TN 68 TRANSPORT PACKAGING

CHAPTER 3

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CHAPTER 3

THERMAL EVALUATION

3.1 Discussion

The TN-68 packaging is designed to passively reject decay heat under normal conditions of transport and hypothetical accident conditions while maintaining appropriate packaging temperatures and pressures within specified limits. Objectives of the thermal analyses performed for this evaluation include:

- Determination of maximum and minimum temperatures with respect to cask materials limits to ensure components perform their intended safety functions;
- Determination of temperature distributions to support the calculation of thermal stresses;
- Determination of the cask cavity gas temperature to support containment pressurization calculations;
- Determination of the maximum fuel cladding temperature.

Chapter 2 presents the principal design bases for the TN-68 packaging.

The design features of the TN-68 basket are described in Section 1.2. The basket consists of an assembly of 68 stainless steel fuel compartments with aluminum and neutron poison (borated aluminum or boron carbide / aluminum metal matrix composite) plates sandwiched between them. The compartments are joined by a fusion welding process to 1.75 in. wide stainless steel plates. Above and below the plates are slotted neutron poison plates which form an egg-crate structure. The aluminum basket rails are bolted to the basket periphery to provide a conduction path from the basket to the cavity wall. This thermal design feature of the basket allows the heat from the fuel assemblies to be conducted along the basket structure (including the neutron poison plates) to the basket rails and dissipated to the cask cavity wall.

Another thermal design feature is the conduction path created by the aluminum boxes in the neutron shielding layer described in Section 1.2. The neutron shielding is provided by a resin compound cast into long slender aluminum containers placed around the cask shell and enclosed within a smooth outer shell. By butting against the adjacent shell surfaces, the aluminum containers allow decay heat to be conducted across the neutron shield.

The design of the steel encased wooden impact limiters is described in Section 1.2. These components are considered in the thermal analysis because of their contribution as a thermal insulator. Since the impact limiters cover the lid and bottom regions of the cask, it is assumed that all decay heat is rejected radially out of the package. Similarly, the impact limiters protect the lid and bottom regions from the external heat load applied during the hypothetical thermal accident event.

A crib frame surrounds the TN-68 packaging preventing access to the outer surfaces of the cask body. The crib frame is sufficiently far away from the packaging such that it has a negligible effect on the heat dissipated from the package.

To establish the heat removal capability, several thermal design criteria are established for the TN-68. These are:

- Containment of radioactive material and gases is a major design requirement. Seal temperatures must be maintained within specified limits to satisfy the leak tight containment function during normal transport conditions. A maximum temperature limit of 536°F (280°C) is set for the Helicoflex seals (double metallic O-rings) in the containment vessel closure lid (Reference 10).

- To maintain the stability of the neutron shield resin during normal transport conditions, an allowable temperature range of -40 to 300° F (-40 to 149° C) is set for the neutron shield.

- Maximum temperatures of the containment structural components must not adversely affect the containment function.

- In accordance with 10CFR71.43(g) the maximum temperature of accessible package surfaces in the shade is limited to 185 °F (85 °C).

- The maximum fuel cladding temperature is determined as a function of the initial fuel age using the guidelines provided by the Commercial Spent Fuel Management Program[(1)](CSFM). The temperature threshold accounts for the effects of cladding temperature, decay time, burnup and fission gas build-up at 40 GWD/MTU. Waterside corrosion of 0.002 in.(radially) has been assumed. For normal conditions of transport, a fuel temperature limit of 343°C (649°F) has been established. During accident conditions, the fuel temperature limit is 570°C (1058°F) as recommended in Reference 12.

The ambient temperature range for normal transport is –20 to 100 °F (-29 to 38 °C) per 10CFR71(b). In general, all the thermal criteria are associated with maximum temperature limits and not minimum temperatures. All materials can be subjected to the minimum environment temperature of -40 °F (-40 °C) without adverse effects as required by 10CFR71(c)(2).

The TN-68 is analyzed based on a maximum heat load of 21.2 kW from 68 fuel assemblies. The analyses consider the effect of the decay heat flux varying axially along a fuel assembly. The heat flux profile for a BWR fuel assembly with a peak power factor of 1.2 and an active length of 144 in. is used for the evaluation. The use of the minimum active length of 144 in. along with a 1.2 peaking factor typically bounds the peak heat flux for the design basis fuel. A description of the detailed analyses performed for normal transport conditions is provided in Section 3.4 and accident conditions in Section 3.5. A summary of the analysis for normal conditions is provided in Table 3-1. The thermal evaluation concludes that with this design heat load, all design criteria are satisfied.

3.2 Summary of Thermal Properties of Materials

1. BWR Fuel

Temperature °F	Thermal Conductivity Btu/hr-in-°F		Specific Heat Btu/lb-F	Density lb/in^3
	Transverse	Axial		
195.8	0.0157		0.055	0.149
200.0		0.0580		
268.4	0.0178			
365.9	0.0206			
400.0		0.0646		
463.7	0.0239			
561.8	0.0277			
600.0		0.0709		
660.3	0.0319			
758.9	0.0367			
800.0		0.0769	0.055	0.149

The effective thermal conductivity is the lowest calculated value for the BWR fuel array that may be stored in this cask and corresponds to the GE 10x10 BWR assembly with channel. The fuel conductivity analysis is presented in Appendix 3.7.1. The values for BWR fuel density and specific heat are typical values obtained from Reference 13.

2. 6061 Aluminum (used for basket rails and shims) (Reference 2)

Temperature °F	Thermal Conductivity Btu/hr-in-°F	Specific Heat Btu/lb-F	Density lb/in^3
70	8.01	0.218	0.096
100	8.08	0.219	
150	8.17	0.223	
200	8.25	0.225	
250	8.32	0.228	
300	8.38	0.230	
350	8.44	0.233	
400	8.49	0.234	0.096

3. Poison Plates

Specific Heat Btu/lb-F	Density lb/in^3
0.214	0.098

Properties are from Reference 5 for aluminum. The thermal conductivities are specified in Section 3.3 for the neutron poison plates and will be verified by test.

4. Stainless Steel Type 304/304L (used for basket plates and fuel compartment) (Reference 2)

Temperature °F	Thermal Conductivity Btu/hr-in-°F	Specific Heat Btu/lb-F	Density lb/in^3
70	0.63	0.111	0.289
100	0.73		
200	0.78	0.124	
400	0.87	0.130	
600	0.94	0.134	
800	1.02	0.140	
1000	1.10		0.289

5. SA-350, Grade LF3 Carbon St (used for cask containment shell) (Reference 2)

Temperature °F	Thermal Conductivity Btu/hr-in-°F	Specific Heat Btu/lb-F	Density lb/in^3
70	1.97	0.106	0.283
100	1.99	0.110	
200	2.03	0.118	
400	2.02	0.128	
600	1.93	0.137	
800	1.81	0.149	
1000	1.67	0.165	
1200	1.52	0.189	
1400	1.28	0.406	0.283

6. Helium (used for gaps within the cask cavity) (Reference 8)

Temperature °F	Thermal Conductivity Btu/hr-in-°F
80	0.0073
440	0.0103
620	0.0116
800	0.0129
1160	0.0152

For the transient analyses, the thermal mass is relatively small and neglected. The density and specific heat are not used.

7. SA-516 Grade 70 Carbon Steel (used for gamma shield shell, outer shell and lid)(Reference 2)

Temperature °F	Thermal Conductivity Btu/hr-in-°F	Specific Heat Btu/lb-F	Density lb/in^3
70	1.91	0.109	0.280
200	1.98	0.118	
400	1.99	0.129	
600	1.91	0.139	
800	1.80	0.152	
1000	1.68	0.169	
1200	1.52	0.206	
1400	1.29	0.184	0.280

8. Air (Reference 5)

Temperature °F	Thermal Conductivity Btu/hr-in-°F	Prandtl Number	Kinematic Viscosity (in^2/sec)
-100	0.0009	0.740	0.0118
81	0.0013	0.708	0.0261
261	0.0016	0.694	0.0409
441	0.0019	0.680	0.0588
621	0.0022	0.680	0.0796
981	0.0028	0.689	0.1275

For the transient analyses, the thermal mass is relatively small and neglected. The density and specific heat are not used.

9. Neutron Shielding (Polyester Resin) (Reference 3)

Thermal Conductivity Btu/hr-in-°F	Specific Heat Btu/lb-F	Density lb/in^3
0.0083	0.311	0.057

The analyses use interpolated values when appropriate for intermediate temperatures where the temperature dependency of a specific parameter is deemed significant. The interpolation assumes a linear relationship between the reported values.

Thermal radiation effects at the external surface of the cask are considered. The thermal analysis assumes that the cask body external surface within the neutron shield length is painted white. The emissivity of white paint varies between 0.93-0.95 and the solar absorptivity varies between 0.12-0.18 (References 5 & 11). To account for dust and dirt, the thermal analysis uses a solar absorptivity of 0.30 and an emissivity of 0.90 for the exterior surfaces in the thermal models. After a fire, the cask surface will be partially covered in soot (emissivity = 0.95, Reference 9).

3.3 Technical Specifications for Components

The cask components for which a thermal technical specification is necessary are the seals and the neutron poison plates.

The seals used in the packaging are the Helicoflex seals (double metallic O-rings). The seals will have a minimum and maximum temperature rating of -40°F and 536°F respectively.

The neutron poison plates will have the following minimum conductivity:

Temperature		Conductivity	
°C	°F	W/m-°C	Btu/hr-in-°F
20	68	120	5.78
100	212	145	6.98
250	482	150	7.22
300	571	150	7.22

3.4 Thermal Evaluation for Normal Conditions of Transport

The normal conditions of transport are used for the determination of the maximum fuel cladding temperature, TN-68 component temperatures, confinement pressure and thermal stresses. These steady state environmental conditions correspond to the maximum daily averaged ambient temperature of 100°F and the 10CFR Part 71.71(c) insolation averaged over a 24 hour period.

3.4.1 Thermal Model

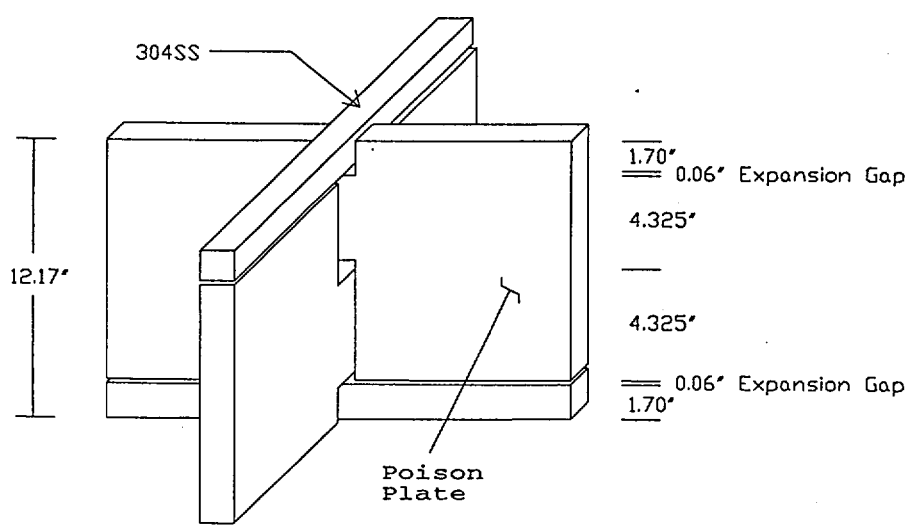
The TN-68 cask model is a finite element model developed using the ANSYS computer code (Reference 4). ANSYS is a comprehensive thermal, structural and fluid flow analysis package. It is a finite element analysis code capable of solving steady-state and transient thermal analysis problems in one, two or three dimensions. Heat transfer via a combination of conduction, radiation and convection can be modeled by ANSYS. The three-dimensional geometry of the cask was modeled. Solid entities were modeled by SOLID70 and SHELL57 three-dimensional thermal elements.

The three-dimensional model represents a 90° symmetric section of the TN-68 cask with basket, and includes the geometry and material properties of the basket components, the basket rails, the cask shells, the neutron shielding (resin in aluminum containers) and the outer shell (see Figures 3-1 and 3-2). The model simulates the effective thermal properties of the fuel with a homogenized material occupying the volume within the basket where the 144 inch active length of the fuel is stored. The length of model is 160 inches, which corresponds to the neutron shield length. Adiabatic boundary conditions are assumed at the ends of the model. The inner and outer cask body shells will be assembled with an interference fit. This will assure thermal contact at the shell interface. From Reference 5, the thermal interface conductance increases with contact pressure and reduces with rougher surface finish. At a minimal contact pressure of 5 psi, Reference 5 reports a conductance of 375 Btu/hr-ft^2. For further conservatism, this interface conductance is reduced to 200 Btu/hr-ft^2 for the thermal evaluation. The neutron shielding consists of 60 long slender resin-filled aluminum containers placed between the cask body and outer steel shell. The aluminum containers are confined between the cask body and outer shell, and butt against the adjacent shells. For conservatism, an air gap of 0.01 in. at thermal equilibrium is assumed to be present between the aluminum resin boxes and the adjacent shells. Radiation across these gaps is neglected. The finite element plot of the model is shown in Figure 3-4.

The basket structure consists of an assembly of 68 stainless steel boxes (6.00 in. square) joined by a proprietary fusion welding process to 1.75 in. wide stainless steel plates. Above and below the plates are slotted poison plates (0.31 in. thick) which form an egg-crate structure (see Figure 3-3). The steel boxes are modeled as 6.15 in. square, and the poison plates are modeled as 0.30 in. thick, to conservatively bound the maximum opening and plate thickness that may be fabricated. The welded design causes the poison plates to be tightly sandwiched between adjacent box sides. The basket portion of the thermal model simulates the conduction paths provided by the poison material and stainless steel plates, the stainless steel boxes and the fuel (modeled as a homogenous material). Aluminum basket rails, bolted to the basket periphery, increase the surface area for heat dissipation while providing structural support for the basket.

Generally, good surface contact is expected between adjacent components within the basket structure. However to bound the heat conductance uncertainty between adjacent components, the following gaps at thermal equilibrium are assumed:

a. 0.01 in. gap between an aluminum/stainless steel plate and its adjacent stainless steel fuel compartment.
b. 0.125 in. total gap between the rails and the basket periphery.
c. 0.125 in. thermal equilibrium gap between the basket rails and the cask cavity wall.
d. The gaps modeled in the axial direction and the axial lengths of the basket plates are shown in the schematic below:

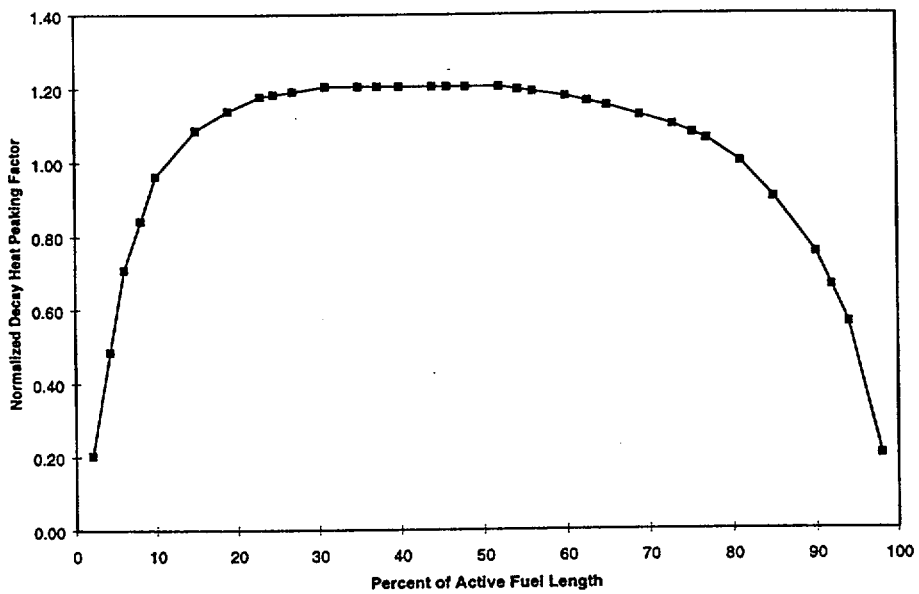


A 0.94 in. total width gap exists between the plates in the slots.

All heat transfer across the gaps is by gaseous conduction. Other modes of heat transfer are neglected.

Decay Heat Load

The decay heat load is applied as volumetric heat generation in the elements that represent the homogenized fuel. This heat load corresponds to a total heat load of 21.2 kW from 68 assemblies (0.312 kW/assy.) with a peaking factor of 1.2. A typical heat flux profile for spent BWR fuel with an axial peaking factor of 1.2 was used to distribute the decay heat load in the axial direction within the active length region of the model. This heat flux profile is shown below.



Heat Dissipation

Heat is transferred in this model by conduction through the cask wall and the aluminum encapsulated resin based neutron shielding to the outer shell. Heat is dissipated from the surface of the cask by a combination of radiation and natural convection.

For the purposes of determining the natural convection coefficient, the TN-68 package was assumed to behave as a horizontal cylinder of diameter 98 inches. Heat dissipation by natural convection from horizontal cylinders is described by the following equation for the average Nusselt number (Reference 6):

$$\overline{N}_{u_L} = \overline{H}_c \frac{L}{k} = 0.13(Gr_L Pr)^{1/3} \quad for\ Gr_L > 10^9$$

where,

Gr_L = Grashof number = $\rho^2 g\beta(T_s - T_a)L^3/\mu^2$
ρ = density, lb/ft^3
g = acceleration due to gravity, ft/sec^2

β = temperature coefficient of volume expansion, 1/R
μ = absolute viscosity, lb/ft-sec
L = characteristic length, ft
Pr = Prandtl number
H_c = natural convection coefficient

Simplifying in terms of H_c,

$$\overline{H}_c = (k)(0.13)[\rho^2 g\beta(T_s - T_a)Pr/\mu^2]^{1/3} \ Btu/hr\text{-}ft^2\text{-}^oF$$

The heat transfer coefficient, H_r, for heat dissipation by radiation, is given by the equation:

$$H_r = G_{12}\left[\frac{\sigma(T_1^4 - T_2^4)}{T_1 - T_2}\right] Btu/hr\text{-}ft^2\text{-}^oF$$

where,

G_{12} = the gray body exchange coefficient
= (surface emissivity) (view factor)
T_1 = ambient temperature, °R
T_2 = surface temperature, °R

The total heat transfer coefficient $H_t = H_r + H_c$, is applied as a boundary condition on the outer surfaces of the finite element model. No heat dissipation (adiabatic boundary condition) is assumed at the ends of the model.

Solar Heat Load

The total insolation for a 12-hour period in a day is 1475 Btu/ft^2 for curved surfaces per 10CFR Part 71.71(c). This insolation is averaged over a 24-hr period (daily averaged value) and applied as a constant steady state value (0.427 Btu/hr-in.^{2}) to all external surfaces of the thermal model. A solar absorptivity of 0.30 is used for the painted outer surface of the cask. Daily averaging of the solar heat load is justified based on the large thermal inertia of the TN-68 transport cask.

Maximum Accessible Temperature in the Shade

The accessible surface for the TN-68 packaging consists of the bars that comprise the crib frame and the outer surfaces of the impact limiters.

The crib frame, or equivalent barrier, surrounding the packaging will be sufficiently far away from the TN-68 packaging such that heat transfer to the barrier from the cask is negligible. In addition, as the impact limiters function as thermal insulators, heat transfer from the cask to the surface of the impact limiters is also negligible.

Therefore, the maximum accessible surface temperature of the enclosure in the shade will be 100 °F, the maximum ambient temperature during normal conditions of transport. This is well below the design criteria of 185 °F (85 °C).

Maximum Fuel Cladding Temperature

The finite element model includes a representation of the spent nuclear fuel that is based on a fuel effective conductivity model. The decay heat of the fuel with a peaking factor of 1.2 was applied directly to the fuel elements. The maximum fuel temperature reported is based on the results of the temperature distribution in the fuel region of the model. As described in Appendix 3.7.1, the homogenized fuel properties are chosen to match both the temperature drop between basket walls and fuel assembly center pin, and the effective conductivity of the fuel assemblies.

Average Cavity Gas Temperature

The cavity gas temperature is calculated using the temperatures at the hottest cross section of the cask. For simplicity and conservatism, it is assumed to be the average value of the maximum basket and cask inner shell temperatures.

3.4.2 Maximum Temperatures

A steady state thermal analysis is performed using the maximum decay heat load of 0.312 kW per assembly (21.2 kW total), 100°F ambient temperature and the maximum insolation. The TN-68 packaging temperature distribution is shown in Figure 3-5. The cask body temperature distribution as calculated in the TN-68 model is shown in Figure 3-7. Details of the temperature distribution in the basket region are shown in Figure 3-6. A summary of the calculated cask component temperatures is listed in Table 3-1.

3.4.3 Minimum Temperatures

Under the minimum temperature condition of -40°F (-40°C) ambient, the resulting packaging component temperatures will approach -40°F if no credit is taken for the decay heat load. Since the package materials, including containment structures and the seals, continue to function at this temperature, the minimum temperature condition has no adverse effect on the performance of the TN-68.

Temperature distributions in a minimum ambient temperature of -40°F and no insolation is performed. Table 3-2 lists the results of the analyses.

3.4.4 Maximum Internal Pressure

The maximum cask cavity internal pressure during normal conditions of transport is calculated in Chapter 4.

3.4.5 Maximum Thermal Stresses

The maximum thermal stresses during normal conditions of transport are calculated in Chapter 2.

3.4.6 Evaluation of Cask Performance for Normal Conditions of Transport

The thermal analysis for normal transport concludes that the TN-68 cask design meets all applicable requirements. The maximum temperatures calculated using conservative assumptions are low. The maximum temperature of any containment structural component is less than 300°F (149°C). The maximum seal temperature (234°F, 112°C) during normal transport is well below the 536°F long-term limit specified for continued seal function. The maximum neutron shield temperature is below 300°F (149°C) and no degradation of the neutron shielding is expected. The predicted maximum fuel cladding temperature is well within allowable fuel temperature limit of 649°F (343°C). The comparison of the results with the allowable ranges is tabulated below:

	Temperature, °F		
Component	**Maximum**	**Minimum**	**Allowable Range**
Seal	234	-40	-40 to 536
Neutron Shield	244	-40	-40 to 300
Fuel Cladding	490	-40	649 max.

3.5 Thermal Evaluation for Accident Conditions

The TN-68 cask is evaluated under the hypothetical accident sequence of 10CFR71.73. The front impact limiters protect the TN-68 cask lid containing the lid and port seals from the thermal accident environment. However, a direct conduction path exists from the cask surfaces exposed to the accident environment to the seal. A shield ring may be placed between the front impact limiter and the neutron shield shell if required for shielding. The shield ring would provide thermal protection during the fire accident and is neglected in the model. Since maintaining the seal temperature below 536 °F is a design requirement two analytical models are developed as discussed in sections 3.5.1 and 3.5.2.

3.5.1 Fire Accident Evaluation

The fire thermal evaluation is performed primarily to demonstrate the containment integrity of the TN-68. This is assured as long as the metallic lid seals remain below 536°F and the cavity pressure is less than 100 psig. Two models, a cask cross-section model and an axisymmetric model, are used for the evaluation. The first model represents the midsection of the cask, where the decay heat generation is at its highest. The second model represents the cask lid region details and upper portion of the cask body to evaluate the performance of the seal and aluminum boxes under fire conditions.

Analysis in Chapter 2 confirms that free drop and puncture damage do not measurably alter the thermal performance of the packaging. The steel encased wood impact limiters are locally deformed from the 30 foot drop, but they remain firmly attached to the cask. The impact limiters in their partially crushed condition still serve as effective thermal insulation. Under exposure to the thermal accident environment the wood at the periphery of the impact limiter shell would char but not burn. Hence, surfaces covered by the impact limiter are assumed to be insulated from the thermal event.

The Sandia Report reports an average convective heat transfer coefficient of 4.5 Btu/hr-ft^2-°F for a railroad tank car fire test (Reference 7). The same parameter is utilized for the fire accident evaluation.

3.5.2 Cask Cross Section Model.

The TN-68 finite element model developed in Section 3.4.1 is modified for the fire accident conditions. The cross section model is created by selecting the nodes and elements in the hottest region along with the temperature distribution. During the pre-fire phase, convection and radiation from the external surface of this model are as in normal conditions of transport (100°F ambient). During the fire phase, a constant convective heat transfer coefficient of 4.5 Btu/hr-ft^2-°F is used. Also the 0.01 in. gaps between the neutron shield and the adjacent shells were removed to assume contact to maximize heat input into the model from the fire. As per 10CFR71.73, a 30 minute 1,475°F flame temperature with an emittance of 0.9 and a surface absorptivity of 0.8 is used during the fire accident condition. See Section 3.4.1 for a detailed description of the model including the method used to calculate the maximum fuel cladding temperature and the average cavity gas temperature. The decay heat load used in this analysis

corresponds to a total heat load of 21.2 kW from 68 assemblies (0.312 kW/assy.) with a peaking factor of 1.2.

During the cooldown period after the fire condition, heat dissipation from the outer surface is by radiation and natural convection to an ambient temperature of 100°F. After a fire, the cask surface will be partially covered in soot (emissivity = 0.95, Reference 9). Per Reference 5, white paint has an emissivity of 0.93 and a solar absorptivity of 0.12-0.18. An emissivity of 0.9 was used for the cask external surfaces after the fire accident condition in order to bound the problem.

3.5.3 Axisymmetric Model

To demonstrate the integrity of the seals in the lid during the fire accident and that the aluminum resin boxes do not melt, an axisymmetric finite element model of the TN-68 is developed. The model includes the geometry and material properties of the lid with the upper portion of the cask body. Figure 3-8 shows the model details in the lid region. Figure 3-9 is an element plot of the 2-D axisymmetric ANSYS model.

An air gap (0.07 inch) is assumed between the lid and the cask body. Heat transfer between the cask and lid is via conduction across the air gaps. The environmental boundary conditions for the pre-fire, fire and post-fire are applied in the same manner as for the cask cross section model.

The front impact limiter acts as a thermal insulator during the fire accident. Regions of the lid and cask body outer surfaces covered by the impact limiter are modeled with adiabatic boundary conditions. The impact limiter spacer slightly reduces the portions of the cask body insulated by the impact limiter as shown in Figure 3-8.

The initial pre-fire temperature of each component within the model corresponds to its maximum temperature during normal conditions (see Table 3-1).

3.5.4 Summary of Results

Table 3-3 presents the maximum temperatures of the cask components during the fire event. The maximum temperatures calculated for the seal and the fuel cladding are 360°F and 552°F respectively. The transient average cavity gas temperature peaks at 432°F. The corresponding peak cavity pressure assuming 100% fuel failure is evaluated in Chapter 4 and is less than 100 psig. The maximum temperature distribution in the basket is shown in Figure 3-10. Figure 3-11 shows the temperature distribution of the axisymmetric model at the end of the fire.

3.5.5 Evaluation of Package Performance During Fire Accident Conditions

It is concluded that the TN-68 maintains containment during the postulated fire accident. The results of the analysis show that no melting of the aluminum resin boxes occurs. The neutron shield will off-gas during the fire event. A pressure relief valve is provided on the outer shell to prevent the pressurization of the outer shell. The shielding integrity of the neutron shielding is assumed to be lost after the fire event and the resulting accident dose rates have been evaluated

in Chapter 5. The maximum seal temperature is well below the 536°F long-term limit specified for continued seal function and the fuel cladding temperature is below the short-term limit (Section 3.1) of 1058°F (570°C).

A comparison of the results with the temperature limits is tabulated below:

Component	Temperature, °F	
	Maximum	Limit
Seal	360	536
Fuel Cladding	552	1058

3.6 References

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TABLE 3-1

COMPONENT TEMPERATURES IN THE TN-68 PACKAGE

Component	Normal Transport			Fire Accident	
	Maximum (°F)	Minimum* (°F)	Allowable Range(°F)	Peak(°F)	Allowable Range(°F)
Outer Shell	204	-40	**	998	**
Radial Neutron Shield	244	-40	-40 to 300	N/A	N/A
Inner Shell	262	-40	**	370	**
Basket Rail	319	-40	**	389	**
Basket Plate	469	-40	**	534	**
Gamma Shell	260	-40	**	379	**
Fuel Cladding	490	-40	649 max.	552	1058 max.
Cask Bottom	254	-40	**	326	**
Seal	234	-40	-40 to 536	360	-40 to 536
Average Cavity Gas	366	-40	N/A	422	N/A

* Assuming no credit for decay heat and a daily average ambient temperature of -20°F
** The components perform their intended safety function within the operating range.

TABLE 3-2

TEMPERATURE DISTRIBUTION IN THE TN-68 PACKAGE
(-40°F AMBIENT TEMPERATURE)

Component	Maximum Component Temperature
Outer Shell	80 °F
Aluminum Boxes	123 °F
Resin	123 °F
Gamma Shield Shell	141 °F
Inner Shell	143 °F
Basket Rails	206 °F
Fuel Cladding	388 °F
Cask Bottom Plate*	135 °F
Seal**	113 °F
Basket	362 °F

* Corresponds to maximum cavity wall temperature at the bottom end of cask body model.
** Corresponds to maximum cavity wall temperature at the top end of cask body model.

TABLE 3-3

MAXIMUM TRANSIENT TEMPERATURES DURING FIRE ACCIDENT

Component	Maximum Transient
Outer Surface	998°F (End of Fire)
Seal	360°F (2.0 hours)
Radial Neutron Shield	981°F (0.50 hours)
Gamma Shell	379°F (1.3 hours)
Basket Rail	389°F (4.9 hours)
Inner Shell	370°F (2.3 hours)
Basket	534°F (20.8 hours)
Fuel Cladding	552°F (21.3 hours)
Average Cavity Gas	422°F (2.3 hours)

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FIGURE 3-1

SCHEMATIC OF THE CASK BODY

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE 3-2

THERMAL MODEL
RADIAL CROSS SECTION

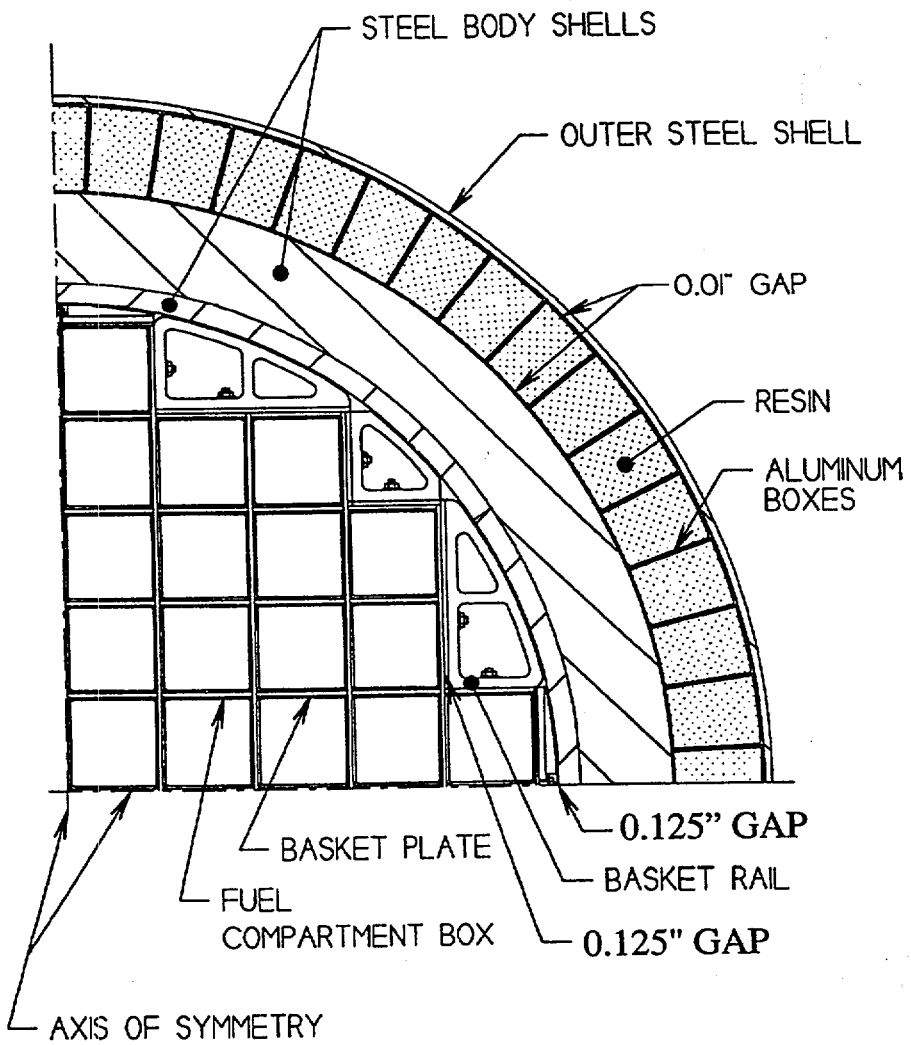

FIGURE 3-3

BASKET, PARTIAL ISOMETRIC VIEW

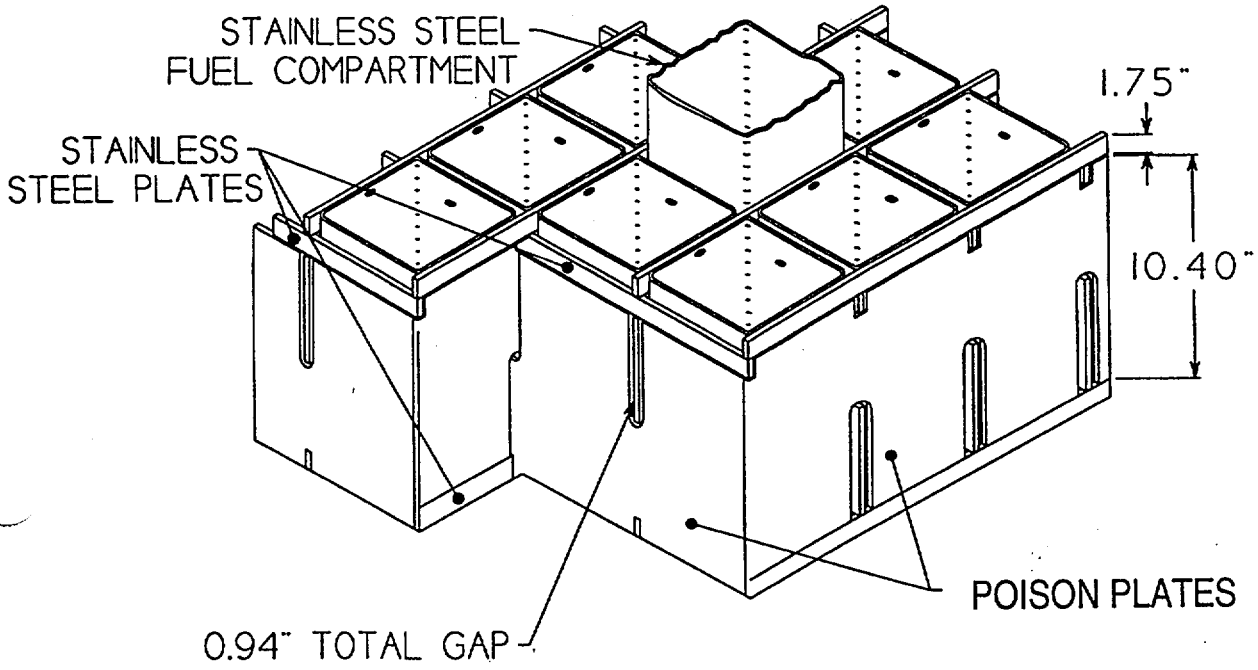

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FIGURE 3-4

FINITE ELEMENT PLOT OF THE TN-68 CASK THERMAL MODEL

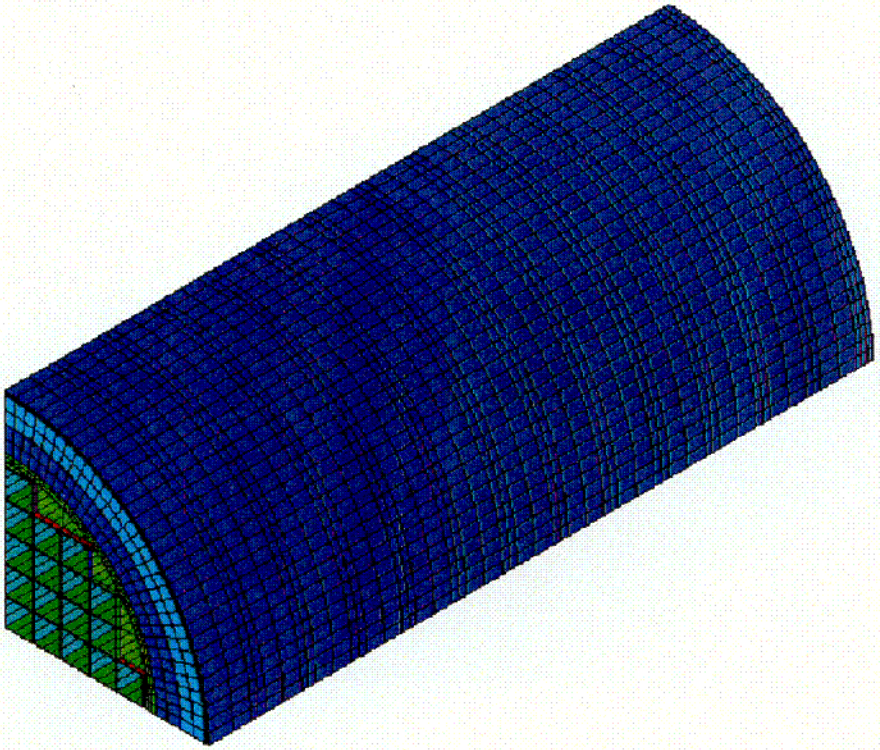

FIGURE 3-5

MAXIMUM TEMPERATURE DISTRIBUTION IN THE TN-68 CASK,
NORMAL CONDITIONS OF TRANSPORT

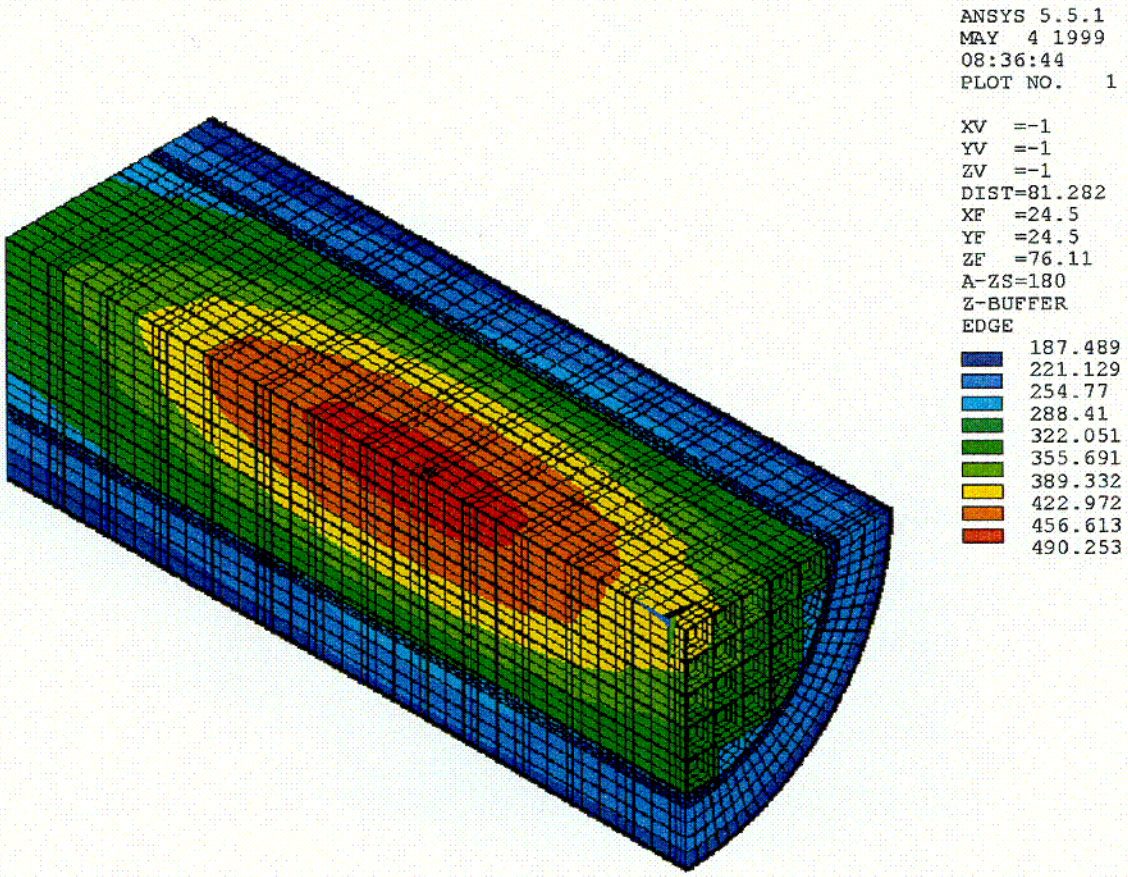

FIGURE 3-6

MAXIMUM TEMPERATURE DISTRIBUTION IN THE BASKET STRUCTURE, HOTTEST CROSS-SECTION

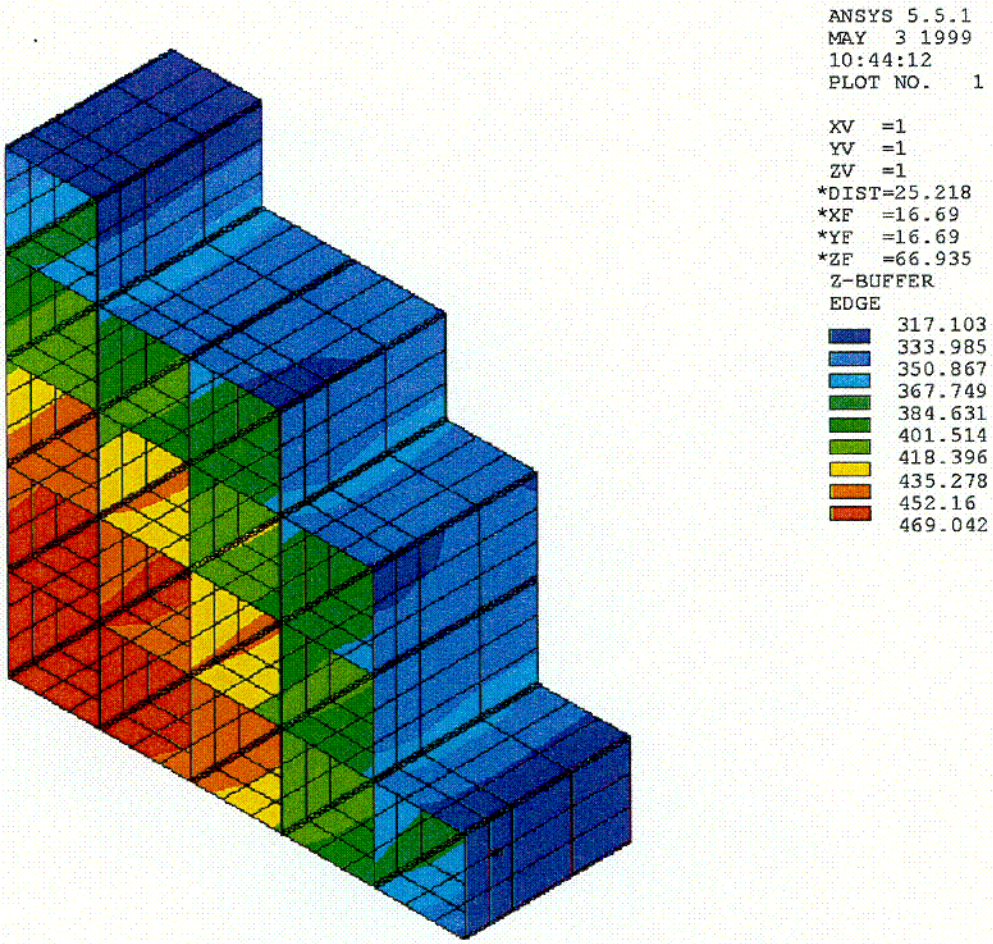

FIGURE 3-7

MAXIMUM TEMPERATURE DISTRIBUTION IN THE TN-68 CASK BODY, NORMAL CONDITIONS OF TRANSPORT

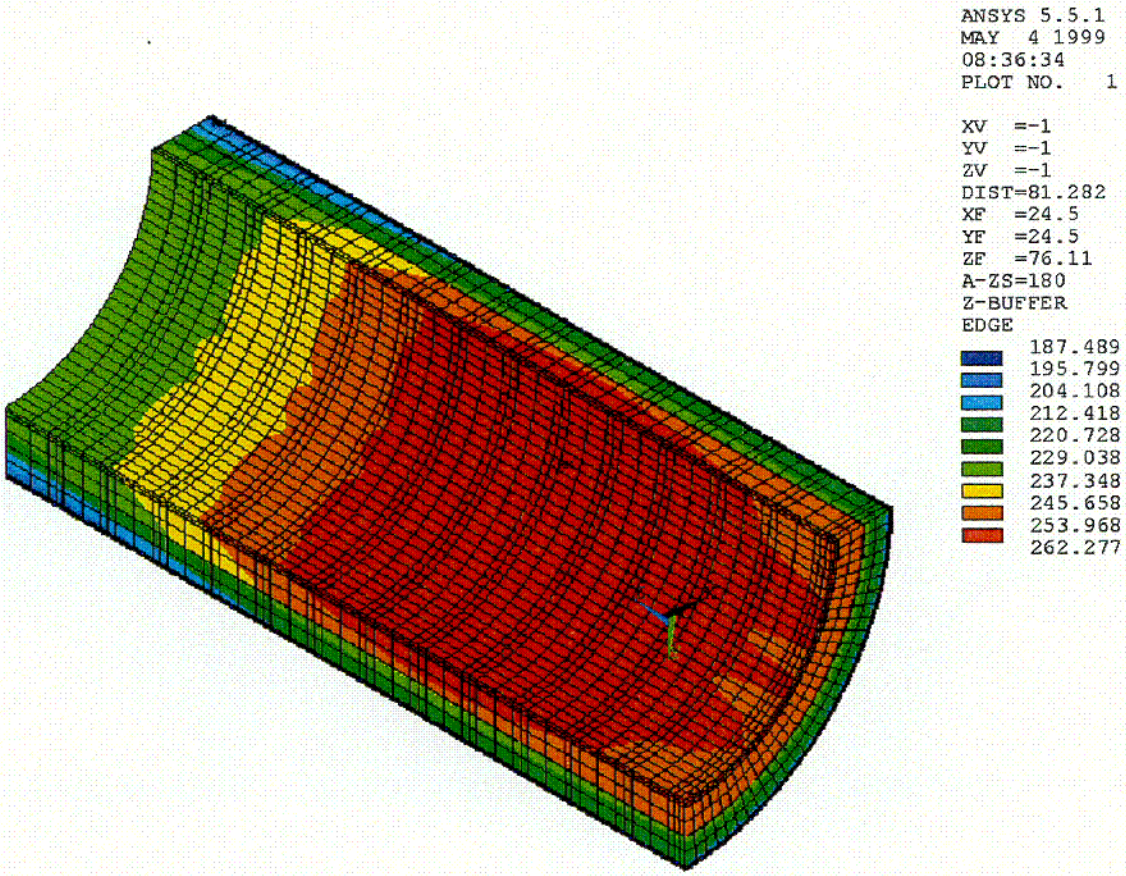

FIGURE WITHHELD UNDER 10 CFR 2.390

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FIGURE 3-9

FINITE ELEMENT PLOT OF AXISYMMETRIC THERMAL MODEL

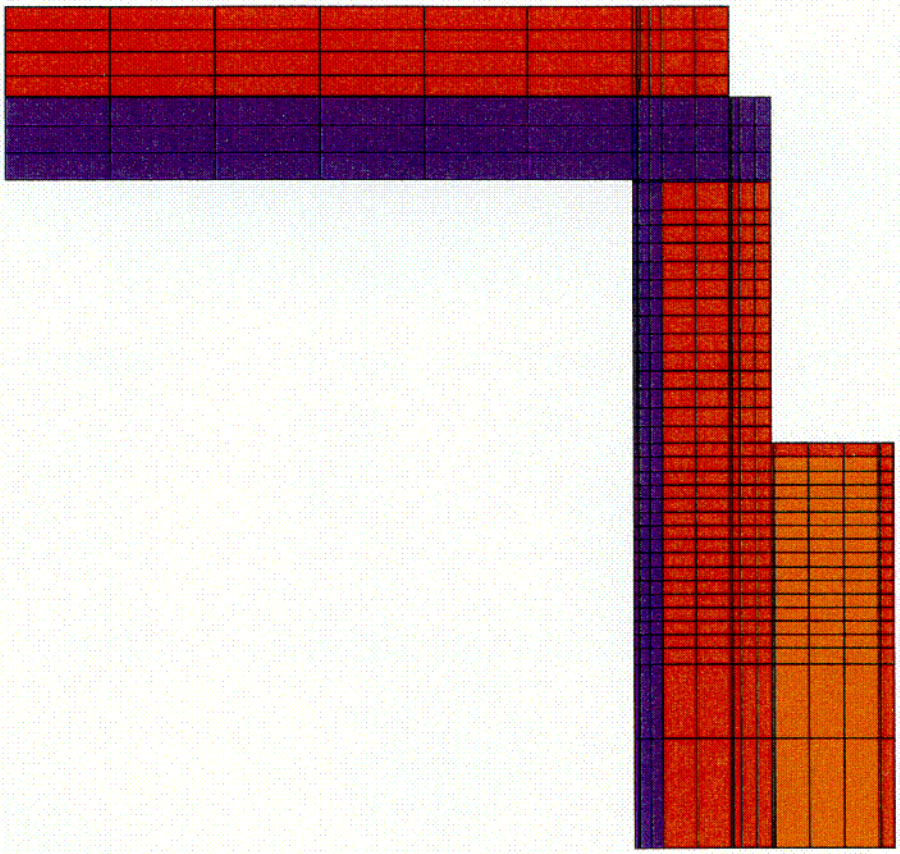


FIGURE 3-10

MAXIMUM TEMPERATURE DISTRIBUTION IN THE BASKET REGION DURING THE FIRE EVENT

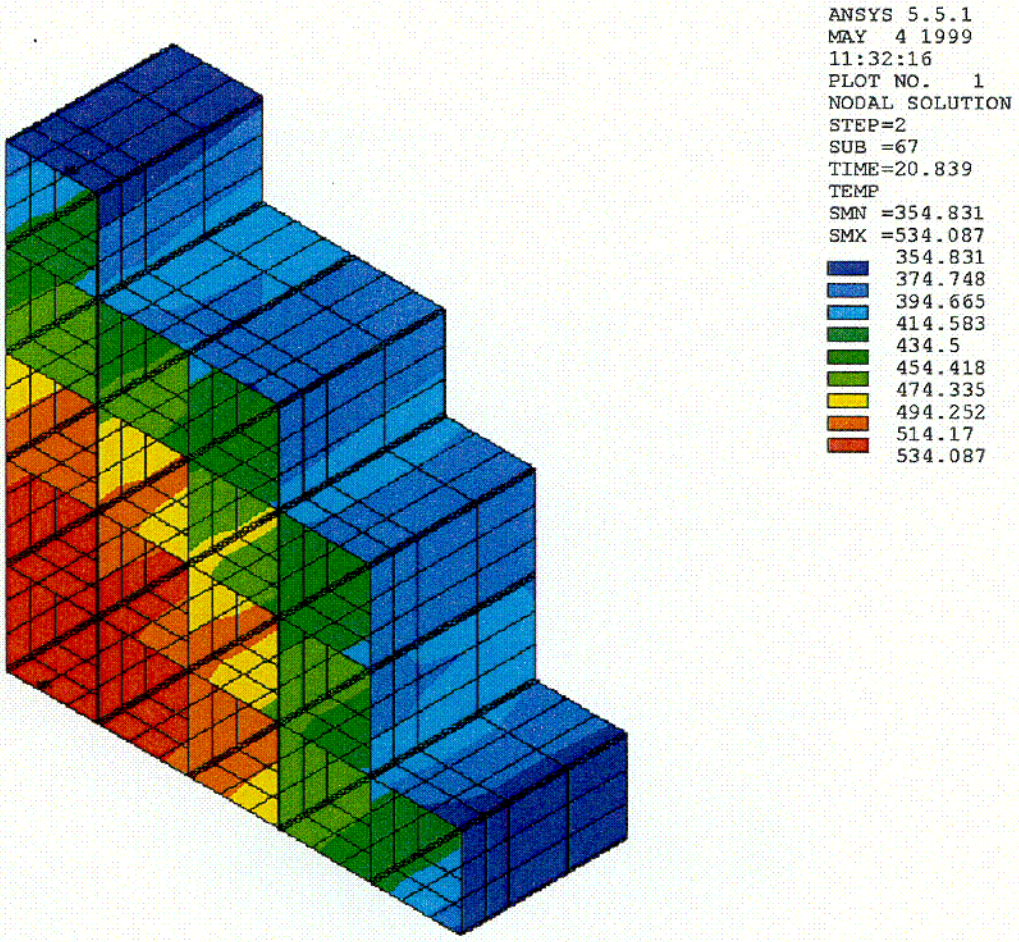
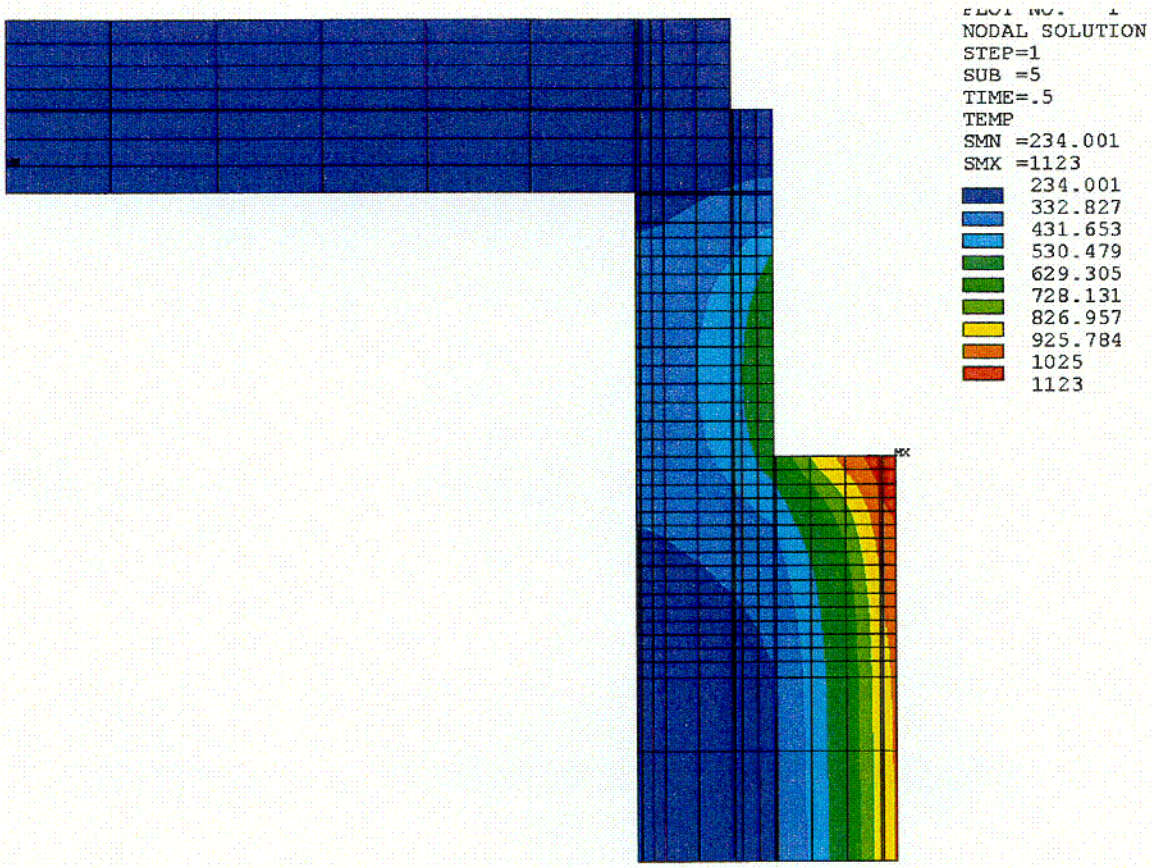


FIGURE 3-11

TEMPERATURE DISTRIBUTION IN THE AXISYMMETRIC MODEL
AT THE END OF FIRE



TN 68 TRANSPORT PACKAGING

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3.7.1-2 Finite Element Model (Element Plot)

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APPENDIX 3.7.1

EFFECTIVE THERMAL CONDUCTIVITY FOR THE FUEL ASSEMBLY

3.7.1.1 Discussion

In order to determine the appropriate effective fuel assembly thermal conductivity, a study was performed of the BWR fuel to be stored in the TN-68 cask to select the fuel assembly with the lowest effective thermal conductivity. Use of these properties would conservatively predict bounding maximum temperatures for the TN-68. The results of the study showed that the GE12 10 x 10 BWR fuel assembly with channel has the lowest conductivity.

