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TITAN Legal Weight Truck Cask Preliminary Design Report

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3. THERMAL EVALUATION

3.1 Discussion

Design requirements specify that under the Heat condition or test, a Normal Condition of transport, the peak or maximum cladding temperature must remain below 716°F, and the cask must remain leak tight. During and after the hypothetical thermal accident condition, the cask must maintain its containment integrity as specified in 10 CFR Part 71 (Reference 3.1.1). The cask is sealed by two elastomeric O-rings which are located between the closure lid and the cask cylinder. In order for the cask to maintain its integrity, temperatures at these O-ring locations should be kept under 500°F.

The analysis of the cask for these two conditions was performed with TRUMP (Reference 3.1.2), a general purpose computer program for transient and steady-state temperature distribution in multidimensional systems. Two separate TRUMP models were used since the limiting temperatures occur in different parts of the cask for the normal and accident conditions. The spent fuel decay heat load of 580 W per assembly was assumed in both evaluations.

The results of this analysis indicated that, under the Normal Heat Condition, the maximum cladding temperature would be 423°F, well below the limiting temperature of 716°F, and that in the course of the 30 minute thermal accident, the maximum temperature at the O-ring locations would be 258°F. Thus, the current design meets the thermal limits when subjected to the conditions as specified in paragraphs 71.71 and 71.73 of 10 CFR Part 71.

3.2 Summary of Thermal Properties of Materials

Thermal properties of all materials used in this analysis are listed in Table 3.2-1.

3.3 Technical Specifications of Components

O-ring seal temperature ---- less than 500°F
Peak cladding temperature ---- less than 716°F

Table 3.2-1

Thermal Properties of Materials

Thermal Property of Materials (Constant Property)

	DENSITY (LB/ft^3)	SPECIFIC HEAT (Btu/lb-°F)	THERMAL CONDUCTIVITY (Btu/sec-ft-°F)
ALUMINUM	168.0	0.208	See Table 3.1-b
TITANIUM	280.0	Table 3.1-b	"
URANIUM METAL	1176.8	"	"
BORO-SILICON	100.224	0.04	0.0003889
STAINLESS STEEL	485.0	Table 3.1-b	Table 3.1-b
FUEL (UO2)	684.5	0.059	"

Thermal Property of Materials (Temperature Dependent)

ALUMINUM

TEMP. (°F)	THERMAL CONDUCTIVITY (Btu/sec-ft-°F)
70.0	0.026694
100.0	0.0269167
150.0	0.0272222
200.0	0.02750
250.0	0.027722
300.0	0.0279444
350.0	0.0281389
400.0	0.028305
2000.0	0.028500

FUEL (UO2)

TEMP. (°F)	THERMAL CONDUCTIVITY (Btu/sec-ft-°F)
200.0	0.00125
400.0	0.000972
600.0	0.000778
800.0	0.000694
1000.0	0.00061
1200.0	0.000556
2400.0	0.000306

Table 3.2-1
Thermal Properties of Materials (Continued)

Thermal Property of Materials (Temperature Dependent)

TITANIUM

TEMP. (°F)	SPECIFIC HEAT (Btu/lb-°F)	TEMP. (°F)	THERMAL CONDUCTIVITY (Btu/sec-ft-°F)
68.0	0.13	68.0	0.0012222
400.0	0.14	200.0	0.0013056
800.0	0.16	400.0	0.0014722
1200.0	0.18	600.0	0.0016944
1600.0	0.21	800.0	0.0019722
		1000.0	0.00225
		1200.0	0.0025833

DEPLETED URANIUM

TEMP. (°F)	SPECIFIC HEAT (Btu/lb-°F)	THERMAL CONDUCTIVITY (Btu/sec-ft-°F)
72.0	0.028	0.0047778
400.0	0.0316	0.0048611

STAINLESS STEEL

TEMP. (°F)	SPECIFIC HEAT (Btu/lb-°F)	THERMAL CONDUCTIVITY (Btu/sec-ft-°F)
300.0	0.1170	0.00250
600.0	0.1300	0.00280
900.0	0.1370	0.00319
1200.0	0.14	0.00353
1500.0	0.1400	0.00389

3.4 Thermal Evaluation for Normal Conditions of Transport

3.4.1 Thermal Model

3.4.1.1 Analytical Model

The 2-D 180° model shown in Figure 3.4-1 of the cask cylinder and basket structure system and the axi-symmetric R-Z model with a detail description in the cask's closure lid area shown in Figure 3.4-2, were used for the steady-state analysis. In the 180° model, the cask cylinder section is represented by 16x7 (circumferential x radial) nodes (elements) with appropriate gap conductance between the titanium shells and the depleted uranium. The inner basket model has a stainless steel grid structure, stainless steel liners and boral plates. The gap conductances were applied to the liner/Boral, Boral/basket and the rib/cask interfaces. The gap conductance values are listed in Table 3.4-1. In the R-Z model, the aluminum honeycomb impact limiter is explicitly modelled with the correct aluminum content. The impact limiter is thermally connected to the cask surface through a gap conductance and radiation heat transfer.

The honeycomb impact limiter is divided into nodes in the longitudinal and transverse directions. Each node has the correct volume content of aluminum. The longitudinal connection is modeled with the connection area equal to the honeycomb's aluminum cross-section area and a connection length equal to the actual length of honeycomb. The connection in the transverse direction is modeled with the connection area equal to the cross section area of honeycomb in the transverse direction and a connection length equal to the actual heat path length which is 4/3 of the physical distance between the nodes. The 4/3 distance factor is the ratio of the aluminum path length to the straight line distance within the honeycomb structure.

The following assumptions were made for the analysis of the LWT cask for the normal heat condition:

1. The cask contains three Westinghouse PWR spent fuel assemblies with each assembly generating 580 W of decay heat.

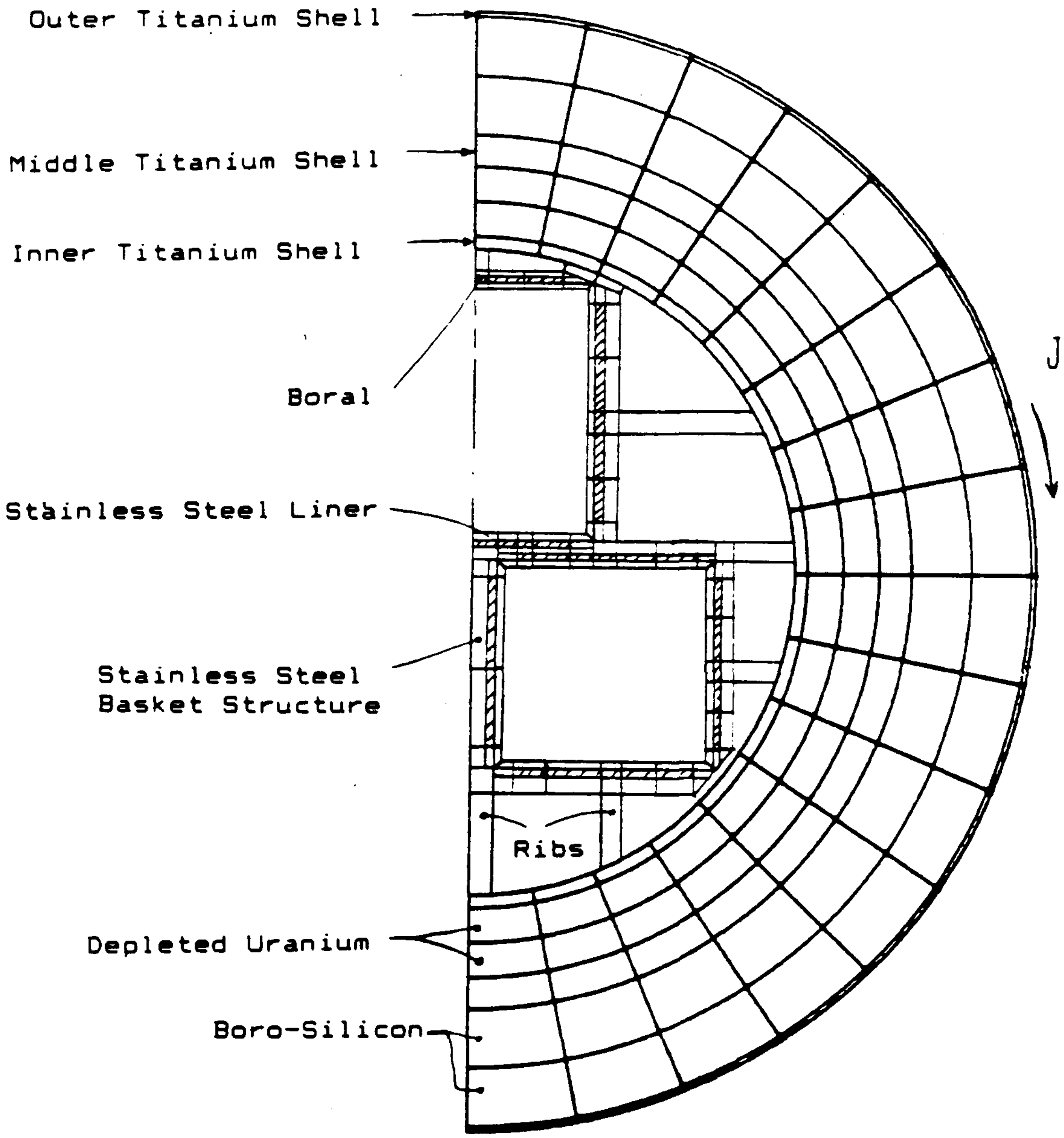

Figure 3.4-1. 2-D 180° Model for Steady-State Analysis

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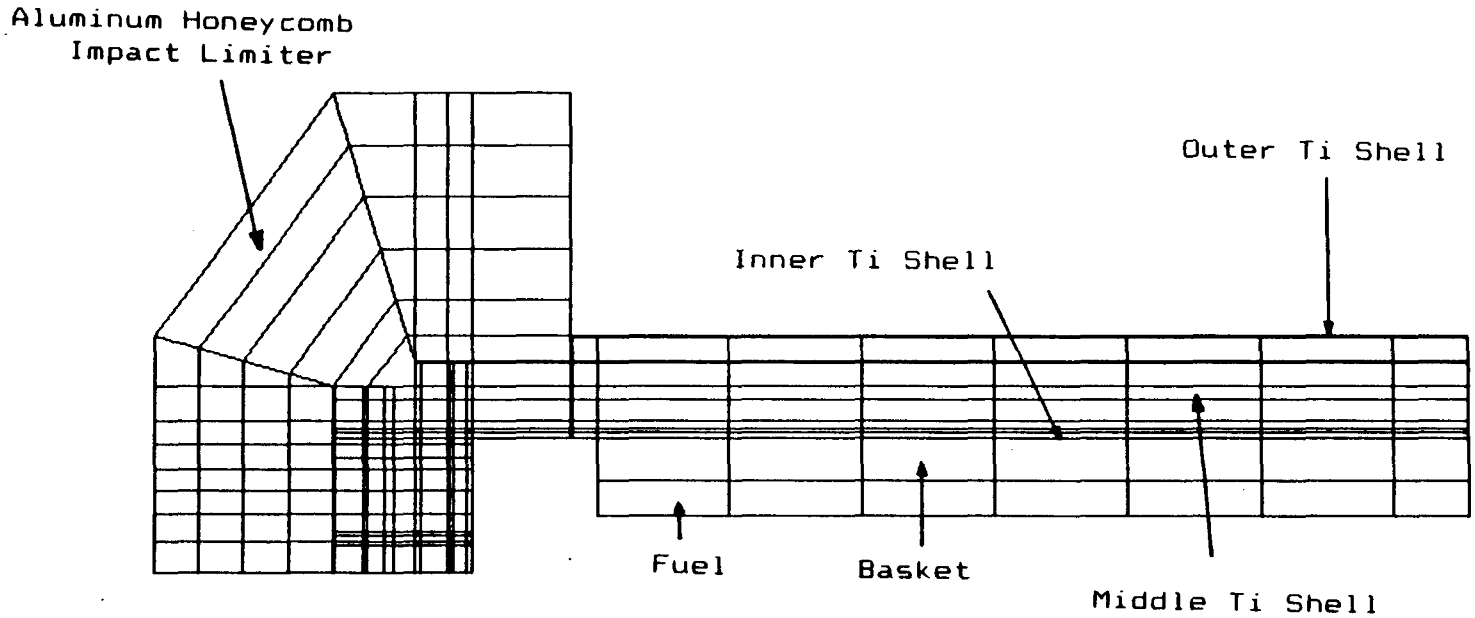

Figure 3.4-2. 2-D Axi-symmetric Cask Model for Transient Analysis

Table 3.4-1

Gap Conductance Applied to Cask Model

Interface Location	Gap Conductance (Btu/hr-ft^2-°F)	Gap Dimension
Basket/Cask		
Top	44.0	He - 0.030 in.
:	52.8	He - 0.025 in.
:	52.8	He - 0.025 in.
:	66.0	He - 0.020 in.
:	88.0	He - 0.015 in.
:	88.0	He - 0.015 in.
:	132.0	He - 0.010 in.
:	264.0	He - 0.005 in.
:	264.0	He - 0.005 in.
Bottom	660.0	He - 0.002 in.
Basket/Boral	264.0	He - 0.005 in.
Boral/Liner	264.0	He - 0.005 in.
Depleted Uranium /Titanium	7.0	Air - 0.030 in.
Cask/Impact Limiter	3.5	Air - 0.060 in.

2. The axial power peaking factor of 1.2 was used for the 180° model and the axial peaking factor shown in Figure 3.4-3 was applied to the R-Z model.

3. The ambient air temperature is 100°F at steady state.

4. Natural convection and radiation cooling exist at the outer surface of the cask.

5. A uniform heat flux of 122.88 Btu/hr-ft^2, derived from Paragraph 71.71 of 10 CFR Part 71, was applied to the cylindrical portion and other curved surfaces of the cask. This avoids the need to consider the periodicity of the solar insolation and results in a conservative estimate of the temperature distribution (References 3.1.1 and 3.4.1).

6. The radiation absorptance and the emissivity of the titanium cask surface are 0.8 and 0.2, respectively.

7. The radiation heat transfer between the cask inner wall and basket and the natural convection heat transfer by the helium in the cask cavity were neglected.

Based on these assumptions, the following boundary conditions were applied to the model:

- The uniform heat flux of 122.88 Btu/hr-ft^2 was applied to the outer surface of the cask cylinder with a surface radiation absorptance of 0.8.

- The natural convective heat transfer coefficient (Reference 3.4.2) at the surface was:

  $$H_{conv} = 0.18 * (DT)^{(1/3)} \quad (Btu/hr\text{-}ft^2\text{-}°F)$$

  where, DT is the temperature difference between the cylinder surface and the air, which had an ambient temperature of 100°F.

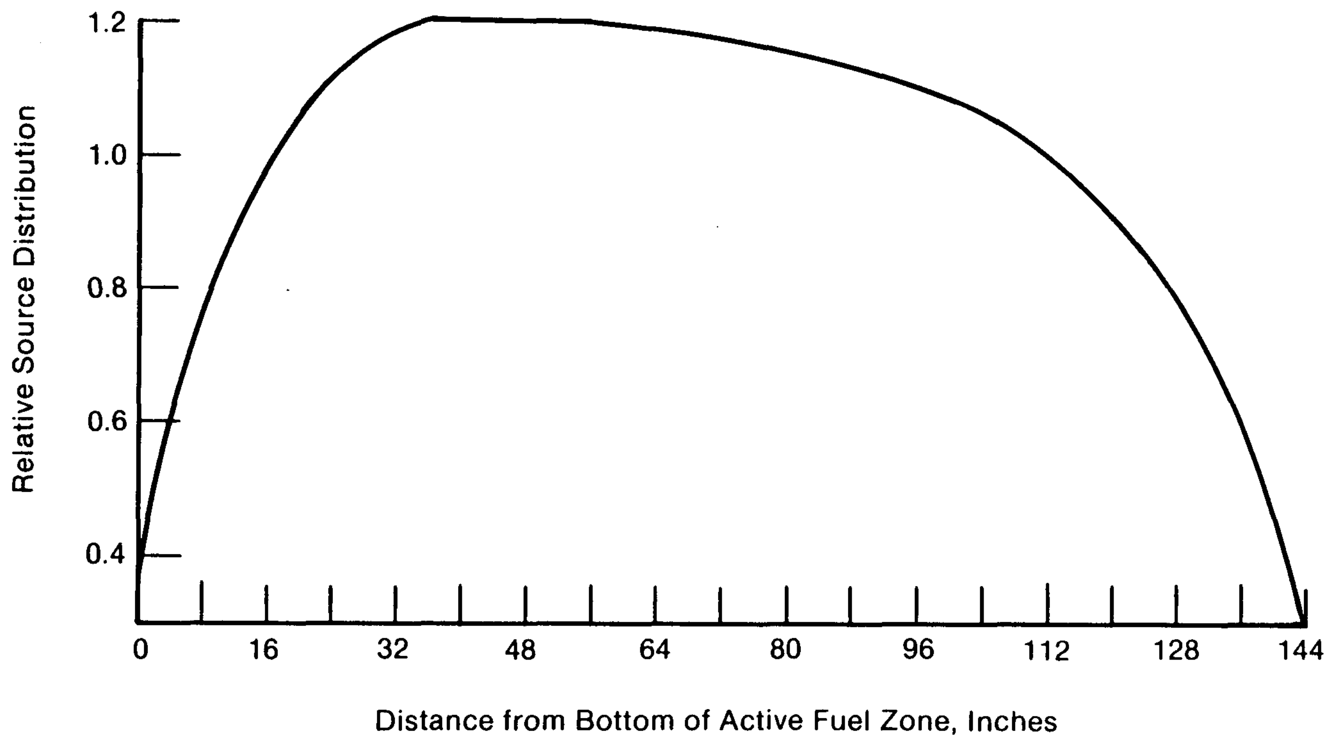

Figure 3.4-3. Axial Power Distribution for Spent Fuel

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- An emissivity of 0.2 was applied at the outer surface of the cask for cooling by radiation.

3.4.2 Maximum Temperatures

The radial and circumferential temperature distributions in the cask were computed using the model shown in Figure 3.4-1. The results are shown in Figure 3.4-4 and 3.4-5, respectively. As can be seen in the figure, the temperatures range from 280°F at the inner titanium shell to 225°F at the outer titanium shell. The basket and liner temperature distribution are shown in Figures 3.4-6 and 3.4-7.

The maximum cladding temperature was estimated by the following correlation, (Reference 3.4.3):

$$T = T_{can} + Q * (243.26 - 0.6752 * T_{can} + 0.0006677 * T_{can}^2)$$

where Q is an assembly power in Kw (=0.58), and T_{can} is the maximum liner temperature.

For the condition considered, T_{can}, the maximum liner temperature, was found to be 374°F resulting in the maximum cladding temperature of 422.8°F which satisfies the limiting temperature for the cladding of 716°F. Selected radial and axial temperature distributions in the top half of the cask are shown in Figure 3.4-8 and 3.4-9. These distributions were taken at locations shown in Figure 3.4.10. The temperatures were computed using the model shown in Figure 3.4-2. As seen in Figure 3.4-8, the temperature at the seal location for the Normal Heat Condition is expected to be approximately 200°F.

3.4.3 Minimum Temperatures

The minimum temperature seen by the cask is the Normal "Cold" Condition of transport per Paragraph 71.71(c)(2) of 10 CFR Part 71, which specifies an ambient temperature of -40°C (-40°F) in still air and shade. For the cold environment of -40°F ambient temperature, Regulatory Guide 7.8 (Reference 3.4.4) requires that insolation be ignored, the decay heat is zero and the internal pressure is a minimum (atmospheric). Thus the minimum temperature for the cask is -40°F.

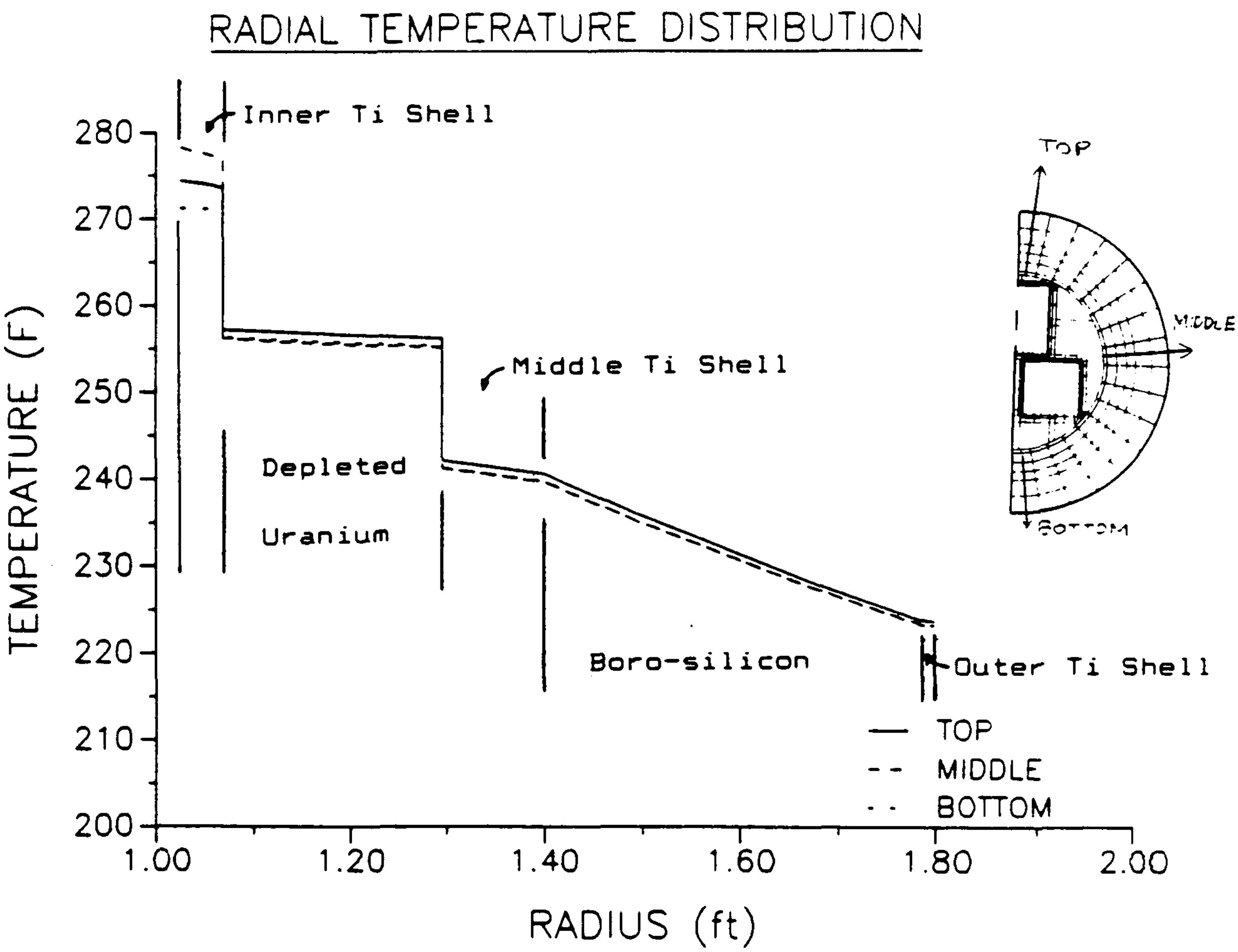

Figure 3.4-4. Radial Temperature Distribution in Cask

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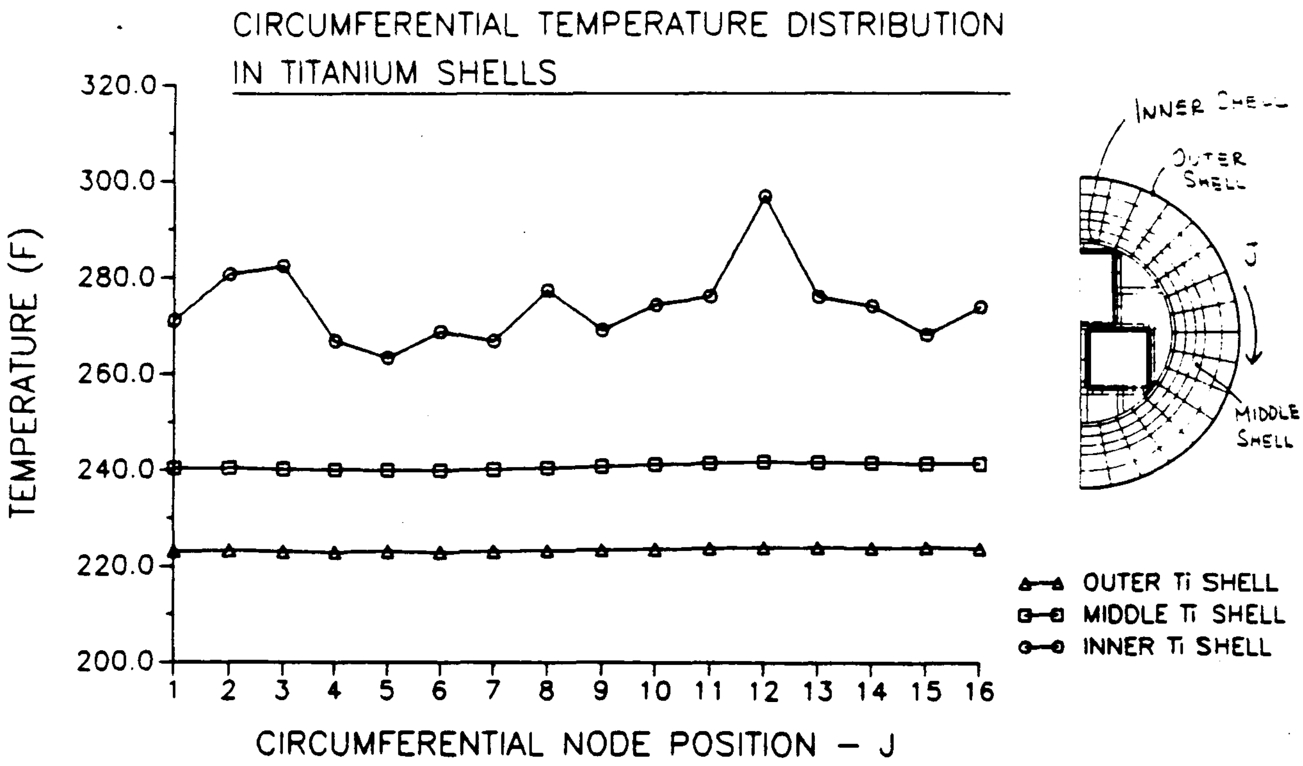

Figure 3.4-5. Circumferential Temperature Distribution in Cask

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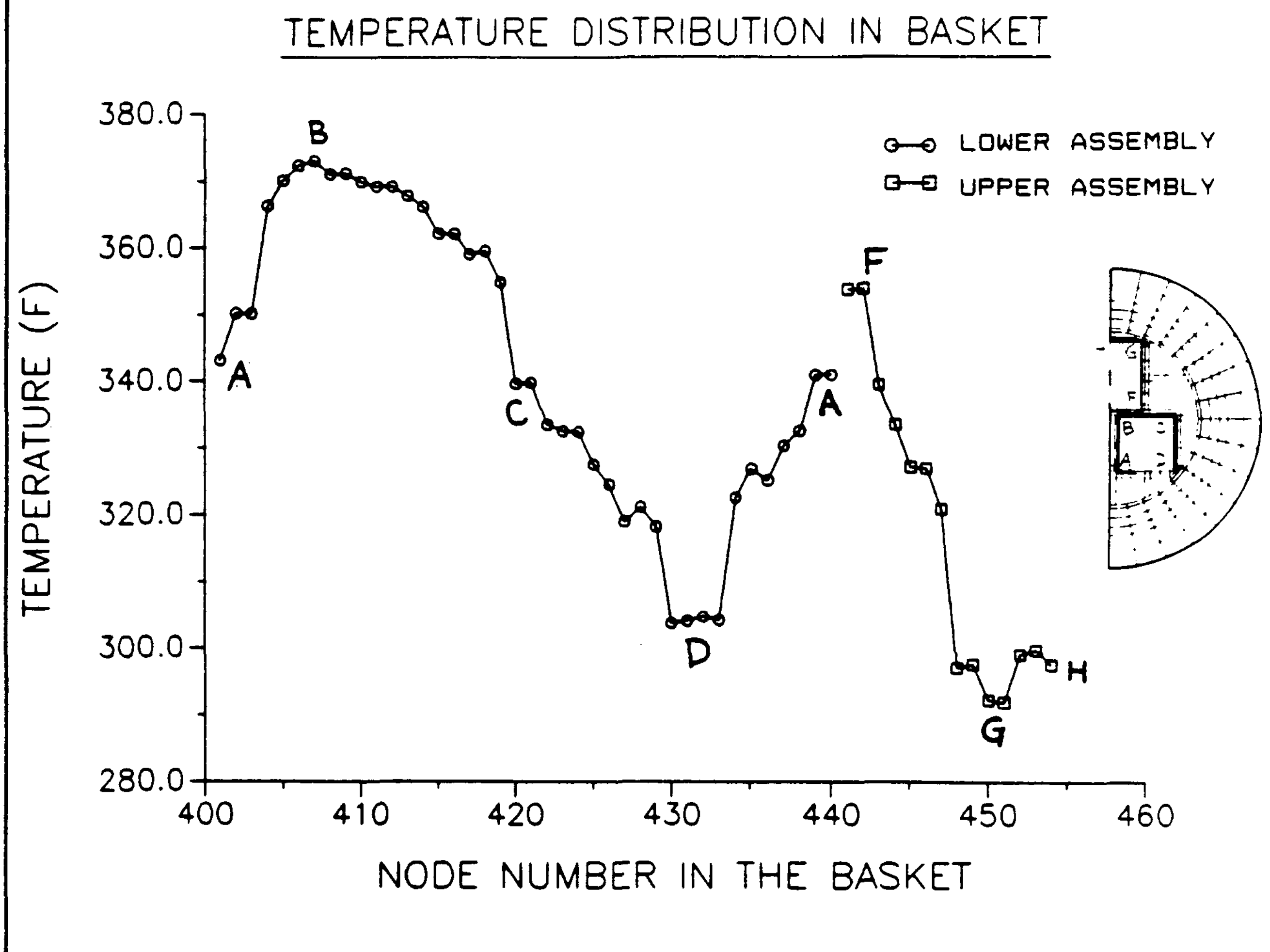

Figure 3.4-6. Temperature Distribution in Basket

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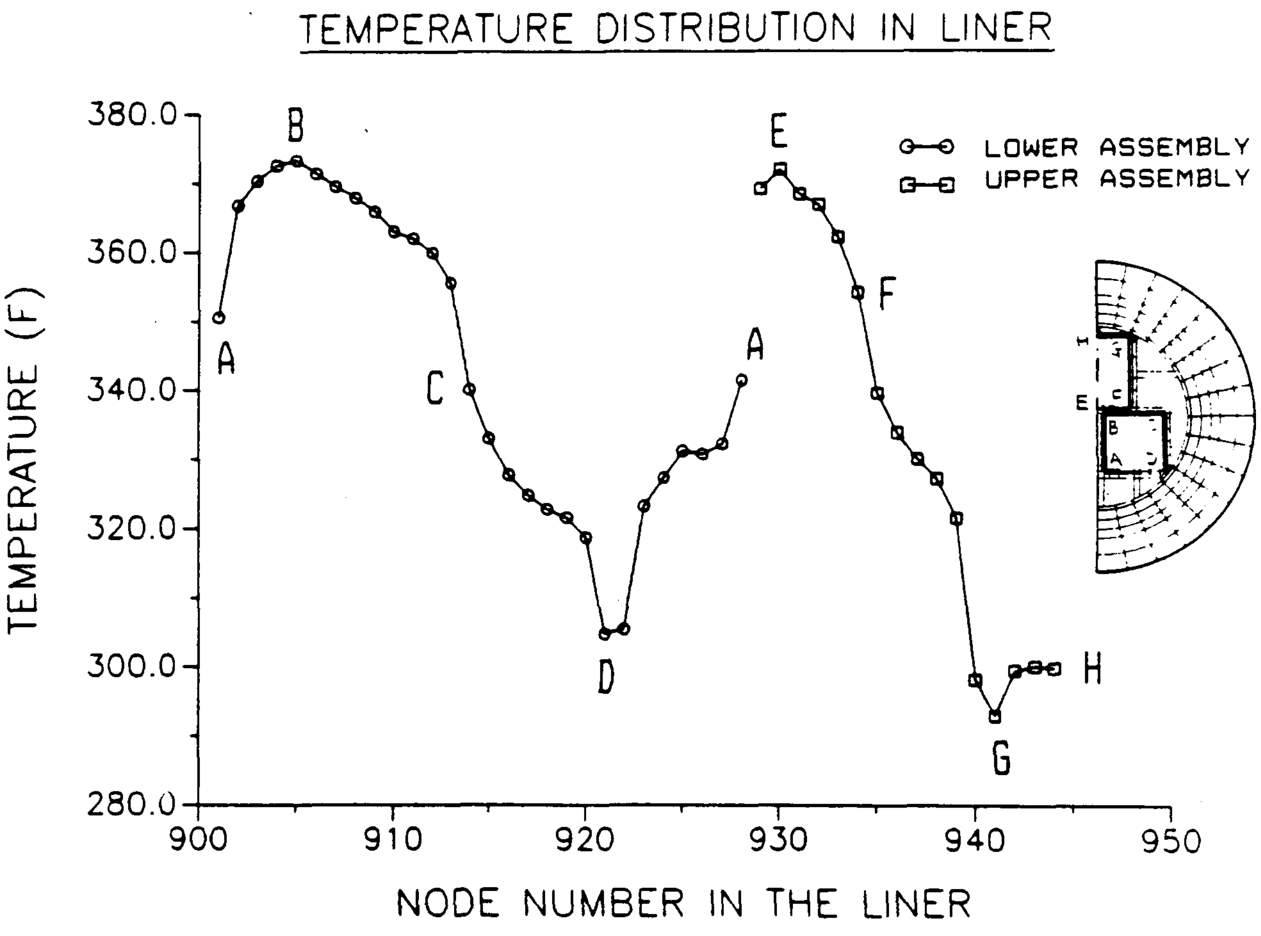

Figure 3.4-7 Temperature Distribution in Liner

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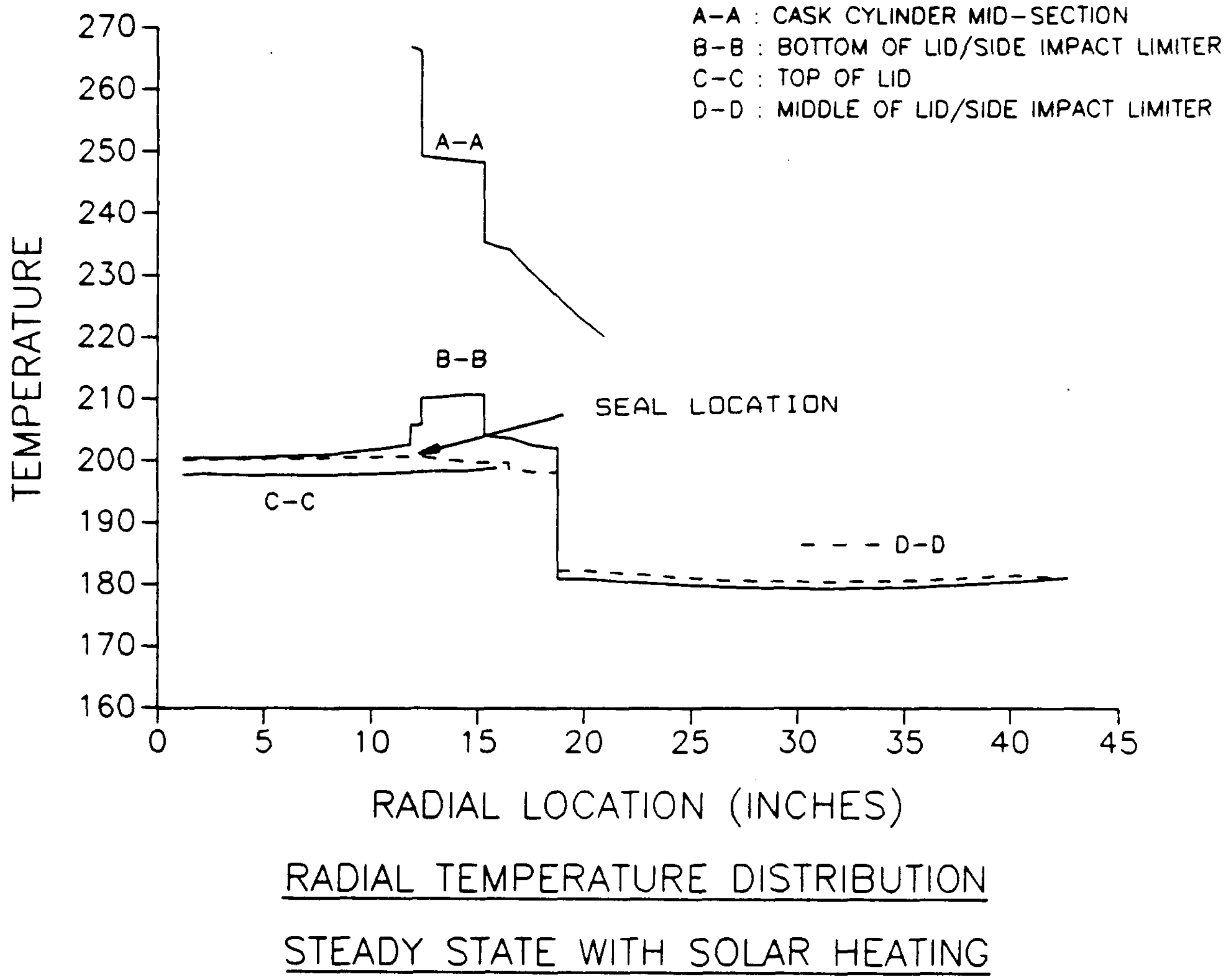

Figure 3.4-8. Radial Temperature Distribution in Cask with Solar Insolation

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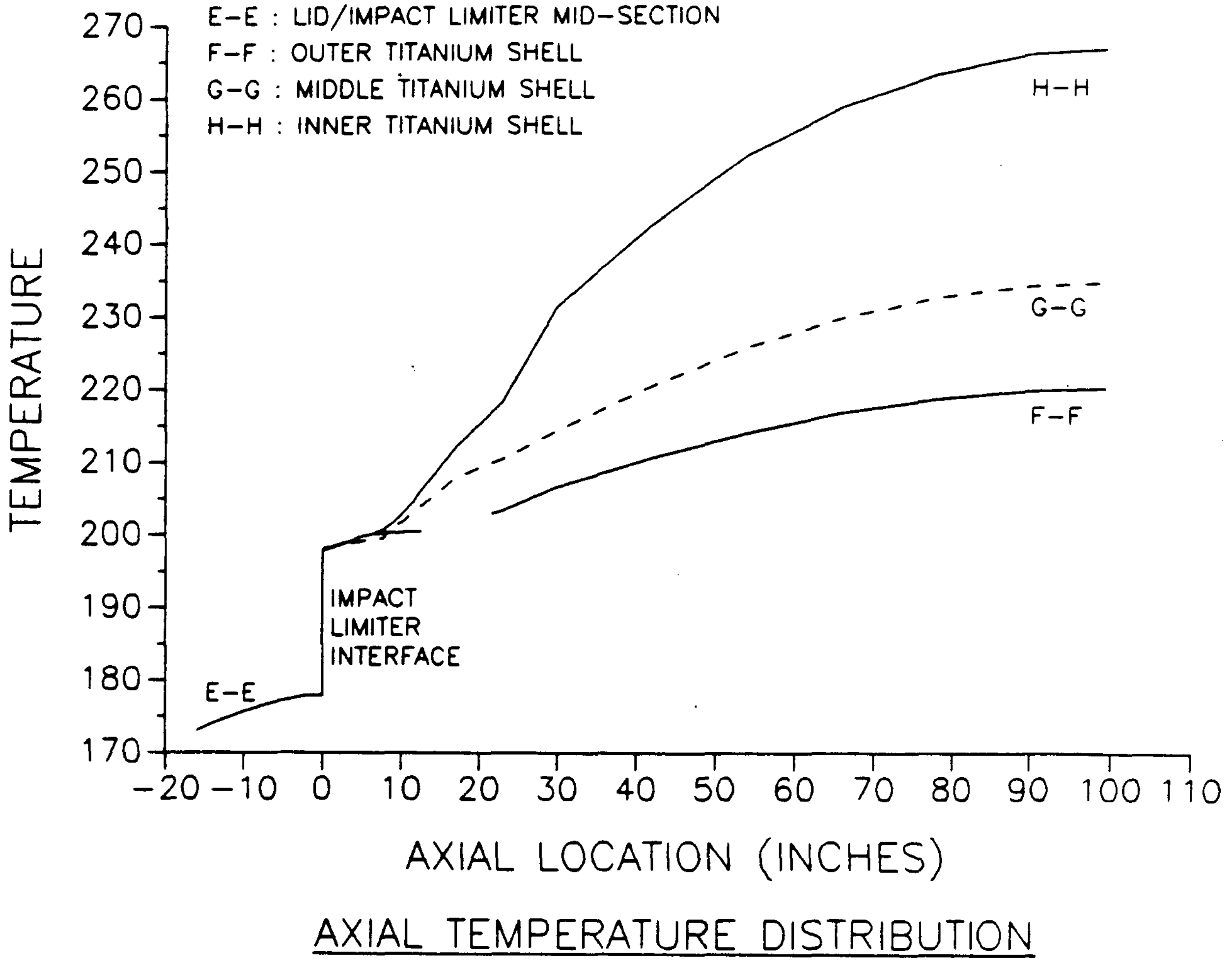

Figure 3.4-9 Axial Temperature Distribution in Cask with Solar Insolation

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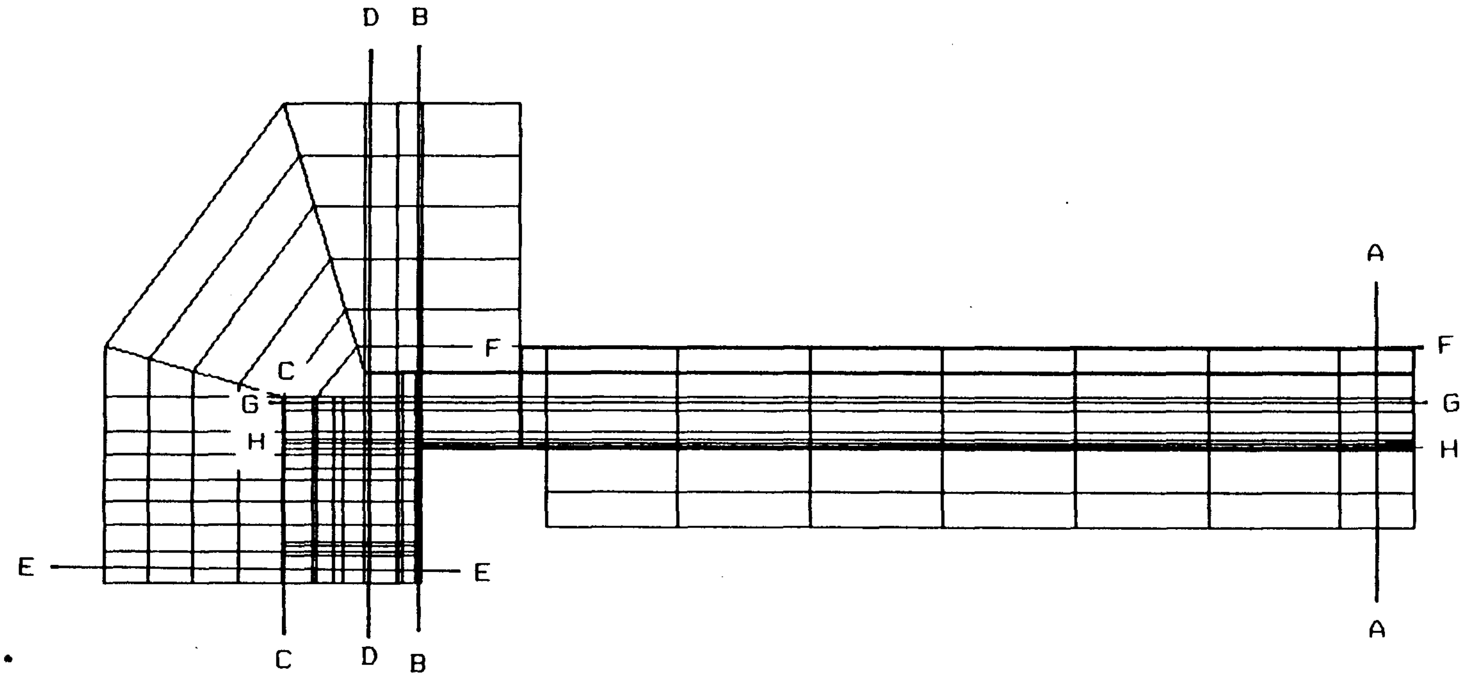

Figure 3.4-10. Location Where Temperature Distributions are Taken

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3.4.4 Maximum Internal Pressure

The maximum internal pressure of the TITAN LWT cask is based on an average gas temperature of 400°F which results from the Heat condition (i.e., 100°F ambient temperature, full isolation and maximum decay heat). The amount of gas (mols) in the cask is taken as the sum of the initial backfill of helium in the cavity at the time of cask closure, the backfill of helium in the fuel rods at the time of manufacture, and the fission gases (xenon and krypton) released from the fuel during irradiation. Regulatory Guide 7.8 (Reference 3.4.4) states the "For commercial nuclear power plant fuels, the release of all the pressurized gases inside the irradiated fuel should be considered in determining the maximum resultant containment vessel pressure." This has been done.

Table 3.4-2 provides data used in computing the volume of three of the various PWR assemblies or seven of the various BWR assemblies. The basic data for these computations was taken from the ORNL data base (Reference 3.4.5). Table 3.4-3 gives the fuel volumes, basket volumes, gross cavity volumes types of fuel for which the cask is designed. The net void volume ranges from 71.7% of the gross cavity volume (for the B&W 17x17) to 76.5% (for the EN 14x14). Table 3.4-4 gives the amount of fission gases that could potentially be released (in gram-mols) from the fuel, the amount of helium initially charged into the rods of either 3 PWR assemblies or 7 BWR assemblies that could potentially be released, and the amount of helium backfill (at the time of closure) in the cask. The pressure that could result if all of this gas reached a temperature of 400°F in the net volume of the cask cavity is given in the last column of Table 3.4-2. As shown in the table, the pressures for the various assemblies would range from 33.7 psia to 49.1 psia (34.4 psig). The pressure selected as the maximum normal operating pressure (MNOP) is thus 35 psig.

3.4.5 Maximum Thermal Stress

The maximum thermal stresses resulting from the range of temperatures that can be experienced during the normal conditions of transport for the three PWR fuel basket and the principal inner and outer titanium shells have been determined.

Table 3.4-2

Fuel Assembly Volumes

	Spent Fuel	Fuel Rod Dia. (in)	Fuel Rod Length (in)	Number of Rods	Weight of Rod (lbs)	Total Weight of Rods (lbs)	Total Volume of Rods (in**3)	Total Weight Assembly (lbs)	Weight of Other Parts (lbs)	Volume of Other Parts (in**3)	Total Volume Assembly (in**3)	Total Volume 3-PWR 7-BWR (in**3)
PWR	W 17 x 17	0.374	151.6	264	5.37	1417.6	4395.6	1482	64.3	242.7	4638.3	13,914.8
	W 15 X 15	0.422	151.9	204	6.85	1397.4	4331.4	1461	63.6	238.9	4570.3	13,710.8
	W 14 X 14	0.422	152.4	179	6.68	1195.7	3813.6	1272	76.3	374.5	4188.1	12,564.3
	BW 17 X 17	0.379	152.7	264	4.9	1293.6	4545.2	1505	211.4	766.5	5311.8	15,935.4
	BW 15 X 15	0.43	153.7	208	7.0	1456.0	4639.7	1515	59.0	291.0	4930.7	14,792.1
	CE 16 X 16	0.382	161.0	224	5.7	1276.8	4131.1	1430	153.2	574.8	4705.9	14,117.7
	CE 14 X 14	0.44	147.0	164	6.9	1131.6	3663.8	1270	138.4	520.3	4184.2	12,552.5
	EN 17 X 17	0.36	152.0	264	4.8	1269.8	4082.5	1348	78.2	315.3	4397.7	13,193.2
	EN 15 X 15	0.424	152.1	204	6.7	1370.9	4377.8	1433	62.1	245.6	4623.5	13,870.5
	EN 14 X 14	0.424	149.1	179	6.7	1202.9	3766.4	1271	68.1	265.6	4032.0	12,096.0
BWR	GE 8 X 8	0.49	160.0	63	N/A	N/A	1899.9	N/A	N/A	N/A	1899.9	13,299.0
	GE 7 X 7	0.57	155.2	49	N/A	N/A	1940.1	N/A	N/A	N/A	1940.1	13,580.5
	EN 8 X 8	0.484	163.4	62	9.2	567.3	1863.2	587.8	20.5	78.5	1941.7	13,592.1
	EN 7 X 7	0.57	158.2	48	12.3	590.4	1936.1	619.1	28.7	111.9	2048.0	14,336.1

Table 3.4-3

Fuel Basket and Net Cask Volumes

	Spent Fuel	Total Volume 3-PWR 7-BWR (in**3)	Volume of Basket (in**3)	Volume of Cask (in**3)	Void Volume of Cask (in**3)
PWR	W 17 x 17	13,914.8	6666.4	79,810.0	59,228.8
	W 15 X 15	13,710.8	6666.4	79,810.0	59,432.8
	W 14 X 14	12,564.3	6666.4	79,810.0	60,579.3
	BW 17 X 17	15,935.4	6666.4	79,810.0	57,208.2
	BW 15 X 15	14,792.1	6666.4	79,810.0	58,351.5
	CE 16 X 16	14,117.7	6666.4	79,810.0	59,025.9
	CE 14 X 14	12,552.5	6666.4	79,810.0	60,591.1
	EN 17 X 17	13,193.2	6666.4	79,810.0	59,950.4
	EN 15 X 15	13,870.5	6666.4	79,810.0	59,273.1
	EN 14 X 14	12,096.0	6666.4	79,810.0	61,047.6
BWR	GE 8 X 8	13,299.0	7585.9	79,810.0	58,925.1
	GE 7 X 7	13,580.5	7585.9	79,810.0	58,643.6
	EN 8 X 8	13,592.1	7585.9	79,810.0	58,632.0
	EN 7 X 7	14,336.1	7585.9	79,810.0	57,888.0

Table 3.4-4

Maximum Possible Gas Pressures

	Spent Fuel	Void Volume of Cask (in**3)	UO2 per Assembly (kg)	Fission Gas Released (gm-mol)	7-BWR 3-PWR	Fuel Rod Plenum Volume (in**3)	Initial Gas Pressure (psia)	Helium Released (gm-mol)	Backfill Helium in Cask (gm-mol)	Total (gm-mol)	Pressure in Cask at 400'F (psia)
PWR	W 17 x 17	59,228.8	463.6	19.5		1.25	514.7	25.4	38.1	83.0	49.1
PWR	W 15 X 15	59,432.8	462.7	19.5		1.25	489.7	18.7	38.2	76.3	45.1
PWR	W 14 X 14	60,579.3	398.0	16.8		1.72	474.7	21.8	38.9	77.5	44.9
PWR	BW 17 X 17	57,208.2	456.2	19.2		0.819	449.7	14.5	36.8	70.5	43.2
PWR	BW 15 X 15	58,351.5	463.6	19.5		1.308	429.7	17.5	37.5	74.5	44.8
PWR	CE 16 X 16	59,025.9	426.0	17.9		1.092	389.7	14.2	37.9	70.1	41.7
PWR	CE 14 X 14	60,591.1	386.0	16.3		1.273	389.7	12.2	38.9	67.4	39.0
PWR	EN 17 X 17	59,950.4	401.1	16.9		0.462	304.7	5.6	38.5	61.0	35.7
PWR	EN 15 X 15	59,273.1	432.0	18.2		0.569	304.7	5.3	38.1	61.6	36.4
PWR	EN 14 X 14	61,047.6	379.0	16.0		0.469	304.7	3.8	39.2	59.0	33.9
BWR	GE 8 X 8	58,925.1	176.8	17.4		2.83	59.7	3.7	37.9	59.0	35.1
BWR	GE 7 X 7	58,643.6	183.8	18.1		2.87	14.7	0.7	37.7	56.5	33.8
BWR	EN 8 X 8	58,632.0	176.8	17.4		1.09	59.7	1.4	37.7	56.5	33.8
BWR	EN 7 X 7	57,888.0	183.8	18.1		1.54	14.7	0.4	37.2	55.6	33.7

Thermal stress for the three PWR fuel basket was evaluated using the WECAN finite element analysis discussed in Section 2.10.4. The maximum thermal stress of 8717 psi would occur at the mid-panel intersection with the bottom vertical panel.

Using Figure 3.4-9, the weighted average inner shell and outer (1.25" thick) shell temperature changes were determined. For the inner shell, the average ΔT, is 177.6°F. For the outer shell the average ΔT_2 is 154.6°F. For Grade 9 titanium at these temperatures, the modulus of elasticity (E) is 14,500,000 psi and the thermal expansion coefficient (2) is 5.1×10^{-6} in/in-°F. The inner shell cross sectional area (A_1) is 38.108 in^2 and the outer shell cross sectional area (A_2) is 125.153 in^2.

Using the equations from Reference 3.4.7 for two parallel bars of same length, fixed at each end, the inner shell and outer shell thermal stresses were calculated.

$$K = \frac{1 - [\alpha_2 (T_2 - T_0)/\alpha_1 (T_1 - T_0)]}{1 + (A_1 E_1 / A_2 E_2)}$$

where:

$$\alpha_1 = \alpha_2 = 5.1 \times 10^{-6} \text{ in/in - °F}$$
$$E_1 = E_2 = 14.5 \times 10^6 \text{ psi}$$
$$A_1 = \frac{\pi}{4} (24.76^2 - 23.76^2) = 38.1 \text{ in}^2$$
$$A_2 = \frac{\pi}{4} (33.12^2 - 30.62^2) = 125.2 \text{ in}^2$$

Therefore K = 0.099

$$\sigma_1 = -K \alpha E (\Delta T) = -(0.099)(0.0000051)(14{,}500{,}000)(177.6)$$

$$\sigma_1 = -1299 \text{ psi}$$

$$\sigma_2 = - \left(\frac{A1}{A2}\right) (\sigma_1) = - \left(\frac{38.108}{125.153}\right) (-1299) = 396 \text{ psi}$$

These stresses are small compared to the yield of about 58 ksi.

3.4.6 Evaluation of Package Performance for Normal Conditions of Transport

The LWT cask meets the specified temperature limits with a considerable margin. The calculated maximum cladding temperature of 423°F is nearly 300°F below the established limit for spent fuel in an inert environment. The maximum normal operating pressure of 35 psig is modest (and conservatively calculated). There are no anticipated problems with differential thermal expansions of the various materials of the system. The clearance between the basket and the inside surface of the cask cavity is such as to preclude interference during the heat condition. The same condition exists with respect to the DU/titanium interface. The Boro-Silicone has a recommended temperature limit of 400°F (Reference 3.4.5) which is well above the temperature encountered during the Normal Heat Condition. Being an elastomeric material, it will easily conform to dimensional changes of the titanium which encloses it.

3.5 Hypothetical Accident Thermal Evaluation

3.5.1 Thermal Model

3.5.1.1 Analytical Model

For the transient, the model shown in Figure 3.4-2 is used. The following assumptions were made for the accident evaluation:

1. The cask contains three Westinghouse PWR spent fuel assemblies with each assembly generating 580 W of decay heat.

2. The axial power peaking factor of the spent fuel used for the analysis is shown in Figure 3.4-3.

3. The ambient air temperature is 100°F initially.

4. For the analysis of the thermal test given in this report, solar radiation was neglected. Paragraph 71.73(c) (3) of 10 CFR Part 71 states, "The effects of solar radiation may be neglected prior to, during, and following the test." It is recognized that the Second Proposed Revision 1 to Regulatory Guide 7.8 states that "The Normal and Hypothetical Accident Conditions should be considered to have initial conditions of ambient temperature at -20°F (-29°C) with no insolation and of ambient temperature at 100°F (38°C) with maximum insolation." The analysis supporting the final design will conform to the Regulatory Guide. The preliminary design analysis was done in compliance with the Part 71 requirements.

5. Temperature is 1475°F with an emissivity of 0.9, as recommended in Paragraph 71.73 (c)(3) of Reference 3.1.1.

6. Natural convection during the thermal test is assumed.

7. The longitudinal thickness as of the honeycomb in the top and diagonal sections of the impact limiter were reduced to 30 percent of their original dimensions, simulating a drop damage.

Based on these assumptions, the following boundary conditions were applied to the model:

- No solar heating was applied prior, during or after the 30 minute thermal accident simulation.

- The natural convective heat transfer coefficient (Reference 3.4.2) at the surface is:

$$H_{conv} = 0.18 * (DT)^{(1/3)} \quad (Btu/hr\text{-}ft^2\text{-}\ °F)$$

where, DT is a temperature difference between the cylinder surface and the ambient air.

- o During the first 30 minutes of the transient, the ambient air temperature is set at 1475°F with an emissivity of 0.9 but is assumed to be 100°F immediately afterward.

- o The radiation cooling with a 0.2 emissivity was applied at the outer surface of the cylinder after the 30 minute thermal accident period.

- o The radiation absorptance of the outer surface is 0.8.

- o After the surface temperature reaches the melting temperature of aluminum (1100°F), the outer edge of the aluminum honeycomb will melt, severing the conduction path from the stainless steel liner to the honeycomb. After the honeycomb edge melting, the heat transfer to the honeycomb becomes inefficient, because 1) the stainless steel liner acts as a thermal shield by not allowing the honeycomb to be exposed to the thermal event, 2) there is only radiation between the stainless steel liner and the honeycomb and the conduction/convection connections through the air trapped in the cavity. The time to melt the aluminum honeycomb was estimated to be approximately 35 minutes, as shown in Section 3.6.1. Therefore, it is assumed that the outer edge of aluminum honeycomb remains at its melting temperature of 1100°F while the thermal accident lasts. This simulates the melting process of the honeycomb.

- o During the cooling period after the 30 minute thermal accident, it was assumed that the gap between the honeycomb and the outer stainless steel liner is present so as to minimize the cooling of the impact limiter.

3.5.2 Package Conditions and Environment

The preliminary thermal evaluation for the hypothetical thermal event has been based on the assumption that the impact limiters are still attached to the cask following the 30 foot drop test and the 40 inch drop onto the 6 inch diameter pin.

The analyses supporting the final design will include the following cases:

1. Cask with crushed impact limiters. The crushing will be assumed to have occurred as a consequence of a c.g. over corner drop. Further, it will be assumed that during the puncture test, the bar strikes the center of the flattened area on the crushed impact limiter, fully compressing the aluminum honeycomb to a density equivalent to solid aluminum.

2. Cask with a portion of the impact limiter material removed. It will be assumed that the puncture test has resulted in removing a 6 inch diameter section of the neutron shielding from the closure lid as close to the edge as possible.

3.5.3 Package Temperatures

The initial radial and axial temperature distributions in the closure lid and cask body are shown in Figures 3.5-1 and 3.5-2. The locations for which the temperatures are calculated are indicated in Figure 3.4-10. These results were compared against the steady-state 180° model results. Throughout the transient, the nodal temperatures were tracked in the selected locations shown in Figure 3.5-3. The transient temperature responses of these selected locations in the closure lid and the cylinder are shown in Figures 3.5-4 and 3.5-5. The transient temperature responses of these selected locations are also tabulated in Table 3.5-1. After the 30 minute thermal accident, the location of the highest temperature moved to the interior in such a way that some time after the accident, when the outside has cooled, the interior reaches a higher temperature than the outside as seen in Figures 3.5-4 and 3.5-5. The maximum temperatures of selected locations and the times at which these temperatures occur are given in Table 3.5-2. The maximum basket and cladding temperatures were estimated in Section 3.6.2 to be 436°F and 468.5°F, respectively.

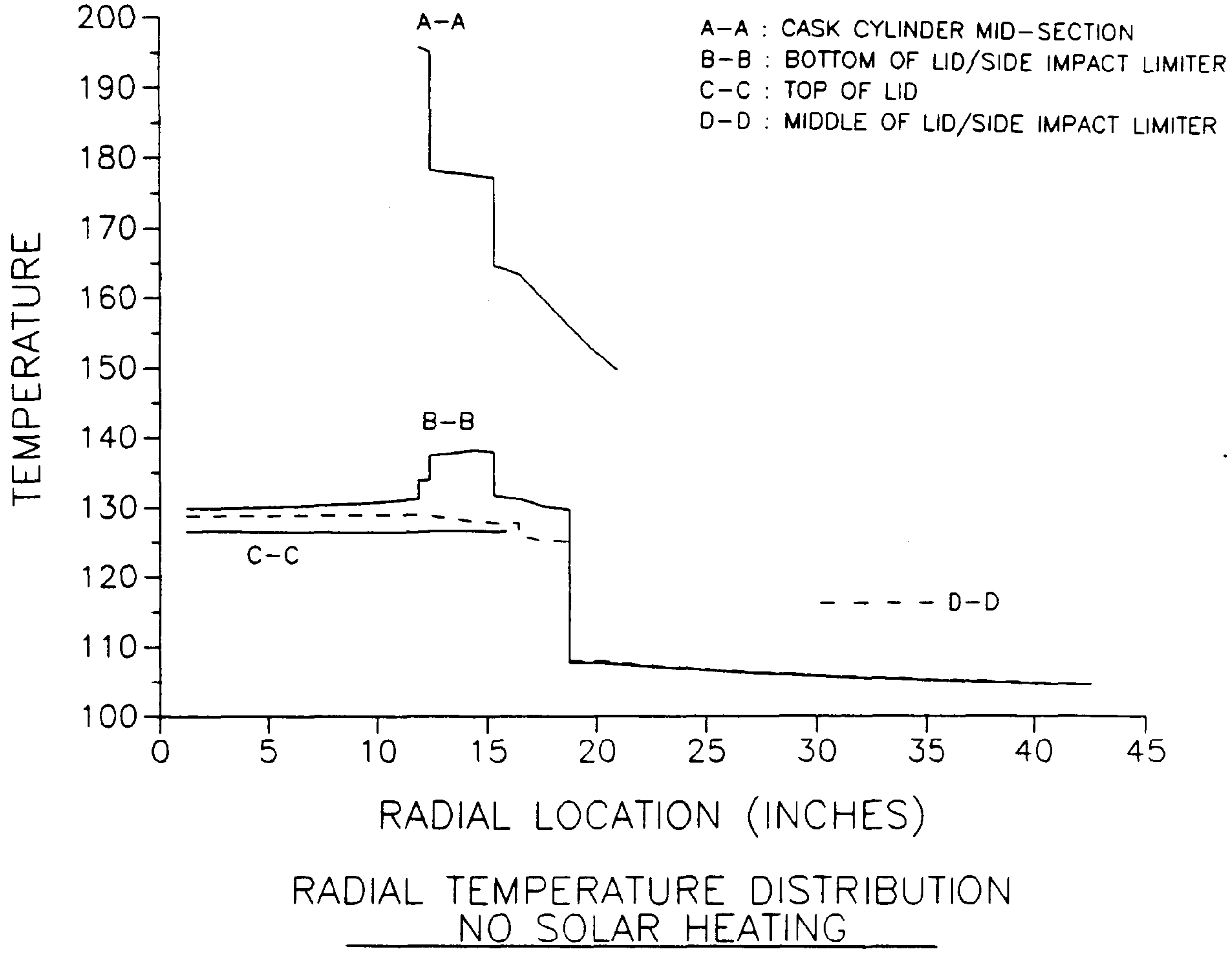

Figure 3.5-1. Radial Temperature Distribution in Cask at T=0 Sec.

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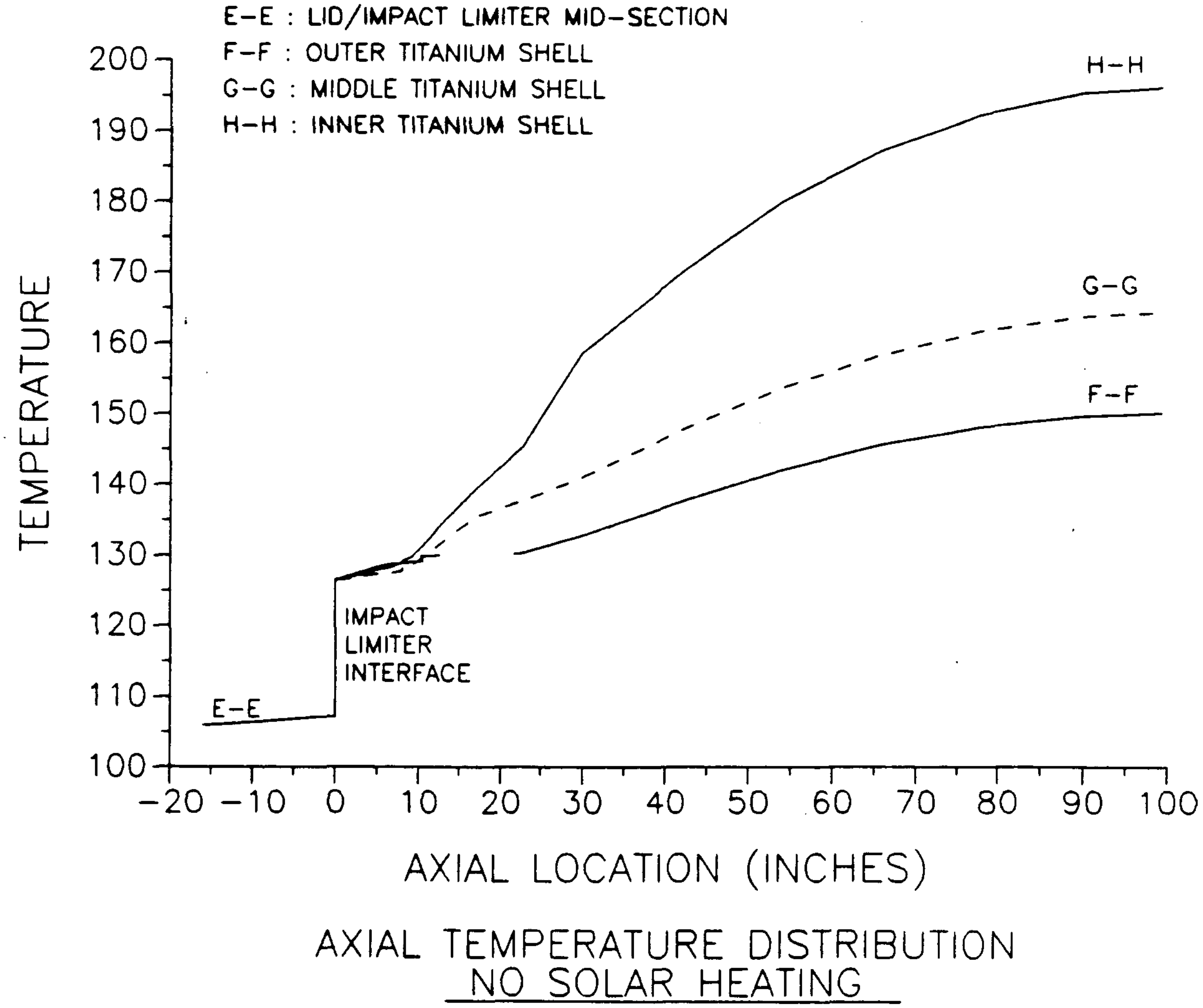

Figure 3.5-2. Axial Temperature Distribution in Cask at T=0 Sec.

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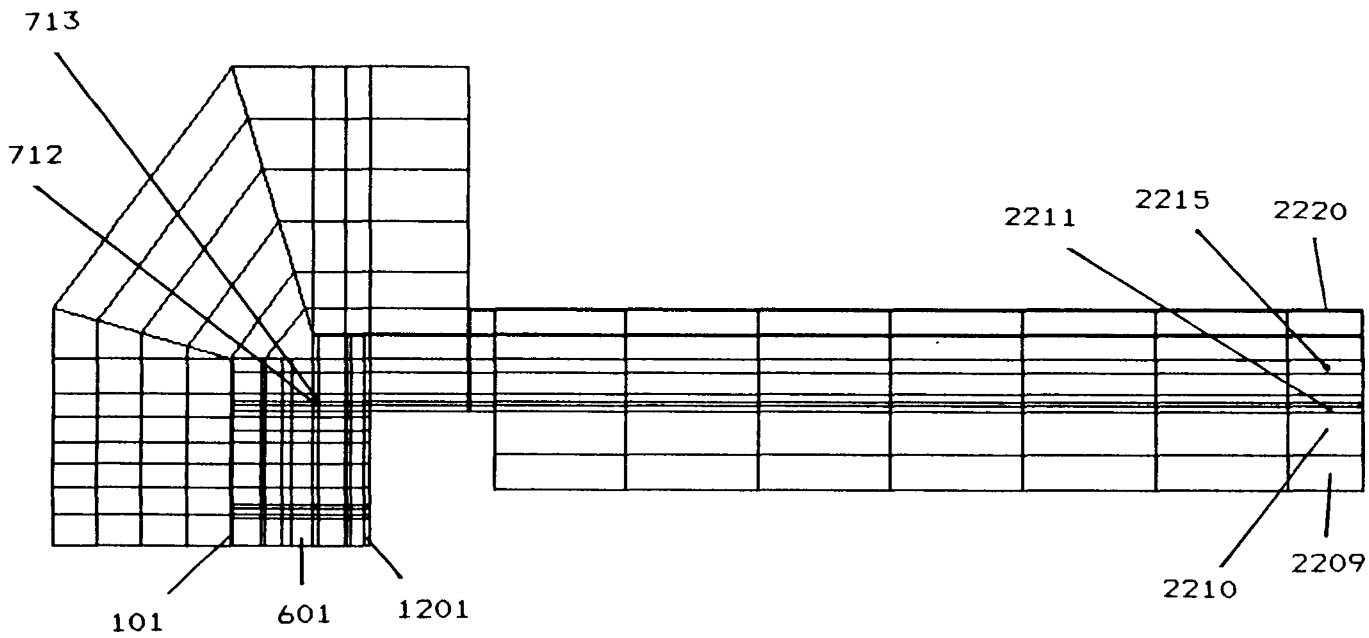

Figure 3.5-3. Nodal Locations Where Temperature Response are Taken

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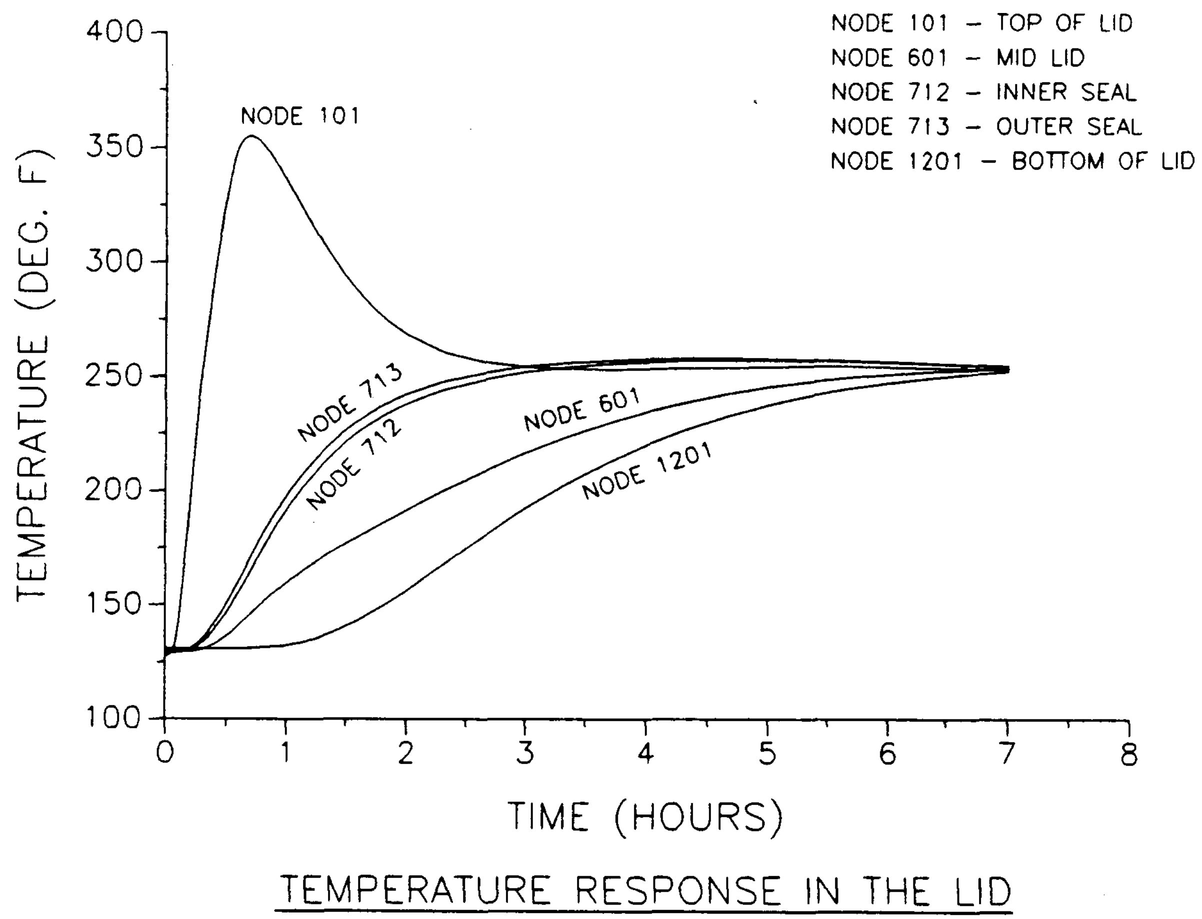

Figure 3.5-4. Transient Temperature Response in Lid Section

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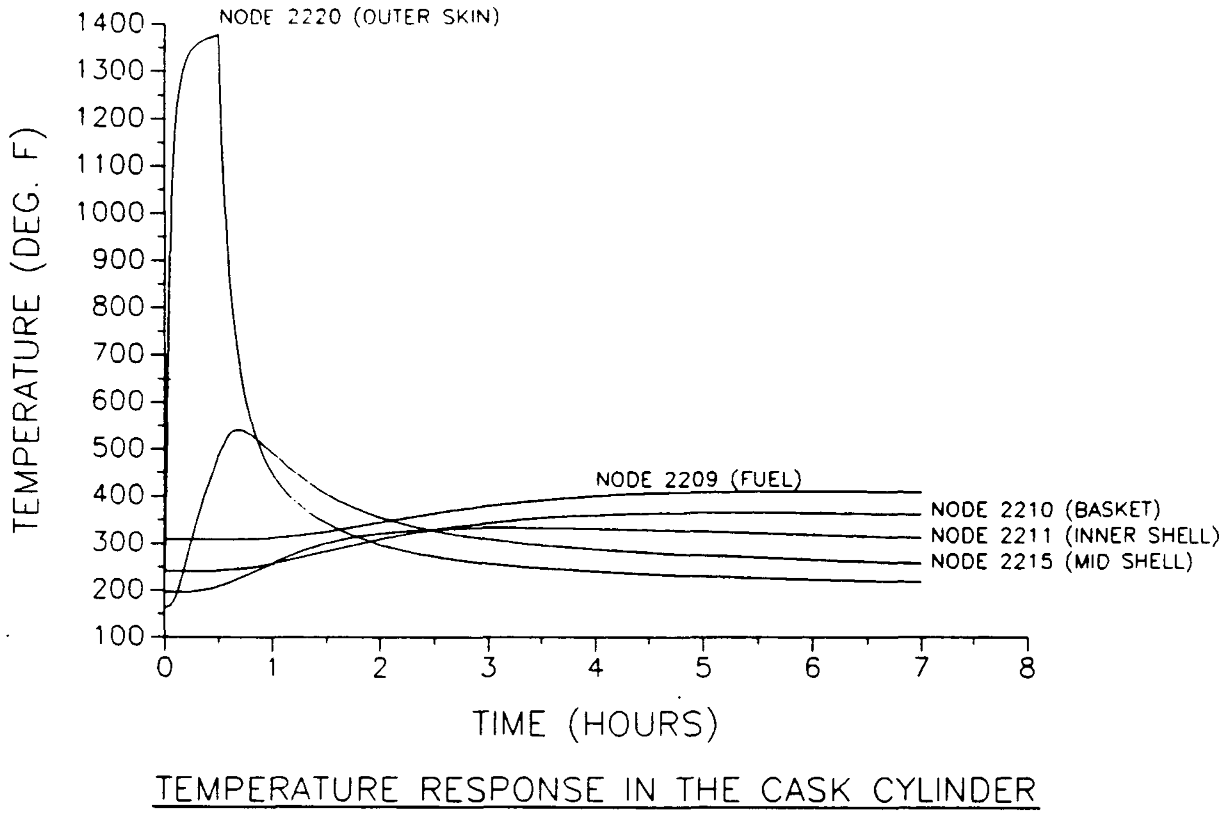

Figure 3.5-5. Transient Temperature Response in Cylinder Section

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Table 3.5-1
Temperature Response of Select Locations

Time (min.)	NODE # 101	NODE # 601	NODE # 712	NODE # 713	NODE # 1201	NODE # 2209	NODE # 2210	NODE # 2211	NODE # 2215	NODE # 2220
66	326.4	163.8	199	205	133.4	313.6	261.4	268.5	471.5	416.5
68	323.3	165.1	201.3	207.3	133.8	314.3	263.1	271.7	464.9	407.6
70	320.2	166.2	203.6	209.5	134.2	315.1	264.8	274.8	458.4	399.5
72	317.2	167.4	205.7	211.6	134.7	315.8	266.5	277.7	452.2	392
74	314.2	168.5	207.8	213.6	135.2	316.7	268.3	280.6	446.1	385.1
76	311.3	169.7	209.8	215.5	135.8	317.5	270.1	283.3	440.2	378.7
78	308.4	170.8	211.7	217.4	136.4	318.4	271.9	286	434.6	372.6
80	305.6	171.8	213.5	219.1	137	319.4	273.7	288.5	429.1	367
82	302.9	172.9	215.3	220.8	137.7	320.4	275.5	290.9	423.9	361.7
84	300.3	173.9	217	222.4	138.4	321.4	277.3	293.2	418.8	356.6
86	297.8	174.9	218.6	224	139.2	322.5	279.1	295.5	413.9	351.9
88	295.4	175.9	220.2	225.5	140	323.5	281	297.6	409.3	347.4
90	293	176.9	221.7	226.9	140.8	324.7	282.8	299.6	404.8	343.1
92	290.8	177.9	223.1	228.3	141.6	325.8	284.6	301.6	400.5	339
94	288.7	178.9	224.5	229.6	142.5	327	286.4	303.4	396.3	335.1
96	286.6	179.9	225.8	230.8	143.4	328.2	288.2	305.2	392.3	331.4
98	284.7	180.9	227.1	232	144.4	329.4	290	306.9	388.5	327.9
100	282.8	181.9	228.3	233.1	145.4	330.7	291.7	308.5	384.8	324.5
102	281	182.8	229.4	234.2	146.4	331.9	293.5	310	381.3	321.2
104	279.3	183.8	230.5	235.3	147.4	333.2	295.2	311.5	377.9	318.1
106	277.7	184.7	231.6	236.3	148.4	334.5	296.9	312.8	374.6	315.2
108	276.2	185.7	232.6	237.2	149.5	335.8	298.6	314.1	371.5	312.3
110	274.7	186.6	233.6	238.1	150.6	337.1	300.3	315.4	368.5	309.6
112	273.3	187.6	234.6	239	151.7	338.5	301.9	316.6	365.6	306.9
114	272	188.5	235.5	239.9	152.8	339.8	303.6	317.7	362.8	304.4
116	270.8	189.5	236.4	240.7	154	341.2	305.2	318.7	360.1	302
118	269.6	190.4	237.2	241.4	155.1	342.5	306.8	319.7	357.5	299.6
120	268.5	191.3	238	242.2	156.3	343.9	308.3	320.7	355.1	297.4
140	260.3	200.4	244.5	248	168.7	357.2	322.5	327.6	334.5	278.9
160	256.1	208.9	248.9	251.8	180.9	369.5	334.2	331.3	319.7	266
180	254	216.6	251.9	254.3	192.4	380.1	343.5	332.8	308.7	256.6
200	253.3	223.4	253.9	255.9	202.8	388.8	350.8	333	300.1	249.5
220	253.1	229.3	255.3	256.9	212	395.7	356.3	332.3	293.2	243.9
240	253.3	234.3	256.2	257.5	220	400.9	360.4	331	287.4	239.4
260	253.6	238.6	256.7	257.7	226.8	404.8	363.2	329.3	282.5	235.6
280	253.8	242.1	256.9	257.8	232.5	407.5	364.9	327.4	278.2	232.4
300	254	245	256.9	257.6	237.3	409.2	365.8	325.3	274.4	229.6
320	254	247.3	256.7	257.2	241.2	410.2	366	323	270.9	227.1
340	254	249.2	256.4	256.8	244.4	410.5	365.7	320.7	267.8	224.8
360	253.8	250.6	255.9	256.2	246.9	410.3	364.9	318.3	264.8	222.6
380	253.5	251.7	255.3	255.5	248.9	409.7	363.7	315.9	262.1	220.7
400	253.1	252.4	254.6	254.8	250.5	408.8	362.3	313.5	259.5	218.8
420	252.6	252.9	253.9	254	251.6	407.6	360.7	311.1	257.1	217.1

Table 3.5-1 (Continued)
Temperature Response of Select Locations

Time (min.)	NODE # 101	NODE # 601	NODE # 712	NODE # 713	NODE # 1201	NODE # 2209	NODE # 2210	NODE # 2211	NODE # 2215	NODE # 2220
1	127.5	129.5	129.4	129.1	130.8	307.8	241	196.3	164.7	571.8
2	127.8	129.5	129.4	129.1	130.8	307.8	241	196.3	165.3	813.4
3	128.8	129.5	129.3	129.1	130.8	307.8	241	196.3	167.3	967.9
4	130.9	129.5	129.3	129.1	130.8	307.8	241	196.3	171.3	1072.8
5	134.3	129.5	129.3	129.1	130.8	307.8	241	196.3	177.7	1146.6
6	139.3	129.5	129.4	129.2	130.8	307.8	241	196.3	186.3	1199.5
7	146	129.5	129.4	129.3	130.8	307.8	241	196.3	196.7	1238.2
8	154.4	129.5	129.5	129.4	130.8	307.8	241	196.3	208.5	1266.7
9	164.2	129.5	129.6	129.6	130.8	307.8	241	196.4	221.4	1288.2
10	174.7	129.6	129.7	129.8	130.8	307.8	241	196.5	235.1	1304.6
11	185.6	129.6	130	130.2	130.8	307.8	241	196.6	249.2	1317.3
12	196.4	129.7	130.3	130.7	130.8	307.8	241	196.7	263.7	1327.2
13	207	129.7	130.6	131.2	130.8	307.8	241	196.9	278.2	1335.1
14	217.1	129.8	131.1	131.8	130.8	307.8	241	197.1	292.7	1341.5
15	226.8	130	131.6	132.6	130.8	307.8	241	197.3	307.1	1346.7
16	236	130.2	132.2	133.4	130.8	307.8	241.1	197.6	321.3	1351
17	244.7	130.4	132.8	134.2	130.8	307.8	241.1	197.9	335.2	1354.7
18	253	130.6	133.5	135.2	130.8	307.8	241.1	198.3	348.9	1357.8
19	260.9	130.9	134.3	136.2	130.8	307.8	241.2	198.8	362.2	1360.5
20	268.4	131.2	135.2	137.2	130.8	307.8	241.2	199.3	375.2	1362.9
21	275.6	131.6	136.1	138.4	130.8	307.8	241.3	199.9	387.8	1365
22	282.4	132	137	139.5	130.8	307.8	241.3	200.5	400.2	1367
23	289	132.4	138.1	140.8	130.8	307.8	241.4	201.2	412.1	1368.7
24	295.3	132.9	139.1	142	130.8	307.8	241.5	202	423.8	1370.4
25	301.3	133.4	140.3	143.3	130.8	307.9	241.6	202.8	435.1	1371.9
26	307.2	133.9	141.4	144.7	130.8	307.9	241.7	203.7	446.1	1373.4
28.5	320.9	135.4	144.5	148.3	130.8	307.9	242	206.2	472.2	1376.7
29.5	326.1	136.1	145.9	149.8	130.8	307.9	242.2	207.3	482.1	1377.9
30	328.6	136.4	146.5	150.5	130.8	307.9	242.3	207.9	486.9	1378.5
32	338.1	137.8	149.3	153.7	130.8	308	242.7	210.4	505.5	1121.9
34	345.4	139.3	152.3	156.9	130.8	308	243.2	213.1	521.1	976.6
36	350.4	140.8	155.3	160.3	130.8	308.1	243.7	216	532.2	873
38	353.3	142.5	158.4	163.6	130.9	308.3	244.3	219.1	538.7	792.5
40	354.8	144.1	161.6	167	130.9	308.4	245.1	222.4	541.7	727.6
42	355.2	145.8	164.8	170.4	131	308.6	245.9	225.8	541.8	674.5
44	354.7	147.5	168	173.7	131	308.8	246.7	229.3	539.8	630.6
46	353.5	149.1	171.1	177	131.1	309	247.7	232.9	536.2	593.8
48	351.8	150.8	174.2	180.2	131.2	309.3	248.8	236.6	531.4	562.8
50	349.7	152.4	177.3	183.3	131.3	309.6	249.9	240.3	525.8	536.4
52	347.3	154	180.3	186.3	131.5	310	251.1	244	519.6	513.8
54	344.7	155.5	183.2	189.3	131.7	310.4	252.4	247.7	513	494.2
56	341.9	157	186	192.1	131.9	310.8	253.8	251.4	506.2	477.1
58	338.9	158.5	188.8	194.9	132.1	311.3	255.2	254.9	499.2	462.1
60	335.9	159.9	191.5	197.6	132.4	311.8	256.7	258.5	492.2	448.7
62	332.8	161.2	194.1	200.1	132.7	312.4	258.2	261.9	485.2	436.8
64	329.6	162.6	196.6	202.6	133	313	259.8	265.3	478.3	426.2

Table 3.5-2
Maximum Temperature During Thermal Accident Simulation

Node	Location	Maximum Temperature (°F)/Time (min)
101	Top of T_1-Lid	355.2 / 42.0
601	Lid Mid-Section	252.9 / 420.0 *
712	Inner Seal	256.9 / 300.0
713	Outer Seal	257.8 / 280.0
1201	Bottom of Lid	251.6 / 420.0 *
2220	Outer T_1-Shell	1378.5 / 30.0
2215	Middle T_1-Shell	541.8 / 42.0
2211	Inner T_1-Shell	333.0 / 200.0
2210	Average Basket	366.0 / 300.0
2209	Average Cladding	410.5 / 340.0
Estimated Maximum Basket		436.0 / 300.0
Estimated Maximum Cladding		468.5 / 340.0

* - The maximum temperature was not reached in 7 hr. period.

3.5.4 Maximum Internal Pressures

With maximum basket and cladding temperatures given in the previous section, the average gas temperature during the thermal accident would not be appreciably different from that during the normal heat condition. Even if insolation were included in the initial conditions, these temperatures (basket and clad) would be only of the order of 70°F higher. The MNOP was evaluated for an average temperature of 400°F. If the average gas temperature reached 500°F during the transient, the pressure would be

$$49.7 \text{ psia } \left(\frac{500 + 460}{400 + 460}\right) = 55.5 \text{ psia or } 40.8 \text{ psig}$$

This is a modest increase and of no consequence to the cask integrity.

3.5.5 Maximum Thermal Stresses

The maximum thermal stresses produced in the main shells of the cask due to differential thermal expansion of these shells during the thermal accident were computed using the formulas given in Section 3.4.5. The maximum temperature difference occurs 42 minutes into the transient. The values of the parameters for the inner (0.5" thick) and outer (1.25" thick) shells are:

$T_1 = 225.8°F$
$\alpha_1 = 5.1 \times 10^{-6}$ in/in - °F
$E_1 = 14.5 \times 10^6$ psi
$T_2 = 541.8°F$
$\alpha_2 = 5.4 \times 10^{-6}$ in/in - °F
$E_2 = 12 \times 10^6$ psi

These temperature produce a compressive stress of 5624 psi in the outer shell and a tensile stress of 18,500 psi in the inner shell. These secondary stresses are small compared to a Grade 9 titanium yield stress of about 44 ksi at 540°F. Other internal stresses (for example in the closure lid and upper flange area) will be determined during the final design phase.

3.5.6 Evaluation of Package Performance for the Hypothetical Accident Thermal Conditions

The LWT cask is expected to accommodate the accidental thermal condition without compromising its principal functions of providing shielding and containment of radioactive material. This conclusion is based on the following:

1. Sufficient shielding is provided to limit the dose rate to one rem per hour at one meter from the external surface of the package during and following the accident. The gamma shielding will be unaffected (the DU may crack, but will remain in place). The Boro-Silicone neutron shielding is, according to the manufacturer, (Reference 3.4.6) self-extinguishing with glowing combustion. The average time to self-extinguish is 0 seconds. The average extent of burning is 0.2 inches. The neutron shielding would thus be expected to remain largely intact during the thermal accident. The shielding provided by the cask should be essentially unaffected.

   However, even if the neutron shielding were to disappear as a consequence of the thermal accident, the dose rates would still be below one rem/hour at one meter from the package surface

   Figure 5.4-7 shows a dose rate due to primary gammas of less than 200 mrem/hr at the outer surface of the 1.25 inch thick shell at the cask mid-point. The corresponding neutron dose rate, from Figure 5.4-17, is 400 mrem/hr. Thus along the side of the cask, the post accident dose rate limits can be met without any neutron shielding.

   At the top along the centerline, the primary gamma dose rate is about one rem/hr at the inside surface of the neutron shielding (see Figure 5.4-8). The neutron dose rate at the same location is less than 100 mrem/hr as shown in Figure 5.4-16. The total dose rate of about 1.1 rem/hr would be reduced to less than 1 rem/hr at one meter from the top (closure lid end) surface.

At the bottom, along the centerline of the cask, the neutron dose rate is approximately 200 mrem/hr at the inside surface of the neutron shielding (See Figure 5.4-18). The primary gamma dose rate at the same location is approximately 1.0 rem/hr (Figure 5.4-10). Thus without any neutron shielding on the bottom, the dose rate would be approximately 1.2 rem/hr which would be attenuated by geometry and distance such that the dose rate would be less than 1 rem/hr at one meter even if the neutron shielding were not present.

2. Containment requirements are expected to be easily met if the impact limiters remain attached to the cask following the drop and puncture events. As shown in Figure 3.5-4, the material in the vicinity of the seals remains below 255°F which is well below the short term temperature limits of elastomer seals. Excluding the outer surface of the closure lid, the principal structural plates of the closure lid experience temperature transients that match each other rather closely and thus it is expected that detailed analyses would show that there would be no distortions that would cause the lid-to-flange joint to open. Thus the thermal accident is not expected to have any significant effect on containment.

3. The analysis shows that the maximum temperature difference between the outer main shell (1.25" thick) and the inner shell (0.5" thick) would be 316°F, 42 minutes after the start of the transient. The resulting thermal stresses would be relatively small. The inner shell would have an average tensile stress of 18.5 ksi and the main outer shell would have a compressive stress of 5.6 ksi. The 0.190" thick outer cover for the neutron shielding would see quite high compressive loads because it comes to a peak temperature of 1378°F at 30 minutes into the transient while the 1.25" thick outer shell is at 487°F. These temperatures are well within the allowable limits for titanium alloys.

4. The fuel and basket temperatures increase slightly during the transient. The average basket temperature of 352°F maximum is not significantly different than the steady-state temperature during the Normal Heat Condition.

3.6 Appendix

3.6.1 References

3.1.1 10 CRF Part 71, "Packaging and Transportation of Radioactive Material," May 31, 1988.

3.1.2 Edwards, A. L. , "TRUMP: A Computer Program for Transient and Steady-State Temperature distributions in Multidimensional Systems," Lawrence Livermore Laboratory, UCRL-14754, Rev. 3, September 1, 1972.

3.4.1 "IAEA Safety Series - 7," A-546.2, p.62.

3.4.2 Kreith, F. , Principles of Heat Transfer, 1976.

3.4.3 AESD-TME-3162, "Spent Fuel Dry Storage Testing at E-MAD, (March 1978 - March 1982)," R. Unterzuber, R. D. Milnes, B. A. Marinkovich, G. M. Kubancsek, contributed by Westinghouse Electric Corporation, September 1982.

3.4.4 Second Proposed Revision 1 to Regulatory Guide 7.8 "Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material," September 1988.

3.4.5 "Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation," DOE/RW-0184, Volume 1, December, 1987.

3.4.6 Bulletin S-83N, "BORO-SILICONE Shielding," Reactor Experiments, Inc. October 1, 1987 (BORO-SILICONE is a Reactor Experiments trademark).

3.4.7 B. E. Gatewood, Thermal Stresses, McGraw Hill, 1957, pp 3.

3.6.2 Estimate of Time Required to Melt Honeycomb During Transient

The side impact limiter has the highest heat load since it receives heat flow from both radial and vertical surfaces. The volume of the side honeycomb impact limiter is calculated to be 26.3 ft^3. The weight of aluminum in the side impact limiter is 26.3 * 0.048 (volume fraction of aluminum in honeycomb) * 168 = 212 lb. Thus the heat required to fuse aluminum = 170 * 212 = 36040 Btu.

The radiation form factor from the 1475°F thermal source to the stainless steel liner is: $((1-.9)/.9 + 1 + (1-.8)/.8)^{-1} = .735$.

The radiation form factor from the stainless steel liner to honeycomb is: $((1-.3)/.3 + 1 + (1-.3)/.3)^{-1} = .176$.

A combined radiation form factor, (assuming the reference area is the area of liner), is:

$$(1/.735 + 1/.176)^{-1} = .142.$$

Therefore, the heat flow from the thermal source to the honeycomb:

$$= \text{Area} * .142 * .1713 * (1935^4 - 1560^4) * 10^{-8} = 90600 \text{ Btu/hr}$$

Where, Area = 46 ft^2 (Area of liner exposed to the thermal event).

Heat flow from the honeycomb to the cask :

$$= A * H_{gap} * DT + A * .18 * .1713 * (1160^4 - 710^4) * 10^{-8}$$
$$= 23537.3 \text{ Btu/hr}$$

where, A = 11.454 ft^2 (Area of side impact limiter/cask interface) and H_{gap} = 3.5 Btu/hr-ft^2-F (60 mil air gap).

Heat of fusion (aluminum) = 170 Btu/lb

Time to melt is therefore, 212 * 170 /(90600 - 23537.3) = .54 hr = 32.2 min.

3.6.3 Estimate of Maximum Basket and Cladding Temperatures During Transient

The steady-state TRUMP model as described in Section 3.4.1 was run without solar heating for comparison with the initial condition of the accident simulation model. In a later model, the basket and the fuel were represented by two lumped nodes which indicate their average temperatures. Thus, the maximum temperatures during the transient must be estimated by first calculating the difference between the steady-state maximum temperature and the initial condition average temperature, and second by adding the difference to the observed maximum temperature during the transient. Therefore, the maximum temperature for the basket/liner is:

$$DT_{adjust} = T^{max}_{basket}(\text{steady-state}) - T^{av}_{basket}(t=0) = 310 - 240$$
$$= 70°F$$

and

$$T^{max}_{basket} = T^{av}_{basket}(t=300 \text{ min.}) + DT_{adjust} = 366 + 70 = \underline{436°F}$$

The maximum temperature for the cladding can be similarly estimated as:

$$DT_{adjust} = T^{max}_{clad}(\text{steady-state}) - T^{av}_{clad}(t=0) = 365 - 307$$
$$= 58°F$$

and

$$T^{max}_{clad} = T^{av}_{clad}(t=340 \text{ min.}) + DT_{adjust} = 410.5 + 58 = \underline{468.5°F}$$

4. CONTAINMENT

4.1 Containment Boundary

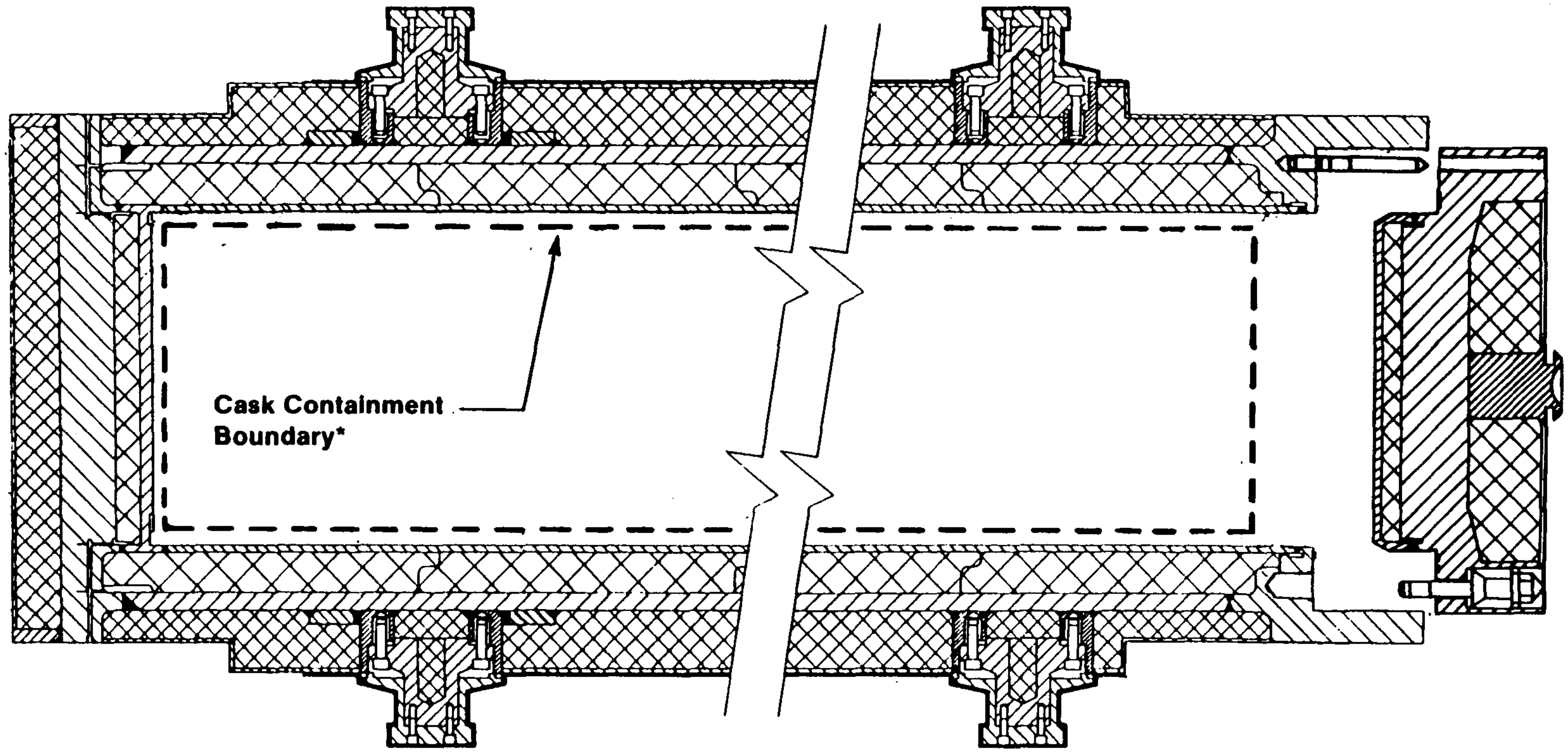
The containment boundary of the TITAN LWT cask, illustrated in Figure 4.1-1, is formed by the following components:

- The cask body inner titanium shell and bottom head assembly inner titanium wall. This includes the inner circumferential weld which joins the shell to the bottom head assembly. The cask body includes an integral top forging which is welded to the inner shell.
- The inner titanium wall of the cask closure lid, which is secured to the cask body by sixteen Alloy 718 bolts.
- The inner seal of the cask lid. The seal is a Viton O-ring, located in a machined groove in the cask closure lid forging.
- The purge and gas sampling penetration and the drain penetration.

Each seal is provided with a redundant backup seal. All seals can be leak-tested.

4.1.1 Containment Vessel

The containment vessel is comprised of those components described above which form the containment boundary. The design criteria for the containment vessel are provided in Section 2.1.2 and summarized in Table 4.1-1.



*Note:
Containment Boundary at Penetration Not Shown

Figure 4.1-1. Cask Containment Boundary

768370-48A

Table 4.1-1

Containment Vessel Design Criteria Summary

Design Life	25 years
Number and Type of Fuel Assemblies	3 PWR or 7 BWR
Internal Cask Atmosphere	Helium at 1 atmosphere
Ambient Temperature	100°F
Decay Heat Load (Maximum)	1740 watts
Solar Heat Load (Maximum)	123 BTU/hr-ft^2
Design Pressure (Internal)	35 psig
Design Pressure (External)	286 psig
Design Temperature (Maximum)	300°F
Design Temperature, Fuel Cladding, Long Term	706°F
Surface Protection	None
a) Interior Cask Body Surfaces	
b) Exterior Exposed Surfaces	
c) Sealing Surfaces	
Applicable Design Rules	ASME B&PVC, Section III Subsection NB Regulatory Guides 7.6 & 7.8

4.1.2 Containment Penetrations

The TITAN LWT cask has only two containment penetrations in addition to the closure lid. These are a purge/gas sampling penetration and a drain penetration. These penetrations are shown in Figures 4.1-2 and 4.1-3, respectively, and are both located in the closure lid. The penetrations are described in Section 1.2.1. A summary of the containment penetration component specifications is presented in Table 4.1-2.

4.1.3 Seals and Welds

The seal design for the cask closure lid includes a double O-ring seal arrangement. The inner seal is considered part of the containment boundary. The outer seal provides a means of leak testing the inner seal and acts as a backup seal. The seal design is illustrated in Figure 4.1-4.

A dovetail groove for each seal is machined into the lid forging. The contact surfaces are machined to a 63 microinch surface finish. The material for the seals is Viton. The closure seal is established by contact pressure applied by the closure bolt loads, which keep the mating surfaces in contact.

The penetration cover seals, illustrated in Figure 4.1-5, are a double seal arrangement. Each penetration has an inner cover which is bolted in place and an outer cover which is bolted in place. Each cover has a Viton O-ring seal. The seal design is essentially the same as the lid closure seal. The inner seals of the gas sampling penetration and the drain penetration are part of the containment boundary. The outer seals provide a means of leak-testing these seals and act as back-up seals. A summary of the seal component specifications is presented in Table 4.1-3.

Seals are used in each of the three seal verification test ports and in the quick-disconnect couplings. These seals are not part of the containment boundary.

