APPENDIX 2.10.5
LEAD POURING PROCEDURE REQUIREMENTS

2.10.5.1 SCOPE

2.10.5.1.1 The purpose of this document is to identify basic material and procedural requirements for the lead pouring special process to assure that the completed FSV packaging assembly meets the design specifications.

The Fabricator shall submit a lead pour procedure addressing in detail all equipment, controls, materials and step by step sequence of the lead pour operation to the Purchaser for approval.

Lead pouring shall be made through at least four pour holes through the bottom plate on the cask. The pour holes shall be equally spaced and be at least 2.0 inches in diameter.

2.10.5.1.2 Fabricator shall take all necessary action required to meet the following major requirements:

2.10.5.1.2.1 Lead pour shall be free of voids, gaps or impurities which would reduce the effective lead shielding below minimum thickness specified on TN Drawings 1090-SAR-1 through 1090-SAR-4.

2.10.5.1.2.2 Lead pour shall be done with the cask in an inverted position (flange down). The cooldown after lead pour shall be sufficiently slow that lead creep can occur, minimizing compressive loading of the inner containment cylinder.

2.10.5.1.2.3 The weld of the outer shell to the bottom shall be made after the lead pour to allow the inner and outer shells to expand and contract at different rates during the lead pouring and cooldown process. The weld area shall be suitably protected

during the lead pouring operation to prevent lead from leaking into it.

2.10.5.1.2.4 Radial alignment fixtures which retain the concentricity of the two shells may be used during the lead pour.

2.10.5.2 MATERIAL REQUIREMENTS

2.10.5.2.1 Lead material shall be in accordance with ASTM B29 Chemical Grade Pig Lead specification.

2.10.5.2.2 The lead material shall be packaged and stored with proper precautions to prevent contamination.

2.10.5.2.3 Certified Mill Test Reports of lead material used shall be maintained on file and copies submitted to Purchaser for review.

2.10.5.3 EQUIPMENT REQUIREMENTS

2.10.5.3.1 Sufficient melting capacity shall be provided to allow a continuous lead pour. No more than 85% of each kettle shall be poured to insure dross free metal transfer.

2.10.5.3.2 Necessary equipment shall be provided for pouring from the bottom of the kettle.

2.10.5.3.3 Kettles shall be set at a level such that a positive gravity feed from the kettle to the cask can be maintained. Lead pouring pipes shall be maintained at the necessary temperature level to prevent freezing of the lead during the pour.

2.10.5.3.4 The necessary ladles and skimming equipment shall be maintained and handled in such a manner that the lead contamination will be kept to a minimum while adding lead during the lead shrinkage cycle.

2.10.5.3.5 The temperature of the kettle shall be monitored by use of a thermocouple or other suitable method. The temperature will be taken and recorded prior to the start of the lead pour to verify that the lead pour temperature is within the proper range.

2.10.5.3.6 Cask temperature shall be measured by thermocouple contact pyrometers and/or temple-sticks (mercury free).

2.10.5.3.7 All temperature control devices shall be within the calibration due date.

2.10.5.4 PRE-POUR INSPECTION

2.10.5.4.1 Surfaces of the cask to be in contact with lead shall be free of loose mill scale, weld slag, liquids (water, oil, etc.) and other foreign materials.

2.10.5.4.2 The exterior of the unit shall be visually inspected for obvious variances that may be important to note prior to lead pour. Any major discrepancy will be reported to the Purchaser and Fabricator prior to the lead pour for their disposition.

2.10.5.4.3 Measurements of the inner diameter of the inner containment cylinder shall be recorded at sufficient locations before and after lead pouring to verify that it has not buckled during cooldown.

2.10.5.4.4 Dimensional inspection shall be conducted to provide data necessary in determining the variances in the lead cavity.

2.10.5.4.5 Bracing of the unit will be accomplished as required to provide stable and level conditions for the lead pour. If welded on bracing is needed, approval shall be obtained from the Purchaser prior to welding.

2.10.5.4.6 Pour holes should be located where there is no possible chance of air entrapment. Where there is a chance of possible air entrapment, steps shall be taken to preclude development of any voids.

2.10.5.4.7 The weld preparation for the final weld of the outer shell to the bottom plate shall be completed and suitably protected.

2.10.5.5 PREHEATING OF ASSEMBLY

2.10.5.5.1 Preheat monitoring and control shall be accomplished using thermocouples, contact pyrometers or mercury free temple-sticks.

2.10.5.5.2 Preheating shall be done with gas fire oven, rings or other devices which achieve uniform heat application and control.

2.10.5.5.3 Preheat temperatures for stainless steel components should be in the 850°-900° F range. Temperature ranges and limits shall be detailed in the Fabricator's procedure.

2.10.5.5.4 When preheating is applied, the assembly shall be brought to temperature gradually; keeping a temperature spread of approximately 50°F to avoid significant thermal stresses.

2.10.5.6 FILLING THE CONTAINER

2.10.5.6.1 The lead temperature range at the time of pouring shall be 750-800°F.

2.10.5.6.2 The lead flow rate should be adjusted for continuous flow in order to fill the container as rapidly as possible.

2.10.5.6.3 The lead filling pipes shall be located to:

a) minimize splashing as the lead enters the container;

b) prevent the molten lead from impinging on the container walls, thereby reducing the probability of distortion from hot spots.

2.10.5.7 LEAD COOLING

2.10.5.7.1 The molten lead in the container must be cooled in a manner that assures only one solidification front. The solidification process must start at the bottom. This front will be controlled by:

a) monitoring the temperature profile of the cask.

b) probing the unit with a long rod and noting the location of the solidification front.

2.10.5.7.2 At all times emphasis shall be kept on keeping the surface free of dross and the pour hole from freezing over.

2.10.5.7.3 Molten lead at a minimum temperature of 750°F shall be added to the container as the shrinkage occurs. Direct heat shall be carefully applied in the solidification process to insure that the fill area is free of voids.

2.10.5.7.4 Dross shall be removed from the fill surface and the lead surface fire-polished to finish the pouring procedure.

2.10.5.7.5 The cooldown rate after lead solidification shall be controlled so that it reaches room temperature gradually.

2.10.5.8 CLOSING POUR HOLES

2.10.5.8.1 Lead in pour holes shall be fire polished and level at a depth of approx. 1-1/4 inch from the outer surface.

2.10.5.8.2 Remove any lead from exterior surface of plate, and wash down with detergent.

2.10.5.8.3 Install an ASME SA-240 Type 304 stainless steel plug into the pour hole such that it covers the lead and is flush with the outside surface. The plug shall fit snugly in the pour hole.

2.10.5.8.4 Weld the plug to the bottom plate with 3/4 inch welds.

2.10.5.8.5 Inspect the welds by the liquid penetrant method and the visual method.

2.10.5.9 FINAL WELDING OF OUTER SHELL TO BOTTOM PLATE

2.10.5.9.1 Thoroughly clean weld prep area.

2.10.5.9.2 Measures shall be taken to minimize localized lead melting during the welding process.

2.10.5.9.3 After completion of the weld, the weld shall be inspected by the Ultrasonic and Liquid Penetrant Methods.

2.10.5.10 QUALITY ASSURANCE REQUIREMENTS

2.10.5.10.1 Each purchase lot and melt/heat of pig lead per ASTM B29 Chemical Grade shall have CERTIFIED TEST REPORTS. These shall contain the signature and title of the authorized representative of the agency performing the

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THERMAL EVALUATION

3.1 DISCUSSION

The TN-FSV is designed to passively reject payload decay heat under normal conditions of transport and hypothetical accident conditions while maintaining appropriate packaging temperatures and pressures within specified limits. An evaluation of the TN-FSV thermal performance is presented in this chapter. Objectives of the thermal analyses performed for this evaluation are:

- Determination of packaging temperatures with respect to containment system material limits;
- Determination of packaging component temperature gradients to support calculation of thermal stresses;
- Determination of the cask cavity temperature to support containment pressurization calculations;
- Determination of the fuel storage container temperature;
- Determination of the maximum fuel temperature.

The packaging components considered in the thermal evaluation are the cask body, the lid, the trunnions, the thermal shield and impact limiters. The body consists of cylindrical stainless steel shells which surround a lead shell. The lid and the bottom are fabricated from stainless steel plate material as described in Chapter 1.0. Temperatures calculated for the components in the packaging support thermal stress calculations and permit selection of appropriate temperature dependent mechanical properties used in the structural analyses. Temperatures are also calculated to demonstrate that specified limits for seal materials are not exceeded.

The design of the stainless steel encased wooden impact limiters is described in Chapter One. There are no temperature limits specified for the impact limiters, however, 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.

Several thermal design criteria have been established for the TN-FSV.

- Containment of radioactive material is a major design requirement for the TN-FSV. Therefore, seal temperatures must be maintained within specified limits to satisfy the required containment criteria (Chapter Four) under normal and accident conditions. A maximum equilibrium seal temperature of 300°F is set for the silicone O-rings under normal and accident conditions. This limit applies to all containment boundary seals used in the cask closure lid and containment penetrations.

- In accordance with 10CFR71.43(g) (Ref. 3-1) the maximum temperature of accessible package surfaces in the shade is limited to 180°F for exclusive use shipments.

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

- Maximum temperatures of the packaging must not adversely affect the shielding function.

In general, all the thermal criteria are associated with maximum temperatures. The ability of the containment system structural materials to function properly under lowest service temperature conditions is discussed in Section 2.6.

The TN-FSV is analyzed based on a maximum heat load of 360 watts. The analysis considers the decay flux to be uniformly distributed on the cavity surfaces radially adjacent to the fuel blocks. The thermal evaluation concludes that with this total heat load and distribution all design criteria are satisfied.

A description of the detailed analysis performed for normal conditions of transport is provided in Section 3.4 and for hypothetical accident conditions, in Section 3.5. A summary of results from these analyses is shown in Table 3-1.

For the normal conditions of transport the temperatures of the inner shell and lead are approximately at the same temperature. Table 3-1 shows that the lead and inner shell are at 166 and 167°F respectively. This is due to the fact that the heat load is low and therefore the gradients through the cask body are small.

For the thermal accident condition the results of the analysis are seen in Figure 3-9. This shows that the lead and inner shell temperatures are no more than 20°F to 30°F apart during the entire thermal event.

3.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

Table 3-2 lists the thermal properties of materials used in the thermal analyses. The values are listed as given in the corresponding references. The analysis uses interpolated values 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 packaging are considered. The external surfaces of the TN-FSV are assumed to have an emissivity of 0.85, a typical value for weathered stainless steel surfaces (Reference 3-3). For solar absorptivity, a conservative value of 0.85 is used.

TABLE 3-1

SUMMARY OF RESULTS

Normal Conditions of Transport	
Maximum Temperatures	
Outer Surface (thermal shield)	161°F
Outer Shell	165°F
Lead Shield	166°F
Inner Shell/Cavity Wall	167°F
Fuel Storage Container	183°F
Fuel	213°F
Lid/Drain Seals	166°F
Average Cavity Gas Temperature	175°F
Maximum Outer Surface Temperature Without Insolation	108°F
Accident Conditions	
Maximum Transient Temperatures	
Outer Surface (thermal shield)	1316°F
Outer Shell	283°F
Lead (behind trunnions)	478°F
Inner Shell/Cavity Wall	245°F
Fuel Storage Container	261°F
Fuel	291°F
Cavity "Cold Wall" (Peak)	<200°F
Lid/Drain Seals	<200°F
Average Cavity Gas Temperature (Peak)	253°F

TABLE 3-2

MATERIAL THERMAL PROPERTIES

	STAINLESS STEEL[3-2]	LEAD[3-3]	AIR[3-3]
Used in	body, lid thermal shield	body	thermal shield
Density ($lb/in.^3$)	0.29	0.41	not used
Specific heat (Btu/lbm-°F)	0.114 @ 100°F 0.122 @ 300°F 0.130 @ 600°F 0.134 @ 900°F 0.140 @ 1350°F	0.031	not used
Thermal Conductivity (Btu/hr-ft-°F)	8.7 @ 100°F 9.8 @ 300°F 11.3 @ 600°F 12.7 @ 900°F 14.7 @ 1350°F	20.5 @ 32°F 19.5 @ 261°F 18.0 @ 621°F	0.0145 @ 68°F 0.0214 @ 392°F 0.0280 @ 752°F 0.0312 @ 932°F 0.0440 @ 1832°F
Emissivity	0.85 (Weathered)		

3.3 TECHNICAL SPECIFICATIONS OF COMPONENTS

The only packaging component for which a thermal technical specification is necessary are the seals. The seals used in the packaging should be made of silicone equivalent to Parker O-ring compound S604-70. The seals will have a minimum and maximum temperature rating of -40°F and 300°F respectively.

3.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF TRANSPORT

3.4.1 Thermal Model

The thermal evaluation of the TN-FSV packaging is performed using the ANSYS computer program (Reference 3-4). ANSYS is a large scale, general purpose finite element computer code which can be used for steady state or transient thermal analyses.

3.4.1.1 Cask Body Analytical Model

The geometry and dimensions used for the cask body thermal model of the TN-FSV are shown in Figure 3-1. The model is a two dimensional axisymmetric representation of the packaging and includes the body, lid, bottom and the thermal shield. The model allows heat transfer in the radial and axial directions.

The thermal model uses the two dimensional isoparametric thermal solid element, STIF55, to simulate conduction in the packaging components and the air gap in the thermal shield. Radiation and natural convection heat transfer across the air is conservatively neglected. In addition, the conduction path provided by the wire in the air gap is neglected. Because of the low decay heat load, the temperature drop due to the contact resistance between the stainless steel shells and the lead shielding has an insignificant effect on temperature distribution and therefore is neglected. Further, good contact is expected even though it is assumed the lead is not bonded to the steel. Figure 3-2 is the ANSYS finite element plot of the model.