3.7.1.2 Worst Case Fuel Payload

The TN-68 shall be capable of handling a wide variety of spent BWR fuel assemblies. The worst thermal case is the fuel assembly that will result in the hottest cladding temperature for a given fuel assembly heat load. The Wooton-Epstein Correlation (Reference 1) was used to determine which fuel assembly was the worst. The Wooton-Epstein Correlation for fuel stored in a horizontal configuration in an air based environment is defined as:

$$q = \sigma(C_1)(C_2)A(T_F^4 - T_G^4) + (0.118)A(T_F - T_G)^{4/3}$$

N = Number of pins per side
A = Surface Area of the Fuel Assembly Perimeter
= 4(Pin Diameter) x N x Active Length
C_1= Fuel Assembly Emissivity Factor
$= 1/(1/\varepsilon_{pin}+1/\varepsilon_{compartment}-1)$
C_2= Fuel Assembly Form Factor
$= 4/(N+2)$ if N is even, $4*N/(N+1)^2$ if N is odd
σ = Stefan-Boltzman Constant
$= 0.1714 \times 10^{-8}$ Btu/hr-ft^2-°R^4

Fuel cladding temperatures, T_F for the fuel assembly types that can be stored in the TN-68 were calculated assuming 312 W/assy. heat load and a compartment temperature, T_G, of 400°F. Note that for the purposes of this calculation, features such as water tubes and fuel channels are neglected. The results are tabulated below:

	Fuel Assembly Type			
	GE7x7	GE8x8	GE8x8R	10x10 (GE12)
Number of Pins	7	8	8	10
Pin Diameter	0.0469 ft	0.0411 ft	0.0403 ft	0.0337 ft
Active Length	12 ft	12 ft	12 ft	12 ft
Comp. Emissivity	0.3	0.3	0.3	0.3
Pin Emissivity	0.8	0.8	0.8	0.8
Area	15.764 ft^2	15.776 ft^2	15.456 ft^2	16.16 ft^2
C_1	0.279	0.279	0.279	0.279
C_2	0.4375	0.4000	0.4000	0.3333
Convection Mult	0.118	0.118	0.118	0.118
T_G	860°R	860°R	860°R	860°R
T_F	928.35°R	931.09°R	932.29°R	935.15°R
Q_{calc}	1164.86	1164.88	1164.84	1164.92

Therefore, the GE12 10x10 fuel assembly with channel represents the worst case fuel assembly for the thermal analysis.

3.7.1.3 GE12 10 x 10 Fuel Geometry Parameters

Fuel Array	10x10
Number of Fuel Rods	92
Number of Water Rods	2
Active Fuel Length, in.	144
Rod Pitch, in.	0.510
Rod Outside Diameter, in.	0.404
Clad Thickness, in.	0.026
Clad Material	Zr-2
Water Rod Material	Zr-2
Water Rod Outside Diameter	0.98
Water Rod Inner Diameter	0.92
Fuel Pellet Material	UO_2
Fuel Pellet Diameter, in.	0.345
Fuel Assy, Width w/Channel, in.	5.44
Channel Material	Zr-2
Channel Wall Thickness, in.	0.065 – 0.120

3.7.1.4 Summary of Properties

a. UO_2 (Reference 2)

Temperature		Thermal Conductivity, k	
(K)	(°F)	(W/m-K)	(Btu/hr-in-°F)
300	80.33	8.241	0.397
500	440.33	5.865	0.282
1000	1340.33	3.439	0.167

b. Zircaloy-2 (Reference 2)

Temperature		Thermal Conductivity, k	
(K)	(°F)	(W/m-K)	(Btu/hr-in-°F)
300	80.33	12.7000	0.6114
500	440.33	15.3000	0.7367
600	620.33	16.5000	0.7943
800	980.33	18.9000	0.9099

c. Helium (Reference 2)

Temperature		Thermal Conductivity, k	
(K)	(°F)	(W/m-K)	(Btu/hr-in-°F)
300	80.33	0.1520	0.0073
400	260.33	0.1842	0.0089
500	440.33	0.2138	0.0103
600	620.33	0.2415	0.0116
800	980.33	0.2927	0.0141

d. Stainless steel (Reference 2)

Emissivity, $\varepsilon_{ss} = 0.3$

e. Spent fuel rod (Reference 2)

Emissivity, $\varepsilon_r = 0.8$

3.7.1.5 Thermal Model

3.7.1.5.1 Transverse Effective Conductivity

The purpose of the effective conductivity in the transverse direction of a fuel assembly is to relate the temperature drop of a homogeneous heat generating square to the temperature drop across an actual assembly cross section for a given heat load. This relationship is established by the following equation obtained from Reference 3:

$$k_e = \frac{Q}{4L_a(T_o - T_s)} 0.2947$$

where:

k_e = Effective thermal conductivity (Btu/hr-in.-°F)
Q = Assembly head generation (Btu/hr)
L_a = Assembly active length (in.)
T_o = Maximum temperature (°F)
T_s = Surface temperature (°F)

A discrete finite element model of the fuel assembly in the TN-68 fuel compartment is developed using the ANSYS computer code (Reference 4). This two-dimensional model simulates heat transfer by radiation and convection and includes the geometry of the fuel rods, water rods and fuel channel. Helium is used as the fill gas in the fuel assembly. A fuel assembly decay heat load of 0.312 kW with a peaking factor of 1.2 is used for heat generation. An active length of 144 in. (366 cm) is assumed.

The finite element model is used to calculate the maximum radial temperature difference with isothermal boundary conditions. The helium fill gas and fuel are modeled using 2-D PLANE55 thermal solid elements, and the zircaloy fuel tubes and channel walls are modeled using 2-D LINK32 conducting bars. The LINK32 elements follow the circumference of the fuel rods to allow heat to flow around the individual rods and to set up the creation of the radiation super-element. Resistance to radial heat flow from each individual rod and channel wall due to the zircaloy tubes is assumed to be negligible. The result of variation within the channel wall thickness range is insignificant. The compartment wall is also modeled using LINK32 elements and used only to set up the surrounding surface for the creation of the radiation matrix super-element using the /AUX12 processor in ANSYS. The compartment wall elements are deleted prior to solution of the thermal problem. The thermal properties used are as described in Section 3.7.1.4, and the fuel assembly geometry is described in Section 3.7.1.3. Figure 3.7.1-1 is the schematic of the 10 x 10 BWR assembly. The ANSYS finite element model of the assembly is shown in Figure 3.7.1-2.

Several computational runs were made using isothermal boundary temperatures ranging from 175 to 750°F. The results of the finite element analysis and the effective fuel assembly conductivity calculation using the equation above are tabulated below:

Temperature (°F)			Effective Thermal Conductivity		
Boundary	Maximum	ΔT	Btu/hr-in-°F	Btu/hr-ft-°F	W/m-K
175	217	41	0.016	0.189	0.327
250	287	37	0.018	0.213	0.369
350	382	32	0.021	0.247	0.428
450	477	27	0.024	0.287	0.497
550	574	24	0.028	0.332	0.575
650	671	21	0.032	0.383	0.662
750	768	18	0.037	0.441	0.762

3.7.1.5.2 Axial Effective Conductivity

The axial fuel assembly conductivity is calculated by taking credit for the conduction paths provided by the fuel cladding, the water tube and the helium in the fuel compartment. The axial conductivity provided by the fuel pellets and fuel channel is conservatively neglected.

The effective conductivity weighted by its fractional area is:

$$K_e = \frac{1}{A_e}\left(K_{Zr-2}A_{Zr-2} + K_{He}A_{He} + K_{Wt}A_{Wt}\right)$$

where the cross section area of the individual components are:

Total fuel element cross sectional area: $A_e = (6.15)^2 = 37.8225\,\text{in.}^2$

The compartment size is calculated as 6.15 in. square instead of 6.00 in. square to conservatively bound the maximum opening that may be fabricated.

Zr-2 Cladding: $A_{Zr-2} = 92\frac{\pi}{4}\left((0.404)^2 - (0.404 - 2\times 0.026)^2\right) = 2.841\,\text{in}^2$

Water tube: $A_{Wt} = 2\frac{\pi}{4}\left((0.98)^2 - (0.92)^2\right) = 0.179\ \text{in}^2$

Helium: $A_{He} = A_e - A_{Zr\text{-}2} - A_{wt} - A_{fuel}$

$$A_{fuel} = 92\pi\left(\frac{0.345}{2}\right)^2 = 8.600\,\text{in}^2$$

$$A_{He} = (6.15)^2 - 2.841 - 8.600 - 0.179 = 26.203\,\text{in}^2$$

The results of the calculation of the effective axial conductivity as a function of temperature are tabulated below.

	Helium		Zircaloy		Water Tube	
Temp.	A_{He}	K_{He}	A_{Zr}	K_{Zr}	A_{Wt}	K_{Wt}
°F	in^2	Btu/hr-in-°F	in^2	Btu/hr-in-°F	in^2	Btu/hr-in-°F
200	26.203	0.0084	2.841	0.6531	0.179	0.6514
400	26.203	0.0100	2.841	0.7227	0.179	0.7183
600	26.203	0.0115	2.841	0.7878	0.179	0.7873
800	26.203	0.0128	2.841	0.8520	0.179	0.8520

3.7.1.6 Conclusion

The following minimum, temperature dependent, effective conductivities have been calculated for the BWR fuel to be stored in the TN-68 cask:

Average Temperature of Fuel °F	Thermal Conductivity Btu/hr-in-°F	
	Transverse	Axial
195.8	0.0157	
200.0		0.0580
268.4	0.0178	
365.9	0.0206	
400.0		0.0646
463.7	0.0239	
561.8	0.0277	
600.0		0.0709
660.3	0.0319	
758.9	0.0367	
800.0		0.0769

Note: The average fuel temperature is the linear average of the boundary and maximum temperatures reported in Section 3.7.1.5.

3.7.1.7 References

1. R. O. Wooton and H. M. Epstein, Heat Transfer from a Parallel Rod Fuel Element in a Shipping Container, Unpublished Report, Battelle Memorial Institute, 1963.

2. NUREG/CR-0497, *A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, MATPRO - Version 11* (Revision 2), EG&G Idaho, Inc., TREE-1280, August 1981.

3. *Spent Nuclear Fuel Effective Thermal Conductivity Report,* DI Number BBA000000-01717-5705-00010 Rev. 0, CRWMS M&O, Prepared for US DOE, Yucca Mountain Site Characterization Project Office, July 1996.

4. ANSYS, Inc., ANSYS Engineering Analysis System User's Manual for ANSYS Revision 5.5, Houston, PA

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FIGURE 3.7.1-1

SCHEMATIC OF 10 X 10 BWR ASSEMBLY

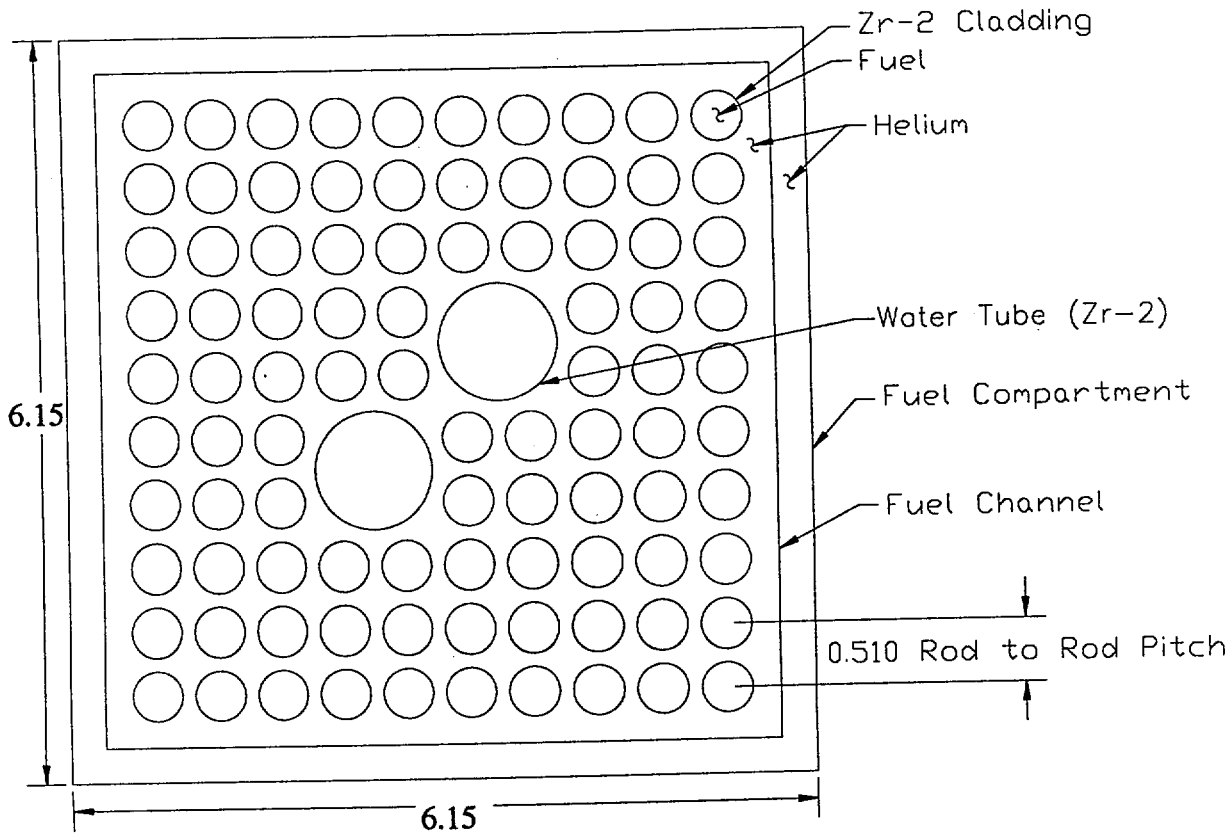
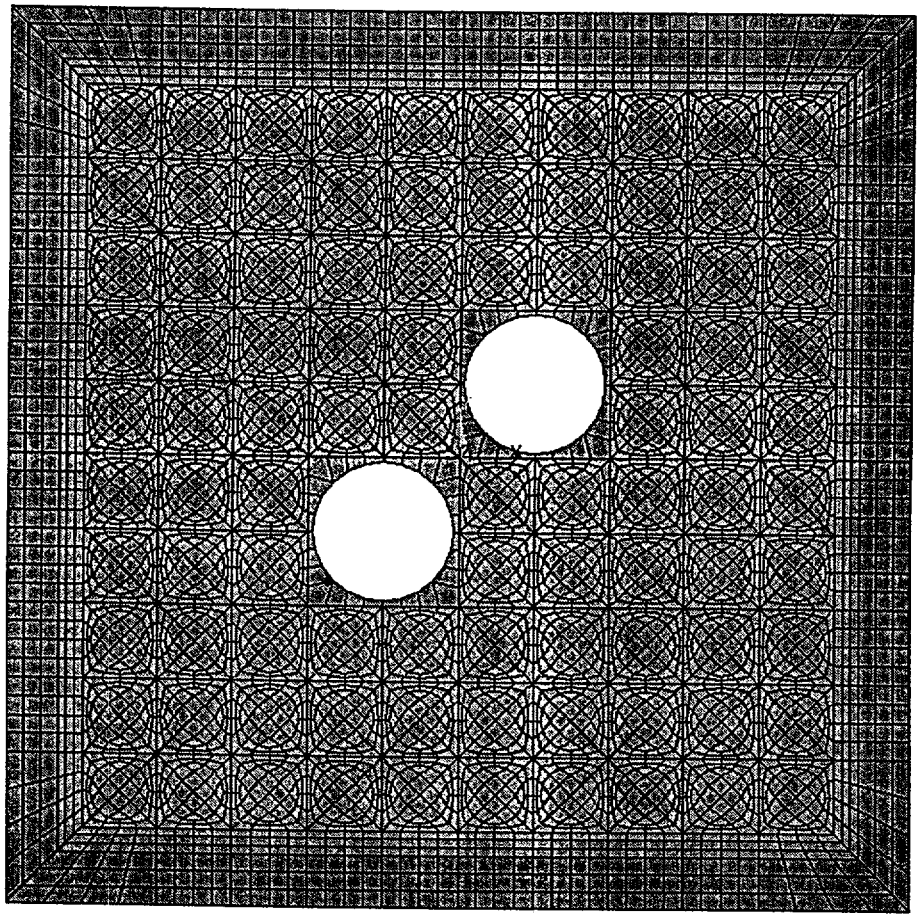


FIGURE 3.7.1-2

FINITE ELEMENT MODEL (ELEMENT PLOT)

1



TN 68 TRANSPORT PACKAGING

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CHAPTER 4

CONTAINMENT

4.1 CONTAINMENT BOUNDARY

The containment boundary consists of the inner shell and bottom plate, shell flange, lid outer plate, lid bolts, penetration cover plate and bolts and the inner metallic O-rings of the lid seal and the two lid penetrations (vent and drain). The containment boundary is shown in Figure 4-1. The construction of the containment boundary is shown on drawings 972-71-2, 3 and 4 provided in Chapter 1. The containment vessel prevents leakage of radioactive material from the cask cavity. It also maintains an inert atmosphere (helium) in the cask cavity. Helium assists in heat removal and provides a non-reactive environment to protect fuel assemblies against fuel cladding degradation which might otherwise lead to gross rupture.

4.1.1 Containment Vessel

The TN-68 containment vessel consists of: an inner shell which is a welded, carbon steel cylinder with an integrally-welded, carbon steel bottom closure; a welded flange forging; a flange and bolted carbon steel lid with bolts; and vent and drain covers with bolts. The overall containment vessel length is 189.0 in. with a wall thickness of 1.5 in. The cylindrical cask cavity has a diameter of 69.5 in. and a length of 178 in.

The containment shell and bottom closure materials are SA-203 Grade E and the shell flange is SA-350 Grade LF3. The containment lid material is SA-203 Grade E or SA-350 Grade LF3.

The cask design, fabrication and testing are performed under Transnuclear's Quality Assurance Program which conforms to the criteria in Subpart H of 10CFR71.

The materials of construction meet the requirements of Section III, Subsection NB-2000 and Section II, Material specifications or the corresponding ASTM Specifications. The containment vessel is designed to the ASME Code, Section III, Subsection NB, Article 3200. The containment vessel is fabricated and examined in accordance with NB-2500, NB-4000 and NB-5000. Also, weld materials conform to NB-2400 and the material specification requirements of Section II, Part C of ASME B&PV.

The containment vessel is hydrostatically tested in accordance with the requirements of the ASME B&PV Code, Section III, Article NB-6200 with the exception that the containment vessel is installed in the gamma shield shell during testing. The containment vessel is supported by the gamma shield during all design and accident events.

Even though the code is not strictly applicable to transport casks, it is the intent to follow Section III, Subsection NB of the Code as closely as possible for design and construction of the containment vessel. The casks may, however, be fabricated by other than N-stamp holders and materials may be supplied by other than ASME Certificate Holders. Thus the requirements of

NCA are not imposed. TN's quality assurance requirements, which are based on 10CFR71 Subpart H and NQA-1 are imposed in lieu of the requirements of NCA-3850. This SAR is prepared in place of the ASME design and stress reports. Surveillances are performed by TN and utility personnel rather than by an Authorized Nuclear Inspector (ANI).

The weld of the bottom inner plate to the containment shell is a Category C, Type 2 corner weld in accordance with Figure NB-4243-1 of the ASME Code. In accordance with NB-5231, Type 2 Category C full penetration corner welded joints require the fusion zone and the parent metal beneath the attachment surface to be ultrasonically examined after welding. If this weld is performed on the containment vessel after assembly with the outer shell, the UT inspection will be performed on a best efforts basis. It may not be possible to do a complete UT inspection, since the outer diameter of the shell is inaccessible. The joint will be examined by the radiographic method and either the liquid penetrant or magnetic particle methods in accordance with the ASME Code Subsection NB.

Paragraph NB-4213 requires the rolling process used to form the inner vessel be qualified to determine that the required impact properties of NB-2300 are met after straining by taking test specimens from three different heats. If the plates are made from less than three heats, each heat will be tested to verify the impact properties.

The materials of the TN-68 packaging will not result in any significant chemical, galvanic or other reaction as discussed in Chapter 2.

4.1.2 Containment Penetrations

There are two penetrations through the containment vessel, both in the lid. One is the drain port and the other is the vent port. A double O-ring seal mechanical closure is provided for each penetration. Each penetration contains a quick disconnect coupling for ease of operation.

4.1.3 Seals and Welds

The containment boundary welds consist of the circumferential welds attaching the bottom closure and the top flange to the vessel shell. Also, the longitudinal weld(s) on the rolled plate, closing the cylindrical vessel shell, and the circumferential weld(s) attaching the rolled shells together are containment welds.

Double metallic seals are utilized on the lid and the two lid penetrations. Helicoflex HND or equivalent seals may be used. The seals are shown in Figure 4-2. The Helicoflex metallic face seals of the lid and lid penetrations possess long-term stability and have high corrosion resistance. These high performance seals are comprised of two metal linings formed around a helically-wound spring. Additionally, all metallic seal seating areas are stainless steel overlayed for improved surface control. The overlay technique has been used for Transnuclear's storage and transport casks.

The metallic seals consist of an inner spring, a lining, and a jacket. The spring is Nimonic 90 or and equivalent material. The lining and jacket are stainless steel and aluminum, respectively.

The internal spring and lining maintain the necessary rigidity and sealing force, and provide some elastic recovery capability. The outer aluminum jacket provides a ductile material against the sealing surfaces. The jacket also provides a connecting sheet between the inner outer seals. Holes in this sheet allow for attachment screws and for communication between the OP transport cover and the space between the seals. This sheet, which is about 0.020 inch thick, has insufficient strength to transmit radial forces great enough to overcome the axial compressive forces on the seals, which are over 1000 lb/inch of seal length. The overpressure port seal is a single metallic seal of the same design, Helicoflex HN200 or equivalent.

All TN-68 surfaces which mate with the metallic seals are stainless steel.

The lid and penetration seals described above are contained in grooves. A high level of sealing over the storage period is assured by utilizing seals in a deformation-controlled design. The deformation of the seals is constant since bolt loads assure that the mating surfaces remain in contact. The seal deformation is set by the original diameter and the depth of the groove.

The nominal diameter of the lid seal is 6.6 mm, and the nominal groove depth is 5.6 mm. At 1 mm compression, the sealing force is 245 N/mm (1399 lb/inch)[1]. The total force of the double seal is 633,800 lb. The total preload of the 48 lid bolts is 6,490 kips, which is greater than the combined force of the seals, internal pressure, and normal conditions impact load 1,737 kips (Appendix 2.10.2).

The nominal diameter of the port seals is 4.1 mm, and the nominal groove depth is 3.2 mm. At 0.9 mm compression, the sealing force is 200 N/mm (1142 lb/inch). The total force of the double seal is 37,900 lb. The total preload of the 8 cover bolts is 64,000 lb, which is greater than the combined force of the seals and internal pressure, 40,000 lb.

The sealing force is maintained by the seal's internal spring. Due to creep, the sealing pressure decreases with increasing temperature as shown in the following table[11]. The ratios P_T/P_{20} compare the seal pressure temperature T °C to the seal pressure at 20°C. The long-term temperature limit is the point at which the sealing pressure becomes zero due to creep (P_{Tmax} = 0). The maximum normal temperature experienced by the seals in the TN-68 is 243 °F (Table 3–1).

Seal	$P_{119\,C}/P_{20\,C}$ (119 °C = 247 °F)	$P_{200\,C}/P_{20\,C}$ (200 °C = 392 °F)	Temperature limit
Lid, 6.6 mm	(439/670) = 66%	(250/670) = 37%	340 °C (644 °F)
Ports, 4 mm	(364/600) = 61%	(170/600) = 28%	280 °C (536 °F)

P_{20C} and P_{200C} from Reference 1; $P_{119\,C}$ by linear interpolation; sealing pressure P in N/mm^2 (referred to as "intrinsic power P_u" in reference 1).

Helicoflex metallic seals are all capable of limiting leak rates to less than 1 x 10^{-7} ref cm^3/sec. After loading, all lid and cover seals are leak tested in accordance with ANSI N14.5. The acceptable total cask leakage (both inner and outer seals combined) is 1 x 10^{-5} ref cm^3/sec.

4.1.4 Closure

The containment vessel contains an integrally-welded bottom closure and a bolted and flanged top closure (lid). The flanged lid plate is attached to the cask body with 48 bolts. The bolt torque required to seal the metallic seals located in the lid and maintain confinement under normal and accident conditions is provided in Drawing 972-71-2. The closure bolt analysis is presented in Appendix 2.10.2.

As previously mentioned, the lid contains two penetrations which are sealed by flanged covers fastened to the lid by 8 bolts each. The bolt torque required to seal the metallic seals in the penetration covers and maintain confinement under normal and accident conditions is provided in Drawing 972-71-2.

4.2 REQUIREMENTS FOR NORMAL CONDITIONS OF TRANSPORT

In accordance with 10 CFR 71.51, a Type B package must be designed, constructed and prepared for shipment so that "no loss or dispersal of radioactive contents, as demonstrated to a sensitivity of 10^{-6} A_2 per hour" will occur under the tests specified in 10 CFR 71.71 for normal conditions of transport.

The guidelines of ANSI N14.5 were used to determine the leakage test criteria which demonstrate that the TN-68 meets the "no-loss" requirements of 10 CFR 71.51.

4.2.1 Containment of Radioactive Material

4.2.1.1 Source Terms

Three sources are considered to determine the releasable airborne material from the TN-68 cask (Reference 4, NUREG/CR-6487).

- Residual activity on the cask interior surfaces as a result of loading operations (and, if applicable, previous shipments);
- Fission and activation-product activity associated with corrosion-deposited material (crud) on the fuel assembly surfaces, and
- Radionuclides within the individual fuel rods comprising the fuel assemblies.

The first source, residual contamination on the interior surfaces of the cask is neglected. Reference 4 indicates that this is negligible as compared to the crud deposition on the fuel rods.

The second source, crud, is basically the radioactive "flaky" material that is formed on the outside surface of the fuel rods due to the radioactive and corrosive environment of the BWR reactor. This material can be loosely bound to the fuel rod surface and may be dislodged during transportation and be available for release from the cask.

The third source is from the fuel itself. A breach in the fuel cladding may allow radionuclides to be released from the fuel to the interior of the cask. There are three types of releases associated with the breaches in the fuel rod cladding: gaseous radionuclides, volatiles and fuel fines.

As a conservative approach, to simplify calculations, it is assumed that crud spallation and cladding breaches occur instantaneously after fuel loading and closure operations. Therefore, all radioactivity is readily available for release if a leak occurs.

Source Activity from the Fuel

The fuel transported in the TN-68 transport packaging may have an initial enrichment of 3.3 wt% U-235, 40,000 MWD/MTU bundle average exposure and 10 year cooled provided the fuel acceptance criteria of Section 1.2.3 have been met. It is conservative to assume that 7x7

assemblies with these fuel parameters are loaded in the TN-68 cask transport packaging. (The 7x7 fuel assemblies have the largest initial uranium loading of all the BWR fuel types analyzed for the TN-68 and therefore the greatest source term.) The radionuclide inventory is presented in Table 4-1 (Reference 5).

Source Activity from Release of Volatiles

The source activity concentration inside the TN-68 due to the release of volatiles is calculated using the following equation (Reference 4).

$$C_{volatiles} = \{N_A f_B A_V f_v\} / V$$

where: N_A = number of assemblies
f_B = fraction of rods that develop cladding breaches
A_V = specific activity of volatiles in the fuel assembly, Ci/assembly
f_V = fraction of volatiles in a fuel rod released if a fuel rod develops a cladding breach
V = void volume inside the containment vessel, cm^3

Table 4-2 presents the results of this calculation.

Source Activity from Release of Gaseous Isotopes

The source activity concentration inside the TN-68 cask due to the release of gaseous isotopes is calculated using the following equation (Reference 4).

$$C_{gases} = \{ N_A f_B A_G f_G \} / V$$

where: N_A = number of assemblies
f_B = fraction of rod that develop cladding breaches
A_G = specific activity of gases in the fuel assembly, Ci/assembly
f_G = fraction of gases in a fuel rod released if a fuel rod develops a cladding breach
V = void volume inside the containment vessel, cm^3

Table 4-2 presents the results of this calculation.

Source Activity from Release of Fuel Fines

The source activity concentration inside the TN-68 due to the release of fuel fines is calculated using the following equation (Reference 4).

$$C_{gases} = \{ N_A f_B A_F f_F \} / V$$

where: N_A = number of assemblies
f_B = fraction of rod that develop cladding breaches

A_F = specific activity of fuel fines in the assembly, Ci/assembly
f_G = fraction of fuel fines released if a fuel rod develops a cladding breach
V = void volume inside the containment vessel, cm^3

Table 4-2 presents the calculated concentration of fuel fines inside the TN-68 for normal transport conditions.

Source Activity due to Crud Spallation

The fuel transported in the TN-68 transport packaging may be cooled a minimum of 10 years (provided the fuel acceptance criteria of Section 1.2.3 have been met). The activity density that results inside of the TN-68 as a result of crud spallation is calculated using the equation below (Reference 4).

$$C_{crud} = \{ f_C S_C N_R N_A S_{AR} \}$$

where

C_{crud} = activity density inside containment vessel as a result of crud spallation, Ci/cm^3
f_C = crud spallation factor
V = free volume inside the containment vessel, cm^3
S_C = crud surface activity, Ci/cm^2
N_R = number of fuel rods per assembly
N_A = number of assemblies in the cask
d = rod outer diameter, cm
r = rod outer radius, cm
l = rod length, cm
S_{AR} = surface area per rod, cm^2

The 10 x 10 fuel assembly is used to determine the maximum crud source since it has the largest overall surface area. Table 4-2 presents the results of this calculation.

4.2.1.2 Determination of A_2 Values

The A_2 value of a mixture of radioactive nuclides is determined as follows:

$$A_{2\,mixture} = [\Sigma (f_i / A_{2i})]^{-1}$$

where: f_i is the fraction of total activity due to isotope i, and
A_{2i} is the A_2 value for isotope i.

Using the methodology of 10 CFR 71 and Reference 4, the A_2 values are determined for each source (Table 4-3). The results provided in Tables 4-2 and 4-3 are combined to determine an effective A_2 for the TN-68 (Table 4–4).

4.2.1.3 Determination of Permissible Leakage Rates

To determine the leakage rates, the four sources are combined to for the total source term:

$$C_{total} = C_{crud} + C_{volatiles} + C_{gases} + C_{fines}$$

From Refence 5, the permissible release rate, R, from the TN-68 is:

$$R = L \times C$$

where:

L = volumetric gas leakage rate (cm^3 / s)
C = curies per unit volume of the radioactive material that passes through the leak path.
R = A_2 x 2.78 x 10^{-10} /second for normal transport conditions

For normal conditions, the permissible leakage rate is 2.47E-05 cc/sec (Table 4-5). This value is converted to units of ref-cm^3/sec by first calculating equivalent hole size. From ANSI N14.5 (Reference 6):

$$L_u = (F_c + F_m)(P_u - P_d)(P_a/P_u) \text{ cc/sec at } T_u, P_u$$

Other definitions:

L_u	= upstream volumetric leakage rate, cc/sec = 2.47E-05 cc/sec
F_c	= coefficient of continuum flow conductance per unit pressure, cc/atm-sec
F_m	= coefficient of free molecular flow conductance per unit pressure, cc/atm-sec
P_u	= fluid upstream pressure, atm abs = 2.26 atm abs
P_d	= fluid downstream pressure, atm abs = 1.0 atm abs
D	= leakage hole diameter, cm
a	= leakage hole length, cm = 0.5 cm (assuming leak path length is on the order of the metal seal width)
μ	= fluid viscosity, cP = 0.0257 cP
T	= fluid absolute temperature, 186°C = 459 K (average cavity gas temperature from Table 3-1)
M	= molecular weight, g/mol = 4 g/mol (from ANSI N14.5, Table B.1)
P_a	= average stream pressure = ½ ($P_u + P_d$), atm abs = 1.63 atm abs

$$L_u = (F_c + F_m)(P_u - P_d)(P_a/P_u) \text{ cc/sec}$$

where:

$$F_c = (2.49\text{x}10^6 \times D^4)/(a\mu) \text{ cc/atm-sec}$$
$$F_m = \{3.81\text{x}10^3 \times D^3 \times (T/M)^{0.5}\} / \{aP_a\} \text{ cc/atm-sec}$$

Substituting:

$F_c = (2.49x10^6 \times D^4)/(0.5 \times 0.0257) = 1.94E+08\ D^4$

$F_m = \{3.81x10^3 \times D^3 \times (459/4)^{0.5}\} / \{0.5 \times 1.63\} = 5.01E+04\ D^3$

$L_u = (F_c + F_m)(P_u - P_d)(P_a/P_u)$ cc/sec

$2.47E-05 = (F_c + F_m)\ (2.26 - 1.0)\ (1.63 / 2.26)$

$2.47E-05 = (F_c + F_m)\ (0.91)$

$F_c + F_m = 2.71E-05$

Solving the equations above for D, yields a hole diameter of 5.56×10^{-4} cm.

This equivalent hole size, is then used to calculate the reference air rate at standard conditions. Assuming all upstream test conditions correspond to standard conditions:

$L_u = (F_c + F_m)(P_u - P_d)(P_a/P_u)$ cc/sec

where:

$F_c = (2.49x10^6 \times D^4)/(a\mu)$ cc/atm-sec

$F_m = \{3.81x10^3 \times D^3 \times (T/M)^{0.5}\} / \{aP_a\}$ cc/atm-sec

Substituting:

$F_c = \{2.49E+06 \times (5.56E-04)^4\}/(0.5 \times 0.0185) = 2.57E-05$

$F_m = \{3.81E+03 \times (5.56E-04)^3 \times (298/29.0)^{0.5}\} / \{0.5 \times 0.505\} = 8.31E-06$

$L_{std} = (F_c + F_m)(P_u - P_d)(P_a/P_u)$ cc/sec

$L_{std} = (2.57E-05 + 8.31E-06)(1.0 - 0.01)(0.505 / 1.0)$

$L_{std} = 1.70E-05$ ref cm^3 / s

4.2.2 Pressurization of Containment Vessel

The TN-68 cask cavity is drained, dried and evacuated prior to backfilling with helium at the end of loading. The operational procedure guidelines for conducting these activities are provided in Chapter 7. The mechanism contributing to containment pressurization are ideal gas heating and release of fission gas from the fuel rods. The cavity gas mixture (assuming 3% fuel rod failure) is primarily 98.2% helium (from cask backfill operations and from rod pre-pressurization) with the balance consisting of xenon (1.6%), krypton (0.2%), iodine (0.1%), and possibly vaporized bromine. This gas mixture is not explosive.

The maximum containment pressure under normal conditions of transport is based on the following conditions:

- 30% release rate of fission gas from fuel pellets into the gap between the fuel pellets and the cladding.

- 3% failure rate of fuel rod cladding during the transport period.

- maximum cavity gas temperature of 366°F (186°C) under hot environment conditions.

- the gas volume (plenum and pellet to cladding volume) inside the fuel rods is conservatively neglected when calculating the cask free volume.

The pressure in the cask under normal conditions of transport is determined. The cask pressure does not exceed 2.2 atm abs prior to shipment. The temperature inside the cask is 366°F or 186°C. After the cask has been prepared for shipment, it is assumed that instantaneously 3% of the fuel rods have failed, the pressure inside the cask cavity is:

$P_{3\%\ rod\ failure}$ = (68assemblies×0.03×4.425E-03kg moles×0.08314 × 459 K)/(6.0 m^3)
$P_{3\%\ rod\ failure}$ = 0.06 bars = 0.06 atm abs

MNOP = $P_{initial}$ + $P_{3\%\ rod\ failure}$
MNOP = 2.2 atm abs + 0.06 atm abs = 2.26 atm abs = 33.2 psia

Therefore, the maximum normal operating pressure for the TN-68 is 33.2 psia (18.5 psig). Casks designs with MNOP greater than 5.0 psig must be subjected to a structural pressure test in accordance with 10 CFR 71.85(b). The test pressure must be at least 1.5 times MNOP. The TN-68 will be subjected to a hydrostatic test at a pressure of 125 psig in accordance with ASME BP&V Code Section NB-6200. This test is described in Chapter 8.

4.2.3 Containment Criterion

As will be demonstrated in Section 4.3.2, the reference leak rate for normal conditions, 1.78E-05 ref cm^3/s, is a significantly lower rate than the accident leakage rate. However, the acceptance criterion for fabrication verification and periodic verification leak test of the TN-68 containment boundary shall be 1.0 x 10^{-5} ref cm^3/sec. This is conservative by 70% over the calculated value. The test must have a sensitivity of at least one half the acceptance criterion, or 5 x 10^{-6} ref cm^3/s.

4.3 CONTAINMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

The containment requirement under hypothetical accident conditions specified by 10 CFR 71.51(a)(2). It states "there would be no escape of krypton-85 exceeding 10 A_2 in 1 week, no escape of other radioactive material exceeding a total amount A_2 in 1 week." It is assumed for purposes of the accident condition evaluation that 100% of the fuel rods fail thereby releasing all of the available fission gas in the fuel rod gas gap to the cask cavity.

Calculation of the fission gas inventory is discussed in Section 4.2.1.

4.3.1 Fission Gas Products

For hypothetical accident conditions, the cladding of 100% of the fuel rods is assumed to fail. The free fission gas inventory available for release from the fuel rods is described Section 4.2.1. Table 4-1 shows the free activity available for release from the fuel rods. Table 4-2 shows the activity concentration from each of the sources available for release from inside the TN-68.

4.3.2 Containment of Radioactive Material

The TN-68 is designed to meet the hypothetical accident requirements of 10 CFR 71.51. The A_2 values are calculated using the methodology of 10 CFR 71.71 and NUREG/CR-6487. The A_2 values are provided in Tables 4-3 and 4-4. The release fractions for the radionuclides are taken from NUREG/CR-6487.

4.3.3 Containment Criterion

The allowable leak rates under hypothetical accident conditions are calculated using the methodology of NUREG/CR-6487 and previously presented in Section 4.2.3. The permissible leak rates under hypothetical accidents are presented in Table 4-5.

Because the reference leak rate for normal conditions is lower than that for accident conditions, the leak test criterion developed in Section 4.2.3 demonstrates that the containment criteria for both normal and accident conditions are met. The structural and thermal consequences of hypothetical accident loading conditions do not adversely affect the performance of the containment boundary structure or seals.

The impact limiters remain in place on the cask after the hypothetical accident as concluded in Chapter 2 for the limiting 15° oblique drop orientation. During the hypothetical accident the impact limiters provide insulation for the seals of the penetrations underneath them, including the lid seal, vent and drain ports, and the OP port.

Table 3-1 lists the maximum temperature of the seals during a hypothetical thermal accident (see Chapter 3). Temperatures are shown for those areas protected by the insulating effect of the impact limiters, and other areas exposed directly to the accident temperatures environment. None of these temperatures exceeds the seal limit of 536°F. The pressure inside the cask cavity

also remains well below the design pressure of 100 psig as shown below. (Assuming 100% fuel rod failure.)

$P_{HAC} = P_{initial} + P_{100\%\ rod\ failure}$

$P_{initial}$ = (2.2 atm abs)(490 K / 459 K) = 2.3 atm abs

$P_{100\%\ rod\ failure}$ = (68 assem×4.425E-03 kgmole×0.08314 bar-m^3/kgmole K×490 K) / (6.0 m^3)
$P_{100\%\ rod\ failure}$ = 2.0 bar = 2.0 atm abs

P_{HAC} = 2.2 + 2.0 = 4.2 atm abs = 61.7 psia = 47.0 psig

In addition, the cavity gas mixture (assuming 100% fuel rod failure) consists of 62.7% helium (from cask backfill operations and from rod pre-pressurization), 32.5% xenon, 3.4% krypton, 1.4% iodine and possibly some vaporized bromine. This gas mixture is not explosive.

4.4 SPECIAL REQUIREMENTS

Solid plutonium in the form of reactor elements is exempt from the double containment requirements of 10 CFR 71.63.

4.5 REFERENCES

1. Helicoflex Catalog ET 507 E5930.

2. 10 CFR 71, "Packaging and Transportation of Radioactive Materials."

3. ASME Boiler and Pressure Vessel Code, 1995 Code with 1996 Addenda.

4. NUREG/CR-6487, "Containment Analysis for Type B Packages used to Transport Various Contents," Lawrence Livermore National Laboratory, 1996.

5. Transnuclear, Inc., "TN-68 Dry Storage Cask, Safety Analysis Report," Revision 4, May 1999.

6. ANSI N14.5-1997, "American National Standard for Radioactive Material – Leakage Tests on Packages for Shipment," February 1998.

TABLE 4-1
RADIONUCLIDE INVENTORY

	Ci/Assembly[1]
Volatiles	
Sr 90	1.36E+04
Cs134	1.30E+03
Cs137	2.02E+04
Total - Volatiles	3.51E+04
Gases	
H 3	6.40E+01
Kr 85	1.03E+03
I129	7.62E-03
Total - Gases	1.09E+03
Fines	
Pu238	8.19E+02
Pu239	6.32E+01
Pu240	1.09E+02
Pu241	1.81E+04
Am241	4.06E+02
Cm244	6.25E+02
Y 90	1.36E+04
Ba137m	1.90E+04
Pm147	2.10E+03
Sm151	7.57E+01
Eu154	1.32E+03
Eu155	4.61E+02
Total - Fines	5.67E+04

1 Values are from Reference 5, based on a 7x7 fuel assembly (40,000 MWD/MTU burnup, 3.3 wt% U-235 initial bundle average enrichment, and 10 year cooled).

TABLE 4-2

ACTIVITY CONCENTRATION BY SOURCE

Source	Fraction available for release from the fuel rod[1] (f_V / f_G / f_F / f_C)	Fraction of rods that develop cladding breach[1]	Activity Concentration in TN-68 cask (Ci/cc)[2,3,4]
Normal Transport Conditions			
Volatiles	2E-04	0.03	2.39E-06
Gases	0.3	0.03	1.12E-04
Fines	3E-05	0.03	5.78E-07
Crud[5]	0.15	not applicable	6.97E-05
Hypothetical Accident Conditions			
Volatiles	2E-04	1.0	7.96E-05
Gases	0.3	1.0	2.18E-04
Gases - Kr-85 only	0.3	1.0	3.50E-03
Fines	3E-05	1.0	1.93E-05
Crud	1.0	not applicable	4.65E-04

1 Values taken from NUREG/CR-6487 (Reference 4).

2 68 assemblies per cask.