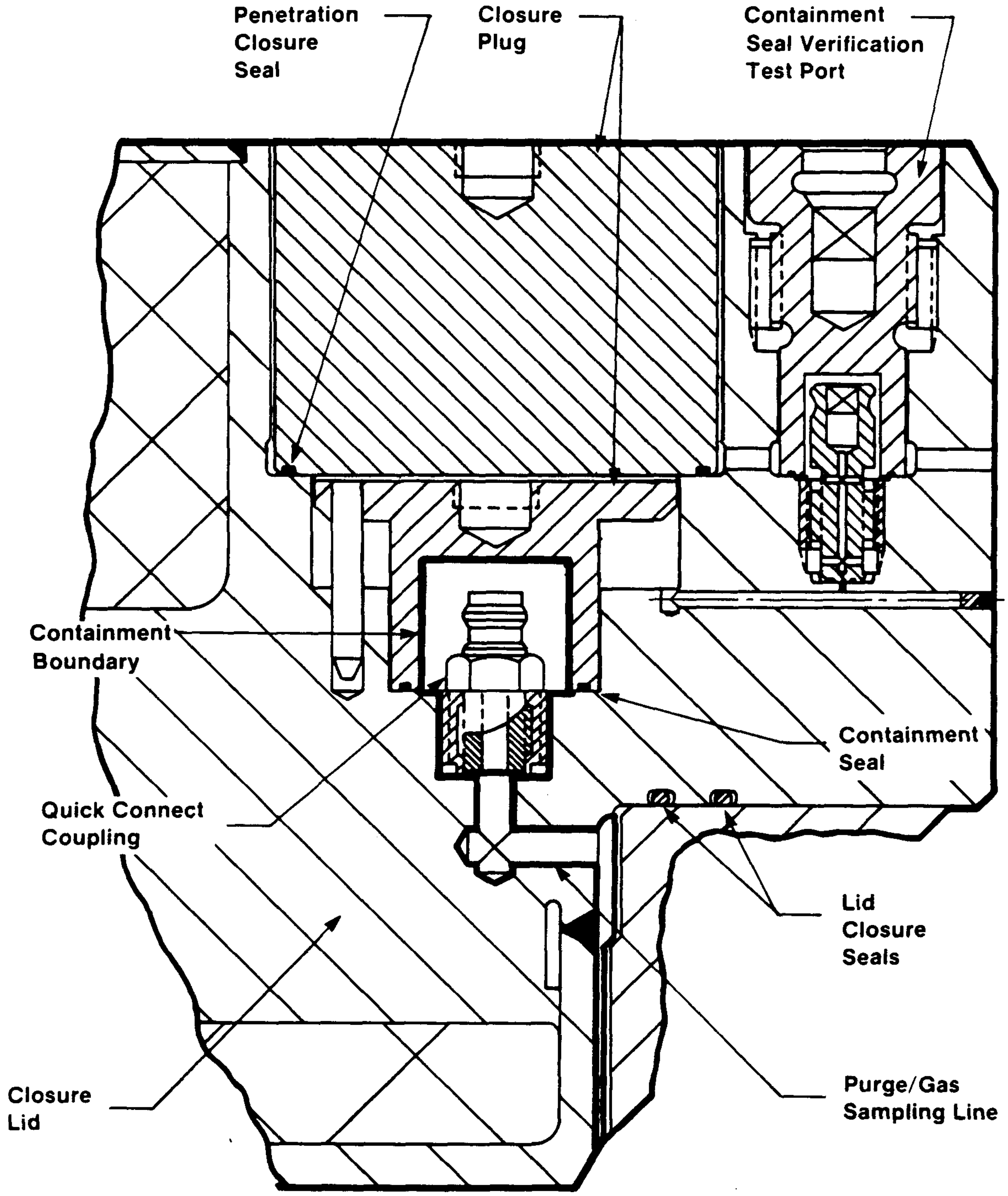


Figure 4.1-2 Purge and Gas Sampling Penetration

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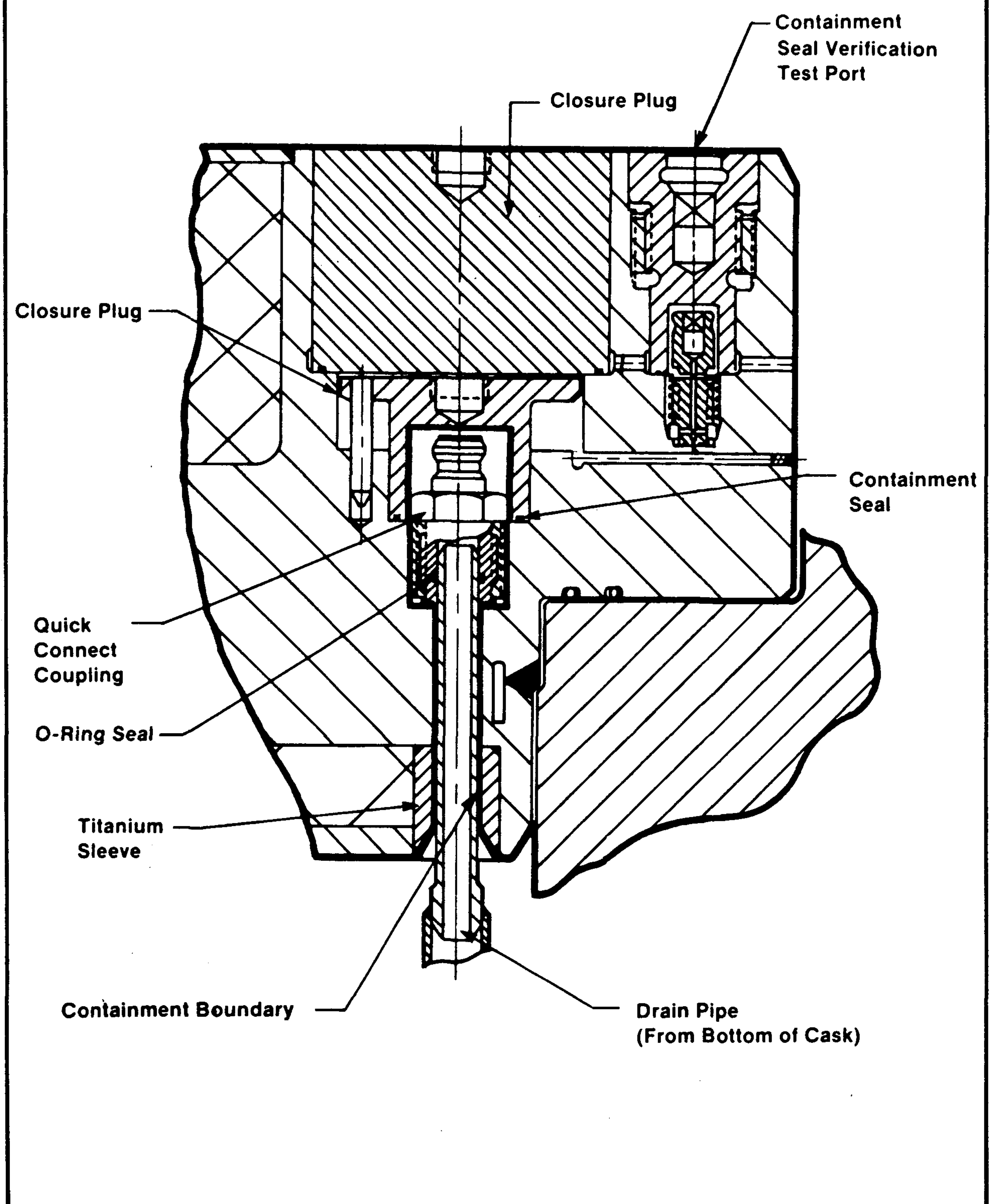


Figure 4.1-3. Drain Port

Table 4.1-2
Specifications for
Penetration Components

Penetration	Component	Specification
Purge/gas sampling port	Quick connect coupling (male)	Snap-Tite Coupling Nipple* (P/N S28-1N10-(5/8)56-V)
Drain port	Quick connect coupling (male)	Snap-Tite Coupling Nipple* (P/N S28-1N10-(1 3/8)56-V)

* See Vendor information in Section 4.5

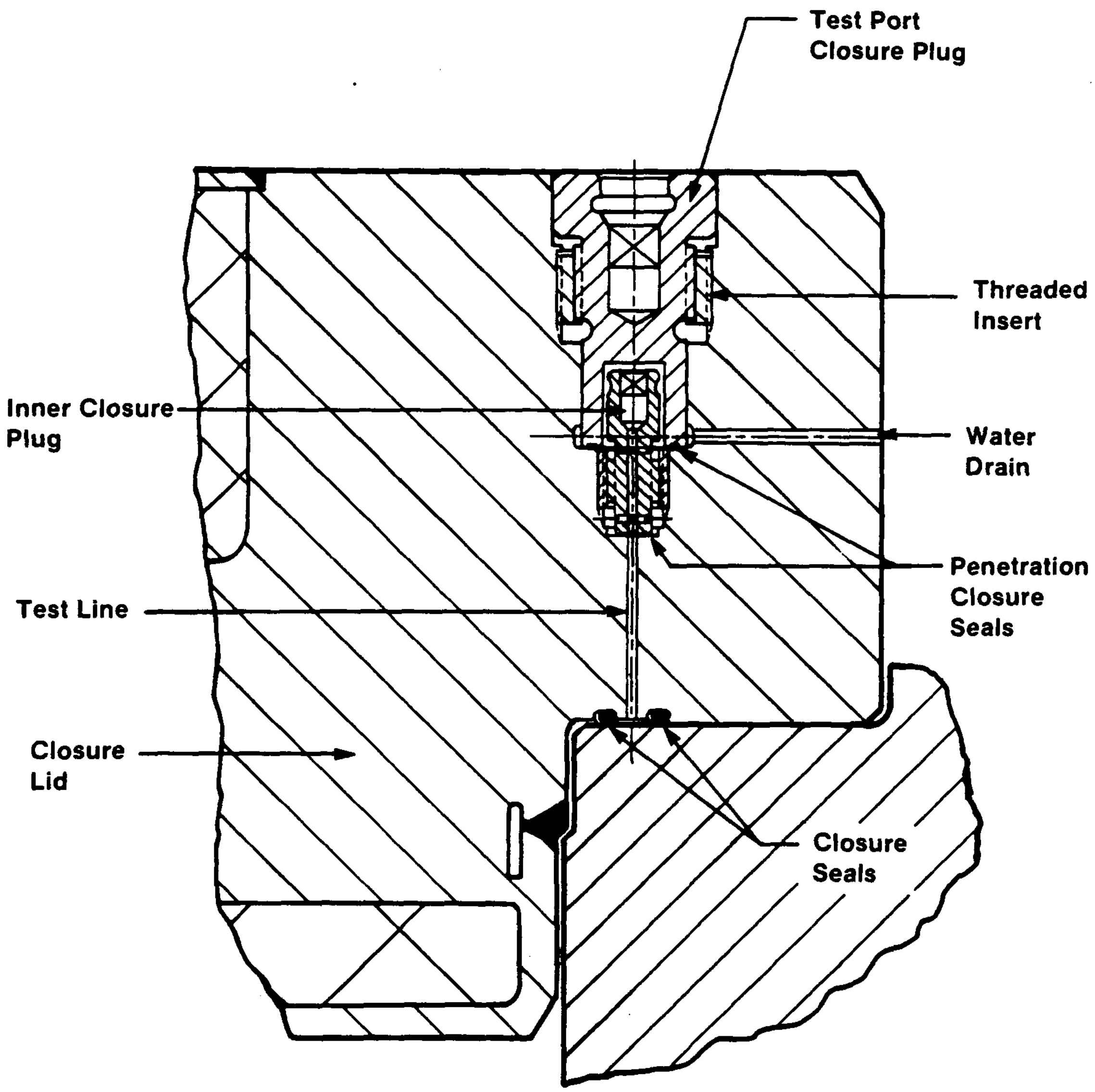


Figure 4.1-4. Closure Lid Seal and Seal Verification Test Port

768370-13A

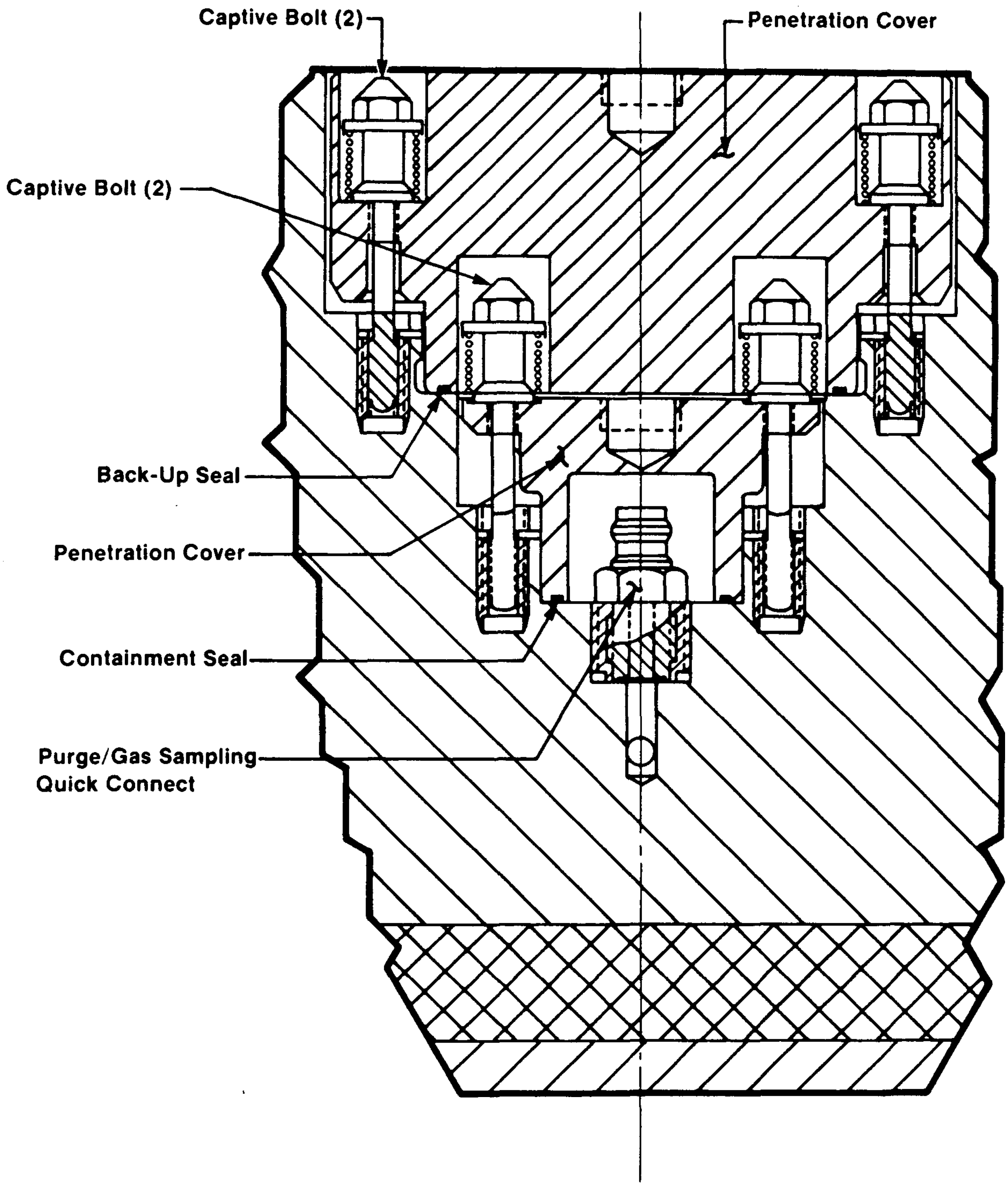


Figure 4.1-5. Penetration Covers

768370-49A

Table 4.1-3
Seal Design Characteristics

Seal	Design Characteristics
Closure lid inner O-ring seal	Parker #2 - 474 Viton V835-75 O-ring
Closure lid outer O-ring seal	Parker #2 - 475 Viton V835-75 O-ring
Purge/gas sampling port inner O-ring seal	Parker #2 - 136 Viton V835-75 O-ring
Purge/gas sampling port outer O-ring seal	Parker #2 - 158 Viton V835-75 O-ring
Drain port inner O-ring seal	Parker #2 - 136 Viton V835-75 O-ring
Drain port outer O-ring seal	Parker #2 - 158 Viton V835-75 O-ring

Parker #2 Viton O-ring	Operating Temperature Range: -40°F to 400°F Maximum Pressure: 1500 psi

Six welds are a part of the containment boundary. These welds are listed below:

- Inner titanium shell-to-bottom forging weld - a circumferential, single vee groove weld.

- Inner titanium shell-to-top forging weld - a circumferential, single vee groove weld.

- Closure lid inner titanium plate-to-titanium forging weld - a circumferential, single vee-groove weld.

- Drain port titanium bushing-to closure lid inner plate weld - a circumferential, bevel weld.

- Drain port titanium bushing-to-closure lid forging weld - a circumferential, bevel weld.

- Cask body inner shell longitudinal weld - a longitudinal weld that runs the length of the shell. This weld will not be present if the inner shell is manufactured as an extrusion rather than rolled and welded.

An additional circumferential weld would be present in the upper titanium shell if this item is rolled and welded. All material is Grade 9 Titanium. These welds are part of the containment vessel and will conform to the design criteria specified in Section 2.1.2.

4.1.4 Closure

The closure devices for the cask include the closure lid, penetration cover for purge/gas sampling line, penetration cover for drain line, and covers for the three seal verification ports. A summary of the closure design features is presented in Table 4.1-4.

Table 4.1-4
Closure Design Features

Closure	Seal Material	Flange or Cover Diameter[t]	Flange or Cover Thickness[t]	Bolts Number	Bolts Diameter[t]	Bolts Torque*
Cask Lid	Viton (V835-75)	37.50	12.60	16	1.875	2100-2300
Purge/Gas Sampling Port**	Outer-Viton (V835-75)	7.625	4.00	2	0.375	57-63
	Inner-Viton (V835-75)	4.375	2.625	2	0.375	57-63
Drain Port**	Outer-Viton (V835-75)	7.625	4.00	2	0.375	57-63
	Inner-Viton (V835-75)	4.375	2.625	2	0.375	57-63
Seal Verification Test Ports(3)	Viton (V835-75)	2.30	4.00	--	--	95-105

t = Dimensions in inches
* = Value in ft-lbs.

** Port has double closure

4.2 Requirements for Normal Conditions of Transport

The TITAN LWT cask will withstand pressures and temperatures in excess of those encountered in normal transport. The maximum normal operating pressure (MNOP) will be less than 35 psig with an average gas temperature of about 340°F. The containment boundary will have temperatures ranging from 200°F to 280°F for the Heat Condition. The elastomer seals will have a temperature of 200°F for the Heat Condition. Under these conditions, there should be no release of radioactive materials.

The MNOP is based on the assumption that all of the gases (fission gases and the initial charge of helium) contained in the rods escape into the cask cavity. The 35 psig MNOP is also based on an average gas temperature of 400°F which is conservative. The gross volume of the 180 inch long by 23.76 inch diameter cavity is 46.2 cubic feet. Three W 17x17 assemblies would have a volume of about 8 cubic feet. Of the various assemblies to be transported in the cask, the W 17x17 assembly yields the highest pressures. The basket would occupy about 3.8 cubic feet leaving a net volume of the cask loaded with 3 PWR assemblies of 34.3 cubic feet. The MNOP is based on the following quantities (sources) of gas:

Helium backfill (at 100°F):	38.1 gm mols
Helium released from fuel rods (initial fill):	25.4 gm mols
Fission gases released (Xenon and Krypton):	19.5 gm mols

The bolt loads are established on the basis that there be no separation of the closure lid-to-flange joint as a consequence of differential thermal expansion or mechanical loads resulting from the various Normal and Accident Conditions.

4.2.1 Containment of Radioactive Material

The cask will be designed to a leaktight capability as defined in Reference 4.5.1. However, as the results of the DOE's ongoing source term evaluation program become available, leakage testing requirements will be set to satisfy containment requirements of 10 CFR 71.51 (a).

4.2.2 Pressurization of Containment Vessel

There are no sources or mechanisms other than those used in the determination of the maximum normal operating pressure that can cause pressurization of the cask.

4.2.3 Containment Criterion

As stated in Section 4.2.1 above, compliance with the maximum allowable release rate requirement of $1x10^{-6}$ A_2/hour for Normal Conditions of Transport will be satisfied by demonstrating leaktightness which is a leak rate of less than 1×10^{-7} std cm^3/sec with an upstream (internal) pressure of 1 atm. absolute and a downstream pressure (between the double O-rings) of 0.01 atm. absolute.

4.3 Containment Requirements for the Hypothetical Accident Conditions

Under the accident conditions, the pressure in the cask cavity would not exceed the MNOP. The thermal event is evaluated for an initial condition that does not include an insolation heat load. Thus, while the temperatures increase during the thermal accident, the maximum average cladding temperature would be 410°F and the average basket temperature would be 366°F. Thus, it is concluded that the gas temperatures would not exceed those for which the MNOP was established.

The thermal accident is evaluated assuming that the impact limiters remain attached to the cask. Even if the impact limiter is crushed, the temperatures of the material in the vicinity of the seals will remain below 260°F which is well within the temperature limits of the O-ring material.

Figure 3.5-4 shows the predicted temperature history of the seals for the thermal accident. The outer seal begins with an initial temperature of 129°F, reaches about 150°F at 30 minutes, increases to 198°F at 1 hour, 242°F at 2 hours, and peaks at 258°F about 4.3 hours after start of the thermal event.

Based on the analysis using SCANS, there are no loads on the cask structure that would indicate that the closure would separate from the flange during any of the drop accidents. During the final design, this section will include figures from detailed finite element analyses which will illustrate the deformed geometry in the cask in the closure lid-to-flange interface region to demonstrate that no separation would be predicted.

With no separation during the drop and puncture accidents, and acceptable temperatures during the thermal event, it is concluded that the containment criterion for accident conditions will be met. Compliance with that criterion will be by demonstration of leaktightness following the drop and puncture testing on a 1/2 scale model and by analysis for the thermal test.

4.3.1 Fission Gas Products

The fission gas products that could be available for release in the containment vessel under the Hypothetical Accident Conditions are the same as those for Normal Conditions of Transport since it is assumed in the evaluation of the Maximum Normal Operating Pressure that all the fission gases have escaped from the spent fuel rods.

4.3.2 Containment of Radioactive Material

The total rate of release of radioactive material that is permitted following the Hypothetical Accident Conditions is 5900 times that allowed during Normal Conditions of Transport (A_2/week or 5.9 $A_2 \times 10^{-3}$/hr for accident conditions vs 1×10^{-6} A_2/hr for Normal Conditions of Transport). It is noted that the total krypton-85 inventory in 3 PWR spent fuel assemblies, 10 years out of reactor with 35,000 MWD/MTIHM burnup is 7430 ci. So the requirement that there be no leakage of krypton-85 exceeding 10,000 Ci in one week is satisfied even if it is assumed that all of the krypton-85 was released from the fuel and escaped from the cask. The amount of krypton-85 in 7 BWR assemblies is even less (5680 ci).

The considerations of specific activity of the medium that might be released through a leak in the containment boundary are the same as those discussed in Section 4.2.1.

4.3.3 Containment Criterion

The cask will be designed to a leaktight capability as defined in ANSI N14.5. However, as the results of the DOE's ongoing source term evaluation program become available, leakage testing requirements will be set to satisfy containment requirements of 10 CFR 71.51(a).

4.4 Special Requirements

Paragraph 71.63 of 10 CFR Part 71 requires shipments of more than 20 Ci of plutonium to be packaged as a solid and to be packaged in a separate inner container placed within the outer packaging which meets the containment criteria for Normal and Accident Conditions. Since shipments of reactor fuel elements are exempt from the requirements of this section, there are no special requirements for the TITAN LWT cask.

4.5 Appendix

4.5.1 References

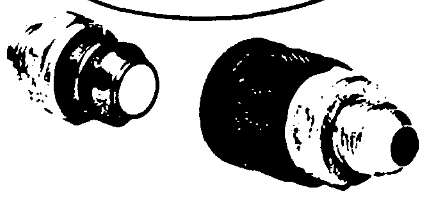
4.5.1 ANSI N14.5-1987, "American National Standards for Radioactive Materials - Leakage Tests on Packages for Shipment".

4.5.2 Vendor Literature

This section contains vendor catalog information on the quick-disconnect couplings used for the cask purge/gas sampling and drain penetrations.

Snap-tite Quick Disconnect Couplings for Applications Requiring Virtually No Air Inclusion or Spillage

28-1 Series
For Pressures to 1000 psi



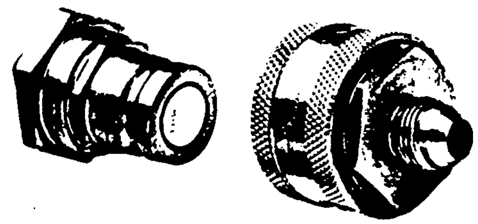
Featuring:
- Low pressure drop
- No spill - minimum air inclusion
- Maximum flow capacity
- Lightweight - compact design
- 1/4" - 2" size range
- Aluminum or Stainless Steel construction
- Smooth push to connect
- Color coded positive lock indicator standard on all models
- Multitude of end fittings: MS33656, MS33657, MS33649, MS33514, MS33515, NPSF, NPT, and SAE
- Wide range of seal material
- Performance meets or exceeds MIL-C-7413B and MIL-C-25427A

This space-age quick disconnect is machined and tested to meet or exceed critical standards. Snap-tite meets MIL-Q-9858A quality control system and exceeds MIL-I-45208 inspection system. Lightweight, maximum flow and minimal pressure drop are design parameters where the 28-1 Series is unsurpassed. The small envelope size permits less weight and Snap-tite's excellent internal design assures maximum flow with minimum pressure drop. Operating pressure rating for 1/4" through 1" sizes is1000 psi; 1 1/4" through 2" sizes, 600 psi.

A smooth automatic, push-to-connect feature, ideal for one hand operation when one half is mounted, sets the 28-1 Series apart from all others. The unit can be connected against a closed system, has no seal transition and provides a green color-coded lock indicator.

Typical applications:
- **Low pressure hydraulic systems**
- **High purity systems**
- **Fuel systems**
- **Electronic coolant**
- **High reliability systems**

29 Series
For Pressures to 5500 psi



Featuring:
- Low pressure drop
- No spill - minimum air inclusion
- Maximum flow capacity
- High pressure design
- 1/8" - 1 1/4" size range
- Aluminum or Stainless Steel construction
- Smooth push to connect
- Multitude of end fittings: MS33656, MS33657, MS33649, MS33514, MS33515, NPSF, NPT and SAE
- Wide range of seal material
- Performance meets or exceeds MIL-C-7413B and MIL-C-25427A

Snap-tite's 29 Series Quick Disconnect offers full-flow characteristics, can handle high pressure as well as gravity flow systems, and contains minimal seals for greater reliability. Snap-tite meets MIL-Q-9858A quality control system and exceeds MIL-I-45208 inspection system.

29 Series has established an excellent performance record over the past 20 years.

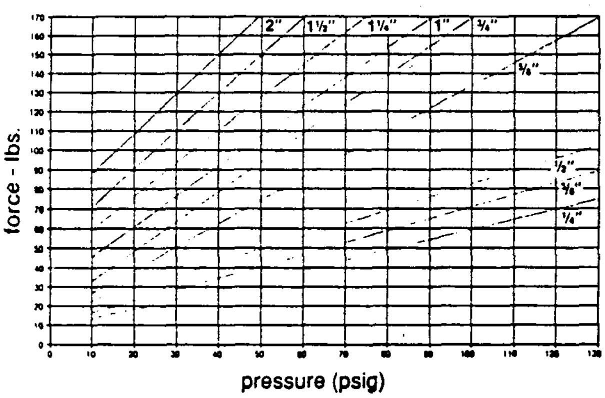
In addition to hydraulic applications, the 29 Series Quick Disconnect is the ideal choice where minimal spillage or air inclusion, safety, cleanliness and precise function in high pressure hydraulic systems are prime requisites.

Like all Snap-tite Quick Disconnect Couplings, the 29 Series connects and disconnects quickly and positively, providing positive shutoff automatically. A smooth automatic, push-to-connect feature, ideal for one hand operation when one half is mounted, sets the 29 Series apart from all others.

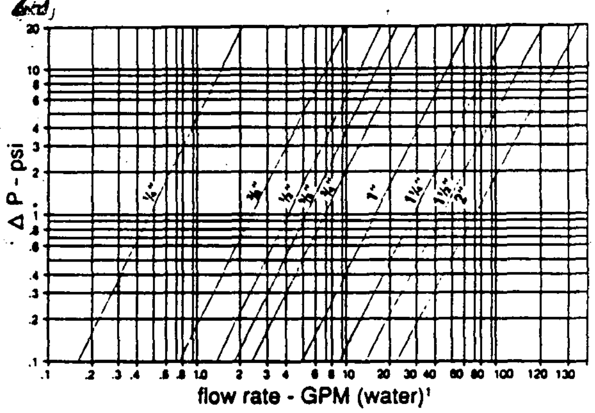
Typical applications:
- **High pressure hydraulic systems**
- **High purity systems**
- **Fuel systems**
- **Electronic coolant**
- **High reliability systems**

28-1 Series
Performance Data

1 **Force to Connect**



2 **Pressure Loss vs. Flow**



¹ Pressure loss vs. flow is in water with specific gravity of 1.0. For fluids with sg of .85 multiply by 1.58; for fluids with sg of .83, multiply by 1.60. Temperatures 100°F.

3

Pressure Ratings			
Coupling Size	Code	**Aluminum** Working Pressure (psig)	**Stainless Steel** Working Pressure (psig)
1/4	-4	1000	1000
3/8	-6	1000	1000
1/2	-8	1000	1000
5/8	-10	1000	1000
3/4	-12	1000	1000
1	-16	1000	1000
1 1/4	-20	600	600
1 1/2	-24	600	600
2	-32	600	600

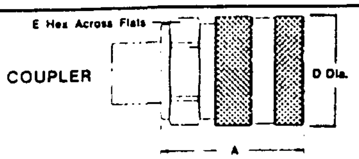
Pressure ratings were established under static pressure conditions. Therefore, pressure ratings for any given flow, pressure surge and/or vibration may vary these ratings.
Proof pressure = 1.5 x working pressure **Burst pressure = 2.5 x working pressure**

4

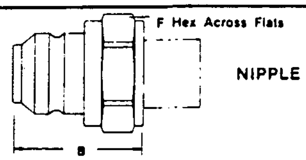
Air Inclusion on Connect, Spillage on Disconnect			
Coupling Size	Code	Air Inclusion* (cubic centimeters)	Spillage* (cubic centimeters)
1/4	-4	.05	.01
3/8	-6	.18	.03
1/2	-8	.28	.04
5/8	-10	.31	.13
3/4	-12	.48	.15
1	-16	.80	.30
1 1/4	-20	1.57	.40
1 1/2	-24	2.00	.70
2	-32	3.00	1.00

***NOTE: Air inclusion at 0 psig internal pressure; spillage at 15 psig internal pressure.**

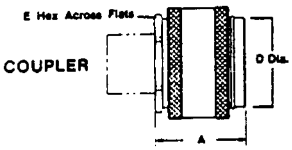
Dimensions and Weights



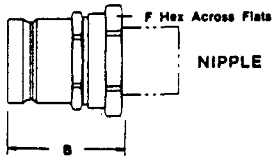
28-1 Series



Body Size	Code	Coupler A	Coupler D	Coupler E	Coupler Weight[1]	Connected Length	Nipple B	Nipple F	Nipple Weight[1]
1/4	-4	1.25	.88	.75	.05	1.80	1.10	.75	.02
3/8	-6	1.44	1.12	1.00	.11	2.02	1.20	1.00	.05
1/2	-8	1.61	1.34	1.13	.17	2.15	1.25	1.13	.06
5/8	-10	1.70	1.58	1.38	.23	2.45	1.50	1.38	.14
3/4	-12	2.05	1.80	1.50	.36	2.89	1.71	1.50	.15
1	-16	2.70	2.32	2.00	.74	3.72	2.19	2.00	.29
1 1/4	-20	2.56	2.63	2.25	.76	3.51	2.16	2.25	.30
1 1/2	-24	3.05	2.96	2.88	1.33	3.76	2.16	2.75	.33
2	-32	3.55	3.92	3.25	1.97	4.73	2.84	3.25	.36



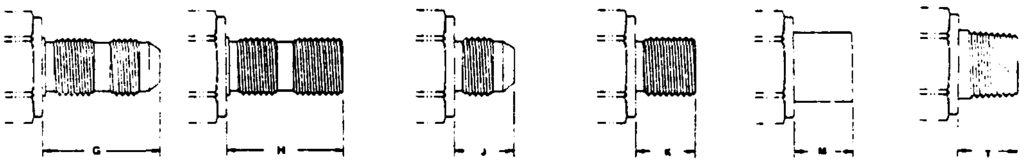
29 Series



Body Size	Code	Coupler A	Coupler D	Coupler E	Coupler Weight[1]	Connected Length	Nipple B	Nipple F	Nipple Weight[1]
1/8	-2	1.20	.88	.75	.05	1.71	1.07	.69	.03
1/4	-4	1.31	1.38	1.13	.13	2.01	1.30	1.13	.07
3/8	-6	1.57	1.63	1.25	.20	2.36	1.48	1.38	.10
1/2	-8	1.53	1.94	1.50	.31	2.64	1.76	1.63	.16
1/2 x 5/8	8-10	1.53	1.94	1.50	.42	2.37	1.76	1.63	.23
3/4	-12	1.89	2.35	1.88	.49	2.81	2.09	1.88	.31
1	-16	2.45	2.44	2.13	.70	3.60	2.50	2.13	.36
1 1/4	-20	2.52	2.88	2.69†	.86	3.64	2.90	2.75†	.71

† Two wrench flats.

Common End Fitting Configurations, Dimensions and Weights



57	15	56	14	F	M
MS33657 Bulkhead Flared or **EB** SAE Bulkhead	MS33515 Bulkhead Flareless	MS33656 Male Flared or **EM** Male SAE	MS33514 Male Flareless	Female Pipe NPSF or **49** Female O'ring Boss MS33649	Male Taper Pipe NPT

Size	Code	G	Wt.[1]	H	Wt.[1]	J	Wt.[1]	K	Wt.[1]	M	Wt.[1]	T	Wt.[1]
1/8	-2	.95	.01	.86	.01	.45	.01	.38	.01	.52	.01	.38	.01
1/4	-4	1.05	.01	.97	.01	.55	.01	.45	.01	.69	.02	.56	.01
3/8	-6	1.13	.01	1.02	.01	.56	.01	.47	.01	.78	.04	.56	.01
1/2	-8	1.28	.03	1.16	.02	.66	.01	.56	.01	.94	.08	.75	.03
5/8	-10	1.42	.04	1.30	.03	.76	.02	.63	.02	—	.11	—	.03
3/4	-12	1.59	.06	1.41	.05	.86	.03	.69	.03	1.00	.13	.75	.03
1	-16	1.59	.09	1.41	.07	.91	.05	.69	.03	1.19	.19	.94	.06
1 1/4	-20	1.64	.14	1.41	.12	.96	.08	.69	.06	1.25	.29	.97	.09
1 1/2	-24	1.66	.17	1.41	.13	1.08	.11	.69	.07	1.28	.40	1.00	.11
2	-32	1.94	.21	1.61	.19	1.33	.15	.69	.10	1.34	.56	1.03	.16

[1] Weights are for aluminum. For Stainless Steel multiply aluminum weight by 2.7 All dimensions and weights are for reference only and are subject to change without notice. Dimension tolerances: A,B,D,E & F ± .03; Connected length ± .06

How to Order

Using the following simple steps, you can generate ordering numbers for Series 28-1 and 29 Quick Disconnect Couplings.

TYPICAL PART NO. A28-1C8-8F ___

MATERIAL	SERIES	BODY	COUPLER OR NIPPLE	END FITTING	TYPE OF END FITTING	SEALS
Aluminum Stainless Steel	28-1 29	C- Coupler N- Nipple	Code Size 2 = 1/8* 4 = 1/4 6 = 3/8 8 = 1/2 10 = 5/8 12 = 3/4 16 = 1 20 = 1 1/4 24 = 1 1/2† 32 = 2† * Available 29 only † Available 28-1 only	Code Size 2 = 1/8 4 = 1/4 6 = 3/8 8 = 1/2 10 = 5/8 12 = 3/4 16 = 1 20 = 1 1/4 24 = 1 1/2 32 = 2	57—Bulkhead MS33657 15—Bulkhead MS33515 56—Male MS33656 14—Male MS33514 49—Female MS33649 F—NPSF Female Straight Pipe Thread (through 1") M—NPT Male Taper Pipe Thread EM—SAE Male EB—SAE Bulkhead	**—Nitrile (AMS 3215) V—Viton (MIL-R-25897) JF—Nitrile (MIL-P-5315) MHO—Nitrile(MIL-P-29732) EPR—Ethylene propylene rubber **Standard unless otherwise specified; no letter designation required. For other seal compounds consult factory.

Accessories

28-1 Series
Dust Caps

- For Couplers...
A28-1DCC - Specify size code
- For Nipples...
A28-1DCN - Specify size code

29 Series
Dust Caps/Plugs

- For Couplers...
ADP29 - Specify size code
- For Nipples...
ADC29 - Specify size code

Pressure Cap
- For Nipple...
APC29 - Specify size code

Material designation: A - Aluminum S - Stainless Steel

Distributed by:

Snap-tite INC.
Quick Disconnect Division
201 Titusville Road
Union City, PA 16438
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5. SHIELDING EVALUATION

This section provides the shielding evaluation performed to support the preliminary design of the TITAN LWT cask. Because the source terms are larger for 3 PWR spent fuel assemblies than for 7 BWR spent fuel assemblies, the shielding thicknesses for the cask are governed by the PWR case. The evaluation which follows addresses the cask loaded with 3 PWR assemblies.

5.1 Discussion and Results

5.1.1 Discussion - Cask Shield Configuration

The cask is designed to serve as a transport cask for irradiated fuel. The present shield design meets all the requirements of 10 CFR Parts 71 and 72 and 49 CFR Part 173.

The shielded cask is shown in Figures 5.1-1 through 5.1-4. Figures 5.1-1 and 5.1-2 show the cask top half and top half details, respectively. The bottom half and bottom half details of the cask are shown in Figures 5.1-3 and 5.1-4. The shield thicknesses are summarized in Table 5.1-1.

The shielding materials, depleted uranium (DU) gamma shielding and Boro-Silicone neutron shielding, were chosen to meet the objective of maximizing the number of assemblies that can be transported, while staying within the weight limit. For example, DU was selected for the gamma shielding because it reduces the weight of the gamma shield by approximately 10% when compared to Pb and 63% when compared to steel.

As the result of these shield innovations, the cask has the capacity to handle 3 "worst case" PWR fuel assemblies meeting the criteria described in Section 5.2.

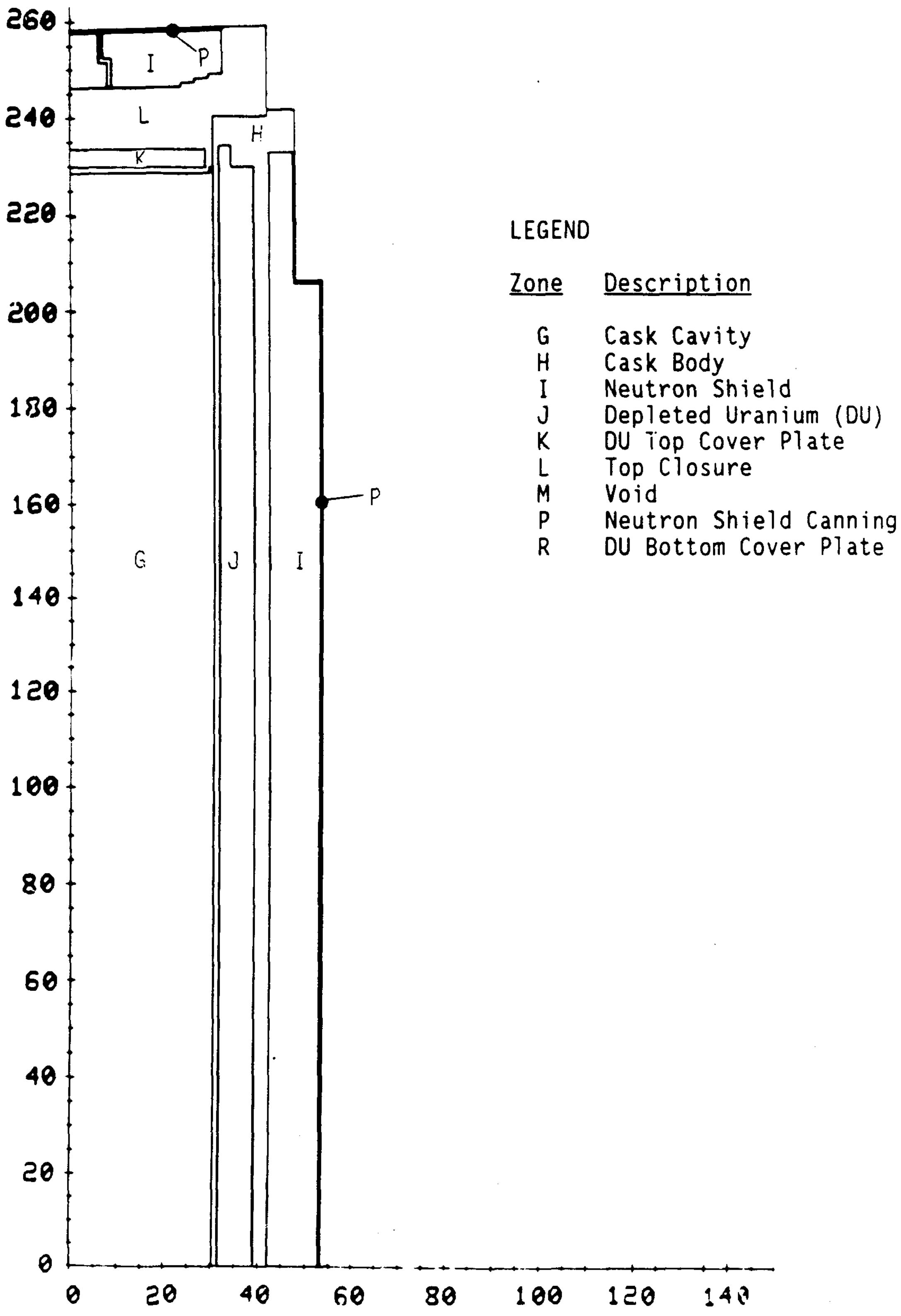


Figure 5.1-1 LWT Cask Model Top Half

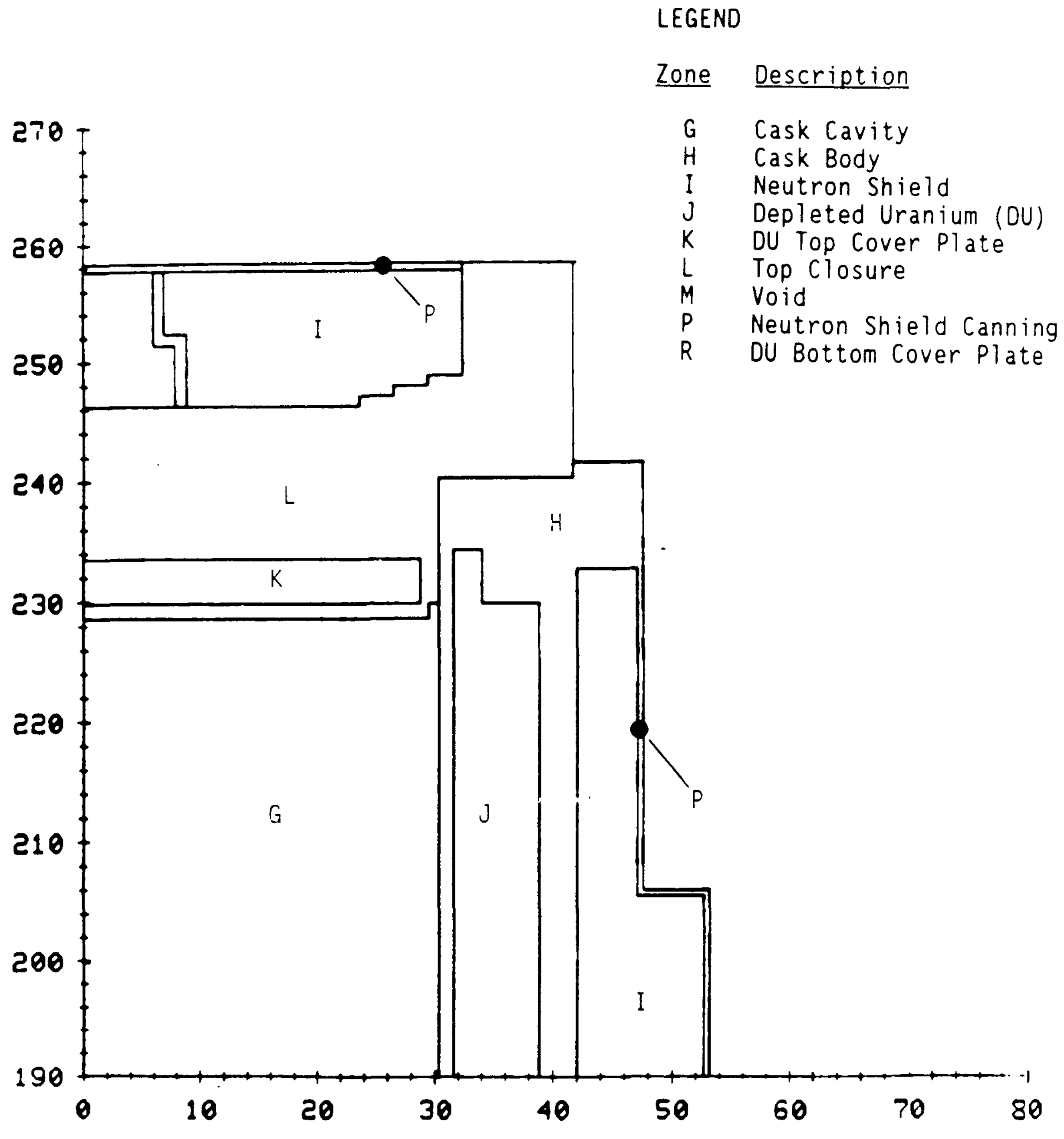
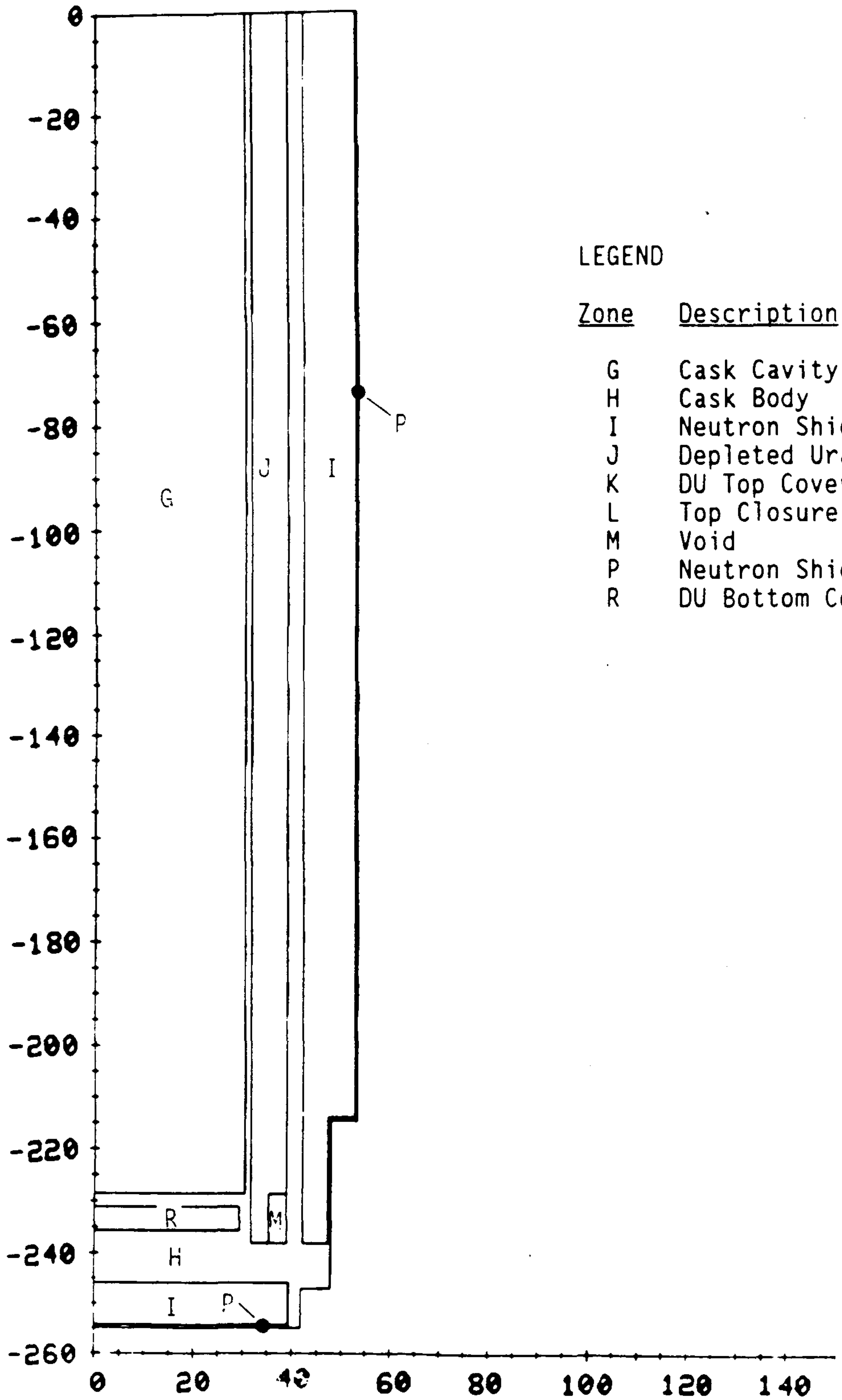


Figure 5.1-2 LWT Cask Model Top Half Details



Zone	Description
G	Cask Cavity
H	Cask Body
I	Neutron Shield
J	Depleted Uranium (DU)
K	DU Top Cover Plate
L	Top Closure
M	Void
P	Neutron Shield Canning
R	DU Bottom Cover Plate

Figure 5.1-3 LWT Cask Model Bottom Half

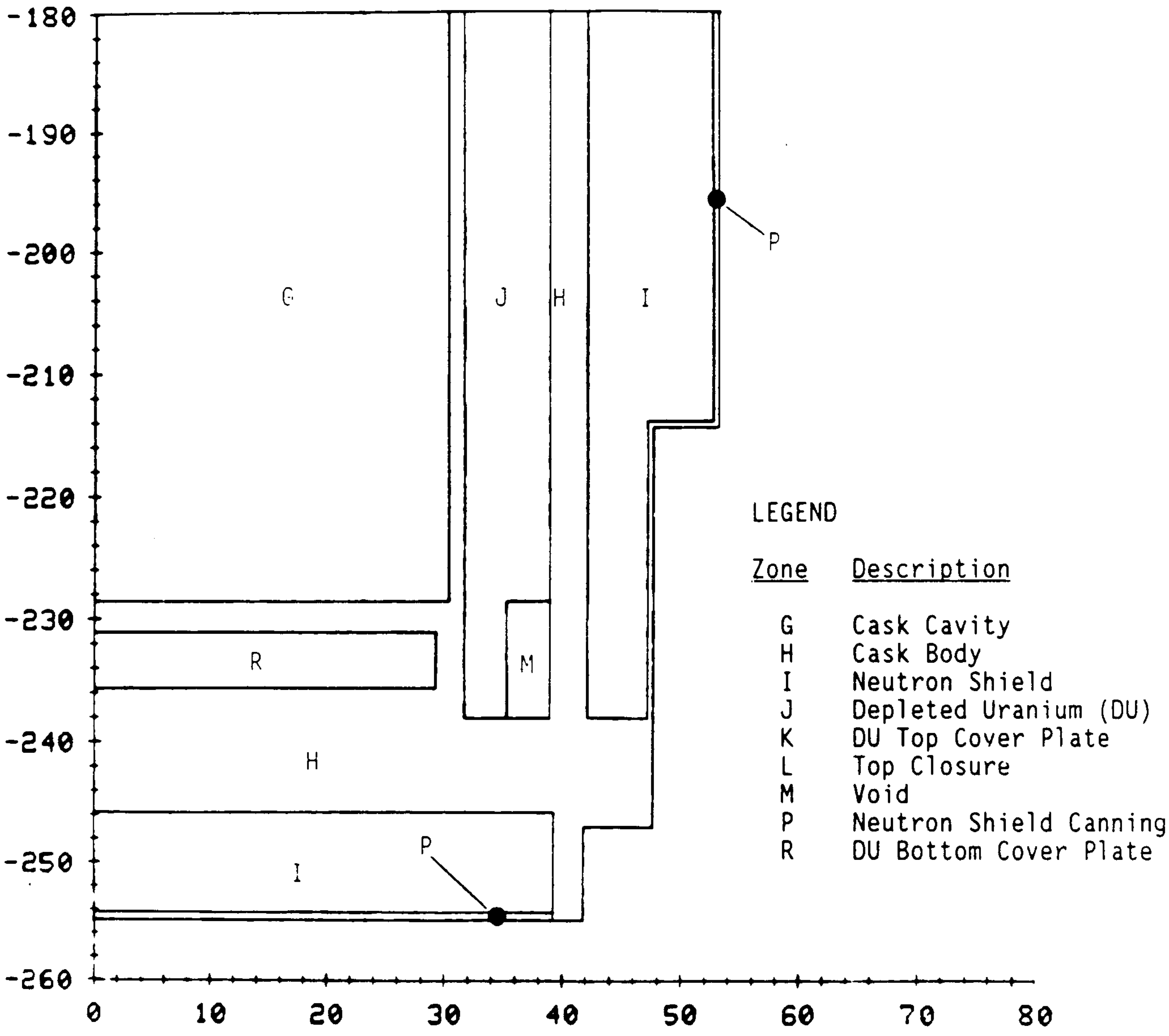


Figure 5.1-4 LWT Cask Model Bottom Half Details

Table 5.1-1
Shield Thicknesses for the LWT Cask
with Intact PWR Spent Fuel Assemblies

	Units	
	English, inches	Metric, cm
CASK:		
Overall Height (External)	202.95	515.49
Overall Diameter (External)	41.90	106.43
Cask Cavity		
Inside Diameter	23.88	60.66
Inside Height	180.00	457.20
CASK BODY:		
Titanium Thicknesses:		
Wall Thickness (Adjacent to UO_2)	0.50	1.27
(Between DU and Boro-Si)	1.25	3.175
Bottom Thickness (Adjacent to Cavity)	1.00	2.54
(Between DU and Boro-Si)	4.00	10.16
Top Closure Thickness		
Impact Cover	5.00	12.70
Seal Cover	0.50	1.27
GAMMA SHIELD (DU):		
Radial	2.87	7.29
Top	1.46	3.71
Bottom	1.785	4.534
NEUTRON SHIELD (Boro-Silicone):		
Radial Thickness	4.20	10.67
Top Closure Thickness	4.50	11.43
Bottom Thickness	3.315	8.42
NEUTRON SHIELD SHELL:		
Shell Thickness at Cask Diameter	0.190	0.483
Shell Thickness at Cask Ends	0.250	0.635

5.1.2 Summary of Results

The dimensions and material thicknesses used in the evaluation of the cask's performance against the external radiation standards for all packages are summarized in Table 5.1-1. The maximum dose rates that result from these shield thicknesses are summarized in Table 5.1-2 for the planes of interest (surface and 2 meters from the vehicle surface). Detailed calculational results for the environment surrounding the cask are given in Section 5.4. The anticipated dose rates at one meter from the external surface of the package following the Hypothetical Accident Conditions are discussed in Section 3.5.6.

5.2 Radiation Source Specification

The cask shield design is based on a composite PWR fuel assembly chosen to provide the most severe radiological source combination. The neutron and gamma sources are based on the following spent fuel characteristics:

Spent Fuel Characteristic	Description
Fuel Type	PWR - Fuel Section characterized by W 17x17, Head and Foot pieces characterized by W 15x15.
Spent Fuel Transport Array	Intact - 3 spent fuel assemblies placed in 3 cells of the basket assembly.
Irradiation History	35000 MWD/MTU
Fuel Enrichment	3.0 Weight Percent U-235
Decay Time	10 Years
Spent Fuel Heat Generation	< 2 kW total

The composite fuel assembly described above was chosen on the following basis:

- o The fuel assembly uranium and Zircaloy mass in the cask is based on the W 17x17 fuel assembly. The W 17x17 assembly has the largest weight of uranium of any of the assemblies evaluated with the exception of the B&W 15x15 (See Table 5.2-1). The W 17x17 assembly, however, contains more Alloy 718 in the core than the B&W 15x15.

Table 5.1-2
Summary of Maximum Dose Rates
and Radiation Level Limitations
(mrem/hr)

	Package Surface*			2 Meters from Surface of Vehicle			At Location of Driver Position
	Side	Top	Bottom	Side	Top	Bottom	Top
Normal Conditions							
Primary Gamma	37.58	107.81	158.05	6.90	1.37	2.40	0.70
Secondary Gamma	1.76	0.06	0.11	0.08	0.00	0.00	0.00
Neutron	16.28	0.87	2.46	0.89	0.03	0.05	0.01
Total	55.62	108.34	160.62	7.87	1.40	2.45	0.70
10 CFR 71 §71.47	200.00	200.00	200.00	10.00	10.00	10.00	2.00

*Includes impact limiters.

Table 5.2-1
Mass of Uranium Fuel Per Assembly
(Kgm)

Fuel Designer	W	W	W	W	W	CE	CE	B&W	B&W	EXXON	EXXON	EXXON
Fuel Rod Array	14x14	15x15	15x15 OFA	17x17	17x17 OFA	14x14	16x16	15x15 (WM)	17x17 (CE)	15x15 (WM)	14x14	17x17
Mass of Uranium	407.0	469.0	462.7	463.6	426.0	386.0	426.0	463.6	456.0	432.0	379.0	401.1
Mass of Associated Zircaloy-4	104.6	110.1	118.1	116.4	119.2	110.5	127.0	129.6	127.7	137.7	119.6	133.0
Mass of O_2	53.4	62.2	62.2	62.3	57.3	51.9	57.3	62.3	61.3	58.1	51.0	53.9
Mass of Hardware*	22.0	27.5	16.3	29.9	20.3	27.7	38.3	31.7	37.5	22.1	27.0	23.4

* Calculated by difference between "total" weight and calculated UO_2, Zircaloy-4 masses.

- The 3% by weight enrichment was chosen as the design basis to maximize the neutron flux required to achieve the design power level. The "maximum" flux results in increased production of several long half-life isotopes which are important to the gamma and neutron source terms after 10 years decay time. These include Co^{60}, Cs^{134}, Eu^{154} and Cm^{244}.

- The W 15x15 fuel assembly structural components above and below the fuel were chosen rather than the W 17x17 fuel assembly because of their relative mass and the predominance of Co bearing materials (304 stainless steel and Alloy 718).

The gamma and neutron source terms due to fission products, actinides, and activation isotopes associated with the fueled portion of the fuel assembly were determined using the ORIGEN II (Reference 5.2.1) computer program. The ORIGEN II input parameters associated with the analysis are shown in Table 5.2-2.

ORIGEN II solves for the isotopic generation and depletion of fission product and actinide nuclides including transmutation of nuclides. The matrix exponential method is used in the solution of the coupled, linear, first order ordinary differential equations with constant coefficients. The nuclear data file used in ORIGEN II is documented in Reference 5.2.2 and includes the Evaluated Nuclear Data File Version IV data for individual fission product isotope decay characteristics. This data file is the industry standard and has been developed for use in source term and decay heat analysis of PWR's.

The decay power associated with the PWR spent fuel assemblies has been predicted with the ORIGEN II computer program. The total decay power of an intact spent fuel array (3 spent fuel assemblies), including neutron activation isotope decay power, is 1.74 kw after a 10 year decay time. The intact fuel assembly array data includes the fuel assembly structure contribution and represents the predicted data for intact fuel assemblies.

Table 5.2-2

Parameters Used to Calculate LWT Cask Radiation Source Terms Using Origen II

Assembly Type	W PWR 17x17 Fuel W PWR 15x15 Hardware
Power Level (MW/Assembly)	19.74
Power Level (MW/MTU)	42.6
Irradiation Time (Days)	822.0
Burnup (Mwd/MTU)	35000
Mass of Uranium (Kg/assembly)	463.6
Mass of Zircaloy (Kg/assembly)	116.4
Mass of Hardware (Kg/assembly) in Fuel Zone	5.00
Mass of Hardware (Kg/assembly)* Above and Below Fuel	31.5

* Source term separately calculated based on neutron flux environments above and below fuel.

5.2.1 Gamma Ray Source Terms

The gamma ray source term of the cask spent fuel array is based on the ORIGEN II analysis of the irradiation of 3 PWR assemblies for the fuel design parameters and burnup discussed in Section 5.2. The ORIGEN II results for 3 intact spent fuel assemblies are listed in Table 5.2-3. The ORIGEN II data are grouped into 18 energy groups with the average energy listed with each group energy release rate, Mev/second, at 10 year decay time in Table 5.2-4. Source term values in Table 5.2-4 include the bremstrahlung produced due to the beta particles emitted during fission product decay and subsequent bremstrahlung production due to the beta particle slowing down in the fuel matrix. The Alloy 718/stainless steel neutron activation gamma sources due to the grids in an intact PWR assembly have been included in the source term. The fission product inventory and gamma ray source term at ten years after shutdown are primarily dependent on the fuel burnup characteristics and the operating neutron flux level. The quantity of specific neutron activation products which are produced due to transmutation of specific fission product isotopes or fuel rod and/or assembly structural material isotopes are in proportion to the operating neutron flux level.

The principal fission product isotopes and neutron activation product isotopes providing the gamma ray source at 10 years after shutdown are summarized in Table 5.2-3. The Co^{60} isotope is a principal neutron activation product and is primarily produced due to neutron capture in the trace quantities of Co^{59} in the structural materials of the PWR fuel assembly (including the Zircaloy clad). The Cs^{134} isotope is produced by neutron activation of the stable Cs^{133} fission product. The Eu^{154} isotope is primarily produced due to the transmutation chains of the fission product isotopes Sm^{147}, Sm^{149}, Sm^{151}, and Sm^{152} which result in the production of Sm^{153} which then decays to Eu^{153}.

The "cross section" set used for the primary and secondary gamma ray dose rate sets includes 20 groups. The group structure is shown in Table 5.2-7. The primary gamma source term uses 15 of these groups. The secondary gamma production, resulting from neutron interactions with the cask internals and

Table 5.2-3
Principal Isotopes Contributing to the Primary Gamma Source of Package
3 Assemblies - 10 Years Decay Time

Isotope	Curies	Isotope	Curies
Kr^{85}	7.05×10^{3}	I^{129}	4.77×10^{-2}
Sr^{90}	8.19×10^{4}	Cs^{134}	8.64×10^{3}
Y^{90}	8.19×10^{4}	Cs^{137}	1.23×10^{5}
Nb^{93m}	1.20	Ba^{137m}	1.16×10^{5}
Nb^{94}	1.55	Ce^{144}	2.27×10^{2}
Rh^{106}	9.18×10^{2}	Pr^{144}	2.27×10^{2}
Ag^{110m}	3.09×10^{-1}	Pm^{147}	1.30×10^{4}
Cd^{113m}	5.73×10^{1}	Sm^{151}	4.41×10^{2}
Sn^{119m}	3.11×10^{-1}	Eu^{152}	3.99
Sn^{121m}	1.05	Eu^{154}	7.74×10^{3}
Sn^{126}	1.20	Mn^{54}	1.09×10^{-1}
Sb^{125}	2.04×10^{3}	Co^{60}	2.80×10^{3}
Sb^{126}	1.69×10^{-1}		
Sb^{126m}	1.20	Total (This Table)	4.46×10^{5}
Te^{125m}	4.97×10^{2}	Grand Total	4.50×10^{5}

Table 5.2-4

Photon Number and Energy Release for LWT Cask*

Activation, Actinide, and
Fission Product Contributions
3 Assemblies - 10 Years Decay Time

Energy	Photons/sec	Mev/sec	Height of Zone
1.5×10^{-2}	2.78×10^{15}	4.17×10^{13}	365.76 cm
2.5×10^{-2}	6.04×10^{14}	1.51×10^{13}	
3.75×10^{-2}	7.39×10^{14}	2.77×10^{13}	
5.75×10^{-2}	5.58×10^{14}	3.21×10^{13}	
8.50×10^{-2}	3.29×10^{14}	2.80×10^{13}	
0.125	3.22×10^{14}	4.03×10^{13}	
0.225	2.72×10^{14}	6.12×10^{13}	
0.375	1.34×10^{14}	5.02×10^{13}	
0.575	4.96×10^{15}	2.85×10^{15}	
0.850	4.34×10^{14}	3.69×10^{14}	
1.25	3.65×10^{14}	4.56×10^{14}	
1.75	5.11×10^{12}	8.94×10^{12}	
2.25	1.15×10^{11}	2.59×10^{11}	
2.75	7.89×10^{9}	2.17×10^{10}	
3.50	9.97×10^{8}	3.49×10^{9}	
5.00	1.90×10^{7}	9.51×10^{7}	
7.00	2.19×10^{6}	1.53×10^{7}	
11.0	2.52×10^{5}	2.77×10^{6}	

*Does not include assembly head and foot pieces.

cask body, produce gamma rays with a wide range of energies and the cross section set is used to evaluate the transport of both secondary and primary gamma rays.

The gamma ray source terms for the fuel assembly structural regions above and below the core results from the activation of the stainless steel and Alloy 718 portion of these structures. The source terms used for these regions for the dose rate analysis are shown in Table 5.2-5.

The only activation source in these structural zones which has a significant gamma ray contribution after a decay time of 10 years is Co^{60}. The Co^{60} source term is based on a maximum Co content of the stainless steel of 0.08% for 304 stainless steel, 0.47% for Alloy 718 and 0.002% for Zircaloy-4. (The material compositions and volume fractions for these zones are shown in Table 5.3-2 and Table 5.3-3).

5.2.2 Neutron Source Terms

The neutron source term of the spent fuel array is based on analysis of the fuel irradiation using ORIGEN II and the ENDF/B-IV data library (Reference 5.2.2). The principal neutron source is the curium-244 produced by transmutation and decay of uranium, plutonium, americium, and curium isotopes starting with the neutron capture of uranium-238. The neutron source term associated with the spent fuel array used in the analysis of the cask includes the inherent neutron source produced by spontaneous fission and (a, n) reactions with the oxide fuel form of the fuel pellets. In addition, the inherent neutron source due to spontaneous fission and (a, n) sources is increased by the subcritical multiplication of the spent fuel array. A subcritical multiplication factor of 1.43 is defined for the spent fuel arrangement of the cask design. This multiplication factor is based on a predicted k_{eff} of 0.3. The neutron source data listed in Table 5.2-6 identifies the total neutron source term at a 10 year decay time for an intact spent fuel array of 3 assemblies. In the neutron transport analysis of the cask, the neutron source term is uniformly distributed in the radial direction and the axial neutron source distribution

Table 5.2-5

Photon and Energy Release for Assembly Components Above and Below Fuel

Activation Products (Co^{60})

3 Assemblies - 10 Year Decay Time

Component	Energy	Photons/sec	Mev/sec	Height of Zone (cm)
Assembly Inlet	1.25	2.57×10^{12}	3.21×10^{12}	6.96
Bottom End Plugs	1.25	6.39×10^{12}	7.99×10^{12}	3.04
Fuel Rod Springs	1.25	7.60×10^{12}	9.51×10^{12}	16.80
Top End Plugs	1.25	1.16×10^{12}	1.45×10^{12}	3.04
Assembly Outlet	1.25	1.09×10^{12}	1.36×10^{12}	8.89

Table 5.2-6

Neutron Source Isotopes and Neutron Production

10 Years Decay Time - 3 Assemblies

Isotope	Curies	Neutrons/Sec
Pu^{238}	3.90×10^3	4.33×10^6
Pu^{239}	4.41×10^2	3.21×10^5
Pu^{240}	7.80×10^2	3.71×10^6
Pu^{242}	3.06	1.35×10^6
Am^{241}	2.60×10^3	2.50×10^6
Am^{243}	3.42×10^1	2.92×10^4
Cm^{242}	8.07	6.33×10^4
Cm^{243}	3.03×10^1	4.05×10^4
Cm^{244}	3.09×10^3	4.27×10^8
Cm^{246}	9.24×10^{-2}	2.68×10^6
	TOTAL	4.42×10^8
Total, w/Subcritical Multiplication		6.31×10^8

is based on the time average axial power shape in the active core region. The axial power shape used in the cask neutron transport analysis is shown in Figure 5.2-1.

The 47 group neutron energy group structure for the cross section data set and the neutron spectrum (neutrons/energy group) used in the cask design is shown in Table 5.2-7 and Table 5.2-8, respectively. The neutron source term identified in Table 5.2-6 was used as a multiplying constant to determine the absolute spectrum. (See Reference 5.2.3 for spectral source information.)

5.3 Model Specification

5.3.1 Description of Radial and Axial Shielding Configuration

The model used to calculate the required cask shield is based on a composite PWR assembly. The composite assembly is based on the following components.

- o Fueled Portion of Fuel Assembly - W 17x17 fuel with 3.0% enriched U-235 fuel.

- o W 15x15 head and foot pieces.

The composite fuel assembly is a worst case for the following standpoints:

- o The mass of UO_2 in the fuel per assembly is higher than any other assemblies except the B&W 15x15, (see Table 5.2-1) and the Alloy 718 content in the fueled region is higher than that of the B&W 15x15.

- o The W fuel assembly structural components, head and foot pieces plus fuel assembly grids are primarily constructed of stainless steel and Alloy 718 which contain cobalt and has a total mass equivalent to that of most other assemblies.

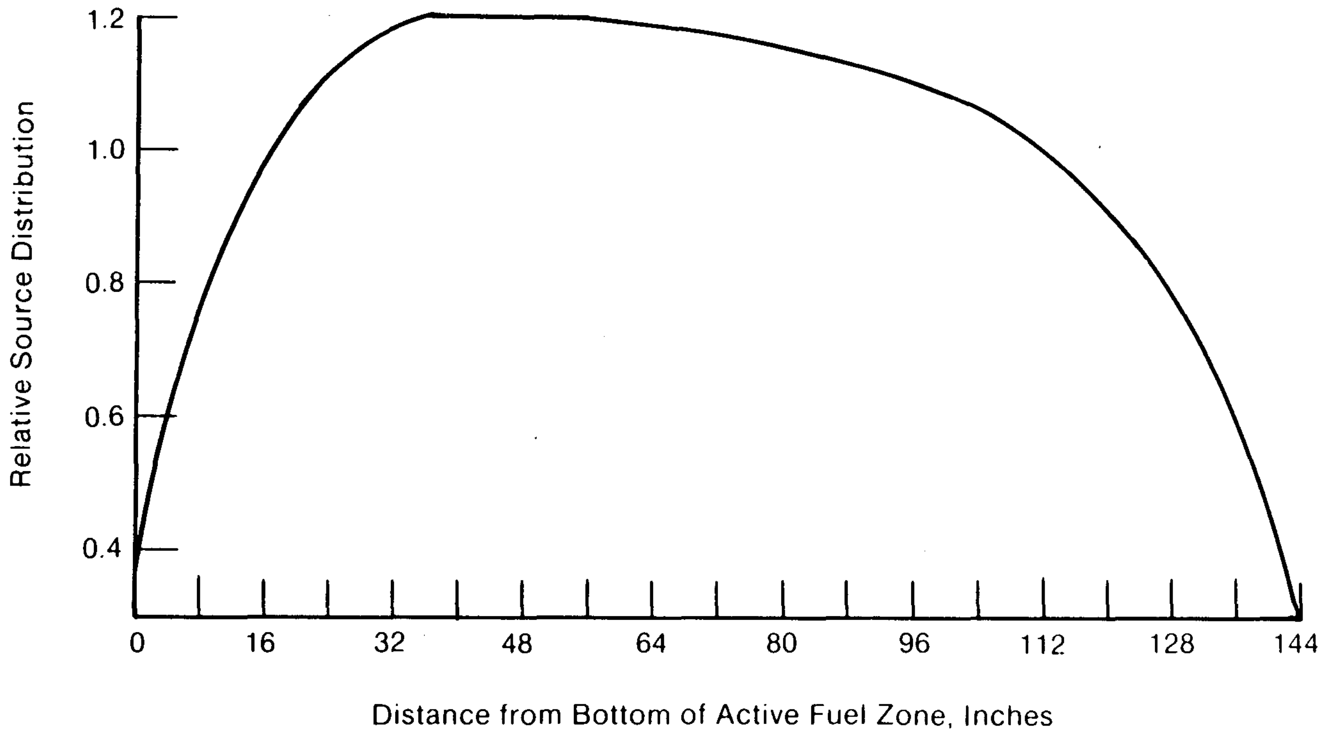


Figure 5.2-1 Axial Source Distribution in Active Fuel Zone of Spent Fuel Array

Table 5.2-7

Multigroup Energy Group Structure for Sailor/Bugle 80 Data Files

Group	Neutron Energy (Mev)*	Group	Gamma Energy (Mev)*
1	1.733E+01	1	14.000
2	1.419E+01	2	10.000
3	1.221E+01	3	8.000
4	1.000E+01	4	7.000
5	8.607E+00	5	6.000
6	7.408E+00	6	5.000
7	6.065E+00	7	4.000
8	4.966E+00	8	3.000
9	3.679E+00	9	2.000
10	3.012E+00	10	1.500
11	2.725E+00	11	1.000
12	2.466E+00	12	0.800
13	2.365E+00	13	0.700
14	2.346E+00	14	0.600
15	2.231E+00	15	0.400
16	1.920E+00	16	0.200
17	1.635E+00	17	0.100
18	1.353E+00	18	0.060
19	1.003E+00	19	0.030
20	8.208E-01	20	0.020
21	7.427E-01		0.010
22	6.081E-01		
23	4.979E-01		
24	3.688E-01		
25	2.972E-01		
26	1.832E-01		
27	1.111E-01		
28	6.738E-02		
29	4.987E-02		
30	3.183E-02		

* Values are upper bound energies for each group.

Table 5.2-7 (Continued)

Multigroup Energy Group Structure for Sailor/Bugle-80 Data Files

Group	Neutron Energy (Mev)*	Group	Gamma Energy (Mev)*
31	2.606E-02		
32	2.418E-02		
33	2.188E-02		
34	1.503E-02		
35	7.102E-03		
36	3.355E-03		
37	1.585E-03		
38	4.540E-04		
39	2.144E-04		
40	1.013E-04		
41	3.727E-05		
42	1.068E-05		
43	5.043E-06		
44	1.855E-06		
45	8.764E-07		
46	4.140E-07		
47	1.000E-07		
	1.000E-11		

* Values are upper bound energies for each group.

Table 5.2-8

Neutron Source Spectrum by Energy Group

Group	Upper Bound Energy of Group (Mev)	Yield Fraction of Group
1	1.733E+01	8.369E-05
2	1.419E+01	2.899E-04
3	1.221E+01	1.456E-03
4	1.000E+01	3.046E-03
5	8.607E+00	6.349E-03
6	7.408E+00	1.699E-02
7	6.065E+00	3.101E-02
8	4.966E+00	7.889E-02
9	3.679E+00	7.327E-02
10	3.012E+00	4.146E-02
11	2.725E+00	4.376E-02
12	2.466E+00	1.875E-02
13	2.365E+00	3.769E-03
14	2.346E+00	2.281E-02
15	2.231E+00	6.922E-02
16	1.920E+00	6.862E-02
17	1.635E+00	8.712E-02
18	1.353E+00	1.151E-01
19	1.003E+00	6.405E-02
20	8.208E-01	2.811E-02
21	7.427E-01	4.874E-02
22	6.081E-01	3.960E-02
23	4.979E-01	4.481E-02
24	3.688E-01	2.352E-02
25	2.972E-01	3.400E-02
26	1.832E-01	1.804E-02
27	1.111E-01	8.900E-03
28	6.738E-02	4.317E-03
29	4.987E-02	1.225E-03
30	3.183E-02	7.019E-04

Table 5.2-8 (Continued)

Neutron Source Spectrum by Energy Group

Group	Upper Bound Energy of Group (Mev)	Yield Fraction of Group
31	2.606E-02	2.139E-04
32	2.418E-02	2.508E-04
33	2.188E-02	6.690E-04
34	1.503E-02	6.016E-04
35	7.102E-03	1.962E-04
36	3.355E-03	3.030E-05
37	1.585E-03	2.600E-05
38	4.540E-04	3.183E-06
39	2.144E-04	1.033E-06
40	1.013E-04	3.858E-07
41	3.727E-05	5.849E-08
42	1.068E-05	3.537E-08
43	5.043E-06	1.148E-08
44	1.855E-06	4.289E-09
45	8.764E-07	8.320E-10
46	4.140E-07	2.700E-10
47	1.000E-07	1.290E-10
	1.000E-11	

To maximize the conservatism in the analysis, the cask model was divided into two models. The cask models interface at cavity midplane to provide a top half model and a bottom half model. This modeling approach allowed positioning of the fuel assembly on the bottom of the cavity to provide conservatism in the bottom cask shield, and positioning the top of the fuel assembly to touch the lower plate of the cavity top shielding, also providing conservatism. In addition, dividing the cask model into two allowed more detailed modeling of the cask design details, otherwise restricted due to limitations on the number of mesh permitted in the DOTIIIW models. The radial activity profile was assumed to be uniform throughout the cavity source region. The fuel physical characteristics are summarized in Table 5.3-1. The cask sources, structure, and shields are modeled as shown on Figures 5.3-1 through 5.3-4. Figure 5.3-5 shows the model of the basket assembly. The dose point locations exterior to the shield are shown on Figures 5.3-1 through 5.3-4.

5.3.2 Shield Regional Densities

The material properties and compositions of all materials used in the shielding evaluation are shown in Table 5.3-2. This table identifies the chemical composition of the materials and the density of the material in its manufactured form. Table 5.3-3 provides the number densities used in the calculation for each of the materials listed in Table 5.3-2 together with the volume fraction of that material in the zone of the fuel/cask where it is incorporated.

The chemical compositions of the materials are based on standard data in References 5.3.1 and 5.3.2.

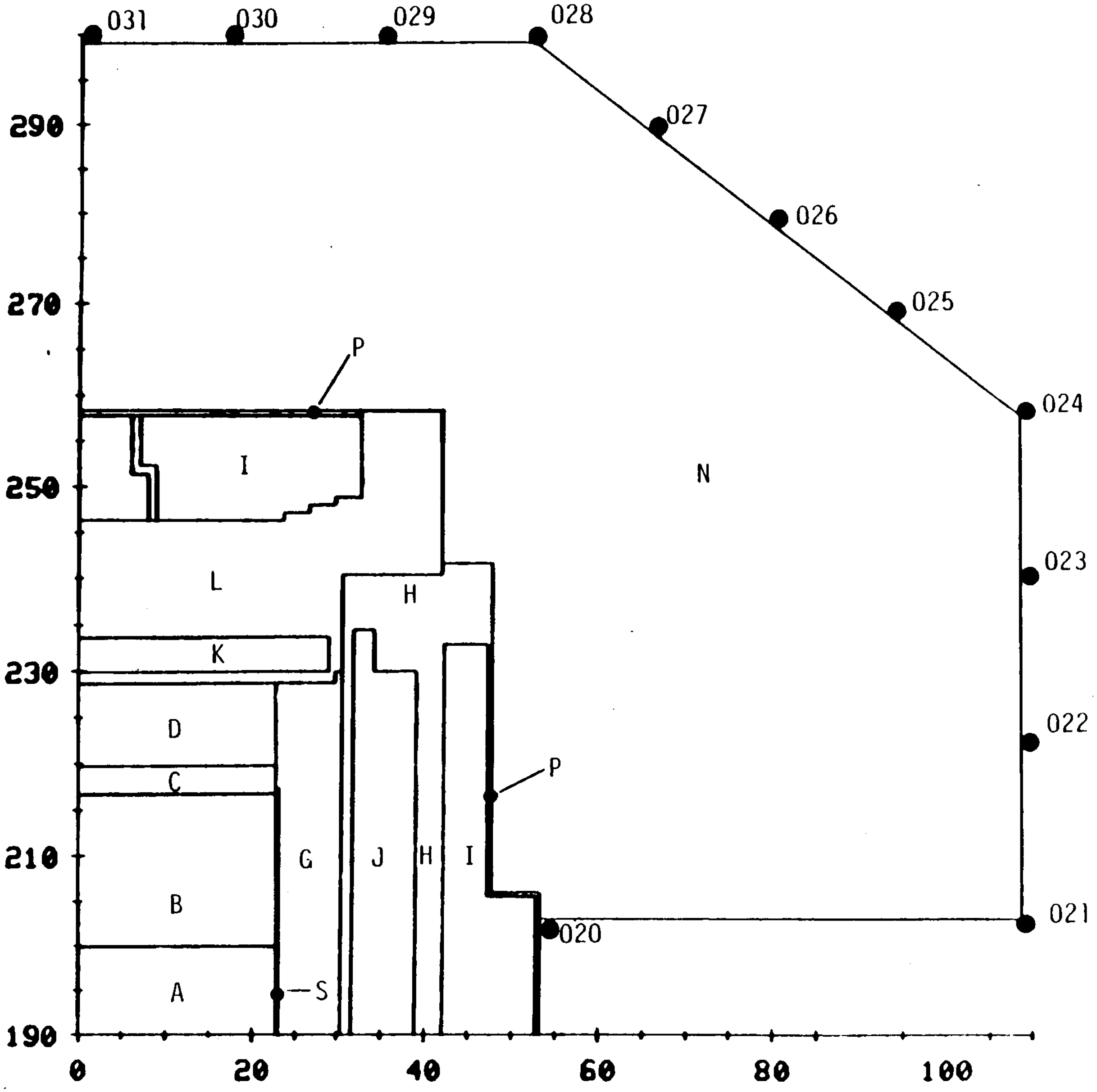
5.4 Shielding Evaluation

5.4.1 Shielding Analytical Methods

The analysis methodology used for determining the shielding requirements of the cask configuration is based on the proven technology of discrete ordinates radiation transport and point kernel integration methods. Analyses performed

Table 5.3-1
Physical Characteristics of PWR Spent Fuel

	English, inches	Metric, cm
SPENT FUEL PARAMETERS:		
Fuel Pellet:		
Theoretical Density	95%	-
U-235 Enrichment	3.0 w/o	-
Diameter	0.3225	0.8192
Fuel Rod:		
Active Fuel Length	144	365.76
Clad Material	Zircaloy-4	-
Clad Thickness	0.0225	0.0572
Diameter	0.374	0.9500
Fuel Assembly:		
Type	W PWR	-
Array	17 x 17	-
Number of Fuel Rods	264	-
Burnup, MWD/MTU	35,000	-
Post Irradiation Time, Years	10	-
Fuel Rod Pitch - Square Array	0.496	1.260
Length	159.25	404.50
Cross Section, Square	8.434	21.4224
Equivalent Radial/Axial Description of Stored Fuel:		
Equivalent Diameter	56.41	143.29
Total Assembly Length	159.25	404.50



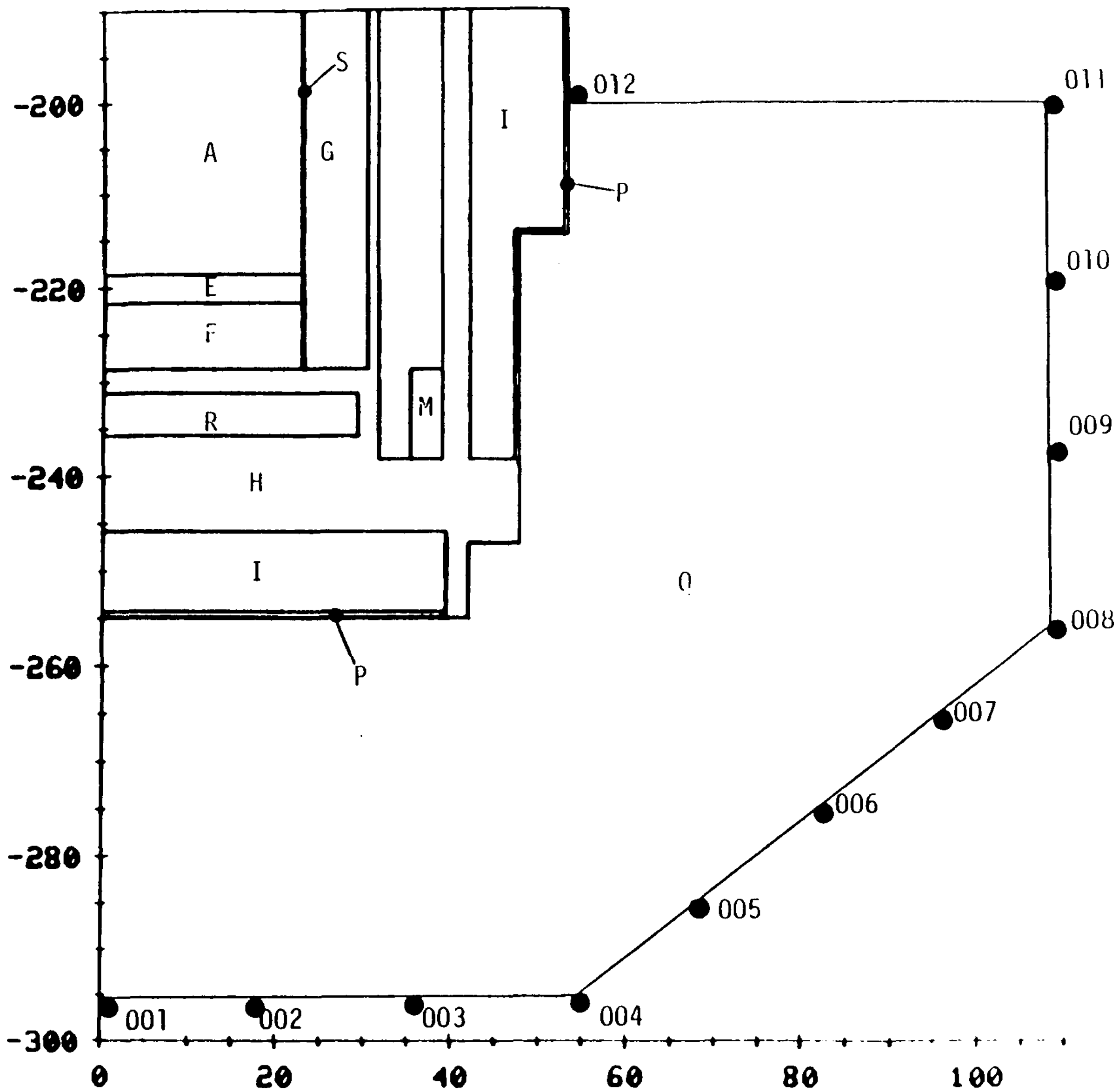
LEGEND

Zone	Description
A	Spent Fuel Array
B	Fuel Rod Springs
C	Top End Plugs
D	Fuel Assembly Outlet
E	Bottom End Plugs
F	Fuel Assembly Inlet
G	Cask Cavity
H	Cask Body
I	Neutron Shield
J	Depleted Uranium (DU)
K	DU Top Cover Plate
L	Top Closure
M	Void
N	Top Limiter
P	Neutron Shield Canning
Q	Bottom Limiter
R	DU Bottom Cover Plate
S	Fuel Assembly Basket

Dose Point Locations

001 - 031		Cask Surface
V01 - V47		2 Meters From Vehicle Surface
D01		Driver

Figure 5.3-1 LWT Cask R-Z Geometry Model Top Half Details With Surface Detector Point Locations



LEGEND

Zone	Description
A	Spent Fuel Array
B	Fuel Rod Springs
C	Top End Plugs
D	Fuel Assembly Outlet
E	Bottom End Plugs
F	Fuel Assembly Inlet
G	Cask Cavity
H	Cask Body
I	Neutron Shield
J	Depleted Uranium (DU)
K	DU Top Cover Plate
L	Top Closure
M	Void
N	Top Limiter
P	Neutron Shield Canning
Q	Bottom Limiter
R	DU Bottom Cover Plate
S	Fuel Assembly Basket

Dose Point Locations

001 - 031	Cask Surface
V01 - V47	2 Meters From Vehicle Surface
D01	Driver

Figure 5.3-2 LWT Cask R-Z Geometry Model Bottom Half Details with Surface Detector Point Locations

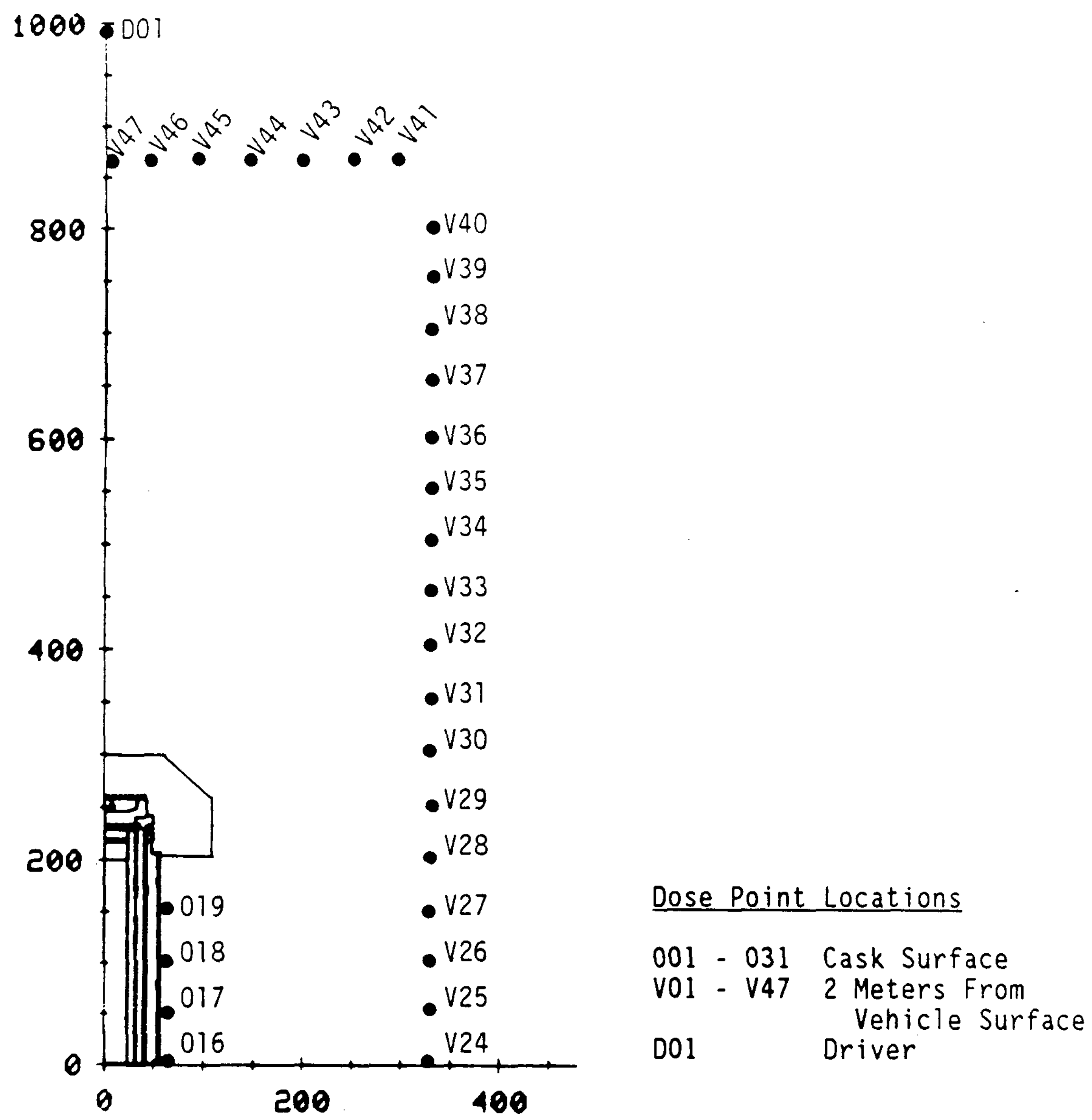


Figure 5.3-3 LWT Cask R-Z Geometry Model Top Half With Off-Surface Detector Point Locations

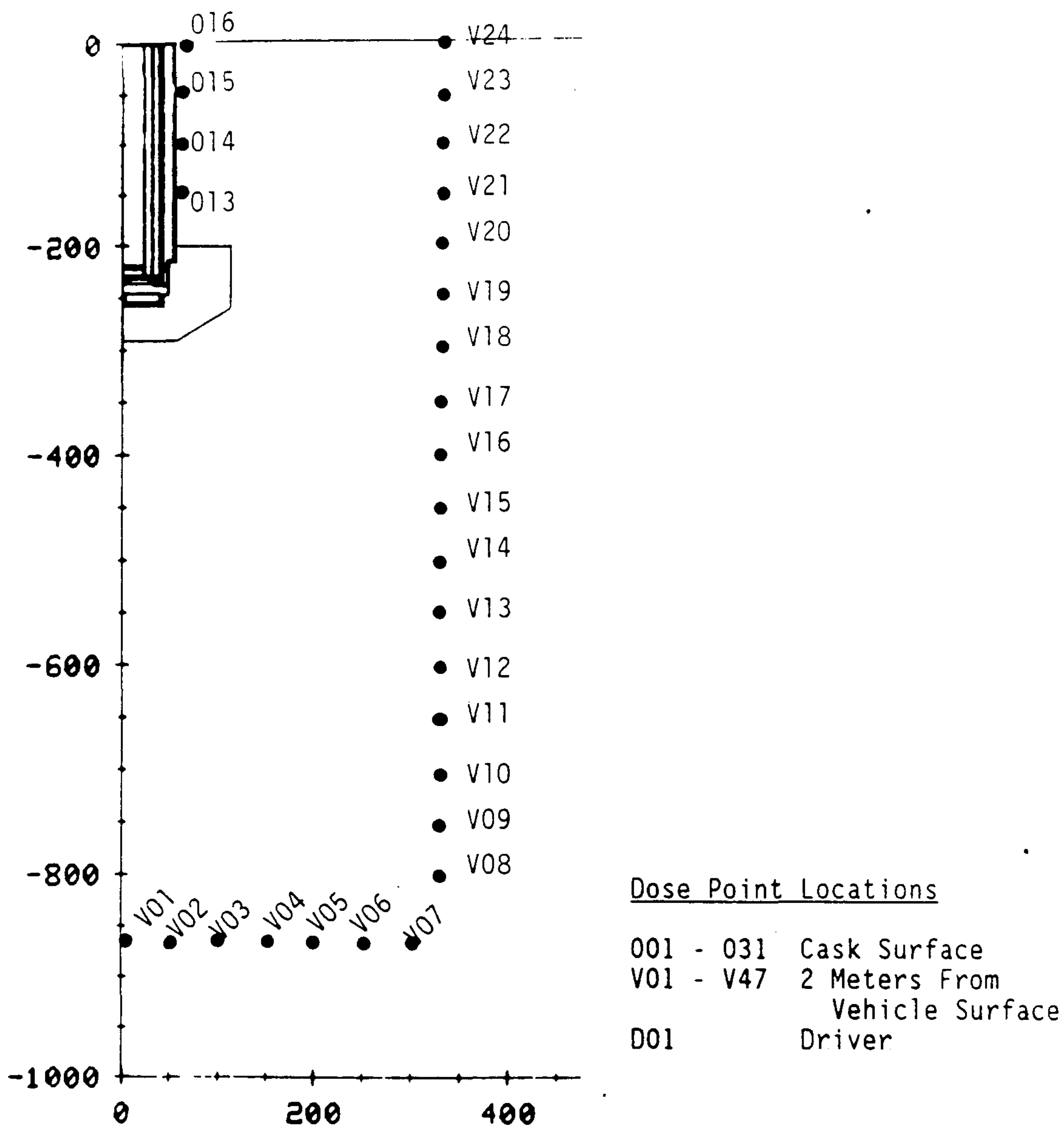


Figure 5.3-4 LWT Cask R-Z Geometry Model Bottom Half With Off-Surface Detector Point Locations

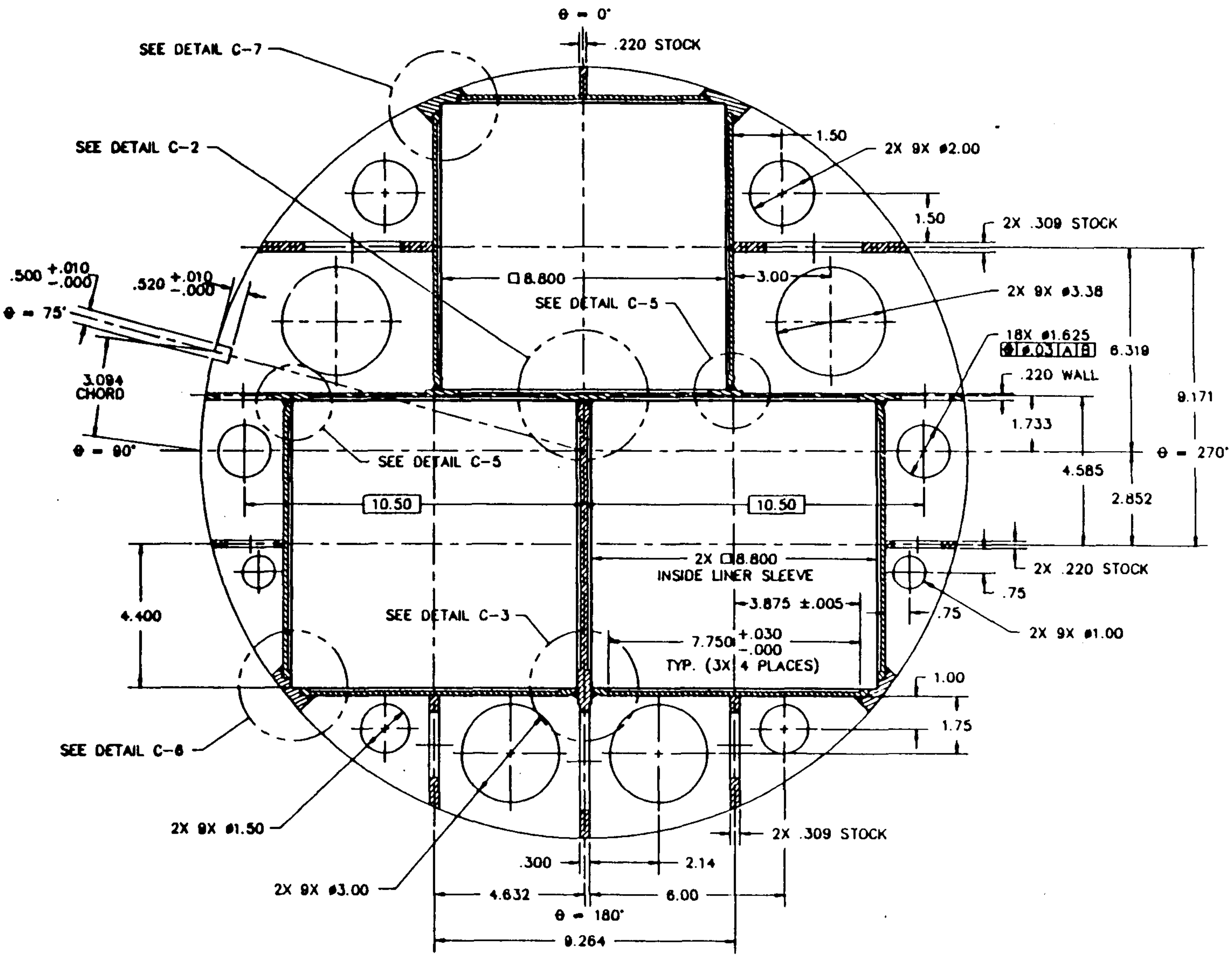


Figure 5.3-5 Fuel Basket Assembly Arrangement

Table 5.3-2
Material Specifications and Properties

Component/Material Specification	Density gm/cc	Composition, w/o			
Boral™ (Core and Clad)	2.63 (0.025 gm/cm^2 B-10 in core)	4.11	B^{10}	5.72	C
		16.52	B^{11}	73.65	Al
UO_2 Fuel Pellet (3% Enrichment for Shield Evaluation)	10.355	2.644	U-235		
		85.5	U-238		
		11.856	O		
Zircaloy-4 Fuel Cladding	6.56	0.10	Cr		
		0.21	Fe		
		1.30	Sn		
		98.39	Zr		
Cask Body, Shield Shells Ti Alloy - ASTM Grade 2	4.484	94.25	Ti		
		3.00	Al		
		2.50	V		
		0.15	Fe		
		0.09	O		
		0.008	C		
		0.002	N		

Table 5.3-2 (Continued)
Material Specifications and Properties

Component/Material Specification	Density gm/cc	Composition, w/o			
Basket, Fuel Structures, Stainless Steel Type 304	8.03	0.08	C	0.10	N
		2.00	Mn	68.495	Fe
		0.045	P	19.00	Cr
		0.030	S	9.25	Ni
		1.00	Si		
Principal Gamma Shield, Depleted Uranium	18.7	0.22	U^{235}		
		99.78	U^{238}		
Principal Neutron Shield, Boro-Silicone*	1.59	4.87	H	1.68	Si
		1.06	B	0.53	Na
		57.19	O	0.52	Mg
		24.65	Al	0.28	Fe
		9.03	Ca	0.19	S

* Boro-Silicone is a trademark of Reactor Experiments, Inc..

Table 5.3-3
Material Atom Densities and Volume Fractions

Material Description	Material Composition	Volume Fraction* of Zone	Number Density Atoms/cc x 10^{-24}
Fuel Assemblies	UO_2	0.2559	
	U^{235}		7.01×10^{-4}
	U^{238}		2.24×10^{-2}
	O		4.62×10^{-2}
Cask Structure,	Ti Alloy	1.0000	
Shield Shells	Ti		5.32×10^{-2}
	Al		3.00×10^{-3}
	V		1.32×10^{-3}
	Fe		7.22×10^{-5}
	O		1.51×10^{-4}
	C		1.80×10^{-5}
	N		3.87×10^{-6}

* Volume fraction is that fraction of cask zone occupied by the described material. Atom density is based on the densities and weight percent given in Table 5.2-1.

Table 5.3-3 (Continued)
Material Atom Densities and Volume Fractions

Material Description	Material Composition	Volume Fraction* of Zone	Number Density Atoms/cc x 10^{-24}
Fuel Clad	Zircaloy-4	0.0967	
		Zr	4.26×10^{-2}
		Sn	4.33×10^{-4}
		Fe	1.48×10^{-4}
		Cr	7.60×10^{-5}
	Boral	0.0372	(Includes core and clad)
		B-10	7.14×10^{-3}
		B-11	2.87×10^{-2}
		C	8.96×10^{-3}
		Al	3.29×10^{-2}
Basket	Stainless	1.0	
	Steel	Fe	5.93×10^{-2}
		Cr	1.77×10^{-2}
		Ni	7.62×10^{-3}
		Mn	1.76×10^{-3}
		Si	1.72×10^{-3}
		N	3.45×10^{-4}
		C	3.22×10^{-4}
		P	7.03×10^{-5}
		S	4.53×10^{-5}

Table 5.3-3 (Continued)

Material Atom Densities and Volume Fractions

Material Description	Material Composition	Volume Fraction* of Zone	Number Density Atoms/cc x 10^{-24}
Gamma Shield	Depleted Uranium	1.0	
		U-235	1.05×10^{-4}
		U-238	4.72×10^{-2}
Neutron Shield	Boro-Silicone	1.0	
		H	4.63×10^{-2}
		B	9.41×10^{-4}
		O	3.42×10^{-2}
		Al	8.75×10^{-3}
		Ca	2.16×10^{-3}
		Si	5.72×10^{-4}
		Na	2.23×10^{-4}
		Mg	2.06×10^{-4}
		Fe	4.74×10^{-5}
		S	5.63×10^{-5}
Bottom Nozzles	SS304	0.180	
Bottom End Plugs	Zircaloy-4 SS304	0.0503 0.195	For Zircaloy-4 and SS304 Atom Densities see Fuel Clad and Basic Structure Descriptions of this Table
Springs	Zircaloy-4 SS304/Inconel	0.0808 0.061	
Top End Plugs	Zircaloy-4 SS304	0.0256 0.195	--
Top Nozzle	Zircaloy-4 SS304	0.00086 0.2765	--

to support the selection of shielding configurations (i.e., arrangement and dimensions) included the determination of neutron and gamma sources for the types of fuel assemblies to be stored in the cask and the evaluation of the neutron and photon transport in the shielding configurations. An overall schematic of the analysis methodology used in the study is shown in Figures 5.4-1 and 5.4-2. As shown, the methodology included the following:

- o Development of neutron and photon cross section data for the materials.

