SAFETY ANALYSIS REPORT
FOR
TRANSPORT OF THE YANKEE NUCLEAR POWER STATION
REACTOR PRESSURE VESSEL PACKAGE

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1.0 GENERAL INFORMATION

1.1 Introduction

The package described in this Safety Analysis Report (SAR) will be used for the transport of the Yankee Reactor Pressure Vessel (RPV) from the Yankee plant to a licensed, low level radioactive waste repository. The proposed model number for this radioactive materials package is YNPS RPV PACKAGE. The package will be used for transport one time only. The RPV was part of the Nuclear Steam Supply System for the Yankee Nuclear Power Station (YNPS), which operated safely and successfully from 1960 to 1992. Located in Rowe, Massachusetts, YNPS is owned by the Yankee Atomic Electric Company, which operates the plant under a Possession Only License.

A Decommissioning Plan for YNPS (Reference 1.3.1) was submitted to the NRC on December 20, 1993 and approved on February 14, 1995 (Reference 1.3.9). An Environmental Report for the decommissioning of YNPS also was submitted on December 20, 1993 (Reference 1.3.2). The waste packaging and transportation plan described in this SAR are consistent with the Decommissioning Plan and Decommissioning Environmental Report. The RPV removal operations will be performed at YNPS in accordance with the provisions of the Possession Only License and 10 CFR 50.59, "Changes, Tests, and Experiments" (Reference 1.3.3). Yankee Atomic Electric Company has prepared this SAR in support of an application for a Certificate of Compliance for Radioactive Materials Packages in accordance with 10 CFR 71 (Reference 1.3.4). The package consists of the RPV, an internal solidified waste package, concrete fill, and the single use shipping cask. The package will be transported in accordance with 10 CFR 71 and applicable Department of Transportation regulations.

1.2 Package Description

1.2.1 Packaging

The RPV single use shipping cask is a three-inch thick steel cylinder with an integral flat circular bottom and flat cover. A one-inch thick steel cylinder for supplemental shielding is located opposite the RPV core region. A cross sectional view of the cask and its contents is presented in Figure 1.2-1. Additional details of the cask are presented in Figures 1.2-2 through 1.2-5. The entire cask, except for the top cover and miscellaneous penetration covers, will be a single, welded integral unit. The top cover and miscellaneous penetration covers will also become an integral, welded part of the cask once the RPV is placed inside. The top cover has three functions. It provides shielding, is a part of the RPV lift fixture, and is the shipping cask cover. Refer to Figure 1.2-4 for a plan view of the cask cover.

The shipping cask cover will be bolted to the RPV using the upper threaded section of the existing RPV head studs, which are currently in protective storage at the plant. Prior to moving the RPV, the main coolant piping will be cut, and the RPV nozzles will be plugged with 8-inch thick carbon steel plugs. The RPV, with the solidified internal waste package inside, will

be lifted from containment and placed inside the cask while both the RPV and the shipping cask are in the vertical position. The cask cover will be welded to the cask. The entire inside of the RPV will then be filled with concrete having an average density of 25-30 pounds per cubic foot (pcf). The annulus space between the outside RPV and the inner surface of the cask will be filled with 75 - 85 pcf average density co

The package shielding will be provided by the steel cylindrical shell, nozzle plugs, and the encapsulated concrete. In addition to shielding, the concrete will immobilize the RPV within the cask, evenly distribute normal transportation loads, and preclude any possibility that the contents of the package could be removed from the cask. Combinations of steel and concrete similar to the RPV package have been used successfully to transport large radioactive waste packages.

The cask will be fabricated from ASTM A516, Grade 70 carbon steel using either ring forgings or rolled plate. Virtually all welding and other fabrication work will be performed in a qualified shop. Only the seal welds and the weld of the cover onto the cask will be made at YNPS. The top cover weld has been designed such that post weld heat treatment is not required. Fabrication will be performed in accordance with the provisions of the YAEC 10 CFR 71, Subpart H, Quality Assurance Program.

The gross weight of the package is approximately 656,000 pounds or 328 tons.

1.2.2 Operational Features

Once assembled, the RPV package has no operational features. Fill and vent ports in the cask and cask cover will be seal-welded and rendered inoperable prior to off-site cask transport. The reactor head studs will be covered by welded-in-place pipe caps. Lifting clevises on the package top cover will be removed prior to off-site transport. There are no valves, connections, piping, accessible openings, or seals associated with the RPV package.

1.2.3 Contents of Packaging

1.2.3.1 General

The RPV was fabricated primarily of carbon steel. The internal surfaces and Main Coolant (MC) nozzles are clad with stainless steel. The RPV is cylindrical with a 10'-5" outside diameter and an 8-inch wall thickness. It has a bottom hemispherical head. The upper section of the RPV consists of a thickened flange ring, nozzle ring, and 28 support lugs. During plant operation, the RPV closure head was bolted to the flange ring with two concentric, silver alloy O-rings providing sealing. The closure head will be disposed of separately and is not part of the RPV package. The O-rings will remain in place on the RPV. The reactor fuel and internals have been removed and will not be part of the RPV package. The overall length of the RPV is about 26'-9", not including the closure head. Refer to Figure 1.2-6 for a cross-sectional view of the RPV. The section of the RPV below the support lugs is insulated with two layers of calcium

silicate blocks, each 1½ inches thick, which contain asbestos. The insulation is enclosed by a stainless steel wrapper on the outside surface, which is held in place by 1/8"-inch thick by 1-inch wide stainless steel bands circling the RPV.

The insulation on the bottom hemispherical head is supported from 20 plates welded along the circumference of the RPV. These plates support a ring fabricated of steel angles. Radial bands extending from the ring to the apex of the hemispherical head support the insulation. The insulation is encased on the outside face by a stainless steel hardware cloth (10 wires/inch, 0.025-inch diameter). All of the RPV insulation below the support lugs is inaccessible and will remain inaccessible until after the RPV is lifted from its installed location. Even then, removal of the insulation would be radiation exposure intensive, therefore, it will remain on the RPV as part of the package.

An extensive campaign to clean the interior of the RPV was conducted after the cutting and packaging of the reactor internals was completed. A small volume (approximately 1.0 ft^3) of contaminated residue ("dross") composed of very fine metal particles, generally smaller than 5 microns in size, remains after the cleanup campaign. The dross is present only on the bottom head of the RPV. The volume of the dross was calculated based on a maximum depth at the center of the bottom head, tapering to a zero depth at the outer radius. The volume calculation is supported by visual observations of the interior of the bottom head recorded on video tape. The dry density of the dross is calculated to be about 2.7 g/cc (Reference 1.3.10).

Prior to removal of the RPV, the dross will be processed and solidified in its own container (solidification liner) within the RPV (Figure 1.2-1). The solidification process will be performed in accordance with a CNSI Process Control Program, and will homogeneously distribute the dross within a concrete matrix of known volume and density such that the 10 CFR 71 requirements for low specific activity (LSA) are met by the final waste form. The solidification process is discussed in more detail in Section 7.1.2. The solidified dross, solidification liner, mixer, pump, and associated hardware comprise the solidified internal waste package. The waste package will remain in the RPV and will be part of the RPV package.

Each main coolant nozzle will be sealed using an eight-inch thick shield plug. The plugs will be installed prior to lifting the RPV. Once the RPV is placed in the shipping cask, low density concrete (25-30 pcf average) will be used to fill the interior of the RPV. The low density concrete will stabilize the position of the solidified dross, fix any contamination on the internal surfaces of the RPV in place, and ensure that there is no free standing water in the RPV interior. Concrete (75-85 pcf average) will also be placed in the annulus between the exterior of the RPV and the shipping cask.

1.2.3.2 Activation Analysis

An activation analysis determined three-dimensional activation profiles by nuclide of the carbon steel RPV walls, stainless steel internal clad, and stainless steel insulation wrapper components. The analysis is based on the operating history of YNPS, detailed neutron flux and

spectra calculated from neutron transport methods, and activation cross sections from activation libraries associated with the ORIGEN code system (Reference 1.3.7).

The process for calculating the vessel activation was as follows:

1) Determine average neutron flux and spectrum over a particular region from the R-θ calculation.

2) Determine operational history for irradiation/decay of the region.

3) Determine the material composition including trace element (impurity) concentrations of the region.

4) Run ORIGEN-2 for 21 cycles of power operation and refueling outages, assuming average cycle power during power operation, using both thermal spectrum and fuel spectrum one-group activation cross sections. Nuclide decay was calculated for various times from the final shutdown date.

5) Weight results of each calculation from 4) based on the calculated spectrum in the region as determined by the R-θ calculation and determine region average activities by nuclide.

6) Perform R-Z transport calculation to obtain axial dependent thermal flux in the vessel region.

7) Normalize the thermal fluxes over the axial core height from 6) to determine relative thermal flux as a function of axial height.

This process was performed to calculate the azimuthally-averaged vessel activation by nuclide for the vessel and vessel clad regions in the beltline region (Table 1.2.1), as well as the average axial distribution of the activation products in the vessel (Figure 1.2-7). Steps 1) through 5) were also repeated 20 times to get detailed radial activation data for 20 individual radial regions through the vessel. Steps 6) and 7) were also repeated to get detailed axial distributions at 10 radial locations in the vessel.

Neutron transport calculations were performed using the DORT discrete ordinates transport code (Reference 1.3.5) in conjunction with the SAILOR 47-neutron group cross section library (Reference 1.3.6). Calculations were made in both fine mesh R-θ and R-Z geometries. Both geometric models extend radially from the core centerline to a point 10 inches into the reinforced concrete biological shield wall which surrounds the RPV and Neutron Shield Tank (see Figure 7.1-1). The R-θ model contains 198 radial mesh and 90 azimuthal mesh, based on a 45 degree modelling representation which describes one-eighth of the reactor core. The R-Z model contains 144 radial mesh and 234 axial mesh spanning approximately a 19 foot length of the vessel. R-θ calculations were based on a P_3 Legendre expansion of the scattering cross

sections and a symmetric S_{16} quadrature (160 angles). R-Z calculations are also based on a P_3 Legendre expansion of the scattering cross sections and a symmetric S_{10} quadrature (70 angles).

A conservative neutron source based on pinwise power distributions and relative assembly powers from YNPS operating Cycle 21, the final operating cycle, is assumed in the analysis. The Cycle 21 power distribution was chosen to determine the analytical fixed source distribution. The Cycle 21 distribution bounds all previous power operation distributions. This assumption therefore maximizes the calculated flux incident on the RPV.

Detailed pointwise flux information, determined from the R-θ calculations described above, were used to determine azimuthally averaged neutron fluxes and 47 group spectra over the beltline (active fuel region) of the RPV wall and internal stainless steel clad components. This flux data, in combination with plant operating history, was used with the ORIGEN 2.1 code (Reference 1.3.7) to determine nuclide specific activities for the RPV and clad. The operating history of the plant is based on 21 cycles of power operation and refueling shutdown time periods. Cycle average power levels are assumed during the irradiation periods. The base material compositions of the RPV and clad are established using nominal base material concentrations. Trace element concentrations are based on the average trace element concentrations from NUREG/CR-3474 (Reference 1.3.8). The NUREG trace element concentrations are based on average concentrations measured from samples of various RPV materials throughout the industry.

Table 1.2.1 contains the beltline average specific activities by nuclide for the RPV and clad regions. Activities for the insulation wrapper are not included since they are small in comparison to the RPV wall and clad, and will not affect characterization of the package. Activities of the insulation wrapper were derived for the shielding analysis and are discussed in Section 5.2. Various decay periods are presented with dates ranging from permanent plant shutdown at the end of Cycle 21 (EOC CY21 in Table 1.2.1) on October 1, 1991 through January 1, 2000. The specific activities are presented in curies/gram. Figure 1.2-7 shows the average axial thermal flux profile for the RPV. This profile is normalized over the beltline region so that it can be used in a manner consistent with the component activations from Table 1.2.1.

The activation products in the reactor vessel are uniformly distributed in the reactor vessel material in all spatial directions. Since the majority of the activation nuclides are produced by thermal neutron reactions, their spacial distribution can be approximated by the distribution of the thermal flux. The thermal neutron flux population at a particular vessel location is the primary parameter for determining the activation parent reactions and corresponding sister products, with changes in neutron spectrum and activation cross section through the vessel contributing in a secondary manner.

The normalized azimuthal distributions over the beltline region of the cobalt-59 reaction rate and the thermal flux, which were determined from the neutron fluxes from the R-θ geometry calculation, are shown in Figure 1.2-8 for the vessel inner and outer surfaces. The cobalt

reaction rate distribution is representative of the distributions of other activated nuclides in the vessel. This figure illustrates that the normalized cobalt-59 reaction rate (or cobalt-60 production rate) and the thermal flux are directly proportional and that there is only minimal variation in the azimuthal distributions from the inner surface to the outer surface of the vessel. The activation products are, therefore, uniformly distributed around the azimuthal coordinates of the vessel cylinder.

The axial distribution of the activation products, as determined from the neutron fluxes calculated from the R-Z geometry, was derived from the axial thermal flux profiles in the vessel region. Figure 1.2-9 shows the normalized thermal flux distributions of the inner and outer surfaces of the vessel over the active core height. As can be seen from this figure, the activation products are uniformly distributed in a smooth cosine shaped profile over the active core height. The profiles of the axial distribution of the activation products vary only slightly between the inner and outer surfaces of the vessel wall. The axial distribution profiles of other radial locations within the vessel wall are bounded by the axial profiles of Figure 1.2-9.

The total activity through the reactor vessel in the radial direction in the beltline region of the vessel is shown in Figure 1.2-10. The radial distribution was calculated from 20 evenly-spaced mesh points from the R-θ geometry and represents the average activity over the active core height of the reactor. As shown in Figure 1.2-10, the activity is uniformly distributed in a smooth profile through the vessel wall. The reduction in activity from the inner to outer vessel wall is due primarily to neutron absorption in the vessel wall steel.

1.2.3.2.1 Comparison to Measured Data

Measurements of the dose rates on the inner and outer surfaces of the vessel were made to assess the adequacy of the calculated activity sources in the vessel which were assumed in the vessel classification and shielding analyses.

Measurements of the dose rates on the inner surface of the vessel were performed at 16 evenly-spaced intervals around the inner circumference of the vessel in one foot axial increments from the main coolant nozzle to near the bottom of the vessel. Measurements were performed with the vessel empty, except for water, on two separate occasions, at a distance of six inches from the vessel surface. The first measurements (1/29/94) were taken prior to completion of segmentation and the cleanup effort performed for the vessel internals removal program. The second measurement (7/20/94) was performed after the cleanup program.

Results of the above measurements are summarized in Figure 1.2-11. This figure shows the azimuthal variation of the measured dose rates averaged over the beltline region of the vessel. This figure shows consistent spatial variation of the measured doses from the two measurements with the second measurement showing somewhat higher absolute values. Comparison of the normalized measured dose rates to the normalized cobalt-59 reaction rate (Figure 1.2-8 expanded to 360 degrees) is shown in Figure 1.2-12. The comparison shows excellent agreement in the spatial variation of the measured and expected dose profiles with only

minor variation for different quadrants. On an absolute basis, the average dose rates for the first two measurements over the beltline region of the vessel were 50.9 R/hr and 60.0 R/hr, respectively. Based on the calculated vessel source over a similar region the calculated dose rate was 87 R/hr (see Section 5.0). Comparison of the axial variation shown in Figure 1.2-13 also shows good agreement between the measured and predicted distributions. These comparisons provide confidence that the calculations appropriately modelled the reactor source and vessel regions and that the calculated source at the inner surface was conservative.

Dose rate measurements were also made on the exterior of the vessel on 1/6/95. Contact dose rates on the outer surface of the reactor vessel insulation were taken at two azimuthal locations on the outer circumference of the vessel. Only two measurements were taken due to access restrictions in the reactor cavity region and personnel ALARA concerns. These measurements provided axial distribution information at two locations and provided absolute dose rates which could be used to ascertain peak location dose rates. Measurements taken on the outside of the vessel were corrected from the measurement location to the peak azimuthal location by using the predicted cobalt-60 production rate distribution on the outside of the reactor vessel. Figure 1.2-14 show the results of the calculation with a conservative bounding curve applied to the corrected measurements. This bounding curve represents the maximum expected outer surface dose rates at the peak azimuthal location. As can be seen, the maximum value for all locations is conservatively bounded at 8.0 R/hr.

The beltline averaged calculated Co^{60} source as a function of radial position in the vessel was used as the input source in a one-dimensional gamma ray transport calculation using the DORT code to calculate contact dose rates. Two calculations were performing using vessel beltline average sources based on dates of 1/1/94 (assumed in shielding evaluation) and 1/1/95 (~ measurement date). The average dose rates from the above calculations were then expanded assuming the axial distribution on the outside of the vessel shown in Figure 1.2-9. The results of this process superimposed on the measured data previously presented is provided in Figure 1.2-15. The measured data at the expected peak location is clearly bounded by the calculated data. This provides reasonable assurance that the calculated vessel source, when used to calculate dose rates at the outside of the vessel, has been properly distributed and is clearly conservative relative to measured values. The shielding analysis calculated dose rate has also been included on this figure for informational purposes, with further discussion provided in Section 5.0.

1.2.3.3 <u>Source Characterization</u>

The source characterization assumes that the activity concentration at any given position is directly proportional to the flux at that position. Thus, the activity concentration will show the same degree and magnitude of variation over the length of the RPV as that displayed in Figure 1.2-7, with the maximum activity concentration located in an approximately 3.3-foot high region within the RPV beltline. The data in Figure 1.2-7 were used to develop an average flux ratio value that served as an adjustment factor for the beltline average activity concentrations.

Source characterization of the RPV was based on the activity concentrations calculated for the RPV wall components in the activation analysis (see Section 1.2.3.2). The activity concentrations for the clad and carbon steel components of the RPV wall as of January 1, 1994 were decay-adjusted to July 1, 1995, and "blended" based on relative thickness to obtain effective average activity concentrations for the RPV beltline. The effective average activity concentrations for the beltline were adjusted based on the analysis of the thermal neutron flux profile data for the region between 241.1 cm and 636.2 cm above the bottom of the RPV to obtain the average activity concentrations for the RPV as a whole. The total RPV activity due to neutron activation of the RPV (Table 1.2.2) was determined by multiplying the average activity concentrations for the RPV as a whole by the total RPV mass (300,000 pounds). The total neutron induced activity is estimated to be 3197 Ci (as of July 1, 1995) which is distributed among 14 radionuclides. Approximately 99 percent of the radioactivity attributable to neutron activation is due to the presence of Co^{60}, Fe^{55}, and Ni^{63}.

Characterization of the RPV also accounts for an estimated 8.23 Ci due to internal and external surface contamination. Activity from surface contamination was calculated as follows. The isotopic distribution of internal surface contamination is based on the laboratory analysis of the surface of a clip taken from inside the reactor vessel in September 1993. The magnitude of the activity of the internal surface contamination was determined by applying the clip's isotopic distribution per cm^2 to the reactor vessel internal surface area (7.92×10^5 cm^2) and adjusting for radioactive decay. The isotopic distribution of contamination of the external surface area of the vessel is based on the distribution presented in Column 3 (headed "Remaining Systems and Structures") of Table 3.1-2 (entitled "Nuclide Distributions for Systems and Structures") of the YNPS Decommissioning Plan (SAR Reference 1.3.1). The magnitude of the activity of the external surface contamination was determined by applying the distribution from Table 3.1-2 to the external surface area of the reactor vessel (8.66×10^5 cm^2) and adjusting for radioactive decay. The total activity for surface contamination was obtained by summing the contributions from internal and external surface contamination. The surface contamination activity is distributed among 31 radionuclides (Table 1.2.2). Six nuclides (Co^{60}, Fe^{55}, Ni^{63}, Cs^{137}, Pu^{241}, and Sr^{90}) account for 98 percent of the surface contamination radioactivity. The total activity for the RPV (i.e., the sum of activity due to neutron activation and surface contamination) is 3205 Ci as of July 1, 1995.

The total RPV activity for each nuclide was used to evaluate the RPV against A_2 values given in Table A-1 of 10 CFR 71. The maximum activity concentrations for the 3.3-foot high region within the beltline of the RPV with maximum activity concentration were compared to the allowable concentration limits for LSA material to provide a bounding assessment for all regions of the RPV. The allowable concentration for each isotope was determined using the definition of LSA material provided in 10 CFR 71.4.

The dross has been characterized from sample analysis, dose rate measurements, and calculations. The radionuclide distribution applied to the dross was based on a sample analysis performed by the Yankee Atomic Environmental Laboratory on July 18, 1994.

The total activity of the dross is calculated to be 1419 Ci (as of July 1, 1995) and, as shown in Table 1.2.2, is distributed among 15 radionuclides. Approximately 99.75 percent of the activity is due to the presence of Co^{60}, Fe^{55}, Ni^{63}, and Mn^{54}.

The isotopic distribution of the RPV and of the solidified dross, decay adjusted to July 1, 1995, are shown in Tables 1.2.3 and 1.2.4, respectively. These tables also show the A_2 value from 10 CFR 71, Appendix A, and the allowable concentration per gram of contents for each isotope. The allowable concentration for each isotope was determined using the definition of LSA material provided in 10 CFR 71.4. The specific activity of the solidified dross is based on a final waste form with a volume of 175 ft^3 and a density of 100 pcf.

The total activity in the RPV package, which includes activation of the RPV walls, surface contamination and activity of the dross material, as of July 1, 1995, is 4624 Ci. The isotopic concentrations in the RPV are approximately 35 percent of the LSA limit and the concentrations in the solidified concrete matrix containing the dross are approximately 61 percent of the LSA limit. In each case, the radioactivity content exceeds the Type A quantity limit, i.e. the summation of the A_2 fractions is greater than 1.0. The contents of the RPV package are therefore classified as "greater than Type A quantity, LSA material." The RPV package will be transported as exclusive use.

Therefore, the RPV package, consisting of the RPV, internal solidified waste package containing the dross and mixing hardware, concrete, and shipping cask, is exempted from the requirements of 10 CFR 71.51 by 10 CFR 71.52.

1.3 References

1.3.1 Letter to USNRC from YAEC (BYR 93-087), "Decommissioning Plan for Yankee Nuclear Power Station," December 20, 1993.

1.3.2 Letter to USNRC from YAEC (BYR 93-086), "Environmental Report for the Decommissioning of Yankee Nuclear Power Station," December 20, 1993.

1.3.3 USNRC Rules and Regulations, Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities."

1.3.4 USNRC Rules and Regulations, Title 10, Part 71, "Packaging and Transport of Radioactive Material," November, 1992.

1.3.5 CCC-484, "DORT Two-Dimensional Discrete Ordinates Transport Code," Oak Ridge National Laboratory, November, 1989.

1.3.6 DLC-76, "SAILOR Coupled, Self-Shielded, 47-Neutron, 20 Gamma-Ray, P_3, Cross Section Library for Light Water Reactors," G.L. Simmons and R. Roussin, March, 1983.

1.3.7 CCC-371, "ORIGEN 2.1 Isotope Generation and Depletion Code, Matrix Exponential Method," Oak Ridge National Laboratory, August 1991.

1.3.8 NUREG/CR-3474, "Long Lived Activation Products in Reactor Materials," J.C. Evans, et. al, August, 1984.

1.3.9 Letter to YAEC from USNRC (NYR 95-016), "Order Approving the Decommissioning of the Yankee Nuclear Power Station," dated February 14, 1995.

1.3.10 YAEC, Calculation No. YRC-1071, "Density of Metal Dross," dated January 23, 1995.

Table 1.2.1

Summary of RPV and Clad Activations

NUCLIDE / DATE	EOC CY21[a] (Ci/g)	01/01/94 (Ci/g)	01/01/00 (Ci/g)
REACTOR PRESSURE VESSEL			
H3	.165E-06	.145E-06	.104E-06
C14	.293E-08	.293E-08	.292E-08
Mn54	.120E-04	.194E-05	.150E-07
Fe55	.165E-03	.907E-04	.183E-04
Co60	.122E-04	.907E-05	.412E-05
Ni59	.175E-08	.175E-08	.175E-08
Ni63	.248E-06	.244E-06	.233E-06
Sr90	.317E-18	.300E-18	.258E-18
Nb94	.687E-10	.687E-10	.687E-10
Tc99	.161E-09	.161E-09	.161E-09
Cs137	.143E-25	.143E-25	.143E-25
Eu152	.433E-07	.386E-07	.285E-07
Eu154	.435E-08	.363E-08	.224E-08
Eu155	.404E-10	.295E-10	.128E-10
TOTAL	.584E-03	.102E-03	.228E-04
REACTOR VESSEL CLAD			
H3	.654E-06	.577E-06	.412E-06
C14	.274E-06	.274E-06	.273E-06
Mn54	.184E-04	.297E-05	.230E-07
Fe55	.965E-03	.529E-03	.107E-03
Co60	.684E-03	.508E-03	.231E-03
Ni59	.713E-06	.713E-06	.713E-06

Table 1.2.1

(Continued)

NUCLIDE / DATE	EOC CY21[(a)] (Ci/g)	01/01/94 (Ci/g)	01/01/00 (Ci/g)
Ni63	.925E-04	.910E-04	.869E-04
Sr90	.266E-17	.252E-17	.219E-17
Nb94	.118E-08	.118E-08	.118E-08
Tc99	.183E-09	.184E-09	.183E-09
Cs137	.819E-21	.819E-21	.819E-21
Eu152	.113E-06	.101E-06	.745E-07
Eu154	.165E-07	.138E-07	.848E-08
Eu155	.140E-08	.102E-08	.443E-09
TOTAL	.787E-02	.113E-02	.426E-03

NOTES:

(a) EOC CY21 date is October 1, 1991.

Table 1.2.2

Distribution of Radioactivity in RPV Package

Nuclide	Radioactivity (Ci)[a]				
	Reactor Vessel			Dross	Package Total
	Neutron Activation	Surface Contamination	Total		
H3	5.543E+00	---	5.543E+00	---	5.543E+00
C14	2.629E-01	1.807E-03	2.647E-01	8.462E-03	2.732E-01
Cr51	---	7.704E-20	7.704E-20	---	7.704E-20
Mn54	2.234E+01	2.310E-02	2.236E+01	1.426E+01	3.662E+01
Fe55	2.591E+03	2.485E+00	2.593E+03	7.502E+02	3.343E+03
Fe59	---	9.188E-15	9.188E-15	---	9.188E-15
Co57	---	8.300E-04	8.300E-04	1.749E-02	1.832E-02
Co58	---	3.109E-11	3.109E-11	---	3.109E-11
Co60	5.180E+02	3.791E+00	5.218E+02	5.033E+02	1.025E+03
Ni59	4.524E-01	1.511E-07	4.524E-01	---	4.524E-01
Ni63	5.790E+01	8.610E-01	5.876E+01	1.496E+02	2.084E+02
Zn65	---	---	---	1.193E+00	1.193E+00
Sr89	---	1.800E-06	1.800E-06	---	1.800E-06
Sr90	1.271E-11	3.006E-01	3.006E-01	5.013E-02	3.507E-01
Zr95	---	1.710E-12	1.710E-12	---	1.710E-12
Nb94	3.346E-03	---	3.346E-03	---	3.346E-03
Nb95	---	7.564E-13	7.564E-13	---	7.564E-13
Tc99	6.454E-03	---	6.454E-03	---	6.454E-03
Ru103	---	7.786E-17	7.786E-17	---	7.786E-17
Ru106	---	1.969E-02	1.969E-02	---	1.969E-02
Ag108m	---	3.945E-09	3.945E-09	---	3.945E-09

Table 1.2.2

(Continued)

Nuclide	Radioactivity (Ci)[a]				
	Reactor Vessel			Dross	Package Total
	Neutron Activation	Surface Contamination	Total		
Ag110m	---	2.916E-03	2.916E-03	---	2.916E-03
Sb124	---	8.669E-13	8.669E-13	---	8.669E-13
Sb125	---	3.869E-02	3.869E-02	---	3.869E-02
Cs134	---	1.856E-02	1.856E-02	---	1.856E-02
Cs137	4.259E-16	3.396E-01	3.396E-01	2.728E-01	6.124E-01
Ce141	---	1.086E-19	1.086E-19	---	1.086E-19
Ce144	---	1.927E-02	1.927E-02	---	1.927E-02
Eu152	1.454E+00	---	1.454E+00	---	1.454E+00
Eu154	1.412E-01	---	1.412E-01	---	1.412E-01
Eu155	9.647E-04	---	9.647E-04	---	9.647E-04
Pu238	---	3.395E-03	3.395E-03	1.424E-03	4.819E-03
Pu239/40	---	8.455E-03	8.455E-03	3.893E-03	1.235E-02
Pu241	---	3.069E-01	3.069E-01	1.257E-01	4.326E-01
Am241	---	5.034E-03	5.034E-03	3.219E-03	8.253E-03
Cm242	---	2.521E-05	2.521E-05	6.160E-05	8.681E-05
Cm243/44	---	2.034E-03	2.034E-03	1.761E-03	3.795E-03
Total:	3.197E+03	8.228E+00	3.205E+03	1.419E+03	4.624E+03

NOTES:

(a) Radioactivity has been decay-adjusted to July 1, 1995.

<u>Table 1.2.3</u>

<u>Classification of the Reactor Pressure Vessel</u>

Isotope	RPV Activity (Ci)[a]	RPV Specific Activity (mCi/g)	A_2 Value (Ci)	LSA Limit (mCi/g)	Fraction LSA Limit	Fraction A_2 Value
H3	5.543E+00	1.676E-04	2.000e+01	3.000E-01	5.587E-04	2.772E-01
C14	2.647E-01	8.069E-06	6.000E+01	3.000E-01	2.690E-05	4.412E-03
Fe55	2.593E+03	7.851E-02	1.000E+03	3.000E-01	2.617E-01	2.593E+00
Ni59	4.524E-01	1.368E-05	9.000E+02	3.000E-01	4.560E-05	5.027E-04
Ni63	5.876E+01	1.807E-03	1.000E+02	3.000E-01	6.023E-03	5.876E-01
Sr89	1.800E-06	1.169E-10	1.000E+01	3.000E-01	3.897E-10	1.300E-07
Sr90	3.006E-01	1.953E-05	4.000E-01	5.000E-03	3.906E-03	7.515E-01
Pu238	3.395E-03	2.206E-07	3.000E-03	1.000E-04	2.206E-03	1.132E+00
Pu239/40	8.455E-03	5.494E-07	2.000E-03	1.000E-04	5.494E-03	4.228E+00
Pu241	3.069E-01	1.994E-05	1.000E-01	5.000E-03	3.988E-03	3.069E+00
Am241	5.034E-03	3.271E-07	8.000E-03	1.000E-04	3.271E-03	6.293E-01
Cm242	2.521E-05	1.638E-09	2.000E-01	5.000E-03	3.276E-07	1.261E-04
Cm243/44	2.034E-03	1.321E-07	9.000E-03	1.000E-04	1.321E-03	2.260E-01
Cr51	7.704E-20	5.006E-24	6.000E+02	3.000E-01	1.669E-23	1.284E-22
Mn54	2.236E+01	6.772E-04	2.000E+01	3.000E-01	2.257E-03	1.118E+00
Fe59	9.188E-15	5.970E-19	1.000E+01	3.000E-01	1.990E-18	9.188E-16
Co57	8.300E-04	5.393E-08	9.000E+01	3.000E-01	1.798E-07	9.222E-06
Co58	3.109E-11	2.020E-15	2.000E+01	3.000E-01	6.733E-15	1.555E-12
Co60	5.218E+02	1.592E-02	7.000E+00	3.000E-01	5.305E-02	7.454E+01
Zr95	1.710E-12	1.111E-16	2.000E+01	3.000E-01	3.703E-16	8.550E-14
Nb94	3.346E-03	1.012E-07	5.000E-02	1.000E-04	1.012E-03	6.692E-02
Nb95	7.564E-13	4.915E-17	2.000E+01	3.000E-01	1.638E-16	3.782E-14
Tc99	6.454E-03	1.952E-07	2.500E+01	3.000E-01	6.507E-07	2.582E-04
Ru103	7.[illegible]-17	5.059E-2[illegible]	2.500E+01	3.000E-01	1.686E-20	3.114E-18
Ru106	1.96[illegible]	1.279E-06	7.000E+00	3.000E-01	4.263E-06	2.813E-03

Table 1.2.3

(Continued)

Isotope	RPV Activity (Ci)[a]	RPV Specific Activity (mCi/g)	A_2 Value (Ci)	LSA Limit (mCi/g)	Fraction LSA Limit	Fraction A_2 Value
Ag108m	3.945E-09	2.563E-13	1.000E+03	3.000E-01	8.543E-13	3.945E-12
Ag110m	2.916E-03	1.895E-07	7.000E+00	3.000E-01	6.317E-07	4.161E-04
Sb124	8.669E-13	5.633E-17	5.000E+00	3.000E-01	1.878E-16	1.734E-13
Sb125	3.869E-02	2.514E-06	2.500E+01	3.000E-01	8.380E-06	1.548E-03
Cs134	1.856E-02	1.206E-06	1.000E+01	3.000E-01	4.020E-06	1.856E-03
Cs137	3.396E-01	2.207E-05	1.000E+01	3.000E-01	7.357E-05	3.396E-02
Ce141	1.086E-19	7.056E-24	2.500E+01	3.000E-01	2.352E-23	4.344E-21
Ce144	1.927E-02	1.252E-06	7.000E+00	3.000E-01	4.173E-06	2.753E-03
Eu152	1.454E+00	4.397E-05	1.000E+01	3.000E-01	1.466E-04	1.454E-01
Eu154	1.412E-01	4.271E-06	5.000E+00	3.000E-01	1.424E-05	2.824E-02
Eu155	9.647E-04	2.918E-08	6.000E+01	3.000E-01	9.727E-08	1.608E-05
Total	3.205E+03	9.722E-02	---	---	3.450E-01	8.927E+01

NOTES:

(a) Activity as of July 1, 1995.