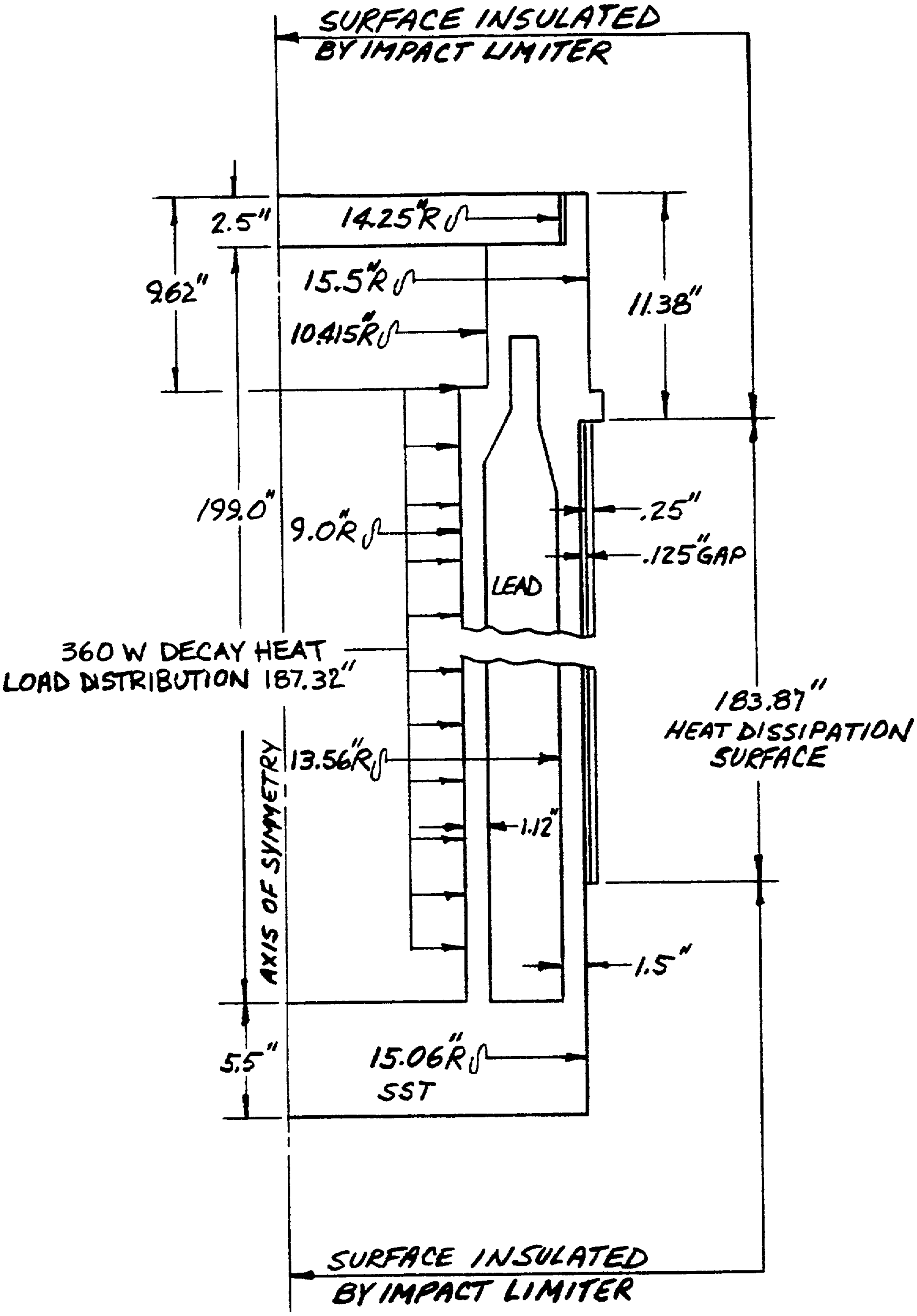

FIGURE 3-1

THERMAL MODEL FOR TN-FSV CASK

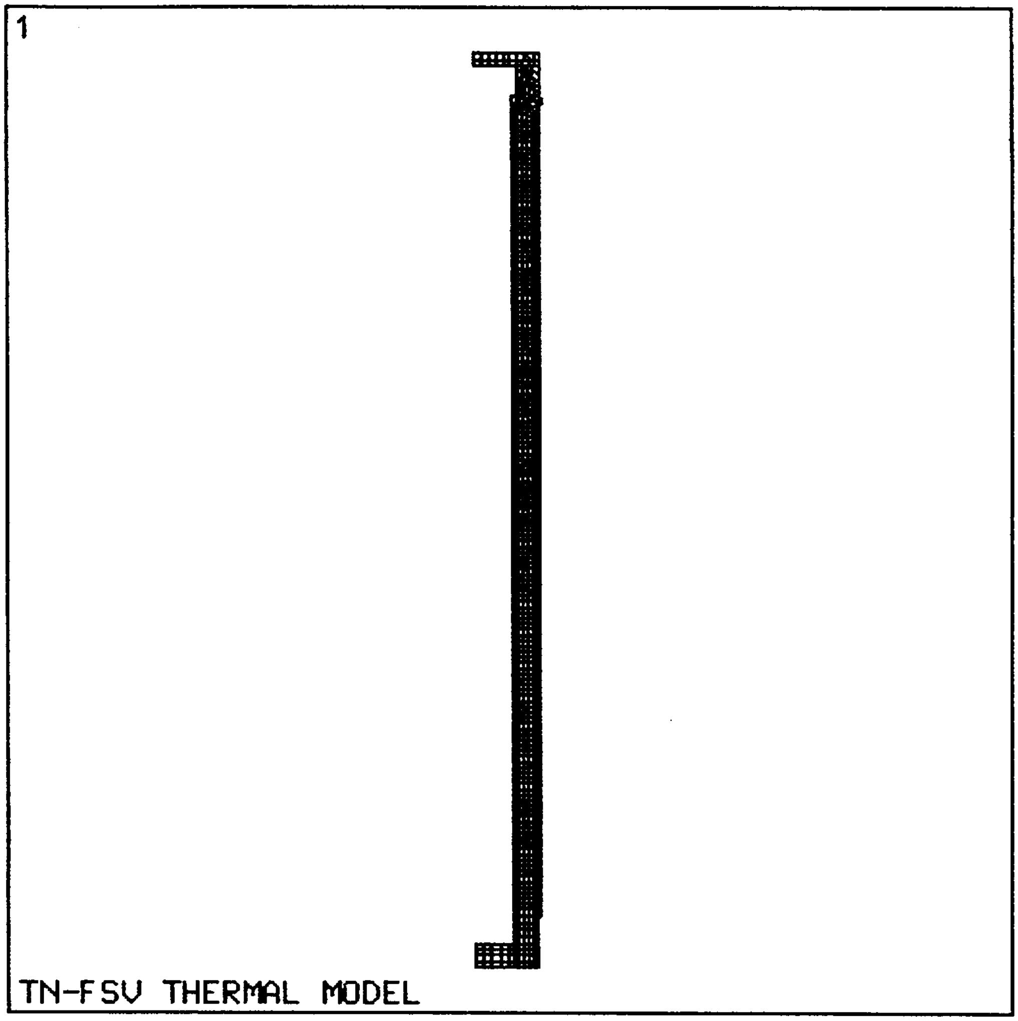

FIGURE 3-2

ANSYS FINITE ELEMENT MODEL FOR THE TN-FSV CASK

The contents of the packaging (the Fort St. Vrain fuel in the fuel storage container) are not included in the model. The temperature calculation for the contents is performed in Section 3.4.2.

The heat flux is applied along the cavity wall surfaces (adjacent to the 187.32 in. axial length of the fuel). The total heat load is 360 watts. Insolation on a curved surface corresponding to 1475 Btu/ft^2 for 12 hours a day is included. For conservatism, the maximum solar heat flux is applied to the surfaces exposed to the environment rather than the average value over a 24 hour period.

Heat dissipation from the surfaces exposed to the environment of the model is by radiation and natural convection. An emissivity of 0.85 and a solar absorptivity of 0.85 are used in the analysis. The total heat transfer coefficient, H_t, for heat dissipation by radiation and natural convention is:

$$H_t = H_c + H_r$$

where,

H_c = Natural convection coefficient
= $0.18\ (T_s - T_a)^{1/3}$ Btu/hr-ft^2-°F for horizontal cylindrical surfaces[3-5]

H_r = Radiation heat transfer coefficient
= $(0.1714E{-8})(F_{12})[(T_s+460)^4-(T_a+460)^4]/(T_s-T_a)$
= Btu/hr-ft^2-°F

F_{12} = Gray body shape factor

= Outer surface emissivity for large surroundings in comparison with surface area[3-5]

= 0.85

T_a = Ambient temperature = 100°F

T_s = Surface temperature of the packaging, °F

Table 3-3 gives the values of H_t as a function of T_s for horizontal cylindrical surfaces in a 100°F ambient.

3.4.1.2 Thermal Test Model

The low decay heat load and the resulting low temperatures preclude the necessity for a thermal test of the TN-FSV packaging.

3.4.2 Maximum Temperatures

The evaluation of maximum temperatures includes the 360 watt payload decay heat, 100°F ambient temperature and insolation. A separate case is analyzed with no insolation to determine the maximum accessible surface temperature in the shade as specified by 10CFR71.43(g). The resulting calculated packaging temperatures are shown in Table 3-4. The temperature distribution in the model is presented in Figure 3-3. The maximum body (cavity wall) temperature computed is 167°F.

TABLE 3-3

CONVECTION AND RADIATION HEAT TRANSFER COEFFICIENTS FOR NORMAL CONDITIONS OF TRANSPORT

Surface Temperature (°F)	Heat Transfer Coefficient* (Btu/hr - ft^2 - °F)		
	Convection	Radiation	Total
T_s	H_c	H_r	H_t
101	0.1800	1.0244	1.2044
110	0.3878	1.0493	1.4371
120	0.4886	1.0777	1.5663
130	0.5593	1.1067	1.6660
140	0.6156	1.1364	1.7520
150	0.6631	1.1669	1.8300
160	0.7047	1.1980	1.9027
175	0.7591	1.2459	2.0050
200	0.8355	1.3295	2.1650

* Horizontal curved surfaces, emissivity = 0.85, 100°F ambient

TABLE 3-4

THERMAL ANALYSIS RESULTS FOR NORMAL CONDITIONS OF TRANSPORT

Packaging Component Maximum Temperatures (°F)

Outer surface, with insolation	161°
without insolation	108°
Outer Shell	165°
Lead Shell	166°
Inner Shell/Cavity Wall	167°
Seals	166°
Fuel Storage Container	183°
Fuel	213°

Average cavity gas temperature is 175°F

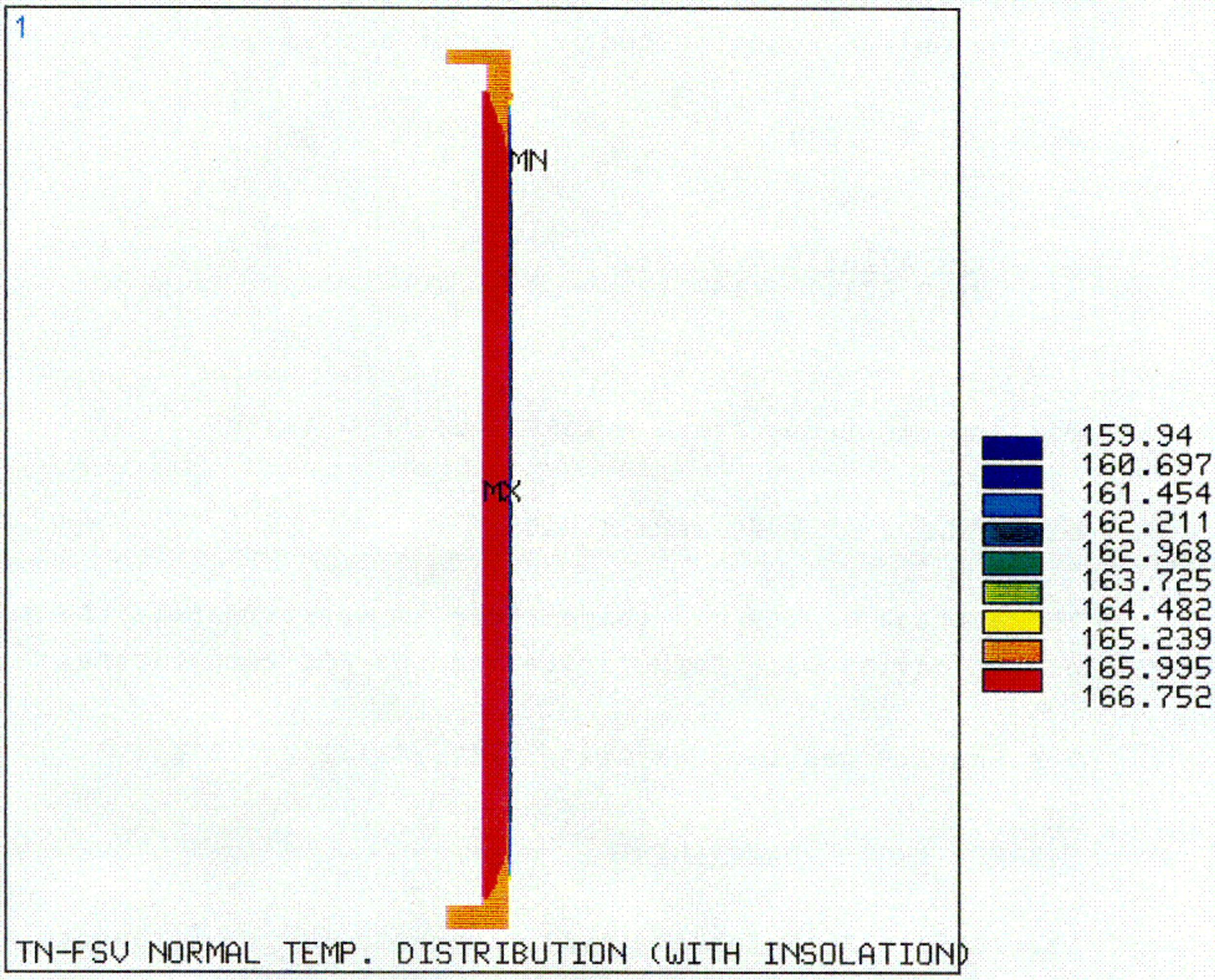

FIGURE 3-3

MAXIMUM TEMPERATURE DISTRIBUTION
NORMAL CONDITIONS OF TRANSPORT

C-01

a. Maximum Fuel Storage Container (FSC) Temperature

Cavity ID, d_o = 18.0 in.
FSC OD, d_i = 17.62 in.
Gap material is air.
Conductivity of air, k, at 175°F = 0.001411 Btu/hr-in-°F.
Decay heat, Q = 360W x 3.142 = 1228 Btu/hr.
Axial length of fuel, L = 187.32 in.
Assume only conduction heat transfer across gap for conservatism.

$$\Delta T = \frac{Q \ln(d_o/d_i)}{2\pi k L}$$

$$= \frac{(1228)\ln(18.0/17.62)}{2\pi(0.001411)(187.32)}$$

$$= 16°F$$

Maximum FSC temperature = 16°F + 167°F = 183°F

b. Average Cavity Gas Temperature

The average cavity gas temperature is approximately the average of the cavity wall temperature and the FSC temperature.

Avg. cavity gas temperature = (167+183)/2 = 175°F

c. Maximum Fuel Temperature

The ISFSI SAR (Reference 3-6), Appendix A3-1.1, presents the thermal analysis of the MVDS. The maximum fuel temperatures are reported in Section A3.1.4.1. For average rated fuel (85W per element, the ΔT between the fuel element centerline and the fuel storage container is (291-261) = 30°F.

The maximum fuel storage container temperature calculated is 183°F and the heat load per element is 60W (6 elements per FSC). It is conservative to assume the ΔT will not exceed 30°F.

$$\text{Maximum fuel temperature} = 183 + 30$$
$$= 213°F$$

3.4.3 Minimum Temperatures

Under the minimum temperature condition of -40°F ambient, the resulting packaging component temperatures will approach -40°F at equilibrium. Since the packaging materials, including containment structures, impact limiters and seals, continue to function at this temperature, the minimum temperature condition has no adverse affect on the performance of the TN-FSV.

3.4.4 Maximum Internal Pressures

The maximum internal pressure is calculated in Chapter 4, assuming:

- Cavity gas is saturated with water vapor. The partial water vapor pressure is based on the minimum cavity wall ("cold wall") temperature of 166°F.
- Average cavity gas temperature is 175°F.
- Cask is closed and sealed at 70°F and 1 atm (14.7 psi)

The calculation includes pressure increases due to ideal gas heating and partial pressure of water vapor.

3.4.5 Maximum Thermal Stresses

The maximum thermal stresses during normal conditions of transport are calculated in Section 2.6 of Chapter 2.

3.4.6 Evaluation of Package Performance

The thermal analysis for normal conditions concludes that the TN-FSV design meets all applicable requirements. The maximum temperatures calculated using conservative assumptions are relatively low. The maximum temperature of any containment structural component is 167°F which has an insignificant effect on the mechanical properties of the containment materials used. The maximum lead temperature (166°F) is well below allowable values. The seal temperature (166°F) during normal transport conditions is well below the 300°F long term limit specified for continued seal function. The maximum accessible surface temperature of 108°F in the shade is below the specified 180°F limit.

3.5 HYPOTHETICAL ACCIDENT THERMAL EVALUATION

3.5.1 Thermal Analysis Models

The cask body is protected from the thermal accident environment by the thermal shield on the sides and by the impact limiters on the ends. However, the trunnions do provide a direct conduction path to the lead shielding. Since the prevention of lead melt is a design requirement, two analytical models were developed as discussed in Sections 3.5.1.1 and 3.5.1.2.

