of materials) and another fuel tube model with a gas gap replacing the BORAL plate (a total of 4 layers of materials). Additionally, the BORAL plate in the BWR fuel tube is composed of 16.46% B_4C and 83.54% aluminum, whereas the BORAL plate in the PWR fuel tube is composed of a 62.34%—37.66% composition of B_4C and aluminum.

The effective conductivity of the fuel tube and BORAL plate, which is used in the three-dimensional canister model, is determined by a two-dimensional fuel tube model. As shown in Figure 3.4-8, this model includes the fuel channel, gas gaps between the fuel channel and fuel tube. the fuel tube, the BORAL plate (including the core matrix sandwiched by aluminum claddings), gas gaps on both sides of the BORAL plate, and a gas gap between the stainless steel cladding for the BORAL plate and the support disk or heat transfer disk.

Additionally, the effective conductivity of the fuel tube without the BORAL plate, which is used in the three-dimensional canister model, is determined by another two-dimensional fuel tube model. As shown in Figure 3.4-9, this model includes the fuel channel, gas gaps between the fuel channel and stainless steel fuel tube, the fuel tube, and a gas gap between the stainless steel cladding and the support disk or heat transfer disk.

ANSYS PLANE55 conduction elements and LINK31 radiation elements are used to construct the models. The model with the BORAL plate consists of 10 layers of conduction elements and 8 radiation elements that are defined at the gas gaps (two per gap). The model without the BORAL plate consists of four layers of conduction elements and four radiation elements that are defined at the gas gaps (two per gap). The thickness of the models (x-direction) is the distance measured from the inside dimension of the fuel channel to the inside dimension of the slot in the support disk (assuming that the fuel tube is located at the center of the disk slot). In the model containing the BORAL plate, the tolerance of the BORAL plate core thickness, 0.0045 inch, is used as the gap size for both sides of the BORAL plate. The height of the models is defined to be the same dimension as the thickness of the models.

In each analysis, a heat flux is applied at the left side of the model and the temperature at the right boundary of the model is constrained. The heat flux is determined on the basis of the design heat load of 16 kW with a peaking factor of 1.22. The maximum temperature of the

model (at the left boundary) and the temperature difference (ΔT) across the model are calculated by using ANSYS. The effective conductivity is determined by using the following formula:

$$q = K_{eff}(A/L)\ \Delta T$$

or

$$K_{eff} = qL/(A\ \Delta T)$$

where:

- q = heat rate applied to inner surface of fuel tube (Btu/hr)
- A = area (in^2)
- L = thickness of composite tube model (in)
- ΔT = temperature difference across the model (°F)
- K_{eff} = effective conductivity (Btu/hr-in-°F).

The temperature-dependent conductivity (K_{eff}) in each analysis is determined by varying the temperature constraint at the boundary of the model and then re-solving for the temperature difference. The effective conductivity for the parallel path is calculated on the basis of area ratio of material.

3.4.1.3 Cask Impact Limiter Thermal Model

As described in Sections 3.4.1.1 and 3.4.1.2, the cask impact limiters are not explicitly modeled in the 3D cask models. In these models, the cask ends enclosed by the impact limiters are modeled as being adiabatic surfaces. The cask impact limiters are evaluated thermally for normal operating conditions in this section. Specifically, the volumetric average temperature of the redwood material in the cask impact limiters is calculated using an ANSYS finite element model. Taking advantage of the symmetrical geometry of the cask impact limiters about the major axis of the cask, the finite element model is an axisymmetric representation of one of the impact limiters with the cask oriented in a horizontal position. This represents the orientation of the impact limiters during normal transport. The cask impact limiter thermal model is shown in Figure 3.4-10.

The finite element model of the cask impact limiter is constructed of PLANE55 axisymmetric thermal elements, and radiation and conduction heat transfer across air gaps within the model are accounted for using effective thermal conductivity properties for air using the method described in Section 3.2.2.3. Air gaps are modeled between the cask and impact limiter based upon

nominal dimensions. Additionally, a 0.125-in. thick layer of Fiberfrax® Ceramic Fiber Paper is modeled between the impact limiter redwood and the cask mating surface of the impact limiter. A heat flux of 0.13 Btu/h-in², which represents the package contents, is applied to the interior surface of the cask lid. This heat flux is obtained from the thermal results for the 3D cask model with the PWR canister and air as the canister cover gas (described in Section 3.4.1.1) by conservatively assuming the heat transfer rate to the cask lid is equal to the heat transfer rate to the canister shield lid.

Heat fluxes representing the normal conditions solar heat loads are applied to the cylindrical and vertical flat end surfaces of the impact limiter as shown in Figure 3.4-10. The solar heat flux applied to the vertical flat surfaces of the impact limiter 0.0769 Btu/h-in² model (which is in the normal transport orientation) are calculated in the same manner described in Section 3.4.1.1.1 using the prescribed solar heat flux value of 737 Btu/12-hr-ft². A solar heat flux of 0.154 Btu/hr-in² is applied to the cylindrical portions of the cask and impact limiter modeled.

A steady-state heat transfer analysis is performed using the ANSYS model described in this Section. The volumetric average temperature of the cask impact limiter redwood material (T_{avg}) is calculated from the results of the thermal steady state analysis.

3.4.1.4 Personnel Barrier Thermal Model

According to 10 CFR 71.43(g), a package must be designed, constructed, and prepared for transport such that in still air at 100°F and shade, no accessible surface of the package has a temperature exceeding 185°F in an exclusive use shipment. Compliance with 10 CFR 71.43(g) is demonstrated by performing a computational fluid dynamics (CFD) analysis on a finite element model of the air between the cask surface (i.e., neutron shield shell) and the personnel barrier using ANSYS/FLOTRAN. The finite element model is constructed of two-dimensional FLUID141 elements and is presented in Figure 3.4-11.

Because of geometrical symmetry, only one-half of the cask and the air around the cask is modeled. In addition to the natural convection of the air, thermal radiation heat transfer from the cask outer surface to the personnel barrier is considered in this model. It is conservative to only model the air between the cask surface and the personnel barrier because it results in a higher air velocity and more heat is carried to the top of the personnel barrier. Along the centerline of the model, the horizontal velocity component is specified to be zero. The nodes at the location of the

personnel barrier (except the top side) are conservatively defined as wall conditions (Velocity = 0) to force all of the heat out from the top of the barrier. At the inlet (bottom side of the model), the pressure is set to atmospheric pressure with the temperature constrained to 100°F. The portion of the model corresponding to the cask surface constrains both the horizontal and vertical components of the velocity to be zero.

The cask and personnel barrier are not explicitly modeled in this analysis—only the air surrounding the cask is modeled. It is conservative that the personnel barrier is not explicitly modeled because it will not have a temperature greater than the temperature of the air in contact with it. The temperatures of nodes in the model that correspond to the air adjacent to the cask surface are constrained as boundary conditions of the model. The temperature is considered to be linearly distributed, with the bottom and top temperatures equal to 267°F and 244°F, respectively.

Since the personnel barrier is not explicitly modeled, its temperature is considered to be the temperature of the air at coordinates that correspond the location of the personnel barrier surface. The maximum temperature of the personnel barrier occurs at the top most location at the centerline of the model. The temperatures at key points from the analysis using the model described above are shown in Figure 3.4-12.

3.4.1.5 Test Model

The methods previously described have been used in previous transport cask licensing and are sufficient to show that the Universal Transport Cask meets the criteria set forth in Section 3.4. Therefore, no thermal test model is created.

3.4.2 Maximum Temperatures

Using the thermal models described in Sections 3.4.1.1 and 3.4.1.2, temperatures for the PWR and BWR cask body, canister, basket, and fuel rod cladding are determined for three normal conditions of transport: (1) maximum decay heat, 100°F ambient temperature, and solar insolance; (2) maximum decay heat, -40°F ambient temperature, and no insolance; and (3) no decay heat, -40°F ambient temperature, and no insolance. The maximum temperatures of the principal PWR and BWR cask components, canister, basket components, and fuel rod cladding are shown in Tables 3.4-1 and 3.4-2 for the first two environmental conditions listed above. For the third environmental

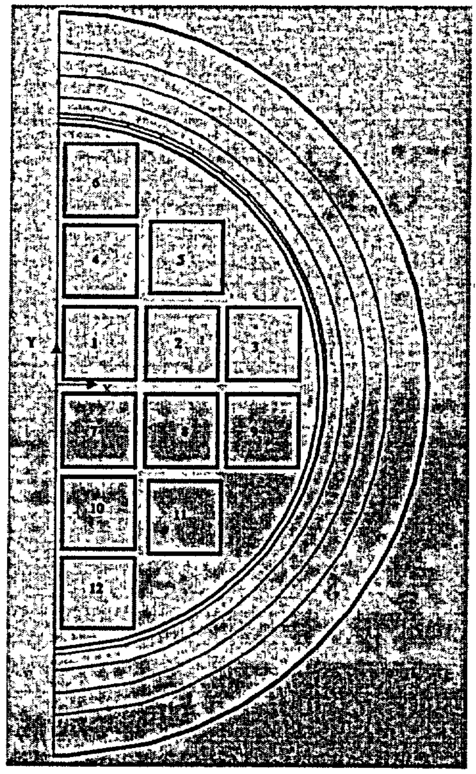
condition (i.e., no decay heat, -40°F ambient temperature, and no insolance), no analysis is necessary because all package temperatures will equilibrate to -40°F. The cask body maximum allowable component temperatures are shown in Section 3.3.2 and Table 3.4-3.

Using the thermal model described in Section 3.4.1.3, the volumetric average temperature of the redwood in the impact limiters is 135°F.

3.4.2.1 Preferential Loading

This section provides the evaluation of preferential fuel loading in accordance with Section B 2.1.2 of Chapter 12 of the Safety Analysis Report for the UMS® Universal Storage System, Docket Number 72-1015. As stated in Section 12B2.1.2, loading of the fuel assemblies designated for a given canister must be administratively controlled to ensure that the fuel cladding temperature limits are not exceeded for any fuel assembly, unless all of the designated fuel assemblies have a cooling time of 7 years or more. Fuel with the shortest cooling time (and, therefore, having a higher allowable cladding temperature) is placed in the center of the basket. Fuel with the longest cooling time (and, therefore, having a lower allowable cladding temperature) is placed in the periphery of the basket. Canisters containing fuel assemblies, all of which have a cooling time of 7 years, or more, do not require preferential loading.

For the transport conditions, three (3) thermal analyses have been performed using the three-dimensional cask model for the cask containing PWR fuel (Section 3.4.1.1.1). Note that the PWR configuration is selected since it is governing. The BWR configuration is considered to be bounded by the evaluation of the PWR configuration. The basket locations refer to those shown in the following figure.



For this analysis, heat load per assembly is based on Table 3.4-16: 5-year cooled fuel: 0.833 (20/24) kW; 6-year cooled fuel: 0.8125 (19.5/24) kW; 7-year cooled fuel: 0.742 (17.8/24) kW and 15-year cooled fuel: 0.70 (16.8/24) kW.

Case 1 considers the loading pattern having 5-year cooled fuel in the center of the basket (Locations 1 and 7), 7-year cooled fuel in the periphery positions (Location 3, 5, 6, 9, 11, and 12) and 6-year cooled fuel in the intermediate positions (Locations 2, 4, 8, and 10). The heat load for each fuel assembly is determined based on the maximum allowable heat load as shown in Table 3.4-16. The allowable temperatures are obtained from Table 3.4-15, based on the burnup and cool time corresponding to the analyzed heat load. The calculated maximum temperature at each fuel position and the corresponding allowable temperature is:

Basket Location	1	2	3	4	5	6	7	8	9	10	11	12
Fuel Cool Time (Years)	5	6	7	6	7	7	5	6	7	6	7	7
T_{max} (°F)	654	613	544	628	573	571	636	594	524	575	518	485
$T_{allowable}$ (°F)	705	693	653	693	653	653	705	693	653	693	653	653

Case 2 considers the loading pattern having 5-year cooled fuel in the center of the basket (Locations 1 and 7) and 7-year cooled fuel in all other basket positions (Locations 2 to 6 and 8 to

12). The calculated maximum temperature at each position and the corresponding allowable temperature are:

Basket Location	1	2	3	4	5	6	7	8	9	10	11	12
Fuel Cool Time (Years)	5	7	7	7	7	7	5	7	7	7	7	7
T_{max} (°F)	642	595	536	609	564	562	624	576	516	557	510	478
$T_{allowable}$ (°F)	705	653	653	653	653	653	705	653	653	653	653	653

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

Basket Location	1	2	3	4	5	6	7	8	9	10	11	12
Fuel Cool Time (Years)	15	7	7	7	7	7	15	7	7	7	7	7
T_{max} (°F)	610	583	528	596	555	553	593	564	508	546	502	471
$T_{allowable}$ (°F)	631	653	653	653	653	653	631	653	653	653	653	653

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

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

The preferential loading does not result in any slot containing fuel with a heat load greater than 0.833 (20/24) kW. As summarized for the above cases, the maximum fuel cladding temperature in Case 1 is bounded by the maximum fuel cladding temperature in Table 3.4-1. Since the total heat for the uniform heat load case bounds the three cases of preferential loading, the component temperatures provided in Table 3.4-1 bound the component temperatures resulting from the preferential loading cases.

3.4.3 Minimum Temperatures

The minimum temperatures of the cask and components occur with no heat load and -40°F. These conditions yield a uniform -40°F temperature throughout the Universal Transport Cask package. All package components are capable.

3.4.4 Maximum Internal Pressures

In the following sections, the maximum internal operating pressures for normal conditions of transport are calculated for the PWR and BWR Transportable Storage Canisters and for the Universal Transport Cask cavity. The maximum operating pressure for the canister and cask cavity are summarized in Table 3.4-4.

3.4.4.1 Maximum Internal Pressure for PWR Fuel Canister and Transport Cask

The internal pressures within the PWR fuel canister and transport cask are a function of fuel type, fuel condition (failure fraction), burnup, canister type, and the backfill gases in the canister and cask cavity. Gases included in the pressure evaluation include rod-fill, rod fission and rod backfill gases, canister and cask backfill gases, and burnable poison generated gases. Each of the fuel types expected to be loaded into the UMS® system is separately evaluated to arrive at a bounding canister pressure.

Fission gases include all fuel material generated gases including long-term actinide decay generated helium. Based on detailed SAS2H calculations of the maximum fissile material mass assemblies in each canister class, the quantity of gas generated by the fuel rods rises as burnup and cool time is increased and enrichment is decreased. To assure the maximum gas is available for release, the PWR inventories are extracted from conservatively high 60,000 MWD/MTU burnup cases at an enrichment of 1.9 wt. % ^{235}U and a cool time of 40 years. Gases included are all krypton, iodine, and xenon isotopes in addition to helium and tritium (^{3}H). Molar quantities for each of the maximum fissile mass assemblies are summarized in Table 3.4-5. Fuel generated gases are scaled by fissile mass to arrive at molar contents of other UMS® fuel types.

Fuel rod backfill pressure varies significantly between the PWR fuel types. The maximum reported backfill pressure is listed for the Westinghouse 17x17 fuel assembly at 500 psig. With the exception of the B&W fuel assemblies, which are limited to 435 psig, all fuel assemblies evaluated are set to the maximum 500 psig backfill reported for the Westinghouse assembly.

Backfill quantities are based on the free volume between the pellet and the clad and the plenum volume. The fuel rod backfill gas temperature is conservatively assumed to have an initial temperature of 68°F.

Burnable poison rod assemblies (BPRAs) placed within the UMS® canister may contribute additional molar gas quantities due to (n, alpha) reactions of fission generated neutrons with ^{10}B during in-core operation. ^{10}B forms the basis of a portion of the neutron poison population. Other neutron poisons, such as gadolinium and erbium, do not produce a significant amount of helium nuclides (alpha particles) as part of their activation chain. Primary BPRAs in existence include Westinghouse Pyrex (borosilicate glass) and WABA (wet annular burnable absorber) configurations, as well as B&W BPRAs and shim rods employed in CE cores. The CE shim rods replace standard fuel rods to form a complete assembly array. The quantity of helium available for release from the BPRAs is directly related to the initial boron content of the rods and the release fraction of gas from the matrix material in question. Release from either of the low temperature, solid matrix materials is likely to be limited, but no release fractions were available in open literature. Therefore, a 100% release fraction is assumed based on a boron content of 0.0063 g/cm ^{10}B per rod, with the maximum number of rods per assembly. The maximum number of rods is 16 for Westinghouse core 14 x 14 assemblies, 20 rods for Westinghouse and B&W 15 x 15 assemblies, and 24 rods for Westinghouse and B&W 17 x 17 assemblies. The length of the absorber is conservatively taken as the active fuel length. CE core shim rods are modeled at 0.0126 g/cm ^{10}B for 16, 12, and 12 rods applied to CE manufactured 14 x 14, 15 x 15 and 16 x 16 cores, respectively.