3 Cavity free volume is equal to 6.0E+06 cm^3

4 Source term for the volatiles, gases and fines are based on the 7x7-49/0 fuel assembly. Source term for the crud is based on the 10x10-92/2 assembly (Fuel rod dimensions are provided in Chapter 5).

5 Crud source is based on a surface area of 1.32E+03 cm^2 / rod and an initial surface activity of 1254E-03 Ci/cm^2 at the time of discharge. At discharge, typically, fuel crud is composed of isotopes of cobalt, manganese, chromium and iron. After a 10 year cooling time, the only isotope of radiological significance is Co-60. A decay factor of 0.27 is included in the values listed above.

TABLE 4-3
DETERMINATION OF EFFECTIVE A_2 BY SOURCE

Volatile Isotopes	Activity (Ci/Assembly)	Activity Fraction, FA	A_2 (Ci)	FA / A_2 (1 / Ci)
sr 90	1.36E+04	3.87E-01	2.7	1.44E-01
cs134	1.30E+03	3.70E-02	13.5	2.74E-03
cs137	2.02E+04	5.75E-01	13.5	4.26E-02
				1.89E-01 = 1 / A_2
				A_2, volatile, BWR = 5.29

Normal Transport Conditions

Gaseous Isotopes	Activity (Ci/Assembly)	Activity Fraction, FA	A_2 (Ci)	FA / A_2 (1 / Ci)
h 3	6.40E+01	5.85E-02	1080	5.42E-05
kr 85	1.03E+03	9.41E-01	270	3.49E-03
i129	7.62E-03	6.97E-06	Unlimited	0
				3.54E-03 = 1 / A_2
				A_2, gas, BWR = 282

Hypothetical Accident Conditions

Gaseous Isotopes	Activity (Ci/Assembly)	Activity Fraction, FA	A_2 (Ci)	FA / A_2 (1 / Ci)
h 3	6.40E+01	1.00E+00	1080	9.26E-04
i129	7.62E-03	1.19E-04	Unlimited	0
				9.26E-04 = 1 / A_2
				A_2, gas, BWR = 1080

Fines Isotopes	Activity (Ci/Assembly)	Activity Fraction, FA	A_2 (Ci)	FA / A_2 (1 / Ci)
pu238	8.19E+02	1.44E-02	5.41E-03	2.67E+00
pu239	6.32E+01	1.12E-03	5.41E-03	2.06E-01
pu240	1.09E+02	1.92E-03	5.41E-03	3.55E-01
pu241	1.81E+04	3.19E-01	0.270	1.18E+00
am241	4.06E+02	7.16E-03	5.41E-03	1.32E+00
cm244	6.25E+02	1.10E-02	1.08E-02	1.02E+00
y 90	1.36E+04	2.40E-01	5.41	4.44E-02
ba137m	1.90E+04	3.35E-01	13.5	2.48E-02
pm147	2.10E+03	3.71E-02	24.3	1.52E-03
sm151	7.57E+01	1.34E-03	108	1.24E-05
eu154	1.32E+03	2.33E-02	13.5	1.73E-03
eu155	4.61E+02	8.13E-03	54.1	1.50E-04
				6.83E+00 = 1 / A_2
				A_2, fine, BWR = 1.46E-01

Crud Isotopes	Activity (Ci/Assembly)	Activity Fraction, FA	A_2 (Ci)	FA / A_2 (1 / Ci)
Co-60	4.10E+01	1.00E+00	10.8	9.26E-02
				9.26E-02 = 1 / A_2
				A_2, crud, BWR = 10.80

TABLE 4-4
NORMAL TRANSPORT AND HYPOTHETICAL ACCIDENT CONDITIONS
EFFECTIVE A_2 VALUES

NORMAL CONDITIONS

7x7-49/0						
Source	Releasable Activity (Ci/cc)	Fraction Activity FA	Effective A_2 (Ci)	FA/A_2 (1/Ci)		
volatiles	2.39E-06	1.30E-02	5.29	2.45E-03		
gases	1.12E-04	6.06E-01	282	2.14E-03		
fines	5.78E-07	3.14E-03	1.46E-01	2.14E-02		
crud	6.97E-05	3.78E-01	10.80	3.50E-02		
Total	1.84E-04			6.11E-02	=	$1/A_2$
				A_2 ,BWR, NTC	=	16.38

ACCIDENT CONDITIONS

7x7-49/0						
Source	Releasable Activity (Ci/cc)	Fraction Activity FA	Effective A_2 (Ci)	FA/A_2 (1/Ci)		
volatiles	7.96E-05	1.02E-01	5.29	1.92E-02		
gases (w/o Kr-85)	2.18E-04	2.79E-01	1080	2.58E-04		
fines	1.93E-05	2.47E-02	1.46E-01	1.69E-01		
crud	4.65E-04	5.95E-01	10.80	5.51E-02		
Total	7.81E-04			2.43E-01	=	$1/A_2$
				A_2 ,BWR, HAC	=	4.11

TABLE 4-5
NORMAL TRANSPORT AND HYPOTHETICAL ACCIDENT CONDITIONS
PERMISSIBLE LEAKAGE RATES FROM THE TN-68

Transport Conditions	Effective A2 (Ci)	Allowable Release Rate (Ci/sec)	Concentration (Ci/cc)	Leakage Rate (cc/sec)
NTC	16.38	4.55E-09	1.84E-04	**2.47E-05**
HAC - w/o Kr85	4.11	6.79E-06	7.81E-04	8.69E-03
HAC - Kr85	270	4.46E-03	3.50E-03	1.27E+00

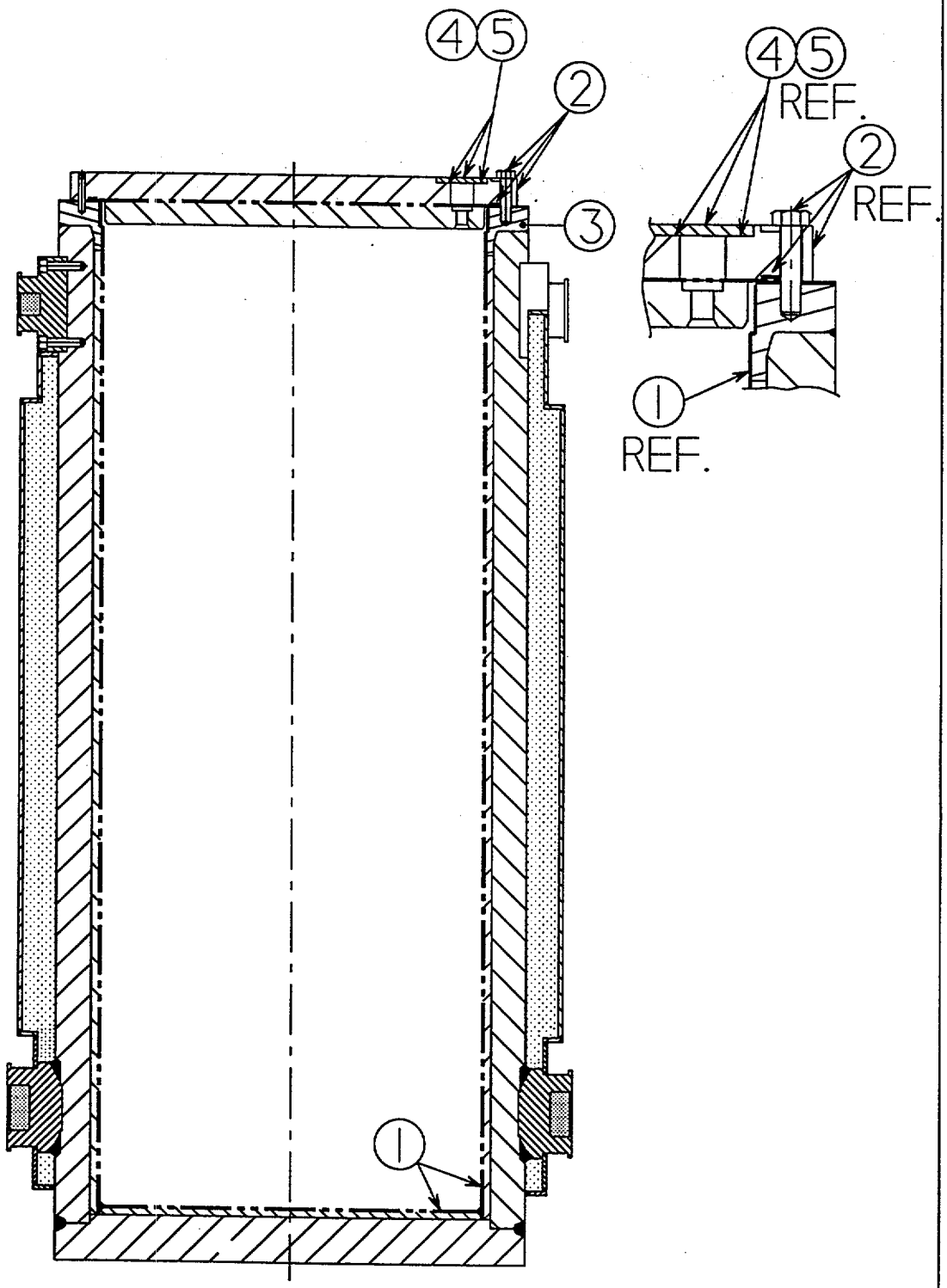


FIGURE 4-1
TN-68 CONTAINMENT
BOUNDARY COMPONENTS

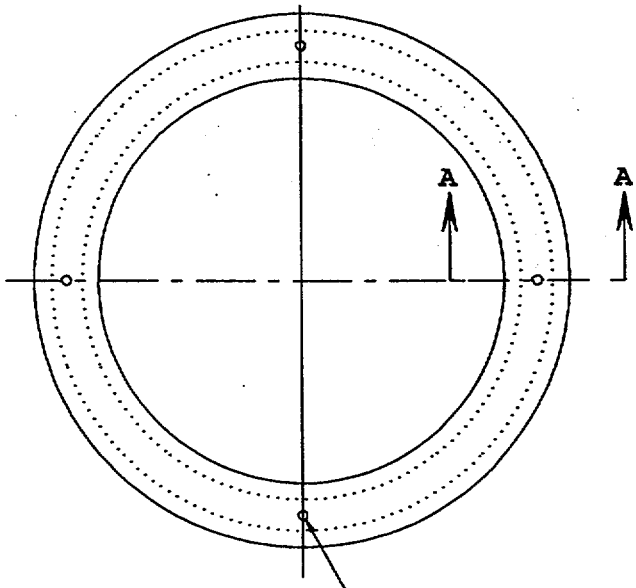
REV. 0 4/99

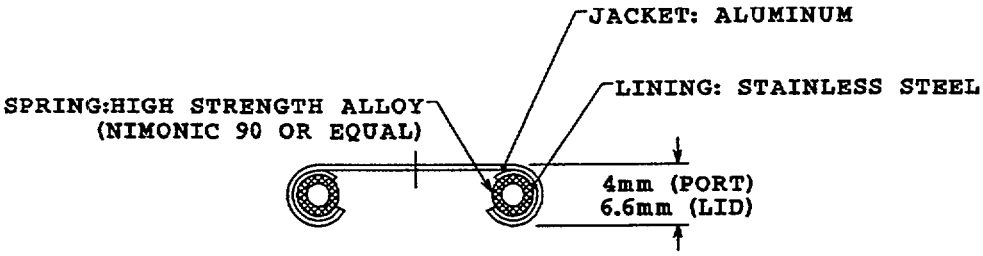
Notes:

1. Figure not to scale. Features exaggerated for clarity.

2. Phantom lines (— · · — · · —) indicates containment boundary.

3. Containment boundary components are listed below:
   1 Cask body and inner shell.
   2 Lid assembly outer plate, closure bolts and inner o-rings.
   3 Bolting flange.
   4 Vent port cover plate, bolts and seals.
   5 Drain port cover plate, bolts and seals.





SECTION A-A

FIGURE 4-2
LID. VENT PORT & DRAIN PORT
METAL SEALS

REV. 0 4/99

TN 68 TRANSPORT PACKAGING

CHAPTER 5

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CHAPTER 5

SHIELDING EVALUATION

5.1 DISCUSSION AND RESULTS

Shielding for the TN-68 cask is provided mainly by the cask body. For the neutron shielding, a borated polyester resin compound surrounds the cask body radially. Additional shielding is provided by the steel shell surrounding the resin layer and by the steel and aluminum structure of the fuel basket.

For transport, wood filled impact limiters are installed on the top and bottom of the cask and provide additional shielding for the top and bottom ends in addition to some radial shielding for the areas above and below the radial neutron shield. Figure 5.1-1 shows the configuration of shielding in the cask. Table 5.1-1 lists the compositions of the shielding materials.

The fuel assemblies acceptable for storage in the TN-68 are listed in Section 1.2.3. This listing of fuel assemblies was collapsed into seven basic designs. Using the SAS2H/ORIGEN-S modules of SCALE[1], source terms for the seven basic fuel designs are calculated. Each basic design has an initial bundle-average enrichment of 3.3 wt% and a total maximum bundle-average burnup of 40,000 MWD/MTU. The most conservative source term (of the seven basic fuel designs) is used in the subsequent shielding calculations.

Through this analysis, the GE 7x7 fuel array with a maximum bundle-average burnup of 40,000 MWD/MTU and an initial bundle-average enrichment of 3.3 wt% is identified as the most conservative fuel source. Section 5.2 describes the source specification and Section 5.4 describes the shielding analysis performed for the TN-68 cask.

Normal conditions are modeled with the TN-68 intact. This shielding calculation is performed using the Monte Carlo computer code MCNP[5]. Dose rates on the side, top and bottom of the TN-68 cask are calculated for the various sources (active fuel-gamma and neutron and irradiated hardware-gamma) and summed to a total gamma and neutron dose rate.

Accident conditions assume that the neutron shield, the impact limiters and the ancillary shield ring are removed. This evaluation bounds the accident conditions since it is shown in Chapter 2 that the impact limiters and the radial neutron shield remain on the cask. Shielding calculations for accident conditions are also performed using MCNP.

The expected maximum dose rates (for normal and accident conditions) from the TN-68 cask are provided in Table 5.1-2. Although the dose rate evaluation is performed using 20 year cooled fuel, thermal (Chapter 3) and containment (Chapter 4) evaluations show that 10 year minimum cooled design basis fuel is acceptable for transport. It is therefore expected that design basis fuel cooled less than 20 years will be acceptable for transport as long as the measured dose rates meet the 10CFR71.47.

5.2 SOURCE SPECIFICATION

There are five principal sources of radiation associated with cask storage that are of concern for radiation protection:

- Primary gamma radiation from spent fuel;
- Primary neutron radiation from spent fuel (both alpha-n reactions and spontaneous fission);
- Gamma radiation from activated fuel structural materials;
- Capture gamma radiation produced by attenuation of neutrons by shielding material of the cask; and
- Neutrons produced by sub-critical fission in fuel.

The TN-68 is designed to transport GE BWR fuel types; from the GE Series 2 and 3 (7x7 fuel array), the GE Series 4 through 10 (8x8 fuel array), the current GE Series 12 (10x10 fuel array), and the current GE Series 11 and 13 (9x9 fuel array). The fuel assemblies acceptable for transport in the TN-68 are described in Section 1.2.3. This listing of fuel assemblies was collapsed into seven basic designs provided below. The various fuel assembly designs were separated according to fuel assembly array, the maximum metric tons of uranium, and the number of water rods. These three parameters are the significant contributors to the SAS2H/ORIGEN-S model. The largest uranium loading results in the largest source term at the design basis enrichment and burnup.

Fuel Array Type	Number of Fueled Rods	Number of Water Rods	Metric Tons Uranium per Assembly
7 x 7	49	0	0.1977
8 x 8	63	1	0.1880
8 x 8	62	2	0.1856
8 x 8	60	4	0.1825
8 x 8	60	1	0.1834
9 x 9	74	2	0.1766
10 x 10	92	2	0.1867

Table 5.2-1 provides additional fuel assembly design characteristics for the seven basic fuel designs. The SAS2H/ORIGEN-S modules of the SCALE code are used to generate a gamma and neutron source term for each fuel assembly design. Each basic design has an initial bundle-average enrichment of 3.3 wt% U235 and the fuel zone is irradiated at a constant specific power of 5 MW/assembly to a total bundle-average burnup of 40,000 MWD/MTU. A conservative three-cycle operating history is utilized with 30 day down time each cycle except for no down time in the last cycle.

The source terms are generated for the active fuel regions, the plenum region, and the end regions. Irradiation of the fuel assembly structural materials (including the channel, plenum, and end fittings) are included in the irradiation of the fuel zone. The fuel assembly hardware materials and masses on a per assembly basis are listed in Table 5.2-2. Table 5.2-3 provides the material composition of fuel assembly hardware materials. Cobalt impurities are included in the SAS2H model. In particular, the cobalt impurities in Inconel, Zircaloy and Stainless Steel are 0.649%, 0.001% and 0.08%, respectively.

The masses for the materials in the top end fitting, the plenum, and the bottom fitting regions are multiplied by 0.1, 0.2 and 0.15, respectively. [4] These factors are used to correct for the spatial and spectral changes of the neutron flux outside of the fuel zone. The material compositions of the fuel assembly hardware are included in the SAS2H/ORIGEN-S model on a per assembly basis.

Axial variation in the moderator density along the BWR fuel assembly was considered by including a volume averaged density for the moderator around the fuel pins. The following axial variation of temperatures and moderator densities were used to calculated the volume average moderator density for use in the BWR source term models[1]:

Distance from bottom of Active Fuel Length	Average Density in Zone (g/cc)	Average Water Temp (K)
30.83	0.743	552
43.17	0.600	558
55.5	0.494	558
67.83	0.417	558
80.17	0.360	558
98.67	0.309	558
123.33	0.264	558
148	0.234	558
Assembly data –water, volume-average density	0.4234 g/cc	558 K

Gamma and neutron source terms are calculated for each of the basic fuel designs. Table 5.2-4 presents the gamma and neutron source terms for a 20 year cooling time. The 7x7 fuel assembly is the most conservative source. This design basis fuel is used for the shielding analysis.

The inventory of fission gases, volatile nuclides and fines used for the containment analyses is presented in Chapter 4.

5.2.1 Axial Source Distribution

Axial source term peaking factors are determined based on typical axial burnup distributions for BWR assemblies and based upon typical axial water density distribution that occurs during core operation. Using the base SAS2H/ORIGEN-S input for the 7x7 BWR, selected as the design basis assembly above, neutron and gamma source terms are generated for axial zones as a

function of burnup and moderator density. This estimates both the non-linear behavior of the neutron source with burnup and the core operating moderator density effects on the actinide isotopics (neutron source).

In-core data from an operating BWR facility forms the basis for the evaluation. The data provided the burnup and moderator density for 25 axial locations along the fuel assembly. Five assemblies located in different locations in the reactor core were utilized to generate a burnup (peaking factor) distribution for the assembly. Figure 5.2-1 represents this distribution.

For water densities, the nodal data provided was examined and 7 assemblies with the lowest densities were selected for evaluation. Of these seven, the assembly with the lowest densities was chosen. The water density data provided shows densities ranging from 0.7608 g/cc at the bottom node to 0.3607 at the top node.

The peaking factors and water densities for the 25 axial locations were collapsed into 12 axial zones and utilized in determining the source terms and axial profiles of the sources for the shielding evaluation. The top and bottom 10% of the assembly was divided into two zones each and the middle 80% divided into 8 equal zones. The peaking factors ranged from 0.2357 and 0.2410 at the bottom and top respectively, to a maximum of 1.20 just below the middle. The water densities ranged from 0.3609 at the top zone to 0.7603 at the bottom.

The burnup and water density axial distribution data was utilized to prepare a 12 axial zone fuel assembly model. Twelve SAS2H calculations were performed for the design basis fuel with the power and water density being variables for each zone. The specific power input was the product of the nominal specific power, (5 MW) and the peaking factor. The water density was that value calculated for the zone as described above. Therefore, the fuel assembly was divided into 12 zones, with each zone having a unique gamma and neutron source term, specifically calculated for the burnup and water density in that zone. This data is presented in Table 5.2-7. (Note: the axial profile data is for 10 year cooled fuel, but the profile is equally applicable for 20 year cooled fuel.)

5.2.2 Gamma Source

Fission product activities and activation activities for the 7x7 fuel assembly (initial bundle average enrichment of 3.3wt%, maximum bundle-average burnup of 40,000 MWD/MTU and 20 year decay) are provided in Tables 5.2-5 and 5.2-6, respectively. The primary gamma source spectrum is listed in Table 5.2-5.

The gamma source spectra are presented in the 18-group structure consistent with the SCALE 27n-18γ cross section library. The conversion of the source spectra from the default ORIGEN-S energy grouping to the SCALE 27n-18γ energy grouping is performed directly through the ORIGEN-S code. The SAS2H/ORIGEN-S input file for the 7x7 fuel assembly is provided in Section 5.5.

The gamma source for the fuel assembly hardware is primarily from the activation of cobalt. This activation contributes primarily to SCALE Energy Groups 36 and 37.

Based on the weight fraction of cobalt in each zone of the fuel assembly model (as adjusted by the appropriate flux ratio), the gamma source term in SCALE Energy Groups 36 and 37 are redistributed accordingly. The gamma source for the plenum region, the top fitting region and the bottom fitting region is provided in Table 5.2-5.

An axial burnup profile has been developed as discussed in Section 5.2.1 above. Table 5.2-7 provides design axial gamma peaking factors and source terms that were utilized in the MCNP shielding model.

5.2.3 Neutron Source

Table 5.2-6 provides the total neutron source spectra for the 7 x 7 fuel assembly (initial bundle average enrichment of 3.3wt%, maximum bundle-average burnup of 40,000 MWD/MTU and 20 year decay. The SAS2H/ORIGEN-S code provides the neutron spectra in the SCALE 27n-18γ energy groups. Table 5.2-4 also provides the total neutron source. The SAS2H/ORIGEN-S input file for the 7x7 fuel assembly is provided in Section 5.5.

The neutron source is not linearly dependent with burnup, and therefore analyses were performed to determine the axial neutron source distribution (Section 5.2.1). The axial neutron source distribution as a function of burnup and water density is shown in Table 5.2-7.

5.3 MODEL SPECIFICATION

The monte carlo code MCNP is used for calculating the gamma and neutron doses immediately around the cask.

5.3.1 Description of Radial and Axial Shielding Configuration

A single geometric model was developed for MCNP. This model was used to calculate both the axial and radial dose rates. In order to determine the total dose rate around a single cask, three separate runs were performed, each with a different source; 1) primary gamma, 2) neutron and 3) hardware gamma (end fittings). A model without impact limiters was utilized for the accident condition evaluation.

Sections 5.3.1.1 and 5.3.1.2 describe the shielding model (for the vicinity immediately around the cask) developed for the TN-68 under normal, off-normal and accident conditions.

5.3.1.1 Radial and Axial Shielding Configuration under Normal Conditions of Transport

Under normal conditions, one shielding configuration is used for the TN-68 design. The model is illustrated in Figures 5.3-1 and 5.3-2. The dimensions of this shielding model correspond to the dimensions of the TN-68 design. The only exception is that the upper and lower trunnions are modeled as flush with the resin shield. (Actually the trunnions extend slightly past the resin shield.) This minor modeling change allowed the model to have surface detectors at the resin shield. This configuration actually results in slightly less shielding present in the area of the trunnions.

The axial locations of the plenum and the end fittings for the fuel assembly are taken from Reference 4; these are the same regardless of fuel assembly type.

The modeled active fuel length is 144 inches and the plenum length is 16.5 inches. The basket's peripheral layer of stainless steel and aluminum are represented by a layer of material with the equivalent thickness. The aluminum rails and shims are also included as an equivalent layer of material within the cask cavity.

Around the fuel region are two thin layers of stainless steel (0.19") and aluminum (0.31") to simulate the periphery of the basket. Also within the cask cavity is 1.21 inches equivalent aluminum to represent the aluminum rails and shims around the basket.

The basket hold down ring is also modeled inside the cask cavity. This hold down ring is 1 inch thick carbon steel with four reduced thickness (7/16 inch) sections. The cross members of the hold down ring are neglected in the model.

The impact limiters are modeled as wood surrounded by a 0.25" thick steel shell. The interior steel gussets are neglected. The wood is all assumed to be redwood except for the 68" diameter plug in the end of each limiter. An aluminum spacer utilized under the top impact limiter is included in the model.

A 1/2 inch thick ancillary steel shield ring is also modeled. The shield ring rests on the radial resin shield, extending up to the top impact limiter. If the measured dose rates exceed the transportation limit, the ancillary shield ring may be utilized to lower the dose rates to acceptable limits.

The fuel region is assumed to consist of uranium dioxide. The fuel cladding and steel and aluminum basket are included in the homogenized fuel region. The fuel channels are not included in the homogenization. (However, the fuel channels are included in the source term.) The neutron poison material of the basket is also neglected. The stainless steel intermittent basket plates are assumed to be aluminum in the basket homogenization. (Aluminum has a lower density than steel and therefore will provide less self-shielding.) The fuel and basket region is modeled as a cylinder within the cask cavity.

The plenum region is assumed to consist of the cladding, plenum springs and the steel and aluminum basket. The hydrogen getters within the plenum are neglected. The basket is homogenized through the plenum region. Similarly, the bottom fitting region is homogenized with the basket. The top fitting hardware is homogenized through the same reduced cavity diameter as the other regions. The basket is not included in the homogenization of the top fitting region. The basket hold down ring is modeled in the top fitting region. The steel cross members of this ring are not included in the homogenization.

Voids are neglected within the fuel assembly. The voids within the cask cavity are modeled.

5.3.1.2 Radial and Axial Shielding Configuration under Hypothetical Accident Conditions of Transport

For accident conditions, it is conservatively assumed that the top and bottom impact limiters are gone. Also, the ancillary shield ring and the neutron shield are assumed to be removed. The model utilizes the same regional densities and shield thickness as the model for normal conditions.

5.3.2 Shield Regional Densities

For the MCNP model, four source areas, shown in Figures 5.3-1 and Figure 5.3-2 are utilized: fuel zone, plenum, upper fitting and lower fitting. The sources are uniformly homogenized over the reduced cavity diameter and the appropriate length. The fuel basket is homogenized over the source diameter and appropriate length (of the fuel zone, plenum and bottom fitting).

The radial resin and aluminum boxes are homogenized into a single composition based on the mass of each component. Measured dose rates around the TN-24P[7], the TN-40, and the TN-32 casks have shown no streaming effects around the neutron shield. This is because the neutrons will not generally travel in a direct path, but scatter, such that the majority of the neutrons will not be able to travel through the aluminum box wall for the full 6 inches of resin box thickness. The material input for the MCNP model is listed in Table 5.3-1.

5.4 SHIELDING EVALUATION

Dose rates around the TN-68 are determined by choosing the most conservative source and using it within a three dimensional MCNP model. The MCNP dose is calculated as surface flux (F2) tallies and converted into dose rates using energy dependent dose conversion factors[6], (Tables 5.4-1and 5.4-2). The shielding evaluation accounts for subcritical neutron multiplication. The generation of secondary gamma dose due to neutron interactions in the shielding materials, principally the neutron shield resin, is neglected because the resin is surrounded by a steel shell and previous evaluations have shown the secondary gamma dose to be small fraction (< 3%) of the total calculated dose.

For the doses around the TN-68, the source is divided into four separate regions: fuel, plenum, top fitting, and bottom fitting. The model is utilized in three separate computer runs consisting of contributions from the following sources:

- Primary gamma radiation from the active fuel (axial and radial directions).
- Neutron radiation from the active fuel region (axial and radial directions).
- Gamma radiation from activated hardware within the top fitting, plenum region and bottom fitting (axial and radial directions).

The sources in the active fuel region (gamma and neutron) are uniform radially but vary axially. The sources in the structural hardware regions (plenum, top fitting, and bottom fitting) are uniform both radially and axially.

Detector surfaces were placed in several radial and axial locations in order to evaluate the dose rate around the cask body. These surfaces provide an averaged surface dose rate based on the size of the detector (surface). The surfaces are subdivided into segments in order to determine the location and magnitude of maximum dose rates. Approximately 25 cm length "detector" segments were utilized both axially and radially.

For normal conditions, the contribution of each source to each dose point is summed to calculate the total gamma and/or neutron dose for each location. Table 5.1-2 presents the maximum calculated dose at contact, at the vehicles outer edge (assumed 10 ft wide vehicle), and at 2 m from the vehicle's outer edge. The calculated neutron and gamma dose rates at the various dose points are illustrated in Figures 5.4-1 through 5.4-4.

For accident conditions, Table 5.1-2 also presents the maximum calculated doses at 1 m from the cask body.

The source term evaluation was performed using SCALE 4.3, "Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers"[1] by Oak Ridge National Laboratory. The dose rate analysis was performed using MCNP, "MCNP4B2 Monte Carlo N-Particle Transport Code System"[5] by Los Alamos National Laboratory. SCALE 4.3 is implemented on a Hewlett Packard 9000/715 Workstation. MCNP is implemented on Pentium based PCs using Windows NT. These program(s) have been verified in accordance with the Transnuclear quality assurance program.

Selected input for MCNP are included in Section 5.6.

5.5 REFERENCES

1. SCALE 4.3, "Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers," CCC-545, ORNL, March 1997.

2. Croff, et al, "Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Codes," ORNL/TM-6051, Oak Ridge National Laboratories, September 1978.

3. Moore and Notz, "Physical Characteristics of GE BWR Fuel Assemblies," ORNL-TM-10902, June 1989.

4. Luksic, 'Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal,' PNL-6906, UC-85, June 1989.

5. MCNP4B2, "Monte Carlo N-Particle Transport Code System." Los Alamos National Laboratory, CCC-660, RSIC.

6. "Data for Use in Protection Against External Radiation," Publication 51, International Commission on Radiological Protection, Annals of the ICRP, 17, No. 2/3, Pergamon Press, Oxford, 1987.

7. EPRI-NP-5128, "The TN-24P PWR Spent-Fuel Storage Cask: Testing and Analyses," prepared by Pacific Northwest Laboratory, Virginia Power Company and EG&G Idaho National Engineering Laboratory, April 1987.

5.6 APPENDIX

5.6.1 SAS2H/ORIGENS Input File

```
=sas2h     parm=(halt03,skipshipdata)
7x7-49-0rv1.inp, 3.3 w/o U235, 40,000 MWD/MTU, 5-60 year cooling
27groupndf4  latticecell
uo2      1   0.95  840 92234 0.0294 92235 3.3 92236 0.0152
92238 96.6555 end
zircalloy 2   1.0     620  end
h2o      3  den=0.432  1.0  558   end
zircalloy 5   1.0     552  end
h2o     11  den=0.669  1.0  552   end
end comp
squarepitch  1.8745  1.23698  1  3  1.43002  2  1.26746  0  end
npin/assm=49  fuelength=365.76  ncycles=3  nlib/cyc=1  printlevel=10
lightel=10    inplevel=2     numzones=4  end
 500  7.4031  3  7.5091  5  7.7957  11  8.5982
power=5.00   burn=527.2   down=30   end
power=5.00   burn=527.2   down=30   end
power=5.00   burn=527.2   down=1461  end
n  0.0432   si 0.0106   ti  0.0106  cr 0.375  mn 0.0228  fe  0.854
co  0.00456   ni 0.422   sn  1.30   zr 84.9
end
=origens
0$$  a4  21  a8  26  a10  51  71  e
1$$  1  1t
cooling to 60 years and fission product gamma reordering
3$$  21  0  1  a33 -86  e
54$$ a8  1  e  t
35$$ 0  t
56$$ 0  8  a13 -2  5   3  e
57** 4.0 e  t
cooling to 60 years and fission product gamma re-ordering
single reactor assembly
60** 5.0  8.0  10.0  20.0  30.0  40.0  50.0   60.0
65$$  a4 1 a7 1 a10 1 a25 1 a28 1 a31 1 a46 1 a49  1  a52  1  e
61** f.0000001
81$$ 2  51  26  1  e
82$$ f6  t
fission product gamma spectra in scale 18 groups
fission product gamma spectra in scale 18 groups
fission product gamma spectra in scale 18 groups
fission product gamma spectra in scale 18 groups
fission product gamma spectra in scale 18 groups
fission product gamma spectra in scale 18 groups
fission product gamma spectra in scale 18 groups
fission product gamma spectra in scale 18 groups
56$$ f0  t
end
=origens
0$$  a4  21  a8  26  a10  51  71  e
1$$  1  1t
cooling to 60 years and actinide gamma re-ordering
3$$  21  0  1  a33 -86  e
```

```
54$$ a8 1 e t
35$$ 0 t
56$$ 0 8 a13 -2 5 3 e
57** 4.0 e t
cooling to 60 years and actinide gamma re-ordering
single reactor assembly
60** 5.0 8.0 10.0 20.0 30.0 40.0 50.0 60.0
65$$ e
61** f.0000001
81$$ 2 51 26 1 e
82$$ f5 t
actinide gamma spectra in scale 18 groups
actinide gamma spectra in scale 18 groups
actinide gamma spectra in scale 18 groups
actinide gamma spectra in scale 18 groups
actinide gamma spectra in scale 18 groups
actinide gamma spectra in scale 18 groups
actinide gamma spectra in scale 18 groups
actinide gamma spectra in scale 18 groups
56$$ f0 t
end
=origens
0$$ a4 21 a8 26 a10 51 71 e
1$$ 1 1t
cooling to 60 years and light element gamma re-ordering
3$$ 21 0 1 a33 -86 e
54$$ a8 1 e t
35$$ 0 t
56$$ 0 8 a13 -2 5 3 e
57** 4.0 e t
cooling to 60 years and light element gamma re-ordering
single reactor assembly
60** 5.0 8.0 10.0 20.0 30.0 40.0 50.0 60.0
65$$ e
61** f.0000001
81$$ 2 51 26 1 e
82$$ f4 t
light element scale group structure
light element scale group structure
light element scale group structure
light element scale group structure
light element scale group structure
light element scale group structure
light element scale group structure
light element scale group structure
56$$ f0 t
end
```

5.6.2 MCNP Neutron Model Input File

```
TransNuclear TN-68 cask:   Near-Field model; neutron source (12 zones)
c    Secondary photons not included.
c    Cask cells have IMP=1; Geometry splitting/routlette used outside
c    cask.  Volumetric F2 detectors used. TRANSPORT CONFIGURATION
c ************************  BLOCK 1: CELL CARDS ************************
c GEOMETRY (r-z)* includes 1/2" ancillary shield ring between resin & top
limiter
c
c  ^ z-axis
c  .
c  | ------------------   |  tally surfaces @
c  .____________________.  |     contact,
c  |    impact limiter |  |     rail car surface,
c  ._______________.     |  |     2m from rail car
c  |---------------+--+  |  |
c  .---------------+   |  |  |
c  |...............|   |__|  |
c  .  BASKETS      |   |+    |
c  |...............|   ||    |          VOID
c  .               |   ||    |
c  |               |   ||    |
c  .   FUEL        |   ||    |
c  O ------------|--||-----------------------------> y-axis
c  |  (12 sub      |   ||    |
c  .   regions)    |   ||    |
c  |               |   ||    |
c  .               |   ||
c  |...............|   ||
c  .  BASKET       |   |__.
c  |---------------+   |  |
c  .=----------------+  |
c  |   impact limiter   |
c  .____________________|
c  |                                                  M. Mason (4/9/99)
c
c
c ****** Cask cells
   1  9   -7.8212  1 -2 -21                  imp:n,p=1  $ Fe cask bottom
   2  9   -7.8212  2 -13 201 -21             imp:n,p=1  $ Fe cask side
   211 9  -7.8212  -32 21 10 -160 #407 #408    imp:n,p=1  $ ancil shield ring
   3  7   -1.918   2 -5 -26                  imp:n,p=1  $ bottom basket
   4  6   -1.158   7 -8 -26                  imp:n,p=1  $ top plenum basket
   5  5   -0.491   8 -11 -26                 imp:n,p=1  $ top fitting
   6  8   -7.92    26 -27 2 -28              imp:n,p=1  $ ss side basket
   7  10  -2.702   27 -201 2 -28             imp:n,p=1  $ Al side basket/rails
   8  0    11 -12 -26                        imp:n,p=1  $ top void - part1
   9  0   (28 -12 26 -24):(19 -12 24 -201) imp:n,p=1    $ top void - part2
   10  9   -7.8212  (28 -19 24 -201)(#13 #14 #15 #16) imp:n,p=1    $ hold
 down rin
   11  9   -7.8212  12 -14 -201               imp:n,p=1  $ Fe cask lid - part1
   12  9   -7.8212  13 -14 201 -29            imp:n,p=1  $ Fe cask lid - part2
   13  0   195 -25 -191 192 24 -19 28         imp:n,p=1  $ cutout in hold down ring
   14  0  -195 -25 -191 192 24 -19 28         imp:n,p=1  $ cutout in hold down ring
   15  0   196 -25  193 -194 24 -19 28        imp:n,p=1  $ cutout in hold down ring
```

```
  16  0  -196 -25  193 -194 24 -19 28  imp:n,p=1 $ cutout in hold down ring
  19  9   -7.8212  21 -23 9 -10 #400 #402 #407 #408 #409 #410
                   #411 #412           imp:n,p=1 $ top side-shld Fe shell
  20  9   -7.8212  22 -23 4 -9 #400 #402 #409 #410 #411 #412
                  #420 #422 #429 #430 #431 #432 imp:n,p=1 $ side side-shld
Fe shell
  21  9   -7.8212  21 -23 3 -4 #420 #422 imp:n,p=1 $ bot side-shld Fe shell
  22  12  -1.687   21 -22 4 -9 #400 #402 #403 #404 #405 #406 #407 #408
          #409 #410 #411 #412 #420 #422 #423 #424 #425 #426 #427 #428
          #429 #430 #431 #432          imp:n,p=1 $ side resin/Al shield
 23  0            150 -3 21 -23        imp:n,p=1 $ void under side shld
 24  0    32 -23 10 -160 #407 #408 #409 #410 #411 #412
                      imp:n,p=1                $ void above side shld - pt1
c 25  1  -0.0013   30 -23 18 -17       imp:n,p=1  $ air above side shld - pt2
c **** impact limiters  *****
c  bottom limiter
  80 9  -7.8212  (-1 155 -253):(1 -151 -253 21) imp:n,p=1 $ inside skin
  81 9  -7.8212  (153 -152 -250):(152 -151 -250 251) imp:n,p=1 $ outside
skin
  82 9  -7.8212     151 -150 -250 21      imp:n,p=1 $ outside skin
  83 14 -0.387      155 -151 -251 253     imp:n,p=1 $ redwood
  84 14 -0.387      154 -155 -251 252     imp:n,p=1 $ redwood
  85 14 -0.387      154 -155 -252         imp:n,p=1 $ redwood
  86 14 -0.387      152 -154 -251 252     imp:n,p=1 $ redwood
  87 15 -0.125      152 -154 -252         imp:n,p=1 $ balsa
c  top limiter
 90 9  -7.8212  (166 -165 -253):(161 -166 -253 21) imp:n,p=1 $ inside skin
 91 9  -7.8212  (162 -163 -250):(161 -162 -250 251) imp:n,p=1 $ outside skin
  92 9  -7.8212     160 -161 -250 21      imp:n,p=1 $ outside skin
  93 14 -0.387      161 -165 -251 253     imp:n,p=1 $ redwood
  94 14 -0.387      165 -164 -251 252     imp:n,p=1 $ redwood
  95 14 -0.387      165 -164 -252         imp:n,p=1 $ redwood
  96 14 -0.387      164 -162 -251 252     imp:n,p=1 $ redwood
  97 15 -0.125      164 -162 -252         imp:n,p=1 $ balsa
  98 10 -2.702 (14 -166 -21):(-14 13 29 -21)  imp:n,p=1 $ al spacer
c  trunnions
  400 0    195 300 -23 304 -10 #409 #411   imp:n,p=1 $ resin cutout at top
trun
  402 0   -195 -301 -23 304 -10 #410 #412  imp:n,p=1 $ resin cutout at top
trunn
  403 9   -7.8212 195 305 -300 -22 304 -9
                #407  #411                 imp:n,p=1 $ trunnion flat fe
plate
  404 9   -7.8212 -195 -306 301 -22 304 -9
               #408  #412                  imp:n,p=1 $ trunnion flat fe
plate
  405 9   -7.8212 195 305 -22 303 -304      imp:n,p=1  $ flat plate under
top tru
  406 9   -7.8212 -195 -306 -22 303 -304    imp:n,p=1  $ flat plate under
top tru
  407 9   -7.8212 307 -300 -330 #2 #411 imp:n,p=1  $ top trunnion base
  408 9   -7.8212 -308 301 -330 #2 #412 imp:n,p=1  $ top trunnion base
  409 9   -7.8212 300 -309 -331 #411  imp:n,p=1  $ top trunnion load surface
  410 9   -7.8212 -301 310 -331 #412  imp:n,p=1  $ top trunnion load surface
  411 12  -1.687  311 -309 -332       imp:n,p=1  $ top trunnion resin plug
  412 12  -1.687 -312  310 -332       imp:n,p=1  $ top trunnion resin plug
```

```
  420 0    195  320 -23 -323 3 #429 #431     imp:n,p=1 $ resin cutout at
bottom t
  422 0   -195 -321 -23 -323 3 #430 #432     imp:n,p=1 $ resin cutout at
bottom t
  423 9   -7.8212 195 324 -320 -22 -323 4 #431
                     imp:n,p=1 $ bottom trunnion flat fe plate
  424 9   -7.8212 -195 -325 321 -22 -323 4 #432
                imp:n,p=1 $ bottom trunnion flat fe plate
  425 9   -7.8212 195 324 -22 -322 323      imp:n,p=1  $ flat plate under bot
tru
  426 9   -7.8212 -195 -325 -22 -322 323    imp:n,p=1  $ flat plate under bot
tru
  427 9   -7.8212  326 -324 -333 #2       imp:n,p=1  $ bottom trunnion base
  428 9   -7.8212 -327  325 -333 #2       imp:n,p=1  $ bottom trunnion base
  429 9   -7.8212  320 -328 -334 #431       imp:n,p=1  $ bottom trunnion load
sur
  430 9   -7.8212 -321  329 -334 #432       imp:n,p=1  $ bottom trunnion load
sur
  431 12  -1.687  311 -328 -335             imp:n,p=1  $ bottom trunnion
resin pl
  432 12  -1.687 -312  329 -335             imp:n,p=1  $ bottom trunnion
resin pl
c **** fuel regions
  40 4   -3.231   5  -39 -26           imp:n,p=1  $ FUEL region 1 (bottom)
  401 4  -3.231   39 -40 -26           imp:n,p=1  $ FUEL region 2
  41 4   -3.231   40 -41 -26           imp:n,p=1  $ FUEL region 3
  42 4   -3.231   41 -42 -26           imp:n,p=1  $ FUEL region 4
  43 4   -3.231   42 -43 -26           imp:n,p=1  $ FUEL region 5
  44 4   -3.231   43 -44 -26           imp:n,p=1  $ FUEL region 6
  45 4   -3.231   44 -45 -26           imp:n,p=1  $ FUEL region 7
  46 4   -3.231   45 -46 -26           imp:n,p=1  $ FUEL region 8
  47 4   -3.231   46 -47 -26           imp:n,p=1  $ FUEL region 9
  48 4   -3.231   47 -48 -26           imp:n,p=1  $ FUEL region 10
  481 4  -3.231   48 -49 -26           imp:n,p=1  $ FUEL region 11
  49 4   -3.231   49 -7  -26           imp:n,p=1  $ FUEL region 12 (top)
c ***** outside cells above/below cask
  140 0            170  -60 -172       imp:n,p=0  $ air beneath cask-pt2
  142 0           -153   60 -250       imp:n,p=1  $ air beneath cask-pt1
  145 0            163  -61 -250       imp:n,p=1  $ air above cask-pt1
  146 0             61 -171 -172       imp:n,p=0  $ air above cask-pt2
c ***** Cells outside radial cask surface
c 600 2   -2.32    150 -1  23 -60      imp:n,p=1 $ concrete inner air
  601 0            150 -160  23 -62    imp:n,p=1 $ inner air (void)
  602 0            150 -160  62 -63    imp:n,p=1 $ inner air (void)
  603 0   (60 -61 250 -64):(150 -160 63 -250) imp:n,p=1 $ inner air (void)
c 604 2   -2.32    150 -1  60 -152    imp:n,p=1 $ concrete outer air
  605 0             60 -61 -172 64     imp:n,p=0    $ outer air (void)
c 606 0            17 -61  60 -152    imp:n,p=1 $ outer air (void)
c 607 0            61 -151 60 -152    imp:n,p=1 $ outer air (void)
  190 0           -170:171:172         imp:n,p=0    $ problem boundary

c ***********************  BLOCK 2: SURFACE CARDS ************************
c **** Horizontal cask planes
  1   pz  -226.42       $ cask bottom - ground surface
  2   pz  -201.65       $ cask bottom - top of bot Fe plate
  3   pz  -192.76       $ side Fe jacket - outside lower bottom
  4   pz  -190.86       $ side Fe jacket - inside lower bottom
```

```
  5   pz  -182.88      $ top bottom basket/bottom of fuel
  7   pz   182.88      $ bottom of plenum basket/top of fuel
  8   pz   224.72      $ top of plenum basket
  9   pz   211.74      $ side Fe jacket - inside top
  10  pz   213.64      $ side Fe jacket - outside top
  11  pz   245.90      $ top of top fitting
 12  pz   250.47      $ cask top - bot of lid
 13  pz   261.90      $ cask side - top of Fe side
 14  pz   274.60      $ cask top - top of Fe
c  15  pz   284.76      $ top of polyprop on top of cask
c  16  pz   318.74      $ top Fe cover - bot surface
c  17  pz   319.38      $ top Fe cover - top surface
  18  pz   266.35      $ flange top
  19  pz   248.56      $ top hold down ring
  191 py    18.42       $ half height of hold down ring cutout
  192 py   -18.42       $ half height of hold down ring cutout
  193 px   -18.42       $ half height of hold down ring cutout
  194 px    18.42       $ half height of hold down ring cutout
  195 px     0.0        $ ambiguity surface
  196 py     0.0        $ ambiguity surface
  28  pz   214.91      $ bottom hold down ring
  150  pz  -194.04      $ top of bottom limiter
  151  pz  -194.68      $ inside skin bottom limiter
  152  pz  -314.68      $ inside skin bottom limiter
  153  pz  -315.32      $ bottom of bottom limiter
  154  pz  -275.94      $ top of balsa disk bottom limiter
  155  pz  -227.06      $ top of inside skin bottom limiter
  160  pz   246.03      $ bottom of top limiter
  161  pz   246.67      $ inside skin top limiter
  162  pz   366.67      $ inside skin  top limiter
  163  pz   367.31      $ top of top limiter
  164  pz   327.93      $ bottom of balsa disc top limiter
  165  pz   279.05      $ top of inside skin top limiter
 166  pz   278.41      $ top of Al spacer top limiter
  300  px   116.28      $ OD of top trunnion base (flat)
  301  px  -116.28      $ OD of top trunnion base (flat)
  302  py    44.37      $ half width of trunnion flat
  303  pz   171.09      $ bottom of plate under top trunnion
  304  pz   173.00      $ top of resin recess under top trunnion
  305  px   114.38      $ ID of flat plate at top trunnion
  306  px  -114.38      $ ID of flat plate at top trunnion
  307  px   102.44      $ top trunnion socket
  308  px  -102.44      $ top trunnion socket
  309  px   123.85      $ top trunnion OD ( to match body OD)
  310  px  -123.85      $ top trunnion OD ( to match body OD)
  311  px   115.34      $ ID of top trunnion resin plug
  312  px  -115.34      $ ID of top trunnion resin plug
  320  px   117.01      $ OD of bottom trunnion base (flat)
  321  px  -117.01      $ OD of bottom trunnion base (flat)
  322  pz  -111.72      $ top of plate over bottom trunnion
  323  pz  -113.62      $ bottom of resin recess over bottom trunnion
  324  px   115.10      $ ID of flat plate at bottom trunnion
  325  px  -115.10      $ ID of flat plate at bottom trunnion
  326  px   104.34      $ bottom trunnion socket
  327  px  -104.34      $ bottom trunnion socket
  328  px   123.19      $ bottom trunnion OD (to match body OD)
  329  px  -123.19      $ bottom trunnion OD  (to match body OD)
```

```
c ***** cylindrical cask surfaces
  201  cz    88.27        $ cask inner surface
  21  cz   107.32        $ cask outer surface
   22  cz   122.56        $ side Fe jacket -- inside
  23  cz   124.47        $ side Fe jacket -- outside
  24  cz    85.73        $ inside radius of hold down ring
  25  cz    87.15        $ inside radius of cutouts in ring
c 25  cz    89.22        $ top polyprop disk radius
  26  cz    83.92        $ inside radius ss basket
27  cz    84.40        $ inside radius Al backet/rails
  29  cz   101.45        $ inside radius top cover
c  30  cz   102.40        $ outside radius top cover
  32   cz  108.60        $ ancillary shield ring (1/2")
  250 cz   182.88        $ outside radius of impact limiter
  251 cz   182.24        $ inside radius of impact limiter
 252 cz    86.36        $ radius of balsa disk
  253 cz   107.96        $ outside radius inside skin
  330 c/x   0 213.64 21.59        $ top trunnion base
  331 c/x   0 213.64 12.38        $ top trunnion load surface
 332 c/x   0 213.64 9.65        $ top trunnion resin plug
 333 c/x   0 -156.8 21.59        $ bottom trunnion base
  334 c/x   0 -156.8 17.78        $ bottom trunnion load surface
 335 c/x   0 -156.8 9.84        $ bottom trunnion resin plug
c ***** surfaces for fuel regions
 39  pz  -164.59        $ top of fuel region 39
 40  pz  -146.30        $ top of fuel region 40
 41  pz  -109.73        $ top of fuel region 41
 42  pz  -73.15         $ top of fuel region 42
 43  pz  -36.53         $ top of fuel region 43
 44  pz  -0.0           $ top of fuel region 44
 45  pz   36.53         $ top of fuel region 45
 46  pz   73.15         $ top of fuel region 46
 47  pz   109.73        $ top of fuel region 47
 48  pz   146.30        $ top of fuel region 48
 49  pz   164.59        $ top of fuel region 49
  c ***** problem boundaries
  170  pz  -500.E2        $ bottom of air (problem boundary)
  171  pz   500.E2        $ top of air (problem boundary)
  172  cz   500.E2        $ radial air limit (problem boundary)
 c ****** surfaces for detector segmentation
  60  pz  -318.0         $ bottom tally surface
  61  pz   370.0         $ top tally surface
  62  cz   125.0         $ radial tally surface  (outer shell)
  63  cz   152.0         $ radial tally surface   (rail car edge)
  64  cz   352.0         $ radial tally surface   (2 m from rail car)
  71  pz  -190.0         $ segmentation plane
  72  pz   200.0         $ segmentation plane
  81  cz    25.00        $ segmentation cylinder
  82  cz    50.00        $ segmentation cylinder
  83  cz    75.00        $ segmentation cylinder

 c ***********************  BLOCK 3: DATA CARDS ***************************
 c
 c  --- volumetric neutron source in 12 axial zones for TN-68 cask
 c      7x7 fuel assemblies; 40,000 MWd/Mt average burnup; 20y cooling time
 SDEF  par=1 pos 0 0 0  axs=0 0 1  rad=d1  ext=d2   erg=d3  cel=d4
  SI1  0   83.92          $ range of radius sampling: 0 to Rmax
```

```
 SP1  -21 1                $ radial distriubtion: here r^1
 SI2 -182.88 182.88        $ range of axial sampling
 SP2  -21  0               $ axial distribution: here z^0
 SI3  H   0.1   0.4     0.9    1.4    1.85    3.0    6.434  20  $ energy bins
 SP3      0.0 .03750 .1915 .1763 .1311 .2355 .2098 .01828  $ bin prob.
 SI4 L 40 401 41 42 43 44 45 46 47 48 481 49
 SP4   0.000092 0.0085273 0.084808 0.13715 0.15279 0.15769
       0.15617 0.13858 0.10747 0.051003 0.0055721 0.00013994
 SB4   0.05     0.05     0.1      0.1      0.1      0.1
       0.1      0.1      0.1      0.1      0.05     0.05
c
c
c ---- Detector types and locations -- neutrons and NO secondary gammas
c -- doses on cask's radial surface (F2 segmented surface detectors)
c FM2    2.008E18      $ convert Sv/neutron to mrem/h for fuel zones
c      6.187E7 x 68 X 1.326 (NF) X 3600 X 1E5 = 2.008E18
c TF2 3j 6
FC2  Doses on outer shell averaged over subsurfaces
F2:n  62
FS2   -71 -39 -40 -41 -42 -43 -44 -45 -46 -47 -48 -49 -72 -8 -11
SD2  3.0E7 19956.97 14364.93 28722.01 28729.86 28761.28
       28690.59 28690.59 28761.28 28729.86 28722.01
        14364.93 27810.95 19415.04 16634.73 3.0E7
 FC12   Doses at the rail car edge averaged over subsurfaces
 F12:n  63
 FS12  -71 -39 -40 -41 -42 -43 -44 -45 -46 -47 -48 -49 -72 -8 -11
 SD12  3.0E7 24267.67 17467.76 34925.96 34935.52 34973.72
         34887.76 34887.76 34973.72 34935.72 34925.96
        17467.76 33818.11 23608.69 20227.84 3.0E7
 FC22    Doses at 2 meters from rail car averaged over subsurfaces
 F22:n   64
 FS22  -152 -154 -155 -71 -39 -40 -41 -42 -43 -44 -45 -46 -47
        -48 -49 -72 -8 -11 -165 -164 -162
  SD22   1.0E8 85680.53 108106.98 81964.90 56198.82 40451.65 80881.18
          80903.30 80991.77 80792.71 80792.71 80991.77 80903.30
          80881.18 40451.65 78315.63 54672.76 46843.41
          73317.23 108106.98 85680.53 8.0E7
c
c -- doses along cask's top
 FC32 Doses at top limiter surface averaged over subsurfaces
 f32:n   61          $ surface tally
 fs32   -81 -82 -83 -29 -23 -63 -250 -64
 sd32   1963.50 5890.49 9817.48 14662.13 16338.41 23911.35
          32487.51 284185.03 7.8E7
c
c -- doses along cask's bottom
 FC42 Doses at bottom limiter surface averaged over subsurfaces
 f42:n   60          $ surface tally
 fs42   -81 -82 -83 -29 -23 -63 -250 -64
  sd42   1963.50 5890.49 9817.48 14662.13 16338.41 23911.35
        32487.51 284185.03 7.8E7
c
c mode n p
 phys:n 20.0 0.0
 cut:n j 0.0
c phys:p 0 1 1
 nps  4000000
```

```
c  void
c
c ------------------------------------------------------------------
c  ambient neutron dose equiv. H*(10mm) Sv (from T-D3 of S&F)
c ------------------------------------------------------------------
de0     2.500E-08 1.000E-07 1.000E-06 1.000E-05 1.000E-04 1.000E-03
        1.000E-02 2.000E-02 5.000E-02 1.000E-01 2.000E-01 5.000E-01
        1.000E+00 1.500E+00 2.000E+00 3.000E+00 4.000E+00 5.000E+00
        6.000E+00 7.000E+00 8.000E+00 1.000E+01 1.400E+01 1.700E+01
        2.000E+01
df0     8.000E-12 1.040E-11 1.120E-11 9.200E-12 7.100E-12 6.200E-12
        8.600E-12 1.460E-11 3.500E-11 6.900E-11 1.260E-10 2.580E-10
        3.400E-10 3.620E-10 3.520E-10 3.800E-10 4.090E-10 3.780E-10
        3.830E-10 4.030E-10 4.170E-10 4.460E-10 5.200E-10 6.100E-10
        6.500E-10
c ------------------------------------------------------------------
c  ambient photon dose equiv. H*(10mm) Sv (from T-D1 of S&F)
c ------------------------------------------------------------------
c de24   1.000E-02 1.500E-02 2.000E-02 3.000E-02 4.000E-02 5.000E-02
c       6.000E-02 8.000E-02 1.000E-01 1.500E-01 2.000E-01 3.000E-01
c       4.000E-01 5.000E-01 6.000E-01 8.000E-01 1.000E+00 1.500E+00
c       2.000E+00 3.000E+00 4.000E+00 5.000E+00 6.000E+00 8.000E+00
c        1.000E+01
c df24   7.690E-14 8.460E-13 1.010E-12 7.850E-13 6.140E-13 5.260E-13
c       5.040E-13 5.320E-13 6.110E-13 8.900E-13 1.180E-12 1.810E-12
c       2.380E-12 2.890E-12 3.380E-12 4.290E-12 5.110E-12 6.920E-12
c       8.480E-12 1.110E-11 1.330E-11 1.540E-11 1.740E-11 2.120E-11
c        2.520E-11
c
c     ***** MATERIAL CARDS
c     ************************************************************
c      AIR: ANSI/ANS-6.4.3, Dry air;  density = 0.0012 g/cm^3
c           Composition by mass fraction
c     ************************************************************
m1     7014.50c  -.75519
       8016.60c  -.23179
       6000.60c  -.00014
      18000.35c  -.01288
c
c     ************************************************************
c      CONCRETE: ANSI/ANS-6.4.3;   density = 2.32 g/cm^3
c                Composition by mass fraction
c     ************************************************************
m2      1001.50c  -.0056
        8016.60c  -.4983
       11023      -.0171
       12000      -.0024
       13027.50c  -.0456
       14000.50c  -.3158
       16000      -.0012
       19000.50c  -.0192
       20000.50c  -.0826
       26000.50c  -.0122
c
c     ************************************************************
c      SOIL: [Jacob, Radn. Prot. Dos. 14, 299, 1986]
c            density = 1.625 g/cm^3;  Composition by mass fraction
```

```
c     ************************************************************
m3     1001.50c  -.021
       6012.50c  -.016
      19000.50c  -.013
      26000.50c  -.011
      20000.50c  -.041
      13027.50c  -.050
      14000.50c  -.271
       8016.60c  -.577
c
c     ************************************************************
c      Fuel-Basket TN-68 Cask  (Table 5.3-1)
c           Density = 3.231 g/cm^3;  Composition by atom fraction
c     ************************************************************
m4     92238.50c  0.14291
       92235.50c  0.00494
       40000.60c  0.09981
       28000.50c  0.02423
       26000.50c  0.18629
       25055.50c  0.00545
       24000.50c  0.05470
       13027.50c  0.18597
        8016.60c  0.29570
c
c     ************************************************************
c      Top Fitting TN-68 Cask (Table 5.3-1)
c           Density = 0.491 g/cm^3;  Composition by atom fraction
c     ************************************************************
m5     26000.50c  0.50712
       28000.50c  0.06595
       25055.50c  0.01483
       24000.50c  0.14890
       40000.60c  0.26320
c
c     ************************************************************
c      Plenum/Basket TN-68 (Table 5.3-1)
c          Density = 1.158 g/cm^3;   Composition by atom fraction
c     ************************************************************
m6     26000.50c  0.34907
       28000.50c  0.04535
       40000.60c  0.17975
       25055.50c  0.01021
       24000.50c  0.10246
       13027.50c  0.31316
c
c     ************************************************************
c      Bottom/Basket TN-68 (Table 5.3-1)
c          Density = 1.918 g/cm^3;  Composition by atom fraction
c     ************************************************************
m7     26000.50c  0.48631
       28000.50c  0.06329
       25055.50c  0.01423
       24000.50c  0.14285
       13027.50c  0.23378
       40000.60c  0.05954
c
c     ************************************************************
```

```
c     Basket Periphery (SS304) TN-68 (Table 5.3-1)
c          Density = 7.92 g/cm^3;   Composition by atom fraction
c    ************************************************************
m8    26000.50c  0.68826
      25055.50c  0.02013
      24000.50c  0.20209
      28000.50c  0.08952
c
c    ************************************************************
c     Carbon Steel TN-68 (Table 5.3-1)
c          Density = 7.8212 g/cm^3;  Composition by atom fraction
c    ************************************************************
m9    26000.50c  0.95510
       6000.60c  0.04490
c
c    ************************************************************
c      Outer Basket/Rails TN-68 (Table 5.3-1)
c          Density = 2.702 g/cm^3;   Composition by atom fraction
c    ************************************************************
m10   13027.50c  1.00000
c
c    ************************************************************
c     Polypropylene Disk TN-68 (Table 5.3-1)
c        Density = 0.90 g/cm^3;   Composition by atom fraction
c    ************************************************************
m11    6012.50c .33480
       1001.50c .66520
c
c    ************************************************************
c     Resin/Aluminum Composite for TN-68 (Table 5.3-1)
c         Density = 1.687 g/cm^3;   Composition by atom fraction
c    ************************************************************
m12   13027.50c  0.10331
       6012.50c  0.24658
       8016.60c  0.21985
       1001.50c  0.42207
       5010.60c  0.00164
       5011.60c  0.00655
c
c    ************************************************************
c     Berm (Silica + water) for ISFSI Site (SAR Page 7a-5);
c         density = 1.400 g/cm^3;  Composition by atom fraction
c    ************************************************************
m13    14000.50c   0.26524
        8016.60c   0.59855
        1001.50c   0.13621
c
c    ************************************************************
c     Redwood for Impact Limiter  (Standard Composition SCALE4.4)
c         density = 0.387 g/cm^3;  Composition by atom fraction
c    ************************************************************
m14     6012.50c   0.2857
        8016.60c   0.2381
        1001.50c   0.4762
c
c    ************************************************************
c     Balsa for Impact Limiter  (Standard Composition SCALE4.4)
```

```
c          density = 0.125 g/cm^3;  Composition by atom fraction
c     ************************************************************
m15      6012.50c   0.2857
         8016.60c   0.2381
         1001.50c   0.4762
c
c  prdmp 2j 1
c  print
```