3.5.1.1 Cask Body Thermal Model

The analysis assumptions and model consider the packaging condition following the hypothetical accident sequence of 10CFR71.73. The two-dimensional, axisymmetric model developed for the normal conditions of transport analysis is used with minor modifications to

obtain the maximum component temperatures under accident conditions. This cask model is described in Section 3.4.1.1.

To permit radiation heat transfer across the air gap in the thermal shield, the ANSYS radiation superelement, STIF50, is added to the model. Surface emissivities of 0.85 are used for the stainless steel surfaces within the air gap. The air gap is 0.125 in. (nominal) at fabrication. During the thermal accident, when the thermal shield is hotter than the outer shell, thermal expansion will cause this gap to expand, increasing the thermal resistance across the gap. For conservatism a constant gap of 0.125 in. is assumed. This assumption makes the analysis conservative for the accident evaluation.

Analysis in Chapter Two and testing on similar designs confirm 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 limiters are assumed to be insulated from the thermal event.

Heat dissipation from the outer surface is by radiation and natural convection to an ambient at 100°F before and after the thermal accident. The total heat transfer coefficient, H_t, is calculated in the same manner as for normal conditions of transport described in Section 3.4.1.1. Table 3-5 gives the values of H_t as a function of surface temperatures, T_s, for an ambient temperature of 100°F.

Initial conditions before the thermal accident are established by performing a steady state analysis with a packaging heat load of 360 watts and an ambient temperature of 100°F. Effects of solar radiation are neglected.

TABLE 3-5

CONVECTION AND RADIATION HEAT TRANSFER COEFFICIENT BEFORE AND AFTER THE THERMAL ACCIDENT

Temperature T_s (°F)	Total Heat Transfer Coefficient H_t (Btu/hr-ft^2-°F)
101	1.2044
110	1.4371
120	1.5663
130	1.6660
140	1.7520
150	1.8300
160	1.9027
175	2.0050
200	2.1650
225	2.3177
250	2.4670
300	2.7640
400	3.3809
500	4.0582
600	4.8168
700	5.6715
800	6.6344
900	7.7164
1000	8.9277
1100	10.2781
1200	11.7772

During the thermal accident, heat absorption at the outer surface by radiation and convection on the basis of still ambient air at 1475°F is given by the following equation (Reference 3-4):

$$q_s = H_c(T_f-T_s)+0.1714E-8(F_o)[E_f(T_f+460)^4-(T_s+460)^4]$$

where,

q_s = Heat flux into packaging from thermal environment

H_c = Convection heat transfer coefficient
= $0.18\ (T_f-T_s)^{1/3}$ Btu/hr-ft^2 - °F

T_f = Radiation environment temperature
= 1475°F

T_s = Surface temperature of the packaging, °F

F_o = Outer packaging surface absorptivity
= 0.8

E_f = Radiation environment emissivity
= 0.9

The total heat transfer coefficient for heat absorption is:

$$H_t = H_c+0.1714E-8(F_o)[E_f(T_f+460)^4-(T_s+460)^4]/(T_f-T_s)$$

Table 3-6 gives the value of H_t as a function of T_s. H_t is used as a boundary input during the 30 minute duration of the thermal radiation environment.

TABLE 3-6

CONVECTION AND RADIATION HEAT TRANSFER COEFFICIENT DURING THE THERMAL ACCIDENT

Temperature T_s (°F)	Total Heat Transfer Coefficient H_t (Btu/hr-ft^2-°F)
101	14.485
110	14.557
120	14.637
130	14.718
140	14.800
150	14.882
160	14.966
175	15.093
200	15.308
225	15.529
250	15.756
300	16.225
400	17.230
500	18.324
600	19.505
700	20.762
800	22.078
900	23.415
1000	24.698
1100	25.765
1200	26.208

Following exposure to the thermal radiation environment, the cooldown of the packaging is calculated in the same manner as that used to calculate heat transfer during normal conditions of transport (Section 3.4.1.1). The total heat transfer coefficients, H_t, used for the range of package temperatures representative of post-thermal event conditions are shown on Table 3-5.

3.5.1.2 Trunnion Thermal Model

The geometry and dimensions used for the trunnion thermal model are shown in Figure 3-4. The model is an approximate two-dimensional axisymmetric representation of the TN-FSV trunnion and models the trunnion, the lead below and around the trunnion and the inner shell. As shown in Figure 3-4, the region within a 10 in. radius from the trunnion centerline is modeled including portions of the thermal shield, air gap and outer shell.

The thermal model is developed using the two dimensional isoparametric thermal solid element, STIF55, which simulates conduction in the model. Radiation heat transfer across the air gap in the thermal shield is simulated with the ANSYS radiation superelement, STIF50. Figure 3-5 shows the finite element plot of the model.

Boundary conditions established for the thermal accident analysis in Section 3.5.1.1 are used for this analysis. A uniform model temperature of 114°F is assumed. This corresponds to the maximum cask body temperature for the thermal accident.

3.5.2 Packaging Conditions and Environment

The condition of the TN-FSV following the free drop and puncture drop accidents is discussed in Section 2.7.1.5 and 2.7.2, respectively.

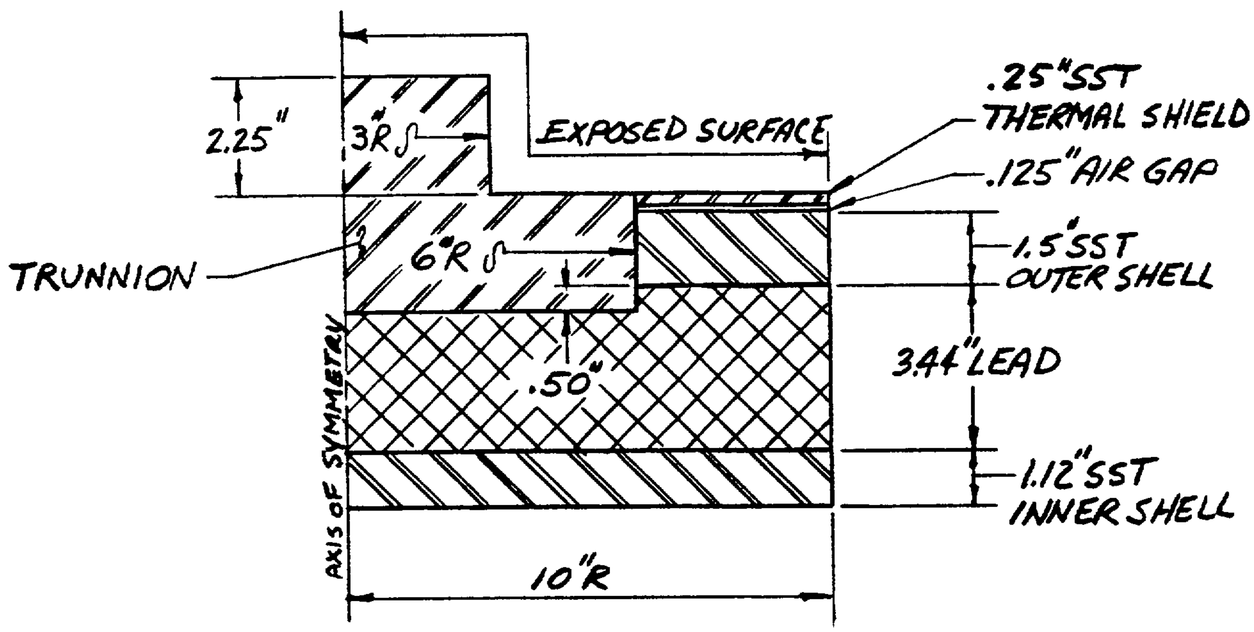

FIGURE 3-4

THERMAL MODEL FOR THE TRUNNION REGION

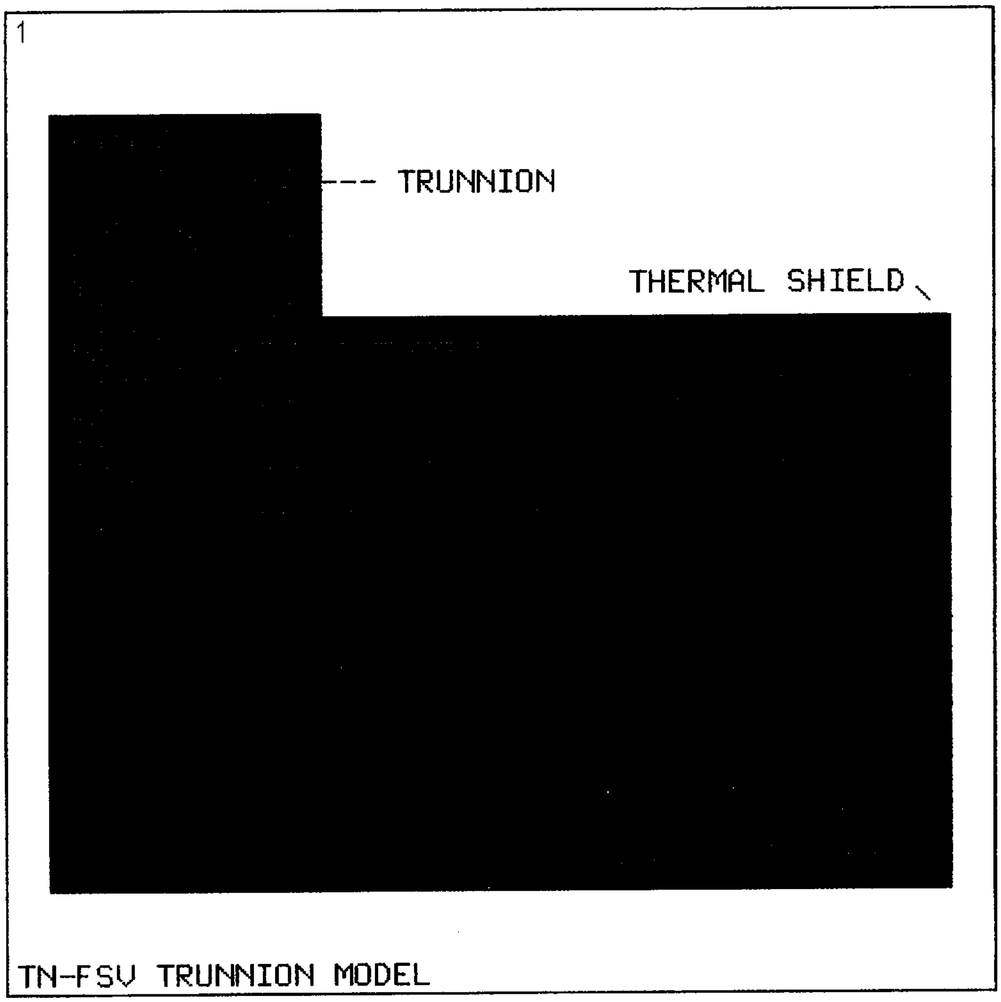

FIGURE 3-5

ANSYS FINITE ELEMENT FOR THE TRUNNION REGION

The packaging is assumed to be initially at steady state in an ambient temperature of 100°F with 360 watts of payload decay heat. The effects of solar radiation are neglected prior to, during, and following the accident. During the accident the packaging is exposed to a radiation environment of 1475°F for 30 minutes. The radiation environment is characterized by an emissivity of 0.9 and the outer surface of the packaging is assumed to have an absorptivity of 0.8. Heat absorption by natural convection on the basis of still air at 1475°F is included. Cooldown of the packaging after 30 minutes of exposure to the accident environment is based on 100°F ambient air temperature and a surface emissivity of 0.85.

3.5.3 Package Temperatures

The maximum transient temperatures of the analyses performed using the cask and packaging thermal model are presented in Table 3-7. The temperature distribution at the end of the accident is shown in Figure 3-6 for the cask body model and Figure 3-7 for the trunnion model. Each color band shows the range of temperatures for the particular region.

The results of the analyses show that the lead temperature reaches a maximum of 478°F (248°C) 35 min. after the start of the thermal accident. This temperature occurs at a single point behind the trunnion at its centerline and is significantly below the melting point of 620°F (327°C) for lead. The maximum temperature distribution (trunnion model) in the lead region at this time is shown in Figure 3-8. The temperature is significantly lower at locations other than this point. The maximum seal temperatures are below 200°F under thermal accident conditions. Figure 3-9 shows the temperature - time history for average midwall locations in the thermal model.

TABLE 3-7

THERMAL ANALYSIS RESULTS FOR ACCIDENT CONDITIONS

Packaging Components Maximum Transient Temperatures

	Temperature	Time*
Outer surface (thermal shield)	1316°F	30 min.
Outer Shell	283°F	31 min.
Lead Shell - behind trunnion	478°F	35 min.
- mid-section of cask	257°F	55 min.
Inner Shell/Cavity Wall	245°F	55 min.
Seals (Lid/Drain)	<200°F	5.5 hr.
Fuel Storage Container	261°F	--
Fuel	291°F	--