The canister backfill gases are conservatively assumed to be at 250°F, which is below the maximum canister shell temperature of 285°F after 9 hours of vacuum drying. The initial pressure of the canister backfill gas is 1 atm (0.0 psig). The cask backfill temperature and pressure are assumed to be 68°F and 1 atm. Free volume inside each PWR canister class is listed in Table 3.4-6. Also included are the total canister and cask free volumes. The listed free volumes do not include fuel assembly components since these components vary for each assembly type and fuel insert. By subtracting the rod and guide tube volumes and all hardware component volumes from the listed free volume, the free volume of the canisters including fuel assemblies and a load of 24 BPRAs can be determined. For the Westinghouse BPRAs, the Pyrex volume is employed since it displaces more volume than the WABA rods.

The total pressure for each of the UMS® payloads is found by calculating the releasable molar quantity of each gas (30% of the fission gas, 100% of the rod backfill, BPRA and shim rod gases adjusted for the 3% fuel failure fraction and the canister and cask backfill gases), and summing the quantities directly. The quantity of gas is then employed in the ideal gas equation in conjunction with the average gas temperature at normal operating conditions to arrive at system pressures. The normal condition average temperature of the gas within the PWR canister and cask is considered to be 453°F. Each of the UMS® PWR fuel types is individually evaluated for normal condition pressure, and the maximum normal condition canister and cask pressures are determined to be 6.15 psig and 6.91 psig, respectively. A summary of the maximum pressure in the canister and in the cask for each PWR canister class is shown in Table 3.4-7. The table also includes the fuel type producing the listed maximum pressures.

The maximum normal condition cask pressure for a PWR payload (West. 17 Std.) is calculated as follows:

$$P = \frac{nRT}{V}$$

$$n = (0.03 \times n_{rodbackfill}) + (0.03 \times 0.30 \times n_{fuelgas}) + (0.03 \times n_{BPRA}) + n_{canisterbackfill} + n_{caskbackfill}$$

$$n = (0.03 \times 134) + (0.03 \times 0.30 \times 965) + (0.03 \times 133) + 184 + 45 = 246 \text{ moles}$$

$$P = \frac{246\text{mol} \times 0.08205 \frac{\text{atm}-\text{l}}{\text{mol}-\text{K}} \times 510\text{K}}{6995\text{l}} = 1.47 \text{ atm} = 21.61 \text{ psia} = 6.91 \text{ psig}$$

Similar the maximum canister pressure for a PWR payload (B&W 17x17) transportable storage canister is:

$$n = (0.03 \times n_{rodbackfill}) + (0.03 \times 0.30 \times n_{fuelgas}) + (0.03 \times n_{BPRA}) + n_{canisterbackfill}$$

$$n = (0.03 \times 169) + (0.03 \times 0.30 \times 962) + (0.03 \times 132) + 195 = 213 \text{ moles}$$

$$P = \frac{213\text{mol} \times 0.08205 \frac{\text{atm}-\text{l}}{\text{mol}-\text{K}} \times 510\text{K}}{6275\text{l}} = 1.42 \text{ atm} = 20.8 \text{ psia} = 6.15 \text{ psig}$$

3.4.4.2 Maximum Internal Pressure for BWR Fuel Canister and Transport Cask

BWR canister and cask maximum pressures are determined in the same manner as those documented for the PWR cases. Primary differences between PWR and BWR analysis include a maximum normal condition average gas temperature of 366°F, rod backfill gas pressures of 132 psig, and pressurizing gases are limited to fission gases (including helium actinide decay gas),

rod backfill gases, and canister and cask backfill gas. The 132 psig employed in this analysis is significantly higher than the 6 atmosphere maximum pressure reported in open literature. BWR assemblies do not contain an equivalent to the PWR BPRAs and, therefore, do not require ^{10}B helium generated gases to be added. Fissile gas inventories for the maximum fissile material assemblies in each of the three BWR lattice configurations (7 x 7, 8 x 8, and 9 x 9) are shown in Table 3.4-8. Free volumes, without fuel components, in UMS® canister classes 4 and 5 are shown in Table 3.4-9. Cask and canister maximum pressures for each canister class are listed in Table 3.4-10. The maximum normal condition pressure of 3.47 psig is based on a GE 7 x 7 assembly designed for a BWR/2-3 reactor and burned to 60,000 MWD/MTU. Cask maximum pressure for the GE 7 x 7 fuel is 3.65 psig. High burnups, greater than 45,000 MWD/MTU, are typically obtained from updated assembly designs such as the GE 9 x 9 assembly. The normal condition pressure for a UMS® canister containing the GE 9 x 9 fuel assembly with 79 fuel rods is 3.33 psig. Similar fuel masses and displaced volume account for similar system pressures.

The maximum normal condition cask pressure for a BWR payload (GE 7x7 UMS Class 4) is calculated as follows:

$$P = \frac{nRT}{V}$$

$$n = (0.03 \times n_{rodbackfill}) + (0.03 \times 0.30 \times n_{fuelgas}) + n_{canisterbackfill} + n_{caskbackfill}$$

$$n = (0.03 \times 75) + (0.03 \times 0.30 \times 940) + 194 + 9 = 214 \text{ moles}$$

$$P = \frac{214 \text{mol} \times 460\text{K}}{6460\text{l}} \times 0.08205 \frac{\text{atm}-\text{l}}{\text{mol}-\text{K}} = 1.25 \text{ atm} = 18.35 \text{ psia} = 3.65 \text{ psig}$$

Similarly, the maximum canister pressure for a BWR payload (GE 7x7 UMS Class 4) transportable storage canister is:

$$n = (0.03 \times n_{rodbackfill}) + (0.03 \times 0.30 \times n_{fuelgas}) + n_{canisterbackfill}$$

$$n = (0.03 \times 75) + (0.03 \times 0.30 \times 940) + 194 = 205 \text{ moles}$$

$$P = \frac{205 \text{mol} \times 0.08205 \frac{\text{atm}-\text{l}}{\text{mol}-\text{K}} \times 460\text{K}}{6275\text{l}} = 1.24 \text{ atm} = 20.8 \text{ psia} = 3.47 \text{ psig}$$

3.4.5 Maximum Thermal Stresses

The ANSYS computer code is used to obtain temperatures for use in the structural analyses of Chapter 2.0. These temperatures are presented in Tables 3.4-1 and 3.4-2. The thermal stress calculations for normal conditions of transport are performed in Sections 2.6.1 and 2.6.2.

3.4.6 Maximum Allowable Cladding Temperature and Canister Heat Load

The maximum allowable cladding temperatures are calculated for PWR and BWR systems based on fuel assembly type, maximum burnup, and minimum initial cool time. Allowable heat loads are determined by relating cladding temperature to canister heat load.

Cladding stresses are calculated for a set of representative PWR and BWR assemblies at 40,000 MWD/MTU and 380°C. The limiting, highest stress assemblies, the Westinghouse 14x14 and GE 9x9 (150-inch fuel region), are then evaluated at various burnups to determine the maximum allowable fuel cladding temperature based on PNL-6364 criteria [28]. Maximum allowable cladding temperatures are generically calculated for PWR and BWR burnups ranging from 35,000 MWD/MTU to 45,000 MWD/MTU. PWR burnups are extended to 50,000 MWD/MTU to envelop the Maine Yankee specific inventory. After applying a bias to the maximum allowable cladding temperatures, the maximum allowable heat load is calculated as a function of burnup and minimum initial cool time.

3.4.6.1 Maximum Allowable Cladding Temperature

Based on PNL-6364, the cladding temperature limit is expressed as a function of initial dry storage temperature, initial cladding stress at the dry storage temperature, and initial storage time. For this evaluation, the transport temperatures and transport times are applied.

The initial cladding stress is a function of the rod internal pressure, temperature, diameter of the fuel rod, and fuel cladding thickness. The initial cladding stress (σ_{mhoop}) for a particular assembly is calculated as [28]:

$$\sigma_{mhoop} = \frac{(P)(D_{mid})}{2t} \times \alpha \times \frac{T_2}{T_1} \times \frac{69.684}{10,000}$$

where:

σ_{mhoop} = dry storage cladding hoop stress, MPa
P = internal gas pressure of the rod, psi
T_1 = temperature at which P was determined, K
t = cladding wall thickness, in.
D_{mid} = cladding midwall diameter, in.
α = a factor, 0.95 for PWR rods or 0.90 for BWR rods
T_2 = allowable storage temperature for σ_{mhoop}, K

To account for cladding oxidation during in-core fuel assembly operation and storage of the fuel in the spent fuel pool, the nominal cladding thickness is reduced by 0.06 mm and 0.125 mm for PWR and BWR fuel rods, respectively [3]. For higher burnup PWR fuels (i.e., rod peak burnup up to 50,000 MWD/MTU), Maine Yankee experience is that the maximum oxide layer thickness on the fuel cladding is 120 microns [30]. The allowable cladding temperature calculations at 50,000 MWD/MTU therefore employ an oxide layer thickness of 0.012 cm.

The pressure in the fuel assembly rods is produced by the combination of fill gas and fission gas. For a given fuel assembly design, the fill gas quantity is fixed and does not vary with discharge burnup. Based on the initial pressure and temperature of the fill gas, the number of moles of gas are calculated using the ideal gas law:

$$PV = NRT$$

where:

P = Pressure
V = Volume (free volume inside fuel rod)
N = Number of moles of gas
R = Universal gas constant
T = Temperature of the gas

The number of moles of fill gas are added to the fission gas quantity and converted to a cladding internal pressure at storage conditions.

The fission gas quantity pressurizing the fuel cladding is calculated on the basis of the burnup and a fission gas release fraction. While the amount of fission gas produced is a predictable

quantity (directly correlated to the number of fissions required to produce the desired burnup); the release fraction of the gas from the pellet into the pellet-cladding void depends on fill gas pressure and reactor operating conditions.

The number of fissions (Z) is related to the burnup by:

$$Z = X \text{ Burnup } \frac{\text{MWd}}{\text{MTU}} \times 1.0 \times 10^{6} \frac{\text{W}}{\text{MW}} \times 86{,}400 \frac{\text{sec}}{\text{d}} \times \frac{1 \text{ MeV}}{1.602 \times 10^{-13} \text{ J}} \times \frac{1 \text{ Fission}}{200 \text{ MeV}} \times \frac{1 \text{ Mole}}{6.02 \times 10^{23} \text{ Atoms}} \times \text{Mass} \frac{\text{MTU}}{\text{Assembly}} \times \frac{\text{Assembly}}{\# \text{ Rods}}$$

Multiplying the number of fissions by 0.3125 (0.25 x 1.25) atoms/fission then derives the quantity of fission gas produced. Olander's "Fundamental Aspects of Nuclear Reactor Fuel Elements" [11] lists the number of gas atoms from a single fission as 0.25. Based on a detailed SAS2H isotope generated fission gas inventory, this fraction is increased by 25% to account for decay chains not included in Olander (particularly those leading to ^{136}Xe). By employing a conservative fission gas fraction rather than the SAS2H output itself, the allowable cladding temperature calculation is decoupled from source term calculations.

Based on Sandia Report 90-2406, "A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements" [10], gas release fractions from the fuel pellets are assumed to be 12% for PWR fuel rods and 25% for BWR fuel rods. Relying on a gas diffusion model (as applied to pre-pressurized light water reactor fuel rods), the Sandia report indicates a release fraction of approximately 1% for PWR rods and approximately 2% for BWR rods [10]. Experimental release fractions reach as high as 16% for PWR rods and 25 % for BWR rods [10]. The higher release fractions are associated with unpressurized fuel rods or those rods run at uncharacteristically high temperatures and linear heat generation rates. While these rods show higher release rates, they are not expected to produce higher "burned fuel" pressures, since the partial pressure of the fill gas is not present, thereby allowing a larger number of fission gas molecules to accumulate before reaching limiting cladding pressure. The 12% PWR fission gas release fraction excludes the unpressurized Maine Yankee rod data while including the 43,000 MWD/MTU Calvert Cliff data to approximate the upper bound 45,000 MWD/MTU burnup. An additional analysis is performed comparing the 12% PWR and 25% BWR release fractions to the element specific release fractions in Reg. Guide 1.25 [29]. The 12% PWR release fraction results in gas releases similar to those indicated by the Regulatory Guide, while the BWR 25% release fraction is twice the Regulatory Guide indicated gas release. Note that both the Sandia

report and the Regulatory Guide release fractions are for punctured fuel rods where the release of the pressurizing gas allows additional gaseous isotopes to migrate from the fuel matrix. Using the 12% PWR and 25% BWR fuel rod release fractions, therefore, results in a conservative cladding pressurization assumption for the intact rod analysis. For higher burnup PWR fuels (i.e., rod peak burnup up to 50,000 MWD/MTU), Maine Yankee experience is that the maximum gas release rate (fuel pellet to rod plenum in intact fuel rods) is less than 3% [30]. Therefore, the 12% release fraction established for standard PWR fuel burned up to 45,000 MWD/MTU is conservatively applied to the higher burnup PWR fuel.

Fuel rod free volume is calculated based on the fuel characteristics in Tables 3.4-11 and 3.4-12 for PWR and BWR fuel, respectively. Not all assemblies requested for loading are included in the tables, since assemblies with significantly higher free volume or lower fuel mass are bounded by the cladding stress evaluations presented.

Substituting the internal gas pressure resulting from the releasable gas inventories produced by 40,000 MWD/MTU burned fuel into the initial cladding stress (σ_{mhoop}) equation at a temperature of 380°C results in the assembly-specific maximum cladding stresses shown in Table 3.4-11 and Table 3.4-12. The Westinghouse 14 x 14 and GE 9 x 9 (150-inch fuel region) are the limiting PWR and BWR assembly types at 113.9 and 70.5 MPa stress levels, respectively.

The stress levels in the limiting assemblies are then evaluated at burnups ranging from 35,000 MWD/MTU to 50,000 MWD/MTU for PWR fuel and 35,000 MWD/MTU to 45,000 MWD/MTU for BWR fuel at temperatures of 300°C and 400°C for PWR fuels and 300°C and 450°C for BWR fuel. The evaluation results are presented in Table 3.4-13. This data is overlaid on generic stress versus limiting temperature curves to arrive at cool time and burnup-specific maximum cladding allowable temperatures. The data, shown in Table 3.4-14, from which the generic curves are constructed, is taken from Table 3.1 of PNL-6189 [27].

The cladding temperature limit curves for the limiting PWR and BWR fuel assemblies are provided in Figures 3.4-13 and Figure 3.4-14. The intercept of each of the curves represents the maximum allowable cladding temperature at a given cool time and maximum assembly burnup. Fuel rod peak cladding stress level and the allowable cladding temperature are calculated using the assembly average burnup, even though some rods experience a higher burnup than the average. The average burnup is used, since the quantity of fission gas formation and the fuel rod gas temperature are conservatively determined. As shown in Table 3.4-15, allowable cladding

temperature varies only slightly over a wide range of burnup for a given required cooling time. Consequently, the variation in cladding stress with burnup is also small.

3.4.6.2 Maximum Allowable Canister Heat Load

Thermal analysis was performed at three heat loads for PWR fuel and one heat load for BWR fuel to determine the corresponding maximum fuel cladding temperature. Only one heat load is analyzed for BWR fuel because the maximum computed clad temperature is 548°F (286.7°C) at the maximum heat load of 16 kW, which is already lower than the minimum allowable temperature limit for any BWR fuel (Table 3.4-15). Therefore, BWR fuel was not further analyzed and a fixed maximum decay heat of 16 kW is allowable for transport of BWR fuel.