Table 1.2.4

Classification of Solidified Dross

Isotope	Dross Activity (Ci)[a]	Dross Specific Activity (mCi/g)	A_2 Value	LSA Limit (mCi/g)	Fraction LSA Limit	Fraction A_2 Value
C14	8.462E-03	1.066E-06	6.000E+01	3.000E-01	3.553E-06	1.410E-04
Fe55	7.502E+02	9.451E-02	1.000E+03	3.000E-01	3.150E-01	7.502E-01
Ni63	1.496E+02	1.885E-02	1.000E+02	3.000E-01	6.282E-02	1.496E+00
Sr90	5.013E-02	6.315E-06	4.000E-01	5.000E-03	1.263E-03	1.253E-01
Pu238	1.424E-03	1.794E-07	3.000E-03	1.000E-04	1.794E-03	4.747E-01
Pu239/40	3.893E-03	4.904E-07	2.000E-03	1.000E-04	4.904E-03	1.947E+00
Pu241	1.257E-01	1.584E-05	1.000E-01	5.000E-03	3.167E-03	1.257E+00
Am241	3.219E-03	4.055E-07	8.000E-03	1.000E-04	4.055E-03	4.024E-01
Cm242	6.160E-05	7.760E-09	2.000E-01	5.000E-03	1.552E-06	3.080E-04
Cm243/44	1.761E-03	2.218E-07	9.000E-03	1.000E-04	2.218E-03	1.957E-01
Mn54	1.426E+01	1.796E-03	2.000E+01	3.000E-01	5.988E-03	7.130E-01
Co57	1.749E-02	2.203E-06	9.000E+01	3.000E-01	7.345E-06	1.943E-04
Co60	5.033E+02	6.340E-02	7.000E+00	3.000E-01	2.113E-01	7.190E+01
Zn65	1.193E+00	1.503E-04	3.000E+01	3.000E-01	5.010E-04	3.977E-02
Cs137	2.728E-01	3.437E-05	1.000E+01	3.000E-01	1.146E-04	2.728E-02
Total	1.419E+03	1.788E-01	---	---	6.132E-01	7.933E+01

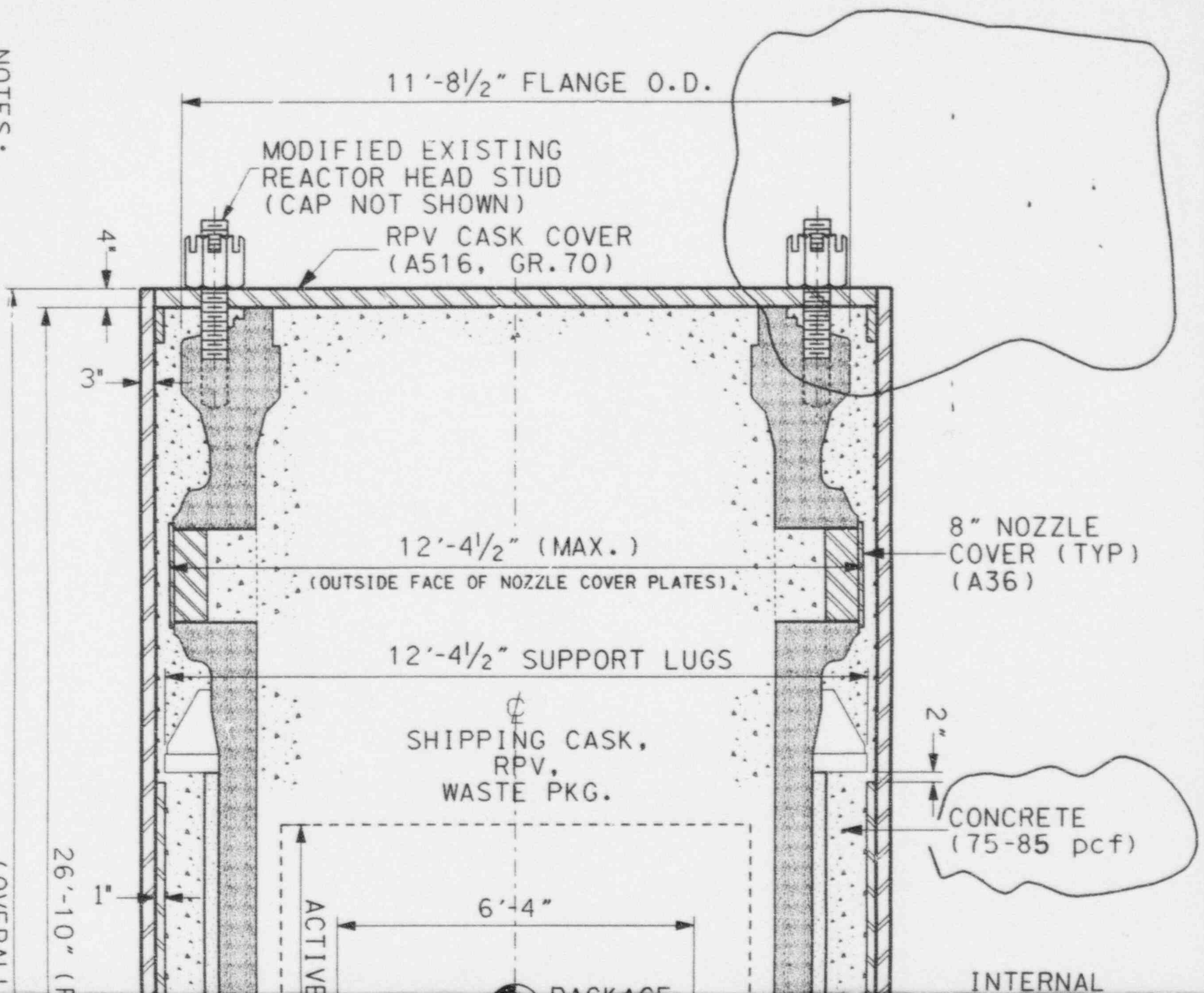
NOTES:

(a) Activity as of July 1, 1995.

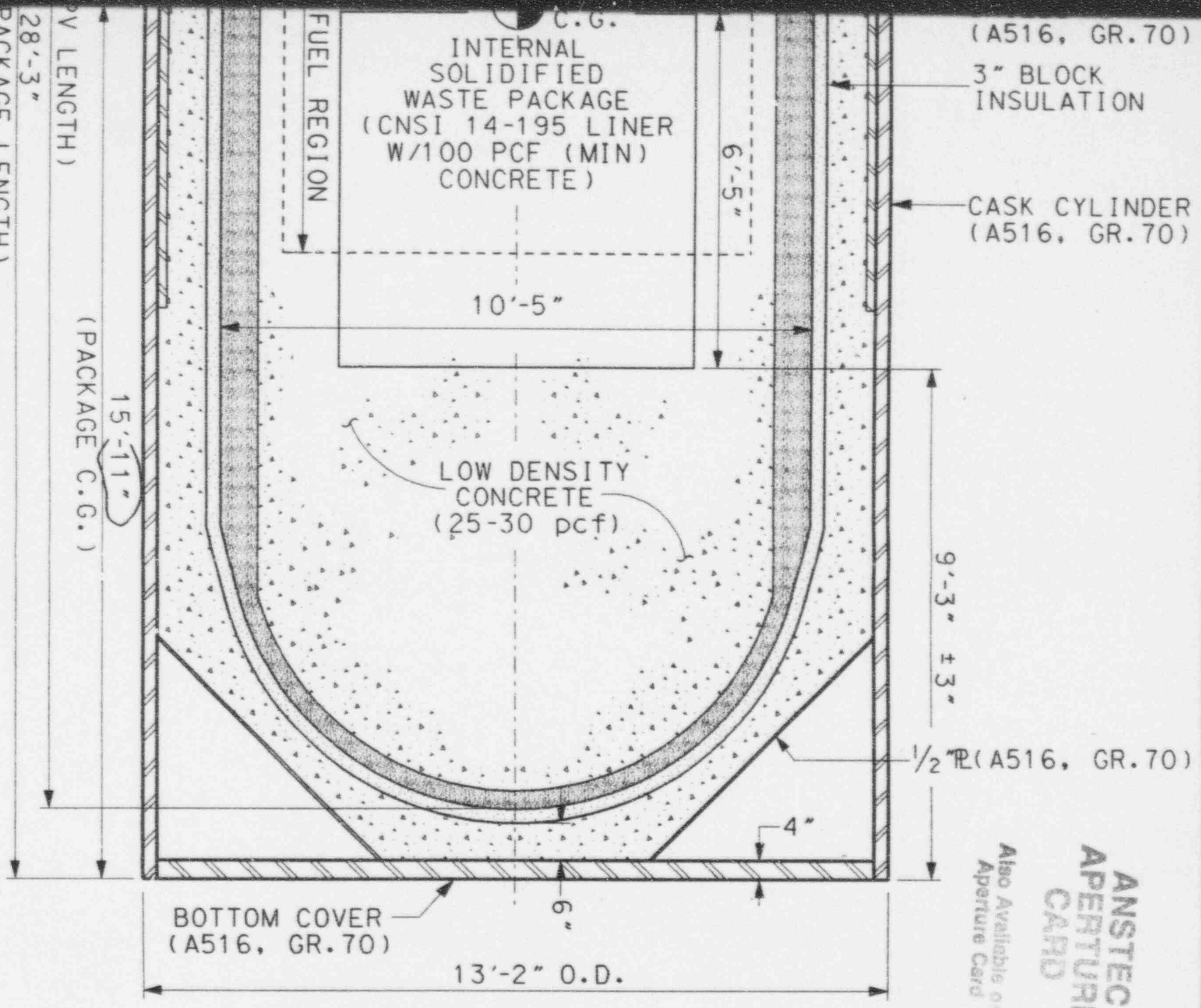
e:\dgn\yrdecom\yrb90005.dgn

NOTES:

1. SHIPPING CASK MATERIALS OF CONSTRUCTION PROVIDED IN PARENTHESES ( ).
2. LIFTING CLEVIS TO BE REMOVED PRIOR TO OFF-SITE TRANSPORT OF LOADED CASK.
3. MAT'LS OF CONST., 14-195 LINER: TOP & BOTTOM - A36 (5/16"), SIDES - A569 (11 GA.)
4. MAXIMUM LOADED PACKAGE WEIGHT: 328 TONS.



REV	DESCRIPTION
1	ADD CONC DENSITY FOR INT. WASTE PKG
2	REVISED FOR CONCRETE DENSITY AND CLEVIS CHGS.



YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION

FOR: YANKEE ATOMIC ELECTRIC CO.
ROWE, MA

TITLE: FIG 1.2-1-YNPS RPV PACKAGE, ELEVATION CROSS SECTION

JOB NO.

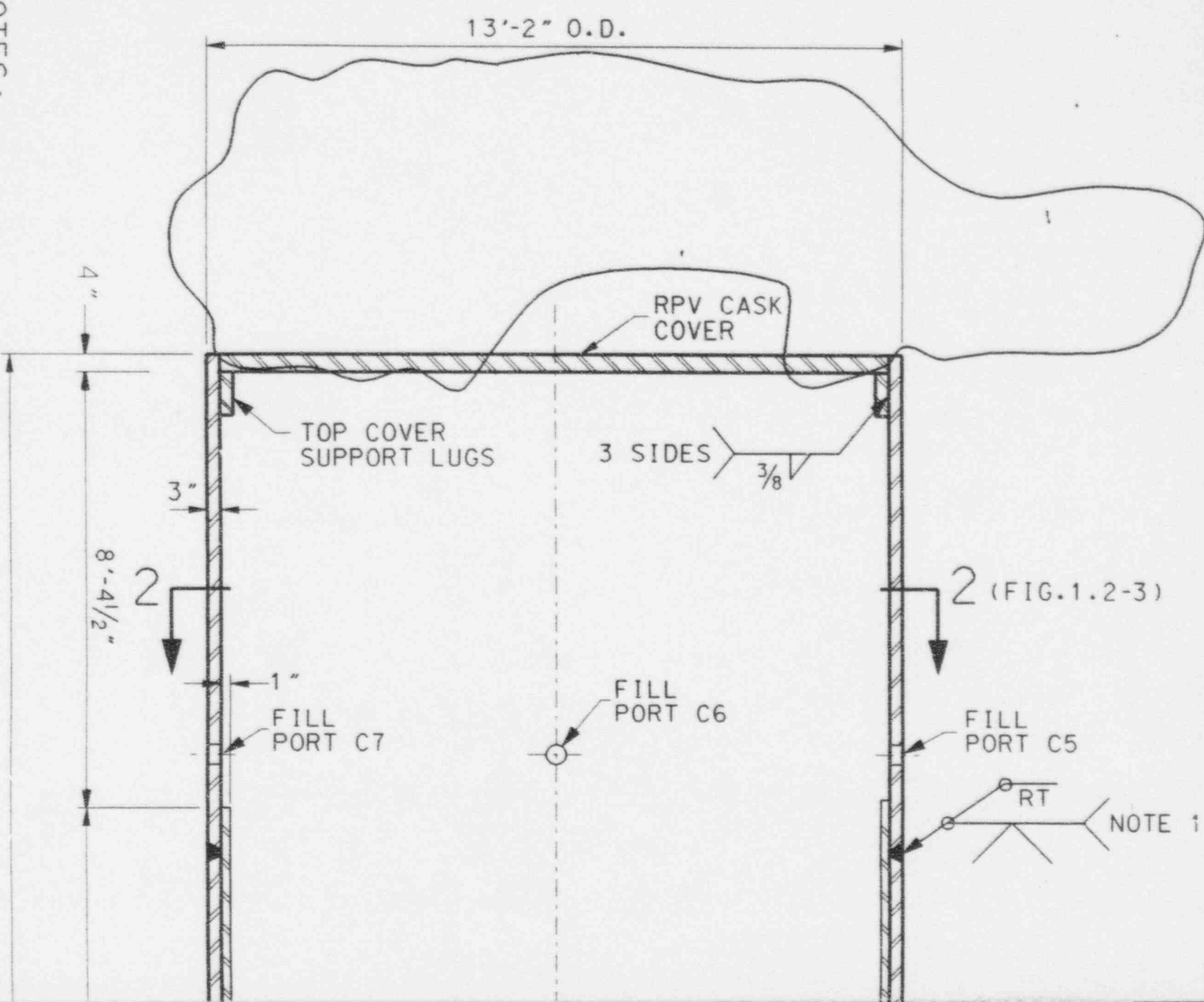
DWG. NO. YR-B-90-005

DRL	JAK
12-7-95	12-8-95
DRL 11/18/96	JAK 11/18/96
CHKD.	APPD.

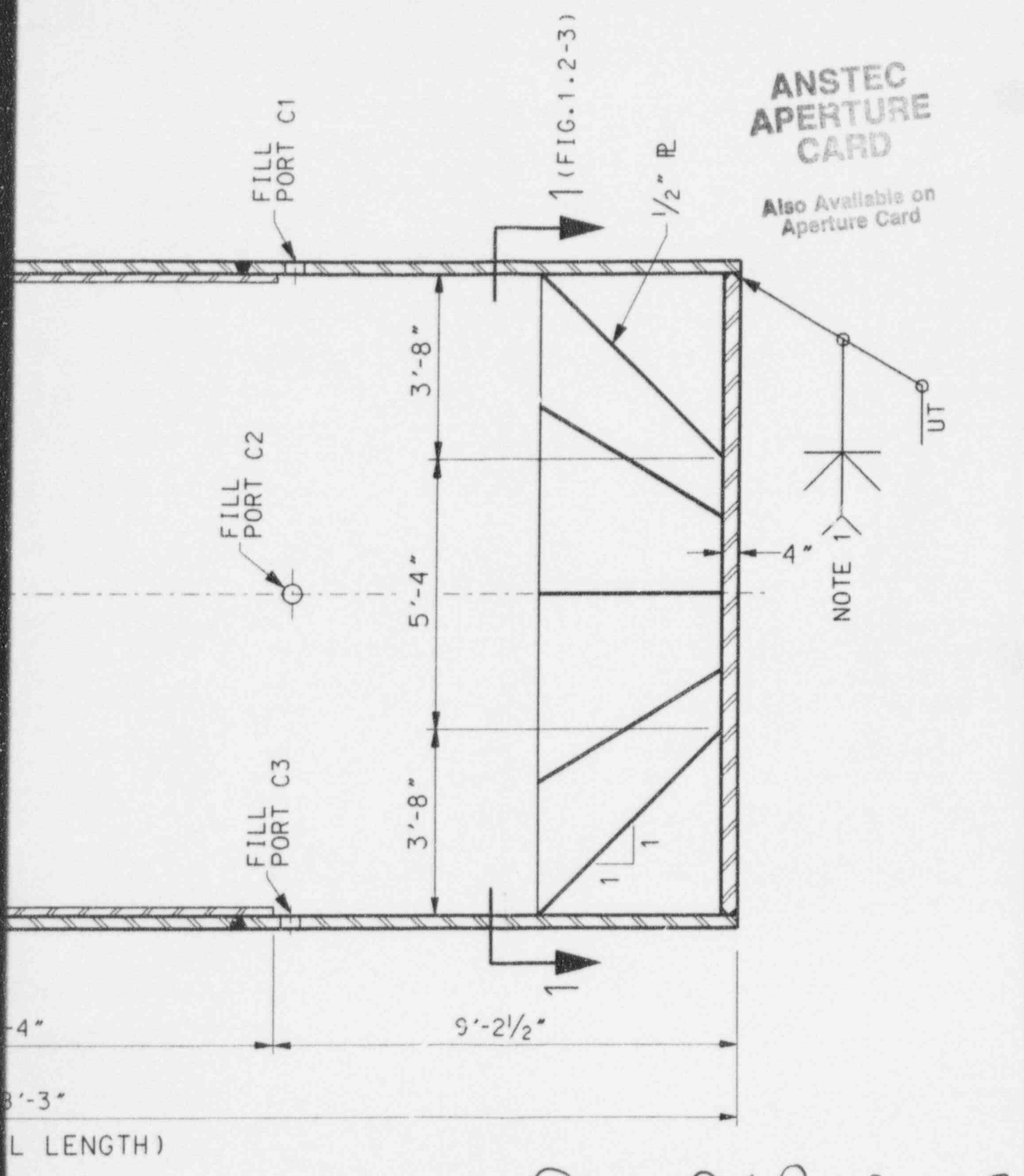
9611190240-01

NOTES:

1. CONTAINMENT WELD. CONFIGURATION IS FOR ILLUSTRATION ONLY. WELD WILL BE FULL PENETRATION IN ACCORDANCE WITH ASME VIII WITH FINAL CONFIGURATION DETERMINED BY FABRICATOR.
2. ALL ITEMS, EXCEPT LIFTING CLEVISES, TO BE FABRICATED FROM A516, GR.70 MATERIAL.
3. LIFTING CLEVIS (NOT SHOWN), TO BE REMOVED PRIOR TO OFF-SITE TRANSPORT OF LOADED PACKAGE.

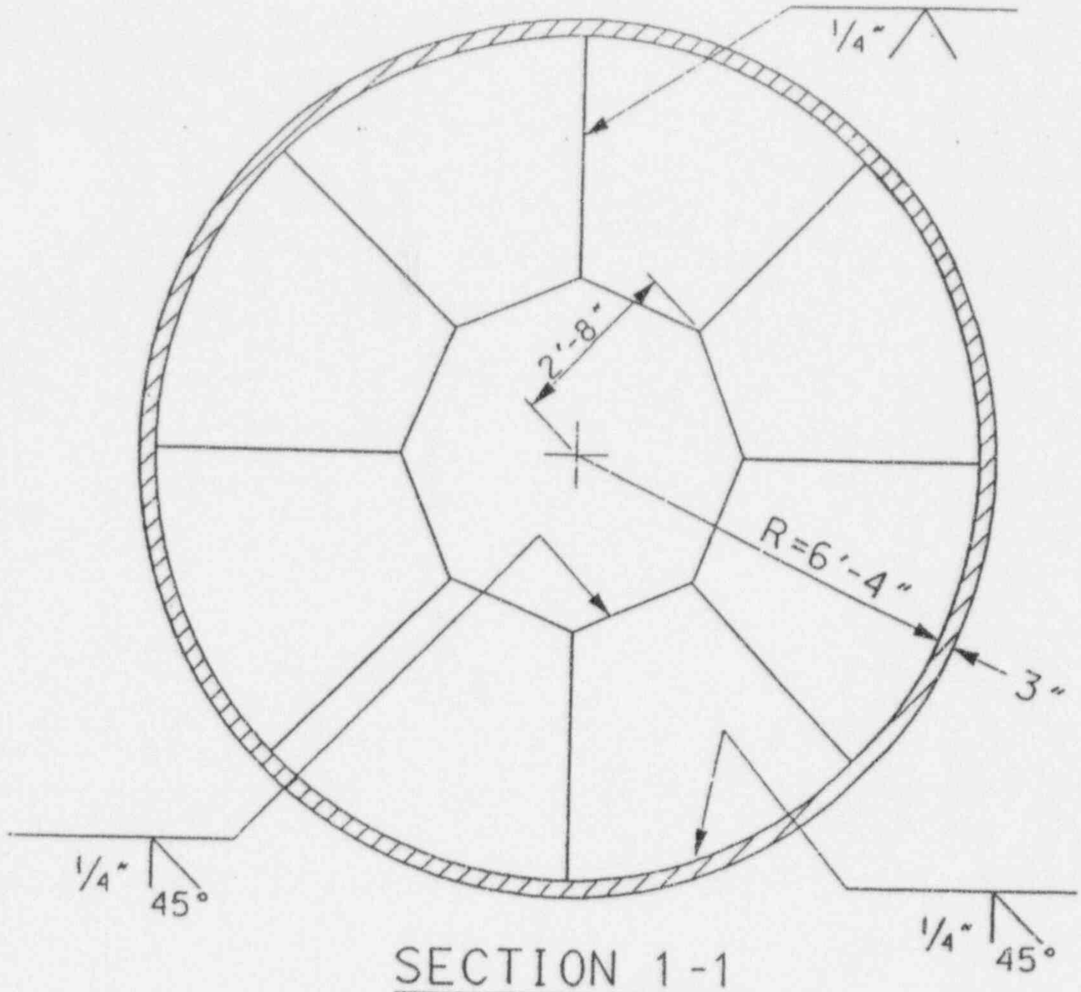


REV	DESCRIPTION		
1	REVISED FOR CHANGE IN LIFTING CLEVIS	8-16	[illegible]
0	ORIGINAL ISSUE PER W.O. DD34	B	

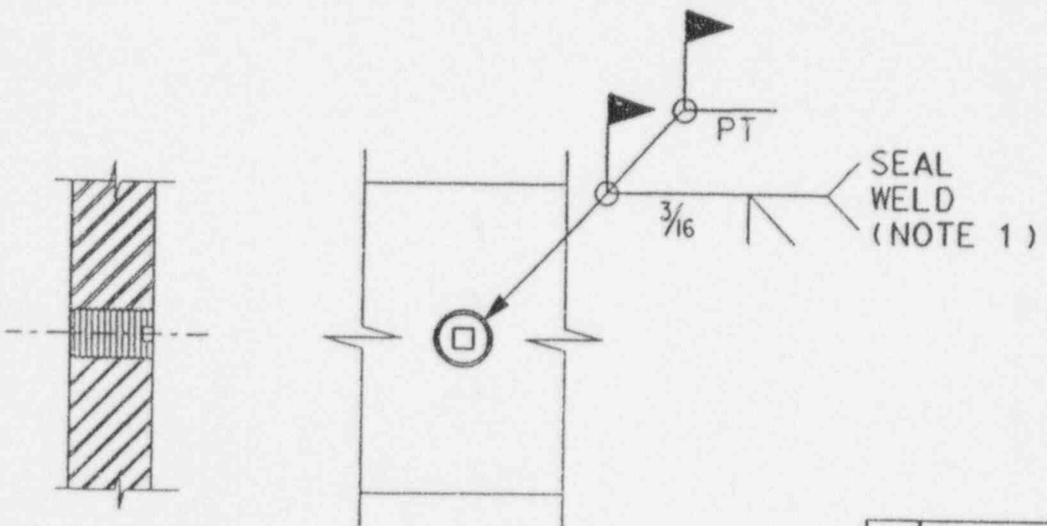


9611190240-02

11/11/96	YANKEE ATOMIC ELECTRIC COMPANY 580 MAIN STREET BOLTON, MA. NUCLEAR SERVICES DIVISION	TITLE: FIG. 1.2-2-YNPS RPV PACKAGE, SHIPPING CASK CROSS SECTION	
8-16-95	FOR: YANKEE ATOMIC ELECTRIC CO. ROWE, MA	JOB NO.	DWG. NO. YR-B-90-006
DRL			
APPD.			

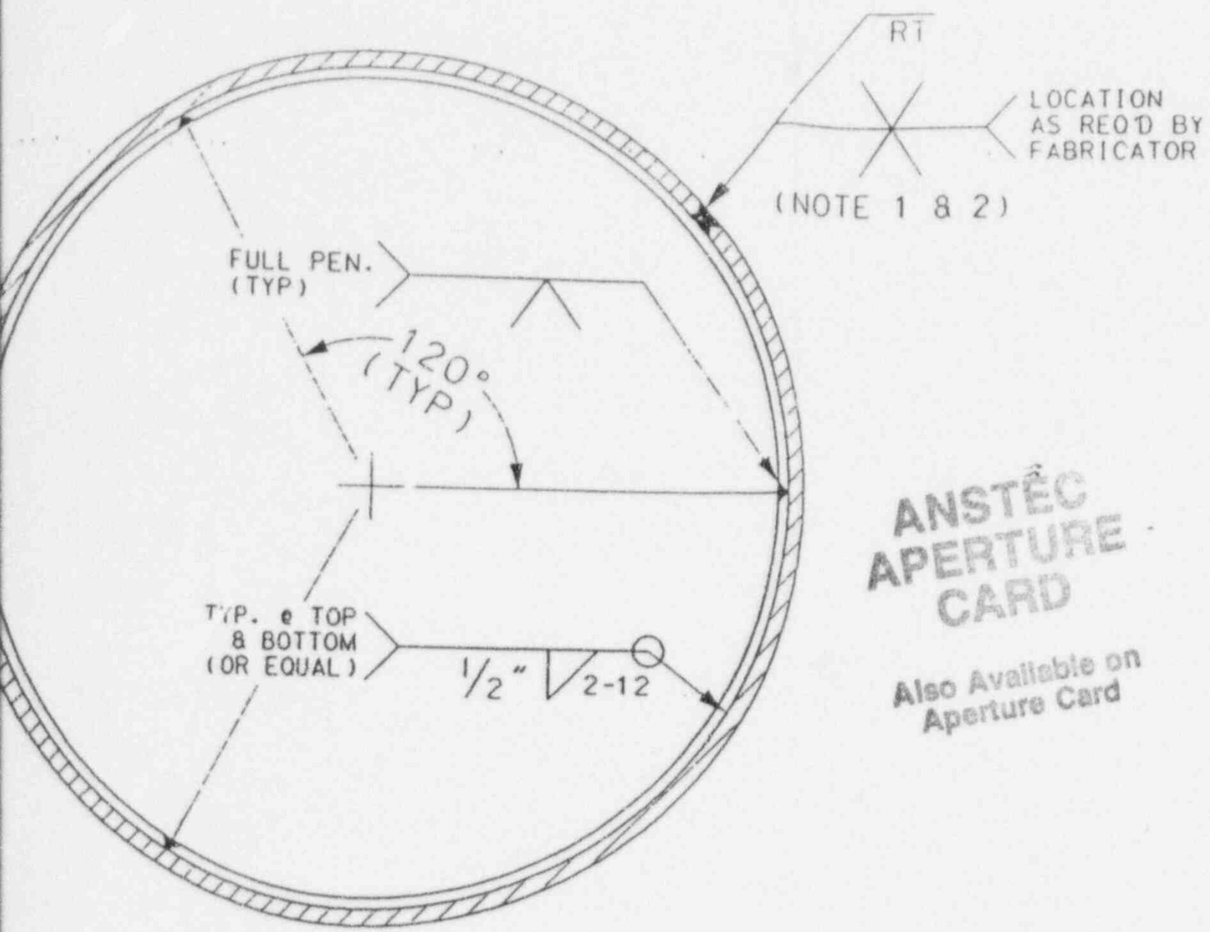


SECTION 1-1
(FIGURE 1.2-2)



FILL/VENT PORT PLUG
DETAIL

REV	DESCRIPTION	BY
0	ORIGINAL ISSUE PER W.O. DD34	8-16-95 AnR



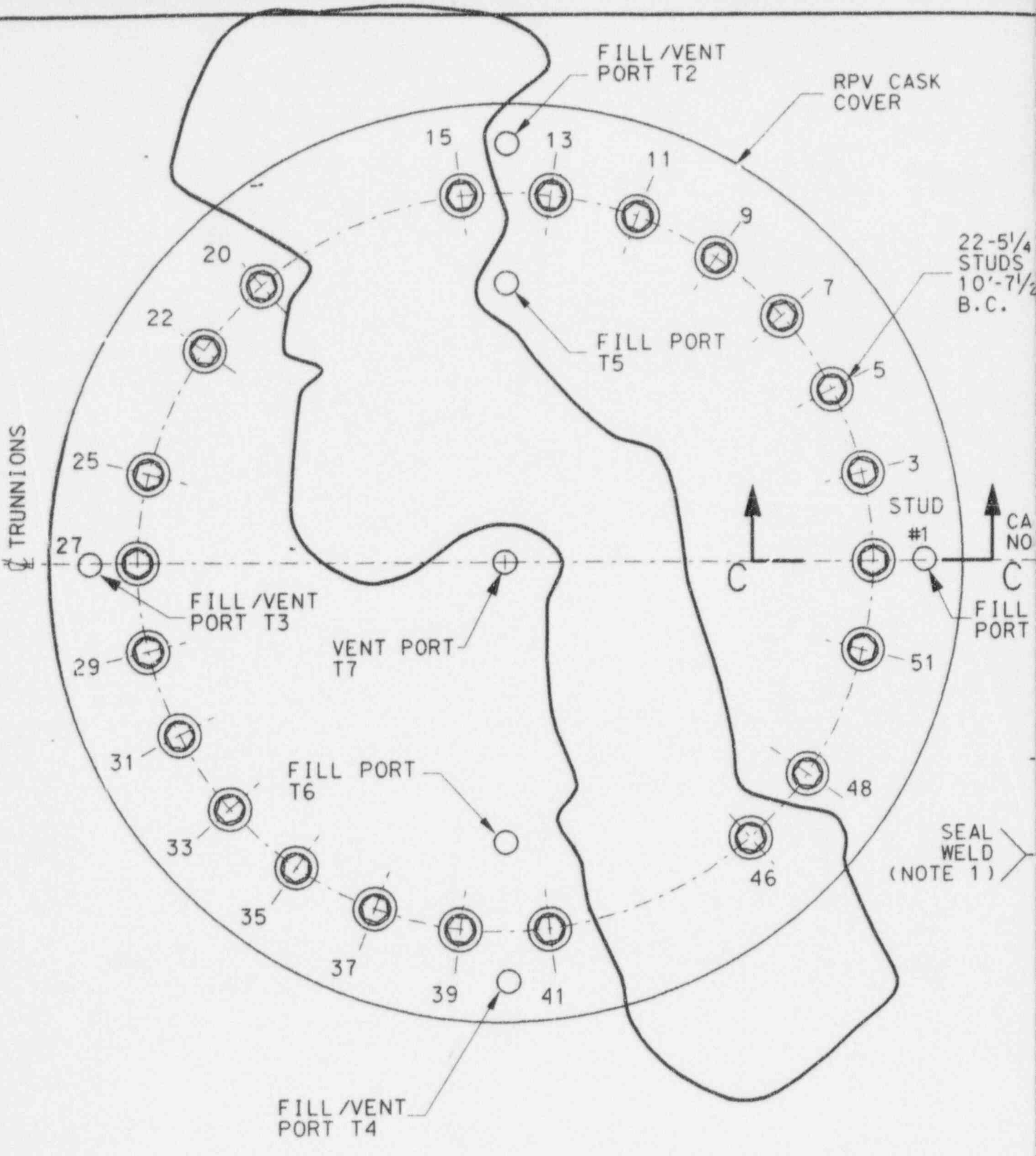
SECTION 2-2
(FIGURE 1.2-2)

NOTE:

1. CONTAINMENT WELD.
2. CONFIGURATION SHOWN IS FOR ILLUSTRATION ONLY. WELD WILL BE FULL PENETRATION IN ACCORDANCE WITH ASME VIII WITH FINAL CONFIGURATION DETERMINED BY FABRICATOR.
3. CASK CYLINDER & INNER SHIELD: A516, GR.70.