- o Prediction of the shielding performance of the shielding configurations to meet design constraints on dose rates.

As shown in Figure 5.4-1, the analysis performed in the study included the studies required to develop a conceptual design of a cask to meet the design requirements. Shown in Figure 5.4-2, is the analysis methodology required to evaluate design details which is similar to the conceptual design method except that the analysis of the effect of heterogeneities in the shield configuration (shield region interfaces) required two-dimensional radiation transport methods. The following discussion provides a description of the nuclear data files and computer programs used in the study.

<u>Neutron and Gamma Ray Cross Sections</u>

Nuclear data files used in the analysis of cask shield requirements were from two data sources. The shield design analyses were performed using the following data files.

SAILOR (REFERENCE 5.2.3)

A 47 neutron, 20 gamma-ray multigroup cross section library developed for neutron and gamma-ray transport studies of LWR reactor systems. This library is being used extensively for studies of LWR reactor vessel neutron irradiation studies. This library was used to determine cask shielding

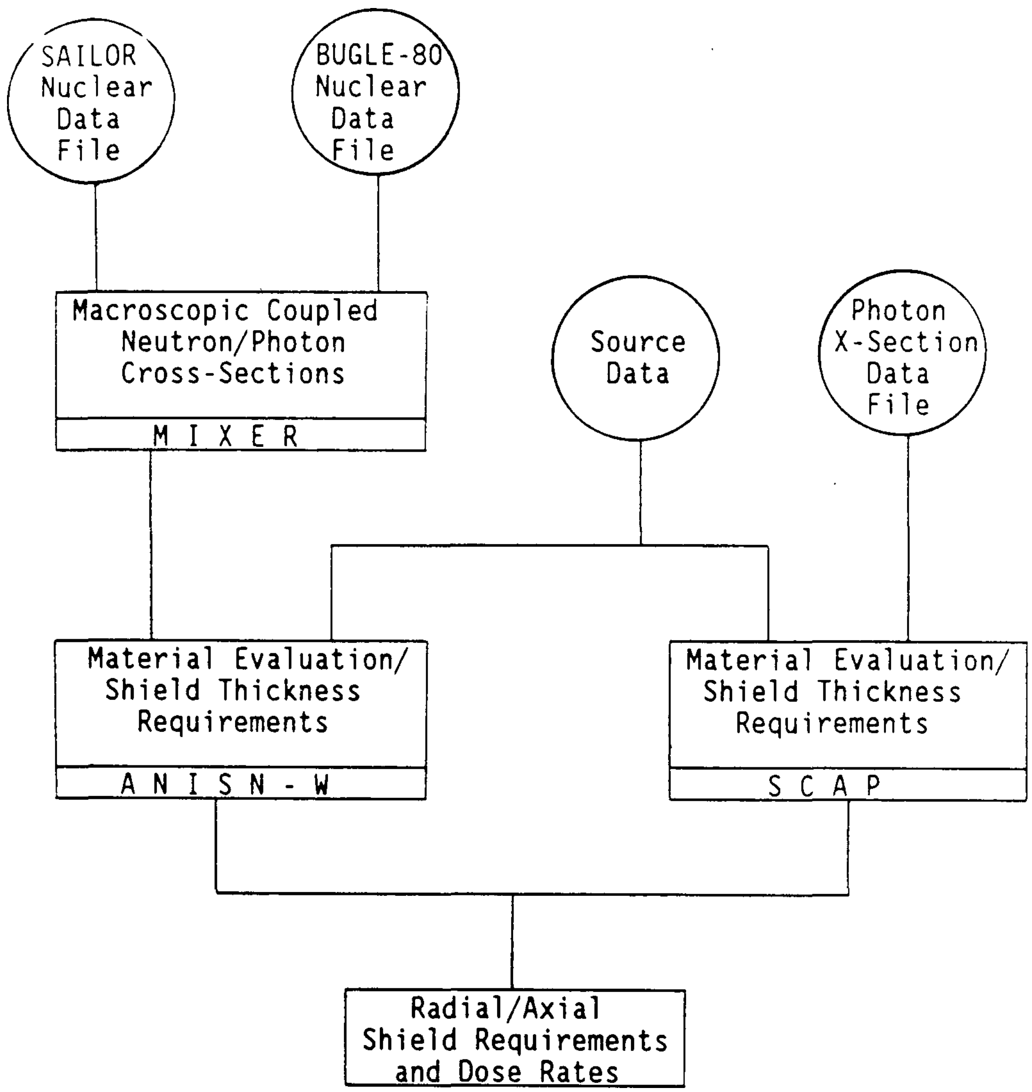


Figure 5.4-1 Analysis Methodology for LWT Cask for Conceptual Design Analysis

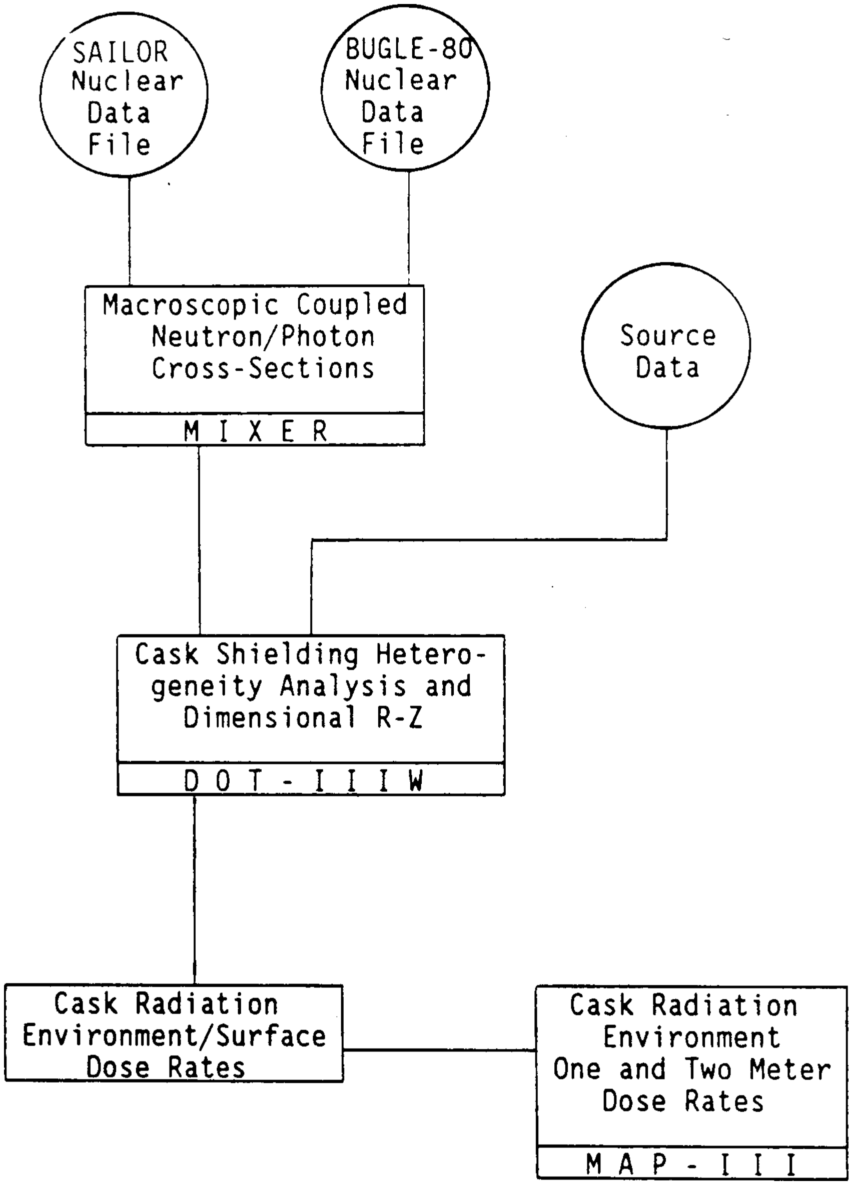


Figure 5.4-2 Analysis Methodology for LWT Cask for Design Details Analysis

performance in one- and two-dimensional model analyses. This library is derived with current state-of-the-art cross section preparation techniques and includes isotopic self-shielding corrections in the neutron cross section preparation.

BUGLE-80 (REFERENCE 5.4.1)

A 47 neutron, 20 gamma-ray, P_3 order of scattering, multigroup cross section library developed for general use in LWR and shipping cask neutron and gamma ray transport analysis. This set is similar to the SAILOR library and predates the SAILOR file. This library was developed by the ANS 6.1.2 Working Group on multigroup cross sections and is incorporated in an ANS standard. BUGLE-80 was used in conjunction with the SAILOR library to conduct analyses of cask shielding performance.

Each of the above cross section data files include multigroup cross section data in formats consistent with the ANISN-W and DOTIIIW discrete ordinates transport methods. The energy group structure of the SAILOR and BUGLE-80 data sets is listed in Table 5.2-7. The anisotropic scattering of neutrons and gamma rays in the multigroup format are approximated by a Legendre expansion of the scattering cross sections. All the above data sets are derived from the ENDF/B-IV data file and include the gamma ray production cross sections due to neutron interactions (e.g., neutron radiative capture, neutron inelastic scatter). Use of coupled multigroup neutron and gamma ray cross section data in ANISN-W or DOTIIIW provides predictions of the neutron flux and dose rates as well as the gamma ray dose rates due to neutron interactions with the materials of the fuel assembly array and cask assembly. As shown in Figures 5.4-1 and 5.4-2, the MIXER computer program is used to prepare macroscopic neutron and gamma ray cross section data from the microscopic library data files described above. MIXER accepts isotopic atom densities as input and prepares multigroup P_3 cross section files for use in the ANISN-W or DOTIIIW computer program.

The third nuclear data library used in the conceptual design analysis was the compilation of gamma ray interaction cross sections as a function of gamma ray energy. This data file is incorporated in the SCAP-II computer program (Reference 5.4.2) and consists of the elemental pair production and photoelectric cross sections. The SCAP program uses this information in combination with the Klein-Nishina equations for inelastic scattering of gamma rays with a free electron to calculate the total gamma ray cross section (linear absorption coefficient) at the source energies in the SCAP problem.

Figure 5.4-1 shows the radiation transport methods used to solve for the neutron and gamma ray attenuation and dose rates in the conceptual design configurations of the cask. The ANISN-W and SCAP-II computer programs were used to define the final design configuration of the cask for more detailed two dimensional analyses. Each of these computer programs used in the conceptual design and final design analyses is described in the following paragraphs.

ANISN-W (Reference 5.4.3) solves the one-dimensional Boltzman transport equation with general anisotropic scattering in either cylindrical geometry or slab geometry. The method of discrete ordinates is used to solve for the angular dependence of the neutron or gamma ray flux with anisotropic scattering treated as a Legendre expansion of the scattering cross section. The energy dependence of particles (neutrons or gamma rays) is treated by a multigroup approach with the individual group cross sections obtained by group averaging methods.

SCAP-II (Reference 5.4.2) uses the point kernel integration method to calculate the gamma ray attenuation and dose rate at specified dose points in the cask shielding configurations. The SCAP program emulates the QAD series of programs and incorporates improvements to increase the geometry capability, input preparation, and the use of a data file of gamma ray cross sections consistent with the discrete ordinates methods nuclear data file described previously.

DOTIIIW is an enhancement of the DOTIIW computer program (Reference 5.4.4) to incorporate improved data management and computer programming on the CRAY-1S mainframe computer. The DOT series of programs solves the two-dimensional Boltzman transport equation with general anisotropic scattering. Geometry capabilities are the R-Z, R-O, and X-Y geometries. The method of discrete ordinates is used to solve for the angular dependence of the neutron or gamma ray flux with anisotropic scattering treated as a Legendre expansion of the scattering cross section. The energy dependence of particles (neutrons or gamma rays) is treated by a multigroup approach with the individual group cross sections obtained by group averaging methods. An S_6, 30 angles, angular quadrature set was used for the preliminary analysis. Consideration will be given to using P_8 for the final design.

MAP-III (Reference 5.4.5) solves for the neutron and gamma transport in attenuating media or void external to the DOTIIIW R-Z geometry model. MAP-III performs a numerical integration of the multigroup angular dependent neutron or gamma ray leakage from the surface of a DOTIIIW geometry to predict neutron or gamma ray flux and dose rate at detector points external to the DOTIIIW geometry. MAP-III extends the capability of the discrete ordinates method.

Neutron and gamma ray flux-to-dose conversion factors used in ANISN-W, DOTIIIW and SCAP are based on the ANSI standard, ANSI/ANS-6.1.1-1977 (N666), as documented in Reference 5.4.6. The neutron and gamma ray flux-to-dose rate conversion factors used in ANISN-W and DOTIIIW are listed in Table 5.4-1.

5.4.2 Shielding Analytical Models

The LWT cask design model is illustrated in Figures 5.3-1 to 5.3-4. Analytical models of the cask were developed in the two dimensional R-Z geometry capability of the DOTIIIW computer program and in the one-dimensional cylindrical capability of the ANISN-W computer program. In all the geometry models, the spent fuel array was modeled as a homogeneous medium. The spent fuel array consists of: 1) the spent fuel rods and fuel assembly structure of intact fuel, and 2) the stainless steel basket for the 3 fuel assemblies.

Table 5.4-1

Multigroup Neutron and Gamma Ray Flux-to-Dose Rate Conversion Factors - Sailor/Bugle-80 Energy Groups

Group	Neutron*	Group	Gamma-Ray**
1	2.143E-01	1	1.102E-02
2	1.957E-01	2	8.771E-03
3	1.634E-01	3	7.662E-03
4	1.471E-01	4	6.926E-03
5	1.471E-01	5	6.191E-03
6	1.482E-01	6	5.417E-03
7	1.536E-01	7	4.627E-03
8	1.487E-01	8	3.721E-03
9	1.317E-01	9	2.930E-03
10	1.306E-01	10	2.320E-03
11	1.265E-01	11	1.834E-03
12	1.252E-01	12	1.604E-03
13	1.254E-01	13	1.442E-03
14	1.256E-01	14	1.153E-03
15	1.263E-01	15	7.587E-04
16	1.274E-01	16	3.793E-04
17	1.286E-01	17	2.607E-04
18	1.304E-01	18	3.171E-04
19	1.254E-01	19	8.002E-04
20	1.161E-01	20	1.952E-03
21	1.977E-01		
22	9.728E-02		
23	8.086E-02		
24	6.399E-02		
25	4.703E-02		
26	3.022E-02		
27	1.957E-02		
28	1.322E-02		
29	8.200E-03		
30	8.200E-03		

* Units of mrem/hour per n/cm^2-second.

** Units of mrem/hour per γ/cm^2-second.

Table 5.4-1 (Continued)

Multigroup Neutron and Gamma Ray Flux-to-Dose Rate Conversion Factors - Sailor/Bugle-80 Energy Groups

Group	Neutron*	Group	Gamma-Ray**
31	7.353E-03		
32	6.867E-03		
33	5.723E-03		
34	3.732E-03		
35	3.575E-03		
36	3.642E-03		
37	3.772E-03		
38	3.995E-03		
39	4.100E-03		
40	4.265E-03		
41	4.450E-03		
42	4.559E-03		
43	4.575E-03		
44	4.499E-03		
45	4.352E-03		
46	4.022E-03		
47	3.675E-03		
	1.000E-11		

* Units of mrem/hour per n/cm^2-second.

** Units of mrem/hour per γ/cm^2-second.

A cross sectional view of the basket is shown in Figure 5.3-5. The basket is a stainless steel box structure which has Boral^{TM} plates attached on the sides of the cell. The homogenized spent fuel array used in the one and two dimensional models was developed on the basis of a 23.28 cm (9.167 inch) center-to-center spacing for the storage cell with the unit cell as illustrated on Figure 5.3-4. The volume fractions used in defining the unit cell are listed in Table 5.3-3. Atom densities used in generating the multigroup cross sections for the spent fuel array materials and the cask materials are listed in Table 5.3-3.

The volume fractions were determined on a per cell basis. The cell boundary is defined as the mid-point of the grid basket. The calculation is then defined as follows:

$$\text{Volume Fraction} = \frac{\text{Cross sectional area of component}}{\text{Cross sectional area of cell}}$$

To adequately model the geometry details of the LWT cask, while staying within the mesh limitations of the DOTIIIW computer program, the cask model was divided into two models at the elevation of the cavity midplane. Figures 5.1-1 and 5.1-3 show the empty cask models for the top and bottom halves, respectively. Figures 5.3-1 and 5.3-2 illustrate the details of the top and bottom models which include the cylindrical homogeneous spent fuel zone, the fuel assembly structures above and below the fuel zone, the cask cavity, the cask body, the neutron shielding, the neutron shield shell, and the top closure configuration. The various fuel zones shown on Figure 5.3-1 are: Zone A which is the active fuel region, Zone B which is the gas plenum region, and Zones C and D, which represent the remainder of the head piece regions. Figure 5.3-2 illustrates the details of the bottom cask model, including Zone A which is the active fuel region and Zones E and F which represent the foot piece regions. As shown, the two dimensional models include the interfaces between the cask body and shield covers. The outlines of the top and bottom impact limiters are included in Figures 5.3-1 through 5.3-4 to illustrate surface detector point locations. The analytical models illustrated in Figures 5.3-1 and 5.3-2 were used in performing either neutron or gamma ray transport analysis of the cask design. The gamma ray transport analysis of the primary gamma ray source term in the spent fuel array required a separate DOTIIIW analysis to distinguish primary gamma dose rate contributions from secondary gamma contributions. A total of four DOTIIIW analyses were required: 1) primary gamma for the top half cask geometry, 2) primary gamma for the bottom half cask geometry, 3) a neutron case for the top half of the geometry, and 4) a neutron case for the bottom half of the cask. The secondary gammas were not analyzed in a coupled neutron/secondary gamma analysis since previous cask analyses showed the secondary gamma contribution to be minimal. To provide conservatism in the analyses of the top and bottom shields, it was assumed in the case of the bottom half of the cask that the fuel assemblies were sitting on the bottom of the cavity. In the case of the upper shields, the fuel assemblies were assumed to be touching the bottom plate at the top of the cavity.

The neutron, secondary gamma, and primary gamma dose rates were estimated using the MAP-III code at detector points : 1) on the surface of the impact limiters, 2) at 2 meters from the surface of the vehicle, and 3) at the driver position. MAP-III integrates the multigroup angular dependent neutron and gamma ray surface leakage to predict dose rates at these locations. Secondary gamma results from the prior analysis have been included in the total dose rates provided in Tables 5.4-2 and 5.4-3.

5.4.3 Radiation Dose Rate Results

Primary gamma dose rate results are illustrated in Figures 5.4-3 through 5.4-10. Radial traverses of primary gamma dose rates on the top and bottom cask surfaces are shown in Figures 5.4-3 and 5.4-4. Figures 5.4-5 and 5.4-6 provide axial traverses of the side surface primary gamma dose rate at three different radii. Each side surface axial traverse is appropriate for a limited elevation range. Identification of each of the three side surfaces is illustrated on the cask geometries shown in Figures 5.4-8 and 5.4-10. Isocontour plots of the primary gamma dose rate for the top and bottom halves of the DOTIIIW cask geometry are shown in Figures 5.4-7 through 5.4-10. Figure 5.4-7 shows isocontours for the cask top half model, with an enlargement of the top half details provided in Figure 5.4-8. Similarly, cask bottom half isocontour plots are provided in Figures 5.4-9 and 5.4-10. Radially, the top and bottom surface primary gamma dose rates peak away from

Table 5.4-2
Summary of Dose Rates - Cask Surface
(mrem/hr)

Dose Point Location*	Primary Gamma	Neutron	Secondary Gamma	Total
001	127.149	2.608	0.116	129.873
002	158.052	2.456	0.111	160.618
003	100.250	2.017	0.083	102.350
004	100.744	2.330	0.056	103.130
005	36.323	1.987	0.065	38.374
006	37.337	2.950	0.082	40.369
007	16.104	2.675	0.115	18.894
008	72.743	3.038	0.141	75.921
009	56.039	3.349	0.200	59.588
010	27.625	3.721	0.266	31.612
011	23.750	4.141	0.342	28.233
012	26.260	8.550	0.956	35.766
013	34.283	13.914	1.531	49.728
014	36.572	15.207	1.721	53.500
015	36.152	15.184	1.730	53.066
016	37.578	16.281	1.756	55.615
017	36.255	15.743	1.694	53.692
018	32.654	14.008	1.521	48.183
019	23.960	10.127	1.186	35.273
020	23.377	4.086	0.493	27.956
021	13.053	2.511	0.222	15.786
022	16.780	2.413	0.165	19.358
023	20.117	1.854	0.118	22.088
024	29.931	1.910	0.083	31.924
025	18.749	1.573	0.064	20.386

*See Figures 5.3-1 through 5.3-4 for location of dose points.

Table 5.4-2 (Continued)
Summary of Dose Rates - Cask Surface
(mrem/hr)

Dose Point Location*	Primary Gamma	Neutron	Secondary Gamma	Total
026	76.731	1.884	0.046	78.661
027	24.591	1.185	0.031	25.807
028	52.031	1.277	0.026	53.333
029	62.572	0.720	0.040	63.332
030	107.811	0.868	0.056	108.735
031	89.333	0.776	0.058	90.166

*See Figures 5.3-3 and 5.3-4 for location of dose points.

Table 5.4-3

Summary of Dose Rates - 2 Meters from Vehicle and at Driver Position

(mrem/hr)

Dose Point Location*	Primary Gamma	Neutron	Secondary Gamma	Total
V01	1.485	0.027	0.001	1.513
V02	2.061	0.033	0.001	2.095
V03	2.396	0.048	0.002	2.446
V04	2.343	0.055	0.002	2.401
V05	2.205	0.058	0.003	2.266
V06	2.050	0.059	0.004	2.113
V07	1.999	0.060	0.004	2.063
V08	2.070	0.069	0.005	2.144
V09	1.691	0.073	0.006	1.769
V10	1.549	0.085	0.007	1.641
V11	1.535	0.109	0.009	1.653
V12	1.715	0.145	0.011	1.871
V13	2.037	0.194	0.015	2.246
V14	2.382	0.253	0.020	2.655
V15	2.473	0.318	0.027	2.818
V16	1.803	0.358	0.036	2.197
V17	5.048	0.511	0.046	5.605
V18	6.151	0.698	0.061	6.910
V19	6.900	0.894	0.079	7.873
V20	4.026	1.019	0.095	5.141
V21	4.539	1.203	0.112	5.854
V22	4.919	1.361	0.125	6.405
V23	4.576	1.431	0.132	6.140
V24	4.579	1.436	0.133	6.149
V25	4.307	1.362	0.128	5.796

*See Figures 5.3-3 and 5.3-4 for location of dose points.

Table 5.4-3 (Continued)

Summary of Dose Rates - 2 Meters from Vehicle and at Driver Position

(mrem/hr)

Dose Point Location*	Primary Gamma	Neutron	Secondary Gamma	Total
V26	4.207	1.244	0.117	5.568
V27	3.734	1.074	0.102	4.910
V28	3.075	0.876	0.086	4.037
V29	3.500	0.718	0.068	4.286
V30	3.223	0.546	0.052	3.821
V31	2.611	0.401	0.039	3.052
V32	1.511	0.284	0.030	1.825
V33	3.088	0.240	0.022	3.351
V34	2.833	0.184	0.017	3.034
V35	2.349	0.139	0.012	2.501
V36	1.939	0.104	0.009	2.052
V37	1.416	0.078	0.007	1.501
V38	1.129	0.061	0.006	1.196
V39	0.958	0.051	0.005	1.014
V40	1.037	0.045	0.004	1.087
V41	1.125	0.039	0.003	1.167
V42	1.200	0.036	0.003	1.240
V43	1.288	0.034	0.002	1.324
V44	1.372	0.031	0.002	1.404
V45	1.373	0.024	0.001	1.399
V46	1.220	0.014	0.001	1.234
V47	1.046	0.011	0.000	1.058
D01	0.696	0.007	0.000	0.704

*See Figures 5.3-3 and 5.3-4 for location of dose points.

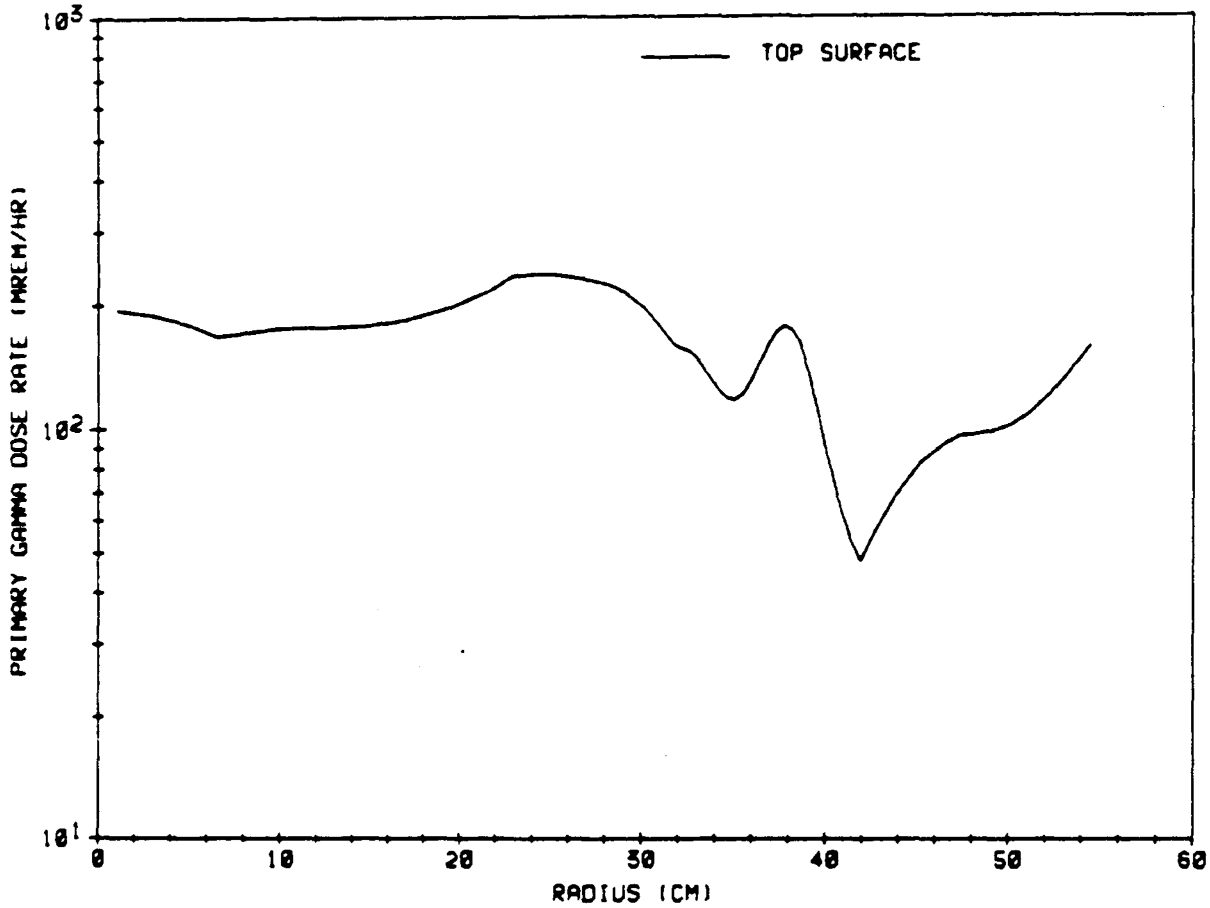


Figure 5.4-3 External Surface Primary Gamma Dose Rates - Top Surface

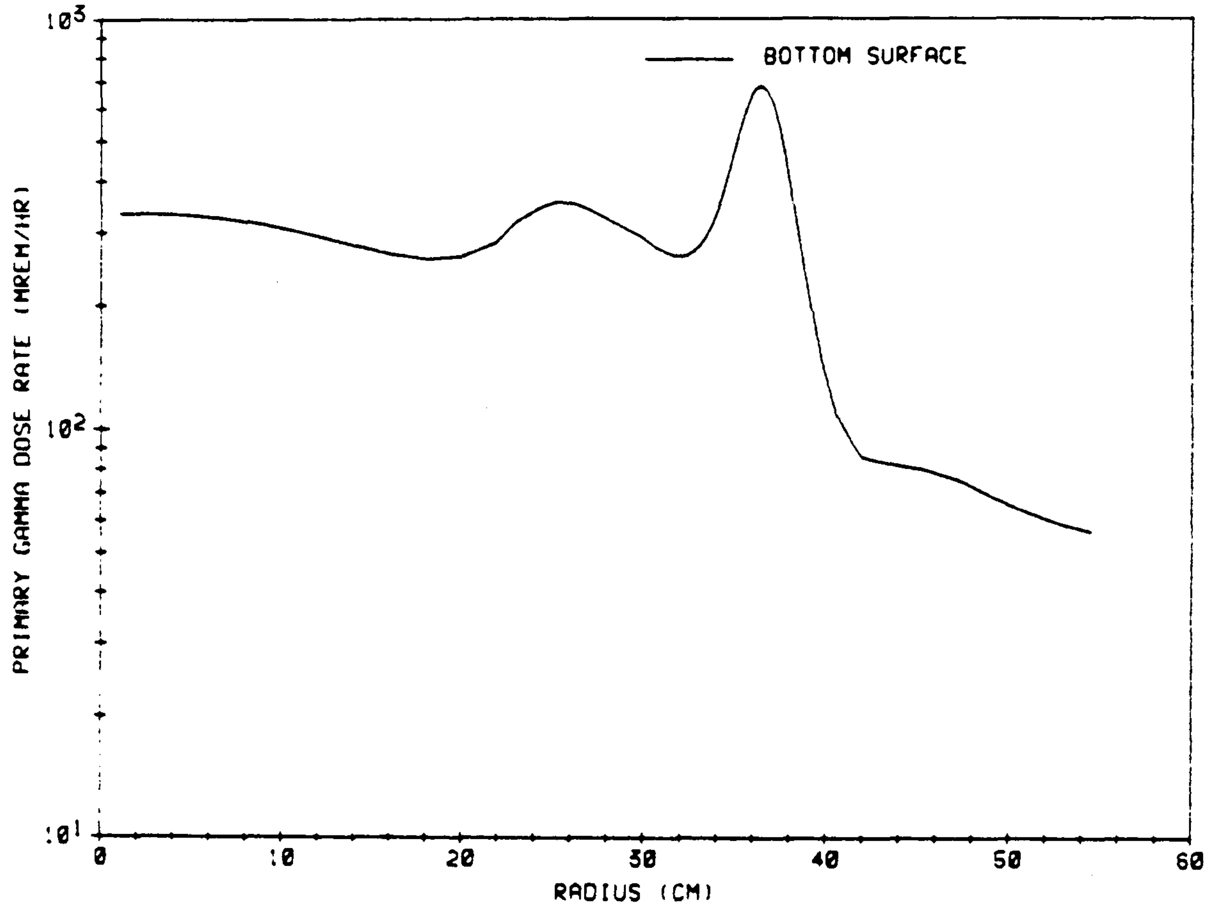


Figure 5.4-4 External Surface Primary Gamma Dose Rates - Bottom Surface

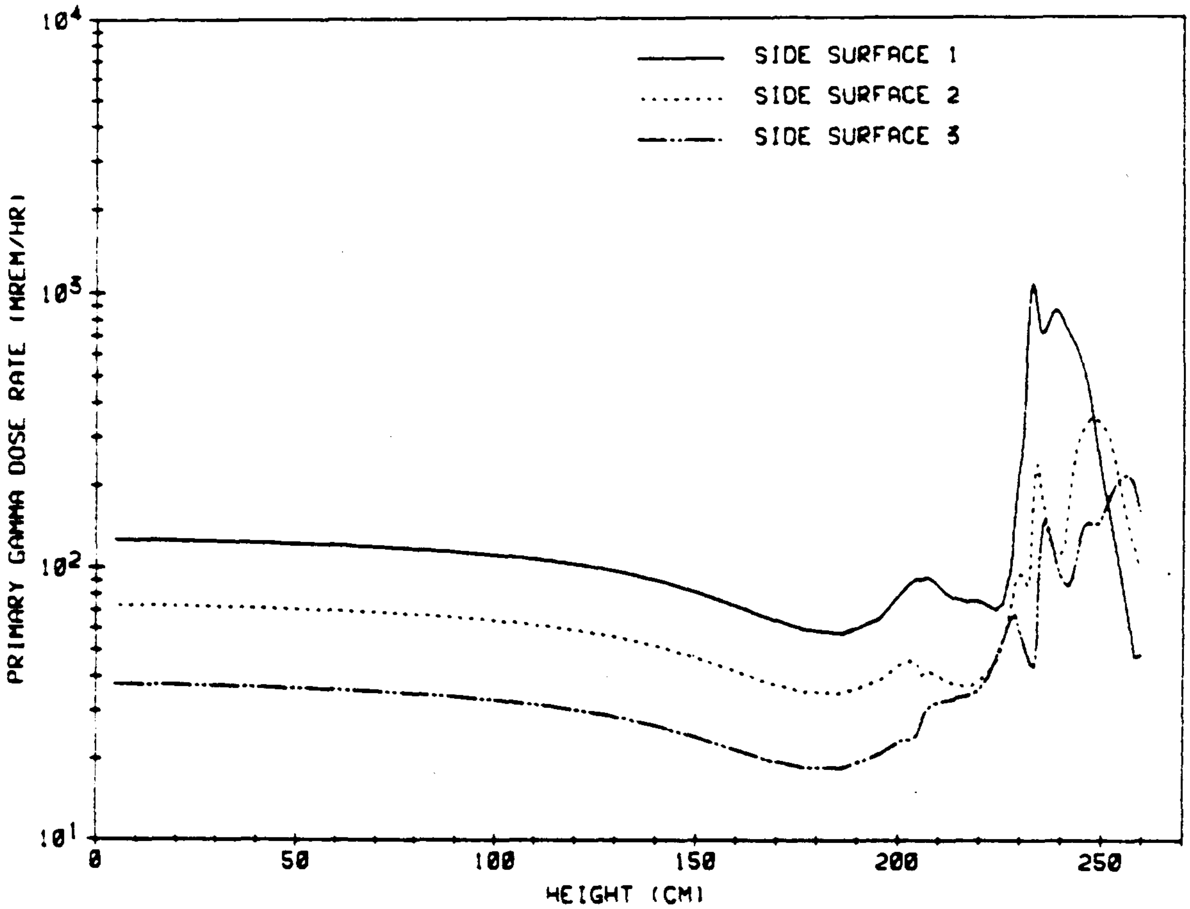


Figure 5.4-5 External Surface Primary Gamma Dose Rates - Side Surface Top Half

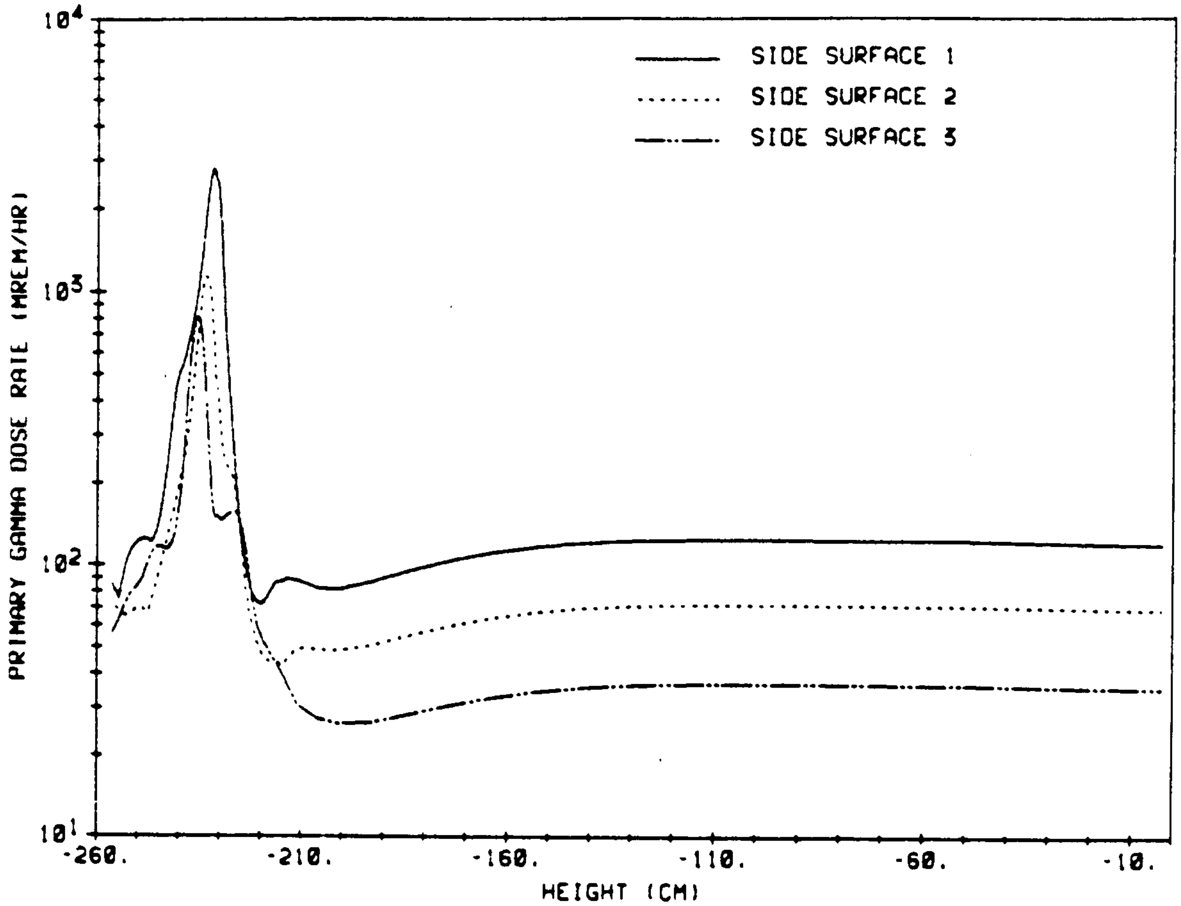


Figure 5.4-6 External Surface Primary Gamma Dose Rates -
Side Surface Bottom Half

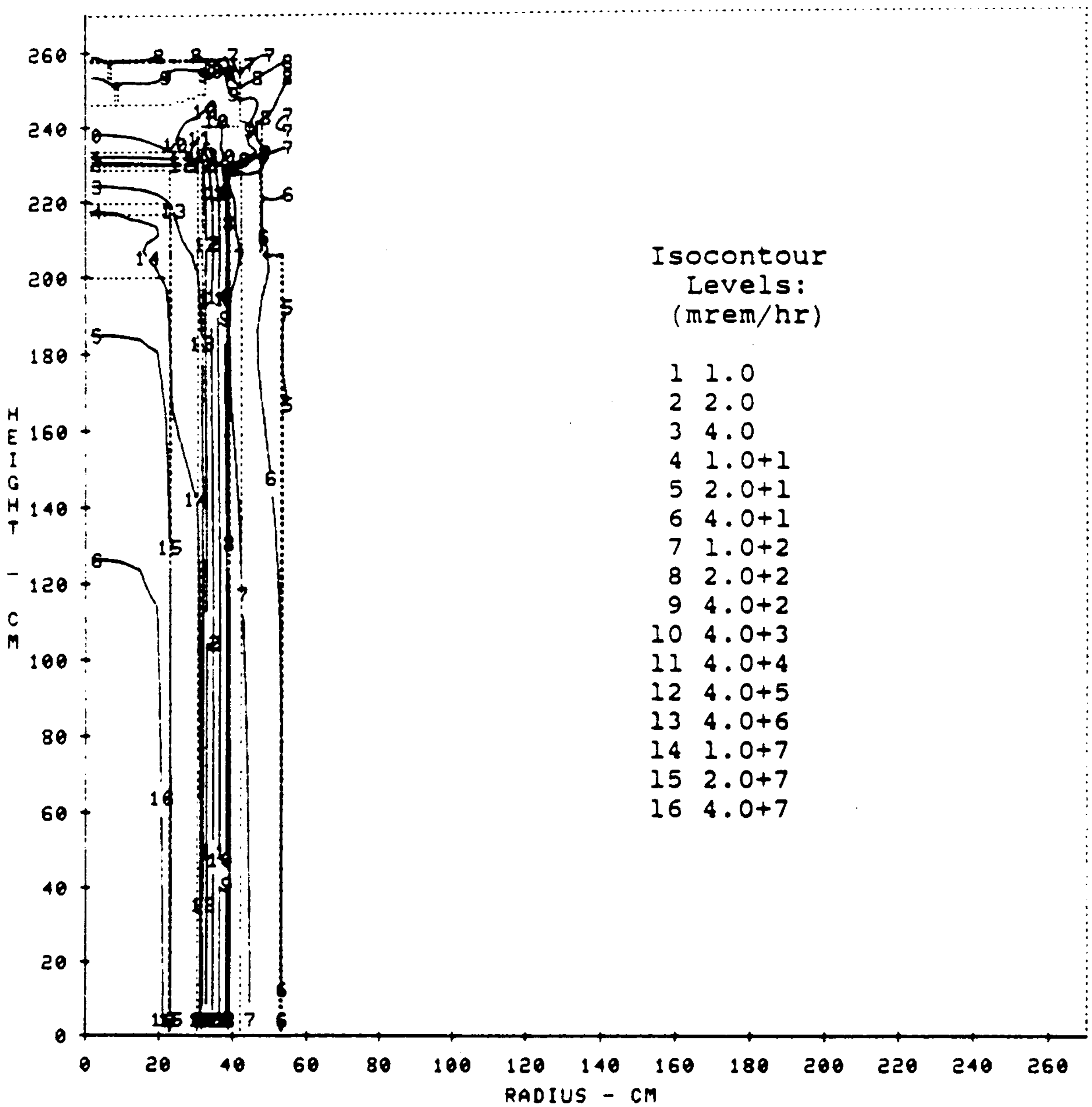


Figure 5.4-7 Primary Gamma Dose Rate Isocontours - Top Half of Cask

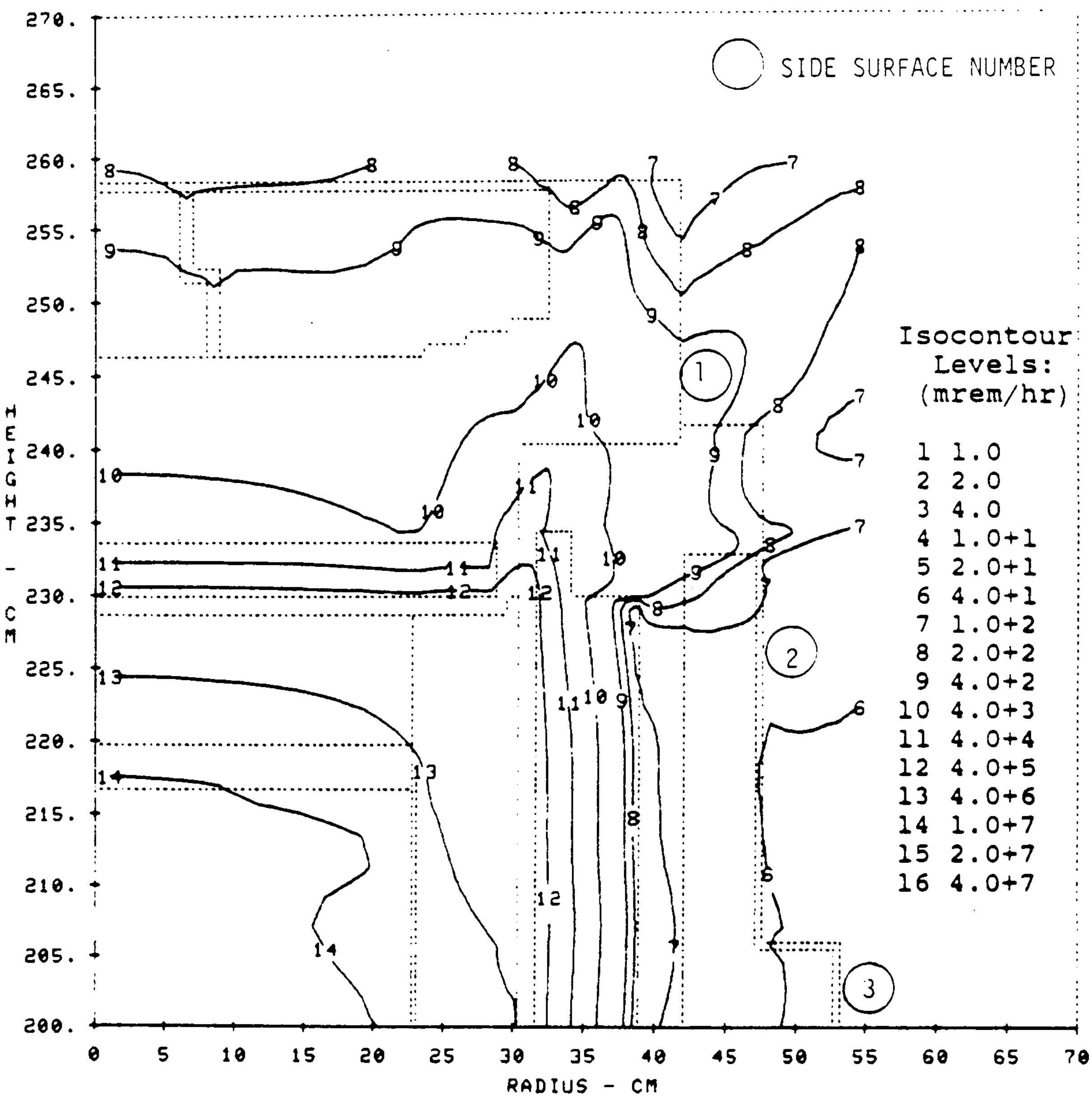


Figure 5.4-8 Primary Gamma Dose Rate Isocontours - Cask Top Half Details

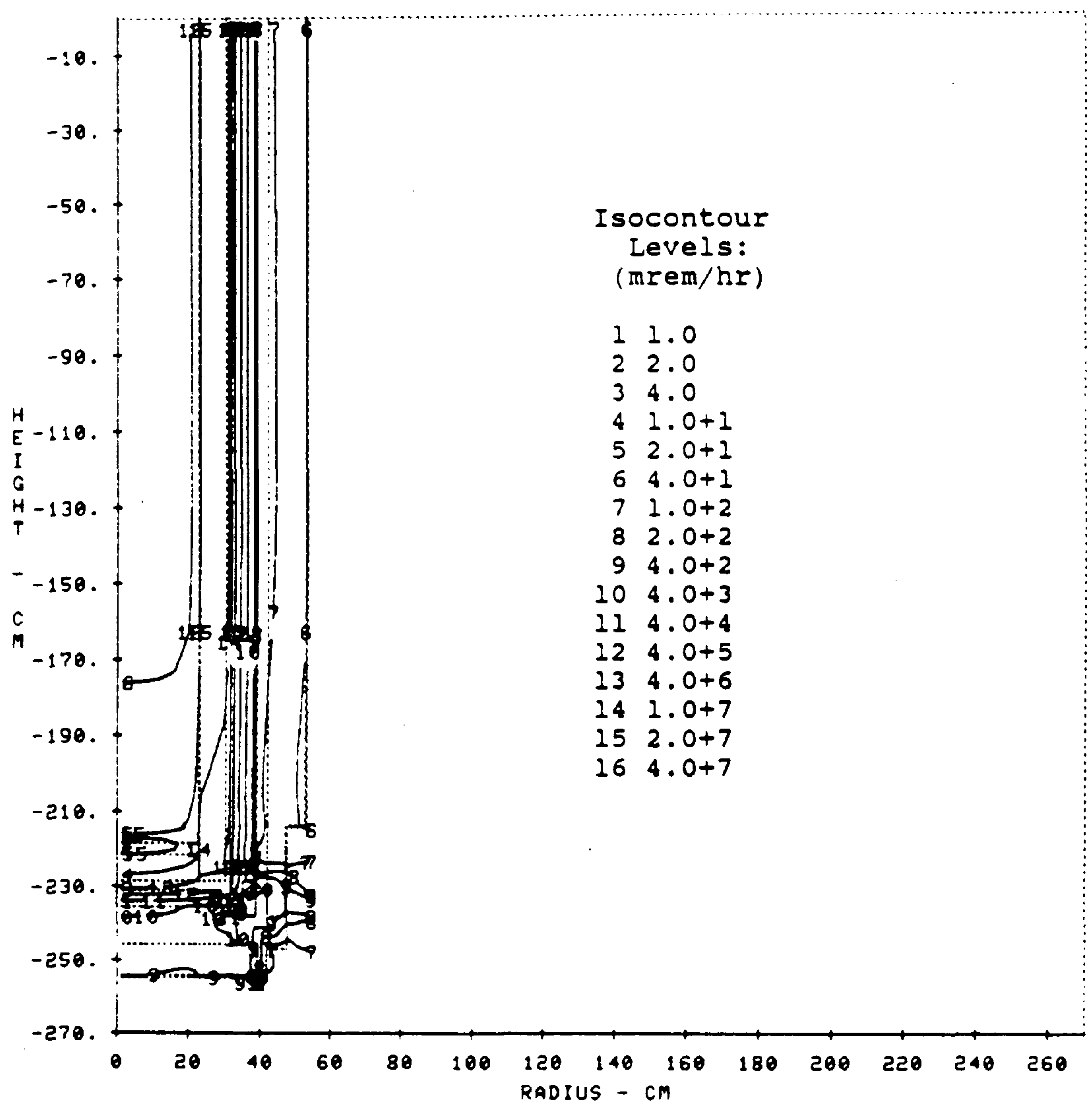


Figure 5.4-9 Primary Gamma Dose Rate Isocontours - Bottom Half of Cask

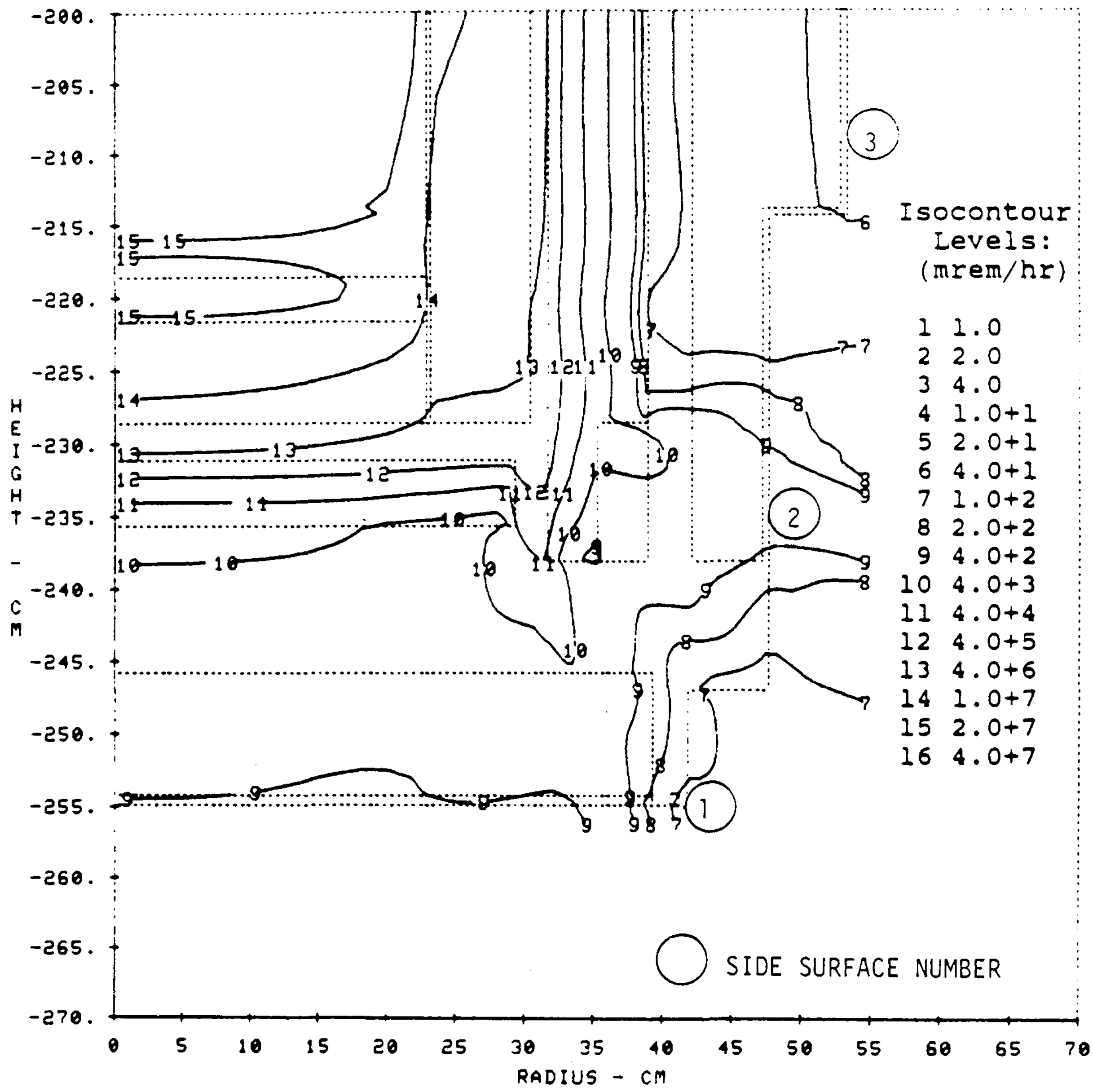


Figure 5.4-10 Primary Gamma Dose Rate Isocontours -
Cask Bottom Half Details

the cask centerline. At the top surface of the cask the peak occurs about halfway between the centerline and the outer cask radius. The thickness of the top Boro-Silicone is tapered at this location. On the bottom surface of the cask, the peak occurs just inside the outer radius of the bottom Boro-Silicone. Axially, the primary gamma dose rate peaks occur in the vicinity of the upper and lower cask ends. The axial peak at the bottom of the cask occurs outboard of the bottom depleted uranium. At the top end of the cask the axial peak occurs above the uppermost elevation of the radial Boro-Silicone.

Figures 5.4-11 through 5.4-18 provide neutron dose rates results. Top and bottom surface radial traverses and side surface axial traverses are shown in Figures 5.4-11 through 5.4-14, respectively. The three side surfaces referred to in Figures 5.4-11 through 5.4-14 are identified in the geometries shown in Figures 5.4-16 and 5.4-18. Figures 5.4-15 and 5.4-16 show isocontour plots of the neutron dose rate for the top half of the DOTIIIW cask geometry and for enlarged details of the top half. Figures 5.4-17 and 5.4-18 show isocontour plots of the neutron dose rate for the bottom half of the DOTIIIW cask geometry and for enlarged details of the bottom half. Radially, the top neutron surface dose rates peak about halfway between the outer radius of the top Boro-Silicone and the outer radius of the cask. The bottom surface neutron dose rate peaks at the centerline of the cask. The side surface neutron dose rate peaks at the elevation of the cavity midplane.

Secondary gamma dose rates illustrated in Figures 5.4-19 through 5.4-26 are from a previous cask design analysis. Surface dose rates are shown in Figures 5.4-19 through 5.4-22. Top and bottom surface plots are shown in Figures 5.4-19 and 5.4-20, with side surface plots provided in Figures 5.4-21 and 5.4-22. Secondary gamma dose rate isocontours are provided in Figures 5.4-23 and 5.4-26 for the top and bottom cask models. On the top and bottom surfaces, the secondary gamma dose rate peaks at the cask centerline. Secondary gamma dose rates peak on the side surface at cavity midplane. Secondary gamma dose rates based on the present design would be expected to be lower than those illustrated in Figures 5.4-19 through 5.4-26, with the same curve shapes and similar isocontour patterns.

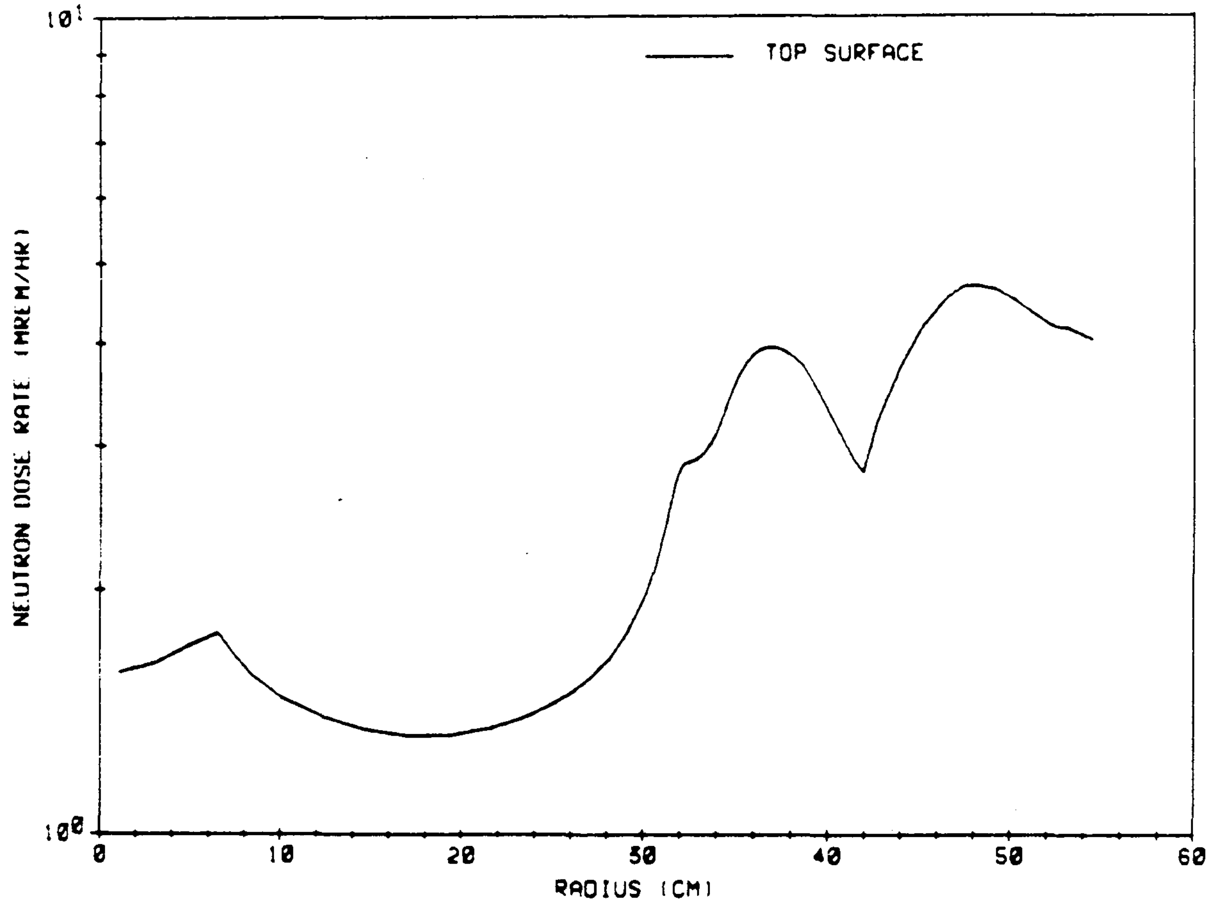


Figure 5.4-11 External Surface Neutron Dose Rates - Top Surface

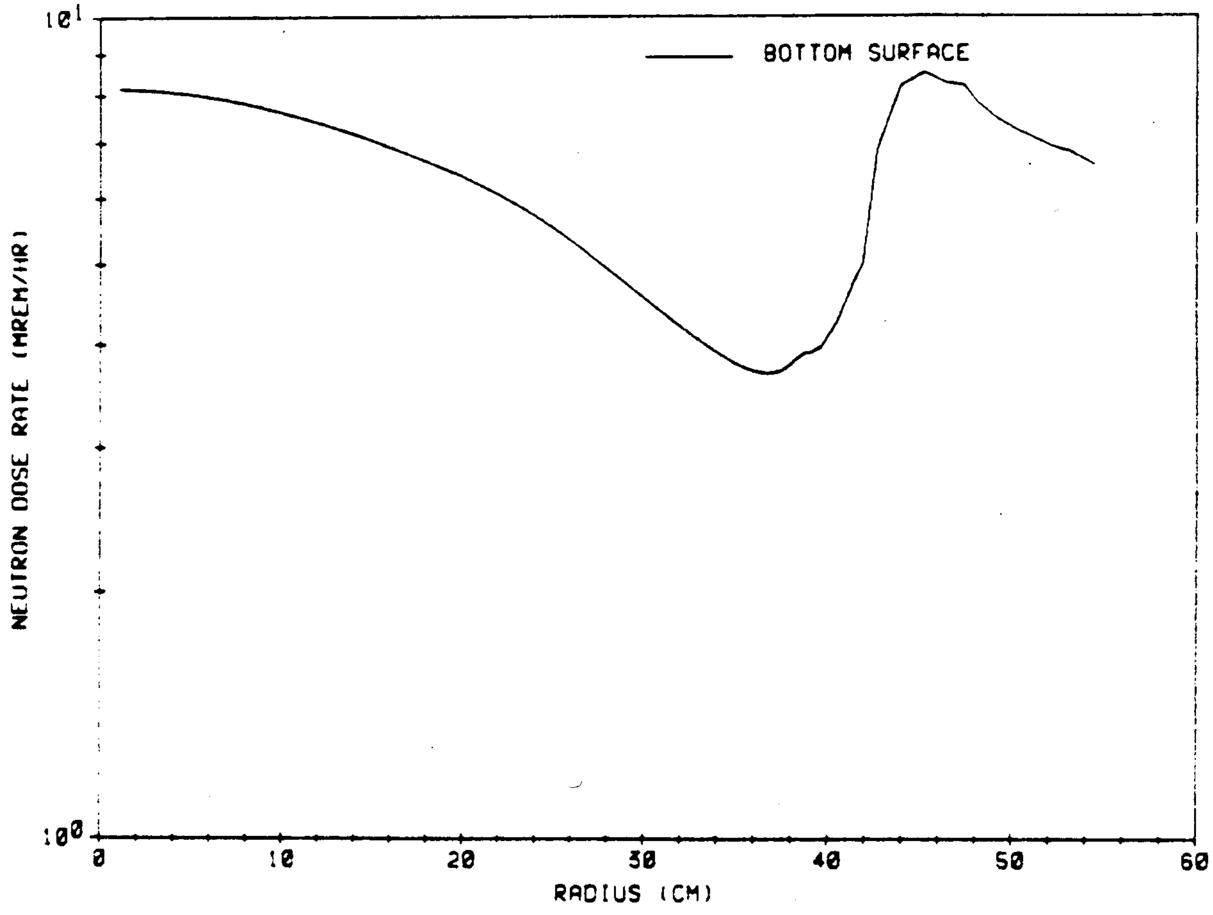


Figure 5.4-12 External Surface Neutron Dose Rates - Bottom Surface

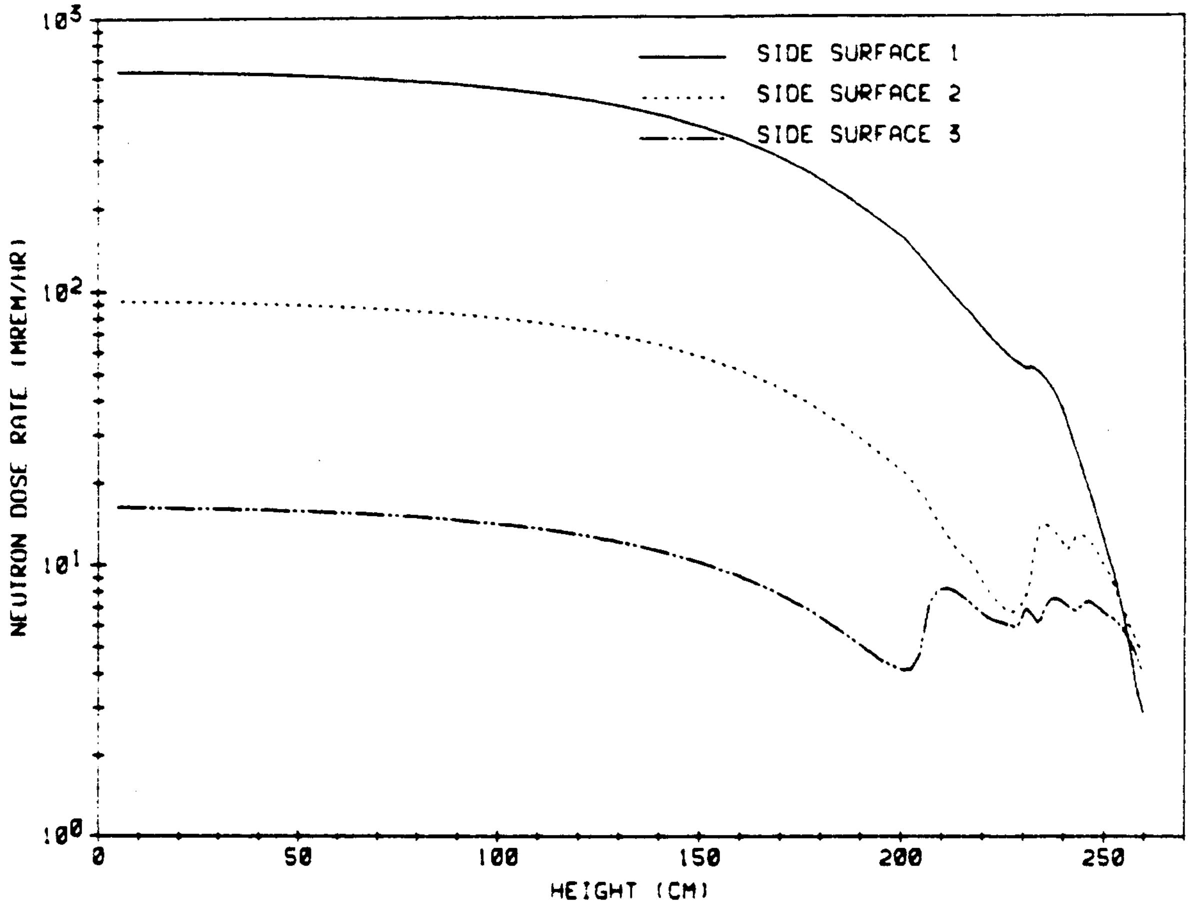


Figure 5.4-13 External Surface Neutron Dose Rates - Side Surface Top Half

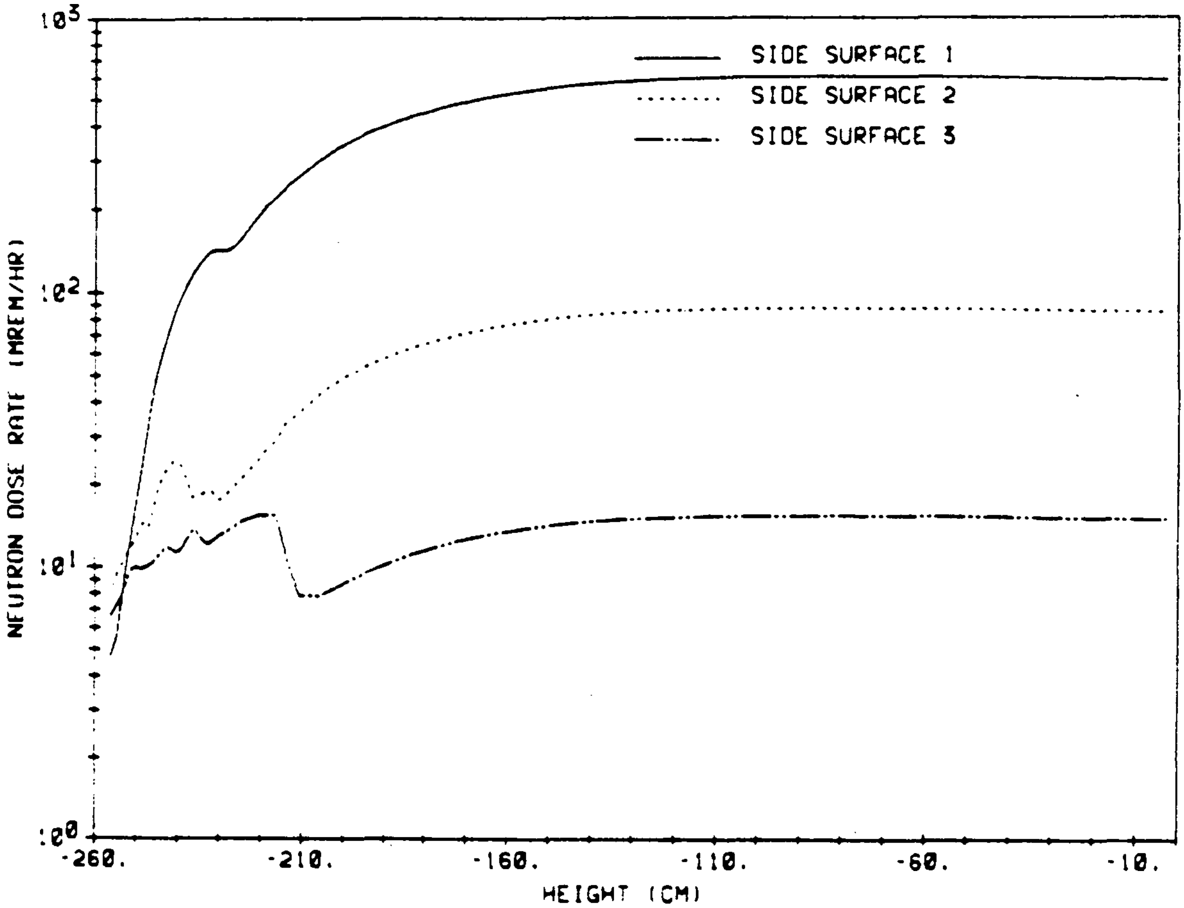


Figure 5.4-14 External Surface Neutron Dose Rates - Side Surface Bottom Half

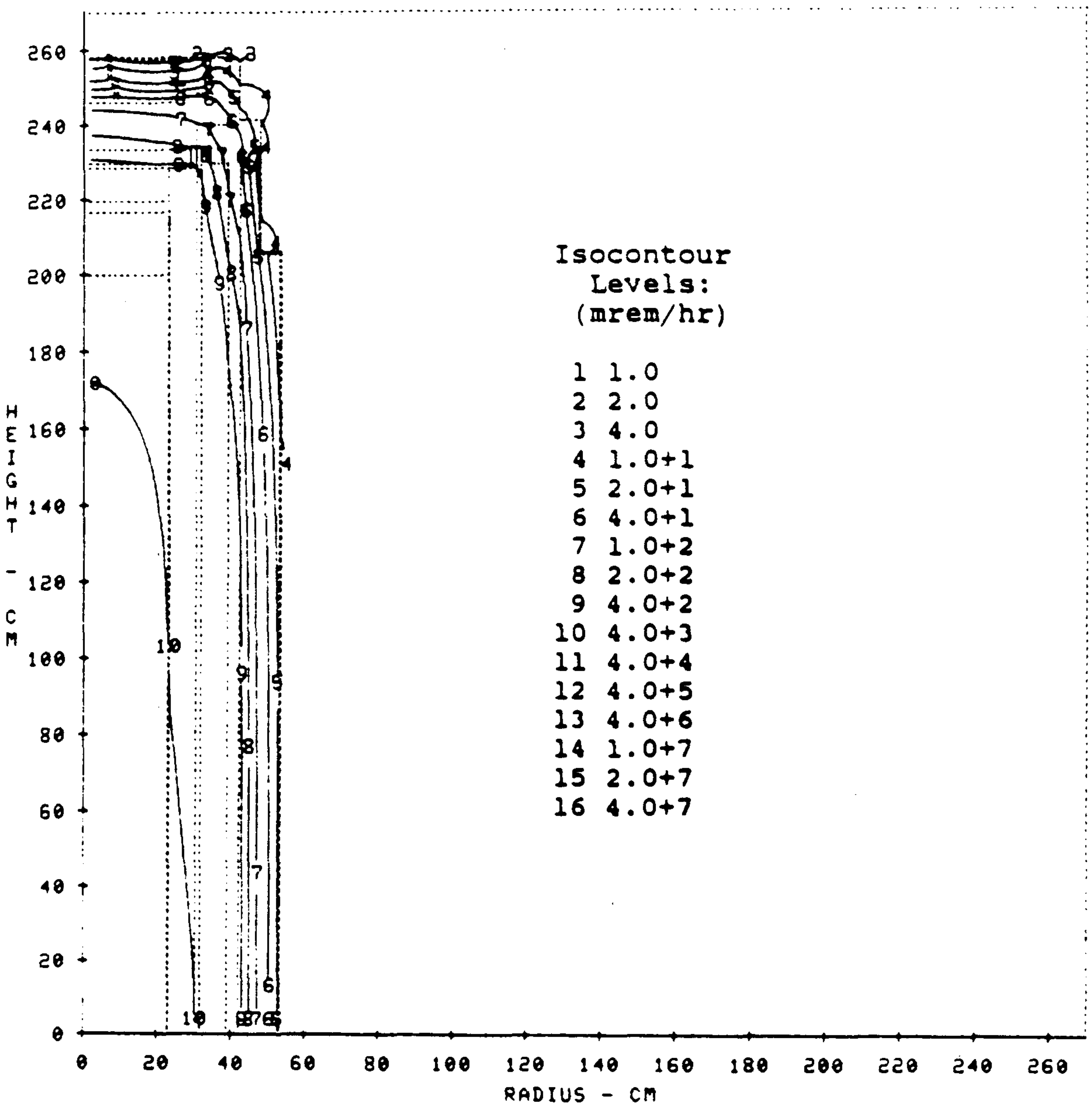


Figure 5.4-15 Neutron Dose Rate Isocontours - Top Half of Cask

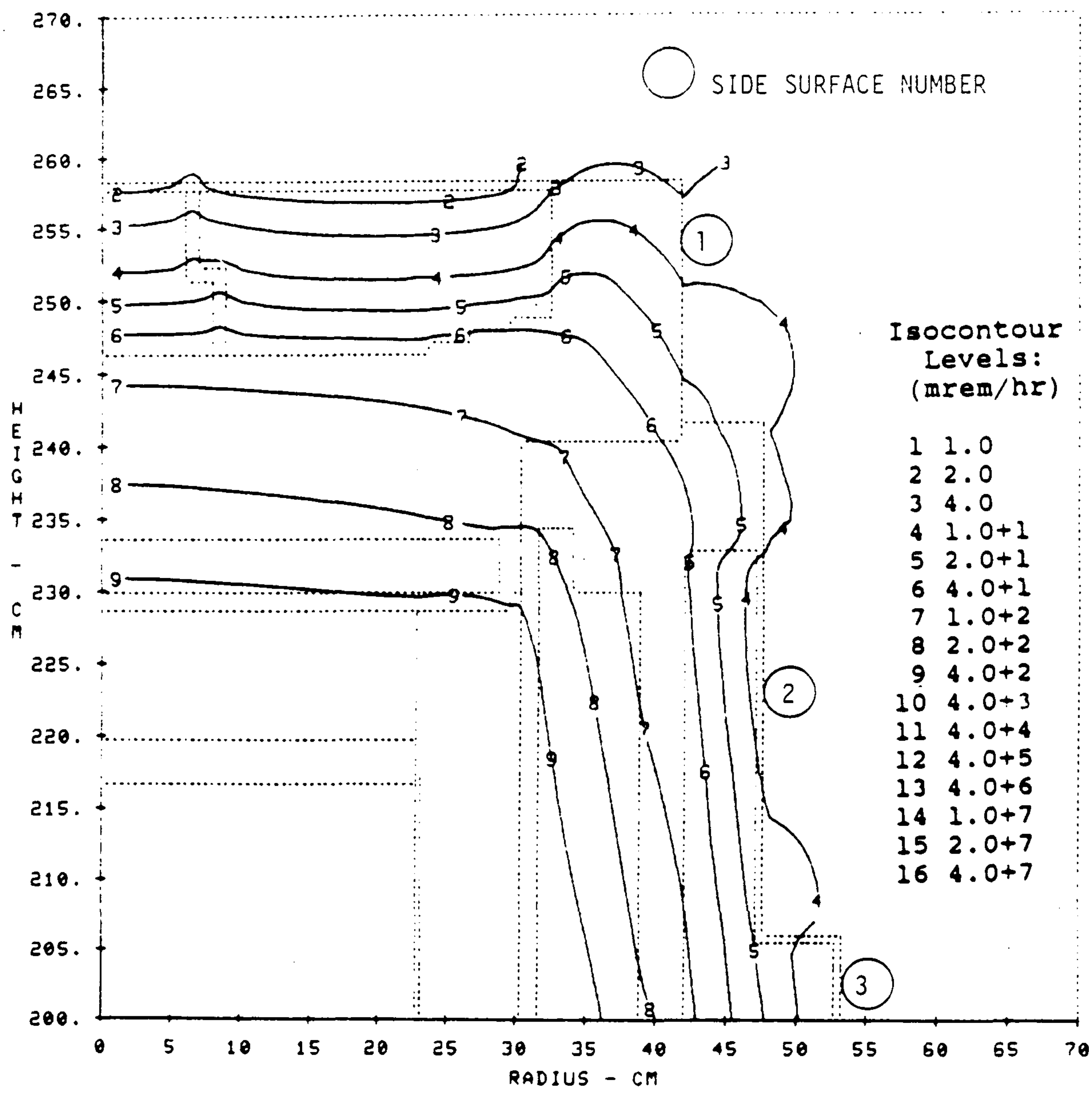


Figure 5.4-16 Neutron Dose Rate Isocontours - Cask Top Half Details

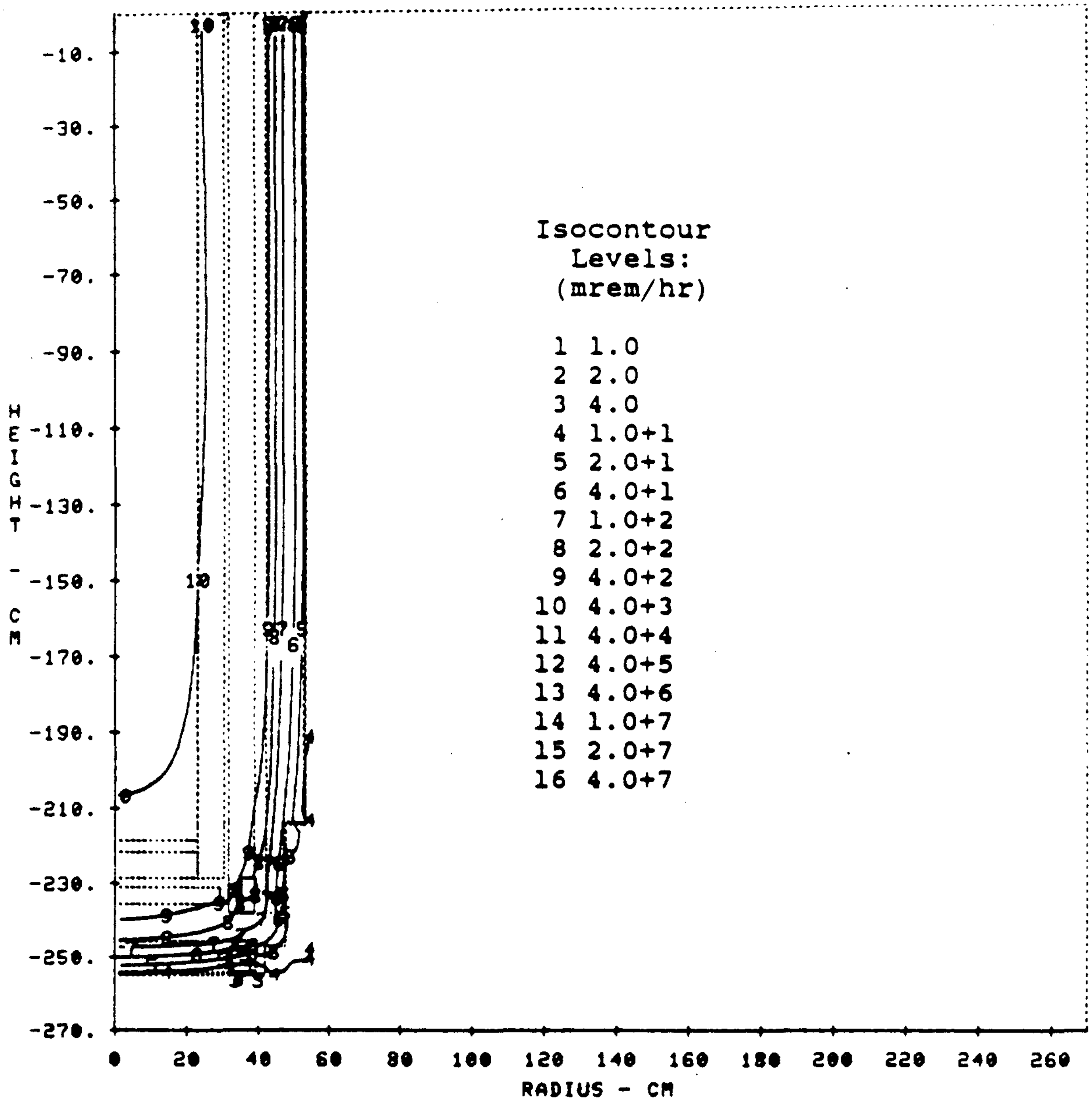


Figure 5.4-17 Neutron Dose Rate Isocontours - Bottom Half of Cask

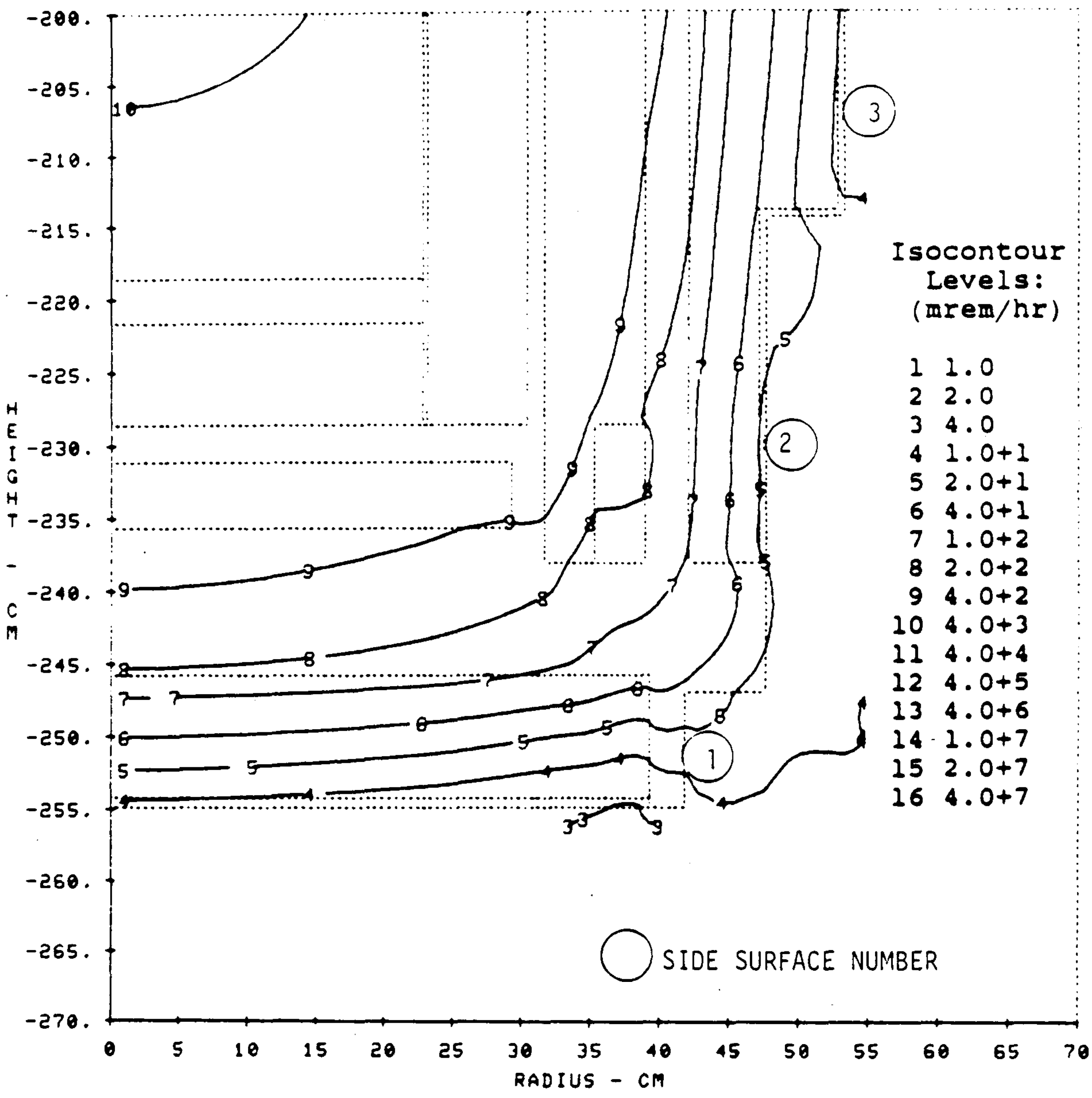


Figure 5.4-18 Neutron Dose Rate Isocontours - Cask Bottom Half Details

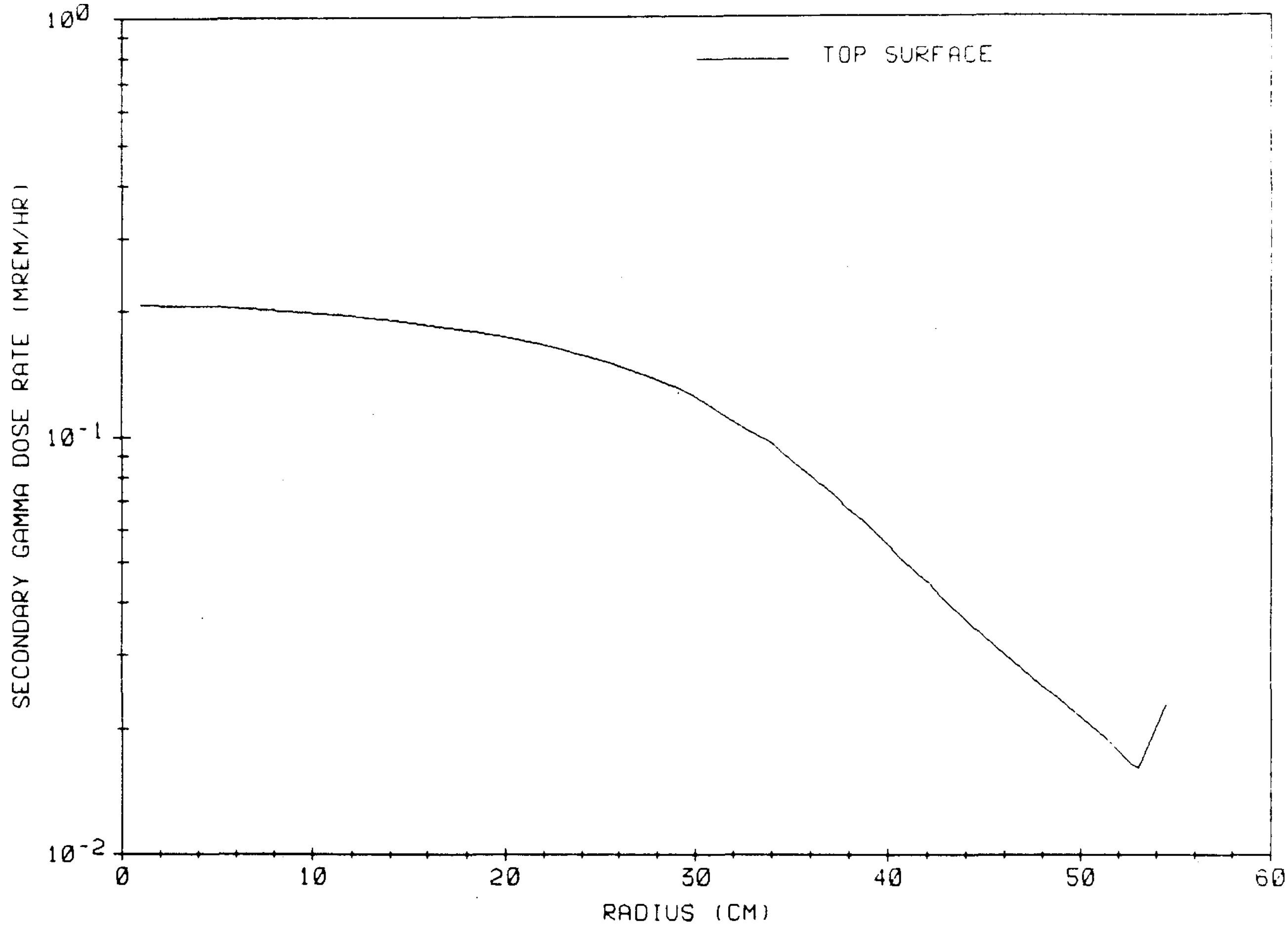


Figure 5.4-19 External Surface Secondary Gamma Dose Rates - Top Surface

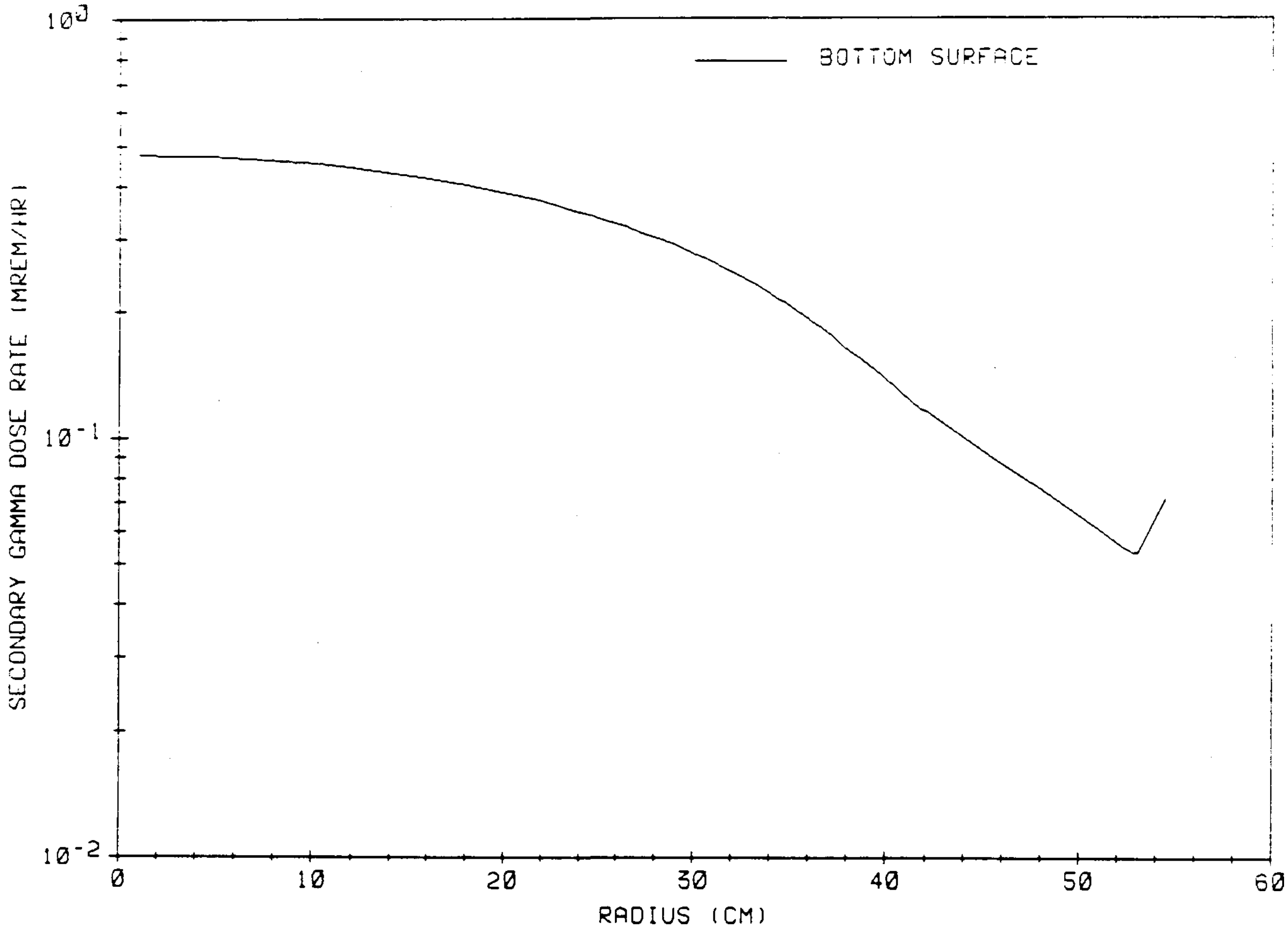


Figure 5.4-20 External Surface Secondary Gamma Dose Rate - Bottom Surface

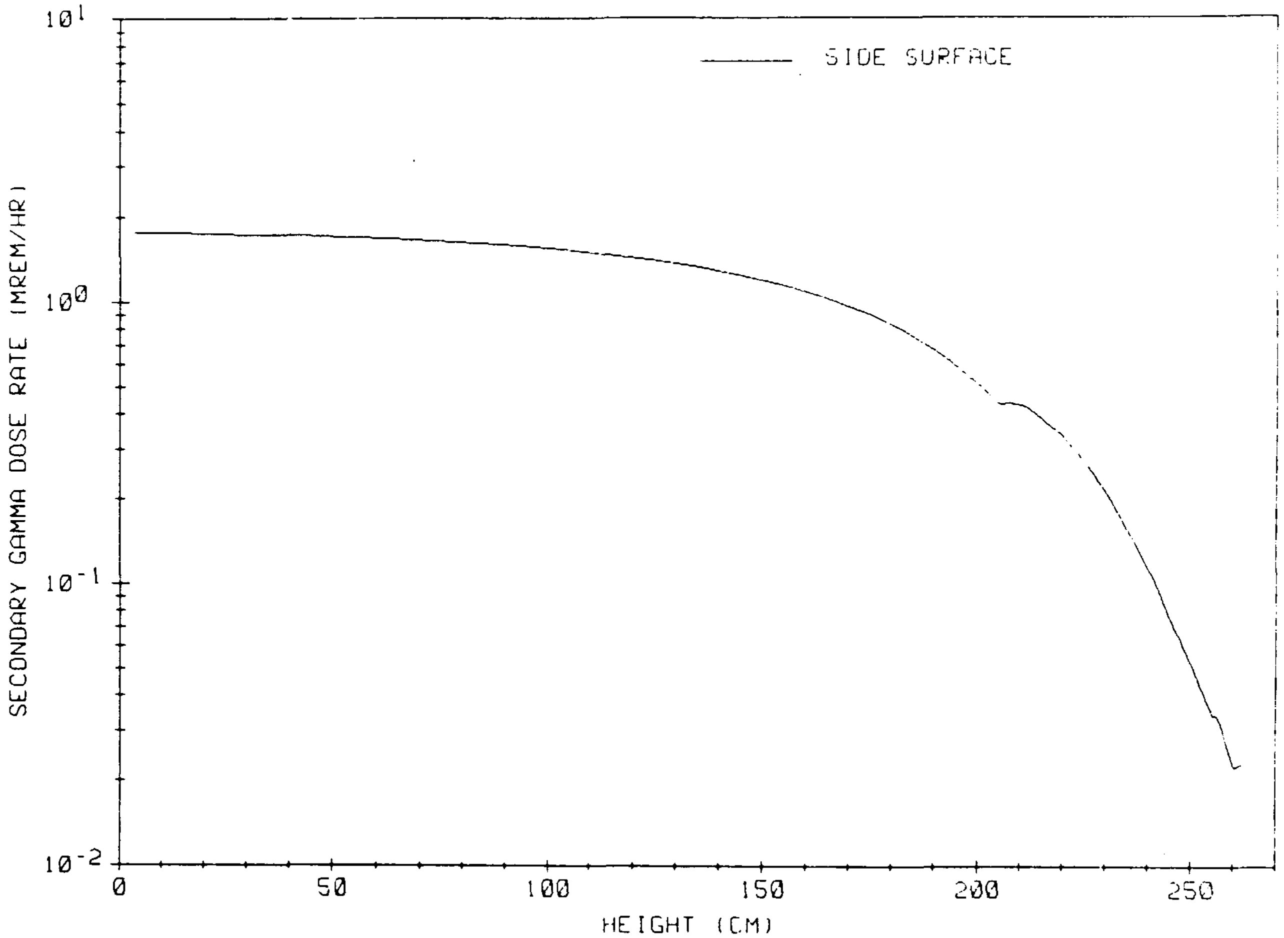


Figure 5.4-21 External Surface Secondary Gamma Dose Rates - Side Surface Top Half

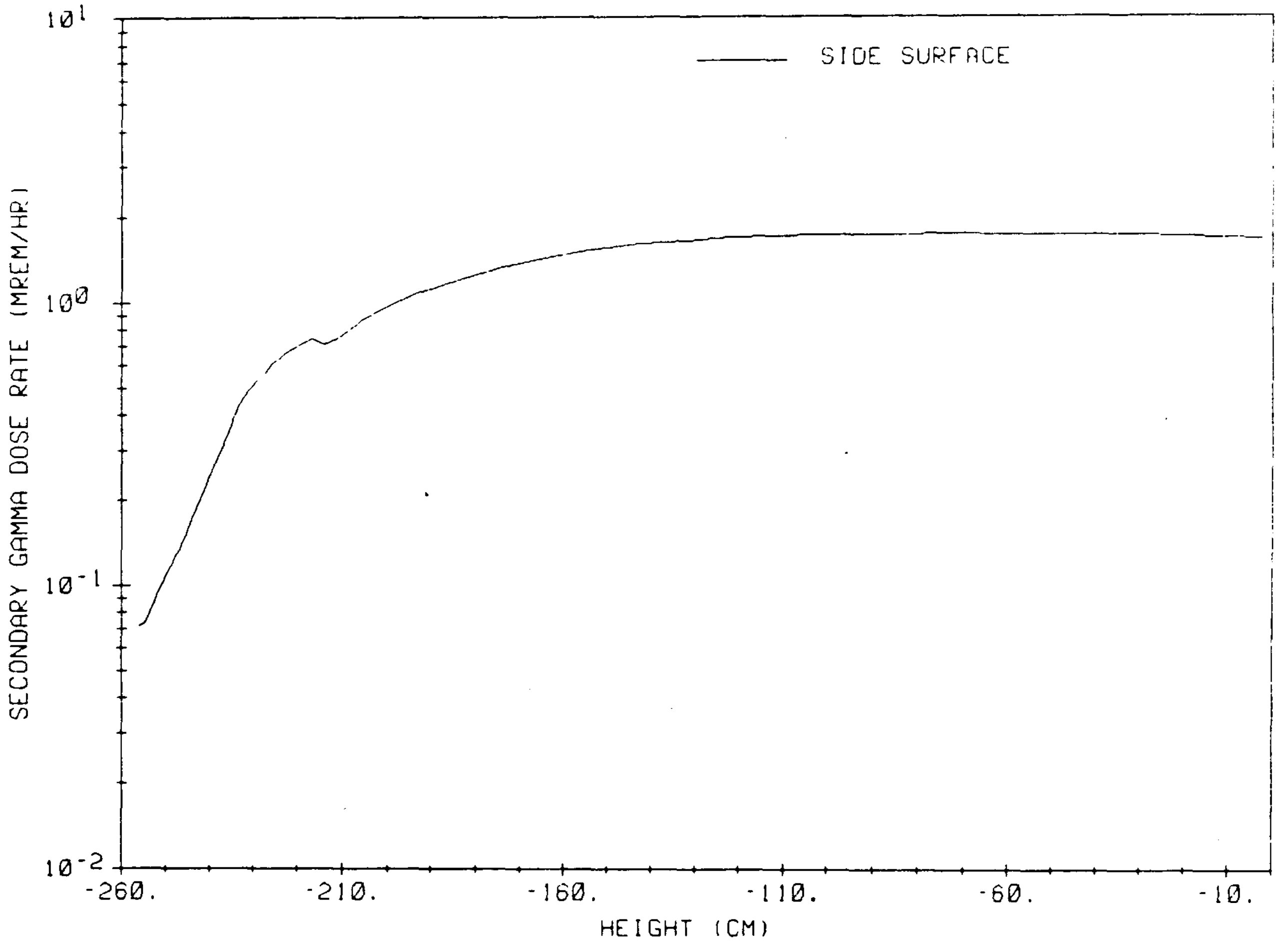


Figure 5.4-22 External Surface Secondary Gamma Dose Rates - Side Surface Bottom Half

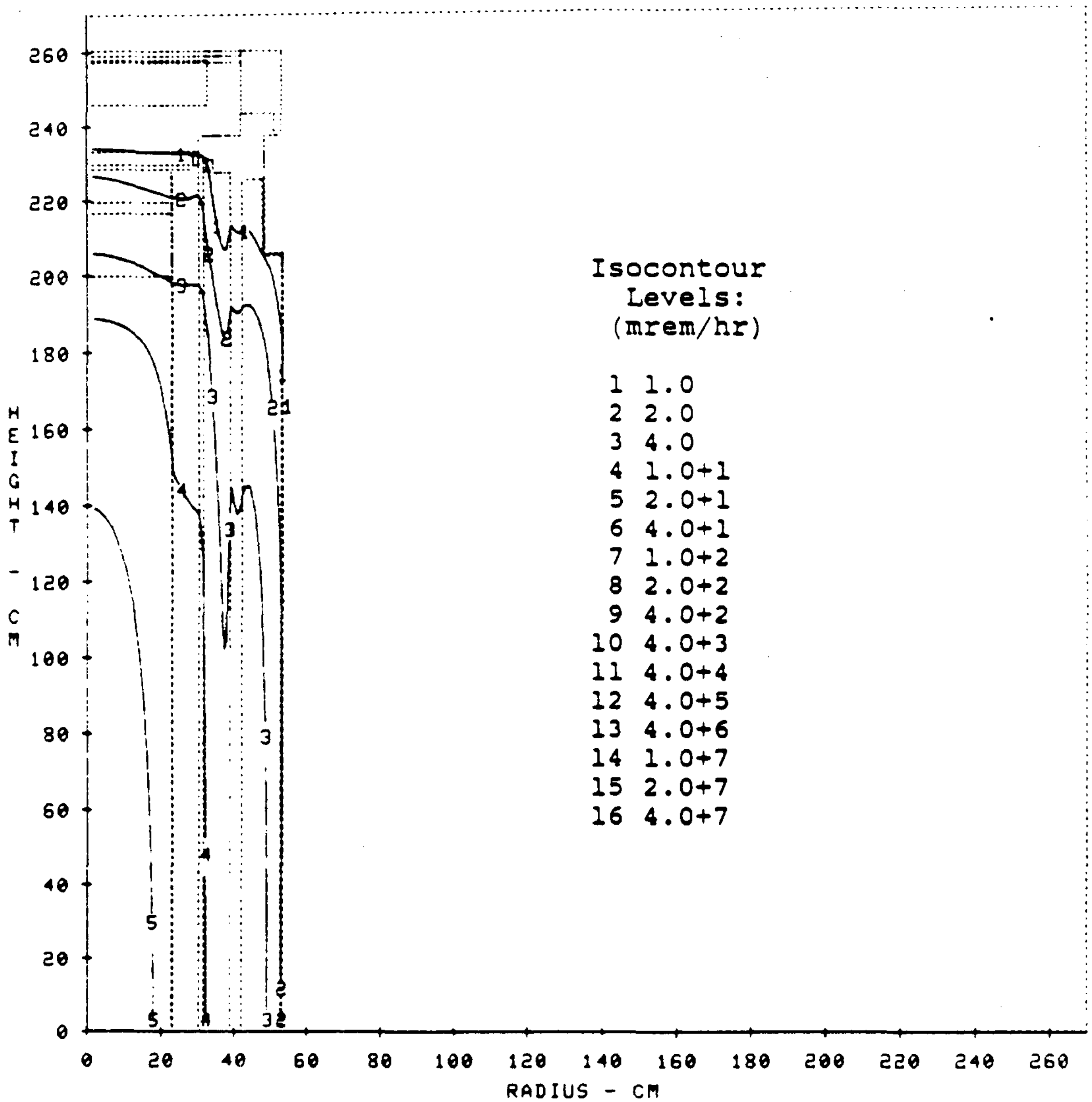


Figure 5.4-23 Secondary Gamma Dose Rate Isocontours - Top Half of Cask

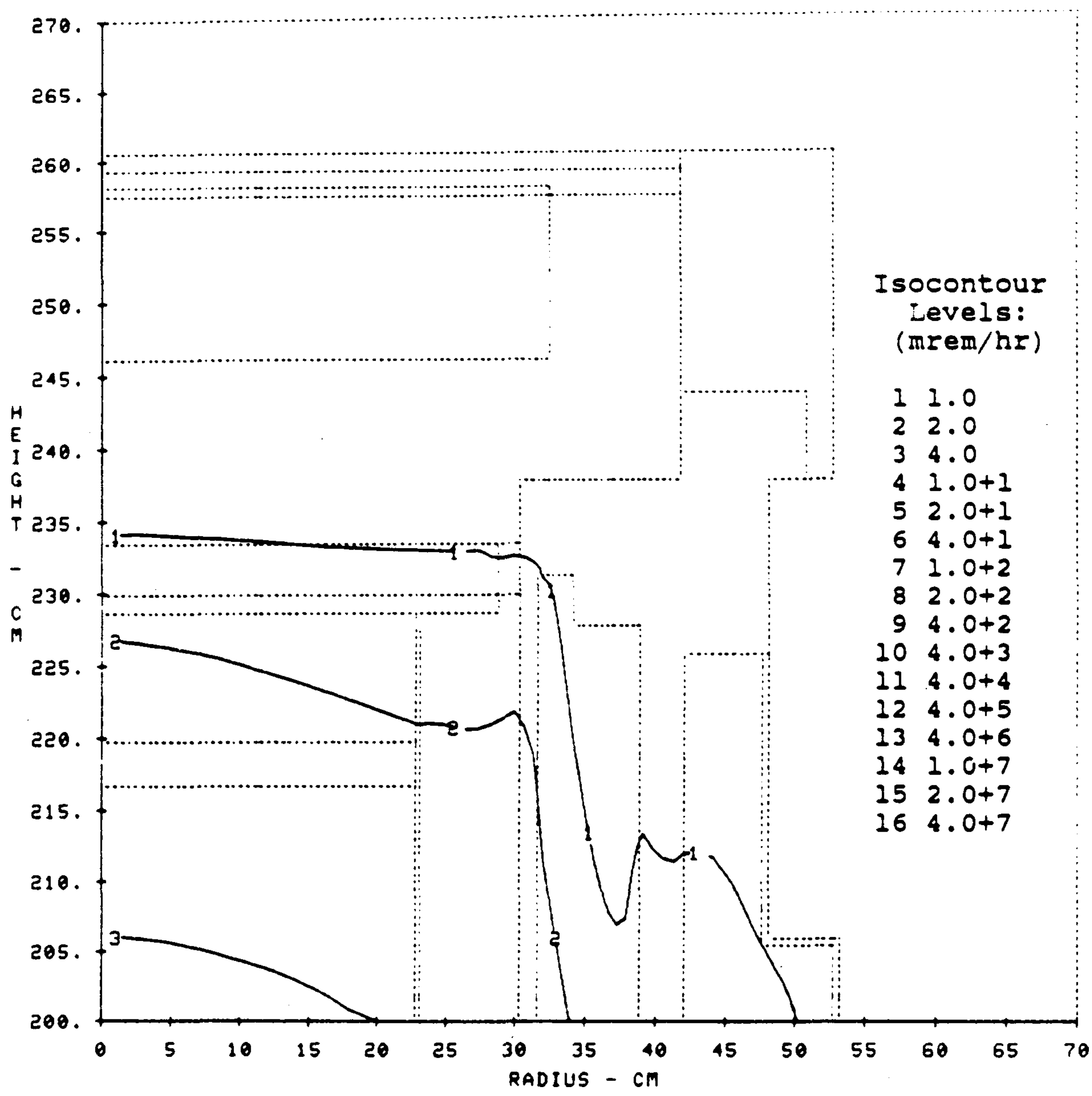


Figure 5.4-24 Secondary Gamma Dose Rate Isocontours - Cask Top Half Details

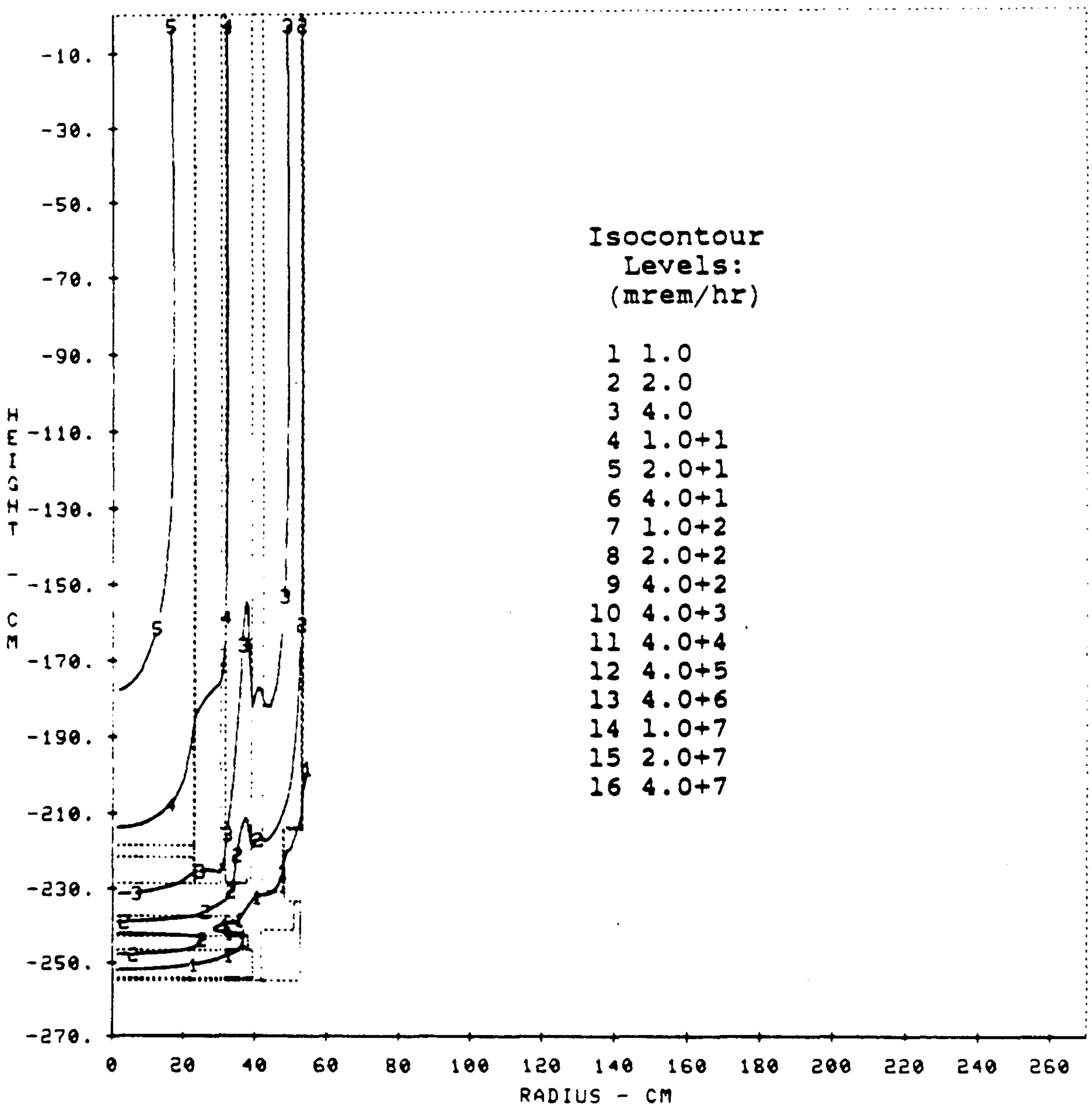


Figure 5.4-25 Secondary Gamma Dose Rate Isocontours - Bottom Half of Cask

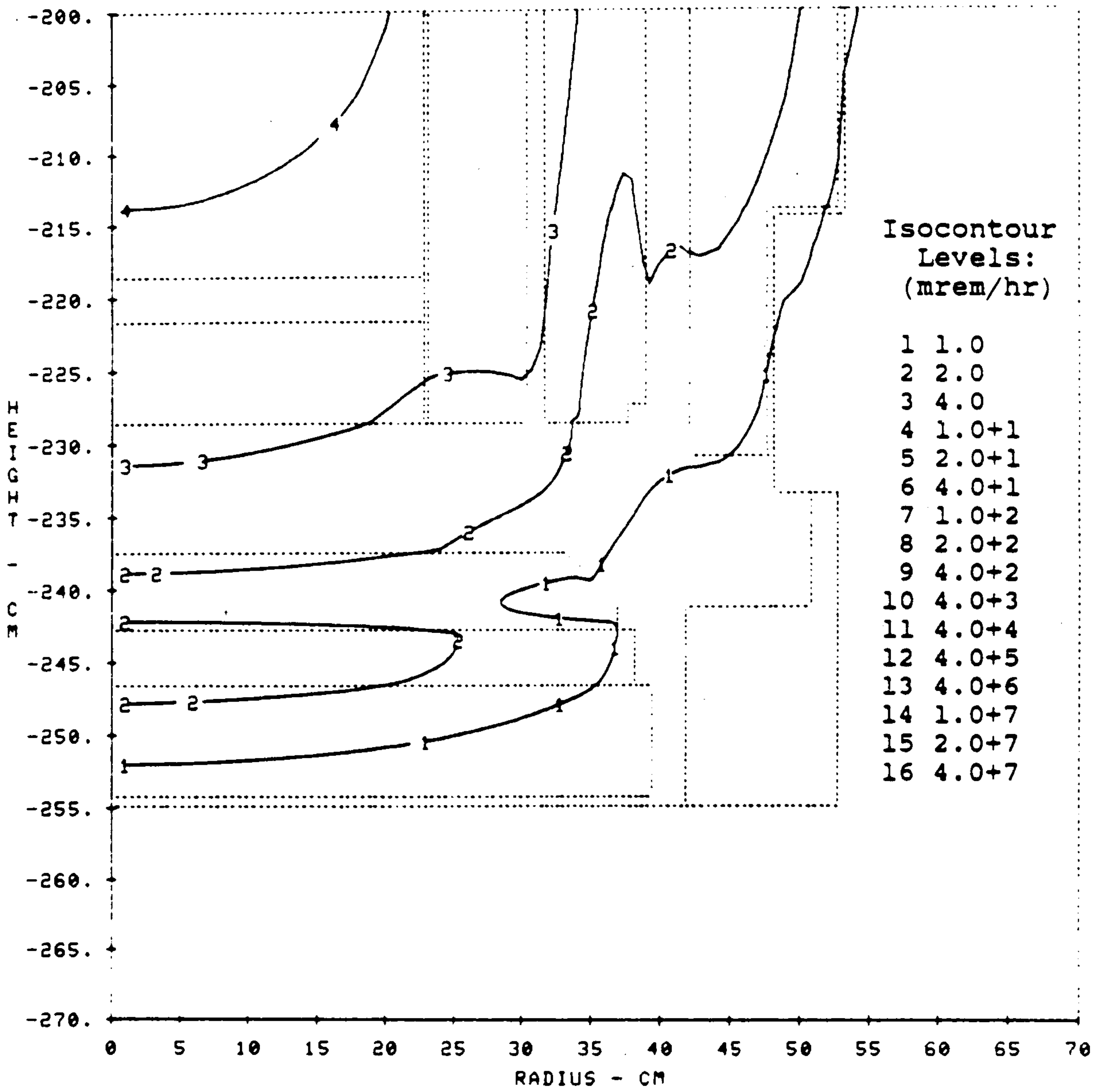


Figure 5.4-26 Secondary Gamma Dose Rate Isocontours - Cask Bottom Half Details

Dose rates at specific dose point locations are listed by contributor in Tables 5.4-2 and 5.4-3. Dose rates are provided for positions on the cask surface, 2 meters from the surface of the vehicle, and at the driver position. Figures 5.3-1 through 5.3-4 illustrate the locations of the various dose points. Tables 5.4-2 and 5.4-3 list the dose rates attributed to the primary gamma, neutron, and secondary gamma contributions, as well as the total dose rate at each position.