The thermal models and methods, described in Section 3.4.1, used to determine the temperature of fuel cladding and system components for the design basis heat load are applied to determine the cladding temperature at reduced heat loads. The ANSYS calculated temperatures that provide input for correlating allowable cladding temperature to allowable heat load are:

Fuel Type	Fuel Clad Temperature (°F)	Fuel Clad Temperature (°C)	Heat Load (kW)
PWR	537	280.6	14
PWR	610	321.1	17
PWR	677	358.3	20
BWR	548	286.7	16

The PWR temperature versus heat load curve is plotted in Figure 3.4-15. To provide adequate design margin, the maximum allowable cladding temperatures are reduced by a temperature bias, shown in Table 3.4-17, prior to their use in the calculation of maximum allowable canister heat load. Maximum allowable canister heat loads are calculated for initial cool times ranging from 5 to 15 years and burnups ranging from 35,000 MWD/MTU to 50,000 MWD/MTU for PWR fuel and 35,000 MWD/MTU to 45,000 MWD/MTU for BWR fuel. The results of the PWR and BWR analysis are presented in Table 3.4-16. Since these temperatures are based on the PWR and BWR assemblies having the highest cladding stress levels, the maximum heat loads can be applied to all UMS® design basis contents.

3.4.7 Evaluation of Package Performance for Normal Conditions of Transport

Results of thermal analysis of the Universal Transport Cask containing PWR and BWR fuel under normal conditions of transport are summarized in Tables 3.4-1 through 3.4-3. The maximum fuel rod cladding temperature is maintained below 810°F (432°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 2.0 to be less than the allowable stresses. As shown in Section 3.4.2, the personnel barrier temperature of 153°F is below the allowable temperature of 185°F for exclusive use shipment. Therefore, the Universal Transport Cask can safely transport the design basis fuel under the normal conditions of transport specified in 10 CFR 71.71.