Average cavity gas temperature (peak value) is 276°F

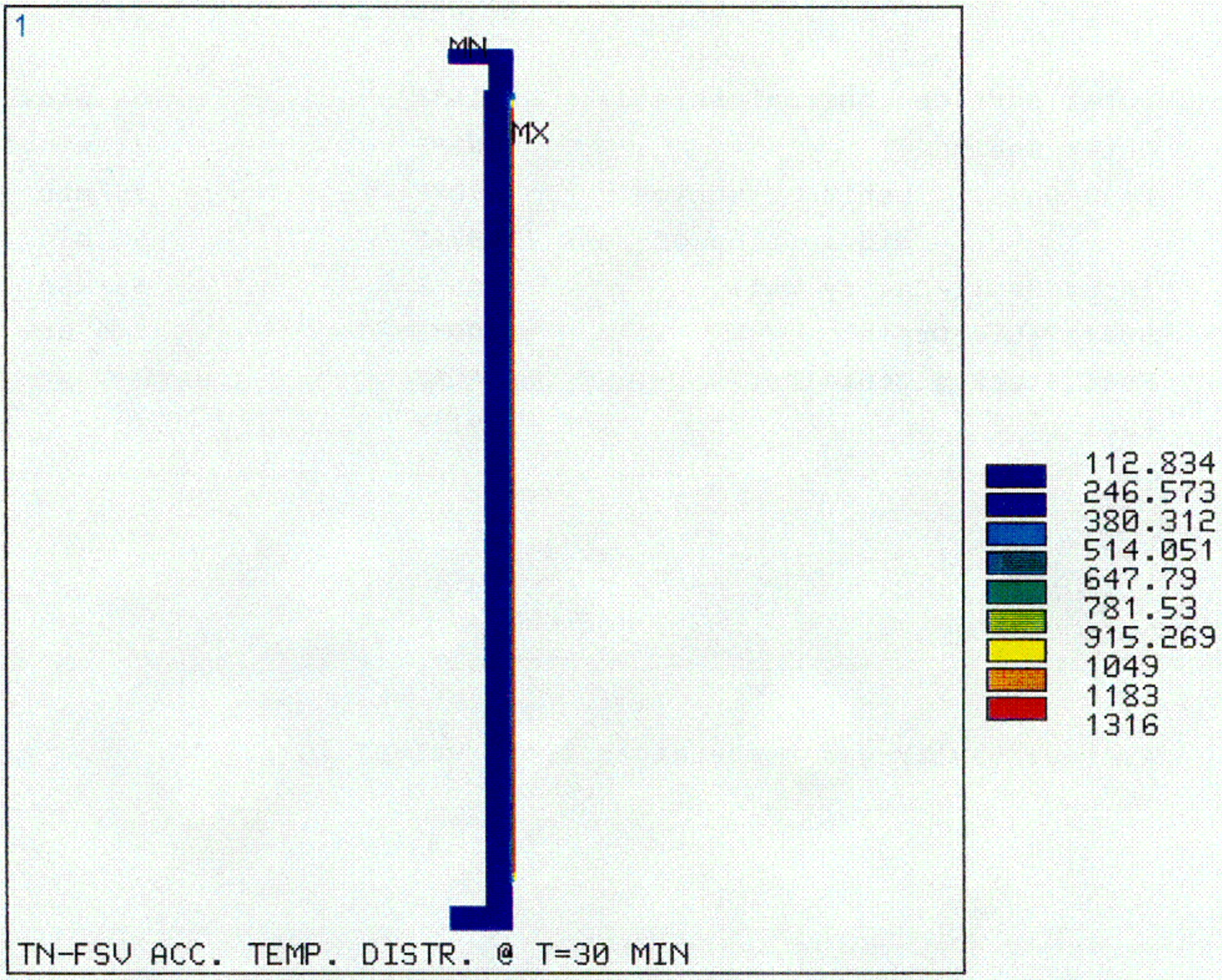

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FIGURE 3-6

TEMPERATURE DISTRIBUTION AT THE END OF THERMAL ACCIDENT (TIME = 0.5 HR.)
CASK BODY MODEL

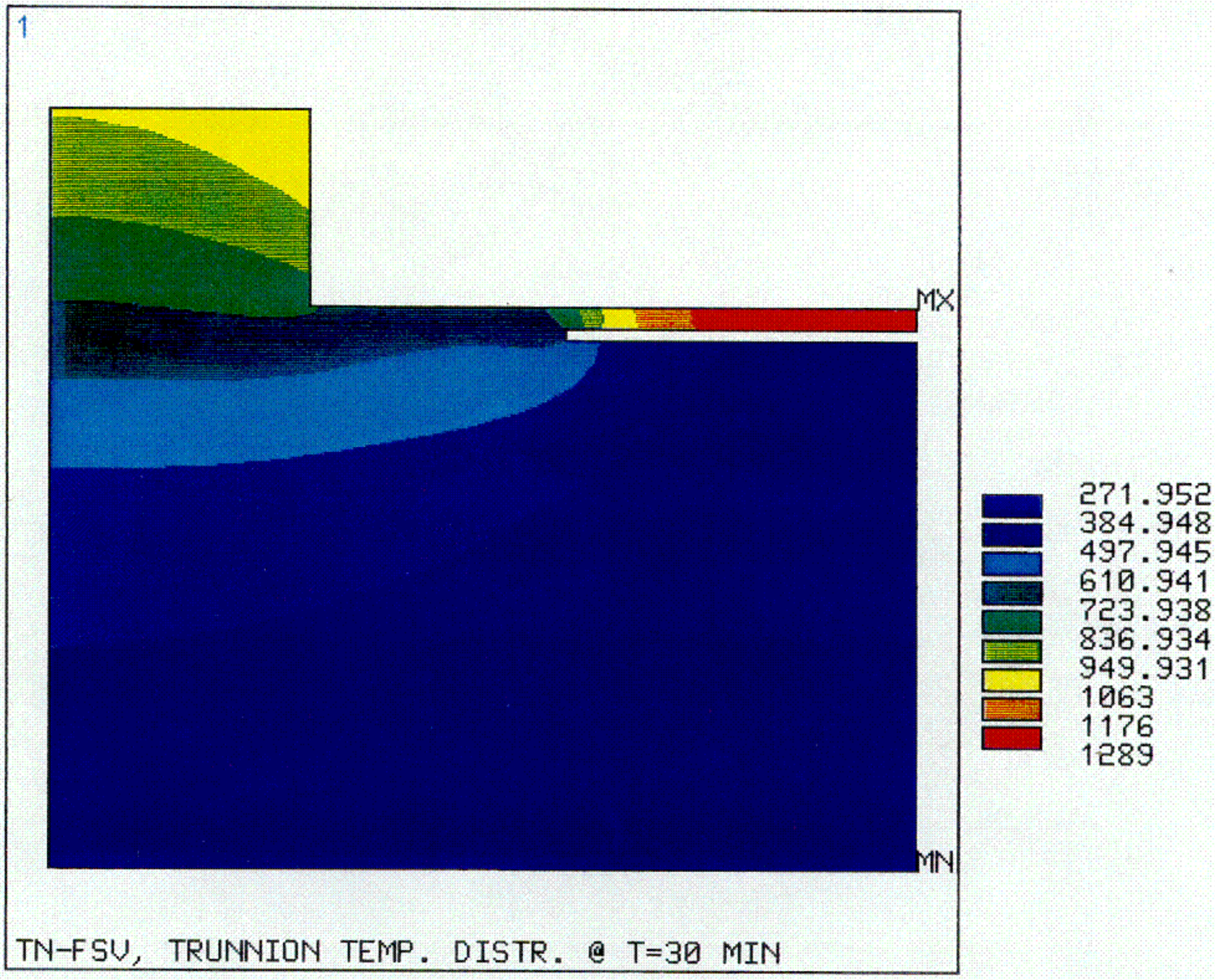

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FIGURE 3-7

TEMPERATURE DISTRIBUTION AT THE END OF THERMAL
ACCIDENT (TIME = 0.5 HR.)
TRUNNION MODEL

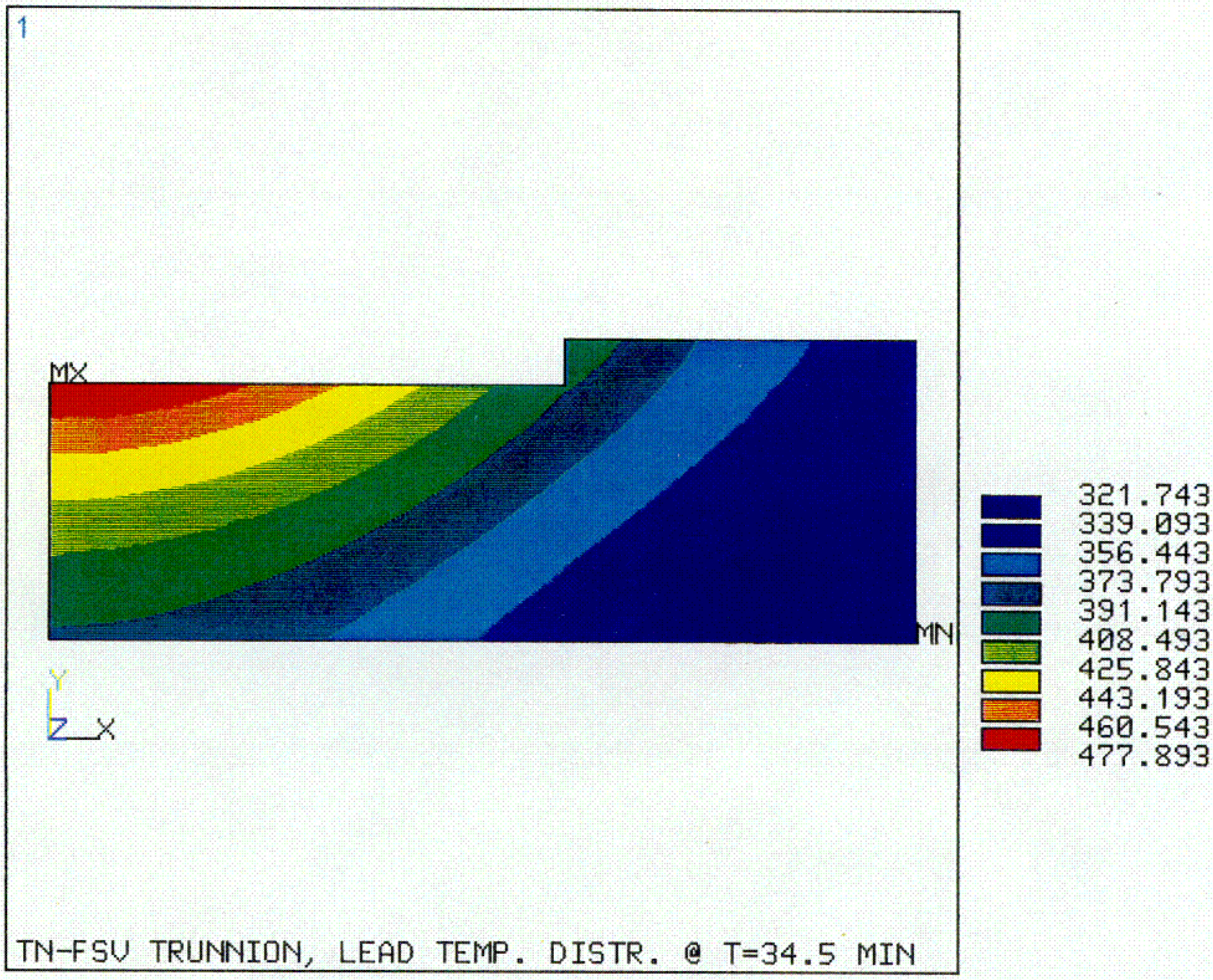

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FIGURE 3-8

MAXIMUM TEMPERATURE DISTRIBUTION
LEAD REGION OF THE PACKAGING

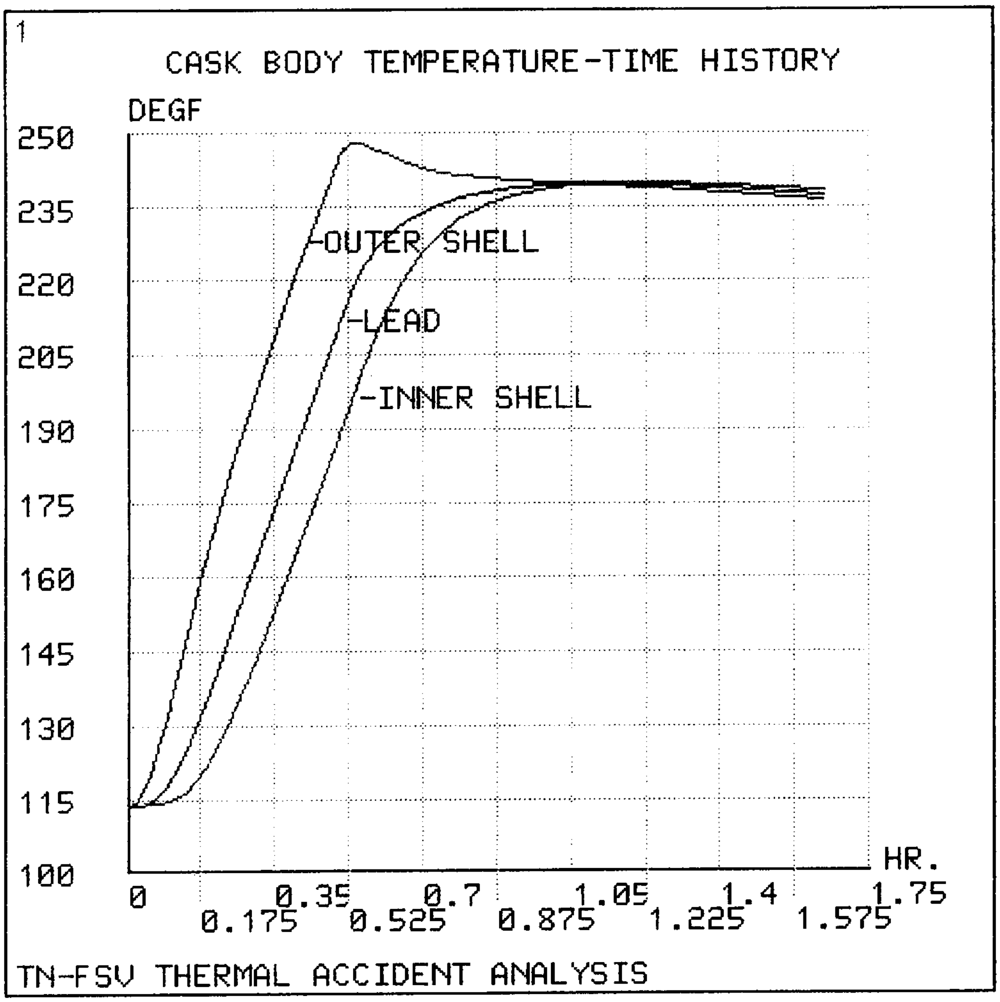

FIGURE 3-9

MIDWALL TEMPERATURE - TIME HISTORY
THERMAL ACCIDENT CONDITIONS

As calculated in Section 3.4.2, the Fuel Storage Container (FSC) is 16°F hotter than the cavity wall temperature, resulting in a peak FSC temperature of 261°F. The maximum fuel temperature is 30°F hotter than the FSC. This results in a peak fuel temperature of 291°F. The peak value of the average cavity gas temperature is approximately the average of the cavity wall temperature and the FSC temperature, and is 253°F.

3.5.4 Maximum Internal Pressure

The maximum cask cavity internal pressure during the hypothetical thermal accident is calculated in Chapter 4 based on an average gas temperature of 276°F and a "cold wall" temperature of 200°F (for the water vapor pressure). It is assumed that after the drop accident all the fission gas is released into the cavity free volume.

3.5.5 Maximum Thermal Stresses

The maximum thermal stresses during the hypothetical thermal accident are calculated in Section 2.7 of Chapter Two. The thermal-stress analysis is performed using the cask model described in Section 3.4.1.1 and the temperature results reported in Section 3.5.3.

3.5.6 Evaluation of Package Performance

The lead temperature remains below the melting point and the analyses presented in Chapters 2, 3, and 4, show that the package will withstand the hypothetical thermal accident without compromising the structural integrity of the package.

3.6 REFERENCES

3-1 Title 10, Code of Federal Regulations, Part 71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions," 1989.

3-2 ASME B&PV Code, Section III, Appendices, 1989 Ed.

3-3 Kreith, et. al., "Principles of Heat Transfer," 4th Edition, Harper and Row Publishers, 1986.

3-4 DeSalvo G. J. and Swanson, J. A., "ANSYS Engineering Analysis System," User's Manual Rev. 4.4, Swanson Analysis Systems, Inc., Houston, 1989.

3-5 Shappert, L. B., "Cask Designers Guide, A Guide for the Design, Fabrication and Operation of Shipping Casks for Nuclear Applications," ORNL-NSIC-68, UC-80-Reactor Technology, Oak Ridge National Laboratory, 1970.

3-6 Thermal-Hydraulic Analysis of the MVDS, Appendix A3-1.1, Safety Analysis Report for the Fort St. Vrain ISFSI, Revision 2, Issue Date 29th July 1991.

CHAPTER FOUR

CONTAINMENT

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4.1 CONTAINMENT BOUNDARY

The containment boundary of the TN-FSV is formed by the containment vessel walls, bottom, flange and lid up to the innermost of the two concentric lid O-rings, the O-ring itself, the cover and O-ring on the vent port, and the O-ring and cover of the drain port, as shown in Figure 4-1.

4.1.1 Containment Vessel

The containment vessel consists of the inner shell, the bottom, the closure flange, the lid plate, and the closure bolts. The inner shell is formed from 1.12 inch thick plate. To this are welded the bottom, a 5.5 inch thick forging, and the closure flange. The cavity length is 199 inches. The top 7.12 inches of the cavity has a diameter of 20.83 inches and the remaining, a diameter of 18.0 inches.

The lid is a 2.5 inch thick plate, and is bolted to the closure flange by twelve 1 inch diameter bolts. For the purposes of containment, the spacer plate attached to the inside of the lid is not considered part of the containment boundary.

The shell, bottom, closure flange, and lid are fabricated from type 304 stainless steel.