5.4.4 Discussion of Results

The LWT cask shield meets all of the radiation dose rate requirements of 10 CFR Part 71 and 49 CFR Part 173. The shielding thicknesses for the cask top and bottom were selected on the basis of allocating 90% of the dose rate to primary gammas and 10% from neutrons and secondary gammas. For the sides of the cask, the split is 70% primary gamma and 30% neutron and secondary gamma. In order to meet the criteria of 200 mrem/hr on the cask surface, primary gamma surface dose rates should be of the order of 180 mrem/hr on the top and bottom surfaces and 140 mrem/hr on the side surface. The combination of neutron and secondary gamma dose rates should be approximately 20 mrem/hr on the top and bottom surfaces and 60 mrem/hr on the side surface. The surface dose given in Table 5.4-2 are well below the 200 mrem/hr limit. Though shielding thicknesses may appear to be overly conservative relative to surface dose rate criteria, these thicknesses are required to meet the 2 meter from the vehicle dose rate criteria of 10 mrem/hr.

Based on the data provided in Tables 5.4-2 and 5.4-3, the cask meets the criteria at all detector points.

5.5 Appendix

5.5.1 References

5.2.1 CCC-371 (ORNL/TM-7175), "A User's Manual for the ORIGEN II Computer Code," Oak Ridge National Laboratory, Oak Ridge, Tennessee, July 1980.

5.2.2 (ORNL/TM-6055) "Updated Decay and Photon Libraries for the ORIGEN Code," Oak Ridge National Laboratory, Oak Ridge, Tennessee, February 1979.

5.2.3 DLC-76, "SAILOR: Coupled, Self-Shielded 47 Neutron, 20 Gamma Ray, P_3, Cross Section Library for Operating Reactors," Contributed by Science Applications, Inc., LaJolla, California; Oak Ridge National Laboratory, Oak Ridge, Tennessee; and Electric Power Research Institute, Palo Alto, California; ORNL Radiation Shielding Information Center, Oak Ridge, Tennessee, March 1983.

5.3.1 "Worldwide Guide to Equivalent Irons and Steels," American Society for Metals, Metals Park, Ohio 44073, 1979.

5.3.2 "Worldwide Guide to Equivalent Nonferrous Metals and Alloys," American Society for Metals, Metals Park, Ohio 44073, 1979.

5.4.1 DLC-75, "BUGLE-80: Coupled 47 Neutron, 20 Gamma Ray, P_3, Cross Section Library for LWR Shielding Calculations," Contributed by Oak Ridge National Laboratory, Radiation Shielding Information Center, Oak Ridge, Tennessee, June 1980.

5.4.2 CCC-418, "SCAP, Single Scatter, Albedo Scatter, and Point Kernel Analysis Program in Complex Geometry," Contributed by Westinghouse Advanced Reactors Division, Madison, PA; ORNL Radiation Shielding Information Center, Oak Ridge, Tennessee, 1982.

5.4.3 CCC-255, "ANISN-W, A One-Dimensional Discrete Ordinates Transport Computer Program," Contributed by Westinghouse Advanced Reactors Division, Madison, Pennsylvania; ORNL Radiation Shielding Information Center, Oak Ridge, Tennessee, 1971.

5.4.4 CCC-89, "DOT-IIIW, A Two-Dimensional Discrete Ordinates Transport Computer Program," Contributed by the Westinghouse Advanced Reactors Division, Madison, Pennsylvania; ORNL Radiation Shielding Information Center, 1980. (DOTIIIW is an unpublished enhancement of DOTIIW for the CRAY-1S computer).

5.4.5 WANL-PR-(LL)-034, "Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation," Volume 5, Two-Dimensional Discrete Ordinates Transport Techniques, R. K. Disney, et al., Westinghouse Astronuclear Laboratory, August 1970.

5.4.6 ANSI/ANS-6.1.1-2977, "Neutron and Gamma Ray Flux-to-Dose Factors," Prepared by the American Nuclear Society Standards Committee, Working Group ANS-6.1.1, American Nuclear Society, LaGrange Park, Illinois, 1977.

5.5.1 DOE/RW-0184, "Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation," Volume 3, U.S. Department of Energy, December 1987.

6. CRITICALITY EVALUATION

The criticality evaluation that was performed in support of the preliminary design of the LWT cask is presented in this section.

6.1 Discussion and Results

As described in Section 1.2.1, the TITAN LWT cask design employs two basic design components which are evaluated in the criticality analysis. The first is the cask body which has the basic shape of a right circular cylinder. The walls and ends of the body consist of layers of Boro-Silicone and depleted uranium shielding which are sandwiched between layers of Grade 9 Titanium. From a criticality standpoint, the second basic component of the cask design is the fuel basket. The basket is placed inside of the cask cavity to support and position the fuel assemblies. To accommodate the different fuel types there are two basket designs. One basket design will hold three PWR fuel assemblies while the second design will hold seven BWR fuel assemblies.

The design basis used for preventing criticality of fuel assemblies outside of a reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array will be less than 0.95.

The cask design is based on meeting this requirement for the transport of selected Westinghouse, B&W, CE, Exxon, and GE PWR and BWR fuel assemblies (see Section 6.2 for fuel parameters). Criticality of fuel assemblies in the fuel shipping cask is prevented by the design of the basket which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies in the cask and inserting neutron poison material between assemblies.

Results of the criticality analysis show that the LWT cask meets the criticality design basis under the Fissile Class I conditions. Table 6.1-1 summarizes the results of this analysis under Normal and Accident Conditions with the most reactive fuel type in place. The conditions and results comply with the performance requirements specified in Paragraphs 71.55 and 71.57 of 10 CFR Part 71.

Table 6.1-1

Summary of Criticality Evaluation

Fissile Classs I

Normal/Accident Conditions	
Number of undamaged packages calculated to be subcritical	∞
Optimum hydrogenous moderation	
Interspersed	1.0 gm/cm^3
Reflection	1.0 gm/cm^3
Package size	1.402×10^6 cm^3
Maximum fuel enrichment	4.5 w/o
Maximum K_∞	
PWR fuel	0.9477
BWR fuel	0.8022

6.2 Package Fuel Loading

The following fuel assembly types meet the criticality acceptance criteria for transport in the LWT cask as intact assemblies.

PWR Fuel Assemblies

Westinghouse Electric	17x17
Westinghouse Electric	15x15
Westinghouse Electric	14x14
Babcock & Wilcox	17x17
Babcock & Wilcox	15x15
Combustion Engineering	16x16
Combustion Engineering	14x14
Exxon Nuclear	17x17
Exxon Nuclear	15x15
Exxon Nuclear	14x14

BWR Fuel Assemblies

General Electric	8x8
General Electric	7x7
Exxon Nuclear	8x8
Exxon Nuclear	7x7

The fuel parameters (from Reference 5.5.1) are listed in Table 6.2-1 for each fuel type. No credit is taken for burnup. The fuel is assumed to be fresh with a maximum enrichment of 4.5 w/o U^{235} for each fuel type.

6.3 Model Specification

6.3.1 Description of Calculational Model

The TITAN LWT cask design is shown schematically in Figure 6.3-1. The PWR and BWR fuel basket designs used in this analysis are shown in Figures 6.3-2, 6.3-3 and 6.3-4. Nominal dimensions and materials used in the design are shown on the Figures. The fuel basket designs are modeled exactly in the

Table 6.2-1
Fuel Parameters Employed in Criticality Analysis

Parameter	W 17x17 OFA	W 17x17 STANDARD	EXXON 17x17
Number of Fuel Rods per Assembly	264	264	264
Rod Zirc-4 Clad O.D. (inch)	0.360	0.374	0.360
Clad Thickness (inch)	0.0225	0.0225	0.025
Fuel Pellet O.D. (inch)	0.3088	0.3225	0.303
Fuel Pellet Density (% of Theoretical)	96	96	96
Fuel Pellet Dishing Factor	0.0	0.0	0.0
Rod Pitch (inch)	0.496	0.496	0.496
Number of Zirc-4 Guide Tubes	24	24	24
Guide Tube O.D. (inch)	0.474	0.482	---
Guide Tube Thickness (inch)	0.016	0.016	---
Number of Instrument Tubes	1	1	1
Instrument Tube O.D. (inch)	0.474	0.482	---
Instrument Tube Thickness (inch)	0.016	0.016	---

Table 6.2-1 (Continued)
Fuel Parameters Employed in Criticality Analysis

Parameter	W 14x14 STANDARD	W 14x14 OFA
Number of Fuel Rods per Assembly	179	179
Rod Zirc-4 Clad O.D. (inch)	0.422	0.400
Clad Thickness (inch)	0.0243	0.0243
Fuel Pellet O.D. (inch)	0.3659	0.3444
Fuel Pellet Density (% of Theoretical)	96	96
Fuel Pellet Dishing Factor	0	0
Rod Pitch (inch)	0.556	0.556
Number of Zirc-4 Guide Tubes	16	16
Guide Tube O.D. (inch)	0.539	0.526
Guide Tube Thickness (inch)	0.017	0.0170
Number of Instrument Tubes	1	1
Instrument Tube O.D. (inch)	0.4220	0.3990
Instrument Tube Thickness (inch)	0.0240	0.0235

Table 6.2-1 (Continued)

Fuel Parameters Employed in Criticality Analysis

Parameter	W 15x15 OFA	EXXON 15X15
Number of Fuel Rods per Assembly	204	204
Rod Zirc-4 Clad O.D. (inch)	0.422	0.424
Clad Thickness (inch)	0.024	0.030
Fuel Pellet O.D. (inch)	0.3659	0.3565
Fuel Pellet Density (% of Theoretical)	96	96
Fuel Pellet Dishing Factor	0	0
Rod Pitch (inch)	0.553	0.563
Number of Zirc-4 Guide Tubes	20	20
Guide Tube O.D. (inch)	0.532	---
Guide Tube Thickness (inch)	0.017	---
Number of Instrument Tubes	1	1
Instrument Tube O.D. (inch)	0.532	---
Instrument Tube Thickness (inch)	0.017	---

Table 6.2-1 (Continued)
Fuel Parameters Employed in Criticality Analysis

Parameter	CE 14x14	CE 16x16	EXXON 14x14
Number of Fuel Rods per Assembly	176	236	176
Rod Clad O.D. (inch)	0.440	0.382	0.440
Rod Clad Material	Zirc	Zirc	Zirc
Clad Thickness (inch)	0.026	0.025	0.031
Fuel Pellet O.D. (inch)	0.3795	0.325	0.370
Fuel Pellet Density (% of Theoretical)	—	96	94
Fuel Pellet Dishing Factor	0.0	0.0	0.99
Rod Pitch (inch)	0.580	0.506	0.580
Number of Zirc-4 Guide Tubes	5	5	5
Guide Tube O.D. (inch)	1.111	0.98	1.115
Guide Tube Thickness (inch)	0.038	0.04	0.04
Guide Tube Material	Zirc	Zirc	Zirc
Number of Instrument Tubes	0	0	0

Table 6.2-1 (Continued)
Fuel Parameters Employed in Criticality Analysis

Parameter	B&W 15x15	B&W 17x17
Number of Fuel Rods per Assembly	208	264
Rod Zirc-4 Clad O.D. (inch)	0.430	0.379
Clad Thickness (inch)	0.0265	0.0235
Fuel Pellet O.D. (inch)	0.370	0.324
Fuel Pellet Density (% of Theoretical)	96	96
Fuel Pellet Dishing Factor	0.0	0.0
Rod Pitch (inch)	0.568	0.502
Number of Zirc-4 Guide Tubes	16	24
Guide Tube O.D. (inch)	0.530	0.465
Guide Tube Thickness (inch)	0.016	0.0175
Number of Instrument Tubes	1	1
Instrument Tube O.D. (inch)	0.493	0.465
Instrument Tube Thickness (inch)	0.026	0.0175

Table 6.2-1 (Continued)
Fuel Parameters Employed in Criticality Analysis

Parameter	EXXON 7x7	EXXON 8x8
Number of Fuel Rods per Assembly	48	63
Rod Clad O.D. (inch)	0.570	0.494
Rod Clad Material	Zirc-2	Zirc-2
Clad Thickness (inch)	0.036	0.036
Fuel Pellet O.D.(inch)	0.49	0.4195
Fuel Pellet Density (% of Theoretical)	95	95
Fuel Pellet Dishing Factor	0	0
Rod Pitch (inch)	0.738	0.641
Number of Water Rod Tubes	0	1
Water Rod O.D. (inch)	---	---
Water Rod Tube Thickness (inch)	---	---
Water Rod Tube Material	---	---

Table 6.2-1 (Continued)
Fuel Parameters Employed in Criticality Analysis

Parameter	GE 7x7	GE 8x8	GE 8x8R
Number of Fuel Rods per Assembly	49	63	62
Rod Clad O.D. (inch)	0.563	0.493	0.483
Rod Clad Material	Zirc-2	Zirc-2	Zirc-2
Clad Thickness (inch)	0.032	0.034	0.032
Fuel Pellet O.D. (inch)	0.487	0.416	0.410
Fuel Pellet Density (% of Theoretical)	96	96	96
Fuel Pellet Dishing Factor	0	0	0
Rod Pitch (inch)	0.738	0.640	0.640
Number of Water Rods	0	1	2
Water Rod O.D.	---	0.591	0.591
Water Rod Tube Wall Thickness	---	0.030	0.030
Water Rod Material	---	Zirc-2	Zirc-2

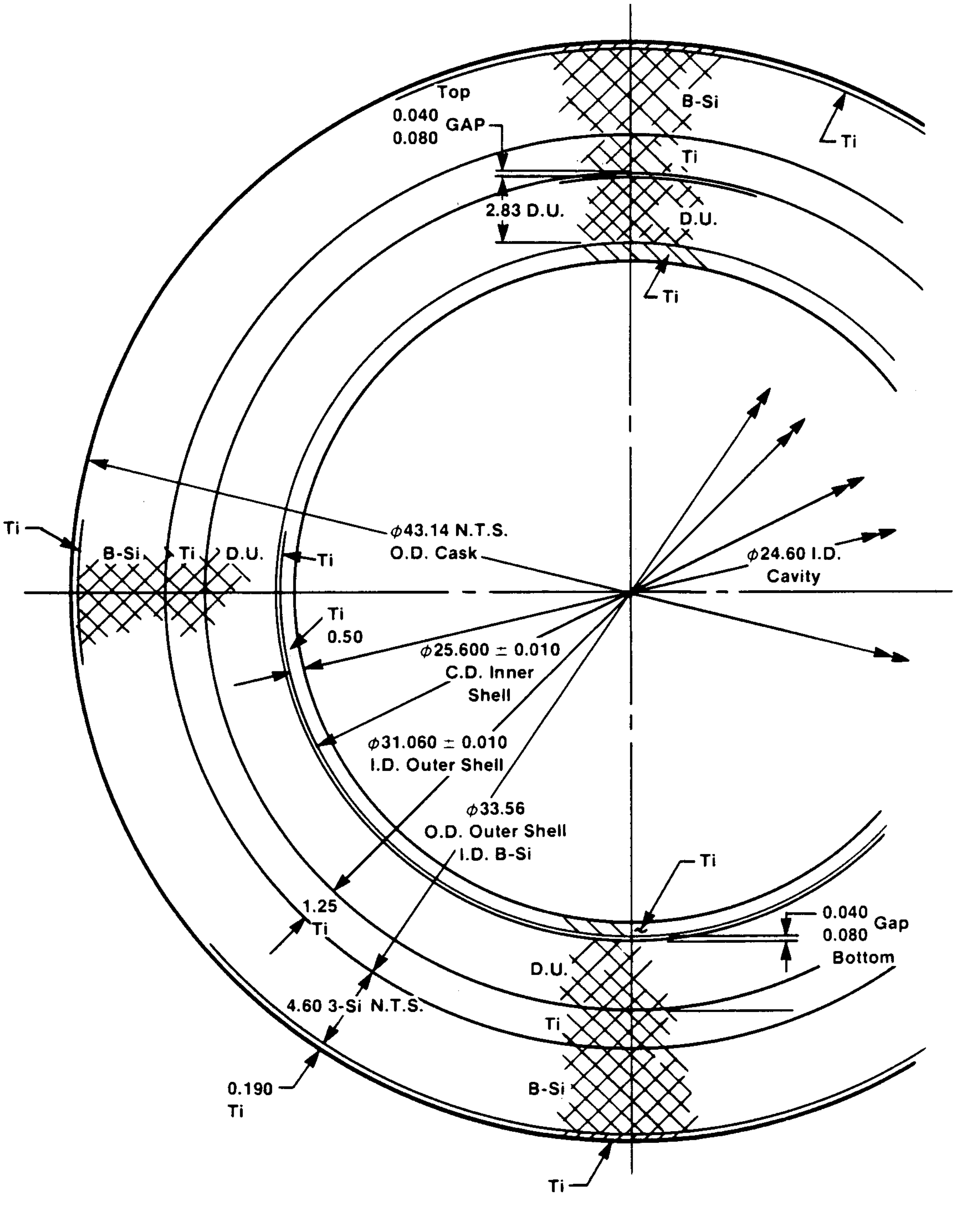


Figure 6.3-1. LWT Cask Nominal Dimensions

768370-50A

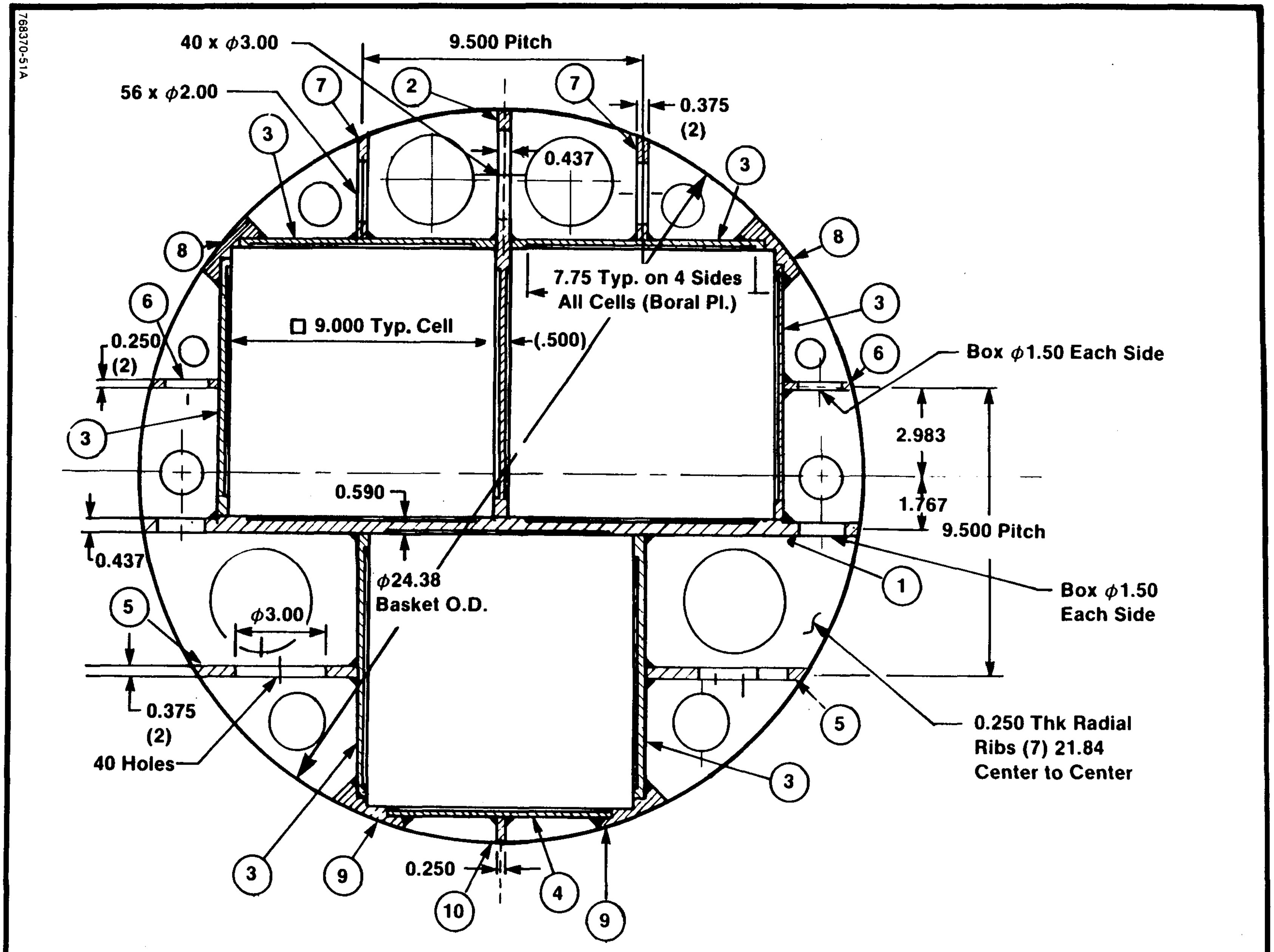


Figure 6.3-2. LWT Cask PWR Fuel Basket Nominal Dimensions

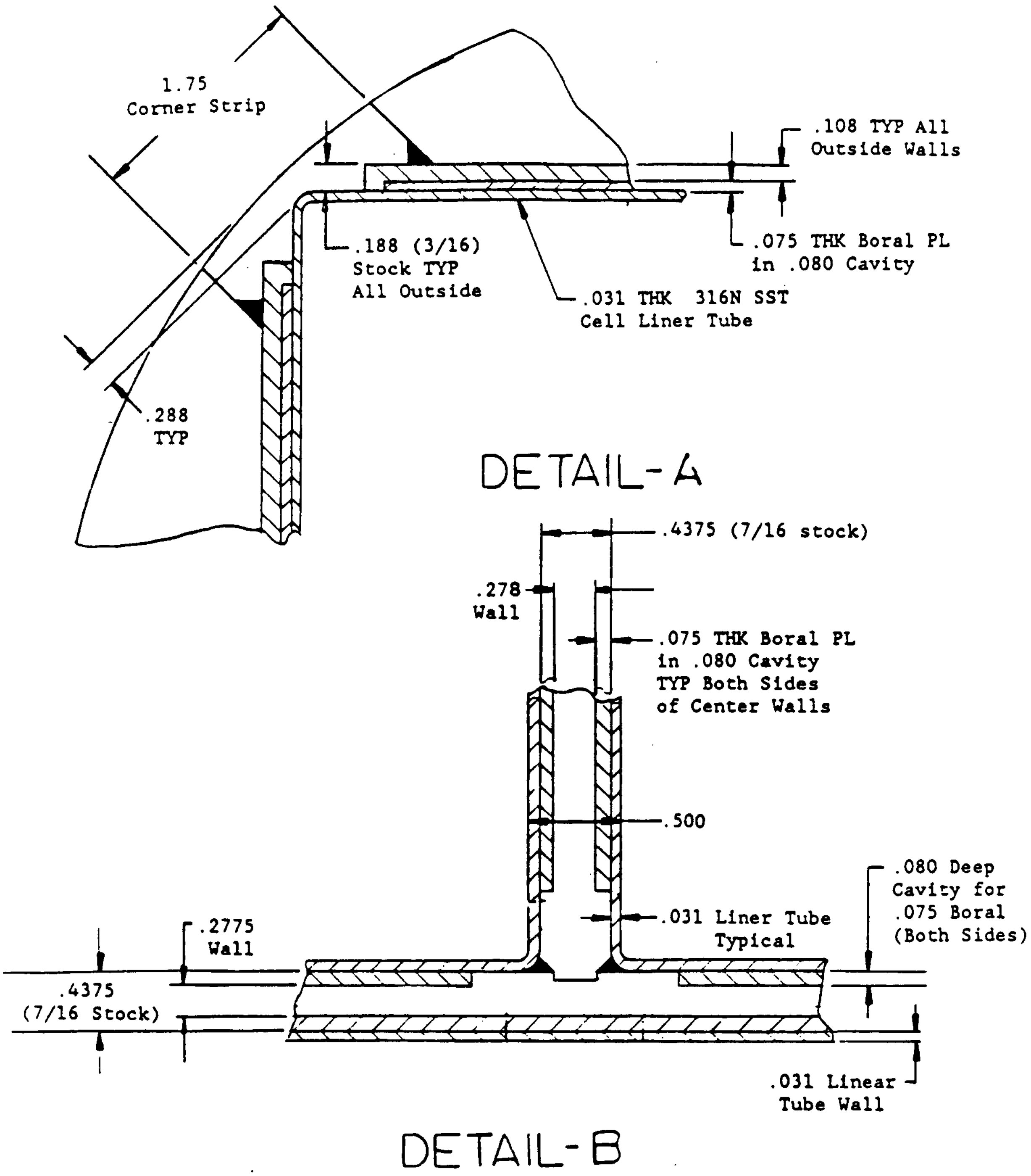


Figure 6.3-3. LWT Cask PWR Fuel Basket Corner Details and Dimensions

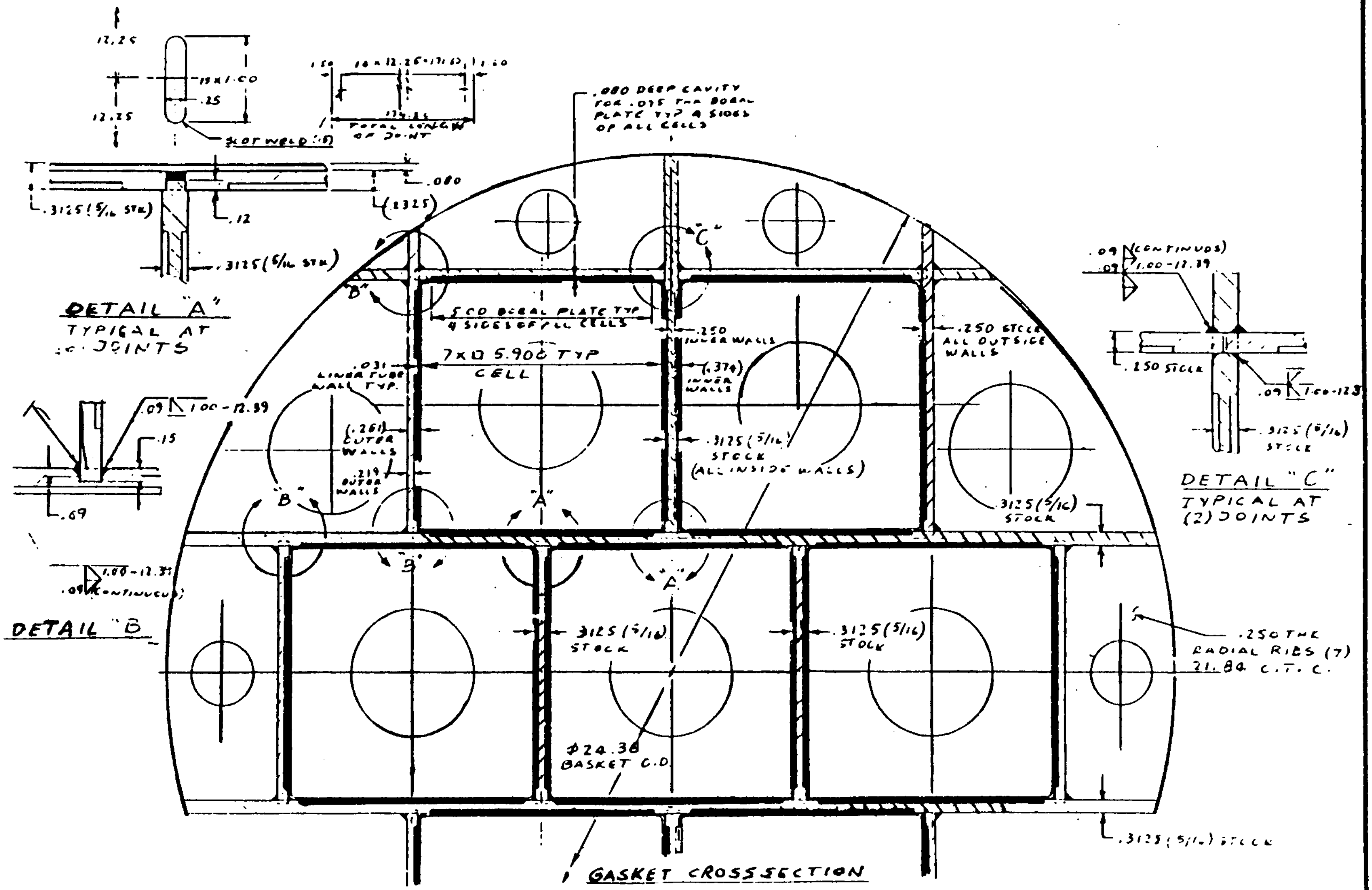


Figure 6.3-4. LWT Cask BWR Fuel Basket Nominal Dimensions

calculational models. The cask body, however, is modeled as a rectangular box having the same inside volume and material volumes as that of the right circular cylinder cask design. Figure 6.3-5 shows a schematic of the cask calculational model.

6.3.2 Package Regional Densities

The material densities (gm/cm^3) and the atomic number densities (atoms/barn-cm) for materials used in the calculational model are given in Table 6.3-1.

6.4 Criticality Calculation

6.4.1 Calculational Method

The calculational method employed to insure the criticality safety of fuel assemblies in the LWT cask uses the AMPX system of codes, References 6.4.1 and 6.4.2, for cross-section generation, and KENO IV, Reference 6.4.3, for reactivity determination.

The 227 energy group cross-section library that is the common starting point for all cross-sections used for the benchmarks and the shipping cask is generated from ENDF/B-V data (Reference 6.4.1). The NITAWL program (Reference 6.4-2) includes, in this library, the self-shielded resonance cross-sections that are appropriate for each particular geometry. The Nordheim Integral Treatment is used. Energy and spatial weighting of cross-sections is performed by the XSDRNPM program (Reference 6.4.2) which is a one-dimensional S_n transport theory code. These multigroup cross-section sets are then used as input to KENO IV (Reference 6.4.3) which is a three dimensional Monte Carlo theory program designed for reactivity calculations.

6.4.2 Fuel Loading Optimization

The following assumptions were used to develop the worst case KENO model for the cask using three PWR fuel assemblies per cask and seven BWR assemblies per cask:

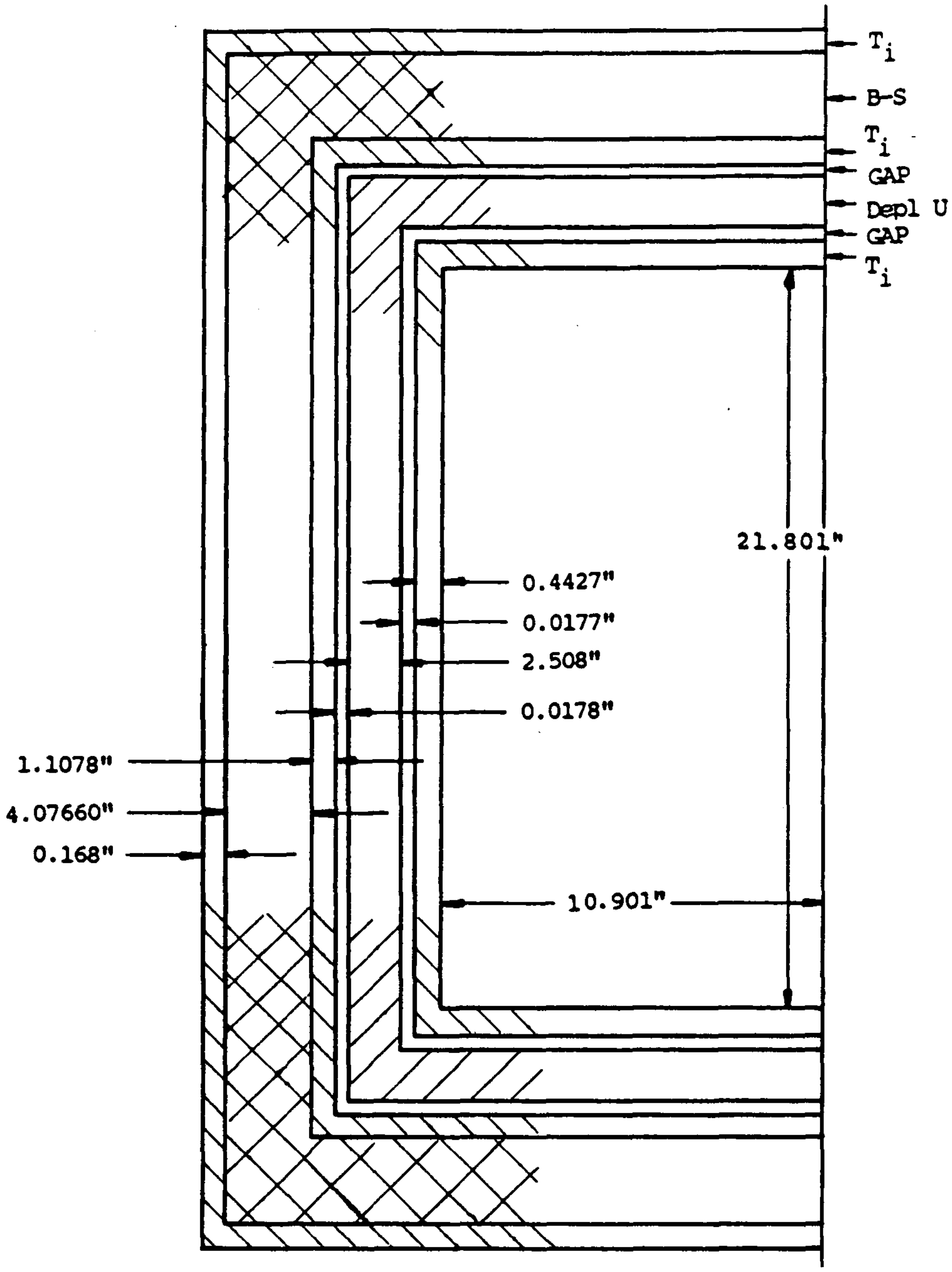


Figure 6.3-5 LWT Cask Calculational Model Dimensions

Table 6.3-1
Material Densities Employed in Criticality Analysis

Material	gm/cm^3	atoms/barns-cm
UO_2 (4.5 w/o U^{235})	10.52	
U^{235}		0.001069
U^{238}		0.022405
O		0.046949
Zircaloy (Zr)	6.55	0.043326
H_2O	1.00	
H		0.06685
O		0.03343
Boral	2.51	
B-10		0.009471
B-11		0.038203
C		0.011919
Al		0.045330
Boral Aluminum Clad (Al)	2.71	0.060485
Titanium (Ti)	4.50	0.05670
Depleted Uranium Metal	18.79	
U^{235}		0.000096
U^{238}		0.047330
C		0.000280
Boro-Silicone (less 25% for conservatism)	1.59	
B-10		0.00014
B-11		0.00057
H		0.03370
O		0.02110
Al		0.00502
Si		0.00449
C		0.00645
Stainless Steel	8.00	
Cr		0.015750
Mm		0.001754
Fe		0.058170
Ni		0.010259

1. Calculations of fuel assemblies in storage and shipping configurations have shown that the Westinghouse 17x17 OFA fuel and the GE 7x7 fuel assemblies yield a K_{eff} as high or higher than does any other PWR or BWR fuel assembly types listed in Section 6.2 when all fuel assemblies have the same U^{235} enrichment. Thus, the W 17x17 OFA fuel assembly was analyzed in the PWR cask basket and the GE 7x7 fuel assembly was analyzed in the BWR cask basket to determine the maximum cask reactivity.

2. All fuel rods contain uranium dioxide at an enrichment of 4.5 w/o U^{235} over the entire length of each rod.

3. No credit is taken for any U^{234}, U^{236} or burnable absorber in the fuel, nor is any credit taken for the buildup of fission product poison material.

4. The moderator is pure water at a temperature of 68°F. A conservative value of 1.0 gm/cm^3 is used for the full water density case. Calculations have shown that less than full water density will not result in a higher reactivity.

5. No credit is taken for any spacer grids or spacer sleeves.

6. The cask array is infinite in all directions which does not allow neutron leakage from the array.

7. The poison material loading in the Boro-Silicone shielding is reduced by 25 percent below its nominal loading.

8. A minimum poison material loading of 0.020 and 0.010 grams B-10 per square centimeter is used in the poison panels of the PWR and BWR fuel baskets respectively. This includes a 25% reduction in the nominal poison loading.

The maximum cask K_{eff} under Normal Conditions also includes asymmetric positioning of the fuel assemblies within the fuel basket such that all assemblies are shifted towards the center of the basket. This minimizes the separation between fuel assemblies in the basket and increases reactivity.

The maximum cask K_{eff} under accident conditions is equal to the maximum cask K_{eff} under Normal Conditions.due to the following conditions:

1. The Boro-Silicone and depleted uranium shielding reduces neutron leakage through the cask walls such that the cask reactivity is unaffected by the presence of any other loaded cask. As a result, the cask reactivity will remain unchanged whether one cask or an infinite number are placed together.

2. The fuel assemblies in the cask are modelled as close as possible in the nominal case. As a result any realistic change in the basket configuration will result in increased spacing between assemblies and a reduction in the cask reactivity.

3. A reduction in the cask volume will bring the neutron absorbing Boro-Silicone and depleted uranium shielding material closer to the fuel assemblies. This change will tend to reduce the cask reactivity. However for small changes (such as a 5% volume change) it will have an insignificant effect on the cask reactivity.

4. The presence of the poison material in the basket and cask design removes the conditions necessary for "optimum moderation" so that K_{eff} continually decreases as moderator density decreases from 1.0 gm/cm^3 to 0.0 gm/cm^3.

These conditions and model assumptions meet the requirements for fuel shipping casks under Normal and Accident Conditions as specified in Paragraphs 71.55 and 71.57 of 10 CFR Part 71.

6.4.3 Criticality Results

Based on the analysis described above, the following equation is used to develop the maximum K_{eff} for the cask:

$$K_{eff} = K_{worst} + B_{method} + B_{homo} + \quad [\ (ks)^2 worst + (ks)^2\ method\ ]^{1/2}$$

where:

K_{worst} = worst case KENO K_{eff} that includes close packed fuel assemblies

B_{method} = method bias determined from benchmark critical comparisons (See Section 6.5)

B_{home} = bias to account for homogenization of fuel assembly in KENO. (Only used in BWR fuel model, PWR fuel modelled explicitly.)

ks_{worst} = 95/95 uncertainty in the worst case KENO K_{eff}

ks_{method} = 95/95 uncertainty in the method bias (See Section 6.5)

Substituting calculated values in the order listed above, the result for the cask with PWR fuel in place is:

$$K_{eff} = 0.9341 + 0.0083 + \sqrt{[(0.0050)^2 + (0.0018)^2]} = 0.9477$$

Using the same equation as described above to develop the maximum K_{eff} for the cask with BWR fuel in place and substituting calculated values, the result is:

$$K_{eff} = 0.8018 + 0.0083 - 0.0117 + \sqrt{[(0.0033)^2 + (0.0018)^2\ ]} = 0.8022$$

Since K_{eff} is not greater than 0.95 including uncertainties at a 95/95 probability/confidence level, the acceptance criteria for criticality is met with fuel enriched to 4.5 w/o in the cask. The criticality analysis of the configuration shown in Figures 6.3-1 through 6.3-4 was performed in the fall of 1988. It was done early to confirm that flux traps would not be required and to establish the B^{10} loading requirements for the Boral. Both of these items could have affected the cask cavity size and hence the weight of the cask. Since the detailed analyses were performed, there have been modifications to the basket and cask body.

Table 6.4-1 summarizes the principal changes along with the effect that these changes would have on reactivity. Sensitivity studies and evaluations indicate that these changes will not cause the maximum K_{eff} to exceed 0.95. This, of course, will be confirmed during the final design phase.

Analysis of consolidated fuel storage configurations have shown that the consolidated fuel rod geometry is less reactive than the normal fuel assembly due to the significant reduction of the water or neutron moderation in the array.

6.5 Critical Benchmark Experiments

6.5.1 Benchmark Experiments and Applicability

A set of 33 critical experiments has been analyzed using the previously discussed calculational method to demonstrate its applicability to criticality analysis and to establish the method bias and variability. The experiments range from water moderated, oxide fuel arrays separated by various materials such as B4C, steel and water that simulate LWR fuel shipping and storage conditions (Reference 6.5.1) to dry, harder spectrum uranium metal cylinder arrays with various interspersed materials such as Plexiglas and air (Reference 6.5.2) that demonstrate the wide range of applicability of the method. Table 6.5-1 summarizes the results of these experiments.

6.5.2 Details of Benchmark Calculations

The 33 critical experiments used in the benchmarking calculations were obtained from the two critical experiment reports discussed above. All 21 of the oxide fuel array critical experiments in Reference 6.5.1 were included in the benchmark calculations while only 12 of the uranium metal cylinder array critical experiments in Reference 6.5.2 were included. All nuclear and

Table 6.4-1

Effect of Design Changes on Reactivity

	Orig.	Mod.	Effect on Reactivity
1. **Cask Body**			
Shell ID (In.)	24.60	23.76	(-)
Depleted Uranium Thickness (In.)	2.83	2.87	(-)
Boro-Silicone Thickness (In.)	4.60	4.20	(+)
Total Shell Thickness (In.)	9.27	9.07	
2. **PWR Basket**			
Total Wall Thickness (In.)	0.5	0.468/0.371	(+)
Boron Loading (gm B10/cm^2)	0.027	0.030	(-)
No. of Poison Panels	12	12	0
Cell ID (In.)	9.0	8.8	0
3. **BWR Basket**			
Total Wall Thickness (In.)	0.374	0.251/0.19	(+)
Boron Loading (gm B10/cm^2)	0.013	0.020	(-)
No. of Poison Panels	28	4	(++)
Cell ID (In.)	5.9	5.9	0

geometric input data used for each experiment and benchmark calculations are documented in complete detail in those reports.

6.5.3 Results of Benchmark Calculations

Table 6.5-1 summarizes the results for each of the benchmark calculations. The average K_{eff} of the benchmarks is 0.992. The standard deviation of the bias value is 0.0008 delta K. The 95/95 one sided tolerance limit factor for 33 values is 2.19. Thus, there is a 95 percent probability with a 95 percent confidence level that the uncertainty in reactivity, due to the method, is not greater than 0.0018 delta K.

6.6 Appendix

6.6.1 References

6.4.1 W. E. Ford III, CSRL-V: Processed ENDIFIB-V 227-Neutron-Group and Pointwise Cross-Section Libraries for Criticality Safety, Reactor and Shielding Studies, ORNL/CSD/TM-160, June 1982.

6.4.2 N. M. Greene, AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDFIB, ORNL/TM-3706, March 1976.

6.4.3 L. M. Petrie and N. F. Cross, KENO IV -- An Improved Monte Carlo Criticality Program, ORNL-4938, November 1975.

6.5.1 M. N. Baldwin, Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel, BAW-1484-7, July 1979.

6.5.2 J. T. Thomas, Critical Three-Dimensional Arrays of U(93.2) Metal Cylinders, Nuclear Science and Engineering, Volume 52, pages 350-350, 1973.

Table 6.5-1

Benchmark Critical Experiments

	General Description	Enrichment w/o U235	Reflector	Separating Material	Soluble Boron ppm	Keff
1.	UO2 rod lattice	2.46	water	water	0	0.9857 +/- .0028
2.	UO2 rod lattice	2.46	water	water	1037	0.9906 +/- .0018
3.	UO2 rod lattice	2.46	water	water	764	0.9896 +/- .0015
4.	UO2 rod lattice	2.46	water	B4C pins	0	0.9914 +/- .0025
5.	UO2 rod lattice	2.46	water	B4C pins	0	0.9891 +/- .0026
6.	UO2 rod lattice	2.46	water	B4C pins	0	0.9955 +/- .0020
7.	UO2 rod lattice	2.46	water	B4C pins	0	0.9889 +/- .0027
8.	UO2 rod lattice	2.46	water	B4C pins	0	0.9983 +/- .0025
9.	UO2 rod lattice	2.46	water	water	0	0.9931 +/- .0028
10.	UO2 rod lattice	2.46	water	water	143	0.9928 +/- .0025
11.	UO2 rod lattice	2.46	water	stainless steel	514	0.9967 +/- .0020
12.	UO2 rod lattice	2.46	water	stainless steel	217	0.9943 +/- .0019
13.	UO2 rod lattice	2.46	water	borated aluminum	15	0.9892 +/- .0023
14.	UO2 rod lattice	2.46	water	borated aluminum	92	0.9884 +/- .0023
15.	UO2 rod lattice	2.46	water	borated aluminum	395	0.9832 +/- .0021
16.	UO2 rod lattice	2.46	water	borated aluminum	121	0.9848 +/- .0024
17.	UO2 rod lattice	2.46	water	borated aluminum	487	0.9895 +/- .0020
18.	UO2 rod lattice	2.46	water	borated aluminum	197	0.9885 +/- .0022
19.	UO2 rod lattice	2.46	water	borated aluminum	634	0.9921 +/- .0019
20.	UO2 rod lattice	2.46	water	borated aluminum	320	0.9920 +/- .0020
21.	UO2 rod lattice	2.46	water	borated aluminum	72	0.9939 +/- .0020
22.	U metal cylinders	93.2	bare	air	0	0.9905 +/- .0020
23.	U metal cylinders	93.2	bare	air	0	0.9976 +/- .0020
24.	U metal cylinders	93.2	bare	air	0	0.9947 +/- .0025
25.	U metal cylinders	93.2	bare	air	0	0.9928 +/- .0019
26.	U metal cylinders	93.2	bare	air	0	0.9922 +/- .0026
27.	U metal cylinders	93.2	bare	air	0	0.9950 +/- .0027
28.	U metal cylinders	93.2	bare	plexiglass	0	0.9941 +/- .0030
29.	U metal cylinders	93.2	paraffin	plexiglass	0	0.9928 +/- .0041
30.	U metal cylinders	93.2	bare	plexiglass	0	0.9968 +/- .0018
31.	U metal cylinders	93.2	paraffin	plexiglass	0	1.0042 +/- .0019
32.	U metal cylinders	93.2	paraffin	plexiglass	0	0.9963 +/- .0030
33.	U metal cylinders	93.2	paraffin	plexiglass	0	0.9919 +/- .0032

References 6.5.1 and 6.5.2

7. OPERATING PROCEDURES

An outline of the operating procedures for the LWT cask system at both the reactor and the receiving facility is provided in this section. The information presented in this Section is preliminary and will be further developed later in the design process.

7.1 Procedures for Loading the Package

Loading the LWT Cask for transport involves (1) cask receipt and preparation, (2) cask immersion in pool, (3) fuel assembly loading, (4) cask closure, dewatering and removal from the pool, (5) inerting and decontamination, and (6) placement on the transporter. The total cask turnaround time at the reactor site is estimated to be 10.4 hours for PWR fuel and 11.4 hours for BWR fuel.

7.1.1 Cask Receipt and Preparation

This section delineates the procedure for cask receipt and preparation for spent fuel loading. Estimated process times are provided for each operation.

7.1.1.1 Perform health physics survey of trailer and personnel barrier (0.25 hr.).

7.1.1.2 Inspect trailer, tractor and personnel barrier for damage (0.20 hr.).

7.1.1.3 Unbolt four tie-down bolts securing the personnel barrier from the trailer bed. Utilizing an overhead crane and sling, lift the personnel barrier free and clear of the trailer and cask. Place personnel barrier in an appropriate accessible area for interim storage. (Estimated weight of personnel barrier is 300 pounds) (0.25 hr.).

7.1.1.4 Position trailer below crane main ways, set brakes and block wheels against movement in either direction (0.15 hr.).

7.1.1.5 Unbolt four 5/8-11 UNC bolts securing the impact limiter to the cask body. Withdraw the impact limiter free and clear from the cask. The impact limiter will remain on the trailer. Repeat the process for the rear impact limiter. (Estimated weight of impact limiter is 1250 pounds) (0.50 hr.).

7.1.1.6 Perform Health Physics smear test of trailer and cask. Document results (0.25 hr.).

7.1.1.7 Inspect cask and support system for damage. Any road dirt or contamination will be removed prior to immersion into the pool (0.25 hr.).

7.1.1.8 Unbolt front support system clamps (2 bolts each side). Flip clamps outboard until they rest on the supports (0.10 hr.).

7.1.1.9 Unbolt rear support system clamp bolt and pull the detent pin from each side. Swing the clamp towards the cask closure end and allow it to reach its rest position (180 degree swing) (0.10 hr.).

7.1.1.10 Using an overhead crane attach the lifting yoke assembly to the two top trunnions. No other trunnions are vertically accessible. Lock the tool onto the trunnions. Lift and rotate the cask to the vertical position and lift from the rear support system. (Estimated weight of the loaded cask is 50800 pounds without impact limiters and lifting the yoke assembly weighs 1500 pounds.) (0.25 hr.).

7.1.1.11 Transfer cask to the cask loading area/decontamination area (0.25 hr.).

7.1.1.12 Wash down cask surfaces to acceptable levels prior to entry into pool, if required (0.38 hr.).

7.1.2 Cask Immersion in Pool

7.1.2.1 Fill the cask with demineralized water to the level of the top of the fuel basket (0.25 hr.).

7.1.2.2 Using the lifting yoke assembly and overhead crane lift the cask to the pool area where cask is to be submerged. Use shroud or demineralized water bucket to minimize contamination of cask outer surfaces (0.25 hr.).

7.1.2.3 Submerge cask into pool. After seating the cask on the pool bottom and the lifting yoke assembly fully seated position indicator is on, disengage the lifting yoke assembly from the cask (0.05 hr.).

7.1.2.4 Lift the lifting yoke assembly from the pool and wash down all components as they exit the pool (0.15 hr.) (Optional Step).

7.1.2.5 Unbolt sixteen 1 3/8-6 UNC cask closure lid bolts. Remove the cask closure lid and place it such that seal damage will not occur. Inspect the seals for damage. (Estimated weight of the cask closure lid is 1820 pounds) (0.25 hr.).

7.1.3 Fuel Assembly Loading

7.1.3.1 Install the fuel assembly lead-in fixture to the top of the cask (0.15 hr.).

7.1.3.2 Verify the fuel assembly basket and fuel assembly spacers and are in-place using the dummy fuel assembly go, no-go fixture check that proper insertion is possible. Use the spent fuel pool bridge crane to perform this operation. After checking each location and when in conformance proceed with operations (0.38 hr.) (Optional Step).

7.1.3.3 Using the spent fuel pool bridge crane, load either 3 PWR or 7 BWR spent fuel assemblies into the cask. Visually verify that fuel assemblies are fully inserted. Use of an underwater camera and light is optional if available.

7.1.3.4 Remove the fuel assembly lead-in fixture. As the tool is removed from the pool wash down all associated components (0.15 hr.).

7.1.4 Cask Closure Through Removal from Pool

7.1.4.1 Perform visual inspection to assure that no obstruction or debris are present on the cask closure lid flange surface. Remove drain port plugs from closure lid (0.15 hr.).

7.1.4.2 Position the lid by aligning the match marks on the cask body and lid head. Allow the guide pins to become fully inserted prior to releasing the lid (0.25 hr.).

7.1.4.3 Hand tighten sixteen 1 3/8-6 UNC closure lid bolts to secre the lid in place.

7.1.4.4 Using the overhead crane, attach the lifting yoke assembly to the cask trunnions and lock the tool arms. Position feedback is provided when the lifting yoke assembly is fully seated and when the arms are locked in place over the trunnions (0.25 hr.).

7.1.4.5 Raise the lifting yoke assembly slowly 3 to 4 inches until the lifting yoke assembly is supporting the weight of the cask. The fully seated light indicator will go out and only the arm locked position indicator should be displayed (0.05 hr.).

7.1.4.6 Raise the cask from the pool, washing down all components as the emerge from the pool (0.25 hr.).

7.1.4.7 Torque sixteen 1 3/8-6 UNC closure lid bolts to 2100 to 2300 ft-lbs (0.50 hr.).

7.1.4.8 Attach the pressurization line to the cask closure lid, and attach a hose to the drain port perform gross check of cask seals and dewater the cask through the drain line on the lid. Attach the vacuum line to the cask closure lid to dry and evacuate the cask internal cavity (1.0 hr.).

7.1.4.9 Backfill the cask with helium gas and fill the cask to atmospheric pressure (0.25 hr.).

7.1.5 Decontamination

7.1.5.1 Perform leak test of the cask closure lid and penetration seals (0.30 hr.).

7.1.5.2 Radiation monitor the package per 49 CFR 173.441 requirements and verify that surface contamination levels meet the requirements of 49 CRF 173.443 (0.38 hr.).

7.1.6 Placement onto Transport Means

7.1.6.1 Transfer the cask to the transporter location. Lower the cask into the rear support system trunnion saddle. Rotate the cask to the horizontal position. Secure the trunnion clamps and install the impact limiters and personnel barrier as previously described (1.2 hr.).

7.1.6.2 Survey the truck, cask and personnel barrier per DOT Regulation 173.441 and 173.443. Visually inspect the truck and supports system welds. Complete all shipping manifests (0.38 hr.).

7.1.6.3 Check transporter and cask for proper DOT labeling and placarding. Release truck from site (0.05 hr.).

7.2 Procedure for Unloading the Package

In general, the procedure for unloading the cask is the reverse of those described above. Since the unloading process is to be performed in a dry, hot cell environment, no dewatering will occur. Prior to cask closure lid removal, a gas sample shall be taken for analysis. Upon examination and acceptable contamination levels established the lid can be remotely removed. The cask unloading shall take place below a hot cell with the cask physically mated to a cell port. The mating will be accomplished using the seal ring illustrated in Figure 7.2-1. The estimated time to unload the cask is 6.85 hours for PWR fuel and 7.85 hours for BWR fuel. The following represent the operations required for cask unloading at the receiving facility and their associated estimated process time.

7.2.1 Transport cask to fuel building, remove personnel barrier, release tiedowns (0.45 hr.)

7.2.2 Perform receipt inspection, radiation surveys.

7.2.3 Remove Impact Limiters and collect swipe samples (0.50 hr.)

7.2.4 Unload cask in decon area (0.25 hr.)

7.2.5 Clean cask (0.50 hr.)

7.2.6 Install cask seal ring (0.15 hr.)

7.2.7 Move cask to hot cell and mate cask to cell port (0.25 hr.)

7.2.8 Remove hot cell port plug. Open gas sampling port on cask lid and perform gas sampling. (0.50 hr.)

7.2.9 Remove closure lid (0.25 hr.)

7.2.10 Install fuel basket lead-in fixture (PWR or BWR) and crud barrier (0.25 hr.)

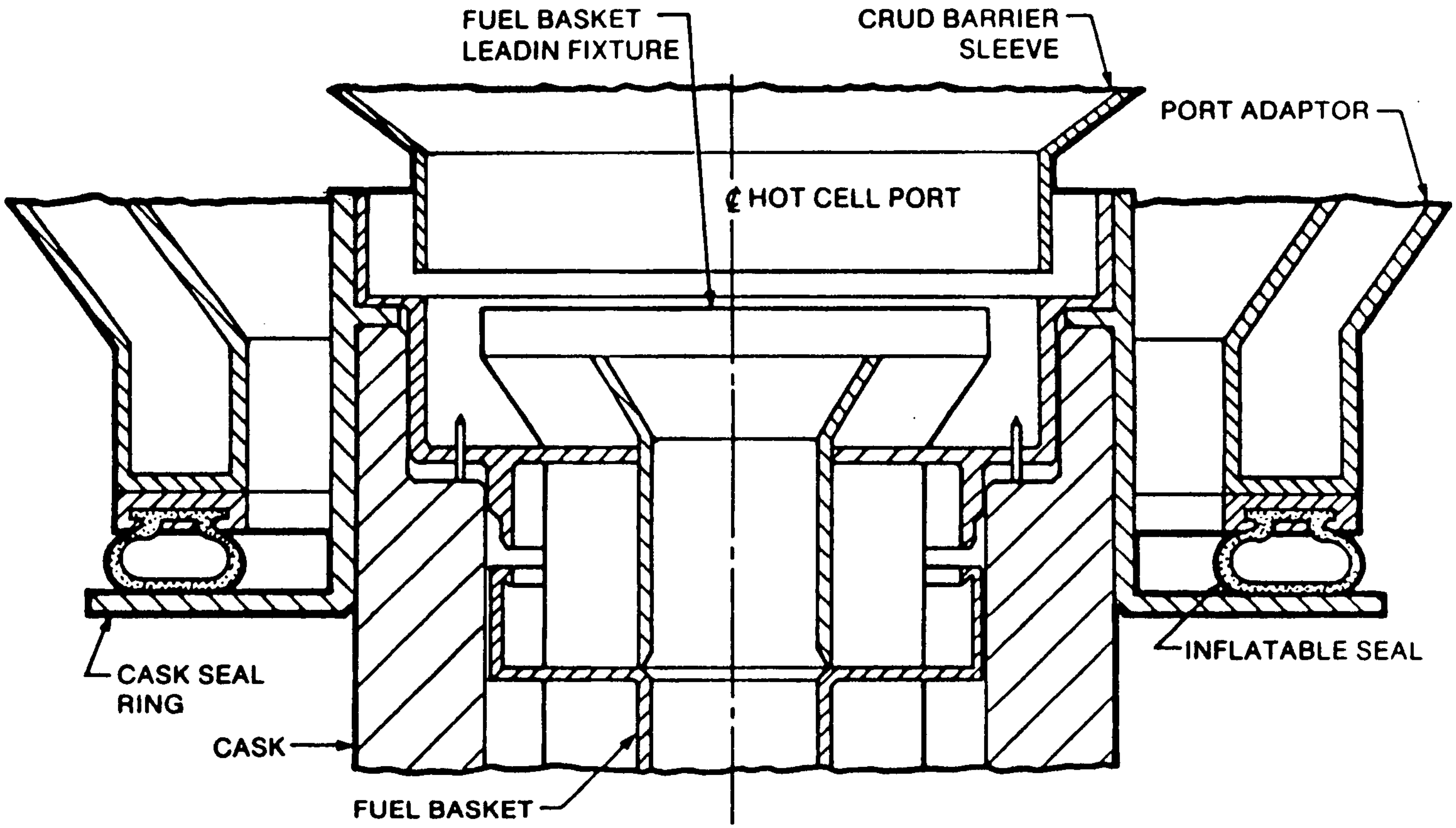


Figure 7.2-1 Cask Seal Ring

7.2.11 Fuel unloading (3 PWRs or 7 BWRs) (0.75 hr. PWR, 1.75 BWR)

7.2.12 Remove fuel basket lead-in and crud barrier and check/clean seal surface (0.25 hr.)

7.2.13 Inspect lid seals and install closure lid, torque bolts to 2100-2300 ft-lbs. (0.50 hr.)

7.2.14 Install hot cell port plug and move cask to decon area (0.25 hr.)

7.2.15 Wash down cask exterior (0.25 hr.)

7.2.16 Perform leak test of cask seals (0.30 hr.)

7.2.17 Load cask on transporter, tiedown cask, and install impact limiters and personnel barrier (1.2 hr.)

7.3 Preparation of an Empty Package for Transport

Casks which have been used to transport spent fuel and have been unloaded are handled per the requirements of 49 CFR 173.427.

7.4 Intermodal Transfer

Transfer of the cask from one shipping mode to another (e.g., truck to rail) involves (1) removal of the personnel barrier and release of the cask tie-down devices on the trailer support system, (2) with the cask in a horizontal position, and the impact limiters attached, vertically raising the cask, (3) placement of the cask onto the intermodal transfer skid and (4) securing the tie-down devices of the skid to the cask trunnions. For the following operations it is assumed that the skid has been secured to a rail car and transfer is between a truck and rail car. Based upon this assumption, the estimated transfer time is 2.0 hours.

7.4.1 Perform health physics survey of trailer and personnel barrier (0.25 hr.).

7.4.2 Inspect trailer, tractor and personnel barrier for damage (0.20 hr.).

7.4.3 Unbolt four tie-down bolts securing the personnel barrier from the trailer bed. Utilizing an overhead crane and sling lift the personnel barrier free and clear of the trailer and cask. Place personnel barrier in an appropriate accessible area for interim storage. (Estimated weight of personnel barrier is 300 pounds) (0.25 hr.).

7.4.4 Position trailer near jib crane, set brakes and block wheels against movement in either direction (0.05 hr.).

7.4.5 Perform Health Physics smear test of trailer and cask. Document results (0.25 hr.).

7.4.6 Inspect cask and support system for damage (0.10 hr.).

7.4.7 Unbolt front support system clamps (2 bolts each side). Flip clamps outboard until they rest on the supports (0.10 hr.).

7.4.8 Unbolt rear support system clamp bolt and pull the detent pin from each side. Swing the clamp towards the cask closure end and allow it to reach its rest position (180 degree swing) (0.10 hr.).

7.4.9 Attach four shackles to each of the four cask trunnions used for cask tie down during transport. Place the shackle pin through the trunnion flange hole and secure the pin to shackle (0.10 hr.).

7.4.10 Lift the cask in the horizontal position (0.05 hr.).

7.4.11 Using the overhead crane, transfer the cask to the intermodal transfer skid. Align the cask over top of the supports on the skid and slowly lower the cask. The bottom trunnions (opposite the closure end) will engage first. Allow the cask to fully seat in the front saddle and remove the four shackles from the trunnions (0.15 hr.).

7.4.12 Secure the trunnion clamps as previously described (0.20 hr.).

7.4.13 Using the overhead crane, install the personnel barrier onto the rail car bed. Bolt the four tie-down bolts to the trailer bed to secure the personnel barrier to the bed.

7.4.14 Survey the rail car and cask per DOT Regulation 173.441 and 173.443. Visually inspect the railcar and support system welds. Complete all shipping manifests (0.15 hr.).

7.4.15 Check railcar and cask for proper DOT labeling and placarding. Release the shipment for transport (0.05 hr.).

7.4.16 Unloading the package will follow the procedure outlined in Section 7.2.

7.5 Fuel Basket Replacement

The TITAN LWT cask is provided with interchangeable fuel baskets to accept either 3 PWR or 7 BWR spent fuel assemblies. The following is an outline of the procedure to change out one fuel basket for another. It is envisioned that the basket changeout will be performed in a cask maintenance facility. Once the cask is opened (cask closure lid removed) the estimated time for fuel basket replacement is 1.4 hours for PWR basket replacement and 2.2 hours for BWR basket replacement.

7.5.1 Removal of the impact limiters, cask upending, handling and closure lid removal will be performed as previously described. It is recommended that fuel basket replacement be performed in a dry environment.

7.5.2 Remove any fuel spacers located within each fuel assembly storage location (0.30 hr. for PWR and 0.70 hr. for BWR baskets).

7.5.3 Grip the fuel basket inner ring and vertically lift the basket from the cask. (The estimated basket weights are 1685 lbs. (PWR) and 1575 lbs. (BWR)) (0.50 hr.).

7.5.4 For basket replacement, align the basket and cask match marks as this will orient the fuel basket keyway with the key on the cask ID (0.15 hr.).

7.5.5 Slowly lower the basket until the basket is fully seated within the cask cavity. Install fuel spacers. Cask is now ready for fuel loading operations (0.45 hr. for PWR and 0.85 hr. for BWR baskets).

8. TECHNICAL CERTIFICATION ISSUES REQUIRING NRC RESOLUTION

This section presents the current status with regard to resolution of technical certification issues that have been raised by the Transportation Branch of the Nuclear Regulatory Commission relative to the TITAN LWT cask design.

8.1 Current Open Issues

Westinghouse has had three meetings with the NRC on the TITAN LWT cask. Those meetings were held on May 24, 1988, November 15, 1988 and May 25, 1989. During those interactions with the Transportation Branch, several issues were raised which have already been resolved but two significant issues remain to be resolved.

8.1.1 Titanium as the Structural Material for the Cask

The NRC considers the use of titanium for the structural material of the TITAN LWT cask as the central issue as this material has no precedent in transportation cask certification. They will require information on the weldability, fabricability, corrosion resistance, and fracture toughness of this material; resolution of whether current ASME Code, Section III rules for establishing allowable stress limits are applicable to the material which has a relatively high yield-to-ultimate strength ratio, and relatively low modulus of elasticity. In addition the NRC has taken the following specific positions:

a. ASME approval of Grade 9 titanium should be sought for use under Section III, Subsection NB. An ASME specification for the material (i.e., Section II of the ASME B&PV Code) should be obtained.

b. Westinghouse should consider NUREG/CR 1815 and NUREG/CR 3826 for guidance on fracture mechanics/fracture toughness properties. It must be shown that Grade 9 titanium will arrest a crack and that initial crack lengths 3 to 4 times the material thickness will not result in brittle fracture.

Response: Westinghouse has responded to this issue through several actions aimed at gaining approval for the use of the material. These include gathering available property data for Grade 9 titanium, submitting a request for an ASME Code Case which would establish the Grade 9 titanium as a suitable material for use under Section III of the B&PV Code, having an independent team review the suitability of the use of the material for cask structures, and developing a test program which will provide all the remaining physical and mechanical property data required for certification.

a. Physical and Mechanical Properties: On November 1, 1988, Westinghouse forwarded to the NRC a summary of properties for Grade 9 titanium. It was recognized at the time that the available information was not complete and that a materials test program would be required to provide the requisite information. The property data sent to the NRC are included as Appendix 8.3.1.

b. ASME approval: A Code Case was submitted on January 13, 1989 to the ASME for approval of Grade 9 titanium for use in Section III, Division 1, Classes 1, 2 and 3 construction. The inquiry and draft response was considered by the Subcommittee on Material Specifications (SC II) on May 2, 1989 and the inquiry was then sent to the Subgroup on Materials, Fabrication, and Examination for Section III. This group approved the inquiry on September 13, 1989. This material will be acceptable for use under Section III of the Code when acted upon by the Subcommittee on Nuclear Power (SC III) and the ASME Code Main Committee. The inquiry is included in Section 2, Appendix 2.10.5.

c. Independent review: A team of ASME Code experts and titanium material experts was assembled to address NRC concerns regarding the material characteristics in two areas: 1) the appropriateness of applying ASME Code Section III criteria for allowable stress limits for Grade 9 titanium which had ultimate tensile strength and yield strength values in close proximity to one another, and also had modulus of elasticity and elongation values that were roughly half those for stainless steel, and 2) the ability of the alloy to meet the fracture toughness requirements stipulated by the NRC for transportation casks.

The team included Dr. William Cooper, Teledyne Engineering Services Co., chairman; Dr. Sumio Yukawa, consultant; Mr. David Thomas, RMI Company; Mr. Terry Webster, Teledyne Wah-Chang Albany; and Dr. John Stevens, Sandia National Laboratories. Dr. Cooper and Dr. Yukawa were selected as recognized authorities on the ASME Code criteria and for Dr. Yukawa's expertise in fracture toughness. Dr. Yukawa is also the Chairman of the ASME NUPACK Committee on materials. Mr. Thomas and Mr. Webster were included for their extensive experience in the development of Grade 9 titanium material property data and knowledge of titanium alloys. Dr. Stevens was selected for his understanding of the transportation cask requirements and experience with titanium alloys.

The team conducted their review on June 21 - 23, 1989 at the Westinghouse offices in Pittsburgh. On June 23, they issued "Report on the Review of the Suitability of Grade 9 titanium for the Legal Weight Truck Cask." (See Appendix 8.3.2.) The team concluded, "...it is our collective opinion that Grade 9 titanium is a suitable material for use in transportation casks which must meet present NRC requirements and guidance." The team recommended obtaining additional information in several areas. These areas are:

- Uniform elongation data from tensile tests up to 300°F.

- Tensile properties of welds and heat affected zones for weldments made with expected welding practices.

- Low cycle fatigue data at temperatures to 300°F.

- o Creep data for weld and heat affected zone materials.

The team further concluded that it was their expectation that the additional information would reinforce their opinion that Grade 9 titanium is suitable for spent fuel transportation casks.

d. Materials test program: A materials test program has been developed to provide all needed material property data, both physical and mechanical. This program will provide the data recommended by the review team as well as other property data required to support the thermal and structural evaluations of the cask. The Grade 9 titanium test program is included in Appendix 8.3.3.

When the test data become available (in approximately 6 months) a meeting will be held with the NRC staff which will be devoted exclusively to the titanium issue. Westinghouse is confident that the test data in conjunction with the approval for use by the ASME Section III committee will alleviate the NRC concerns about the suitability of the material for the cask structures.

8.1.2 Aluminum Honeycomb Impact Limiters

The NRC will require testing of the aluminum honeycomb impact limiters to verify the load-deflection characteristics used in analysis of free drop accidents. They have requested data on the strength and aging characteristics of the adhesive and whether it is stronger than the aluminum. The NRC staff has expressed an interest in data from the following tests:

- o 1/2 scale drop tests including end, side, c.g. over corner and an oblique drop. The staff wishes to review the information going into the selection of the particular angle to be chosen for the oblique drop. Drawings of the test article must be produced so that it can be demonstrated that they were an accurate simulation of the full-scale design.

- o tests to determine the adhesive's resistance to shearing.

The staff expressed the feeling that the 1/4 scale tests (which will precede the tests of 1/2 scale prototypic impact limiters mounted on a simulated cask made of solid steel) should be made with curved (i.e., scaled) honeycomb sections.

Response: Westinghouse recognizes that the performance of the aluminum honeycomb impact limiters must be experimentally determined. A two-phase test program has been developed to produce the requisite information. The test plan incorporates all of the aspects of concern to the NRC. Westinghouse expects that the test models will demonstrate the viability of the impact limiter design and confirm the crush characteristics used for the preliminary design evaluation. The plan for Phase I of the impact limiter tests is included in Appendix 8.3.4.

8.2 Resolved Issues

In addition to these two issues which have not been resolved because planned tests have not been completed, there have been several other issues which were raised during the meetings with the NRC. These issues have been essentially resolved but are mentioned here for completeness.

8.2.1 Treatment of Depleted Uranium in the Structural Evaluation

The NRC position is that if the strength of the DU would be of benefit, to ignore it. If, on the other hand, the presence of the DU introduces additional loads (such as those that may exist as a consequence of differential thermal expansion) it must be considered.

Resolution: Westinghouse understands the NRC position and structural evaluations have and will adhere to this position.

8.2.2 Scale Model Verification Tests

The NRC initially expressed a preference for full scale testing as they felt that such testing would gain better public acceptance. The NRC has since accepted the approach of using a 1/2 scale model for design verification testing. They have asked to review the specifics of the test program, and will expect to see drops on the end, side, "c.g." over a corner and an oblique drop. The staff wishes to review the information supporting the selection of the oblique drop angle.

Resolution: The tests demonstrating compliance with the free drop and puncture tests of the set of hypothetical accident conditions will use a 1/2 scale model test article. Compliance with the thermal (fire accident) and immersion tests will be by analysis. Prior to finalizing the test plan, a meeting will be held with the NRC to provide agreement on the specifics of the drop and puncture tests.

8.2.3 Structural Analysis of the Free Drop Accidents

The NRC has taken the position that analysis of the impacts associated with the free drop accidents must include dynamic effects. It is not sufficient to simply apply the impact g loads to static finite analysis models.

Resolution: The NRC has accepted the approach of applying dynamic amplification factors to static analyses using detailed finite analysis models. It was further agreed that the dynamic amplification factors can be determined by ratioing the force and moment results from dynamic and static analyses of the cask using the SCANS code.

8.3 Appendices

This section contains information on the following:

- Grade 9 titanium physical and mechanical properties
- Independent review team report of Grade 9 titanium for use in the TITAN LWT cask
- Grade 9 titanium test program
- Impact limiter test program

8.3.1 Grade 9 Titanium Physical and Mechanical Properties

This appendix provides the physical and mechanical data forwarded to the NRC on November 1, 1988.

SUMMARY OF PROPERTIES FOR TITANIUM ALLOY 3Al-2.5V (ASTM GRADE 9)

The main structural material for the Titan cask will be Ti 3Al 2.5V which is referred to in ASTM specifications as Grade 9. The purpose of this summary is to document the status of the physical and mechanical properties of this material.

The material forms to be used in the cask include sheet, plate, forgings, and welding fittings. It is planned that the heat treatment shall consist of an intermediate-temperature alpha-beta anneal (1475 degrees F, 30 minutes) followed by an air cool.

The chemical compositions of the applicable ASTM specifications for Grade 9 are given in Table 1.

The properties of interest are:

1. Physical properties
    - Density
    - Specific heat
    - Thermal Conductivity
    - Thermal Diffusivity
    - Emmisivity

2. Mechanical properties
    - Tensile Strength
    - Yield Strength
    - Modulus of Elasticity
    - Poisson's Ratio
    - Coefficient of thermal expansion
    - Fracture toughness

In addition to the data given on the following pages and tables, Westinghouse will be contracting for services to provide test data to supplement the data on the Ti-3Al-2.5V alloy that are already in hand.

PHYSICAL PROPERTIES OF

Ti-3Al-2.5V ALLOY

1. Density (lb/in^3 (g/cm^3)) 0.162 (4.48)

   Reference: Metals Handbook, 9th Edition, American Society of Metals, page 399.

2. Melting Point (degrees F (degrees C)) 3100 (1704)

   Reference: Metals Handbook, 9th Edition, American Society of Metals, page 400.

3. Phase Transformation Temperature (degrees F (degrees C)) 1715 (935)

   Reference: Same as item 2, above.

4. Specific Heat see Table 2

5. Thermal Conductivity see Table 3

6. Thermal Diffusivity see Table 4

7. Emmisivity

   Emissivity 0.2

   Solar Absorptivity 0.8

   Reference: Handbook of Heat Transfer, Warren M. Rohsenow and James P. Hartnett, McGraw-Hill Book Co., 1973, page 3-22, Table 4.

MECHANICAL PROPERTIES OF

Ti-3Al-2.5V ALLOY

1. Room temperature tensile properties:

   Minimum values (as given in the ASTM specifications) for room temperature tensile strength, yield strength, elongation and reduction in area for the various applicable product forms is given in Table 5.

2. Tensile properties as a function of temperature:

   Tensile, yield and allowable stress (Sm) values as a function of temperature, are given in Table 6. Except for the allowable stress, the values are identical to those for the Code Case which applies to Section VIII of the ASME B&PV Code (Table 7). An application for adoption of a code case applicable to Section III will be initiated with the values shown in the table as the proposed values. It is proposed that the values at room temperature be used for temperatures below room temperature.

3. Modulus of Elasticity:

   The room temperature Modulus of Elasticity is 15.0 million psi and at 1450 degrees F, the Modulus of Elasticity is 14.0 million psi.

   Reference: "Ti 3Al 2.5V Seamless Tubing Engineering Guide," Second Edition, Clyde E. Forney, Jr. and John H. Schemel, Sandvik Special Metals Corporation, May 1987.

   The Metals Handbook, 9th Edition, pp 400 also gives a value of 15 million psi for the elastic modulus.

4. Poisson's Ratio: 0.31

   Reference: Technical Report DTNSRDC/SME-81/18, "Investigation of Ti-3Al-2.5V for Seawater Piping Applications," by Robert E. Maerch and Ivan L. Caplan, David W. Taylor Naval Ship R&D Center, June 1981 page 4, Table 2. (Third Party Proprietary)

5. Coefficient of Thermal Expansion:

   The linear coefficient of thermal expansion from room temperature to various temperatures is given in Table 8.

6. Fracture Toughness

Charpy V-notch impact energy data for extruded plates in the alpha-beta annealed condition:

Test Temperature (degrees F)	Charpy Impact Energy (ft-lb)
200	86
RT	75
32	64
-80	51

The fracture properties at room temperature are:

J-IC (in-lb/in^2)	779
K-Ic (equivalent) (ksi-in^.5)	115

Reference: Technical Report DTNSRDC/SME-81/18, "Investigation of Ti-3Al-2.5V for Seawater Piping Applications," Robert E. Maersch and Ivan L. Caplan, David W. Taylor Naval Ship R&D Center, June, 1981, pages 12 and 13, Tables 7 and 9. (Third party proprietary)

TABLE 1

CHEMICAL COMPOSITION OF

Ti-3Al-2.5V (ASTM Grade 9)

ELEMENT	Composition, % ASTM Standard Specification B 265 (Sheet/ Plate)	B 348 (Bar & Billet)	B 381 (Forging) (Gr F-9)	B 337 (Pipe)	B 363 (Welding Fittings)
Nitrogen, max	0.02	0.02	0.02	0.02	0.02
Carbon, max	0.010	0.05	0.05	0.05	0.01
Hydrogen, max	0.015	0.0125	0.015	0.013	0.015
Iron, max	0.25	0.25	0.25	0.25	0.25
Oxygen, max	0.15	0.12	0.12	0.12	0.15
Aluminum	2.5-3.5	2.5-3.5	2.5-3.5	2.5-3.5	2.5-3.5
Vanadium	2.0-3.0	2.0-3.0	2.0-3.0	2.0-3.0	2.0-3.0
Residuals (each)	0.1	0.1	0.1	0.1	0.1
Residuals (total)	0.4	0.4	0.4	0.4	0.4
Titanium	remainder	remainder	remainder	remainder	remainder

Note: 1. Values for B 265, Grade 9, are proposed values that are expected to be published by ASTM in mid 1989

2. Max hydrogen for billets (B 348) is 0.0100 %

3. Permissible raw materials for welding fittings include pipe, plate, bar and billet, and forgings. Values for plate are shown.

TABLE 2
SPECIFIC HEAT OF TITANIUM

TEMP (C)	TEMP (F)	(1) UNALLOYED TITANIUM Cp J/kg·K	Cp Btu/lb·F	(1) Ti·6Al·4V Cp J/kg·K	Cp Btu/lb·F	(2) Ti·3Al·2.5V Cp J/kg·K	Cp Btu/lb·F
20	68	520	0.12	580	0.14	550	0.13
205	400	560	0.13	610	0.15	585	0.14
425	800	628	0.15	670	0.16	649	0.16
650	1200	720	0.17	760	0.18	740	0.18
870	1600	810	0.19	930	0.22	870	0.21

NOTES: (1) DATA FROM METALS HANDBOOK, 9TH EDITION, AMERICAN SOCIETY OF METALS
(2) AVERAGE BETWEEN THE VALUES FOR UNALLOYED TI AND Ti·6Al·4V

TABLE 3
THERMAL CONDUCTIVITY OF TITANIUM

		(1) UNALLOYED TITANIUM		(2) Ti-6Al-4V		(3) Ti-3Al-2.5V		(4) Ti-3Al-2.5V
TEMP (C)	TEMP (F)	k W/m-K	k B/Ft-Hr-F	k W/m-K	k B/Ft-Hr-F	k W/m-K	k B/Ft-Hr-F	k B/Ft-Hr-F
20	68	21.9	12.7	6.6	3.8	7.6	4.4	4.8
93	200	20.8	12.0	7.3	4.2	8.1	4.7	5.3
205	400	19.8	11.5	9.1	5.3	9.2	5.3	6.2
315	600	19.4	11.2	10.6	6.1	10.5	6.1	6.8
425	800	19.4	11.2	12.6	7.3	12.4	7.1	
540	1000	19.8	11.4	14.6	8.4	14.1	8.1	
650	1200	20.1	11.6	17.5	10.1	16.1	9.3	

NOTES:
(1) ASME BOILER AND PRESSURE VESSEL CODE, SECTION III DIV 1 APPENDIX I, TABLE I-4.0
(2) DATA FROM METALS HANDBOOK, 9TH EDITION, AMERICAN SOCIETY OF METALS
(3) FROM AEROSPACE STRUCTURAL MATERIALS HANDBOOK, MARCH, 1980, CODE 3725, FIG 2.013
(4) DATA FROM "Ti 3Al 2.5V SEAMLESS TUBING ENGINEERING GUIDE", 2ND EDITION CLYDE E. FORNEY, JR. AND JOHN H. SCHEMEL, SANDVIK SPECIAL METALS CORP, MAY 1987

TABLE 4
THERMAL DIFFUSIVITY OF TITANIUM

TEMP (C)	TEMP (F)	(1) UNALLOYED TITANIUM (FT^2/HR)	(2) Ti-6Al-4V (FT^2/HR)	(3) Ti-3Al-2.5V (FT^2/HR)	(1) 18Cr-8Ni (FT^2/HR)
20	68	0.359	0.096	0.121	0.151
93	200	0.331	0.103	0.125	0.156
205	400	0.300	0.125	0.136	0.165
315	600	0.283	0.140	0.145	0.174
425	800	0.275	0.162	0.159	0.184
540	1000	0.271	0.175	0.171	0.194
650	1200	0.270	0.199	0.185	0.203

NOTES: (1) FROM ASME B&PV CODE, SECTION III, DIV 1, APPENDIX I, TABLE I-4.0
(2) FROM METALS HANDBOOK, 9TH EDITION, AMERICAN SOCIETY OF METALS
(3) BASED ON SPECIFIC HEAT FROM TABLE 2 AND THERMAL CONDUCTIVITY FROM TABLE 3

TABLE 5

ASTM PROPERTIES FOR TITANIUM
GRADE 9

SPECIFICATION DESIGNATION	TITLE	TENSILE PROPERTIES: TENSILE STRENGTH (ksi)	YIELD STRENGTH (ksi)	ELONGAT'N MIN (%)	REDUCT'N IN AREA MIN, (%)
B 265	TITANIUM AND TITANIUM ALLOY STRIP, SHEET, AND PLATE	90	70	15	
B 348	TITANIUM AND TITANIUM ALLOY BARS AND BILLETS	90	70	15	25
B 381 (GR F-9)	TITANIUM AND TITANIUM ALLOY FORGINGS	90	70	15	25
B 363	SEAMLESS AND WELDED UNALLOYED TITANIUM AND TITANIUM ALLOY WELDING FITTINGS	90	70	15	

Notes:
1. Values given for B 265 are proposed values expected to be approved mid-1989
2. Reduction in area is not given for strip, sheet and plate. Bend test requirements are 5T for material under 0.070" and 6T for material 0.070" to 0.187". Bend test is not applicable for material over 0.187" in thickness.

TABLE 6

TITANIUM GRADE 9

SECTION III ALLOWABLE STRESSES
(TO BE PROPOSED AS ASME SECTION III CODE CASE)

TEMP	YIELD STRENGTH (KSI)			TENSILE STRENGTH (KSI)			ALLOWABLE
(DEGREES F)	RATIO	MPV	2/3 MPV	RATIO	MPV	1/3 MPV	(KSI)
RT	1.00	70.0	46.7	1.00	90.0	30.0	30.0
100	0.97	67.9	45.3	0.97	87.3	29.1	29.1
150	0.93	65.1	43.4	0.93	83.7	27.9	27.9
200	0.88	61.6	41.1	0.88	79.2	26.4	26.4
250	0.83	58.1	38.7	0.84	75.6	25.2	25.2
300	0.79	55.3	36.9	0.80	72.0	24.0	24.0
350	0.75	52.5	35.0	0.75	67.5	22.5	22.5
400	0.71	49.7	33.1	0.71	63.9	21.3	21.3
450	0.67	46.9	31.3	0.68	61.2	20.4	20.4
500	0.64	44.8	29.9	0.64	57.6	19.2	19.2
550	0.62	43.4	28.9	0.62	55.8	18.6	18.6
600	0.59	41.3	27.5	0.61	54.9	18.3	18.3
650	0.56	39.2	26.1	0.60	54.0	18.0	18.0
700	0.56	39.2	26.1	0.59	53.1	17.7	17.7

NOTE: See Section 2.10.5, page 2-247 for corrections to allowable stress intensity

TABLE 7

TITANIUM GRADE 9

SECTION VIII ALLOWABLE STRESSES

TEMP (DEGREES F)	YIELD STRENGTH (KSI)			TENSILE STRENGTH (KSI)			ALLOWABLE (KSI)
	RATIO	MPV	2/3 MPV	RATIO	1.1 MPV	1.1/4 MPV	
RT	1.00	70.0	46.7	1.00	90.0	22.5	22.5
100	0.97	67.9	45.3	0.97	90.0	22.5	22.5
150	0.93	65.1	43.4	0.93	90.0	22.5	22.5
200	0.88	61.6	41.1	0.88	87.1	21.8	21.8
250	0.83	58.1	38.7	0.84	83.2	20.8	20.8
300	0.79	55.3	36.9	0.80	79.2	19.8	19.8
350	0.75	52.5	35.0	0.75	74.3	18.6	18.6
400	0.71	49.7	33.1	0.71	70.3	17.6	17.6
450	0.67	46.9	31.3	0.68	67.3	16.8	16.8
500	0.64	44.8	29.9	0.64	63.4	15.8	15.8
550	0.62	43.4	28.9	0.62	61.4	15.3	15.3
600	0.59	41.3	27.5	0.61	60.4	15.1	15.1
650	0.56	39.2	26.1	0.60	59.4	14.9	14.9
700	0.56	39.2	26.1	0.59	58.4	14.6	14.6

REFERENCE: LETTER, R. T. WEBSTER, TELEDYNE WAH CHANG TO B. NAIR, W·NWD, MAY 23, 1988

TABLE 8
THERMAL EXPANSION OF Ti 3Al 2.5V

TEMPERATURE		(1) UNALLOYED TITANIUM		(2) Ti 6Al-4V ALLOY		(3) Ti-3Al-2.5V
(C)	(F)	MEAN COEF. RT TO TEMP (micro m/m per C)	MEAN COEF. RT TO TEMP (micro in/in per F)	MEAN COEF. RT TO TEMP (micro m/m per C)	MEAN COEF. RT TO TEMP (micro in/in per F)	MEAN COEF. RT TO TEMP (micro in/in per F)
100	212	8.70	4.83	9.50	5.28	5.06
200	392	9.35	5.19	9.80	5.44	5.32
300	572	9.50	5.28	10.05	5.58	5.43
400	752	9.70	5.39	10.30	5.72	5.56
500	932	9.82	5.46	10.55	5.86	5.66
600	1112	10.00	5.56	10.80	6.00	5.78
700	1292	10.15	5.64	11.00	6.11	5.88

REFERENCE: METALS HANDBOOK, 9TH EDITION, AMERICAN SOCIETY OF METALS
(1) UNALLOYED TITANIUM: FIGURE 1, PAGE 372
(2) Ti-6Al-4V: FIGURE 20, PAGE 390 (HIGHEST VALUES USED)
(3) Ti-3Al-2.5V: AVERAGE OF VALUES FOR UNALLOYED AND Ti-6Al-4V

8.3.2 Independent Review Team Report of Grade 9 Titanium for use in the TITAN LWT Cask

This appendix includes the report of the independent review team convened to assess the suitability of Grade 9 titanium as the structural material for the TITAN LWT cask.

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WESTINGHOUSE ELECTRIC CORPORATION
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REPORT ON THE REVIEW OF THE SUITABILITY OF GRADE 9 TITANIUM FOR THE LEGAL WEIGHT TRUCK CASK

JUNE 23, 1989

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1.0 INTRODUCTION

The Nuclear Waste Department of the Energy Systems Business Unit of the Westinghouse Electric Corporation (Westinghouse) is involved with the U.S. Department of Energy (DOE) in the development of a spent fuel transportation cask with the objective of having a licensed, tested, and proven cask fleet by the end of this century. The cask system is being developed within the framework of the existing regulations and guidelines of the U.S. Nuclear Regulatory Commission (NRC), including specific applications of the requirements of the Boiler and Pressure Vessel Code (BPVC or Code) of the American Society of Mechanical Engineers (ASME).

Westinghouse has conducted a feasibility study (Reference 1) on an innovative cask design (TITAN) for Legal Weight Truck (LWT) shipments which would utilize a structural material with a high strength-to-weight ratio as an alternative to the austenitic stainless steels presently used for such casks. The structural material selected is Grade 9 Titanium, which is also known as Ti-3Al-2.5V because the major constituents are Titanium (Ti), Aluminum (Al), and Vanadium (V). Westinghouse has established the Review Team which prepared this report to review the Alternative Material Feasibility Study performed as a part of the Westinghouse development effort and to conduct those additional efforts required to:

> Evaluate the suitability of Grade 9 titanium (Ti-3Al-2.5V) as a spent fuel transportation cask structural and containment material from the standpoint of meeting the NRC requirements and guidelines. Special focus shall be placed on critical mechanical properties such as tensile and yield strengths, ductility, and fracture toughness. In addition, an evaluation shall be made of the appropriateness of using existing BPVC, Section III rules for the establishment of allowable stress intensities and allowable stress values for the alloy.

This report has been prepared by the Review Team in response to this assignment.

2.0 CONCLUSION

Based on the efforts of the members of the Review Team, it is our collective opinion that Grade 9 Titanium is a suitable material for use in transportation casks which must meet present NRC requirements and guidance.

The Review Team did, however, identify several areas where the determination of additional information is recommended. These areas are:

- Uniform elongation data from tensile tests up to 300^{o}F. (See Section 4.2.1)
- Tensile properties of welds and heat affected zones for weldments made with expected welding practices. (See Section 4.1.1)
- Low cycle fatigue data at temperatures to 300^{o}F. (See Section 4.3.1)
- Creep data for weld and heat affected zone materials. (See Section 4.1.2)

This additional information has been recommended to supplement the existing body of data for Grade 9 Titanium. It is the expectation of the Review Team that the information will reinforce our opinion that Grade 9 Titanium is suitable for spent fuel transportation casks.

A potential limitation on the maximum acceptable material thickness results from present interpretations of NRC requirements and of the limited fracture toughness data. As discussed in Section 4.2.2, a maximum thickness of about 3" is presently predicted. However, Section 4.2.2 also identifies alternative approaches which could alleviate the thickness limitation while retaining assurance that any flaws are stable even during accident conditions.

3.0 METHOD

3.1 Procedures

The Review Team was established by Mr. B. R. Nair, Westinghouse, Lead Technical Manager, TITAN Cask Project in consultation with the chairman of the Review Team. The objective was to convene a small group which included two individuals (Cooper and Yukawa) with specific experience in the application of engineering materials to critical structures and knowledge of the Code procedures and philosophy; two individuals (Thomas and Webster) with detailed technical knowledge of the properties and application experience with Grade 9 Titanium; and, an individual (Stephens) from a National Laboratory with experience in the design of transportation casks.