EP		YANKEE ATOMIC ELECTRIC COMPANY 580 MAIN STREET BOLTON, MA. NUCLEAR SERVICES DIVISION	TITLE: FIG. 1.2-3-YNPS RPV PACKAGE, SHIPPING CASK SECTIONS AND DETAILS
CHKD.	APPD.	FOR: YANKEE ATOMIC ELECTRIC CO. ROWE, MA	JOB NO. DWG. NO. YR-B-90-007

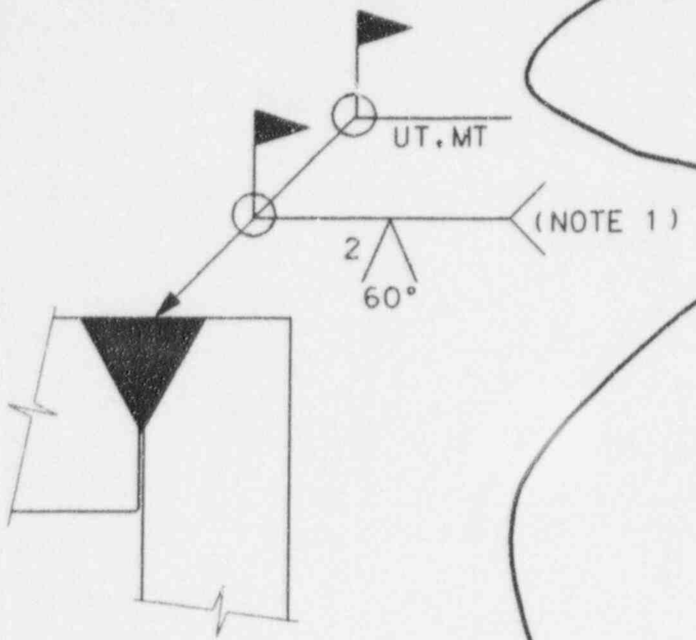
9611190240-03



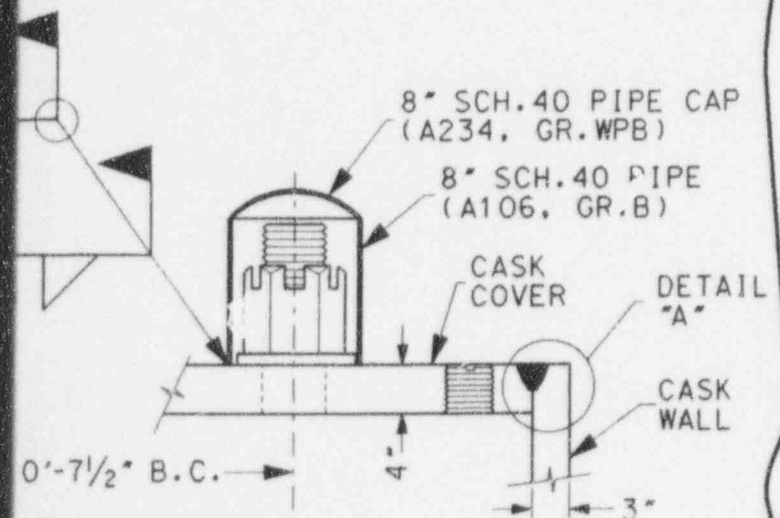
PLAN

(STUD COVERS AND LIFTING CLEVIS NOT SHOWN)

REV	DESCRIPTION	BY
1	REVISED FOR CHANGE IN LIFTING CLEVIS	11-8- WT
0	ORIGINAL ISSUE PER W.O. DD34	8-16- PMR



DETAIL "A"



SECTION C-C

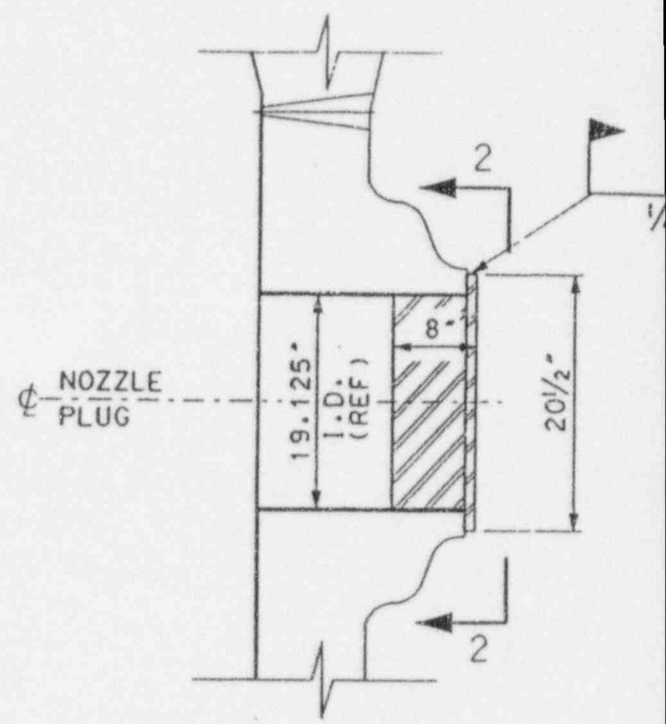
9611190240-04

NOTE:

1. CONTAINMENT WELD
2. SEE FIGURE 1.2-3 FOR DETAILS OF TYPICAL FILL/VENT PORT PLUG.
3. TOP COVER: A516, GR.70
4. LIFTING CLEVIS TO BE REMOVED PRIOR TO OFF-SITE TRANSPORT OF LOADED PACKAGE.

11/15/96	11/14/96	YANKEE ATOMIC ELECTRIC COMPANY 580 MAIN STREET BOLTON, MA. NUCLEAR SERVICES DIVISION	TITLE: FIG. 1.2-4-YNPS RPV PACKAGE, SHIPPING CASK COVER PLAN VIEW AND DETAILS
DRL	JAK		
8-16-95	8-16-95		
JEP	DRL	FOR: YANKEE ATOMIC ELECTRIC CO. ROWE, MA	JOB NO.
CHKD.	APPD.		DWG. NO. YR-B-90-008

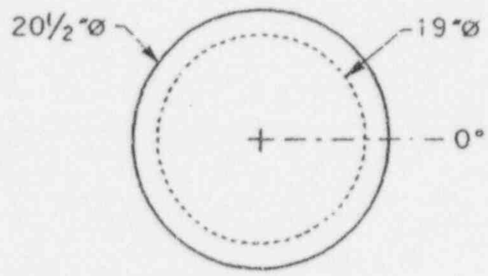
e:\dan\vrdecom\vrb90008.dan



NOZZLE WITH PLUG

0	ORIGINAL ISSUE PER W.O. DD34
REV	DESCRIPTION

1" WELD @ 45°
135°, 225°, 315°



SECTION 2-2
(NOZZLE NOT SHOWN)

NOTES:

1. FABRICATE NOZZLE PLUGS FROM A36 MATERIAL.

9611190240.05

8-16-95	8-16-95	8-16-95
	JEP	DBL
	CHKD.	APPD.

YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.

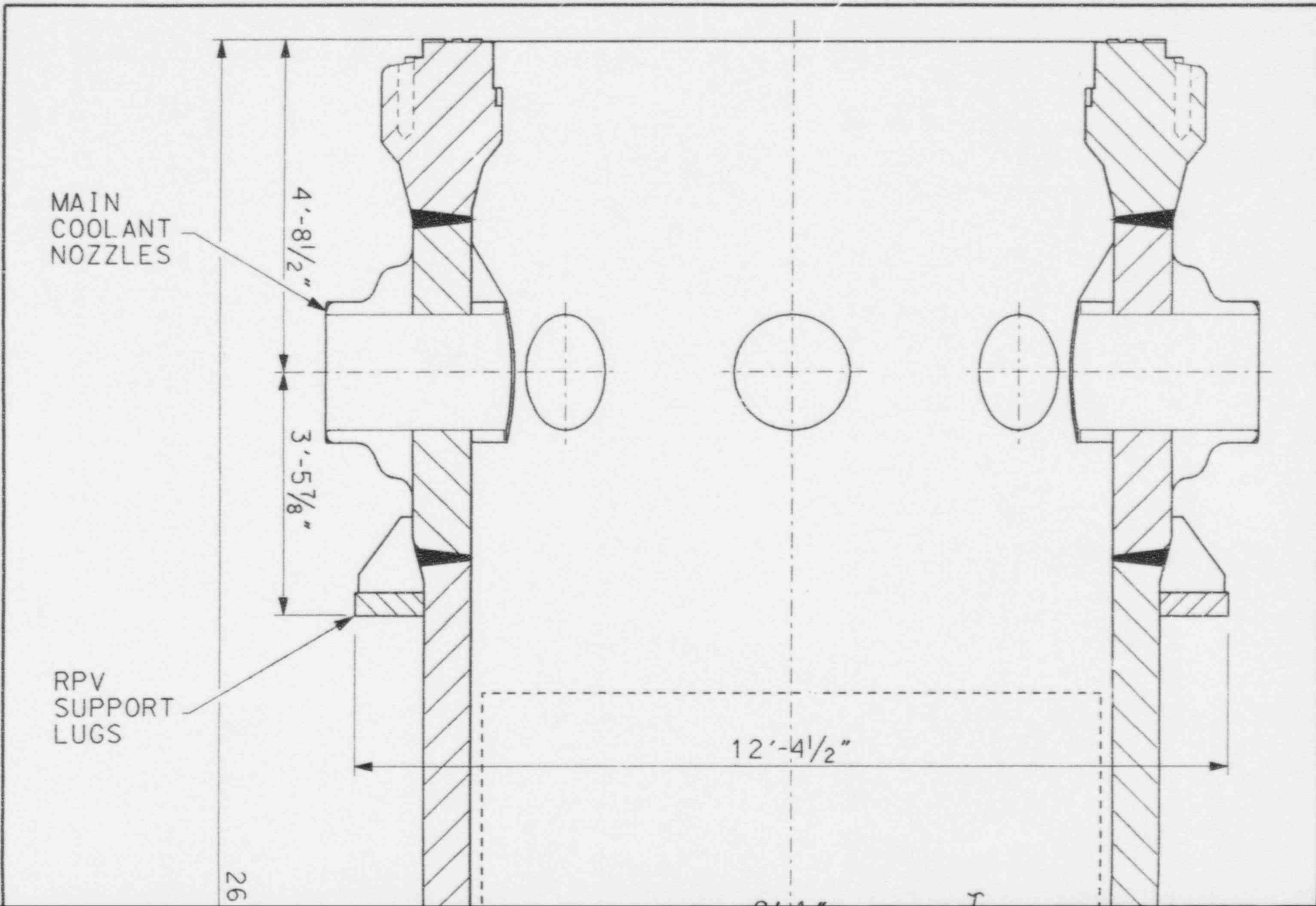
NUCLEAR SERVICES DIVISION

FOR: YANKEE ATOMIC ELECTRIC CO.
ROWE, MA

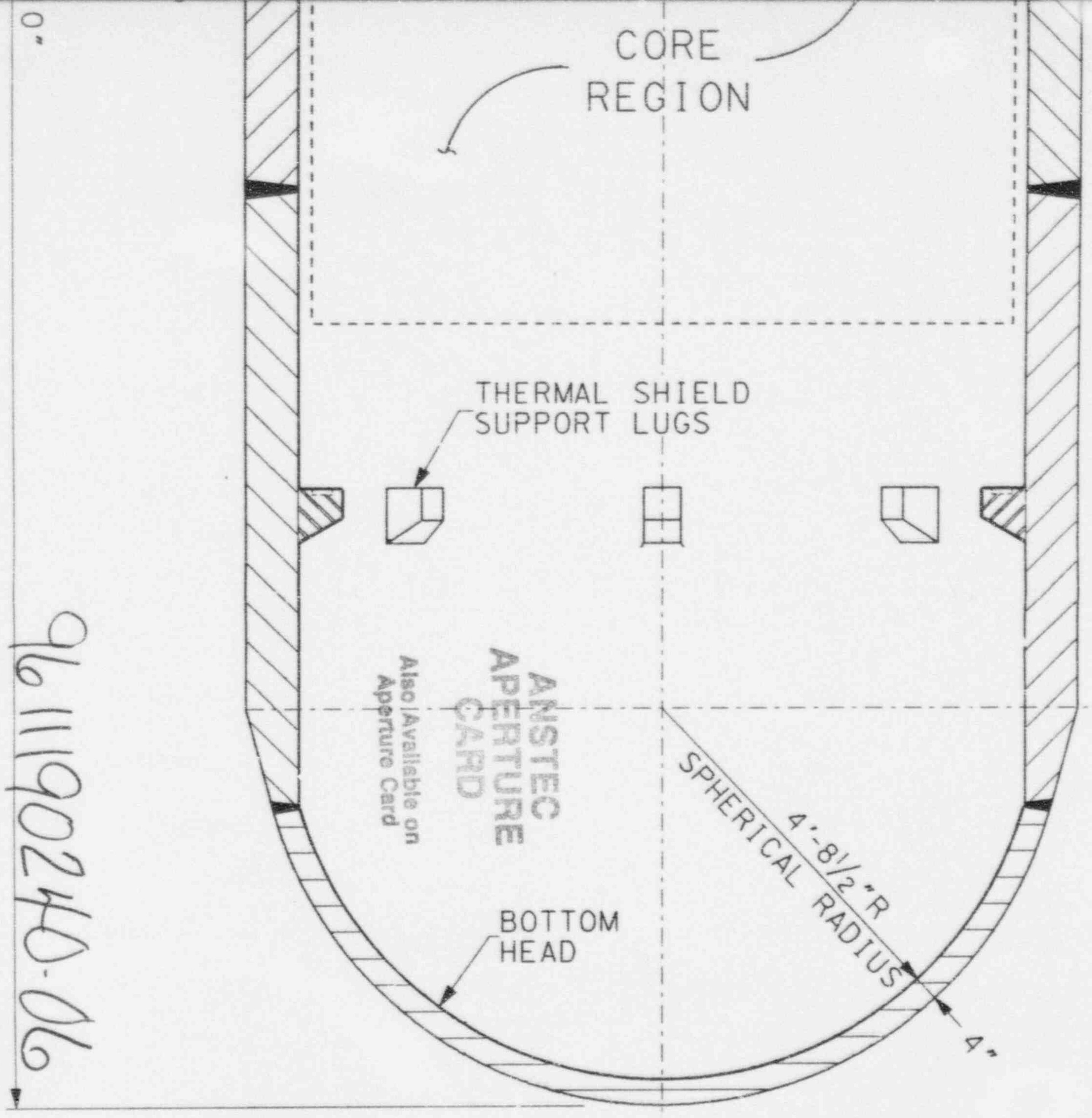
TITLE: FIG. 1.2-5-YNPS RPV PACKAGE, MAIN COOLANT NOZZLE PLUGS

JOB NO.

DWG. NO. YR-B-90-009



MAIN
COOLANT
NOZZLES
4'-8 1/2"
3'-5 7/8"
RPV
SUPPORT
LUGS
12'-4 1/2"
26



9611190240-06

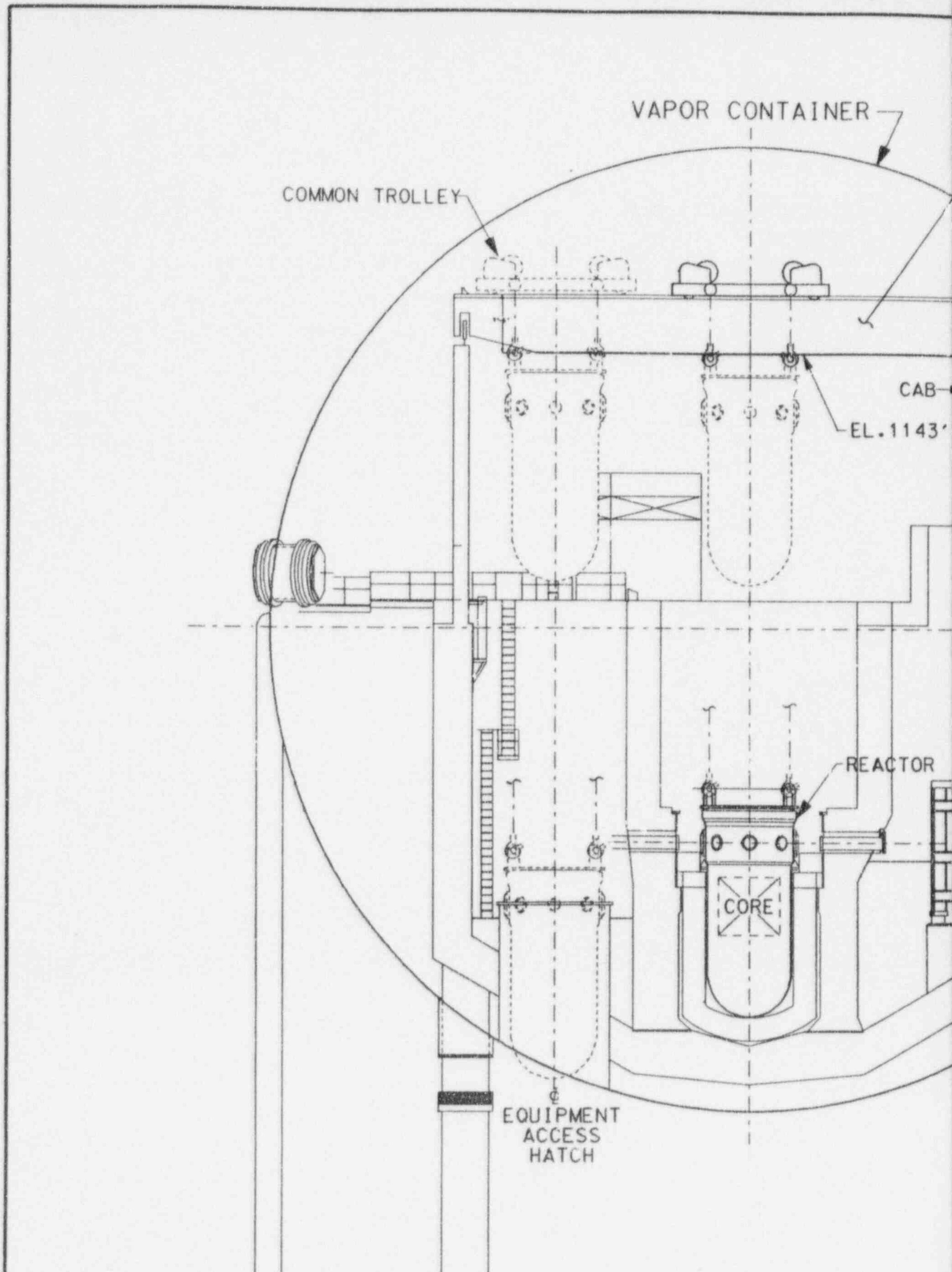
TITLE: YNPS RPV - CROSS SECTION

SCALE: NONE

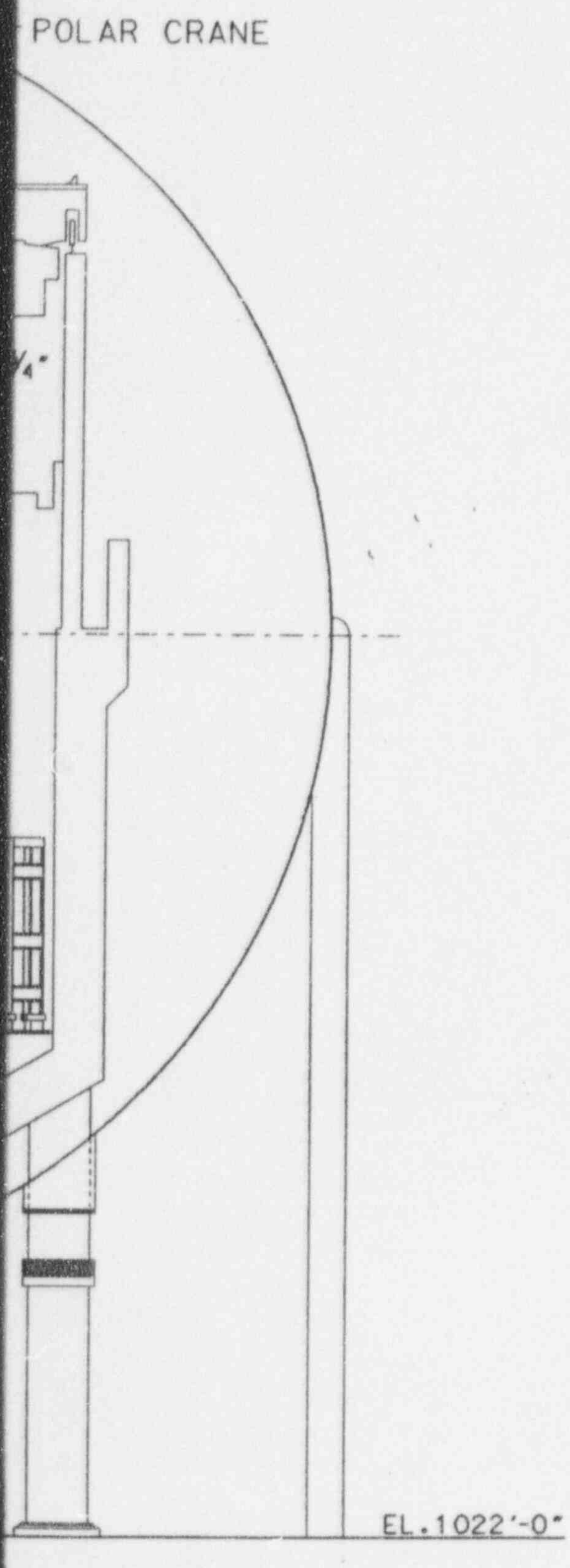
FIGURE 1.2-6

YRDECOMM/FIG126.DGN

e:\dan\yrdecom\fig126.dgn



VAPOR CONTAINER
COMMON TROLLEY
CAB
EL.1143'
REACTOR
CORE
EQUIPMENT
ACCESS
HATCH



9611190240-07

TITLE:

RPV LIFT LOAD PATH

SCALE: NONE

FIGURE 7.1-3

e:\dan\vrdecom\sar713.dan

R7826

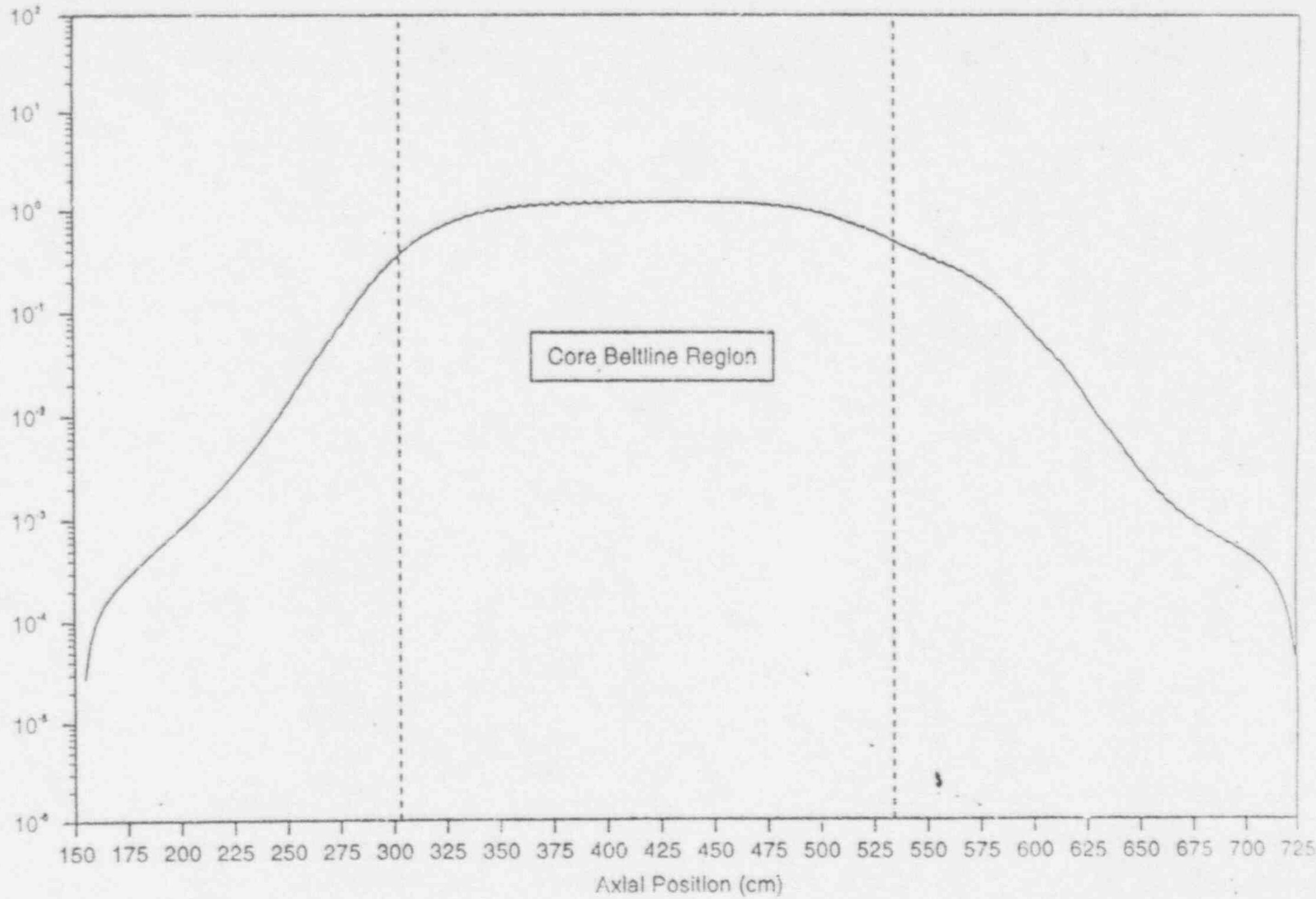


Figure 1.2-7

Thermal Neutron Flux Profile Data

R7826

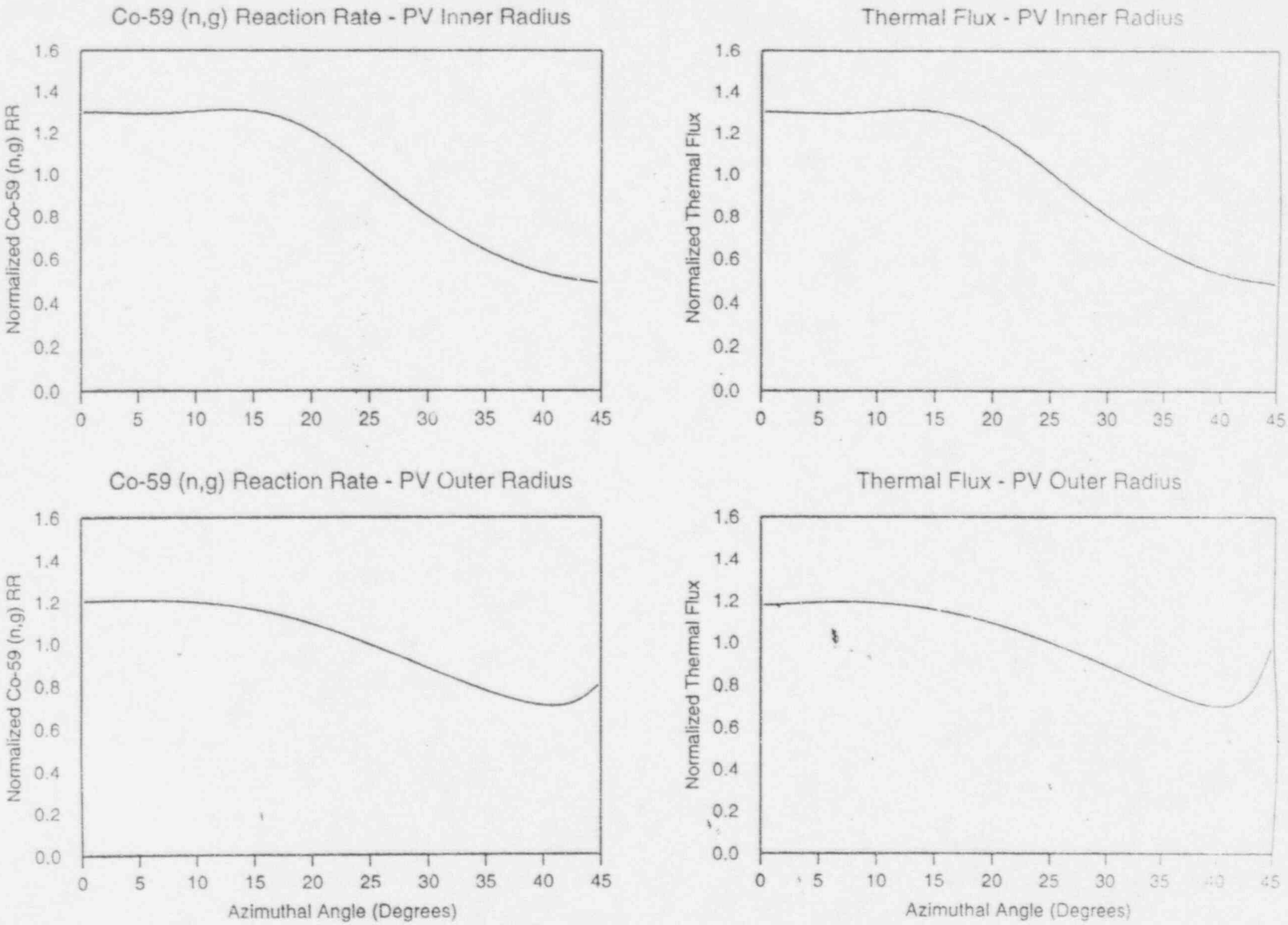


Figure 1.2-8

Azimuthal Activation/Flux Variation in the Pressure Vessel

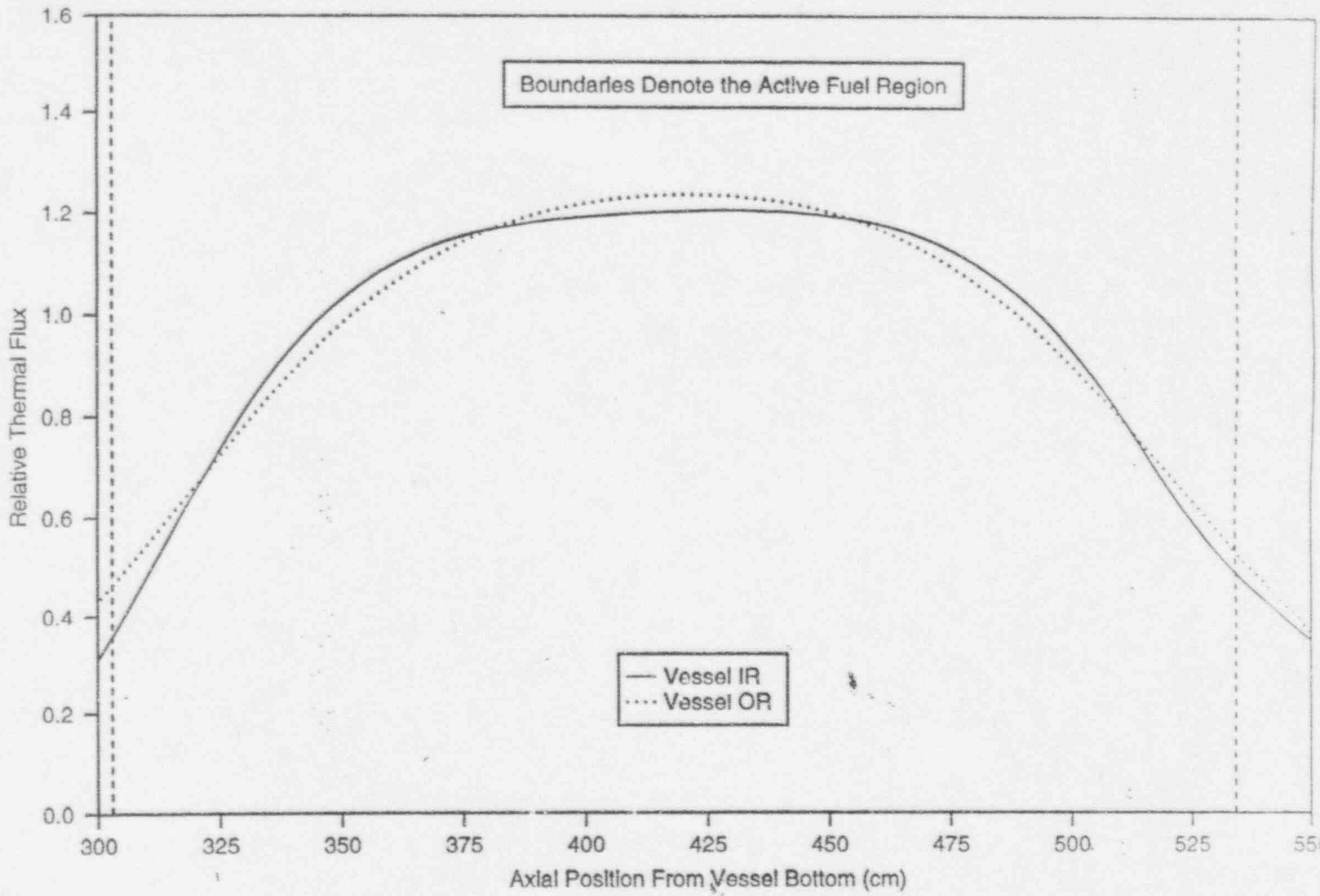


Figure 1.2-9

Yankee Rowe Activation Analysis, Relative Axial Thermal Flux Profiles - Vessel (10 Mesh)