TABLE 5.1-1

TN-68 CASK SHIELD MATERIALS

Component	Material	Density (g/cm^3)	Thickness (inches)
Cask Body Wall	Carbon Steel	7.82	7.50
Lid	Carbon Steel	7.82	9.50
Bottom	Carbon Steel	7.82	9.75
Resin [a]	Polyester Resin Styrene Aluminum Hydrate Zinc Borate	1.58	6.00
Aluminum Box	Aluminum	2.7	0.12
Outer Shell	Carbon Steel	7.82	0.75
Basket [b]	Stainless Steel (Inserts) Stainless Steel (Strips) Aluminum Neutron Poison Material [c]	7.92 7.92 2.7	0.19 0.31 0.31
Impact Limiter	Carbon Steel Redwood Balsa Wood	7.82 0.387 0.125	0.25 19.25[d] 15.25[d]
Basket Hold Down Ring	Carbon Steel	7.82	1.0
Ancillary Shield Ring	Carbon Steel	7.82	0.5

Notes:

a The neutron shielding is borated polyester resin compound with a density of 1.58 g/cc. The four major constituents are listed in the table.

b The stainless steel inserts of the basket are 0.19 inches thick. The remaining portion of the basket consists of 10.4-inch sections containing neutron poison material intermittent with 4-inch stainless steel sections. See Chapter 3 for the basket design.

c This is modeled as aluminum for shielding purposes (see Chapter 6 for details).

d Thickness of wood is variable.

TABLE 5.1-2

SUMMARY OF DOSE RATES
(Exclusive Use)

Normal Conditions[1]	Package Surface mSv/h (mrem/h)			Vehicle Edge mSv/h (mrem/h)			2 Meter from Vehicle mSv/h (mrem/h)		
Radiation	Top	Side	Bottom	Top	Side	Bottom	Top	Side	Bottom
Gamma	0.072 (7.2)	0.944 (94.4)	0.053 (5.3)	-	0.501(50.1)	-	-	0.074 (7.44)	-
Neutron	0.002 (0.2)	0.157 (15.7)	0.003 (0.31)	-	0.072 (7.2)	-	-	0.008 (0.80)	-
Total	0.074 (7.4)	1.10 (110)	0.056 (5.6)	-	0.57 (57.3)	-	-	0.082 (8.24)	-
Limit	10 (1000)	10 (1000)	10 (1000)		2 (200)			0.1 (10)	

Hypothetical Accident Conditions[2]	1 Meter from Package Surface mSv/h (mrem/h)		
Radiation	Top	Side	Bottom
Gamma	0.508 (50.8)	0.609 (60.9)	0.236 (23.6)
Neutron	0.099 (9.9)	2.12 (212)	0.317 (31.7)
Total	0.608 (60.8)	2.73 (273)	0.553 (55.3)
Limit	10 (1000)	10 (1000)	10 (1000)

(1) Doses from normal conditions are calculated with the optional ancillary shield ring in place.
(2) The ancillary shield ring, the neutron shield, and the impact limiters are removed

TABLE 5.2-1

BWR FUEL ASSEMBLY DESIGN CHARACTERISTICS

Transnuclear, ID	7 x 7 - 49/0	8 x 8 - 63/1	8 x 8 - 62/2	8 x 8 - 60/4	8 x 8 - 60/1	9 x 9 - 74/2	10x10- 92/2
GE Designations	GE2 GE3	GE4	GE-5 GE-Pres GE-Barrier GE8 Type I	GE8 Type II	GE9 GE10	GE11 GE13	GE12
Max Length (in)[a]	176.2	176.2	176.2	176.2	176.2	176.2	176.2
Max Width (in)[a]	5.44	5.44	5.44	5.44	5.44	5.44	5.44
Rod Pitch (in)	0.738	0.640	0.640	0.640	0.640	0.566	0.510
No of Fueled Rods	49	63	62	60	60	66 full 8 partial	78 full 14 partial
Maximum Active Fuel Length (in)	144	146	150	150	150	146" full 90" partial	150" full 93" partial
Fuel Rod OD (in)	0.563	0.493	0.483	0.483	0.483	0.440	0.404
Clad Thickness (in)	0.032	0.034	0.032	0.032	0.032	0.028	0.026
Fuel Pellet OD (in)	0.487	0.416	0.410	0.410	0.411	0.376	0.345
No of Water Rods	0	1	2	4	1	2	2
Water Rod OD (in)	---	0.493	0.591	2 @ 0.591 2 @ 0.483	1.340	0.980	0.980
Water Rod ID (in)	---	0.425	0.531	2 @ 0.531 2 @ 0.419	1.260	0.920	0.920
Maximum MTU/assembly[b]	0.1977	0.1880	0.1886	0.1825	0.1834	0.1766	0.1867
Minimum Plenum Volume (in^3)	2.066	1.595	1.273	1.273	1.291	1.184	0.995
Fill Gas	He	He	He	He	He	He	He
Maximum Initial Rod Pressurization (psig)	10	10	80	80	80	155	155

[a] Unirradiated length and width.
[b] The maximum MTU/assembly is calculated based on the theoretical density. The calculated value is higher than the actual.

TABLE 5.2-1a

BWR FUEL ASSEMBLY DESIGN CHARACTERISTICS
(continued)

Transnuclear, ID	7 x 7 - 49/0	8 x 8 - 63/1	8 x 8 - 62/2	8 x 8 - 60/4	8 x 8 - 60/1	9 x 9 - 74/2	10x10- 92/2
GE Designations	GE2 GE3	GE4	GE-5 GE-Pres GE-Barrier GE8 Type I	GE8 Type II	GE9 GE10	GE11 GE13	GE12
Max Length (in)[a]	176.2	176.2	176.2	176.2	176.2	176.2	176.2
Plenum Length (in)	16.47	14.47	10.47	10.47'	10.47	14.47	10.47
Top Fitting Length (in)	8.34	8.34	8.34	8.34	8.34	8.34	8.34
Bottom Fitting Length (in)	7.39	7.39	7.39	7.39	7.39	7.39	7.39

TABLE 5.2-2

BWR FUEL ASSEMBLY HARDWARE CHARACTERISTICS

Item	Material	Average Mass (kg/assembly)
Fuel Zone		
Cladding	Zircaloy	49.2
Spacers	Zircaloy	1.95
Spacer Springs	Inconel	0.36
Fuel-Gas Plenum Zone		
Cladding	Zircaloy	4.89
Springs	Stainless Steel	1.05
Top End Fitting Zone		
Upper Tie Plate	Stainless Steel	2.08
Lock Tab Washers & Nuts	Stainless Steel	0.05
Expansion Springs	Inconel	0.43
End Plugs	Zircaloy	1.26
Bottom End Fitting Zone		
Finger Springs	Inconel	0.05
End Plugs	Zircaloy	1.26
Lower Tie Plate	Stainless Steel	4.70
Channel		
Channel Sleeve	Zircaloy	37.1
Channel Spacer & Rivet[a]	Stainless Steel	0.13
Channel Fastener[a]		
Guard	Stainless Steel	0.46
Spring & Bolt	Inconel	0.13
Total		105.1

[a] The channel spacer, rivet and fastener are located at top end fitting zone.

TABLE 5.2-3

MATERIAL COMPOSITIONS FOR FUEL ASSEMBLY HARDWARE MATERIALS

Material[a]	Element	Weight %
Zircaloy	Zirconium	98.225
	Tin	1.5
	Chromium	0.1
	Nitrogen	0.05
	Cobalt	0.001
Stainless Steel (SS304)	Iron	69.5
	Chromium	19.0
	Nickel	9.5
	Manganese	1.92
	Cobalt	0.08
Inconel	Nickel	73
	Chromium	15
	Iron	7
	Titanium	2.5
	Silicon	1.85
	Cobalt	0.649

a Material compositions are taken from the SCALE Standard Composition Library, however, cobalt impurities are taken from Reference 2.

TABLE 5.2-4

BWR FUEL ASSEMBLY SOURCE (with CHANNELS)
BUNDLE AVERAGE ENRICHMENT 3.3 wt% U235,
40,000 MWD/MTU, 20 YEAR COOLING TIME

Total Gamma Source (γ/sec/assembly)

TN Design ID	Total
7x7-49-0	9.47E14
8x8-63-1	9.03E14
8x8-62-2	8.92E14
8x8-60-4	8.78E14
8x8-60-1	8.82E14
9x9-74-2	8.52E14
10x10-92-2	8.98E14

Total (α,n) plus Spontaneous Fission Neutron Source (n/sec/assembly)

TN Design ID	Total
7x7-49-0	6.22E07
8x8-63-1	5.69E07
8x8-62-2	5.47E07
8x8-60-4	5.58E07
8x8-60-1	5.58E07
9x9-74-2	4.80E07
10x10-92-2	5.01E07

TABLE 5.2-5
PRIMARY GAMMA SOURCE SPECTRUM
SCALE 18 GROUP STRUCTURE
GENERAL ELECTRIC 7x7, BUNDLE AVERAGE ENRICHMENT 3.3wt% U235,
40,000 MWD/MTU, AND 20 YEAR COOLING TIME
WITH CHANNELS

		γ/sec/assembly			
Scale Group	Energy Interval, MeV	Active Fuel Zone	Plenum Zone[a]	Top Fitting Zone[a]	Bottom Fitting Zone[a]
28	8.00E+00 to1.00E+01	3.470E+04			
29	6.50E+00 to 8.00E+00	1.634E+08			
30	5.00E+00 to 6.50E+00	8.333E+05			
31	4.00E+00 to 5.00E+00	2.077E+06			
32	3.00E+00 to 4.00E+00	6.284E+06			
33	2.50E+00 to 3.00E+00	2.159E+08			
34	2.00E+00 to 2.50E+00	1.370E+09			
35	1.66E+00 to 2.00E+00	2.698E+10			
36	1.33E+00 to 1.66E+00	1.337E+12	3.364E+10	1.016E+11	1.135E+11
37	1.00E+00 to 1.33E+00	1.427E+13	1.191E+11	3.782E+11	4.020E+11
38	8.00E-01 to 1.00E+00	8.331E+12			
39	6.00E-01 to 8.00E-01	4.849E+14			
40	4.00E-01 to 6.00E-01	1.032E+13			
41	3.00E-01 to 4.00E-01	1.038E+13			
42	2.00E-01 to 3.00E-01	1.606E+13			
43	1.00E-01 to 2.00E-01	5.414E+13			
44	5.00E-02 to 1.00E-01	2.779E+13			
45	1.00E-02 to 5.00E-02	2.678E+14			

[a] Cobalt-60 is the gamma source of significance in the fuel assembly hardware.

TABLE 5.2-6

NEUTRON SOURCE DISTRIBUTION
GENERAL ELECTRIC 7x7,
BUNDLE AVERAGE ENRICHMENT 3.3wt% U-235,
40,000 MWD/MTU, AND 20 YEAR COOLING TIME
WITH CHANNELS
TOTAL (α,n PLUS SPONTANEOUS FISSION) NEUTRON SOURCE
SCALE STRUCTURE USING SPECTRA FOR URANIUM DIOXIDE

Scale Group	Energy Interval, MeV	n/sec/assembly
1	6.43E+00 to 2.00E+01	1.131E+06
2	3.00E+00 to 6.43E+00	1.298E+07
3	1.85E+00 to 3.00E+00	1.457E+07
4	1.40E+00 to 1.85E+00	8.112E+06
5	9.00E-01 to 1.40E+00	1.901E+07
6	4.00E-01 to 9.00E-01	1.185E+07
7	1.00E-01 to 4.00E-01	2.320E+06

TABLE 5.2-7

SOURCE TERM SUMMARY

SAS2H Source Terms
Summary
Neutron and Gamma Source As a Function of Burnup, Water Density and Active Core Height

7x7 Fuel Assembly 40,000 MWd/MtU Average Burnup 10 Years Cool Time

Power (MW) 5 Cycle Length (days) 527.2

Output File Name	Zone	Frac Core Height	Peaking Factor	Burnup (MWd/MtU)	SAS2H Power (MW)	Water Density (g/cc)	Neutron Source (n/s)	Neutron Peaking Factor	Gamma Source (g/s)	Gamma Peaking Factor
7x7-9-36.output	12	0.95-1.0	0.2410	9640	1.205	0.3609	1.661E+04	0.0028	1.574E+13	0.2303
7x7-25-36.output	11	0.90-0.95	0.6330	25320	3.165	0.3631	6.500E+05	0.1093	4.275E+13	0.6255
7x7-36-37.output	10	0.8-0.9	0.8973	35891	4.486	0.3701	6.005E+06	0.5047	1.238E+14	0.9053
7x7-43-39.output	9	0.7-0.8	1.0766	43065	5.383	0.3861	1.274E+07	1.0707	1.499E+14	1.0964
7x7-46-41.output	8	0.6-0.7	1.1515	46061	5.758	0.4118	1.647E+07	1.3842	1.535E+14	1.1227
7x7-47-43.output	7	0.5-0.6	1.1912	47649	5.956	0.4375	1.859E+07	1.5624	1.663E+14	1.2164
7x7-48-47.output	6	0.4-0.5	1.2000	48000	6.000	0.4708	1.877E+07	1.5775	1.674E+14	1.2244
7x7-48-53.output	5	0.3-0.4	1.2000	48000	6.000	0.5251	1.819E+07	1.5288	1.671E+14	1.2223
7x7-47-59.output	4	0.2-0.3	1.1836	47345	5.918	0.5945	1.649E+07	1.3859	1.644E+14	1.2027
7x7-43-70.output	3	0.1-0.2	1.0750	43001	5.375	0.7008	1.005E+07	0.8447	1.484E+14	1.0854
7x7-31-75.output	2	0.05-0.1	0.7746	30985	3.873	0.7541	1.002E+06	0.1683	5.245E+13	0.7674
7x7-9-76.output	1	0.0-0.05	0.2357	9426	1.178	0.7603	1.100E+04	0.0018	1.542E+13	0.2256
Average/Total			0.9917	39670	4.959	0.5016	1.190E+08	1.0000	1.367E+15	1.0000
Uniform Case		0.0-1.0	1	40000	5	0.432	8.976E+07		1.382E+15	
Ratio to Non-Uniform Case							1.326		0.989	

TABLE 5.3-1

MATERIALS INPUT FOR MCNP

Zone	Material	Density (g/cc)	Element/Nuclide	Library Identifier	Atomic Number Density (atoms/barn-cm)
Fuel/Basket	UO_2	1.855	U-235	92235	1.405E-04
			U-238	92238	4.065E-03
			O	8016	8.411E-03
	Zircaloy	0.430	Zr	40302	2.839E-03
	SS304	0.707	Cr	24304	1.556E-03
			Mn	25055	1.550E-04
			Fe	26304	5.299E-03
			Ni	28304	6.892E-04
	Aluminum	0.239	Al	13027	5.290E-03
Plenum/Basket	Zircaloy	0.359	Zr	40302	2.370E-03
	SS304	0.614	Cr	24304	1.351E-03
			Mn	25055	1.346E-04
			Fe	26304	4.602E-03
			Ni	28304	5.985E-04
	Aluminum	0.185	Al	13027	4.129E-03
Top Fitting	Zircaloy	0.182	Zr	40302	1.202E-03
	SS304	0.309	Cr	24304	6.800E-04
			Mn	25055	6.774E-05
			Fe	26304	2.316E-03
			Ni	28304	3.012E-04
Bottom Fitting/Basket	Zircaloy	0.205	Zr	40302	1.353E-03
	SS304	1.475	Cr	24304	3.246E-03
			Mn	25055	3.234E-04
			Fe	26304	1.105E-02
			Ni	28304	1.438E-03
	Aluminum	0.238	Al	13027	5.312E-03
Basket Periphery	SS304	7.92	Cr	24304	1.743E-02
			Mn	25055	1.736E-03
			Fe	26304	5.936E-02
			Ni	28304	7.721E-03
Basket Periphery/Shim/Rails	Aluminum	2.702	Al	13027	6.031E-02
Cask Body/shield ring impact limiter skin	Carbon Steel	7.8212	Fe	26000	8.350E-02
			C	6012	3.925E-03
Resin/Aluminum	Resin (1.58 g/cc) & Al (2.702 g/cc)	1.687	O	8016	2.245E-02
			Al	13027	1.055E-02
			C	6012	2.518E-02
			H	1001	4.310E-02
			B-10	5010	1.662E-04
			B-11	5011	6.692E-04

TABLE 5.3-1 (continued)

MATERIALS INPUT FOR MCNP

Zone	Material	Density (g/cc)	Element/Nuclide	Identifier	Atomic Number Density (atoms/barn-cm)
Impact Limiter	Balsa Wood	0.125	C	6012	2.787E-03
			O	8016	2.323E-03
			H	1001	4.646E-03
Impact Limiter	Redwood	0.387	C	6012	8.630E-03
			O	8016	7.192E-03
			H	1001	1.438E-02

TABLE 5.4-1

RESPONSE FUNCTIONS FOR GAMMA

Photon Energy (MeV)	Response (10^{-12} Sv cm^2)
0.01	0.0769
0.015	0.846
0.02	1.01
0.03	0.785
0.04	0.614
0.05	0.526
0.06	0.504
0.08	0.532
0.10	0.611
0.15	0.890
0.20	1.18
0.30	1.81
0.40	2.38
0.50	2.89
0.60	3.38
0.80	4.29
1.0	5.11
1.5	6.92
2.0	8.48
3.0	11.1
4.0	13.3
5.0	15.4
6.0	17.4
8.0	21.2
10.0	25.2

TABLE 5.4-2

RESPONSE FUNCTIONS FOR NEUTRON

Neutron Energy (MeV)	Response (10^{-12} Sv cm^2)
2.5E-8	8.0
1.0E-7	10.4
1.0E-6	11.2
1.0E-5	9.2
1.0E-4	7.1
1.0E-3	6.2
1.0E-2	8.6
2.0E-2	14.6
5.0E-2	35.0
1.0E-1	69.0
2.0E-1	126
5.0E-1	258
1.0	340
1.5	362
2.0	352
3.0	380
4.0	409
5.0	378
6.0	383
7.0	403
8.0	417
10.0	446
14.0	520
17.0	610
20.0	650

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE 5.2-1

AXIAL BURNUP PROFILE FOR DESIGN BASIS FUEL

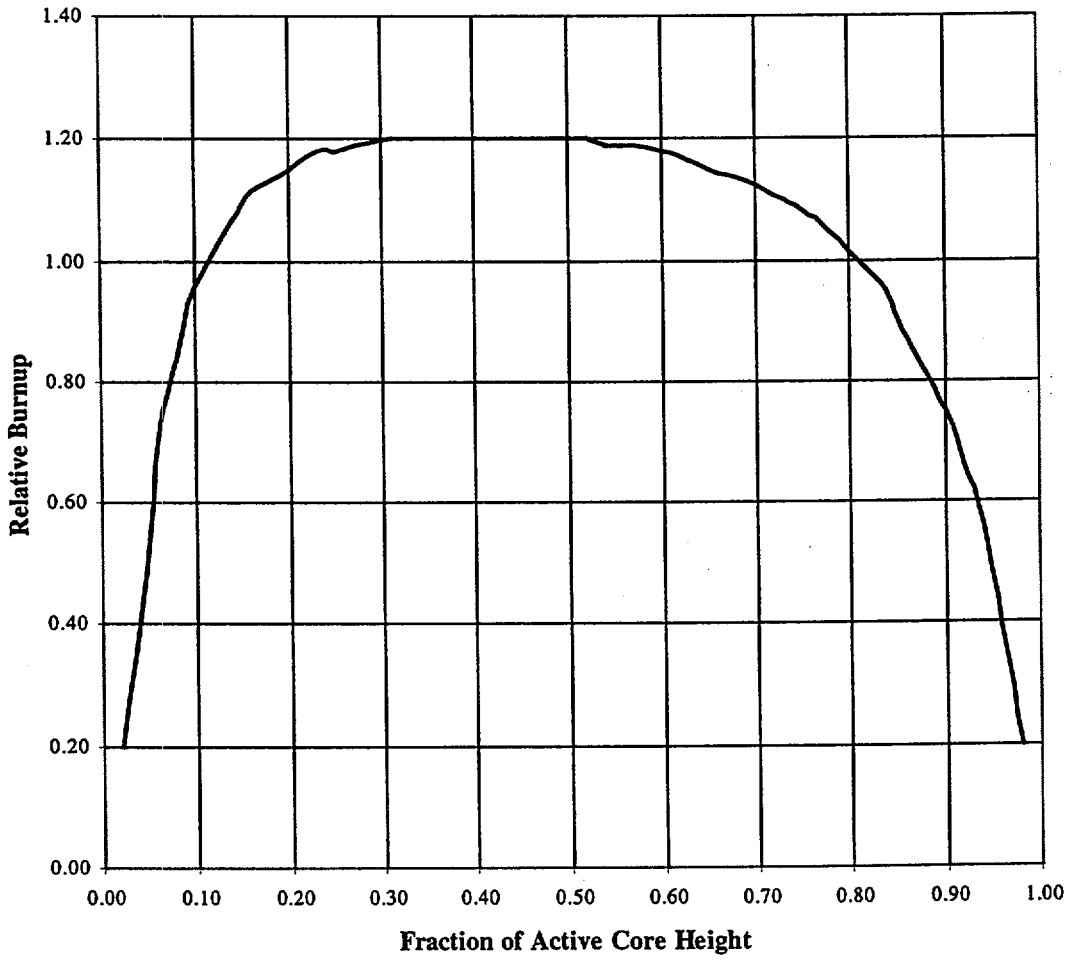


FIGURE 5.3-1

MCNP TOP HALF MODEL
VIEW THROUGH TRUNNIONS

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE 5.3-2

MCNP BOTTOM HALF MODEL
VIEW THROUGH TRUNNIONS

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE 5.4-1

TN-68 RADIAL GAMMA DOSE RATE PROFILE

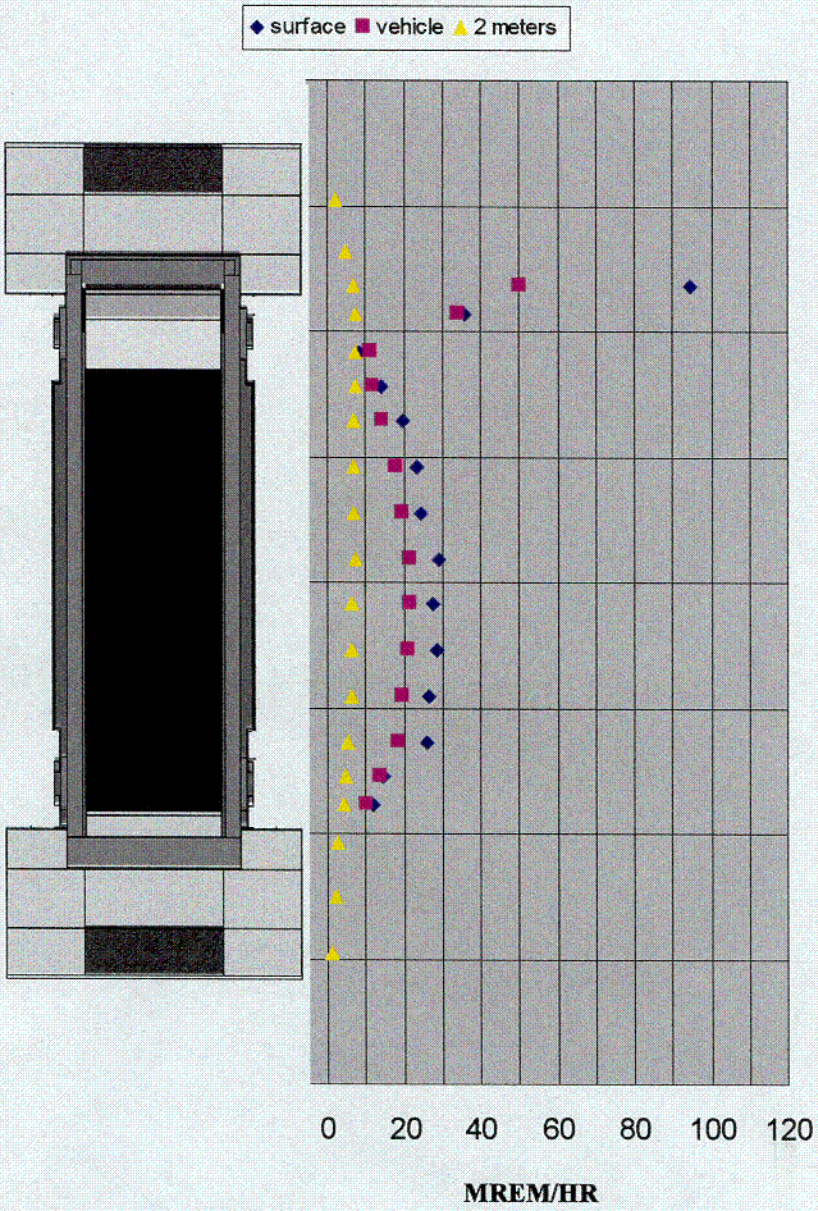


FIGURE 5.4-2

TN-68 RADIAL NEUTRON DOSE RATE PROFILE

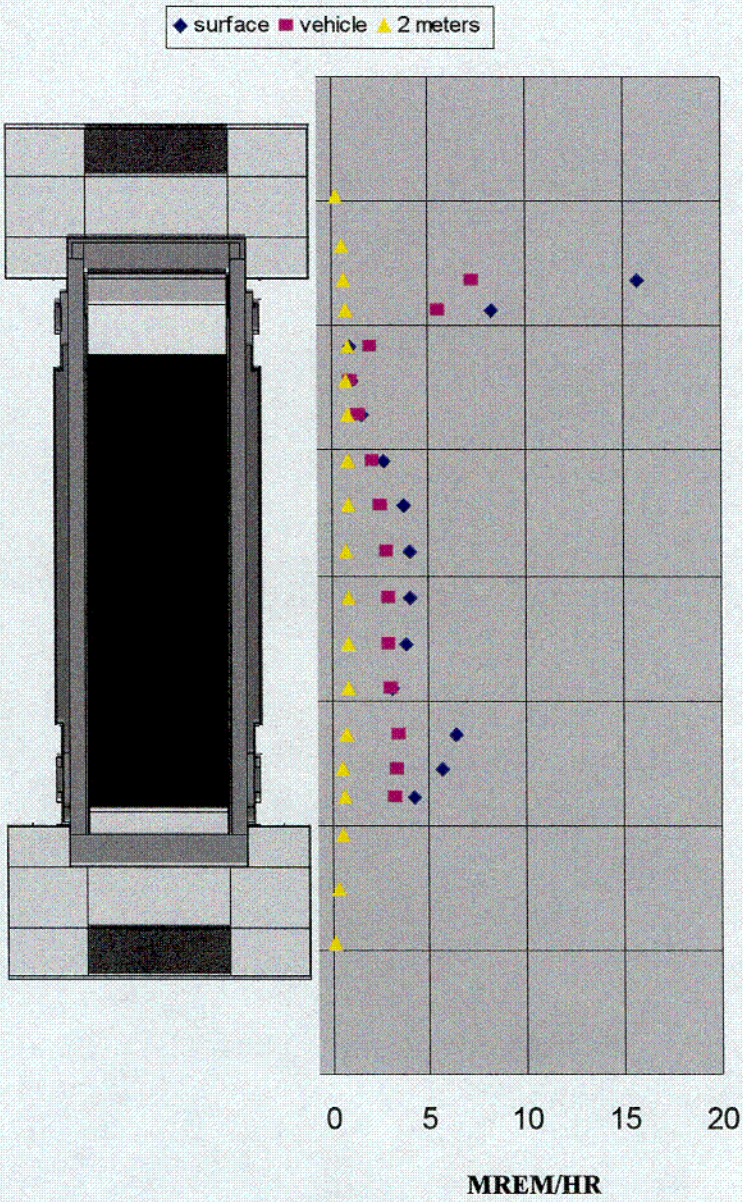


FIGURE 5.4-3

TN-68 GAMMA DOSE RATE
IMPACT LIMITER SURFACE

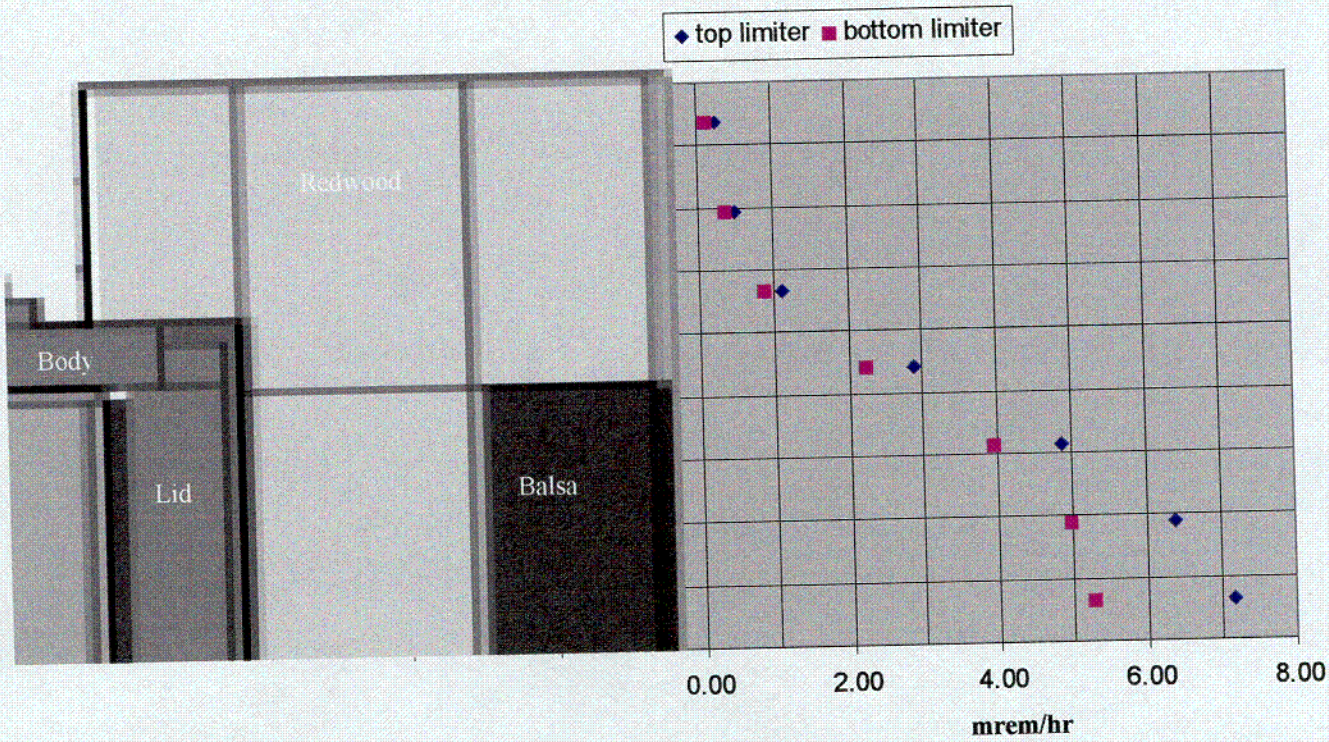
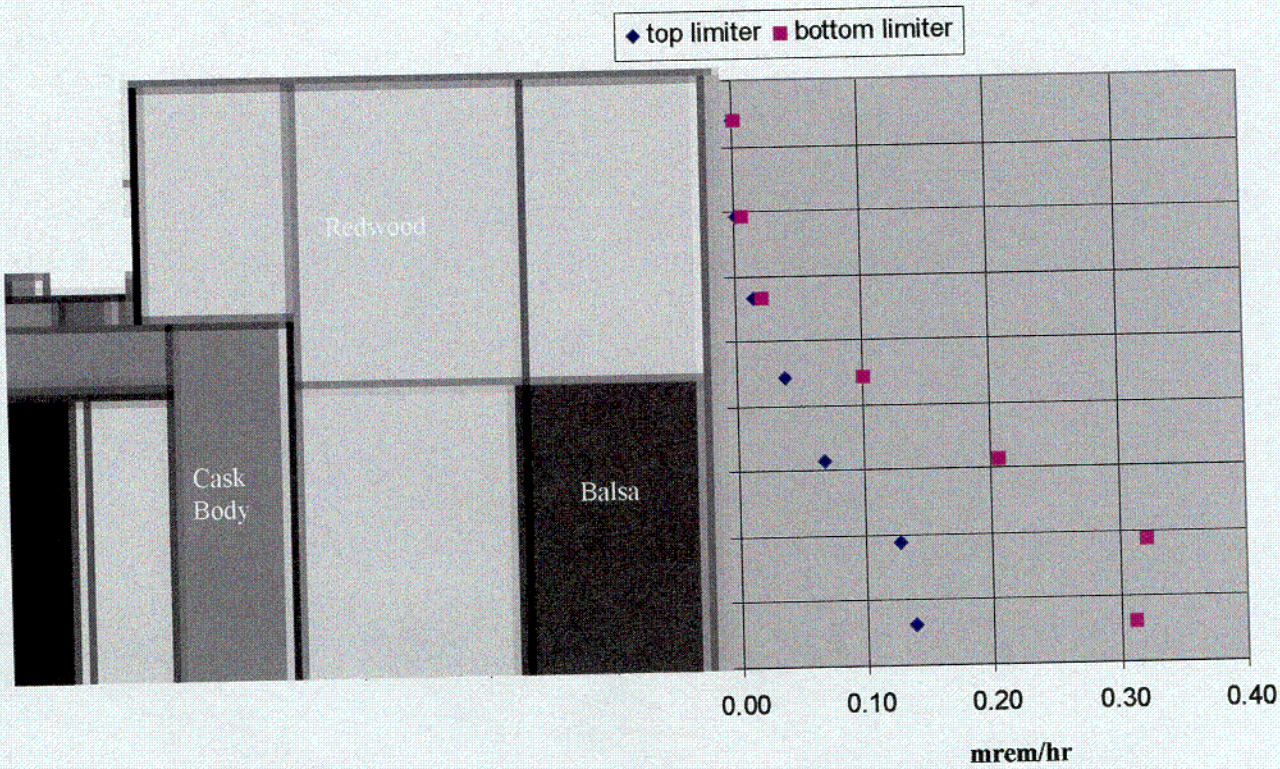


FIGURE 5.4-4

TN-68 NEUTRON DOSE RATE
IMPACT LIMITER SURFACE



TN 68 TRANSPORT PACKAGING

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CHAPTER 6

CRITICALITY EVALUATION

6.1 Discussion and Results

Criticality control in the TN-68 is performed by the basket structural components, which maintain the relative position of the spent fuel assemblies under normal and accident conditions as demonstrated in Chapter 2, and by the neutron absorbing plates between the basket compartments.

The TN-68 contents are limited to the GE fuel designs listed in Chapter 2, with a maximum lattice-average enrichment of 3.7 wt % U235. Fuel assemblies with or without channels are acceptable. Any fuel channel thickness from 0.065 to 0.120 inch is acceptable on any of the fuel designs. Criticality control does not require special loading patterns or special rotational orientation of the fuel assemblies. The TN-68 may be loaded with pool water at maximum density, i.e., at 4 °C.

The criticality evaluation is organized as follows:

a) Determination of the most reactive lattice (Section 6.4.2A). All of the design basis fuels are evaluated with uniform enrichment to determine the most reactive geometry. The effect of water in the fuel pins and of fuel channels is also evaluated.

b) Uniform enrichment model validation (Section 6.4.2B). The results using fuel lattice models with pin-by-pin enrichment variation are compared with results obtained using uniform enrichment models. Vanished lattices are also compared with the uniform enrichment models. If the uniform enrichment model underpredicts k_{eff}, the modeling bias will be determined and the Upper Subcritical Limit will be reduced by that amount.

c) TN-68 single package evaluation (Section 6.4.2C). The uniform enrichment model of the most reactive fuel lattice is used for this evaluation. The TN-68 package is evaluated for the following conditions:
   - varied water density,
   - varied fuel compartment inside dimension and pitch between compartments,
   - off-center placement of fuel in the compartments.
   - combined effects: most reactive normal configuration
   - hypothetical accident condition.

d) TN-68 package infinite array evaluation (Section 6.4.2D).
   - most reactive normal configuration bounds the hypothetical accident condition

e) Benchmarking (Section 6.5). An upper subcritical limit (USL) is determined by subtracting from unity an administrative margin of 0.05, the bias determined from benchmark calculations and any modeling bias.

All calculations assume fresh fuel composition and ignore burnable poisons.

Results

The maximum k_{eff} under all evaluated conditions is 0.9250 ± 0.0008. The upper subcritical limit (USL) is 0.9331. In all cases, $k_{eff} + 2\sigma < USL$.

6.2 Package Fuel Loading

The allowable contents are listed in Chapter 2. Fuel characteristics used in the criticality calculations are listed in Table 6.2-1. Because an infinite array of packages is evaluated for both normal and accident conditions, the transport index for criticality safety is zero.

Where fuel pins have variable axial enrichment, the average is calculated for each axial zone (lattice), and the lattice with the highest average enrichment is used to characterize the entire bundle for criticality purposes. The average enrichment is defined as the simple arithmetic average of pin enrichments:

$$E_{avg} = \sum_{i=1,n} E_i \Big/ n$$

Where E_i is the enrichment of pin i, and n is the number of fuel pins in the lattice. There is no averaging of the axial enrichment variation in this evaluation; "bundle average" enrichments, which are an average enrichment over the entire fuel bundle, including natural uranium blankets, are not used to qualify fuel for transport in the TN-68.

To maintain subcriticality, the maximum lattice-average enrichment of the fuel bundle must be less than or equal to 3.7 weight % U235.

The acceptable contents of the TN-68 do not include failed fuel other than fuel with hairline cracks or pinholes in the cladding. Fuel bundles from which fuel pins are missing are not allowable contents unless the missing pin is replaced.