Figure 3.4-1 Three-Dimensional PWR Cask Finite Element Model

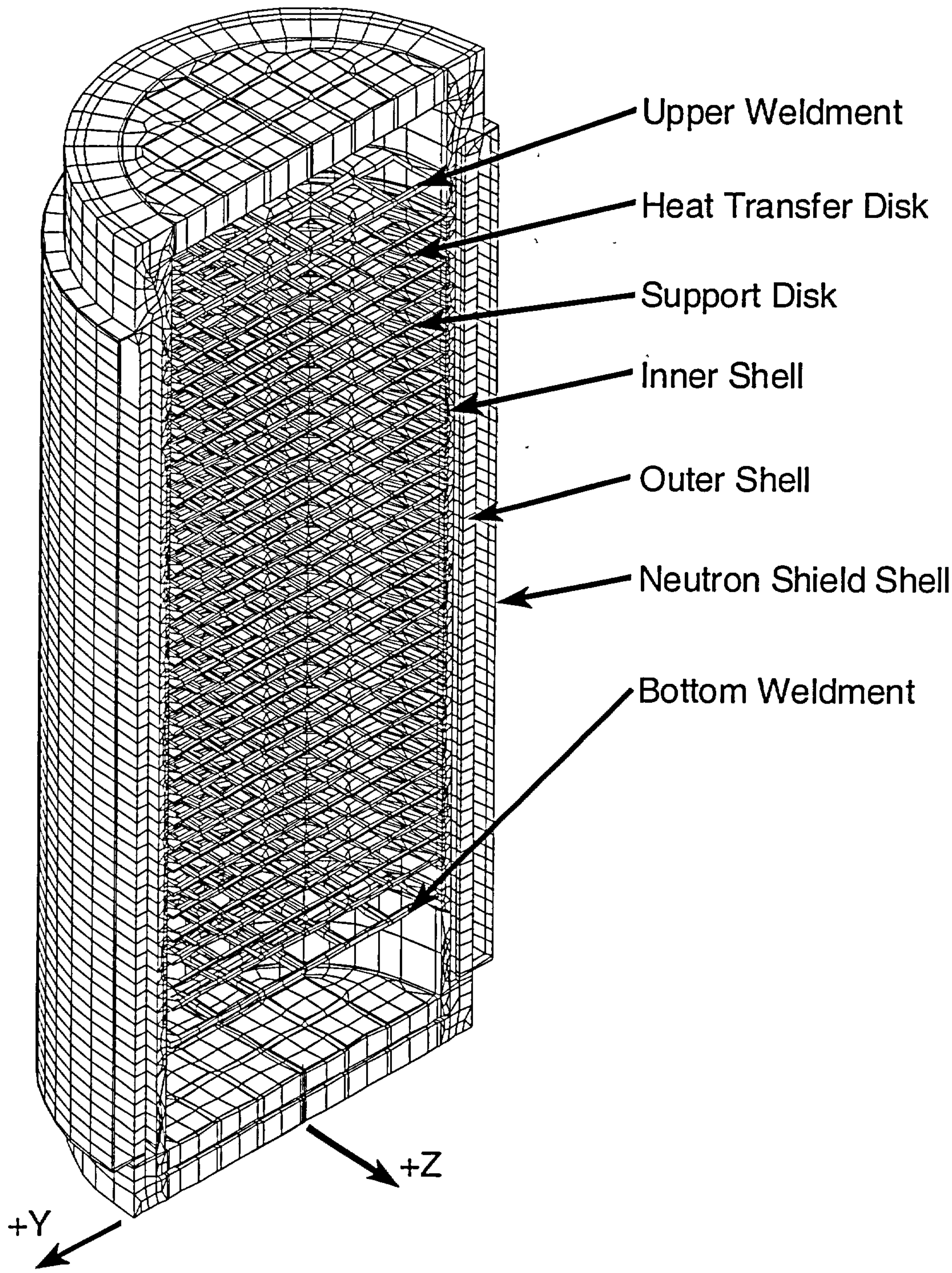


Figure 3.4-2 Design Basis PWR Fuel Assembly Axial Power Distribution

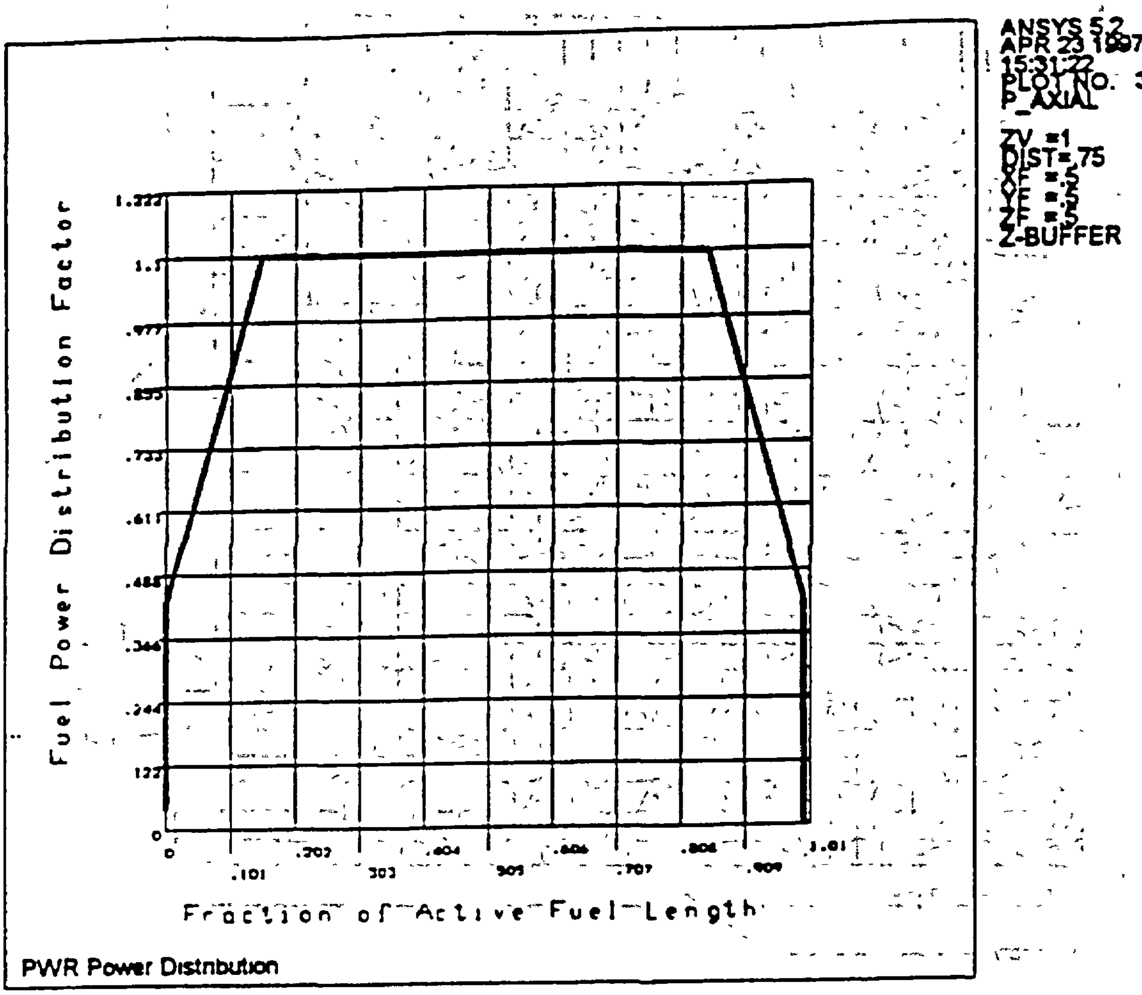


Figure 3.4-3 PWR 14x14 Fuel Assembly Two-Dimensional Finite Element Model

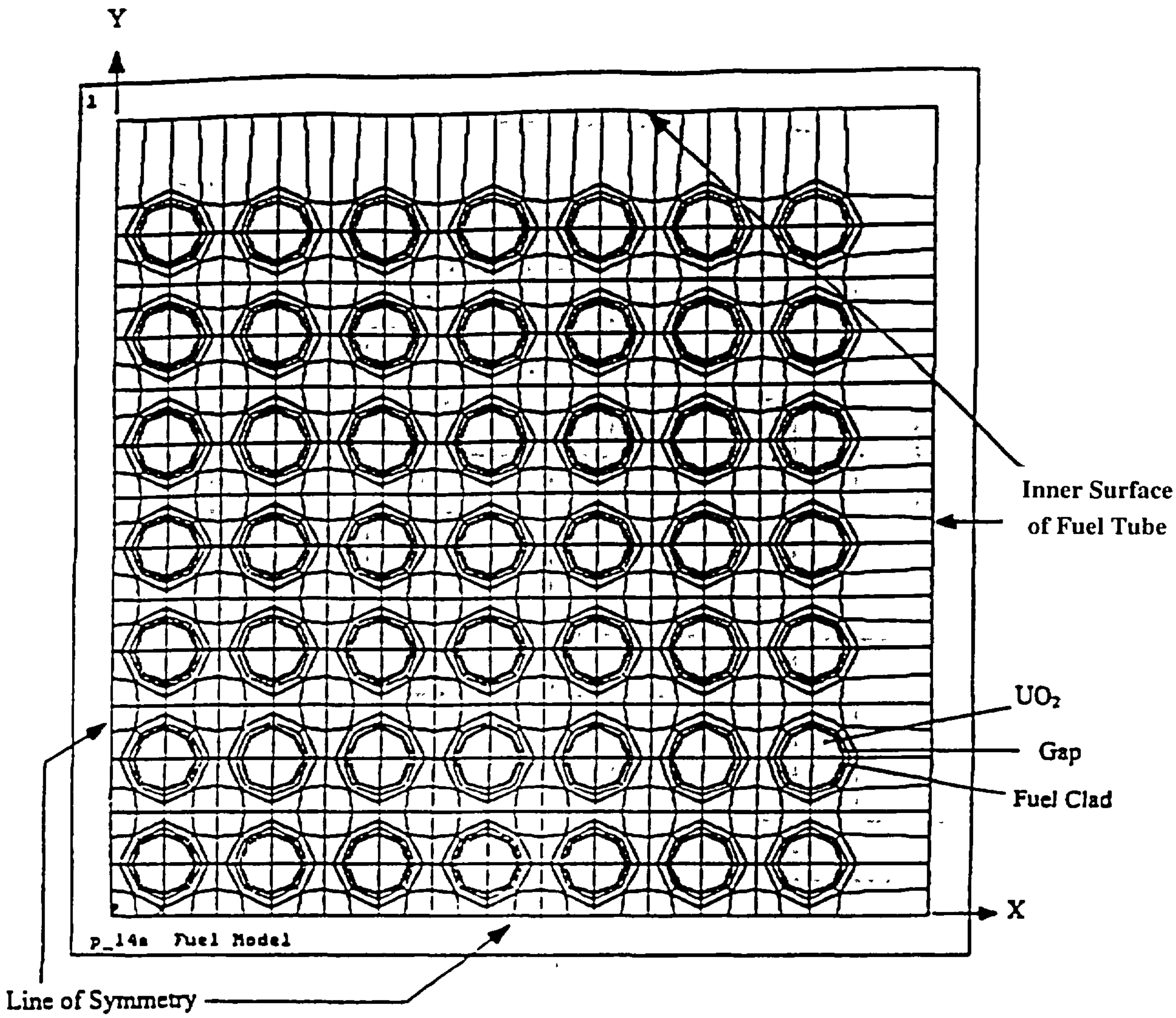


Figure 3.4-4 Two-Dimensional PWR Fuel Tube Model

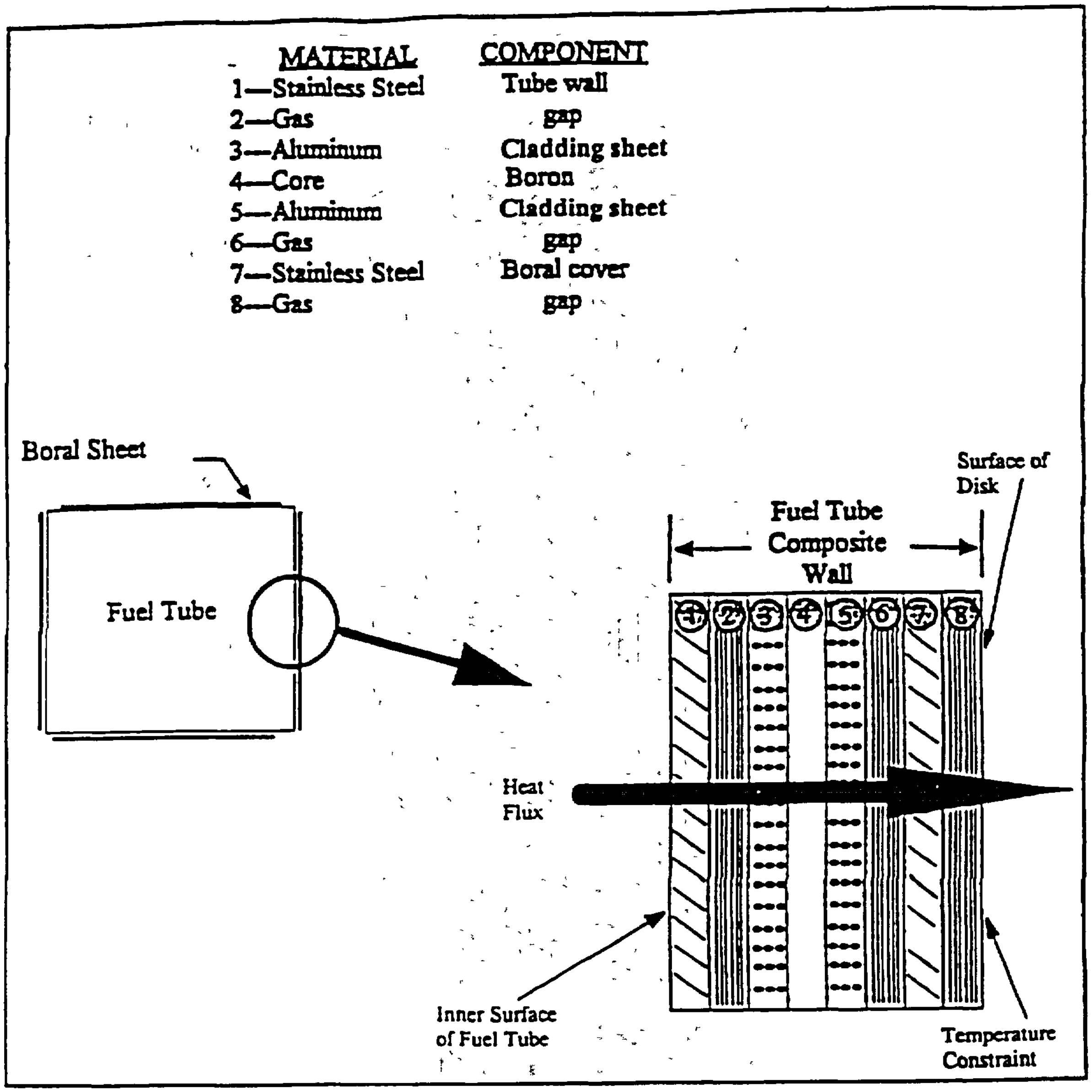


Figure 3.4-5 Three-Dimensional BWR Cask Finite Element Model

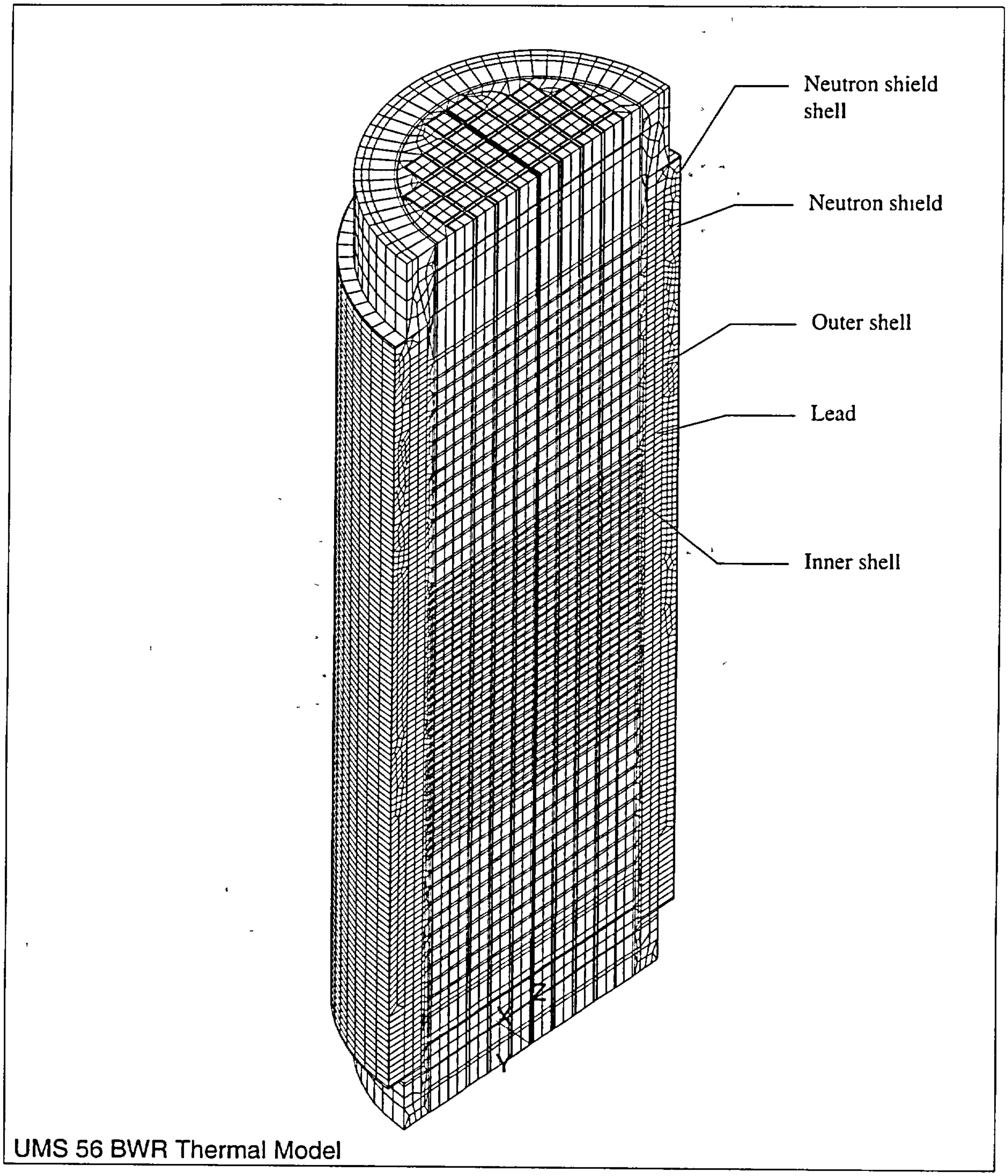


Figure 3.4-6 Design Basis BWR Fuel Assembly Axial Power Distribution

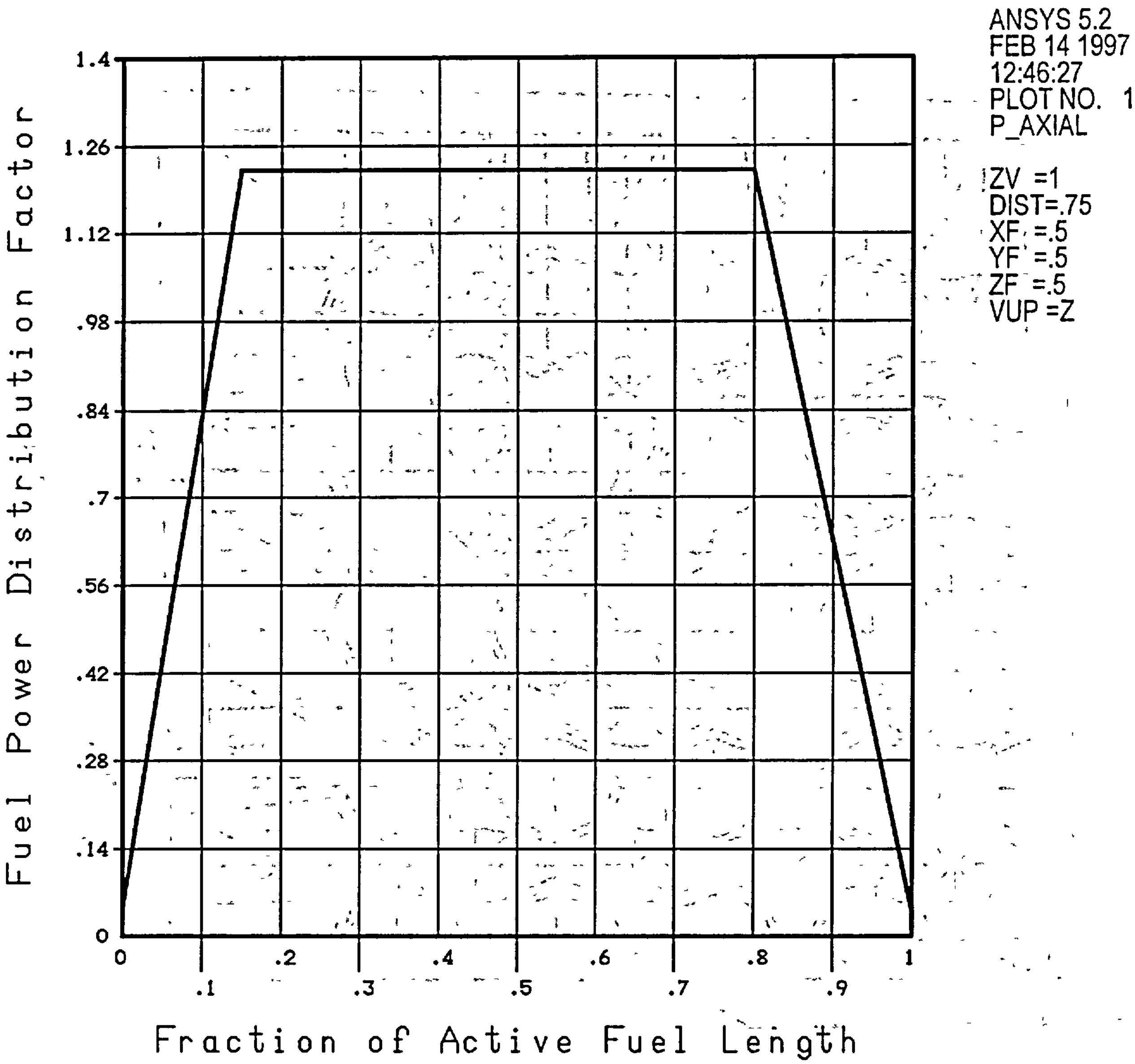


Figure 3.4-7 BWR 9x9 Fuel Assembly Two-Dimensional Finite Element Model