Westinghouse provided each of the members of the Review Team with copies of References 1-5. The first of these is the report on the Alternative Material Feasibility Study and included, in Appendix C, References 6-8 which provide detailed material property data. In addition to these references, one or more of the members of the Review Team considered the contents of References 9-16 in their review.

The meeting of the Review Team on which this report is based was held June 21-23, 1989 at the Westinghouse offices. The meeting consisted of sessions attended by both Westinghouse and Review Team personnel and of executive sessions attended only by members of the Review Team. Westinghouse support services were available to assist in appropriate tasks.

Each of the matters included in Section 4.0 of this report were discussed in both types of sessions. Drafts of each of the sections were prepared by individual Review Team members, with assistance from appropriate Westinghouse support personnel. The drafts were reviewed with the other Review Team members and preliminary agreement reached or questions formulated. The drafts were then provided to Westinghouse personnel for their review as to factual content and to provide for the presentation to the Review Team of additional information considered desirable by the Westinghouse personnel. The contents of this report were then prepared by the Review Team.

3.2 Limitations

The Review Team activity was limited in accordance with the scope stated in 1.0. In particular:

a. For this review the alloy, Grade 9 Titanium (Ti-3Al-2.5V), was understood to be in the mill annealed condition. (Mill annealed meaning heat treated between 1100^{o}F and 1450^{o}F). Properties obtained from conditions other than mill annealed, such as beta annealed, were not considered by the Review Team.

b. The review was conducted within the context of differentiating between the containment and the structural functional uses of Grade 9 Titanium. The specific classification of the various parts of the cask with respect to these two functional uses of Grade 9 Titanium was not included in the review.

c. In conducting this review, we have focused on the potential use of Titanium Grade 9 insofar as it has material properties which would require design methodology procedures which differ from current ferrous alloy cask design. We were not requested to conduct a comprehensive review of the cask design. Rather, we have attempted to identify the important materials properties which will affect the use of Grade 9 Titanium in the current TITAN cask design. The major materials property areas of concern identified by the group are covered as major topics in Section 4.

d. We have not considered the possibility of radiation damage. No data on this effect are available for Grade 9 Titanium. However, we know of no data for other titanium alloys which suggest that this will be a problem in the use of Grade 9 Titanium.

Discussions between Westinghouse and Review Team personnel went beyond these specific limitations in order that the Review Team understood the application. However, the Review Team reached no conclusions on matters not within their scope.

4.0 DISCUSSION

The objective of this section is to identify the issues considered by the Review Team and to summarize the most important factors affecting the conclusions of the team. The first paragraph under each of the following subheadings is phrased in the form of the question considered by the team.

4.1 Determination of Code-Type Allowable Stresses

The Code follows specific procedures in relating certain material properties to the allowable stress values used in the Code design rules. These have been considered and the appropriateness of the Code procedures and the specific numerical values have been reviewed.

4.1.1 Tensile Test Results

Do the trend curves for the yield strength and the tensile strength contained in Attachment 2 to Reference 2 provide a reasonable interpretation of available data?

The procedure followed in preparation of the trend curves is that used by the Code Committee for establishing allowable design stress values. Tensile data were available from four heats of Grade 9 Titanium and four different product forms which exceed Code requirements for establishing allowable stresses. These data are plotted as a function of temperature, a best fit curve is drawn to represent the data, and the curve is reduced, everywhere, by the ratio of the specified minimum value at room temperature to the fitted curve value at room temperature. The specified minimum values are those contained in the applicable ASME specification or the ASTM specification if the Code has not adopted the material. The Code currently has adoption of SB348 containing Grade 9 similar to to ASTM B348-83 (Reapproved 1987) for bars and billets out for letter ballot. This specification establishes minimum room temperature values of 70 ksi and 90 ksi, respectively for yield and ultimate strengths; and minimum values of 15% and 25%, respectively for the elongation in 4D and the reduction of area.

The recently revised ASTM Product Specifications (which include Grade 9 Titanium) B381, Forgings; B337, Seamless and Welded Pipe; B338, Tubing; and B363, Welding Fittings have been approved for adoption by Section II of the Code Committee. B265, Strip, Sheet and Plate, includes Grade 9 Titanium in the latest revision of the specification which was recently approved by ASTM Society ballot. Each of the ASTM Product Specifications have the same minimum tensile values as are specified in SB348.

4.1.2 Creep and Creep Rupture Values

Are the available creep and stress rupture data sufficient to assure that neither of these properties will control the Code allowable stresses in the temperature range applicable to the subject cask?

The maximum normal service temperature for the transportation cask is 300^{o}F. Using available data from the current Section VIII Code Case, tensile properties, and not creep or rupture properties, have been shown to form the basis for allowable stress values up to a temperature of 600^{o}F. Representative creep data which support this statement are given in the Appendix to this report. Creep and stress rupture are therefore not important in the determination of allowable stresses for the transportation cask.

The Review Team was, however, concerned with the tendency for titanium alloys, specifically Grade 9 Titanium, to show significant creep strain at stresses greater than $0.7S_y$ at room temperature or higher. While the bulk of the cask containment system would be designed well below $0.7S_y$, it was felt that the locally high stresses at the threads used to seal the containment shell could lead to stress relaxation and possible leakage of the seal. While the design does call for Alloy 718 threaded inserts, which could alleviate the stress concentration, stress relaxation of the titanium in the thread area must be specifically treated. This issue is discussed in Section 4.3.4.

It is recognized that the accident condition sequence postulated for the cask includes exposure to a 1475^{o}F fire for 30 minutes. However, the Review Team considers this condition to be outside the scope of the Code allowable stresses. Further discussion of this accident condition and its affect on the cask design are given in Section 4.3.3.

4.1.3 Determination of Allowable Stress Values

Are the Code, Section III, procedures for the determination of allowable stress values appropriate for application to Grade 9?

At temperatures below those where creep or stress rupture values control, as discussed in Section 4.1.2, the general Code procedures establish the allowable stress or stress intensity value as the lower of certain factors on the tensile strength or of two-thirds of the minimum (specified or as determined from the trend curve) yield strength. An exception is made for certain materials, such as the austenitic stainless steels which are strongly strain hardening and for which significant service experience is available, in that the factor on yield strength at temperature is increased to 90%.

Based on the yield and tensile strength values as a function of temperature being considered to be correct, (See Section 4.1.1), the values determined from the tensile strength are controlling. The Review Team considers the procedures used in the Code to determine allowable stress values, or allowable stress intensity values, from the minimum specified tensile strength or from the elevated temperature trend-curve-derived values to be appropriate.

Based on the discussion in Section 4.1.2, the Review Team considers the allowable values derived from the Code procedures to be reasonable up to a temperature in excess of that to which the Code values are applicable in the design of the cask.

Code allowable values for accident conditions are termed Service Level D allowables. Code allowable values are adopted in Regulatory Guide 7.6 with a few changes. All of these allowable values are expressed either in terms of a factor on the Service Level A allowable value or as a factor on one of the tensile properties, yield strength or ultimate tensile strength. The use of a fraction of the ultimate tensile strength limits the use of materials with a high ratio of yield to tensile strength. In addition to this protection, the limits used in the Code discussed in Section 4.2.1 considered the possible use of such materials. The yield-to-tensile strength ratio of Grade 9 Titanium is no higher than for several ferritic steels approved for Section III, Class 1, applications.

The Review Team considers the data discussed in Section 4.1.1 to provide the necessary material property data for those allowable values which are based on the yield strength or on the ultimate tensile strength. This consideration includes the Review Team agreement with the numerical factors applied to these properties in order to obtain the tabulated allowable values.

4.2 Ductility and Fracture Toughness

It is, first, necessary to express what is meant by the terms "ductility" and "fracture toughness" as viewed by the Review Team.

The Review Team considers the meaningful measures of ductility to be the strain at maximum load and the reduction of area at failure, both determined by tensile testing. The first of these values is of use in establishing the true stress - true strain curve for use in inelastic analyses and the second may be related to crack initiation as the result of overstraining, if the effects of triaxial stresses are considered. In contrast, the percentage elongation at failure in a tensile test is considered to be useful only as a quality control measure.

The Code (Sections III and VIII) does not require fracture toughness testing of Grade 9 Titanium because it is a nonferrous material. However, for regulatory acceptance, it is necessary to demonstrate that the material has acceptable fracture toughness. The Review Team considers the meaningful measures of the fracture toughness of this material in the temperature range of interest to be data obtained from a J-integral versus crack extension or a Crack Opening Displacement (COD) test. Such data are useful in determining J_{Ic}, K_{Ic}-equivalent and tearing modulus values. Charpy V-notch test data are considered primarily useful as a quality control measure. The Drop-Weight Test, as is used in References 4 and 5, and similar tests used to determine a "Nil-ductility Temperature", is not applicable to this material.

This discussion is not intended to imply that the material property data obtained with respect to fracture toughness must be applied in specific fracture mechanics evaluations in the design of these casks. However, such data are of value in understanding the behavior under accident conditions by comparison with the behavior of other materials.

Also, this specific discussion may understate the importance of Charpy V-notch data and notched tensile test data which are available and which indicate that the ductile-to brittle transition temperature, as generally defined based on these properties, is very far below the temperature range of this cask application for Grade 9 Titanium.

4.2.1 Ductility

Are there sufficient data available on ductility?

No specific data are available with respect to the strain at maximum load in the tensile tests. Such data will become available from the material test programs generally planned by Westinghouse. In the Review Team's opinion, the absence of such data is not considered to be a limiting factor in evaluating the suitability of Grade 9 Titanium.

One of the major applications of such data is to establish that a plastic instability will not occur as a consequence of the membrane stretching of the material which may result from postulated accident conditions. Present estimates of such biaxial membrane strains produce values which are small when compared to values of concern with reasonably anticipated uniform strain values.

The other use of such data is related to permissible multipliers on the tabulated allowable stress values in determining the allowable values under accident conditions. These multipliers are the 2.4 factor permitted by Code Appendix F in establishing allowable membrane stresses and the associated 3.6 multiplier used in determining the limit on primary plus membrane stresses. These limiting values were developed by consideration of such data as that in Reference 12. The test results and interpretations of that paper include alloys with Yield/Tensile strength ratios higher than Grade 9 Titanium.

4.2.2 Fracture Toughness

Does Grade 9 Titanium have acceptable levels of fracture toughness?

One way of showing that Grade 9 Titanium has acceptable fracture toughness is by demonstrating that it meets requirements equivalent to those that have been proposed for ferritic steels. There are three proposed requirements that can be considered:

1. NRC draft Regulatory Guides for ferritic steel shipping containers with maximum wall thickness of four inches (Ref. 13) and wall thickness greater than four inches (Ref. 14).

2. ASME Section III, Division 3, proposed requirements for ferritic steels.

3. ASME Section III, Division 1, Class 2, toughness requirements for ferritic steels (NC-2000).

The essence of each of these is:

1. Draft Regulatory Guide is based on the requirement that the ratio K_{ID}/S_{YD} is equal to or greater than the square root of t; where K_{ID} = dynamic fracture toughness; S_{YD} = dynamic yield strength; and, t = thickness.

2. ASME III, Division 3, requirements for dynamic fracture toughness are based on hypothetical semi-elliptical surface flaw of depth = t/10 but not less than 0.25", length = 6x depth, applied stress equal to $2S_m$ (safety factor of 2).

3. NC-2000 requirements are stated in terms of C_V energy or lateral expansion values for thickness up to 2 1/2 inches, but the underlying basis is the quasi-static fracture toughness, K_{Ic}, required for an applied stress equal to 5/8 of the yield strength and 2t long through-wall flaw. Requirements for thicknesses greater than 2 1/2 inches use a different fracture mechanics method but the method is not easily adaptable to a material other than ferritic steel. Therefore, the requirements for thickness greater than 2 1/2 inches are calculated by using the basis for thickness up to 2 1/2 inches.

It should be noted that NRC draft Regulatory Guide requirements apply only to the base material with no specific requirements for the weld metal or the heat affected zone (HAZ) whereas Division 3 and Division 1, Class 2, requirements apply to the base metal, weld metal and HAZ.

The calculated fracture toughness required by each of these bases for Grade 9 Titanium using S_m = 30 ksi and S_{YD} = 70 ksi are as follows:

Thk., in.	NRC Draft Regulatory Guide K_{ID}, ksi in.	ASME III, Div. 3 K_{ID}, ksi in.	ASME III NC-2000 K_{IC}, ksi in.
5/8	55		42
1	70	55	55
2	101	55	78
2.5	111	55	87
3	121	60	
4	140	70	110
5	157	78	

Experimental data from the Navy (not for general public) indicates that the fracture toughness required in all of these criteria up to about 3 inch thickness are attainable in Grade 9 Titanium at room temperature if dynamic and quasi-static fracture toughness values are similar. However, potential limitations exist at larger thicknesses, lower temperatures, and welds and HAZ's.

Several alternatives are possible to alleviate the potential limitations:

1. Processing to increase the fracture toughness of the parts where required.

2. Evaluate the fracture toughness requirements in terms of J values to adjust for elastic modulus differences.

3. Use the J value at some small amount of crack extension (such as 1 mm, 0.04 inches) for the determination of the fracture toughness value.

4. Redefine the requirements by using elastic-plastic instability analysis.

Additional testing is required to better define the properties and to assist in choosing whether or not any of these alternative methods are implemented.

4.3 Other Significant Features of the Material

The discussions in Sections 4.1 and 4.2 cover specific properties and expected behavior or actions based on that property. This subsection is intended to discuss other matters in which the selection of Grade 9 Titanium may affect the design or the response of the cask to operating or accident conditions.

4.3.1 Fatigue

Are the available fatigue data sufficient for the present application and sufficient to meet Code requirements?

Sufficient room temperature strain- or stress-controlled fatigue data are available from the Navy to assure that a design can be developed with Grade 9 Titanium. Additional data at elevated temperatures must be developed before the proposed Code Case of Reference 2 is complete.

4.3.2 Corrosive Environments

Are available corrosion resistance data sufficient for the present application?

Titanium alloys in general, and specifically Grade 9 Titanium, have excellent corrosion resistance to naturally occurring environments (Reference 16). As such the Review Team does not consider corrosion as an issue in the application of Grade 9 Titanium for the transportation cask.

4.3.3 Fire Accident

The postulated event, as discussed in Section 4.1.2, involves exposure to a 1475 F condition for 30 minutes. Are the available data sufficient to assure that there is no significant consequence of the selection of Grade 9 Titanium for cask construction?

Westinghouse analysis of the fire exposure shows that the maximum temperature of the containment vessel and the intermediate structural vessel will be 350°F and 550°F, respectively. These temperatures are well within the Section VIII Code Case temperature limits of 600°and would pose no threat to the titanium vessels.

Even if the containment and/or intermediate vessel reached the fire temperature, 1475°F, no serious degradation of the Grade 9 Titanium would occur. Fire fighting fluids would likewise have little affect since the alloy undergoes no change in properties from 1475°F to ambient temperature.

4.3.4 Relaxation of the Bolted Closure Seal

Is relaxation of the closure bolt preload a potential limitation to the use of Grade 9 Titanium?

Previous work on the Ti-6Al-4V alloy (Reference 15) has shown that stress relaxation at room temperature can occur in this alloy at stresses which are approximately $0.7S_y$ or greater. Room temperature, 200°F and 250°F creep tests should include stresses in the range of 70-90% of the yield stress at a given temperature.

An estimate of the possible degree of room temperature stress relaxation has been made using the stress and time exponents for Ti-6Al-4V alloy (References 15) and room temperature creep data for Grade 9 Titanium (Reference 16). These results suggest that Grade 9 Titanium is resistant to stress relaxation at room temperature in this design. Similar analyses should be made at other temperatures.

4.3.5 Compatible Plating Materials

Are there any plating materials which should be excluded from the cask design due to incompatibility with Grade 9 Titanium?

It is the opinion of the Review Team that Zn, Ag and Cd platings be avoided on components such as the Alloy 718 fasteners (Reference 16). Use of these platings could possibly lead to embrittlement of Grade 9 Titanium. Current design calls for use of Cr-plated Alloy 718 bolts, which should not pose a problem.

5.0 REFERENCES

1. "TITAN Legal Weight Truck Cask Alternative Material Feasibility Study," Westinghouse NWD-TR-008, Rev. 0, July 1988 (Proprietary)

2. "Request for Approval of Grade 9 Titanium for Use Under the Rules of ASME Code, Section III," Westinghouse Letter SE:RRM:89:009, January 13, 1989

3. Westinghouse Structural Design Criteria

4. "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Up TO Four Inches Thick," NUREG/CR-1815, LLNL, 1984

5. "Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers Greater Than Four Inches Thick," NUREG/CR-3826, LLNL, 1981

6. Properties of Ti 3Al-2.5V extracted from Aerospace Structural Materials handbook

7. Properties of Ti 3Al-2.5V provided by Timet Corporation

8. "Properties of Titanium for Industrial Applications with Emphasis on Ti 3Al-2.5V", P. A. Russo and S. R. Seagle, Industrial Applications of Titanium and Zirconium, Third Conference, ASTM STP 830, 1984

9. "Packaging and Transportation of Radioactive Material," Title 10 Code of Federal Regulations Part 71.

10. "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," Regulatory Guide 7.6, REVISION 1, USNRC

11. "Load Combinations for the Structural Analysis of Shipping Casks," Regulatory Guide 7.8, USNRC

12. "Experimental Effort on Bursting of Constrained Disks as Related to the Effective Utilization of Yield Strength," W. E. Cooper, E. H. Kottcamp, G. A. Spiering, ASME Paper 71-PVP-49.

13. "Fracture Toughness Criteria for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of Four Inches (0.1 m)", U.S. NRC Draft Regulatory Guide, June 1983.

14. "Fracture Toughness Criteria for Ferritic Steel Shipping Containers with a Wall Thickness Greater than Four Inches (0.1 m)", U.S. NRC, Draft Regulatory Guide, June 1986.

15. "Stress Relaxation of a Titanium (Ti-6Al-4V) Threaded Joint", J. J. Stephens, J. W. Munford, SAND87-1818, Sandia National Laboratories, Albuquerque, May 1988.

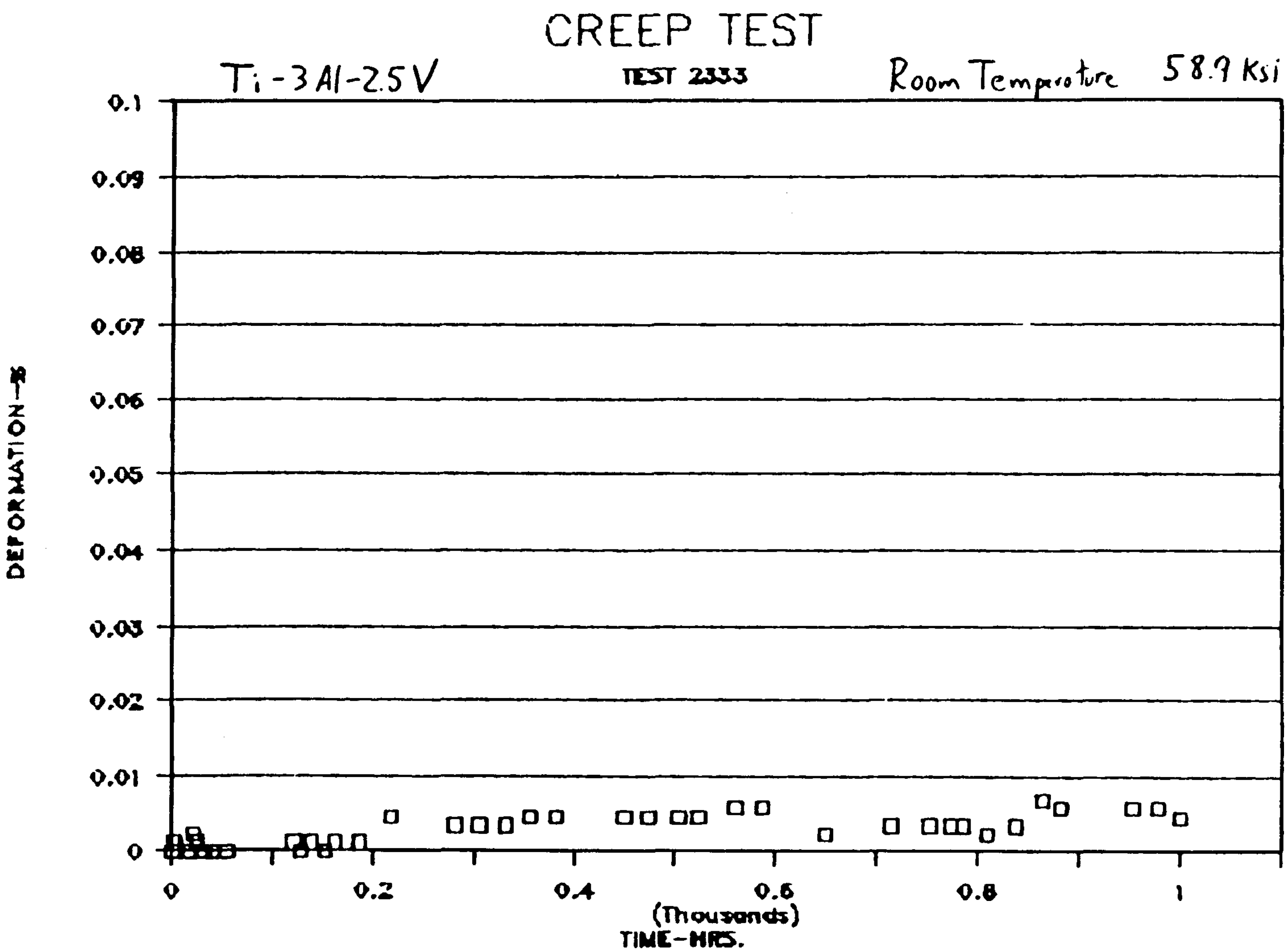
16. R. W. Schutz, D. E. Thomas, "Corrosion of Titanium and Titanium Alloys", ASM Metals Handbook Volume 13, ASM International, Metals Park, OH, 1988.

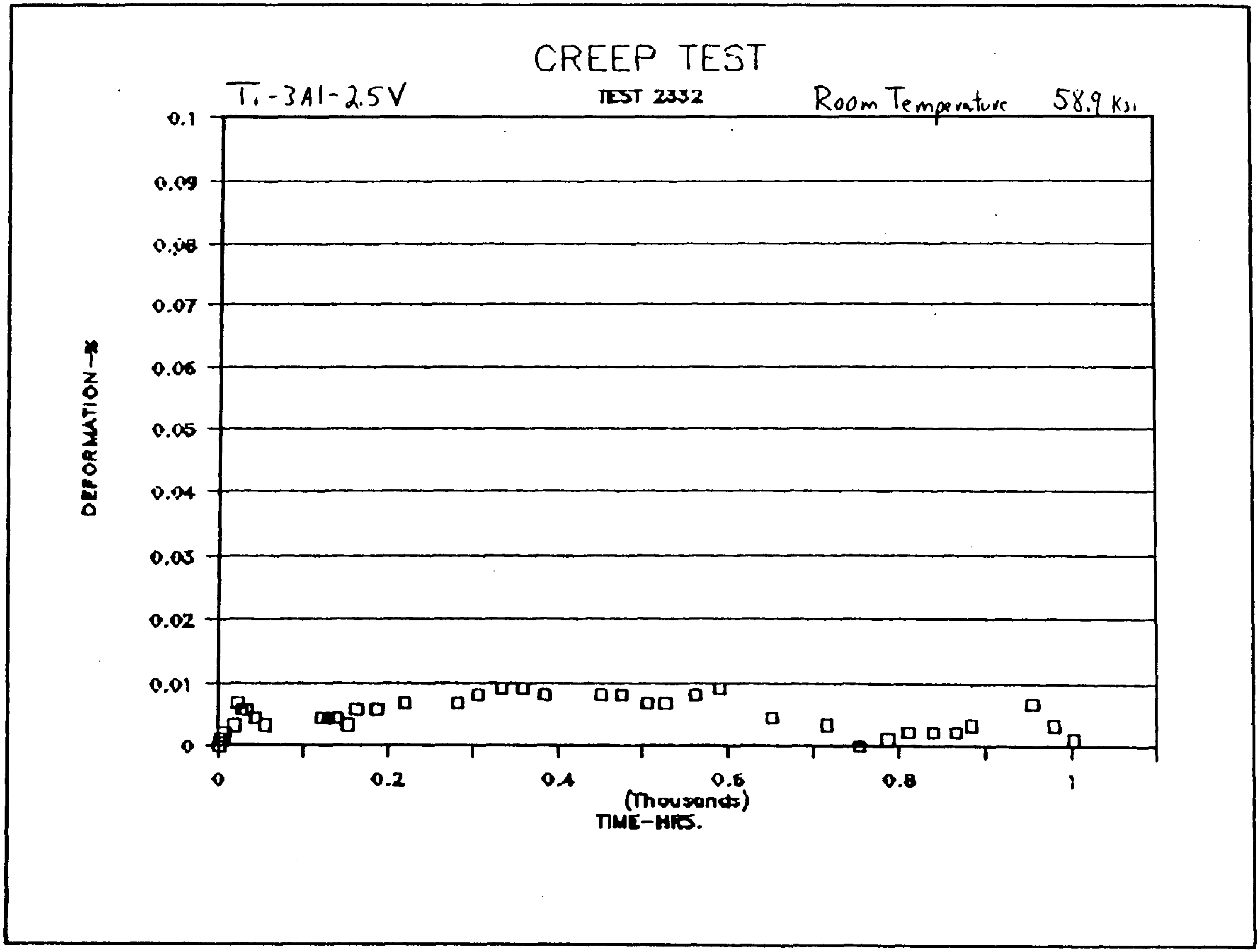
APPENDIX

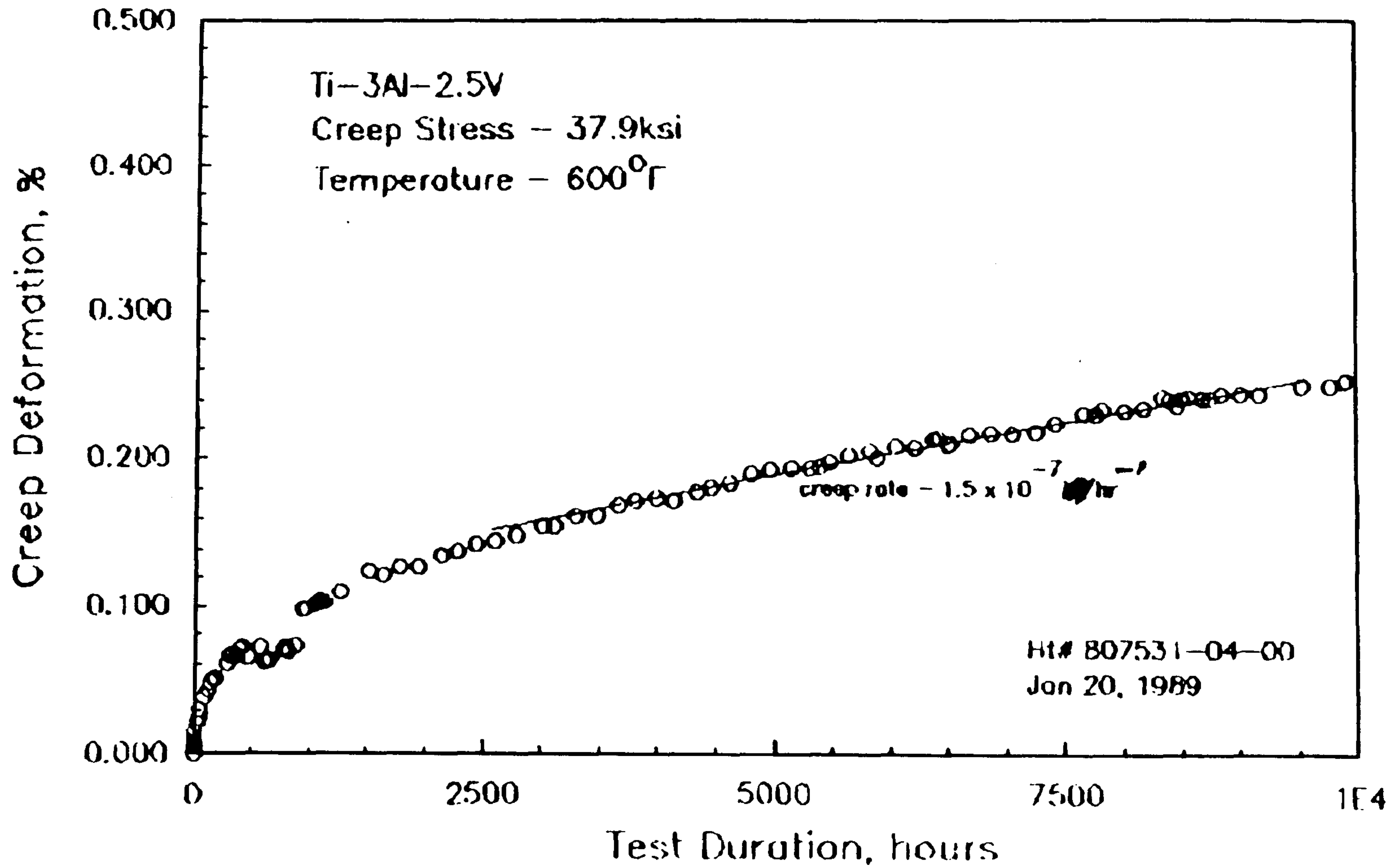
Creep Data for

Grade 9 Titanium

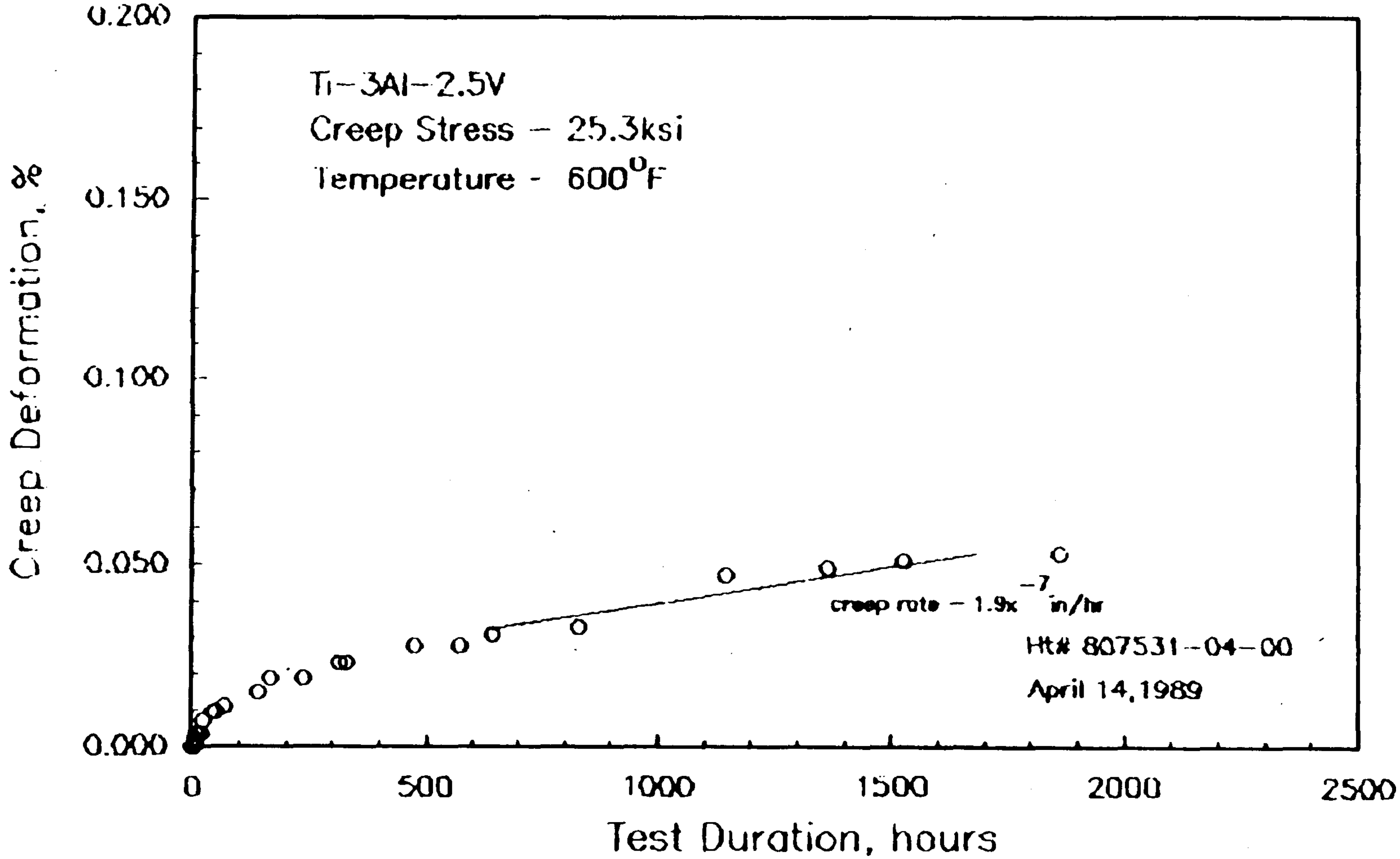
Provided by RMI Company

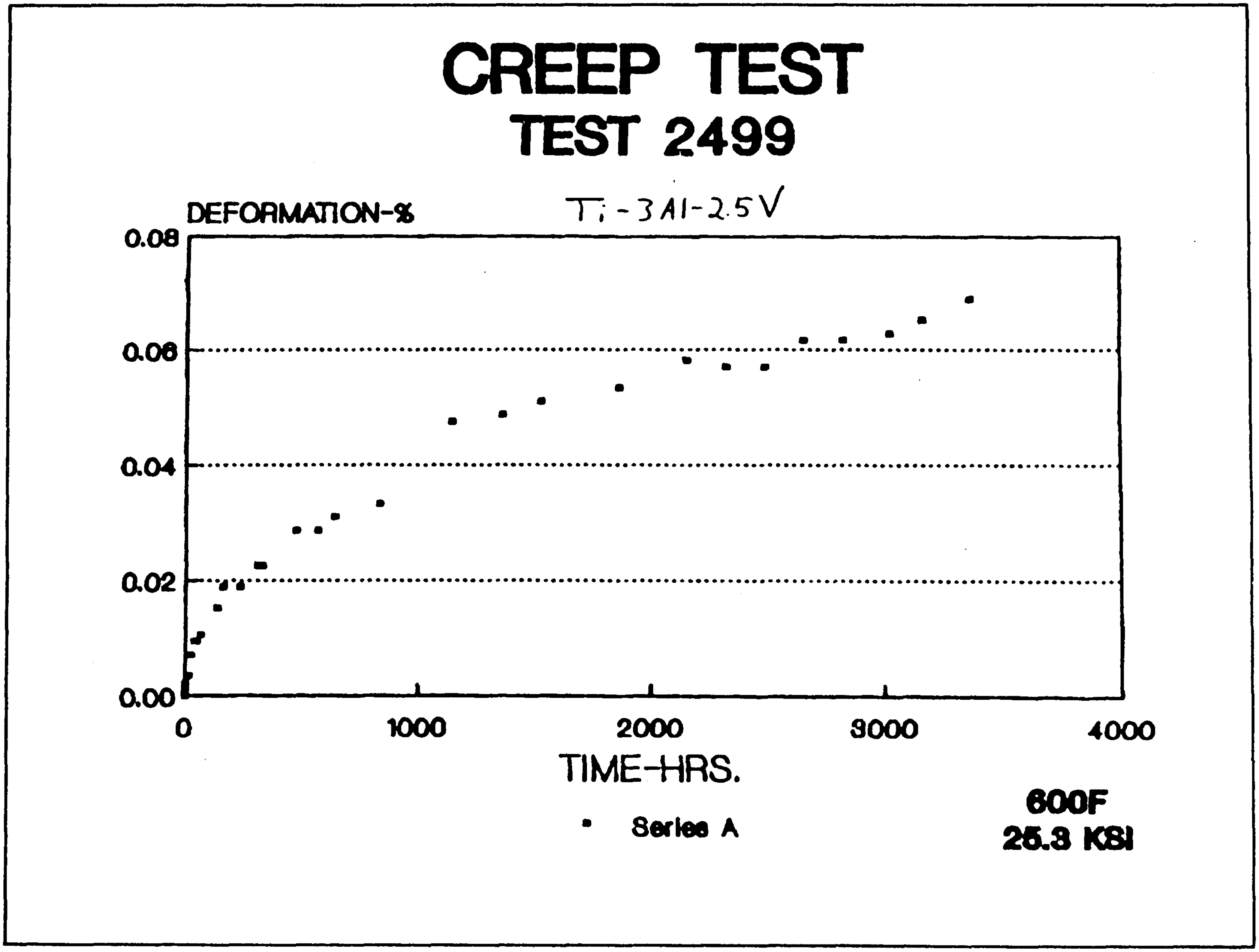

CREEP TEST
Ti-3Al-2.5V
TEST 2333
Room Temperature
58.9 ksi
DEFORMATION-%
0.1
0.09
0.08
0.07
0.06
0.05
0.04
0.03
0.02
0.01
0
0
0.2
0.4
0.6
0.8
1
(Thousands)
TIME-HRS.

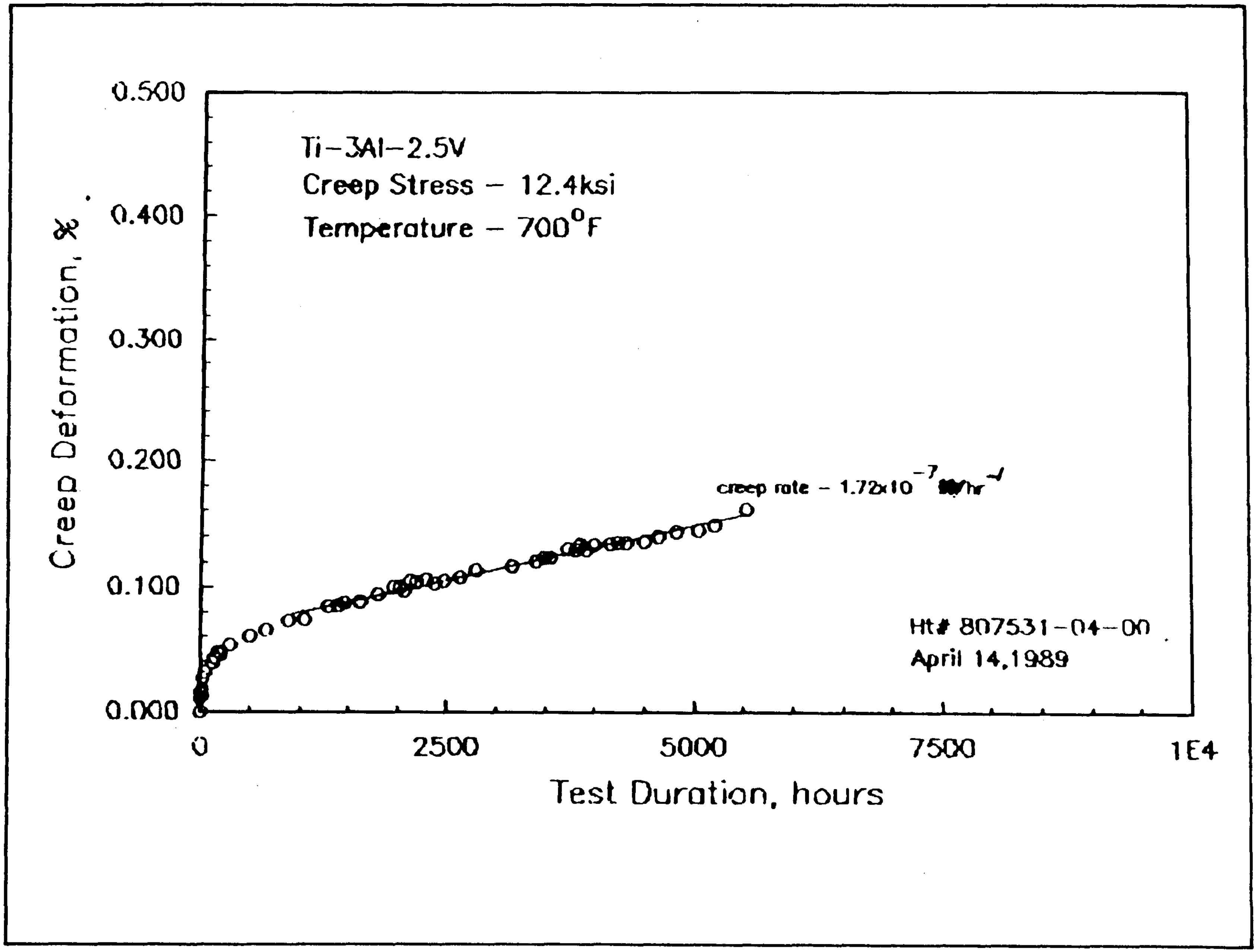

CREEP TEST
Ti-3Al-2.5V
TEST 2332
Room Temperature
58.9 Ksi
DEFORMATION–%
0.1
0.09
0.08
0.07
0.06
0.05
0.04
0.03
0.02
0.01
0
0
0.2
0.4
0.6
0.8
1
(Thousands)
TIME–HRS.

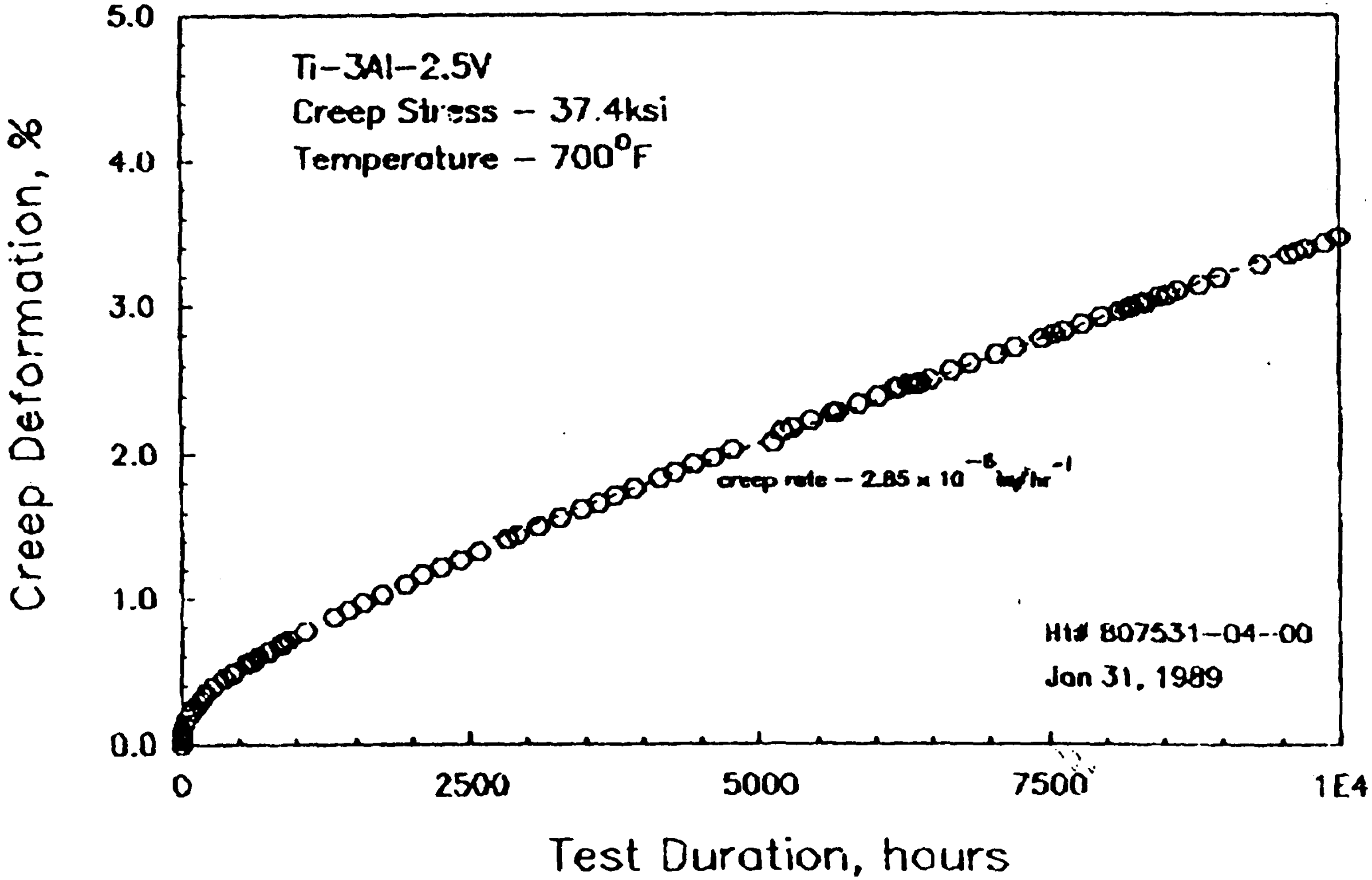


Ti–3Al–2.5V
Creep Stress – 37.9ksi
Temperature – 600°F
creep rate – 1.5 x 10^{-7} /hr
Ht# 807531–04–00
Jan 20, 1989
Creep Deformation, %
0.000
0.100
0.200
0.300
0.400
0.500
0
2500
5000
7500
1E4
Test Duration, hours


Ti−3Al−2.5V
Creep Stress − 25.3ksi
Temperature - 600°F
Creep Deformation, %
Test Duration, hours
creep rate − 1.9x10⁻⁷ in/hr
Ht# 807531−04−00
April 14,1989
0.200
0.150
0.100
0.050
0.000
0
500
1000
1500
2000
2500


CREEP TEST
TEST 2499
Ti-3Al-2.5V
DEFORMATION-%
0.08
0.06
0.04
0.02
0.00
0
1000
2000
3000
4000
TIME-HRS.
Series A
600F
25.3 KSI


Ti-3Al-2.5V
Creep Stress - 12.4ksi
Temperature - 700°F
creep rate - 1.72x10^-7 /hr
Ht# 807531-04-00
April 14,1989
Creep Deformation, %
Test Duration, hours
0.000
0.100
0.200
0.300
0.400
0.500
0
2500
5000
7500
1E4



Ti–3Al–2.5V
Creep Stress – 37.4ksi
Temperature – 700°F
Creep Deformation, %
Test Duration, hours
Ht# 807531–04–00
Jan 31, 1989
0.0
1.0
2.0
3.0
4.0
5.0
0
2500
5000
7500
1E4

8.3.3 Grade 9 Titanium Test Program

This section contains two separate scopes of work for the conduct of tests on specimens of Grade 9 titanium. The test results will be used to support the ASME Code Case inquiry and design activities.

1.0 Scope of Work

Perform the following tests on ASTM Grade 9 Titanium (Ti-3Al-2.5V) in accordance with the below-referenced tests to be performed, specifications and requirements:

1.1 Tests To Be Performed

Test Description	Temperature	Number of Specimens
Tensile Test including Modulus of Elasticity, σ vs. ε, RA, El.	-40°F	6
	RT	6
	150°F, 300°F	12
Charpy V-Notch	-40°F, RT, 150°F, 300°F	24
Poisson's Ratio	-40°F, RT, 150°F, 300°F	24
Specific Heat	-40°F to 300°F	4
Thermal Conductivity	-40°F to 600°F	4
Thermal Expansion	-40°F to 600°F	4
Emissivity	-40°F to 300°F	5

1.2 Test Material Procurement

The test organization is responsible for procurement of a sufficient quantity of ASTM Grade 9 Titanium (Ti-3Al-2.5V) to perform the tests defined in Section 1.1, Tests To Be Performed. Specimens for testing shall be from three heats. The specimens will be both plate and billet in the annealed condition (low temperature alpha & beta anneal, 1475°F for 30 minutes, air cool). Westinghouse shall verify the material certification for the test specimens are in conformance to ASTM Grade 9 Titanium (Ti-3Al-2.5V) specifications prior to proceeding with any testing. The test organization shall notify Westinghouse 2 weeks prior to material receipt.

1.3 Applicable Standards For Testing

The applicable ASTM Specification for property determination for ASTM Grade 9 Titanium (Ti-3Al-2.5V) is provided below. Where procedures or specifications are not available, the test organization shall submit to Westinghouse the proposed test procedure or specification for approval along with their response to this RFQ. The following presents the available applicable ASTM Standards:

ASTM E8-87a: Standard Test Methods of Tension Testing Metallic Materials.

ASTM E21-79: Standard Recommended Practice for Elevated Temperature Tension Tests of Metallic Materials

ASTM E23-86: Standard Methods for Notched Bar Impact Testing of Metallic Materials

ASTM E132-86: Standard Test Method for Poisson's Ratio at Room Temperature.

ASTM E813-87: Standard Test Method for J_{Ic}, a Measure of Fracture Toughness. Elevated and low temperature tests would be run to the applicable sections of ASTM E813-87.

ASTM E967-83 and ASTM E968-83: Specific Heat

ASTM E228-85: Standard Test Method for Linear Thermal Expansion with a Vitreous Silica Dilatometer.

1.4 Quality Assurance

The Supplier shall perform the scope of work under a Quality Assurance Program which meets the pertinent basic and supplemental requirements of ANSI/ASME NQA-1 (1986). A qualification audit of the Supplier may be performed prior to initiation of work if the Supplier is not on the Westinghouse "Qualified Suppliers List".

1.5 Westinghouse Contact

The Cognizant Technical Manager is Mr. Bala R. Nair, who can be reached on (412) 374-2401.

1.6 Test Information From Suppliers

The Supplier shall submit the following information for each test listed in Section 1.1

- Type of test and temperature
- Calibration date of instruments
- Photographic records of test specimens before and after testing (black & white format) and the test setup
- Material mill certifications
- Comparable published ASTM material properties
- Raw data from each test
- Test Record books (copies acceptable)
- Location, time, date and responsible test engineer
- Unique specimen identification

1.0 Scope of Work

Provide a Firm Fixed Price (with details, substantiating your price) and schedule for the performance of the following tests on ASTM Grade 9 Titanium (Ti-3Al-2.5V) in accordance with the below-referenced tests to be performed, specifications and requirements:

1.1 Tests To Be Performed

1.1.1. Tensile test at -40°F, RT, 150°F, 300°F, and 600°F to obtain:

- o true stress-strain curves up to failure
- o fracture strains

Six specimens for each test temperature shall consist of two specimens from each of the three heats. (See Section 1.2) A total of 30 specimens will be required.

1.1.2 J_{IC} at -40°F, RT, and 300°F.

Six specimens for each test temperature shall consist of two specimens from each of the three heats. A total of 18 specimens will be required.

1.1.3 K_{ID} at -40°F, RT, 150°F, 300°F, and 600°F.

Six specimens for each test temperature shall consist of two specimens from each of the three heats. A total of 30 specimens will be required. Use pre-cracked instrumented charpy specimens and perform testing in accordance with the draft ASTM procedure (E 2401 (81-1) letter ballot).

1.1.4 Tensile tests for weld metal and weldments including heat affected zone at -40°F, RT, 150°F, 300°F, and 600°F to obtain.

- o yield strength
- o yield point
- o tensile strength
- o elongation
- o reduction area

Specimens will be provided under a seperate contract. A total of 30 specimens will be provided for testing.

1.1.5 Low cycle strain fatigue at RT, 300°F and 600°F.

Thirty six specimens for each test temperature shall consist of twelve specimens from each of the three heats. A total of 108 specimens will be required.

1.1.6 Creep and creep rupture for weld metal and weldments including heat affected zone at RT, 300°F and 600°F.

Specimens will be provided under a seperate contract. A total of 18 specimens will be provided for testing.

WB89-8

1.1.7 Tensile test at 600°F to obtain:

- Yield strength
- Yield point
- Tensile strength
- Elongation
- Reduction of Area

Six specimens shall consist of two specimens from each of the three heats. A total of six specimens will be required.

1.2 Test Material Procurement

The test organization is responsible for procurement of a sufficient quantity of ASTM Grade 9 Titanium (Ti-3Al-2.5V) to perform the tests defined in Section 1.1, Tests To Be Performed. Specimens for testing shall be from three heats. The specimens will be both plate and billet in the annealed condition (low temperature Alpha & Beta anneal, 1475°F for 30 minutes, air cool). Westinghouse shall verify the material certification for the test specimens are in conformance to ASTM Grade 9 Titanium (Ti-3Al-2.5V) specifications prior to proceeding with any testing. The test organization shall notify Westinghouse 2 weeks prior to material receipt.

1.3 Applicable Standards For Testing

The applicable ASTM Specifications for property determination of ASTM Grade 9 Titanium (Ti-3Al-2.5V) are provided below. Where procedures or specifications are not available, the test organization shall submit to Westinghouse the proposed test procedure or specification for approval along with their response to this RFQ. The following presents the available applicable ASTM Standards:

ASTM E8-87a: Standard Test Methods of Tension Testing Metallic Materials.

ASTM E21-79: Standard Recommended Practice for Elevated Temperature Tension Tests of Metallic Materials

ASTM E606-80: Recommended Practice for Constant-Amplitude Low Cycle Fatigue Testing

ASTM E139-83: Recommended Practice for Conducting Creep, Creep-Rupture, and Stress-Rupture Tests of Metallic Materials

ASTM E813-87: Standard Test Method for J_{IC}, a measure of fracture toughness. Elevated and low temperature tests would be run to the applicable sections of ASTM E813-87.

1.4 Quality Assurance

The Supplier shall perform the scope of work under a Quality Assurance Program which meets the pertinent basic and supplemental requirements of

ANSI/ASME NQA-1 (1986). A qualification audit of the Supplier may be performed prior to initiation of work if the Supplier is not on the Westinghouse "Qualified Suppliers List".

1.5 Westinghouse Contact

The Cognizant Technical Manager is Mr. Bala R. Nair, who can be reached on (412) 374-2401.

1.6 Supplier Submittals for RFQ

Response to the RFQ shall contain the following:

- Firm Fixed Price and cost breakdown to substantiate price
- Schedule from receipt of order to completion of testing including lead time for raw material
- Compliance with ASTM specifications and any other test procedures or specifications for the tests listed under Section 1.1 of this RFQ.

1.7 Test Information From Suppliers

The Supplier shall submit the following information for each test listed in Section 1.1

- Type of test and temperature
- Calibration date of instruments
- Photographic records of test specimens before and after testing (black & white format) and the test setup
- Material mill certifications
- Comparable published ASTM material properties
- Raw data from each test
- Test Record books (copies acceptable)
- Location, time, date and responsible test engineer
- Unique specimen identification
- Test specimens to be furnished to Westinghouse NWD
- Deviation from procedures shall be documented and approved by NWD prior to implementation
- Submittal of above shall be on Westinghouse NWD Document Submittal Forms
- The results and their acceptability
- The actions taken with regard to any deviations noted
- The person evaluating the test results

8.3.4 Impact Limiter Test Program

A two-phase engineering test program will be implemented for demonstrating the performance of the aluminum honeycomb impact limiters used on the cask.

During Phase 1, static load-deflection tests will be conducted on aluminum material samples, and on quarter-scale impact limiter assemblies for various loading orientations. This section contains the draft test plan for the Phase 1 test program.

During Phase 2, half-scale impact limiters attached to a cask body mockup will be drop tested (30 foot free drop) at five different impact orientations (side, end, C.G. over corner, and oblique). A total of three pairs of impact limiters will be used for testing.

The engineering test program will provide the necessary data to demonstrate impact limiter performance and verify analytical predictions. This will provide a high degree of assurance of success of the half-scale cask design verification testing that will be implemented subsequently in accordance with the requirements of 10 CFR Part 71.

The Phase 1 Engineering Test program is scheduled for completion in March 1991 while the Phase 2 program is anticipated to be completed by the end of July 1991.

DRAFT

TITAN LEGAL WEIGHT TRUCK CASK

ALUMINUM HONEYCOMB IMPACT LIMITER
PHASE I TEST PLAN

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1.0 INTRODUCTION

1.1 Background

The TITAN Legal Weight Truck (LWT) Cask design includes aluminum Alloy 5052 honeycomb impact limiters. As shown in Figure 1, the cask is provided with two impact limiters that are bolted to the cask body. The impact limiters reduce the deceleration loads on the cask body resulting from design drop accident conditions and absorb the energy of the dropped cask.

The impact limiter is constructed of honeycomb material with two different crush strengths (densities) and the honeycomb cells are oriented to provide optimum energy absorption characteristics for the various loading conditions. Honeycomb with a density of 10.6 lb/ft^3 is used to absorb the end drop and the corner drop loadings, and honeycomb with a density of 8.1 lb/ft^3 is used to absorb the side drop loading. The honeycomb segments are bonded with adhesive and the structure is covered with an inner and outer skin of 0.031-inch thick stainless steel.

Analytical methods have been developed to predict the load-deformation behavior of the impact limiters (References 1 through 3). Experimental confirmation of analytical predictions is essential in order to demonstrate the performance of the impact limiters. Drop testing of a 1/2 scale model of the cask will verify the adequacy of the cask design including the impact limiters. But before those relatively expensive tests are conducted, experimental evidence of the suitability of the impact limiter design (its shape, construction, crush strengths, etc.) must be obtained so that there is a high degree of confidence that the verification testing will successfully demonstrate compliance with 10 CFR Part 71 requirements.

This first phase of testing will provide data necessary for the conduct of Phase II which will consist of drop tests on 1/2 scale impact limiters attached to a 1/2 scale simulated (solid steel) cask.

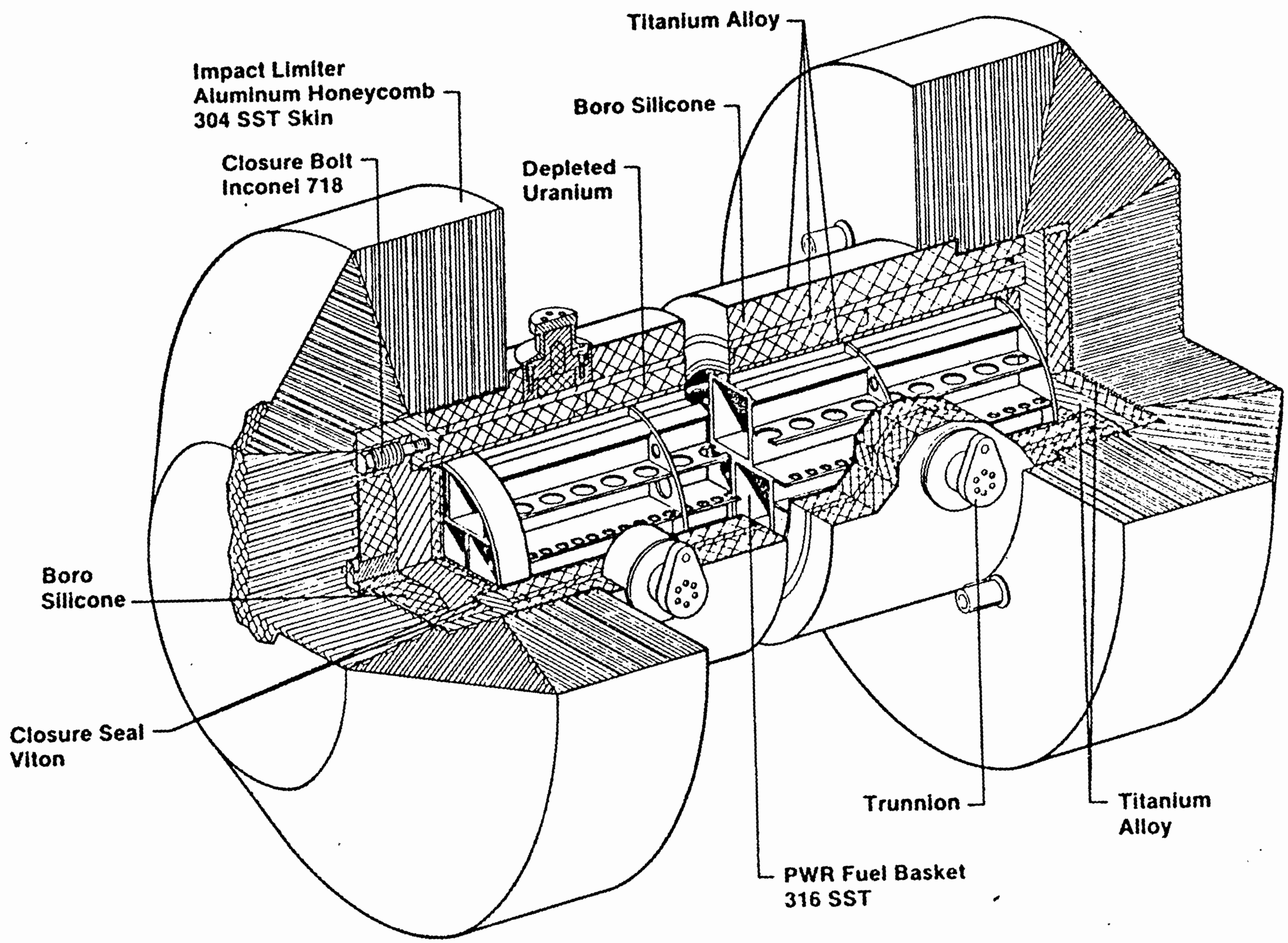

Figure 1. TITAN Legal Weight Truck Cask

1.2 Justification for the Test

The application of aluminum honeycomb material for cask impact limiters is relatively new. Existing vendor data show that the load-deflection characteristics of the aluminum honeycomb combined with its light weight make the material an attractive choice for impact limiter service in transportation casks. The vendor data, supplemented with the proposed testing, will provide reasonable confidence that the aluminum honeycomb is a viable material for the impact limiters. The program of tests proposed includes static and dynamic load testing of rectangular blocks of honeycomb material and dynamic load testing of quarter-scale impact limiters. The material tests will be performed at various temperatures to determine temperature effects on the honeycomb. This test program is necessary for the following reasons:

1. It must be experimentally determined that the load-deflection characteristics of the honeycomb materials are not significantly affected by temperature. Thus load-deflection tests will be performed at temperatures spanning the range of temperatures that can be seen operationally. Tests will therefore, be done at -20°F, room temperature and 200°F.

2. It must be experimentally determined that the strength of the adhesives which bond the honeycomb itself, honeycomb sections to each other and the skin to the honeycomb sections will be adequate to preclude premature failure of the impact limiter structure. Consistent, predictable crushing characteristics depend upon the integrity of the adhesive joints. This integrity must be demonstrated by test for the configurations, temperatures, and loading conditions which may be encountered.

3. Drop accidents of the cask will result in loading sections of the impact limiter(s) in directions oblique to the axis of individual honeycomb cells. It must be confirmed that the methods used to predict the load-deflection characteristics are valid for the impact limiters for the TITAN LWT cask.

4. Analyses performed with the SCANS code have shown that secondary impact loads can be affected by the unloading characteristics specified for the impact limiters. Experimental data are required to properly specify those characteristics for the various drop orientations.

2.0 TEST OBJECTIVES

The objectives of this testing program are to:

1. Measure the static and dynamic load-deflection characteristics of prototype aluminum honeycomb material test specimens (both loading and unloading) over the applicable range of impact limiter temperature.

2. Measure the dynamic load-deformation characteristics of quarter-scale models of the actual impact limiters for various loading orientations or directions.

3. Demonstrate that the adhesive will be strong enough to prevent premature separation of the honeycomb material, of the sections of the limiter assembly, and failure of the skin.

4. Demonstrate that the methodology for predicting load deflection characteristics are applicable to the complex shapes and loading conditions potentially encountered by the TITAN LWT Cask impact limiters.

These objectives will be met by two different sets of tests. The first set consists of simple blocks of honeycomb loaded along the axis of the honeycomb cells. The second set consists of loading quarter scale models of the TITAN LWT Cask impact limiters at various angles to the cask axis. These two types of tests are described in the next section.

3.0 TEST HARDWARE DESCRIPTION

3.1 Material Tests

The test hardware consists of aluminum Alloy 5052 honeycomb material. Two types will be tested:

a. 5052 alloy hexagonal honeycomb 750 psi crush strength material, Hexcel 1/8 5052-.003, 8.1 lb/ft^3, or equivalent.

b. 5052 alloy aluminum honeycomb, 10.66 lb/ft^3, 1400 psi crush strength material, Hexcel Rigidcell corrugated honeycomb CR-ALC-3/16-.004, or equivalent.

Material "a" will be 5 inches square by 5.96 inches long (along the cell axis). Material "b" is 4 inches square by 4 inches long.

A total of twenty-seven specimens of each material is required. Each material will be tested at room temperature, at -20°F, and at 200°F. Three tests of each material will be performed at each temperature. The tests will be performed statically, at 20 feet per second, and at 44 feet per second.

3.2 Quarter-Scale Impact Limiter Tests

Six quarter scale impact limiters will be fabricated as shown in Figure 2. The honeycomb sections will be joined using adhesive as shown in the figure and will be enclosed by a 0.031-inch thick Type 304 stainless steel skin. The design of the quarter scale impact limiter will simulate the details of the design of the full-scale impact limiters. This includes the number of expanded/corrugated blocks, joint configurations, adhesive types, adhesive applications methods, curing times, etc. These details shall be documented and submitted to Westinghouse. Each limiter is provided with four mounting lugs which are bolted to the test support fixture.

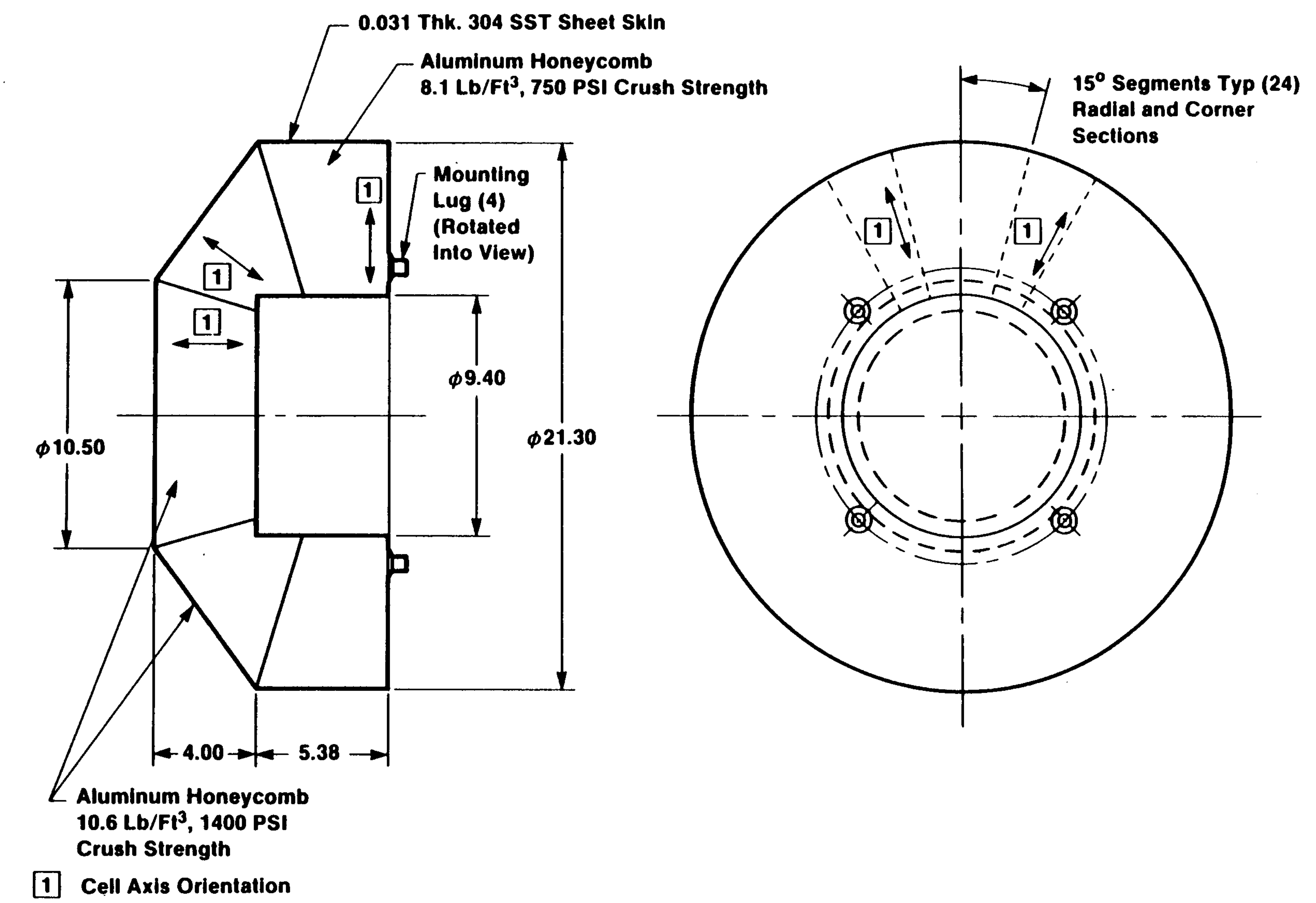

Figure 2. 1/4 Scale Aluminum Honeycomb Impact Limiter For Load-Deflection Testing

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4.0 TEST EQUIPMENT

A brief description of the equipment that will be used for the testing is given below.

4.1 Material Testing

The static load-deflection tests will be performed on a tensile/compression testing machine. A load rating of 25 kips (minimum) and a stroke range of at least 6 inches are required. The testing machine stroke rate shall be controlled.

The dynamic load-deflection tests will be performed by dropping weights on the test specimens. The test machine shall be capable of impacting the test specimen at 20 feet per second and at 44 feet per second.

The test apparatus shall include instrumentation capable of measuring and recording load as a function of deflection for all of the tests.

4.2 Quarter-Scale Impact Limiter

The quarter-scale impact limiter dynamic testing will be performed on a dead weight drop testing machine. The machine shall accommodate the test specimen and support fixture described in Section 5.2. The test machine shall be capable of impacting the impact limiter at 20 feet per second. The test apparatus shall generate load-deflection curves for all tests.

5.0 TEST DESCRIPTION

Test procedures shall be prepared for each type of test by the testing organization and provided to Westinghouse for approval prior to the start of testing.

The test procedures shall include test objectives and provisions for assuring that prerequisites for the given test have been met, that adequate instrumentation is available and used, that necessary monitoring is performed, and that suitable environmental conditions are maintained. Prerequisites shall include the following, as applicable:

- Calibrated instrumentation
- Test equipment of the required type and capacity
- Personnel trained in the use and operation of the equipment and data acquisition systems
- Implementation of safety precautions
- Data acquisition systems, including photography and video tape equipment

Appropriate sections of standard testing procedures such as those developed by the American Society for Testing Materials (ASTM), and American National Standards Institute (ANSI) may be used wherever possible in lieu of specially written test procedures. Such standard procedures must include adequate instructions to assure the required quality of work.

Test procedures shall specify appropriate quality assurance inspection and verification points, including those specified by contract provisions for customer representatives.

Test records shall identify the following, as applicable to the specific test:

- o Item tested (uniquely identified)
- o Date of test
- o Name of test operator or data recorder
- o Type of observation
- o Results and comparison with predicted values
- o Action taken in conjunction with any deviation noted
- o Name of person evaluating the test results

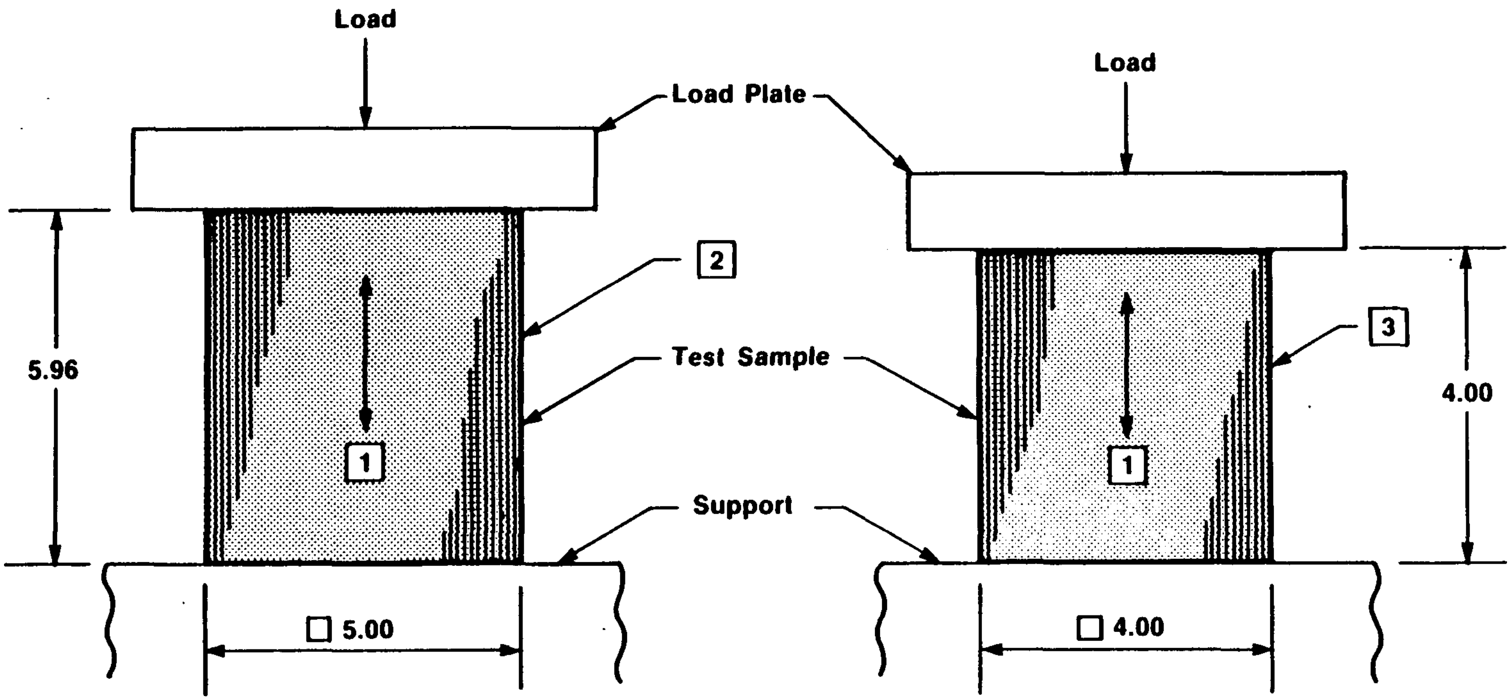
5.1 <u>Material Testing</u>

A summary of the material testing program is shown in matrix form in Table 1. A schematic of the test set-up is illustrated in Figure 3. Testing will be conducted in a "shop floor" environment. Test specimens will be at the temperatures described in Table 1. The tests will be run statically and dynamically (20 ft/sec and 44 ft/sec). The general procedure for testing is as follows.

<u>Static Testing</u>

5.1.1 The test section is placed in a hydraulic press. The load is applied to the test specimen. The load is increased by indexing the press at 5 inches/minute. NOTE: The load can be indexed at a higher or lower rate, depending on the compression machine capabilities. A higher rate is desirable.

5.1.2 The applied load is measured from hydraulic pressure or load cell readings and recorded simultaneously with the deflections.



Notes:

1 Cell Axis Orientation

2 5052 Alloy Hexagonal Aluminum Honeycomb
1/8-5052-.002, 8.1 LB/FT^3, 750 PSI Crush Strength

3 5052 Rigicell Aluminum Corrugated Honeycomb
CR-ALC-3/16-.004, 10.6 LB/FT^3, 1400 PSI Crush Strength

Figure 3. Aluminum Honeycomb Material Load-Deflection Test

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TABLE 1

Material Test Matrix

Specimen	Static Test			Dynamic Test (20 ft/sec)			Dynamic Test (44 ft/sec)		
	Room Temp	-20°F	200°F	Room Temp	-20°F	200°F	Room Temp	-20°F	200°F
750 psi Crush Strength Honeycomb	3	3	3	3	3	3	3	3	3
1400 psi Crush Strength Honeycomb	3	3	3	3	3	3	3	3	3

5.1.3 Displacements shall be measured with linear variable displacement transducers or other device producing a signal that can be recorded simultaneously with the load.

5.1.4 The loading is increased until the specimen is crushed to 30% of its initial height.

5.1.5 For tests at other than room temperature, the outer surface of the specimen must be at the desired temperature prior to testing. Provisions shall be made to minimize temperature changes during testing. Temperatures shall be maintained to within ± 5°F.

5.1.6 The test specimens shall be examined for damage and photographed. The inspection results shall be documented.

Dynamic Testing

5.1.7 The test specimen is secured to a support fixture.

5.1.8 The test machine drop weights and drop heights are established for the desired test case based on the required impact velocity and energy required. The load is released and dropped onto the impact limiter.

5.1.9 The applied load shall be measured to an accuracy of 5% and recorded.

5.1.10 Displacements of the load to impact limiter interface shall be are measured with linear variable displacement transducers or equivalents and recorded simultaneously with the load.

5.1.11 The impact limiters shall be examined for damage. The inspection results shall be documented.

5.2 Quarter-Scale Impact Limiter Testing

A summary of the quarter-scale impact limiter testing is presented in Table 2. The test loads will be applied to the test specimens in five different orientations:

- o side (Figure 4)
- o 17.5° oblique (Figure 5)
- o 53.5° oblique (Figure 6)
- o CG-over-corner (80.6°) (Figure 7)
- o end (Figure 8)

Each test will be performed two times, for a total of ten tests. Each test specimen will be used twice (except for the end drop). The specimens will be rotated 180° in the support fixture and loaded on the opposite side. The end-loaded specimen cannot be reused. Therefore a total of six specimens are required for the ten tests.

It is anticipated that the quarter-scale impact limiters will be tested at room temperature. However, based on the results of the materials tests, which are performed at various temperatures, it may be decided to run the tests at either -20°F or +200°F, or a combination. This decision will be made by the Westinghouse representative prior to the start of quarter-scale impact limiter testing.

The general procedure for testing is as follows:

5.2.1 The test specimen is secured to the support fixture.

5.2.2 The support fixture is oriented for the given test condition.

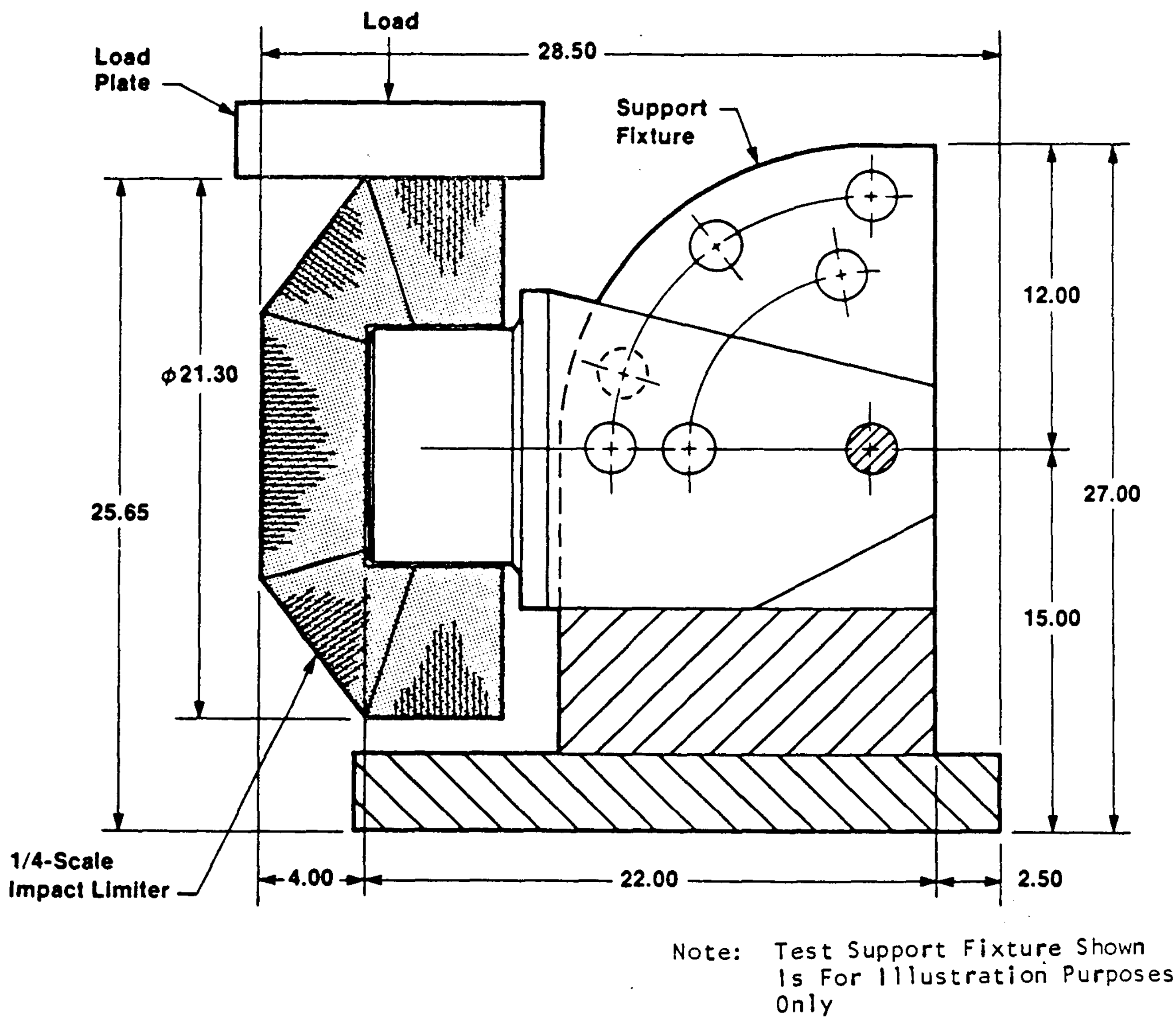

Figure 4. 1/4 Scale Aluminum Honeycomb Impact Limiter Load-Deflection Test, Side Drop Orientation

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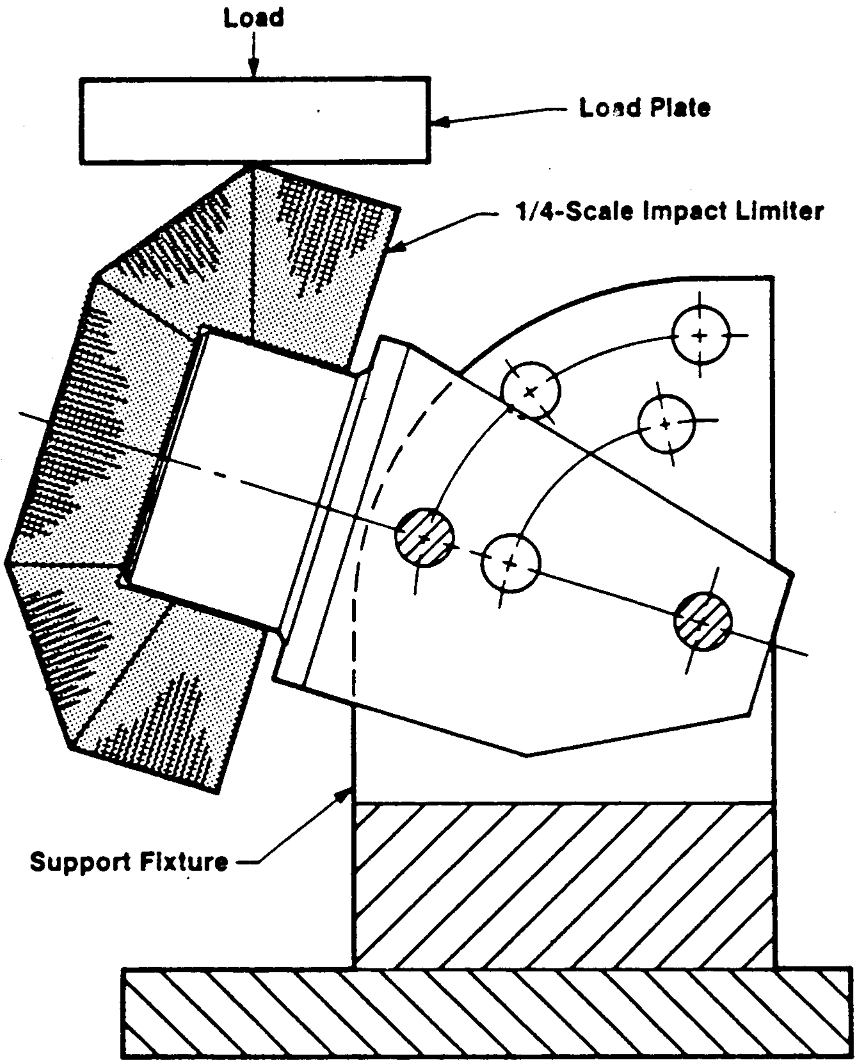

Note: Test Support Fixture Shown Is For Illustration Purposes Only

Figure 5. 1/4-Scale Aluminum Honeycomb Impact Limiter Load-Deflection Test, 17.5° Orientation

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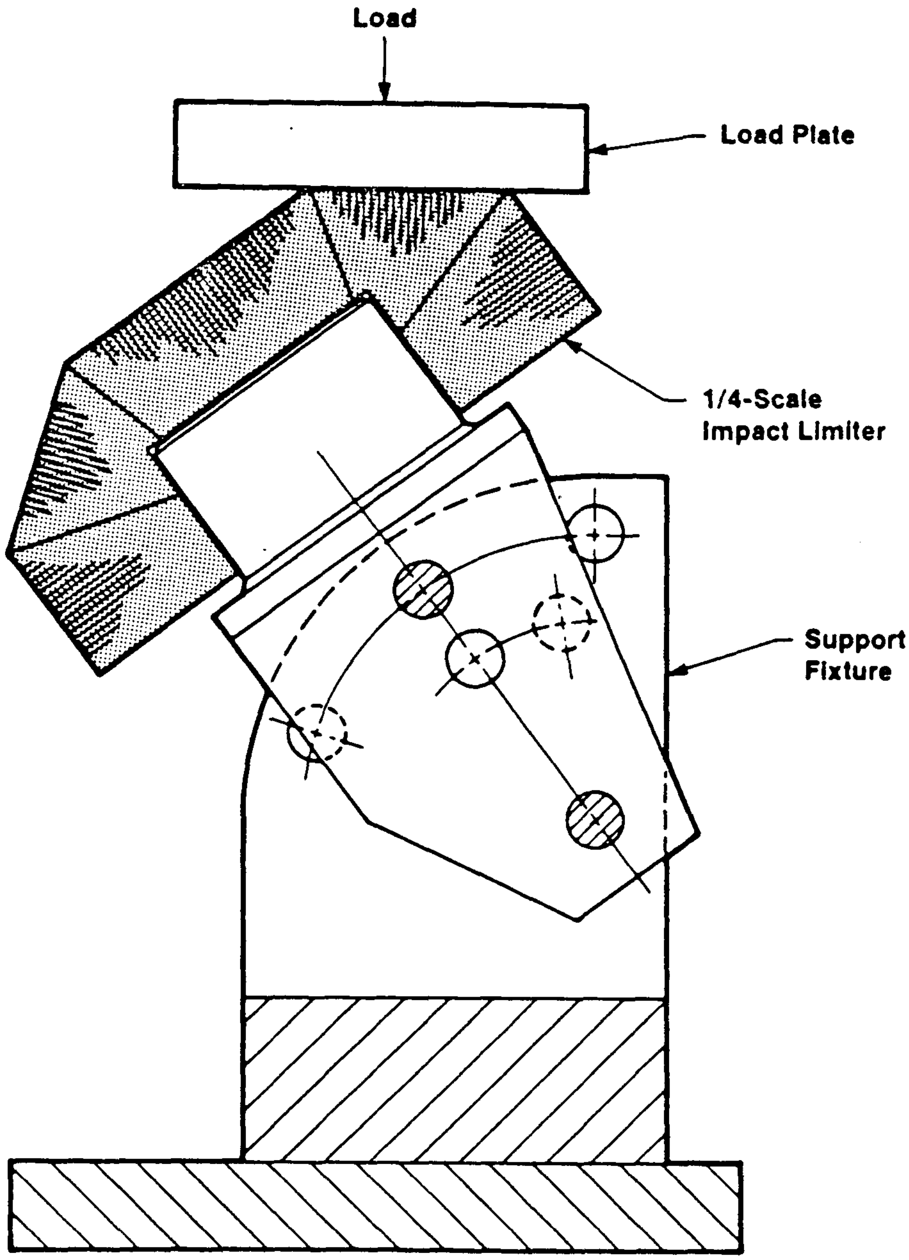

Note: Test Support Fixture Shown Is For Illustration Purposes Only

Figure 6. 1/4-Scale Aluminum Honeycomb Impact Limiter Load-Deflection Test, 53.5° Orientation

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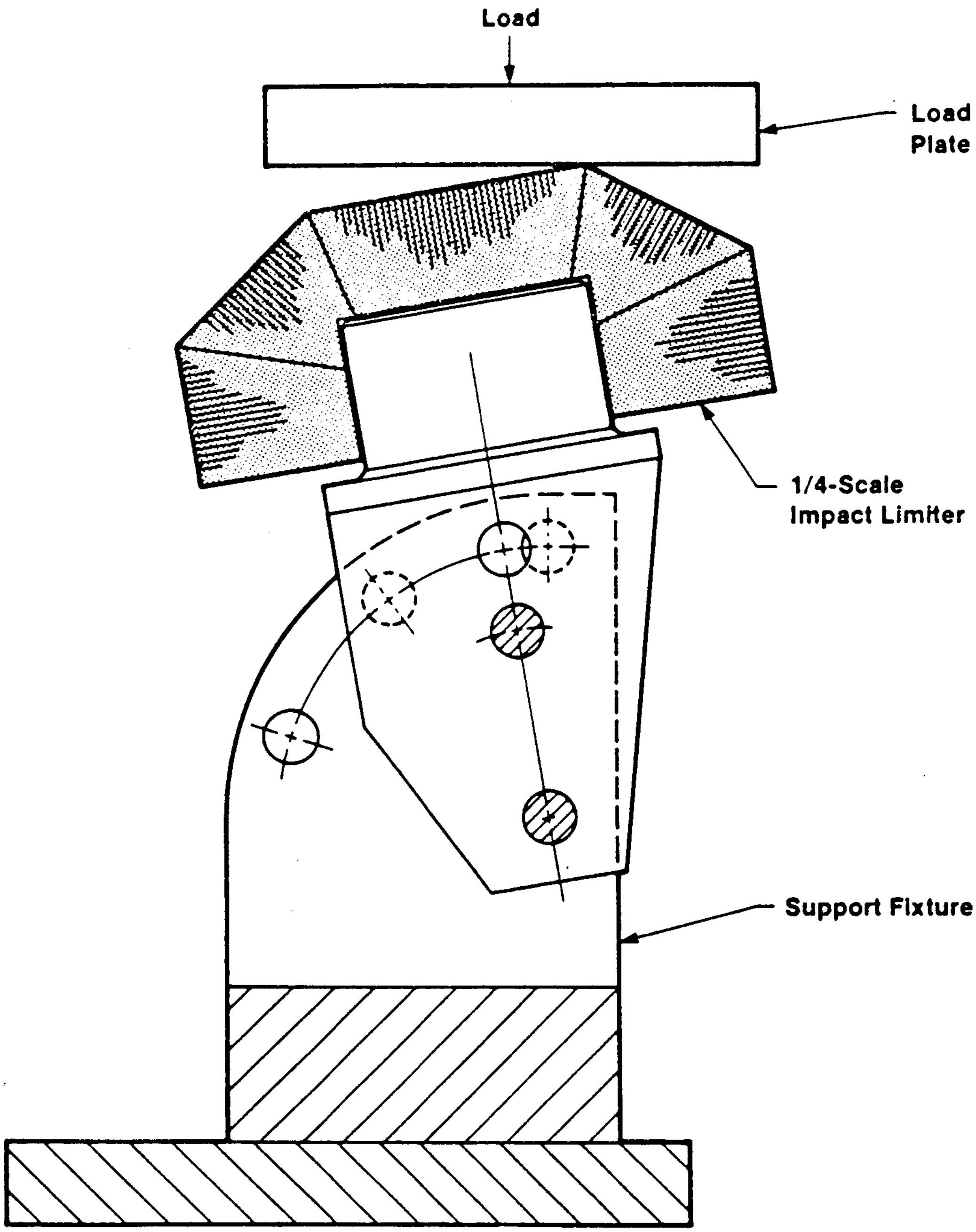

Note: Test Support Fixture Shown Is For Illustration Purposes Only

Figure 7. 1/4-Scale Aluminum Honeycomb Impact Limiter Load-Deflection Test, 80.6°, C.G.-Over-Corner Orientation

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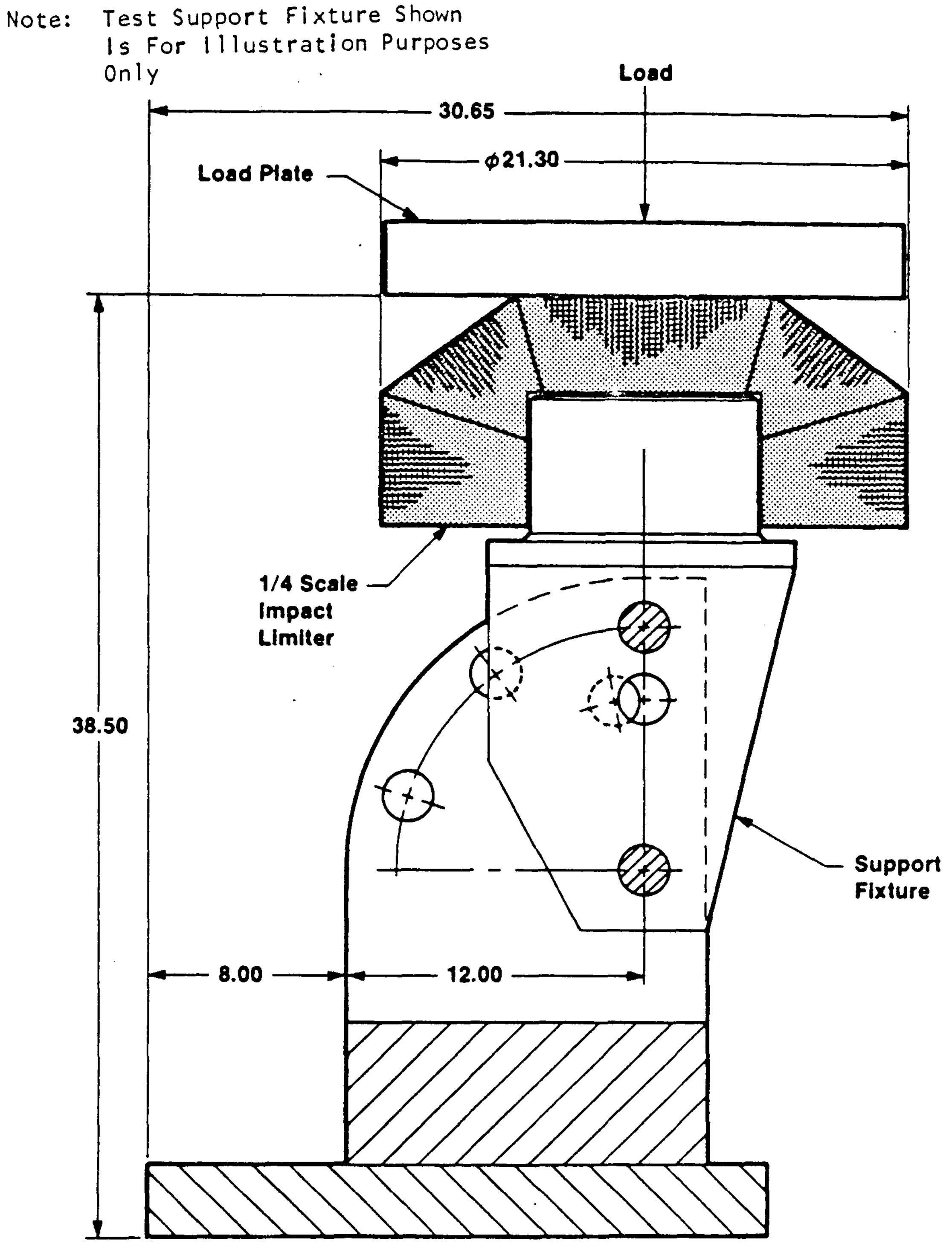

Figure 8. 1/4 Scale Aluminum Honeycomb Impact Limiter Load-Deflection Test, End Drop Orientation

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TABLE 2

Quarter-Scale Impact Limiter
Test Matrix

Drop Test	Number of Tests	Crush Depth	Estimated Peak Load
Side Drop	2	4.1 in	83,000 lb
17.5° oblique	2	3.4 in	75,000 lb
53.5° oblique	2	3.7 in	130,000 lb
CG-Over-Corner Drop (80.6°)	2	4.2 in	214,000 lb
End Drop	2	2.2 in	246,000 lb

5.2.3 A conbination of drop weight and height shall be selected is provided 304,000 inch-lbs. of energy at initial contact. The height shall be selected to provide a velocity of at least 20 ft/sec at initial contact.

5.2.4 The applied load shall be measured to an accuracy of 5% and recorded.

5.2.5 Displacements of the load to impact limiter interface shall be measured with linear variable displacement transducers or equivalent transducers and recorded simultaneously with the load.

5.2.6 The impact limiters shall be examined for damage. The inspection results shall be documented.

5.3 Data Recording

Still photographs will be taken of all test set-ups and of the test pieces before and after each test. All the tests will be videotaped. A minimum of one copy of the videotape and three sets of black and white photographs will be provided to Westinghouse. All test data will be provided to Westinghouse.

Tools, gages, instruments, and other measuring equipment used for testing shall have the precision, accuracy, and range required to establish conformance with specified requirements and shall be calibrated and adjusted to maintain precision and accuracy within necessary limits. All calibrations should use standards traceable to the National Institute of Science and Technology.

5.4 Quality Assurance

The testing will be performed in accordance with a quality assurance program that meets applicable requirements of ANSI/ASME NQA-1. A pre-award quality audit will be performed by Westinghouse to assure compliance with NQA-1.

Chemical and physical certifications shall be provided for all aluminum honeycomb material used in testing.

A certificate of compliance and test results for the adhesive mechanical properties shall be provided. The material shall conform to MIL-A-25463 and Federal Specification MMM-A-132. The adhesive shear tests shall be conducted at room temperature, -20°F, and 200°F.

Testing shall be witnessed by a Westinghouse representative. A minimum of 72 hours notice shall be given to Westinghouse prior to the commencement of testing.

6.0 DATA EVALUATION AND REPORTING

The test data will be used to confirm the feasibility of the use of aluminum honeycomb for impact absorption. This feasibility will be demonstrated if:

1. The measured load-deflection curves for the straight test (materials test) specimens exhibit a nearly flat force plateau over most of the deformation.

2. The large deflection and strains do not cause catastrophic failure.

3. No substantial premature failure of the adhesives during testing.

4. The structures exhibit good energy absorption characteristics.

A report will be prepared by Westinghouse which documents the test articles, test methods and procedures, and the test data. Photos of the test arrangement and equipment and test articles before and after testing will be included. In addition, the report will include a test evaluation section written around the items listed in the above paragraph and a section on recommendations and conclusions.

7.0 SCHEDULE

The schedule for the proposed engineering testing is given in Figure 9. The entire test program, including preparation of the test report, is expected to be completed in 5.5 months from the date of approval of the program by DOE.

8.0 REFERENCES

1. Hexcel Technical Service Bulletin, TSB 120, "Mechanical Properties of Hexcel Honeycomb Materials," Hexcel Structural Products, Dublin, CA, 1988

2. Hexcel Technical Service Bulletin, TSB 122, "Preliminary Selection of Honeycomb Energy Absorption Systems," Hexcel Structural Products, Dublin, CA, 1988.

3. "Development of Circumferential Honeycomb Impact Limiters for Defense High Level Waste Shipping Cask," A. Zimmer, et. all., Waste Management '88, Tucson, Arizona, March 1988.

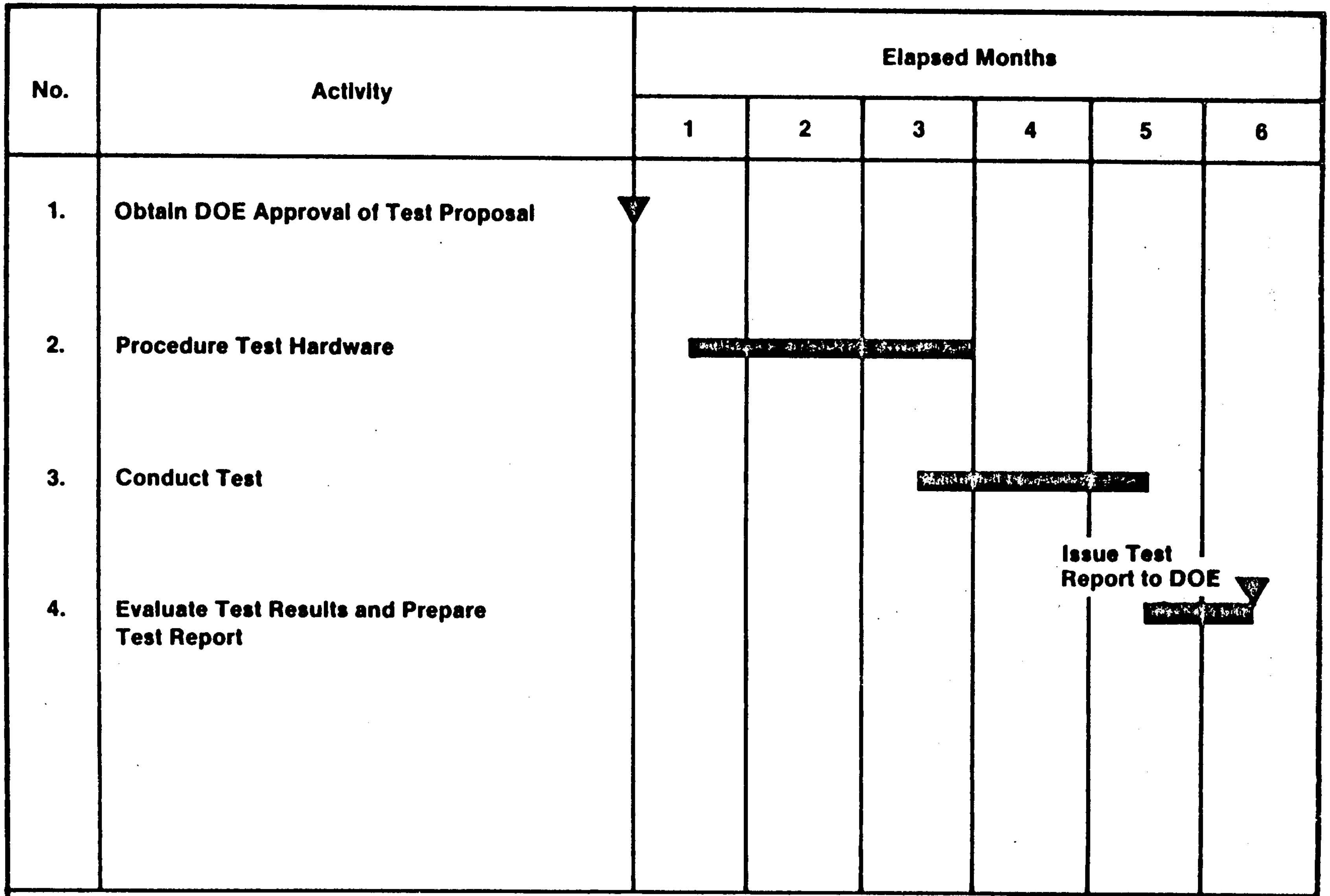

Figure 9. Schedule for Aluminum Honeycomb Impact Limiter Phase I Engineering Test

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9. SAFETY/QUALITY ASSURANCE ISSUES

9.1 Safety Issues

No safety issues have been identified for the TITAN LWT Cask and Ancillary Equipment.