R7826

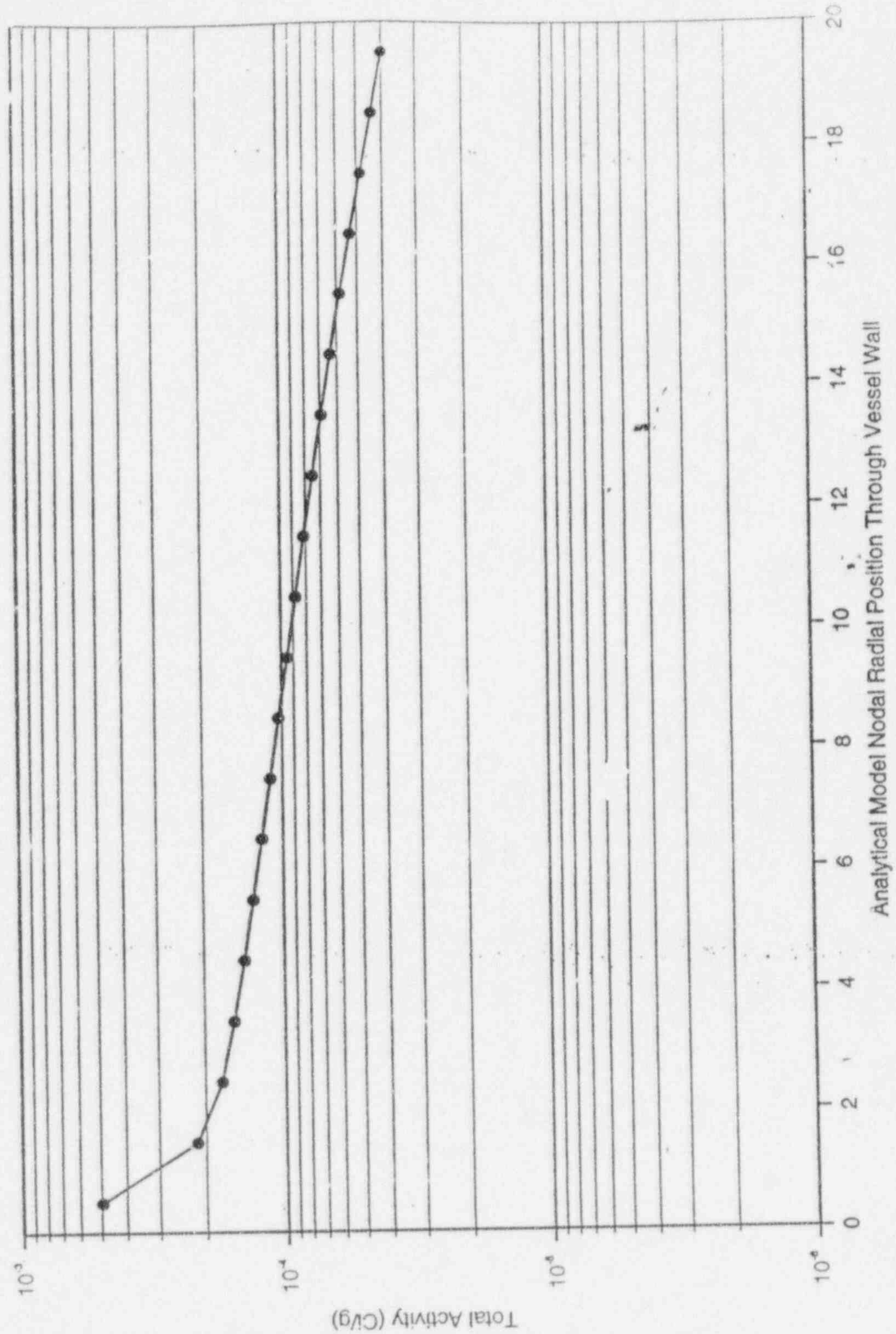


Figure 1.2-10

Total Curie/Gram Activity Versus Radial Position in the Yankee Pressure Vessel, Nuclide Decay Time = 1/1/94

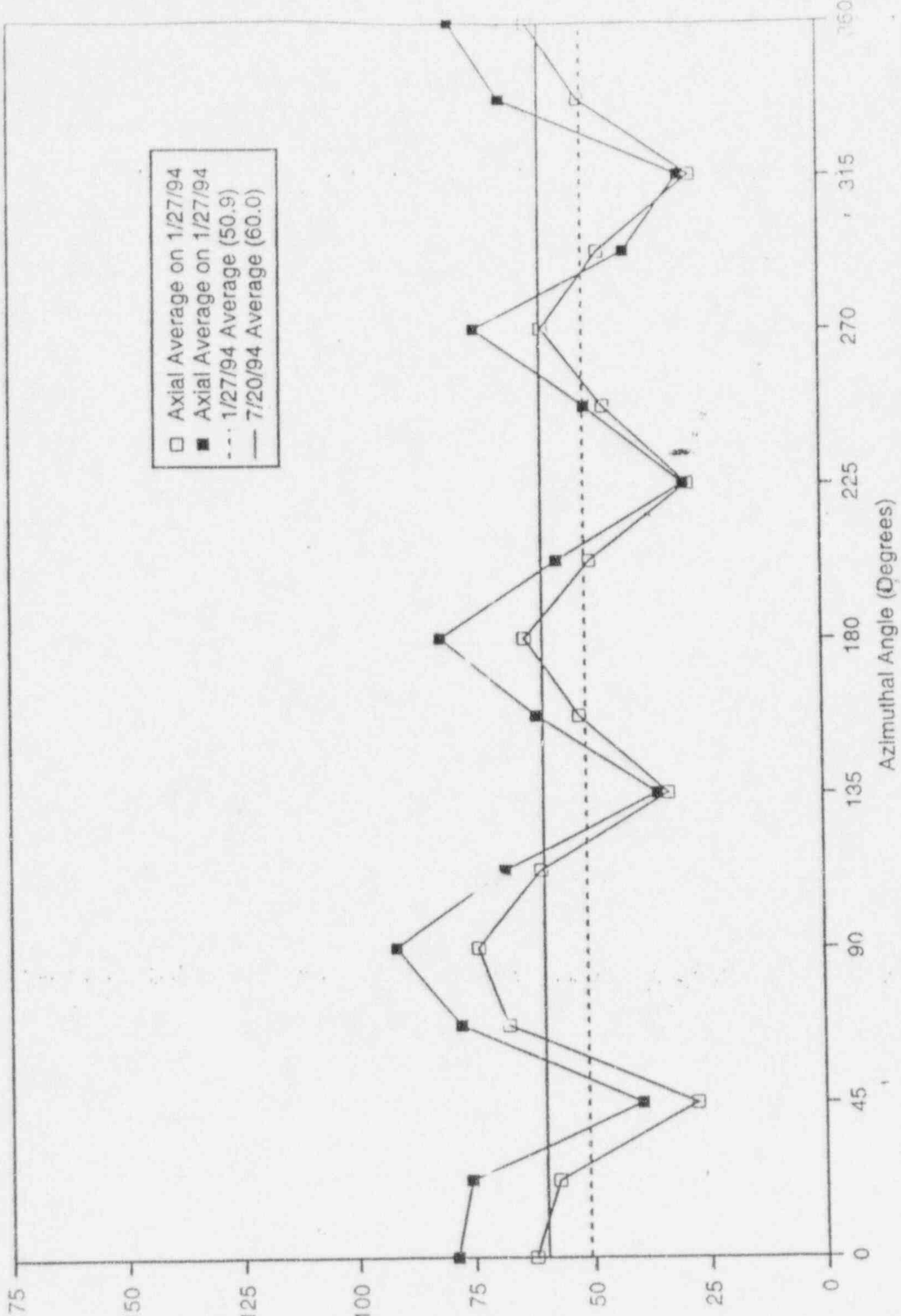


Figure 1.2-11

Reactor Vessel Inner Radius - Beltline (7 Ft) Region, Axially Averaged Dose Rates (R/Hr) on 1/27/94 and 7/20/94

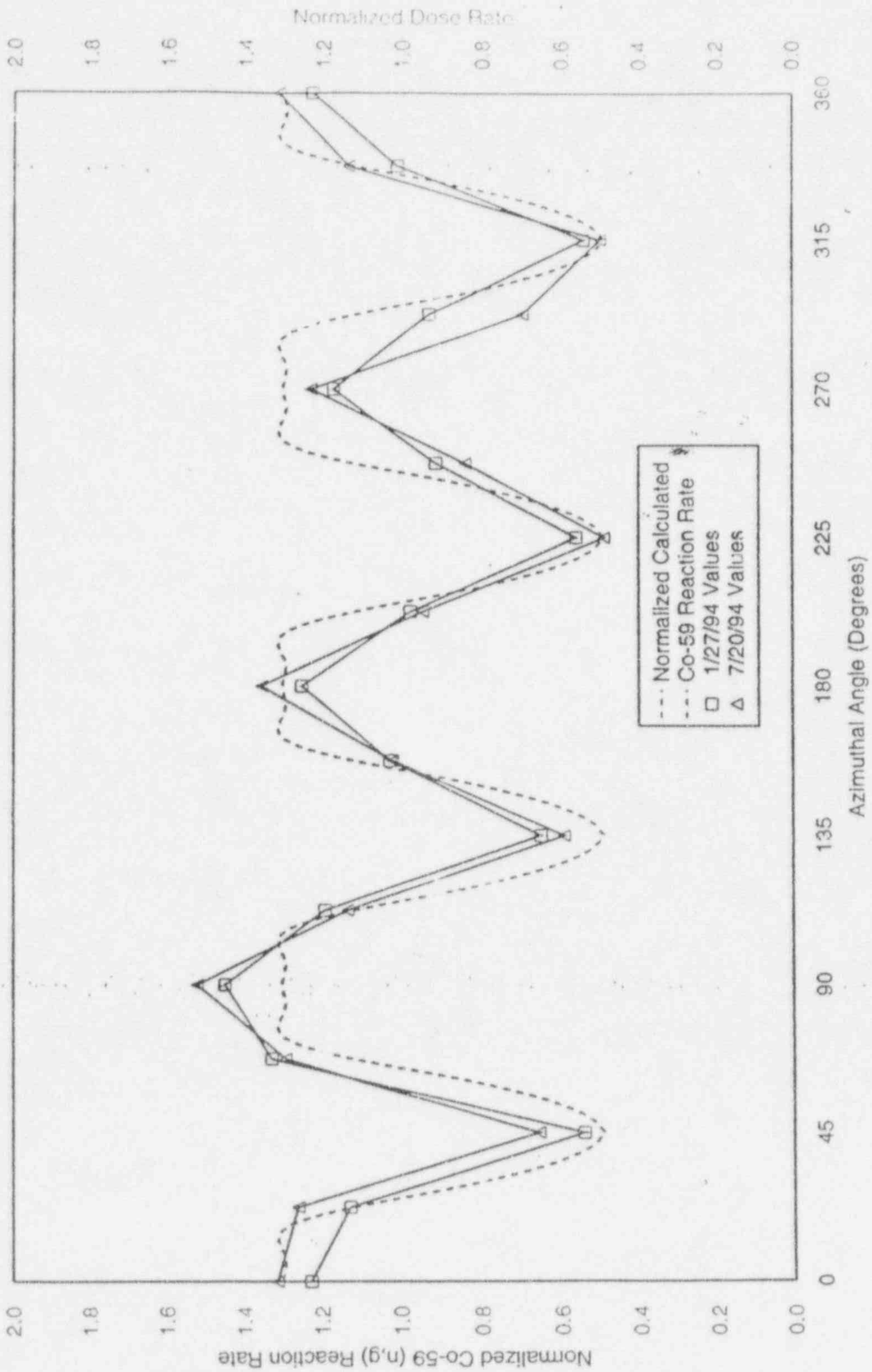


Figure 1.2-12

Reactor Vessel Inner Radius - Beltline Region, Comparison of Normalized Measured Dose Rate to Normalized Calculated Co-59 (n,g) Reaction Rate

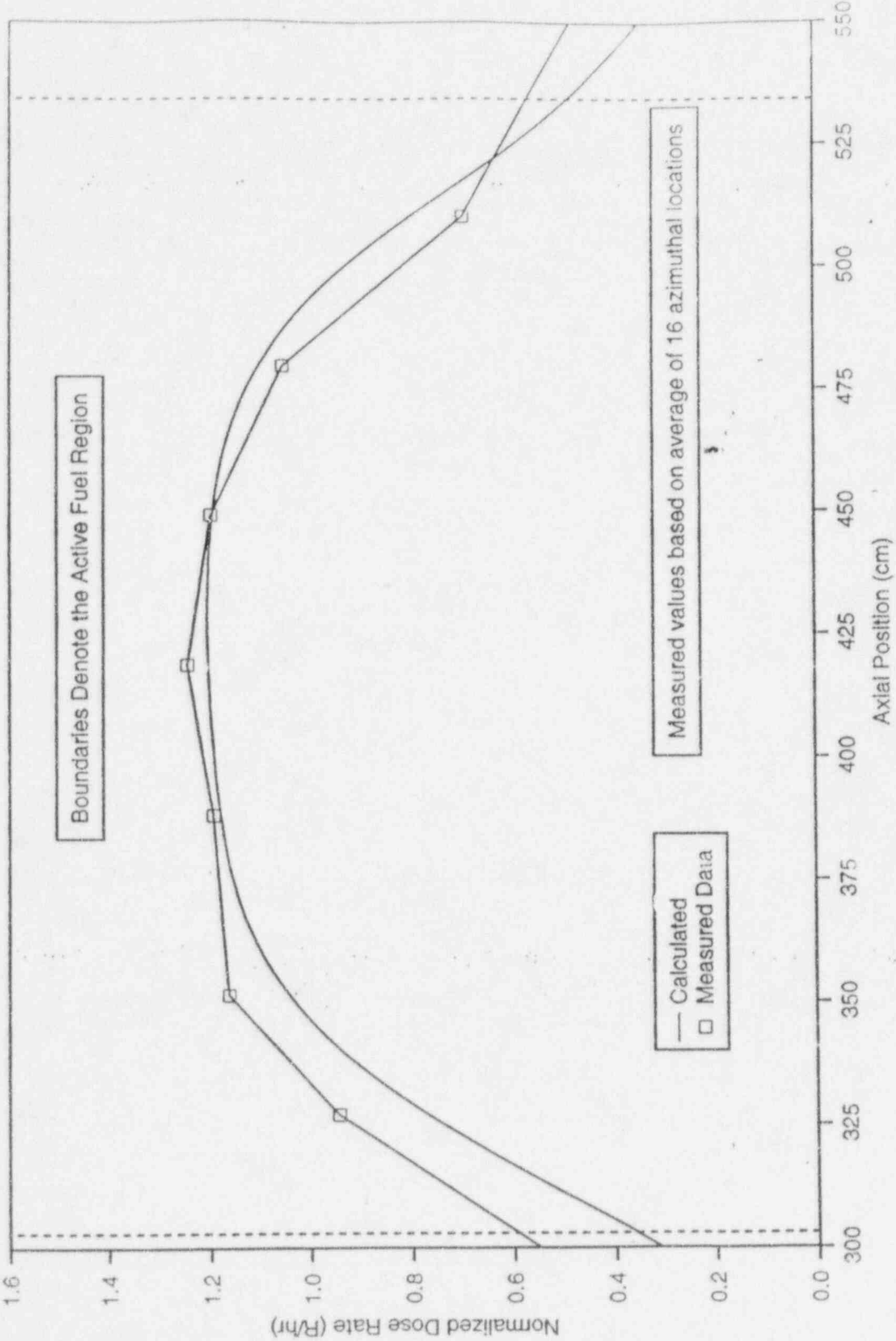


Figure 1.2-13

Comparison of Normalized Axial Dose Profiles - Vessel Inner Radius

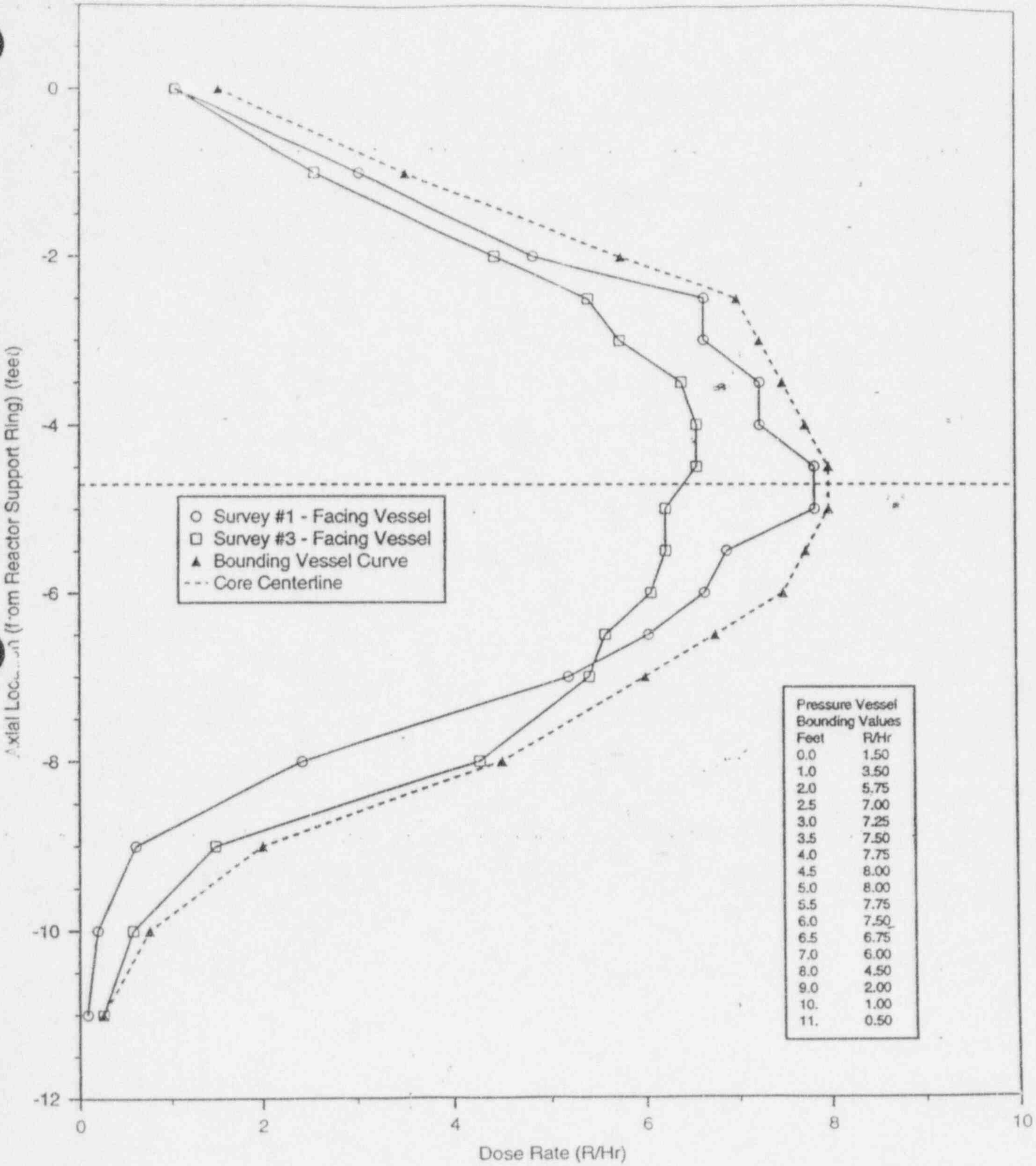


Figure 1.2-14

Reactor Vessel/Neutron Shield Tank Cavity - Dose Survey, Adjusted Vessel Dose Rates (R/Hr) to Determine Boundary Values

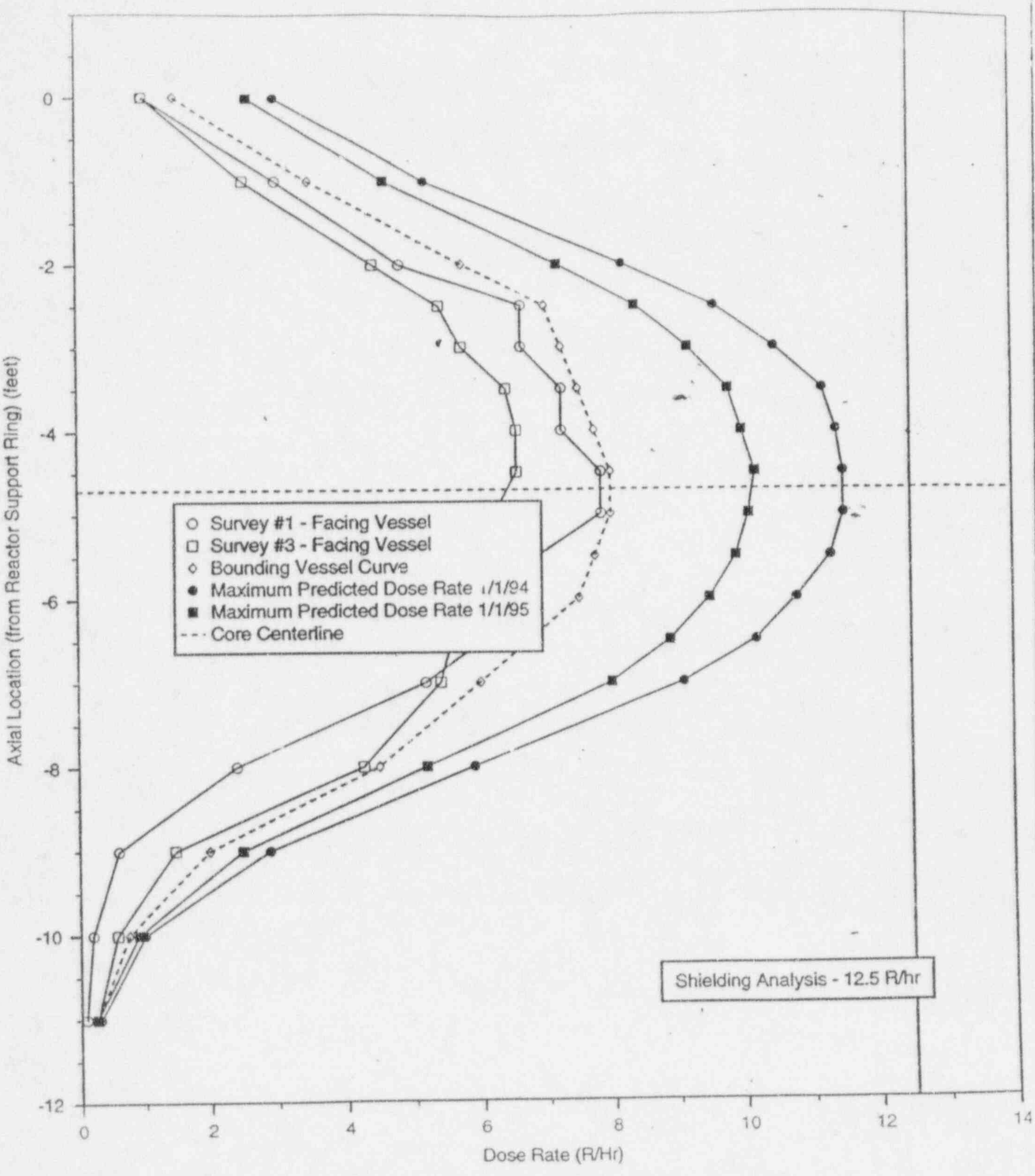


Figure 1.2-15

Comparison of Predicted to Measured Vessel Dose Rates at the Peak Azimuthal Location on the Vessel Exerior (R/Hr)

APPENDIX 1.4.1

CALCULATION YRC-1074, CURIE ESTIMATE FOR REACTOR PRESSURE VESSEL

APPENDIX 1.4.2

CALCULATION YRC-1076, CLASSIFICATION OF RPV DROSS

APPENDIX 1.4.3

CALCULATION YRC-1077, CLASSIFICATION OF REACTOR VESSEL

2.0 STRUCTURAL EVALUATION

The Yankee RPV package contains only LSA materials, and will be transported as exclusive use. However, the package contains greater than a Type A quantity of low specific activity (LSA) radioactive material. The structural evaluation for this type of package must demonstrate that the design meets the performance requirements of 10 CFR 71, Subparts E and F, for normal conditions of transport specified in 10 CFR 71.71. This type of package is exempted from the requirements of 10 CFR 71.51 by 10 CFR 71.52.

2.1 Structural Design

2.1.1 Discussion

The principal components of the Yankee RPV package are the reactor pressure vessel, internal solidified waste package, concrete fill, and external radiation shielding (shipping cask). The external radiation shielding will consist of a three-inch thick steel cylinder with four-inch thick closed ends. An additional one-inch thick steel cylinder will be located opposite the RPV core region. The annulus between the RPV and the cask will be filled with concrete. The interior of the RPV will be filled with low density concrete for added shielding. Figure 1.2-1 shows the individual components of the package. The cask, together with the concretes, provide the required shielding. The cask by itself is the containment vessel.

2.1.2 Design Criteria

The cask will be designed in accordance with the requirements of the ASME Boiler and Pressure Vessel (B&PV) Code, Section VIII Division 2 (Reference 2.10.1). The design criteria therein is consistent with the criteria presented in Regulatory Guide 7.6 (Reference 2.10.2). This criteria can be used with a linear elastic analysis of the containment vessel (cask). Under normal conditions of transport, as specified in 10 CFR 71.71, the maximum stress intensity in the cask resulting from the primary membrane stress should be less than the design stress intensity, S_m. The maximum stress intensity resulting from the sum of the primary membrane stress and the primary bending stress should be less than $1.5S_m$.

The normal conditions of transport as specified in 10 CFR 71.71 consist of initial conditions, and conditions and tests. The initial condition is that the ambient temperature preceding and following the tests remains constant at that value between -20°F and 100°F which is most unfavorable. The conditions and tests are as follows:

(1) Heat: 100°F + insolation
(2) Cold: -40°F
(3) Reduced external pressure: 3.5 psia
(4) Increased external pressure: 20 psia
(5) Vibration: normally incident to transport

(6) Water Spray: simulates exposure to 2 inches per hour rainfall for one hour minimum
(7) Free Drop: one-foot drop onto unyielding surface. Drop orientation must be such that maximum damage is expected.
(8) Corner Drop: applicable only to fiberboard or wood packages
(9) Compression: applicable only to packages weighing up to 11,000 pounds
(10) Penetration: hemispherical end of steel cylinder, 1¼ inches in diameter and weighing 13 pounds, dropped from height of 40 inches onto package.

Table 1 in Regulatory Guide 7.8 (Reference 2.10.3) lists the load combinations that should be used in the structural evaluation of the package. Table 1 also indicates the initial conditions which are to be combined with a particular normal condition. For example, the shock and vibration should be combined with initial conditions of -20°F ambient temperature, zero insolation, zero decay heat, minimum internal pressure, and fabrication stresses. The loadings and loading combinations used to evaluate the cask are consistent with Regulatory Guide 7.8. The individual loads and load combinations used in evaluating the cask are listed in Tables 2.6.1 and 2.6.2, respectively.

Regulatory Guide 7.11 (Reference 2.10.4) presents fracture toughness criteria for containment vessel material with a maximum wall thickness of 4 inches. The RPV cask will be fabricated of ASTM A-516, Grade 70, normalized steel made to a fine austenitic grain size practice. The cask is classified by Regulatory Guide 7.11 as a Category III container. The maximum wall thickness of the cask is 4 inches. Therefore, the fracture toughness requirements of Regulatory Guide 7.11 are met without testing. Additional material toughness requirements are imposed by the ASME B&PV Code, Section VIII. Specifically, for the materials used in the cask, AM-204 of Section VIII, Division 2 requires impact testing at the design temperature (-40°F). The material toughness requirements of AM-204 will be followed during cask fabrication.

The ASME B&PV Code, Sections III and VIII are used to determine material properties and allowable stresses. The allowable stresses for the tiedown system, none of which is a structural part of the package, are based on the 1989 AISC Specification (Reference 2.10.11).