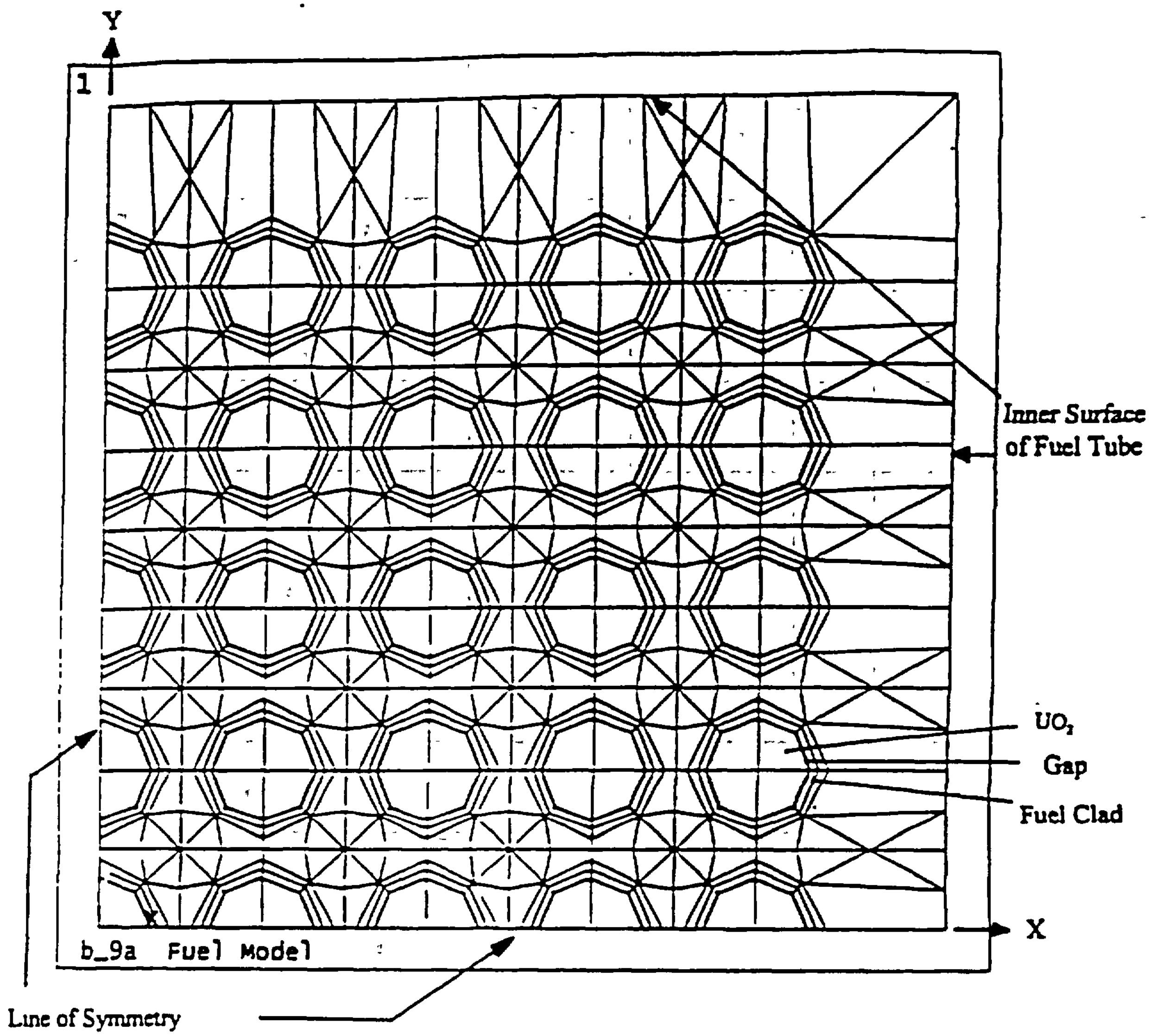


Figure 3.4-8 Two-Dimensional BWR Fuel Tube (with BORAL) Model

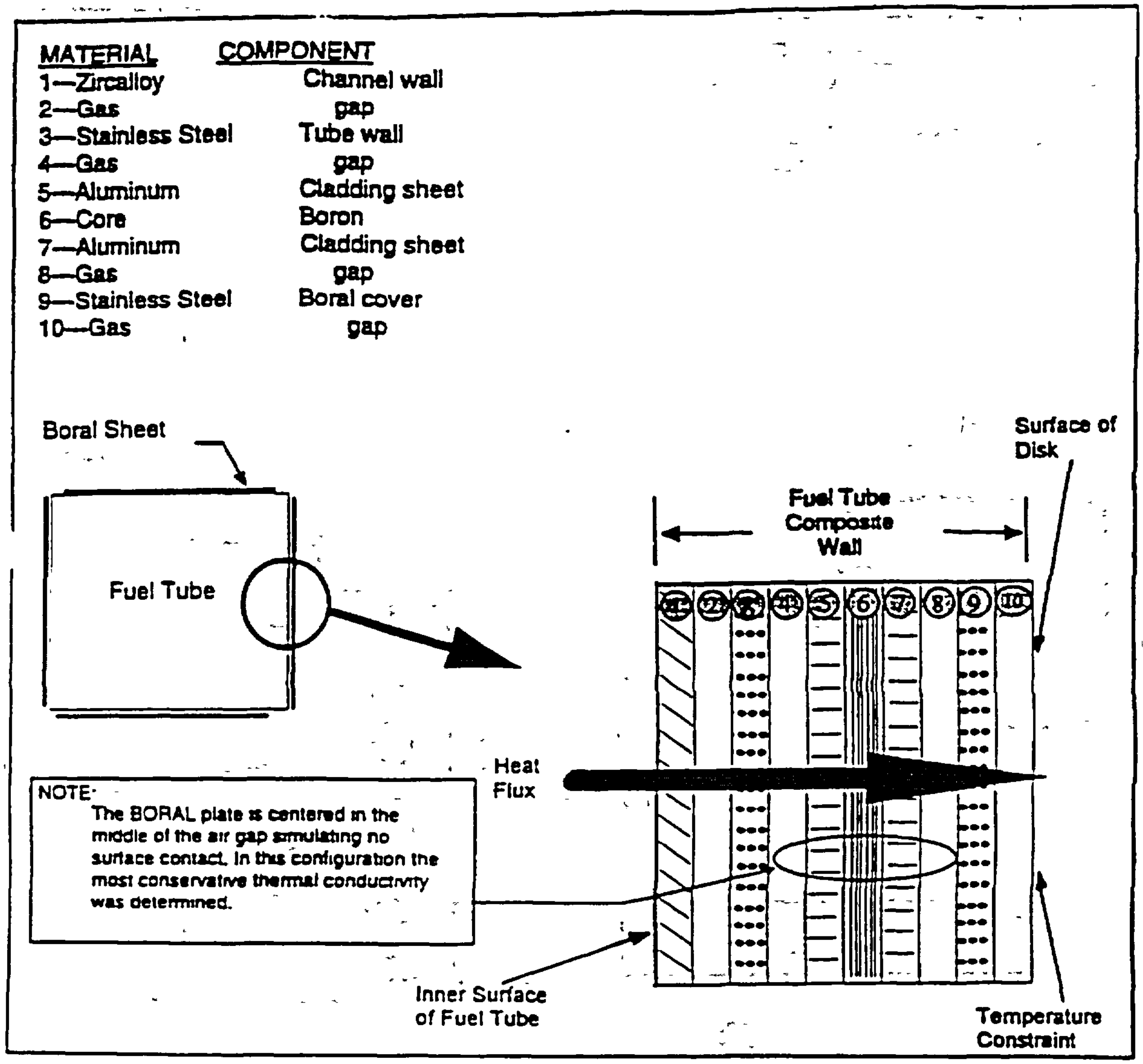


Figure 3.4-9 Two-Dimensional BWR Fuel Tube (without BORAL) Model

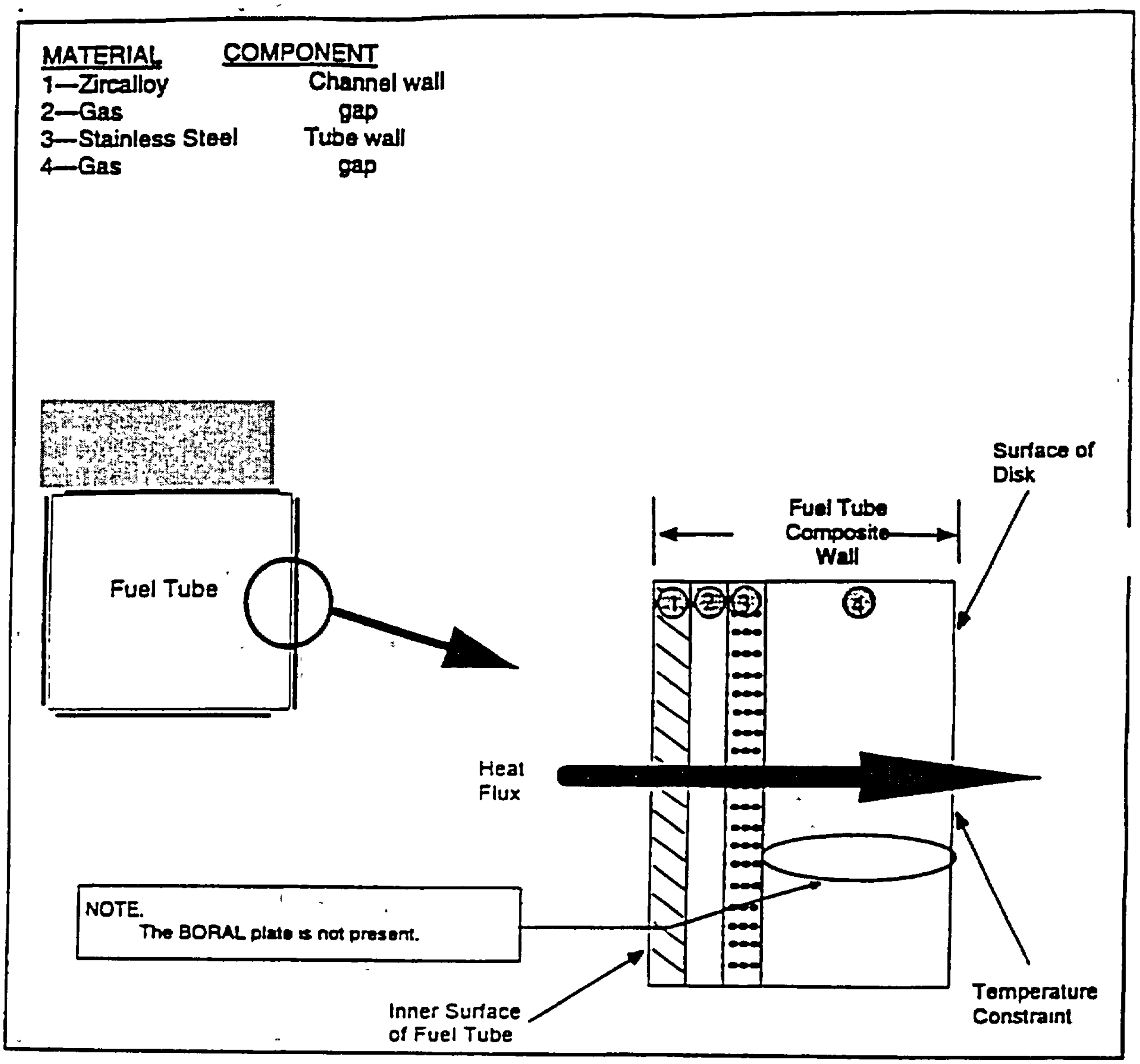


Figure 3.4-10 Cask Impact Limiter Thermal Model

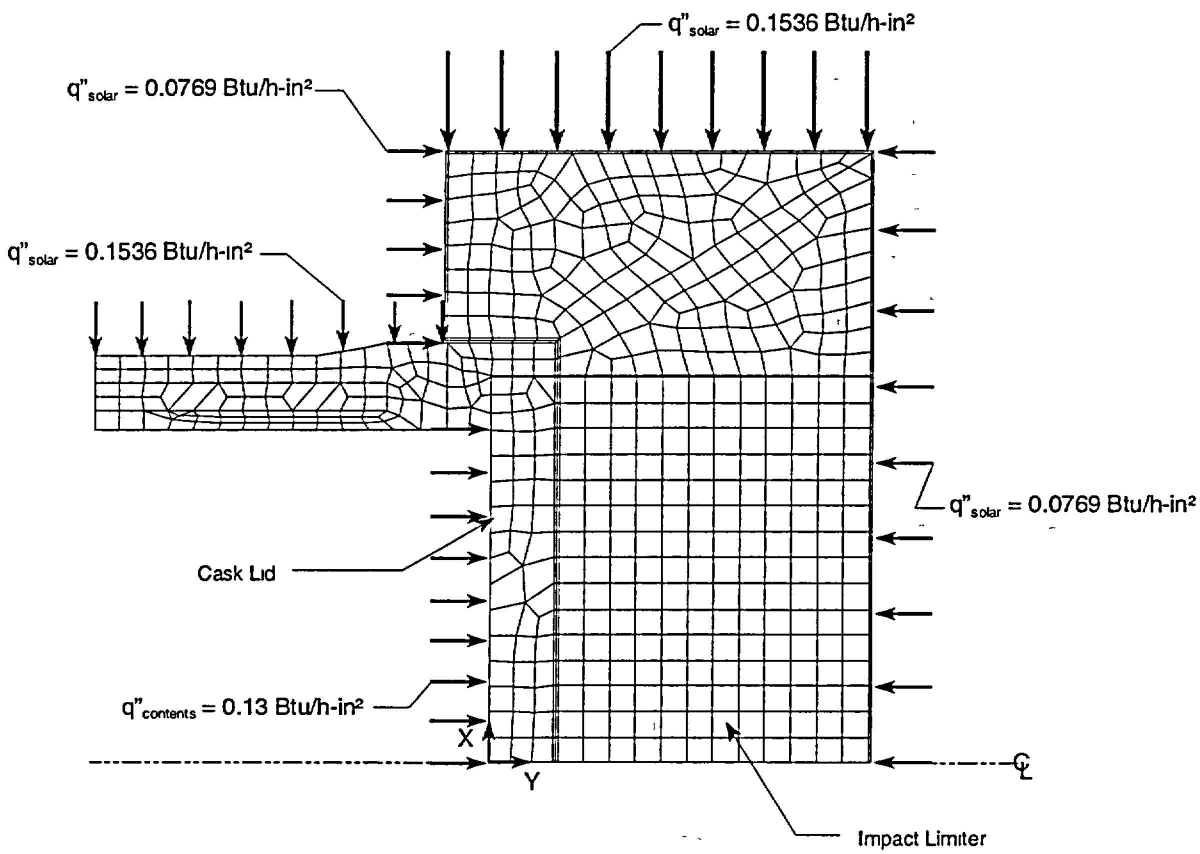


Figure 3.4-11 Personnel Barrier Thermal Model

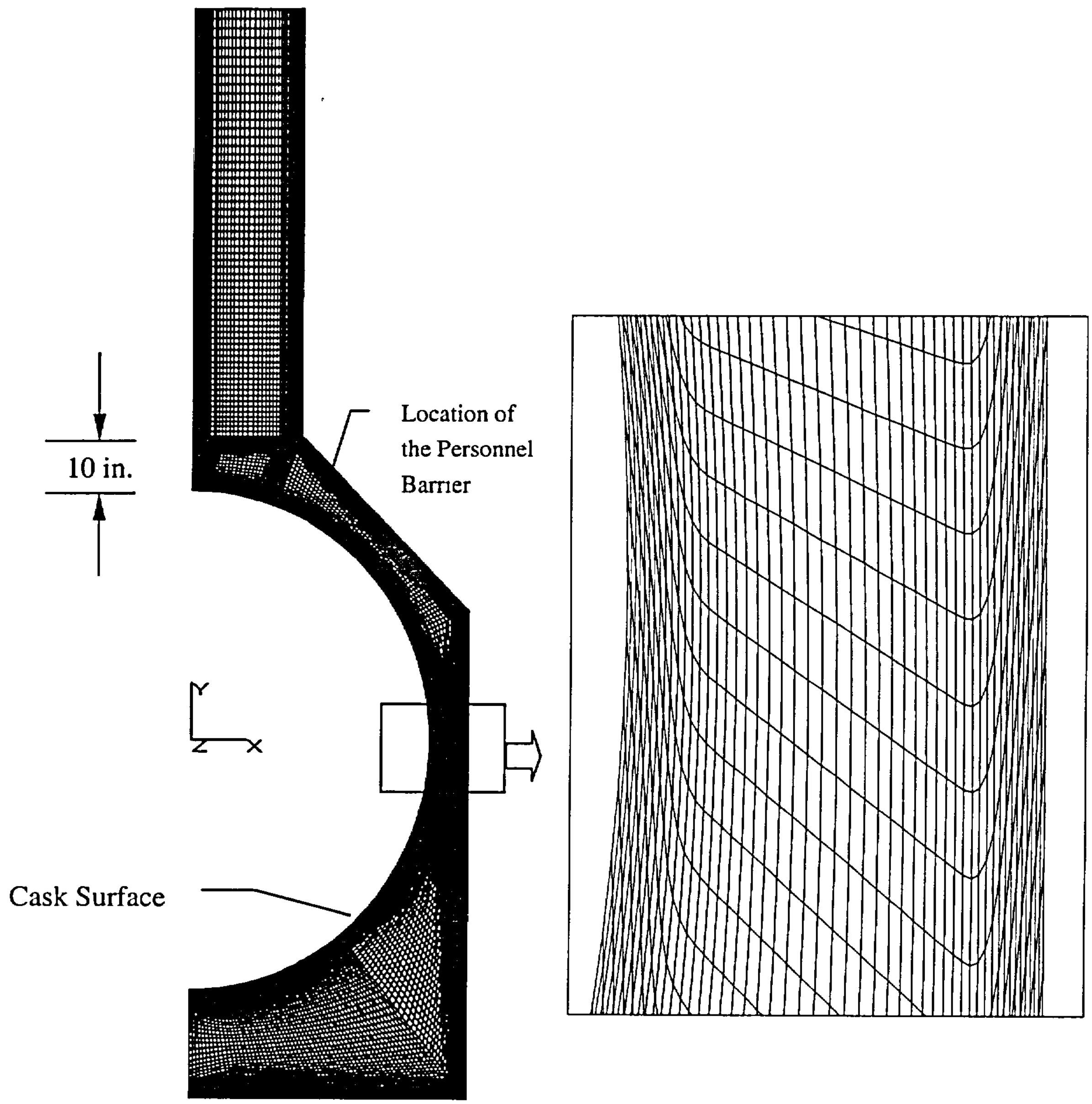
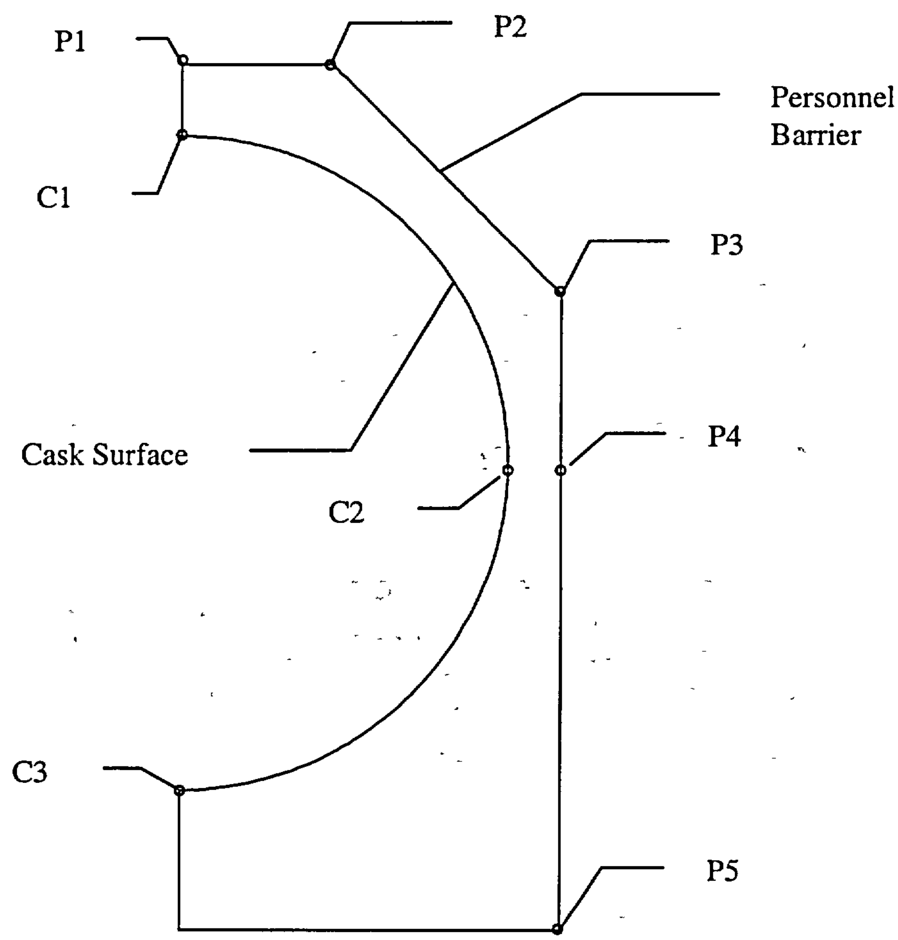


Figure 3.4-12 Temperature Results at Key Points of the Personnel Barrier



	Boundary Conditions			Calculated Temperature (°F)				
Location	C1	C2	C3	P1	P2	P3	P4	P5
Temperature	244	256	267	153	108	133	131	100