6.3 Model Specification

6.3.1 Description of Calculational Model

Infinite length fuel, basket, and packaging body are modeled by taking one "layer" of the basket, as shown in Figure 6.3-1, and applying periodic reflection to the top and bottom. Fuel pins are modeled individually. Water rods in the fuel are modeled explicitly, except for fuels where large water rods displace several fuel rods. In those cases, due to modeling constraints, the water rods are modeled as water cells, without the Zircaloy tube. The stainless steel basket compartment tubes, stainless steel bars, neutron poison plates, egg-crate gaps, and thermal expansion gaps are modeled explicitly as shown in Figures 6.3-2 through 6.3-8. The dimensions used for the neutron poison plates and the gaps are shown in Table 6.3-1, which compares the model dimensions with the design dimensions. The thermal expansion gaps and the vertical egg-crate slots are wider in the model than in the TN-68 design, and the area of the neutron poison plate is correspondingly smaller.

The dimensional variations in the basket that can affect criticality are the thickness of the tube and plate materials, and the inside dimension of the tubes. Except as noted, all cases are run with the nominal compartment inside dimension, 6 inches. Section 6.4.2C demonstrates that k_{eff} decreases as the compartment size increases, and it also demonstrates criticality safety for the minimum compartment size, 5.97 inches. The tolerances on the plate and tube materials are very small (see Table 6.3-1). The undersize allowance is negligible from the criticality standpoint, and the oversize tolerance would decrease reactivity. Furthermore, in practice it is not possible to assemble the basket compartments and plates perfectly line to line as they are in the criticality model, so the compartments will be slightly further apart than the plate thicknesses alone would indicate. These considerations confirm that the material thicknesses used in the calculations are appropriate.

The aluminum basket rails, packaging body, and outer water reflector are included, and the outer neutron shield is neglected. The complete package model is shown in Figure 6.4-4. The following differences between the model and the design have been evaluated:

Model	Design
0.156 inch thick outer absorber plates	0.305 inch thick outer absorber plates
egg crate slots repeated at all compartment corners, including ends of plates	plates have thermal expansion gap up to 0.5 inch wide at the ends of each plate
aluminum basket rails all around	stainless steel rail at 4 axes of basket
4 outermost plates are 0.156 thick, borated	4 outermost plates are 0.156 inch thick, non-borated aluminum
Basket rail holes circular	Basket rail holes triangular

The first four changes are evaluated by the model shown in Figure 6.3-9. In this model,

- half thickness borated aluminum plates are added at the perimeter
- this added material is only half the compartment width; the volume of the plates still left out at the intersections is greater than the volume of neutron absorber that would be removed if the plates were cut 0.5 inch shorter at their ends
- stainless steel plates are inserted at the axes
- at the four axes, the outermost basket plate is replaced with non-borated aluminum

A comparison of this model with the baseline indicates that the model is equivalent to, or conservative with respect to the design:

baseline model: $k_{eff} = 0.9151 \pm 0.0016$

design approximation model: $k_{eff} = 0.9123 \pm 0.0016$.

The triangular openings in the design basket rails have a larger cross sectional area than the cylindrical holes used in the criticality model. By reducing the amount of water at the basket perimeter, the moderation of reflected neutrons is reduced. This in turn reduces the effectiveness of the borated aluminum plate at the basket perimeter in capturing returning neutrons. Less water at the perimeter causes increasing reactivity, and therefore the model with the cylindrical holes is conservative.

The description above refers to the model used for the single package evaluation, Section 6.4.2 C. The models used for the most reactive fuel evaluation and the uniform enrichment validation are somewhat different. The differences are described in the respective calculation descriptions. For array evaluations, the cylindrical water reflector is replaced with a close fitting, water cuboid with mirror reflection on all sides.

6.3.2 Package Regional Densities

Materials are converted to atom densities by the Material Information Processor in the CSAS25 code sequence[1]. The mass densities supplied to the code are reported in Table 6.3-2.

Two materials are modeled for the poison plates. The first is an alloy of aluminum and about 1.7 wt% boron, the boron being enriched to about 95 wt % B10. Because this material is subjected to extensive acceptance testing as described in Chapter 8, the calculations take credit for 90% of the minimum specified boron 10 areal density. The density used in the calculation is 0.034 g B10/cm^3, and the plate thickness is 0.792 cm. The corresponding areal density is 0.034(0.792) = 0.0269 gB10/cm^2. The minimum specified boron content, using 90% credit is 0.0269/0.9 = 0.030 g B10/cm^2.

The second material evaluated is a boron-carbide/aluminum metal matrix composite, consisting of about 15 volume % boron carbide particles and 85 % aluminum. This is subject to less rigorous acceptance testing as described in Chapter 8. In this case, the calculations take 75% credit for the B10, and the minimum specified boron content for this material is 0.036 g B10/cm^2.

Studies have been conducted to show that neither of these materials will degrade significantly as both have excellent resistance to thermal and radiation alteration in the service environments of interest to this application. Both are solid, non-friable materials physically similar to their base aluminum alloys. Neither material includes any organic components or binders. They are held in place and protected from damage by the surrounding stainless steel bar and tube structure. The basket structure encloses the neutron absorber plates on all six sides.

6.4 Criticality Calculation

6.4.1 Calculational or Experimental Method

All calculations are performed using the CSAS25 sequence from the SCALE4.3 code system[1] with the SCALE 27-group ENDF/B-IV cross section library. Within this sequence, resonance correction based on the fuel pin cell description is performed by NITAWL using the Nordheim Integral method, and k_{eff} is determined by the KENOVa code using the Monte Carlo technique. A sufficiently large number of neutron histories is run so that the standard deviation is below 0.0020 for all calculations.

6.4.2 Fuel Loading or Other Contents Loading Optimization

A. Determination Of The Most Reactive Fuel Lattice

All lattices listed in Table 6.2-1 are evaluated with the maximum TN-68 design basis lattice enrichment, 3.7%, in all pins. The lattices are analyzed with and without water in the fuel pellet-cladding annulus, and with and without fuel channels. All lattices are analyzed with the minimum and maximum fuel channel thicknesses, 0.065 and 0.120 inch thick, and one intermediate thickness. The lattices are centered in the fuel compartments.

The package model for this evaluation differs from the TN-68 design in the following ways:

- the boron 10 content in the poison plates is lower,
- the egg-crate (vertical) slots run the full height of the poison plate,
- the fuel, basket, and packaging body are infinite length (periodic reflection on "z" faces of model),
- the basket rails are a homogenized 50/50 volume % mixture of water and aluminum, and
- the stainless steel bars between compartments are modeled as carbon steel.

In all other respects, the model is the same as that described in Sections 6.3.1 and 6.3.2. The sole purpose of this model is to determine the *relative* reactivity of different lattices in a configuration similar to the TN-68. The model is shown in Figure 6.4-1.

A typical input file is included in Appendix 6.7. The results of these calculations are listed in Table 6.4-1. The most reactive fuel lattice evaluated for the TN-68 is the GE generation 12 lattice, 10x10 array, with water in the fuel rods and with the 0.065 inch thick fuel channel.

B. Uniform Enrichment Model Validation

Except for the earliest fuels, BWR fuel lattices do not actually have the same enrichment fuel in each fuel pin. It is necessary to validate the use of fuel lattice models in which all fuel pins have the same enrichment, because the most reactive fuel evaluation and the TN-68 criticality use this model. To do this, k_{eff}'s of variable pin-enrichment lattice models and of equivalent uniform enrichment lattice models are calculated and compared. The variable enrichment pin models are all normalized so that the lattice average enrichment is 3.7 %. The pin enrichment patterns and the normalized equivalents are shown in Appendix 6B. In all those patterns, the control blade corner is in the upper right, and the highest enrichment corner is in the lower left; they are oriented this way in the quarter-basket model so that the highest enrichment zones face the package longitudinal axis. The equivalent average enrichment for the uniform enrichment model is defined in Section 6.2.

The package model for this evaluation differs from the TN-68 design in the following ways:

- the boron 10 content in the poison plates is lower,
- the egg-crate (vertical) slots run the full height of the poison plate,
- the fuel, basket, and packaging body are infinite length (periodic reflection on "z" faces of model),
- the basket rails are a homogenized 50/50 volume % mixture of water and aluminum,
- the stainless steel bars between compartments are modeled as carbon steel, and
- the package is modeled with a square cross section and 64 fuel assemblies. This is a significant difference from the actual package, but again, the sole purpose of this model is to determine the *relative* reactivity of different fuel models in a configuration similar to the TN-

68. The square cross section was used to simplify modeling the case where the highest enrichment zones of the variable pin enrichment fuel lattices are all rotated toward the package longitudinal (z) axis. This is done by modeling a quarter of the package cross section, and then using mirror reflection along the x and y axes. This orientation is more reactive than either random or uniform rotational orientation of the lattices. The model is shown in Figure 6.4-2.

The results of the calculation are listed in Table 6.4-2. The case designations may be correlated to the pin enrichment patterns by referring to Appendix 6.8. The last six cases in the table are vanished lattice cases corresponding to the six cases immediately before them. These are the lattices above the partial length fuel rods; the partial length rod has vanished, and is replaced by water, as shown in Figure 6.4-3. Because of the improved moderation, the vanished lattice can be more reactive than the complete lattice. Typical input files for the varied enrichment model of both the full and vanished lattices are included in Appendix 6.7.

Examination of the difference between k_{eff} calculated with the uniform enrichment model and k_{eff} calculated with the varied enrichment model indicates that the uniform enrichment model has an average positive (conservative) bias of $0.0032 \pm 0.0037\ \Delta k_{eff}$. Subtracting 2σ results in a negative bias of 0.0042 which will be applied in the determination of the Upper Subcritical Limit in Section 6.5.3.

C. TN-68 Single Package Evaluation

This evaluates the TN-68 in a variety of configurations intended to bound all normal and accident conditions. The following conditions are evaluated for the three most reactive fuels as determined from Table 6.4-1: 10x10, 0.065 channel, 7x7(2, 2b), 0.120 channel, and 8x8(9, 9b, 10), 0.120 channel. All fuels are modeled with uniform pin enrichment. There is water in the fuel pellet-cladding annulus and inside the package.

- Baseline: Fuel centered in 6 inch compartments, 100% water density. TN-68 model and material densities per Sections 6.3.1 and 6.3.2.
- The inside dimension of the compartment is increased to 6.05 inches. All compartments move correspondingly further apart. This case verifies that the minimum compartment size is the most reactive dimensional configuration. By moving the compartments closer together, it reduces the neutron leakage, and by reducing the thickness of the water layer between the lattice and the compartment wall, it reduces the effectiveness of the neutron poison plates between the compartments.
- Variation of water density throughout: The water density in the fuel and the entire basket is varied from 1 to 100%, including water at the maximum density of 1.000 g/cm^3, which occurs at 4 °C. The water reflector remains at full density (0.9982 g/cm^2) for all cases.
- Fuel lattices off-center in the compartments: Several channel thicknesses and a lattice with no channel are investigated. All lattices are shifted toward the longitudinal axis of the basket until the fuel channel or the outer pin cells of the lattice contact two compartment walls. This is not a credible configuration, but is intended to bound all cases of off-center fuel.

These evaluations confirm that

- optimum internal moderation occurs for full density water; the difference between 1.00 g/cm^3 and 0.9982 g/cm^3 is negligible.
- the most reactive fuel configuration is the fuel with a 0.120 inch thick channel shifted off the compartment center toward the package center, and
- a smaller basket compartment sizes results ins a more reactive basket configuration.

Based on these conclusions, the combination of optimum moderation, most reactive fuel configuration, and minimum compartment size, 5.97 inches, is evaluated for both the borated aluminum and the metal matrix composite neutron absorbers.

The results are reported in Table 6.4-3. The model for the combination of off-center fuel with minimum compartment size is shown in Figure 6.4-4, and sample input files are included in Appendix 6.7.

Lastly, the analysis considers the hypothetical drop accident, and the consequent accelerations experienced by the fuel assemblies. The evaluations in Chapter 2 demonstrate that the basket geometry is maintained in the hypothetical accident. Appendix 2.10.7 provides a structural analysis of the fuel under hypothetical accident conditions. Based on the conclusions of Appendix 6A, the accident criticality analysis assumes that the fuel pins do not shear through, but the fuel pin spacer grids collapse, resulting in the fuel rods moving closer together. Therefore, reduced pin pitch is evaluated as the credible tipover accident configuration. The 7x7, 8x8, 9x9, and 10x10 lattices are modeled with pin pitch uniformly reduced. For the 7x7, 8x8, and 10x10 the baseline models above are used. For the 9x9, a similar baseline model is used, based on the most reactive 9x9 fuel in Table 6.4-1. The results, listed in Table 6.4-4, confirm that for all lattices, k_{eff} in the TN-68 basket decreases with decreasing pin pitch. Therefore, the hypothetical accident is bounded by the normal conditions evaluated above.

D. Package Array Evaluation

An infinite rectangular array of packages with optimum internal moderation and interspersed water is modeled to demonstrate criticality safety for both the normal and damaged package arrays.

The array evaluation uses the combined worst case contents configuration and optimum internal moderation from the single package normal conditions evaluation. Fuel assemblies are assumed to be intact, which is the most reactive condition as shown by the single package hypothetical accident evaluation. The same three fuel designs evaluated for the single package are evaluated for the array.

The input files from the single package evaluation are modified by changing the outer water cylinder to a water filled cuboid tangent to the outside of the packaging, and applying mirror neutron reflection to all faces. To find optimum interspersed moderation, the water density in the external cuboid (the space between the packages) is varied, while the internal moderator density is held constant at its optimum value.

The neutron shield and the impact limiters are neither modeled, nor considered in the spacing of packages in the array. The impact limiters in particular would have the effect of increasing the package spacing if they were included.

The results are presented in Table 6.4-5. As expected for a heavy-walled package, there is little difference between the single package and array results. Variation of the interspersed water density results in random variations in k_{eff}, with no discernable trend.

6.4.3 Criticality Results

The highest result for the single package, normal and accident conditions is $k_{eff} = 0.9240 \pm 0.0008$, which occurs for 10x10 fuel with 0.120 thick channels, intact fuel assemblies shifted toward package center, and 100% density water internal moderation.

The highest result for an infinite array of packages, normal and accident conditions is $k_{eff} = 0.9250 \pm 0.0008$, which occurs for 10x10 fuel with 0.120 thick channels, intact fuel assemblies shifted toward the package center, 100% density water internal moderation and 75% density water interspersed moderation.

6.5 Critical Benchmark Experiments

6.5.1 Benchmark Experiments and Applicability

The critical experiments and input files are taken from NUREG/CR-6361[5]. The input files are obtained from ORNL, and modified to change the cross section library to the SCALE 27 group library that is used in all the TN-68 criticality evaluations. Experiments which feature simple arrays, separator plates, steel reflector walls, water holes, and borated poison plates are selected. Experiments with features that are not characteristic of the TN-68 transport packaging are not used. Such features include soluble boron, poisons other than boron, poison rods, reflector walls other than steel, and flux traps. The 73 critical experiments chosen, and their descriptive characteristics are listed in Table 6.5-1.

6.5.2 Details of Benchmark Calculations

An upper subcritical limit (USL) is determined using Method 1, "confidence band with administrative margin", described in Section 4.1.1 of NUREG/CR-6361. The USLSTATS program, Version 1.3.4, distributed by Oak Ridge National Laboratory and described in NUREG/CR-6361 is used to perform the statistical analysis. The administrative margin is 0.05, and the confidence level $1-\gamma_1$ will be 0.95. It is assumed that the actual value of k_{eff} in all the experiments is exactly 1.

The characteristics water/fuel volume, hydrogen to fissile atom ratio (H/X), fuel pin pitch, and enrichment are listed in Tables 2.1 and 3.5 of NUREG/CR-6361. One additional characteristic, boron 10 concentration in the separator plates, is calculated in Table 6.5-2. A comparison of the range of these characteristics in the experiments, and the corresponding values for the TN-68 and

its contents verifies that the TN-68 falls within the range covered by the critical experiments. See Table 6.5-4.

Criticality calculations and benchmarks were performed on the same platform, a Hewlett Packard 9000/715 Workstation.

6.5.3 Results of Benchmark Calculations

The quantitative characteristics of the critical experiments and results of the benchmark calculations are listed in Table 6.5-3.

Eight subsets of the results are analyzed to determine if there is a trend in the bias (calculated k_{eff} -1) as a function of an experimental variable. In all subsets, the data test normal, although the sample size for the boron density is too small for this determination to be conclusive. A least mean squares linear regression is performed to fit the data of k_{eff} as a function of each independent variable, and the Pearson correlation coefficient r is determined. A coefficient of zero indicates no correlation, and a coefficient of |1| indicates exact correlation. The results are listed in Table 6.5-4. The values of the correlation coefficient, as well as a visual examination of the data plots, indicate that there is very little correlation between the bias and any of the experimental variables, and therefore, no discernable trend. The best correlation between bias and an experimental variable occurs for enrichment. The data and the linear regression results for the ratio of water volume to fuel volume in the pin cell are plotted in Figure 6.5-1.

The minimum value of the USL from all the data sets is 0.9373, which occurs for pin cell water to fuel pellet volume ratio, as shown in Table 6.5-4.

There is no modeling bias due to the package model as discussed in Section 6.3.1. The modeling bias due to the use of a uniform pin enrichment model of the lattice is -0.0042, as discussed in Section 6.4.2B. This bias is added to the USL determined from the benchmark evaluation:

$$0.9373 - 0.0042 = 0.9331$$

This is the USL value used in the TN-68 criticality analysis.

6.6 References

1. SCALE-4.3, Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers, CCC-545, ORNL, March 1997.

2. ORNL/TM-10902, Physical Characteristics of GE BWR Fuel Assemblies, 1989.

3. GE Proprietary Data
   a. Initial Core Fuel Design Summary for Peach Bottom Atomic Power Station Unit 3, NEDC-10816, March 1973
   b. Peach Bottom 2 Reload 1 Nuclear Design Report, NEDE 21051, September 1975
   c. Peach Bottom 3 Reload 2 Nuclear Design Report, NEDE 21759, December 1977
   d. Peach Bottom 2 Reload 3 Nuclear Design Report, NEDE 23896, July 1978
   e. Fuel Bundle Design Report GE8B-P8DQB321-11GZ-80M-4WR-150-T
   f. Fuel Bundle Design Report GE8B-P8DQB319-9GZ-80M-4WR-150-T
   g. Fuel Bundle Design Report GE9B-P8DWB310-11GZ-80M-150-T
   h. Fuel Bundle Design Report GE9B-P8DWB324-10GZ-80M-150-T
   i. Fuel Bundle Design Report GE9B-P8DWB320-10GZ-80M-150-T
   j. Fuel Bundle Design Report GE9B-P8DWB324-10GZ1-80M-150-T
   k. Fuel Bundle Design Report GE9B-P8DWB328-11GZ-80M-150-T
   l. Fuel Bundle Design Report GE11-P9HUB307-5G5.0/4G4.0-100M-146-T-LTA
   m. Fuel Bundle Design Report GE11-P9HUB334-10GZ1-100M-146-T
   n. Fuel Bundle Design Report GE11-P9HUB367-11GZ-100M-146-T
   o. Fuel Bundle Design Report GE11-P9HUB387-12GZ3-100T-146-T
   p. Fuel Bundle Design Report GE11-P9HUB405-13GZ1-100T-146-T
   q. Fuel Bundle Design Report GE13-P9DTB400-13GZ-100T-146-T
   r. Fuel Bundle Design Report GE13-P9DTB397-13GZ-100T-146-T
   s. Fuel Bundle Design Report GE13-P9DTB392-15GZ-100T-146-T
   t. Fuel Bundle Design Report GE13-P9DTB39-13GZ-100T-146-T
   u. Fuel Bundle Design Report 8DRB284-7G4.0-100M-150
   v. Fuel Bundle Design Report 8DRB285-4G2.0-100M-150
   w. Fuel Bundle Design Report 8DRB299-7G4.0-100M-150
   x. Fuel Bundle Design Report P8DRB299-3G5.0/4G4.0-100M-150

4. Power Authority of the State of New York, Inquiry No. Q-02-1961 (RFQ for FitzPatrick Dry Spent Fuel Storage System)

5. NUREG/CR-6361, Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages, 1997

6. NAC UMS Safety Analysis Report, Rev. 0, EA790-SAR-001, Docket No. 71-9270.

Appendix 6.7 Sample Computer Input Files

Sample Input File, Most Reactive Lattice Evaluation

```
=CSAS25
TN68  case 12wc, G12 fuel, wet annulus, 0.065 channel
27GROUPNDF4  LATTICECELL
UO2       1 0.965   293.  92235 3.7  92238 96.3  END
ZIRCALLOY   2  1.0  END
H2O       3  1.0  END
SS304      4  1.0  END
CARBONSTEEL  5  1.0  END
AL        6  DEN=2.668    END
B-10      6  DEN=0.025   END
H2O       7  1.0  END
H2O       8  0.5 END
AL        8  0.5 END
END COMP
SQUAREPITCH  1.2954  0.8763  1  3  1.0262  2  0.8941 7  END
TN68
READ PARAM RUN=yes PLT=yes TME=5000 GEN=203 NPG=1000 END PARAM
READ GEOM
UNIT  1   com='fuel rod'
CYLINDER  1  1  0.4382  2P15.456
CYLINDER  7  1  0.4470  2P15.456
CYLINDER  2  1  0.5131  2P15.456
CUBOID   3  1  4P 0.6477  2P15.456
UNIT  2   com='water rod'
CUBOID   3  1  4P 0.6477 2P15.456
UNIT 3   com='fuel compartment'
ARRAY  1   -6.4770   -6.4770   -15.456
CUBOID   7  1  4P6.703  2P15.456
CUBOID   2  1  4P6.868  2P15.456
CUBOID   7  1  4P7.62  2P15.456
CUBOID   4  1  4P8.095 2P15.456
CUBOID   6  1  4P8.491 2P15.456
HOLE  7  0  8.293 15.4559
HOLE  7  0  -8.293 15.4559
HOLE  8   8.293  0  -15.4559
HOLE  8  -8.293  0  -15.4559
HOLE  9   8.293  0   15.4559
HOLE  9  -8.293  0   15.4559
HOLE 10   7.4549    8.293    -2.3755
HOLE 10  -7.4549    8.293    -2.3755
HOLE 10   7.4549   -8.293    -2.3755
HOLE 10  -7.4549   -8.293    -2.3755
HOLE 11   8.293    7.4549    2.0695
HOLE 11  -8.293    7.4549    2.0695
HOLE 11   8.293   -7.4549    2.0695
HOLE 11  -8.293   -7.4549    2.0695
UNIT 4
ARRAY  3   -16.982   -8.491   -15.456
UNIT 5
ARRAY  4    -8.491   -16.982  -15.456
UNIT 6
ARRAY  5   -50.946   -8.491   -15.456
UNIT 7   com='stainless spacer and horizontal gap'
CUBOID  5  1 2P8.491 2P0.19799 0 -4.445
CUBOID  7  1 2P8.491 2P0.19799 0 -4.75
UNIT 8   com='stainless spacer'
CUBOID  5  1 2P0.19799 2P8.491 4.445  0
UNIT 9   com='horizontal gap'
CUBOID  7  1 2P0.19799 2P8.0949 0 -0.305
UNIT 10  com='vertical egg crate gap'
CUBOID  7  1  2P0.64  2P0.19799  2P13.08
UNIT 11  com='vertical egg crate gap'
CUBOID  7  1  2P0.19799  2P0.64  2P13.08
GLOBAL UNIT 12
```

```
ARRAY 2  -67.928  -50.946  -15.456
CYLINDER 8 1 88.26 2P15.4561
HOLE 6 0 59.4371   0
HOLE 6 0 -59.4371   0
HOLE 4 0 76.4192  0
HOLE 4 0 -76.4192  0
HOLE 5 76.4191 0 0
HOLE 5 -76.4191 0 0
CYLINDER  5 1 107.31 2P15.4561
CYLINDER  7 1 137.31 2P15.4561
CUBOID    0 1 4P137.31 2P15.4561
END GEOM
READ ARRAY
ARA=1 NUX=10 NUY=10 FILL F1 A34 2 2 A44 2 2 A56 2 2 A66 2 2 END FILL
ARA=2 NUX=8 NUY=6 FILL F3 END FILL
ARA=3 NUX=2 FILL F3 END FILL
ARA=4 NUY=2 FILL F3 END FILL
ARA=5 NUX=6 FILL F3 END FILL
END ARRAY
READ BNDS ZFC=PER END BNDS
READ PLOT
TTL='CLOSE UP top'
XUL=-1 YUL=18. ZUL=10.86 XLR=18. YLR=-1 ZLR=10.86
UAX=1 VDN=-1
NAX=1000 LPI=10.0 END
SCR=YES
END PLOT
END DATA
END
```

Sample Input File, Varied Enrichment, Complete Lattice

```
=CSAS25
TN68 case 11var2, G11 fuel, wet annulus, 0.100 channel, varied enrichment
27GROUPNDF4 LATTICECELL
UO2       1 0.965  293. 92235 4.076 92238 95.924 END
ZIRCALLOY  2 1.0 END
H2O       3 1.0 END
SS304      4 1.0 END
CARBONSTEEL 5 1.0 END
AL        6 DEN=2.668  END
B-10      6 DEN=0.025  END
H2O       7 1.0 END
H2O       8 0.5 END
AL        8 0.5 END
UO2       9 0.965  293. 92235 1.651 92238 98.349 END
UO2      10 0.965  293. 92235 2.476 92238 97.524 END
UO2      11 0.965  293. 92235 2.270 92238 97.730 END
UO2      12 0.965  293. 92235 3.302 92238 96.698 END
UO2      13 0.965  293. 92235 3.096 92238 96.904 END
UO2      14 0.965  293. 92235 3.921 92238 96.079 END
UO2      15 0.965  293. 92235 2.683 92238 97.317 END
UO2      16 0.965  293. 92235 2.889 92238 97.111 END
UO2      17 0.965  293. 92235 3.715 92238 96.285 END
UO2      18 0.965  293. 92235 3.508 92238 96.492 END
END COMP
SQUAREPITCH 1.4376 0.9550 1 3 1.1176 2 0.9754 7 END
MORE DATA res=9 cylinder 0.4775 dan(9)=0.24577753
      res=10 cylinder 0.4775 dan(10)=0.24577753
      res=11 cylinder 0.4775 dan(11)=0.24577753
      res=12 cylinder 0.4775 dan(12)=0.24577753
      res=13 cylinder 0.4775 dan(13)=0.24577753
      res=14 cylinder 0.4775 dan(14)=0.24577753
      res=15 cylinder 0.4775 dan(15)=0.24577753
      res=16 cylinder 0.4775 dan(16)=0.24577753
      res=17 cylinder 0.4775 dan(17)=0.24577753
      res=18 cylinder 0.4775 dan(18)=0.24577753 END MORE DATA
TN68
READ PARAM RUN=yes PLT=yes TME=5000 GEN=203 NPG=1000 END PARAM
```

```
READ GEOM
UNIT 1  com='fuel rod'
CYLINDER 1 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 2  com='water rod'
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 3  com='fuel compartment'
ARRAY 1  -6.4692  -6.4692 -15.456
CUBOID  7 1 4P6.703 2P15.456
CUBOID  2 1 4P6.957 2P15.456
CUBOID  7 1 4P7.62 2P15.456
CUBOID  4 1 4P8.095 2P15.456
CUBOID  6 1 4P8.491 2P15.456
HOLE 4 0 8.293 15.4559
HOLE 4 0 -8.293 15.4559
HOLE 5  8.293 0 -15.4559
HOLE 5 -8.293 0 -15.4559
HOLE 6  8.293 0  15.4559
HOLE 6 -8.293 0  15.4559
HOLE 7  7.4549   8.293   -2.3755
HOLE 7 -7.4549   8.293   -2.3755
HOLE 7  7.4549  -8.293   -2.3755
HOLE 7 -7.4549  -8.293   -2.3755
HOLE 8  8.293   7.4549   2.0695
HOLE 8 -8.293   7.4549   2.0695
HOLE 8  8.293  -7.4549   2.0695
HOLE 8 -8.293  -7.4549   2.0695
UNIT 4  com='stainless spacer and horizontal gap'
CUBOID 5 1 2P8.491 2P0.19799 0 -4.445
CUBOID 7 1 2P8.491 2P0.19799 0 -4.75
UNIT 5  com='stainless spacer'
CUBOID 5 1 2P0.19799 2P8.491 4.445 0
UNIT 6  com='horizontal gap'
CUBOID 7 1 2P0.19799 2P8.0949 0 -0.305
UNIT 7  com='vertical egg crate gap'
CUBOID 7 1 2P0.64 2P0.19799 2P13.08
UNIT 8  com='vertical egg crate gap'
CUBOID 7 1 2P0.19799 2P0.64 2P13.08
UNIT 9  com='fuel rod'
CYLINDER 9 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 10  com='fuel rod'
CYLINDER 10 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 11  com='fuel rod'
CYLINDER 11 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 12  com='fuel rod'
CYLINDER 12 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 13  com='fuel rod'
CYLINDER 13 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 14  com='fuel rod'
CYLINDER 14 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
```

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UNIT 15  com='fuel rod'
CYLINDER 15 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 16  com='fuel rod'
CYLINDER 16 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 17  com='fuel rod'
CYLINDER 17 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 18  com='fuel rod'
CYLINDER 18 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
GLOBAL UNIT 19
ARRAY 2 0 0  -15.456
CUBOID  8 1 73 0 73  0 2P15.456
CUBOID  5 1 92 0 92  0 2P15.456
CUBOID  7 1 122 0 122 0 2P15.456
END GEOM
READ ARRAY
ARA=1 NUX=9 NUY=9 FILL 16 6R1 18 15
8R1 12 7R1 17 12 4R1 2 2 12 1 14
3R1 3R2 1 1 13 3R1 2 2 1 1 17 12
3R1 12 3R1 16 11 18 1 17 1 1 17 16 16 10
15 12 12 14 13 12 11 10 9 END FILL
ARA=2 NUX=4 NUY=4 FILL F3 END FILL
END ARRAY
READ BNDS ZFC=PER -XY=MIR END BNDS
READ PLOT
TTL='CLOSE UP'
XUL=-1 YUL=20. ZUL=0 XLR=20. YLR=-1 ZLR=0
UAX=1 VDN=-1
NAX=1000 LPI=10.0 END
SCR=YES
END PLOT
END DATA
END
```

Sample Input File, Varied Enrichment, Vanished Lattice

```
=CSAS25
TN68 case 11van2, G11 fuel, wet annulus, 0.100 channel, vanish lattice
27GROUPNDF4  LATTICECELL
UO2      1 0.965  293. 92235 4.076 92238 95.924 END
ZIRCALLOY  2 1.0 END
H2O      3 1.0 END
SS304     4 1.0 END
CARBONSTEEL 5 1.0 END
AL       6 DEN=2.668  END
B-10     6 DEN=0.025  END
H2O      7 1.0 END
H2O      8 0.5 END
AL       8 0.5 END
UO2      9 0.965  293. 92235 1.651 92238 98.349 END
UO2      10 0.965  293. 92235 2.476 92238 97.524 END
UO2      11 0.965  293. 92235 2.270 92238 97.730 END
UO2      12 0.965  293. 92235 3.302 92238 96.698 END
UO2      13 0.965  293. 92235 3.096 92238 96.904 END
UO2      14 0.965  293. 92235 3.921 92238 96.079 END
UO2      15 0.965  293. 92235 2.683 92238 97.317 END
UO2      16 0.965  293. 92235 2.889 92238 97.111 END
UO2      17 0.965  293. 92235 3.715 92238 96.285 END
```

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UO2      18 0.965   293.  92235 3.508  92238 96.492  END
END COMP
SQUAREPITCH  1.4376  0.9550  1  3  1.1176  2  0.9754 7  END
MORE DATA res=9  cylinder 0.4775 dan(9)=0.24577753
      res=10 cylinder 0.4775 dan(10)=0.24577753
      res=11 cylinder 0.4775 dan(11)=0.24577753
      res=12 cylinder 0.4775 dan(12)=0.24577753
      res=13 cylinder 0.4775 dan(13)=0.24577753
      res=14 cylinder 0.4775 dan(14)=0.24577753
      res=15 cylinder 0.4775 dan(15)=0.24577753
      res=16 cylinder 0.4775 dan(16)=0.24577753
      res=17 cylinder 0.4775 dan(17)=0.24577753
      res=18 cylinder 0.4775 dan(18)=0.24577753 END MORE DATA
TN68
READ PARAM RUN=yes PLT=no TME=5000 GEN=203 NPG=1000 END PARAM
READ GEOM
UNIT  1   com='fuel rod'
CYLINDER  1  1  0.4775  2P15.456
CYLINDER  7  1  0.4877  2P15.456
CYLINDER  2  1  0.5588  2P15.456
CUBOID    3  1  4P 0.7188  2P15.456
UNIT  2   com='water rod'
CUBOID    3  1  4P 0.7188  2P15.456
UNIT 3   com='fuel compartment'
ARRAY  1   -6.4692   -6.4692  -15.456
CUBOID    7  1  4P6.703  2P15.456
CUBOID    2  1  4P6.957  2P15.456
CUBOID    7  1  4P7.62  2P15.456
CUBOID    4  1  4P8.095 2P15.456
CUBOID    6  1  4P8.491 2P15.456
HOLE  4  0  8.293 15.4559
HOLE  4  0  -8.293 15.4559
HOLE  5   8.293  0  -15.4559
HOLE  5  -8.293  0  -15.4559
HOLE  6   8.293  0   15.4559
HOLE  6  -8.293  0   15.4559
HOLE  7   7.4549    8.293    -2.3755
HOLE  7  -7.4549    8.293    -2.3755
HOLE  7   7.4549   -8.293    -2.3755
HOLE  7  -7.4549   -8.293    -2.3755
HOLE  8   8.293    7.4549    2.0695
HOLE  8  -8.293    7.4549    2.0695
HOLE  8   8.293   -7.4549    2.0695
HOLE  8  -8.293   -7.4549    2.0695
UNIT  4   com='stainless spacer and horizontal gap'
CUBOID  5  1 2P8.491 2P0.19799 0 -4.445
CUBOID  7  1 2P8.491 2P0.19799 0 -4.75
UNIT 5   com='stainless spacer'
CUBOID  5  1 2P0.19799 2P8.491 4.445  0
UNIT 6   com='horizontal gap'
CUBOID  7  1 2P0.19799 2P8.0949 0 -0.305
UNIT 7  com='vertical egg crate gap'
CUBOID  7  1  2P0.64  2P0.19799  2P13.08
UNIT 8  com='vertical egg crate gap'
CUBOID  7  1  2P0.19799  2P0.64  2P13.08
UNIT  9   com='fuel rod'
CYLINDER  9  1  0.4775  2P15.456
CYLINDER  7  1  0.4877  2P15.456
CYLINDER  2  1  0.5588  2P15.456
CUBOID    3  1  4P 0.7188  2P15.456
UNIT  10   com='fuel rod'
CYLINDER 10  1  0.4775  2P15.456
CYLINDER  7  1  0.4877  2P15.456
CYLINDER  2  1  0.5588  2P15.456
CUBOID    3  1  4P 0.7188  2P15.456
UNIT  11   com='fuel rod'
CYLINDER 11  1  0.4775  2P15.456
CYLINDER  7  1  0.4877  2P15.456
CYLINDER  2  1  0.5588  2P15.456
CUBOID    3  1  4P 0.7188  2P15.456
```

```
UNIT 12  com='fuel rod'
CYLINDER 12 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 13  com='fuel rod'
CYLINDER 13 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 14  com='fuel rod'
CYLINDER 14 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 15  com='fuel rod'
CYLINDER 15 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 16  com='fuel rod'
CYLINDER 16 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 17  com='fuel rod'
CYLINDER 17 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
UNIT 18  com='fuel rod'
CYLINDER 18 1 0.4775 2P15.456
CYLINDER 7 1 0.4877 2P15.456
CYLINDER 2 1 0.5588 2P15.456
CUBOID  3 1 4P 0.7188 2P15.456
GLOBAL UNIT 19
ARRAY 2 0 0  -15.456
CUBOID  8 1 73 0 73  0 2P15.456
CUBOID  5 1 92 0 92  0 2P15.456
CUBOID  7 1 122 0 122 0 2P15.456
END GEOM
READ ARRAY
ARA=1 NUX=9 NUY=9 FILL 16 6R1 18 15
1 2 1 1 2 1 1 2 12 7R1 17 12 4R1 2 2 12 1 14
1 2 1 3R2 1 2 13 3R1 2 2 1 1 17 12
3R1 12 3R1 16 11 18 2 17 1 2 17 16 2 10
15 12 12 14 13 12 11 10 9 END FILL
ARA=2 NUX=4 NUY=4 FILL F3 END FILL
END ARRAY
READ BNDS ZFC=PER -XY=MIR END BNDS
READ PLOT
TTL='CLOSE UP'
XUL=-1 YUL=20. ZUL=0 XLR=20. YLR=-1 ZLR=0
UAX=1 VDN=-1
NAX=1000 LPI=10.0 END
SCR=YES
END PLOT
END DATA
END
```

Sample Input File TN-68 Single Package Evaluation, Off Center Fuel with Optimum Moderation, 7x7 fuel with no channel, 6.00 compartment, borated aluminum

```
=CSAS25
TN68 case offctr7nude, GE7x7 fuel, wet annulus, no channel, off-center fuel
27GROUPNDF4 LATTICECELL
UO2      1 0.965  293. 92235 3.7 92238 96.3 END
```

```
ZIRCALLOY   2 1.0 END
H2O       3 1.0 END
SS304      4 1.0 END
CARBONSTEEL 5 1.0 END
AL        6 DEN=2.659   END
B-10       6 DEN=0.034   END
H2O        7 1.0 END
AL        8 1.0 END
END COMP
SQUAREPITCH 1.8745 1.237 1 3 1.43 2 1.267 7 END
TN68
READ PARAM RUN=yes PLT=no TME=5000 GEN=403 NPG=2000 END PARAM
READ GEOM
UNIT 1  com='fuel rod'
CYLINDER 1 1 0.6185 2P15.456
CYLINDER 7 1 0.6337 2P15.456
CYLINDER 2 1 0.7150 2P15.456
CUBOID   3 1 4P 0.9373 2P15.456
UNIT 2  com='water rod, not used'
CYLINDER 3 1 0.6337 2P15.456
CYLINDER 2 1 0.7239 2P15.456
CUBOID   3 1 4P 0.9373 2P15.456
UNIT 3  com='fuel compartment upper left quadrant'
ARRAY 1  -5.50221  -7.620  -15.456
CUBOID   7 1 4P7.62 2P15.456
CUBOID   4 1 4P8.095 2P15.456
CUBOID   6 1 4P8.491 2P15.456
HOLE 7 0 8.293 15.4559
HOLE 7 0 -8.293 15.4559
HOLE 8  8.293 0 -15.4559
HOLE 8 -8.293 0 -15.4559
HOLE 9  8.293 0  15.4559
HOLE 9 -8.293 0  15.4559
HOLE 10  7.4549    8.293    5.14
HOLE 10 -7.4549    8.293    5.14
HOLE 10  7.4549   -8.293    5.14
HOLE 10 -7.4549   -8.293    5.14
HOLE 11  8.293    7.4549   -5.445
HOLE 11 -8.293    7.4549   -5.445
HOLE 11  8.293   -7.4549   -5.445
HOLE 11 -8.293   -7.4549   -5.445
HOLE 12  7.4549    8.293   -13.155
HOLE 12 -7.4549    8.293   -13.155
HOLE 12  7.4549   -8.293   -13.155
HOLE 12 -7.4549   -8.293   -13.155
HOLE 13  8.293    7.4549    12.85
HOLE 13 -8.293    7.4549    12.85
HOLE 13  8.293   -7.4549    12.85
HOLE 13 -8.293   -7.4549    12.85
UNIT 4
ARRAY 3  -16.982  -8.491  -15.456
UNIT 5
ARRAY 4   -8.491  -16.982 -15.456
UNIT 6
ARRAY 5  -50.946  -8.491  -15.456
UNIT 7  com='stainless spacer and horizontal gap'
CUBOID 5 1 2P8.491 2P0.19799 0 -4.445
CUBOID 7 1 2P8.491 2P0.19799 0 -4.75
UNIT 8  com='stainless spacer'
CUBOID 5 1 2P0.19799 2P8.491 4.445 0
UNIT 9  com='horizontal gap'
CUBOID 7 1 2P0.19799 2P8.0949 0 -0.305
UNIT 10 com='vertical egg crate gap'
CUBOID 7 1 2P0.64 2P0.19799 2P5.565
UNIT 11 com='vertical egg crate gap'
CUBOID 7 1 2P0.19799 2P0.64 2P5.565
UNIT 12 com='vertical egg crate gap'
CUBOID 7 1 2P0.64 2P0.19799 2P2.3
UNIT 13 com='vertical egg crate gap'
CUBOID 7 1 2P0.19799 2P0.64 2P2.3
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UNIT 14   com='basket rail hole'
CYLINDER  7  1  5.5  2P15.4561
UNIT 15   com='basket rail hole'
CYLINDER  7  1  4.4  2P15.4561
UNIT 16   com='basket rail hole'
CYLINDER  7  1  3.0  2P15.4561
UNIT 17   com='fuel compartment upper right quadrant'
ARRAY  1  -7.620   -7.620   -15.456
CUBOID   7  1  4P7.62  2P15.456
CUBOID   4  1  4P8.095 2P15.456
CUBOID   6  1  4P8.491 2P15.456
HOLE  7  0  8.293 15.4559
HOLE  7  0  -8.293 15.4559
HOLE  8   8.293  0  -15.4559
HOLE  8  -8.293  0  -15.4559
HOLE  9   8.293  0   15.4559
HOLE  9  -8.293  0   15.4559
HOLE 10   7.4549    8.293    5.14
HOLE 10  -7.4549    8.293    5.14
HOLE 10   7.4549   -8.293    5.14
HOLE 10  -7.4549   -8.293    5.14
HOLE 11   8.293    7.4549   -5.445
HOLE 11  -8.293    7.4549   -5.445
HOLE 11   8.293   -7.4549   -5.445
HOLE 11  -8.293   -7.4549   -5.445
HOLE 12   7.4549    8.293   -13.155
HOLE 12  -7.4549    8.293   -13.155
HOLE 12   7.4549   -8.293   -13.155
HOLE 12  -7.4549   -8.293   -13.155
HOLE 13   8.293    7.4549    12.85
HOLE 13  -8.293    7.4549    12.85
HOLE 13   8.293   -7.4549    12.85
HOLE 13  -8.293   -7.4549    12.85
UNIT 18   com='fuel compartment lower right quadrant'
ARRAY  1     -7.620  -5.50221  -15.456
CUBOID   7  1  4P7.62  2P15.456
CUBOID   4  1  4P8.095 2P15.456
CUBOID   6  1  4P8.491 2P15.456
HOLE  7  0  8.293 15.4559
HOLE  7  0  -8.293 15.4559
HOLE  8   8.293  0  -15.4559
HOLE  8  -8.293  0  -15.4559
HOLE  9   8.293  0   15.4559
HOLE  9  -8.293  0   15.4559
HOLE 10   7.4549    8.293    5.14
HOLE 10  -7.4549    8.293    5.14
HOLE 10   7.4549   -8.293    5.14
HOLE 10  -7.4549   -8.293    5.14
HOLE 11   8.293    7.4549   -5.445
HOLE 11  -8.293    7.4549   -5.445
HOLE 11   8.293   -7.4549   -5.445
HOLE 11  -8.293   -7.4549   -5.445
HOLE 12   7.4549    8.293   -13.155
HOLE 12  -7.4549    8.293   -13.155
HOLE 12   7.4549   -8.293   -13.155
HOLE 12  -7.4549   -8.293   -13.155
HOLE 13   8.293    7.4549    12.85
HOLE 13  -8.293    7.4549    12.85
HOLE 13   8.293   -7.4549    12.85
HOLE 13  -8.293   -7.4549    12.85
UNIT 19   com='fuel compartment lower left quadrant'
ARRAY  1   -5.50221   -5.50221    -15.456
CUBOID   7  1  4P7.62  2P15.456
CUBOID   4  1  4P8.095 2P15.456
CUBOID   6  1  4P8.491 2P15.456
HOLE  7  0  8.293 15.4559
HOLE  7  0  -8.293 15.4559
HOLE  8   8.293  0  -15.4559
HOLE  8  -8.293  0  -15.4559
HOLE  9   8.293  0   15.4559
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HOLE 9 -8.293 0 15.4559
HOLE 10 7.4549 8.293 5.14
HOLE 10 -7.4549 8.293 5.14
HOLE 10 7.4549 -8.293 5.14
HOLE 10 -7.4549 -8.293 5.14
HOLE 11 8.293 7.4549 -5.445
HOLE 11 -8.293 7.4549 -5.445
HOLE 11 8.293 -7.4549 -5.445
HOLE 11 -8.293 -7.4549 -5.445
HOLE 12 7.4549 8.293 -13.155
HOLE 12 -7.4549 8.293 -13.155
HOLE 12 7.4549 -8.293 -13.155
HOLE 12 -7.4549 -8.293 -13.155
HOLE 13 8.293 7.4549 12.85
HOLE 13 -8.293 7.4549 12.85
HOLE 13 8.293 -7.4549 12.85
HOLE 13 -8.293 -7.4549 12.85
UNIT 20
ARRAY 6 -16.982 -8.491 -15.456
UNIT 21
ARRAY 7 -8.491 -16.982 -15.456
UNIT 22
ARRAY 8 -50.946 -8.491 -15.456
GLOBAL UNIT 23
ARRAY 2 -67.928 -50.946 -15.456
CYLINDER 8 1 87.96 2P15.4561
HOLE 6 0 59.4371 0
HOLE 22 0 -59.4371 0
HOLE 4 0 76.4192 0
HOLE 20 0 -76.4192 0
HOLE 5 76.4191 0 0
HOLE 21 -76.4191 0 0
HOLE 14 25.0 75.5 0
HOLE 14 75.5 25.0 0
HOLE 15 57.4 57.4 0
HOLE 16 72.8 40.0 0
HOLE 16 40.0 72.8 0
HOLE 14 -25.0 75.5 0
HOLE 14 -75.5 25.0 0
HOLE 15 -57.4 57.4 0
HOLE 16 -72.8 40.0 0
HOLE 16 -40.0 72.8 0
HOLE 14 25.0 -75.5 0
HOLE 14 75.5 -25.0 0
HOLE 15 57.4 -57.4 0
HOLE 16 72.8 -40.0 0
HOLE 16 40.0 -72.8 0
HOLE 14 -25.0 -75.5 0
HOLE 14 -75.5 -25.0 0
HOLE 15 -57.4 -57.4 0
HOLE 16 -72.8 -40.0 0
HOLE 16 -40.0 -72.8 0
CYLINDER 7 1 88.26 2P15.4561
CYLINDER 5 1 107.31 2P15.4561
CYLINDER 7 1 137.31 2P15.4561
CUBOID 0 1 4P137.31 2P15.4561
END GEOM
READ ARRAY
ARA=1 NUX=7 NUY=7 FILL F1 END FILL
ARA=2 NUX=8 NUY=6 FILL 4R19 4R18 2Q8 4R3 4R17 2Q8 END FILL
ARA=3 NUX=2 FILL 3 17 END FILL
ARA=4 NUY=2 FILL 18 17 END FILL
ARA=5 NUX=6 FILL 3R3 3R17 END FILL
ARA=6 NUX=2 FILL 19 18 END FILL
ARA=7 NUY=2 FILL 19 3 END FILL
ARA=8 NUX=6 FILL 3R19 3R18 END FILL
END ARRAY
READ BNDS ZFC=PER END BNDS
END DATA
END
```

Sample Input File TN-68 Infinite Array Evaluation, Combined Worst Configuration and Optimum Moderation, 10x10 fuel, 5.97 compartment, borated aluminum absorber

```
=csas25
TN68 case 10A75, GE 10x10, Infinite array
27GROUPNDF4  LATTICECELL
UO2        1 0.965   293.  92235 3.7  92238 96.3  END
ZIRCALLOY   2 1.0  END
H2O        3 1.0  END
SS304      4 1.0  END
CARBONSTEEL 5 1.0  END
AL        6 DEN=2.659   END
B-10       6 DEN=0.034   END
H2O        7 1.0  END
AL        8 1.0 END
H2O        9 0.75  END
END COMP
SQUAREPITCH 1.2954 0.8763 1 3 1.0262 2 0.8941 7 END
TN68
READ PARAM RUN=yes PLT=no TME=5000 GEN=403 NPG=2000 END PARAM
READ GEOM
UNIT 1  com='fuel rod'
CYLINDER 1 1 0.4382 2P15.456
CYLINDER 7 1 0.4470 2P15.456
CYLINDER 2 1 0.5131 2P15.456
CUBOID   3 1 4P 0.6477 2P15.456
UNIT 2  com='water rod'
CUBOID   3 1 4P 0.6477 2P15.456
UNIT 3  com='fuel compartment upper left quadrant'
ARRAY 1  -5.9028  -7.0512  -15.456
CUBOID   7 1 7.2772 -6.1288 6.1288 -7.2772  2P15.456
CUBOID   2 1 7.582 -6.4336 6.4336 -7.582  2P15.456
CUBOID   7 1 4P7.582  2P15.456
CUBOID   4 1 4P8.057 2P15.456
CUBOID   6 1 4P8.453 2P15.456
HOLE 7 0 8.255 15.4559
HOLE 7 0 -8.255 15.4559
HOLE 8  8.255 0 -15.4559
HOLE 8 -8.255 0 -15.4559
HOLE 9  8.255 0  15.4559
HOLE 9 -8.255 0  15.4559
HOLE 10  7.4169    8.255    5.14
HOLE 10 -7.4169    8.255    5.14
HOLE 10  7.4169   -8.255    5.14
HOLE 10 -7.4169   -8.255    5.14
HOLE 11  8.255    7.4169   -5.445
HOLE 11 -8.255    7.4169   -5.445
HOLE 11  8.255   -7.4169   -5.445
HOLE 11 -8.255   -7.4169   -5.445
HOLE 12  7.4169    8.255   -13.155
HOLE 12 -7.4169    8.255   -13.155
HOLE 12  7.4169   -8.255   -13.155
HOLE 12 -7.4169   -8.255   -13.155
HOLE 13  8.255    7.4169   12.85
HOLE 13 -8.255    7.4169   12.85
HOLE 13  8.255   -7.4169   12.85
HOLE 13 -8.255   -7.4169   12.85
UNIT 4
ARRAY 3  -16.906  -8.453  -15.456
UNIT 5
ARRAY 4   -8.453  -16.906 -15.456
UNIT 6
ARRAY 5  -50.718  -8.453  -15.456
UNIT 7  com='stainless spacer and horizontal gap'
CUBOID 5 1 2P8.453 2P0.19799 0 -4.445
CUBOID 7 1 2P8.453 2P0.19799 0 -4.75
UNIT 8  com='stainless spacer'
```