10 CFR 71.45(a) requires, in part, that "any other structural part of the package which could be used to lift the package must be capable of being rendered inoperable for lifting the package during transport ...." To comply with 10 CFR 71.45(a), any part of the package which could be used for lifting will be removed or rendered inoperable prior to off-site transport.

2.2 Weights and Centers of Gravity

| The total weight of the RPV package, which includes the shipping cask, concrete,
| internal solidified waste package, and reactor pressure vessel is approximately 656,000

pounds, or 328 tons. Calculated weights of the individual package components are provided in Table 2.2.1.

The center of gravity of the package is along the longitudinal centerline, approximately 15'-11" from the bottom of the shipping cask. Locations of the centers of gravity of the individual package components are provided in Table 2.2.1. The overall package center of gravity is shown on Figure 1.2-1. The detailed calculations for component weights and centers of gravity are provided in Appendix 2.11.

2.3 Mechanical Properties of Materials

The following materials are used in the construction of the shipping cask for the YNPS RPV package:

(a) ASTM A516, Grade 70: Shipping Cask Cylinder, Internal Shield, Top Cover, Bottom Cover

(b) ASTM A106, Grade B: Pipe for Reactor Head Stud Caps

(c) ASTM A234, Grade WPB: Pipe Caps for Reactor Head Stud Caps

(d) AWS E70 Series: Welding Electrodes

Material properties and allowable stress intensities (S_m) for the A516 cask material are obtained from the ASME B&PV Code, Section VIII, Division 2, Part AM. Material properties for the E70 weld material are obtained from AWS D1.1 (Reference 2.10.6). A summary of the material properties and allowable stress intensities for the cask and welding materials at temperatures of interest are listed in Table 2.3.1.

2.4 General Standards for All Packages

The requirements for general standards for all packages are given in 10 CFR 71.43. The YNPS RPV package meets all these requirements, as discussed in the paragraphs below.

2.4.1 Minimum Package Size

The smallest overall dimension of a package as required by 10 CFR 71.43(a) must be not less than 10 cm (4 inches). The smallest overall dimension of the RPV package is 13'-2". Therefore, this requirement is met.

2.4.2 Tamperproof Feature

10 CFR 71.43(b) requires that the outside of the package incorporate a feature which is not readily breakable, and which, while intact, provides evidence that the package has not been opened. All penetrations in the package will be plugged with steel plates or covered with steel pipe caps, which will be welded in place. The top and bottom covers will be welded in place to the cask cylinder. The above features prevent any opening of the package. Therefore, the requirements of 10 CFR 71.43(b) are met.

2.4.3 Positive Closure

10 CFR 71.43(c) requires that the package include a containment system securely closed by a positive fastening device which cannot be opened inadvertently. As discussed in Section 2.4.2 above, the RPV package is completely welded closed, and therefore cannot be opened inadvertently. Therefore, the requirements of 10 CFR 71.43(c) for positive closure are met.

2.4.4 Chemical and Galvanic Reaction

10 CFR 71.43(d) requires that a package must be of materials and construction such that there will be no significant chemical, galvanic, or other reactions between the package components, or between the components and package contents. The principal materials used for the package are carbon steel plate, welding electrodes, and concrete. The package contents (RPV, concrete fill, and solidified waste package) are composed of carbon steel with a stainless steel liner (RPV), and a carbon steel liner filled with concrete (solidified waste package). All these materials are compatible with each other, and no significant chemical or galvanic reactions are expected. Therefore, the requirements of 10 CFR 71.43(d) are met.

2.4.5 Package Valves

10 CFR 71.43(e) requires that package valves or other devices which could allow radioactive contents to escape, must be protected against unauthorized operation, and except for pressure relief devices, must be provided with a leak proof enclosure. The YNPS RPV package is, as discussed in Section 2.4.2 above, completely welded closed and has no valves or other devices which could allow radioactive contents to escape. Therefore, the requirements of 10 CFR 71.43(e) for package valves and other devices are met.

2.4.6 Accessible Surface Temperature

10 CFR 71.43(g) requires that an exclusive use package, such as the RPV package, must be designed and constructed so that in still air at 100°F and in the shade, the maximum accessible surface temperature is less than 180°F. As demonstrated in Chapter 3, the maximum accessible surface temperature of the package under these conditions will be

102.5°F. Therefore, the requirements of 10 CFR 71.43(g) for maximum accessible surface temperature are met.

2.4.7 Venting

Features which would permit continuous venting of the package are not allowed by 10 CFR 71.43(h). As discussed above, the YNPS RPV package is completely sealed, with no provisions for venting. Therefore, the requirements of 10 CFR 71.43(h) are met by the package design.

2.5 Lifting and Tiedown Standards for All Packages

2.5.1 Lifting Devices

The shipping cask will have clevises welded to the top cover. Trunnions will be bolted to the cask cylinder for upending and downending operations. The clevises will be used during removal of the RPV from containment and placing the RPV into the shipping cask and downending of the cask. The clevises will be removed from the cover prior to off-site transport of the RPV package. The trunnions will also be removed prior to off-site transport of the package. There are no other mechanical or welded attachments which could be used to lift the RPV package. Therefore, the requirements of 10 CFR 71.45(a) are met.

2.5.2 Tiedown Devices

The RPV package tiedown system will consist of two saddle assemblies, two endstop assemblies, and wire ropes. A conceptual design for the tiedown system is shown on Figure 2.5-1. No part of the tiedown system is a structural part of the package. The tiedown system is designed in accordance with the AISC Manual of Steel Construction, Ninth Edition (Reference 2.10.11) for inertia loads imposed by the RPV package. The package inertia loads are based on accelerations 50 percent higher than required by AAR, Section 1 (Reference 2.10.5). See Table 2.6.4.

The effects of the tiedown system were included in the STARDYNE finite element model of the package. As shown in Table 2.6.3, all stresses in the package are less than the allowables discussed in Section 2.1.2.

2.6 Normal Conditions of Transport

As demonstrated in this section, the YNPS RPV package meets the acceptance criteria stated in Section 2.1 and will demonstrate no loss or dispersal of radioactive contents, no significant increase in external radiation, and no substantial reduction in the effectiveness of the packaging when subject to the normal conditions of transport specified in 10 CFR 71.71, as supplemented by Regulatory Guide 7.8 (Reference 2.10.3).

Two finite element models using the STARDYNE computer code (Reference 2.10.10) are employed to evaluate the package for the conditions specified in 10 CFR 71.71(c). A 3-dimensional plate element model of the entire shipping cask is used to evaluate the package for heat, cold, reduced external pressure, increased external pressure, and vibration. A 2-dimensional plane strain model is used to evaluate the top cover and cover-to-cylinder weld for vibrations in the package longitudinal direction. Plots of the models are shown in Figures 2.6-1 through 2.6-3. The effect of impact of a vertical steel cylinder (10 CFR 71.71(c)(10)) is evaluated using hand calculations.

A summary of the load cases considered in the package evaluation is provided in Table 2.6.1. A summary of the combinations of the load cases used to evaluate the package is provided in Table 2.6.2. The load cases were combined using the guidance provided in Regulatory Guide 7.8.

Table 2.6.3 provides a summary of the maximum stress intensities in the package as a result of the normal conditions of transport.

Detailed calculations for the structural evaluation of the shipping cask under normal conditions of transportation are provided in Appendix 2.11.

2.6.1 <u>Heat</u>

10 CFR 71.71(c)(1), Normal Conditions of Transport, Heat, requires an evaluation of the package for an ambient temperature of 100°F with a total insolation (for a curved surface) of 400 g cal/cm^2 during a 12-hour period. This condition results in a maximum internal package temperature of 193.6°F and a cask outside surface temperature of 171.6°F. Details of the thermal evaluation are discussed in Chapter 3.

The cask shell and top and bottom covers are evaluated with a thermal gradient through the cask wall of 22°F resulting from temperatures of 172°F on the outside and 194°F on the inside. Expansion of trapped water vapor is not a concern due to the available expansion space provided by the air entrained in the concretes. There are no restraints to thermal movement.

The package internal pressure at the specified temperature conditions is calculated to be 19.53 psia, resulting in a ΔP of 19.53 - 14.7 = 4.83 psi. Calculation of the internal pressure is discussed in Section 3.4.4. A ΔP of 5.0 psi is applied to the finite element model as an internal pressure.

The package was evaluated for the above thermal and pressure conditions using the STARDYNE finite element model. The maximum combined membrane plus bending stress intensity resulting from these conditions is 3.81 ksi, which is 11 percent of the allowable stress intensity of 34.95 ksi.

2.6.2 Cold

10 CFR 71.71(c)(2), Normal Conditions of Transport, Cold, requires an evaluation of the package for an ambient temperature of -40°F. The package is evaluated assuming a thermal gradient through the cask walls of 22°F resulting from temperatures of -40°F on the outside and -18°F on the inside. The thermal gradient from the inside to the outside of the package is assumed to be the same as in the hot condition. There are no constraints to package contraction. Expansion of trapped water vapor due to freezing is not a concern due to the available expansion space provided by the normal air entrained in the concretes.

Internal pressure in the package at -40°F, calculated using the ideal gas law, is 11.03 psia. This results in a ΔP of 11.03 - 14.7 = -3.67 psi. A ΔP of 4.0 psi is applied as an external pressure in the finite element model.

The STARDYNE finite element model of the package was used for evaluation. The maximum combined stress intensity resulting from the thermal and pressure conditions described in this section is 5.14 ksi, which is 15 percent of the allowable stress intensity of 34.95 ksi.

2.6.3 Reduced External Pressure

10 CFR 71.71(c)(3), Normal Conditions of Transport, Reduced External Pressure, requires an evaluation of the package for an ambient external pressure of 3.5 psia. Per Regulatory Guide 7.8, this condition is to be evaluated assuming the package is at 100°F, the maximum initial temperature condition. The effects of insolation per 71.71(c)(1) is also included.

The internal pressure in this condition is 19.53 psig. This results in a ΔP of 19.53 - 3.5 = 16.03 psi. A ΔP of 16.5 psi is applied as an internal pressure to the finite element model, and the resulting stress intensities are combined with the stress intensities due to thermal conditions. The maximum stress intensity due to the combined conditions is 6.42 ksi, which is 18 percent of the allowable stress intensity of 34.95 ksi.

2.6.4 Increased External Pressure

10 CFR 71.71(c)(4), Normal Conditions of Transport, Increased External Pressure, requires an evaluation of the package for an ambient external pressure of 20 psia. Per Regulatory Guide 7.8, this condition is to be evaluated assuming the package is at -20°F, the minimum initial temperature condition.

The internal pressure at -20°F is 11.55 psi. This results in a ΔP of 11.55 - 20.0 = -8.45 psi. A ΔP of 9.0 psi is applied to the finite element model as an external pressure, and the resulting stress intensities are combined with the stress intensities due to an ambient

temperature of -20°F. The maximum stress intensity due to these conditions is 6.07 ksi, which is 17 percent of the allowable stress intensity of 34.95 ksi.

2.6.5 <u>Transport Shock and Vibration</u>

10 CFR 71.71(c)(5), Normal Conditions of Transport, Vibration, requires an evaluation of the package for vibration normally incident to transportation.

The package will be transported using a hydraulic platform trailer and a rail car. Accelerations required by AAR, Section 1, Open Top Rules (Reference 2.10.5), are used to evaluate the package. These accelerations are greater than those provided in ANSI N14.2, "Proposed American National Standard, Tiedown for Truck Transport of Radioactive Materials" (Reference 2.10.13). A comparison of ANSI N14.2, AAR, and tiedown system design accelerations is provided in Table 2.6.4.

The 2-dimensional plane strain model (Figure 2.6-3) was used to evaluate the top cover and cover-to-cylinder weld under longitudinal accelerations. The 3-dimensional plate element model (Figures 2.6-1 and 2.6-2) was also used. The maximum stresses reported below and in Table 2.6.3 are the maximum from either model.

The stress intensities due to the accelerations are combined, per Regulatory Guide 7.8, separately with stress intensities resulting from maximum heat and from normal (-20°F) cold. Each direction of acceleration is evaluated individually per AAR and ANSI N14.2.

The maximum stress intensity occurs during the combination of Normal Cold conditions with longitudinal acceleration. The maximum stress intensity is 30.94 ksi, which is 89 percent of the allowable stress intensity of 34.95 ksi.

2.6.6 <u>Water Spray</u>

10 CFR 71.71(c)(6), Normal Conditions of Transport, Water Spray, requires an evaluation of the package for a water spray that simulates exposure to a rainfall of approximately two inches per hour for at least one hour.

The shipping cask for the YNPS RPV is fabricated from 3-inch and 4-inch thick A516, Grade 70 steel plate, and is of welded construction. All openings are seal welded. Therefore, the water spray will have no effect on the package.

2.6.7 <u>Free Drop</u>

10 CFR 71.71(c)(7), Normal Conditions of Transport, Free Drop, requires an evaluation of the package for a free drop through 0.3m (1 foot) onto a flat, essentially unyielding surface, striking the surface in a position for which maximum damage is expected. Considering the special handling and operational controls, a one-foot free drop is not a

Normal Condition of Transport for the RPV package. Further details are provided below and in Chapter 7.

2.6.7.1 Package Handling

Off site package handling will occur when the package is transferred from a roadway hauler to a rail car, and vice-versa. This process, described in Section 7.5.1, will be carefully controlled. The transfer process will be performed by the roadway transport contractor and the railroad crew under YAEC supervision. Transfer of the package from a rail car to a roadway hauler, if necessary, will be performed using a similar process.

Based on the transfer process, the control of the process, and the design of the transfer mechanism, reasonable assurance is provided that a free drop of the RPV package will not occur during the transfer process.

2.6.7.2 Roadway Transport

The movement of the package on the roadway will be performed using a dedicated heavy hauler, similar to that used successfully for transport of the YNPS steam generators. The movement will proceed at a speed of less than 5 mph. The transport will be escorted by public safety officials. The package will not be transported during hazardous weather conditions.

Culverts along the public road portion of the haul route will be evaluated in accordance with the AASHTO Standard Specification for Highway Bridges (Reference 2.10.12). Any repairs or upgrades required by the evaluations will be performed prior to package transport. Slope stability along the public roadway, the Sherman Dam crest roadway, and at the Sherman Dam spillway bridge will also be evaluated. The existing Sherman Dam spillway bridge will be evaluated and replaced if necessary prior to transport of the RPV package. Both the evaluation of the existing bridge, and if necessary, design of the replacement bridge will be performed in accordance with AASHTO (Reference 2.10.12). All evaluations pertaining to the roadway haul route will be reviewed by the appropriate regulatory agencies prior to transport.

The tiedown system used for roadway transport is designed to AISC standards (Reference 2.10.11), which are more conservative than those required by ANSI N14.2 (Reference 2.10.13) for truck transport. Design accelerations, as shown in Table 2.6.4, are higher than those required by ANSI N14.2. Design beyond ANSI criteria increases the margin of safety during transport. As discussed in Section 7.4, tiedown of the package will be inspected prior to roadway transport.

Based on the evaluations performed for the haul route, the design of the tiedown system, and the method used for hauling, reasonable assurance is provided that a free drop of the RPV package will not occur during roadway transport.

2.6.7.3 Rail Transport

Rail movement of the RPV package will be performed using a special, dedicated train along a predetermined route. Maximum speed will be limited to that requested by each of the rail companies responsible for transport. Rail transport will not occur during hazardous weather.

The accelerations used to design the tiedown system are 50 percent higher than those required by AAR. Design beyond the AAR criteria increases the safety margin during rail transport. Tiedown of the package will be inspected and accepted by rail car inspectors who are AAR rules qualified prior to rail transport.

The provisions made for rail transport provide reasonable assurance that a free drop of the RPV package will not occur.

2.6.7.4 Summary

In summary, based on:

(a) The process used for transfer of the RPV package from roadway hauler to rail car, control of the process, and design of the transfer mechanisms, and
(b) The roadway haul process, evaluation of the roadway haul route, and design of the tiedown system, and
(c) The railway haul process and design of the tiedown system,

reasonable assurance is provided that a free drop of the RPV package will not occur during transport, and is therefore not a normal condition of transport.

2.6.8 Corner Drop

This condition of transport applies only to fiberboard or wood rectangular packages not exceeding 110 pounds, or wood cylindrical packages not exceeding 220 pounds. The YNPS RPV package is a steel, cylindrical package weighing approximately 656,000 pounds. This condition of transport is therefore not applicable.

2.6.9 Compression

This condition of transport applies only to packages weighing up to 5000 kilograms (11,000 pounds). The YNPS RPV package weighs approximately 656,000 pounds. This condition of transport is therefore not applicable.

2.6.10 Penetration

10 CFR 71.71(c)(10), Normal Conditions of Transport, Impact, requires an evaluation of the package for the impact of the hemispherical end of a vertical steel cylinder with a diameter of 1¼ inches, and a weight of 13 pounds, dropped from a height of 40 inches.

The effect of the impact on the package was evaluated using both the Ballistic Research Laboratory and Stanford Research Institute formulas as presented in ASCE Manual No. 58, "Structural Analysis and Design of Nuclear Plant Facilities" (Reference 2.10.7).

The Ballistic Research Laboratory formula predicts that a plate greater than or equal to 0.015-inches thick will not be perforated by the given missile. The Stanford Research Institute formula predicts that a plate greater than or equal to 0.006-inches thick will not be perforated by the given missile. The minimum thickness of the shipping cask is 3.0 inches. Therefore, sufficient margin is provided to prevent perforation of the package by the vertical steel cylinder.

2.7 Hypothetical Accident Conditions

The Yankee RPV package contains only LSA materials, and will be transported as exclusive use. This type of package is exempted from the requirements of 10 CFR 71.51 by 10 CFR 71.52. Therefore, the tests of 10 CFR 71.73, "Hypothetical Accident Conditions", required by 10 CFR 71.51, are not applicable to the package.

2.8 Special Form

The contents of the RPV package are not classified as "special form". Therefore, the requirements of 10 CFR 71.75 are not applicable.

2.9 Fuel Rods

There are no fuel rods contained in the YNPS RPV package. Therefore, this section of Regulatory Guide 7.9 is not applicable.

2.10 References

2.10.1 ASME Boiler and Pressure Vessel Code, Sections III and VIII, 1989 Edition.

2.10.2 USNRC Regulatory Guide 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," Revision 1, March 1978.

2.10.3 USNRC Regulatory Guide 7.8, "Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Materials," Revision 1, March 1989.

2.10.4 USNRC Regulatory Guide 7.11, "Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels With a Maximum Wall Thickness of 4 Inches (0.1m)," June 1991.

2.10.5 Association of American Railroads, Operations and Maintenance Department, Mechanical Division, Section No.1, "General Rules Governing the Loading of Commodities On Open Top Cars", Republished September 1, 1994.

2.10.6 ANSI/AWS D1.1-88, Structural Welding Code - Steel.

2.10.7 ASCE, Manual No. 58, "Structural Analysis and Design of Nuclear Plant Facilities," 1980.

2.10.8 Not Used

2.10.9 ASME Boiler and Pressure Vessel Code, Section II, "SA-516 Specification for Pressure Vessel Plates, and Carbon Steel for Moderate and Lower Temperature Service," 1989 Edition.

2.10.10 STARDYNE User's Information Manual, Revision E.

2.10.11 AISC, Manual of Steel Construction, Allowable Stress Design, Ninth Edition.

2.10.12 AASHTO, "Standard Specification for Highway Bridges," Fifteenth Edition.

2.10.13 ANSI N14.2, "Proposed American National Standard, Tiedown for Truck Transport of Radioactive Materials," Draft 5, Revision 6, February 1986.

Table 2.2.1

Package Component Weights and Centers of Gravity

PACKAGE CENTER OF GRAVITY		
COMPONENT	WEIGHT (lbs)	DISTANCE (ft)[1]
Cask Cylinder (3" thick)	140,150	14.125
Cask Top Cover, Studs & Nuts	24,940	28.083
Cask Bottom (4" thick)	20,540	0.167
Interior Shield (1" thick)	16,650	14.374
Cone Plates (½" thick)	2,990	2.778
Solidified Waste Package	28,000	14.396
Reactor Pressure Vessel	300,000	17.46
Nozzle Plugs	5,500	23.208
Insulation	1,700	10.271
Concrete Between RPV and Cask[2]		
Below RPV Lugs	48,960	10.021
Above RPV Lugs	24,800	23.813
Concrete - RPV Interior[3]		
RPV Bottom Head	4,170	3.777
RPV Cylinder and Flange Ring	36,960	16.437
TOTALS	655,360	
CENTER OF GRAVITY = 15.90 FT. (Above Cask Bottom) = 190.8 IN.		

NOTES:

(1) All distances are given from the bottom of the cask.
(2) Density of 80 pcf used in calculations.
(3) Density of 30 pcf used in calculations.

Table 2.3.1

Material Properties and Allowable Stress Intensities

Material	Ambient Temp. Condition (°F)	Allow. Stress Intensity, S_m [(1)]	Coef. of Thermal Expansion, α [(2)]	Modulus of Elasticity, E [(3)]
A516, Grade 70	-40°F	23.3 ksi	5.53×10^{-6} (@100°F)	29.95×10^{6}
	-20°F	23.3 ksi	5.53×10^{-6} (@100°F)	29.95×10^{6} (@ -40°F)
	100°F	23.3 ksi (@ 188°F)	5.86×10^{-6} (@188°F)	28.95×10^{6} (@ 188°F)
E70 Electrodes[(4)]	-40°F	23.3 ksi[(4)]	[(5)]	[(5)]
	-20°F	23.3 ksi[(4)]	[(5)]	[(5)]
	100°F	23.3 ksi[(4)]	[(5)]	[(5)]

NOTES:

(1) Allowable (membrane) stress intensities for the cask cylinder and covers, in ksi, are from the ASME B&PV Code, Section VIII, Division 2, Part AM, Table ACS-1 (A516, Grade 70).

(2) Coefficients of thermal expansion, α, are obtained from the ASME B&PV Code, Section VIII, Division 2, Part AM, Table AMG-1.

(3) Moduli of Elasticity, E, in ksi, are obtained from the ASME B&PV Code, Section VIII, Division 2, Part AM, Table AMG-2 for temperatures ≥ 70°F, and the ASME B&PV Code, Section III, Division 1 Appendices, Table I-6.0 for temperatures < 70°F.

(4) Per AWS D1.1 (Reference 2.10.6) and the ASME B&PV Code, Section VIII, Division 2, weld material is limited to the same allowables as the base material for matching materials.

(5) Material properties of weld material are the same as base material properties.

Table 2.6.1

Summary of Load Cases

Load Case No.	Description
1	Maximum Heat: 100°F Ambient + Insolation
2	Maximum Cold: -40°F Ambient
3	Normal Cold: -20°F Ambient
4	Increased External Pressure: ΔP = 9.0 psi[(1)]
5	Decreased External Pressure: ΔP = 16.5 psi[(2)]
6	Lateral Acceleration During Transport: a = 2.0g
7	Vertical Acceleration During Transport: a = 2.0g (Up)
8	Longitudinal Acceleration During Transport: a = 3.0g

NOTES:

(1) ΔP noted is that used in analysis. Actual ΔP for this condition is 8.45 psi.

| (2) ΔP noted is that used in analysis. Actual ΔP for this condition is 16.03 psi.

Table 2.6.2

Summary of Load Combinations

Load Comb. No.	Load Case Combination(1)	Description
1	1 + 0.303[5]	Maximum Heat + ΔP (5.0 psi)(2)
2	2 + 0.444[4]	Maximum Cold + ΔP (4.0 psi)(3)
3	3 + 4	Normal Cold + Increased External Pressure (ΔP = 9.0 psi)
4	1 + 5	Maximum Heat + Decreased External Pressure (ΔP = 16.5 psi)
5	1 + 0.303[5] + 6	Maximum Heat + ΔP + X1 Acceleration (ΔP = 5.0 psi)
6	1 + 0.303[5] + 7	Maximum Heat + ΔP + X2 Acceleration (ΔP = 5.0 psi)
7	1 + 0.303[5] + 8	Maximum Heat + ΔP + X3 Acceleration (ΔP = 5.0 psi)
8	3 + 0.35[4] + 6	Normal Cold + ΔP + X1 Acceleration (ΔP = 3.15 psi)
9	3 + 0.35[4] + 7	Normal Cold + ΔP + X2 Acceleration (ΔP = 3.15 psi)
10	3 + 0.35[4] + 8	Normal Cold + ΔP + X3 Acceleration (ΔP = 3.15 psi)

NOTES:

(1) Numbers refer to load cases described in Table 2.6.1.

(2) ΔP is the differential between normal atmospheric pressure (14.7 psia) and internal cask pressure at maximum heat.

(3) ΔP is the differential between normal atmospheric pressure (14.7 psia) and internal cask pressure at maximum cold.

Table 2.6.3

Summary of Maximum Stress Intensities

Load Comb. No.(1)	Description	Element No.(2)	Max. Stress Intensity(3)	Allow. Stress Intensity(4)
1	Maximum Heat + ΔP	261-280	3.81 ksi	34.95 ksi
2	Maximum Cold (-40°F) + ΔP	261-280	5.14 ksi	34.95 ksi
3	Normal Cold (-20°F) + Increased External Pressure	261-280	6.07 ksi	34.95 ksi
4	Maximum Heat + Decreased External Pressure	321-340	6.42 ksi	34.95 ksi
5	Maximum Heat + ΔP + Lateral Vibration	278	15.76 ksi	34.95 ksi
6	Maximum Heat + ΔP + Vertical Vibration	261, 270	4.63 ksi	34.95 ksi
7	Maximum Heat + ΔP + Longitudinal Vibration	274/157(5)	29.76 ksi	34.95 ksi
8	Normal Cold + ΔP + Lateral Vibration	278	16.75 ksi	34.95 ksi
9	Normal Cold + ΔP + Vertical Vibration	261,270	6.02 ksi	34.95 ksi
10	Normal Cold + ΔP + Longitudinal Vibration	274/157(5)	30.94 ksi	34.95 ksi

NOTES:

(1) Numbers refer to load combinations described in Table 2.6.2.
(2) Element numbers refer to STARDYNE 3-D finite element model (Figures 2.6-1 and 2.6-2) unless noted otherwise.
(3) Maximum Stress Intensity is the maximum combined membrane plus bending stress intensity.
(4) Allowable combined membrane plus bending stress intensity = 1.5 S_m = 34.95 ksi for A516, Grade 70 material.
(5) Element numbers provided are for 3-D Model/2-D Model.

Table 2.6.4

Shock and Vibrations During Transport

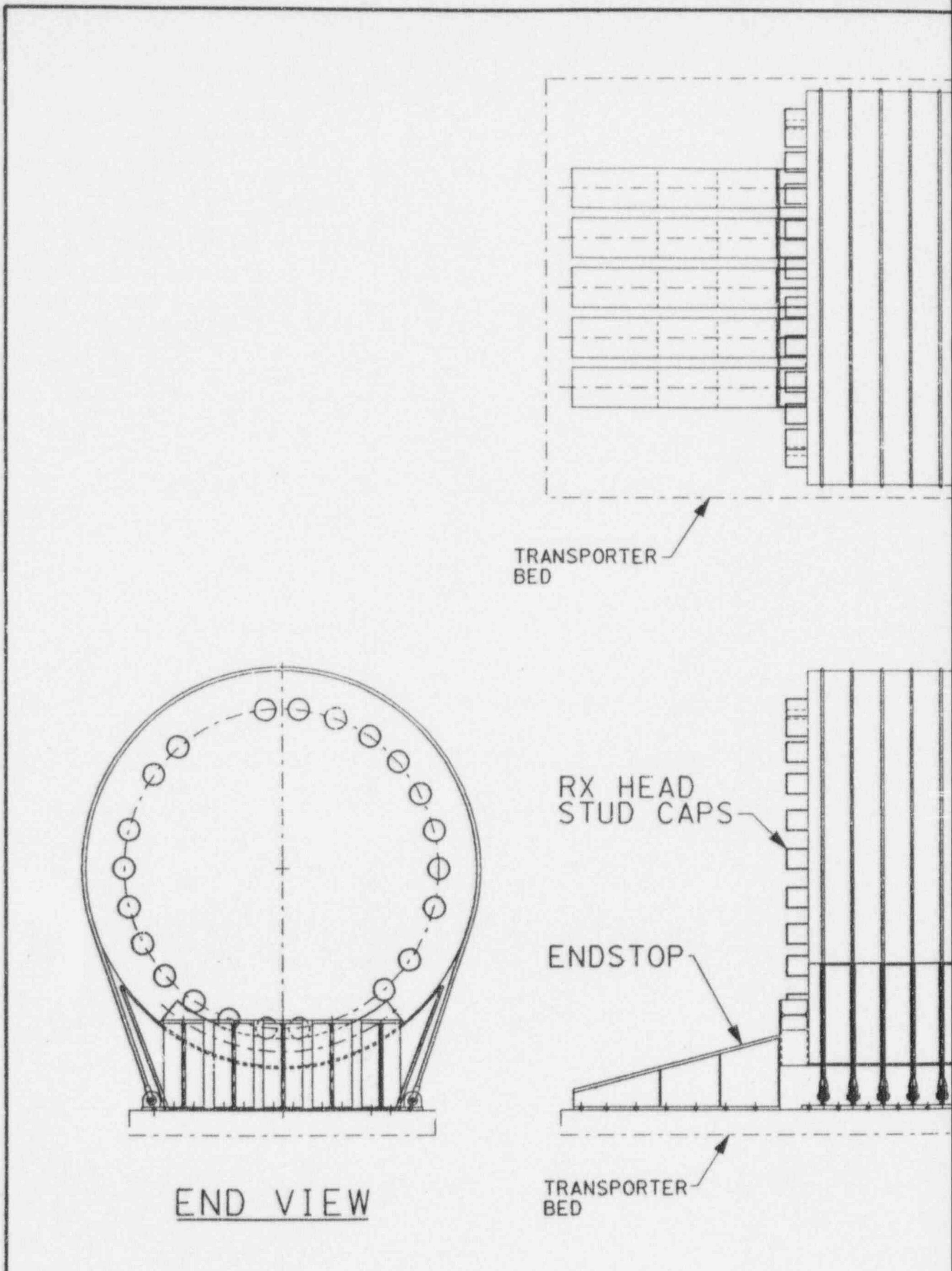
Comparison of Regulatory Criteria for Tie Down Systems

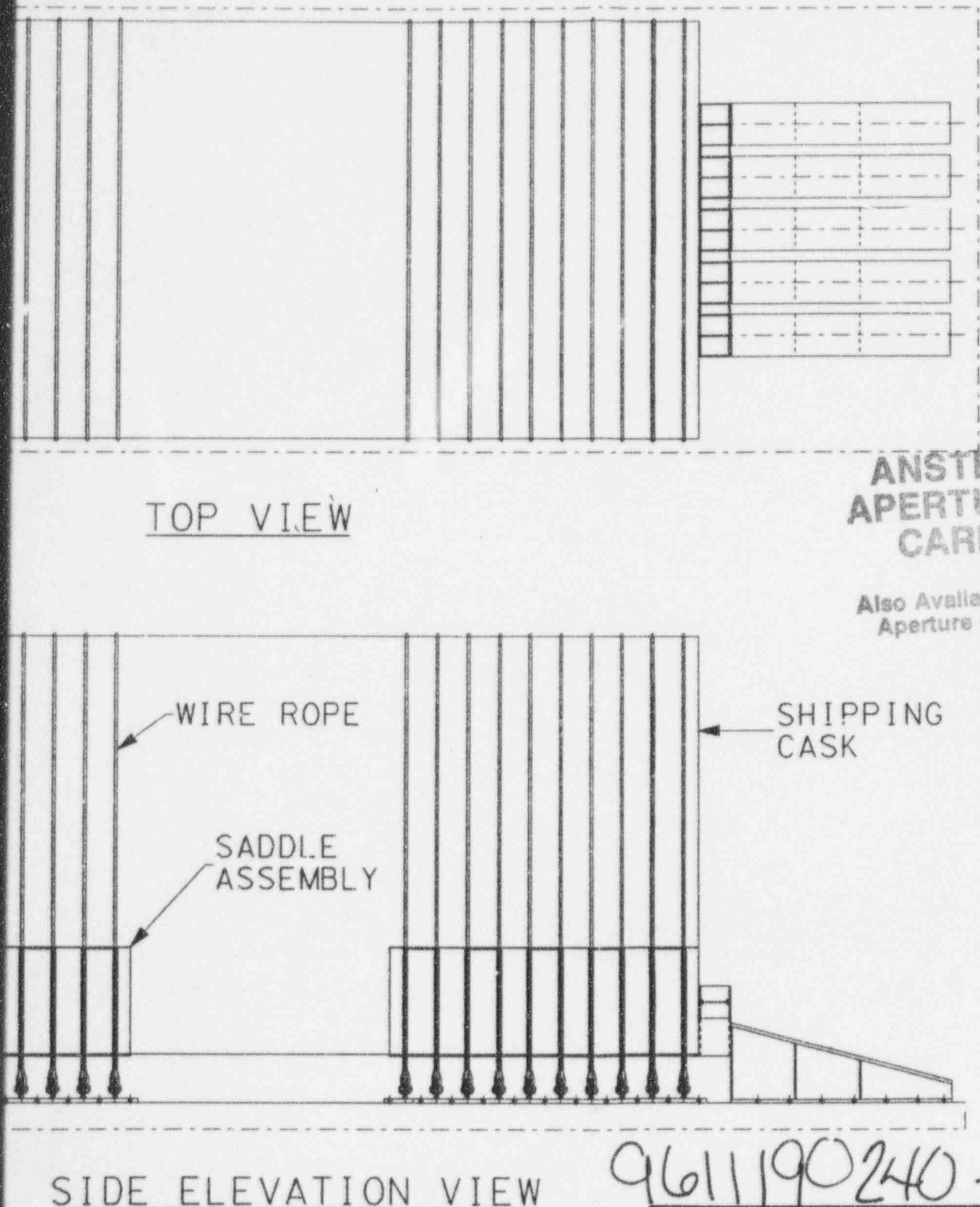
DIRECTION	ANSI N14.2 SPRING SUSPENSION	ANSI N14.2 AIR SUSPENSION	AAR OPEN TOP RULES[1]	TIEDOWN SYSTEM DESIGN[2]
Longitudinal	2.3g	1.8g	3.0g	4.5g
Lateral	1.6g	1.1g	2.0g	3.0g
Vertical	3.5g (Up)	1.5g(Up)/ 2.5g(Down)	2.0g (Up)	3.0g (Up)

NOTES:

(1) Accelerations specified by AAR Section 1, Open Top Rules, are used for evaluation of the RPV package.