9.2 Quality Assurance Issues

The Preliminary Design of the TITAN LWT Cask and Ancillary Equipment has been implemented in conformance with the requirements of the following Quality Assurance documents:

- Quality Assurance Program Plan, NWD-TR-005, Revision 1, approved by the U.S. Nuclear Regulatory Commission Transportation Branch on March 30, 1988.
- TITAN Cask Project Quality Program Plan, NWD-TR-021, Revision 0.
- Nuclear Waste Department Quality Assurance Manual, QAP 1.
- Nuclear Waste Department Procedures Manual.

All of the design and analyses activities in support of the LWT Cask and Ancillary Equipment have been classified as Quality Level 1 (as defined in QMP DOE/ID-10178 and NWD-TR-021) and the appropriate quality assurance program requirements and procedural controls have been applied to the performance of these activities.

Two Quality Assurance audits of the TITAN Cask program were conducted by EG&G-ID, since the award of the contract to Westinghouse on May 13, 1988. The first audit was conducted during July 11-14, 1988 and the scope of the audit was to verify compliance with the NRC - approved Quality Assurance Program. The second audit was conducted during

March 21-24, 1989 and its scope was to verify compliance with the DOE-ID Quality Management Plan DOE/ID-10178 and ANSI/ASME NQA-1. No findings or conditions adverse to quality were identified as a result of the two audits.

An internal Quality Assurance audit of the TITAN Cask Project was conducted during April 27 - May 3, 1989. The scope of that audit was to verify compliance with the NWD Quality Assurance Program. The overall conclusion of that audit was that the NWD Quality Assurance Program was being implemented in an effective manner.

There are therefore no Quality Assurance issues that have been identified on the TITAN Cask Project.

APPENDIX A
TRADEOFF STUDIES AND EVALUATIONS

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EVALUATION OF THE OPTIMAL CAPACITIES OF A COMMON USE CASK DESIGN VERSUS SINGLE USE CASK DESIGNS

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1.0 INTRODUCTION

The current Westinghouse Titan Legal Weight Truck (LWT) cask design approach uses a single cask system with interchangeable baskets to accommodate both PWR and BWR spent fuel assemblies. The results of a study to evaluate the optimal capacity of such a common use cask design versus single use cask designs that can transport only one type of spent fuel (PWR or BWR) are presented in this report.

The common use cask design developed in support of the Alternative Material Feasibility Study (Reference 1) was used in the evaluation. That design, and supporting shielding calculations, were extrapolated to the single use cask configuration to provide a consistent basis for comparison.

The following conclusions are drawn from the study:

- Single use cask designs will permit improved payload capacities over a common use cask design. The single use casks will be able to accommodate 4 PWR fuel assemblies and 9 BWR fuel assemblies compared to 1 to 3 PWR fuel assemblies and 5 to 7 BWR fuel assemblies possible with common use casks of various designs.

- The bulk of the PWR fuel assemblies and all of the BWR fuel assemblies can be shipped in two single use cask configurations with significant benefits in system costs compared to using a single common use cask design.

- A separate cask configuration for shipping the Combustion Engineering (CE) spent fuel assemblies that are longer than 176 inches needs to be developed in parallel with the single use cask designs.

A brief description and inventories of the PWR and BWR fuel assemblies required to be accommodated by the LWT cask system is included in Section 2 of this report. The results of the evaluations are presented in Section 3, and the conclusions and recommendations from the study are provided in Sections 4 and 5, respectively. The references cited in the report are listed in Section 6.

2.0 SPENT FUEL DESCRIPTION AND INVENTORIES

The LWT cask is required to accommodate the PWR fuel assemblies shown in Table 1, and the BWR fuel assemblies shown in Table 2 (Reference 2). The fuel assemblies are listed in order of increasing assembly lengths. The inventory of spent fuel at reactor sites as of the end of 1985 includes about 17000 PWR assemblies and 26000 BWR assemblies, considering only those fuels listed in Tables 1 and 2. The projected inventories for the period between 1986 and 2020 are 101,216 PWR assemblies and 127,175 BWR assemblies, based on no new orders for nuclear power plants and with extended burnup of the fuel (Reference 3).

Of special interest in the context of the current study are the three CE 16x16 PWR assemblies included at the end of Table 1. Single use casks are economically viable only if they have a payload capacity greater than that of a common use cask. For a single use PWR cask, this additional capacity is achievable only by eliminating these CE 16x16 assemblies, which are approximately 11 inches longer than the others, from the payload. They are used in the San Onofre 2, San Onofre 3, Arkansas Nuclear One 2, and Palo Verde 1, 2, and 3 power plants. The inventory of these CE spent fuels as of the end of 1985 includes 65 assemblies from San Onofre 2 and 226 assemblies from Arkansas Nuclear One 2. The projected inventories from the six CE nuclear power plant units are expected to build up at the rate of approximately 106 assemblies every 16 months from each of the San Onofre and Palo Verde units, and approximately 80 assemblies every 16 months from Arkansas Nuclear One 2 (Reference 4). At these rates, the spent fuel storage pools at those reactor sites should be filled to capacity within 12 years. This is significant because it indicates the need to design a cask system to handle the CE spent fuel under Initiative I of the From-Reactor Cask Development Program.

TABLE 1

PWR FUEL ASSEMBLIES REQUIRED TO BE ACCOMMODATED IN LWT CASK

ASSEMBLY VENDOR	ARRAY SIZE	Version	ASSEMBLY CROSS-SECTION INCHES	ASSEMBLY LENGTH INCHES
Westinghouse Electric	14 x 14	Std/SC	7.760	137.06
Westinghouse Electric	15 x 15	Std/SC	8.420	137.06
Babcock & Wilcox	15 x 15	St. Stl.	8.466	137.63
Combustion Engineering	14 x 14	Ft. Calhoun	8.100	146.00
Exxon/ANF	14 x 14	Ft. Calhoun	--	147.00
Exxon/ANF	15 x 15	Comb. Engrg	8.250	148.85
Combustion Engineering	14 x 14	Std.	8.100	157.00
Westinghouse Electric	14 x 14	Model C	8.030	157.24
Exxon/ANF	14 x 14	Comb. Engrg	8.110	157.24
Combustion Engineering	16 x 16	St. Lucie 2	8.100	158.13
Exxon/ANF	15 x 15	Westinghouse	8.426	159.70
Westinghouse Electric	14 x 14	Std/ZA	7.760	159.71
Westinghouse Electric	14 x 14	OFA	7.760	159.71
Westinghouse Electric	14 x 14	Std/ZCB	7.760	159.71
Exxon/ANF	17 x 17	Westinghouse	8.426	159.71
Westinghouse Electric	15 x 15	Std/ZC	8.434	159.71
Westinghouse Electric	17 x 17	Std	8.434	159.77
Westinghouse Electric	17 x 17	OFA	8.434	159.77
Westinghouse Electric	15 x 15	OFA	8.424	159.71
Westinghouse Electric	17 x 17	VANTAGE 5	8.426	160.10
Exxon/ANF	14 x 14	Westinghouse	7.763	160.13
Exxon/ANF	14 x 14	Top Rod	7.763	160.13
Babcock & Wilcox	15 x 15	Mark BZ	8.536	165.63
Babcock & Wilcox	15 x 15	Mark B	8.536	165.63
Babcock & Wilcox	17 x 17	Mark C	8.536	165.72
Combustion Engineering	16 x 16	San Onofre	8.100	176.80
Combustion Engineering	16 x 16	ANO 2	8.100	176.80
Combustion Engineering	16 x 16	SYSTEM 80	8.100	178.25

(From Reference 2)

TABLE 2

BWR ASSEMBLIES REQUIRED TO BE ACCOMMODATED IN LWT CASK

ASSEMBLY VENDOR	ARRAY SIZE	Version	ASSEMBLY CROSS-SECTION INCHES	ASSEMBLY LENGTH INCHES
General Electric	7 x 7	Humboldt Bay	4.000	95.00
General Electric	8 x 8	/2, 3	5.518	171.00
Exxon/ANF	7 x 7	GE	5.440	171.25
General Electric	7 x 7	/2,3:V2	5.518	171.40
General Electric	7 x 7	/2,3:V1	5.518	171.40
General Electric	7 x 7	/4,5	5.518	176.00
General Electric	8 x 8	/4,5:V1	5.518	176.20
General Electric	8 x 8	/4,5:V2	5.518	176.20
Exxon/ANF	8 x 8	JP-3	5.440	178.50
Exxon/ANF	8 x 8	JP-4,-5	5.440	178.50

(From Reference 2)

3.0 EVALUATION OF OPTIMAL CAPACITIES OF ALTERNATIVE TITAN LWT CASK CONFIGURATIONS

The results of the evaluations performed to establish the optimal payload capacities of common use and single use cask configurations are presented in this section. The cask design based on the alternative structural material (Reference 1) was used in the evaluations. Sufficient scoping structural and shielding analyses were performed in support of the feasibility study presented in Reference 1 to ensure that payload predictions based on the results of the analyses can be met in the final design with a high degree of assurance. The results of those analyses performed in support of a common use cask design were extrapolated to the single use cask designs to arrive at cask weights and payload capacity estimates with a reasonable level of confidence.

3.1 Common Use Cask for PWR and BWR Assemblies

A single cask design can be optimized to accommodate a maximum of 3 PWR fuel assemblies or 7 BWR assemblies with a cask cavity diameter of 24.6 inches and cavity length of 180 inches (Reference 1). This cask configuration will accept all of the PWR and BWR fuel assemblies required to be accommodated by Reference 2. The estimated weights of the cask and transporter are shown in Table 3.

3.2 Single Use Cask for PWR Assemblies

A single use cask designed to accommodate all of the PWR assemblies listed in Table 1 will have the same payload capacity as the common use cask. However, by excluding the three CE 16x16 fuel assemblies that are over 176 inches long, it is possible to optimize a cask design with a cavity diameter of about 26.8 inches and cavity length of 167 inches to accommodate a maximum of 4 PWR assemblies depending on the cask materials selected. Table 4 shows the estimated weight of the cask and transporter for this configuration. This cask system will be able to transport approximately 84 percent of the total PWR spent fuel inventory with significant benefits in total system costs as

TABLE 3

Estimated Weight of Common Use LWT Cask

ITEM	Weight (lb.)
Gross Vehicle Weight	80,000
Tractor	16,000
Trailer and Support Systems	9,000
Empty Cask	
- with PWR Basket	48,300
- with BWR Basket	48,500
Loaded Cask	
- with 3 PWR assemblies	52,845
- with 7 BWR assemblies	52,840
Weight Margin	
- with PWR fuel	2,155
- with BWR fuel	2,160

(From Reference 1)

Note: The weight of a single use cask to transport CE PWR assemblies longer than 176 inches will be the same as that for the common use cask (PWR case) given above.

TABLE 4

Estimated Weight of Single Use LWT Cask for PWR Assemblies

ITEM	Weight (lb.)
Gross Vehicle Weight	80,000
Tractor	16,000
Trailer and Support Systems	9,000
Empty Cask	48,000
Loaded Cask with 4 PWR assemblies	54,060
Weight Margin	940

Note: This cask will not accommodate the three Combustion Engineering 16 x 16 fuel assemblies that are over 176 inches long.

discussed in Section 4. The CE spent fuel that cannot be accommodated by this single use cask design can be transported in a separate single use cask having the same cavity dimensions and shielding requirements as the common use cask design described in Section 3.1.

3.3 Single Use Cask for BWR Assemblies

A single use cask design can be optimized to accommodate a maximum of 9 BWR fuel assemblies with a cask cavity diameter of about 26.8 inches and cavity length of 180 inches depending on the cask materials selected. The increase in capacity compared to the common use cask is because of the slightly reduced shielding required for BWR spent fuel compared to PWR spent fuel which provides the weight margin to accommodate two additional assemblies. The estimated weights of the cask and transporter are given in Table 5.

TABLE 5

Estimated Weight of Single Use LWT Cask for BWR Assemblies

ITEM	Weight (lb.)
Gross Vehicle Weight	80,000
Tractor	16,000
Trailer and Support Systems	9,000
Empty Cask	48,200
Loaded Cask with 9 BWR assemblies	53,780
Weight Margin	1,220

4.0 DISCUSSION OF RESULTS AND CONCLUSIONS

The results of the evaluation presented in the preceding section show that for a cask design based on the alternative structural material proposed by Westinghouse:

- A common use cask will be able to accommodate a maximum of 3 PWR fuel assemblies or 7 BWR fuel assemblies without any restrictions as to the size of the assemblies depending upon the cask material used (see Reference 1).

- A single use cask designed to accept all PWR fuel assembly sizes except for three CE fuel assemblies that are over 176 inches long can accommodate a maximum of 4 PWR fuel assemblies depending on the cask material used.

- A single use cask design identical to the common use cask configuration for PWR fuel could be used to transport those CE fuel assemblies and can accommodate a maximum of 3 PWR fuel assemblies depending on the cask material used.

- A single use cask design for BWR fuel can accommodate a maximum of 9 BWR fuel assemblies depending on the cask material used.

The significant savings in total life cycle costs possible with the use of single use casks compared to common use casks can be seen in Figure 1 taken from Reference 5. The life cycle costs are based on the following assumptions and input data for a generic cask design:

- Life Cycle activities modeled for a total shipment of 110,000 MTU

- Modal mix of one-third truck and two-thirds rail shipments

- 60% PWR, 40% BWR

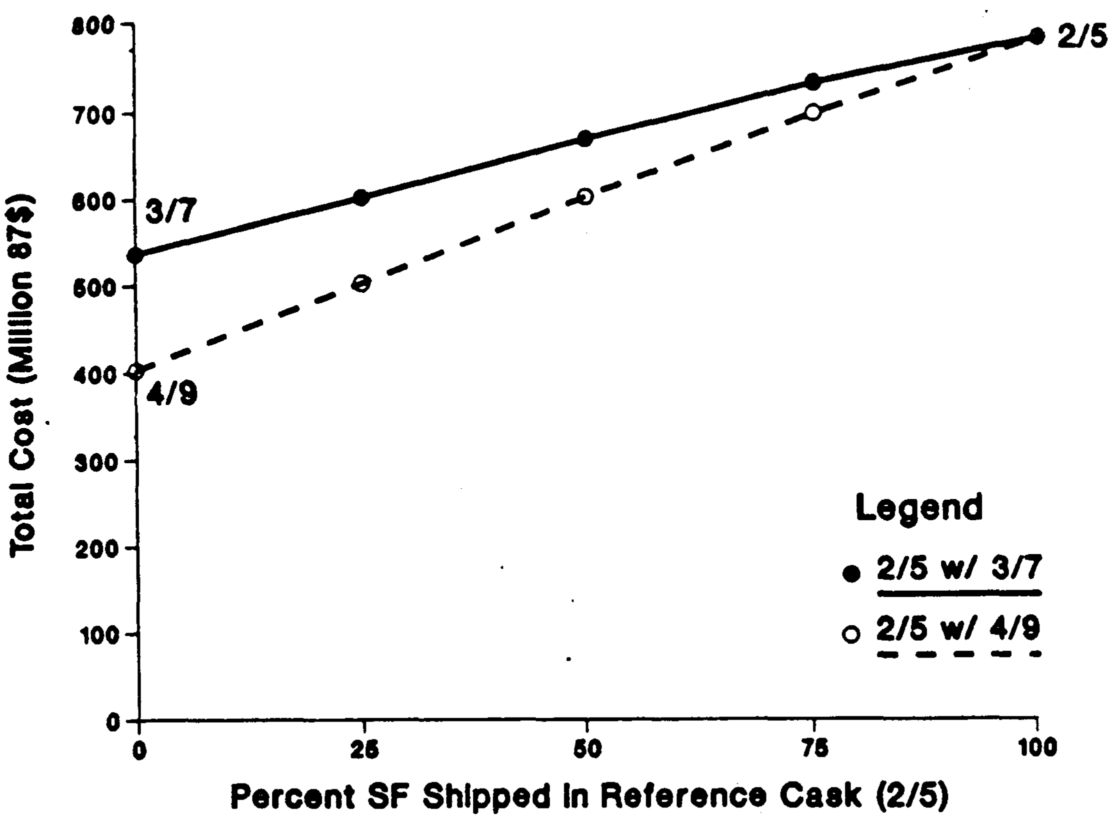

Figure 1 Truck Cask System Life Cycle Costs

- One origin and one destination
- Development/Certification, Cask Fleet acquisition costs
- Cask Handling/Operation/Inspection costs
- Cask Decommissioning/Salvage costs

Figure 1 shows that if all truck shipments of spent fuel are made in single use casks (capacity of 4 PWRs, 9 BWRs) instead of common use casks (capacity of 3 PWRs, 7 BWRs), a total savings of approximately $140,000,000 can be obtained in total life cycle costs. Hence even if 16 percent of the PWR spent fuel (representing the CE fuel assemblies that have to be accommodated in a separate cask) is shipped in casks having a reduced capacity (3 PWRs), the savings in life cycle costs would outweigh the additional development costs of that cask. It is noted that the actual life cycle costs for the cask system proposed by Westinghouse will be slightly different and the data in Figure 1 is presented to indicate the comparative magnitude of expected savings.

It is concluded from the current study that a three cask system described above would provide the most cost effective means of transporting PWR and BWR spent fuels using LWT shipments.

5.0 RECOMMENDED ACTIONS BY DOE AND NWD

Based on the results of the evaluations and conclusions presented in the preceding sections, the following recommendations are offered:

1. Evaluations should be performed by DOE in the context of the overall spent fuel transportation system to assess the need for shipping the CE 16x16 spent fuel assemblies that are over 176 inches long in a LWT cask (versus OWT cask or Rail/Barge cask).

2. A two or three cask system designed for single use service should be implemented in lieu of the reference Westinghouse common use cask system, based on the results of the preceding evaluations.

6.0 REFERENCES

1. "Titan Legal Weight Truck Cask Alternative Material Feasibility Study", NWD-TR-008, Rev. 0, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, July 1988, (Westinghouse Proprietary).

2. Contract No. DE-AC07-88ID12699 for Cask Systems Development, Initiative I, Development of From-Reactor Casks: Task 4A Legal Weight Truck Cask Development, Task 4B Overweight Truck Cask Development, issued by U. S. Department of Energy.

3. "Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation", DOE/RW-0184, U. S. Department of Energy, Office of Civilian Radioactive Waste Management, December, 1987.

4. Personal Communication, N. Breckenridge, Combustion Engineering, to B. R. Nair, Westinghouse Nuclear Waste Department, August 2, 1988.

5. Hofmann, P., "Comparisons of Life Cycle Costs," Spent Fuel Burnup Credit Workshop, February, 1988.

APPENDIX I

Additional Information on Dedicated Versus Common Use Cask Designs

This Appendix provides additional information, including the cost and schedule impacts of implementing dedicated LWT cask designs, to support a DOE decision on this issue.

The conclusions of the earlier study were that: (1) dedicated cask designs will permit improved payload capacities over a common use cask design, (2) significant benefits in life cycle operating costs would be obtained with dedicated casks, and (3) the payload and associated cost benefits are possible only if the LWT casks are not required to transport those Combustion Engineering 16 x 16 fuel assemblies that are longer than 166 inches.

The payload capacity estimates for dedicated casks in that study were based on extrapolation of shielding calculations and the design configuration of the common use cask during the early stages of its preliminary design and were therefore necessarily conservative. With the completion of the preliminary design and optimization of the cask geometry and shield thicknesses, we have been able to estimate the payload capacities for dedicated casks with a higher degree of accuracy. Revised calculations show that dedicated LWT casks will be able to accommodate 4 PWR assemblies and 10 BWR assemblies while staying well within the weight allocation of 54,000 lb. for the loaded cask. This represents an increase by 1 BWR assembly over the earlier estimate of 4 PWR assemblies and 9 BWR assemblies.

Westinghouse continues to maintain its position that dedicated LWT cask designs be implemented instead of a common use cask design. A DOE decision on this issue in early 1990 will enable implementation with minimum cost and schedule impact to the cask development program.

TITAN LEGAL WEIGHT TRUCK CASK

IMPACT OF
REDUCING THE ALLOWABLE 2-METER DOSE RATE
FROM 10 MREM/HR TO 2 MREM/HR

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1.0 INTRODUCTION

The Statement of Work for the Cask Systems Development, Initiative 1, requires that the contractor conduct a number of trade-off and impact evaluations on cask payload capacities and costs. One of the design considerations to be evaluated is the reduction of the allowable 2-meter dose rate from 10 mrem/hr to 2 mrem/hr.

The purpose of this report is to provide an evaluation of the impact of the reduction of the allowable dose rate on the payload capacity of the Westinghouse Titan Legal Weight Truck (LWT) cask and to provide an estimate of the impact on total life cycle costs that would be associated with the change in payload capacity.

The Titan LWT cask has an allocated weight of 54,000 lbs out of a gross vehicle weight of 80,000 lbs. At this stage of the preliminary design of the Titan cask, Westinghouse believes that a common use cask (i.e., capable of transporting both PWR and BWR assemblies) can be designed, within this weight allocation, to transport 3 PWR assemblies or 7 BWR assemblies. This payload capacity is achieved through the use of depleted uranium (DU) for the gamma shielding material, Boro-Silicone for the neutron shielding material and titanium, Grade 9, for the main structural material of the cask. The impact limiters are toroidal shells fabricated from aluminum.

The weight of the current preliminary design is, however, marginal. Thus, a reduction in the allowable dose rate by a factor of five will, of course, mean a reduction in the payload capacity. The evaluation of the impact on payload capacity of such a reduction in allowable dose rate is presented in Section 2. An evaluation of associated increases in life cycle costs is presented in Section 3. The conclusions are presented in Section 4 with references given in Section 5.

2.0 EVALUATION OF PWR/BWR PAYLOADS CAPACITIES

Preliminary analyses indicate that the Westinghouse Titan LWT Cask (a common use cask capable of transporting both PWR and BWR spent assemblies) can accommodate a maximum of 3 PWR assemblies or 7 BWR assemblies with a cask cavity diameter of 23.76 inches and cavity length of 180 inches (Reference 1). The estimated weight of the cask loaded with 3 PWR assemblies and the estimated weight of the cask loaded with 7 BWR assemblies are shown in Column 1 of Table 1. Table 1 gives the weight margin of the cask to be 760 lbs for the PWR assemblies and 800 lbs for the BWR assemblies. These weights are based on a design having shielding which meets the allowable 2-meter dose rate of 10 mrem/hr. Scoping shielding analyses show that the weight of the cask increases by approximately 19% when the allowable 2-meter dose rate is decreased by a factor of 5. The new estimated weight of the cask loaded with 3 PWR assemblies or 7 BWR assemblies is shown in Column 2 of Table 1. The increase in cask weight is created by the additional gamma and neutron shielding added to the sides, top, and bottom of the cask. Therefore, in order to maintain the cask weight requirement of 54,000 lbs (Reference 2), the cask payload capacity must be reduced. Reducing the cask payload capacity decreases the radiation source that must be shielded and also allows the cask cavity diameter to be reduced if necessary.

Since a PWR assembly has a source term approximately four times higher than a PWR assembly, the PWR payload capacity was determined first. The BWR capacity payload was established next by determining the number of BWR assemblies that could be accommodated in the new cask cavity.

2.1 Evaluation of PWR Payload Capacity

Since the weight of a cask which can accommodate a payload capacity of 3 PWR assemblies or 7 BWR assemblies is marginal, it is obvious that any increase in shielding required to reduce the dose rates by

Table 1

Estimated Weight of the LWT cask for Various
Assembly Payload Capacities

	3 PWR/ 7 BWR	3 PWR/ 7 BWR	2 PWR/ 4 BWR
1. Dose Rate at 2-meters (mrem/hr)	10	2	2
2. Gamma Shielding (inches of DU)			
Side	2.87	3.08	3.01
Top	1.35	1.56	1.49
Bottom	2.30	2.51	2.44
3. Neutron Shielding (inches of Boro-Silicon)			
Side	4.60	9.09	7.60
Top	4.50	8.99	7.50
Bottom	3.00	7.49	6.00
4. Cavity Diameter (inches)	23.76	23.76	21.6
5. Total Cask Weight (pounds)			
PWR	53,250	63,400	52,450
BWR	53,200	63,350	52,050
6. Margin (based on 54,000 pound allocation)			
PWR	760	(9,400)	1,550
BWR	800	(9,350)	1,950

a factor of 5 will necessarily result in a reduction of the payload capacities. Therefore, the first step in the evaluation was to reduce the PWR payload capacity to 2 PWR assemblies.

This payload reduction decreases the radiation source by one-third and allows a reduction in the cask cavity diameter of approximately 2 inches. This reduction in the radiation source still requires an increase in shielding thickness. However, the weight associated with the increase in shielding thickness is offset by the decrease in the cask diameter. Column 3 of Table 1 provides an estimate of the weight of a cask designed to transport 2 PWR assemblies with a dose rate of 2 mrem/hr at 2-meters. The weight margin of the cask loaded with 2 PWR assemblies would be about 1,550 lbs. Therefore, the PWR payload capacity of a cask designed to limit the dose rate to 2 mrem/hr would be 2 PWR spent fuel assemblies having a burnup no higher than 35,000 MWD/MTU. The cask cavity diameter required for these 2 PWR assemblies would be approximately 21.6 inches.

2.2 Evaluation of BWR Payload Capacity

If the cask cavity diameter is 21.6 inches, 4 BWR assemblies can be accommodated. Since the radiation source for 4 BWR assemblies is equivalent to the radiation source of 1 PWR assembly, recalculation of the gamma and neutron shielding is not required. The weight of the cask without the basket and the assemblies remains the same. Column 3 of Table 1 provides an estimate of the weight of the new cask with 4 BWR assemblies. The weight margin of the new cask loaded with 4 BWR assemblies is 1,950 lbs. Therefore, a common use cask could be designed to transport 4 BWR assemblies within the weight allocation.

3.0 EVALUATION OF LIFE CYCLE COSTS

The total truck system life cycle costs for a common use cask with a capacity of 3 PWR assemblies and 7 BWR assemblies and a reference cask with a capacity of 2 PWR assemblies and 5 BWR assemblies are given in Figure 1 (Reference 3). Figure 1 shows that the cost of shipping the spent fuel in the reference cask would be approximately 43.5% higher than shipping the spent fuel in the commom use cask. The life cycle costs are based on the following assumptions and input data:

- A total shipment of 110,000 MTU
- A mix of one-third truck and two-thirds rail shipments
- A mix of 60% PWR assemblies and 40% BWR assemblies
- One origin and one destination

The following costs are included in the estimated truck cask system life cycle cost:

- Development and certification
- Cask fleet acquisition
- Cask handling, operation and inspection
- Cask decommissioning and salvage

The truck cask system life cycle cost for the new PWR/BWR payload capacity is developed by using the same assumptions and input data used for the 3 PWR/7 BWR and 2 PWR/5 BWR cases. The development and certification costs, cask fleet acquisition costs, cask handling, operation and inspection costs, and cask decommissioning and salvage

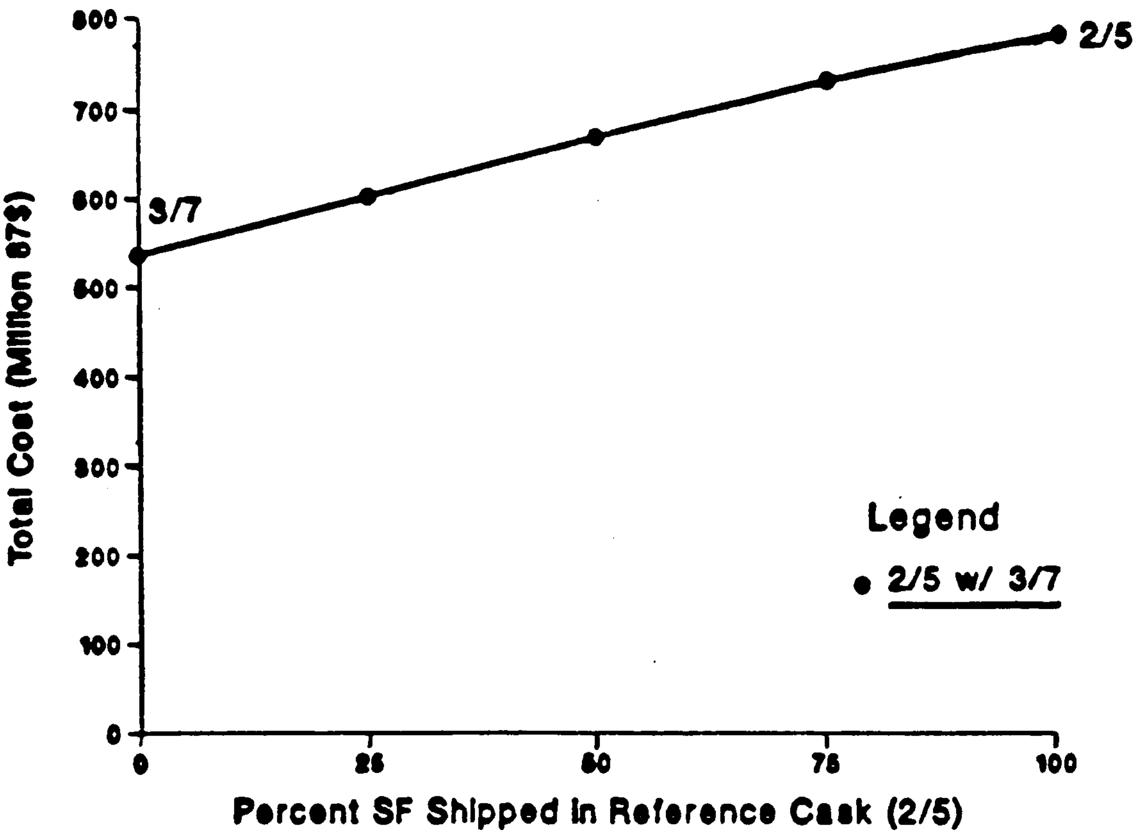

Figure 1 Truck Cask System Life Cycle Costs

costs are estimated to be approximately the same for the reduced cavity casks. The only significant change is the number of PWR assemblies and BWR assemblies that can be transported in one truck shipment. The number of PWR assemblies per shipment is reduced from 3 to 2 and the number of BWR assemblies per shipment is reduced from 7 to 4. This reduction increases the cost of shipping the PWR assemblies by 50% and increases the cost of shipping the BWR assemblies by 75%. The total truck system life cycle cost increases by 60%. Figure 2 shows the relationship of the 2 PWR/4 BWR case to the 3 PWR/7 BWR case.

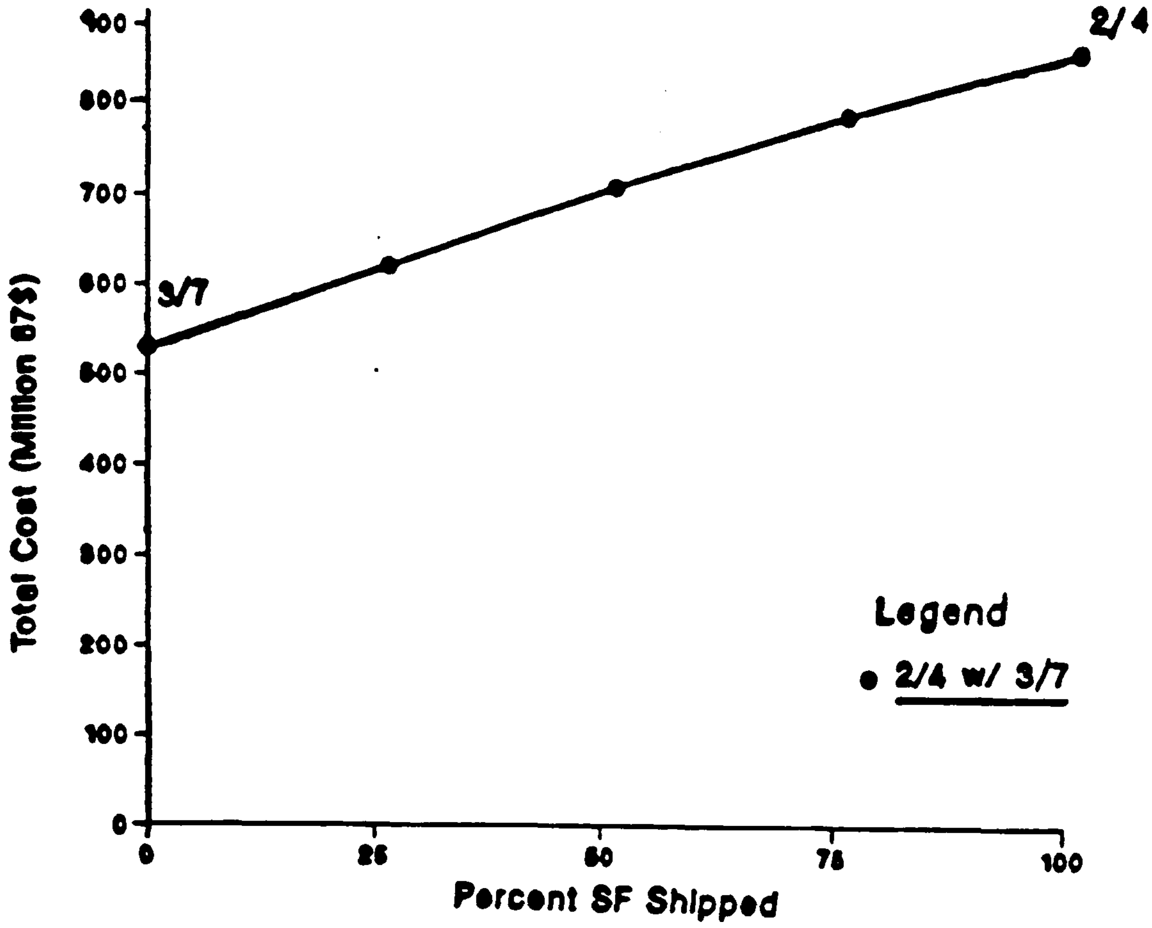

Figure 2 Comparsion of 3 PWR/7 BWR Case to 2 BWR/4 BWR Case

4.0 CONCLUSIONS

The two principal conclusions which may be drawn from the evaluation are:

- A reduction in the allowable 2-meter dose rate by a factor of five would have a large impact on payload. It is expected that the number of PWR assemblies that could be transported in a legal weight truck cask would decrease from 3 to 2 and reduce the number of BWR assemblies from 7 to 4.

- If one third of the total of 110,000 MTU would still be transported by truck in spite of this reduction in payload capacity, it is expected that the total truck system life cycle cost would increase by 60% thus adding a cost of approximately $330,000,000 to the program.

5.0 REFERENCES

1. "Titan Legal Weight Truck Cask Alternative Material Feasibility Study", NWD-TR-008, Rev. 0, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, July 1988, (Westinghouse Proprietary).

2. "Titan Legal Weight Truck Cask Design Requirements", NWD-TR-007, Rev. 0, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, July 1988, (Westinghouse Proprietary).

3. Hofmann, P., "Comparisons of Life Cycle Costs," Spent Fuel Burnup Credit Workshop, February, 1988.

TITAN LEGAL WEIGHT TRUCK CASK

IMPACT ON PAYLOAD

OF

ALLOWANCE OF FUEL BURNUP CREDIT FOR CRITICALITY EVALUATIONS

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1.0 INTRODUCTION

The purpose of this report is to provide an evaluation of the effect of burnup credit on payload for the TITAN cask. This evaluation is addressed from the standpoint of impact on payload and basket design if burnup credit is disallowed, and what modifications, if any, to the design would optimize the payload if burnup credit is allowed (Reference 1).

The TITAN LWT cask is a common use cask capable of transporting either 3 PWR or 7 BWR spent fuel assemblies. The criticality evaluations that have been completed in support of the Preliminary Design have not taken any credit for burnup.

2.0 DISCUSSION

2.1 Current Criticality Evaluation

Criticality analyses have been performed for a preliminary design of the Titan cask. The results show that the K-eff (including method bias and statistical uncertainties to a 95/95 probability/confidence level) is 0.9477 and 0.8022 for the TITAN cask containing three PWR or seven BWR fuel assemblies respectively. These analyses were based on the following assumptions:

1. Calculations of fuel assemblies in storage and shipping configurations have shown that the Westinghouse 17X17 OFA and the GE 7X7 fuel assemblies yield a Keff as high or higher than does any other PWR or BWR fuel assembly types for which the cask is to be designed when all fuel assemblies have the same U-235 enrichment. Thus, the W 17X17 OFA fuel assembly was analyzed in the PWR cask basket and the GE 7X7 fuel assembly was analyzed in the BWR cask basket to determine the maximum cask reactivity.

2. All fuel rods contain uranium dioxide at an enrichment of 4.5 w/o U-235 over the entire length of each rod (i.e., no credit was taken for burnup).

3. No credit is taken for any U-234, U-236 or burnable absorber in the fuel, nor is any credit taken for the buildup of fission product poison material.

4. The moderator is pure water at a temperature of 68 degrees F. A conservative value of 1.0 gm/cc is used for the full water density case.

5. No credit is taken for any spacer grids or spacer sleeves.

6. The cask array is infinite in all directions which does not allow neutron leakage from the array.

7. The poison material loading in the borated silicone shielding (the neutron shielding on the outside of the cask) is reduced by 25 percent below its nominal loading (1.06 w/o).

8. A minimum poison material loading of 0.020 and 0.010 grams B-10 per square centimeter is used in the poison panels of the PWR and BWR fuel baskets respectively. This includes a 25% reduction in the nominal poison loading.

The maximum cask K-eff under normal conditions also includes asymmetric positioning of the fuel assemblies within the fuel basket such that all assemblies are shifted towards the center of the basket. This minimizes the separation between fuel assemblies in the basket and increases reactivity.

The maximum cask K-eff under accident conditions is equal to the maximum cask K-eff under normal conditions due to the following conditions:

1. The borated silicone and depleted uranium shielding reduces neutron leakage through the cask walls such that the cask reactivity is unaffected by the presence of any other loaded cask. As a result, the TITAN cask reactivity will remain unchanged whether one cask or an infinite number are placed together.

2. The fuel assemblies in the cask are modelled as close as possible in the nominal case. As a result, any realistic change in the basket configuration will result in increased spacing between assemblies and a reduction in the cask reactivity.

3. A reduction in the cask volume will bring the neutron absorbing borated silicone and depleted uranium shielding materials closer to the fuel assemblies. This change will tend to reduce the cask reactivity. However for small changes (such as a 5% volume change) it will have an insignificant effect on the cask reactivity.

4. The presence of the poison material in the basket and cask design remove the conditions necessary for "optimum moderation" so that K-eff continually decreases as moderator density decreases from 1.0 gm/cc to 0.0 gm/cc.

These conditions and model assumptions meet the requirements for fuel shipping cask under normal and accident conditions as specified in 10 CFR Part 71, Sections 71.55 and 71.57.

2.2 Effect of Burnup Credit on Cask Payload

At this stage of the preliminary design, the total weight of the TITAN cask (including three of the heaviest PWR assemblies) is 53,400 pounds. The cask has a cavity which is 180 inches long and 23.76 inches in diameter. The thickest web of the PWR basket is 0.492 inches thick. This thickness includes a 0.250 inch thick stainless steel (Type 316N) plate, two Boral panels (0.085 inches thick each) and the walls of two liner tubes (0.031 inches thick each). Figure 1 shows the cross-section of the PWR basket.
The payload of large casks, such as those used for rail or barge transport which may accommodate on the order of 20 PWR assemblies, can potentially be increased by 15% or more (Reference 2) if credit is taken for burnup. This increase in payload is achieved by reductions in the basket web thickness which results in an increase in the payload by some amount while retaining a given cask cavity size (or perhaps even reducing the cavity size to accommodate the increased weight of the additional assemblies). For legal weight truck casks, however, an incremental change in the number of assemblies would most certainly involve a significant increase in the cavity diameter with a corresponding increase in cask weight. Decreases in basket web thicknesses, which may be achieved if fuel burnup can be considered, will not be sufficient to allow the addition of another assembly in the original cavity diameter.

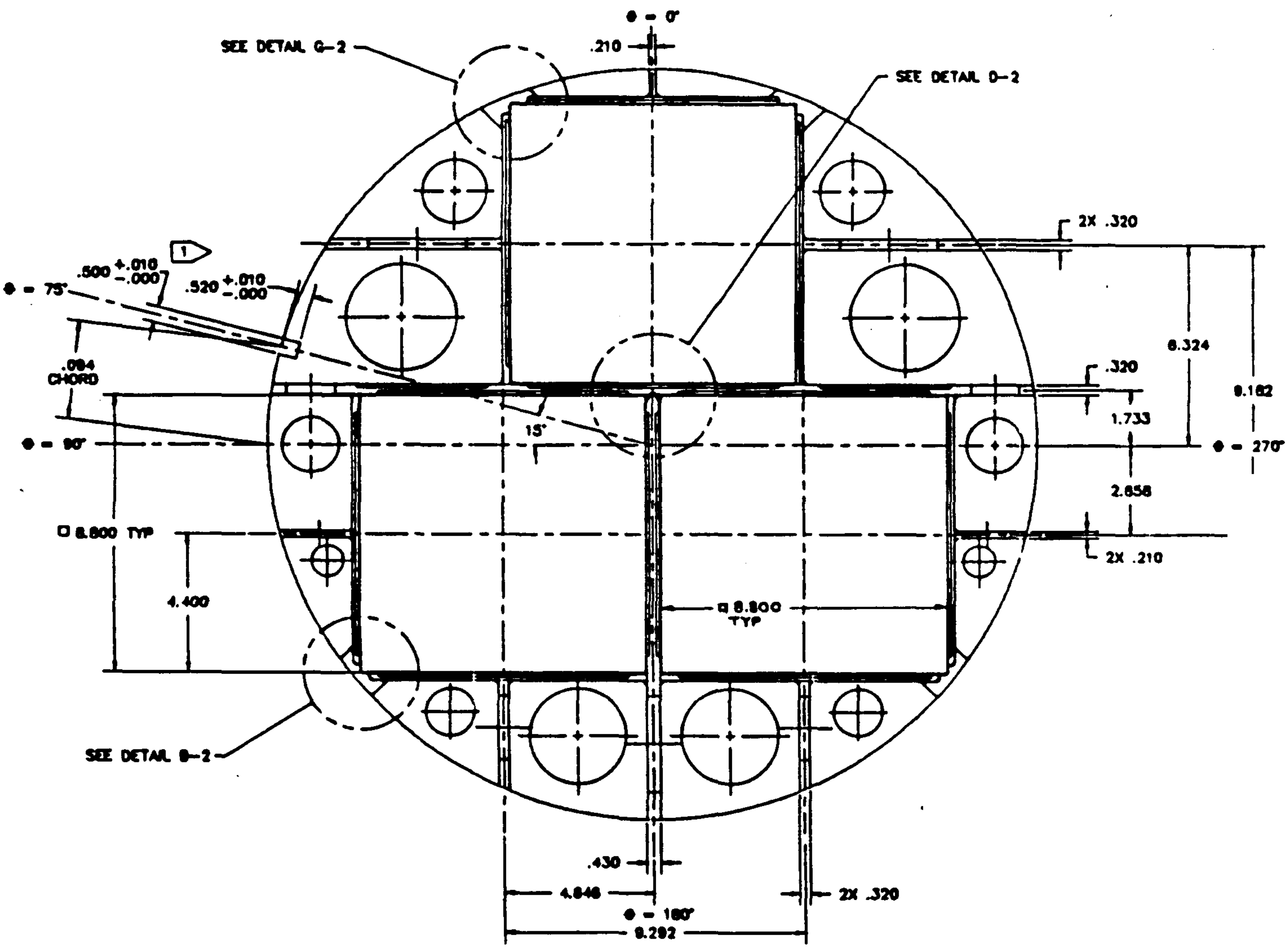

Figure 1 PWR Basket Cross Section

For example, if the TITAN cask were to contain 4 PWR assemblies, the inside diameter of the cask would have to increase to approximately 27 inches. The weight of the cask would increase accordingly. It is clear that even if the Boral could be removed entirely from the design by taking credit for burnup, the reduction in the web thickness would not result in bringing the cavity diameter back to the current inside diameter of the cask (23.76 inches) and thus the weight of the cask would be unacceptable.

Thus it can be stated with certainty that the payload of the current preliminary design of the TITAN cask which accommodates 3 PWR assemblies or 7 BWR assemblies without taking credit for burnup cannot be increased to 4 PWR assemblies even if credit could be taken for burnup.

If burnup credit is disallowed, there would be no impact on the payload of the TITAN cask. Likewise, if burnup credit is allowed, there are no modifications to the design which would permit increasing the payload. Because the K-eff can be held below the 0.95 limit without the need for a flux trap in the basket design, the only possible reduction in the cask cavity diameter would come from a reduction in the pitch of the basket as a consequence of eliminating (or reducing) the Boral panels. Clearly this possible reduction is insufficient to accommodate a 4th PWR assembly and at the same time stay roughly within the current diameter of the cavity. Actually, some reduction in the cask diameter would have to be achieved because a 4th PWR assembly would add 1500 pounds to the total weight, and the basket itself would get heavier.

3.0 CONCLUSIONS

The following conclusions are drawn from this study:

1. Taking credit for fuel burnup will not impact the payload of the TITAN cask. There are no modifications to the design which would increase the payload if burnup credit was allowed. The reason is that changes to the basket design that may be possible with burnup credit would not offset the basic need for increasing the cavity diameter when going from a 3 to 4 PWR assembly payload. Such an increase would result in the cask weight exceeding the allocated limit.

2. The only incentive for taking credit for burnup would be to reduce the cask cavity diameter in the event that such a reduction becomes necessary to meet the cask's weight allocation with the current payload objective of 3 PWR or 7 BWR assemblies.

4.0 REFERENCES

1. Letter, I. K. Hall to Distribution, "Guidance on Trade-off and Impact Evaluations," November 16, 1988, IKH-165-88

2. Robert Jones, Presentation at the Fuel Burnup Credit Workshop, February 23, 1988, "Survey of Previous and Current Industry Efforts Regarding Burnup Credit."

TITAN LEGAL WEIGHT TRUCK CASK

IMPACT

OF

TRANSPORTING FUEL AGED 5 YEARS AFTER DISCHARGE

VERSUS

DESIGN BASIS (TEN-YEAR-OLD) FUEL

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1.0 INTRODUCTION

The Statement of Work for the Cask Systems Development, Initiative 1, requires that the contractor conduct a number of trade-off and impact evaluations on cask payload capacities. One of the design considerations to be evaluated is the transportation of fuel aged 5 years after discharge versus design basis (ten-year-old) fuel.

The purpose of this evaluation is to address the question of whether or not the baseline design is adequate to haul any five-year-old fuel and if so, how much? An additional consideration is to define those modifications to the basket design that would be required to transport five-year-old fuel (Reference 1).

The Titan LWT cask has an allocated weight of 54,000 lbs out of a gross vehicle weight of 80,000 lbs. The cask is intended for transporting fuel that is at least 10 years out of reactor. The current design of the Titan cask can transport either 3 ten-year-old PWR or 7 ten-year-old BWR assemblies.

Because of the higher radiation source terms associated with five-year-old fuel, transporting five-year-old fuel instead of design basis (ten-year-old) fuel will impact the shielding requirements and mean a reduction in the payload capacity. The evaluation of the impact on payload capacity is presented in Section 2. The conclusions are presented in Section 3 with references given in Section 4.

2.0 PAYLOAD OF BASELINE DESIGN WITH FIVE-YEAR-OLD FUEL

Preliminary analyses indicate that the Westinghouse Titan LWT Cask (a common use cask capable of transporting both PWR and BWR assemblies) can accommodate a maximum of 3 PWR or 7 BWR assemblies with a cask cavity diameter of 23.76 inches and cavity length of 180 inches (Reference 2). The estimated weight of this baseline cask is 53,250 lbs when it is loaded with 3 PWR assemblies and 53,200 lbs when it is loaded with 7 BWR assemblies. The weight margin of the cask is 750 lbs for the PWR assemblies and 800 lbs for the BWR assemblies. These weights are based on a cask designed for ten-year-old fuel.

2.1 Payload with Baseline Cask and Basket Designs

Preliminary DOT analysis shows that the top nozzle of a PWR assembly is the most radioactive part of the assembly due to the cobalt found in the end fittings or nozzles. The Co 60 activity of a five-year-old PWR assemby is 93% higher than a ten-year-old PWR assembly. The shielding on the closure end is sized to just meet the 200 mrem/hr surface dose limit. The higher source of 1.25 MeV gammas exceeds that limit. Thus, the baseline cask optimized for 3 ten-year-old PWR assemblies can not accommodate even 1 five-year-old PWR assembly without additional gamma shielding. Therefore, no five-year-old PWR assemblies can be accommodated in the baseline cask with the baseline PWR basket. It is expected that the baseline cask and the baseline BWR basket can accommodate some five-year-old BWR fuel. It is estimated that the baseline cask can accommodate 2 five-year-old BWR assemblies.

2.2 Payload of Baseline Cask Design with Modified Basket Design

Five-year-old spent fuel can be transported in the baseline cask provided the cask is modified to incorporate additional shielding for the assembly end fittings. With new basket designs, 1 five-year-old PWR assembly or 4 five-year-old BWR assemblies could be placed in the baseline cask. The baskets must be designed so that the entire assembly is placed inside the basket. Four .2 inch thick plates of depleted uranium encased in stainless steel must be placed around the top and bottom of the basket so that the top and bottom nozzles of the assemblies are always shielded. The plates must be 15 inches long at the top of the basket and 12 inches long at the bottom of the basket. The plates need to be this long so that the same basket design would accommodate the various PWR configurations (lengths, nozzle sizes, etc). The top and bottom of the basket must also include a .2 inch thick plates of depleted uranium encased in stainless steel. Both plates must overlap the side plates of the basket so that streaming will not occur. The bottom plate of the basket would be welded to the basket and remain stationary. The top of the basket would be a removable lid designed so that spacers could be attached to it. Spacers would be needed to hold the various assembly configurations in place. A sketch of the PWR basket is shown in Figure 1. The BWR basket would be similar to the PWR basket. The only difference would be the number of assembly spaces. A sketch of the BWR basket is shown in Figure 2.

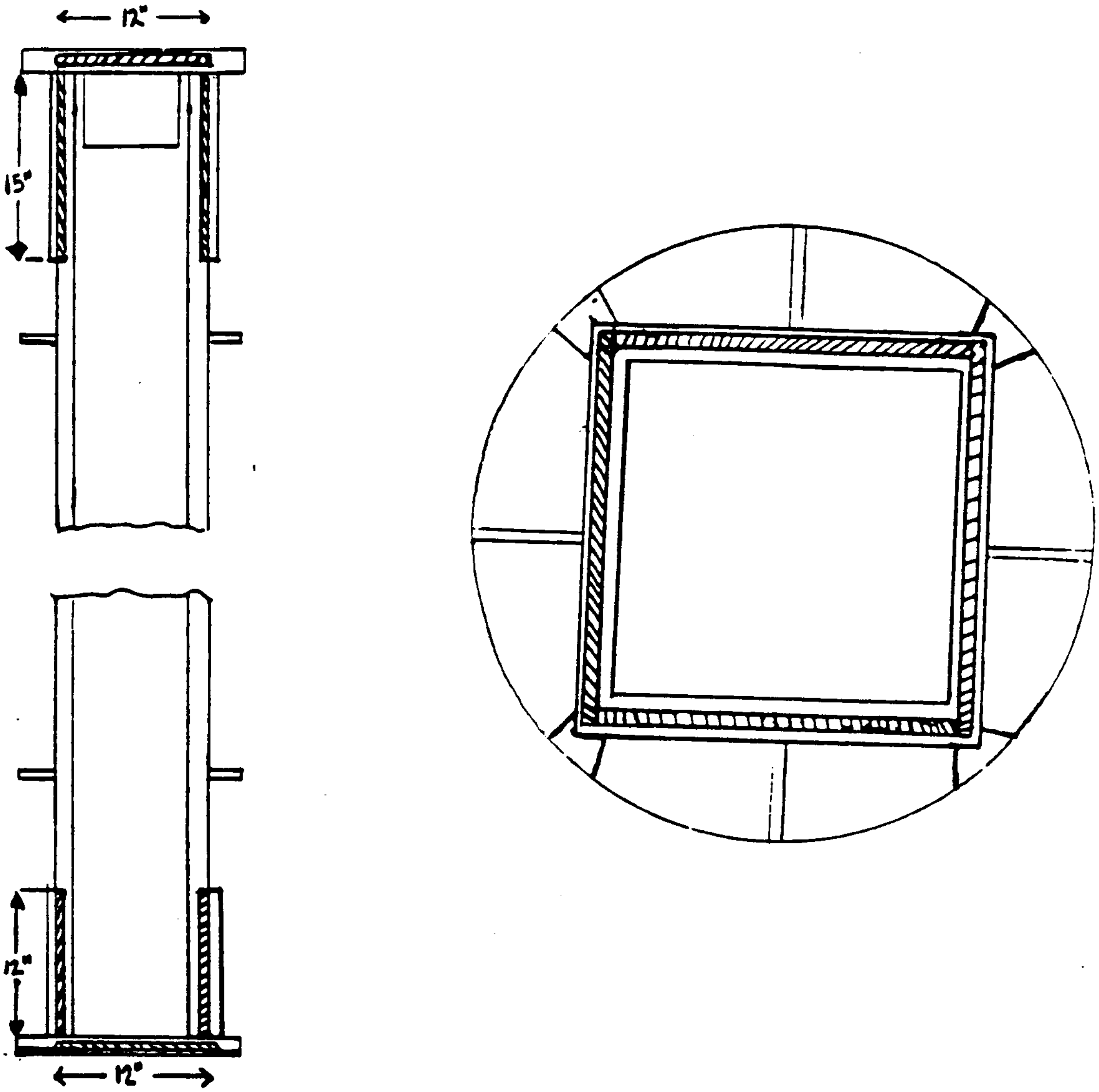

Figure 1 PWR Basket

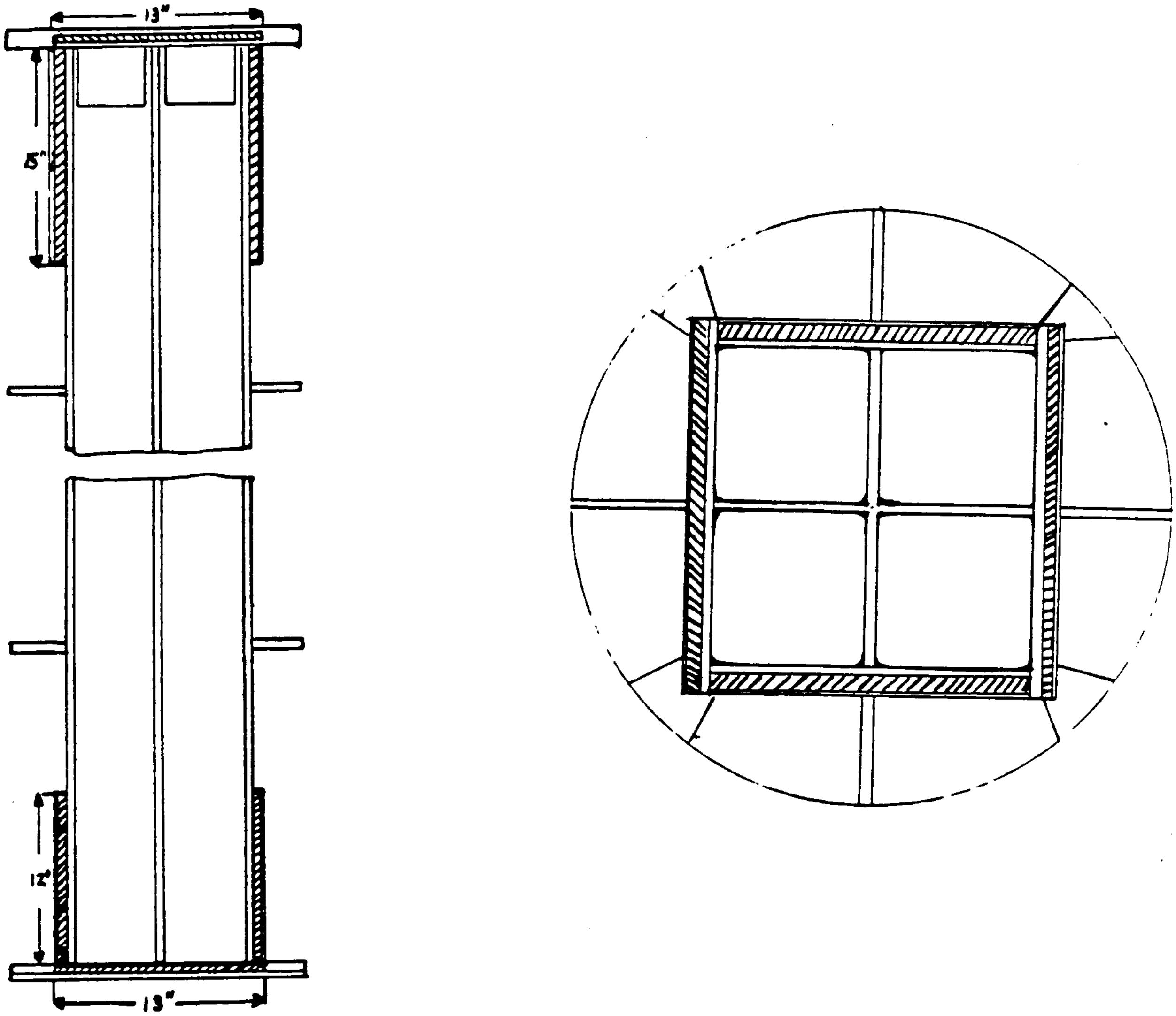

Figure 2 BWR Basket

3.0 CONCLUSIONS

The two principal conclusions which may be drawn from the evaluation are:

- Five-year-old PWR fuel can not be shipped in the baseline cask designed for ten-year-old PWR fuel unless the basket design is changed.

- Even if the basket design is modified, the transportation of fuel aged 5 years after discharge verses design basis (ten-year-old) fuel would have a large impact on payload. If the basket design is changed, it is expected that the number of PWR assemblies would decrease from 3 to 1 and the number of BWR assemblies would decrease from 7 to 4. It is expected that the basket could be modified (by adding depleted uranium to shield the fuel assembly end fittings) such that either 1 PWR or 4 BWR assemblies could be transported with the baseline design.

4.0 REFERENCES

1. Letter, I.K. Hall to Distribution, "Guidance on Trade-off and Impact Evaluations", November 16, 1988, IKH-165-88.

2. "Titan Legal Weight Truck Cask Alternative Material Feasibility Study", NWD-TR-008, Rev. 0, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, July 1988, (Westinghouse Proprietary).

3. "Titan Legal Weight Truck Cask Design Requirements", NWD-TR-007, Rev. 0, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, July 1988, (Westinghouse Proprietary).

Titan Legal Weight Truck Cask

Impact of Shipping Consolidated Fuel

on Cask Payload

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1.0 INTRODUCTION

The Statement of Work for the Cask Systems Development, Initiative 1, requires that the contractor conduct a number of trade-off and impact evaluations on cask payload capacities. One of the design considerations to be evaluated is the transportation of consolidated fuel.

The purpose of this evaluation is to determine the impact of fuel consolidation on cask payload capacities if canisters containing fuel with consolidation ratios of 1.2:1 to 2.0:1 are shipped in the cask. The Titan LWT cask has an allocated weight of 54,000 lbs out of a gross vehicle weight of 80,000 lbs. The cask is intended for transporting spent nuclear PWR fuel that has a maximum burnup of 35,000 MWD/MTU or BWR fuel with 30,000 MWD/MTU burnup and is at least 10 years out-of-reactor. The current design of the Titan cask can transport either 3 intact PWR assemblies or 7 intact BWR assemblies. Consolidation of the intact fuel assemblies should increase payload capacities. If canisters are fabricated with outside dimensions equivalent to the envelope dimensions of the intact fuel assembly, the canisters would fit into the reference fuel basket and thus more fuel rods would fit into the reference cask. However, a higher fuel loading increases the weight, the radiation source, and the thermal power. This report addresses the effect of these aspects on the amount of consolidated fuel that could be shipped in the consolidated form. The study approach, i.e. assumptions and limitations, is presented in Section 2. The results are presented in Section 3, conclusions in Section 4, and references in Section 5.

2.0 STUDY APPROACH

In this study, fuel consolidation ratios and assembly payloads were obtained for each of the following fuel assembly types:

PWR Fuel Assemblies	
Westinghouse Electric	17x17
Westinghouse Electric	15x15
Westinghouse Electric	14x14
Babcock & Wilcox	17x17
Babcock & Wilcox	15x15
Combustion Engineering	16x16
Combustion Engineering	14x14
Exxon	17x17
Exxon	15x15
Exxon	14x14
BWR Fuel Assemblies	
General Electric	8x8
General Electric	7x7
Exxon	8x8
Exxon	7x7

Maximum PWR fuel consolidation ratios and equivalent fuel assembly payloads were determined for baskets containing either three canisters or two canisters. Maximum BWR fuel consolidation ratios and equivalent fuel assembly payloads were determined for baskets containing either five canisters or six canisters. Fuel assembly payloads were determined for canisters with a fuel consolidation ratio of 1.8 and for canisters packed with the maximum number of rods.

2.1 Assumptions

The evaluation of the TITAN cask's capability for transporting consolidated fuel was based on the following assumptions:

- The baseline cask is the cask described in the Preliminary Design Report (Reference 1). Shielding materials are depleted uranium and Boro-Silicone. The maximum decay power is 1740 watts.

- The baskets to be used in conjunction with shipping consolidated fuel would be new baskets designed for the increased loads. Special basket designs would be used which have numbers of cells consistent with the number of canisters that could be shipped.

- The canisters would have approximately the same external cross section as the corresponding intact fuel assembly. The canisters would be square and made from stainless steel. The walls of the canisters would be 0.125" thick; the bottom lid would be 0.625" thick; and the top lid would be 0.375" thick. The weight of the canisters would vary between 180 lbs and 220 lbs for the PWR assemblies, and 130 lbs and 150 lbs for the BWR assemblies.

- The number of consolidated fuel rods that could be shipped is assumed to be divided equally among the number of canisters being considered. For example, if the cask could handle 300 rods, and three canisters would be required, then each canister would hold 100 rods, etc. This means that each canister placed in the cask would have the same consolidation ratio.

2.2 Limitations

The amount of consolidated fuel that could be placed in the cask could potentially be limited by the weight available for consolidated fuel, the 2-meter dose rate requirement, the decay power of the consolidated fuel, and the criticality.

2.2.1 Weight Limit

The weight allocation for the TITAN cask loaded with consolidated fuel is 54,000 lbs (Reference 2). The baseline cask weighs 46,989 lbs without the basket and consolidated fuel. This leaves 7011 lbs available for the basket and the loaded canisters. The basket weight would be dependent on the weight of the loaded canisters. The weight of the basket increases as the weight of the loaded canisters increases. This is because the cell walls of the basket must be increased so that the basket can withstand the higher dynamic loads.

2.2.2 Shielding Limit

The 2-meter dose rate must not exceed 10 mrem/hr. Scoping shielding calculations show that up to 4 PWR assemblies or 10 BWR assemblies could be placed in the baseline cask without exceeding this limit.

2.2.3 Structural/Thermal Limit

The critical stuctural member of the TITAN cask is the 1.25 inch thick outer shell of the cask body. Preliminary structural analysis of the 15 degree oblique hypothetical 30 foot free drop accident, using the SCANS computer code in the dynamic mode, showed a maximum axial plus bending stress of 76,336 psi in this shell. One of the initial conditions for this event includes full decay heat and insolation which results in a temperature for this shell of 240 degrees F. The allowable stress for this loading condition is equal to the minimum ultimate strength which is equal to 76,320 psi at this temperature. This would indicate that any increase in shell temperature (i.e., increase in decay power over 1740 watts total decay power in the cask) would reduce the allowable stresses and cause negative structural margins. On this basis, a power limit of 1740 watts has been established for the cask.

However, the temperature (decay power) effect on allowable stresses is not strong for modest changes in decay power as shown by the following. The relationship between shell temperatures and decay power is given in Figure 1. The allowable stress as a function of temperature is given in Figure 2. These two figures are then cross plotted to yield allowable stress vs decay power in Figure 3. If, as will be discussed later, the decay power would not be limiting and enough consolidated fuel were loaded to utilize the entire weight margin, the decay power for the Babcock & Wilcox 17x17 fuel (the worst case) would be just 15% higher than the reference 1740 watts or 2000 watts. The corresponding allowable stress would be reduced by only 520 psi (0.7%) to 75,800 psi. It could thus be concluded that the effect on the material properties would be small for the changes in decay power that would be seen if the full weight limit were used. Hence, decay power is not the limiting factor. It should be noted that because temperature margins for the consolidated fuel are large, there is no real concern about staying within the cladding temperature limit if the cask is used to transport consolidated fuel.

2.2.4 Criticality Limit

There is no real criticality concern with transporting consolidated fuel in the TITAN cask. Consolidated fuel, being more closely packed, is under-moderated and would be less reactive than the intact fuel.

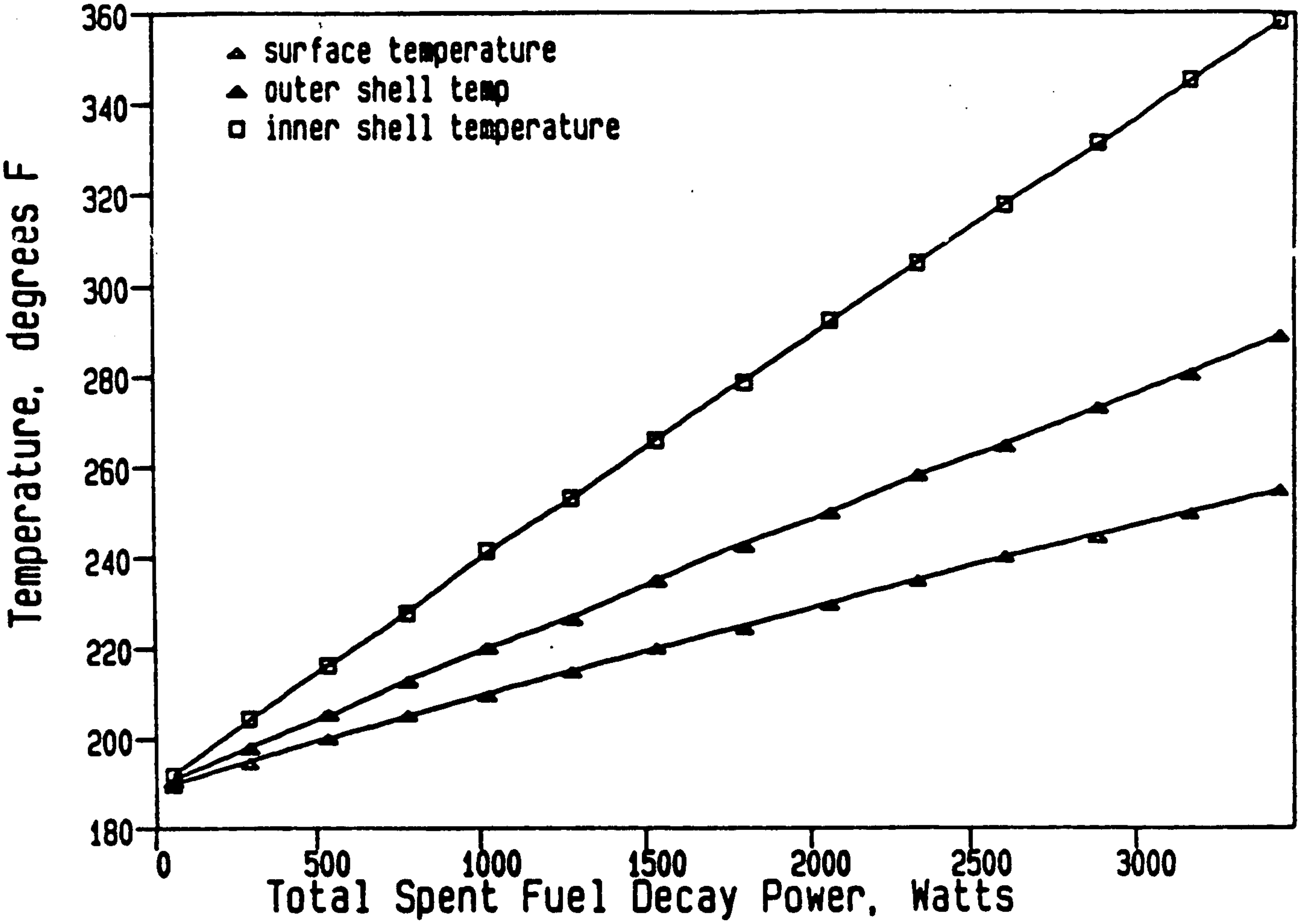

Figure 1
Relationship Between Shell Temperatures and Decay Power

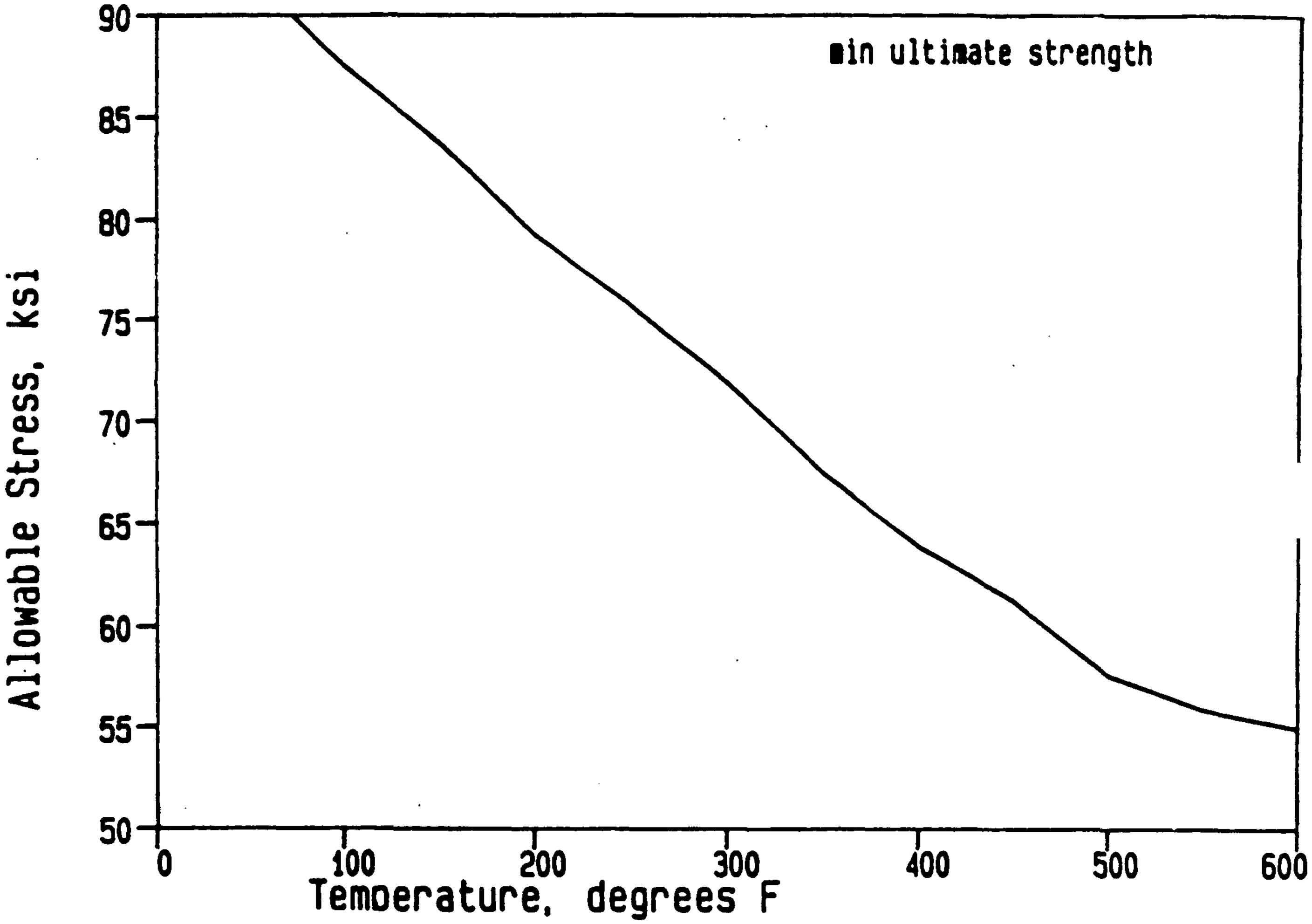

Figure 2
Allowable Stress as a Function of Temperature

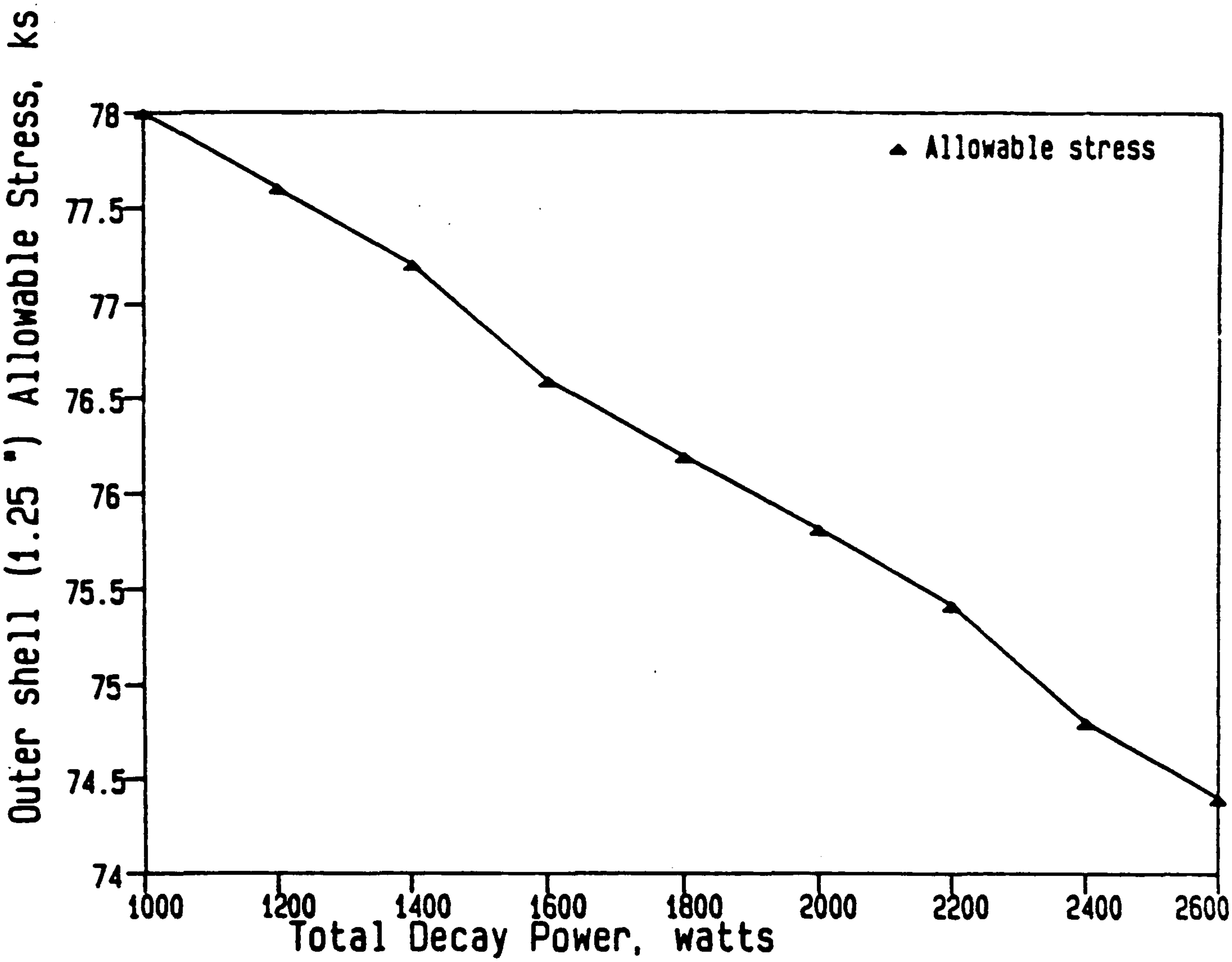

Figure 3
Allowable Stress versus Decay Power

3.0 RESULTS

After considering the effect of each of the above limits, it was determined that the most critical limit for the consolidated fuel was the weight limit. This limit yields the lowest cask payloads. Consolidation ratios and payloads were determined for the weight limit and a power limit of 1740 watts.

3.1 Cask Payloads with PWR Fuel

As stated previously, two basket types were evaluated for the PWR fuel: a three canister basket and a two canister basket. It turns out, that two canisters of consolidated fuel yields a larger payload than a cask holding three canisters. In addition, the rods would be more tightly packed in two canisters, thus minimizing fuel rod damage. For both baskets, there would be four canister sizes: a 8.5"x8.5", a 8.4"x8.4", a 8.1"x 8.1", and a 7.7"x7.7". The 8.5"x8.5" canister would hold the Babcock & Wilcox 17x17 and 15x15 consolidated fuel. The 8.4"x8.4" canister would hold the following consolidated fuels:

o Westinghouse	17x17
o Westinghouse	15x15
o Exxon	17x17
o Exxon	15x15

Rods from the Combustion Engineering 16x16 and 14x14 assemblies would be consoldiated in the 8.1" square canisters. The 7.7"x7.7" canister would hold fuel from the Westinghouse 14x14 and the Exxon 14x14 assemblies.

3.1.1 Weight Tables

Tables 1, 2, 3, and 4 provide the following information:

- o Fuel assembly type
- o Fuel assembly manufacturer and size
- o Weight of the fuel rods

Table 1
Weight Limit for 3 Canisters of PWR Fuel

Fuel Type	Fuel Assembly	Rod Weight (lbs)	Rods / Assembly	Canister Size (inches)	Max # Rods Canister	Weight of Fuel (lbs)	Rods / Canister	Ratio	Number of Assemblies
	W 17 x 17 STD	5.37	264	8.4 x 8.4	504	4362	271	1.03	3.1
	W 15 x 15 STD/ZC	6.85	204	8.4 x 8.4	389	4362	212	1.04	3.1
	W 14 x 14 STD/ZCB	6.68	179	7.7 x 7.7	323	4362	218	1.22	3.6
P	B & W 17 x 17 MARK C	4.90	264	8.5 x 8.5	504	4362	297	1.12	3.4
W	B & W 15 x 15 MARK B	7.00	208	8.5 x 8.5	389	4362	208	1.00	3.0
R	CE 16 x 16 ONOFRE	5.70	224	8.1 x 8.1	449	4362	255	1.14	3.4
	CE 14 x 14 STD	6.90	164	8.1 x 8.1	340	4362	211	1.28	3.9
	Exxon 17 x 17 WE	4.81	264	8.4 x 8.4	504	4362	302	1.15	3.4
	Exxon 15 x 15 WE	6.72	204	8.4 x 8.4	389	4362	216	1.06	3.2
	Exxon 14 x 14 WE	6.72	179	7.7 x 7.7	323	4362	216	1.21	3.6

Table 2
Weight Limit for 2 Canisters of PWR Fuel

Fuel Type	Fuel Assembly	Rod Weight (lbs)	Rods / Assembly	Canister Size (inches)	Max # Rods Canister	Weight of Fuel (lbs)	Rods / Canister	Ratio	Number of Assemblies
	W 17 x 17 STD	5.37	264	8.4 x 8.4	504	4585	427	1.62	3.2
	W 15 x 15 STD/ZC	6.85	204	8.4 x 8.4	389	4585	335	1.64	3.3
	W 14 x 14 STD/ZCB	6.68	179	7.7 x 7.7	323	4310	323	1.80	3.6
P	B & W 17 x 17 MARK C	4.90	264	8.5 x 8.5	504	4585	468	1.77	
W	B & W 15 x 15 MARK B	7.00	208	8.5 x 8.5	389	4585	328	1.57	3.1
R	CE 16 x 16 ONOFRE	5.70	224	8.1 x 8.1	449	4585	402	1.80	3.6
	CE 14 x 14 STD	6.90	164	8.1 x 8.1	340	4585	332	2.00	4.0
	Exxon 17 x 17 WE	4.81	264	8.4 x 8.4	504	4585	477	1.81	3.6
	Exxon 15 x 15 WE	6.72	204	8.4 x 8.4	389	4585	341	1.67	3.3
	Exxon 14 x 14 WE	6.72	179	7.7 x 7.7	323	4335	323	1.80	3.6

Table 3

Weight Limit for Canisters with a Consolidation Ratio of 1.8

Fuel Type	Fuel Assembly	Rod Weight (lbs)	Rods / Assembly	Canister Size (inches)	Max # Rods Cansiter	Weight of Fuel (lbs)	Rods / Canister	Ratio	Number of Assemblies	Number of Canisters
PWR	W 17 x 17 STD	5.37	264	8.4 x 8.4	504	2550	475	1.80	1.8	1
	W 15 x 15 STD/ZC	6.85	204	8.4 x 8.4	389	2517	367	1.80	1.8	1
	W 14 x 14 STD/ZCB	6.68	179	7.7 x 7.7	323	4310	323	1.80	3.6	2
	B & W 17 x 17 MARK C	4.90	264	8.5 x 8.5	504	2329	475	1.80	1.8	1
	B & W 15 x 15 MARK B	7.00	208	8.5 x 8.5	389	2620	374	1.80	1.8	1
	CE 16 x 16 ONOFRE	5.70	224	8.1 x 8.1	449	4585	402	1.80	3.6	2
	CE 14 x 14 STD	6.90	164	8.1 x 8.1	340	4085	296	1.80	3.6	2
	Exxon 17 x 17 WE	4.81	264	8.4 x 8.4	504	4584	477	1.80	3.6	2
	Exxon 15 x 15 WE	6.72	204	8.4 x 8.4	389	2465	367	1.80	1.8	1
	Exxon 14 x 14 WE	6.72	179	7.7 x 7.7	323	4335	323	1.80	3.6	2

Table 4

Weight Limit for Canisters Containing the Maximum Number of Rods

Fuel Type	Fuel Assembly	Rod Weight (lbs)	Rods / Assembly	Canister Size (inches)	Max # Rods Canister	Weight of Fuel (lbs)	Rods / Canister	Ratio	Number of Assemblies	Number of Canisters
	W 17 x 17 STD	5.37	264	8.4 x 8.4	504	2708	504	1.91	1.9	1
	W 15 x 15 STD/ZC	6.85	204	8.4 x 8.4	389	2665	389	1.91	1.9	1
	W 14 x 14 STD/ZCB	6.68	179	7.7 x 7.7	323	4310	323	1.80	3.6	2
P	B & W 17 x 17 MARK C	4.90	264	8.5 x 8.5	504	2472	504	1.91	1.9	1
W	B & W 15 x 15 MARK B	7.00	208	8.5 x 8.5	389	2725	389	1.87	1.9	1
R	CE 16 x 16 ONOFRE	5.70	224	8.1 x 8.1	449	2557	449	2.00	2.0	1
	CE 14 x 14 STD	6.90	164	8.1 x 8.1	340	2345	340	2.07	2.1	1
	Exxon 17 x 17 WE	4.81	264	8.4 x 8.4	504	2422	504	1.91	1.9	1
	Exxon 15 x 15 WE	6.72	204	8.4 x 8.4	389	2615	389	1.91	1.9	1
	Exxon 14 x 14 WE	6.72	179	7.7 x 7.7	323	4335	323	1.80	3.6	2

- o The number of rods in the fuel assembly
- o The canister size
- o The maximum number of rods the canister will hold
- o The weight of the consolidated fuel, i.e. the amount of consolidated fuel that would actually be placed in the cask.
- o The number of rods in each canister for the given weight of consolidated fuel.
- o The fuel consolidation ratio for the canisters
- o The number of assemblies in the cask.

Tables 3 and 4 also give the number of canisters that could be placed in the cask for each fuel type.

Table 1 shows the effect of distributing the consolidated fuel (from a weight standpoint) among three canisters. There is no real advantage to using the TITAN cask to transport consolidated PWR fuel distributed in three canisters. The consolidation ratios are very low, ranging from 1.03 for the Westinghouse 17x17 fuel to 1.28 for the Combustion Engineering 14x14 fuel. The number of assemblies in the cask is only slightly increased for the Westinghouse 17x17 fuel and the Westinghouse 15x15 fuel. There is no increase in payload for the Babcock & Wilcox 15x15 fuel. At best, the payload for the cask would increase from 3.0 to 3.9 assemblies for the Combustion Engineering 14x14 fuel.

Table 2 shows the effect of placing the consolidated fuel in two canisters. With two canisters, consolidation ratios ranging from 1.57 for the Babcock & Wilcox 15x15 fuel to 2.03 for the Combustion Engineering 14x14 fuel could be achieved. Compared to an allowable payload of three intact assemblies. The payload of consolidated rods ranges from 3.1 Babcock & Wilcox 15x15 assemblies to 4.0 Combustion Engineering 14x14 assemblies if the fuel rods are transported in two canisters.

Table 3 shows the payload effect of limiting the consolidation ratio to 1.8. A fuel consolidation ratio of 1.8 decreases the number of assemblies that could be shipped in the cask in some cases. For example, only one canister of Westinghouse 17x17 fuel could be shipped in the cask if the fuel consolidation ratio had to be 1.8. The same is true for the Westinghouse 15x15, the Exxon 15x15, the Babcock & Wilcox 17x17, and the Babcock & Wilcox 15x15 fuel assembly designs. For the Combustion Engineering 14x14 design, two canisters of rods could be shipped but the number of assemblies would decrease from 4.0 to 3.6.

Table 4 shows the effect of limiting the payload to canisters having only the maximum number of rods the canister could hold. Only one canister of consolidated fuel could be shipped in the cask for most of the fuel types. However, two canisters could be shipped in the cask if the canisters were loaded with rods from either Westinghouse 14x14 and the Exxon 14x14 assemblies. For these two fuel types, the limiting factor is the number of rods the canister can hold and not the weight of the consolidated fuel.

3.1.2 Structural/Thermal Tables

Table 5, 6, 7, and 8 contain the same information listed in the first four Tables plus the metric tons uranium per assembly and the decay power of the consolidated fuel. These tables show the effect on payload if the payload is limited to 1740 watts and have been included to provide a basis for comparison with the weight limited payload.

Table 5 shows the effect of placing the consolidated fuel in a three canister basket. For the Westinghouse 17x17 fuel, the Westinghouse 15x15 fuel, and the Babcock & Wilcox 15x15 fuel, there would be no reason to consolidate since rods from only 3 assemblies could be placed in the cask. However, for the other fuel assemblies the fuel consolidation ratios would increase slightly from 1.03 for Babcock & Wilcox 17x17 fuel to 1.22 for the Combustion Engineering 14x14 fuel. For the Combustion Engineering 14x14 fuel and the Exxon 14x14 fuel, the number of assemblies would increase from 3.0 to 3.6.

Table 5

Decay Power Limit for 3 Canisters of PWR Fuel

Fuel Type	Fuel Assembly	Rod Weight (lbs)	Rods / Assembly	Canister Size (inches)	Max # Rods Canister	Weight of Fuel (lbs)	Rods / Canister	Ratio	Number of Assemblies	MTU / Assembly	Decay Power (Watts)
	W 17 x 17 STD	5.37	264	8.4 x 8.4	504	4302	267	1.00	3.0	0.464	1740
	W 15 x 15 STD/ZC	6.85	204	8.4 x 8.4	389	4192	204	1.00	3.0	0.469	1740
	W 14 x 14 STD/ZCB	6.68	179	7.7 x 7.7	323	4133	206	1.15	3.5	0.407	1740
P	B & W 17 x 17 MARK C	4.90	264	8.5 x 8.5	504	3990	271	1.03	3.1	0.456	1740
W	B & W 15 x 15 MARK B	7.00	208	8.5 x 8.5	407	4362	208	1.00	3.0	0.464	1718
R	CE 16 x 16 ONOFRE	5.70	224	8.1 x 8.1	449	4218	247	1.10	3.3	0.426	1740
	CE 14 x 14 STD	6.90	164	8.1 x 8.1	340	4125	199	1.22	3.6	0.386	1740
	Exxon 17 x 17 WE	4.81	264	8.4 x 8.4	504	4362	302	1.15	3.4	0.401	1704
	Exxon 15 x 15 WE	6.72	204	8.4 x 8.4	389	4362	216	1.06	3.2	0.432	1700
	Exxon 14 x 14 WE	6.72	179	7.7 x 7.7	323	4362	216	1.21	3.6	0.379	1700

Table 6

Decay Power Limit for 2 Canisters of PWR Fuel

Fuel Type	Fuel Assembly	Rod Weight (lbs)	Rods / Assembly	Canister Size (inches)	Max # Rods Canister	Weight of Fuel (lbs)	Rods / Canister	Ratio	Number of Assemblies	MTU / Assembly	Decay Power (Watts)
	W 17 x 17 STD	5.37	264	8.4 x 8.4	504	4302	401	1.52	3.0	0.464	1740
	W 15 x 15 STD/ZC	6.85	204	8.4 x 8.4	389	4192	306	1.50	3.0	0.469	1740
	W 14 x 14 STD/ZCB	6.68	179	7.7 x 7.7	323	4133	309	1.73	3.5	0.407	1740
P	B & W 17 x 17 MARK C	4.90	264	8.5 x 8.5	504	3990	407	1.54	3.1	0.456	1740
W	B & W 15 x 15 MARK B	7.00	208	8.5 x 8.5	407	4418	316	1.52	3.0	0.464	1740
R	CE 16 x 16 ONOFRE	5.70	224	8.1 x 8.1	449	4218	370	1.65	3.3	0.426	1740
	CE 14 x 14 STD	6.90	164	8.1 x 8.1	340	4124	299	1.82	3.6	0.386	1740
	Exxon 17 x 17 WE	4.81	264	8.4 x 8.4	550	4455	463	1.75	3.5	0.401	1740
	Exxon 15 x 15 WE	6.72	204	8.4 x 8.4	389	4465	332	1.63	3.3	0.432	1740
	Exxon 14 x 14 WE	6.72	179	7.7 x 7.7	323	4340	323	1.80	3.6	0.379	1691

Table 7

Decay Power Limit for Canisters with a Consolidation Ratio of 1.8

Fuel Type	Fuel Assembly	Rod Weight (lbs)	Rods / Assembly	Canister Size (inches)	Max # Rods Canister	Weight of Fuel (lbs)	Rods / Canister	Ratio	Number of Assemblies	Number of Canisters	MTU / Assembly	Decay Power (Watts)
	W 17 x 17 STD	5.37	264	8.4 x 8.4	504	2550	475	1.80	1.8	1	0.464	1031
	W 15 x 15 STD/ZC	6.85	204	8.4 x 8.4	389	2517	367	1.80	1.8	1	0.469	1045
	W 14 x 14 STD/ZCB	6.68	179	7.7 x 7.7	323	2158	323	1.80	1.8	1	0.407	908
P	B & W 17 x 17 MARK C	4.90	264	8.5 x 8.5	504	2329	475	1.80	1.8	1	0.456	1016
W	B & W 15 x 15 MARK B	7.00	208	8.5 x 8.5	407	2620	374	1.80	1.8	1	0.464	1032
R	CE 16 x 16 ONOFRE	5.70	224	8.1 x 8.1	449	2292	402	1.80	1.8	1	0.426	946
	CE 14 x 14 STD	6.90	164	8.1 x 8.1	350	4085	296	1.80	3.6	2	0.386	1723
	Exxon 17 x 17 WE	4.81	264	8.4 x 8.4	550	2292	477	1.80	1.8	1	0.401	895
	Exxon 15 x 15 WE	6.72	204	8.4 x 8.4	389	2465	367	1.80	1.8	1	0.432	961
	Exxon 14 x 14 WE	6.72	179	7.7 x 7.7	323	4342	323	1.80	3.6	2	0.379	1692

Table 8

Decay Power Limit for Canisters Containing the Maximum Number of Rods

Fuel Type	Fuel Assembly	Rod Weight (lbs)	Rods / Assembly	Canister Size (inches)	Max # Rods Canister	Weight of Fuel	Rods / Canister	Ratio	Number of Assemblies	Number of Canisters	MTU / Assembly	Decay Power (Watts)
	W 17 x 17 STD	5.37	264	8.4 x 8.4	504	2708	504	1.91	1.9	1	0.464	1095
	W 15 x 15 STD/ZC	6.85	204	8.4 x 8.4	389	2665	389	1.91	1.9	1	0.469	1106
	W 14 x 14 STD/ZCB	6.68	179	7.7 x 7.7	323	2158	323	1.80	1.8	1	0.407	908
P	B & W 17 x 17 MARK C	4.90	264	8.5 x 8.5	504	2472	504	1.91	1.9	1	0.456	1078
W	B & W 15 x 15 MARK B	7.00	208	8.5 x 8.5	389	2725	389	1.87	1.9	1	0.464	1073
R	CE 16 x 16 ONOFRE	5.70	224	8.1 x 8.1	449	2557	449	2.00	2.0	1	0.426	1055
	CE 14 x 14 STD	6.90	164	8.1 x 8.1	340	2345	340	2.07	2.1	1	0.386	989
	Exxon 17 x 17 WE	4.81	264	8.4 x 8.4	504	2422	504	1.91	1.9	1	0.401	946
	Exxon 15 x 15 WE	6.72	204	8.4 x 8.4	389	2615	389	1.91	1.9	1	0.432	1019
	Exxon 14 x 14 WE	6.72	179	7.7 x 7.7	323	4335	323	1.80	3.6	2	0.379	1689

Table 6 shows the effect of placing the consolidated fuel in two canisters. Fuel consolidation ratios ranging from 1.5 for the Westinghouse 15x15 fuel to 1.82 for the Combustion Engineering 14x14 fuel could be achieved. Again, there would be no reason to consolidate the Westinghouse 17x17 fuel, the Westinghouse 15x15 fuel, and the Babcock & Wilcox 15x15 fuel. The Combustion Engineering 14x14 fuel increase from 3.0 to 3.6 assemblies. At turns out, that the number of rods the canister could hold is the limiting factor for the Exxon 14x14 fuel.

Table 7 shows the effect of limiting fuel consolidation ratio to 1.8. If the decay power is limited to 1740 watts, a consolidation ratio of 1.8 decreases the number of assemblies that could be shipped in the cask in all but two cases. For the Combustion Engineering 14x14 fuel and Exxon 14x14 fuel, two canisters could be shipped.

Table 8 shows the effect of placing a canister with the maximum number of rods the canister could hold into the cask. Only one canister of consolidated fuel could be shipped in the cask for most of the fuel types if the canister contained the maximum number of rods it could hold. Table 8 shows, however, that two canisters of Exxon 14x14 fuel could be shipped in the cask. Again, the limit would be the number of rods the canister could hold.

3.2 Cask Payloads with BWR Fuel

For consolidated BWR fuel, five and six canister payloads were evaluated. It turns out, that both options would be optimum for the BWR consolidated fuel depending on the particular fuel design. Both the weight limit and the decay power limit considerations resulted in the same number of canisters. General Electric 8x8, General Electric 7x7, and Exxon 8x8 fuel would be best contained in five canisters. A six canister payload would provide the best basket design for the Exxon 7x7 fuel because more fuel rods can be packed in six canisters than in five canisters. The General Electric 8x8 and General Electric 7x7 fuel would be contained in 5.5"x 5.5" canisters. The Exxon 8x8 and the Exxon 7x7 would be contained in 5.2"x 5.2" canisters.

3.2.1 Weight Table

Table 9 contains the same information listed in Section 3.1.1. For the General Electric 8x8 fuel, the fuel consolidation ratio would be 1.84 and the number of assemblies that could be shipped in the cask would increase from 7.0 to 9.2 assemblies. Each canister would be 95% full. For the General Electric 7x7 fuel, the fuel consolidation ratio would be 1.71 and the number of assemblies that could be shipped in the cask would increase from 7.0 intact assemblies to 8.6 assemblies if the rods are consolidated. The canister would be 99% full. For the Exxon 8x8 fuel, the fuel consolidation ratio would be 1.55 and the number of assemblies that could be shipped in the cask would increase from 7.0 to 9.3 assemblies. The canister would be 91% full. For the Exxon 7x7 fuel, the fuel consolidation ratio would be 1.35 and the number of assemblies that could be shipped in the cask would increase from 7.0 to 8.5 assemblies. The canister would be 94% full.

3.2.2 Structural/Thermal Tables

Table 10 contains the same information listed in Section 3.1.2. If the thermal power is limited to 1740 watts, the General Electric 8x8 fuel could be cosolidated to a ratio of 1.61 and the number of assemblies that could be shipped in the cask would increase from 7.0 to 8.0 assemblies. Each canister would be 83% full. The fuel consolidation ratio would be 1.55 for the General Electric 7x7 fuel, and the number of assemblies that could be shipped in the cask would increase from 7.0 to 7.7 assemblies. Each canister would be 89% full. For Exxon 8x8 fuel, the fuel consolidation ratio would be 1.61 (95% full) and the number of assemblies that could be shipped in the cask would increase from 7.0 to 8.0 assemblies. For Exxon 7x7 fuel, the fuel consolidation ratio would be 1.29 (86% full) and the number of assemblies that could be shipped in the cask would increase from 7.0 to 7.7 assemblies.

TABLE 9
BWR WEIGHT LIMIT

Fuel Type	Fuel Assembly	Rod Weight (lbs)	Rods / Assembly	Canister Size (inches)	Max # Rods Cansiter	Weight of Fuel (lbs)	Rods / Canister	Ratio	Number of Assemblies	Number of Canisters
B	G E 8 x 8 (4,5 V2)	9.07	62	5.5 x 5.5	120	5160	114	1.84	9.2	5
W	G E 7 x 7 (4,5)	12.30	49	5.5 x 5.5	85	5160	84	1.71	8.6	5
R	Exxon 8 x 8 JP-4,5	8.95	62	5.2 x 5.2	105	5160	96	1.55	9.3	5
	Exxon 7 x 7 GE	12.30	48	5.2 x 5.2	72	5010	68	1.41	8.5	6

TABLE 10
BWR FUEL DECAY POWER LIMIT

Fuel Type	Fuel Assembly	Rod Weight (lbs)	Rods / Assembly	Canister Size (inches)	Max # Rods Cansiter	Weight of Fuel	Rods / Canister	Ratio	Number of Assemblies	Number of Canisters	MTU / Assembly	Decay Power (Watts)
B	G E 8 x 8 (4,5 V2)	9.07	62	5.5 x 5.5	120	4525	100	1.61	8.0	5	0.177	1740
W	G E 7 x 7 (4,5)	12.30	49	5.5 x 5.5	85	4665	76	1.55	7.7	5	0.184	1740
R	Exxon 8 x 8 JP-4,5	8.95	62	5.2 x 5.2	105	4465	100	1.61	8.0	5	0.177	1740
	Exxon 7 x 7 GE	12.30	48	5.2 x 5.2	72	4570	62	1.29	7.7	6	0.184	1740

4.0 CONCLUSIONS

Based on the evaluations discussed above, the following conclusions can be drawn:

1. Either BWR or PWR consolidated fuel can be transported in the TITAN cask.

2. The limit on the amount of consolidated fuel that can be transported is the determined by total weight. The reference cask has sufficient shielding to accommodate the increased spent fuel payload and criticality is not an issue because the consolidated fuel is less reactive in the compacted configuration. The thermal/structural aspect is not expected to be limiting within the increase in decay heat that would result within the weight limitation (approximately 15% over the reference maximum weight of three intact assemblies).

3. For consolidated PWR fuel, the maximum payload would consist of two canisters loaded with rods from 3.1 to 4.1 assemblies, depending on the particular consolidated fuel being considered. Less consolidated fuel can be transported in three canisters than in two because even with just two canisters, the canisters would not be entirely full and a third canister would just decrease the consolidated fuel payload by an amount equivalent to the additional weight of the third canister.

4. For consolidated BWR fuel, the maximum payload would consist of either five or six canisters loaded with rods from 8.5 to 9.3 assemblies, depending on the particular consolidated fuel being considered. Because of the weight limit, less consolidated fuel can be transported in seven canisters than in the five or six canisters.

5. Transporting spent fuel in canisters that are not entirely full would not be recommended because the rods would be unsupported and could become easily damaged (the irradiated cladding is quite brittle) from normal transportation and handling loads. It is believed that if at-reactor fuel consolidation is performed, the canisters will be fully loaded. If this is the case, only one canister of PWR fuel or 4-5 canisters of BWR fuel (depending on the type) could be transported in the reference TITAN cask.

5.0 REFERENCES

1. Letter IKH-165-88, I.K. Hall to Distribution, "Guidance on Trade-off and Impact Evaluations", November 16, 1988.

2. "Titan Legal Weight Truck Cask Alternative Material Feasibility Study", NWD-TR-008, Rev. 0, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, July 1988, (Westinghouse Proprietary).

3. "Titan Legal Weight Truck Cask Design Requirements", NWD-TR-007, Rev. 0, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, July 1988.

4. "TITAN Legal Weight Truck Cask Preliminary Design Report", NWD-TR-025, Rev. 1, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, September 1989.

TITAN LEGAL WEIGHT TRUCK CASK

THE EFFECTS OF NON-STANDARD AND FAILED FUEL AND NON-FUEL-BEARING COMPONENTS ON CASK PAYLOADS

January 1990

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1.0 INTRODUCTION

The results of the evaluation to assess the impacts of non-standard and failed fuel and non-fuel bearing components on the payload capacity of the Westinghouse TITAN Legal Weight Truck (LWT) cask are presented in this report.

The TITAN LWT cask is optimized to transport the fuel assembly types listed in Section 1.A. of the Cask Interface Guidelines (Reference 1). The cask will transport either 3 PWR assemblies (including the Combustion Engineering System 80 fuel) or 7 BWR assemblies. The cask is designed to accommodate ten year old PWR fuel having a burnup of 35 GWD/MTU or ten year old BWR fuel having a maximum burnup of 30 GWD/MTU. The cask cavity is 23.76 inches in diameter and 180 inches long. Either PWR or BWR fuel can be accommodated by using interchangeable baskets appropriate for the type of fuel being shipped.

The cask design configuration that was used in the evaluation is slightly different from the preliminary design presented in Reference 2. The two changes that are of significance from the standpoint of the evaluation are: 1) the use of BISCO NS-4-FR as the neutron shielding material instead of BOROSILICONE, and 2) an increase in the structural shell thickness from 1.25 inches to 1.40 inches. The preliminary design of the LWT cask will be revised to incorporate these design changes and it was therefore considered appropriate to use the new design for purposes of this evaluation.

2.0 DISCUSSION

A discussion of the approach used in the evaluation of the impact of non-standard and failed fuel, and non-fuel bearing components on cask payload capacity and the results obtained are presented in this section.

2.1 Nonstandard Fuel

A review of the non-standard fuel assembly listing provided in Reference 1 and the information contained in Volume 3 of Reference 3 and Reference 4 revealed a few discrepancies in assembly types that are currently out in the field. Table 1 presents the non-standard fuel assembly data given in Reference 1. Table 2 presents the fuel assembly types from References 3 and 4 that were considered in the evaluation.

The limited quantity fuels are generally low burnup and will have been in storage for decades before they are shipped to a repository or a Federal Interim Storage facility. Consequently, the level of radiation would not be such as to challenge the thermal or shielding design limits and those aspects were not considered further. The focus of this study was to estimate the total number of fuel assemblies of a particular type that could be shipped and whether the reference basket designs are adequate or new basket designs were required.