With metal matrix composite absorber, material 6 becomes

```
AL       6 0.875  END
B4C      6 0.09273  END
```

```
CUBOID  5  1 2P0.19799 2P8.453 4.445  0
UNIT 9   com='horizontal gap'
CUBOID  7  1 2P0.19799 2P8.0569 0 -0.305
UNIT 10  com='vertical egg crate gap'
CUBOID  7  1  2P0.64  2P0.19799  2P5.565
UNIT 11  com='vertical egg crate gap'
CUBOID  7  1  2P0.19799  2P0.64  2P5.565
UNIT 12  com='vertical egg crate gap'
CUBOID  7  1  2P0.64  2P0.19799  2P2.3
UNIT 13  com='vertical egg crate gap'
CUBOID  7  1  2P0.19799  2P0.64  2P2.3
UNIT 14   com='basket rail hole'
CYLINDER  7  1  5.5  2P15.4561
UNIT 15   com='basket rail hole'
CYLINDER  7  1  4.4  2P15.4561
UNIT 16   com='basket rail hole'
CYLINDER  7  1  3.0  2P15.4561
UNIT 17   com='fuel compartment upper right quadrant'
ARRAY  1  -7.0512   -7.0512   -15.456
CUBOID    7  1  6.1288 -7.2772 6.1288 -7.2772  2P15.456
CUBOID    2  1  6.4336 -7.582 6.4336 -7.582  2P15.456
CUBOID    7  1  4P7.582  2P15.456
CUBOID    4  1  4P8.057 2P15.456
CUBOID    6  1  4P8.453 2P15.456
HOLE  7  0  8.255 15.4559
HOLE  7  0  -8.255 15.4559
HOLE  8   8.255  0  -15.4559
HOLE  8  -8.255  0  -15.4559
HOLE  9   8.255  0   15.4559
HOLE  9  -8.255  0   15.4559
HOLE 10   7.4169    8.255    5.14
HOLE 10  -7.4169    8.255    5.14
HOLE 10   7.4169   -8.255    5.14
HOLE 10  -7.4169   -8.255    5.14
HOLE 11   8.255    7.4169   -5.445
HOLE 11  -8.255    7.4169   -5.445
HOLE 11   8.255   -7.4169   -5.445
HOLE 11  -8.255   -7.4169   -5.445
HOLE 12   7.4169    8.255   -13.155
HOLE 12  -7.4169    8.255   -13.155
HOLE 12   7.4169   -8.255   -13.155
HOLE 12  -7.4169   -8.255   -13.155
HOLE 13   8.255    7.4169    12.85
HOLE 13  -8.255    7.4169    12.85
HOLE 13   8.255   -7.4169    12.85
HOLE 13  -8.255   -7.4169    12.85
UNIT 18   com='fuel compartment lower right quadrant'
ARRAY  1    -7.0512  -5.9028  -15.456
CUBOID    7  1  6.1288  -7.2772 7.2772 -6.1288  2P15.456
CUBOID    2  1  6.4336 -7.582  7.582  -6.4336 2P15.456
CUBOID    7  1  4P7.582  2P15.456
CUBOID    4  1  4P8.057 2P15.456
CUBOID    6  1  4P8.453 2P15.456
HOLE  7  0  8.255 15.4559
HOLE  7  0  -8.255 15.4559
HOLE  8   8.255  0  -15.4559
HOLE  8  -8.255  0  -15.4559
HOLE  9   8.255  0   15.4559
HOLE  9  -8.255  0   15.4559
HOLE 10   7.4169    8.255    5.14
HOLE 10  -7.4169    8.255    5.14
HOLE 10   7.4169   -8.255    5.14
HOLE 10  -7.4169   -8.255    5.14
HOLE 11   8.255    7.4169   -5.445
HOLE 11  -8.255    7.4169   -5.445
HOLE 11   8.255   -7.4169   -5.445
HOLE 11  -8.255   -7.4169   -5.445
HOLE 12   7.4169    8.255   -13.155
HOLE 12  -7.4169    8.255   -13.155
HOLE 12   7.4169   -8.255   -13.155
```

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HOLE 12 -7.4169   -8.255   -13.155
HOLE 13  8.255    7.4169    12.85
HOLE 13 -8.255    7.4169    12.85
HOLE 13  8.255   -7.4169    12.85
HOLE 13 -8.255   -7.4169    12.85
UNIT 19   com='fuel compartment lower left quadrant'
ARRAY 1  -5.9028  -5.9028   -15.456
CUBOID   7 1 7.2772 -6.1288 7.2772 -6.1288  2P15.456
CUBOID   2 1 7.582 -6.4336 7.582 -6.4336  2P15.456
CUBOID   7 1 4P7.582  2P15.456
CUBOID   4 1 4P8.057 2P15.456
CUBOID   6 1 4P8.453 2P15.456
HOLE 7 0 8.255 15.4559
HOLE 7 0 -8.255 15.4559
HOLE 8  8.255 0 -15.4559
HOLE 8 -8.255 0 -15.4559
HOLE 9  8.255 0  15.4559
HOLE 9 -8.255 0  15.4559
HOLE 10  7.4169   8.255    5.14
HOLE 10 -7.4169   8.255    5.14
HOLE 10  7.4169  -8.255    5.14
HOLE 10 -7.4169  -8.255    5.14
HOLE 11  8.255    7.4169   -5.445
HOLE 11 -8.255    7.4169   -5.445
HOLE 11  8.255   -7.4169   -5.445
HOLE 11 -8.255   -7.4169   -5.445
HOLE 12  7.4169   8.255   -13.155
HOLE 12 -7.4169   8.255   -13.155
HOLE 12  7.4169  -8.255   -13.155
HOLE 12 -7.4169  -8.255   -13.155
HOLE 13  8.255    7.4169    12.85
HOLE 13 -8.255    7.4169    12.85
HOLE 13  8.255   -7.4169    12.85
HOLE 13 -8.255   -7.4169    12.85
UNIT 20
ARRAY 6  -16.906  -8.453  -15.456
UNIT 21
ARRAY 7  -8.453  -16.906  -15.456
UNIT 22
ARRAY 8  -50.718  -8.453  -15.456
GLOBAL UNIT 23
ARRAY 2  -67.624  -50.718  -15.456
CYLINDER 8 1 87.96 2P15.4561
HOLE 6  0 59.1711  0
HOLE 22 0 -59.1711  0
HOLE 4  0 76.0772  0
HOLE 20 0 -76.0772  0
HOLE 5  76.0771 0 0
HOLE 21 -76.0771 0 0
HOLE 14 25.0 75.5 0
HOLE 14 75.5 25.0 0
HOLE 15 57.4 57.4 0
HOLE 16 72.8 40.0 0
HOLE 16 40.0 72.8 0
HOLE 14 -25.0 75.5 0
HOLE 14 -75.5 25.0 0
HOLE 15 -57.4 57.4 0
HOLE 16 -72.8 40.0 0
HOLE 16 -40.0 72.8 0
HOLE 14 25.0 -75.5 0
HOLE 14 75.5 -25.0 0
HOLE 15 57.4 -57.4 0
HOLE 16 72.8 -40.0 0
HOLE 16 40.0 -72.8 0
HOLE 14 -25.0 -75.5 0
HOLE 14 -75.5 -25.0 0
HOLE 15 -57.4 -57.4 0
HOLE 16 -72.8 -40.0 0
HOLE 16 -40.0 -72.8 0
CYLINDER 7 1 88.26  2P15.4561
```