(1) Accelerations used in designing the tiedown system are equivalent to the AAR Section 1 accelerations increased by 50 percent.


TRANSPORTER BED
RX HEAD STUD CAPS
ENDSTOP
TRANSPORTER BED
END VIEW



9611190240-08

TITLE: YNPS RPV PACKAGE-CONCEPTUAL TIE DOWN SYSTEM

SCALE: NONE

FIGURE 2.5-1

e:\dan\vrdecom\fla251.dan

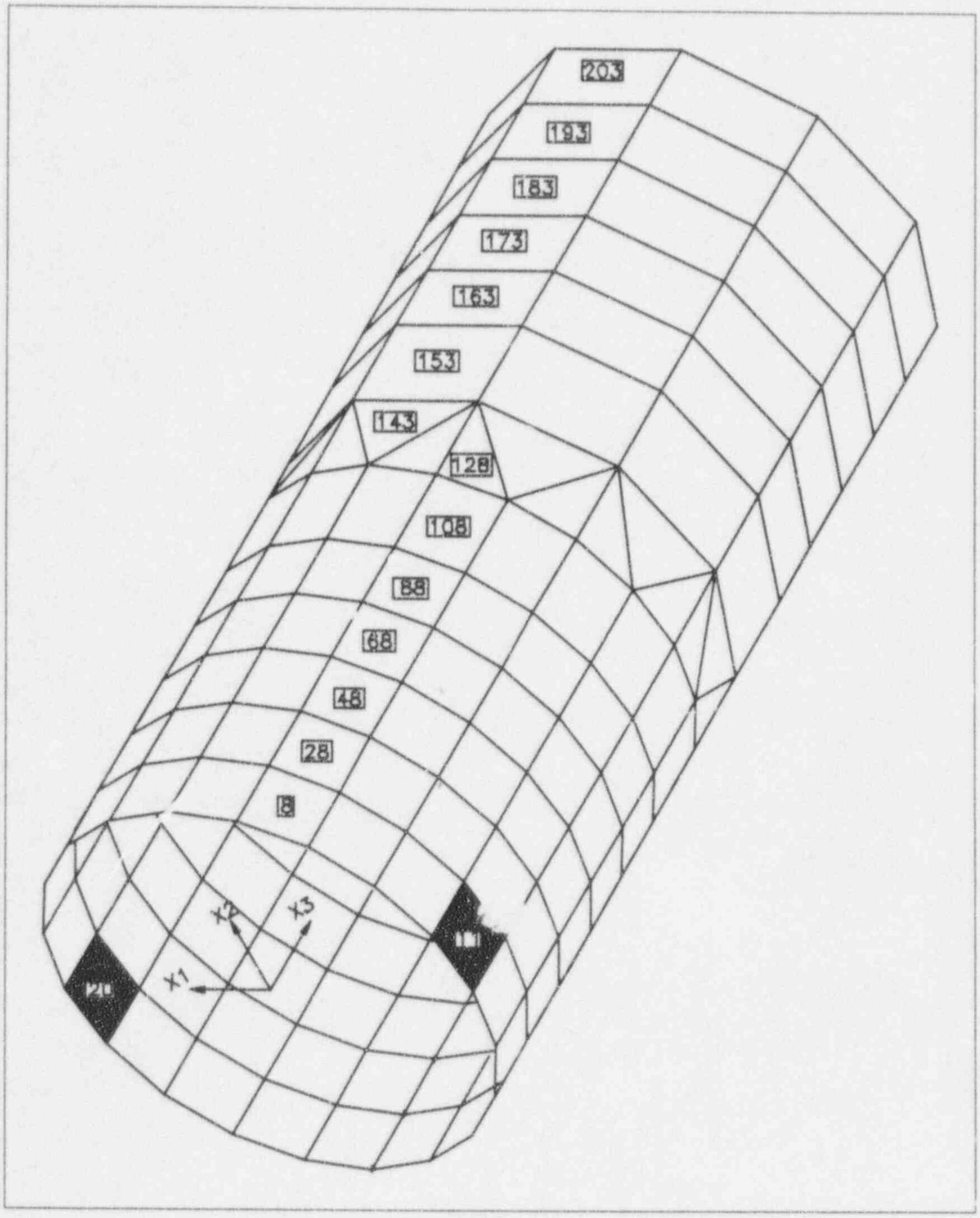


Figure 2.6-1

STARDYNE Finite Element Model
Cask Cylinder

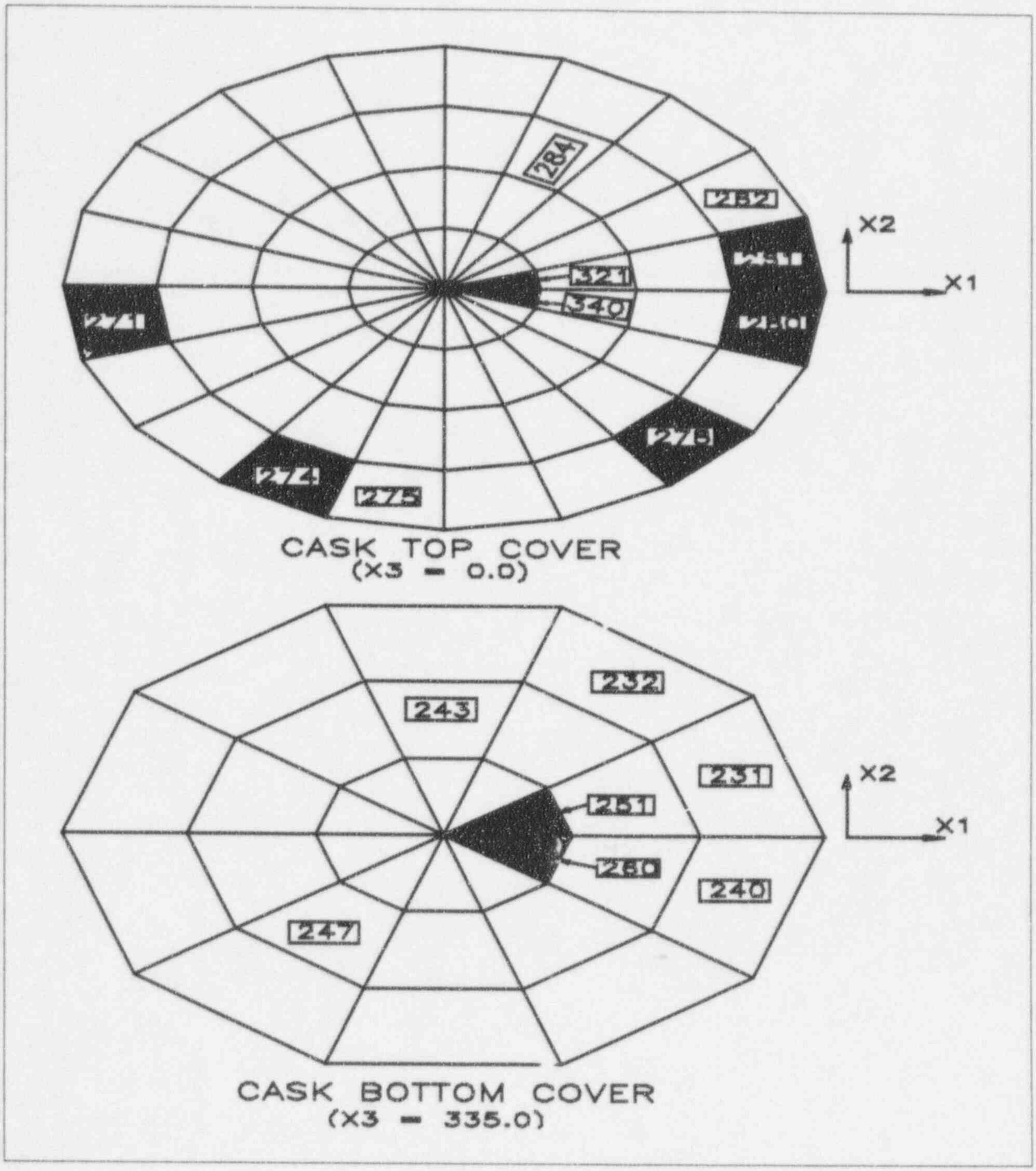


Figure 2.6-2

STARDYNE Finite Element Model
Cask Top and Bottom Covers

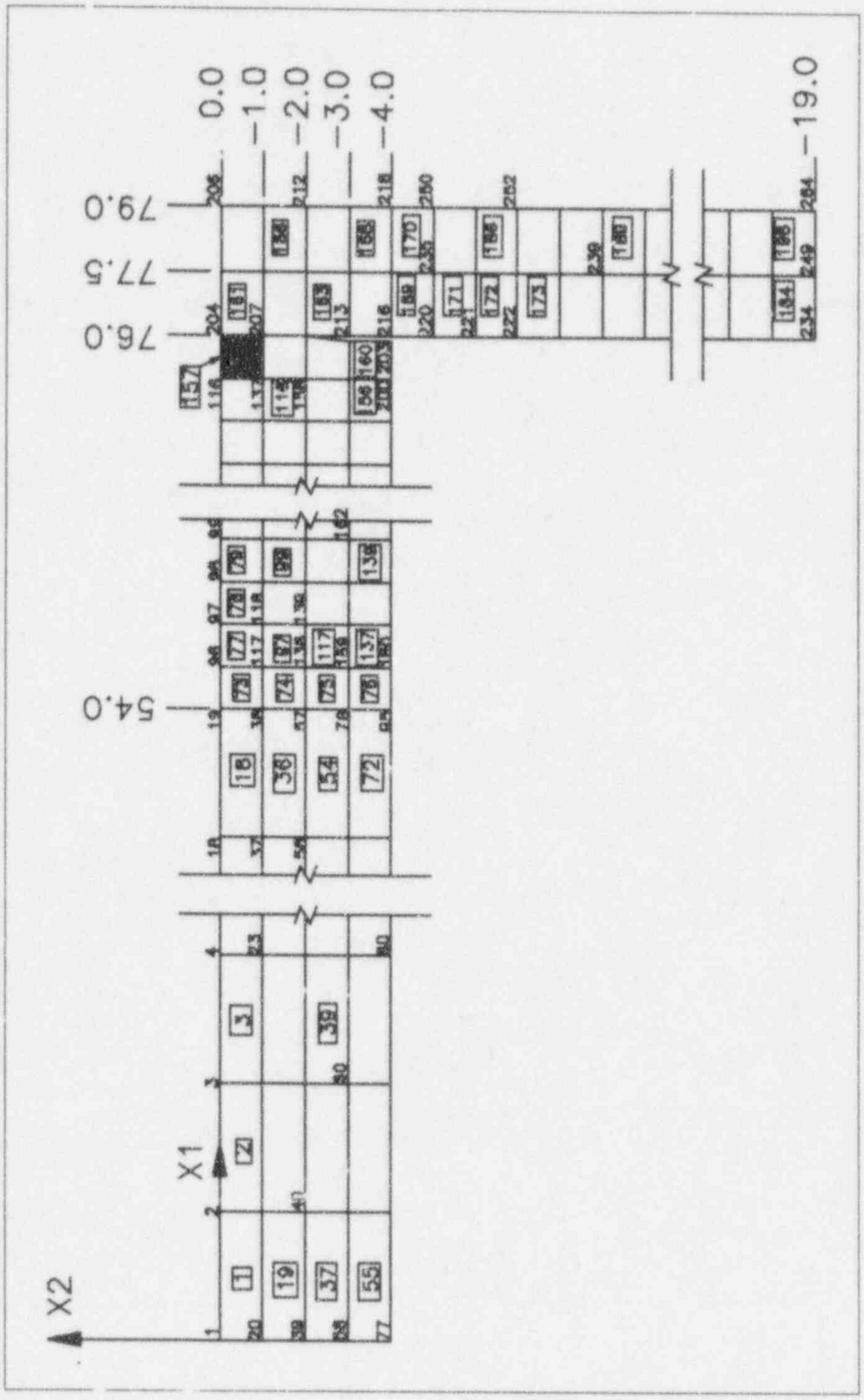


Figure 2.6-3

STARDYNE Finite Element Model - 2-D Plane Strain

APPENDIX 2.11
CALCULATION YRC-1062
EVALUATION OF REACTOR PRESSURE VESSEL SHIPPING CASK FOR 10CFR71 NORMAL CONDITIONS OF TRANSPORT

CALCULATION/ANALYSIS REVIEW FORM

CALCULATION NO. YRC-1062 REVISION NO. 1

COMMENTS	RESOLUTION
Page 24 - Top Cover weight for the FE model is listed as 0.400 but value used in the models was 0.350.	Value on figure changed from 0.400 to 0.350 TRL 11/1/96 [initials] 11-4-96
Pages 26 & 27- Revise "Hot Thermal" paragraph for changes in ΔT from thermal calculation.	Revised TRL 11/1/96 [initials] 11-4-96
Page 26 - Provide information on wire rope location and size.	Reference added for wire rope TRL 11/1/96 [initials] 11-4-96
Page 38 - Fatigue usage determination should consider cycles taking place during storage.	Section on fatigue revised TRL 11/1/96 [initials] 11-4-96
Page B2 - Remove nodes 501 through 706 from model for X1 load case.	Node removed from input file TRL 11/1/96 [initials] 11-4-96
Generic - minor editorial comments, as noted.	Comments incorporated TRL 11/1/96 [initials] 11-4-96

Identify method(s) of review:

- ☑ Calculation/analysis review
- ☐ Alternative calculational method
- ☐ Qualification testing

Resolution By: TR LeFrancois 11/1/96
Preparer/Date

Comments Continued on Page: N/A

Concurrence with Resolution Lilian Schendel / 11-4-96
(LILIAN V. SCHENDEL) Reviewer/Date

EVALUATION OF COMPUTER CODE USE

CALCULATION NO. YRC-1062 REVISION NO. 1

List the computer codes used and complete the following:

Code Name/Version	Approved per WE-108[1]		Appropri-ateness Verified[2]		Outstanding SPRS[3]	
	Yes	No	Yes	No	Yes	No
STARDYNE-4.4, WINDOWS/DOS		X	X		X	

[1] Refer to Section 4.1.4.4, Bullet 3, of this procedure.

[2] Refer to Section 4.1.4.4, Bullet 2, of this procedure.

[3] Refer to WE-108, Section 4.4

If a computer code was not verified per WE-10ɞ, or if there are outstanding SPRs, state below why it is appropriate.

Code Name	Appropriateness
STARDYNE-4.4,	Outstanding SPRs reviewed, none are applicable to
WINDOWS/DOS	use of code herein. Code has been verified in
	Calc. No. YRC-1140 in accordance with WE-103. All
	uses of the code herein fall within the verification
	as performed in YRC-1140.

RECORD OF REVISION

REVISION: 0

PURPOSE: Design and evaluate the Reactor Pressure Vessel (RPV) shipping cask in accordance with the normal conditions of transport per 10 CFR 71.71(c)(1),(2),(3),(4),(5), and (10).

ADDED PAGES: N/A

MODIFIED PAGES: N/A

REVISION: 1

PURPOSE: The Reactor Pressure Vessel (RPV) shipping cask designed in Revision 0 is reevaluated using lower density concretes in the RPV interior (30 pcf vs. 35 pcf) and the RPV/Cask annulus (80 pcf vs. 150 pcf).

ADDED PAGES: 40

MODIFIED PAGES: 1,4-9,16,19-21,23-27,31-35,38,39,B2, B5,B8,B9, B11,B13,B14,B16,B17,B20-B23,B25,C2

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PURPOSE

The YNPS is permanently shut down. Irradiated/contaminated components, including the reactor pressure vessel (RPV) will be removed and shipped to low level radioactive waste disposal facilities. Transport of the RPV to the disposal facility will be made in a package meeting the requirements of 10 CFR Part 71 (Reference 1). This calculation designs the shipping cask for the package in accordance with ASME, Section VIII for "normal conditions of transport" specified in 10 CFR 71, as supplemented by Regulatory Guides (R.G.) 7.6, 7.8, and 7.9 (References 4, 5, 6).

The shipping cask is evaluated as above with lower density concretes in the RPV interior (30 pcf vs. 35 pcf) and the RPV/Cask annulus (80 pcf vs. 150 pcf). The cask thermal evaluation is performed using the results of the revised thermal analysis (Reference 11). 1

CONCLUSION

The shipping cask, as shown in Figures 1.2.1 through 1.2.4 of the SAR (Reference 14), is designed in accordance with ASME VIII, Division 2, for the normal conditions of transport per 10 CFR 71.71(c). Maximum stresses are less than the limits of ASME Section VIII and R.G. 7.6 (Reference 4). The above conclusions are not changed by the use of lower density concretes. 1

REFERENCES

1. 10 CFR Part 71, *Packaging and Transportation of Radioactive Material,* November 30, 1992.
2. ASME Boiler & Pressure Vessel Code, 1989 ed., Sections II, III, and Section VIII, Divisions 1 and 2.
3. STARDYNE User Information Manual, Revision E.
4. USNRC, Regulatory Guide 7.6, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels,* Rev. 1, March, 1978.
5. USNRC, Regulatory Guide 7.8, *Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material,* Revision 1, March, 1989.
6. USNRC, Regulatory Guide 7.9, *Standard Format and Content of Part 71 Applications for Approval of Packaging for Radioactive Materials,* Proposed Revision 2.
7. ANSI N14.2, *Proposed American National Standard, Tiedown for Truck Transport of Radioactive Materials,* Draft 5, Rev. 0, Feb. 1986.
8. Drawings: As Referenced
9. ASCE Manual No. 58, *Structural Analysis and Design of Nuclear Plant Facilities,* 1980.

REFERENCES (continued)

10. E-Mail, C. Child to Dist., *14-195 Liner Data Retransmittal*, January 20, 1995.
11. YAEC Calculation No. YRC-1058, *Thermal Evaluation for Reactor Vessel Shipment*, Revision 1.
12. YAEC Calculation No. YRC-1066, *Design of RPV Shipping Cask Transport Tie-Down System*, Revision 1.
13. Association of American Railroads, Operations and Maintenance Department, Mechanical Division, Section No. 1, *General Rules Governing the Loading of Commodities on Open Top Cars*, Republished September 1, 1994.
14. YAEC, *Safety Analysis Report for Transport of the Yankee Reactor Pressure Vessel Package*, Revision 3.
15. YAEC Calculation No. YRC-1140, *Verification of STARDYNE 4.4, Windows/DOS for RPV Transportation Calculations*, Revision 0.

COMPUTER CODES

STARDYNE 4.4, Windows/DOS is used herein. STARDYNE 4.4 was validated in YRC-1140 (Reference 15) in accordance with WE-103 using an American Megatrends 486DX2-66 with integral math co-processor and MS-DOS, Version 6.22. The computing environment used for this calculation is an American Megatrends 486DX2-66 Mhz with integral math co-processor and MS-DOS, Version 6.22. There is therefore no change in computing environment. Computer code file names, sizes, and dates as used herein are the same as those used in YRC-1140 for the validation of STARDYNE 4.4. STARDYNE 4.4 is used in this calculation within the restrictions/parameters established by YRC-1140. The use of STARDYNE 4.4 for this calculation is therefore appropriate and validated per WE-103.

SAFETY CLASSIFICATION

The shipping cask is NNS, however, it will be used to ship radioactive materials, and, therefore must meet the requirements of 10 CFR Part 71, including the quality assurance requirements of Subpart H. Subpart H requires that the cask design be performed in accordance with a qualified QA program. Therefore, the cask will be considered safety related, and all calculations will be performed in accordance with WE-103.

METHOD OF ANALYSIS AND DESIGN INPUT

GENERAL

The cask is evaluated using linear elastic analysis methods. Both manual and computer (STARDYNE) calculations are used. The cask will be designed in accordance with ASME VIII, Division 2, Part AD, "Design Requirements" (Reference 2).

ALLOWABLE STRESSES

Maximum stress intensity, S, for normal conditions of transport is limited to $1.0S_m$ for membrane stress and $1.5S_m$ for membrane + bending stress (References 2 and 4). The maximum stress intensity is defined (Ref. 2, 4) as 2 x the maximum shear stress, which is equal to the largest algebraic difference between the maximum and minimum principal stresses.

GEOMETRY & MATERIAL PROPERTIES

The geometry of the cask is shown in Figure 1.2.1 of the SAR (Reference 14). The cask will be fabricated from A516, Grade 70 material (Reference 14). Material properties (S_m, E, α) are from ASME III, Division I Appendices and ASME VIII, Part AM.

DESIGN LOADING CONDITIONS

Design loads provided below for "Normal Conditions of Transport" are obtained from 10 CFR 71.71, as interpreted by RG 7.8.

10 CFR 71.71(c)(1), HEAT

T_{MAX} = 100°F (Ambient) + Heating in excess of 100°F due to insolation and internal heating.

10 CFR 71.71(c)(2), COLD

T_{MIN} (Ambient) = -20°F (normal)/ -40°F (maximum)

10 CFR 71.71(c)(4), INCREASED EXTERNAL PRESSURE

$P_{AMBIENT}$ = 20.0 psia

10 CFR 71.71(c)(3), REDUCED EXTERNAL PRESSURE

$P_{AMBIENT}$ = 3.5 psia

10 CFR 71.71(c)(5), TRANSPORT SHOCK AND VIBRATION

Accelerations provided by AAR Section 1 (Reference 13) are used for shock and vibration normally incident to transportation. These values are higher than those required by ANSI N14.2 (Reference 7) for truck transport. Neither ANSI nor AAR require any combinations of the accelerations.

a (longitudinal) = 3.0G
a (lateral) = 2.0G
a (vertical) = 2.0G (Up)

DESIGN LOADING CONDITIONS (continued)

10 CFR 71.71(c)(6), WATER SPRAY

There is no effect from a water spray which simulates a 2"/hr rainfall on a 3" thick welded steel cask. This loading condition is not evaluated.

10 CFR 71.71(c)(7), FREE DROP

This condition is not a normal condition of transport for the YNPS RPV package - see the SAR (Reference 14), Section 2.6.7.

10 CFR 71.71(c)(8), CORNER DROP

This is applicable to fiberboard or wood containers only.

10 CFR 71.71(c)(9), COMPRESSION

This is applicable for packages weighing ≤ 5,000 kg (11,023 lbs). The package will weigh more than 700,000 lbs. This loading is not evaluated.

10 CFR 71.71(c)(10), PENETRATION

The package must be evaluated for impact of a 1¼" dia., 13 lb steel cylinder dropped from a height of 40 inches.

INTERNAL PRESSURE

Initial internal pressure @ 70°F is taken as 1 atmosphere (14.7 psia). Internal pressures at different temperatures are calculated in Ref. 11.

LOADING COMBINATIONS

RG 7.8, Table 1 (Reference 5) is used to define the appropriate loading combinations.

DESCRIPTION	COMBINATION
Hot Environment (100°F)	Heat + Max. Internal Pressure
Cold Environment (-40°F)	Max. Cold + Min. Internal Pressure
Increased External Pressure	Normal Cold + Incr. External Pressure + Min. Internal Pressure
Decreased External Pressure	Heat + Reduced Ext. Pressure + Max. Internal Pressure
Shock and Vibration - Hot	Heat + Acceleration + Max. Internal Pressure
Shock and Vibration - Cold	Normal Cold + Acceleration + Min. Internal Pressure

MATERIAL PROPERTIES

See page 27 for coefficients of thermal expansion and moduli of elasticity used in the evaluations.

PACKAGE WEIGHT

CASK

$$W_{CYL} = \frac{\pi}{4}(D_o^2 - D_i^2)(\gamma H) = \frac{\pi}{4}(158^2 - 152^2)(0.283\ lb/inch^3)(28'\text{-}3'' x\ 12)$$

$\underline{W_{CYL} \cong 140{,}150\ lbs}$

$\underline{W_{INNER\ SHIELD}} = (\pi/4)(152^2 - 150^2)(0.283)(10'\text{-}4''\ x\ 12) \cong \underline{16{,}650\ lbs}$

$\underline{W_{BOT.\ DISK}} = (\pi/4)D_o^2 t\gamma = (\pi/4)(152)^2(4'')(0.283) \cong \underline{20{,}540 lbs}$

$\underline{W_{CASK\ COVER}} = [(\pi/4)(152'')^2(4'')](0.283) \cong \underline{20{,}540\ lbs}$

$W_{CONE\ PLATES} = (Surface\ Area\ Frustrum)(t)(\gamma) = \pi(R_1 + R_2)S(t)(\gamma)$

$S = (3'\text{-}8'')\sqrt{2} = 5.185'$; $SA_{FRUSTRUM} = \pi\left(\frac{12'\text{-}8'' + 5'\text{-}4''}{2}\right)(5.185') = 146.6$

$\underline{W_{CONE\ PLATES}} = 146.6\ ft^2[(1/2)/12](490\ lb/ft^3) \cong \underline{2990\ lbs}$

$\underline{W(Studs, RPV \rightarrow Cover)} = [\frac{\pi}{4}(5\frac{1}{4}'')^2(22'')(0.283)](22\ studs) \cong \underline{3500\ lbs}$

$\underline{W(Nuts)}$ = 22 Nuts(43 lbs/Nut) ≅ $\underline{900\ lbs}$ (B+W Dwg. 34993)

Total Cask Weight = $W_{CYL} + W_{INNER\ SHIELD} + W_{BOT.DISK} + W_{COVER} + W_{CONE\ PLS} + W_{STUDS} + W_{NUTS}$

= 140,150 +16,650 +20,540 +20,540 +2990 +3500+900

Total Cask Weight = 205,270 lbs

INTERNAL SOLIDIFIED WASTE PACKAGE

OD = 76", H = 77", t = 11 ga (~0.125"), Top & Bottom Covers = 5/16" (Reference 10)

$$W_{LINER} = [\pi(76'')(.125'')(77'') + 2(\frac{\pi}{4})(76^2)(\frac{5}{16})]0.283 = 1450\ lbs$$

$$W_{CONTENTS} = \frac{\pi}{4}(76-2(0.125))^2(77)(\frac{120}{1728}) = 24{,}098\ lbs$$

$W_{MISC} \cong 10\%\ of\ Contents \cong 2500\ lbs$

$W_{TOTAL} = 1{,}450 + 24{,}098 + 2500 = 28{,}048\ lbs$, USE 28,000 lbs

$$Outside\ V_{LINER} = \frac{\pi}{4}(76'')^2(77'')/1728 = 202\ ft^3$$

PACKAGE WEIGHT
REACTOR PRESSURE VESSEL (RPV)

- Vessel = 300,000 lbs (Babcock & Wilcox Dwg. 34975E)

- *Nozzle Plugs* = $(\frac{\pi}{4})(0.283)[7''(19\frac{1}{8}'')^2 + (1'')(23'')^2](8\ Plugs) \approx 5500$

- Insulation (See B&W dwg. 34971E for RPV dimensions)

γ = 11 lb/ft^3

Cylinder: Ht = (26'-9-19/32")-(4'-8-1/2")-(0'-4")-(8'-2-1/2")
≈ 13'-7" (Bottom of Lugs to Bottom Head)
OD(Insul.) = (9'-1") + 2(0'-8") + 2(0'-3") = 131",
ID(Insul.) = 131"- 6" = 125"

W = 11 pcf (13'-7")(π/4)[(131^2 - 125^2)/144]
= 1250 lbs

Bottom Head: W = {(2π/3)[(63-3/8")^3 - (60-3/8")^3]}(11 pcf)/1728
= 460 lbs

Total Vessel Weight = Vessel + Nozzle Plugs + Insulation
= 300,000 + 5,500 + (1250 + 460)

TOTAL REACTOR VESSEL WEIGHT (Including Vessel, Plugs, & Insulation) = 307,210 lbs

CONCRETE (Between Vessel and Cask)

- Interior Vol. of Cask = (π/4)(152/12)^2[(28'-3")-(0'-4")-(0'-4")]
- (π/4)(152^2-150^2)(10'-4")/144
= 3442 ft^3

- Exterior Vol. of RPV and Insulation

Bottom Head: R = (4-8-1/2) + (0-4) + (0-3) = 5-3-1/2
V = (2π/3)(5-3-1/2)^3 = 308.5 ft^3 (w/insulation)

Cylinder: V = (π/4)(131/12)^2(13-7) = 1271.4 ft^3

PACKAGE WEIGHT

CONCRETE (Between Vessel and Cask) (continued)

Flange Ring: (No Insul.) V = Vol. Ring + Interior Vol.

W(flng. ring) = 300k - 63k - 78k -76k - 18k = 65k
(RPV) (Mk13) (Mk 8) (Mk 5) (Bot)

V(flng. ring) = 65,000 lb/490 pcf = 132.7 ft^3

V(interior) = (2-6-1/2)(π/4)(8'-8")^2 = 150 ft^3
(B&W Dwg. 34974E-5)

V(Flange Ring) = 132.7 + 150 = 282.7 ft^3

References:
Shell Course Mk. 5 Wt - B&W Dwg. 34977E-3
Shell Course Mk. 8 Wt - B&W Dwg. 34976E-4
Shell Course Mk. 13 Wt - B&W Dwg. 34978E-3
Bottom Head Wt. - See Page 17

Nozzle Ring: (No Insul.) V = Vol. Ring + Interior Vol.