Table 2 shows the non-standard fuel assembly types that can be accommodated in the LWT cask and whether the reference design baskets would suffice or special baskets are required. The only non-standard fuel that cannot be transported by the TITAN LWT cask is the 199 inch long Westinghouse 17 x 17 XLR assembly. Table 3 provides a summary of the payload capacities for these limited quantities of fuels.

2.2 Failed Fuel

Failed fuel can be broadly categorized into two types: 1) fuel assemblies that leak but which are dimensionally close to non-failed fuel assemblies, and 2) fuel assemblies that are physically deformed (with or without leaks) and cannot fit into the cask fuel baskets. The TITAN LWT cask can accommodate failed fuel of the first type without requiring encapsulation of the fuel assembly. However, the usual practice at the utilities is to either encapsulate an entire failed fuel assembly or individually encapsulate the failed fuel rods. In accordance with the Reference 5 guidelines, this evaluation was directed towards estimating the LWT cask payload capacities for these two cases of encapsulated failed fuel.

For the case where an entire fuel assembly is encapsulated in a canister, it is assumed that the canister has external dimensions about 0.4 inches larger than the width of the assembly. Some of these canisters will not fit into the reference design baskets and new basket designs would be required.

For the case where defective fuel rods are individually encapsulated, it is assumed that each rod is placed in a tube having 0.75 inch outer diameter and 0.049 inch thick wall. These tubes, in turn, are placed in a canister having external dimensions no larger than the envelope of the corresponding intact fuel assembly.

Table 4 provides a summary of the payload capacities for canisterized failed fuel assemblies and for individually encapsulated failed rods.

Depending on the type of fuel, up to 3 PWR or 7 BWR encapsulated failed fuel assemblies and between 294 and 378 encapsulated failed fuel rods can be accommodated in the LWT cask.

Table 1

Limited Quantity Spent Nuclear Fuels (From the Cask Interface Guidelines)

PWR Spent Fuels:

Westinghouse Electric	16 x 16
Westinghouse Electric	13 x 13
Babcock and Wilcox	14 x 14
Combustion Engineering	14 x 14 XL
Combustion Engineering	15 x 15
Gulf United Nuclear	17 x 17
Westinghouse Electric	17 x 17 XL

BWR Spent Fuels:

General Electric	11 x 11
General Electric	9 x 9
General Electric	6 x 6
Exxon Nuclear	11 x 11
Exxon Nuclear	10 x 10
Exxon Nuclear	9 x 9
Exxon Nuclear	6 x 6
Allis Chalmers	10 x 10
Nuclear Fuel Services	9 x 9
United Nuclear	6 x 6
Westinghouse Electric	8 x 8

Table 2
Existing Limited Quantity Spent Fuel

Fuel Type	Fuel Assembly	Assembly Width (inches)	Assembly Length (inches)	Number Assm's in Storage	Fit in Baseline Cask?	New Basket Req'd?
PWR	W 13 x 14	6.3	138	160	Yes	Yes
	W 17 x 18	7.6	112	76	Yes	No
	W 17 x 17	8.4	199	new	No	No
	CE 14 x 14	8.1	146	290	Yes	No
	CE 15 x 15	8.2	148	273	Yes	No
	CE 15 x 16	7.6	112	40	Yes	No
	Exxon 15 x 16	7.6	112	228	Yes	No
	United Nuclear 15 x 16	7.6	112	89?	Yes	No
BWR	GE 6 x 6	4.7	95	176	Yes	Yes
	GE 6 x 6	4.3	134	365	Yes	Yes
	GE 9 x 9	6.5	82	143	Yes	Yes
	GE 11 x 11	6.5	82	6	Yes	Yes
	Exxon 6 x 6	4.7	95	126	Yes	Yes
	Exxon 6 x 6	4.3	134	66	Yes	Yes
	Exxon 9 x 9	6.5	82	4	Yes	Yes
	Exxon 9 x 9	5.3	171	new	Yes	No
	Exxon 10 x 10	5.6	102	178	Yes	No
	Exxon 11 x 11	6.5	84	128	Yes	Yes
	United Nuclear 6 x 6	4.7	134	457	Yes	Yes
	Nuclear Fuel 11 x 11	6.5	84	2	Yes	Yes
	Allis Chalmers 5 x 5	3.5	82	188	Yes	Yes
	Allis Chalmers 10 x 10	5.6	102	156	Yes	No

Table 3
Payload Capacities of Nonstandard Fuel
(Limited Quantity Fuel)

FUEL TYPE	FUEL ASSEMBLY TYPE	PAYLOAD WITH REF. BASKET	PAYLOAD WITH SPEC. BASKET	REMARKS
PWR	WEST. 13X14, 17X18	3	4	REQUIRES ONE 4-CELL SPECIAL BASKET DESIGN
	B&W 14X14	3	4	
	UNITED NUCLEAR 15X16	3	4	
	CE 15X16	3	4	
	EXXON/ANF 15X16	3	4	
	CE 15X15	3	NO INCREASE	
BWR	EXXON 10X10	7	NO INCREASE	
	AC 10X10	7	NO INCREASE	
	GE 9X9, 11X11	WILL NOT FIT	10	REQUIRES ONE 5-CELL SPECIAL BASKET
	EXXON 9X9, 11X11	WILL NOT FIT	10	
	NFS 11X11	WILL NOT FIT	10	
	GE 6X6	7	10	REQUIRES ONE 10-CELL SPECIAL BASKET
	EXXON 6X6	7	10	
	UN 6X6	7	10	
	AC 5X5	14	42	SPECIAL BASKET

2.3 Non-fuel-Bearing Components

Volume 1 of Reference 3 classifies non-fuel bearing components as Spent Fuel Disassembly (SFD) hardware and Non-fuel Assembly (NFA) hardware.

SFD hardware includes PWR fuel assembly skeletons (top and bottom nozzles, grid spacers, hold-down springs, etc.); and the top and bottom tie plates, fuel rod compression springs, grid spacers and water rods for BWR assemblies.

NFA hardware includes reactor hardware that is not necessarily tied in with the fuel assemblies. Such hardware is used within or between assemblies, is not permanently attached to an assembly, and has a life in the core that is different from that of the fuel assemblies themselves. This hardware includes such items as BWR fuel channels and control blades, and PWR control rods and burnable poison assemblies in addition to neutron sources, in-core instrumentation and thimble plug assemblies.

SFD hardware weighs about 75 pounds for PWR assemblies and about 20 pounds for BWR assemblies. It is possible to compact these components following disassembly and load up to ten of these compacted skeletons in special canisters which would fit into the reference PWR and BWR baskets. Such canisters filled with compacted skeletons would weigh less than the fuel assemblies and have a negligible decay power. However, because of the activation of the cobalt in the stainless steel and Inconel, the gamma radiation could be high enough to exceed allowable dose rates. It is estimated that the TITAN LWT cask can accommodate SFD hardware from 24 to 30 PWR assemblies or 50 to 70 BWR assemblies. The exact number would depend on the particular chemical composition of the hardware and its irradiation history.

Table 4
Summary of Titan LWT Cask Failed Fuel Payload Capacities

Fuel Type	Fuel Assembly Type	Failed Intact Assemblies: Canister Size (inches)	Failed Intact Assemblies: Canisterized Assemblies	Failed Intact Assemblies: New Basket	Failed Rods: Canister Size (inches)	Failed Rods: Encapsulated Rods	Failed Rods: Number of Canisters	Failed Rods: New Basket
PWR	W 17 x 17 STD	8.8 x 8.8	2	Yes	8.4 x 8.4	360	3	No
	W 15 x 15 STD/ZC	8.8 x 8.8	2	Yes	8.4 x 8.4	360	3	Yes
	W 14 x 14 STD/ZCB	8.1 x 8.1	3	No	7.7 x 7.7	297	3	No
	B & W 17 x 17 MARK C	8.9 x 8.9	2	Yes	8.5 x 8.5	378	3	No
	B & W 15 x 15 MARK B	8.9 x 8.9	2	Yes	8.5 x 8.5	378	3	Ye
	CE 16 x 16 ONOFRE	8.5 x 8.5	3	No	8.1 x 8.1	315	3	No
	CE 14 x 14 STD	8.5 x 8.5	3	No	8.1 x 8.1	315	3	No
	Exxon 17 x 17 WE	8.8 x 8.8	2	Yes	8.4 x 8.4	360	3	No
	Exxon 15 x 15 WE	8.8 x 8.8	2	Yes	8.4 x 8.4	360	3	Yes
	Exxon 14 x 14 WE	8.1 x 8.1	3	No	7.7 x 7.7	297	3	No
BWR	G E 8 x 8 (4,5 V2)	5.8 x 5.8	7	Yes	5.5 x 5.5	322	7	Yes
	G E 7 x 7 (4,5)	5.8 x 5.8	7	Yes	5.5 x 5.5	322	7	Yes
	Exxon 8 x 8 JP-4,5	5.5 x 5.5	7	Yes	5.2 x 5.2	294	7	Yes
	Exxon 7 x 7 GE	5.5 x 5.5	7	Yes	5.2 x 5.2	294	7	Yes

If the NFA hardware is not consolidated, individual intact control rod, burnable poison, neutron source or thimble plug assemblies can be placed in canisters that fit in the reference design cask baskets. Also for those cases where weight and dimensional constraints are not exceeded, the PWR control and poison assemblies can be shipped with the spent fuel assemblies. Similarly, BWR fuel channels can be transported using the reference design BWR basket. Depending on the weight of the fuel assembly and the weight of the channels, the channels can be shipped along with the BWR assemblies. This is not possible with the BWR control blades which would require a canister that is 10.1 inches wide. Two such canisters can be accommodated with a special basket. There are no weight, power or shielding constraints with any of the NFA if it is not compacted.

If the NFA hardware from PWR related assemblies is disassembled, compacted and placed in canisters, it is physically possible to fit more hardware into the cask than can be accommodated from a weight standpoint. The LWT cask can accommodate a combined weight of payload and associated basket weight of approximately 6800 pounds. The baskets typically weigh about 1800 pounds; hence the TITAN cask can accept payload weights of approximately 5000 pounds. For example, the Westinghouse 17X17 control rod assembly weighs 149 pounds, contains 24, 153 inch long control rods and an 8 inch long spider. There are normally 61 control rod assemblies in a core and 2 or 3 sets of these assemblies would be needed over the life of the reactor. All the spiders for a complete set of assemblies could be accommodated in the LWT using three canisters. And individual control rods from 30 assemblies (weighing 4250 pounds) could be packaged in three 225 pound canisters and accommodated in the cask. Thus if disassembled, a complete set of control rods assemblies could be transported in just three loads: one load of spiders and two with the control rods themselves.

3.0 CONCLUSIONS

The Westinghouse LWT cask, with its interchangeable basket design, provides a very high degree of flexibility to accommodate non-standard and failed fuel, and non-fuel bearing materials. The results of the evaluations show that the cask has the capability to transport:

1. All of the nonstandard, limited quantity spent fuel that is currently in storage.

    Being a common use cask, it accommodates all of the BWR fuel assemblies in addition to the PWR assemblies. The only nonstandard fuel that is not transportable in the cask is the extra long (199 inches) Westinghouse 17 x 17 fuel being used at the South Texas plant.

2. Both failed intact assemblies and encapsulated failed fuel rods.

    Generally, failed intact assemblies that must be fitted into a canister for shipment will require a special basket, in which case 2 PWR assemblies or 7 BWR assemblies can be shipped at a time. If the failed fuel rods are individually encapsulated and placed in canisters for storage and shipment, the cask can accommodate between 294 and 378 encapsulated rods depending on their weights.

3. From 24 to 30 compacted PWR assembly skeletons or 50 to 70 compacted BWR skeletons.

   Non-fuel assembly hardware can be shipped intact in the reference design baskets (except for BWR control blades which will require a special basket), or if disassembled and contained in canisters, significant quantities can be shipped. A complete set of control rod assemblies from a typical Westinghouse reactor can be shipped in three loads. An additional option is that PWR control assemblies can be shipped in the reference cask with the spent fuel as long as the combined weight of the control and fuel assemblies does not exceed the loading basis for the basket (1515 pounds per PWR assembly).

None of the materials considered in this study will have heat generation rates approaching that used for the design of the cask. Nor will they involve loadings for which criticality control is an issue. Consequently, for those cases where new baskets would be required or desirable to accommodate these nonstandard payloads, the baskets could be of a simpler design than those used to support the standard spent fuel.

4.0 REFERENCES

1. Contract, "Cask Systems Development, Initiative I, Development of From-Reactor Casks: Task 4.A Legal Weight Truck Cask Development," DE-AC07-88ID12699, May 13, 1988.

2. "TITAN Legal Weight Truck Cask Preliminary Design Report," NWD-TR-025, Rev 1, Westinghouse Electric Corporation, Nuclear Services Division, September, 1989.

3. "Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation," DOE/RW-0184, U. S. Department of Energy, Office of Civilian Radioactive Waste Management, December, 1987.

4. Moore, R. S., et. al., "A Classification Scheme for LWR Fuel Assemblies," ORNL/TM-10901, Oak Ridge National Laboratory, November, 1988.

5. Letter, Hall, I. K., "Guidance on Trade-Off and Impact Evaluations," IKH-165-88, EG&G Idaho, Inc., November 16, 1988.

TITAN LEGAL WEIGHT TRUCK CASK

RECOMMENDED SEALING SURFACE

FOR HOT CELL LOADING/UNLOADING OPERATIONS

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1.0 INTRODUCTION

The Statement of Work for the Legal Weight Truck Cask requires that a study be performed during the Preliminary Design Phase and a recommendation made on the cask sealing surface for hot cell loading and unloading operations. This report summarizes the results of a study that was recently completed to evaluate alternative sealing surfaces and provides recommendations on the preferred design approach.

2.0 OPERATIONAL CONSIDERATIONS

Dry transfer operations involving spent fuel transportation are generally performed by mating the cask closure end to a hot cell port. This interface is normally provided with a seal in order to prevent the spread of contamination outside of the controlled hot cell environment. Once the cask is mated to the hot cell port, the plug port and the closure lid are removed and the fuel assemblies loaded or unloaded. Typically, the hot cell port is provided with a shield collar or adapter ring which mates with the cask.

Dry transfer operations are envisioned to take place at spent fuel interim dry storage facilities, the Monitored Retrievable Storage facility or the repository. As none of these facilities have been designed beyond the conceptual stage, details of the hot cell port interface are not currently available. However, such detailed information is not essential for providing the transportation casks with features required to perform the sealing function.

Inflatable elastomer seals have been successfully used in nuclear facilities for sealing interfaces between hot cells and components. Several facilities in the U. S. including the Waste Isolation Pilot Plant have incorporated such seals for hot cell/cask interfaces. In addition to being simple and reliable, these seals do not impose any special configuration requirements on the interfacing component other than for a reasonably smooth and clean sealing surface. For these reasons, all the concepts considered in this study use inflatable seals.

3.0 EVALUATION OF ALTERNATIVE CASK SEALING SURFACE CONCEPTS

Three approaches were considered in this study. The conceptual designs for each of these are described in this section.

3.1 Face Seal

Figure 1 shows a conceptual arrangement of a face seal between the cask and the hot cell port shield collar. A compact inflatable seal could be used to perform the sealing function. Details of the interface are shown in Figure 2. The LWT cask design allows for a flat and smooth surface approximately 2 inches wide to be provided on the cask face for the seal to bear against. The principal advantages and disadvantages of this concept are given in Table 1.

3.2 Bore Seal

Figures 3 and 4 show the conceptual design of the cask sealing surface based on using a circumferential or bore seal. This concept has the same basic advantages and drawbacks of the face seal concept except that even tighter positioning accuracy will be required.

3.3 Cask Seal Ring

In this approach, a cask seal ring, shown in Figure 5, is installed on top of the cask before it is moved into position under the hot cell port shield collar. The upper end of the seal ring is provided with a wide flange which provides a sealing surface for a large inflatable seal. The interface between the seal ring lower flange and the cask face is sealed by a flat elastomer seal attached to the flange.

TABLE 1

Comparison of Cask Sealing Surface Concepts

CONCEPT	ADVANTAGES	DISADVANTAGES
1. Direct Face Seal between cask and Hot Cell Port Shield Collar	• One Seal Interface	• Cask has to be raised to establish sealed interface. • Size of inflatable seal has to be small because of narrow sealing face on the cask. • Not enough room to install crud barrier. • Relatively accurate diametral and vertical positioning required. • Separate hot cell ports for different cask sizes and geometries required.
2. Direct Bore Seal between cask and Hot Cell Port Shield Collar	• One Seal Interface	• Has all the disadvantages of Concept 1. • Greater positioning accuracy required than for Concept 1.
3. Cask Seal Ring and Inflatable Seal	• Wide Seal surface interface with Hot Cell Port Shield Collar allows for liberal positioning tolerances. • Provides flexibility for installing crud barrier. • More room available for removal and installation of closure lid. • Capability for incorporating large inflatable seal on Hot Cell Port Shield Collar makes it feasible to eliminate raising of cask to establish seal. • The Cask Seal Ring provides flexibility for using same Hot Cell Port for different cask sizes and geometries.	• Two Sealing Interfaces

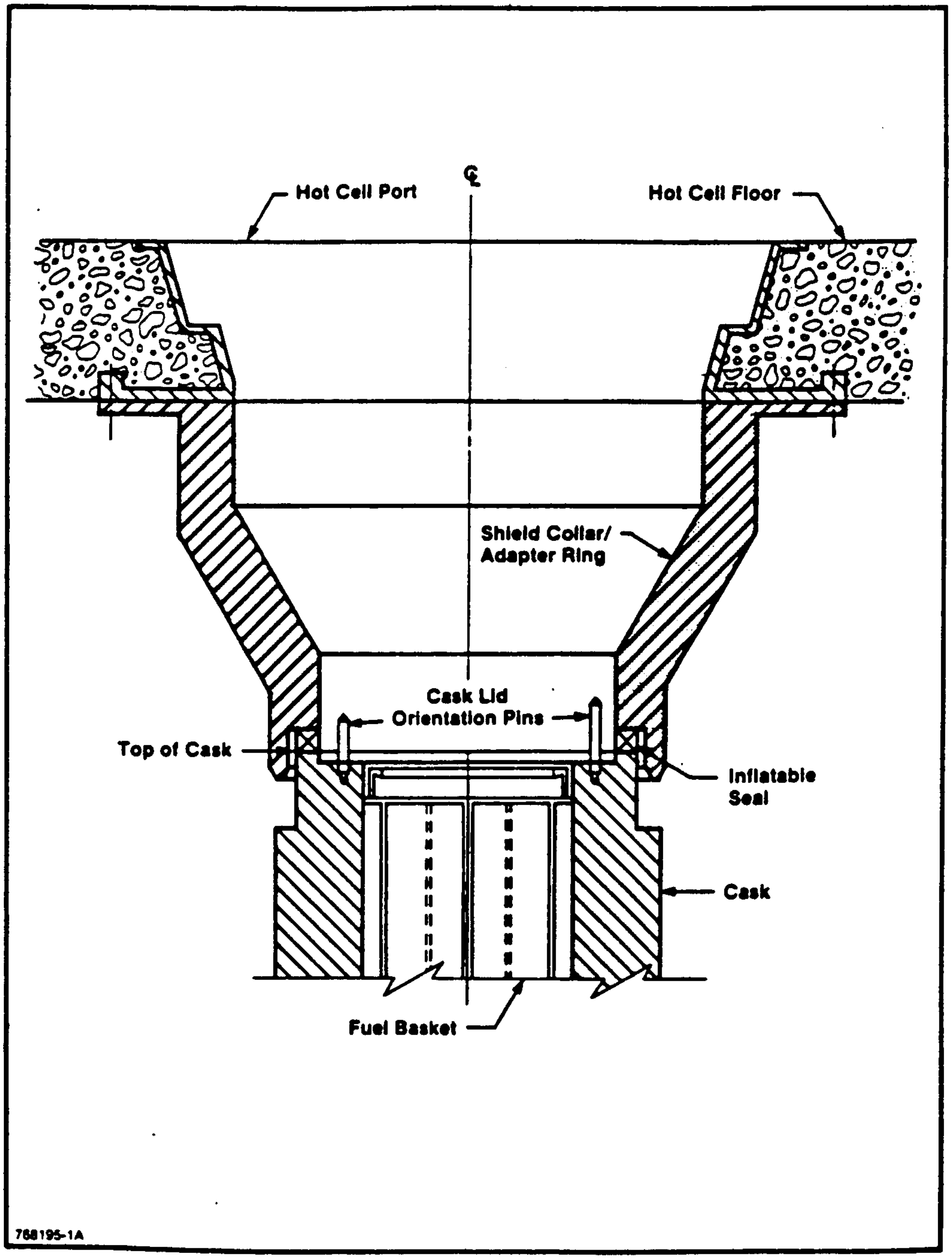


Figure 1. Concept 1 - Direct Face Seal Between Cask and Hot Cell Port Shield Collar

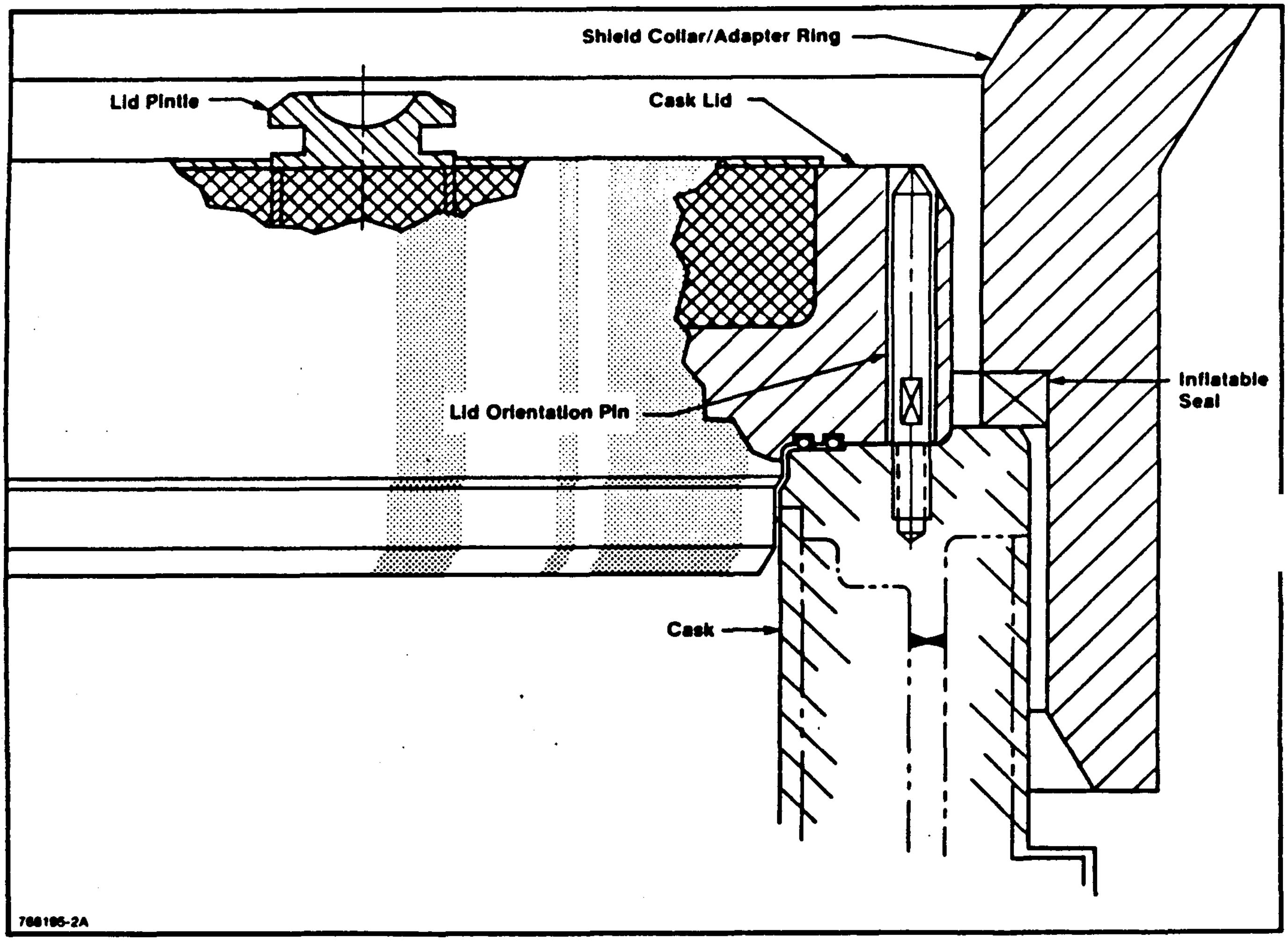


Figure 2 Concept 1 - Detail of Cask Closure Lid and Inflatable Seal

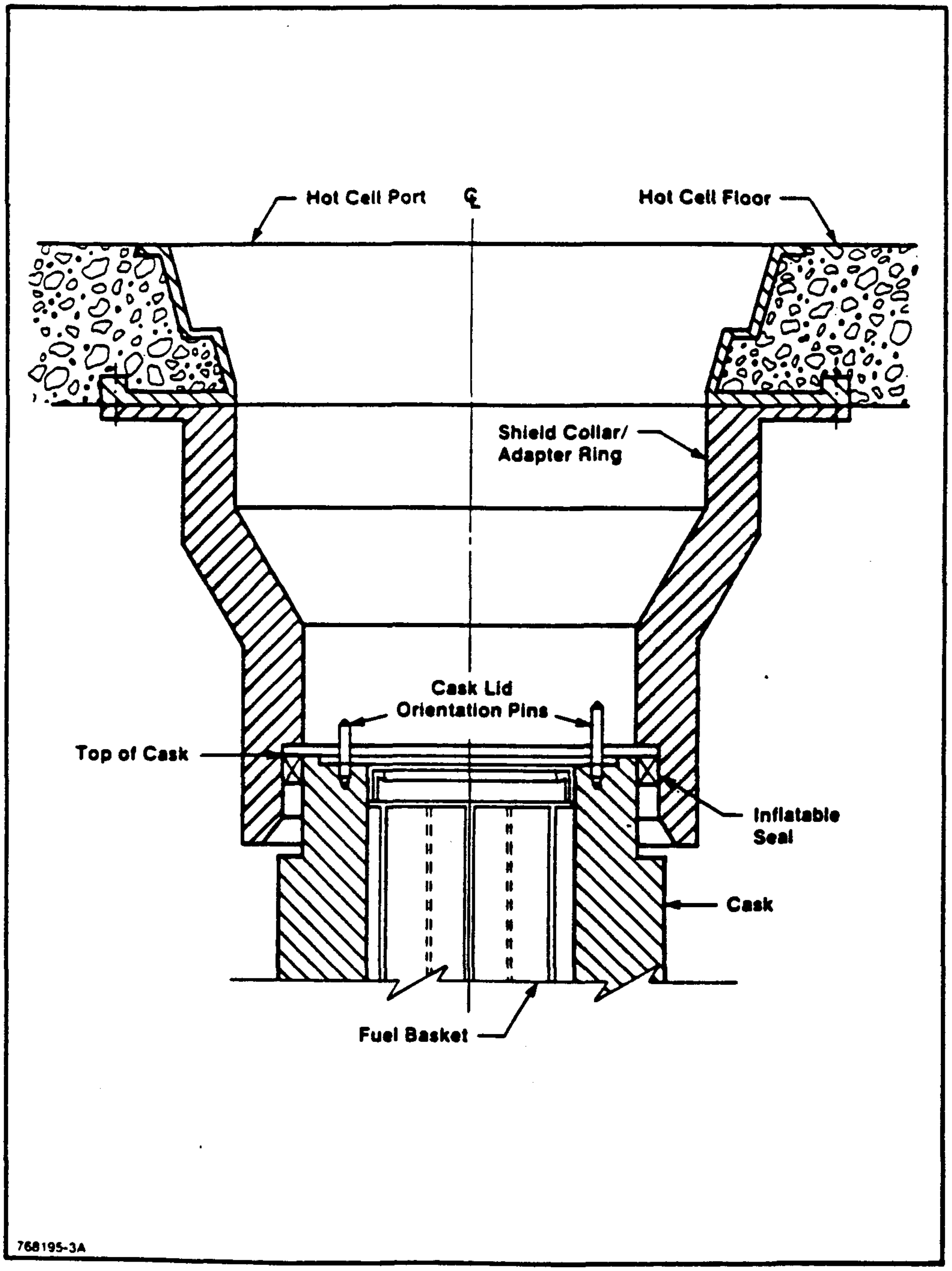


Figure 3 Concept 2 - Direct Seal Between Cask
and Hot Cell Port Shield Collar

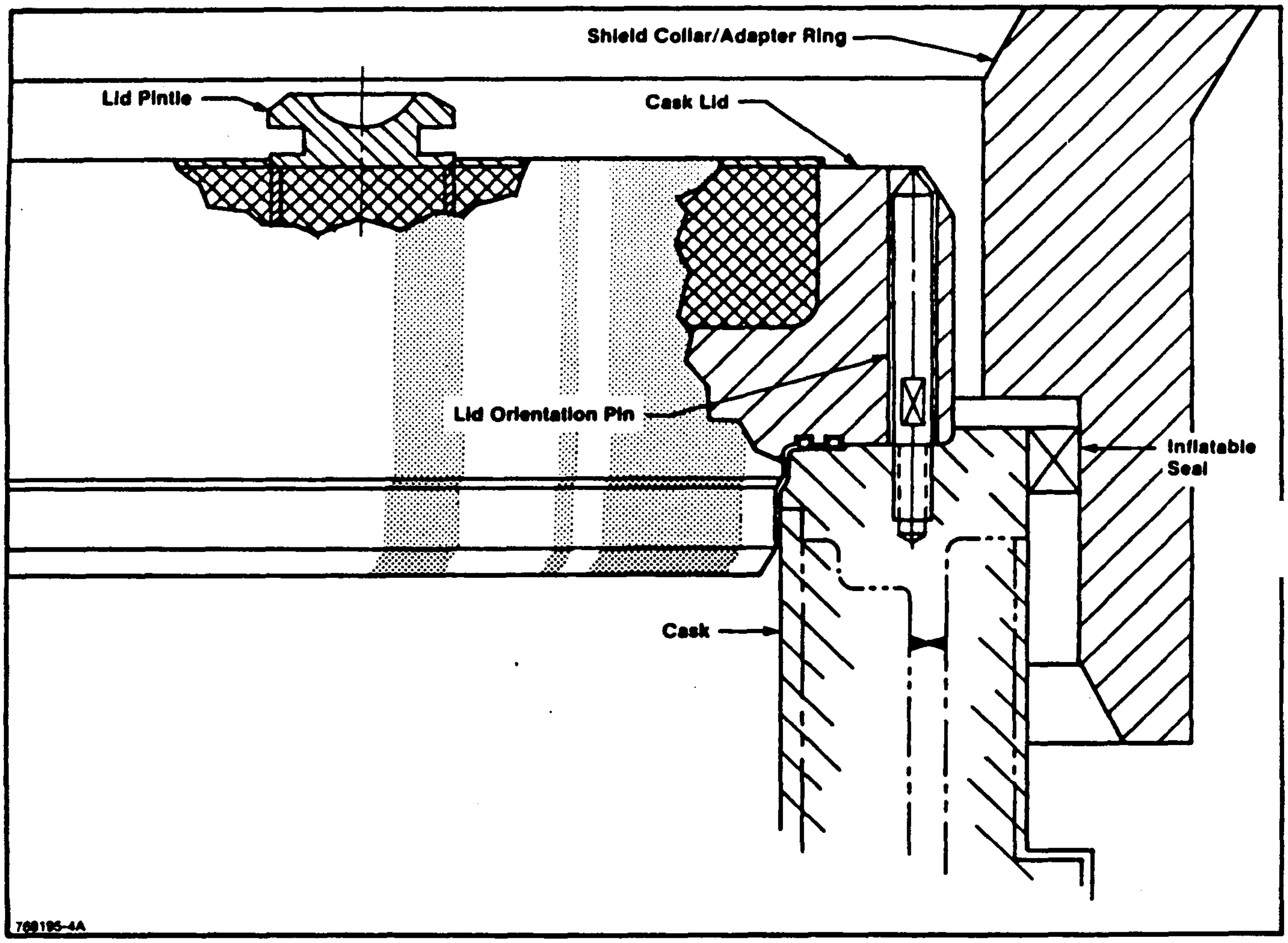


Figure 4. Concept 2 - Detail of Cask Clsure Lid and Inflatable Seal

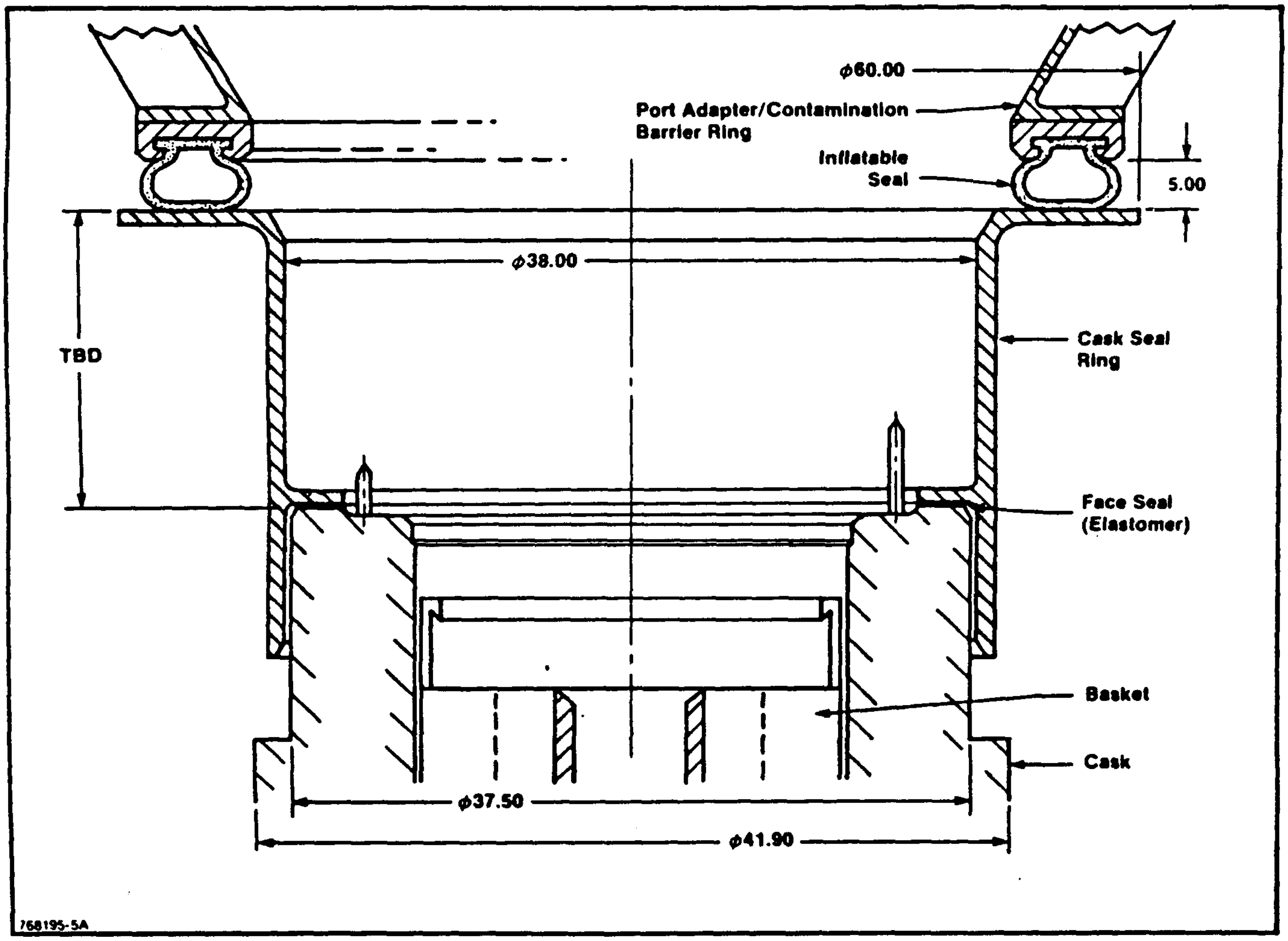


Figure 5. Concept 3 - Cask Seal Ring and Inflatable Seal

The use of a cask seal ring provides operational flexibility in a number of areas. First, the wider opening above the cask allows for the provision of a crud barrier sleeve shown in Figure 6, that prevents contamination of the cask face during fuel transfer. Secondly, the cask seal ring allows casks of different sizes and shapes to use the same hot cell port as the seal rings can be tailored to individual cask types and are relatively less expensive than having a separate hot cell port and shield collar for each cask type. The wide sealing surface provided by the top flange of the seal ring also allows for liberal positioning tolerances.

With the cask seal ring concept, there are now two sealing surfaces rather than one present with the other two concepts. However, this is not considered to be a significant drawback because both seals are effective in preventing egress of contamination.

Table 1 provides a listing of the unique advantages of the cask seal ring concept.

4.0 RECOMMENDED CONCEPT

The clear advantages of the cask seal ring concept, shown in Figure 5, over the other two concepts make it an ideal choice for hot cell loading and unloading operations. It is therefore recommended that this approach be implemented for the LWT cask.

The crud barrier sleeve is a very desirable feature for use during the fuel transfer operations and should be considered for implementation by the facility designers.

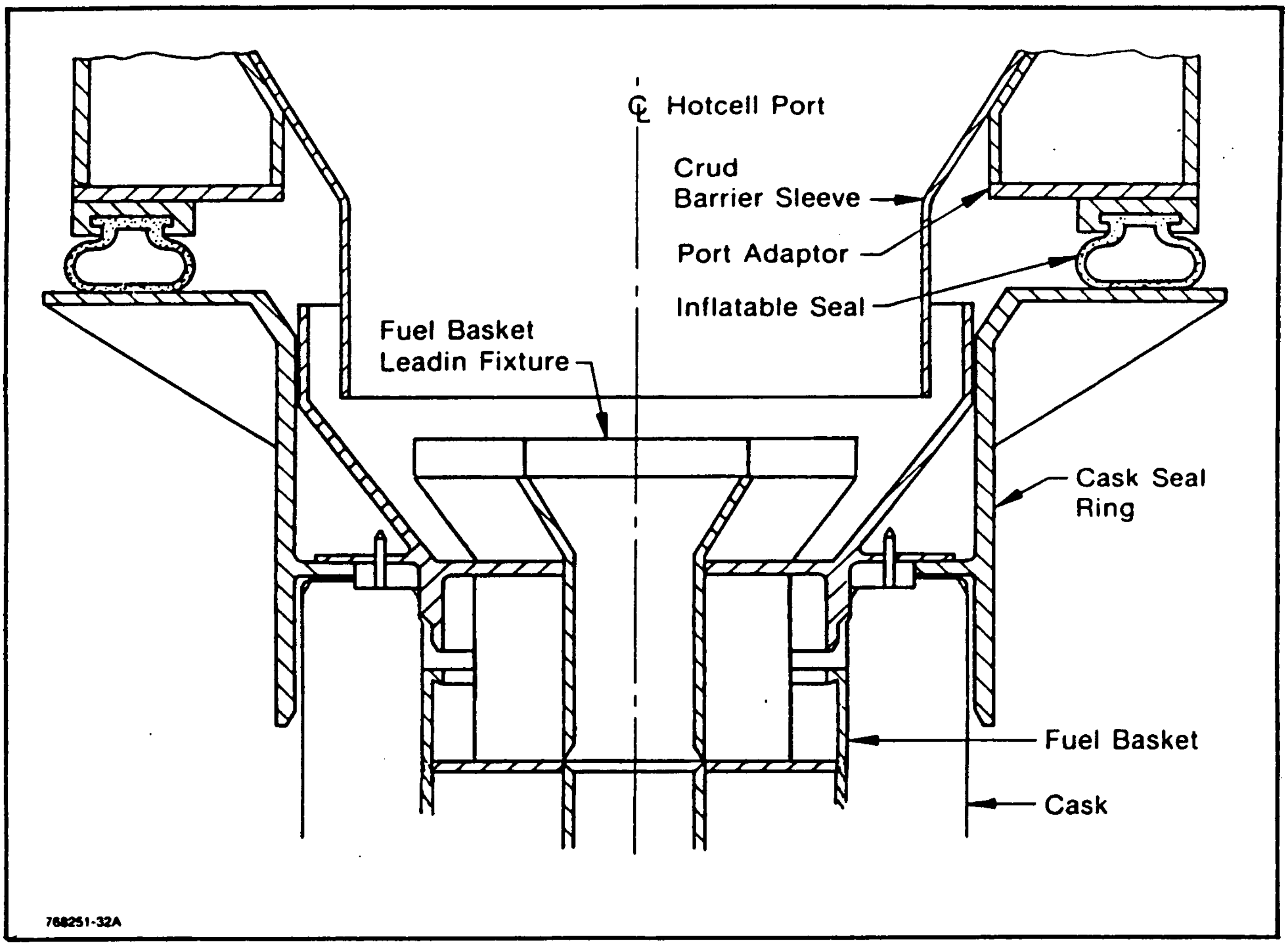


Figure 6. Concept 3 - With Fuel Basket Lead-in Fixture
and Crud Barrier Sleeve

APPENDIX B

RESPONSES TO DOE PRELIMINARY DESIGN REVIEW COMMENTS

RESPONSES TO DOE PRELIMINARY DESIGN REVIEW COMMENTS

This Appendix presents the Westinghouse responses to the comments from the Preliminary Design Review of the TITAN LWT Cask and Ancillary Equipment conducted by the DOE Technical Review Group. These comments were provided in the Review Report (Reference B-1).

The comments and the corresponding responses have been organized by reviewer using the reviewer's intials for identification as follows:

TM	T. McLaughlin
KC	K. Childs
WY	W. Yoon
PB	P. Bennett
HY	H. Yoshimura
WS	W. Stoddart
RJ	R. Jones
RT	R. Thompson
HD	H. Dyer
HS	H. Spaletta
RP	R. Peterson

Reference

B-1 Letter, McLaughlin, T., "Report of Formal Preliminary Design Review for Westinghouse TITAN LWT Cask and Ancillary Equipment," TBM-37-89; EG&G Idaho, November 21, 1989.

COMMENT:

TM-1 The ratcheting growth of the depleted uranium (DU) needs to be investigated to ensure the alloy selected is below growth limits for the proposed operating temperatures. Since the fabrication clearances are small, growth in the DU could result in interferences and increased stress on the titanium shells.

RESPONSE:

TM-1 The differential thermal growth between the DU and titanium shells is accommodated by providing radial and axial gaps in the DU. This allows the growth of the depleted uranium without causing stresses in the titanium shells. In Section 2.6.2 of the Preliminary Design Report it was shown that the radial gaps between the DU and titanium shells are sufficient to prevent loading of the titanium shells due to differential shrinkage during the cold condition. In the axial direction, the differential shrinkage causes a gap to open up in the shielding. Similarly, for the heat condition, gaps between the DU and titanium shells are selected to insure that the titanium shells will not be stressed. Analysis showing that the DU will not load the titanium shells for the heat condition will be included in the final design report.

COMMENT:

TM-2 The contractor must present technically justifiable data that the BORO-SILICONE neutron shield material will not degrade or be lost during accident conditions to support assumptions made for the criticality, thermal and shielding analyses. Radiolysis gas generation of the material should also be examined.

RESPONSE:

TM-2 Westinghouse will not take credit for the neutron shielding material in the shielding and criticality evaluations of the post-accident conditions.

The preliminary shielding analysis showed that the one rem per hour dose rate limit at one meter from the external surface of the package can be met without the neutron shielding in place, as discussed in Section 3.5.6, Pages 3-36 and 37 of the Preliminary Design Report.

Criticality calculations completed since the report was written show that K_{eff} is essentially unaffected by the neutron shielding. These calculations will be included in the final design report. It is, therefore, not necessary to demonstrate that the neutron shield will not degrade or be lost during the accident conditions.

In the case of thermal analysis, the maximum structural shell temperatures will be determined assuming both the presence and absence of the neutron shield and the worst case temperatures will be used in the design.

COMMENT:

TM-3 The data qualifying the cask closure head O-ring seals and their expected life at -40°F should be provided for the final design.

RESPONSE:

TM-3 Westinghouse plans to utilize the experimental data from SNL's Seal Qualification Test Program to support the selection and performance predictions of the closure head O-ring seal material. In addition, the performance of the O-ring seal design configuration specific to the TITAN LWT cask will be demonstrated through design verification testing of a half-scale model cask. This approach was discussed with the NRC at a DOE-sponsored meeting of all the cask contractors and SNL in September 1989 and is acceptable to the NRC. The data qualifying the O-ring seals, including information from the design verification testing, will be included in the final design report.

COMMENT:

TM-4 The vendor's performance data that is used as the design basis for the material of the impact limiter should be provided during the final design. This should include the identification of the materials used to construct the honeycomb, performance data of the adhesive used to bond the core to the stainless steel sheets and bond the segments together, and data on the degradation of the material with age from such items as atmospheric corrosion (consider leakage into honeycomb), radiation resistance, etc. This data should support if it is feasible to assume the honeycomb can last the life of the cask or if inspection and maintenance procedures should be considered in the design.

RESPONSE:

TM-4 The vendor's performance data will be provided in the final design report. The information will include a list of materials of construction, properties of adhesives, radiation resistance, and data on corrosion. Inspection and maintenance procedures will be developed if it cannot be demonstrated that the impact limiter materials cannot last the life of the cask.

COMMENT:

TM-5 The removal and handling of the impact limiters is considered to have a major effect on the final design and performance of the impact limiters. The methods to be used should be resolved early in the final design in order that design and testing of the impact limiter can proceed.

RESPONSE:

TM-5 Westinghouse has developed an impact limiter removal and handling concept which allows for the limiters to remain on the transporter while the cask is removed for loading/unloading operations. This concept was not included in the Preliminary Design Report but was presented at the design review meeting. The approach was a rail-mounted cradle to support and position the impact limiter for attachment to the cask and to move it away from the cask following disassembly. Other concepts will also be evaluated and a selection made early in the final design phase. Any design features required to be provided on the impact limiters will be incorporated, if necessary, in the design of the test articles.

COMMENT:

TM-6 **Thermal expansion of cask in its support system needs to be further addressed to determine if cyclic loading from the trunnions can occur during normal operation. The normal deflection of the trailer should also be considered in combination with any thermal cycling to determine if loading and over stressing of the support system occurs from the cask trunnions.**

RESPONSE:

TM-6 The effects of differential thermal expansion and deflection of the trailer on the cask support system design will be evaluated during the final design phase. Features to accommodate differential thermal expansion include the use of articulated tiedown clamps. The trailer will be a specially engineered structure designed to minimize deflections as prior experience with spent fuel shipments by truck have shown that the large deflections associated with commercial light-weight trailers have been responsible for structural failures. The final design report will include the results of the evaluations to demonstrate that the integrated cask/support/transporter system meets the specified performance requirements.

COMMENT:

TM-7 The accelerations used in the fuel basket analysis should be reviewed during slap down conditions to ensure that the highest accelerations have been identified.

RESPONSE:

TM-7 The maximum lateral acceleration for the one-foot drop normal condition during slapdown of the cask is 18.4 g's for the 15° drop orientation (see Design Report Section 2.6.7). That value was used in the evaluation of the fuel baskets. A dynamic load factor (DLF) of 1.02, which was obtained for the side drop orientation, was used to scale up that acceleration. (The SCANS program does not calculate correct DLFs for shallow angle drops.)

For the 30-foot drop accident condition (see Design Report Section 2.7.1), the lateral acceleration of 49.5 g's at the 0° orientation was used for the design of the fuel baskets along with the corresponding 1.35 DLF. For the 15° orientation, the maximum acceleration is slightly (3.4%) higher, with a value of 51.2 g's. However, it is believed that the DLF is lower than 1.35 based on the SCANS results for the 30° and 45° drop orientations. The accelerations will be reviewed during final design for the shallow angle oblique drops condition to ensure that the highest basket loads have been identified.

COMMENT:

TM-8 The structural analysis should ensure that the load transfer from the DU to the titanium has been fully considered. This includes concentrated loads on the titanium shell during puncture drops from the jointed DU. This analysis should also define the boundary conditions that will be used for applying the static plus dynamic load factor.

RESPONSE:

TM-8 The structural strength of the DU is conservatively ignored in determining the overall beam type behavior of the cask during drop and puncture accidents. Only the mass of the DU is considered. However, when the DU can act to transfer loads between titanium shells, its strength is considered. In the analysis of the cask for a side puncture, (Section 2.7.2 of the Design Report), the compressive strength of the DU is modeled using interface elements to enable the punch load to be transferred to the inner titanium shell from the outer shell. This approach provides for the concentrated loads under the punch to act on the inner titanium shell and to be accounted for in the analysis. This method will be used when the puncture and drop analysis of the closure head and bottom head assembly are performed during final design.

Dynamic analysis of the cask using a relatively simple beam model is used to determine the overall structural response of the cask to the drop accidents. The loads, accelerations, and dynamic load factors (DLF) obtained from these analyses will be applied statically to detailed finite element models of structural components or sections of the cask where greater detailed analysis is required. An example of this approach is the closure head analysis given in Section 2.7.1 of the Design Report. For this detailed finite element model, boundary conditions were selected to obtain the correct structural response for the applied loadings. Along the axes of symmetry, boundary conditions were selected to maintain symmetry (in-plane displacement and slopes set equal to zero). In addition, the length of the cylindrical section was selected to dampen out local stress effects.

COMMENT:

TM-9 There was no indication in the report or presentation that combining of stresses from load combinations per RG 7.8 to use with RG 7.6 criteria has been initiated.

RESPONSE:

TM-9 Tables 2.1-4 and 2.1-6 in the Design Report provide the load combinations that will be used in the evaluation of the cask. These are in agreement with Regulatory Guide 7.8. These load combinations were considered in the preliminary evaluation of the cask. For example, in Section 2.6.7.5 of the Design Report, the primary plus secondary stresses that result from the 1-foot drop accident and differential thermal growth during the heat condition are combined and compared to Regulatory Guide 7.6 limits. In most cases, stresses that result from pressure loadings and fabrication were ignored in the preliminary design phase because they are expected to be low, and will not contribute significantly to the overall combined stresses. It is noted that for all the major loading conditions that cause significant stresses in the cask components, the higher temperature sets the design allowables but does not cause additional stresses. For example, design allowables were taken at 200°F for the heads, 240°F for the outer shell, and 275°F for the inner shell. In this manner, the drop and puncture events were combined with the heat condition.

All the required load combinations will be explicitly calculated during the final design phase and included in the final design report.

COMMENT:

TM-10 The approach being followed for puncture on the closure may lead to very highly stressed bolts.

RESPONSE:

TM-10 The closure head was treated as either a simply-supported circular plate or a fixed edge circular plate for the preliminary design analysis of the puncture accident, (see Design Report Section 2.7.2). This simple but conservative approach was used to verify the thickness of the closure. It is recognized that this approach does not enable the loads on the closure bolts to be determined. For the final design, a detailed finite element analysis of the closure head will be performed to assess the stresses in the bolts during the puncture accident. This analysis will be similar to that completed for the 30-foot accident and reported in Section 2.7.1 of the Preliminary Design Report.

COMMENT:

TM-11 A review should be made of bridge law/axle groupings to ensure load limits will not be exceeded for routes or states of proposed cask operations.

RESPONSE:

TM-11 The cask transporter design was not included within the scope of the Design Report as it has not advanced to the level of a preliminary design. The transporter design, including maximum axle loadings, will be reviewed to ensure that the bridge formula requirements are satisfied as stipulated in the contract performance specifications and interface guidelines. This information will be presented in the design package for the transporter.

COMMENT:

TM-12 The design of the head should be reviewed for operational functionality features. The number and complexity of penetrations in the closure head combined with the temperatures and radiation levels will make handling by operators difficult. It may be possible to combine functions of some head penetrations to reduce operational steps. The occupational exposures from cask operations at the reactor facilities should be reviewed to ensure ALARA philosophy is a major consideration in the head design.

RESPONSE:

TM-12 The cask functions that are required to be performed by the use of penetrations are draining, drying, gas sampling, purging, seal leak testing, and cooldown. Also, each penetration is required to have redundant closure protection and capability for seal leak testing. Westinghouse has provided the minimum number of penetrations that are necessary to accomplish all the required functions. A single penetration is used for drying, gas sampling, and purging. A second penetration is provided in the closure head for draining the cask cavity. Both these penetrations are required for draining operations and for collecting a flowing gas sample. Each of these penetrations is provided with a leak test port for testing the integrity of the penetration seals. In addition, a separate leak test port is provided for checking the integrity of the closure head seals. There are, therefore, a total of five penetrations in the closure head which is the minimum number required to perform all the specified functions.

An estimate of the occupational exposures from cask operations at reactor facilities will be performed prior to completing the final design, as required by the contract. However, the closure head design will be reviewed early in the final design and modifications made to reduce dose rates to the maximum feasible levels. The use of appropriate temporary radiation shields and remote manual tooling will also be recommended to further reduce occupational exposures consistent with ALARA objectives.

COMMENT:

TM-13 The use of aerospace technology for titanium design allowables and fabrication processes should be actively pursued by the contractor. Highly controlled shop fabrication processes plus the need to make numerous identical parts allows the aerospace industry to routinely use titanium. The infusion of these techniques into the cask fabrication processes will provide the contractor with the added assurance the material qualification program has considered all available data and a solid technical basis exists for the data produced.

RESPONSE:

TM-13 Westinghouse has performed a survey of available material specifications and property data on Grade 9 titanium, including those being used in the aerospace industry. In establishing design allowable stress limits, we are required by Reg. Guide 7.6 to use the ASME Code design rules for Section III, Class 1 components. In the area of fabrication processes, Westinghouse has surveyed all of the major titanium fabricators in the U.S. and selected Cameron Offshore Engineering as a potential fabricator for the half-scale and prototype casks. Cameron has been a supplier of titanium components to the aerospace industry as well as the Navy and the off-shore oil drilling industry and has had the benefit of assimilating the fabrication and NDE processes for titanium that were developed by those industries. Cameron will be performing an extensive weld qualification and NDE qualification program using the state-of-the-art in these technologies for the TITAN cask project. These qualified welding and NDE procedures will be used in the fabrication of the half-scale cask test model and the prototype casks. Figure TM-13.1 presents a letter summarizing the experience and capabilities of Cameron and the overall conclusions of their manufacturability review of the TITAN LWT cask.

In addition to the available expertise of Cameron, Westinghouse will be contacting independent aerospace companies such as Boeing to obtain current information on the fabrication technologies that they have developed and ensure that these are factored into the weld and NDE qualification program and the cask design.

Cameron OFFSHORE ENGINEERING, INC.

580 WESTLAKE PARK BLVD. SUITE 1650 HOUSTON, TEXAS 77079
(713) 939-5400 TELEX NO 775422

November 28, 1989

Westinghouse Electric Corporation
Nuclear Waste Dept. M/L 4-2A
Mall Office Building
200 Mall Circle
Monroeville, PA 15146

Attention: Mr. Bala Nair, Project Manager.
TITAN Cask Project

Dear Mr. Nair:

Subject: Summary of Fabrication Review of TITAN Cask

Cameron Offshore Engineering, Inc. (COE) has completed a detailed fabrication review of the TITAN Legal Weight Truck Cask. The use of titanium alloy Grade 9 offers an excellent combination of strength and toughness while displaying very good fabricability and weldability.

Existing state-of-the-art manufacturing methods can be used to fabricate the titanium Grade 9 cask components. The inner and outer bodies can be extruded to produce a seamless product with exceptional structural integrity, while the closure heads can be die forged. Cameron has a long history of providing titanium forged and extruded components for the aerospace, military, and oil industries, and will draw upon this expertise to produce quality titanium components meeting all nuclear requirements.

The titanium mill products required to produce these forgings can readily be melted by RMI Company, in Niles, Ohio, the largest titanium mill in the world.

The machining of the titanium Grade 9 components will be straight forward, since Grade 9 machines comparable to the most popular aerospace titanium alloys.

The welding of titanium Grade 9 will utilize the same techniques as used in the aerospace industry, where critical titanium structural components are routinely welded using tungsten inert gas methods, with existing procedures very well defined and documented. Nondestructive testing procedures are available to inspect welds to an extremely stringent criteria.

Figure TM-13.1 Experience and Capabilities of Cameron Offshore Engineering, Inc.

Westinghouse Electric Corp. -2- November 28, 1989

Cameron has significant expertise in fabricating large titanium components for the offshore oil and gas industry, including welding of up to 4" thick cross-sections. The titanium stress joint (see enclosed photograph) was fabricated by Cameron Offshore Engineering for use in a critical offshore production system, and utilized the world's largest titanium extrusion. State-of-the-art manufacturing processes were developed for this fabrication, many of which are applicable to this cask fabrication.

COE has visited the manufacturing facility of Cameco, a potential depleted uranium supplier, and is confident that the depleted uranium components can be manufactured and inspected to the required tolerances and acceptance criteria. COE recommends assembling the depleted uranium and performing the closure welding at the DU facility, to eliminate the problems associated with shipping the depleted uranium to the cask fabrication facility.

COE feels the technology for fabricating a titanium cask is presently available within the COE team, and we are committed to providing the resources required for the successful execution of the prototype casks fabrication.

COE looks forward to the chance to work closely with Westinghouse on this challenging project,.

Sincerely,

Kevin Gendron

Kevin Gendron, Manager
Titanium Projects

KG/dr

Enclosure

Figure TM-13.1 Experience and Capabilities of Cameron Offshore Engineering, Inc. (cont'd)

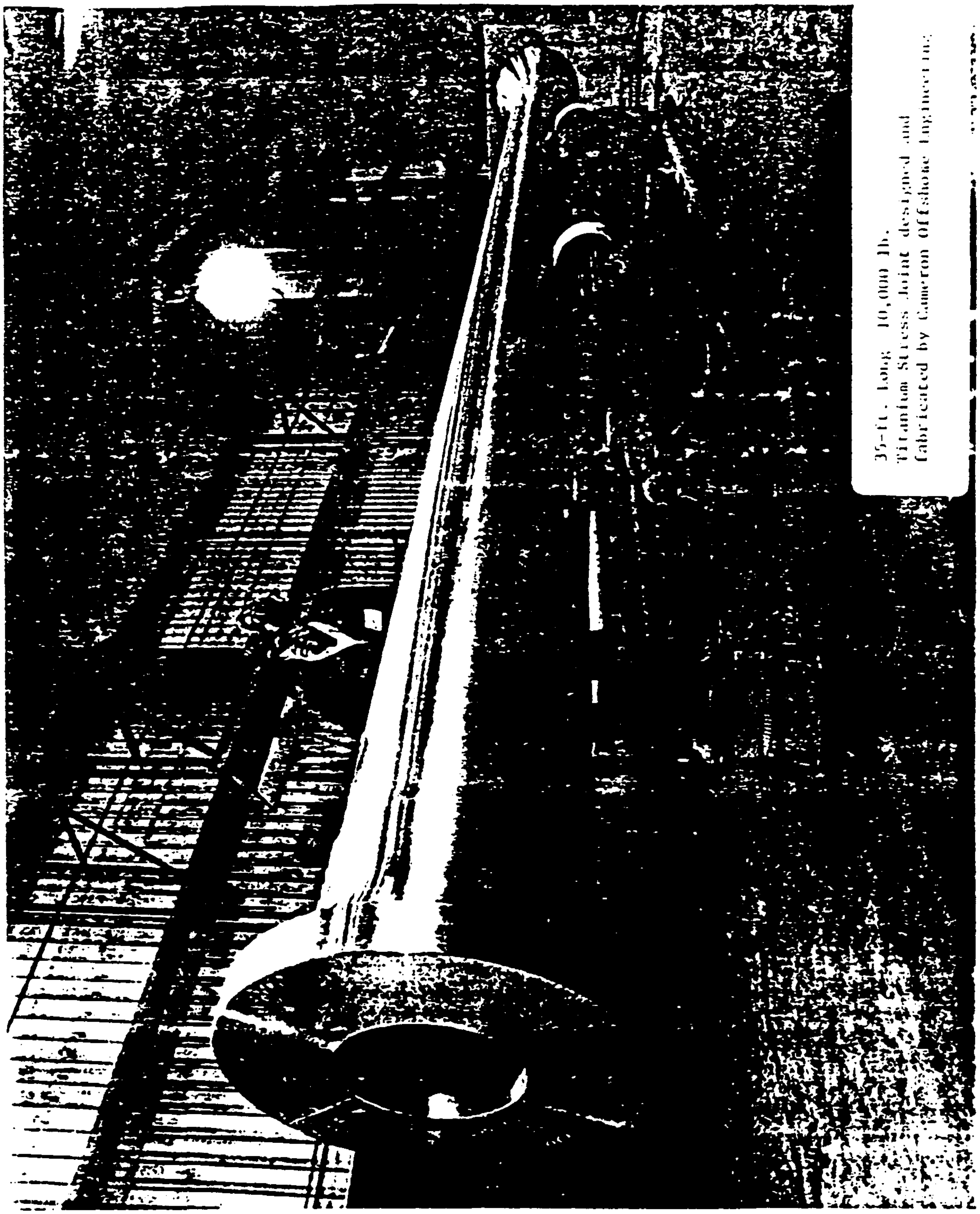


Figure TM-13.1 Experience and Capabilities of Cameron Offshore Engineering, Inc. (cont'd)

COMMENT:

KC-1 The thermal evaluation almost entirely depends on the computer analyses performed with TRUMP. Therefore, it is imperative that all assumptions used in developing the TRUMP models be stated explicitly in order that a judgment can be made about the adequacy of the evaluation. The information included in the report should be sufficient to allow an independent party to recreate the results with TRUMP or another general-purpose heat transfer computer code.

RESPONSE:

KC-1 TRUMP uses finite volume differencing with a resistance formulation. All nodes are characterized by the volume. All connectivities are characterized by the connection length and the corresponding connection area.

The description of the models, the gap conductances and the axial power distribution are given in Section 3.4.1.1, pages 3-4 to 3-8 of the Preliminary Design Report. The thermal properties of the materials are given in Table 3.2-1 of the report. The level of detail will be expanded for the Final Design Report and the SARP. It is possible with the dimensions given in the drawings contained in Section 1.5.4 of the report and the information contained in Section 3 for an independent party to recreate the results with TRUMP or another general purpose heat transfer computer code.

COMMENT:

KC-2 In the 2-D 180° thermal model the ribs have a series of circular cutouts. An explanation of how these are handled in the model is required.

RESPONSE:

KC-2 The circular cutouts are treated by modifying the connection length for those ribs that have cutouts. The thermal resistance was calculated for three regions per rib. The connection length in the region with the holes was modified to yield an equivalent thermal resistance which was then added to the thermal resistances of those sections or regions of the ribs without holes.

In TRUMP, the rib section is modeled with a single node and a modified connection length.

COMMENT:

KC-3 For the R-Z thermal model the report states only that "the aluminum honeycomb impact limiter is explicitly modelled with the correct aluminum content." There is preferential heat transfer in the direction along the axis of the honeycomb cells. A statement about the derivation of effective thermal conductivity in the direction of the axis of the honeycomb and perpendicular to this axis is needed.

RESPONSE:

KC-3 The honeycomb impact limiter is divided into nodes in the longitudinal and transverse directions. Each node has the correct volume content of aluminum. The longitudinal connection is modeled with the connection area equal to the honeycomb's aluminum cross-section area and a connection length equal to the actual length of honeycomb. The connection in the transverse direction is modeled with the connection area equal to the cross section area of honeycomb in the transverse direction and a connection length equal to the actual heat path length which is 4/3 of the physical distance between the nodes. The 4/3 distance factor is the ratio of the aluminum path length to the straight line distance within the honeycomb structure.

COMMENT:

KC-4 **In the R-Z thermal model the assumptions used in developing the model for the fuel and basket need to be explained.**

RESPONSE:

KC-4 The lumped basket and fuel nodes for the R-Z model were developed with the two considerations in mind. They were the maximum temperature of fuel cladding during the transient and the correct transient behavior. The lumped basket nodes (each 1 foot long) represent the basket for that node length and has the stainless steel properties and the volume equal to the volume of 1 foot length of the basket. Each node is thermally connected to the cask inner surface through the gap (the connection area and gap conductivity are the rib cross-section area and the average gap conductivity value from the R-theta model) and to the fuel node through the radiation connection. The connection length was adjusted to yield the basket average temperature which was obtained from the R-theta model steady-state results.

The lumped fuel node was assumed to be all UO_2 and the connection length inside the fuel node was calculated from the cladding thickness. The connection area was the cladding surface area. The radiation form factor between the basket and the fuel nodes was adjusted to yield the maximum cladding temperature calculated for the steady-state condition (from the R-Theta model results).

COMMENT:

KC-5 The assumptions made in the hypothetical accident thermal evaluation may not be conservative. In the steady-state R-Z model for normal conditions the assumption of numerous gap conductances is conservative since it produces the highest internal temperatures. However for the accident conditions the assumption of gap conductances will result in lower internal temperatures than would occur without them. A conservative assumption for the calculation of internal temperatures would be to assume perfect contact during the fire transient, but to include gap conductances in the cool-down transient. Otherwise, the inclusion of these gap conductances needs to be justified.

RESPONSE:

KC-5 The thermal analyses in support of the final design for the normal and hypothetical conditions will be based on conservative assumptions that are appropriate for the determination of the peak temperatures in each component, e.g., closure head seal, structural boundary of the cask, basket, and fuel assembly. The assumption of gaps or perfect contact will be justified in each case in the final design report.

COMMENT:

KC-6 The accident condition modeled in this report is clearly not a worst case since the initial steady state conditions did not include solar radiation and the impact limiter remained essentially intact. Even assuming that the impact limiter remains in place, the possibility of natural convection loops being established within the honeycomb is not considered. This convective loop results from air heated at the outer stainless steel covering flowing through the honeycomb channels to the inner covering, down this surface, and back through the honeycomb channels to the outer stainless steel covering. This is an additional mechanism for transferring heat from the outer stainless steel liner to the cask lid. If the stainless steel covering is breached, this loop would be enhanced.

RESPONSE:

KC-6 The worst case condition for the accident condition thermal analysis is that where the impact limiter(s) are assumed to have separated from the cask. That case will be evaluated for the final design and the results included in the final design report.

COMMENT:

KC-7 Even the additional thermal analyses assuming further damage to the impact limiter which will be included in the final design report do not appear to go far enough. Given the design of the impact limiters, it seems quite conceivable that following an accident the cask lid could be exposed directly to a fire. Drops which produce the worst case from a stress standpoint are not necessarily the worst case from a thermal standpoint. A less severe accident could result in the ripping of the impact limiter from the end of the cask. Unless it can be justified otherwise, the worst case thermal analysis should assume that the end of the cask is exposed directly to the fire, but that the impact limiter blocks radiation to the surroundings during the cool-down transient.

RESPONSE:

KC-7 Thermal analysis of the accident condition will be performed assuming separation of the impact limiter and exposure of the closure head to the fire. The results of the analysis will be included in the final design report.

COMMENT:

WY-1 **It is stated in the text that the W 17x17 fuel assembly was chosen for shielding analysis since it has the largest weight of uranium of any of the assemblies evaluated. Although the discrepancy is small, Table 5.2.1 shows that the W 15x15 assembly has the largest weight of uranium.**

RESPONSE:

WY-1 The W 17x17 fuel assembly was chosen for shielding analysis for its higher Inconel content compared to the W 15x15 assembly even though the 17x17 assembly is slightly lower in uranium weight compared to the 15x15 assembly. The statement in the Preliminary Design Report has been corrected to explain this rationale.

COMMENT:

WY-2 ORIGEN2 does not calculate neutron spectra. The neutron source with the LWR operation spectrum used in the analysis may underestimate neutron dose rates. It is recommended to use the ORIGEN-S code in the final analysis since this code is a significantly updated version of the ORIGEN2 code and it provides neutron spectrum based on measured isotopic spectral data.

RESPONSE:

WY-2 The use of ORIGEN-S in the shielding analysis in support of the final design will be evaluated by Westinghouse. For the preliminary design, it is noted that a 47-neutron group spectrum (with the same group structure as the BUGLE/SAILOR cross sections) was used in the DOT3W analysis to ensure realistic estimates of neutron dose rates.

COMMENT:

WY-3 The radially homogenized spent fuel array used in the analysis was based on the center-to-center spacing for the storage cell (close to the equivalent area-model), leaving a cask cavity between the fuel basket and the cask wall; however, this approach is not conservative since the actual cask has assemblies much closer to the cask wall than the outer surface of the equivalent-area model. Homogenization of the radiation source region(s) over the full cask cavity area is the conservative approach that yields the highest radial dose.

RESPONSE:

WY-3 An R-theta DOT3W analysis at the midplane of the cask cavity was performed to evaluate the effect of cylindricizing the 3 assemblies, while maintaining the same calculated area occupied by the assemblies, versus an explicit model in which the 3-PWR assemblies were modeled just as they would be located during transport. In the explicit model calculation, the difference between the minimum and maximum dose rate values at any point on the circumference of the cask is less than 4%. Approximately the same differences were obtained for the cylindricized model. Homogenization of the source over the entire volume of the cask cavity would provide an overly conservative approach as such homogenization would not include the self-shielding afforded by the 3-PWR assembly configuration in the cask. The approach used in the Preliminary Design Report is therefore sufficiently conservative and accurate and preferable to the suggested over-conservative assumption.

COMMENT:

WY-4 It is not stated in the text what order of discrete angular structures was used in the DOTIIIW calculations. As for angular quadrature sets, at least S_8 or above is recommended for cask calculations. For the Legendre expansion order of scattering distributions, P_e is normally sufficient for neutron shielding problems. For photon transport, a higher expansion order is preferable but is not always used (or available) in practice[c]. Since SAILOR/BUGLE cross section libraries are generated in P_3, it is recommended to use P_3 in the final analysis rather than P_1 as used in the preliminary analysis.

RESPONSE:

WY-4 The order of angular quadrature used in the shielding analysis was inadvertently omitted in the Preliminary Design Report An S_6, 30 angles, angular quadrature set was used in the DOTIIIW calculations as that was judged to be adequate for the preliminary analysis. Consideration will be given to using S_8 for the final design. The order of scattering used in the analyses was P_3, and not P_1 which was a typographical error.

COMMENT:

WY-5 The quality of plots is generally poor. Regions in the geometric plots are not completely described. Irrelevant legends are included and coordinate units are often omitted. Isocontour plots, Figures 5.4-7, 5.4-9, 5.4-15, 5.4-17, 5.4-23, and 5.4-25, need to be improved for better legibility. Limiting the coordinates to smaller distances will help.

RESPONSE:

WY-5 Special attention will be paid in the final design report to the inclusion of plots that provide a greater degree of clarity. The figures mentioned in the above comment were included primarily to show that the surface dose rate profile was relatively flat for most of the cask cylindrical region. Figures 5.4-8 and subsequent even-numbered figures provide information on a much larger scale for those areas where there is significant variation in the dose rates.

COMMENT:

WY-6 The shielding analyst needs to be aware of potential inaccuracies from various sources (methodology, modeling, physics data, etc.) and seek to verify the accuracy of his results or quantify uncertainty on the final dose results. Some conservative assumptions made in the analysis should also be reflected in this regard. Some sensitivity studies done in the past, e.g., References (1) and (2) below may help quantify some uncertainties involved in the analysis.

(1) C. V. Parks, et al., Assessment of Shielding Analysis Methods, Codes, and Data for Spent Fuel Transport/Storage Applications, ORNL/CSD/TM-246, July 1988

(2) M. C. Brady, et al., Comparison of Radiation Spectra from Selected Source-Term Computer Codes, ORNL/CSD/TM-259, April 1989.

RESPONSE:

WY-6 The effect of potential inaccuracies and uncertainties in shielding analysis methodology, modeling, physics data, etc. on the dose rate calculations will be evaluated during the final design to ensure that the assumptions made in the analysis are conservative. The sensitivity studies reported in the referenced documents will be reviewed and appropriate data will be utilized by Westinghouse in that evaluation.

COMMENT:

PB-1 <u>Personnel Barrier</u> (Ref. Fig. 1.4-5, P. 1-54)

- Using 2 lifting hooks, diametrically opposed and apparently at the center of gravity, would not provide stability required for automated installation/removal. A top lifting handle/grip would be better.

- The sling for handling the barrier apparently must be installed manually.

RESPONSE:

PB-1 Two additional lift points will be added to provide the necessary stability required for remote automated handling.

The sling is intended for use only for manual operations at reactor facilities.

COMMENT:

PB-2 Impact LImiters (Ref. Fig. 1.2-23)

- Alignment will be difficult since the limiter has no staged-mating features and the clearance with the cask is small.

- No discussion is given as to how the limiter is moved and what fixture(s) is available on the limiter.

- It is not clear how the limiters are stored/secured on the trailer.

- The bolt fittings are welded to the cask, parallel to the center line, and very near the cask surface. This may cause problems approaching with a tool for the bolting/unbolting, as well as in decontamination and surveying.

- On the front limiter, the tool must also be threaded through two plates of the front restraint cradle to reach two of the four bolts (Fig. 1.3-1). If this is necessary, perhaps moving the bolts further up the cask body would help accessibility.

- The complex shape of the limiters adds to the complexity of modeling the cask surface for swiping operations.

RESPONSE:

PB-2 Alignment of the impact limiters with the cask will be facilitated by providing a slight taper on the ends of the cask that interface with the impact limiter.

A handling concept using a trailer mounted rail and cart was presented at the design review meeting. The limiters remain on the trailer stored on the carts at the ends of the trailer.

The limiter attachment features have to be inside the envelope of the trunnions (i.e., the circle circumscribed around the trunnions) because the impact limiter is sized so that the trunnions will not contact the ground during the hypothetical 30 foot side drop of the cask. The distance from the cask O.D. to the limiter attachment bolt circle is presently 2.65 inches and can be increased to 3.70 inches without exceeding the trunnion envelope. Threading the tool through the support should not be difficult because a tube is welded between the two plates of the support serving as a guide for the tool. In addition, the approximate distance from the trailer deck to the centerline of the bolt is 27.0 inches, providing comfortable room for maneuvering the tool.

The impact limiter shape was established from the standpoint of meeting performance requirements while minimizing weight. Discussions with SNL personnel engaged in the development of remote automated systems for cask operations have confirmed that the impact limiter profile will not present a problem in modeling its geometry for remote swiping operations.

COMMENTS:

PB-3 Tiedowns

- The front tiedowns appear to be easily and quickly installed manually. The stops provide a fixed position which a robot may approach for bale gripping. An improvement from the robotic perspective would include a means to actuate them without changing tools from a gripper to a wrench.

- Detent pins in the rear tiedowns are likely to be difficult to handle remotely, requiring a peg-in-the-hole operation with a tight clearance.

RESPONSE:

PB-3 Westinghouse recommends that the impact wrench used to loosen or tighten the tiedown bolts be provided with a hook to raise and swing away the clamps. Such an approach would be preferable to modifying the tiedown system to be compatible with gripper operation which would provide for a relatively less reliable tiedown system.

The detent pins and engagement holes will be provided with generous lead-in tapers to facilitate installation. The rear tiedown hardware will be checked out as part of the SNL remote automated systems development and any changes that may be required will be incorporated in the final design.

COMMENT:

PB-4 Ports

- For the gas and drain ports, there are 2 plugs each. These are bolted, but details are not given. Also, they apparently must be removed and stored away from the cask. This may require special storage and retrieval operations. (Ref. Fig. 1.2-11)

- Leak check ports appear similar to TRUPACT II ports. A means of automating them should be discussed. Plug manipulation should also be described.

- Remote operation of the plugs seem to have been largely ignored.

RESPONSE:

PB-4 Details of the gas and drain ports have been provided on Sheet 14 of Drawing 1988E43. The designs meet the contractual requirement that each penetration be provided with redundant closure protection. This necessitates the use of two separate plugs that have to be removed and stored. The plugs incorporate features to permit attachment of remote tooling for handling. These features will be checked out during SNL's remote automated systems development and any needed modifications will be incorporated in the final design.

The leak check ports are capable of being remotely operated using the tool design that was presented at the design review meeting. The leak test port and handling tool will also be tested at SNL and modifications made as necessary.

The proposed approach was discussed with SNL personnel involved with the remote automated systems development and found to be acceptable.

COMMENT:

PB-5 Seals

- **It is not clear that the 1/4" elastomeric seals in dovetail grooves are readily inspected and replaced by automated means. Only 1/3 of the surface may be seen without removal, and installation of a floppy seal may require more complex equipment.**

RESPONSE:

PB-5 In response to the above comment, Westinghouse has developed an innovative O-ring seal removal and replacement fixture shown in attached Figure PB-5.1. Using this tool, the seal can be safely removed from the dovetail groove and the entire seal surface as well as the groove surfaces examined for any degradation.

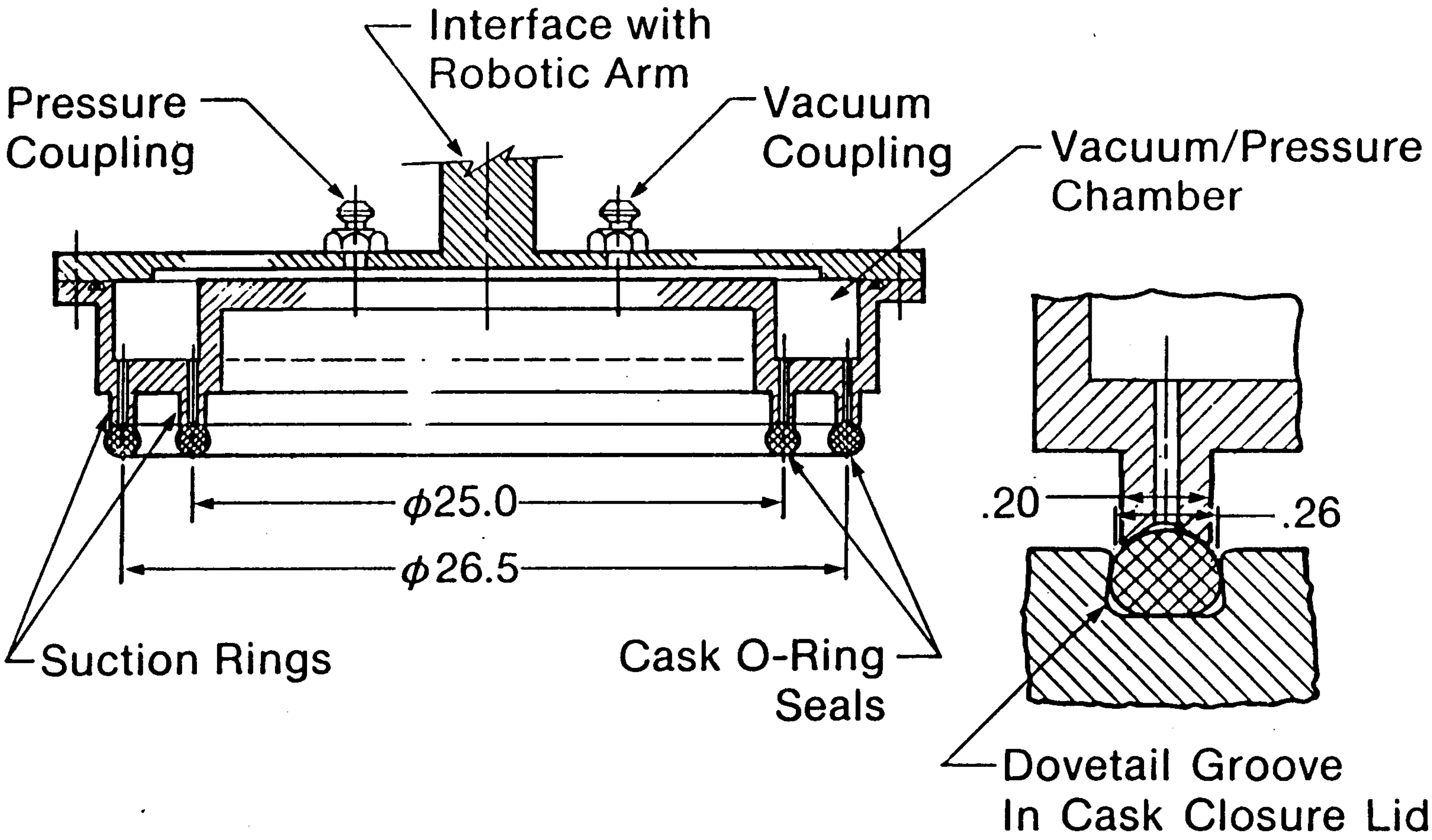


Figure PB-5.1 Seal Removal Tool

COMMENT:

PB-6 Fuel Basket

- It is not clear how the top collar is used for basket handling purposes.

- Handling details for the assembly lead-in fixture are not given. It is not clear whether it is attached or simply laid onto the guidepins.

RESPONSE:

PB-6 The collar at the basket top has an internal ledge which is engaged by an internal grapple. The basket lead-in fixture is placed on top of the cask but is not attached to the cask. It engages the cask lid orientation pins on the cask face.

COMMENT:

PB-7 Trunnions

- The recesses for six bolts in each trunnion may be difficult to survey automatically unless they are covered and sealed. They could also be a contamination problem.

- 2.62 inches may not be sufficient clearance for automated swiping of the trunnions (Ref. Fig. 1.2-4)

- No mention of remote replacement is made.

RESPONSE:

PB-7 The bolt recesses in the trunnions will be filled with RTV to provide a smooth surface that can be surveyed automatically. This approach is currently used in nuclear packagings to facilitate radiation surveys and simplify decontamination.

Westinghouse recommends that specially designed swiping tools that are narrow enough to reach the trunnion surfaces be developed for the From-Reactor casks. This approach is preferable to lengthening trunnions to accommodate existing swiping heads as it creates an undesirable ratcheting effect leading to larger and heavier impact limiters and reduced cask payload.

The trunnions and trunnion sleeves are designed for remote replacement. However, remote replacement of the trunnion sleeves or trunnions is not considered essential as any unplanned replacement operations would be a relatively infrequent event for which hands-on operations are justified and practical.

COMMENT:

PB-8 General

- Visual alignment features are mentioned P. 1-38, #12, and again p. 7-4. However, location and nature of these features is not clear.

- It is not clear how water will be removed from port wells after submersion. If they are blocked off by the assembly lead-in fixture, the method of sealing should be described.

- A means of automating the leak-check apparatus should be discussed.

- No detail of the cask ring seal (hot cell adapter) was given, nor of the handling/attaching/detaching thereof.

RESPONSE:

PB-8 Visual alignment features are shown on Sheet 3 of Drawing 1988E43.

The port wells for the outer closure plugs for the penetrations are provided with drain holes, as shown in Drawing 1988E43, Sheet 15. Additional drain holes will be provided for the plugs and bolt recesses in the final design. It is noted that the fuel assembly lead-in fixture is installed only after the closure lid is removed from the cask and will therefore not block the closure penetration ports.

The tool proposed for handling the leak test penetration ports was presented at the design review. That tool is designed for remote automated operation. The quick-disconnect coupling permits the attachment of the leak test apparatus to the tool using remote automated equipment.

The cask-to-hot cell port adapter was a conceptual design that was presented at the design review. This design will be developed during the final design phase and will incorporate handling features. The tool is designed to be simply placed on the cask and is not attached to it.

COMMENT:

HY-1 The Westinghouse impact limiter design is a combination of cylindrical and truncated cone shapes built up from segmented blocks of honeycomb. Testing is necessary to develop the required force-displacement curves because of the lack of validated analysis techniques. However, it may be difficult to correlate results from a proposed series of static tests performed at 1/4 scale to a series of dynamic tests performed at 1/2 scale because of the geometry of the honeycomb shapes and the as yet undefined strength and crush characteristics of the adhesive bonds. I recommend that the static tests be performed at 1/2 scale to facilitate easier interpretation of the test data with the dynamic results. The adhesive bonds should be evaluated for temperature and dynamic loading.

The dynamic tests would also be useful in evaluating the performance of the impact limiter attachments. The performance of the attachments were not assessed in the design report.

RESPONSE:

HY-1 The impact limiter test plan has been revised to require dynamic testing of 1/4 scale impact limiters instead of static load-deformation tests. The 1/4 scale impact limiter test hardware has been designed to include scaled-down thicknesses of the stainless steel outer covering. This approach will permit extrapolation of the 1/4 scale and 1/2 scale dynamic test results to full scale performance with a high degree of confidence. The dynamic testing of the 1/2 scale models will also evaluate the performance of the impact limiter attachments.

COMMENT:

HY-2 The use of depleted uranium in the cask does not appear to be a certification issue because it is not used as a structural material. However, if the DU imposes additional loads on the structure, its strength must be considered.

RESPONSE:

HY-2 The structural strength of the DU was conservatively ignored in determining the overall structural response of the cask during drop and puncture accidents. Only the mass of the DU was considered. However, in situations where the DU provides a means for transferring loads between the structural shells, its strength was considered in the analysis. An example where the strength of the DU was considered is the punch analysis presented in Section 2.7.2 of the Preliminary Design Report.

COMMENT:

WS-1 The use of the SCANS code for preliminary design is generally an informative activity but DOE has indicated that applicants should not submit SARPs containing SCANS results. This may apply to those submitted to the NRC as well.

RESPONSE:

WS-1 The structural analysis methodology and approach to the dynamic analysis of the cask using the SCANS computer code was discussed with the NRC. The use of SCANS to calculate the overall behavior of the cask during the drop accidents was well received and is acceptable to the NRC. The results from SCANS will provide input to detailed finite element analysis of the cask structural components that will be performed using other computer codes such as WECAN (a Westinghouse Proprietary code).

COMMENT:

WS-2 The Boro-Silicone is placed by a filling operation into the cavity between the outer shell and the middle shell. What assurance can be provided that the material reaches all needed regions of the cavity? The holes through which this material is inserted are closed by plastic caps. Does this lead to a difficulty in decontamination?

RESPONSE:

WS-2 The pouring procedure is qualified by test pours into prototypic models, including regions considered to be potentially difficult to fill. These models are constructed of clear plastic to allow visual observation of the filling operations. The poured material is then destructively examined for presence of voids. Such a qualification procedure has been used in the past and was acceptable to the NRC. The plastic caps will be covered with RTV to facilitate decontamination.

COMMENT:

WS-3 Does the thermal cycling of the depleted uranium gamma shield lead to thermal growth?

RESPONSE:

WS-3 Depleted uranium thermal expansion properties have been well characterized and documented. The average linear coefficient of expansion between 70° to 300°F is 8.5×10^{-6} in/in °F. As the depleted uranium will not be thermally stressed during the cask thermal cycles (between -40°F to 275°F), there should be no thermal ratcheting or growth. Gaps between the DU and the cask titanium shells will accommodate differential thermal expansions and eliminate any thermal stresses in the depleted uranium.

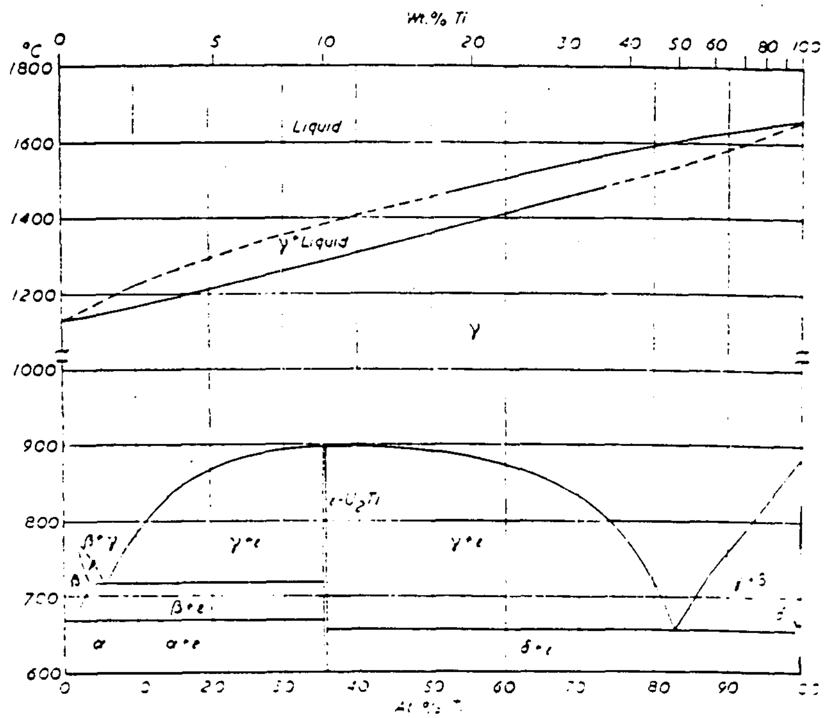
COMMENT:

WS-4 Is the depleted uranium chemically/galvanically compatible with the Titanium alloy shell material?

RESPONSE:

WS-4 The depleted uranium (with 0.2 percent Mo) is completely enclosed by the titanium components of the cask. In addition, provisions are incorporated in the design for inert gas purging of the cavity containing the depleted uranium to ensure high quality of the final closure welds. The purge gas provides assurance that there will be no moisture present in the cavity. Therefore, corrosion will not occur in the depleted uranium. Also, the absence of any moisture ensures that there will be no galvanic corrosion between titanium and the depleted uranium.

Figure WS-4.1 shows that the melting point of the titanium-depleted uranium eutectic is 2000°F which is significantly above the maximum possible metal temperature of 1475°F during the hypothetical fire accident. Titanium provides a significant advantage over stainless steels in this respect as no special protective barrier is required between the titanium and depleted uranium surfaces to prevent formation of a low melting point eutectic.



Ti-U

Figure WS-4.1 Titanium - Depleted Uranium Phase Diagram

COMMENT:

WS-5 Can it be shown that the skid and the cask are not a coupled system as result of being exposed to a highway accident?

RESPONSE:

WS-5 The intermodal transfer skid is used only for securing the LWT cask for transport by rail or barge and will be brought to the intermodal transfer point independently of the cask. Hence, there is no possibility of a highway accident with the cask secured to the skid. It is noted, however, that the tiedown features on the skid are designed for lower transportation accelerations than the cask (and integral tiedown features) to preclude the possibility of the cask and skid acting as a coupled system.

COMMENT:

WS-6 Benchmarking of all computer programs used in the SARP will need to be documented.

RESPONSE:

WS-6 Documentation of bench marking of all computer programs used in the cask evaluations will be provided in the SARP.

COMMENT:

WS-7 The bolting analysis will need to be performed where the gasket and flange stiffness are taken into consideration.

RESPONSE:

WS-7 The preload for the closure bolts is primarily determined by the differential thermal expansion between the titanium closure and the Alloy 718 bolts rather than internal pressure which is small. The calculations to establish closure bolts preload will be included in the final design report. Those calculations will include finite element analysis of the flange and bolting that will account for the stiffness of the closure mating surfaces and the preload in the bolts.

COMMENT:

WS-8 The section on hypothetical puncture indicates that the preparer was not familiar with some of the reference material used to support this section. The equation taken from the cask designers guide was developed for casks with lead as the backing material not graphite as stated in this document. Further applicable data may be found in a two volume study prepared by LLNL and published in 1981 on the topic of puncture. Uranium backing is presented in this report. General information on a development program for uranium shielded casks are contained in reports prepared at the Paducah, Kentucky Gaseous Diffusion Plant.

RESPONSE:

WS-8 Nelm's equation was used for the preliminary design assessment of the titanium shells for the puncture accident. In addition, a detailed finite element analysis of the cask in the vicinity of the punch was completed. Nelm's equation shows a large margin against a shear failure of the 1.25 inch thick middle shell, and the finite element analysis shows that stress limits of Regulatory Guide 7.6 are met. For the final design, more detailed analyses will be completed to show that the cask will not fail during the puncture accident. The recommended reference documents will be reviewed and used as appropriate in the final design structural evaluations.

COMMENT:

WS-9 The approach used to examine the effects of the punch drop need to be further supported. The shear stresses in the vicinity of the punch need to be presented.

RESPONSE:

WS-9 Detailed analyses will be completed in support of the final design to show that the cask will not fail during the puncture accident. In Section 2.7.2 of the Preliminary Design Report, it was shown that a simple shear stress calculation for the shell around the perimeter of the punch using the 20.45 g punch load gives a stress of 46,868 psi which is below the allowable of 47,600 psi. This provides a high degree of confidence that the 1.25 inch thick middle shell can withstand the punch load. Additional analyses during the final design will confirm the adequacy of the shell thickness.

COMMENT:

WS-10 What confirmatory tests are proposed for this design?

RESPONSE:

WS-10 Design verification tests will be performed on a 1/2 scale model of the TITAN cask. These tests will include 30 foot drop tests followed by 40 inch puncture tests. The tests will be performed for drop orientations calculated to produce the maximum structural damage to the cask.

COMMENT:

WS-11 Has the categorization of thermal stresses proposed in this document been accepted by the NRC?

RESPONSE:

WS-11 Regulatory Guide 7.6 states that thermal stresses can be considered as secondary stresses as they are strain-controlled rather than load-controlled. Regulatory Guide 7.6 further limits the magnitude of primary-plus-secondary stresses that occur during all Normal Conditions of transport. The Design Requirements document (NWD-TR-007) categorizes the thermal stresses in the same manner as Regulatory Guide 7.6, and hence should be acceptable to the NRC staff.

COMMENT:

WS-12 USNRC RG 7.6 indicates that fabrication stresses need to be factored into the evaluation for spent fuel casks.

RESPONSE:

WS-12 Regulatory Guides 7.6 and 7.8 require that fabrication stresses be considered in the evaluation of the cask structural components. The Design Requirements document (NWD-TR-007) and the Preliminary Design Report, Tables 2.1.4 and 2.1.6, stipulate that fabrication stresses shall be considered. For the preliminary design, it was judged that fabrication stresses were not significant and hence were not calculated. Fabrication stresses will be calculated and combined with other stresses in the final structural analysis as required by Regulatory Guides 7.6 and 7.8.

COMMENT:

WS-13 Buckling load calculations are needed for the basket under dynamic loads.

RESPONSE:

WS-13 Buckling analysis was not performed for the fuel baskets during the preliminary design as it was judged not to present the critical failure mode. Buckling analysis will be performed and included in the final design report.

COMMENT:

WS-14 Mechanical properties for the 316N SST basket are not presented in Chapter 2.

RESPONSE:

WS-14 The mechanical properties of 316N stainless steel were not presented in Chapter 2 of the Preliminary Design Report because the relevant properties are documented in Section III of the ASME Boiler and Pressure Vessel Code.

COMMENT:

WS-15 What QA Program will be used to assure that the BORAL plates are inserted and remain inserted?

RESPONSE:

WS-15 The Boral plates are installed in recesses machined in the basket cell walls and completely encapsulated and supported in place with welded stainless steel liners. Appropriate hold points will be established in the fabrication sequence to ensure that the Boral plates are installed prior to welding the liners to the basket structure. The use of detectors that employ a neutron source will also be evaluated and specified if found to be worthwhile.

COMMENT:

WS-16 Will the loose fitting basket wear by fretting the inner Titanium alloy shell? If the basket is tight fitting how will it be swapped between PWR and BWR shipments?

RESPONSE:

WS-16 The basket to cask cavity radial clearance is 0.030" which is adequate for interchanging the PWR and BWR baskets. If fretting does occur the SST basket will be the component subject to wear because the titanium cask wall is the harder of the two materials.

COMMENT:

WS-17 What are the temperature limits for the Viton O-rings?

RESPONSE:

WS-17 The Viton O-rings selected for the cask have an operating temperature range of -40°F to 400°F for continuous operation. The material can withstand temperatures of up to 600°F for about 50 hours.

COMMENT:

WS-18 A general chapter on fabrication and acceptance testing is needed in the SARP.

RESPONSE:

WS-18 The SARP will be prepared in accordance with Proposed Revision 2 to Regulatory Guide 7.9, "Standard Format and Content of Part 71 Applications for Approval of Packaging for Radioactive Material" (May, 1986). Chapter 8 of the SARP will include Section 8.1 on Acceptance Testing. A general chapter on fabrication is not required by Regulatory Guide 7.9 but will be added if necessary to address titanium fabrication issues.

COMMENT:

RJ-1 The captive bolts on the lid will retain pool contaminants and will be difficult to clean. Also, I cannot see how the lid bolt recesses will drain when the cask is removed from the pool.

RESPONSE:

RJ-1 The bolt recesses must be flushed with clean water after the lid is raised above the pool surface to wash away the pool contaminants. Drains will be provided in the final design for all bolt recesses.

COMMENT:

RJ-2 **Three guide pins or two unequal length guide pins may result in easier engagement. The guide pin clearance of 0.010" is fairly tight for remote engagement especially with a single-point lifting arrangement.**

RESPONSE:

RJ-2 Two guide pins are provided for orientation of the lid as the lid enters the cask at about the same time as it engages the first pin. The two pins are of unequal length (Dwg. 1988E43, Sh. 8, Detail F-8). The clearance of 0.010" is needed because the lid-to-cask cavity clearance is small (0.0175" nom. radially) and the tight guide pin clearance will prevent lid lock-up on entering the cask. A rigid mast attachment to the grapple will ensure that the single point lifting arrangement will enable remote engagement of the lid with the cask.

COMMENT:

RJ-3 I do not favor the "blind" engagement of the cavity drain line. With a 0.060" basket-to-cavity radial clearance, a 0.005" guide pin-to-lid radial clearance, and a 0.001" dip tube-to-snaptite radial clearance, I believe that proper engagement will be difficult and the potential for tube or tube seal damage is high. Does this arrangement satisfy the interchangeability requirement?

RESPONSE:

RJ-3 The design of the cask cavity drain line and its interface with the closure lid will be carefully reviewed early in the final design to ensure that operational reliability and interchangeability requirements are fully satisfied.

COMMENT:

RJ-4 Does not the drain line penetration create a "hole" in the shielding?

RESPONSE:

RJ-4 The drain line penetration results in a hole in the shielding close to the periphery of the cask I.D. The streaming through this opening is expected to be minimal and will be evaluated during the final design shielding analyses. The use of temporary shields is an option that is available to minimize personnel exposure during cask draining operations and will be recommended if found to be necessary.

COMMENT:

RJ-5 Why are there no controllable valves on the penetrations? The use of Snaptite connectors, even "no spill" models, is regarded by some as far less desirable than a valve.

RJ-5 The Snap-Tite connectors used on the cask penetrations are valves with a quick-disconnect feature. Their major advantage is that such fittings are amenable to remote automated operation. They are provided with leak tight seals and have a leakage performance equal to that of controllable valves. Flow controllability is not a consideration with the cask penetrations. Hence, the quick disconnect fittings are judged to be better for the From-Reactor Cask fleet. In addition, these fittings are readily replaceable in the event of deterioration.

COMMENT:

RJ-6 **How are the DU pieces assembled and stacked? Are they joined end-to-end in any fashion other than just overlapped? Has the Du-Ti long-term compatibility been confirmed?**

RJ-6 The DU is cast and machined in the form of circular rings. Adjacent rings are provided with stepped sections that overlap one another and minimize radiation streaming. The rings are not joined together in any other fashion.

The DU is installed in the annulus which has an inert gas environment. Hence no reactions can take place between the DU and titanium. As there is no moisture present in the annulus there is no possibility of any galvanic corrosion between the two materials. Therefore, there should be no degradation due to any chemical or galvanic interactions between the DU and titanium.

COMMENT:

RJ-7 The basket design, with its many horizontal disk-like standoffs, will accumulate contamination during loading and transport. The drain holes will help, but not eliminate the problem.

RJ-7 Contamination buildup of the basket over a period of time has to be expected and will occur irrespective of the design features that are provided. The fuel assemblies are expected to have a crud buildup that could come loose even as they are inserted or removed. Operating procedures will be recommended for removing such crud from the bottom of the cask after unloading the fuel. In addition, the removable basket design permits the basket to be removed during planned maintenance for extensive decontamination.

COMMENT:

RJ-8 Basket construction will be very difficult (not to mention expensive) due to the many welding operations. Fixturing for proper alignment of all sections will be complex.

RJ-8 An in-depth manufacturability review of the basket design has been initiated by Westinghouse to simplify its fabrication. A simplified basket design is expected to be developed early in the final design phase.

COMMENT:

RJ-9 Is there sufficient radial clearance in the BWR basket to accommodate the standard BWR fuel grapple?

RESPONSE:

RJ-9 Sufficient radial clearance is present in the BWR basket to accommodate the standard BWR fuel grapple.

COMMENT:

RJ-10: The design suggests that the upper end of the fuel assemblies will be nominally 1/4" from the underside of the lid. Is this close proximity a design requirement, and how is assembly irradiation growth factored into spacer selection?

RESPONSE:

RJ-10 Minimization of the distance between the top of the fuel assemblies and the underside of the cask lid is a Westinghouse design objective and not a contractual design requirement. The spacer length will be based on the fuel assembly lengths including irradiation growth.

Locating the fuel assemblies as close as possible to the top of the cask facilitates remote removal of the assemblies and also reduces cask structural loadings due to movement of the assemblies during transport.

COMMENT:

RJ-11 **What is the calculational basis for the trunnion stress summary table (2.5-5). What computational methods were employed?**

RESPONSE:

RJ-11 Stresses in the trunnions and attached shells were obtained using the CYLNOZ computer program. The program uses the equations of Welding Research Council Bulletin 107, "Local Stresses in Spherical and Cylindrical Shells Due to External Loadings."

COMMENT:

RJ-12 Please be more precise in describing how the Dynamic Load Factors were derived.

RESPONSE:

RJ-12 The SCANS computer program uses both quasi-static and dynamic methods to obtain maximum impact responses for the evaluation of the cask during drop accidents. The SCANS quasi-static analysis models the cask as a rigid beam. For dynamic analysis, SCANS treats the cask as a lumped mass elastic beam system. The ratio of loads calculated by the dynamic method to the loads calculated by the quasi-static method is the dynamic amplification factor (due to the flexibility of the cask) or the dynamic load factor.

COMMENT:

RJ-13 How are the DU axial and radial clearances factored into the inner shell and outer shell stress analyses under accident conditions? Does this "rattling around" affect the results?

RESPONSE:

RJ-13 In Section 2.6.2 of the Preliminary Design Report, it was shown that radial gaps are sufficient to prevent differential shrinkage from introducing stresses in the inner titanium shell. The most severe loadings in the cask during the accident event occur when the cask is at its highest temperature. In that case, the maximum radial gap of 0.06 inches will be reduced. Even if a maximum gap of 0.06 inches exists, it will close during the impact event before there is significant deceleration of the cask. Therefore, the maximum decelerations of the DU should be no greater than the rest of the cask structures, and the analyses presented in the report envelopes this effect of closing the gap.

COMMENT:

RJ-14 The SCANS code has been used for much of the cask preliminary design effort. How much confidence does Westinghouse have in this method and what are the plans, if any, for using more sophisticated codes in the final design stage? If more exacting methods are used, to what extent might they affect the cask design?

RESPONSE:

RJ-14 The SCANS code will be used for the final design to calculate the overall cask response during the drop accidents. The program has been thoroughly verified by LLNL and is technically superior to similar programs that have been used in the past to evaluate transportation cask behavior. The loads or accelerations determined from SCANS will be applied statically to finite element models of structural components of the cask which require detailed stress analysis.

There are no plans to run detailed non-linear time-history dynamic analysis of the cask to determine local cask stresses. The approach based on using a relatively simple overall model of the cask for dynamic analysis and performing detailed static analysis of cask components using the dynamic derived loads yields reasonable results and is also acceptable to the NRC.

COMMENT:

RJ-15 The SCANS input implies that the DU is somehow taken into account in a structural sense. Is this true and if so, to what degree?

RESPONSE:

RJ-15 For the SCANS dynamic analysis of the cask, the mass of the DU is taken into account when calculating the response of the cask during free drop accidents. However, the strength of DU is set at a very low value so that it cannot contribute to the overall stiffness of the cask.