```
CYLINDER  5  1  107.31  2P15.4561
CUBOID    9  1  4P107.31  2P15.4561
END GEOM
READ ARRAY
ARA=1  NUX=10 NUY=10 FILL F1 A34 2 2 A44 2 2 A56 2 2 A66 2 2 END FILL
ARA=2  NUX=8 NUY=6  FILL 4R19 4R18 2Q8 4R3 4R17 2Q8  END FILL
ARA=3  NUX=2 FILL  3 17 END FILL
ARA=4  NUY=2 FILL  18 17 END FILL
ARA=5  NUX=6 FILL  3R3 3R17 END FILL
ARA=6  NUX=2 FILL  19 18 END FILL
ARA=7  NUY=2 FILL  19 3 END FILL
ARA=8  NUX=6 FILL  3R19 3R18 END FILL
END ARRAY
READ BNDS  ZFC=PER XYF=MIR END BNDS
END DATA
END
```

Appendix 6.8 Variable Enrichment Patterns, Uniform Enrichment Validation[2, 3]

GE 7x7, fuel model 3

Case: 3var

location	Qty	enrichment	normalized
1	30	2.930	4.331
11	12	1.940	2.868
10	6	1.690	2.498
9	1	1.330	1.966
	Total	average	average
	49	2.508	3.7

10	11	11	11	10	10	9
11	1	1	1	1	11	10
1	1	1	1	1	1	10
1	1	1	1	1	1	11
1	1	1	1	1	1	11
11	1	1	1	1	1	11
11	11	1	1	1	11	10

GE, 8x8, fuel model 4

Case: 4var1

location	Qty	enrichment	normalized
11	1	1.450	1.960
10	4	1.870	2.528
9	14	2.220	3.001
1	44	3.010	4.069
	Total	average	average
	63	2.737	3.7

10	9	9	9	9	9	10	11
9	1	1	1	1	1	9	10
1	1	1	1	1	1	1	9
1	1	1	1	1	1	1	9
1	1	1	w	1	1	1	9
1	1	1	1	1	1	1	9
1	1	1	1	1	1	1	9
9	1	1	1	1	1	9	10

Case: 4var2

location	Qty	enrichment	normalized
11	1	1.450	2.046
10	4	1.870	2.639
9	14	2.140	3.020
1	44	2.870	4.050
	Total	average	average
	63	2.622	3.7

Note: w= water rod

GE, 8x8, fuel model 5

Case: 5var1

location	Qty	enrichment	normalized
11	1	1.450	1.963
10	4	1.870	2.532
9	14	2.220	3.006
1	43	3.010	4.075
	Total	average	average
	62	2.733	3.7

10	9	9	9	9	9	10	11
9	1	1	1	1	1	9	10
1	1	1	1	1	1	1	9
1	1	1	w	1	1	1	9
1	1	1	1	w	1	1	9
1	1	1	1	1	1	1	9
1	1	1	1	1	1	1	9
9	1	1	1	1	1	9	10

Case: 5var2

location	Qty	enrichment	normalized
1	16	3.950	4.822
9	14	3.300	4.029
10	11	3.000	3.663
11	12	2.400	2.930
12	6	2.000	2.442
13	2	1.700	2.075
14	1	1.300	1.587
	Total	average	average
	62	3.031	3.7

12	11	11	11	12	12	13	14
11	9	10	9	10	11	11	13
9	10	1	1	9	10	11	12
9	1	1	w	1	9	10	12
9	1	1	1	w	1	9	11
9	10	1	1	1	1	10	11
10	1	10	1	1	10	9	11
11	10	9	9	9	9	11	12

Case: 5var3

location	Qty	enrichment	normalized
1	18	3.950	4.581
9	7	3.800	4.407
10	6	3.300	3.827
11	12	2.800	3.247
12	8	2.400	2.783
13	4	2.000	2.320
14	1	1.500	1.740
15	6	3.000	3.479
	total	average	average
	62	3.190	3.7

13	12	11	11	11	12	13	14
12	10	9	15	10	11	12	13
11	15	1	1	1	9	11	12
9	1	1	w	1	1	10	11
9	1	1	1	w	1	15	11
10	15	1	1	1	1	9	11
11	1	15	1	1	15	10	12
12	11	10	9	9	11	12	13

GE 8x8 with 4 water rods, fuel model 84

Case: 84var1

location	qty	enrichment	normalized
1	26	3.950	4.313
9	4	3.600	3.931
10	8	3.400	3.713
11	4	3.200	3.494
12	5	3.000	3.276
13	4	2.800	3.058
14	1	2.600	2.839
15	4	2.400	2.621
16	1	2.200	2.402
17	2	2.000	2.184
18	1	1.600	1.747
	total	average	average
	60	3.388	3.7

15	13	12	12	13	15	17	18
11	10	10	9	10	11	14	17
1	9	1	1	1	12	11	15
1	1	1	W	w	1	10	13
1	1	1	w	W	1	9	12
1	1	1	1	1	1	10	12
10	1	1	1	1	9	10	13
16	10	1	1	1	1	11	15

Case: 84var2

location	qty	enrichment	normalized
1	29	3.950	4.283
9	2	3.800	4.120
10	2	3.600	3.903
11	6	3.400	3.686
12	8	3.000	3.253
13	5	2.800	3.036
14	2	2.600	2.819
15	3	2.200	2.385
16	2	1.800	1.952
17	1	1.600	1.735
	total	average	average
	60	3.413	3.7

16	14	12	12	13	13	15	17
12	10	11	9	11	12	13	15
1	1	1	1	1	1	12	13
1	1	1	W	w	1	11	13
1	1	1	w	W	1	9	12
1	1	1	1	1	1	11	12
11	1	1	1	1	1	10	14
15	11	1	1	1	1	12	16

GE 8x8 with 1 large water rod, fuel model 9

Case: 9var1

location	qty	enrichment	normalized
9	1	1.800	1.815
10	2	2.800	2.824
11	5	3.000	3.025
12	7	3.400	3.429
13	2	3.600	3.630
14	2	2.400	2.420
1	41	3.950	3.983
	total	average	average
	60	3.669	3.7

14	11	12	13	12	11	10	9
12	1	1	1	1	1	11	10
1	1	1	1	1	1	1	11
1	1	1	w	w	1	1	12
1	1	1	w	w	1	1	13
1	1	1	1	1	1	1	12
1	1	1	1	1	1	1	11
12	1	1	1	1	1	12	14

Case: 9var2

location	qty	enrichment	normalized
9	1	1.600	1.650
10	2	2.200	2.269
11	4	2.600	2.682
12	2	3.200	3.300
13	7	3.400	3.507
14	2	3.000	3.094
15	1	2.400	2.475
16	4	3.600	3.713
1	35	3.950	4.074
17	2	3.800	3.919
	total	average	average
	60	3.588	3.7

11	14	13	13	12	11	10	9
13	1	1	1	16	16	15	10
1	1	1	1	1	1	16	11
1	1	1	w	w	1	16	12
1	1	1	w	w	1	1	13
1	1	1	1	1	1	1	13
17	1	1	1	1	1	1	14
13	17	1	1	1	1	13	11

Case: 9var3

location	qty	enrichment	normalized
9	1	1.600	1.696
10	2	2.600	2.756
11	2	3.200	3.393
12	16	3.400	3.605
13	4	3.000	3.181
14	5	2.400	2.544
1	28	3.950	4.188
15	2	3.600	3.817
	total	average	average
	60	3.490	3.7

14	13	12	12	12	11	10	9
12	1	13	1	1	14	12	10
1	12	1	1	1	1	14	11
1	1	15	w	w	1	1	12
1	12	1	w	w	1	1	12
1	1	1	1	15	1	13	12
12	12	1	12	1	12	1	13
14	12	1	1	1	1	12	14

GE 9x9 with two large water rods, model 11

Case: 11var1

location	qty	enrichment	normalized
9	1	1.800	1.690
10	5	2.200	2.065
11	7	3.000	2.816
12	8	3.600	3.380
13	4	3.200	3.004
14	19	3.950	3.708
1	23	4.800	4.506
15	7	4.400	4.131
	total	average	average
	74	3.941	3.7

10	11	12	12	12	12	11	10	9
13	14	14	14	1	14	13	11	10
15	1	14	14	15	15	14	13	11
1	14	1	w	w	15	15	14	12
1	1	1	w	w	w	15	1	12
1	14	1	1	w	w	14	14	12
14	1	1	1	1	1	14	14	12
11	1	1	14	1	14	1	14	11
10	11	14	1	1	1	15	13	10

Case: 11var2

location	qty	enrichment	normalized
9	1	1.600	1.651
10	2	2.400	2.476
11	2	2.200	2.270
12	8	3.200	3.302
13	2	3.000	3.096
14	2	3.800	3.921
15	2	2.600	2.683
16	4	2.800	2.889
17	4	3.600	3.715
18	2	3.400	3.508
1	45	3.950	4.076
	total	average	average
	74	3.586	3.7

15	12	12	14	13	12	11	10	9
18	1	17	1	1	17	16	16	10
1	1	1	12	1	1	1	16	11
1	1	1	w	w	1	1	17	12
1	1	1	w	w	w	1	1	13
1	1	1	1	w	w	12	1	14
1	1	1	1	1	1	1	17	12
1	1	1	1	1	1	1	1	12
16	1	1	1	1	1	1	18	15

Case: 11var3

location	qty	enrichment	normalized
9	1	1.600	1.734
10	4	2.200	2.384
11	8	2.600	2.817
12	4	3.200	3.467
13	8	3.000	3.250
14	4	3.400	3.684
15	19	3.600	3.901
1	26	3.950	4.280
	total	average	average
	74	3.415	3.7

11	13	13	13	12	11	10	10	9
15	15	13	15	15	14	11	11	10
1	15	15	12	1	15	15	11	10
1	15	1	w	w	1	15	14	11
1	1	1	w	w	w	1	15	12
1	15	1	1	w	w	12	15	13
1	1	1	1	1	1	15	13	13
14	1	1	15	1	15	15	15	13
11	14	1	1	1	1	1	15	11

GE 9x9 with two large water rods, model 11

Case: 13var1

location	qty	enrichment	normalized
9	1	1.600	1.408
10	4	2.400	2.112
11	2	3.200	2.816
12	12	3.950	3.476
13	4	2.800	2.464
14	4	3.600	3.168
1	29	4.900	4.313
15	14	4.400	3.872
16	4	4.200	3.696
	total	average	average
	74	4.204	3.7

10	13	12	12	12	12	11	10	9
14	12	15	16	1	16	14	13	10
15	1	15	1	15	1	15	14	11
1	15	1	w	w	1	1	16	12
1	1	1	w	w	w	15	1	12
1	15	1	15	w	w	1	16	12
1	1	1	1	1	1	15	15	12
12	1	1	15	1	15	1	12	13
13	12	1	1	1	1	15	14	10

Case: 13var2

location	qty	enrichment	normalized
9	1	2.000	1.736
10	7	2.800	2.430
11	3	3.200	2.777
12	11	3.950	3.428
13	4	3.600	3.125
1	32	4.900	4.253
14	12	4.400	3.819
15	4	4.200	3.645
	total	average	average
	74	4.263	3.7

10	10	12	12	12	12	11	10	9
13	14	14	15	1	15	13	10	10
14	1	14	1	14	1	12	13	11
1	14	1	w	w	1	1	15	12
1	1	1	w	w	w	14	1	12
1	1	1	1	w	w	1	15	12
1	1	1	1	1	1	14	14	12
12	1	1	1	1	14	1	14	10
11	12	1	1	1	1	14	13	10

Case: 13var3

location	qty	enrichment	normalized
9	1	2.000	1.724
10	2	2.800	2.414
11	6	3.200	2.759
12	17	3.950	3.406
13	2	2.400	2.069
1	36	4.900	4.225
14	8	4.400	3.794
15	2	3.600	3.104
	total	average	average
	74	4.291	3.7

13	11	12	12	12	12	11	10	9
15	1	14	12	1	12	12	11	10
14	1	12	1	1	1	12	12	11
1	14	1	w	w	1	1	12	12
1	1	1	w	w	w	1	1	12
1	1	1	1	w	w	1	12	12
1	1	1	1	1	1	12	14	12
14	1	1	1	1	14	1	1	11
11	14	1	1	1	1	14	15	13

Table 6.2-1
Fuel Characteristics for Criticality[Refs 2, 3, 4]

GE fuel generation	model	array	rod pitch	fuel rods	rod od	clad thick	pellet dia	water rods	water rod od	water rod id	
2A	2a	7x7	0.738	49	0.570	0.036	0.488	0	x	x	
2, 2B	2	7x7	0.738	49	0.563	0.032	0.487	0	x	x	
3, 3A, 3B	3	7x7	0.738	49	0.563	0.037	0.477	0	x	x	
4, 4A, 4B	4	8x8	0.640	63	0.493	0.034	0.416	1	0.493	0.425	
5	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	
6, 6B[5]	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	
7, 7B[5]	5	8x8	0.640	62	0.483	0.032	0.410	2	0.591	0.531	
8, 8B - 2w	82	8x8	0.640	62	0.483	0.032	0.411	2	0.591	0.531	
8, 8B - 4w	84	8x8	0.640	60	0.483	0.032	0.411	4	0.591	0.531	2 large water rods
									0.483	0.431	2 small water rods
9, 9B	9	8x8	0.640	60	0.483	0.032	0.411	1	1.34	1.26	
10	9	8x8	0.640	60	0.483	0.032	0.411	1	1.34	1.26	
11	11	9x9	0.566	74	0.440	0.028	0.376	2	0.98	0.92	
13	11	9x9	0.566	74	0.440	0.028	0.376	2	0.98	0.92	
12	12	10x10	0.510	92	0.404	0.026	0.345	2	0.98	0.92	

Notes:
1. All dimensions in inches
2. All fuel channels 5.278 inches inside, and from 0.065 to 0.120 inch thick.
3. All fuels are evaluated with 96.5% theoretical density and 3.7 wt % U235 average enrichment.
4. The fuel rod pitch is for C and D lattice designs. The some S lattice fuels have a smaller pitch, which is less reactive as shown in Table 6.4-4.
5. The fuel designs designated by GE as 6, 6B, 7, and 7B are sometimes referred to as "P" (pressurized) and "B" (barrier).

Table 6.3-1
Comparison of Design Dimensions with Criticality Model

dimensions: inch (cm)

Description	TN-68 Design	Model
compartment inside	5.97 min	5.97 (15.164) 6.00 (15.240) 6.05 (15.368)
compartment wall (ASME SA-480, Table A1.3)	0.1874 ± 0.014	0.187 (0.475)
stainless steel strip ht	1.75	1.75 (4.445)
stainless steel strip thickness (ASME SA-480, Table A1.18)	0.3125 +0.045/-0.010	0.312 (0.792)
poison plates height	10.4	10.3 (26.16)
poison plates thickness	0.300 - 0.310	0.312 (0.792)
horizontal (thermal expansion) gap	0.03 – 0.09	0.12 (0.305)
vertical (egg-crate) slot width	1.25	1.32 (3.352)
vertical (egg-crate) slot height	1.81 + 4.38	1.81 + 4.38 (4.6 + 11.13)
cavity inside radius	34.75	34.75 (88.26)
packaging wall thickness	7.5	7.5 (19.05)

Table 6.3-2
Model Mass Densities

Material	Component *	Density g/cm^3	Notes
Fuel	UO2	10.576	1
Cladding and fuel channel	ZIRCALLOY	6.56	
Water	H2O	0.9982	
Borated Aluminum	AL	2.659	
	B-10	0.034	2
Boron carbide / Aluminum	AL	2.364	3
	C	0.051	
	B-10	0.034	
	B-11	0.150	
Compartment tubes, spacers	SS304	7.92	
Basket rails	AL	2.702	
Containment and gamma shield	CARBONSTEEL	7.9	

* Component identifiers are from the SCALE Standard Composition Library.

Notes:

1. Fuel is modeled as 96.5% of theoretical density: 0.965(10.96) = 10.576 g/cm^3. U234 and U236 are ignored. Enrichment is 3.7 wt % U235 for all calculations except the evaluation of varying pin-by-pin enrichment.
2. B11 is ignored because enriched boron will be used.
3. This is an alternate neutron poison plate material.

Table 6.4-1
Results, Most Reactive Lattice Evaluation

array	GE fuel generation	water in rods	channel	k_{eff}	σ
7x7	2a	no	none	0.9087	0.0017
		yes	none	0.9135	0.0015
		yes	0.065	0.9190	0.0016
		yes	0.080	0.9213	0.0017
		yes	0.120	0.9213	0.0017
7x7	2, 2b	no	none	0.9136	0.0016
		yes	none	0.9199	0.0015
		yes	0.065	0.9207	0.0016
		yes	0.080	0.9219	0.0016
		yes	0.120	0.9229	0.0015
7x7	3, 3A, 3B	no	none	0.9093	0.0018
		yes	none	0.9190	0.0016
		yes	0.065	0.9150	0.0017
		yes	0.080	0.9167	0.0016
		yes	0.120	0.9208	0.0016
8x8	4, 4a, 4b	no	none	0.9032	0.0017
		yes	none	0.9104	0.0016
		yes	0.065	0.9114	0.0015
		yes	0.100	0.9156	0.0015
		yes	0.120	0.9141	0.0015
8x8	5, B, P	no	none	0.9131	0.0016
		yes	none	0.9152	0.0018
		yes	0.065	0.9190	0.0015
		yes	0.100	0.9180	0.0016
		yes	0.120	0.9200	0.0016
8x8	8, 8b, 2 water rods	no	none	0.9156	0.0016
		yes	none	0.9137	0.0016
		yes	0.065	0.9183	0.0017
		yes	0.100	0.9226	0.0017
		yes	0.120	0.9199	0.0016
8x8	8, 8b, 4 water rods	no	none	0.9145	0.0016
		yes	none	0.9145	0.0015
		yes	0.065	0.9171	0.0016
		yes	0.100	0.9197	0.0016
		yes	0.120	0.9218	0.0016

Table 6.4-1, continued
Results, Most Reactive Lattice Evaluation

array	GE fuel generation	water in rods	channel	k_{eff}	σ
8x8	9, 9b, 10	no	none	0.9149	0.0017
		yes	none	0.9159	0.0015
		yes	0.065	0.9212	0.0017
		yes	0.100	0.9220	0.0015
		yes	0.120	0.9235	0.0016
9x9	11, 13	no	none	0.9114	0.0017
		yes	none	0.9175	0.0017
		yes	0.065	0.9204	0.0015
		yes	0.100	0.9212	0.0016
		yes	0.120	0.9223	0.0016
10x10	12	no	none	0.9183	0.0014
		yes	none	0.9226	0.0016
		yes	0.065	0.9270	0.0015
		yes	0.100	0.9250	0.0015
		yes	0.120	0.9268	0.0016

Table 6.4-2
Results, Uniform Enrichment Model Validation

Uniform 3.7% enrichment			Varied enrichment			
Case	**k_{eff}**	**σ**	**Case**	**k_{eff}**	**σ**	**Δ k_{eff}**
7x7 GE3, 0.080 channel	0.9107	0.0017	3var	0.9054	0.0017	0.0053
8x8 GE4, 0.100 channel	0.9152	0.0018	4var1	0.9094	0.0016	0.0058
			4var2	0.9073	0.0016	0.0079
8x8 GE5, 0.100 channel	0.923	0.0016	5var1	0.9122	0.0016	0.0108
			5var2	0.9148	0.0016	0.0082
			5var3	0.9115	0.0017	0.0115
8x8 GE8, 0.100 channel 4 water rods	0.9179	0.0016	84var1	0.9171	0.0016	0.0008
			84var2	0.9149	0.0013	0.0030
8x8 GE9, 0.100 channel	0.9206	0.0015	9var1	0.9189	0.0016	0.0017
			9var2	0.9214	0.0017	-0.0008
			9var3	0.9201	0.0015	0.0005
9x9 GE11, 0.100 channel	0.922	0.0016	11var1	0.9174	0.0016	0.0046
			11var2	0.9196	0.0016	0.0024
			11var3	0.9218	0.0016	0.0002
			13var1	0.9180	0.0016	0.0040
			13var2	0.9208	0.0016	0.0012
			13var3	0.9198	0.0015	0.0022
			11van1(1)	0.9184	0.0015	0.0036
			11van2	0.9211	0.0015	0.0009
			11van3	0.9252	0.0017	-0.0032
			13van1	0.9218	0.0016	0.0002
			13van2	0.9209	0.0015	0.0011
			13van3	0.9204	0.0016	0.0016
				average Δ k_{eff}(2)		0.0032
				std deviation		0.0037

Note:

1. The last six cases are vanished lattices. They are the same as the six immediately preceding cases, except that the partial length fuel rods have vanished and are replaced by water.
2. The uniform enrichment modeling bias = 0.0032 - 2(0.0037) = -0.0042

Table 6.4-3
Results, TN-68 Single Package Normal Criticality Evaluation

10x10 GE12

Case description	k_{eff}	σ	$k_{eff}+2\sigma$
centered 10x10 fuel, 0.065 channel, 6.05 inch cell	0.9109	0.0015	0.9139
centered 10x10 fuel, 0.065 channel, 6 inch cell	0.9151	0.0016	0.9183
Optimum internal moderation with 6 inch cells, 0.065 channels:			
water density 0.01 g/cm^3	0.4004	0.0006	0.4016
water density 0.25	0.5901	0.0012	0.5925
water density 0.50	0.7521	0.0014	0.7549
water density 0.75	0.8530	0.0015	0.8560
water density 0.96	0.9078	0.0015	0.9108
water density 0.98	0.9092	0.0015	0.9122
water density 1.00	0.9136	0.0016	0.9168
Investigation of offcenter fuel with 6 inch cells:			
offcenter 10x10 fuel, no channel, 6 inch cell	0.9109	0.0015	0.9139
offcenter 10x10 fuel, 0.065 channel, 6 inch cell	0.9189	0.0016	0.9221
offcenter 10x10 fuel, 0.100 channel, 6 inch cell	0.9205	0.0015	0.9235
offcenter 10x10 fuel, 0.120 channel, 6 inch cell	0.9221	0.0015	0.9251
Investigation of offcenter fuel with 5.97 inch cells:			
offcenter 10x10 fuel, 0.120 channel, 5.97 inch cell	0.9233	0.0008	0.9249
offcenter 10x10 fuel, 0.120 channel, 5.97 inch cell metal matrix composite absorber	0.9240	0.0008	0.9256

Notes:

1. All cases are evaluated with water in the fuel pellet-cladding annulus of all pins and inside the package. Water density is 100% (0.9982 g/cm^3) except where noted.
2. Upper Subcritical Limit = 0.9331

Table 6.4-3, continued
Results, TN-68 Single Package Normal Criticality Evaluation

7x7 GE 2, 2b

Case description	k_{eff}	σ	$k_{eff}+2\sigma$
centered 7x7 fuel, 0.120 channel, 6.05 inch cell	0.9079	0.0008	0.9095
centered 7x7 fuel, 0.120 channel, 6 inch cell	0.9114	0.0008	0.9130
Optimum internal moderation with 6 inch cells, 0.120 channels			
water density 0.01 g/cm^3	0.4109	0.0005	0.4119
water density 0.25	0.5866	0.0011	0.5888
water density 0.50	0.7430	0.0013	0.7456
water density 0.75	0.8435	0.0016	0.8467
water density 0.96	0.9031	0.0016	0.9063
water density 0.98	0.9050	0.0016	0.9082
water density 1.00	0.9109	0.0016	0.9141
Investigation of offcenter fuel with 6 inch cells:			
offcenter 7x7 fuel, no channel, 6 inch cell	0.9102	0.0008	0.9118
offcenter 7x7 fuel, 0.065 channel, 6 inch cell	0.9141	0.0008	0.9157
offcenter 7x7 fuel, 0.100 channel, 6 inch cell	0.9165	0.0008	0.9181
offcenter 7x7 fuel, 0.120 channel, 6 inch cell	0.9165	0.0007	0.9179
Investigation of offcenter fuel with 5.97 inch cells:			
offcenter 7x7 fuel, 0.120 channel, 5.97 inch cell	0.9200	0.0008	0.9216
offcenter 7x7 fuel, 0.120 channel, 5.97 inch cell, metal matrix composite absorber	0.9199	0.0008	0.9215

Table 6.4-3, continued
Results, TN-68 Single Package Normal Criticality Evaluation

8x8 GE 9, 9b, 10

Case description	k_{eff}	σ	$k_{eff}+2\sigma$
centered 8x8 fuel, 0.120 channel, 6.05 inch cell	0.9073	0.0016	0.9105
centered 8x8 fuel, 0.120 channel, 6 inch cell	0.9103	0.0015	0.9133
Optimum internal moderation with 6 inch cells, 0.120 channels:			
water density 0.01 g/cm^3	0.3844	0.0005	0.3854
water density 0.25	0.5819	0.0011	0.5841
water density 0.50	0.7470	0.0014	0.7498
water density 0.75	0.8510	0.0015	0.8540
water density 0.96	0.9069	0.0013	0.9095
water density 0.98	0.9062	0.0017	0.9096
water density 1.00	0.9111	0.0015	0.9141
Investigation of offcenter fuel with 6 inch cells:			
offcenter 8x8 fuel, no channel, 6 inch cell	0.9074	0.0008	0.9090
offcenter 8x8 fuel, 0.065 channel, 6 inch cell	0.9154	0.0008	0.9170
offcenter 8x8 fuel, 0.100 channel, 6 inch cell	0.9158	0.0008	0.9174
offcenter 8x8 fuel, 0.120 channel, 6 inch cell	0.9159	0.0008	0.9175
Investigation of offcenter fuel with 5.97 inch cells:			
offcenter 8x8 fuel, 0.120 channel, 5.97 inch cell	0.9212	0.0007	0.9226
offcenter 8x8 fuel, 0.120 channel, 5.97 inch cell metal matrix composite absorber	0.9178	0.0008	0.9194

Table 6.4-4
Single Package Accident Evaluation – Collapsed Fuel Assemblies

array	Pitch cm	channel	k_{eff}	σ
10x10	1.2954	0.120	0.9151	0.0016
	1.25		0.8863	0.0016
	1.15		0.8137	0.0015
7x7	1.8745	0.120	0.9114	0.0008
	1.82		0.8888	0.0016
	1.77		0.8624	0.0016
8x8	1.6256	0.120	0.9103	0.0015
	1.58		0.8916	0.0017
	1.53		0.8682	0.0016
9x9	1.4736	0.120	0.9105	0.0017
	1.38		0.8792	0.0015
	1.33		0.8546	0.0016

Table 6.4-5
TN-68 Package Array Accident Evaluation, Intact Fuel

10x10 GE12

Case description	k_{eff}	σ	$k_{eff}+2\sigma$
Infinite array, 100% density interspersed water	0.9247	0.0008	0.9263
Infinite array, 75% density interspersed water	0.9250	0.0008	0.9266
Infinite array, 50% density interspersed water	0.9242	0.0008	0.9258
Infinite array, 25% density interspersed water	0.9228	0.0007	0.9242
Infinite array, 01% density interspersed water	0.9238	0.0008	0.9254

8x8 GE 9, 9b, 10

Case description	k_{eff}	σ	$k_{eff}+2\sigma$
Infinite array, 100% density interspersed water	0.9180	0.0009	0.9198
Infinite array, 75% density interspersed water	0.9193	0.0008	0.9209
Infinite array, 50% density interspersed water	0.9182	0.0008	0.9198
Infinite array, 25% density interspersed water	0.9198	0.0008	0.9214
Infinite array, 01% density interspersed water	0.9172	0.0008	0.9188

7x7 GE 2, 2b

Case description	k_{eff}	σ	$k_{eff}+2\sigma$
Infinite array, 100% density interspersed water	0.9207	0.0008	0.9223
Infinite array, 75% density interspersed water	0.9209	0.0007	0.9223
Infinite array, 50% density interspersed water	0.9174	0.0008	0.9190
Infinite array, 25% density interspersed water	0.9176	0.0008	0.9192
Infinite array, 01% density interspersed water	0.9210	0.0008	0.9226

Table 6.5-1
Characteristics of Critical Experiments from NUREG/CR-6361

case	file	assy pitch	pin lattice	steel wall	borated plate	separator plate	water holes
1	ANS33AL1	*	*			Al	
2	ANS33AL2	*	*			Al	
3	ANS33AL3	*	*			Al	
8	ANS33SLG	*	*				
24	BW1484SL	*	*				
48	EPRU65		*				
50	EPRU75		*				
52	EPRU87		*				
54	NSE71H1		*				
55	NSE71H2		*				
56	NSE71H3		*				
57	NSE71SQ		*				
58	NSE71W1		*				*
59	NSE71W2		*				*
60	P2438AL	*	*			Al	
61	P2438BA	*	*		*	Boral	
63	P2438SLG	*	*				
64	P2438SS	*	*			SS	
66	P2615AL	*	*			Al	
67	P2615BA	*	*		*	Boral	
71	P2615SS	*	*			SS	
77	P2827SLG	*	*				
82	P3314AL	*	*			Al	
83	P3314BA	*	*		*	Boral	
84	P3314BC	*	*		*	Boral	
85	P3314BF1	*	*		*	B-flex	
86	P3314BF2	*	*		*	B-flex	
87	P3314BS1	*	*		*	BSS	
88	P3314BS2	*	*		*	BSS	
89	P3314BS3	*	*		*	BSS	
90	P3314BS4	*	*		*	BSS	
99	P3314SLG	*	*				
100	P3314SS1	*	*			SS	
101	P3314SS2	*	*			SS	
102	P3314SS3	*	*			SS	
103	P3314SS4	*	*			SS	
104	P3314SS5	*	*			SS	
105	P3314SS6	*	*			SS	
106	P3314W1	*	*				*
107	P3314W2	*	*				*
109	P3602BA	*	*	*	*	Boral	
110	P3602BS1	*	*	*	*	BSS	
111	P3602BS2	*	*	*	*	BSS	
118	P3602N11	*	*	*			

Table 6.5-1, continued

case	file	assy pitch	pin lattice	steel wall	borated plate	separator plate	water holes
119	P3602N12	*	*	*			
120	P3602N13	*	*	*			
121	P3602N14	*	*	*			
122	P3602N21	*	*	*			
123	P3602N22	*	*	*			
124	P3602N31	*	*	*			
125	P3602N32	*	*	*			
126	P3602N33	*	*	*			
127	P3602N34	*	*	*			
128	P3602N35	*	*	*			
129	P3602N36	*	*	*			
130	P3602N41	*	*	*			
131	P3602N42	*	*	*			
132	P3602N43	*	*	*			
133	P3602SS1	*	*	*		SS	
134	P3602SS2	*	*	*		SS	
141	P3926SL1	*	*				
142	P3926SL2	*	*				
156	P49-194	*	*				
162	PAT80L1	*	*		*	Boral	
163	PAT80L2	*	*		*	Boral	
164	PAT80SS1	*	*	*	*	Boral	
165	PAT80SS2	*	*	*	*	Boral	
171	W3269SL1		*				
172	W3269SL2		*				
173	W3269W1		*				*
174	W3269W2		*				*
175	W3385SL1		*				
176	W3385SL2		*				
	Total	58	73	22	17	34	6

Note:

1. The case numbers from Table 2.1 of NUREG/CR-6361 are used throughout to refer to the individual experiments.

Table 6.5-2
Critical Experiments, Boron Plate Areal Density

case	file	material	core thickness (cm)	core density (g/cm^3)	weight % boron in core	boron 10 g/cm^2
61	P2438BA	Boral	0.509	2.49	28.7	0.0670
67	P2615BA	Boral	0.509	2.49	28.7	0.0670
83	P3314BA	Boral	0.509	2.49	28.7	0.0670
84	P3314BC	Boral	0.181	2.47	31.88	0.0263
85	P3314BF1	Boraflex	0.226	1.731	32.74	0.0236
86	P3314BF2	Boraflex	0.452	1.731	32.74	0.0472
87	P3314BS1	borated ss	0.298	7.9	1.05	0.0046
88	P3314BS2	borated ss	0.298	7.77	1.62	0.0069
89	P3314BS3	borated ss	0.298	7.9	1.05	0.0046
90	P3314BS4	borated ss	0.298	7.77	1.62	0.0069
109	P3602BA	Boral	0.292	2.5	30.36	0.0408
162	PAT80L1	Boral	0.43	2.6189	22.2	0.0461
163	PAT80L2	Boral	0.43	2.6189	22.2	0.0461
164	PAT80SS1	Boral	0.43	2.6189	22.2	0.0461
165	PAT80SS2	Boral	0.43	2.6189	22.2	0.0461

Notes:
1. Boron 10 is assumed to be 18.431 weight % of natural boron.
2. "Core" refers to the borated part of the plate. For Boral, this does not include the aluminum cladding, and for Boraflex, this does not include the plexiglas plates.

Table 6.5-3
Critical Experiment Results with CSAS25 and 27 Group Library

case	enrich	pin pitch (cm)	water/ fuel volume	H/X	AEF (eV)	boron 10 (g/cm**2)	assy spacing (cm)	second assy spacing	k_{eff}	σ
1	4.74	1.35	2.302	138.4	0.1860		5		1.0024	0.0015
2	4.74	1.35	2.302	138.4	0.1602		2.5		1.0066	0.0016
3	4.74	1.35	2.302	138.4	0.1410		10		1.0007	0.0015
8	4.74	1.35	2.302	138.4	0.1595		5		0.9940	0.0016
24	2.46	1.636	1.841	216.1	0.1094		6.54		0.9891	0.0013
48	2.35	1.905	1.196	163.6	0.2057				0.9880	0.0013
50	2.35	1.905	2.408	329.4	0.0878				0.9951	0.0014
52	2.35	2.21	3.687	504.2	0.0633				0.9961	0.0014
54	4.74	1.35	1.804	108.3	0.2242				0.9923	0.0016
55	4.74	1.26	3.811	228.8	0.0883				0.9984	0.0016
56	4.74	2.26	7.608	456.8	0.0550				0.9994	0.0015
57	4.74	1.26	1.823	110	0.2027				0.9954	0.0016
58	4.74	1.26	1.823	110	0.1824				0.9956	0.0017
59	4.74	1.26	1.823	110	0.1565				0.9961	0.0015
60	2.35	2.032	2.918	398.7	0.0733		8.67		0.9938	0.0013
61	2.35	2.032	2.918	398.7	0.0755	0.067	5.05		0.9945	0.0013
63	2.35	2.032	2.918	398.7	0.0730		8.39		0.9924	0.0014
64	2.35	2.032	2.918	398.7	0.0735		6.88		0.9925	0.0014
66	4.31	2.54	3.883	256.1	0.0927		10.72		0.9968	0.0016
67	4.31	2.54	3.883	256.1	0.0944	0.067	6.72		0.9939	0.0015
71	4.31	2.54	3.883	256.1	0.0931		8.58		0.9957	0.0015
77	2.35	2.032	2.918	398.7	0.0726		8.31		0.9913	0.0013
82	4.31	1.892	1.6	105.4	0.1977		9.04	2.83	0.9911	0.0015
83	4.31	1.892	1.6	105.4	0.2700	0.067	4.8	2.83	0.9953	0.0015
84	4.31	1.892	1.6	105.4	0.2653	0.0263	3.53	2.83	0.9956	0.0016
85	4.31	1.892	1.6	105.4	0.2621	0.0263	3.6	2.83	0.9979	0.0015
86	4.31	1.892	1.6	105.4	0.2664	0.0472	4.94	2.83	0.9980	0.0015
87	2.35	1.684	1.6	218.6	0.1360	0.00456	3.86		0.9922	0.0014
88	2.35	1.684	1.6	218.6	0.1377	0.00691	3.46		0.9871	0.0014
89	4.31	1.892	1.6	105.4	0.2461	0.00456	7.23		0.9948	0.0016
90	4.31	1.892	1.6	105.4	0.2464	0.00691	6.63		0.9984	0.0015
99	4.31	1.892	1.6	105.4	0.1924		10.86	2.83	0.9965	0.0016
100	4.31	1.892	1.6	105.4	0.1955		3.38	2.83	0.9931	0.0015
101	4.31	1.892	1.6	105.4	0.2110		11.55	2.83	0.9978	0.0015
102	4.31	1.892	1.6	105.4	0.2022		4.47	2.83	0.9937	0.0015
103	4.31	1.892	1.6	105.4	0.2160		8.36	2.83	0.9918	0.0014
104	2.35	1.684	1.6	218.6	0.1302		7.8		0.9892	0.0014
105	4.31	1.892	1.6	105.4	0.2354		10.52		0.9960	0.0016
106	4.31	1.892	1.6	105.4	0.1637				1.0003	0.0016
107	2.35	1.684	1.6	218.6	0.1158				0.9930	0.0014
109	4.31	1.892	1.6	105.4	0.2562	0.04085	8.3		0.9969	0.0015
110	2.35	1.684	1.6	218.6	0.1405		4.8		0.9950	0.0013
111	4.31	1.892	1.6	105.4	0.2523		9.83		0.9967	0.0015

Table 6.5-3, continued

case	enrich	pin pitch (cm)	water/ fuel volume	H/X	AEF (eV)	boron 10 (g/cm**2)	assy spacing (cm)	second assy spacing	k_{eff}	σ
118	2.35	1.684	1.6	218.6	0.1419		8.98		0.9957	0.0013
119	2.35	1.684	1.6	218.6	0.1359		9.58		0.9958	0.0014
120	2.35	1.684	1.6	218.6	0.1320		9.66		0.9920	0.0015
121	2.35	1.684	1.6	218.6	0.1253		8.54		0.9900	0.0014
122	2.35	2.032	2.918	398.7	0.0733		10.36		0.9971	0.0014
123	2.35	2.032	2.918	398.7	0.0754		11.2		0.9986	0.0012
124	4.31	1.892	1.6	105.4	0.2635		14.87		1.0018	0.0015
125	4.31	1.892	1.6	105.4	0.2532		15.74		1.0019	0.0016
126	4.31	1.892	1.6	105.4	0.2448		15.87		1.0021	0.0016
127	4.31	1.892	1.6	105.4	0.2391		15.84		0.9963	0.0015
128	4.31	1.892	1.6	105.4	0.2359		15.45		0.9954	0.0015
129	4.31	1.892	1.6	105.4	0.2305		13.82		0.9966	0.0015
130	4.31	2.54	3.883	256.1	0.1011		12.89		1.0006	0.0016
131	4.31	2.54	3.883	256.1	0.0952		14.12		1.0044	0.0016
132	4.31	2.54	3.883	256.1	0.0932		12.44		0.9994	0.0015
133	2.35	1.684	1.6	218.6	0.1328		8.28		0.9935	0.0014
134	4.31	1.892	1.6	105.4	0.2436		13.75		0.9986	0.0015
141	2.35	1.684	1.6	218.6	0.1251		6.59		0.9906	0.0014
142	4.31	1.892	1.6	105.4	0.2321		12.97		0.9944	0.0016
156	4.31	1.598	0.509	33.6	3.0230				0.9951	0.0018
162	4.74	1.6	3.807	228.6	0.1192	0.0461	2		1.0000	0.0016
163	4.74	1.6	3.807	228.6	0.1144	0.0461	2		0.9936	0.0016
164	4.74	1.6	3.807	228.6	0.1199	0.0461	2		1.0019	0.0016
165	4.74	1.6	3.807	228.6	0.1152	0.0461	2		0.9955	0.0016
171	2.72	1.524	1.495	156.1	0.2597				0.9896	0.0015
172	5.7	1.422	1.932	98.3	0.2684				0.9997	0.0015
173	5.7	1.524	1.495	156.1	0.2470				0.9929	0.0014
174	5.7	1.422	1.932	98.3	0.2580				0.9981	0.0015
175	5.74	1.422	1.933	97.6	0.2520				0.9941	0.0016
176	5.74	2.011	5.067	255.9	0.0847				0.9986	0.0015
								avg keff	0.9958	
								std dev	0.0038	

Notes:

1. AEF is the energy (eV) of the average lethargy causing fission
2. H/X is the atom ratio of hydrogen to U235 in the pin cell
3. Water/fuel volume ratio is defined by the pin cell only

Table 6.5-4
Trend Analysis of Benchmark Results

Independent variable	**experiment range**	**TN-68 range**	**number of cases**	**correlation coefficient *r***
Pin pitch, cm	1.26 - 2.54	1.29 - 1.88	73	0.10
Boron areal density in separator plates, g/cm^2	0.005 - 0.067	0.027	17	0.26
Pin cell hydrogen to U235 atom ratio (H/X)	33.6 - 504.2	119 - 146	73	-0.11
Pin cell water / fuel pellet volume ratio	0.509 - 7.608	1.411 - 1.715	73	0.30
Energy of the average lethargy causing fission (AEF), eV	0.055 - 0.27	0.180-0.300 (note 1)	72	0.085
Assembly separation, cm	2 - 15.87	3.86	55	0.26
Assembly separation using second assembly distance, cm	2 - 15.87	3.86	55	0.27
Enrichment, wt% U235	2.35 – 5.74	3.70	73	0.52

Notes:

1. Except for reduced water density cases

Independent variable, x	**USL**	**TN-68**	
		x	**USL**
Pin pitch, cm	$0.9364+1.2243\times10^{-3}x$	1.29	0.9380
Boron areal density in separator plates, g/cm^2	$0.9377+3.1765\times10^{-2}x$	0.027	0.9386
Pin cell hydrogen to U235 atom ratio (H/X)	$0.9392-4.0608\times10^{-6}x$	146	0.9386
Pin cell water / fuel pellet volume ratio	$0.9358+1.0398\times10^{-3}x$ $x < 6.305$	1.411	0.9373
Energy of the average lethargy causing fission (AEF), ev	$0.9379+4.8275\times10^{-3}x$	0.180	0.9388
Assembly separation, cm	$0.9365+2.5985\times10^{-4}x$	3.86	0.9375
Assembly separation using second assembly distance, cm	$0.9367+2.5336\times10^{-4}x$	3.86	0.9377
Enrichments, wt% U235	$0.9325 + 1.8290\times10^{-3}x$	3.70	0.9393

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE 6-3-1
BASKET VIEWS & DIMENSIONS
REV. 0 4/99

Figure 6.3-2
Basket Model Compartment Wall, View G

(Refer to Figure 6.3-1 for view orientation)

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure 6.3-3
Basket Model Compartment Wall, View F

(Refer to Figure 6.3-1 for view orientation)

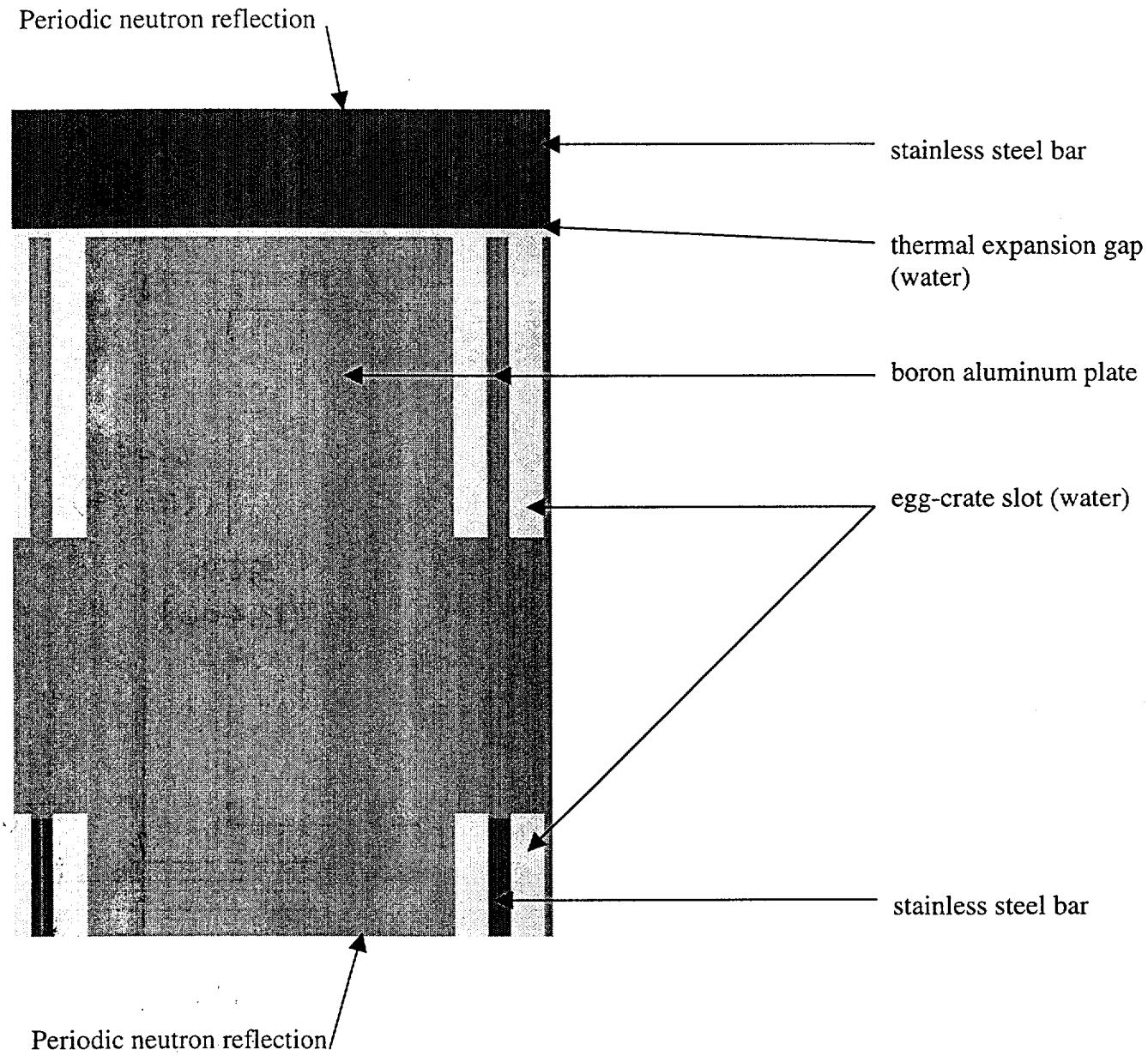


Figure 6.3-4
Basket Compartment Section A, Fuel Model 1lwc,

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure 6.3-5
Basket Compartment Section B, Fuel Model 84

(Refer to Figure 6.3-1 for section location)

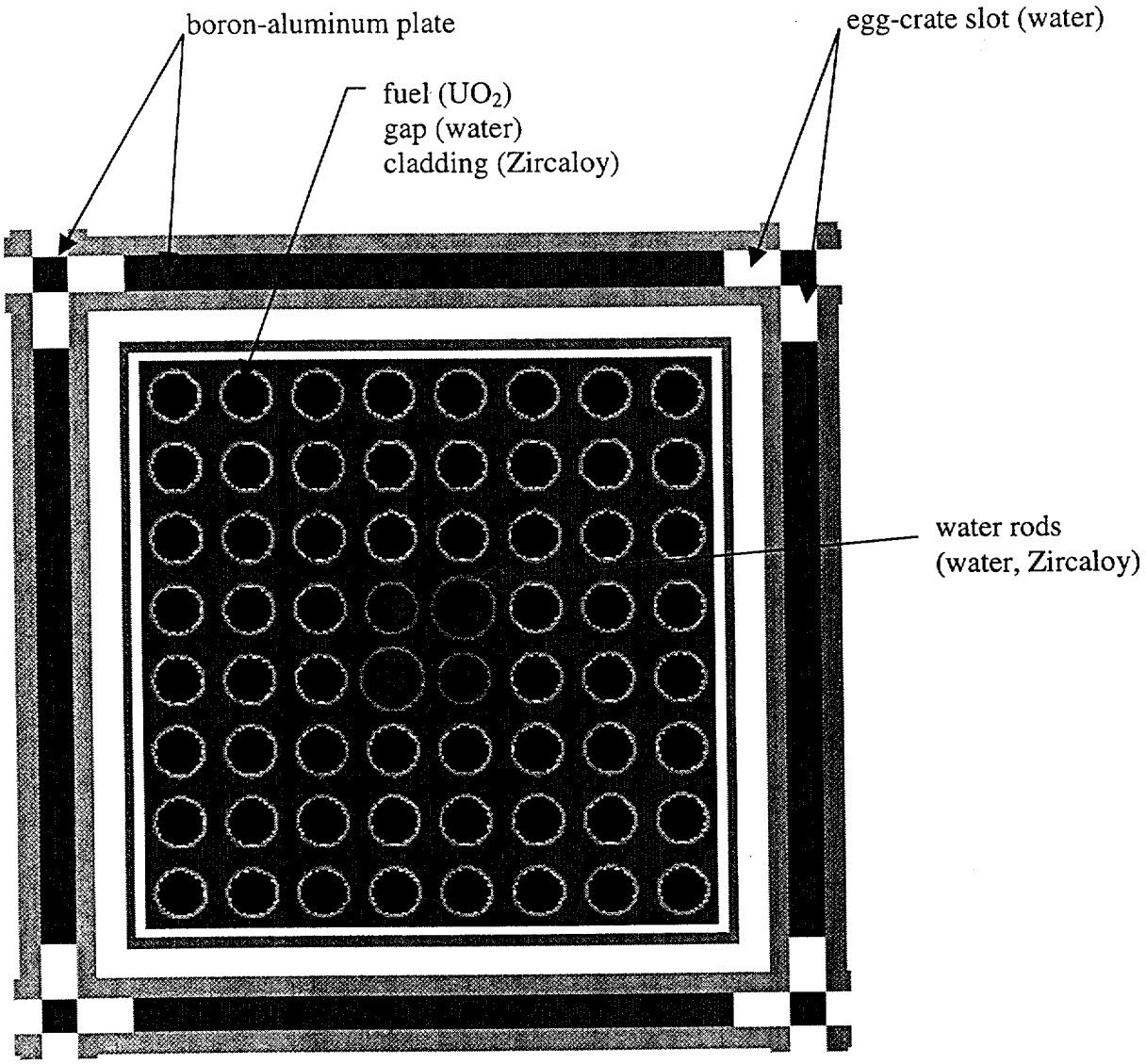


Figure 6.3-6
Basket Compartment Section C, Fuel Model 12wc

(Refer to Figure 6.3-1 for section location)

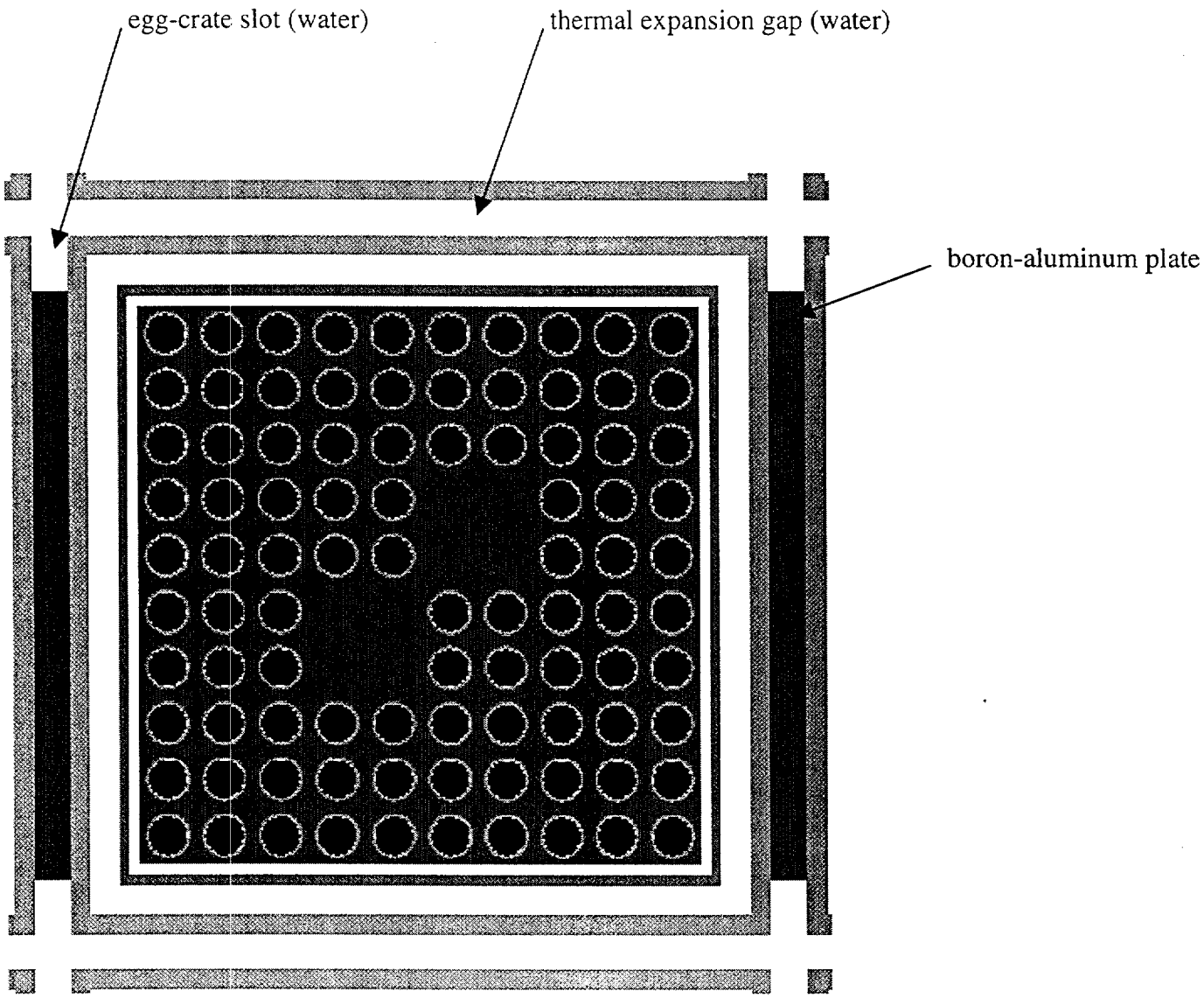


Figure 6.3-7
Basket Compartment Section D, Fuel Model 9wc

(Refer to Figure 6.3-1 for section location)

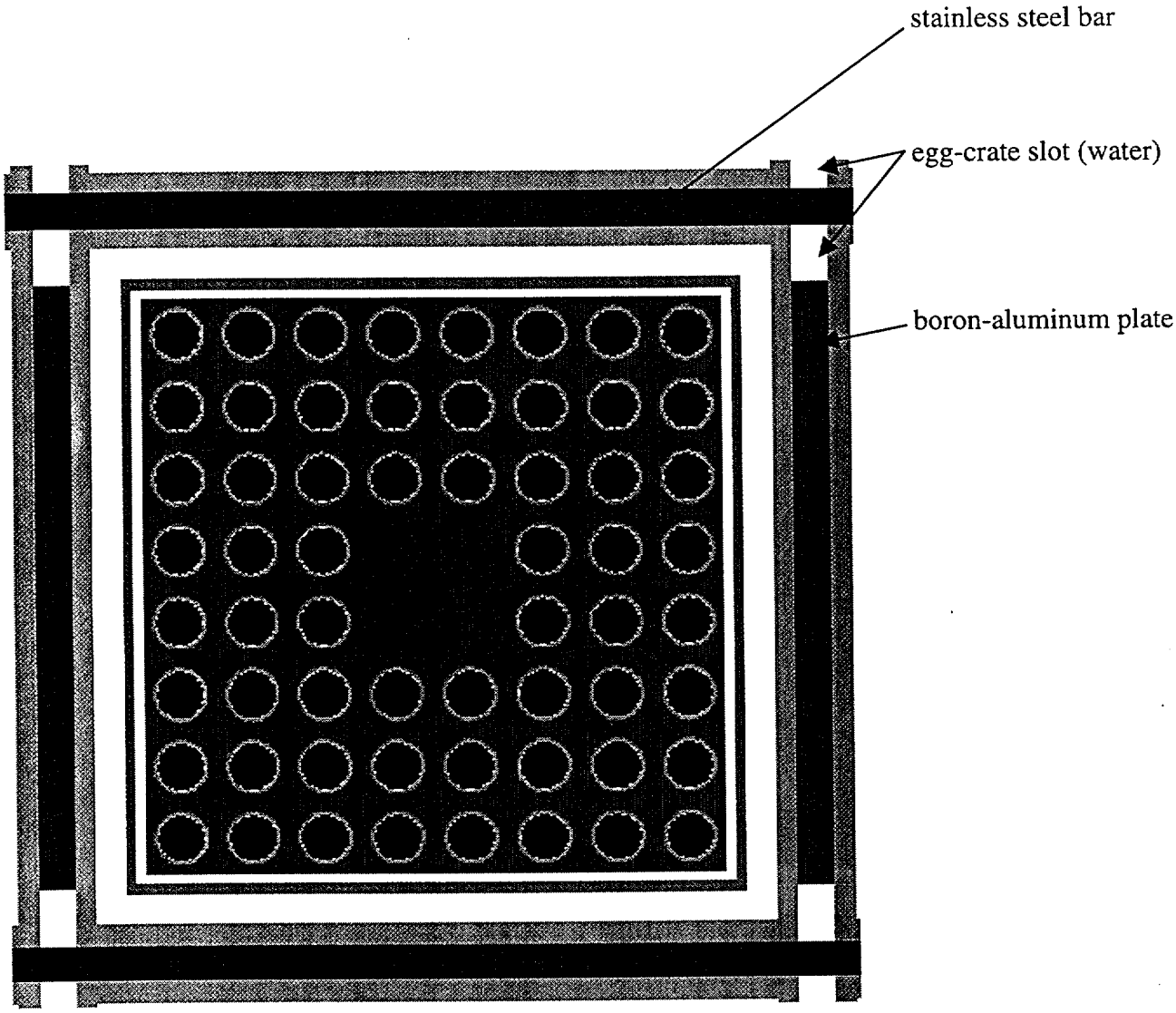


Figure 6.3-8
Basket Compartment Section E, Fuel Model 82wc

(Refer to Figure 6.3-1 for section location)

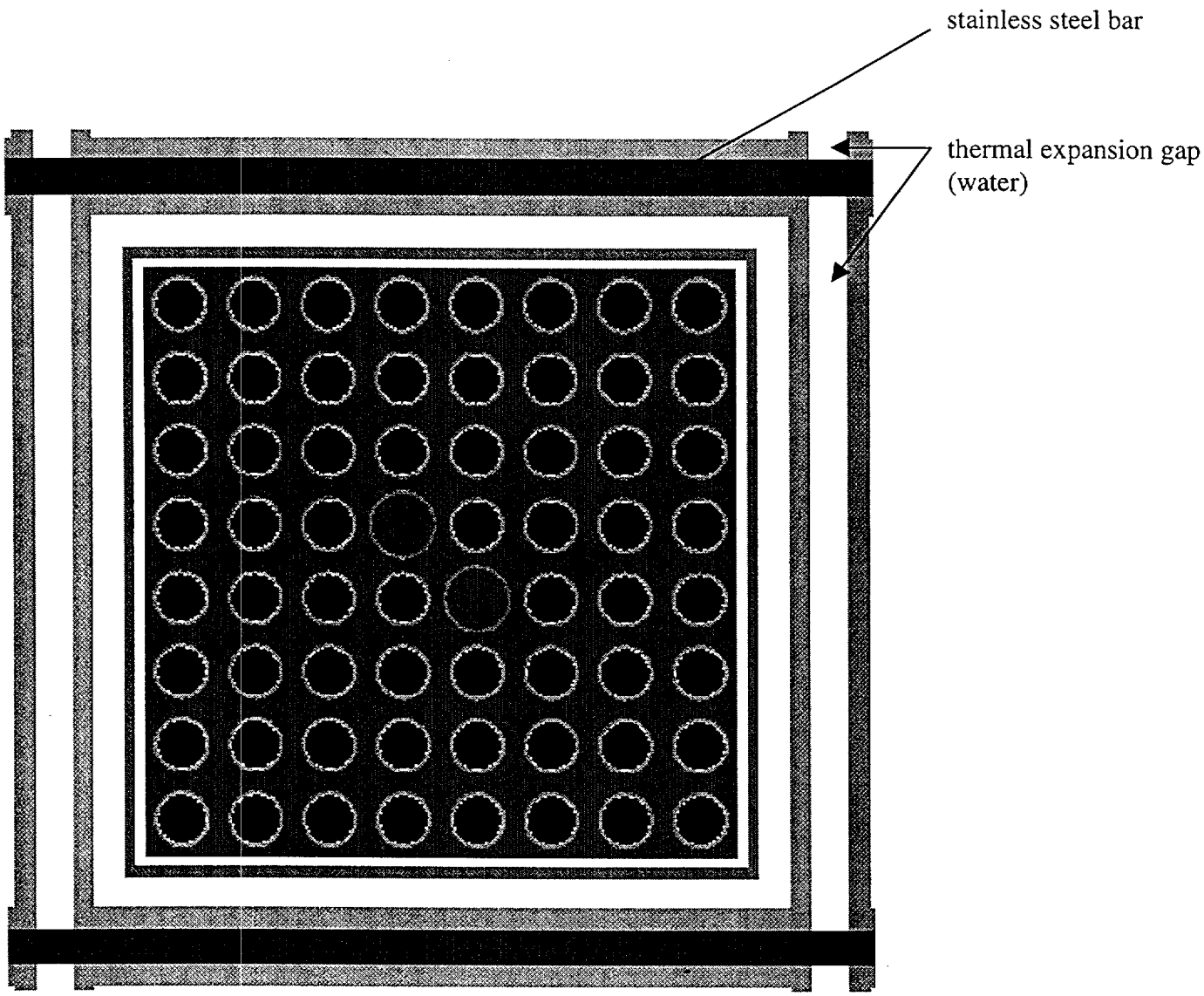


Figure 6.3-9
Criticality Model Validation – Design Approximation Model

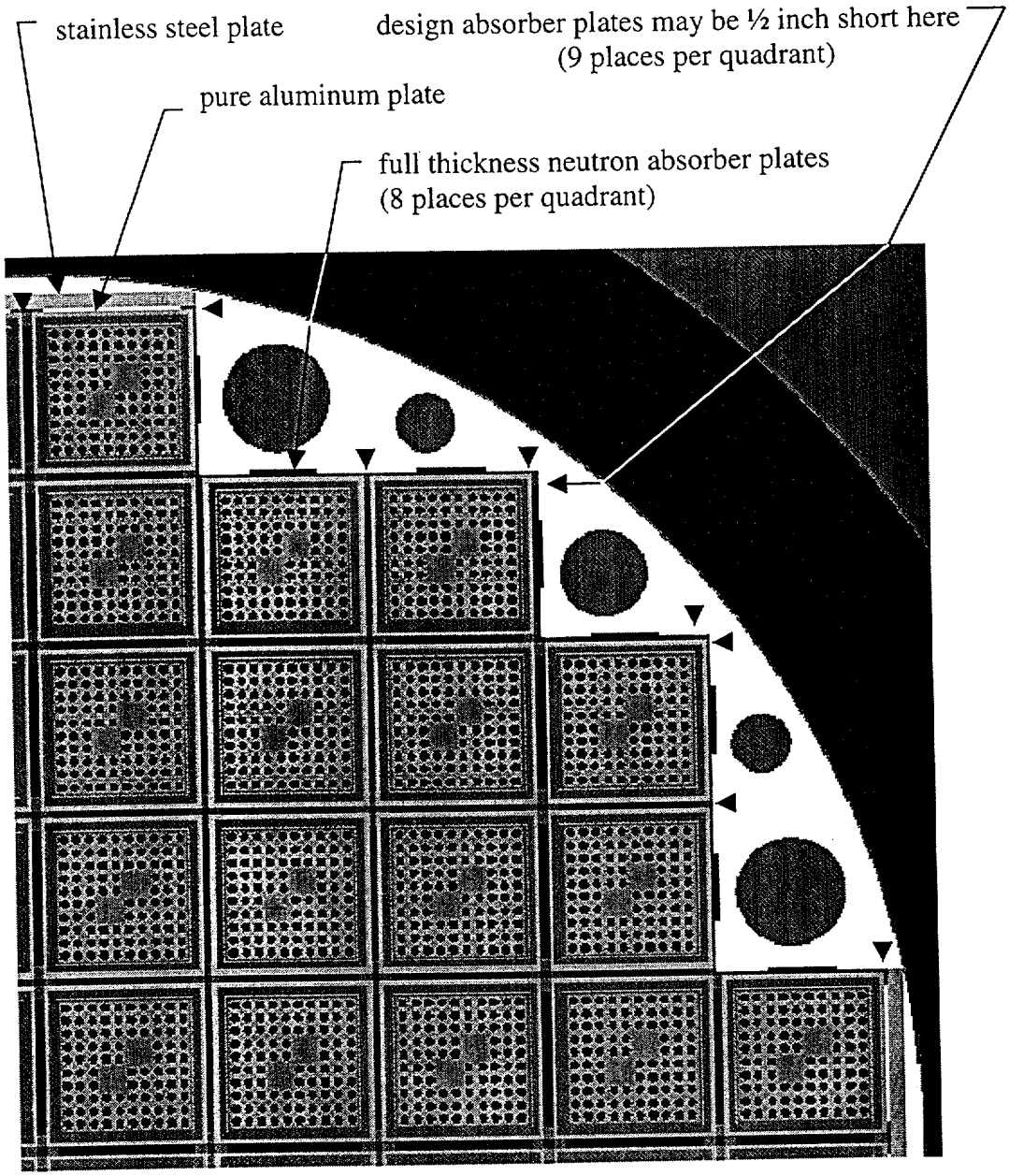
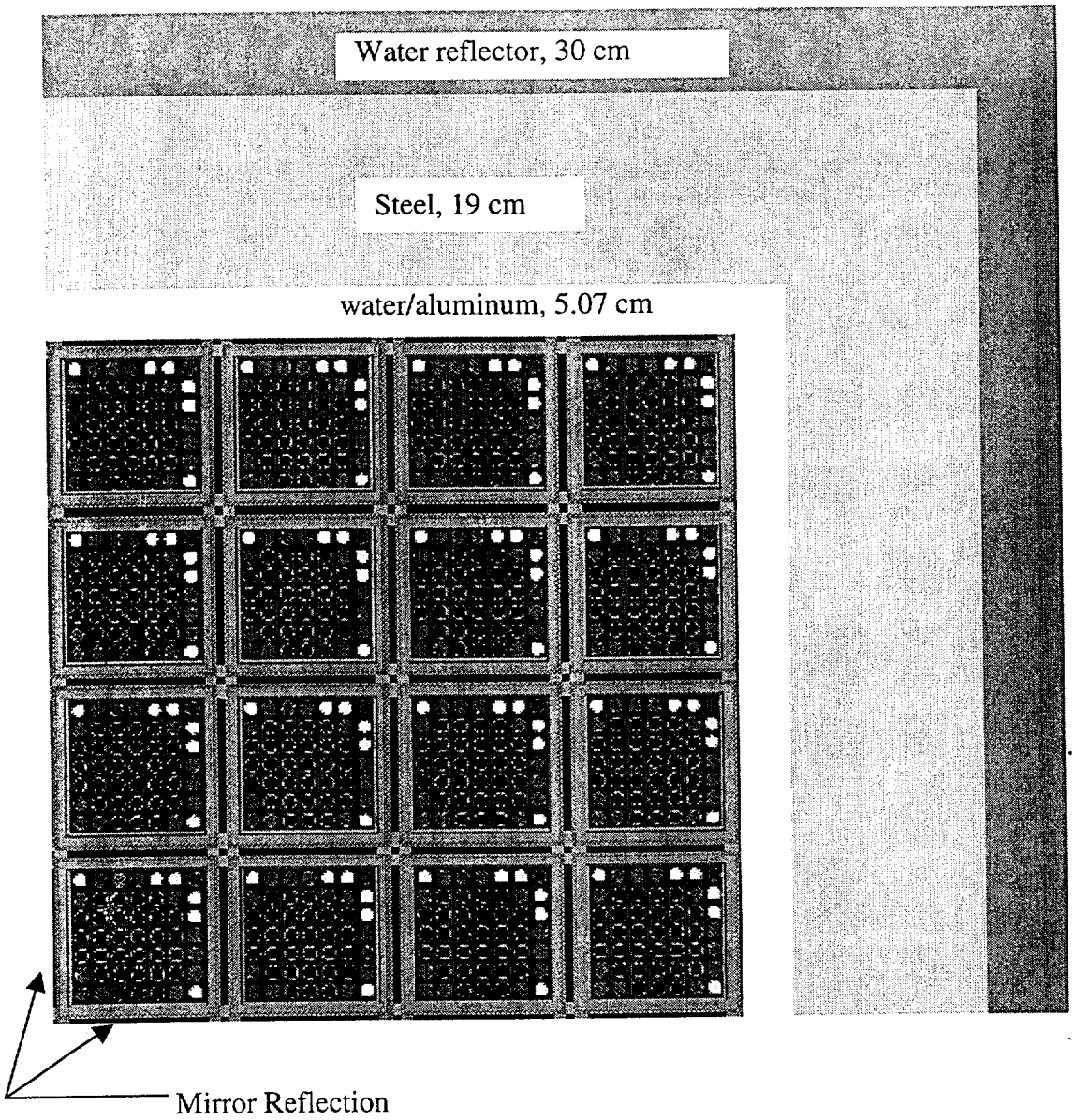


Figure 6.4-1

TN-68 Model Cross-Section, Most Reactive Lattice Evaluation

FIGURE WITHHELD UNDER 10 CFR 2.390

Figure 6.4-2
TN-68 Model Cross-Section, Uniform Enrichment Validation, Case 3var



Model is 30.912 cm in the z direction, with periodic reflection top and bottom

Note: this model is not intended to be an accurate representation of the cask, but rather to provide a comparison of uniform and non-uniform pin enrichment fuel models in a configuration similar to the TN-68. All fuel assemblies are oriented with the highest enrichment corner toward the cask axis.

Figure 6.4-3
Vanished Lattice Model

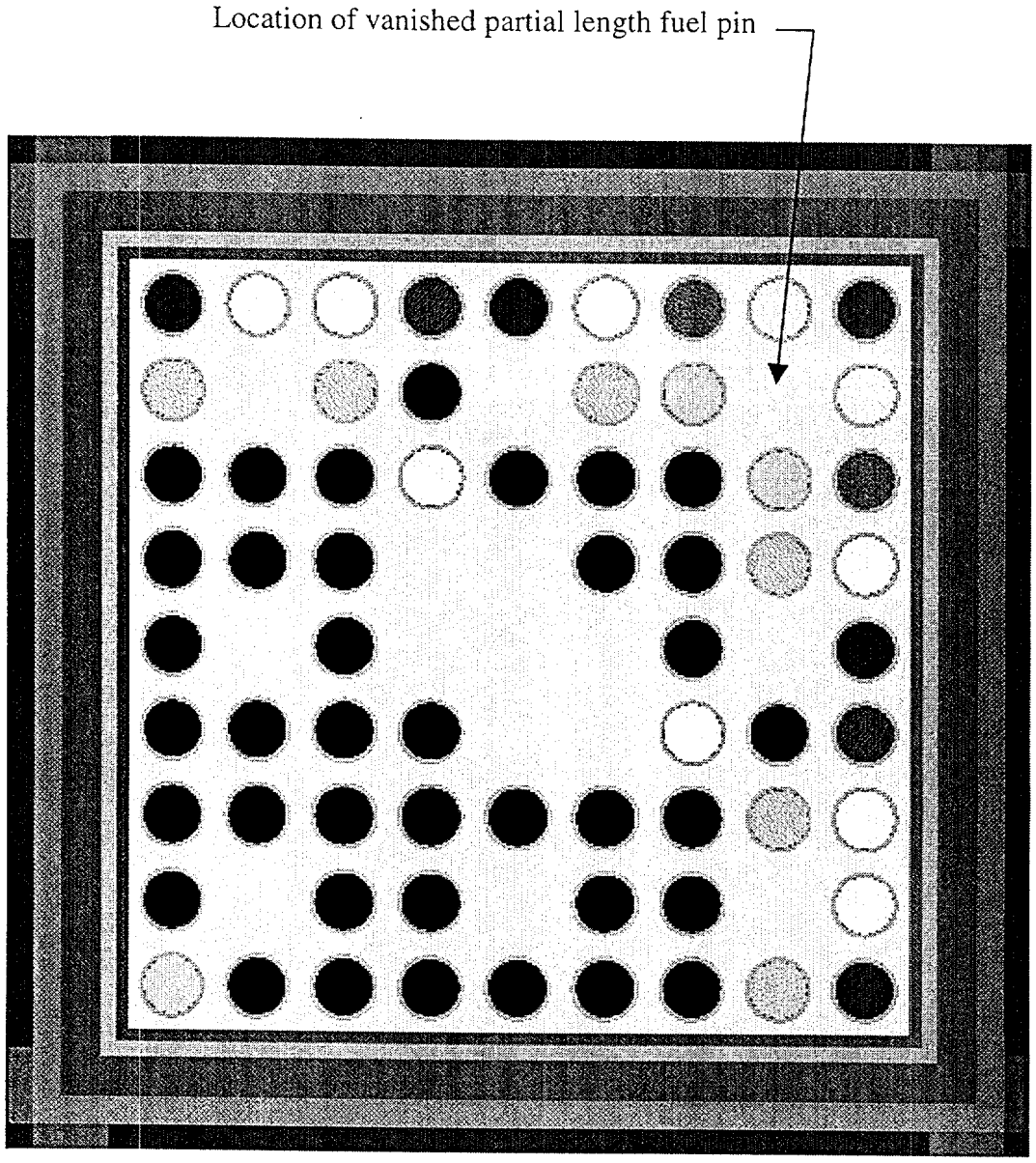
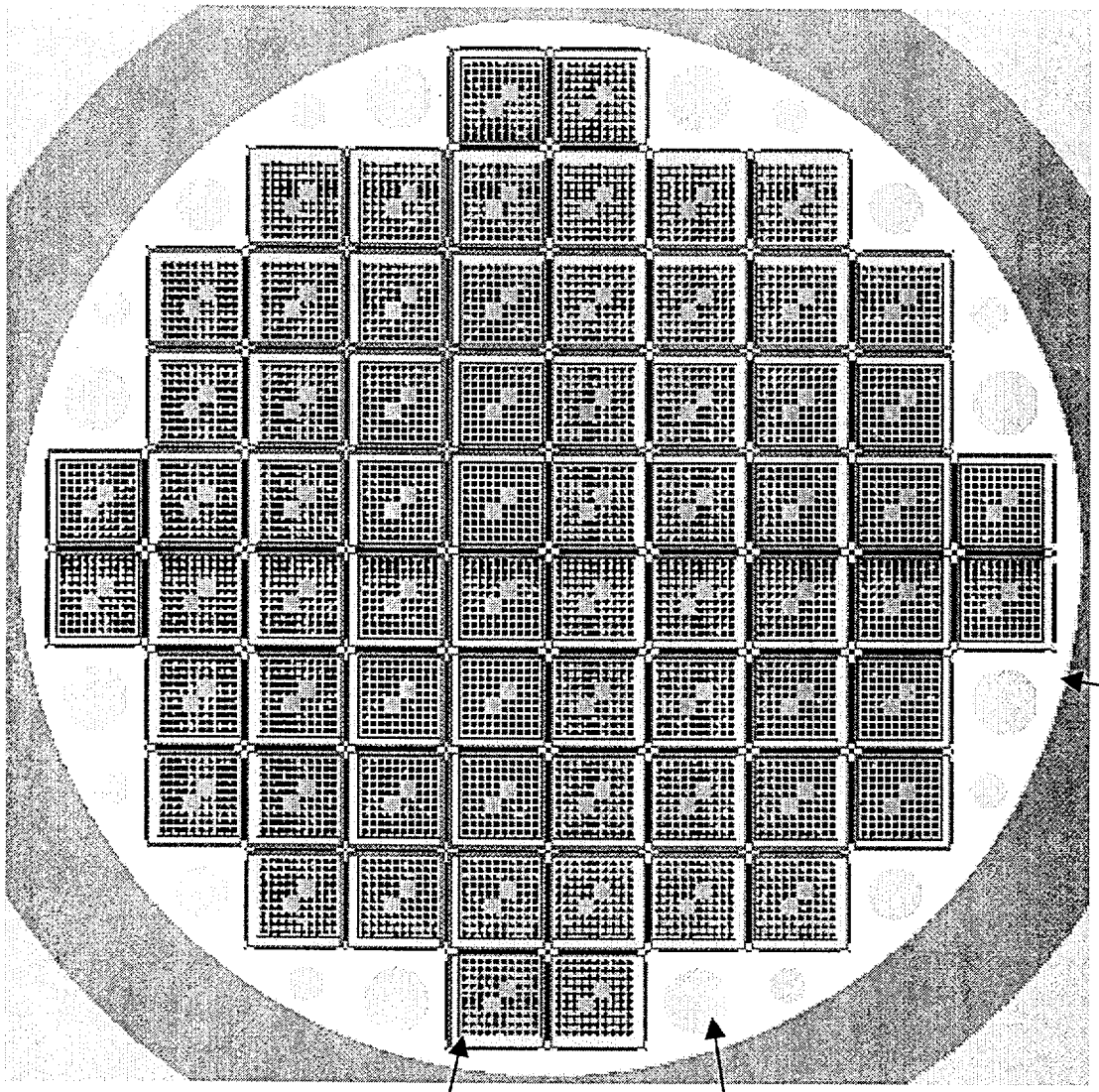


Figure 6.4-4

Model Cross-Section, TN-68 Criticality Evaluation

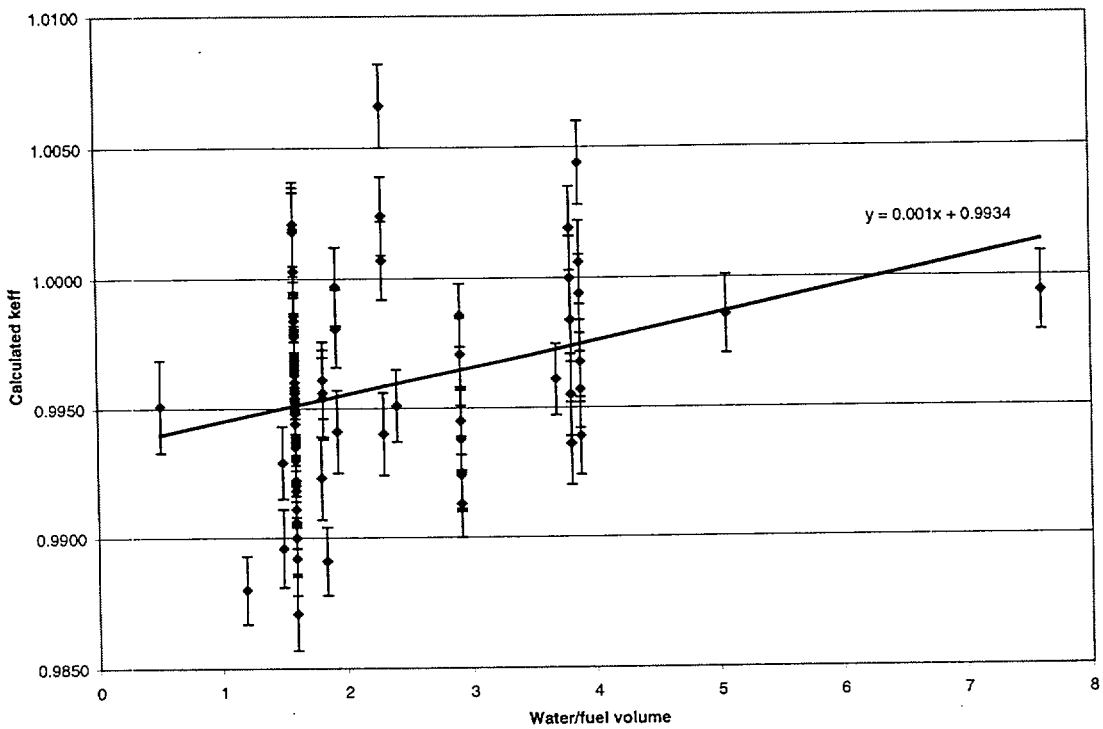


3 mm water between basket rail and wall

Compartment 5.97-6.05 inch inside
All fuel shifted toward cask vertical axis

Basket rails modeled as aluminum with water holes

Figure 6.5-1
Benchmarks - Trend Evaluation for Water /Fuel Volume Ratio



Note: no credit is taken for positive bias

TN 68 TRANSPORT PACKAGING

CHAPTER 7

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CHAPTER 7

OPERATING PROCEDURES

This chapter contains TN-68 loading and unloading procedures that are intended to show the general approach to cask operational activities. A separate Operations Manual (OM) will be prepared for the TN-68 to describe the operational steps in greater detail. The OM, along with the information in this chapter, will be used to prepare the site-specific procedures that will address the particular operational considerations related to the cask. The operations required to convert the cask from its storage configuration to its transport configuration are also described in here.

7.1 PACKAGE LOADING

7.1.1 Preparation for Loading

7.1.1.1 Upon arrival of the empty packaging, on its transport vehicle (rail or heavy haul trailer) and shipping frame, perform a receipt inspection to check for any damages or irregularities. Verify that the records for the packaging are complete and accurate.

7.1.1.2 Remove the security device, the impact limiter attachment bolts, tie-rods, and the associated hardware, as necessary.

7.1.1.3 Remove the front and the rear impact limiters, as well as the front spacer and the ancillary shield ring, using a suitable crane and a two-legged sling or an equivalent.

7.1.1.4 Remove the tie-down strap and trunnion support block caps.

7.1.1.5 Clean the external surfaces of the cask, if necessary, to get rid of the road dirt.

7.1.1.6 Attach the lift beam to the cask handling crane hook, and engage the lift beam to the two upper (top) trunnions.

7.1.1.7 Rotate the cask slowly from the horizontal to the vertical position.

7.1.1.8 Lift the cask from the transport/shipping frame and place it in the cask preparation area.

7.1.1.9 Disengage the lift beam from the cask

7.1.1.10 Remove the neutron shield pressure relief valve and install the plug in the neutron shield vent hole.

7.1.1.11 Remove the lid bolts and the lid.

7.1.1.12 Replace the lid seal using the retaining screws, and inspect the lid sealing surface. Check for defects in the seal contact areas that may prevent a proper seal. (This step may be performed at any time prior to installing the lid on the loaded cask).

7.1.1.13 Replace the seals in the vent, drain and transport covers, and inspect the sealing surfaces. Check for defects in the seal contact areas that may prevent a proper sealing. (This step may be performed at any time prior to installing covers on the loaded cask).

7.1.1.14 Visually inspect the lid bolts and the bolt hole threads to ensure that they do not have any laps, seams, cracks or damaged threads.

7.1.1.15 Remove the hold down ring from the cask cavity.

7.1.1.16 Verify that the basket is installed in the cask, with no evident signs of damage to either. Verify that there is no foreign material in the cask.

7.1.1.17 Move the cask to the cask loading area using the lift beam attached to the top trunnions.

7.1.2 Loading

Note: The term 'cask loading pool' is used to describe the area where the cask is to be loaded.

7.1.2.1 Lower the cask into the cask loading pool, while rinsing the exterior of the cask with demineralized water and filling the interior with demineralized or pool water.

7.1.2.2 Disengage the lift beam and move it aside.

7.1.2.3 Load the pre-selected spent fuel assemblies into the basket compartments. Procedures shall be developed to ensure that the fuel loaded into the cask meets the fuel specifications.

7.1.2.4 Verify the identity of the fuel assemblies loaded into the cask, and document the location of each fuel assembly on the cask loading report.

7.1.2.5 At least one lid penetration (drain or vent port) must be completely open (both cover and quick-disconnect fitting removed) prior to the installation of the lid. Using the lift beam and the lid lifting slings, lower the lid placing it on the cask body flange over the two alignment pins.

7.1.2.6 Engage the lift beam on the upper (top) trunnions, and lift the cask so that the top of the cask is above the water surface in the pool, and install some of the lid bolts. The lid bolts should be hand tight.

Note: Throughout this procedure, all bolt threads are to be coated with Nuclear Grade Neolube or equivalent.

7.1.2.8 Using the drain port in the lid, drain the water from the cask in accordance with the procedures. This may be done either before or after lifting the cask out of the pool. While lifting the cask out of the pool, the exterior of the cask may be rinsed with clean deionized water to facilitate decontamination.

Note: In order to minimize internal hydrogen accumulation, the cask should be drained completely within 18 hours of the start of draining. If this period is exceeded, the cask cavity should be inerted by injecting nitrogen, argon or helium through the open lid penetration, while the draining continues. An initial gas flow rate of 0.6 m^3 per minute (21 cfm) will purge the cask cavity volume in about 10 minutes, after which the flow rate can be reduced to about 3 cfm until the draining is complete.

7.1.2.9 Disconnect the drain line.

7.1.2.10 Move the cask to the decontamination area and disengage the lift beam.

7.1.3 Preparation for Transport

Note: The maximum potential for worker exposure exists during the decontamination of the cask and other operations near the lid, after the water is pumped out of the cask. Worker exposure can be minimized by use of temporary shielding (lead "bean bags", plastic neutron shielding), and by minimizing the exposure time and maximizing the distance, as well as using any measures to facilitate decontamination.

7.1.3.1 Decontaminate the cask until acceptable surface contamination levels are obtained.

7.1.3.2 Install the remaining lid bolts and torque them to 200 ft-lbs. Follow the torquing sequence shown in Figure 7-1. Repeat the torquing process following the sequence of Figure 7-1. Torque to 600 ft-lbs in the second pass, 1000 ft-lbs in the third pass and between 2050 and 2100 ft-lbs in the final pass. A circular pattern of torquing may be used, to eliminate further bolt movement.

7.1.3.3 Remove the plug from the neutron shield vent, and reinstall the pressure relief valve, making sure that it is operable and set.

7.1.3.4 Evacuate the cask cavity using the Vacuum Drying System (VDS) to remove the remaining moisture, and verify the dryness as follows:

a) Using a wand attached to the vacuum drying system, remove any excess water from the seal areas through the passageways at the overpressure drain and vent ports.
b) Remove the quick disconnect from the drain port, and install the drain port cover.
c) With the quick disconnect removed to improve evacuation, connect the VDS to a flanged vacuum connector installed over the vent port. Purge or evacuate the

helium supply lines and evacuate the cask to 4 millibar (4×10^{-4} MPa) or less. Make provisions to prevent or correct any icing of the evacuation lines, if necessary.

d) Isolate the vacuum pump. If, in a period of 30 minutes, the pressure does not exceed 4 millibar (4×10^{-4} MPa), the cask is adequately dried. Otherwise, repeat the vacuum pumping until this criterion is met.

e) Backfill the evacuated cask cavity with helium (minimum 99.99% purity), to slightly above atmospheric pressure. Then, remove the vacuum connector and immediately install the quick disconnect fitting.

f) Attach the vacuum/backfill manifold to the vent port fitting, purge or evacuate the helium supply lines, and re-evacuate the cask to below 100 mbar.

7.1.3.5 Isolate the vacuum pump, and backfill the cask cavity to approximately 2.0 atm abs (14.7 psig) with helium (minimum 99.99% purity).

7.1.3.6 Leak test the inner lid, inner vent and drain port cover seals. The maximum acceptable cask seal leak rate is 1×10^{-5} ref cm^3/sec. The leak test shall be performed in accordance with ANSI N14.5[(2)]. For ports containing quick-disconnects, purge the cavity below the cover with helium, at a minimum flow rate of 80 cubic feet per hour for at least 20 seconds. Install the port cover. (A partial pressure of at least 50% helium will be obtained under the cover.) The vent and drain covers should be torqued to the values specified on drawing 972-71-1 prior to leak testing.

7.1.3.7 If the cask does not pass the leak test, determine the source of the leak. If the leak is in a vent or drain cover, remove the cover and replace the seals. Also examine the sealing surface for any obvious indication of scratches or defects. Repeat the leak test after replacing the seals.

7.1.3.8 If the cask still does not pass the leak test, evaluate the test method or return the cask to the pool and replace the lid seals.

7.1.3.9 Install the overpressure transport cover. Torque the bolts to the value specified on drawing 972-71-2.

7.1.3.10 Re-engage the lift beam to the upper (top) trunnions of the cask.

7.1.3.11 Move the transport vehicle into the loading position.

7.1.3.12 Lift the cask off the decontamination pad, and place the rear trunnions on the rear trunnion supports of the transport frame.

7.1.3.13 Rotate the cask from the vertical to the horizontal position.

7.1.3.14 Install the lower (bottom) trunnion support caps and the tie-down strap.

7.1.3.15 Check if the surface dose rates and the surface contamination levels are within the regulatory limits. Install an optional shield ring adjacent to the top of neutron shield, if required, based on dose limits.

7.1.3.16 Install the spacer on the front end of the cask. Then remove the spacer lifting eye bolts.

7.1.3.17 Install the front and the rear impact limiters onto the cask. Lubricate the attachment bolts with Never-Seez or an equivalent and torque to 200 ft-lb, diametrically in the first pass, and to 400 ft-lb in the final pass.

7.1.3.18 Install thirteen impact limiter attachment tie-rods between the front and the rear impact limiters.

7.1.3.19 Render the impact limiter lifting lugs inoperable by covering the lifting holes or installing a bolt inside the holes to prevent their inadvertent use.

7.1.3.20 Install security seal on one tie-rod and lock sleeve.

7.1.3.21 Install a transportation enclosure.

7.1.3.22 Check the temperature on all accessible surfaces to make sure that it is <185 °F.

7.1.3.23 Perform a final radiation and contamination survey to satisfy the shield test requirements and to assure compliance with 10CFR71.47 and 71.87.

7.1.3.24 Apply appropriate DOT labels and Placards in accordance with 49CFR172. Prepare the final shipping documentation.

7.1.3.25 Release the loaded cask for shipment.

7.2 PACKAGE UNLOADING

7.2.1 Receipt of Package from Carrier

7.2.1.1 Upon arrival of the loaded cask, perform a receipt inspection of the cask to check for any damage or irregularities. Verify that the security seal is intact, and perform a radiation survey.

7.2.1.2 Verify that the records for the packaging are complete and accurate.

7.2.1.2 Remove the security seal, the impact limiters, tie-rods, and the associated hardware.

7.2.1.3 Render the impact limiter lifting lugs operable by removing the covering on the lifting holes or the bolt inside the lifting holes, that prevented their inadvertent use.

7.2.1.4 Remove the front and rear impact limiters as well as the front spacer, using a suitable crane and a two-legged sling or an equivalent.

7.2.1.5 Remove the tie down strap and trunnion support block caps.

7.2.1.6 Attach the lift beam to the cask handling crane hook, and then engage the lift beam to the two upper (top) trunnions.

7.2.1.7 Rotate the cask slowly from the horizontal to the vertical position.

7.2.1.8 Lift the cask from the transport/shipping frame, and place it in the decontamination area.

7.2.1.9 Disengage the lift beam from the cask, and move the crane as well as the lift beam from the area.

7.2.1.10 Clean the external surfaces of the cask, if necessary, to get rid of the road dirt.

7.2.1.11 Remove the neutron shield pressure relief valve, and install the plug in the neutron shield vent hole.

7.2.2 Preparation for Unloading

7.2.2.1 Remove the vent cover.

7.2.2.2 Collect a cavity gas sample, through the vent port quick-disconnect coupling, if required

7.2.2.3 Analyze the gas sample for radioactive material, and add necessary precautions based on the cavity gas sample results.

Note: If degraded fuel is suspected, additional measures, appropriate for the specific conditions, are to be planned, reviewed, and approved by the appropriate

plant personnel, as well as implemented to minimize worker exposures and radiological releases to the environment. These additional measures may include provision of filters, as well as respiratory protection and other methods to control releases and exposures to ALARA.

7.2.2.4 In accordance with the site requirements, vent the cavity gas through the hose until atmospheric pressure is reached.

7.2.2.5 Remove the vent port quick-disconnect and the drain port cover. Attach the vent port adapter.

7.2.2.6 Loosen the lid bolts and remove all but six lid bolts, approximately equally spaced.

7.2.2.7 Attach the cask to the crane using lift beam. Attach the lid lifting equipment.

7.2.2.8 Attach the fill and drain lines to the drain quick-disconnect coupling and the vent port adapter.

7.2.2.9 Ensure that appropriate measures are in place for proper handling of steam. Both fill and drain lines should be designed for a minimum of 100 psig steam, to prevent steam burns and radiation exposures due to a possible line failure.

7.2.2.10 Lower the cask into the spent fuel pool cask pit, while spraying the exterior of the cask with demineralized water to minimize contamination. Lower the cask until the top surface is just above the water level. Note: The cask may be filled with some water before lowering the cask into the pool or while the cask is partially submerged in the spent fuel pool. Vent the cavity pressure, and then remove the drain port cover.

7.2.3 <u>Contents Removal</u>

Note: In BWR spent fuel pools, there may be significant amounts of fuel crud particulate material. Precautions should be taken to ensure that this particulate material does not become airborne or float on the surface of the water, becoming a radiation concern. Precautions may include enhanced filtering of the pool water during the loading and unloading operations, as well as increased ventilation and monitoring of airborne contamination during all spent fuel pool activities.

7.2.3.1 Begin pumping pool or demineralized water into the cask through the drain port, at a rate of 1 gpm, while continuously monitoring the exit-pressure (See Figure 7-2). Continue pumping the water at a rate of 1 gpm for at least eighty minutes. By this time, the water level in the cask will have reached the active fuel length.

7.2.3.2 The flow rate can then be gradually increased, while monitoring the pressure at the outlet. If the pressure gage reading exceeds 55.3 psig, close the inlet valve until the pressure falls below 50 psig. Re-flooding can then be resumed.

7.2.3.3 After verifying that a steady stream of water is coming from the vent line (by checking for bubbles or carefully lifting the hose out of the water), take a sample for chemical analysis.

7.2.3.4 When the cask is full of water, remove the hose from the drain port, and the hose and the vent port adapter from the vent port. Remove the remaining six lid bolts.