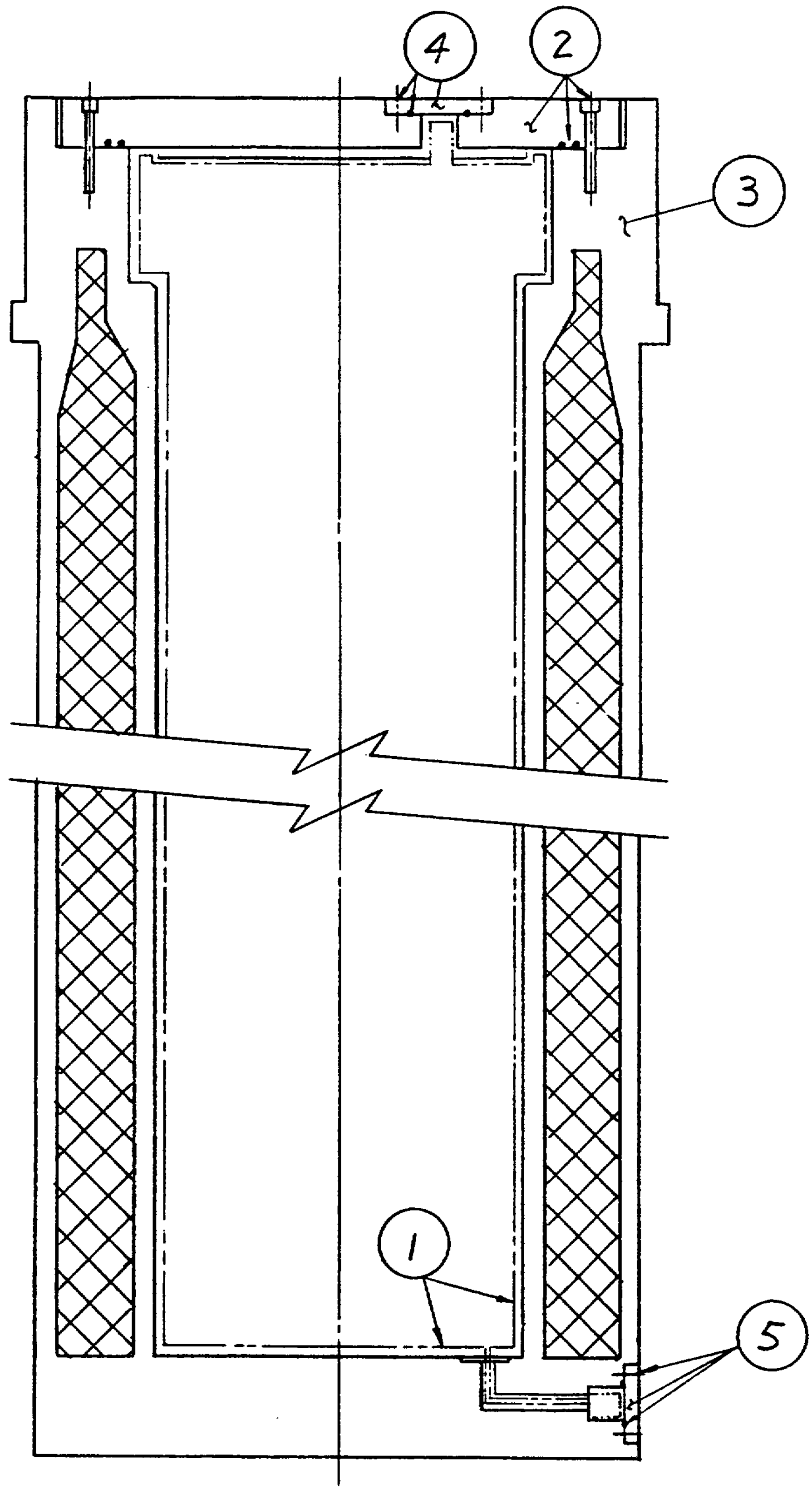

FIGURE 4-1
TN-FSV CONTAINMENT BOUNDARY COMPONENTS

FIGURE 4-1 (Continued)

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

2. Dashed line (-————-————-) indicates containment boundary.

3. Containment boundary components are listed below:

   1. Cask body inner shell and bottom plate

   2. Lid assembly, closure bolts, and inner O-ring

   3. Bolting flange

   4. Vent port cover plate, bolts, and seals

   5. Drain port cover plate, bolts, and seals

4.1.2 Containment Penetration

There are two penetrations through the TN-FSV containment vessel, the drain port through the bottom and the vent port through the lid.

The vent port is a 1.5 inch diameter hole straight through the lid with a quick-connect coupling. Containment is provided by a 0.75 inch thick, 3.81 inch diameter blind flange with four 0.25 inch bolts and one O-ring.

The drain port consists of a 0.5 inch I.D. hole bored through the solid bottom; ending with a 1.88 inch ID hole. A quick-connect coupling is housed inside this 1.88 inch hole. Containment is provided by a 1.0 inch thick 4.19 inch diameter blind flange with an O-ring, which is mounted to the housing with four 0.25 inch bolts.

Both port covers (blind flanges) are fabricated from type 304 stainless steel.

4.1.3 Seals and Welds

The following welds are part of the containment:

- the longitudinal weld of the shell
- the circumferential welds between the bottom and the shell
- the circumferential weld between the flange and the shell

All welds are full penetration welds performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code Section III, Subsection NB-4241 for a Type I butt joint. Non-destructive examination includes radiographic and liquid dye penetrant methods using the acceptance standards of ASME Section III, Subsection NB-5300.

The following seals form part of the containment boundary:

- the inner silicone O-ring on the lid
- the silicone O-ring on the vent port cover
- the silicone O-ring on the drain port cover

All seals are face-mounted in dovetail grooves on the lid and covers. The volume of the grooves is designed to allow sufficient room so the mating metal surfaces can be brought into contact by the bolts, thereby ensuring uniform seal deformation. All surfaces in contact with the seals are machined to a 63 microinch (maximum) R_a surface finish.

All seals are protected from damage during the hypothetical thermal accident by the insulating properties of the impact limiters, which keep all seal temperatures below their operating limit (Ref. 4-1, Table A3-13) as demonstrated in Chapter 3.

The lid is provided with two concentric O-rings with access to the annular space between them provided for leak testing. The outer O-ring is not considered part of the containment boundary. Leak testing of the penetration cover seals can be accomplished by means of a vacuum bell.

4.1.4 Closures

The closure devices consist of the lid, the port covers, and their associated bolts. Bolt installation torque values are selected to achieve compression of the O-rings resulting in metal-to-metal contact between the lid or cover and the corresponding mating surface. As demonstrated in Chapter 2, hypothetical accidents will not result in yielding of any closure bolts, ensuring that the same degree of sealing is provided under both normal and accident conditions.

The lid is closed by twelve 1.0-8UN x 3.50 bolts equally spaced on a 25.2 inch diameter bolt circle. The vent and drain port covers are each closed by four 0.25 inch diameter bolts.

4.2 REQUIREMENTS FOR NORMAL CONDITIONS OF TRANSPORT

The TN-FSV is designed to meet the requirement of 10CFR71.51 for a Type B package 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 10CFR71.71 for normal conditions of transport.

4.2.1 Containment of Radioactive Material

The radioactive contents of TN-FSV are six (6) spent fuel elements from the Fort. St. Vrain High Temperature, Gas-Cooled Reactor. These fuel elements consist of graphite blocks containing fuel rods and coolant passages. The fuel rods contain the nuclear fuel in the form of micro-spheres of either thorium/uranium carbide (fissile particles) or thorium carbide (fertile particles). The uranium is enriched to 93% U-235 and the thorium material was partially converted during reactor operation to produce U-233. These micro-spheres are sealed within a silicon carbide shell which contains the nuclear fuel and any fission products generated during fuel burn up in the reactor. In case of any failures of these silicon carbide shells during reactor operation the fission products will mix with the helium coolant and then be removed by the coolant purification system. Therefore during the normal conditions of transport there is no radioactive material available for release. In fact, a 0.25% fuel failure rate was found during this operation of the FSV plant. Therefore, a 0.25% fuel failure shall be assumed for normal conditions of transport.

4.2.1.1. Fission Gas Products

The maximum fission product activity that will be contained in the TN-FSV at any time would be that associated with six elements having the maximum burnup. Total fission product per fuel element is given in Table 4-1.

Assuming a power peaking factor of 1.8 over the 6 years in the reactor and 1600 days of decay since shutdown, the volatile inventory in the fuel elements would amount to about 392 curies of noble gas (principally Kr-85). In addition, about 1.4 gm moles of stable noble gas and iodine would be present.

This fission product activity is doubly contained, first by the fuel particles and their impervious TRISO coatings, and secondly by the cask containment. Fission products can be annealed from fuel particles, but not in significant quantity at temperatures below 1000°F.

4.2.1.2 Release of Contents

The maximum allowable release rate for the packaging is
$A_2 \times 10^{-6}$ Ci/hr.
Kr-85 is the only radioactive isotope that is of significance for potential release; A_2 valve is 1000. Therefore, the maximum allowable release rate is

$$R_N = 1.0 \times 10^{-3} \text{ Ci/hr}$$

The Kr-85 activity available for release is:

392 Ci/element x 6 elements/cask x 0.0025 = 5.88 Ci

The net void volume in the TN-FSV packaging is 2.67×10^5 cc. Therefore the specific activity is:

$$C_N = \frac{5.88 \text{ Ci}}{2.67 \times 10^5 \text{ cc}} = 2.20 \times 10^{-5} \text{ Ci/cc}$$

TABLE 4-1

SOURCE DATA FOR ONE FUEL ELEMENT

Isotope	Activity (Curies)
Kr-85	392
Sr-90	2,610
Y-90	2,610
Rh-106	99
Ru-106	99
Sb-125	56
Cs-134	1,242
Cs-137	2,754
Ba-137m	2,610
Ce-144	628
Pr-144	628
Pm-147	1,054
Sm-151	9
Eu-154	158
Eu-155	23

The maximum permissible leakage rate for normal conditions is calculated as:

$$L_N = \frac{R_N}{C_N} \times \frac{1}{3600} = \frac{1.0 \times 10^{-3}}{7.92 \times 10^{-2}} = 0.013 \text{ cc/s}$$

However, a conservative fabrication verification leak test will be performed to satisfy a criterion of 1×10^{-3} std cc/s.

Assuming this leakage rate criteria, it is of interest to estimate the diameter of the possible leak path. From Ref. 4-3, equations B2 and B3 can be used to find the hole Size.
For the specified leakage rate, the continuum region dominates and

$$Fc = 1 \times 10^{-3}/0.99 = 1.01 \times 10^{-3} \quad \text{(B2)}$$

If we assume the leak path length is the o-ring diameter (0.7 cm) and the air viscosity μ = 0.0185 cp then:

$$D = \left[\frac{1.01 \times 10^{-3} \times 0.7 \times 0.0185}{2.49 \times 10^{6}}\right]^{0.25} \qquad \text{(B3)}$$

$$D = 1.51 \times 10^{-3} \text{ cm (15 microns)}$$

In Chapter 6, the minimum size of the fissile or fertile particle is approximately 200 microns in diameter. The diameter of the fuel particles with the TRISO coating is approximately 460 microns. It is easily seen that the fuel particles are at least an order of magnitude larger than the leak path and therefore the probability of particles leaking out of the packaging is extremely small.

4.2.2 Pressurization of Containment Vessel

The TN-FSV contents will not include any organic materials. Therefore neither hydrogen nor other gases will be generated by radiolysis and there will be no flammable gas hazard.

Assuming the cask cavity is closed at 70°F and 1 atm and that saturated water vapor is present, an equilibrium cavity pressure is calculated to define the maximum normal operating pressure (MNOP). The quantity of dry air filling the cavity is

$$N = \frac{PV}{RT} = \frac{1.0 \times 0.2673}{0.0821 \times 294} = 0.0111 \text{ kmole}$$

From Table 3-1, the average cavity gas temperature at equilibrium (normal conditions) is 79°C. Therefore the partial pressure of the air in the cavity is:

$$P_{1N} = 1.0 \times \frac{352}{294} = 1.20 \text{ atm}$$

The partial pressure due to saturated water vapor is 0.365 atm (cold wall temperature of 74°C Table 3-1). Since fission gas contribution to the total pressure is negligible, the containment vessel pressure is 1.56 atm or 8.3 psig.

The TN-FSV is tested to at least 1.5 times the MNOP in accordance with 10CFR71.85(b).

4.2.3 Containment Criterion

The following leak tests are specified in accordance with ANSI N14.5, Sections 6.3, 6.4 and 6.5 (Ref. 4-3):

- Fabrication and periodic verification tests shall determine that the leak rate for the entire containment is no greater than 10^{-3} std cm^3/s with a test sensitivity better than 0.5×10^{-4} std cm^3/s.

- Assembly verification tests for the lid seal and port covers with a sensitivity of at least 0.1 std cm^3/s shall find no detectible leak.

4.3 CONTAINMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

The release of radioactive material is limited to A_2 per week and 10,000 Ci Kr85 under the conditions of the hypothetical accident tests of 10CFR71.73, in accordance with 10CFR71.51(2).

4.3.1 Fission Gas Products

From the previous Section 4.2.1.1., there is about 392 Ci of Kr85 and 1.4 gram moles of gas available in each fuel element. Therefore, the total inventory in the TN-FSV packaging is 2352 Ci Kr85 (6x392) and a quantity of 8.4 gram moles of gas (Kr, Xn & I). If it is assumed that there is 100% fuel failure in the accident, the total pressure in the cask can be calculated as follows:

- The partial pressure of the air in the cavity at the accident condition temperature of 253°F (Table 3-1) is:

  $$P_{1A} = 1.0 \times \frac{396}{293} = 1.35 \text{ atm}$$

- The partial pressure due to saturated water vapor at the cold wall temperature of 200°F (Table 3-1):

  $$P_{2A} = 0.775 \text{ atm}$$

- The partial pressure due to the release of the fission gas into the cavity:

  $$P_{3A} = \frac{8.4 \times 10^{-3} \times 0.0821 \times 396}{0.267} = 1.02 \text{ atm}$$

The total cavity pressure is therefore 3.15 atm (31.5 psig).

4.3.2. Containment of Radioactive Material

Kr85 is the only radioactive material that must be considered. If all of the Kr85 is assumed to be available for leakage out of the packaging, the specific activity is:

$$C_A = \frac{2354}{2.67 \times 105} = 8.81 \times 10^{-3} \text{ Ci/cc}$$

The maximum permissible release rate R_A is 10,000 Ci/week.

Therefore the maximum permissible leak rate is:

$$L_A = (R_A/C_A) \times 1.65 \times 10^{-6} = 1.87 \text{ cc/sec}$$

4.3.3 Containment Criterion

As can be seen from above, the requirements for normal condition of transport are limiting and therefore, the leakage tests specified in Section 4.2.3 are applicable.

4.4 REFERENCES

4-1 Parker O-Ring Handbook, ORD-5700, Copyright 1982

4-2 Federal Regulation, 10CFR71 - Packaging and Transportation of Radioactive Material, Pacific Northwest Laboratories, September 1987.

4-3 "American National Standard for Leakage Tests of Packages for Shipment of Radioactive Material," ANSI N14.5 - 1986.

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SHIELDING EVALUATION

5.1 DISCUSSION AND RESULTS

An evaluation of the shielding performance of the TN-FSV is performed to demonstrate compliance with the dose rate limits of 10CFR71.47. This also demonstrates compliance with the accident dose rate limit of 10CFR71.51(2) because the components of the package shielding which are not an integral part of the body (the impact limiters, the lid, and the thermal shield) will remain in position under all accident conditions, as demonstrated in Appendices 2.10.1 and 2.10.2. The contents of the TN-FSV consist of six irradiated FSV fuel elements enclosed within a fuel storage container (FSC). The fuel elements have a maximum burnup of 70,000 MWD/MTU and have been decayed for at least 1600 days since discharge from the reactor.

The most significant shielding design feature of the TN-FSV is the cask body, which consists of an inner layer of stainless steel, followed by lead and an outer layer of stainless steel. The impact limiters, which consist of wood in stainless steel cases, provide additional axial shielding, and the thermal shield, a stainless steel shell, provides additional radial shielding. Additional shielding at the top of the packaging is provided by a depleted uranium plug which is inserted into the closure lid of the FSC. The shield layers and thickness are listed in Table 5-1.