Figure 3.4-13 PWR Fuel Dry Storage Temperature versus Cladding Stress

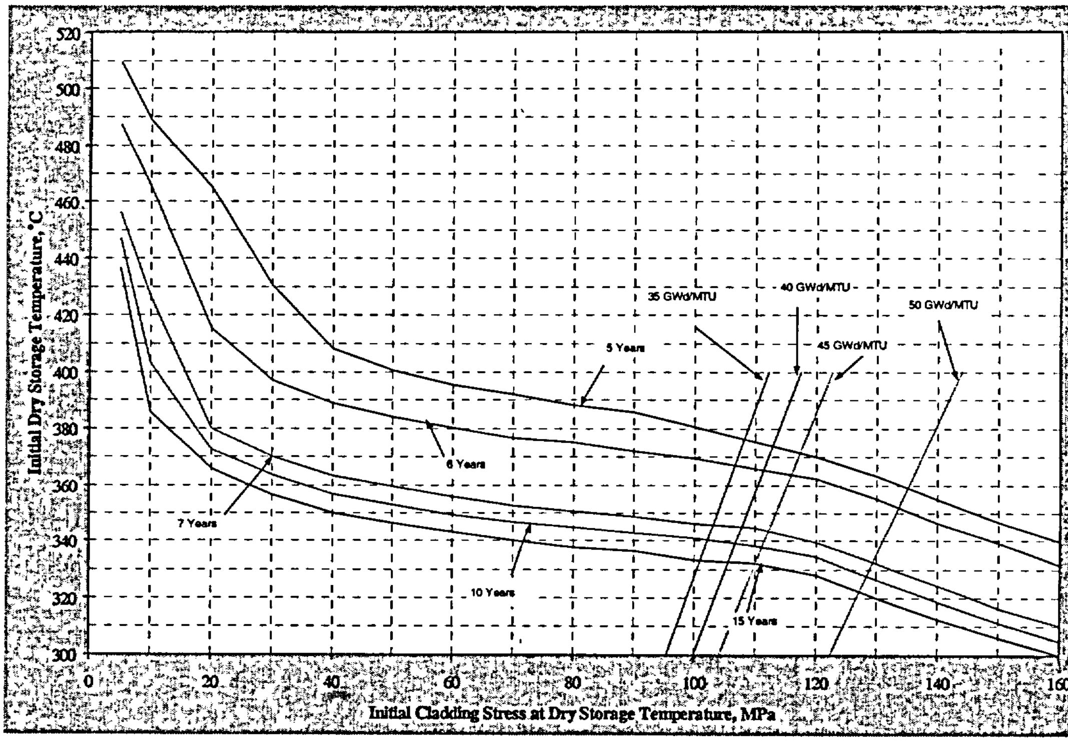


Figure 3.4-14 BWR Fuel Dry Storage Temperature versus Cladding Stress

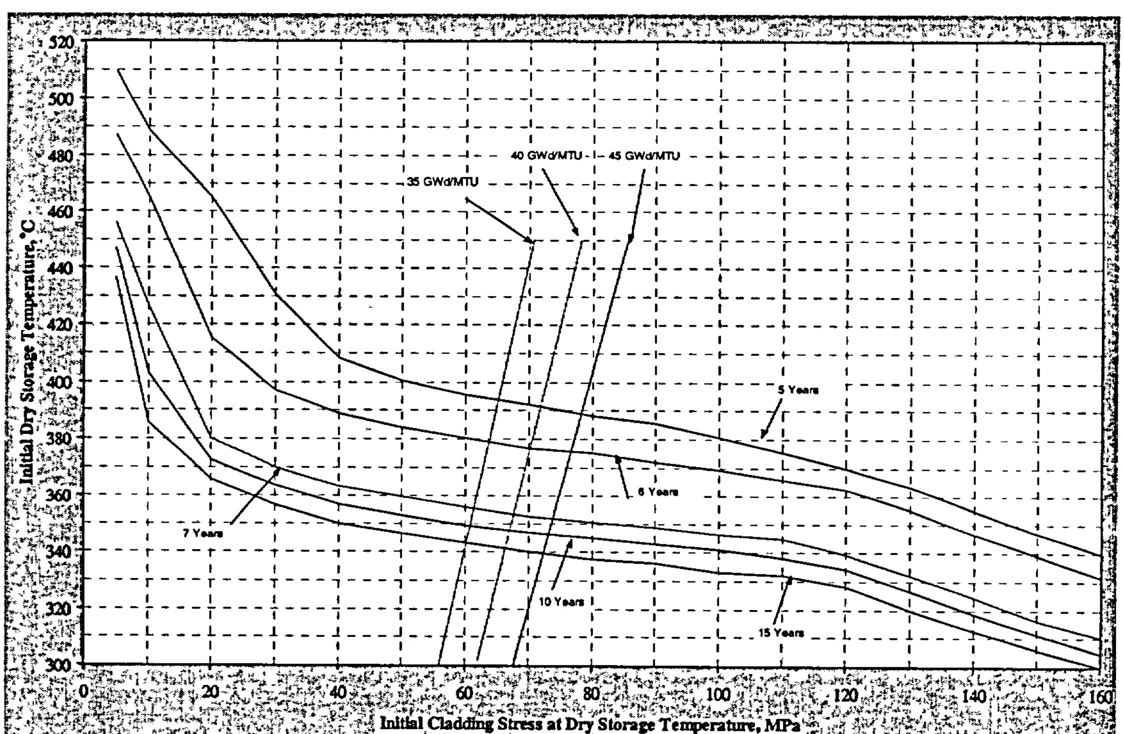


Figure 3.4-15 PWR Fuel Cladding Dry Storage Temperature versus Basket Heat Load

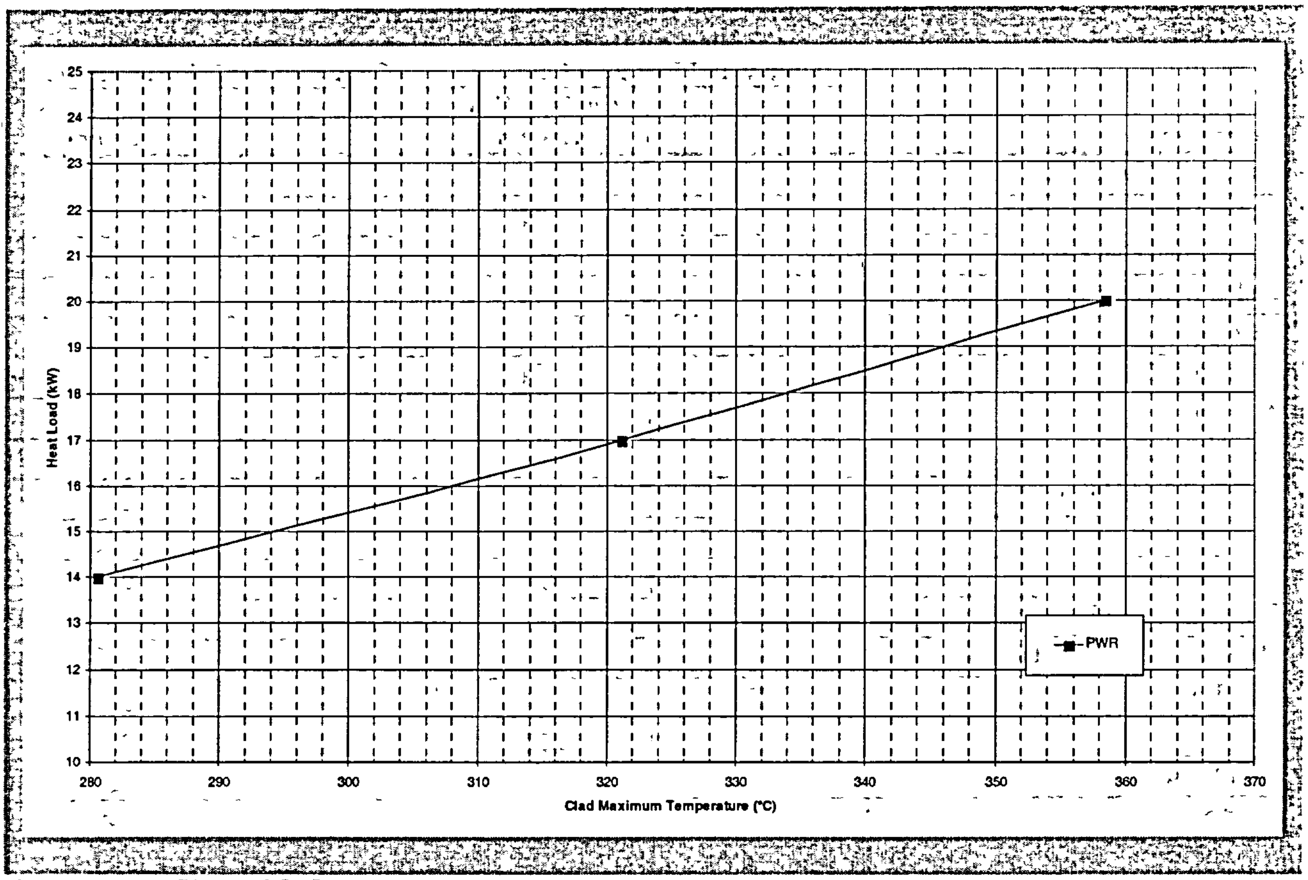


Table 3.4-1 Maximum Component Temperatures - Normal Conditions of Transport, Maximum Decay Heat, Maximum Ambient Temperature

Component	Temperature (°F) Cask with PWR Fuel Canister		Temperature (°F) Cask with BWR Fuel Canister	
	[redacted]	Canister Gas: Helium	[redacted]	Canister Gas: Helium
Cask Lid O-Rings/Vent Port O-ring[1]	[redacted]	266	[redacted]	204
Lower Drain Port O-ring[4]	[redacted]	224	[redacted]	230
Cask Radial Outer Surface	[redacted]	266	[redacted]	256
Radial Neutron Shield	[redacted]	293	[redacted]	286
Lead Gamma Shield	[redacted]	306	[redacted]	298
Aluminum Disk Exterior	[redacted]	268	[redacted]	298
Aluminum Disk Interior	[redacted]	605	[redacted]	515
Support Disk Exterior	[redacted]	255	[redacted]	208
Support Disk Interior	[redacted]	608	[redacted]	512
Canister Shell	[redacted]	408	[redacted]	363
Canister Shield Lid	[redacted]	270	[redacted]	208
Canister Bottom Plate	[redacted]	324	[redacted]	262
Maximum Fuel Rod Cladding	[redacted]	673	[redacted]	548
Cask Bottom	[redacted]	217	[redacted]	228
Bottom Forging	[redacted]	224	[redacted]	230
Inner Shell	[redacted]	344	[redacted]	312
Outer Shell	[redacted]	301	[redacted]	293
Top Forging[2]	[redacted]	250	[redacted]	194
Cask Lid	[redacted]	266	[redacted]	204
Cask Lid Bolt[3]	[redacted]	266	[redacted]	204
Average Gas Temperature in the Canisters[5]	[redacted]	453	[redacted]	366

Conditions:
100°F ambient temperature
20 kW decay heat load, 1 1 peaking factor - PWR
16 kW decay heat load, 1 22 peaking factor - BWR
Solar insolation
Cask cavity gas. helium
Canister cavity gas [redacted] helium [redacted]

1. Cask lid O-rings and vent port O-rings not explicitly modeled—taken to be the maximum cask lid temperature
2. Average temperature
3. Cask lid bolts not explicitly modeled—taken to be the maximum temperature of the cask lid
4 Lower drain port O-ring not explicitly modeled - taken to be the maximum temperature of the bottom forging
5 Calculated as a volumetric average.

[redacted]

Table 3.4-2 Maximum Component Temperatures - Normal Conditions of Transport, Maximum Decay Heat, Minimum Ambient Temperature

Component	Temperature (°F) Cask with PWR Fuel Canister		Temperature (°F) Cask with BWR Fuel Canister	
	[redacted]	Canister Gas: Helium	[redacted]	Canister Gas: Helium
Cask Lid O-Rings/Vent Port O-ring[1]	[redacted]	140	[redacted]	62
Cask Radial Outer Surface	[redacted]	151	[redacted]	132
Radial Neutron Shield	[redacted]	178	[redacted]	162
Lead Gamma Shield	[redacted]	191	[redacted]	174
Maximum Basket[2]	[redacted]	505	[redacted]	404
Canister Shell	[redacted]	289	[redacted]	238
Canister Shield Lid	[redacted]	145	[redacted]	66
Canister Bottom Plate	[redacted]	205	[redacted]	127
Maximum Fuel Rod Cladding	[redacted]	578	[redacted]	440

Conditions:
-40°F ambient temperature
20 kW decay heat load, 1 1 peaking factor - PWR
16 kW decay heat load, 1 22 peaking factor - BWR
No insolation
Cask cavity gas helium
Canister cavity gas [redacted] helium

1 Cask lid O-ring and vent port O-rings not explicitly modeled—taken to be the maximum cask lid temperature
2 Taken to be the greater of the maximum support disk and the maximum aluminum heat transfer disk temperatures
[redacted]

Table 3.4-3 Universal Transport Cask Thermal Performance Summary for Component Operating Temperature

Temperature	Cask with PWR Fuel Canister (helium in cask cavity/helium in canister)	Cask with BWR Fuel Canister (helium in cask cavity/helium in canister)	Allowable Temperature Range
Maximum cladding temperature(°F)	673	548	< 705[1]
Component safe operating temperature ranges			
Cask lid O-rings	-40 to 266°F	-40 to 208°F	-40 to 300°F
Vent port coverplate O-ring	-40 to 266°F	-40 to 208°F	-40 to 300°F
Drain port coverplate-O-rings	-40 to 224°F	-40 to 230°F	-40 to 300°F
Radial NS-4-FR neutron shield	-40 to 293°F	-40 to 286°F	-40 to 300°F
Lead gamma shield	-40 to 306°F	-40 to 298°F	-40 to 600°F
Aluminum heat transfer disk	-40 to 605°F	-40 to 515°F	-40 to 700°F
PWR support disk	-40 to 608°F		-40 to 650°F
BWR support disk		-40 to 517°F	-40 to 700°F

1. The temperature of 705°F (374°C) is based on the maximum allowable cladding temperature established in Table 3.4-15 for the fuels with 5-year cooling time and 40,000 MWD/MTU burnup, which corresponds to maximum allowable canister decay heat of 20 kW for the PWR system (Table 3.4-16). Note that the design basis heat load of 20 kW is used in the thermal evaluation for the PWR fuels. The allowable temperature of 374°C also bounds the allowable temperatures for the design basis heat load for the BWR system (16 kW) as shown in Table 3.4-15.

Table 3.4-4 Maximum Internal Pressures for Transport

Fuel	Cavity	Condition	Pressure (psig)
PWR	Canister	3% fuel rod failure	6.15
		100% fuel rod failure	74.3
	Cask	3% fuel rod failure	6.91
		100% fuel rod failure	69.3
BWR	Canister	3% fuel rod failure	8.47
		100% fuel rod failure	43.8
	Cask	3% fuel rod failure	8.65
		100% fuel rod failure	42.8

Table 3.4-5 PWR Per Assembly Fuel Generated Gas Inventory

Array	Assy Type	MTU	Moles
14x14	WE Standard	0.4144	35.52
15x15	B&W	0.4807	41.32
16x16	CE	0.4417	38.10
17x17	WE Standard	0.4671	40.18

Table 3.4-6 PWR Canister Free Volume (No Fuel or Inserts)

Canister Class	1	2	3
Basket Volume (in^3)	69800	74490	77460
Canister Height (inch)	175.05	184.15	191.75
Canister Free Volume w/o Fuel (liter)	7970	8400	8770
Canister and Cask Free Volume w/o Fuel (liter)	9030	8980	8970

Table 3.4-7 PWR Maximum Normal Condition Pressure Summary

Canister Class	Fuel Type	Canister Pressure (psig)	Cask Pressure (psig)
Class 1	West. 17x17 Standard	6.13	6.91
Class 2	B&W 17x17 Mark C	6.15	6.62
Class 3	CE 16x16	5.81	6.02

Table 3.4-8 BWR Per Assembly Fuel Generated Gas Inventory

Array	Assy Type	MTU	Moles
7x7	GE 7x7 (49 Rods)	0.1985	16.78
8x8	GE 8x8 (63 Rods)	0.1880	16.07
9x9	GE 9x9 (79 Rods)	0.1979	16.86

Table 3.4-9 BWR Canister Free Volume (No Fuel or Inserts)

Canister Class	4	5
Basket Volume (in^3)	73110	74680
Canister Height (inch)	185.55	190.35
Canister Free Volume w/o Fuel (liter)	8500	8740
Canister and Cask Free Volume w/o Fuel (liter)	8710	8930

Table 3.4-10 BWR Maximum Normal Condition Pressure Summary

Canister Class	Fuel Type	Canister Pressure (psig)	Cask Pressure (psig)
Class 4	GE 7 x 7	3.47	3.65
Class 5	GE 7 x 7	3.41	3.55
Class 5	GE 9 x 9	3.33	3.48

Table 3.4-11 PWR Cladding Stress Level Comparison Chart

Fuel Type	B&W 15x15	B&W 17x17	CE 14x14	CE 16x16	WE 14x14	WE 15x15	WE 17x17
Rod OD (inch)	0.43	0.379	0.44	0.382	0.422	0.422	0.374
Cladding Thickness (inch)	0.0265	0.024	0.028	0.025	0.0225	0.0242	0.0225
Pellet OD (inch)	0.3686	0.3232	0.3765	0.325	0.3674	0.3659	0.3225
Active Fuel Length (inch)	144	143	137	150	145.2	144	144
Plenum Length (inch)	7.755	8.318	8.528	9.927	5.790	7.386	6.260
Spring Weight (lb)	0.042	0.026	0.1	0.1	0.07	0.044	0.037
Backfill Pressure (psig)	435	435	500	500	500	500	500
Fuel Mass (MTU)	0.4807	0.4658	0.4037	0.4417	0.4144	0.4646	0.4671
# of Fuel Rods	208	264	176	236	179	204	264
Free Volume (inch^3)	1.427	1.198	1.252	1.052	1.217	1.300	0.882
Pressure (psia) (380°C)	1525	1478	1739	1722	1762	1713	1795
Stress Level (Mpa)	83.1	78.9	91.2	88.5	113.9	101.7	102.1

Table 3.4-12 BWR Cladding Stress Level Comparison Chart

Fuel Type	EX 7x7	EX 8x8	EX 9x9	GE 7x7	GE 8x8a	GE 8x8b	GE 9x9
Rod OD (inch)	0.57	0.484	0.424	0.563	0.493	0.483	0.441
Cladding Thickness (inch)	0.036	0.036	0.03	0.032	0.034	0.032	0.028
Pellet OD (inch)	0.49	0.4045	0.3565	0.487	0.416	0.41	0.376
Active Fuel Length (inch)	144	150	150	144	144	150	150
Plenum Length (inch)	10.200	10.024	9.578	11.190	10.960	9.580	9.580
Spring Weight (lb)	0.13	0.1	0.047	0.083	0.066	0.066	0.047
Backfill Pressure (psig)	44.1	132.0	132.0	44.1	132.0	132.0	132.0
Fuel Mass (MTU)	0.196	0.1793	0.1666	0.1977	0.1855	0.1847	0.1979
# of Fuel Rods	48	62	74	49	63	62	79
Free Volume (inch^3)	2.426	1.708	1.469	3.236	2.181	1.970	1.758
Pressure (psia) (380°C)	1264	1469	1359	971	1236	1345	1286
Stress Level (MPa)	66.7	65.1	65.4	58.2	59.8	68.7	70.5