7.2.3.5 Lower the cask and place it on the bottom of the pool/pit while rinsing the lift beam with demineralized water.

7.2.3.6 Raise the lift beam from the cask, removing the cask lid.

7.2.3.7 Unload the spent fuel assemblies in accordance with the site procedures.

7.2.3.8 At least one lid penetration must be completely open (both cover and quick-disconnect fitting removed) prior to installation of the lid. Using the lift beam and lid lifting slings, lower the lid placing it on the cask body flange, over the two alignment pins.

7.2.3.9 Engage the lift beam on the upper (top) trunnions, and lift the cask out of the pool.

7.2.3.10 Using the drain port in the lid, drain the water from the cask in accordance with the procedures. This may be done either before or after lifting the cask out of the pool. While lifting the cask out of the pool, the exterior of the cask may be rinsed with clean deionized water to facilitate decontamination.

> Note: In order to minimize internal hydrogen accumulation, the cask should be drained completely within 18 hours of the start of draining. If this period is exceeded, the cask cavity should be inerted by injecting nitrogen, argon, or helium through the open lid penetration while the draining proceeds. An initial inert gas flow rate of 0.6 m^3 per minute (21 cfm) will purge the cask cavity volume in about 10 minutes, after which the flow rate can be reduced to about 3 cfm until the draining is complete.

7.2.3.11 Disconnect the drain line from the quick-disconnect couplings.

7.2.3.12 Move the cask to the decontamination area, and disengage the lift beam.

7.3 PREPARATION OF EMPTY PACKAGE FOR TRANSPORT

7.3.1 Decontaminate the cask until acceptable surface contamination levels are obtained.

7.3.2 Lubricate and install the lid bolts and torque them to 200 ft-lbs. Follow the torquing sequence shown in Figure 7-1. Repeat the torquing process following the sequence of Figure 7-1. Torque to 400 ft-lbs in the second pass. A circular pattern of torquing may be used, to eliminate further bolt movement.

7.3.3 Remove the plug from the neutron shield vent, and reinstall the pressure relief valve, making sure that it is operable and set.

7.3.4 Evacuate the cask cavity using the Vacuum Drying System (VDS) to remove the remaining moisture, and verify the dryness as follows:

a) Using a wand attached to the vacuum drying system, remove any excess water from the seal areas through the passageways at the overpressure drain and vent the ports.
b) Remove the quick disconnect from the drain port, and install the drain port cover.
c) With the quick-disconnect removed to improve evacuation, connect the VDS to a flanged vacuum connector installed over the vent port. Purge or evacuate the helium supply lines and evacuate the cask to 4 millibar (4×10^{-4} MPa) or less. Make provision to prevent or correct icing of the evacuation lines.
d) Isolate the vacuum pump. If, in a period of 30 minutes, the pressure does not exceed 4 millibar (4×10^{-4} MPa), the cask is adequately dried. Otherwise, repeat vacuum pumping until this criterion is met.
e) Backfill the evacuated cask cavity with helium (minimum 99.99% purity) to slightly above atmospheric pressure, remove the vacuum connector, and immediately install the quick disconnect fitting.
f) Attach the vacuum/backfill manifold to the vent port fitting, purge or evacuate the helium supply lines, and re-evacuate the cask to below 100 mbar.

7.3.5 Isolate the vacuum pump, and backfill the cask cavity with an inert gas.

7.3.6 Install the overpressure transport cover. Torque the bolts to the value specified on drawing 971-71-1.

7.3.7 Re-engage the lift beam to the upper (top) trunnions of the cask.

7.3.8 Move the transport vehicle into the loading position.

7.3.9 Lift the cask off the decontamination pad, and place the rear trunnions on the rear trunnion supports of the transport frame.

7.3.10 Rotate the cask from the vertical to the horizontal position.

7.3.11 Install the front and rear trunnion tie-downs, by torquing them to 100 ft-lb.

7.3.12 Check if the surface dose rates and the surface contamination levels are within the regulatory limits.

7.3.13 Install the spacer on the front end of the cask. Then remove the spacer lifting eye bolts.

7.3.14 Install the front and the rear impact limiters onto the cask. Lubricate the attachment bolts with Never-Seez or an equivalent, and torque to 200 ft-lbs, diametrically in the first pass, and to 400 ft-lbs in the final pass.

7.3.15 Install thirteen impact limiter attachment tie-rods between the front and the rear impact limiters.

7.3.16 Render the impact limiter lifting lugs inoperable, by covering the lifting holes or installing a bolt inside the holes to prevent their inadvertent use.

7.3.17 Perform a final radiation and contamination survey to satisfy the shield test requirements and to assure compliance with 10CFR71.47 and 71.87.

7.3.18 Install a transportation enclosure.

7.3.19 Apply appropriate DOT labels and Placards in accordance with 49CFR172, and prepare the final shipping documentation.

7.3.20 Release the empty cask for shipment.

7.4 OTHER PROCEDURES

7.4.1 Preparation of Cask Used in Storage for Transport

The TN-68 cask is designed for storage as well as transport. The following steps describe the steps required to convert the TN-68 from its storage configuration to the transport configuration. In some cases, the casks which have been used for storage may not have the transport regulatory plate or nameplate installed on them. These plates must be installed prior to transport. In addition, some casks that have been used for storage, may not have the impact limiter attachment lugs installed on them. The lugs must be welded to the outer shell, prior to using the cask for transport.

7.4.1.1 Review the loading records and ensure that the fuel within the storage cask meets the requirements for the transport.

Note: The following steps may be performed at the ISFSI site.

A. Storage Area.

7.4.1.2 Disconnect the overpressure system from the monitoring panel. Depressurize the overpressure tank and disconnect the tubing at the protective cover.

7.4.1.3 Position the cask transporter over the cask.

Note: The following 3 steps may not be necessary if preparation is done on the storage pad .

7.4.1.4 Engage the lifting arms and lift the cask to the designated lift height.

7.4.1.5 Move the cask to the loading area.

7.4.1.6 Lower the cask down onto the floor, disconnect the cask transporter and remove the transporter from the loading area.

B. Loading Area.

7.4.1.7 Remove the protective cover.

7.4.1.8 Tighten the lid bolts to 2050-2100 ft lb following the torquing sequence shown in Figure 7-1 using at least two passes.

7.4.1.9 Remove the overpressure tank assembly and the top neutron shield.

7.4.1.10 Inspect the sealing surface at the overpressure port. Check for defects in the seal contact area that may prevent proper sealing. Leak test the lid seals through the

overpressure port. The maximum acceptable cask seal leak rate is 1×10^{-5} ref cm^3/sec. The leak test shall be performed in accordance with ANSI N14.5[2].

7.4.1.11 Install the overpressure transport cover. The transport cover should have a new metallic seal. Torque the transport cover bolts to the value specified on Drawing 972-71-1.

7.4.1.12 Remove the storage shield ring if necessary.

7.4.1.13 Check the surface dose rates above the radial neutron shield.

7.4.1.14 Place the lower (bottom) trunnions on the rear trunnion supports of the transport frame.

7.4.1.15 Rotate the cask from the vertical to the horizontal position.

7.4.1.16 Install the lower (bottom) trunnion support caps and the tie-down strap, by torquing them to 100 ft-lb.

7.4.1.17 Check if the surface dose rates and the surface contamination levels are within the regulatory limits. Install an optional shield ring adjacent to the top of neutron shield, if required, based on dose limits.

7.4.1.18 Install the spacer on the front end of the cask. Then remove the spacer lifting eye bolts.

7.4.1.19 Install the front and the rear impact limiters onto the cask. Lubricate the attachment bolts with Never-Seez or an equivalent and torque to 200 ft-lb, diametrically in the first pass, and to 400 ft-lb in the final pass.

7.4.1.20 Install thirteen impact limiter attachment tie-rods between the front and the rear impact limiters.

7.4.1.21 Render the impact limiter lifting lugs inoperable by covering the lifting holes or installing a bolt inside the holes to prevent their inadvertent use.

7.4.1.22 Install security seal on one tie-rod and lock sleeve.

7.4.1.23 Install a transport enclosure.

7.4.1.24 Check the temperature on all accessible surfaces to make sure that it is <185 °F.

7.4.1.25 Perform a final radiation and contamination survey to satisfy the shield test requirements and to assure compliance with 10CFR71.47 and 71.87.

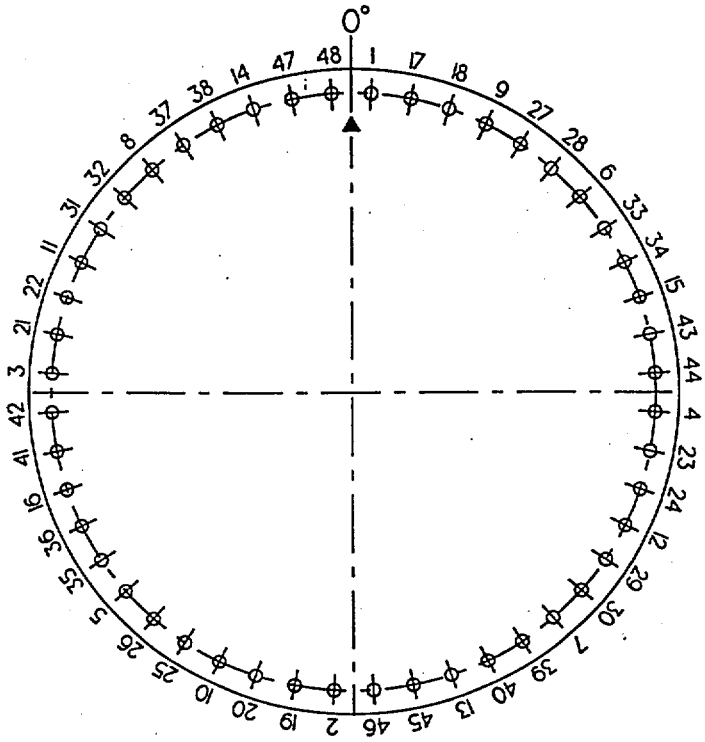
7.4.1.26 Apply appropriate DOT labels and Placards in accordance with 49CFR172. Prepare the final shipping documentation, and release the loaded cask for shipment.

7.5 REFERENCES

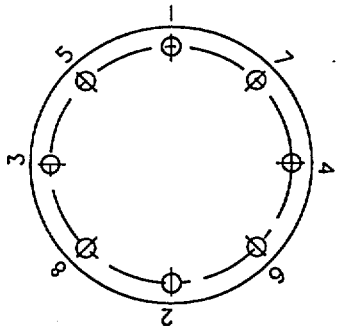
1. ANSI N14.6-1993, American National Standard for Radioactive Materials Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More

2. ANSI N14.5-1997, Leakage Tests on Packages for Shipment of Radioactive Materials

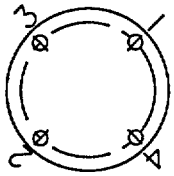
3. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, US Nuclear Regulatory Commission, July, 1980



LID



VENT/ DRAIN COVER



O.P. TRANSPORT COVER

FIGURE 7-1
TORQUING PATTERNS

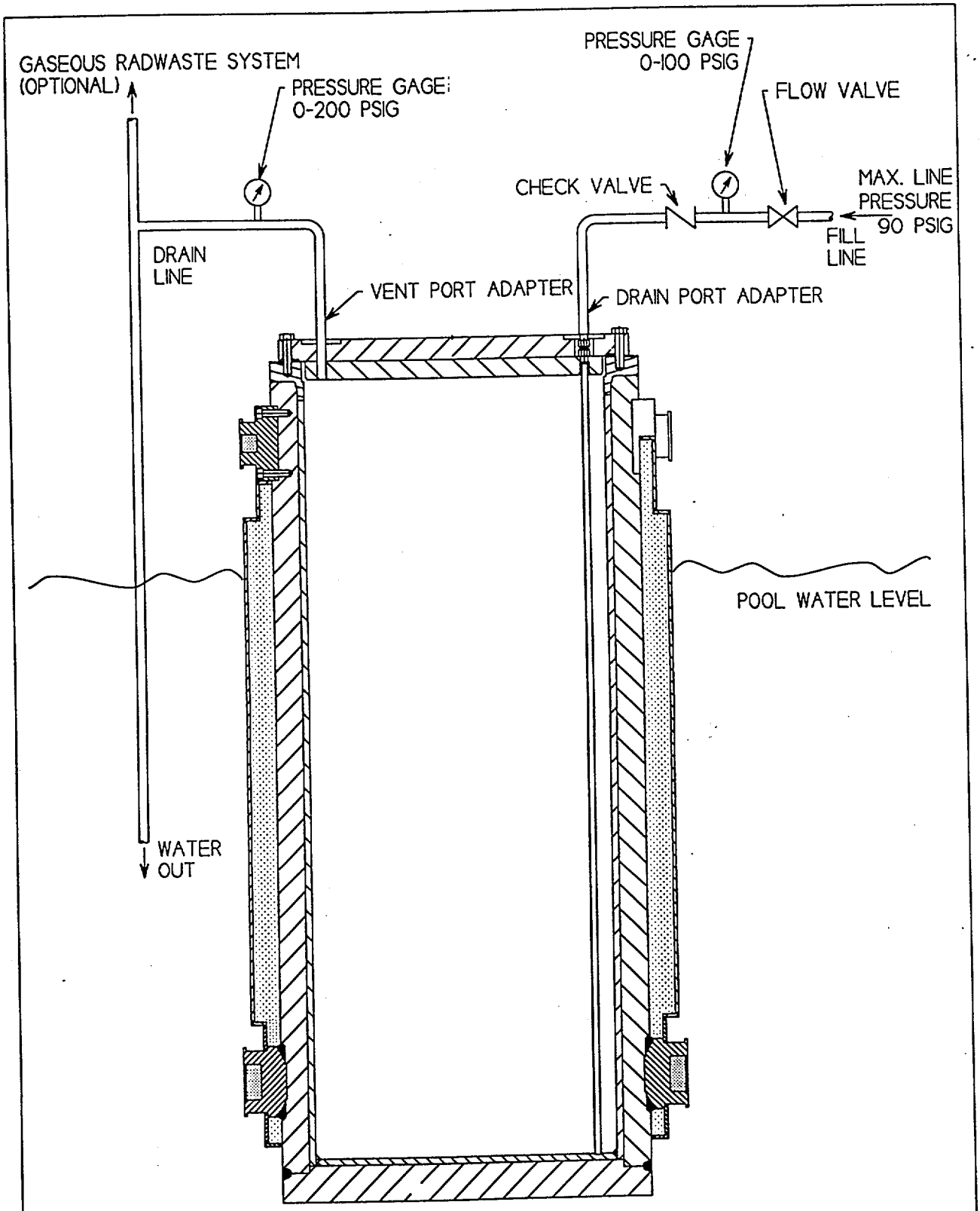


FIGURE 7-2
TYPICAL SETUP FOR FILLING
CASK WITH WATER

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TN 68 TRANSPORT PACKAGING

CHAPTER 8

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CHAPTER 8

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1 Acceptance Tests

The following reviews, inspections, and tests shall be performed on the TN-68 packaging prior to initial transport. Many of these tests will be performed at the Fabricator's facility prior to delivery of the cask to the utility for use. Tests will be performed in accordance with written procedures approved by Transnuclear Inc.

8.1.1 Visual Inspection

Visual inspections are performed at the Fabricator's facility to ensure that the packaging conforms to the drawings and specifications. The visual inspection includes verifying that all specified coatings are applied and the packaging is clean and free of cracks, pinholes, uncontrolled voids or other defects that could significantly reduce its effectiveness. Visual inspection is also performed to verify that the packaging has been fabricated and assembled in accordance with the drawings and other requirements specified in the SAR. Weld inspection is performed in accordance with the applicable ASME code sections. Dimensions and tolerances shown on the drawings provided in Chapter 1 are confirmed by measurements. Prior to shipping, the packaging will be inspected to ensure that it is in good physical condition. This inspection shall include verification that all accessible cask surfaces are free of grease, oil or other contaminants, and that all cask components are in an acceptable condition for use. The sealing surfaces on the flange, lid and covers are inspected to ensure that there are no gouges, cracks or scratches that could result in an unacceptable leakage.

8.1.2 Structural and Pressure Tests

The structural analyses performed on the packaging are presented in Chapter 2. To ensure that the packaging can perform its design function, the structural materials are chemically and physically tested to confirm that the required properties are met. All welding is performed using qualified processes and qualified personnel, according to the ASME Boiler and the Pressure Vessel Code[1]. Base materials and welds are examined in accordance with the ASME Boiler and Pressure Vessel code requirements. NDE requirements for welds are specified on the drawings provided in Chapter 1. All NDE is performed in accordance with written and approved procedures. The inspection personnel are qualified in accordance with SNT-TC-1A[2].

The confinement welds are designed, fabricated, tested and inspected, in accordance with ASME B&PV Code Subsection NB. Exceptions to the code taken regarding the containment vessel are described in Chapter 7. The basket is designed, fabricated, and inspected in accordance with the ASME B&PV Code Subsection NG. Exceptions to the code taken regarding the basket are described in Section 2.1.2.2. Welds of the noncontainment structure are inspected as per the NDE acceptance criteria of ASME B&PV Code, Subsection NF.

Pressure Tests

A pressure test is performed on the packaging assembly (containment vessel installed in the gamma shield shell) at a pressure of 125 psig. This is well above 1.5 times the maximum normal operating pressure of 18.5 psig. The test pressure is held for a minimum of 10 minutes. The test is performed in accordance with ASME B&PV Code, Section III, Subsection NB, Paragraph NB-6200 or NB-6300. The containment vessel is installed in the gamma shield shell during testing. All visible joints/surfaces are visually examined for possible leakage after application of the pressure. Temporary gaskets and seals may be used in place of the metallic seals during the test.

In addition, a bubble leak test is performed at a pressure equal to or greater than 4.5 psig, on the resin enclosure. The purpose of this test is to identify any potential leak passages in the enclosure welds. The bubble leak test pressure is set at 1.5 times the relief valve set pressure.

Load Tests

The lifting trunnions are fabricated and tested in accordance with ANSI N14.6[3] and are designed for nonredundant (single failure proof) lifting. A load test of three times the design lift load is applied to the trunnions for a period of ten minutes, to ensure that the trunnions can perform satisfactorily.

A force equal to 1.5 times the impact limiter weight will be applied to the lifting lugs of each limiter for a period of ten (10) minutes. At the conclusion of the test, the impact limiter lifting lugs (including welds) will be:

a) Visually examined for defects and permanent deformations.
b) Examined by the magnetic particle method for defects. Acceptance Standards will be in accordance with Article NF-5340 of Section III of the ASME Boiler and Pressure Vessel Code.

8.1.3 Leak Tests

Leakage tests are performed on the containment seals at the Fabricator's facility. These tests are usually performed using the helium mass spectrometer method. Alternative methods are acceptable, provided that the required sensitivity is achieved. The leak test is performed in accordance with ANSI N14.5[4]. The personnel performing the leakage test are qualified in accordance with SNT-TC-1A[2].

The permissible leakage rate for the containment boundary is less than or equal to 1×10^{-5} ref cm^3/sec. The sensitivity of the leakage test procedure is at least 5×10^{-6} ref cm^3/sec.

8.1.4 Components Tests

8.1.4.1 Valves, Rupture Discs, and Fluid Transport Devices

There are no valves in the packaging performing a safety related function. The TN-68 design incorporates quick-disconnect couplings for ease of draining and venting. However, these couplings do not form part of the containment boundary. They are covered by bolted closures with metallic seals. There is no required acceptance test for these devices.

8.1.4.2 Gaskets

The lid and all the other containment penetrations are sealed using double metallic seals. The inner seal forms part of the containment boundary. Metallic seals are not temperature sensitive, and are therefore tested at room temperature. Metallic seals of the same type as those to be used for transport are installed for the fabrication leak test, described in Section 8.1.3. The tested seals are replaced before loading the packaging for storage or for transport. Any time the lid or the port cover is removed, the seals are replaced and a leak test is performed prior to spent fuel shipment.

8.1.4.3 Impact Limiter Leakage Test

The following test will be performed, after all the seal welds are completed on the impact limiter, to verify that the impact limiter wood has been protected from any moisture exchange with the environment.

Pressurize each impact limiter container to a pressure of approximately 3 psig using helium. Test all the weld seams and surfaces for leakage in accordance with the methods of ANSI N14.5 and the requirements of Section V of the B&PV Code, using the helium sniffer method or soap bubble test.

8.1.4.4 Functional Tests

The following functional tests will be performed prior to first use of the cask. Generally these tests will be performed at the fabrication facility.

a. Installation and removal of the lid, penetration covers, and other fittings will be observed. Each component will be checked for difficulties in installation and removal. After removal, each component will be visually examined for indications of deformation, galling, ease of use, proper functioning, etc. Any defects will be corrected prior to acceptance of the cask.

b. After installation of the basket, each basket compartment will be checked by gaged to demonstrate that the fuel assemblies will fit in the basket.

8.1.5 Shielding Tests

The analyses performed to ensure the shielding integrity are presented in Chapter 5. The radial neutron shield is protected from damage or loss by the aluminum and steel enclosure. The neutron shield material is a proprietary, borated, reinforced polymer.

The primary function of the resin is to shield against neutrons, which is performed primarily by the hydrogen content in the resin. Both the neutrons and the capture gamma form a small component of the radial dose rate. The resin also provides some gamma shielding, which is a function of the overall resin density, and is not sensitive to composition.

The shielding performance of the resin can be verified adequately by chemical analysis and verification of density. Uniformity is assured by installation process control.

The following are acceptance values for density and chemical composition for the resin. The values used in the shielding calculations of Chapter 5 are included for comparison.

Chapter 5 values		Acceptance Testing Values		
Element	nominal wt %	Element	wt %	acceptance range (wt %)
H	5.05	H	5.05	-10 / +20
B	1.05	B	1.05	± 20

The minimum resin density in acceptance testing is 1.547 g/cm^3. Resin composition or density test results which fall outside of this range will be evaluated to ensure that the shielding regulatory dose limits are not exceeded.

Density testing will be performed on every mixed batch of resin. Chemical analysis will be made on the first batch mixed with a given set of components, and thereafter whenever a new lot of one of the major components is introduced. Major components are aluminum oxide, zinc borate and the polyester resin, which combined make up 92% of the resin by weight.

Qualification tests of the personnel and procedure used for mixing and pouring the polyester resin used for radial neutron shielding are performed. Qualification testing includes verification that the chemical composition and density are achieved, and the process is performed in such a manner as to prevent voids.

Tests are performed at loading to ensure that the radiation dose limits are not exceeded for each cask.

8.1.6 Neutron Absorber Tests

The analyses performed to ensure that the packagings are capable of performing their heat transfer function are presented in Chapter 3.

The neutron absorber plates serve no other function than neutron absorption and heat conduction;

the packaging safety analysis does not rely upon their mechanical strength. The basket structural components surround the plates on all sides. The radiation and temperature environment in the package is not sufficiently severe to damage the aluminum matrix that retains the boron-containing particles. To assure performance of the plates' safety-related function, the only critical variables that need to be verified are thermal conductivity and boron 10 areal density.

The boron 10 areal density of the neutron absorber plates shall be verified using approved procedures. There are two acceptable, neutron absorbing materials, as described in Section 6.3.2. These materials are subject to different types of tests and the associated minimum boron 10 contents, as follows.

A. Boron-Aluminum Alloy Using Enriched Boron, 90% B10 Credit

Material Description:

The neutron absorber consists of borated aluminum (containing about 1.7 wt% boron), which is isotopically enriched to 95 wt% B10. Because of the negligibly low solubility of boron in solid aluminum, the boron appears entirely as discrete second phase particles of AlB_2 in the aluminum matrix. The matrix is limited to any 1000 series aluminum , aluminum alloy 6063, or aluminum alloy 6351 so that no boron-containing phases other than AlB_2 are formed. Titanium may also be added to form TiB_2 particles, which are finer.

The 1.7 wt % boron converts to a nominal areal density of boron 10 as follows:
(2.69 g BAl/cm^3)x(1.7 wt% B)x(95 wt%B10)x(0.305 inch)x(2.54 cm/inch) = 0.0337 g B10/cm^2, which is intentionally slightly above the design minimum of 0.030 g B10/cm^2.

The boron-containing phase is introduced into the system, during the reaction of a proprietary boron-containing salt with the molten aluminum. The individual AlB_2 particles range in size from 5 to 10 microns. If titanium salt is added as well, the resulting TiB_2 particles will range in size from 1 to 5 microns. Both AlB_2 and TiB_2 are thermally stable at all temperatures below the melting point of the aluminum matrix. As such, their effect on the properties of the matrix aluminum alloy are those typically associated with a fine uniform dispersion of an inert, equiaxed, second phase.

The cast ingot may be rolled, extruded, or both, to the final plate dimensions.

Test Coupons:

Each neutron absorber plate is 10.4 inches wide by ~42, 55, and 69 inches long. Coupons, of the full width of the plate (10.4 inches), will be removed between each finished plate and at the ends of the "stock plate", as shown in Figure 8-1. The thermal conductivity coupon may be removed from one of the coupons for neutronic inspection, as shown in the figure. The minimum dimension of the coupon shall be as required for the neutron transmission measurements; 1 to 2 inches is adequate for the typical 1cm diameter neutron beam.

B. Boron Carbide/Aluminum Metal Matrix Composite (MMC), 75% B10 credit

Material Description:

The neutron absorber consists of a composite of aluminum with about 15 volume % boron carbide particulate reinforcement. The material is formed into a billet by powder metallurgical processes and either extruded, rolled, or both to final dimensions. The finished product has near-theoretical density and metallurgical bonding of the aluminum matrix particles. It is uniform from face to face, i.e., it is not a "sandwich" panel.

The 15 volume % boron carbide corresponds to a boron 10 areal density of $0.15(2.52\ g/cm^3\ B_4C)(0.782\ gB/gB_4C)(0.185\ g\ B10/gB)(0.305\ in)(2.54\ cm/in)=0.0424\ gB10/cm^2$, in excess of the specified minimum, $0.036\ g\ B10/cm^2$.

The process specifications for the material shall be subject to qualification testing to demonstrate that the process results in a material that

a) has a uniform distribution of boron carbide particles in an aluminum alloy with few or no voids, oxide-coated aluminum particles, B_4C fracturing, or B_4C/aluminum reaction products.
b) meets the requirements for boron 10 areal density and thermal conductivity, and
c) will be capable of performing its safety-related functions under the thermal and radiological environment of the TN-68 packaging.

The production of plates for use in the TN-68 packaging will be consistent with the process used to produce the qualification test material. "Lessons learned" in the qualification testing, or processing changes that will clearly lead to an improvement in the properties of the material may be incorporated into the production process, provided that they are reviewed and approved by Transnuclear, Inc.

Typical processing consists of:

a) Blending of boron carbide powder with aluminum alloy powder
b) Billet formed by cold isostatic pressing + sintering or by vacuum hot pressing
c) Billet extruded to intermediate or to final size
d) Hot roll, cold roll and flatten as required
e) Annealing

Test Coupons:

Each neutron absorber plate location in the basket is 10.4 inches wide by ~42, 55, and 69 inches long. Coupons will be removed between every other plate or at the end of the plate so that there is at least one coupon contiguous with each plate as shown in Figure 8-2. Coupons will generally be the full width of the plate. End coupons may be slightly shorter due to irregular shape at the end of the stock plate, and thermal conductivity coupons may be removed from the full width coupon as shown in the figure. The minimum dimension of the coupon shall be as required for acceptance test specimens; 1 to 2 inches is generally adequate.

Thermal Conductivity Testing - All Neutron Absorber Plates

The neutron absorber plate material will be tested by qualification tests to verify that the thermal conductivity equals or exceeds the values listed in Section 3.3. Material in production may be tested at only one temperature in that range to verify that the conductivity equals or exceeds the corresponding value in Section 3.3.

Testing may be by ASTM E1225[5], ASTM E1461[7], or an equivalent method, performed on specimens removed from coupons adjacent to the final plates. A lot is defined as all the plates and associated coupons made from a single casting or billet. Initially, tightened thermal conductivity inspection on one specimen per lot shall be performed. Normal inspection may be introduced, once every other lot, based upon the acceptance of 5 preceding lots. Reduced inspection (once every fourth lot) may be introduced if the preceding 5 inspected lots exceed the specified minimum by 10% or more. During normal or reduced inspection, a single measurement below the specified minimum will require returning to the tightened inspection. During reduced inspection, a single measurement less than 10% above the minimum specified will require returning to normal inspection.

If a coupon tests below the minimum specified thermal conductivity, the entire lot is considered non-conforming. A lot that is initially considered non-conforming may be accepted based upon additional sampling and testing of that lot.

Neutronic Acceptance Testing of Boron-Aluminum Alloy (Borated Aluminum):

Effective boron 10 content is verified by neutron transmission testing of the coupons. The transmission through the coupons is compared with transmission through calibrated standards composed of a homogeneous boron compound without other significant neutron absorbers, for example zirconium diboride or titanium diboride. These standards are paired with aluminum shims sized to match the scattering by aluminum in the neutron absorber plates. Provision shall be made so that the neutron transmission test is not always made in the same location on the coupon. For example, the transmission measurement may be made about 1/4 to 1/3 of the distance from the end of the coupon as shown in Figure 8-1. Thus, the random placement of the coupons in the test fixture results in testing at two locations across the plate width. The effective boron 10 content of each coupon, minus 3σ based on the number of neutrons counted for that coupon, must be ≥ 30 mg B10/cm^2.

In the event that a coupon fails the single neutron transmission measurement, four additional measurements will be made on the coupon, and the average of the 5 measurements, less 3σ based on the number of neutrons counted for each measurement on that coupon, must be ≥ 30 mg B10/cm^2.

Macroscopic uniformity of boron-10 distribution is verified by neutron radioscopy or radiography of the coupons. The acceptance criterion is that there be uniform luminance across the coupon. This inspection shall cover the entire coupon.

Normal sampling of coupons for neutron transmission measurements and radiography/radioscopy shall be 100%. Rejection of a given coupon shall result in rejection of the contiguous plate(s). Reduced sampling (50% - every other coupon) may be introduced based upon acceptance of all coupons in the first 25% of the lot. A rejection during reduced inspection will require a return to 100% inspection of the lot. A lot is defined as all the plates produced from a single casting.

Acceptance Testing, Visual - Boron Aluminum Alloy (Borated Aluminum)

The finished plates shall be visually examined to verify that they are free of cracks, porosity, blisters, or foreign inclusions. Such defects, where possible, are removed if it does not result in a dimensional non-conformance.

Justification for the Borated Aluminum Acceptance Test Requirements:

According to NUREG/CR-5661[6]

> "Limiting added neutron absorber material credit to 75% without comprehensive tests is based on concerns for potential 'streaming' of neutrons due to nonuniformities. It has been shown that boron carbide granules embedded in aluminum permit channeling of a beam of neutrons between the grains and reduce the effectiveness for neutron absorption."

Furthermore

> "A percentage of neutron absorber material greater than 75% may be considered in the analysis only if comprehensive tests, capable of verifying the presence *and uniformity* of the neutron absorber, are implemented." [Emphasis added]

The calculations in Chapter 6 use a boron areal density of 27 mg B10/cm^2, 90% of the minimum value required here of 30 mg B10/cm^2. This is justified by the following considerations.

a) The coupons for neutronic inspection are removed between each finished plate, and are the full width of the plate. As such, they are taken from locations that are truly representative of the finished product. Coupons are also removed at the ends of the "stock plate", where under thickness of the plates or defects propagated from the pre-roll ingot would be most likely. The use of representative coupons for inspection is analogous to the removal of specimens from structural materials for mechanical testing.

b) Neutron radiography/radioscopy of coupons across the full width of the plate will detect macroscopic non-uniformities in the boron 10 distribution such as could be introduced by the fabrication process. Such defects usually originate from the ingot and propagate in the direction of rolling or extrusion. For example, an ingot with a skin high in boron and a center depleted in boron will exhibit alternating bands of high and low boron concentration, which can be detected with radiography or radioscopy, parallel to the processing direction.

c) Neutron transmission measures effective boron 10 content directly. The term "effective" is used here because if there are any of the effects noted in NUREG/CR-5661, the neutron

transmission technique will measure not the physical boron 10 areal density, but a lower value. Thus, this technique by its nature screens out the microscopic non-uniformities which have been the source of the recommended 75% credit for boron 10 in criticality evaluations.

d) The use of neutron transmission and radiography/radioscopy satisfies the "and uniformity" requirement emphasized in NUREG/CR-5661 on both the microscopic and macroscopic scales.

e) The normal inspection for neutronic tests is 100%. The provisions for transition to reduced inspection and for return to normal inspection require that each new lot begin with normal inspection. This is more restrictive than the guidelines of ANSI/ASQC Z1.4[9], which allows reduced inspection to continue from lot to lot.

f) The recommendations of NUREG/CR-5661 are based upon testing of a neutron absorber with boron carbide particles averaging 85 microns. The boride particles in the borated aluminum are much finer (5-10 microns). Both the manufacturing process and the neutron radioscopy assure that they are uniformly distributed. For a given degree of uniformity, fine particles will be less subject to neutron streaming than coarse particles. Furthermore, because the material reviewed in the NUREG is a sandwich panel, the thickness of the boron carbide containing center could not be directly verified by thickness measurement. The alloy specified here is uniform throughout its thickness.

g) Visual inspection of the plates verifies that there are no gross mechanical defects that could compromise the neutron absorber's ability to remain intact and in position in the basket.

Neutronic Acceptance Testing - Boron Carbide/Aluminum Metal Matrix Composite

Effective boron 10 content is verified by neutron transmission testing of these coupons, or by chemical, spectrometric, and dimensional inspection.

In the first method, the transmission through the coupons is compared with transmission through calibrated standards composed of a homogeneous boron compound without other significant neutron absorbers, for example zirconium diboride or titanium diboride. These standards are paired with aluminum shims sized to match the scattering by aluminum in the neutron absorber plates. Provision shall be made so that the neutron transmission test is not always made in the same location on the coupon. For example, the transmission measurement may be made about 1/4 to 1/3 of the distance from the end of the coupon as shown in Figure 8.1-2. Thus, the random placement of the coupons in the test fixture results in testing at two locations across the plate width. The effective boron 10 content of each coupon, minus 3σ based on the number of neutrons counted for that coupon, must be ≥ 36 mg B10/cm^2.

In the event that a coupon fails the single neutron transmission measurement, four additional measurements may be made on the coupon, and the average of the 5 measurements, less 3σ based on the number of neutrons counted for each measurement on that coupon, must be ≥ 36 mg B10/cm^2.

In the second method, the grams B10 per gram of total boron and the grams of total boron per grams of boron carbide are determined by spectrometric and chemical analysis of each lot of boron carbide feed powder (ASTM-C791[8] or equal). The grams of boron carbide per gram of finished composite is then determined by chemical analysis of a specimen selected from a random location on the finished coupon (ASTM D-3553[10] or equal). These three values are then multiplied together, with the composite density, and the minimum allowable plate thickness:

$$(g\ B10/g\ B)*(g\ B/g\ B_4C)*(g\ B_4C/g\ MMC)*(g\ MMC/cm^3)*(min\ thickness,\ cm) = g\ B10/cm^2$$

The value for each coupon must be $\geq$ 36 mg B10/cm^2.

A lot is defined as all the plate and coupons made from a single billet. Inspection of coupons for B10 areal density measurements shall begin with tightened inspection (100% of coupons) for the first three lots. Inspection may be switched to normal (50%), if all coupons in the three preceding lots demonstrate a boron 10 areal density of greater than 36 mg B10/cm^2. Reduced sampling (25%) may be introduced, if all coupons from three preceding inspected lots demonstrate a boron 10 areal density of greater than 39.6 mg B10/cm^2. During normal or reduced inspection, a coupon with < 36 mg B10/cm^2 will require a return to the tightened inspection. A coupon below 39.6 mg B10/cm^2 during reduced inspection will require a return to normal inspection.

Measurement of <36 mg B10/cm^2 on a given coupon shall result in rejection of the contiguous plate(s).

Acceptance Testing, Visual Boron Carbide/Aluminum Metal Matrix Composite (MMC):

The finished plates shall be visually examined to verify that they are free of cracks, porosity, blisters, or foreign inclusions. Removal of such defects is permitted, where possible, if the removal does not result in a dimensional non-conformance.

Justification for Acceptance Test Requirements, Metal Matrix Composite

According to NUREG/CR-5661[7]

> " For each calculational model, the atom density of any neutron absorber...should be limited to 75% of the minimum neutron absorber content specified in the application. This minimum neutron absorber content should be verified by chemical analysis, neutron transmission measurements, or other acceptable methods. A percentage of neutron absorber material greater than 75% may be considered in the analysis only if comprehensive tests, capable of verifying the presence and uniformity of the neutron absorber, are implemented."
>
> "...Limiting added neutron absorber material credit to 75% without comprehensive tests is based on concerns for potential 'streaming' of neutrons due to nonuniformities. It has been shown that boron carbide granules embedded in aluminum permit channeling of a

beam of neutrons between the grains and reduce the effectiveness for neutron absorption."

The calculations in Chapter 6 use a boron areal density of 27 mg B10/cm^2. That is 75% of the minimum value required here, 36 mg B10/cm^2. Based on the recommendations above, because only 75% credit is used, comprehensive testing for presence and uniformity of boron 10 is not necessary.

Other considerations are:

a) The coupons for neutronic inspection are removed between every other finished plate, and are generally the full width of the plate. As such, they are taken from locations that are truly representative of the finished product, where every plate is represented by a contiguous coupon. The use of representative coupons for inspection is analogous to the removal of specimens from structural materials for mechanical testing.

b) Uniformity of B10 distribution must be demonstrated by the qualification testing of the material.

c) Initial inspection for B10 areal density tests is 100% of the coupons. The provisions for transition to normal and reduced inspection are consistent with the guidelines of ANSI/ASQC Z1.4[9]. The provisions here, by starting with tightened inspection and by requiring a margin of 10% over the minimum for reduced inspection, are more restrictive than Z1.4.

d) The recommendations of NUREG/CR-5661 are based upon testing of a neutron absorber with boron carbide particles on the order of 80-100 microns. The boron carbide particles in a typical metal matrix composite are much finer (1-25 microns). The powder metal manufacturing process controls and the qualification testing assure that they are uniformly distributed. For a given degree of uniformity, fine particles will be less subject to neutron streaming than coarse particles. Furthermore, because the material reviewed in the NUREG was a sandwich panel, the thickness of the boron carbide containing center could not be directly verified by thickness measurement. The metal matrix composite specified here is uniform throughout its thickness.

e) Visual inspection of the plates verifies that there are no gross mechanical defects that could compromise the neutron absorber's ability to remain intact and in position in the basket.

8.1.7 Thermal Tests

The thermal evaluation presented in Chapter 3 is based on conservative properties taken from reliable sources. In addition, the evaluation presented does not result in temperatures being close to allowable limits. Therefore thermal tests of the TN-68 cask are not required. However, the

thermal conductivity of the neutron absorber plates is tested. This testing is described in Section 8.1.6.

8.2 Maintenance Program

8.2.1 Structural and Pressure Tests

Within 14 months prior to any lift of a TN-68 transport package, the lifting (top) trunnions shall be subject to either of the following:

(1) A test load equal to 300% of the maximum service load per ANSI N14.6[3], paragraph 7.3.1(a). After sustaining the test load for a period of not less than 10 minutes, critical areas, including major load-bearing welds, shall be subjected to visual inspection for defects, and all components shall be inspected for permanent deformation.

(2) Dimensional testing, visual inspection and nondestructive examination of critical areas of the trunnions including the bearing surfaces in accordance with Paragraph 6.3.1(b) of ANSI N14.6. The examination of the trunnion bolts in accordance with ANSI N14.6 will not be performed if the cask is taken directly from storage to be used for transport. This is justified since the cask will only be lifted a few times. The examination will be performed for subsequent shipments.

8.2.2 Leak Tests

After lid or port cover removal, the affected metallic containment seals shall be replaced and leak tested prior to spent fuel shipment to show a leak rate less than 1×10^{-5} ref cm^3/sec. The sensitivity of the leakage test procedure is at least 5×10^{-6} ref cm^3/sec. These tests are usually performed using the helium mass spectrometer method. Alternative methods are acceptable, provided that the required sensitivity is achieved. The leak test is performed in accordance with ANSI N14.5[4]. Because the seals are not to be reused for spent fuel shipment, the preshipment leak tests may be used to fulfil the ANSI N14.5 requirements for maintenance and periodic testing.

No leak tests are required prior to shipment of an empty TN-68 packaging.

8.2.3 Subsystem Maintenance

8.2.3.1 Fasteners

The lid bolts, vent, drain and overpressure transport cover bolts shall be inspected after each use, and annually, for deformed or stripped threads. Damaged parts shall be evaluated for continued use and replace as required. At a minimum, the lid bolts, vent, drain and overpressure transport cover bolts shall be replaced at least once per one hundred fifty (150) shipments. If the cask is used in a storage mode, the fasteners need not be inspected or replaced until after the first shipment.

8.2.3.2 Impact Limiters

Impact limiters will be leak tested once every 5 years to ensure that water has not entered the impact limiters. If the leak test indicates that the impact limiters have a leak, a humidity test will be performed to verify that there is no free water in the impact limiters.

8.2.4 Valves, Rupture Discs, and Gaskets on Containment Vessel

If a port cover or the lid is removed, the seals are replaced prior to spent fuel transport. At the time the TN-68 is first converted from storage to transport use, it is most likely that the fuel that has been in storage will remain in the TN-68 for transport, and therefore, the lid will not be removed. In this case, the lid seal and seals of penetrations which have not been opened will not be replaced. The seals will be leak tested after retorquing the bolts in accordance with Section 7.4.

The metallic seals may be reused for transport of an empty TN-68 packaging.

The leak test port (the overpressure port in the storage configuration) is closed by the overpressure transport cover with a single metallic seal. This flange and seal are not part of the containment boundary. The quick connect couplings in the vent and drain ports is not part of the containment boundary.

There are no valves or rupture discs on the TN-68 packaging containment.

8.2.5 Shielding

There are no periodic tests or inspections required for the TN-68 shielding. Radiation surveys will be performed of the package exterior to ensure that the limits specified in 10 CFR 71.47 are met prior to each shipment.

8.2.6 Thermal

There are no periodic tests or inspections required for the TN-68 heat transfer components.

8.3 References

1. ASME Boiler and Pressure Vessel Code, Section III, 1995 Edition including 1996 addenda.

2. SNT-TC-1A, "American Society for Nondestructive Testing, Personnel Qualification and Certification in Nondestructive Testing," 1992.

3. ANSI N14.6-1993, "American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials," New York

4. ANSI N14.5-1997, "American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials"

5. ASTM E1225, Thermal Conductivity of Solids by Means of the Guarded-Comparative-Longitudinal Heat Flow Technique

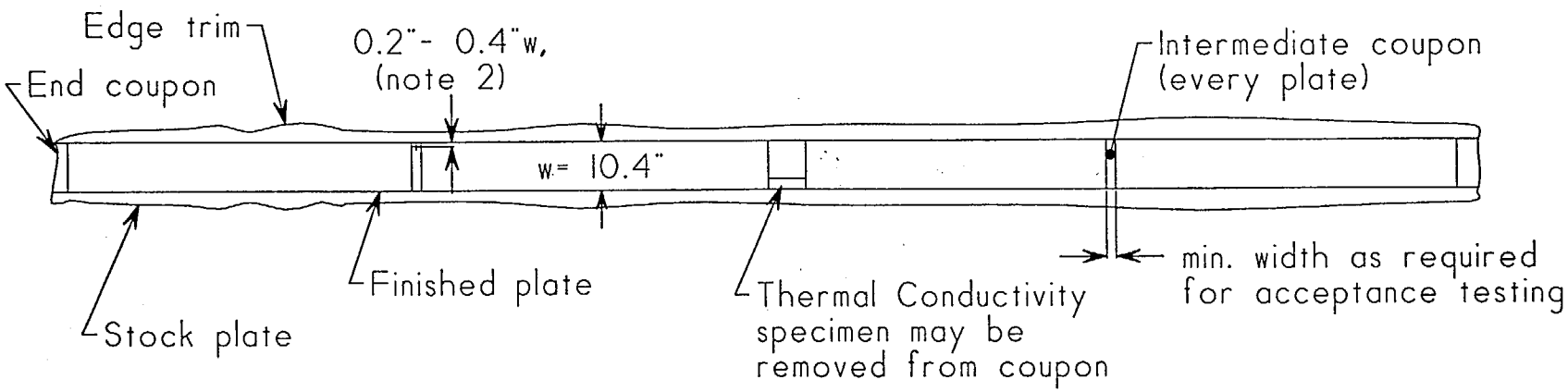
6. NUREG/CR-5661, Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packages, 1997

7. ASTM E1461, Thermal Diffusivity of Solids by the Flash Method

8. ASTM C 791, Standard Methods for Chemical, Mass Spectrometric, and Spectrochemical Analysis of Nuclear-Grade Boron Carbide

9. ANSI/ASQC Z1.4-1993, Sampling Procedures and Tables for Inspection by Attributes

10. ASTM D 3553, Fiber Content by Digestion of Reinforced Metal Matrix Composites



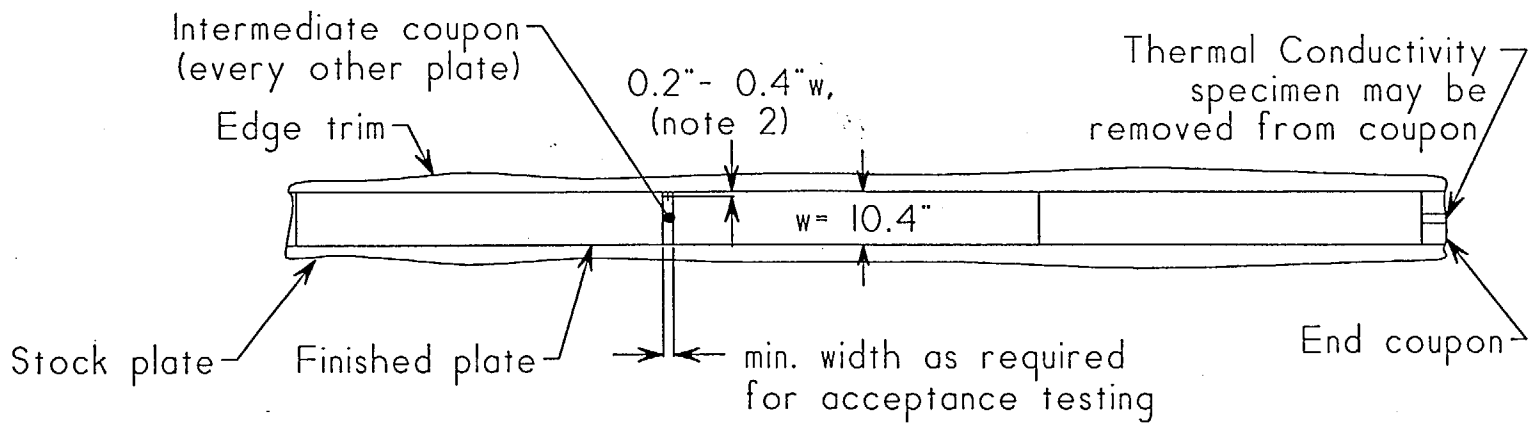
Notes:

1. Thermal conductivity test specimens may be taken from intermediate or end coupon, or from edge trim.

2. Neutron transmission measurements, at this location taken randomly from alternate edges.

3. Radiography/ Radioscopy of full coupon: if thermal conductivity specimen is removed from coupon, radiography of balance of coupon is acceptable.

FIGURE 8-1
TN-68 NEUTRON ABSORBER PLATE COUPONS
BORATED ALUMINUM



Notes:

1. Thermal conductivity test specimens may be taken from intermediate or end coupon, or from edge trim.

2. Neutron transmission measurements, if applicable at this location, taken randomly from alternate edges: chemistry specimen, if applicable from random location on coupon.

FIGURE 8-2
TN-68 NEUTRON ABSORBER PLATE COUPONS
MMC