The shielding analysis of the TN-FSV is performed with regulatory acceptable codes from the SCALE system (Ref. 5-1). Conservative modeling of the source provides an upper bound on the dose rates. Table 5-2 summarizes the calculated dose rates and shows that all applicable limits are satisfied.

TABLE 5-1

TN-FSV SHIELD CONFIGURATION

Axial, Bottom		Axial, top		Radial	
2.0 inch steel	(1)	1.19 inch steel	(1)	0.5 inch steel	(1)
2.69 inch Al		4.38 inch dep U	(1)	1.12 inch steel	
5.5 inch steel		0.56 inch Al		3.38 inch lead	(2)
0.25 inch steel	(3)	2.5 inch steel		1.50 inch steel	
19.37 inch balsa	(3)	0.25 inch steel	(3)	0.25 inch steel	(4)
0.19 inch steel	(3)	19.37 inch balsa	(3)		
		0.19 inch steel	(3)		

(1) Fuel storage canister (See Figure 1-2).

(2) 3.44 nominal, 3.38 minimum

(3) Impact limiter

(4) Thermal shield

TABLE 5-2

CALCULATED DOSE RATES

		Dose Rates, mrem/hr Calculated(1)		
Normal		Gamma	Neutron	Limit
Package Surface	Side	99	1.0	200
	Top	<0.01	nil	200
	Bottom	76.8	nil	200
Two Meters from	Side	9.5	0.07	10
	Top	<0.01	nil	10
	Bottom	6.1	nil	10
Occupied Position(2)		<0.01	nil	2
Accident				
One Meter from Package Surface	Side	<99	<1.0	1000
	Ends	<77	nil	1000

(1) All dose rates are calculated using the minimum lead thickness.

(2) The occupied position is assumed to be in a cab five meters from the package top. No credit is taken for shielding by cab materials.

(3) Because there is no loss of shielding under hypothetical accident conditions, the normal condition surface dose rate results are adequate to demonstrate that the one meter accident dose rates are less than 1000 mrem/hr.

5.2 SOURCE SPECIFICATION

5.2.1 Gamma Source

An ORIGEN-S calculation was performed for the highest burnup FSV fuel element (70,000 MWD/MTU). Table 5-3 shows the ORIGEN-S gamma spectra for a FSV irradiated element decayed for 1600 days. The SCALE 18 group gamma library structure is not the same as the gamma structure output by ORIGEN-S. Therefore, the ORIGEN-S source terms were converted, conserving energy, into the SCALE 18 group gamma structure as shown in Table 5-4. The gamma source terms are increased by a factor of 1.8 to conservatively account for burnup peaking. The homogenized source volume is assumed to be a cylinder 187.3 inches long and 16.6 inches in diameter. The source volume is 6.67E5cc and the resulting volumetric source strength is:

$$\frac{\text{1.20E14 } \gamma\text{/s/ele x 6 ele x 1.8}}{\text{6.67E5cc}} = 1.94 \text{ E9 } \frac{\gamma}{\text{cc-s}}$$

5.2.2 Neutron Source

The ORIGEN-S analyses mentioned above also evaluated the neutron source from the irradiated FSV fuel. The neutra spectra output by ORIGEN-S conforms to the 27 neutron group spectra utilized in the SCALE x-section library. The neutron spectra for a single FSV fuel element, decayed 1600 days is shown in Table 5-5. The volumetric neutron source term is:

$$\frac{\text{2.03E5 n/s/ele x 6 ele x 1.8}}{\text{6.67E5cc}} = \text{3.29 n/cc-s}$$

TABLE 5-3
ORIGEN-S GAMMA SPECTRA
gamma/sec/element

Group	Ave Energy (Mev)	Fission Products	Actinides	Total
1	0.3	4.21E12	2.07E10	4.23E12
2	0.65	1.12E14	2.45E10	1.12E14
3	1.13	3.21E12	7.80E8	3.21E12
4	1.57	8.44E11	1.55E9	8.46E11
5	2.00	1.19E11	4.06E7	1.19E11
6	2.40	3.19E9	3.05E4	3.19E9
7	2.80	4.44E8	1.42E10	1.46E10
8	3.25	6.14E7	1.07E4	6.14E7
9	3.75	2.71E4	6.18E3	3.33E4
				1.20E14

TABLE 5-4
SCALE 18 Group Gamma Spectra

Group	Upper Energy (Mev)	γ/s/element
5	4.0	5.70E7
6	3.0	1.55E10
7	2.5	5.54E10
8	2.0	2.91E11
9	1.66	7.21E11
10	1.33	2.27E12
11	1.0	1.70E13
12	0.8	4.16E13
13	0.6	5.82E13
14	0.4	1.81E12
15	0.3	2.54E12
16	0.2	-----

TABLE 5-5

NEUTRON SPECTRA

Energy (Mev)	n/s/element
6.43 - 20.0	2.96E3
3.0 - 6.43	4.18E4
1.85 - 3.0	6.05E4
1.40 - 1.85	2.72E4
0.9 - 1.40	3.19E4
0.4 - 0.9	3.20E4
0.1 - 0.4	6.22E3
	2.03E5

5.3 MODEL SPECIFICATION

5.3.1 Description of Radial and Axial Shielding Configurations

The one-dimensional shielding analysis uses an infinite cylinder for the radial model and an infinite slab for the axial model. The models use the minimum lead thickness determined from the tolerances on the inner and outer stainless steel vessels. Half the source length is modeled axially, with a reflective boundary at the source midplane. The fuel storage container with its depleted uranium top plug is included in the model. The shielding models are shown in Figures 5-1 and 5-2.

5.3.2 Source and Shielding Regional Densities

The source is assumed to consist only of the carbon in the fuel elements, the uranium is neglected. The average density of the fuel element is assumed to be 1.54 g/cc. (Ref. 5-3).

The steel components of the packaging are modeled as carbon steel or 304 stainless, as applicable. The lead is modeled at 11.18 g/cc, 98.5% of the theoretical density to account for minor volume shrinkage after pouring. The wood in the impact limiters is modeled as balsa at 0.125 g/cm^3, with the chemical composition taken from the SCALE Standard Material Composition Library. All of the wood in the impact limiters was modeled as balsa which is conservative since the redwood in the limiters is twice as dense.

The shielding model densities are summarized in Table 5-6.

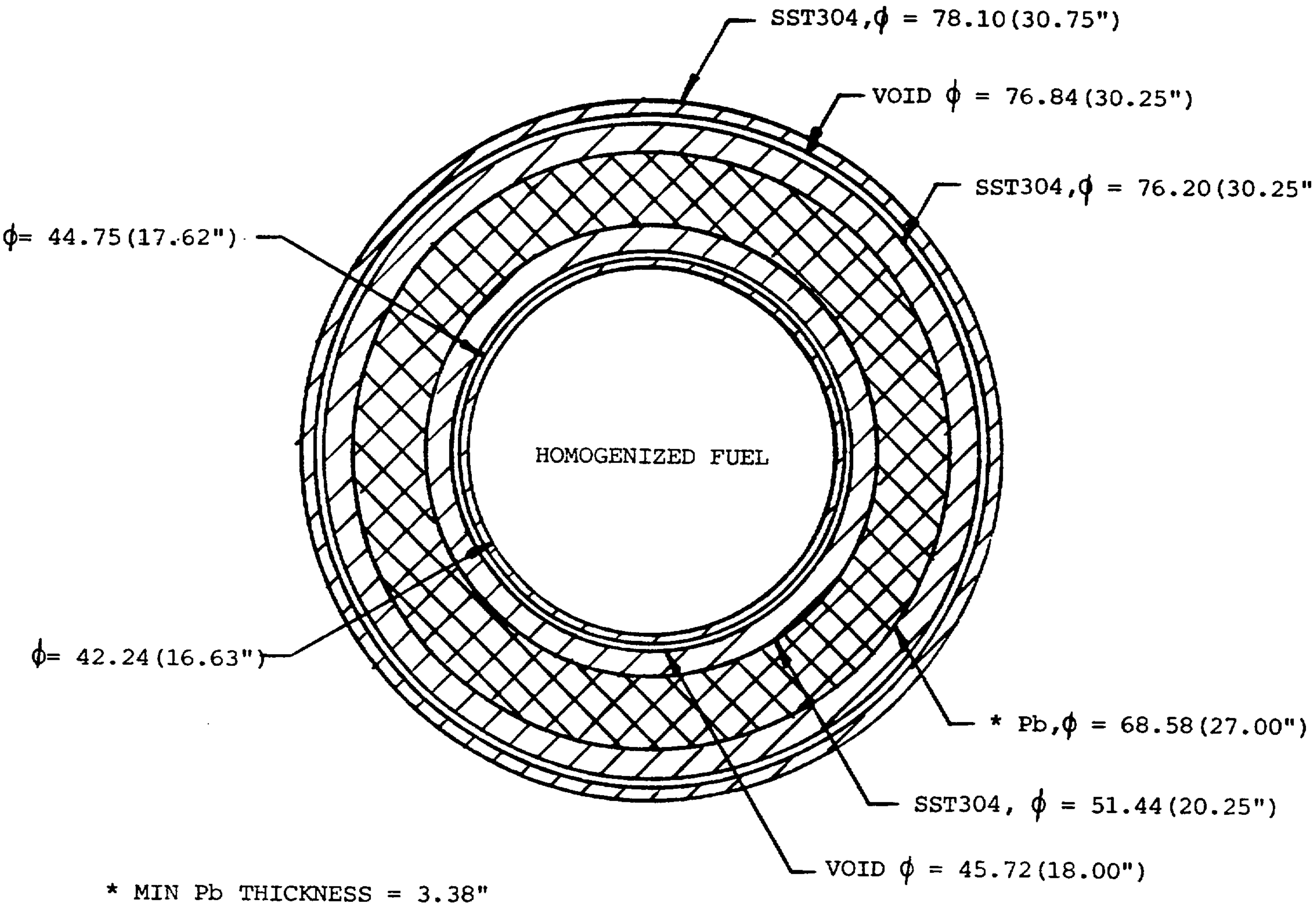

FIGURE 5-1

TN-FSV RADIAL SHIELDING MODEL-INFINITE CYLINDER

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE 5-2

TN-FSV AXIAL SHIELDING MODELS INFINITE SLAB

TABLE 5-6

SOURCE AND SHIELD MATERIAL DENSITIES

ZONE	MATERIAL	DENSITY g/cm^3	DENSITY atoms/b.cm
Source	Carbon	1.54	8.080E-2
Carbon Steel	Iron (99)	7.82	8.350E-2
	Carbon (1)		3.925E-3
Lead	Lead	11.18	3.249E-2
Balsa Wood	Hydrogen (6)	0.125	4.645E-3
	Carbon (44)		2.787E-3
	Oxygen (50)		2.323E-3
304SS	Iron (69.5)	7.92	5.936E-2
	Nickel (95)		7.721E-3
	Chromium (19)		1.743E-2
	Manganese (2)		1.736E-3
Aluminum	Aluminum	2.70	6.024E-2
Depleted Uranium	U-235 (.27)	19.05	1.318E-4
	U-238 (99.73)		4.806E-2

5.4 SHIELDING EVALUATION

The gamma and neutron dose rates are calculated using the SAS1 sequence of the SCALE-4 code. (Ref. 5-1)

The gamma flux at the package surface is determined using the discrete ordinate code XSDRN-PM with P3 scattering expansion and S16 quadrature. The cross sections are taken from the SCALE 27 neutron - 18 gamma library. A working library for XSDRN-PM is prepared by NITAWL. Infinite homogenized x-sections were used in all materials.

To determine the dose rates exterior to the package, the surface angular flux determined by XSDRN-PM is integrated by the XSDOSE code, and converted to a dose rate using the ANSI flux to dose rate factors included in the SCALE library (Ref. 5-2).

5.4.1 Radial Analysis

A cylindrical model with no buckling correction is used for the radial calculation.

Dose rates are determined at the source midplane, at the package surface, and at two meters from the vehicle edge using a cylinder length for XSDOSE corresponding to the cask length of 207 inches. The vehicle width is assumed to correspond to be 8 feet.

5.4.2 Axial Analysis

A disk model with buckling correction for the finite cask diameter is used for the axial calculation. Dose rates are determined on the package axis at the impact limiter surface and at two meters away. The diameter of the disc over which XSDOSE integrates the surface angular flux corresponds to the cask diameter of 30.8 inches.

5.5 REFERENCES

5-1 SCALE-4, Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, CCC-545, Oak Ridge National Laboratory.

5-2 ANSI/ANS-6.1.1-1977, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors".

5-3 Consolidated Design Report for the Model FSV-1 Shipping Cask, Volumes I and II. GADR55

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CHAPTER SIX
CRITICALITY EVALUATION

6.1 DISCUSSION AND RESULTS

6.1.1 Discussion

The TN-FSV is designed for the transport of six (6) spent fuel elements from the Fort St. Vrain, High Temperature Gas-Cooled Reactor. The Fort St. Vrain fuel elements to be transported in the TN-FSV packaging are identical to the fuel elements which are described in the SAR for the licensed Model FSV-1 packaging. It is therefore the intent of the ensuing discussion in this Chapter to utilize the criticality evaluation presented in the Model FSV-1 packaging SAR (Ref. 6-1) as a basis for the criticality evaluation for the TN-FSV packaging. No new analyses have been performed because the previous analyses are applicable and bounding for the TN-FSV packaging. Many details of the Model FSV-1 analyses have not been duplicated here but are readily available in the SAR. The Model FSV-1 is a Fissile Class III package and therefore the TN-FSV shall also be classified as Fissile Class III.

6.1.2 Fuel Description

The fissile material is contained in fuel elements which are hexagonal in cross section with dimensions 14.2 in. across flats by 31.2 in. high as shown in Fig. 1-4. Each fuel element contains coolant and fuel channels which are drilled from the top face of the element. Fuel holes are drilled to within about 0.3 in. of the bottom face and are closed at the top by a 0.5 in. cemented graphite plug. The fuel channels occupy alternating positions in a triangular array within the element structure, are 0.5 in. in diameter and contain the active fuel.

The element structure consists of needle coke and/or isotropic graphite. The fuel itself is in the form of carbide particles coated with layers of pyrolytic carbon and silicon carbide. The fuel bed contains a homogeneous mixture of two types of particles, called fissile and fertile. Fresh fissile particles contain both thorium and 93.5% enriched uranium, while fresh fertile particles contain only thorium. The important parameters of fresh particles are:

Parameter	Fissile	Fertile
Nominal Th-U Ratio	3.6 or 4.25	All Th
Particle Composition	(Th/U)C2	Th C2
Average Fuel Particle Diam, µm	200	450
Average Total Coating Thickness, µm	130	140

Irradiated fuel elements contain, besides fission products, thorium, U-233, U-235, other uranium isotopes and a small quality of plutonium. In the fertile particles, the fissile material is essentially U-233, while the fissile particles contain the residual U-235 and bred U-233.

The effective fissile material enrichment (U-235/U+Th) in fresh fuel for the initial core and reload segments varies between 2% and 12% due to radial and axial fuel zoning requirements. The most reactive fresh fuel element contains a maximum of 1.4 Kg of 93.5% enriched uranium and about 11.3 Kg of thorium. Any irradiated elements will contain a smaller amount of fissile material, since the conversion ratio of the reactor is less than unity.

6.1.3 Results

The TN-FSV and Model FSV-1 packaging are very similar in size and materials of construction. The major difference is that the TN-FSV packaging has a lead shield and the FSV-1 has a depleted Uranium shield. However, both materials act as neutron reflectors/absorbers

as concerns reactivity. In fact, experiments (Ref. 6-2) have shown that both lead and depleted Uranium reflected system have approximately the same reactivity. Therefore, the FSV-1 results given below are also applicable to the TN-FSV packaging.