Table 3.4-13 Cladding Stress as a Function of Fuel Assembly Average Burnup and Temperature

	PWR		BWR	
Burnup	300°C	400°C	300°C	450°C
35,000 MWD/MTU	95.4 Mpa	112.3 Mpa	55.9 Mpa	70.8 Mpa
40,000 MWD/MTU	99.9 Mpa	117.4 Mpa	61.8 Mpa	78.2 Mpa
45,000 MWD/MTU	104.2 Mpa	122.6 Mpa	67.6 Mpa	85.5 Mpa
50,000 MWD/MTU	122.3 Mpa	143.9 Mpa	-	-

Table 3.4-14 Maximum Allowable Initial Storage Temperature (°C) as a Function of Initial Cladding Stress and Initial Cool Time

MPa	5 years	6 years	7 years	10 years	15 years
5	509.2	487.3	455.9	447	436.5
10	488.8	465.5	426.4	403	385.6
20	465.2	415.5	380.1	372.4	366
30	430.4	397	370.1	363.8	356.5
40	408.1	389	363.2	356.6	350
50	400.6	384	359.7	353.1	346.5
60	395.6	380.4	355.9	349.6	343.1
70	391.9	376.5	352.5	347	340
80	388.2	375	350.8	345.2	337.6
90	385.7	372	348.8	342.8	336.1
100	380.7	369.3	346.2	341	333.2
110	375.2	365.9	344.6	338	332.1
120	370	362.4	339.5	334.3	328.2
130	363.5	355.2	332.2	326.6	320
140	355	346.6	324.2	318.6	312.6
150	346.9	339.1	316.5	311.2	306
160	339.6	331.4	310.3	304.7	299.9

Table 3.4-15 Maximum Allowable Cladding Temperature for PWR and BWR Fuel

Cool Time [years]	PWR Clad Temperature Limit [°C]				BWR Clad Temperature Limit [°C]			
	Burnup (MWD/MTU)				Burnup (MWD/MTU)			
	35,000	40,000	45,000	50,000	35,000	40,000	45,000	50,000
5	376	374	371	359	394	391	389	-
6	367	365	364	352	379	376	376	-
7	346	345	343	333	355	353	352	-
10	340	339	338	328	349	348	346	-
15	333	333	332	322	343	341	339	-

Table 3.4-16 Maximum Allowable Decay Heat for PWR and BWR Systems

Cool Time [years]	PWR Decay Heat Limit[1] [kW]				BWR Decay Heat Limit[1] [kW]			
	Burnup (MWD/MTU)				Burnup (MWD/MTU)			
	35,000	40,000	45,000	50,000	35,000	40,000	45,000	50,000
5	20.00	20.00	19.90	19.30	16.00	16.00	16.00	-
6	19.50	19.30	19.20	18.70	16.00	16.00	16.00	-
7	17.80	17.80	17.70	17.20	16.00	16.00	16.00	-
10	17.40	17.30	17.20	16.80	16.00	16.00	16.00	-
15	16.80	16.80	16.70	16.50	16.00	16.00	16.00	-

1 Based on maximum clad temperature and biases shown in Table 3.4-17.

Table 3.4-17 Temperature Bias Applied to Maximum Allowable Decay Heats

Cool Time [years]	PWR Clad Temperature Bias [°C] Burnup (MWD/MTU) 35,000	40,000	45,000	50,000	BWR Clad Temperature Bias [°C] Burnup (MWD/MTU) 35,000	40,000	45,000	50,000
5	-15	-15	-14	-9	-18	-17	-18	[illegible]
6	-15	-15	-16	-10	-18	-17	-19	[illegible]
7	-15	-15	-14	-10	-16	-16	-17	[illegible]
10	-15	-15	-15	-10	-16	-16	-15	[illegible]
15	-15	-16	-16	-9	-15	-16	-16	[illegible]

3.6 Thermal Evaluation for Site Specific Contents

3.6.1 Maine Yankee Site Specific Contents

The standard spent fuel assembly for the Maine Yankee site is the Combustion Engineering (CE) 14x14 fuel assembly. Fuel of the same design has also been supplied by Westinghouse and by Exxon. The standard 14x14 fuel assembly is included in the population of the design basis PWR fuel assemblies for the UMS Transport System (see Table 1.2.5). The maximum decay heat for the Maine Yankee fuel is limited to the design basis heat load for the PWR fuels (20 kW total, or 0.83 kW per assembly). This heat load is bounded by the thermal evaluations in Sections 3.4 and 3.5 for the normal conditions of transport and hypothetical accident conditions, respectively.

The Maine Yankee site specific fuels and GTCC waste are described in Sections 1.3.1.1.1 and 1.3.1.1.2, respectively.

The thermal evaluations of the Maine Yankee site specific fuels and the GTCC waste are provided in Sections 3.6.1.1 and 3.6.1.2, respectively.

Preferential loading of the Maine Yankee site-specific fuel assemblies is governed by the standard fuel inventory requirement presented in the Approved Contents and Design Features for the NAC-UMS® System in Chapter 12 of the Final Safety Analysis Report (FSAR) for the UMS® Universal Storage System, Docket Number 72-1015. Loading fuel assemblies for storage with a cool time of less than 7 years requires a preferential loading arrangement with shorter-cooled fuel placed at the canister interior locations. The corresponding thermal evaluation for the transport system is shown in Section 3.4.2.1. Maine Yankee site-specific preferential loading patterns placing high heat load (1.05 or 0.958 kW) fuel in basket peripheral locations, as allowed in Chapter 12 of the UMS® Universal Storage System FSAR, are not applicable for the transport system. All fuel assemblies loaded in the transport cask must meet the standard configuration transport minimum allowable heat load limits and cool time tables. As such, a transportable storage canister loaded under the Maine Yankee site-specific high heat load preferential loading option will require additional cool time for the peripheral assemblies to meet the transport cask cool time requirements shown in Table 1.3.1-2 (also Table 5.5.1.1-10). This assures that a loaded canister will meet all thermal and shielding limits for transport.

3.6.1.1 Spent Fuel

The Maine Yankee site specific fuels included in this evaluation are:

1. Consolidated fuel rod lattices consisting of a 17x17 lattice fabricated with 17x17 grids, 4 stainless steel support rods and stainless steel end fittings. One of these lattices contains 283 fuel rods and 2 vacancies. The other contains 172 fuel rods, with the remaining locations either empty or containing stainless steel dummy rods.
2. Standard fuel assemblies with a Control Element Assembly (CEA) inserted in each one.
3. Standard fuel assemblies that have been repaired by removing damaged fuel rods and replacing them with stainless steel dummy rods, solid zirconium rods, or 1.95 wt % enriched fuel rods.
4. Standard fuel assemblies that have had the burnable poison rods removed and replaced with hollow Zircaloy tubes.
5. Standard fuel assemblies with in-core instrument thimble assemblies stored in the center guide tube.
6. Standard fuel assemblies that are designed with variable enrichment (radial) and axial blankets.
7. Standard fuel assemblies that have fuel rods removed.
8. Fuel assemblies with damaged fuel rods.

The thermal evaluations of these site specific fuels are provided below. The maximum heat load per assembly is limited to the design basis heat load (0.83 kW) for all Maine Yankee site specific fuels.

1. Consolidated Fuel

There are two (2) consolidated fuel lattices (pseudo assemblies). The maximum decay heat of each consolidated fuel assembly is 0.279 kW. The heat load of the consolidated fuel lattice with 283 fuel pins is bounded by the design basis PWR fuel assembly, since its heat load is only one-third (0.279/0.83) of the design basis heat load.

The second consolidated fuel lattice has 172 fuel rods with 76 stainless steel dummy rods at the outer periphery of the lattice. Due to the presence of the stainless steel rods, the effective thermal conductivities of this assembly may be slightly lower than those of the standard CE 14x14 fuel

assembly. While the stainless steel rods provide better conductance in the axial direction, the radiation heat transfer is less effective at the surface of stainless steel rods, as compared to the standard fuel rods. The radiation is a function of surface emissivity and the emissivity for stainless steel (0.36) is less than one-half of that for Zircaloy (0.75). A parametric study is performed to demonstrate that the thermal performance of the UMS® PWR basket loading configuration consisting of 23 standard CE 14x14 fuel assemblies and the consolidated fuel lattice with stainless rods is bounded by that of the configuration consisting of 24 standard CE 14x14 fuel assemblies. Two finite element models are used in the study: a two-dimensional fuel assembly model and a three-dimensional periodic canister internal model.

The two-dimensional model is used to determine the effective thermal conductivities of the consolidated fuel lattice with stainless steel rods. Considering the symmetry of the consolidated fuel, the finite element model represents a one-quarter section as shown in Figure 3.6.1.1-1. The same methodology used in Section 3.4.1.1.2 for the two-dimensional fuel model for PWR fuel is employed in this model. The model includes the fuel pellets, cladding, helium between the fuel rods, and helium occupying the gap between the fuel pellets and cladding. In addition, the rods at the two outer layers are modeled as solid stainless steel rods to represent the configuration of this consolidated fuel lattice. Modes of heat transfer modeled include conduction and radiation between individual rods for steady-state condition. ANSYS PLANE55 conduction elements and LINK31 radiation elements are used in the model. Radiation elements are defined between rods and from rods to the boundary of the model (inside surface of the fuel tube). The effective conductivity for the fuel is determined by using the same procedure as documented in Section 3.4.1.1.2.

The three-dimensional periodic canister internal model consists of a periodic section of the canister internals. The model contains one support disk with two heat transfer disks (half thickness) on its top and bottom, the fuel assemblies, the fuel tubes and the helium in the canister, as shown in Figure 3.6.1.1-2. The purpose of this model is to compare the maximum fuel cladding temperatures of the following cases:

1) Base Case: All 24 positions loaded with standard CE 14x14 fuel assemblies
2) Case 2: 23 positions with standard fuel, with one consolidated fuel lattice in position 2
3) Case 3: 23 positions with standard fuel, with one consolidated fuel lattice in position 3
4) Case 4: 23 positions with standard fuel, with one consolidated fuel lattice in position 4
5) Case 5: 23 positions with standard fuel, with one consolidated fuel lattice in position 5

Positions 2, 3, 4, and 5 are shown in Figure 3.6.1.1-3. Based on symmetry, these locations represent all of the possible locations for consolidated fuel in the basket.

The fuel assemblies and fuel tubes are represented by homogeneous regions with effective thermal conductivities. The effective conductivities for the consolidated fuel are determined by the two-dimensional fuel assembly model discussed above. The effective conductivities for the CE 14x14 fuel assemblies are established based on the model described in Section 3.4.1.1.2. Properties for the fuel tubes are determined by the two-dimensional fuel tube model in Section 3.4.1.1.3. Volumetric heat generation corresponding to the design basis heat load of 0.83 kW per assembly is applied to the CE14x14 fuel regions in the model. Similarly, a heat generation rate corresponding to 0.279 kW is applied to the consolidated fuel assembly region. The heat conduction in the axial direction is conservatively ignored by assuming that the top and bottom surfaces of the model are adiabatic. A constant temperature of 400°F is applied to the outer surface of the model based on the PWR fuel maximum canister temperature of 399°F for the Transport Cask under normal conditions (Table 3.4-1). Steady state thermal analysis is performed for all five cases and the calculated maximum fuel cladding temperatures in the model are:

	Base Case	Case 2	Case 3	Case 4	Case 5
Maximum Fuel Cladding Temperature (°F)	714	695	698	701	700

The maximum temperatures for Cases 2 through 5 are less than that for the Base Case. Additionally, a sensitivity study analysis was performed to evaluate the effect of the number of stainless steel rods considered in the consolidated fuel assembly thermal model (see Figure 3.6.1.1-1). The thermal model was modified such that only the outer layer of rods are modeled as stainless steel rods with the remainder modeled as fuel rods. This differs from the model presented in Figure 3.6.1.1-1, which considers two layers of rods modeled as stainless steel. All five cases (i.e., "Base Case" and cases 2 through 5) were re-analyzed using the effective thermal conductivities generated by the modified thermal model. The results of this analysis show that the maximum fuel cladding temperature for each of the five cases is within 0.1°F of the results using the original consolidated fuel assembly thermal model. It is concluded that the thermal performance of the configuration consisting of 23 standard CE 14x14 fuel assemblies and one consolidated fuel lattice is bounded by that of the configuration consisting of 24 standard CE

14x14 fuel assemblies. These evaluations show that a consolidated fuel lattice can be located in any basket position, based on heat load. Conservatively, however, the consolidated fuel lattice is limited to loading in one of the four corner locations of the basket.

2. Standard CE 14x14 fuel assemblies with Control Element Assemblies

A Control Element Assembly (CEA) consists of a solid B_4C rod encapsulated in a stainless steel tube. The B_4C material has a conductivity of 1.375 BTU/hr-in-°F. With the CEA inserted into the guide tubes of the CE 14x14 fuel assembly, the effective conductivity in the axial direction of the fuel assembly is increased because solid material replaces helium in the guide tubes. The change in the effective conductivity in the transverse direction of the fuel assembly is negligible since the CEA is inside of the guide tubes. Note that the total heat load, including the small amount of extra heat generated by the CEA, remains below the design basis heat load. Therefore, the thermal performance of the fuel assemblies with CEAs inserted is bounded by that of the standard fuel assemblies.

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

The maximum number of fuel rods replaced by stainless steel rods is four (4) per assembly, which is about 2% of the total number of rods in each assembly (176). The change in the effective conductivity of the assembly is negligible. Note that the conductivity of the stainless steel is similar to that of Zircaloy and better than that of the UO_2. The resultant increase in effective conductivity of the repaired fuel assembly in the axial direction offsets the decrease in the effective conductivity in the transverse direction (due to slight reduction of radiation heat transfer at the surface of the stainless steel rods). The maximum number of fuel rods replaced by solid Zirconium rods is five (5) per assembly. Since the solid Zirconium rod has a higher conductivity than the fuel pin (UO_2 with Zircaloy clad), the effective conductivity of the repaired fuel assembly is increased. The thermal properties for the enriched fuel rod remain the same as for standard fuel rods, so there is no change in effective conductivity of the fuel assembly results from the use of fuel rods enriched to 1.95 wt % ^{235}U. These rods replace other fuel rods in the assembly after the first or second burnup cycles were completed. Therefore, these replacement fuel rods have been burned a minimum of one cycle less than the remainder of the assembly,

producing a proportionally lower per rod heat load. The heat load (on a per rod basis) of the fuel rods in a standard assembly bounds the heat load of the 1.95 wt % ^{235}U enriched fuel rods. Consequently, the loading of modified fuel assemblies is bounded by the thermal evaluation of the standard fuel assembly.

4. Standard fuel assemblies that have had the burnable poison rods removed and replaced with hollow Zircaloy tubes.

There are 16 locations where rods were removed and replaced with hollow Zircaloy tubes. Since the maximum heat load for these assemblies is 0.552 kW per assembly (two-thirds of the design basis heat load) and the number of hollow Zircaloy rods is only about one-tenth (16/176) of the total number of the fuel rods, the thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

5. Standard fuel assemblies with in-core instrument thimble assemblies stored within the center guide tube of each fuel assembly.

Storing an in-core instrument thimble assembly in the center guide tube of a fuel assembly will slightly increase the axial conductance of the fuel assembly (helium replaced by solid material). Therefore, there is no negative impact on the thermal performance of the fuel assembly with this configuration. The thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

6. Standard fuel assemblies that are designed with variable enrichment (radial) and axial blankets.

The thermal conductivities of the fuel assemblies with variable enrichment and axial blankets are considered to be essentially the same as those of the standard fuel assemblies. Since the heat load per assembly is limited to the design basis heat load, there is no significant effect on the thermal performance of the system due to this loading configuration.