W(nozzle ring) = 63,000 lb (B&W 34978E-3)

V(nozzle ring) = 63,000 lb/490 pcf = 128.6 ft^3

V(interior) = (4-9-7/8)(π/4)(9'-1")^2 = 312.5 ft^3
(B&W Dwg. 34971E)

V(Nozzle Ring) = 128.6 + 312.5 = 441.1 ft^3

Total RPV Exterior Volume = 308.5 +1271.4 +282.7 +441.1 = 2303.7ft^3

- Empty Cone Vol. @ Bot. of Cask

$V = V_{CYL} - V_{FRUS}$

$V_{FRUS} = (\pi/3)(3.67')[(6.33')^2 + (2.67')^2 + 6.33'(2.67')]$
$= 246.3\ ft^3$

$V_{CYL} = (\pi/4)(12.67')^2(3.67')$
$= 462.7\ ft^3$

$V = 462.7 - 246.3 = 216.4\ ft^3$

PACKAGE WEIGHT
CONCRETE (Between Vessel and Cask) (continued)

- Concrete Volume = Cask Int. Vol. -RPV Ext. Vol. -Empty Cone Vol.

$$= 3442 - 2304 - 216 = 922 \text{ ft}^3$$

- Concrete Weight = (80 lb/ft^3)(922 ft^3) (See SAR Figure 1.2-1 for concrete density) 1

Total Annulus Concrete Weight = 73,760 lbs

CONCRETE (Vessel Interior)
Bottom Head

$$V_{Interior} = \frac{\pi}{3}h^2(3r - h); \quad h = (4\text{-}8\frac{1}{2}) - 1'\text{-}2'' = 3\text{-}6\frac{1}{2}$$

$$= \frac{\pi}{3}(3\text{-}6\frac{1}{2})^2(3(4\text{-}8\frac{1}{2}) - 3\text{-}6\frac{1}{2}) = 139.0 \ ft^3$$

Flange Ring to Bottom Head

$$V = \frac{\pi}{4}(Vessel\ ID)^2(h); \quad h = (7'\text{-}10'' + 7'\text{-}10'' + 4'\text{-}10'') = 20'\text{-}6'' \ (B{+}W\ 34974E)$$

$$V = \frac{\pi}{4}(9'\text{-}1'')^2(20'\text{-}6'') = 1328 \ ft^3$$

Flange Ring

$$V = (\frac{\pi}{4})(\frac{h}{2})[(8'\text{-}3'')^2 + (\frac{8\text{-}3 + 9\text{-}1}{2})^2]; \quad h = (2\text{-}6\frac{3}{4}) - (0\text{-}8\frac{1}{4}) = 1\text{-}10\frac{1}{2}$$

$$V = \frac{\pi}{4}(\frac{1.875}{2})[(8.25)^2 + (8.67)^2] \approx 106 \ ft^3$$

Total Interior Concrete Volume = 1328.0 + 106.0 + 139.0 - 202.0
= 1371 ft^3

Interior Concrete Weight = 1371 ft^3(30 pcf) (See SAR Figure 1.2-1 for interior concete density) 1

Total Interior Concrete Weight = 41,130 lbs

PACKAGE CENTER OF GRAVITY
VESSEL

(Note: All C.G. distances are from bottom of vessel)

Bottom Head (To Weld) (B&W Dwg. 34974E)

$$V_o = \frac{1}{3}\pi h_o^2(3R_o - h_o),\quad h_o = 5'\text{-}0\frac{1}{2}''(R_o) - 1'\text{-}3'' = 3'\text{-}9\frac{1}{2}''$$

$$= \frac{1}{3}\pi(3-9\frac{1}{2})^2[3(5-0\frac{1}{2}) - (3-9\frac{1}{2})] = 170.63\ ft^3$$

$$V_i = \frac{1}{3}\pi h_i^2(3R_i - h_1),\quad h_i = (4-8\frac{1}{2}) - (1\text{-}3) = 3'\text{-}5\frac{1}{2}''$$

$$= \frac{1}{3}\pi(3-5\frac{1}{2})^2[3(4-8\frac{1}{2}) - (3-5\frac{1}{2})] = 133.60\ ft^3$$

$V\ (Bottom\ Head) = 170.63 - 133.60 = 37.03\ ft^3$

$W\ (Bottom\ Head) = 37.03\ ft^3 \times 490\ lb/ft^3 = 18{,}150\ lbs$, *Say* 18,000 *lbs*

C.G. @ $h_o/2$; h_o = 3-9-1/2; C.G. = 1.896'

Shell Course Mk. 5 (B&W Dwg. 34977E-3)

W = 76,000 lbs; C.G. @ [(7-10)/2] + (3-9-1/2) = 7.708'

Shell Course Mk. 8 (B&W Dwg. 34976E-4)

W = 78,000 lbs; C.G. @ [(7-10)/2] + (7-10) + (3-9-1/2) = 15.542'

Shell Course 13 (Nozzle Ring) (B&W Dwg. 34978E)

W = 63,000 lbs; C.G. @ Nozzles = (26-10) - (4-8-7/8) = 22.094'
(34974E)

Flange Ring (B&W Dwg. 34975E)

W = 65,000 lbs (see page 15);

Estimate C.G. @ 2/3 Ring Height = (26-10) - 1/3(2-6-3/4) = 25.979'
(34974E)

PACKAGE CENTER OF GRAVITY
VESSEL (continued)

RPV CENTER OF GRAVITY			
COMPONENT	WEIGHT	DISTANCE	WEIGHT x DISTANCE
Bottom Head	18000	1.896	34128
Shell Mk. 5	76000	7.708	585808
Shell Mk. 8	78000	15.542	1212276
Nozzle Ring	63000	22.094	1391922
Flange Ring	65000	25.979	1688635
TOTALS	300000		4912769
CENTER OF GRAVITY = 16.38 FT (From Bottom of Vessel)			

RPV CG (From Cask Bottom) = 16.38' + 0.50' (clearance) + 0.33' (bottom disk) + 0.25' (insulation)
= 17.46'

CASK (From Cask Bottom)

- Cylinder

  W = 140,150 lbs (Page 13)

  C.G. = (28-3)/2 = 14.125' (From Cask Bottom)

- Cask Cover, Studs, and Nuts

  W = 20,540 + 3,500 + 900 = 24,940 lbs (Page 13)

  C.G. = (28-3) - (0-4/2) = 28.083' (From Cask Bottom)

- Cask Bottom Disk

  W = 20,540 lbs (Page 13)

  C.G. = (0-4)/2 = 0.167' (From Cask Bottom)

- Internal Shield

  W = 16,650 lbs (Page 13)

PACKAGE CENTER OF GRAVITY
CASK

- Internal Shield (continued)

C.G. = Ht. @ Bottom of Lugs - 2" - Ht. of Shield/2
Ht. @ Bottom of Lugs = 28-3 - 0-4 - 8-2½ = 19.708' (B&W 34975E)
Ht. of Shield = 10-4, 10-4/2 = 5.167'

C.G. = 19.708 - 0.167 - 5.167 = 14.374' (From Cask Bottom)

- Cone Plates

W = 2990 lbs (Page 13)
C.G. = ⅔(Cone Ht.) + (Bot. Disk) = ⅔(3-8) + (0-4)
= 2.778' (From Cask Bottom)

INTERNAL SOLIDIFIED WASTE PACKAGE

W = 28,000 lbs (Page 13)

C.G. = 10'-3¾" + [(77/12)/2] = 13'-6¼" below flange
C.G. = (28'-3" - 0'4" - 13'-6¼") = 14.396' (From Cask Bottom)

CONCRETE (Between RPV and Cask)

- Between RPV Lugs and Bottom Disk

H = 19.708' - 0.333' = 19.375'

(See Pg. 15 for Volumes)

$V_{CYL.INT.} = \pi/4(12'-8")^2(19.375') = 2442\ ft^3$

$V_{RPV\ EXT.} = 1271.4$ (Cylinder) + 308.5 (Bot. Head) = 1580 ft^3

$V_{INNER\ SHIELD} = \pi/4(152^2-150^2)(124")/1728 = 34\ ft^3$

Empty Vol. @ Bot. of Cask = 216 ft^3

Concrete Volume = 2442 - 1580 - 34 - 216 = 612 ft^3;

W = 612 ft^3(80 lb/ft^3) = 48,960 lbs; | 1

C.G. = (19.375'/2) + (0'-4") = 10.021' (From Cask Bottom)

- Above RPV Lugs

W = 73,760 - 48,960 = 24,800 lbs (Page 16 & Above) | 1

PACKAGE CENTER OF GRAVITY

CONCRETE (Between RPV and Cask)

- Above RPV Lugs (continued)

C.G. assumed @ midway between lugs and top of inside of cask

Top of Inside of Cask = 28'-3" - 0'-4" = 27.917'
Lugs = 19.708' (See Above)

C.G. = 19.708' + [(27.917' - 19.708')/2]
= 23.813' (From Cask Bottom)

CONCRETE (Vessel Interior)

- Bottom Head

W = 139.0 ft^3 (30) lb/ft^3) = 4,170 lb (Vol. - Page 16) | 1

C.G. = 4"(Bot. Disk) + 6"(Clearance) + 3"(Insulation) + 4"(Head Thick.) + ⅔(3-6½) (B&W 34975E)

C.G. = 3.777' (From Cask Bottom)

- Cylinder + Flange Ring

W = (1371 - 139)(30) = 36,960 lbs (Vol. - Page 16) | 1

C.G. Assumed @ Midheight of Cylinder and Flange Ring

C.G. = (28-3) -(0-4) -½(26-6 - 3-6½)
= 16.437' (From Cask Bottom)

MISCELLANEOUS

- Nozzle Plugs

W = 5500 lbs (Page 14)

C.G. = 28-3 - 0-4 (Top Cover) - 4-8½ (34975E)
= 23.208' (From Cask Bot.)

- Insulation

W = 1250 + 460 = 1710 lbs, Say 1700 lbs (Page 14)

C.G. assumed @ midpoint between bottom of insulation and bottom of RPV lugs = [(13'-7") + (5'-3-1/2")]/2 + (0'-4") + (0'-6")
CYL. HT. BOT. HEAD BOT. DISK CLEARANCE

C.G. = 10.271' (From Cask Bottom)

SUBJECT REACTOR PRESSURE VESSEL SHIPPING CASK

PREP BY DRL DATE 3/20/95 REVIEW BY JEP DATE 3/21/95 WORK ORDER NO DD28
REV. 1 DRL 11/3/96 JO 11-4-96 DDCR

COMPLETE PACKAGE CG (RPV, Insul., Cask, Concrete, Misc.)

PACKAGE CENTER OF GRAVITY			
COMPONENT	WEIGHT (lbs)	DISTANCE (ft)	WEIGHT x DISTANCE (ft-lbs)
Cask Cylinder (3")	140,150	14.125	1,979,619
Cask Cover, Studs & Nuts	24,940	28.083	700,390
Cask Bottom (4")	20,540	0.167	3,430
Internal Shield (1" th.)	16,650	14.374	239,327
Cone Plates (½")	2,990	2.778	8,306
Internal Solidif. Waste Pkg.	28,000	14.396	403,088
Vessel	300,000	17.46	5,238,000
Nozzle Plugs	5,500	23.208	127,644
Insulation	1,700	10.271	17,461
Concrete Btwn. RPV and Cask (80 pcf)			
Below RPV Lugs	48,960	10.021	490,628
Above RPV Lugs	24,800	23.813	590,562
Concrete - RPV Interior (30 pcf)			
RPV Bot. Head	4,170	3.777	15,[illegible]0
RPV Cyl. + Flange Ring	36,960	16.437	607,512
TOTALS	655,360		10,421,717

CENTER OF GRAVITY = 15.90 FT. (Above Cask Bottom)
= 190.8 IN.

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
MODEL GEOMETRY AND PROPERTIES

General

Two STARDYNE FEMs are used for evaluation. The first FEM models the entire shipping cask, including the covers. STARDYNE QUADH and TRIAB 5 dof, plate elements are used for the model. This model is used to evaluate all load cases. This model assumes that the inertia of the contents of the package (RPV, Concretes, Solidified Waste Pkg.) acts directly with the cask cylinder. This will occur during lateral (X1) or vertical (X2) accelerations. For this to occur during longitudinal (X3) accelerations, sufficient friction, or other load transfer mechanisms, must be developed between both the RPV and annulus concrete, and the annulus concrete and the cask.

The second STARDYNE FEM was developed assuming that the longitudinal inertia loads from the RPV, the interior concrete, and the waste package are transmitted directly to the top cover. Sufficient load transfer is assumed to occur between the annulus concrete and the cask cylinder such that no load is imposed on the cover by the annulus concrete. 2-D plane strain elements are used with the modulus of elasticity (E) and Poisson's ratio adjusted per Section M of the STARDYNE User's Manual (Reference 3). A 1" thick "slice" from the cover centerline to the outside of the cask is modeled. A 19" length of the cask cylinder is included. Symmetrical boundary conditions are used at the cover centerline. Loads from the RPV, internal grout, and waste package are assumed to be transferred to the cover at the RPV flange centerline (54" from the cover centerline).

The STARDYNE POST program is used to combine loads and calculate stress intensities. Plots/sketches of the models are shown in Attachment A.

Maximum stress intensity due to longitudinal accelerations is the maximum stress from either model.

3-D Plate Element Model

- Overall Dimensions

R = R @ mid-thickness of cylinder = [(12'-8")/2] + [(0'-3")/2]
(I.R.) (t)

R = 77.5"

H = Distance From Mid-Cover to Mid-Bottom Disk
= (28'-3") - [(0'-4")/2] - [(0'-4")/2]
= 27'-11" = 335"

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
MODEL GEOMETRY AND PROPERTIES
3-D Plate Element Model (continued)

- Top Cover to Cylinder Weld

The top cover to cylinder weld is a 2" deep, partial penetration weld (See SAR Figure 1.2-4). This weld will resist both shear and moment. Since the full thickness of the top will not be mobilized by the weld, the thickness of the outer ring of top cover elements (Elements 261-280) is reduced to match the thickness of the weld. This modelling provides reasonably accurate, but conservative stresses in the reduced thickness elements.

- Weight, Density & C.G. (See sketch on next page)

- Top Cover

$W = 24,940$

$\gamma = W/V = (24,940)/[\pi(77.5)^2(4)] = 0.330\ lb/inch^3$

- Rings 1, 2

$W = 65,000\ (Flng.\ Ring) + 150 ft^3 (30\ lb/ft^3)\ (Internal\ Conc.) + \left[\frac{\pi(76)^2(30.5'')}{1728} - 282.7\right] 80\ lb/ft^3\ (Annulus\ Conc.)$

$W = 65,000 + 4,500 + 3,006 = 72,506\ lbs$

$\gamma = W/V + 0.283 = \frac{72,506}{\pi(79^2 - 76^2)(48)} + 0.283 = 1.317\ lb/inch^3$

- Rings 3, 4

$W = 63000\ (Nozzle\ Ring) + 5500\ (Nozzle\ Plugs) + 312.5(30)\ (Intern\ Conc.) + \left[\left(\frac{\pi(76)^2(58'')}{1728} - 441.1\right)80\right]\ (Annulus\ Conc.)$

$W = 63,000 + 5,500 + 9,375 + 13,437 = 91,312\ lbs$

$\gamma = W/V + 0.283 = \left[\frac{91,312}{\pi(79^2 - 76^2)48}\right] + 0.283 = 1.585\ lb/inch^3$

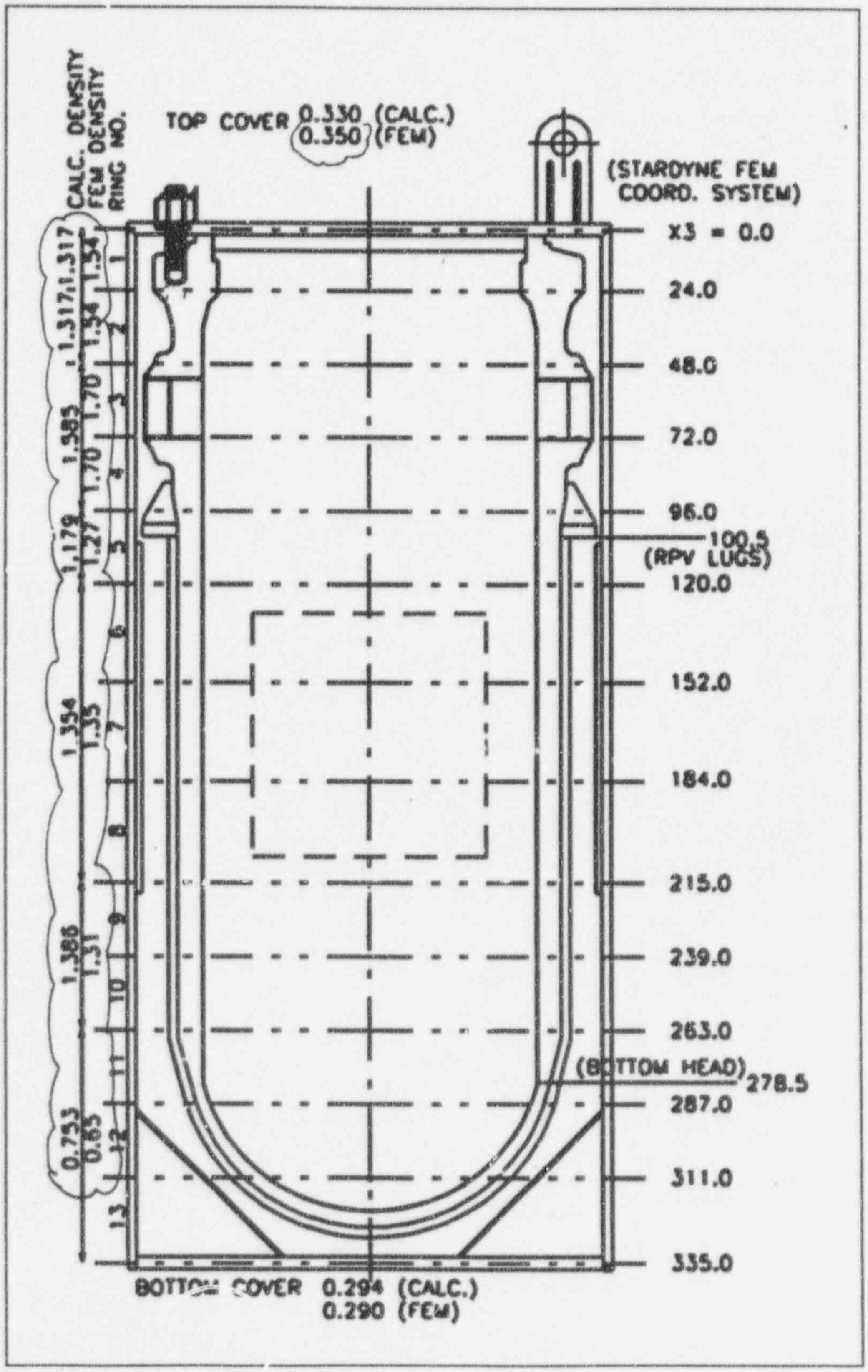
- Ring 5

$W = \pi(62.5^2 - 54.5^2)(24)(0.283)(RPV) + \pi(76^2 - 75^2)(24)(0.283)(Intr.\ Shield) + [\pi(54.5)^2(24)(30)(Intr.\ Conc.) + \pi(74^2 - 65.5^2)(24)(80)(Ann.\ Conc.) + \pi(65.5^2 - 62.5^2)(24)(11)(Insul.)]/1728$

$W = 19,972 + 3,222 + 3,888 + 4,139 + 184 = 31,405\ lbs$

$\gamma = W/V + 0.283 = \left[\frac{31,405}{\pi(79^2 - 76^2)(24)}\right] + 0.283 = 1.179\ lb/inch^3$

Calculated Densities vs. FEM Densities



SUBJECT REACTOR PRESSURE VESSEL SHIPPING CASK

PREP BY DRL DATE 3/20/95 REVIEW BY JEP DATE 3/20/95 WORK ORDER NO DD28
REV. 1 DRL 11/3/96 JS 11-4-96 DDCR

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
MODEL GEOMETRY AND PROPERTIES
3-D Plate Element Model

- Weight, Density & C.G. (continued)

- Rings 6, 7, 8

$W = \pi(62.5^2 - 54.5^2)(114.5-24)(0.283)(RPV) + [16,650-3,222]$ $(Shield) + [[\pi(54.5)^2(90.5) - 202(1728)](30)(Intern\ Conc.) +$ $[\pi(74^2 - 65.5^2) \times (90.5)(80)](Ann.\ Grout) + [\pi(65.5^2 - 62.5^2)$ $(90.5)(11)(Insulation)]]/1728 + (28,000)(Solidif.\ Liner)$

$W = 75,311 + 13,428 + 8,601 + 15,607 + 695 + 28,000 = 141,642\ lbs$

$$\gamma = W/V + 0.283 = [\frac{141,642}{\pi(79^2 - 76^2)(90.5)}] + 0.283 = 1.354\ lb/inch^3$$

- Rings 9, 10

$W = \pi(62.5^2 - 54.5^2)(278.5-215)(0.283)(RPV) + [\pi(54.5)^2(63.5'') \times$ $(30)(Internal\ Conc.) + \pi(76^2 - 65.5^2)(63.5)(80)(Annulus\ Conc.)$ $+ \pi(65.5^2 - 62.5^2)(63.5)(11)(Insulation)]/1728$

$W = 52,842 + 10,287 + 13,722 + 488 = 77,339$

$$\gamma = W/V + 0.283 = [\frac{77,339}{\pi(79^2 - 76^2)(48)}] + 0.283 = 1.386\ lb/inch^3$$

1

- Rings 11, 12, 13

$W = 18,000(Bot.\ Head) + 139ft^3(30)(Bot.\ Head\ Conc.) + [73,760(Pg.$ $3,006(1,2) - 13,437(3,4) - 4,13?(5) - 15,607(6-8) - 13,722(9,10$ $(Annulus\ Conc.) + 460(Insul) + 2990(ConePls)$

$W = 18,000 + 4,170 + 23,849 + 460 + 2990 = 49,469\ lbs$

$$\gamma = W/V + 0.283 = \frac{49,469}{\pi(79^2 - 76^2)(335-263)} + 0.283 = 0.753\ lb/inch^3$$

- Bottom Cover

$\gamma = (79^2/77.5^2)(0.283) = 0.294\ lb/inch^3$

NOTEWELL: The above densities are used as a guide for the STARDYNE finite element model (F.E.M). The F.E.M. is not a perfect cylinder, and if the above calculated densities are specified in the F.E.M., the calculated c.g. and total weight of the model will vary slightly from the values presented on page 21. Therefore, the model densities for the cylinder are varied from the above densities as required to match the calculated model weight and c.g. with the total package weight and overall package c.g. on page 21.

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
MODEL GEOMETRY AND PROPERTIES
3-D Plate Element Model

- Weight, Density & C.G. (continued)

STARDYNE FEM Weight = 654,740 lbs vs. 655,360 calc. (Pg. 21) (OK)
STARDYNE FEM C.G. = 145.6" from middle of top cover
= (28'-3" x 12) - (4"/2) - (145.6") from bottom of cask
= 191.4" vs. 190.8" calculated (Page 21) (OK)

- Boundary Conditions

- General: The shipping cask will be retained on its transporter/ railway car using wire ropes, saddles, and end stops, collectively known as the tiedown system. The tiedown system is evaluated and designed in calculation YRC-1066 (Ref. 12) and shown on drawings 9699-FM-17G,H,I, and J. The wire rope locations and size included in the FEM are consistent with YRC-1066 and the drawings. The wire ropes only resist loads which result in tension in the ropes. The saddles only resist loads which result in compressive loads normal to the saddle at the point of application. The end stops resist loads only in the -X3 (top) or +X3 (bottom) direction. Accounting for the one directional nature of these restraints was done through iteration. The model is initially restrained at all wire rope/ saddle/end stop locations, then restraints were removed at the locations where loads/deflections are in the wrong direction (e.g. wire rope in compression or tensile load on a saddle), and the model rerun with the new restraint configuration.

- Hot Thermal: The wire ropes and saddles are assumed to be at the same temperature as and have coefficients of thermal expansion the same as the cask. Therefore, the ropes and saddles will not restrain thermal growth, and are not included in the model. The longitudinal thermal growth of the cask from ambient (100°F) to maximum (194°F) is minimal (0.2"). The endstops will be gapped and will not provide restraint. The model is restrained only as required to insure structural stability.

- Cold Thermal (Max and Normal), External Pressure (Increased and Decreased): No restraint will be provided by the ropes, saddles, or stops for these loads. The model was restrained only as required to insure structural stability.

- X1, X2, X3 Accelerations: Wire ropes, saddles, and end stops were included as appropriate.

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
MODEL GEOMETRY AND PROPERTIES
2-D Model

- Boundary Conditions

Symmetric boundary conditions are applied to the model at the cover centerline. The model is also restrained at the bottom of the modeled portion of the cylinder. See the sketch in Attachment A.

- Loading

The 2-D model is used for evaluating longitudinal acceleration loads only. Since the model is a model of a 1" thick slice of the top cover and cylinder, the inertia loads from the RPV, interior concrete, and solidified waste package are converted to a line load applied at the centerline of the RPV flange. A concentrated load equal to 1" of the line load is applied at node 95. An acceleration of 3.0g in the model X2 direction is also applied to the model to account for the inertia of self weight.

F_{X2}(Node 95) = $[3.0(W_{RPV} + W_{INT.CONCRETE} + W_{SOLID.DROSS})]/2\pi(54)$

$= [3.0((300 + 5.5 + 1.7) + (4.17 + 36.96) + (28))]/2\pi(54)$

F_{X2}(Node 95) = 3.327 k = 3327 lbs

Material Properties (Both Models)

- Hot Condition*
  All Elements - 172°F (Outside)/194°F (Inside) (Reference 11)

  $\alpha = 5.87 \times 10^{-6}$ in/in/°F (194°F, ASME VIII, Part AM, Tbl AMG -1, Mat'l Group C, Coefficient B)

  $E = 28.95 \times 10^{6}$ psi (172°F, ASME VIII Div. 2, Part AM, Table AMG-2, Carbon Steel)

- Cold Conditions

  Normal (-20°F): Cylinder, Top and Bottom Heads - -20°F (Outside)/+2°F (Inside)**

  Maximum (-40°F): Cylinder, Top and Bottom Heads - -40°F (Outside)/-18°F (Inside)**

  $\alpha = 5.53 \times 10^{-6}$ in/in/°F (100°F, ASME VIII, Part AM, Tbl AMG -1, Mat'l Group C, Coefficient B)***

  $E = 29.95 \times 10^{6}$ psi (ASME III, Div. 1 Append., Table I -6.0, Carbon Steel @ -40°F)

* Conservatively, the temperature calculated at the centerline of the package is applied to the inside of the cask shell. The actual temperature calculated at the inside of the cask shell is 172°F (Ref. 11).

** Conservatively, the 22°F ΔT calculated from the package centerline to the outside of the cask shell is applied across the thickness of the shell.

*** Conservative since α decreases with decreasing temperature.

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
MODEL GEOMETRY AND PROPERTIES
Material Properties (Both Models) (continued)

- Pressure and Accelerations:

  $E = 29.7 \times 10^6$ psi (ASME VIII Div. 2, Part AM, Table AMG-2, Material Group C @ 70°F)

Allowable Stresses (Both Models)

S_m = 23.3 ksi, -40°F to 188°F (ASME VIII Div. 2, Part AM, Table ACS-1)

k = 1.0 for Load Combinations 1-10 (ASME VIII Div. 2, Part AD, AD -150)

Allow. Membrane Stress Intensity (ASME VIII, Appendix 4, 4-131)

$= kS_m = (1.0)23.3 = 23.3$ ksi (L.C. 1-10)

Allow. Membrane + Bending Stress Intensity (ASME VIII, Appendix 4, 4-133)

$= 1.5kS_m = (1.0)34.95 = 34.95$ ksi (Load Combinations 1-10)

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
LOAD CASES
No.

1. 10 CFR 71.71(c)(1), Heat, T = 100°F (ambient) w/insolation,
   - Assume RR car/transporter/restraint system at ambient temp. (100°F) and thermal properties of RR car/transporter/ restraints approx. equal to cask thermal properties. Cask thermal growth is small, therefore the cask is not restrained by the tie-down system.
   - Assume temperature gradient across cask wall = 172°F (outside) 194°F (inside) for all elements (see Reference 11 for cask temperatures). 1

2. 10 CFR 71.71(c)(2), Cold (-40°F)
   - Assume RR car/transporter/restraints at ambient temp. and thermal properties of RR car/transporter/restraints approx. equal to cask thermal properties. Therefore, there is no restraint of cask thermal movement.
   - Assume temperature gradient across cask wall (cylinder, top and bottom disks) = -40°F (outside) to -18°F (inside). Temperature @ zero thermal stress/growth = 70°F. 1

3. 10 CFR 71.71(c)(2), Cold (-20°F)
   - Same assumptions as Load Case 2, except ambient and outside cask temperature = -20°F, inside cask temperature = +2°F. 1

4. 10 CFR 71.71(c)(3), Increased External Pressure
   - $P_{AMBIENT}$ = 20.0 psia
   - P_{CASK} (Internal) = 11.55 psia @ -20°F (Reference 11)

   $\therefore \Delta P = 20.0 - 11.55 = 8.45$ psi (Use 9.0 psi, applied as external pressure)

5. 10 CFR 71.71(c)(4), Decreased External Pressure
   - $P_{AMBIENT}$ = 3.5 psia
   - P_{CASK} (Internal) = 19.5 psia @ 100°F w/insolation (Ref. 11) 1

$\therefore \Delta P = 3.5 - 19.5 = -16.0$ psi (Use 16.5 psi, applied as internal pressure)

STARDYNE FINITE ELEMENT MODEL (F.E.M.)

LOAD CASES (continued)

6. 10 CFR 71.71(c)(5), Acceleration (Lateral - X1)

   - a_{x1} = 2.0g, a_{x2} = 1.0g (Down - deadweight)

7. 10 CFR 71.71(c)(5), Acceleration (Vertical - X2)

   - a_{x2} = 2.0g (Up)

8. 10 CFR 71.71(c)(5), Acceleration (Longitudinal - X3)

   - a_{x3} = 3.0g, a_{x2} = 1.0g (Down - deadweight)

LOAD CASE SUMMARY	
Load Case No.	Description
1	Maximum Heat - 100°F Ambient + Insolation
2	Maximum Cold - -40°F Ambient
3	Normal Cold - -20°F Ambient
4	Increased External Pressure - ΔP = 9.0 psi
5	Decreased External Pressure - ΔP = 16.5 psi
6	Lateral Acceleration During Transport, a_{x1} = 2.0g
7	Vertical Acceleration During Transport, a_{x2} = 2.0g (Up)
8	Longitudinal Acceleration During Transport, a_{x3} = 3.0g

STARDYNE FINITE ELEMENT MODEL (F.E.M.)

LOAD COMBINATIONS

1. Maximum Heat

= Load Case 1 + ΔP @ maximum cask temp.