During the normal conditions of transport, the multiplication constant or Keff is 0.41.

During the hypothetical accident conditions with non-flooded spent fuel elements, the Keff is 0.41 and for the flooded spent fuel particle case the Keff is 0.89.

6.2 PACKAGE FUEL LOADING

6.2.1 Assumptions

The following assumptions were used for the FSV-1 criticality evaluation:

a) The fuel elements contain the most reactive fresh fuel composition anticipated for fuel shipment, i.e., a maximum of 1.4 Kg of 93.5% enriched uranium and about 11.3 Kg of thorium per element resulting in a maximum of 8.4 Kg of uranium and about 68 Kg of thorium per package.

b) The presence of burnable poison or other neutron absorbing material, other than U-235, U-238, thorium, silicon and graphite, is neglected.

c) The fuel is at room temperature.

d) All fission products are neglected.

These assumptions are all conservative. In general, spent fuel elements will contain considerably less fissile material. Also, all other fresh fuel element types are less reactive due to their lower

uranium contents and/or higher thorium content. Hence, elements with a maximum of 1.4 Kg of uranium and a thorium/uranium ratio of at least 8.1/1 are acceptable.

6.3 MODEL SPECIFICATION

a. Geometry

Two geometric models were used to evaluate the criticality situation for the FSV-1 shipping cask. The one-dimensional model is shown in Fig. 6-1, and assumes an infinitely long cylinder. This model is adequate as long as it can be assumed that the fuel is well contained within the fuel elements.

The two-dimensional geometric model, shown in Fig. 6-2, was used for the maximum criticality situation which includes fuel element breakage, internal flooding and an accumulation of fuel particles at the bottom of the cask.

b. Cross Sections

Cross sections were calculated with a GGC-4 code, a description of which is provided in the Appendix. The calculational methods and the basic nuclear cross section data are well established and in use for the high temperature gas-cooled reactor (HTGR) nuclear design.

c. Computer Codes

The DTF-IV transport code was used for one-dimensional and the GAMBLE-5 code was used for the two-dimensional calculations.

d. TN-FSV

The TN-FSV Packaging is very similar to the Model FSV-1 Cask. The TN-FSV is fabricated from stainless steel and lead while the FSV-1 is composed of stainless steel and depleted Uranium. Both casks consist of the heavy shielding (Pb or dpU) surrounded inside and out by shells of stainless steel. Dimensionally, the TN-FSV has a slightly larger diameter (~3 inches) than the FSV-1 due to the thicker lead shielding.

6.4 CRITICALITY CALCULATIONS

6.4.1 Normal Conditions of Transport

During normal transport not more than 6 fuel elements will be stacked end to end within the inner container. No hydrogenous or other moderating material besides the structural and coating graphite will be present.

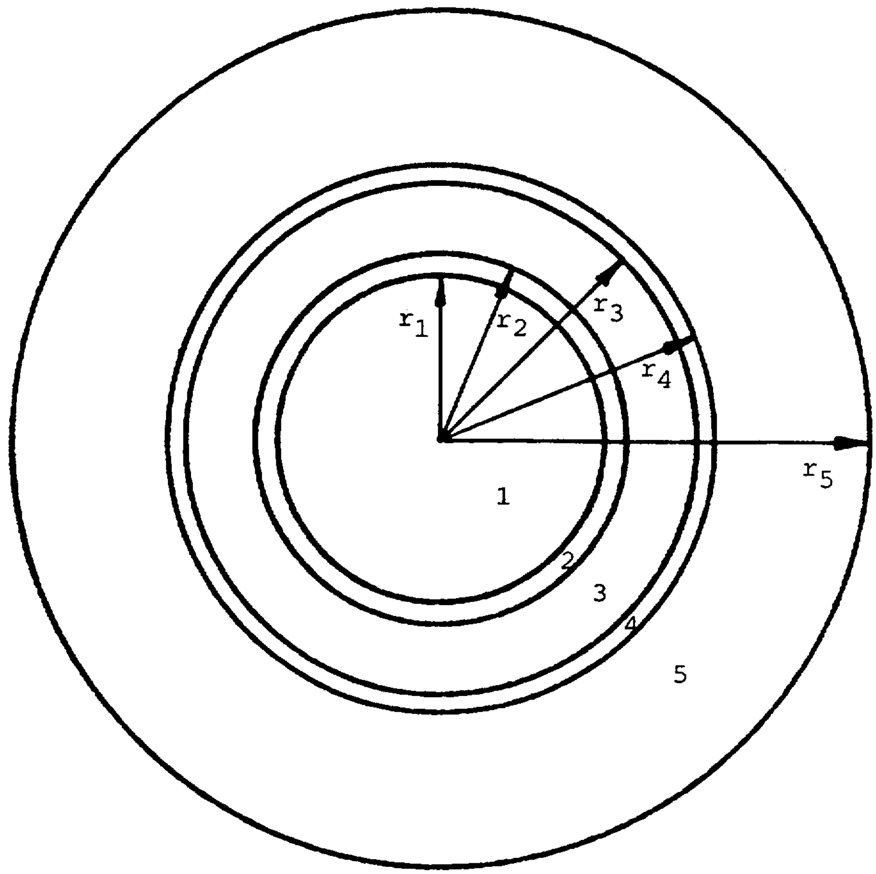
The fuel elements occupy 81.4% of the inner container of the shipping cask. The contents of the elements were homogenized over the inner container of the cask and used in one-dimensional transport calculations.

The calculated multiplication constant (Keff) for the FSV-1 in this condition, assuming an infinitely long container, is 0.41 and no criticality hazard is to be expected under any circumstances.

6.4.2 Hypothetical Accident Conditions

a. Unflooded Fuel

For the hypothetical accident conditions some breakage of the fuel elements has to be expected with possible accumulation fuel particles in a corner of the cask. Detailed drop test data for irradiated fuel elements,



REGION	REGION OUTER RADIUS	REGION CONTENTS
1	r_1 = 21.11 cm.	FUEL ELEMENT (+ WATER)
2	r_2 = 23.97 cm.	IRON
3	r_3 = 32.86 cm.	URANIUM-235 & URANIUM-238
4	r_4 = 35.40 cm.	IRON
5	r_5 = 55.40 cm.	VOID or WATER

FIGURE 6-1
MODEL FSV-1 CONFIGURATION E - ONE DIMENSIONAL MODEL

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE 6-2

MODEL FSV-1 CONFIGURATION - TWO DIMENSIONAL MODEL

however, are not available and some assumptions have to be made concerning the amount of particles released from the fuel elements. Assuming an upper limit of 20% particle loss and accumulation of these particles in coolant holes and the space between container wall and fuel element at the bottom of the container, the multiplication constant is calculated to be less than 0.41, with the cask completely immersed in water.

In the analysis it was assumed that the fuel particles fall out of the fuel channels into the coolant channels and spaces between element and container wall and form a homogeneous mixture with the remaining fuel element graphite. At 10% particle loss the lowest 24 cm of the cask would be filled with fuel particles and fuel elements. The remaining elements would have a correspondingly reduced fuel loading. At 20% particles loss, the lowest 48 cm would be filled with particles.

The following multiplication constants were calculated with the FSV-1 two-dimensional model:

No particle loss	Keff = 0.37
10% particle loss	Keff = 0.37
20% particle loss	Keff = 0.41

The first result agrees well with the data from the one-dimensional transport calculation. The other results show that partial particle loss from the fuel elements and subsequent particle accumulation at the bottom of the cask do not significantly increase the multiplication constant. Even if the whole cask were filled with particles only, the overall multiplication constant of the immersed cask would be less than 0.55. The presence of water at the outside of the cask has no significant effect on the criticality of the system. The iron and uranium

shield acts as a sink for thermal neutrons. Fast neutrons escaping from the fuel region are moderated in the water and absorbed before they can return to the fuel.

b. Flooded Fuel

In order to obtain an upper limit for the multiplication constant of the FSV-1 shipping cask under flooded conditions the following assumptions were made in addition to those stated in Section 6.2.1:

(1) All fuel particles leave the fuel elements (100% particle loss).

(2) The fuel particles accumulate in the fuel holes, coolant holes and void spaces between fuel elements and container wall.

(3) The graphite structure of the fuel elements stays intact allowing the highest concentration of fissile material in the available void space.

(4) All void space between the fuel particles and in the unfueled section of the cask is filled with water.

In particular, the following cases were considered for the FSV-1:

Case 1: All fuel particles accumulate at the bottom of the cask, filling the available void space. The particle packing fraction is 0.65. The overall height of the fueled section is 171 cm.

Case 2, 3, 4: The fuel particles float in water. A homogeneous mixture of water and particles occupies the available void space up to a height of 214 cm. 257 cm and 343 cm, respectively.

Case 5: The fuel particles float in water. A homogeneous mixture of water and particles fills all available void space in the shipping cask.

Fuel Containing Section

Case	Height (cm)	H/U-235	Keff
1	171	143	0.84
2	214	244	0.89
3	257	346	0.89
4	343	551	0.85
5	476	866	0.77

Figure 6-3 shows these results graphically. The most reactive situation, a multiplication constant of about 0.9, is obtained if the total fuel contents, in a mixture of water and particles, occupies the available void spaces of about half the cask (235 cm). From these results it is concluded that no critical arrangement can occur even if all the fuel leaves the fuel elements and the cask is completely flooded and that K_{eff} is less than 0.95.

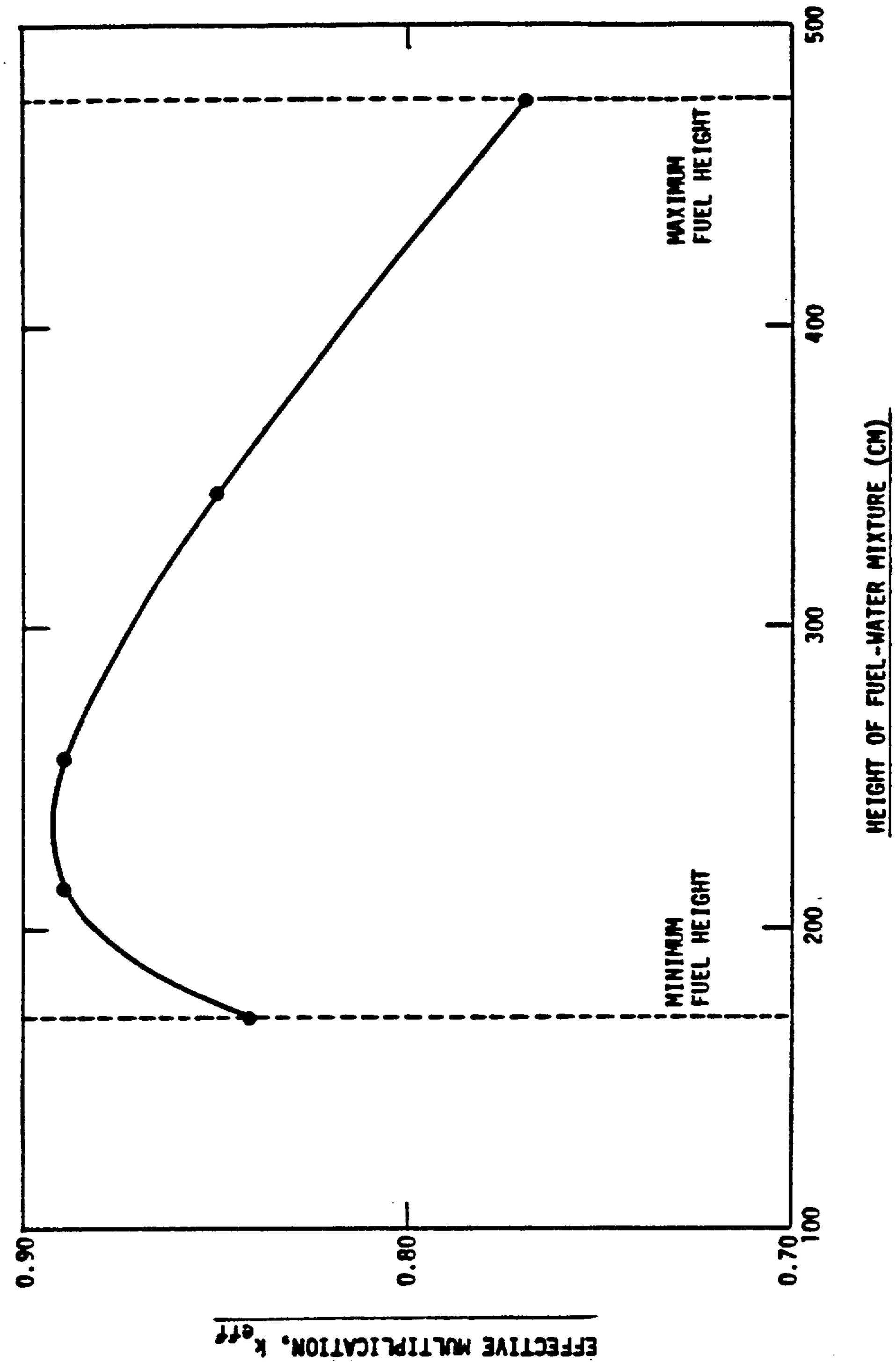


FIGURE 6-3

MODEL FSV-1 WITH INTERNAL FLOODING
CONTENTS: 8.4 Kg 93.5% ENRICHED URANIUM
68.0 Kg THORIUM-232

6.5 CRITICALITY BENCHMARK

The calculational methods used for the FSV-1 criticality analysis are essentially the same as those used for HTGR design and the analysis of the HTGR critical facility. Comparisons of experimental and calculational results are well documented in the Final Hazards Report for the Fort St. Vrain Reactor. These results indicate that the accuracy of the methods used is in the order $\pm$ 0.02 Δk.

6.6 REFERENCES

6-1 Consolidated Design Report for the Model FSV-1 Shipping Cask, Volumes I and II. GADR 55.

6-2 S. R. Biezman, et al. 'Criticality Experiments with Subcritical Clusters of 2.35 Wt% and 4.29 Wt% ^{235}U Enriched UO2 Rods in Water with Uranium or Lead Reflecting Walls' NUREG/CR-0796, Pacific Northwest Laboratories (1979).

CHAPTER SEVEN

OPERATING PROCEDURES

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CHAPTER SEVEN
OPERATING PROCEDURES

This chapter contains TN-FSV loading and handling procedure guidelines which show the general approach to cask operational activities. The information in this chapter will be used to prepare site specific procedures. Operational steps which must be performed in order to maintain the validity of cask transport regulations and safety analysis conclusions are identified by underlined procedural steps.

The procedures in this section are for those activities associated with the loading and transport of canistered spent fuel elements from the Fort St. Vrain High Temperature Gas Cooled Reactor. The loading shall be performed in the independent spent fuel storage installation and unloading of the cask shall be performed dry in a hot cell.

The final steps of cask acceptance testing are performed at the loading site prior to the cask being transported for the initial shipment.

7.1 PROCEDURES FOR LOADING PACKAGE

7.1.1 Receipt of Empty Package

7.1.1.1 Inspect and clean the tractor, semitrailer and shipping cask. Check for damage or irregularities and perform radiation survey.

7.1.1.2 Back the tractor and semitrailer into the truck bay.

7.1.1.3 Remove the personnel barrier.

7.1.1.4 Remove the impact limiter attachment bolts and remove the front and rear impact limiters using a suitable crane and three legged sling.

7.1.1.5 Release the trunnion tie-downs.

7.1.1.6 Release the front saddle tie-downs.

7.1.1.7 Engage the cask lifting apparatus in the recessed lifting sockets. Use handcrank to lock the balls in the sockets.

7.1.1.8 Raise the shipping cask to the vertical position. This may require moving the trailer forward while the cask is being lifted.