7. Standard fuel assemblies that have fuel rods removed from the lattice.

There is one fuel assembly that has 107 rods removed. This fuel assembly has a heat load of 70 watts (only 8% of the design basis heat load of 0.83 kW). For the rest of fuel assemblies that have fuel rods removed from the lattice, the maximum number of removed fuel rods is 14, which is 8% (14/176) of the total number of rods in one fuel assembly. The maximum heat load for any one of these fuel assemblies is conservatively determined to be 0.63 kW. This heat load is 24% less than the design basis heat load of 0.83 kW. Therefore, the thermal performance for the configuration that contains standard fuel assemblies bounds that of the fuel assemblies with removed rods.

8. Damaged Fuel Assemblies

Damaged fuel assemblies are standard fuel assemblies with fuel rods that have known or suspected cladding defects greater than hairline cracks or pinhole leaks. Each damaged fuel assembly will be placed in a Maine Yankee fuel can. The primary function of the fuel can is to confine fuel material within the can and to facilitate handling and retrievability. The Maine Yankee fuel can is shown in Drawings 412-501 and 412-502. The placement of the loaded fuel cans is restricted by operating procedures and/or Technical Specifications to loading into the four fuel tube positions at the periphery of the fuel basket as shown in Figure 3.6.1.1-4. The heat load for each damaged fuel assembly is limited to the design basis heat load of 0.833 kW (20 kW/24).

A steady-state thermal analysis is performed using the three-dimensional cask model described in Section 3.4.1.1.1 simulating 100% failure of the damaged fuel rods held in the Maine Yankee fuel can. The canister is assumed to contain twenty (20) design basis PWR fuel assemblies and damaged fuel assemblies in fuel cans in each of the four corner positions.

A debris compaction length of 104 inches is considered in the analysis based on the volume of fuel rods and a 50% compaction of the debris. Additionally, this 104-inch debris region is assumed to be located at the center of the active fuel region of the design basis PWR fuel assemblies, as shown in Figure 3.6.1.1-4. The entire heat load for a single fuel assembly (i.e., 0.833 kW) is considered to be concentrated in the debris region. The effective thermal conductivities for the design basis PWR fuel assembly (Section 3.4.1.1.2) are used for the debris region. This is conservative since the debris (100% failed rods) is expected to have a higher density (better conduction) and more surface area (better radiation) than an intact fuel assembly.

In addition, the thermal conductivity of helium is used for the remainder of the active fuel length. Boundary conditions corresponding to normal transport are used at the outer surface of the cask (see Section 3.4.1.1.1). The results of the steady-state thermal analysis for 100% fuel rod, fuel cladding and guide tube failure are:

Description	Maximum Temperature (°F)			
	Fuel Cladding	Damaged Fuel	Support Disk	Heat Transfer Disk
Configuration with damaged fuel loaded in four basket corner locations	682	633	618	614
Design basis PWR fuel	673	N/A	608	605
Allowable	750	N/A	650	700

As shown in the previous table, the maximum temperatures for the fuel cladding, damaged fuel assembly, support disks, and heat transfer disks for the configuration with damaged fuel loaded in four (4) basket corner locations are within the allowable temperature range. Additionally, the maximum temperature of the support disk remains bounded by that used in the structural analyses of the fuel basket.

The effect of the compaction of the damaged fuel is most significant for the interior of the basket, and this effect is determined to be 10 °F, as shown in the table above. For the cask body closure lid seal, the effect of the damaged fuel is expected to be insignificant, since the transportable storage canister shield and structural lids, representing a thickness of 10 inches of steel, separate the fuel from the cask body closure lid seals. The canister lids act to spread any concentration of heat from the damaged fuel. The port cover seals are even more remote from the damaged fuel than the cask body lid seals and, therefore, are not considered to be affected by the damaged fuel.

Damaged high burnup fuel must be loaded into damaged fuel cans. These fuel assemblies have more than 1% of rods with oxide layers greater than 80 microns or more than 3% of rods with oxide layers greater than 70 microns and burnup greater than 45,000 MWD/MTU. The cask pressure for this condition is used as input to the containment analysis. Consistent with the containment analysis, a basket release fraction of 20% is applied. This release fraction accounts for up to 12 high burnup assemblies, including up to four classified as damaged. Applying this release fraction to the pressure evaluation in Section 3.4.4.1 yields a normal conditions cask pressure of 15.61 psig, calculated using B&W 17x17 Mark C fuel assembly parameters.

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Table 5.5.1.1-27 Additional Maine Yankee Non-Fuel Hardware – Transport Cask Source Assembly Surface Dose Rates – Accident Conditions – 1m Dose – 10-Year Cool Time from 1/1/1997

Group	Sb-Be Source Dose	Pu-Be Unirradiated Source Dose		Pu-Be Irradiated Source Dose	
	Gamma [mrem/hr]	Gamma [mrem/hr]	Neutron [mrem/hr]	Gamma [mrem/hr]	Neutron [mrem/hr]
1	0.00E+00	4.04E-12	7.22E-06	1.29E-14	2.31E-08
2	0.00E+00	2.27E-11	3.80E-04	7.28E-14	1.22E-06
3	0.00E+00	1.18E-10	9.62E-04	3.79E-13	3.08E-06
4	0.00E+00	2.63E-10	2.43E-04	8.42E-13	7.78E-07
5	0.00E+00	6.06E-10	1.46E-04	1.94E-12	4.67E-07
6	1.10E-07	4.25E-10	6.96E-05	1.95E-07	2.23E-07
7	3.79E-05	4.12E-10	5.61E-06	6.73E-05	1.80E-08
8	0.00E+00	2.94E-10	-	9.33E-13	-
9	2.22E-01	4.63E-13	-	3.94E-01	-
10	1.41E-01	1.06E-10	-	2.50E-01	-
11	4.89E-07	3.64E-11	-	1.28E-06	-
12	6.70E-10	1.44E-11	-	1.19E-09	-
13	4.23E-12	9.50E-16	-	7.49E-12	-
14	5.52E-17	2.41E-21	-	9.79E-17	-
15	6.41E-33	3.35E-37	-	1.14E-32	-
16	0.00E+00	0.00E+00	-	0.00E+00	-
17	0.00E+00	0.00E+00	-	0.00E+00	-
18	0.00E+00	0.00E+00	-	0.00E+00	-
Total	3.63E-01	2.30E-09	1.81E-03	6.44E-01	5.81E-06

5.5.1.2 Maine Yankee Site Specific GTCC Waste

Source terms, decay heat, and dose rates are calculated for the Maine Yankee Greater Than Class C (GTCC) waste to be transported in the Universal Transport Cask. The calculations are performed by first determining the gamma source spectra and decay heat from the GTCC isotopics, then using the gamma source information to calculate radial and axial dose rates. The results of these calculations show that the dose rates produced by the GTCC waste are bounded by the dose rates previously calculated for the design basis spent fuel, as presented in Section 5.1.3. Detailed results for the Maine Yankee GTCC waste are presented in Section 5.5.1.2.4.

5.5.1.2.1 GTCC Waste Transport Configuration

The GTCC waste basket is described in Section 1.3.1.1.2. The GTCC waste material is loaded into a cylindrical shell that is 3 inches thick. The cavity is divided into two loading sections using an insert that is placed into the GTCC basket after loading the bottom section. Each of the two sections may contain up to 10,000 pounds of waste, for a total of 20,000 pounds per canister.

5.5.1.2.2 GTCC Waste Source Term

The radionuclide inventories presented in Table 5.5.1.2-1 are utilized to develop the bounding source spectra and decay heat. These bounding radionuclide inventories are developed from information provided by the utility of all significant isotopes in the activated metal to be transported. The decay heat resulting from this source term is 4,490 watts.

The gamma source spectra presented in Table 5.5.1.2-2 is produced from the radionuclide inventory presented in Table 5.5.1.2-1. This information is used to derive the total gamma source strength which is utilized along with the homogenized source volume to calculate the uniformly distributed source strength per unit volume, as shown in Table 5.5.1.2-3.

The source volume information presented in Table 5.5.1.2-3 is based on the cylindrical region of the basket and the 77-inch (195.58 cm) length of each of the two GTCC canister loading cavities. These dimensions are used to determine the volumetric source strength and homogenized material densities that are used as input in the SAS1 sequence of the SCALE 4.3 computer code package to calculate external dose rates.

5.5.1.2.3 GTCC Waste Shielding Models

The radial and top axial models for the transport cask are described in the following sections. The relevant dimensions of the GTCC basket/canister within the transport cask are taken from the drawings presented in Section 1.3.4.

5.5.1.2.3.1 Radial Models

The 1-D radial models consist of a series of infinitely long concentric cylinders representing a cross section taken at a point along the length of the GTCC source cavity region of the GTCC basket/canister. The nested cylindrical regions consist of the following regions and cumulative thicknesses:

1.	Homogenized waste source region	61.900 cm
2.	GTCC basket stainless steel shield	7.620 cm
3.	Basket support disks	13.690 cm
4.	Canister stainless steel shell	1.588 cm
5.	Gap	0.698 cm
6.	Transport cask stainless steel inner shell	5.080 cm
7.	Transport cask lead gamma shielding	6.871 cm
8.	Transport cask lead gap	0.114 cm
9.	Transport cask stainless steel outer shell	6.985 cm
10.	Transport cask NS-4-FR neutron shielding	11.430 cm
11.	Transport cask stainless steel neutron shield shell	0.635 cm

The SAS1 input is generated using these regions, and specifying a buckling height of 391.16 cm which is equal to the two 77-inch (195.58 cm) loading cavity heights within the GTCC basket.

5.5.1.2.3.2 Top Axial Models

The 1-D SAS1 top axial models consist of a series of infinite slabs representing the GTCC waste canister and cask materials. Both the normal and accident conditions models are symmetric about the bottom of the top source region, assuming that the bottom region source does not penetrate the 3-inch separator plate and top source region sufficiently to contribute to the dose rate at the top of the cask. The slabs modeled for the top axial normal condition models are:

1	Homogenized waste source region	195.580 cm
2	Basket stainless steel top lid	3.810 cm
3	Canister stainless steel shield lid	17.780 cm
4	Canister stainless steel structural lid	7.620 cm
5	Transport Cask stainless steel lid	16.510 cm
6	Impact Limiter stainless steel inner shell	0.635 cm
7	Impact Limiter redwood	75.565 cm
8	Impact Limiter balsa wood	3.810 cm
9	Impact Limiter stainless steel outer shell	0.635 cm

The SAS1 input is generated using these regions, and specifying a buckling diameter of 170.4 cm, which is equal to the diameter of the canister. This buckling is conservative, as the value is larger than the actual source diameter of approximately 124 cm.

5.5.1.2.3.3 Bottom Axial Models

Like the 1-D top axial models, the 1-D bottom axial models represent the canister and cask materials as a series of infinite slabs. Both the normal and accident conditions models are symmetric about the top of the bottom source region, assuming that the top region source does not penetrate the 3-inch separator plate and bottom source region sufficiently to contribute to the dose rate at the bottom of the cask. The slabs modeled for the bottom axial normal condition models are:

1	Homogenized waste source region	195.580 cm
2	Basket stainless steel bottom plate	7.620 cm
3	Canister stainless steel bottom	4.445 cm
4	Transport cask spacer (modeled as void)	42.545 cm
5	Transport cask stainless steel inner bottom plate	10.795 cm
6	Transport cask NS-4-FR neutron shielding	2.540 cm
7	Transport cask stainless steel outer bottom plate	12.700 cm
8	Impact Limiter stainless steel inner shell	0.635 cm
9	Impact Limiter redwood	75.565 cm
10	Impact Limiter balsa wood	3.810 cm
11	Impact Limiter stainless steel outer shell	0.635 cm

The SAS1 input is generated using these regions, and specifying a buckling diameter of 170.4 cm, which is equal to the diameter of the canister. This buckling is conservative, as the value is larger than the actual source diameter of approximately 124 cm.

5.5.1.2.4 GTCC Waste Dose Rate Results

The models described in the previous sections were analyzed using the SAS1 1-D code. The resulting dose rates were multiplied by a 20% peaking factor to provide additional conservatism to the results. This conservative factor accounts for potential non-uniformity in the radionuclide distribution within the GTCC waste material and for potential shifting of the waste material during transport.

The Transport Cask containing Maine Yankee GTCC waste produces external dose rates below those calculated for the design basis PWR fuel loading. This ensures that the GTCC waste will meet all 10CFR71 normal and accident shielding requirements. During normal operation 10 CFR 71.47 requires the package surface dose rate be less than 200 mrem/hr and the dose rate two meters from the transportation vehicle not exceed 10 mrem/hr, including the conservative peaking factor.

As presented in Table 5.5.1.2-4, the highest calculated normal condition package surface dose rate is 33 mrem/hr, which occurs on the radial surface. Likewise, the highest package dose rate at 2 meters from the surface is 9.24 mrem/hr.

During accident conditions, 10 CFR 71.51 limits the dose rate to 1000 mrem/hr at 1 meter from the package surface. The maximum dose rate at 1 meter from the radial surface, as shown in Table 5.5.1.2-5, considering the combined accident of a loss of neutron shielding and lead slump, is 204 mrem/hr.

Table 5.5.1.2-1 Design Basis GTCC Source Term

Radionuclide	Curie Inventory
H-3	3.00E+02 Ci
C-14	1.50E+02 Ci
MN-54	3.50E+02 Ci
FE-55	2.00E+05 Ci
CO-58	1.00E+01 Ci
CO-60	2.90E+05 Ci
NI-59	8.20E+02 Ci
NI-63	9.00E+04 Ci
NB-94	1.00E+01 Ci
TC-99	1.00E+01 Ci
Total	5.82E+05 Ci

Table 5.5.1.2-2 Design Basis GTCC Gamma Source Spectra

Energy Range			Gamma/sec	MeV/sec	Normalized Gamma/sec
8.00E+00	to	1.00E+01	0.00E+00	0.00E+00	0.00E+00
6.50E+00	to	8.00E+00	0.00E+00	0.00E+00	0.00E+00
5.00E+00	to	6.50E+00	0.00E+00	0.00E+00	0.00E+00
4.00E+00	to	5.00E+00	0.00E+00	0.00E+00	0.00E+00
3.00E+00	to	4.00E+00	0.00E+00	0.00E+00	0.00E+00
2.50E+00	to	3.00E+00	1.76E+08	4.84E+08	7.96E-09
2.00E+00	to	2.50E+00	1.13E+11	2.55E+11	5.14E-06
1.66E+00	to	2.00E+00	1.75E+09	3.20E+09	7.92E-08
1.33E+00	to	1.66E+00	4.78E+15	7.15E+15	2.16E-01
1.00E+00	to	1.33E+00	1.69E+16	1.97E+16	7.66E-01
8.00E-01	to	1.00E+00	1.35E+13	1.21E+13	6.09E-04
6.00E-01	to	8.00E-01	3.92E+11	2.74E+11	1.77E-05
4.00E-01	to	6.00E-01	1.71E+11	8.57E+10	7.76E-06
3.00E-01	to	4.00E-01	9.12E+11	3.19E+11	4.13E-05
2.00E-01	to	3.00E-01	6.96E+11	1.74E+11	3.15E-05
1.00E-01	to	2.00E-01	1.40E+13	2.10E+12	6.33E-04
5.00E-02	to	1.00E-01	5.80E+13	4.35E+12	2.63E-03
1.00E-02	to	5.00E-02	2.95E+14	8.84E+12	1.33E-02
TOTALS			2.21E+16	2.69E+16	1.00E+00

Table 5.5.1.2-3 Design Basis GTCC Source Strength

Parameter	Value
Total Source Strength	2.21E+16 gamma/sec
Equivalent Source Radius	61.9 cm
Equivalent Source Length	391.16 cm
Equivalent Source Volume	4.709E+06 cc
Weight Volume Fraction	2.438E-01
Volumetric Source	4.69E+09 gamma/sec/cc

Table 5.5.1.2-4 GTCC Waste Dose Rate Results – Normal Conditions of Transpor

Location	Surface	2 Meters from Surface
Radial	32.66 mr/hr	9.24 mr/hr
Top Axial	0.21 mr/hr	0.06 mr/hr
Bottom Axial	3.11 mr/hr	0.91 mr/hr

Table 5.5.1.2-5 GTCC Waste Dose Rate Results – Accident Conditions

Location	1 Meter from Surface
Radial	203.94 mr/hr
Top Axial	1.18 mr/hr
Bottom Axial	23.51 mr/hr

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