COMMENT:

RJ-16 With respect to the limiter tests:

How are the static data going to be related to dynamic properties?

Is the honeycomb "scaled" in the 1/4 scale tests? How about the adhesive? How do these items scale?

The "G" levels are stated as a single value. Is not the shape of the impact limiter force-time curve important in determining the response of the cask or its components? How will the dynamic behavior be factored into the final design analysis?

RESPONSE:

RJ-16 The Phase I impact limiter test program has been revised since the Preliminary Design Report was issued. The testing will include dynamic (20 ft/sec) load testing of the quarter-scale impact limiters. A summary of the revised test program is presented in Tables RJ-16.1 and RJ-16.2. The material test matrix will test straight block specimens staticly and dynamically (20 ft/sec and 44 ft/sec) at various temperatures. Static versus dynamic correlations and temperature dependent characteristics will be obtained from these tests.

The energy absorption capability of the impact limiter is a linear function of crush volume. The volume of the quarter-scale impact limiters is 1/64 the volume of the full-scale impact limiter. The values given in Table RJ-16.2 are 1/64 of the values calculated in the Preliminary Design Report. The adhesive is not scaled for the testing.

The "g" levels given in the Preliminary Design Report are peak values. The load-deflection curve will be generated during the quarter-scale testing and subsequent larger scale tests and will be factored into the final design analyses.

TABLE RJ-16.1

Material Test Matrix

Specimen	Static Test			Dynamic Test (20 ft/sec)			Dynamic Test (44 ft/sec)		
	Room Temp	-20°F	200°F	Room Temp	-20°F	200°F	Room Temp	-20°F	200°F
750 psi Crush Strength Honeycomb	3	3	3	3	3	3	3	3	3
1400 psi Crush Strength Honeycomb	3	3	3	3	3	3	3	3	3

TABLE RJ-16.2

Quarter-Scale Impact Limiter
Test Matrix

Drop Test	Number of Tests	Crush Depth	Estimated Peak Load
Side Drop	2	4.1 in	83,000 lb
17.5° oblique	2	3.4 in	75,000 lb
53.5° oblique	2	3.7 in	130,000 lb
CG-Over-Corner Drop (80.6°)	2	4.2 in	214,000 lb
End Drop	2	2.2 in	246,000 lb

COMMENT:

RJ-17 The report does not contain the analysis of the attachment of the impact limiters to the cask. Historically, these have been vulnerable connections since retention of the limiter after the drop event is a design requirement.

RESPONSE:

RJ-17 The analysis of the impact limiter attachments to the cask was planned to be performed during the early stages of the final design and the results will be presented in the final design report.

For most drop orientations and impact limiter crush depths, the impact limiter crush force is transmitted to the cask body in direct compression, hence, the forces transmitted to the impact limiter attachments will be near zero. For near vertical and near horizontal orientations of the cask and at very modest crush deformations and forces, the center of pressure of the crush force can lie beyond the outer extremities of the cask body and produce a resultant moment on the impact limiter attachments. It is noted that these moments only exist during very modest crush deformations and crush forces and the resulting loads on the attachments should not be large.

COMMENT:

RJ-18 **No leak test methods were presented for determining that the leak tight criterion is met.**

RESPONSE:

RJ-18 The cask design includes provisions for leak testing of the closure lid seals and all penetration seals. These design features will not be affected by the choice of leak test methods or the test gases. In lieu of the considerable amount of development work that is ongoing in the area of leak test methods and criteria, a detailed discussion of leak test methods is more appropriate in the final design report and the Technical Manual.

COMMENT:

RJ-19 There is an assumption on pg. 2-123 that states that all KE is absorbed by the punch. Is it not shared between the cask and the punch?

RESPONSE:

RJ-19 The kinetic energy will be absorbed by both the cask and the punch. However, because of their relative stiffnesses, the punch will absorb most of the kinetic energy and it is conservatively assumed that it absorbs all the energy. This assumption results in a higher predicted punch load.

COMMENT:

RJ-20 The mechanical properties of DU, especially fracture toughness, suggest that shield cracking could occur under drop and/or puncture events. Where is the shielding analysis for cracked DU? Also, is it possible that axial displacement of one shielding piece relative to an adjacent piece can occur? If so, this produces a gap with only one-half of the effective shielding present.

RESPONSE:

RJ-20 The DU can be expected to crack under the drop and puncture accident conditions. The maximum crack dimensions that could develop are limited by the clearances between the DU and the titanium walls. Evaluations will be performed during the final design and included in the final design report to show that the higher dose rates resulting from the cracks will still be within the allowable limits that have been established for the accident conditions.

The final design shielding analysis will also address gaps resulting from axial displacement of one DU shield ring relative to an adjacent piece. The design of the ring overlaps will be modified so that the overlapping interface is conical (with an included angle of about 60°) rather than cylindrical. This ensures that ring separation will create a gap that extends through about 25 percent of the shield thickness rather than one-half.

COMMENT:

RJ-21 Does the shielding source term include the small amount of fissioning that will occur in the DU shield?

RESPONSE:

RJ-21 The shielding source term used for the preliminary design analysis does not include the small amount of fissioning that will occur in the DU shield as it was judged to have a negligible effect on the results.

COMMENT:

RJ-22 How is the molten aluminum honeycomb treated in the thermal model? Since the thermal characteristics of the cask have changed following the accident (e.g., aluminum melt, N-shielding charring), have these phenomena been modeled and what is the resultant post-fire steady-state temperature?

RESPONSE:

RJ-22 The post-fire steady-state temperatures were not determined for the case of melted impact limiters and charred neutron shielding in the Preliminary Design. These evaluations will be performed during the final design and included in the final design report.

COMMENT:

RJ-23 The lid, by virtue of having two major penetrations, plus numerous containment verification test ports, is a very complex structure. The large number of O-rings, captive bolts, and port covers, creates the potential for operational and maintenance problems. The number of crevices around plugs will dramatically increase the contamination problem even if they are free-draining.

RESPONSE:

RJ-23 The cask lid has the minimum number of penetrations (including leak test ports) that are required to perform the required functions of venting, purging, evacuating, drying, and draining. The redundant closures and associated seals are required and are present in existing spent fuel transportation casks and will not introduce any new operational or maintenance problems. All recesses will have provisions for draining and will need to be flushed with clean water to minimize contamination.

A significant difference in the operational requirements for the From-Reactor casks compared to existing casks is the need for compatibility with remote automated equipment. Design considerations for remote operation require the use of captive bolts, minimization of loose parts, use of alignment pins, etc. which increase the difficulty of decontamination. There is therefore a tradeoff between features that enhance decontamination and features that enhance remote operation. Westinghouse will be working closely with SNL who have ongoing development programs in the areas of remote automated systems and evaluation of surface decontamination to select the optimum combination of features that satisfy both these operational requirements.

COMMENT:

RJ-24 The lid bolt torque of 2100 - 2300 ft-lbs will require mechanical assistance to achieve and will add greatly to the cask turnaround time since such values must be achieved by incremental tightening.

RESPONSE:

RJ-24 The torque was determined based on preliminary estimates of the lid-flange region temperature during a fire accident which required a bolt preload sufficient to compensate for the differential thermal expansion between the Alloy 718 bolts and the titanium lid. The preliminary design thermal analysis shows much lower temperatures in that region which will allow a significant reduction of the bolt preload and the torque required to achieve that preload. It is noted that bolt torquing equipment that are compact and have capacities much higher than 3500 ft/lbs. are readily available and desirable to reduce cask turnaround times.

COMMENT:

RJ-25 It is not reasonable to manually remove the personnel barrier. Handholes are shown, no crane lifting points are indicated.

RESPONSE:

RJ-25 The personnel barrier is designed to be lifted in the manual and remote-automated modes. Two lift points are currently shown on the drawing. Two additional lift points will be added to enhance remote-automated lifting.

COMMENT:

RJ-26 How are the impact limiters removed? No lifting points are indicated.

RESPONSE:

RJ-26 The concept of a trailer-mounted rail and cart which allows withdrawal of the limiters from the cask on the cart and storage on the trailer was presented at the preliminary design review meeting. It is intended to investigate other viable concepts during the final design phase. Necessary lifting points for handling the impact limiters will be added during the final design.

COMMENT:

RJ-27 **Do not use red (or any color) paint for reference marks on the cask or basket. This material will deteriorate quickly, especially that on the baskets. Mark the proper orientation with a chemical or mechanical method that is permanent.**

RESPONSE:

RJ-27 The red (or yellow) paint is applied inside "permanent" machined grooves on the lid and cask flange (see Dwg. No. 1988E43, Sheet 3) and is recommended by SNL as a very helpful visual orientation aid for remote automated operation. Alternative chemical marking methods will be evaluated in the final design.

Basket removal is not a routine operation and could be accomplished without any orientation marks as the basket has distinct recognizable features such as the drain pipe that could be used for orientation. Hence paint will not be used on the basket.

COMMENT:

RJ-28 **What data exists or what tests are planned to demonstrate that titanium alloy is not easily contaminated and/or is readily decontaminated. The operational sequence text is silent on decontamination. It implies that a sleeve or wet-well will be (is) required, yet, the cask design makes no provisions for the attachment and sealing of a contamination-prevention system nor is the lifting device sized to accommodate the weight of such a system.**

RESPONSE:

RJ-28 Titanium alloy is included in SNL's test programs to assess the decontamination and weeping characteristics of cask materials. Data from titanium material producers indicate that titanium alloys can be readily decontaminated using commercially available decon agents.

The use of a wet well will be evaluated during final design and appropriate design modifications to the cask and lifting yokes will be made if such a system is to be used.

COMMENT:

RJ-29 Where is the redundant lifting device?

RESPONSE:

RJ-29 The cask is designed with four redundant lifting trunnions at the top for compatibility with a redundant lifting device if required by a utility. Westinghouse has designed a lifting yoke assembly without redundant lifting capability which is not a contractual requirement.

COMMENT:

RJ-30 **How is the lid lifted and placed remotely?**

RESPONSE:

RJ-30 The lid is lifted and installed remotely using an external grappling device which engages with the pintle in the center of the lid.

COMMENT:

RJ-31 **A review of the operational sequence and corresponding times shows the times to be grossly underestimated. The actual turnaround time could be twice that estimated.**

RESPONSE:

RJ-31 The operating times were estimated on the basis of several assumptions. They include the use of trained personnel, availability of equipment and personnel as and when needed in the operating sequence, and the learning experience acquired from large scale planned shipping campaigns from each reactor plant as anticipated for the From-Reactor spent fuel shipment program. Existing data on operational times are derived from sporadic shipments from utilities and reflect the associated inefficiencies and lack of proper planning. Data available to Westinghouse from the West Valley shipments and Virginia Power Surry Nuclear Power Station shipments using the NLI-1/2 cask shows that the operational time estimates, based on the stated assumptions, are realistic and achievable.

COMMENT:

RJ-32 The selection of titanium for the cask shells, although innovative, is certain to cause regulatory problems. Westinghouse has recognized this and is to be complimented on the thorough approach to data acquisition. Nevertheless, the CASTOR cask experience has demonstrated that a significant technical data base is not sufficient to gain acceptance. What are the Ti issues that have come from the Westinghouse - NRC meetings and how are they being addressed?

RESPONSE:

RJ-32 Westinghouse recognizes that the use of a transportation cask structural material without licensing precedent presents a major challenge in obtaining the NRC certification. The key issues that have been raised by the NRC during the three meetings with Westinghouse are: the documentation of the physical and mechanical properties, qualification of Grade 9 titanium as an ASME Code material, fracture toughness requirements, and design and fabrication to the requirements of Section III of the BPVC for Class 1 components. The extensive material property data development program, initiatives to include Grade 9 titanium in Sections II and III of the BPVC, and weld/NDE qualification program (all described in Section 8 of the Preliminary Design Report) are fully responsive to the issues that have been raised.

The CASTOR cask experience with the NRC has been discouraging because the development program intended to demonstrate the adequacy of the ductile cast iron material resulted in structural failure. These failures tended to reinforce the NRC concerns regarding assurance of consistent quality with castings. Variability of material quality with titanium alloys is not a certification issue because the mill product forms that are used in the cask do not include castings. Hence the data base that will be developed and presented to the NRC should be sufficient to demonstrate the adequacy of the material for transportation casks.

It is noted that the NRC has recently accepted new structural materials for transportation casks. An example is the successful certification of the TransNuclear cask that uses borated stainless steel for the fuel basket.

COMMENT:

RJ-33 **The matter of DU behavior has not been addressed. Although it is not being given any structural credit it is nevertheless present in the system and under certain conditions, such as longitudinal bending, participates in load transfer between shells.**

RESPONSE:

RJ-33 The DU is installed in the cask in the form of cast and machined rings each of which is about two feet in length. These rings will not contribute any strength in longitudinal bending of the cask, though its mass is considered in the evaluation of the stresses. The DU will transmit compressive load between the titanium shells. The punch analysis presented in Section 2.7.2 of the Preliminary Design Report considers this effect.

COMMENT:

RJ-34 No discussion of the tractor/trailer was presented. If one uses a loaded cask system weight of 54,000 pounds and a conventional tractor weight of 15,000 to 20,000 pounds, this leaves only 6,000 to 11,000 pounds for the trailer. Tractor/trailer design will be challenging.

RESPONSE:

RJ-34 The design of the cask transporter was not included in the Preliminary Design Report as it had not advanced to the level of a preliminary design. The maximum loaded cask weight of 54,000 lb. was predicated on an allocation of 16,000 lbs. for the tractor and 10,000 lb. for the transporter and cask tiedown system. A commercial tractor weighing 16,000 lbs. is an achievable goal as indicated by a manufacturer (see Figure RJ-34.1). Similarly, the trailer is envisioned to be an engineered structure and not a commercial item. The TRUPACT-II trailer weighs less than 10,000 lbs. and provides an example of how an adequate engineered trailer can be built weighing less than the allocated limit.

KENWORTH

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A DIVISION OF PACCAR

April 12, 1988

Mr. George V. B. Hall
Principal Engineer
Westinghouse Electric Corporation
Nuclear Waste Dept. - Mail Stop 4-2A
P.O. Box 3912
Pittsburg, PA 15230

Dear Mr. Hall:

Your call today was very interesting and your requirement to haul a payload of 55,000 pounds while remaining with current size, weight and bridge regulations seems quite possible to meet. If we look ten years out, it seems especially possible in light of weight reductions I expect to see in place then.

Today, some of our customers are hauling equivalent loads of bulk commodities but without sleeper equipped trucks. However, if special trailers are built for the containers you mentioned (casks), some trailer weight reduction could be expected which might make up for the weight of a sleeper compartment.

As I said on the phone, it is easy to specify a sleeper equipped Kenworth (T600A Model for instance) which will weigh 16,500 pounds road ready today. With some effort, we might lower that weight, but without definite specifications to work on, an estimate would be pure speculation.

I hope this provides enough information to be of use to you. We at Kenworth would be glad to provide additional information if you need it.

Sincerely,

Larry Orr
Chief Engineer

Figure RJ-34.1 Kenworth Tractor Weight Estimate

COMMENT:

RJ-35 The actual fabrication of the cask, including cost, should be thoroughly reviewed. There are several components, such as the basket and the lid, that will be difficult and/or costly to construct. Perhaps there are ways to simplify the design while preserving the functions.

RESPONSE:

RJ-35 An in-depth manufacturability review of the cask components and baskets has been initiated by Westinghouse to simplify the designs and reduce fabrication and tooling costs. The results of the review will be factored into the designs early in the final design phase.

COMMENT:

RT-1 Drawing 1988E42, Sheet 1, Note 4 - The acceptance criteria for the ultrasonic examinations should be Article NB-5330.

RESPONSE:

RT-1 The drawing note will be changed during final design to incorporate the comment.

COMMENT:

RT-2 **Drawing 1988E42, Sheets 2 through 4 - All final welds should be inspected using liquid penetrant examination per ASME B&PV Section V, with Article NB-5350 acceptance criteria.**

RESPONSE:

RT-2 Drawing Note 5 will be changed during final design to specify liquid penetrant inspection on all final welds.

COMMENT:

RT-3 **Drawing 1988E43, Sheet 3, Zone C2 - Inspections should include Note 7 for liquid penetrant as well as Note 5 for radiographic examination.**

RESPONSE:

RT-3 The drawing will be changed during final design to add Note 7 for liquid penetrant examination.

COMMENT:

RT-4 Drawing 1988E43, Sheet 4, Zone C6 - The full penetration weld should include Note 7 for liquid penetrant examination.

RESPONSE:

RT-4 The drawing will be changed during final design to add Note 7 for liquid penetrant examination.

COMMENT:

RT-5 **Drawing 1988E43, Sheets 5 & 6 - Should the structural attachment welds for the trunnions be full penetration welds (ref. NB-4433) since the fillet configuration does not meet Figure NB-4427.1?**

RESPONSE:

RT-5 The drawing will be changed during final design to incorporate full penetration welds for the trunnion housing-to-cask shell attachment.

COMMENT:

RT-6 **Drawing 19988E43, Sheet 11 - Should the fillet welds be inspected using the liquid penetrant method specified in ASME B&PV Section V with acceptance criteria per NF-5350 or NB-5350 [examples zones D3 & G1]?**

RESPONSE:

RT-6 The drawing will be changed during final design to add Note 7 requiring liquid penetrant examination for all welds.

COMMENT:

RT-7 **Drawing 1988E43, Sheet 12 - Should the fillet welds be inspected using the liquid penetrant method specified in ASME B&PV Section V with acceptance criteria per NF-5350 or NB-5350?**

RESPONSE:

RT-7 The drawing will be changed during final design to add Note 7 requiring liquid penetrant examination for all welds.

COMMENT:

RT-8 Drawing 1988E43, Sheet 13, Zone G6 - Need to specify a weld symbol and inspection requirement.

RESPONSE:

RT-8 This weld was specified in Detail F-8, Sheet 10. Inspection requirements will be covered by Note 7 during final design.

COMMENT:

RT-9 **Drawing 1988E44 - All welds should be liquid penetrant examined following ASME Section V, using Article NB-5350 as acceptance criteria.**

RESPONSE:

RT-9 The drawing will be changed during final design to specify liquid penetrant examination for all final welds.

COMMENT:

RT-10 Drawing 1988E46 - Need a note to specify the welding procedure/welder qualification requirements and also a note to state the inspection requirements with an acceptance criteria.

RESPONSE:

RT-10 The following Notes will be added during final design to Drawing 1988E46:

6. "All welding procedures and welders shall be qualified per the ASME B&PV Code, Section III, Subsection NF."

7. "Liquid penetrant examination shall be in accordance with ASME B&PV Code, Section V, Article 6. Acceptance Standards of ASME B&PV Code, Section III, Article NF-5350 shall apply."

8. "Visual examination shall be in accordance with ASME B&PV Code, Section V, Article 9. Acceptance standards of ASME B&PV Code, Section III, Article NF-5360 shall apply."

COMMENT:

RT-11 Drawing 1988E47 - The load bearing surfaces and welds should be liquid penetrant examined after the load test as well as during fabrication.

RESPONSE:

RT-11 Drawing 1988E47, Sheet 1 of 6, will be revised during final design to include liquid penetrant examination of the load bearing surfaces both after fabrication and load testing. Notes 4, 5 and 8 specify weld inspections after fabrication and load testing.

COMMENT:

RT-12 Drawing 1988E51, Sheet 1 Note 5 - Will undercut and concavity be allowed during visual inspections?

RESPONSE:

RT-12 Drawing 1988E51, Sheet 1, Note 5 will be changed during final design as follows:

"Visual examination shall be in accordance with ASME B&PV Code, Section V, Article 9. Acceptance standards of ASME B&PV Code, Section III, Article NF-5360 shall apply."

COMMENT:

RT-13 Drawing 1988E51, Sheet 1 - Is visual inspection enough for the load bearing welds? What standard was used to determine the required inspection and acceptance criteria?

RESPONSE:

RT-13 Drawing 1988E51, Sheet 1 will be revised during final design to add the following note:

"Liquid penetrant examination shall be in accordance with ASME B&PV Code, Section V, Article 6. Acceptance standards of ASME B&PV Code, Section III, Article NF-5350 shall apply."

Note 5 will be revised to specify the same requirement for visual examination as defined in the response to Comment RT-12.

COMMENT:

RT-14 Drawing 1988E52, Sheet 1 - Is visual inspection enough for the load bearing welds? What standard was used to determine the required inspection and acceptance criteria?

RESPONSE:

RT-14 The non-destructive examination requirements specified for the Support System load bearing welds were reviewed and it was determined that liquid penetrant examination will be desirable in addition to visual examination. A note will be added during final design to Drawing 1988E52, Sheet 1 as follows:

"Liquid penetrant examination shall be in accordance with ASME B&PV Code, Section V, Article 6. Acceptance standards of ASME B&PV Code, Section III, Article NF-5350 shall apply."

Note 5 will be revised at that time to specify the same requirement for visual examination as defined in the response to Comment RT-12.

COMMENT:

HD-1 Subsection 6.1, Page 6-2: The format of Table 6.1-1 should be the same as Table 6-2 in Reg. Guide 7.9.

RESPONSE:

HD-1 The format of Table 6.1-1 is the same as Table 6-2 in Reg. Guide 7.9. The normal and accident parameters in Table 6.1-1 have been grouped together because they are assumed to be identical as discussed in the Preliminary Design Report.

COMMENT:

HD-2 Subsection 6.2 - The description includes consolidated fuel rods; however, no data or information are provided describing this loading. If consolidated fuels are to be shipped, such data and information should be included.

RESPONSE:

HD-2 The cask is optimized for transporting intact fuel assemblies. The criticality analyses performed in support of the Preliminary Design evaluated only intact assemblies. The capability to transport consolidated rods will be achieved through the use of special baskets if required. In general, consolidated rods are less reactive than intact assemblies, but when consolidated rod and canister loading details are developed for the authorized contents of the cask, criticality evaluations will be performed.

COMMENT:

HD-3 Table 6.2-1, Pages 6-4 through 6-10: The source of this data should be referenced for each fuel assembly and additional data should be provided (e.g., active fuel length, pin lattice geometry, etc.). Drawings should be provided.

RESPONSE:

HD-3 The source of the data provided in Table 6.2-1 is DOE/RW-0184 (Reference 5.5.1). Complete data for each fuel assembly will be included in the SARP.

COMMENT:

HD-4 **Subsection 6.3, first paragraph, Page 6-3: It states the "fuel basket designs are modeled exactly in the calculational models." To what extent does the exact modeling apply? Does this mean the fuel elements were modeled exactly?**

RESPONSE:

HD-4 The fuel basket wall materials and dimensions were modeled explicitly in the calculational model and were not homogenized with the water or the fuel elements in the basket. The PWR fuel rods were modeled explicitly while the BWR fuel rods were homogenized with the water surrounding the fuel rods. This was done for calculational convenience. A bias was added to the K_{eff} calculation to account for this modeling difference (see Page 6-20 of the Preliminary Design Report).

COMMENT:

HD-5 Subsection 6.3, top paragraph, Page 6-15: It is not clear why a "rectangular box" was modeled instead of the circular cask design. Does this change introduce conservatism or non-conservatism? The sentence states the "inside volume and material volumes" are the same; does this also apply to the materials external to the cavity?

RESPONSE:

HD-5 A rectangular shape was used to model the cask because of the ease in developing this type of model. KENO calculations show that there is no statistically significant variation in results when compared with a circular cask model. All material volumes in the cask cavity and outside the cavity were maintained.

COMMENT:

HD-6 Page 6-18, Assumption #1: Provide the calculations, or reference the appropriate document, which substantiate the claim that the 17x17 OFA and the GE 7x7 fuel assemblies are worst case models. Is this also true for consolidated fuel configurations?

RESPONSE:

HD-6 Calculations were performed as part of the preliminary design work to establish that the 17x17 OFA and the GE 7x7 fuel assemblies are the worst cases for the criticality evaluation. Results from these calculations will be included in the final design report.

The 17x17 OFA will not be the worst case for consolidated fuel configurations. Consolidated fuel rods were not evaluated because the cask is required to be optimized for intact assemblies. However, consolidated fuel will be significantly less reactive than the fuel in intact assemblies.

COMMENT:

HD-7 Page 6-18, assumption #4: Calculations or justification should be provided that less than full density water does not result in a higher reactivity.

RESPONSE:

HD-7 Calculations were made following the design review to show that less than full density water will not result in a higher reactivity. The results are shown in Figure HD-7.1.

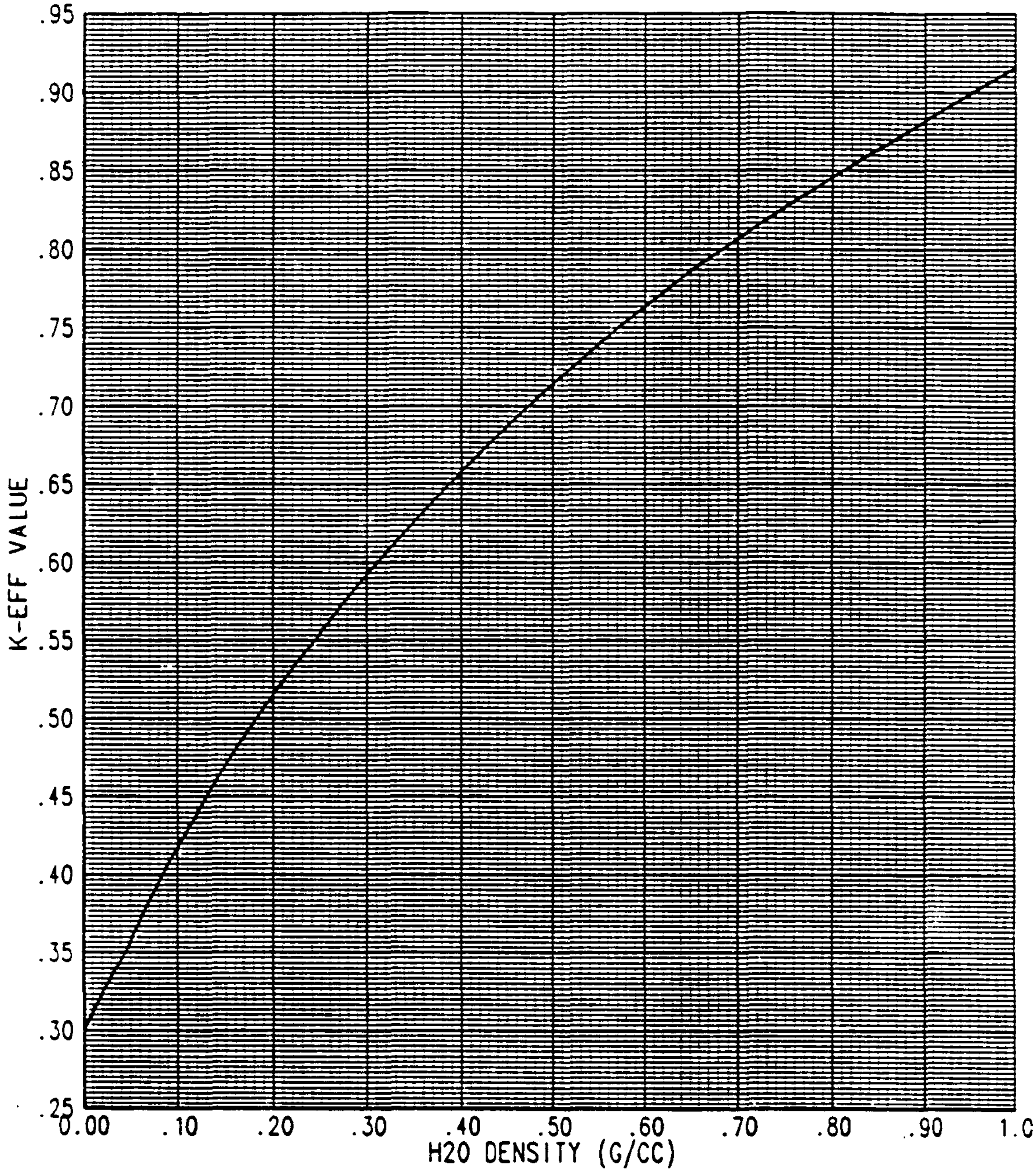

Figure HD-7.1 TITAN Cask Optimum Moderation Check

COMMENT:

HD-8 Page 6-18, Assumption #5: It is not clear how "no credit is taken for any spacer grids or spacer sleeves." Will these be replaced with water in the model, and what is the effect of omitting these materials?

RESPONSE:

HD-8 The fuel assembly grids and sleeves were not included in the models, but their volume was replaced with water. Since these components are composed of stainless steel, Inconel and Zircaloy, ignoring them in the model was a conservative assumption.

COMMENT:

HD-9 Page 6-18, assumption #6: The effects of a water reflector on a single cask, and of a varying water density between a finite array of casks should be evaluated to support this assumption.

RESPONSE:

HD-9 The cask shell was modeled in a rectangular shape. The infinite array of casks has the wall of one cask against the wall of another. As a result, the model of an infinite array of casks has no water between the casks because reflective boundary conditions were placed on the outside walls of the cask.

COMMENT:

HD-10 Page 6-18, last paragraph: Provide supporting evidence to substantiate the claim that reduced spacing increases reactivity, and that asymmetric positioning in the center is more reactive than fuel shifted to one side of the cask.

RESPONSE:

HD-10 Detailed calculations will be included in the final design report to show that reduced spacing increases reactivity, and that asymmetric positioning in the center is more reactive than fuel shifted to one side of the cask.

COMMENT:

HD-11 Page 6-19, first sentence: It is not clear what this statement means. Is k_{eff} for normal conditions the same as for accident conditions?

RESPONSE:

HD-11 The assumptions used to generate the model were the same for the Normal Conditions and the worst case (Accident) Conditions. Therefore the Accident K_{eff} is the same as the Normal K_{eff}.

COMMENT:

HD-12 Page 6-19, Conditions 1, 2, 3, and 4: Provide supporting evidence to substantiate these claims.

RESPONSE:

HD-12 Evidence supporting the claim that the maximum cask K_{eff} under Accident Conditions is equal to the maximum cask K_{eff} under Normal Conditions will be provided in the final design report.

COMMENTS:

HD-13 Subsection 6.4.3, Page 6-19: The results of the calculations performed for this evaluation should be presented.

RESPONSE:

HD-13 The results for these calculations are provided below the equation on Page 6-20 of the Preliminary Design Report.

COMMENT:

HD-14 Subsection 6.4.3, Page 6-20: Provide reference or justification for the statistical validity of the equation to develop maximum k_{eff}.

RESPONSE:

HD-14 This equation adds the bias terms and the root mean square of the uncertainty terms to the calculated K_{eff}. The statistical validity of the equation will be included in the final design report.

COMMENT:

HD-15 Subsection 6.4.3, Page 6-20, last paragraph: The results of flux trap analyses should be presented, or referenced.

RESPONSE:

HD-15 Flux trap analysis were not performed since the preliminary design calculations confirm that flux traps are not needed in the cask basket to meet the K_{eff} limit.

COMMENT:

HD-16 Subsection 6.5: Address the applicability of the bias from the critical experiments with 2.46% enriched uranium to those calculations with 4.5% enriched uranium. The inclusion of the 93.2% enriched uranium experiments in determining the bias is inappropriate.

RESPONSE:

HD-16 The 93.2 w/o enriched criticals are included in the benchmarking to validate KENO for the very low water density problems that would be used in "optimum moderation" studies (water density below 0.3 gm/cc). The final analysis will include benchmark criticals enriched to 4.3 w/o U235. No significant change in the reactivity bias is expected.

COMMENT:

HD-17 General comment to Section 6: The format of Section 6 followed that required by Reg. Guide 7.9; however, much of the data and information required by Reg. Guide 7.9 was not included. For example:

(a) Insufficient information was provided to perform confirmatory calculations.

(b) KENO input was not provided and modeling adequacy could not be confirmed or checked.

(c) The results of the calculations performed were not included.

(d) Discussion and comparison of normal conditions vs accident conditions are not adequately presented.

(e) The discussion on calculational method should be expanded to more clearly describe the cross section processing and the details of the KENO input (e.g., number of neutrons, number of neutron histories, boundary conditions, etc.).

RESPONSE:

HD-17 As stated in the comment, the format of Reg. Guide 7.9 was followed; however, because only a preliminary analysis was performed, detailed analyses and reporting were not implemented. Sufficient information was, however, provided in the Preliminary Design Report such that an independent model could be developed if necessary. The final design report will include detailed discussions on the complete analysis and address the above comments.

COMMENT:

HS-1 Throughout the report, in the drawings and in the attachments reference is made to welding of titanium and other materials. Specifically, the drawings should be include the following for <u>all</u> welds:

- o The type of welding to be used (GTAW, SMAW, etc.),
- o The filler material to be used and a required specification for the filler material, and
- o A welding specification to accomplish the welding.

The requirements are important for all welds and are especially important for titanium welds.

RESPONSE:

HS-1 The drawings will be revised during final design to specify the type of welding and the filler wire to be used for all welds. A note will be added to specify that all welds shall be performed in accordance with specifications (prepared by the cask fabricator) that have been approved by Westinghouse.

It is currently planned to develop detailed specifications and procedures for titanium welding as part of an extensive weld and NDE process qualification program that has been initiated by Westinghouse. These specifications and procedures will be included in the Safety Analysis Report.

COMMENT:

HS-2 In the report, it is stated that it is assumed that the allowable stress for the titanium welds will be the same as for the base metal (pp. 2-121). The allowable stress for the aluminum welds is specified to be one half of the stress for the base material (pp. 2-253). This appears to be inconsistent. Recommend that properties for material weld metal and heat affected zones be defined for <u>each</u> filler material and base material being used. The specific filler material being used should be defined, especially for the titanium components which should use the ELI (extra low interstitial) grades.

It also appears that preparation of a detailed welding specification for the titanium welds and other welds would be appropriate.

RESPONSE:

HS-2 Welding processes for the titanium will be limited to GTAW or plasma arc. Test data for titanium welds made using these processes show that there is no decrease in strength over that of the base metal. On the other hand, welding greatly decreased the strength of aluminum, and the "Specifications for Aluminum Structures" document published by the Aluminum Association shows that allowables for welded members are about one-half of those for base metals. An extensive Grade 9 titanium material property data development program has been initiated by Westinghouse. Mechanical and physical property data will be obtained for both base metal and welds. These will be used in the final design analysis.

Detailed welding specifications for titanium welds will be developed as part of the titanium welding and NDE process qualification program. Only procedures qualified in accordance with Section IX of the ASME B&PV Code will be used. The welding procedures will identify specific filler materials, thicknesses of material for which the procedures have been qualified, and NDE requirements.

COMMENT:

HS-3 The report states that the aluminum honeycomb components will be sealed with stainless steel sheets (pp. 2-41). It appears to me that the sealing method should be defined and qualified to ensure that the seal is maintained over the 25-year life of the components. Atmospheric corrosion and moisture levels and chloride concentrations caused by salted road conditions (p. 36, requirements) could destroy the 0.003- and 0.004-inch-thick aluminum honeycomb core in a very short period of time.

RESPONSE:

HS-3 The honeycomb material is contained inside a SST welded housing. The welds will be seal welds and will be inspected for soundness using visual and liquid penetrant examination methods. All honeycomb materials are provided with corrosion protection and qualified to ASTM B-117.

The impact limiters will be visually inspected for damage after each shipment. In addition, the welds will be periodically inspected during scheduled maintenance operations using visual and liquid penetrant methods. If a damage is detected, the limiter will be repaired by removing the damaged housing area and replacing the affected honeycomb section (if signs of corrosion are visible) and then replacing the removed section of the housing.

COMMENT:

HS-4 **Page 7, Par. 1.3C requires an interior surface finish that comes in contact with the fuel to be greater than or equal to a 125 AA microinch surface. Page 12, Par. 1.11A requires a minimum interior cask surface finish of 32 microinches and an exterior surface finish of 16 microinches or better. These requirements appear inconsistent. Recommend that the surface finish that comes in contact with the fuel and interior cask surfaces be a 32 microinch or better.**

RESPONSE:

HS-4 The requirements referred to are cited verbatim from the contractual requirements. The 125 AA microinch surface finish pertains to the fuel basket interior. The 32 microinch surface finish for the cask interior surface and the 16 microinch finish for the case exterior surface are Westinghouse requirements that are based on practical surface finishes that can be achieved at reasonable cost.

COMMENT:

HS-5 Page 8, Par. 1.3D requires component size be designed with sufficient clearance with allowances for fuel bowing, twisting, bulging, etc. Define clearance requirements in inches.

RESPONSE:

HS-5 Page 8, Par. 1.3D of the Design Requirements Document cites verbatim a contractual requirement. The clearances provided in the basket are adequate for accommodating the vast majority of the spent fuel assemblies that are in storage or anticipated to be in storage at the reactor sites. Fuel assemblies that have excessive bowing, twisting or bulging will have to be shipped specially as non-standard or failed fuel. Westinghouse has elected to use this design approach to provide a cask system with the maximum payload that will entail the minimum life cycle costs.

COMMENT:

HS-6 **Page 10, Par. 1.7D requires cask materials be chosen to preclude unacceptable corrosion. Define what unacceptable corrosion is.**

RESPONSE:

HS-6 This is a contractual requirement has been repeated verbatim in the design requirements document.

COMMENT:

HS-7 Page 12, Par. 1.11C requires cask painting/coating be compatible with facility requirements. Define facility requirements.

RESPONSE:

HS-7 Par. 1.11C of the Design Requirements Document cites verbatim a contractual requirement.

COMMENT:

HS-8 Page 13, Par. 1.11F requires that all materials be compatible with standard decontamination washing solutions. Define the standard decontamination washing solutions.

RESPONSE:

HS-8 Page 13, Par. 1.11F of the Design Requirements Document cites verbatim a contractual requirement.

COMMENT:

HS-9 **Page 13 states a requirement shall be met "where practical." This is not a requirement, either delete the words "where practical" or delete the paragraph.**

RESPONSE:

HS-9 Page 13 of the Design Requirements Document cites verbatim a contractual requirement.

COMMENT:

HS-10 **Page 14, Par. 1.13A states a requirement shall be met "where appropriate." This is not a requirement, either delete the words, "where appropriate" or delete the requirement.**

RESPONSE:

HS-10 Page 14, Par. 1.13A of the Design Requirements Document cites verbatim a contractual requirement.

COMMENT:

HS-11 Page 19: Is the GWV weight limit 40 tons in all states or are some states less than 40 tons?

RESPONSE:

HS-11 The Gross Vehicle Weight limit is 40 tons on interstate highways for all the 48 contiguous states.

COMMENT:

HS-12 Page 20, Par. 2.46 requires sufficient clearance be provided. Define the sufficient clearance.

RESPONSE:

HS-12 Page 20, Par. 2.46 cites verbatim a contractual requirement.

COMMENT:

HS-13 Page 20, Par. 2.4L requires the design to optimize the number of cycles. The words "to optimize" do not adequately define a requirement. Define the minimum number of acceptable cycles.

RESPONSE:

HS-13 Page 20, Para. 2.4L cites verbatim a contractual requirement.

COMMENT:

HS-14 Page 32, Load combination of service load. The values of -20°F and the value of -40°F on Page 23 are confusing to me. Shouldn't they be the same?

RESPONSE:

HS-14 These values are taken from 10 CFR 71 and are correct as stated. The -40°F temperature is specified for the "cold" condition only. The -20°F temperature is used for all other cases.

COMMENT:

HS-15 **Page 36, Par. 3.1.3, requires the cask exterior surfaces be capable of withstanding the effects of moisture levels and chloride concentrations caused by salted road conditions. The moisture actual levels and chloride concentrations should be defined.**

RESPONSE:

HS-15 Page 36, Par. 3.1.3 cites verbatim a contractual requirement. Moisture levels can be 0-100%. The chloride concentrations for design purposes can be fully saturated (60% solution CaCl). These will be added to the Design Requirements Document.

COMMENT:

HS-16 Page 38, Par. 3.2.1 and throughout the Requirements and Design Review documents allows for the use of industry recognized standards. Define specific industry recognized standards that are acceptable throughout the Requirements and Design Review documents.

RESPONSE:

HS-16 Page 38, Par. 3.2.1 cites verbatim a contractual requirement. Industry recognized standards are defined wherever they have been used or cited in the text of the Preliminary Design Report. For example, the AISC Manual for Steel Construction and the Aluminum Construction Manual are cited on page 2-26 of the report for design limits for the cask support and tiedown systems.

COMMENT:

HS-17 Page 41 Par. (e) does not make sense. The words "shall be" should be used before the word "constructed."

RESPONSE:

HS-17 Page 41, Par. (e) reads as follows: "Those containment boundary components constructed from materials that do not undergo a brittle-to-ductile transition with increasing temperature, such as austenitic stainless steel or titanium alloys, will not be limited with respect to use at low temperature."

Sentence is correct as stated.

COMMENT:

HS-18 **Page 63, Par. 1.1B and Par. 1.2. Delete the words, "if necessary" or delete the paragraph.**

RESPONSE:

HS-18 Page 63, Par. 1.1B and Par. 1.2 cite verbatim the contractual requirements.

COMMENT:

HS-19 Page 70, Par. 1.1. Change the word "should" to "shall."

RESPONSE:

HS-19 Page 70, Par. 1.1 cites verbatim a contractual requirement.

COMMENT:

HS-20 Page 70, Par. 1.3D. Define "the low as practical values" required.

RESPONSE:

HS-20 Page 70, Par. 1.3D cites verbatim a contractual requirement.

COMMENT:

HS-21 **Page 71, Par. 1.36. Delete the words "where practical" or delete the paragraph or define where practical.**

RESPONSE:

HS-21 Page 71, Par. 1.36 cites verbatim a contractual requirement.

COMMENT:

HS-22 Page 71, Par. 1.4. Delete the words "this may" or delete the paragraph.

RESPONSE:

HS-22 Page 71, Par. 1.4 cites verbatim a contractual requirement.

COMMENT:

RP-1 The principal concerns with the closure design relate to the size of the vent and drain lines, the size, number, and operational complexity of the ports, and the guide pins.

The vent and drain line sizes are not clearly specified. The vent line should be at least .5 inches and the drain line at least .75 inches for efficient flushing and draining of the cask. Small lines have relatively low flow to begin with, and can become restricted by foreign material. The mate-up of the lid with the drain line should be carefully considered because lack of mating, and sealing, can cause blow-by at the union. The blow-by gives the appearance that the cask is drained when it is still full.

A full size mock-up of a port (they appear to be all of the same size), including the valve, should be constructed to ensure that there is adequate room to operate, remove and install the valve. The amount of time required for these operations must also be considered. The size of the (outer) closure plug seems large, and it is not clear why two closure plugs per port are required. This operations test should recognize the working conditions (gloves). In addition the analysis should calculate the estimated dose in the area of the pintle for the record. There appears to be limited neutron shielding at that point.

The use of the guide pins as proposed should be revisited. In underwater loading the position of these pins will not be visible from above, consequently, the operator may occasionally get lucky, but in general, will bounce the lid on the guide pins trying to achieve engagement. Additional specific comments are provided with Section 7 comments.

RESPONSE:

RP-1 The vent and drain line sizes will be reviewed during the final design and increased if necessary. Similarly, Westinghouse intends to reevaluate the drain system and develop a final design that provides a high degree of efficiency and reliability.

A mockup of one of the cask penetration ports is intended to be tested as part of SNL's remote-automatic system development program. Two closure plugs are used because of the contractual requirement for redundant closure protection. The quick-disconnect couplings used in the penetrations are specially adapted for operation with a tool without having to insert the hand in the plug cavity. This is done by having extensions to the coupling that can be actuated either manually or remotely.

The surface dose rates in the closure lid pintle area will be calculated in the final design.

Visibility of the guide pins will be improved by having slots rather than holes in the lid for pins. Other alternatives will also be considered during final design to enhance closure lid installation on the cask.

COMMENT:

RP-2 The principal concern regarding the basket is the lack of detail in the basket design. The mating of the basket to the bottom of the cask affects draining efficiency and residual contamination. The arrangement of the drain pipe in the basket structure should be such that the basket does not have to be destroyed to repair or replace a blocked drain pipe. The pipe should not be permitted to move down with repeated engagements of the mating fixture during lid installation. The basket, cask bottom and drain pipe arrangement should encourage water to be removed by the drain rather than by vacuuming.

The method of handling the basket, and of the use of the spacers with the baskets, could not be completely determined from the information provided.

RESPONSE:

RP-2 Complete details of the basket (and drain pipe) construction are provided in Drawings 1988E42 and 1988E44. Special grooves machined on the underside of the basket bottom plate ensure that water will drain readily into the sump groove at the cask bottom close to the ID. The well into which the drain pipe protrudes ensures that the residual water will be minimal. As indicated in later responses, Westinghouse will be reevaluating the drain system during the final design.

The basket is provided with a handling collar at the top end that is engaged with an internal grapple. Spacers are provided only at the bottom end of the fuel assemblies.

COMMENT:

RP-3 Numerous specific comments regarding the impact limiter installation/removal operations are presented in the discussions below. In general, consideration must be given to the special alignment problems that result from trying to manipulate a 2,000 pound impact limiter onto the cask when only very small clearances exist. Control of a crane to this clearance will be difficult or impossible to achieve.

The lift point of the limiters should be established over, or in line with, the limiter c-g. Even very small off sets from the true center of gravity will cause the limiter to hang at an angle. There is very little hope that a limiter that does not hang true will go onto, or come off of, the cask.

Consideration should be given to systems of trays or other supports that permit the limiter to be supported at the correct height while being installed or removed. These supports should allow the limiter to be "translated" to the front or rear of the trailer without being supported by an overhead crane. If possible, the limiters should remain on the trailer both for their own protection from damage during handling, and to reduce handling times. It is recognized that in some cases, due to cask lift height limits, one or both limiters must come off of the trailer. For these cases (and to support the current design if it is not changed) impact limiter stands should be designed to support the limiter in a way to cause the least damage. It is not clear that the limiters will be stable when resting on the outer 21 inch surface.

The contamination barrier (weather seal) should be designed to stay with the cask rather than the top limiter. This major source of contamination should travel with the cask and be removed at the cask

work station so that control of contamination is maintained. In addition, covers (contamination barriers) for the trunnion bolts should also be considered. It is likely that water would leak from the bolt holes for an extended period after removal of the cask from the pool.

RESPONSE:

RP-3 The impact limiter is designed to be handled on the transporter using a rail-mounted cradle. This concept was presented at the Design Review and is shown in Figure RP-3.1. With this approach, a crane will not be needed. In addition, lift points will be provided for the impact limiters in the final design to permit lifting and removal off of the transporter.

The weather seal is best retained on the impact limiter rather than the cask. The purpose of this barrier is not to limit radioactive contamination but to preclude road dirt and water from getting into the lid area. Locating the seal on the cask body will only increase the likelihood of radioactive contamination of the barrier.

The bolts for the trunnion sleeves will be covered with plastic covers or by using a sealant such as RTV.

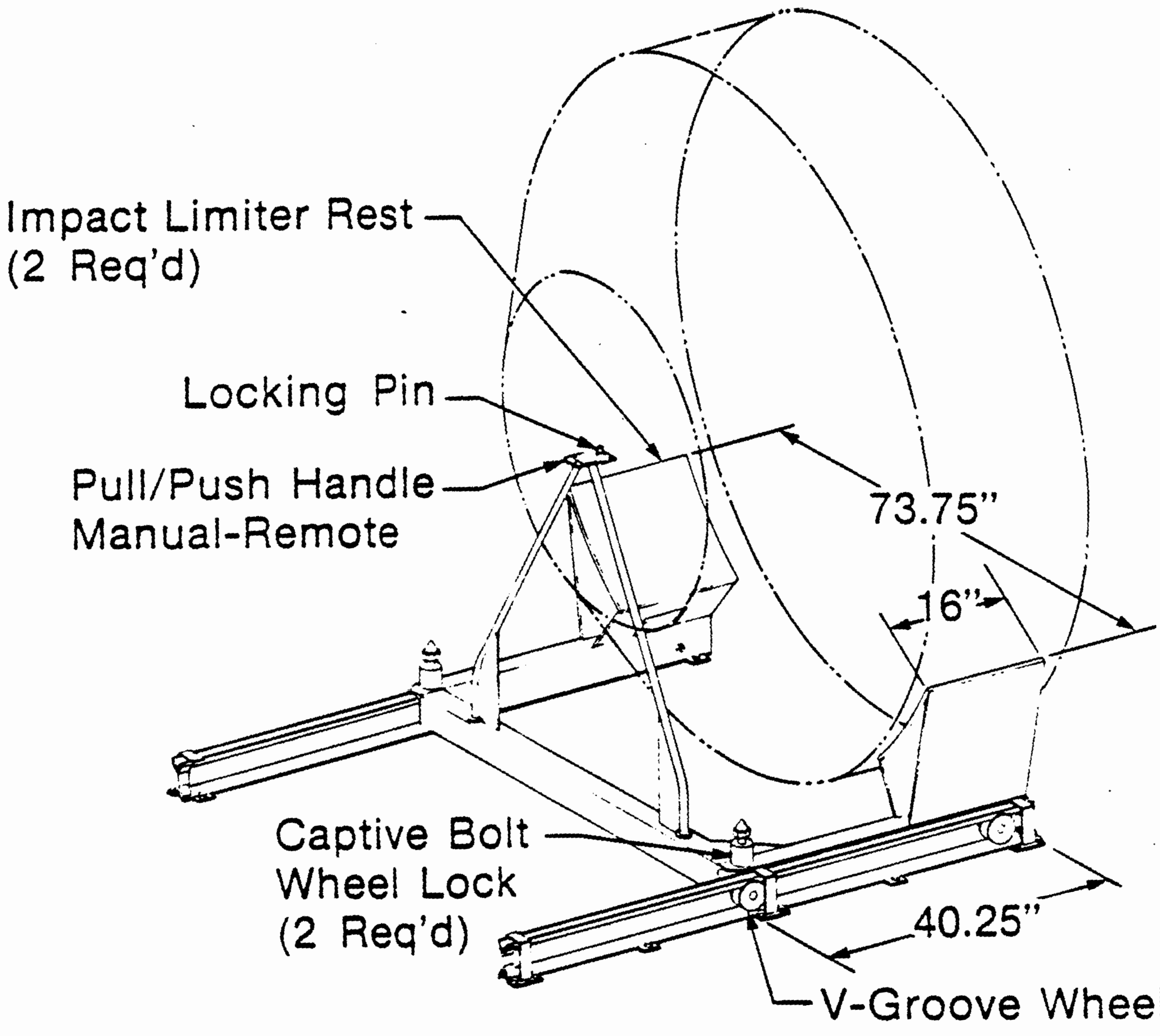

Figure RP-3.1 Impact Limiter Carriage

COMMENT:

RP-4 The personnel barrier should be designed to cover the cask and limiters as this will reduce the weather and environmental effects suffered by the cask, including protection from diesel soot. Covering the impact limiters will serve to "isolate" the cask system from the environment, and without doubt reduce contamination control concerns. It is noted that expanded metal has previously been a high maintenance item in other applications. This should be considered in its selection for use here. Fabric based barriers could also be considered for the increased potential for designs that would not require removal from the trailer. Frequently, removal requires the use of a separate crane, in addition to storage space. Consideration should be given to having the barrier "move" out of the way, but remain on the trailer since this will reduce equipment requirements and save time.

For this design, the barrier must be "square" to the cask during lifting or the impact limiters will be damaged during installation and removal.

RESPONSE:

RP-4 Westinghouse intends to evaluate, during the final design phase, a personnel barrier design based on fabric which will cover the cask and impact limiters, will stay on the trailer, and will not require the use of a crane. Such designs are in use with existing road casks and should prove feasible for the TITAN Cask.

COMMENT:

RP-5 The lifting yoke design should include an integral fixture or a stand to allow the yoke to stand vertically for attachment to the crane hook. Attachment with the yoke laying on the floor is very difficult, and becomes harder as hook size increases. Lift lugs should be added so that the yoke can be lowered from the vertical to the horizontal for placement in a shipping container. The lugs for horizontal lifting must be placed so that control of the yoke is maintained (i.e., the center of gravity of the yoke must be considered). Consideration should be given to the design of fixtures that mate to the mast coupling that allow attachment to sister hooks. Otherwise the yoke has restricted usefulness.

RESPONSE:

RP-5 The lifting yoke design will be modified during the final design phase to: 1) incorporate lift lugs to permit removal from and placement in a shipping container, and 2) incorporate fixtures that mate to the mast coupling that allow attachment to sister hooks. In addition, a stand will be designed to allow the yoke to stand vertically for attachment to the crane hook.

COMMENT:

RP-6 The shielding analysis appears to show very high (900 mrem/hr) radiation readings in the vicinity of the closure. Exposure at this level not only presents a high level of threat to the operators, but will result in operators reaching administrative control levels of exposure (less than NRC limits) in a very short time. Reactors could lose the use of loading personnel for a quarter after only one loading. If the estimated exposure is as high as the graphs seem to indicate, then additional consideration must be given to the design basis parameters, to the calculational methods, and to supplemental shielding steps to reduce exposure.

In addition, consideration must be given to design revisions that reduce the amount of time the operator is in the vicinity of high dose areas including lid bolt torquing, leak testing and decontamination tasks. Thought should be given to moving the cavity penetrations from the closure head to the cask body (side).

RESPONSE:

RP-6 The primary gamma dose rate on the surface of the cask reaches 700 mrem/hr at the location of the surface intersection of the closure lid and the body upper flange. The neutron contribution at the same location is about 28 mrem/hr for a total of 728 mrem/hr at a local point on the surface of the cask. The cask design has been modified since these shielding calculations were performed and the flange has been extended such that the closure lid is totally recessed. This additional material will reduce the surface dose rate to a value between 200 and 400 mrem/hr. It is noted that Figure 5.4-8, shows that the primary gamma dose rate drops from about 700 mrem/hr at the surface intersection of the closure lid and the flange (height of 241.5 cm, radius of 42 cm) to about 200 mrem/hr at about 6 inches from the surface of the closure lid. With the additional material now present in the design around the closure lid, this dose rate is reduced even further.

Special focus will be placed on obtaining further dose rate reductions in this area during the final design through reduction in the streaming paths to the extent feasible. In addition, the use of auxiliary shielding during cask operations in the area will be identified if found to be necessary.

COMMENT:

RP-7 The load bearing welds of the lid/pintle are not accessible. This will present some difficulties with load testing. Is annual load testing proposed? The description of the operation is not adequate. A single attachment point allows the lid to rotate when attached to a cable under water. At a minimum a second point is needed for the attachment of a cable to permit lid control and alignment.

RESPONSE:

RP-7 Provisions will be made in the final design to improve access to the load bearing welds of the pintle for examination during load tests that are envisioned to be performed annually. A second attachment point will be added on the closure lid to prevent rotation.

COMMENT:

RP-8 **Since the closure bolts are captured, what indicates to the operator that the bolt is no longer engaged in the cask flange threads? The pop-up travel provided by the springs seems to be limited in that "full up and clear" of the bolt appears to be very close to "still engaged." These bolts, and the recess, will be major sources of contamination. In addition, the very high torque value may require that the bolts be inspected on each use. Consideration should be given to making these bolts removable.**

RESPONSE:

RP-8 The use of captive bolting is preferred for the closure lid because of the requirement for remote automated operations. Loose bolts will cause problems for remote automated systems and negate the time saving advantages of going to such systems. It is recognized that captive bolting will render decontamination more difficult. However, these bolts and associated hardware can be readily removed for extensive decontamination.

The difference between the "just engaged" and "popped up" positions of the bolt is 0.70 inch which can be readily detected.

The bolt torque values were recalculated using more realistic closure lid temperatures during the fire accident and will be about 1000 ft.-lb. rather than the earlier value of 2100 ft.-lb. With these levels of bolt preload, inspection should not be required with each use.

COMMENT:

RP-9 While the arrangement of the bolts appears to provide excellent opportunities for remote operation, the amount of material required to be removed from the lid forging (16 x approximately 2 inch diameter) plus the test ports, will place the lid at risk in the drop accident analysis. Consideration could be given to use of bolts with a socket hex drive to reduce the amount of material removed. These large openings will also be difficult to decontaminate for shipment. Special attention must be given to the contamination barrier for the entire lid and annulus area.

RESPONSE:

RP-9 The preliminary structural evaluations have shown that the closure lid has a high probability of meeting all the requirements in the final design analysis. The captive bolt designs with conical lead-ins were selected for compatibility with remote automated operation. The use of socket head drives will be evaluated in the final design in conjunction with the SNL's remote-automated systems development program results.

COMMENT:

RP-10 Guide pins that do not project above the cask top surface (cask outer flange) have very limited value. They are hidden from the operator once the lid is close to the cask, and the tendency is for the operator to bounce the lid on the pins.

RESPONSE:

RP-10 The closure lid is designed to be fully recessed into the cask body without any protruding bolt heads or alignment pins. With such a design, the alignment pins will be hidden from the operator when the lid is close to the cask. Proper rotational alignment is facilitated by orientation markings on the lid and cask body. Once the lid is aligned properly with respect to those markings, it can be guided down over the alignment pins. The alignment pin holes have generous lead-ins for this purpose.

COMMENT:

RP-11 The top of the closure lid should have markings that identify the ports.

RESPONSE:

RP-11 Markings that identify the various ports in the closure lid will provided on the top of the lid in the final design.

COMMENT:

RP-12 The planned arrangement of spacers was not presented, however, no method of attaching spacers to the underside of the lid could be identified.

RESPONSE:

RP-12 No spacers are provided at the top of the fuel assemblies. The assemblies will be positioned close to the underside of the lid by using spacers at the bottom end.

COMMENT:

RP-13 **A .375 inch hole is considered small. A .5 inch hole should be used at the top of the cavity. A larger tube/opening should be used for the drain (3/4 inch). The design of the vent and drain should consider the efficiency of the filling and draining operations, and the possible requirements for cool down of the fuel or c;ask, flushing operations, and the recirculation of decontamination fluids.**

RESPONSE:

RP-13 Larger port openings for the purge and drain ports will be evaluated during the final design. The current drain sizing permits the cask cavity to be drained in approximately 15 minutes.

COMMENT:

RP-14 It is not clear that the 2.5 inch opening around the snap-tite allows sufficient room for manual operation of the valve, given that the valve is 6 inches from the top of the lid. A full size mockup of the valve arrangement should be used to verify manual operation, inspection and maintenance. (Operation should consider the fact that hands will be gloved - usually several layers.)

RESPONSE:

RP-14 The couplings will be modified to include extensions that will permit manual operation without having to insert the hand into the recess. This arrangement will also be fully compatible with remote automatic operations. Westinghouse recommends this approach to the alternative of opening the recess diameter because of the limited space available in a truck cask closure lid.

COMMENT:

RP-15 How is pool water removed from the volume around the snap-tite valves (drain and vent ports).

RESPONSE:

RP-15 The closure plugs will be in place when the cask is submerged in the pool. When the cask is taken out of the pool, the water will be drained through a drain hole for the outer plug cavity.

COMMENT:

RP-16 **The method of mating of the drain pipe to the closure id could not be determined from the information provided. Sealing at the union of the lid component with the drain pipe has been a problem. Inadequate sealing can lead to "blow by" during draining, giving the impression that the cask is empty when it is still full of water. Based on the information presented, the drain line may engage the lid before the lid engages the guide pin.**

RESPONSE:

RP-16 Details of the mating of the drain pipe with the closure lid are provided on Drawing 1988E43, Sheets 13-15. The coupling details are shown on Sheet 17. Details of the drain pipe are also given in Drawing 1988E42, Sheet 4 and Drawing 1988E44, Sheet 4. Westinghouse will be reevalating the entire drain system design in the context of the comments and will ensure that the final design will provide a reliable system.

COMMENT:

RP-17 It is not clear that the valve used for verification can be operated manually (it is not clear that manual attachment of the test fixture is required). A mockup should be used to ensure that the necessary manual operations, inspection, and replacement can be performed, considering the conditions.

RESPONSE:

RP-17 The seal verification test port can be operated either manually or with remote automated equipment, in conjunction with the tool that was presented at the design review meeting (see Figure RP-17.1). A mockup of the seal verification test port and tool will be built and tested at SNL under their remote automated systems development program.

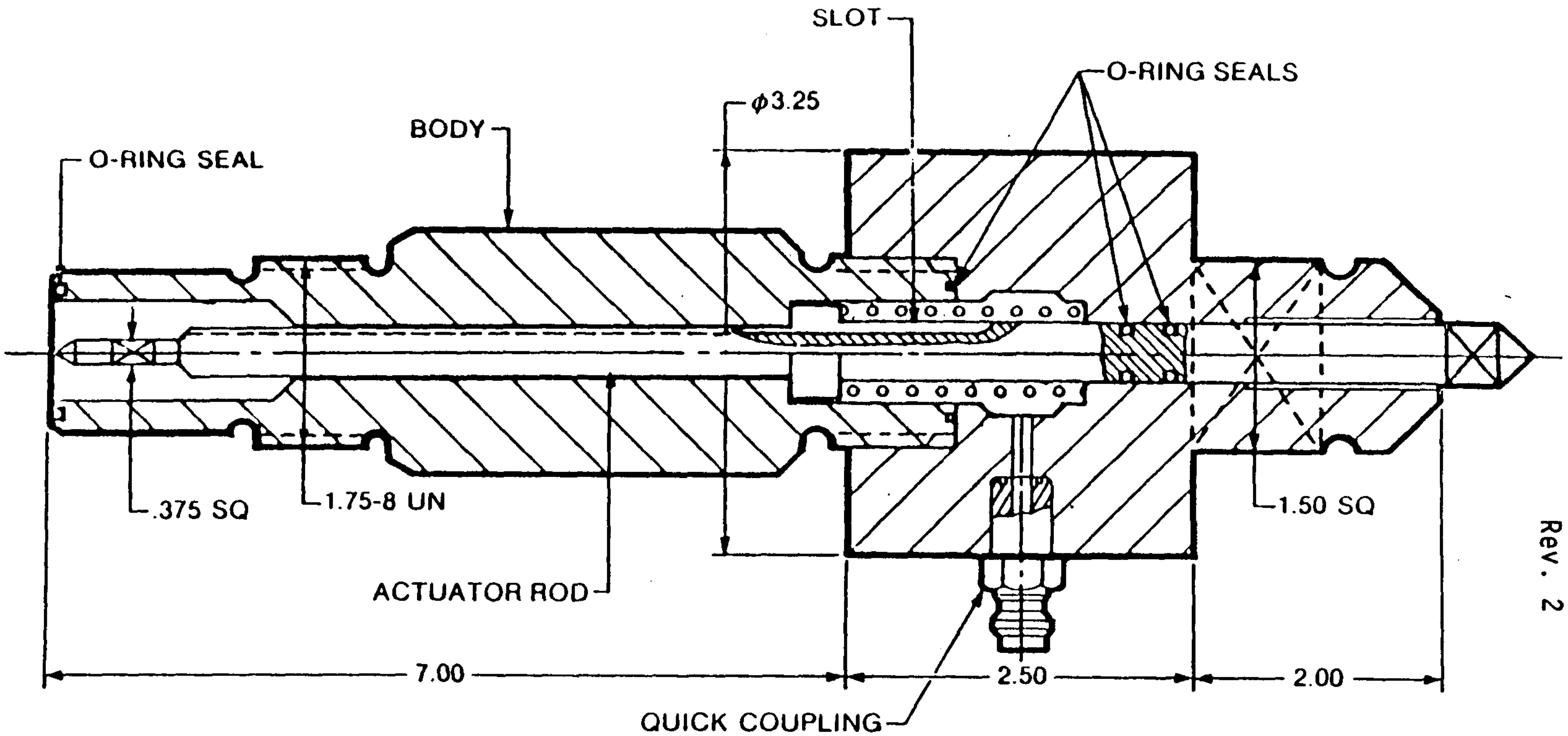

Figure RP-17.1 Seal Test Tooling

COMMENT:

RP-18 It is not clear what the "special tool" is looking for in the leak testing equipment. No leak test procedure has been given. It will likely be easier to pressure test the seal arrangement.

RESPONSE:

RP-18 The special tool is installed on the leak test port after removal of the outer plug. The tool then loosens the inner plug to provide a connecting path to the seal area to measure any leakage. The leak test apparatus is connected to the special tool with a hose and quick disconnect coupling.

Detailed leak test procedures are more appropriately provided in the Technical Manual.

COMMENT:

RP-19 The use of standard hex sockets (rather than square sockets) should be considered, as this reduces the number of "non-standard" tools required to support cask operations at commercial facilities. (Note removal of pool water from these sockets must be considered.)

RESPONSE:

RP-19 The use of standard hex sockets will be considered in the final design. The sockets will be provided with drain holes to remove the pool water.

COMMENT:

RP-20 **It would seem possible that at least one test port could be eliminated by having either the vent or the drain test port also test the lid O-ring annulus, through a vertically drilled line.**

RESPONSE:

RP-20 The use of one test port to test two different seals is undesirable because it would not allow the source of a leak to be determined.

COMMENT:

RP-21 It is not clear if the spacers are intended to go only in the bottom of the cask. If so, this is a good idea, but must be permitted by the top end shielding analysis. It is not clear if the spacers are intended to be safety related components.

RESPONSE:

RP-21 The spacers are intended to be used at the bottom of the cask only. The shielding analysis was performed assuming that the top end of the fuel assemblies are close to the underside of the cask lid and the bottom end of the assemblies contact the bottom of the cask cavity. The spacers are not intended to be safety-related components.

COMMENT:

RP-22 It is not clear how the basket, or the drain pipe, "interfaces" with the bottom of the cask. The bottom of the cask should be configured (locally shaped) to encourage water to move to the vicinity of the drain tube. Even small quantities of water can take a long time to remove by vacuum drying.

RESPONSE:

RP-22 The interface between the drain pipe and the bottom of the cask cavity is shown in Drawing 1988E42, Sheet 4, Drawing 1988E44, Sheet 4, and Drawing 1988E43, Sheets 4 and 7. An annular groove at the bottom of the cask cavity near the ID serves as a sump to collect water. This groove is also provided with a well into which this drain tube will protrude. The amount of water that will be left in the cask is therefore minimal.

COMMENT:

RP-23 Building the drain line into the basket means that a pump must be used to drain the cask cavity if the basket is removed.

RESPONSE:

RP-23 As presently configured, draining the cask cavity without the basket in place will require a pump. The entire drain system will be evaluated during the final design and alternatives such as attaching the drain pipe to the cask cavity will be considered which would eliminate the need for a pump.

COMMENT:

RP-24 In the design, the drain line extends about 6 inches above the basket surface. The line should be "protected" from incidental damage due to contact with wayward fuel assemblies. (Does the funnel protect this line, or interfere with it?) Even small dings could prevent sealing with the mating tube in the lid. The resulting blow-by would not allow the cavity to be drained using the drain tube.

RESPONSE:

RP-24 The top end of the drain tube is protected from damage during fuel assembly insertion and removal by the basket lead-in fixture.

COMMENT:

RP-25 The key way seems needlessly small. A larger key should be considered and it should be at the top of the cask cavity so that it can be seen during basket installation. (It is noted that yellow and orange are good under water colors.)

As an observation, the orientation of the basket with respect to the cask should be shown.

RESPONSE:

RP-25 The use of a larger key will be considered during the final design and will be located as close as possible to the top of the cask cavity. It is noted that basket replacement will be a relatively infrequent operation that will be performed at a cask maintenance facility in a dry environment.

The orientation of the basket with respect to the cask is shown on the cask and basket drawings and also in Figure 1.2-1 of the Preliminary Design Report.

COMMENT:

RP-26 The function and operation of the basket handling collar could not be determined. For example, Step 7.5.3 specifies that the basket inner ring (assumed to be this collar) is "gripped." This operation needs additional detail. Attachment of the basket to the lift gear should be rapid to reduce exposure.

It is assumed that the handling collar also serves to limit travel of the basket (when in the horizontal position) in the cask cavity. If it does not, then basket movement must be addressed.

RESPONSE:

RP-26 The basket handling collar internal ledge is engaged using an integral grapple similar to that used for grappling fuel assemblies. This type of grapple can be remotely operated and is fast-acting.

The basket handling collar also serves to limit the axial movement of the basket.

COMMENT:

RP-27 **The discussion of the Fuel Assembly Lead-in fixture reads as if its use is consider optional. Optional use is a good idea. It is not clear from the drawings or discussion how many Lead-in Fixtures are required. Does one serve all three PWR positions?**

RESPONSE:

RP-27 The fuel assembly lead-in fixture described in the Preliminary Design Report was a conceptual design only but its use was not considered optional. Two fixtures are required - one for use with the PWR basket and the other for use with the BWR basket. A single fixture incorporates lead-ins to all the fuel positions and does not require any repositioning.

COMMENT:

RP-28 The method by which pool water drains from the fuel assembly position is not clear. The design to accomplish this should allow rapid draining.

RESPONSE:

RP-28 The bottom end of each fuel compartment in the basket is open to facilitate rapid draining. Slots are also machined on the underside of the basket to permit drainage of water into the annular groove in the bottom of the cask cavity.

COMMENT:

RP-29 The spacers shown are too large for easy installation. At least 1/4 inch should be allowed on all sides. The means of grappling the spacer for removal must ensure that the spacer will not fall off of the tool during handling.

RESPONSE:

RP-29 The clearance between the fuel assembly spacers and the basket compartments will be reviewed during the final design and increased to the maximum feasible extent to simplify installation and removal. Features will be added to ensure that the spacer will not fall off the tool during handling.

COMMENT:

RP-30 Because the drawing appears to show that there is only 0.06 inch clearance between the basket and cavity wall, careful consideration must be given to the means of lifting the basket. Unless the basket hangs straight, installation will be very difficult and could result in damage to the cask or basket during attempts to free a wedged basket. Further, because of the clearance, a fixture that protects the seal surface during basket handling should be considered.

RESPONSE:

RP-30 Basket removal and replacement operations will be relatively infrequent and will be performed at a cask maintenance facility. The basket is designed to be removed using an internal grapple that engages with the handling collar on the basket and this arrangement is expected to maintain the basket in a reasonably vertical attitude. Nevertheless, the seal surface will be protected during basket handling.

COMMENT:

RP-31 The impact limiter attachment method is considered superior to "through the limiter" arrangements. However, the use of captured bolts does not seem prudent in this application. Since the bolt, spring and housing travel with the cask into the pool, decontamination of this area could be difficult. Decontamination is complicated by the use of captured bolts, since it would appear the bolts must be removed for decontamination of the bolt and fixture.

RESPONSE:

RP-31 Captive bolts are desirable in order to facilitate installation and removal of the impact limiters using remote automated equipment at the receiving facility. It is recognized that the use of such hardware will make decontamination more difficult. There is hence a tradeoff between features enhancing remote operation versus features simplifying decontamination. The impact limiter attachment hardware will be tested as part of SNL's remote automated systems development and modifications that simplify decontamination will be incorporated to the maximum feasible extent.

COMMENT:

RP-32 The method of removing, installing and handling the limiter is not described, and no lift points were found. Provision should be made for removing the limiters from the cask without requiring a crane, and for storing the limiters on the trailer when not in use. Lift points (over c-g) are required in the event that the limiter must be removed from the trailer. See additional comments at Step 7.1.1.5, of the Loading Procedure.

RESPONSE:

RP-32 Westinghouse has developed a rail mounted cradle (see figure attached to Response RP-3) which will permit the impact limiters to be installed and removed from the cask and retained on the transporter without requiring a crane. Additional features for lifting the limiters off of the transporter will be incorporated in the final design.

COMMENT:

RP-33 No handles for maneuvering the limiters or for attachment of tag lines are shown.

RESPONSE:

RP-33 The impact limiters are intended to be maneuvered on the transporter on rail-mounted cradles. Provisions will be included in the final design for lifting the impact limiters off of the transporter.

COMMENT:

RP-34 The intended clearance between the cask and impact limiters could not be found on the design drawings, but is given in Table 3.4-1 as 0.06 inches. This clearance will make installation of the limiters very difficult, if not impossible.

The step on the inside of the bottom limiter will make alignment by crane somewhat difficult since the limiter is already on the cask approximately 5 inches. A tray for installing the limiters is needed.

RESPONSE:

RP-34 A complete evaluation of tolerances between the impact limiter and cask will be completed during final design and appropriate modifications made prior to the manufacture of the design verification test model.

The step currently shown on the inside of the bottom limiter will be eliminated during final design by changing the neutron shielding profile on the cask body so that both limiters will be identical.

The impact limiter will be maneuvered on the transporter using a rail-mounted cradle concept that was presented at the design review and is shown in the figure attached to Response RP-3. The detailed design of the cradle will be developed during final design.

COMMENT:

RP-35 It is not clear that the limiter will be stable when resting on the 21 inch surface. Will the limiter retain its shape if set down? Horizontal lift lugs are also required.

RESPONSE:

RP-35 The impact limiter will be supported on the transporter and maneuvered using a rail mounted cradle shown in Figure RJ-3.1. The cradle is profiled to match the impact limiter curvature and should provide for a stable support. Additional lifting provisions for removing the limiters off of the transporter will be added in the final design.

COMMENT:

RP-36 It is not clear why 1/2 scale testing is being used, since 1/4 scale is typically sufficient.

RESPONSE:

RP-36 A 1/4 scale model of a truck cask is not feasible because of the practical difficulties in fabricating components such as the fuel basket. Justification for the use of a 1/2 scale model was provided to DOE-ID early in the cask development program and has been accepted by both the DOE and the NRC.

COMMENT:

RP-37 It is very difficult to connect yokes to hooks unless the yoke can be supported vertically for hook attachment. This yoke requires a stand or additional structures to allow vertical connection to the hook. In addition, lift attachment points are needed to allow the yoke to be removed from the shipping container in a controlled fashion.

RESPONSE:

RP-37 See response to Comment RP-5.

COMMENT:

RP-38 The design of the yoke will prevent its use with sister hooks without additional fixtures. Has an alternate bail and mast been considered for attachment at the coupling plate?

RESPONSE:

RP-38 The yoke has been designed for use with two different bails: a 125 ton double hook and a 100 ton hot cell crane hook. The bolted attachments permit use of different length masts as well as sister hooks.

COMMENT:

RP-39 The analysis of the bolted connection at the bail/mast coupling plate is not presented. It would seem that the tension on these bolts would be high enough to make one nervous.

RESPONSE:

RP-39 Analysis of the bolted connection at the bail/mast coupling plate is presented in Section 2.10.6 of the Preliminary Design Report. The results show an adequate margin of safety for the bolted joint (+0.101).

COMMENT:

RP-40 The cask is designed to accommodate redundant lift systems, but the yoke design does not seem to have the same adaptability. Are the extra trunnions redundant?

RESPONSE:

RP-40 The contract requires the cask to be designed with a redundant pair of lifting trunnions. The yoke assembly is designed for use with the prototype casks during the demonstration tests at either the shipping or the receiving facility. It is not a contractual requirement to design a yoke assembly to accommodate single failure of the handling system.

COMMENT:

RP-41 The use of proximity indicators is suspect. Unless the indicators are rugged, retain their alignment during yoke handling, and are reliable, they could be more trouble than they are worth. Failure of an indicator to operate could result in the yoke having to be returned to the operating floor for adjustments that can't be checked until the yoke is returned to the cask. In the worst case, the connection of the yoke to the cask will be verified visually, which must be done even if the locked lights are on. A mechanical indicator should be considered.

RESPONSE:

RP-41 Westinghouse agrees that visual observation should supplement the indications provided by the proximity sensors. The yoke assembly design will be revised during final design to incorporate a mechanical indication of yoke arm engagement with the trunnions.

COMMENT:

RP-42 Can the yoke arms be moved manually if air pressure fails? If not, the system should be redesigned so that they can. The "screw" provided to move the arms is partially obscured by the yoke, and totally obscured by the crane block. This feature should be reconsidered.

RESPONSE:

RP-42 The yoke arms can be moved in the event of loss of air pressure by operating the manual override screw using an impact wrench. This feature will be reevaluated during the final design to improve accessibility for operation.

COMMENT:

RP-43 The use of Nitronic-60 for the trunnions and 304 stainless for the lifting yoke may cause galling of the 304.

RESPONSE:

RP-43 The combination of Nitronic-60 and 304 stainless was selected based on the galling threshold bearing stress of 50 ksi (ARMCO Product Data Bulletin NO. S-45). The calculated bearing stress is approximately 40 ksi and galling should not occur in service.

COMMENT:

RP-44 The personnel barrier should be designed to cover the impact limiters, and to be removed without requiring a crane. Covering the impact limiters serves to protect the limiters from the elements and from incidental damage. This would also act to isolate the cask system from the environment, reducing contamination concerns. Reduced handling time will result if the barrier could be moved to expose the cask without requiring a crane, and without removing it from the trailer. For this design, it could split in the center and the two sections moved to each end of the trailer.

(With this design) There is certain to be damage to the limiters during installation and removal of the barrier.

RESPONSE:

RP-44 See response to Comment RP-4.

COMMENT:

RP-45 The moisture content of the boral matrix immediately prior to the seal welding of the cladding is critical in this application, but is not discussed in the specifications. In addition to including this in the specification, a thermal test of finished plates may be desirable.

RESPONSE:

RP-45 Limits on the moisture content of the boral matrix and any required tests will be specified in the final design report.

COMMENT:

RP-46 Do the payload weights include BWR fuel channels or PWR spiders?

RESPONSE:

RP-46 The weights of the BWR fuel channels were not considered in the maximum BWR fuel assembly weight of 640 pounds used in the evaluation of the cask, as the cask contract specifies that nonfuel components do not have to be accommodated if they would decrease the payload of the cask. However, the 640 pound weight limit/assembly used for design will envelope the weight of many of the BWR spent fuel assemblies including their fuel channels.

The payload weight of the PWR assembly is 1515 pounds and does not include the weight of the control rods or spider.

COMMENT:

RP-47 The weights used in Table 2.2-1 on page 29 should be reconciled with those used in Table 2.7-1 on page 2-85.

RESPONSE:

RP-47 For the SCANS analysis of the cask, conservative estimates of cask component weights were used to obtain maximum loadings and stresses. The allowable gross weight of 54,000 pounds for the package was used instead of the reported actual cask weight of 53,044 pounds. The contents/internals weight of 11,076 pounds includes the weight of the fuel basket, Boro-Silicone and payload. Grouping of all these in the contents weight provides conservative analysis results for the heads and cask body.

COMMENT:

RP-48 It is not clear that possible reactions between the titanium and depleted uranium at elevated temperatures has been addressed.

RESPONSE:

RP-48 See response to Comment WS-4.

COMMENT:

RP-49 **The actual analysis performed for the trunnions can not be determined. Are the safety factors on yield and ultimate 3 and 5, or 6 and 10. Factors of 6 and 10 are advantageous in some situations.**

RESPONSE:

RP-49 The design of the cask lifting devices or trunnions which are a structural part of the cask are based on supporting at least three times the weight of the cask without yielding in accordance with the requirements of 10 CFR Part 71.45(a). The design of integrally attached portions of the cask tiedown trunnions is based on withstanding the specified transport loadings of 10 g's longitudinal, 2 g's vertical and 5 g's lateral, per 10 CFR Part 71.45(b)(1), without yielding. Non-integral structural members of the tiedown trunnions are based on the truck transport loadings of 2.3 g's longitudinal, 1.6 g's lateral and 2.0 g's vertical without yielding.

Loadings on the trunnions that can be produced by rail transport will be evaluated during the final design.

COMMENT:

RP-50 **It is not clear if this analysis covers vibration loads, and possible fatigue failure, of the basket or other internal components.**

RESPONSE:

RP-50 For the preliminary design of the cask, a fatigue evaluation considering vibration loads was completed for the main cylindrical shells of the cask (see Sections 2.6.5 of the Preliminary Design Report). The fuel baskets will be lightly stressed for loads that result from vibrations normally incident to transport. Therefore, a fatigue evaluation of the baskets was not performed for the preliminary design phase. For final design, a fatigue evaluation will be carried out for the fuel baskets and all other major structural components of the cask.

COMMENT:

RP-51 Both of these sections show negative margins of safety for the closure design. This should be cause for concern. (This concern is noted in the text and is mentioned here as an observation.) Consideration should be given to reducing the complexity of the closure lid to reduce the amount of structural material removed.

RESPONSE:

RP-51 The detailed analysis of the closure head reported in Section 2.7.1.1 of the Preliminary Design Report shows a slight negative margin of -0.04. This analysis was very conservative because of the model and loading limitations used in the analysis. The model used an earlier configuration of the head that was not fully recessed. It is expected that for the final design, all allowables will be met when the current design configuration of the head and flange is used.

During the final design phase of the cask the material type for neutron shielding will be re-evaluated. If a thinner layer of material can be used it will simplify the closure head design.

Consideration will be given during the final design to simplifying the closure lid design and minimizing the machining requirements.

COMMENT:

RP-52 The torque required on the closure lid bolts to establish preload can not be obtained except through the use of braced power tools. 2100 ft/lbs is an over-turning force for a robot. Consideration must be given to how this torque will be applied.

RESPONSE:

RP-52 The torque required to establish the required preload on the closure lid bolts is expected to be about 1000 ft.-lb. The 2100 ft.-lb value given in the Preliminary Design Report was based on an overly conservative estimate of the closure lid temperature during the fire accident which has now been revised downwards. A power tool will be required and compact units are readily available for such service.

COMMENT:

RP-53 Consideration must be given to reducing the torque required for lid bolts.

RESPONSE:

RP-53 The torque value of 2100 ft.-lb. given in the Preliminary Design Report was determined based on very conservative estimates of the lid-flange region temperature during a fire accident which required a bolt extension sufficient to compensate for the differential thermal expansion between the Alloy 718 bolts and the titanium lid material such that the lid will remain tight against the cask flange. The preliminary design thermal analysis shows significantly lower temperatures which will allow a significant reduction of the bolt preload. The torque required is now estimated to be about 1000 ft.-lb.

COMMENT:

RP-54 It is expected that the closure lid bolt torque will accelerate thread wear on both the bolt and the threaded insert. It is not clear that this aspect of the required torque has been addressed.

RESPONSE:

RP-54 Revised bolt preload calculations have been performed which show that the bolt torque is about 1000 ft.-lb. rather than 2100 ft.-lb. originally estimated. Thread wear will be minimal because of the use of Alloy 718 bolts and threaded inserts which have excellent wear resistance. The use of threaded inserts will also permit ready replacement of the wear surfaces.

COMMENT:

RP-55 **The torque required to seal the unloaded (empty of fuel) cask should be calculated and included in the appropriate handling procedure. A much reduced torque value should be required which will facilitate preparing the unloaded cask for shipment.**

RESPONSE:

RP-55 The torque value for the unloaded cask will be the same as that for the loaded cask because the required torque value is based on the differential thermal expansion between the bolt and the lid at the material temperatures during the fire accident condition. The unloaded cask has also to maintain leak tightness because of the contaminated internals.

COMMENT:

RP-56 **Section 2.10.6 implies that the tie-down system would fail in certain accidents but failure modes are not discussed or analyzed. This analysis should show that the impact limiters are not damaged by the tie-down structure during accidents, and that the impact limiters will not be required to bear the additional loads that an attached trailer structure might impose on the system.**

RESPONSE:

RP-56 The integrally attached portions of cask tiedown trunnions were designed for loadings of 10g longitudinal, 5g lateral and 2g vertical. The remaining portions of the trunnions and the tiedown system itself were designed for loadings of 2.3g longitudinal, 1.6g lateral and 2.0g vertical. If a severe transportation accident occurred, the tiedown system will fail first, releasing the cask from the trailer. Failure of the tiedown structural system should not effect the impact limiters. Allowable stresses for the tiedown system are based on limiting yielding of the aluminum material. However, the ultimate strength of the material is only 20% above the material yield strength. Therefore, the tiedown structures will begin to fail at about 20% over the design load (2.3 g's x 1.2 = 2.76 g's, for example) which is still much lower than the cask 10 g loading.

COMMENT:

RP-57 Section 2.10.4 should address partially loaded basket conditions, for those cases where the cask might carry less than a full load. If the analysis is bounded by the fully loaded basket conditions, then it should be sufficient to merely make the statement.

RESPONSE:

RP-57 The evaluations of the fuel baskets reported in Sections 2.6.7.2 and 2.7.1.2 of the Preliminary Design Report were based on fully loaded baskets. It was assumed that these analyses bounded all other partially loaded basket conditions. However, this assumption will be verified during the final design phase.

COMMENT:

RP-58 **Section 3.4.2 - It is not clear from the analysis if the effects of BWR fuel with fuel channels have been considered.**

RESPONSE:

RP-58 Basket temperatures with BWR fuel were not calculated during the Preliminary Design because the decay power for three PWR assemblies was higher than for seven BWR assemblies. Thermal analysis for the BWR fuel case will be performed during final design and will consider the presence of the fuel channels.

COMMENT:

RP-59 Section 3.4.6 - Since the evaluation has concluded that the accessible cask surface temperature exceeds 180 degrees, then this section should specify a personnel barrier for this package.

RESPONSE:

RP-59 The requirement is that the surface temperature should not exceed 180°F when the cask is in the shade. The accessible surface of the TITAN cask is predicted to have a maximum temperature of 150°F when the cask is in still air at 100°F and in the shade (see Figure 3.5-2, Page 3-28 of the report). This meets the requirement of 10 CFR 71, Paragraph 71.43(g) with a 30°F margin. Hence a personnel barrier is not required from the standpoint of the 180°F temperature limit though the design includes such a barrier.

COMMENT:

RP-60 Section 3.5.2 - It would appear on the surface that assumptions regarding the "intactness" or "point of removal" of impact limiter or neutron shield material would be difficult to support. The natural first question is: "What happens in the absence of the limiters?" It is assumed that this has been done and indicates that the limiters or most of them must stay in place; but, it should be discussed. The second question, which is also unaddressed, is: "What happens to the neutron shield?" a The post fire accident response of the Boro-Silicone should also be discussed. The difficulty is that Item 2 of Section 3.5.6, appears to imply that it does not matter if the impact limiters remain attached or not.

RESPONSE:

RP-60 The thermal analysis of the cask closure head area was performed assuming that the impact limiters stay attached to the cask following the drop events. During final design, thermal analysis will be completed assuming the limiters are not attached to the cask. However, simple extrapolations indicate that the cask closure seals under that condition would still see temperatures below the maximum operating limit of 400°F.

The thermal, shielding, and criticality analyses for the final design will be performed assuming complete loss of the neutron shielding. Preliminary evaluations show that all design limits for the post-accident conditions will be met in the shielding and criticality areas assuming total loss of the neutron shielding. In the case of thermal analysis, calculations will be performed assuming the neutron shield is present and also for the case it is lost. This will enable the worst-case temperatures to be calculated for the closure seals and the structural components of the cask.

COMMENT:

RP-61 Section 4.2.3 - The containment criterion should be revisited. The application of a 10^{-7} criteria is unnecessarily restrictive for the operating conditions. The strategy for containment verification should be that for annual testing high standards must be met (leak rates of 10^{-6} A_2 or less), using very sensitive equipment that has limited portability. For post-loading testing, pressure testing (10^{-3}) should be adequate. This would require less time to perform, fits well with the skills and capabilities of cask handling personnel, and results in lower costs.

RESPONSE:

RP-61 Westinghouse agrees that the application of the leak tightness criterion is unnecessarily restrictive for operating conditions. However, alternative approaches have not been accepted by the NRC at this time, though development work is ongoing in the U.S. in support of less stringent criteria based on computing the specific activity of the media in the cask. When those efforts are successful in obtaining the NRC's concurrence, appropriate alternative leak test methods will be proposed by Westinghouse.

COMMENT:

RP-62 The analysis or discussion of the post-accident condition dose rates could not be found.

RESPONSE:

RP-62 The discussion of post-accident condition dose rates is given in Section 3.5.6, pages 3-36 and 3-37 of the Preliminary Design Report.

COMMENT:

RP-63 The narrative relative to the "side surface" dose rates was impossible to follow, consequently, the figures proporting to show external surface dose rates (gamma and neutron) can only be seen as demonstrating that the surface dose rates are very high.

While Table 5.4-2 and 5.4-3 do appear to show that the cask meets the regulatory requirements for transport, the cask (skin) surface dose rates appear (from figures 5.4-5 and 5.4-13) to approach 1,000 mrem/hr. Sources of radiation exceeding 100 mrem/hr must be maintained locked and access controlled in almost all of the commercial power plants. Access to the cask, including the impact limiters, may need to be limited by a personnel barrier to meet plant administrative procedures. It is recognized that a worst case condition has been analyzed, none the less, cautionary notes must be added to the procedure to protect operators and supplemental shielding may be required for use during handling.

It is recommended that most of this narrative be relegated to "supporting information," since the worst case will limit.

RESPONSE: See response to Comment RP-6

COMMENT:

RP-64 **There is an apparent inconsistency between the discussion of Figures 5.4-5 and 6, and the dose point locations cited in Figures 5.4-8 and 10. The curves in Figures 5.4-5 and 6 show dose rates on surfaces that do not exist according to locations specified in Figures 5.4-8 and 10.**

RESPONSE:

RP-64 The discussion of Figures 5.4-5 and 5.4-6 is given on page 5-45 of the Preliminary Design Report. The figures show "surface" dose rates at three different radii. These "surfaces" do not exist as shown in Figures 5.4-8 and 5.4-10. For example, for "side surface 1," (the solid line in Figure 5.4-6) the curve should have been shown for heights between -247 and -255 cm only. The rest of the curve shows what the dose rate would be if the surface continued over the entire cask length. The curves in Figures 5.4-5 and 5.4-6 should be treated as showing the dose rates at three different radii rather than "side surfaces."

COMMENT:

RP-65 **Many of the time allowances for activities in the handling procedure seem optimistic.**

RESPONSE:

RP-65 The time estimates were based on several assumptions such as (1) learning experience through involvement in a planned, continuous shipping campaign at each reactor plant, (2) availability of trained personnel and equipment as and when needed in the operating sequence, and (3) use of remote automated equipment at the receiving facility. Westinghouse experience with fuel shipments at West Valley, E-MAD and Virginia Power (Surry) confirms that the time estimates are realistic and achievable subject to the assumptions that are stated.

COMMENT:

RP-66 Section 7.1.1 - Step 7.1.1.5. The method of moving the impact limiter is not described. No lift fixtures are shown the limiter drawings. For routine operations, removal of the limiters from the cask should not require a crane and should stay on the trailer when not in use. The fixture for "translating" the limiters and for securing the limiter on the trailer should be a design feature. Lifting attachments must be provided for the limiters in the event that they must be removed from the trailer or replaced.

RESPONSE:

RP-66 See response to Comment RP-3.

COMMENT:

RP-67 Step 7.1.1.12. Add "if required."

RESPONSE:

RP-67 The comment will be incorporated in the Preliminary Design Report.

COMMENT:

RP-68 Step 7.1.1.13. It is likely that radiation levels of the cask internals and the large potential for airborne contamination will preclude removal of the lid in air. There must be a step to fill the cask with water (provides shielding and essentially eliminates air borne) or for raising the lid for seal inspection/replacement after the cask is in the pool.

RESPONSE:

RP-68 The procedure will be changed to fill the cask with the lid in place and to remove the lid in the pool.

COMMENT:

RP-69 Section 7.1.2 - Step 7.1.2.2. Is a "shroud or demineralized water bucket" being designed? No attachment points for a "shroud" could be found. No means for bolting a top (of a water bucket) could be found. Use of the closure bolt holes would prevent later installation of the closure. The finish of the cask should be such that there is a reasonable expectation that manual decontamination of the cask external surface will be satisfactory.

RESPONSE:

RP-69 The design of a shroud or demineralized water bucket is outside the current scope of work.

COMMENT:

RP-70 **Step 7.1.2.4 Removal from the pool should only occur if the yoke is in the way of loading. (Optional step)**

RESPONSE:

RP-70 Step 7.1.2.4 of the procedure in the Preliminary Design Report has been revised to incorporate the comment.

COMMENT:

RP-71 Section 7.1.3 - Step 7.1.3.1. If there is any way that this fixture can be installed while the cask is in the work area, then it should be done then. Installing such equipment underwater can be difficult, and results in wet cables that must be wiped down as they are removed from the pool. (If necessary, the fixture dimensions should be revised to allow dry installation.)

RESPONSE:

RP-71 As the closure lid will be removed in the pool, the lead-in fixture has also to be installed in the pool following lid removal.

COMMENT:

RP-72 **Step 7.1.3.2. The value of this step can not be determined. Failure of fit should occur only in the case of bow, twist or damage. This step should be deleted, or entered as an "option" that could be exercised in dry runs.**

RESPONSE:

RP-72 Step 7.1.3.2 will be specified as an option.

COMMENT:

RP-73 Step 7.1.3.3. "Verification" by underwater camera and lights should not be a "requirement" (Requirements arise by placing the step in the procedure) unless OCRWM intends to provide the camera and lights. Cameras and lights (for cameras) have not historically proven to be necessary.

RESPONSE:

RP-73 Step 7.1.3.3 will be changed to read: "verify visually that fuel assemblies are fully inserted. Use of underwater camera and light is optional if available."

COMMENT:

RP-74 Step 7.1.3.4. Removal is usually easier than installation. Consideration should be given to the fixture finish, as an aid to decontamination, and to elimination of crevices or pockets that trap water or contaminants.

RESPONSE:

RP-74 The fixture will be designed for ease of decontamination. A surface finish of 32 micro inches or better will be specified.

COMMENT:

RP-75 **Section 7.1.4 - Step 7.1.4.1. Visual inspection of these surfaces is adequate. Use of cameras and lights is optional if they are present in the pool. They are not required.**

RESPONSE:

RP-75 Step 7.1.4.1 will be changed to read: "Perform visual inspection to assure that no obstruction or debris are present on the cask closure lid flange surface."

COMMENT:

RP-76 Step 7.1.4.2. What configuration of which equipment is used to move the closure to the cask? A lid lift fixture is not described. Are slings to the cask handling crane hook proposed? If possible the closure lid could be attached by slings to the yoke (arms may need to be slightly longer). This would avoid repeat lifts into or out of the pool.

RESPONSE:

RP-76 The design of tooling to handle the closure lid is outside the current scope of work. It is envisioned that the closure lid pintle can be engaged with an external grapple attached to a post that is supported from the cask handling crane.

COMMENT:

RP-77 **Match marks to be used underwater should be "bright."**

RESPONSE:

RP-77 The match marks are painted using red paint in 0.500" wide x 0.060" deep machined grooves, as specified in Drawing 1988E43, Sheet 3.

COMMENT:

RP-78 Use of guide pins that do not extend above the lid will have very limited value. The tendency will be for the operator to "bounce" the lid on the tops of the pins. A suggested configuration would be to install one very long, and one long, guide pin in the threaded holes when the lid is removed. Cut the holes in the lid to the outer edge of the lid, making them "slots." This will allow the crane operator to see the engagement of the very long guide pin, and then see the engagement of the long one. Hand tightened "shipping bolts" would be installed in these holes during transport. (If already available in the pool, the camera would be used in this step.)

RESPONSE:

RP-78 Westinghouse will evaluate the suggestions during the final design and incorporate features that will enhance installation of the closure lid. The use of slots for the pins rather than circular holes will be incorporated.

COMMENT:

RP-79 **The closure lid drain or purge/gas sample port should be opened with a mating (unvalved) "Snaptite" to eliminate hydraulic lock between the cask and the lid. This will prevent water from washing particulate contamination into the seal area between the lid and cask.**

RESPONSE:

RP-79 Hydraulic locking between the closure lid and the cask will be prevented by incorporating flow grooves on the lid OD that is recessed in the cask cavity. These flow grooves can be sized to provide a larger flow area than possible with the drain or purge ports and will be incorporated in the final design.

COMMENT:

RP-80 **Step 7.1.4.5. Several bolts should be hand tightened in lid as the cask leaves the pool, to ensure that the lid stays on in the event of a problem in moving from the pool to the work stations.**

RESPONSE:

RP-80 A step will be added requiring hand tightening of several bolts prior to cask removal from the pool.

COMMENT:

RP-81 Step 7.1.4.6. The torque specified cannot be achieved without mechanical advantage provided by a special power tool locked against some structure that will not move. This tool must be provided. The analysis that specifically establishes this torque could not be found (Reference page 2-114). This is a very high torque value which cannot be obtained manually. The torque sequence and number of passes must also be specified.

RESPONSE:

RP-81 The torque value given in the Preliminary Design Report was over-estimated based on higher closure lid temperature estimates for the fire accident. The actual torque value is expected to be about 1000 ft.-lb. This will still require the use of a power tool. Analysis supporting the selection of the bolt preload and torque will be included in the final design report. The operating procedures in the Technical Manual will specify the torque sequence and number of passes.

COMMENT:

RP-82 **Step 7.1.4.7. This step should require that a pressurization line be attached to the purge/gas sample port and a hose to the drain port.**

RESPONSE:

RP-82 Step 7.1.4.7 will be revised to require that a pressurization line be attached to the purge/gas sample port and a hose to the drain port.

COMMENT:

RP-83 **Additional information will be needed for the draining, sampling, vacuum drying activities.**

RESPONSE:

RP-83 The information provided in the Preliminary Design Report was intended to provide a broad outline of the operating sequence rather than detailed step-by-step operating procedures. Detailed procedures are more appropriately included in the Technical Manual.

COMMENT:

RP-84 Section 7.1.5 - As an observation, this activity does not contain a "decontamination" step, nor does it remove the "shroud" installed in step 7.1.2.2.

RESPONSE:

RP-84 Decontamination was inadvertantly left out in the Preliminary Design Report. This step has been added. The cask will be removed from the bucket in the pool (the bucket does not leave the pool, but sits on top of a shelf, with the top end just above the surface of the pool).

COMMENT:

RP-85 **Step 7.1.5.1. The use of helium leak tests should only be used if leak rate determination to the 10^{-6} range is required. Pressure testing should be used if rate determination to the 10^{-3} range is satisfactory. It is difficult to understand technically why it should not be since only 35 psig is expected in normal use. Alternatives to helium leak testing should be investigated.**

RESPONSE:

RP-85 Westinghouse agrees that alternatives to helium leak testing should be investigated. The issue is one of being able to establish a suitable specific activity for the media which could potentially leak through a seal system having leak tightness less than that defined as "leak tight." When this is established other easier methods will be investigated and considered.

COMMENT:

RP-86 Section 7.2 - A seal ring, and a location on the cask for installing a seal ring, could not be found.

RESPONSE:

RP-86 A conceptual design of the cask seal ring was presented at the design review meeting and is included as Figure RP-86.1. The figure shows how the ring is mounted on the cask.

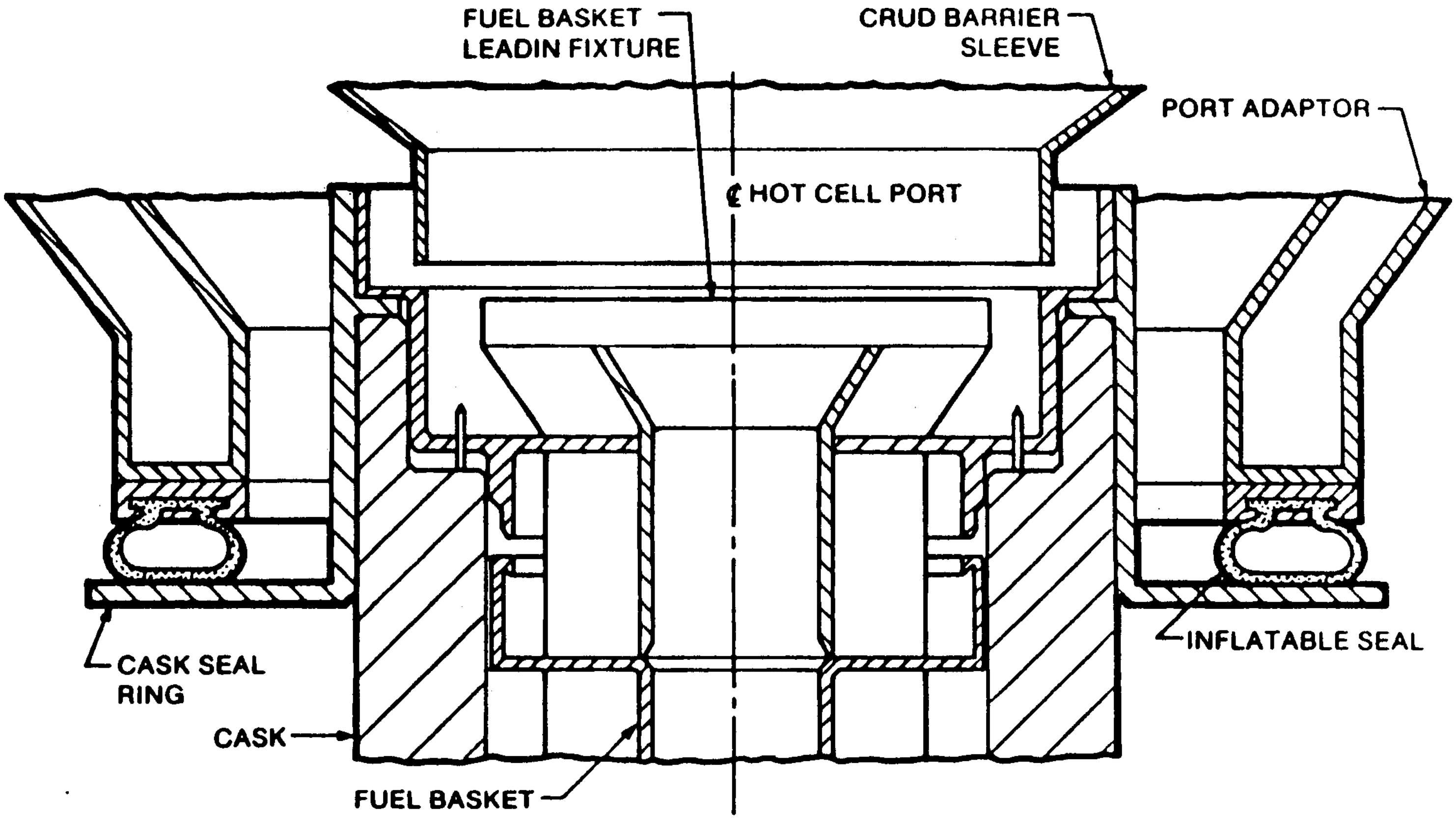

Figure RP-86.1 Cask Seal Ring

COMMENT:

RP-87 Section 7.3 - As an observation, since this package used Depleted Uranium as shielding, it can never be shipped as an EMPTY package.

RESPONSE:

RP-87 The word EMPTY was used in the context that the cask does not contain fuel. The word will be replaced with "unloaded cask."

COMMENT:

RP-88 **Section 7.4 - The personnel barrier should be reinstalled somewhere around 7.4.12.**

RESPONSE:

RP-88 Step 7.4.12 will be revised to specify reinstallation of the personnel barrier.

COMMENT:

RP-89 **No comments on the information provided, however, this section should have provided some information on the projected maintenance and annual inspection requirements for the proposed design. The information that has been provided has limited value.**

RESPONSE:

RP-89 A section on Acceptance Tests and Maintenance Program will be be included in the final design report.

COMMENT:

RP-90 It is noted that the use of Titanium for the cask structure will require that maintenance, inspection, repair and welding procedures be developed and provided for the material.

RESPONSE:

RP-90 Maintenance, inspection, repair and welding procedures for titanium will be including in the Technical Manual.

COMMENT:

RP-91 As an observation, what has become of the issue of the transmission of pin puncture forces through DU to the inner container wall?

RESPONSE:

RP-91 While no credit is taken for the structural strength in bending of the DU, the transmission of pin puncture forces through the DU to the inner containment wall was considered in the Preliminary Design Report (Section 2.7.2, pages 2-125 to 2-128).