7.1.1.9 If necessary, clean the cask external surfaces of road dirt.

7.1.1.10 Raise the cask to above the Cask Load and Unload Port (CLUP).

7.1.1.11 Lower the cask to seat on the CLUP.

7.1.1.12 Install the bottom restraint on the shipping cask.

7.1.1.13 Disengage the cask lifting apparatus from the cask.

7.1.1.14 Lower the CLUP hatch cover into place.

7.1.2 Dry loading

Note: The drain port is not used for loading operations.

7.1.2.1 Install the lid lifting attachments.

7.1.2.2 Remove and visually inspect for damage the twelve (12) socket bolts that hold the lid in place. Replace any bolt found to have stripped or galled threads or any visible deformation of the head or shank. Minor nicks, scrapes or upsets from normal wrench contact are not cause for replacement.

7.1.2.3 Lift the lid from the cask and store.

7.1.2.4 Examine the sealing surface.

7.1.2.5 Visually inspect the cavity of the inner container for any damage or debris. If any is noted, evaluate and take corrective action if necessary.

7.1.2.6 Load the canister with six (6) spent fuel elements into the cask. <u>Record the contents on the cask loading report</u>.

7.1.2.7 <u>Inspect the o-rings in the lid for damage and replace if defects are noted. Record inspection results on the cask loading report</u>.

7.1.2.8 Transfer the lid to a position directly over the cask cavity. Establish correct lid orientation using the orientation marks and lower the lid until fully seated. Visually verify the lid for proper installation.

7.1.2.9 <u>Inspect the lid bolts. Replace defective bolts and note any defect indications on the cask loading report.</u> Apply a light coating of Nuclear Grade Neolube to bolt threads and install all 12 lid bolts. Tighten to hand tight. Torque all lid bolts to 130 ft-lbs. in several stages. Follow an approved torquing sequence. Remove the lid o-ring port plug.

7.1.2.10 Install the Leak Test System (LTS) to the lid gasket port and evacuate the lid gasket interspace until the pressure is reduced to 10, +2, -0 mbar.

7.1.2.11 <u>Perform a pressure rise leakage test for assembly verification of the cask lid</u>. The leakage rate is calculated by

$$L_r = \frac{V * \Delta P * 298}{t * 1013 * T}$$

where V = test volume (cc)

ΔP = measured pressure difference (mbar)

t = elapsed time for the test (sec)

and T = temperature of test (°K)

It is assumed that over the relatively short duration of the test (1-2 minutes), the change in temperature is insignificant.

The acceptance criteria is a leakage rate no greater than 1×10^{-3} std-cm^3/sec. <u>Replace the lid o-ring port plug</u>.

7.1.2.12 Remove the Vent cover, and evacuate the cask using the leak test system. Back fill with dry air.

7.1.2.13 Inspect the vent cover, seal and bolts for damage and replace if defects are noted. Record inspection results on the cask loading report.

7.1.2.14 <u>Install the Vent port cover and bolts</u>. Torque the bolts to 20 in-lbs.

7.1.2.15 Place the test bell over the Vent cover and use the LTS to reduce the pressure between the Vent port O-ring and the O-ring on the test bell to 7-10 mbar. Isolate the vacuum pump and <u>perform a pressure rise leakage rate test of the vent port cover</u>. The acceptance criteria is a leakage rate no greater than 1×10^{-3} std-cm^3/sec.

7.1.2.16 If the sum of the leakage rates from both tests is greater than the allowable rate, the leakage area shall be identified, repaired as needed and the test repeated until the acceptance criteria is satisfied.

Note: Other tests such as a bubble test may be used in lieu of the pressure rise test. The test must have a sensitivity of at least 10^{-1} std cm^3/sec.

7.1.2.17 Remove the lid lifting attachments.

7.1.2.18 Engage the cask lifting apparatus in the recessed lifting sockets. Use the handcrank to lock the balls in the sockets.

7.1.2.19 Remove the restraint from the bottom of the cask.

7.1.2.20 Open the sliding hatch above the truck bay.

7.1.2.21 Lift the cask from the fuel loading port, and lower it into the truck bay.

7.1.2.22 Align the trunnions of the cask with the support pedestals on the trailer.

7.1.2.23 Place trunnions on transport vehicle rear trunnion supports and rotate cask from the vertical to horizontal position.

7.1.2.24 Install and torque the rear trunnion tie-downs and the front saddle tie-downs.

7.1.2.25 Install the front and rear impact limiters and torque attachment bolts diametrically to 40 ft-lbs. Repeat torquing sequence to 80 ft-lbs.

7.1.2.26 Install security seal on the front impact limiter.

7.1.2.27 Perform final radiation and contamination surveys to assure compliance with 10CFR71.47 and 71.87.

7.1.2.28 Apply appropriate labels to the cask and placards to the vehicle in accordance with 49CFR172.

7.1.2.29 Install Personnel Barrier.

7.1.2.30 Prepare final shipping documentation.

7.1.2.31 Release the loaded cask for shipment.

7.2 PROCEDURES FOR UNLOADING PACKAGE

7.2.1 Receipt of Loaded Package

7.2.1.1 Upon arrival of the loaded cask at the receiving site, perform receipt inspection. Inspect for damage, verify security seals are intact and perform radiation survey.

7.2.1.2 Verify that placards, labels and shipping papers are in place and correct.

7.2.1.3 Inspect and clean the tractor, trailer and cask as required

7.2.1.4 Remove the personnel barrier.

7.2.1.5 Remove the security seal from the front impact limiter.

7.2.1.6 Remove the impact limiter attachment bolts and remove the front and rear impact limiters using a suitable crane and three legged sling.

7.2.1.7 Release the front saddle and rear trunnion tie-downs.

7.2.1.8 Attach the cask lifting apparatus to the crane hook.

7.2.1.9 Engage the cask lifting apparatus in the recessed lifting sockets. Use handcrank to lock the balls into the sockets.

7.2.1.10 Lift the cask slowly to the vertical position.

7.2.1.11 Move the cask to the unloading area and lower in the vertical position using a suitable support. Disengage cask lifting apparatus.

7.2.1.12 Attach suitable lifting fixture to three (3) threaded holes in the lid.

7.2.1.13 Remove the vent cover.

7.2.1.14 Install the cavity gas sampling adapter in the vent test port.

7.2.1.15 Operate an evacuation system to draw a sample of the cavity gas from the cask.

7.2.1.16 Disconnect the evacuation system and install the vent cover.

7.2.1.17 Attach a lifting fixture to the lid using the three (3) threaded inserts.

7.2.1.18 Remove the twelve lid bolts.

Caution: The next three steps must be performed remotely.

7.2.1.19 Remove the lid.

7.2.1.20 Use a suitable grapple to remove the canister containing six (6) spent fuel elements from the cask cavity.

7.2.1.21 Visually examine the cask sealing surfaces and interior for any damage or debris.

7.2.1.22 Visually examine the lid, especially the o-ring seals for any damage.

7.2.1.23 Transfer the lid to a position directly over the cask cavity. Establish correct lid orientation using the orientation marks and lower the lid until fully seated. Visually verify the lid for proper installation.

7.2.1.24 <u>Inspect the lid bolts. Replace defective bolts and note any defect indications on the cask loading report.</u> Apply a light coating of Nuclear Grade Neolube to bolt threads and install all 12 lid bolts. Tighten to hand tight. Torque all lid bolts to 130 ft-lbs. in several stages. Follow an approved torquing sequence.

7.2.1.25 Inspect the vent cover, seal and bolts for damage and replace if defects are noted. Record inspection results on the cask loading report.

7.2.1.26 <u>Install the Vent port cover and bolts</u>. Torque the bolts to 20 in-lbs.

7.2.1.27 Remove the lid lifting attachments.

7.2.1.28 Engage the cask lifting apparatus in the recessed lifting sockets. Use the handcrank to lock the balls in the sockets.

7.3 PREPARATION OF EMPTY PACKAGE FOR TRANSPORT

7.3.1 Raise the cask into the vertical position and return to the trailer.

7.3.2 Position the cask above the trunnion supports and lower the cask.

7.3.3 Rotate cask from the vertical to the horizontal position

7.3.4 Install and torque the rear trunnion tie-downs and the front saddle tie-downs.

7.3.5 Perform radiation and contamination survey.

7.3.6 <u>Install the front impact limiter and torque attachment bolts diametrically to 40 ft-lbs. Repeat torquing sequence to 80 ft-lbs</u>.

7.3.7 <u>Apply appropriate transport labels to the cask and placards to the vehicle</u>.

7.3.8 Install personnel barrier.

7.3.9 Release loaded cask for shipment.

CHAPTER EIGHT

ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

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CHAPTER EIGHT
ACCEPTANCE AND MAINTENANCE PROGRAMS

This chapter describes the activities to be performed in compliance with Subpart G of 10CFR71 to assure that the TN-FSV cask conforms to the requirements of this Safety Analysis Report and remains in conformance following loading.

8.1 ACCEPTANCE TESTS

The following reviews, inspections and tests shall be performed on the TN-FSV cask prior to the first transport. Most of these inspections and tests will be performed at the Fabricator's facility prior to delivery to the loading site. These tests will be performed in accordance with written procedures prepared by the Fabricator and approved by Transnuclear.

The shield test (paragraph 8.1.5) shall be performed at the site following loading and prior to transport. This test shall be performed in accordance with approved written procedures.

8.1.1 Visual Inspection

After fabrication and prior to first use, a visual inspection shall be performed to verify that all accessible cask surfaces have no damage (cracks, dents, gouges), are free of grease, oil or other contamination, that there are no gouges, cracks or scratches on cask sealing surfaces which may result in unacceptable leakage, and that all cask components are in acceptable condition for use.

8.1.2 Structural and Pressure Tests

All of these tests shall be performed by the Fabricator. Prior to use, some tests, as indicated, will be repeated.

8.1.2.1 Lid Lifting Attachment Load Test

A force equal to twice the weight of the lid will be applied vertically to the lid lifting holes using a sling assembly for a period of ten (10) minutes.

The lid lifting attachment bolt holes shall be load tested in the following manner. A 3/4-10 UNC-2A eyebolt shall be inserted into each of the three tapped holes in the lid surface. The weights shall be added to the lid such that the total load of the lid and added weight is twice the lid weight. The eyebolts shall be attached to a sling system with a lifting angle of approximately 60 degrees from horizontal. The lid shall be lifted using the sling system and held for a period of ten (10) minutes.

At the conclusion of the test, the bolt holes shall be:

a) Visually examined for defects and permanent deformations.

b) Checked with a go/no-go gauge to verify that no damage to the threads have occurred.

8.1.2.2 Lifting Socket Load Test

a) A force equal to 1.5 times the design lift load will be applied for a period of 10 minutes to the lifting sockets with the cask in a vertical position.

b) Following completion of the load test the internal surface of the sockets shall be examined by the liquid penetrant method for defects. Acceptance standards shall be in accordance with Articles NF-5350 of Section III of the ASME Boiler and Pressure Vessel Code.

8.1.2.3 Hydrostatic Test

After the shielding integrity test, the TN-FSV cask body will be hydrostatically tested using demineralized water in accordance with the requirements of the ASME B&PV Code, Section III, Article NB-6200.

The test pressure of 45 psig will be maintained for a minimum time of ten (10) minutes. The cask body and closures will then be examined for any deformations or leakage.

8.1.2.4 Cask Weight Measurements

The assembled cask, as well as major individual components shall be weighed with a precision of $\pm$ 0.5 percent.

8.1.3 Leak Tests

8.1.3.1 Containment System Fabrication Verification

A Containment System Fabrication Verification leakage rate test of the TN-FSV shall be performed at the Fabricator's facility in accordance with the requirements of ANSI N14.5 and Section V of the ASME B&PV Code.

The following tests will be performed in accordance with approved, written procedures on the containment boundary. The test cavities will be evacuated to a pressure of less than 10^{-2} mbar. The total leakage rate for the containment boundary shall be no greater than 1×10^{-3} std cc/sec with a test procedure sensitivity of no greater than 5×10^{-4} std cc/sec. The test method used shall meet this sensitivity requirement and ANSI N14.5.

a) A preliminary test shall be performed on the interspace between the inner and outer o-rings of the lid through a test port using an o-ring sealed test connector. This will show that the lid is seated properly.

b) The next test shall be performed on the containment boundary formed by the cask cavity and the Drain Port Cover and seal (Hansen coupling removed), and the inner O-ring of the lid. Evacuation shall be through the Vent port using an O-ring sealed test connector.

c) The next test shall be performed after the Vent port cover is installed. The O-ring seal of the cover shall be leak tested using a test bell. The test bell fits over the vent cover and has an O-ring which seals against a machined surface on the cask surface.

An acceptable alternative test method is to perform the test in paragraph b) through the drain port and then test the drain port cover and seal by the method described in paragraph c). The Hansen coupling must be removed for the test described in paragraph b).

8.1.3.2 Impact Limiter Leakage Rate Test

Following final closure welding, each impact limiter will be tested 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 bubble leak test methods. The differential pressure shall be limited to less than 3 psi.

8.1.3.3 Humidity Test

After performing the impact limiter leakage rate test, a humidity test will be performed to determine that there is no in-leakage of water into the impact limiter container during fabrication. The dew point of a gas sample from the impact limiter container will be determined. The dew point measured shall be less than the equilibrium temperature of the impact limiter. The difference between the measured dew point and the impact limiter wall temperature shall be greater than twice the accuracy of the humidity test system.

8.1.4 Component Tests

Installation and removal tests will be performed for the lid, impact limiters, drain and vent port covers, and other fittings and inserts.

Each component will be observed for difficulties in installation and removal. After removal, each component will be visually examined for indications of deformation, galling, ease of use and proper functioning. Any such defects will be corrected prior to acceptance of the cask.

8.1.5 Tests for Shielding Integrity

A gamma scan or equivalent test shall be performed on the cask after lead installation to detect any shielding deficiencies from lead voids equal to or greater than ± 5% of shielding thickness. The test shall be performed on a maximum grid spacing of three (3) inches.

Following initial loading of the TN-FSV cask, a shield effectiveness test shall be performed prior to delivery to a carrier for transport. Measurements shall be made of the neutron and gamma dose rates of the loaded package to verify that the loaded cask meets the transport dose rate limitations of 10CFR71 and the applicable USDOT regulations.

8.1.6 Thermal Acceptance Tests

There are no thermal tests required for the TN-FSV.

8.2 MAINTENANCE PROGRAM

8.2.1 Structural and Pressure Tests

There are no periodic structural tests required on the TN-FSV.

8.2.2 Leak Tests

After the TN-FSV is loaded and before it is released for transport, a series of pressure rise tests shall be performed on the cask openings to verify proper assembly. See Chapter 7.0 for details of these tests.

No leak tests will be done on the empty cask before it is shipped.

After the third use and every twelve months thereafter, the Containment System Fabrication Verification Test, Section 8.1.3.1, shall be repeated, unless the cask is not is service. Prior to use, the cask shall have been tested within the preceding 12-month period.

8.2.3 Subsystems Maintenance

This section does not apply.

8.2.4 Valves, Rupture Discs and Gaskets on Containment Vessel

All gaskets and the Drain port Hansen quick connect plug will be replaced prior to the third use test and annually thereafter unless the cask is not in service. Prior to use, the maintenance shall have been completed within the preceding 12-month period.

No other maintenance is required prior to transport.

8.2.5 Shielding

A shielding integrity test using gamma scan or equivalent shall be performed after fabrication, and a shield effectiveness test shall be performed following initial loading. See paragraph 8.1.5. Dose rate measurements are required to be taken prior to each shipment. There are no periodic shield effectiveness tests required.

8.2.6 Thermal

There are no thermal tests required for the TN-FSV.

8.2.7 Miscellaneous

The lid bolts, vent cover bolts and drain cover bolts (if removed) shall be replaced after every 250 round trip shipments to preclude fatigue failure.