ΔP = 19.5 psia (Reference 11) - 14.7 psia = 4.8 psi (Use 5.0 psi, applied as Internal Pressure) 1

2 Maximum Cold

= Load Case 2 + ΔP_{-40}

ΔP = 14.7 psia - 11.03 psia (Reference 11) = 3.67 psi (Use 4.0 psi, applied as External Pressure)

3. Cold + Increased External Pressure

= Load Case 3 + Load Case 4

4. Heat + Decreased External Pressure

= Load Case 1 + Load Case 5

5. Heat + X1 Acceleration

= Load Combination 1 + Load Case 6

6. Heat + X2 Acceleration

= Load Combination 1 + Load Case 7

7. Heat + X3 Acceleration

= Load Combination 1 + Load Case 8

8. Cold + X1 Acceleration

= Load Case 3 + 0.35(Load Case 4) + Load Case 6

9. Cold + X2 Acceleration

= Load Case 3 + 0.35(Load Case 4) + Load Case 7

10. Cold + X3 Accelerat on

= Load Case 3 + 0.35(Load Case 4) + Load Case 8

11. Normal Cold + ΔP (Note: This is not a load combination required per 10CFR71, but is used to combine results between the 2D and 3D models) 1

= Load Case 3 + 0.35(Load Case 4)

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
LOAD COMBINATIONS

SUMMARY OF LOAD COMBINATIONS		
Load Comb. No.	Load Case Combination	Description
1	1 + 0.303(5)	Maximum Heat + ΔP (ΔP = 5.0 psi)
2	2 + 0.444(4)	Maximum Cold + ΔP (ΔP = 4.0 psi)
3	3 + 4	Normal Cold + Increased External Pressure (ΔP = 9.0 psi)
4	1 + 5	Maximum Heat + Decreased External Pressure (ΔP = 16.5 psi)
5	1 + 0.303(5) + 6	Maximum Heat + ΔP + X1 Acceleration (ΔP = 5.0 psi)
6	1 + 0.303(5) + 7	Maximum Heat + ΔP + X2 Acceleration (ΔP = 5.0 psi)
7	1 + 0.303(5) + 8	Maximum Heat + ΔP + X3 Acceleration (ΔP = 5.0 psi)
8	3 + 0.35(4) + 6	Normal Cold + ΔP + X1 Acceleration (ΔP = 3.15 psi)
9	3 + 0.35(4) + 7	Normal Cold + ΔP + X2 Acceleration (ΔP = 3.15 psi)
10	3 + 0.35(4) + 8	Normal Cold + ΔP + X3 Acceleration (ΔP = 3.15 psi)
11	3 + 0.35(4)	Normal Cold + ΔP (= 3.15 psi), Used for combining 2D/3D model results

1

RESULTS

The maximum stress intensity from X3 acceleration in the 2-D model occurs at element no. 157. This element is located on the top cover, at the intersection of the cover and cylinder. Element 157 is therefore equivalent to the outer row of top cover elements (Elements 261-280) in the 3-D plate element model. To calculate the stress intensities for load combinations 7 and 10, the 2-D model X3 acceleration stress intensity at element 157 is combined with the temperature and pressure stress intensity from the 3-D model at element 274.

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
RESULTS (continued)
Load Combination 7

3D Model

S.I. (Heat + ΔP) = 3811 psi = 3.81 ksi (Load Comb. 1, Elem. 274)

2D Model

S.I. (X3 Accel.) = 25.95 ksi (Element 157)

Max. S.I. for Load Combination 7 = S.I.(Heat+ΔP - 3D Model) + S.I. (X3 Accel.- 2D Model)

= 3.81 ksi + 25.95 ksi

∴Maximum Stress Intensity for Load Combination 7 = 29.76 ksi

Load Combination 10

3D Model

S.I. (Cold + ΔP) = 4986 psi = 4.99 ksi (Load Comb. 11, Elem. 274)

2D Model

S.I. (X3 Accel.) = 25.95 ksi (Element 157)

Max. S.I. for Load Combination 10= S.I.(Cold+ΔP - 2D Model) + S.I. (X3 Accel.- 3D Model)

= 4.99 ksi + 25.95 ksi

∴Maximum Stress Intensity for Load Combination 10 = 30.94 ksi

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
RESULTS
Maximum Stress Intensities by Load Case

MAXIMUM STRESS INTENSITY BY LOAD CASE			
Load Case No.	Description	Element No. (1)	Maximum M+B Stress Ints.
1	Maximum Heat - 100°F Ambient + Insolation	261-280	4.58 ksi
2	Maximum Cold - -40°F Ambient	261-280	4.47 ksi
3	Normal Cold - -20°F Ambient	261-280	4.47 ksi
4	Increased External Pressure - ΔP = 9.0 psi	321-340	2.83 ksi
5	Decreased External Pressure - ΔP = 16.5 psi	321-340	5.20 ksi
6	Lateral Acceleration During Transport, a_{X1} = 2.0g	278	12.48ksi
7	Vertical Acceleration During Transport, a_{X2} = 2.0g (Up)	61, 70	1.76 ksi
8	Longitudinal Acceleration During Transport, a_{X3} = 3.0g	157(2)	25.95 ksi

NOTES:

(1) Element Numbers refer to 3-D model unless noted otherwise.

(2) Element Number refers to 2-D model.

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
RESULTS
Maximum Stress Intensities by Load Combination

MAXIMUM STRESS INTENSITIES BY LOAD COMBINATION				
Load Comb. No.	Description	Element No.(1)	Max. M+B S.I.(2)	Allowable S.I.(3)
1	Maximum Heat + ΔP	261-280	3.81	34.95
2	Maximum Cold (-40°F) + ΔP	261-280	5.14	34.95
3	Normal Cold (-20°F) + Increased External Pressure	261-280	6.07	34.95
4	Maximum Heat + Decreased External Pressure	321-340	6.42	34.95
5	Maximum Heat + ΔP + X1 Acceleration	278	15.76	34.95
6	Maximum Heat + ΔP + X2 Acceleration	261,270	4.63	34.95
7	Maximum Heat + ΔP + X3 Acceleration	274/157 (4)	29.76	34.95
8	Normal Cold + ΔP + X1 Acceleration	278	16.75	34.95
9	Normal Cold + ΔP + X2 Acceleration	261,270	6.02	34.95
10	Normal Cold + ΔP + X3 Acceleration	274/157 (4)	30.94	34.95

NOTES:

(1) Element Numbers refer to 3-D model unless noted otherwise.

(2) Maximum Membrane plus Bending Stress Intensity in ksi.

(3) Allowable Membrane + Bending stress instensity in ksi.

(4) Element Numbers refer to 3D model/2D model

STARDYNE FINITE ELEMENT MODEL (F.E.M.)
RESULTS
Maximum Stress Intensities by Load Combination

MAXIMUM STRESS INTENSITIES BY LOAD COMBINATION				
Load Comb. No.	Description	Element No.(1)	Max. M+B S.I.(2)	Allowable S.I.(3)
1	Maximum Heat + ΔP	261-280	3.81	34.95
2	Maximum Cold (-40°F) + ΔP	261-280	5.14	34.95
3	Normal Cold (-20°F) + Increased External Pressure	261-280	6.07	34.95
4	Maximum Heat + Decreased External Pressure	321-340	6.42	34.95
5	Maximum Heat + ΔP + X1 Acceleration	278	15.76	34.95
6	Maximum Heat + ΔP + X2 Acceleration	261,270	4.63	34.95
7	Maximum Heat + ΔP + X3 Acceleration	274/157 (4)	29.76	34.95
8	Normal Cold + ΔP + X1 Acceleration	278	16.75	34.95
9	Normal Cold + ΔP + X2 Acceleration	261,270	6.02	34.95
10	Normal Cold + ΔP + X3 Acceleration	274/157 (4)	30.94	34.95

NOTES:

(1) Element Numbers refer to 3-D model unless noted otherwise.

(2) Maximum Membrane plus Bending Stress Intensity in ksi.

(3) Allowable Membrane + Bending stress instensity in ksi.

(4) Element Numbers refer to 3D model/2D model

PENETRATION EVALUATION
MISSILE PARAMETERS

- B = Width of Plate Between Rigid Supports
- D = Missile Caliber Density = W/d^3 = 13 lbs/(1.25")^3
  = 6.656 lb/in
- d = Effective Projectile (Missile) Diameter = 1.25"
- e = Perforation Distance
- S_s = Ultimate Tensile Strength of Target Material = 70,000 lb/in^2
- v_0 = Initial Striking Velocity of Missile
  = $(2gh)^{½}$; Where g = 32.2 ft/sec^2, h = drop height = 40 in.
  = $[2(32.2)(40/12)]^{½}$ = 14.65 ft/sec
- W = Projectile (Missile) Weight = 13 lbs

Definitions per ASCE Man. No. 58, 6.4.1.1.2, Symbols (Reference 9). Missile weight, dimensions, and drop height from 10 CFR 71.71(c)(10).

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$$\left(\frac{e}{d}\right)^2 + \frac{3F}{128}\left(\frac{e}{d}\right) = \frac{0.045Dv_0^2}{S_s} \quad (Ref.\ 9,\ Eq.\ 6.48);$$

Where $F = (B/d)$, *except* $F \le 8 \le 100(e/d)$

$$\left(\frac{e}{1.25}\right)^2 + \frac{3(8)}{128}\left(\frac{e}{1.25}\right) = \frac{0.0452(6.656)(14.65)^2}{70{,}000};$$

$$0.64e^2 + 0.15e = 9.224 \times 10^{-4}$$

$e = +0.006'/-0.240''$; $\therefore$ $e = 0.006''$

NOTE: SRI equation is valid for 100(e/d) ≥ 8. Since 100(e/d) = 0.48 << 8, the results predicted by the SRI equation may not be valid.

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$$\left(\frac{e}{d}\right)^{\frac{3}{2}} = \frac{Dv_0^2}{1.12 \times 10^6} \quad (Ref.\ 9,\ Eq.\ 6.46);$$

$$\left(\frac{e}{1.25}\right)^{\frac{3}{2}} = \frac{(6.656)(14.65)^2}{1.12 \times 10^6}; \quad \frac{(e)^{\frac{3}{2}}}{1.3975} = 0.0013$$

$e = (1.783 \times 10^{-3})^{\frac{2}{3}} = 0.015$ *inches*

Minimum Cask Thickness = 3" >> 0.015" >> 0.006"

$\therefore$ Cask is adequate for penetration

ASME VIII, DIVISION 2, PART AD QUALIFICATION

Only those paragraphs of Part AD and Appendix 4 which are applicable to the design of the shipping cask are discussed below. It should be noted that many of the ASME requirements are duplicated by 10 CFR 71, R.G. 7.6, or R.G. 7.8 requirements. The NRC requirements are more conservative than the ASME requirements, and are used for design/qualification of the cask.

Paragraph, Part AD

- AD-104, Minimum Thickness of Shell or Head:
  $t \geq ¼"$, $t_{min} = 3" >> ¼"$ (OK)

- AD-110, Loadings: Specified by 10 CFR 71.71, except
  (b) Weight of Vessel: Evaluated as part of X1 and X3 accelerations
  (e) Reactions of Supporting Lugs, Saddles, etc.: Evaluated in design of tiedown system (Reference 12)

- AD-120, Design Basis [Pressure and Temperature Relationships]: Specified in 10 CFR 71 and R.G. 7.8. The relationships in Table AD-120.1 are not applicable to the cask.

- AD-130, Design Stress Intensity Values: Requires use of S_m values from Part AM. Part AM is used to determine S_m (see Page 28).

- AD-131, Coef. of Thermal Expansion and Modulus of Elasticity: Requires use of values in Tables AMG-1 for thermal expansion, and Tables AMG-2 for E. These tables are used, along with Section III, Division 1 Appendices (Reference 2) for determining α and E. Section III, Division 1 Appendices are used for values at temperatures not provided in Tables AMG-1 and 2 (see Page 27).

- AD-140, Design Criteria [Stress Limits]: Stress limits are specified in R.G. 7.6 as $1.0S_m$ for primary membrane stresses and $1.5S_m$ for primary membrane + bending stresses. This is consistent with AD-140 for "normal" load combinations.

- AD-150, Load Combinations: Load combinations are given in R.G. 7.8.

- AD-160, Fatigue Evaluation

  AD-160.2: $n_{(a)} + n_{(b)} + n_{(c)} + n_{(d)} \leq 1000$

ASME VIII, DIVISION 2, PART AD QUALIFICATION

- AD-160, Fatigue Evaluation (continued)

The 10CFR71 thermal and pressure conditions are based on potential extremes of weather. The shipping cask is not expected to be subject to near the full range of either the design pressures or the design temperatures. For the purposes of this evaluation, it is assumed that the loaded cask could be in storage or transportation for a period of up to 1 year (52 weeks). Conservatively, it is assumed that the cask could be subject to the following temperature/pressure cycles at a rate of one cycle per week for the storage/transportation period.

(a) $n_{(a)}$ = Expected no. of full range pressure cycles
= Assume full range of pressure cycles occur once/week for the duration of the storage/ transportation period (1 year = 52 weeks)
= 52 cycles

(b) $n_{(b)}$ = Expected no. of operating pressure cycles where range of pressure variation is > 20% of design pressure range.
= Assume full range of pressure cycles occur once/week for the duration of the storage/ transportation period (1 year = 52 weeks)
= 52 cycles

(c) $n_{(c)}$ = Effective no. of changes in metal temperature between 2 adjacent points. Assume worst temperature differential, ΔT, = 70°F → -40°F (Page 10). Therefore, ΔT = 110°F, and Factor (AD-160.2(c)) = 2.
= Assume full range of pressure cycles occur once/week for the duration of the storage/ transportation period (1 year = 52 weeks)
= 52 cycles

(d) $n_{(d)}$ = The number of temperature cycles for components involving welds between materials with different coefficients of expansion.
= 0

$n_{(a)} + n_{(b)} + n_{(c)} + n_{(d)} = 52 + 52 + 2(52) + 0 = 208 << 1000$,

Therefore, fatigue evaluation is <u>not</u> required.

- AD-201, Cylindrical Shells

(a) $t_{min} = \frac{PR}{S-0.5P} = \frac{16.5(76)}{23300-0.5(16.5)} = 0.054'' < 3'$

ASME VIII, DIVISION 2, PART AD QUALIFICATION

- AD-201, Cylindrical Shells (continued)

(b) $t_{min} = (0.5PR + F)/(S - 0.5P)$,

F = max. membrane stress in cask cylinder
= 7500 psi (Load Comb. 10, Elem. 13,18)

$$t_{min} = \frac{0.5(16.5)(76) + (7500)(3)}{23,300-(0.5(16.5))} = 0.99'' < 3'' \text{ (OK)}$$

- AD-310, Cylindrical Shells and Tubes (Subject to External Pressure)

$D_o/t = 158''/3'' = 52.7 \geq 10$; $L/D_o = 28.42'/13.17' = 2.16$;

A = 0.00161 (Table 2-AGO-28.0)

A516, Grade 70, F_y = 38 ksi, Factor B = 16,500
(Fig. 2-ACS-28.3)

$P_a = 4B/3(D_o/t) = 4(16,500)/3(52.7) = 417.5$ psi

$P_a >> P_{EXTERNAL}(Max) = 8.45$ psi (Page 29) (OK)

- AD-702, Flat Heads

$$T_{min} = d\sqrt{\frac{CP}{S}};\ d = 152'',\ C = 0.44m \geq 0.27\ (AD\text{-}703(c)),$$
$m = t_r/t_s$
$P = 16.5$ psig (Pg. 26), $S = 23.3$ ksi

$$T_{min} = 152\sqrt{\frac{0.27(16.5)}{23,300}} = 2.10'';\ T = 4.0'' > 2.10''\ \text{(OK)}$$

Appendix 4

- 4-100, General Requirements: (b) Analytical methods provided in Articles 4-2 - 4-9 may be used for vessel design as well as other methods which are more accurate or conservative than those provided in the articles. The vessel is designed using a detailed finite element model, using the STARDYNE computer code. Finite element evaluations are recognized as an accurate means to determine stresses, deflections, and other structural information. The model used herein is sufficiently detailed to provide a very accurate depiction of the shipping cask when subject to design loads. STARDYNE is a commercially available, recognized computer code for evaluating finite element models. Therefore, the analytical techniques used in this calculation are acceptable per Article 4-100(b).

ASME VIII, DIVISION 2, PART AD QUALIFICATION (continued)

- 4-110, Design Acceptability

  (a) Stresses shall not exceed the limits in 4-130. The design will meet the stress limits.
  (b) N/A
  (c) N/A

- 4-111, Basis for Determining Stresses

  The maximum shear stress theory is used as the basis for Appendix 4. The same theory is used in this calculation for evaluation of the cask.

- 4-120, Derivation of Stress Intensities

  Stress intensities are derived in accordance with 4-120.

- 4-130, Basic Stress Intensity Limits

  4-131, General Primary Membrane Stress Intensity (P_m)

  $P_m \le kS_m$, where k is given in Table AD-150.1. This limit is consistent with the requirements of R.G. 7.6 used herein. k is always taken as 1.0.

  4-132, Local Primary Membrane Stress Intensity (P_L)

  $P_L \le 1.5kS_m$, where k is given in Table AD-150.1. In this calculation, P_L is limited to $1.0kS_m$, with k always taken as 1.0. This calculation is therefore conservative.

  4-133, Primary Membrane (General or Local) Plus Primary Bending Stress Intensity (P_m or $P_L + P_B$)

  P_m or $P_L + P_B \le 1.5kS_m$, where k is given in Table AD-150.1. This limit is consistent with the requirements of R.G. 7.6 used herein. k is always taken as 1.0.

  4-134, Primary Plus Secondary Stress Intensity ($P_L + P_B + Q$)

  $P_L + P_B + Q \le 3.0S_m$. In this calculation, this combination is limited to $1.5S_m$. This calculation is therefore conservative.

  4-135, Peak Stress Intensity ($P_L + P_B + Q + F$)

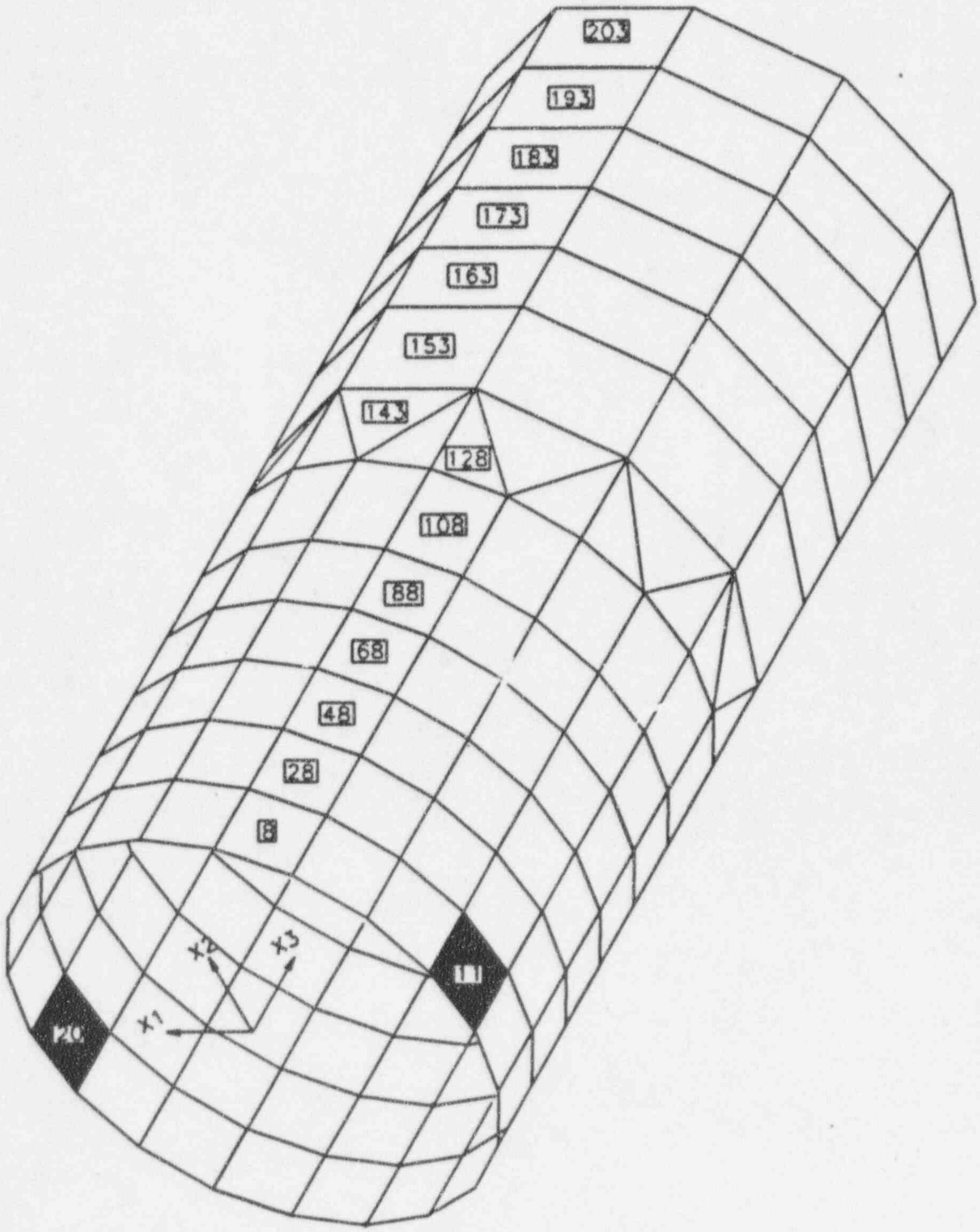
  This is not applicable to this calculation.

ATTACHMENT A

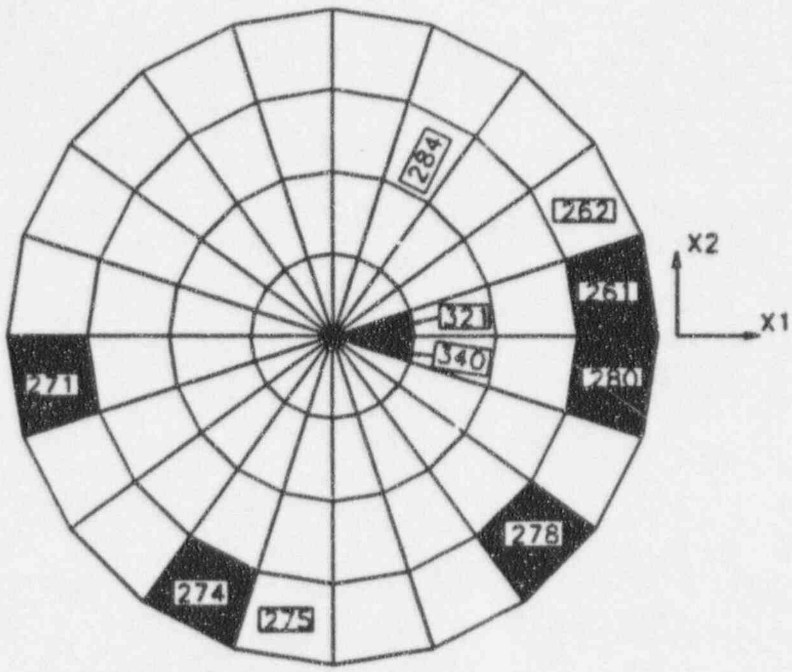
STARDYNE FINITE ELEMENT MODEL

Sketches and Plots

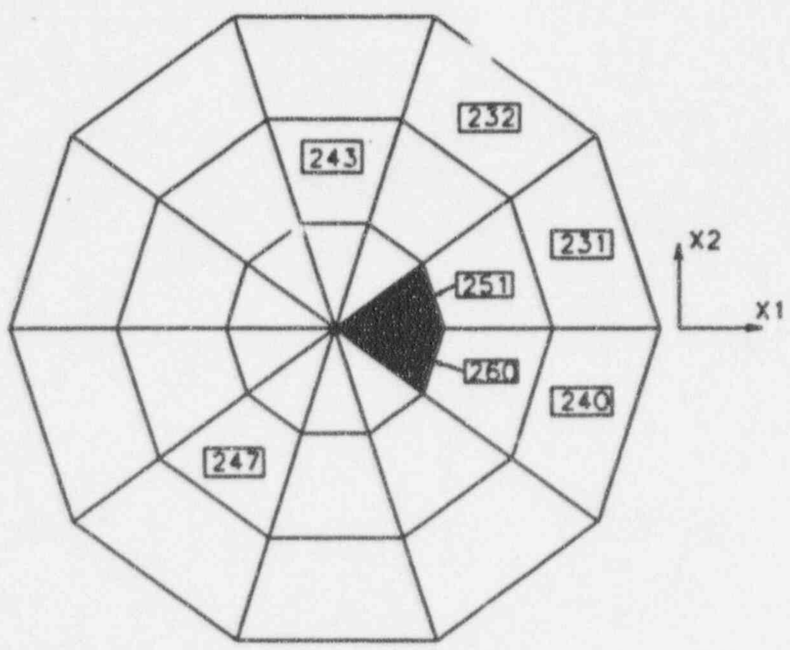
STARDYNE Finite Element Model



STARDYNE Finite Element Model



CASK TOP COVER
(X3 = 0.0)



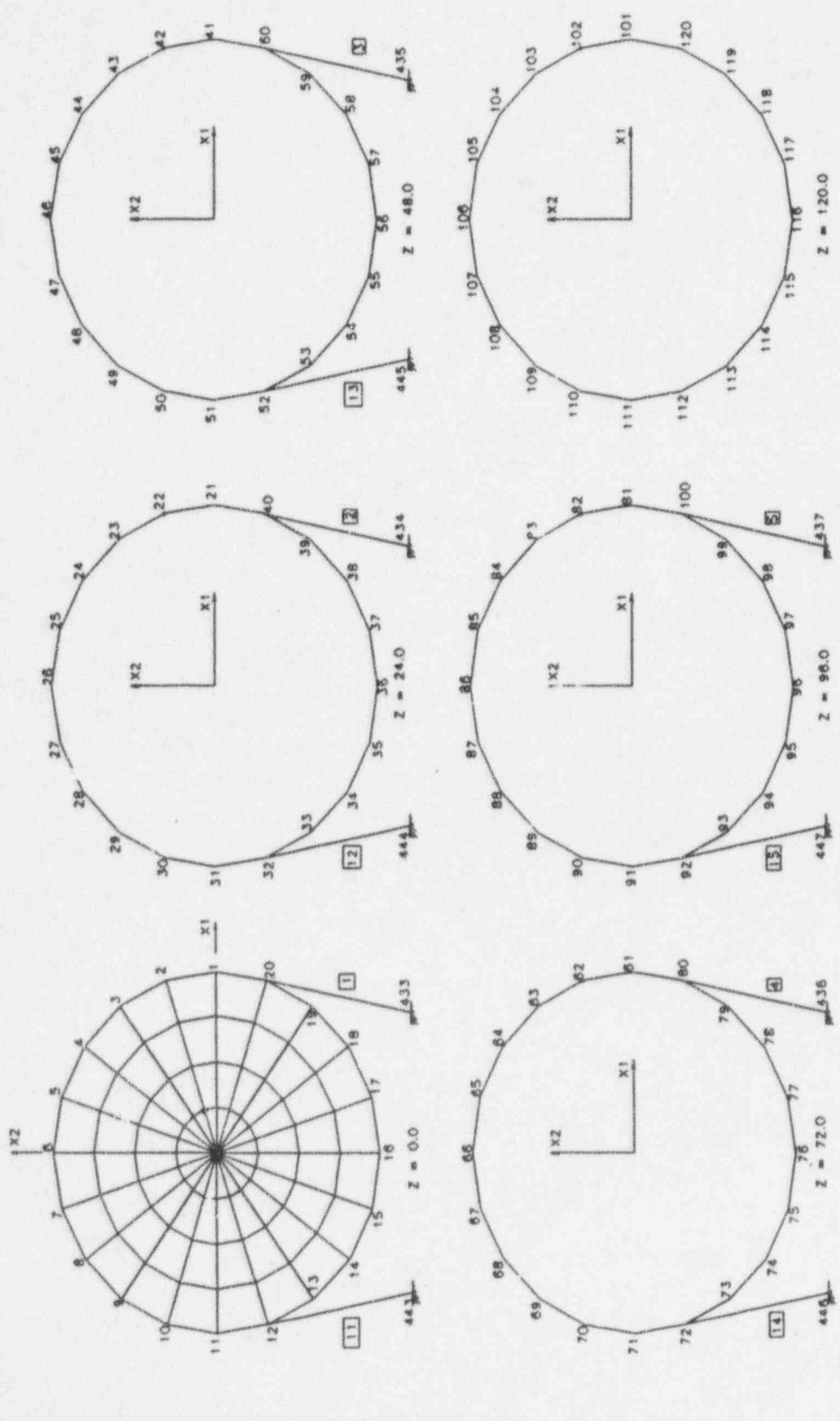
CASK BOTTOM COVER
(X3 = 335.0)

YANKEE ATOMIC ELECTRIC COMPANY CALCULATION NO. YRC-1062 PAGE A4 OF ____

SUBJECT REACTOR PRESSURE VESSEL SHIPPING CASK

PREP BY DRL DATE 3/20/95 REVIEW BY JEP DATE 3/20/95 WORK ORDER NO DD28

STARDYNE FINITE ELEMENT MODEL - NODE NUMBERS

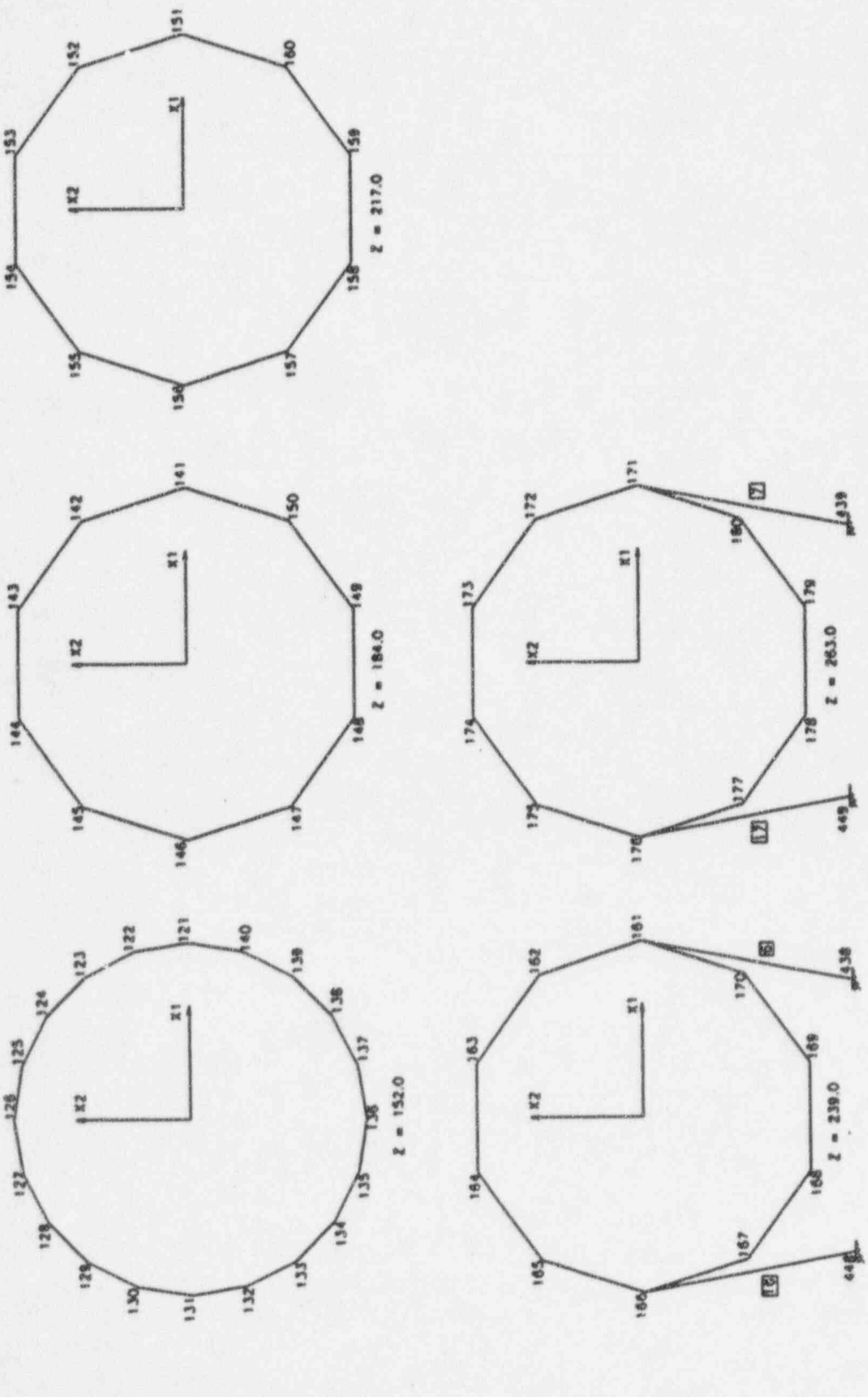


YANKEE ATOMIC ELECTRIC COMPANY CALCULATION NO. YRC-1062 PAGE A5 OF ____

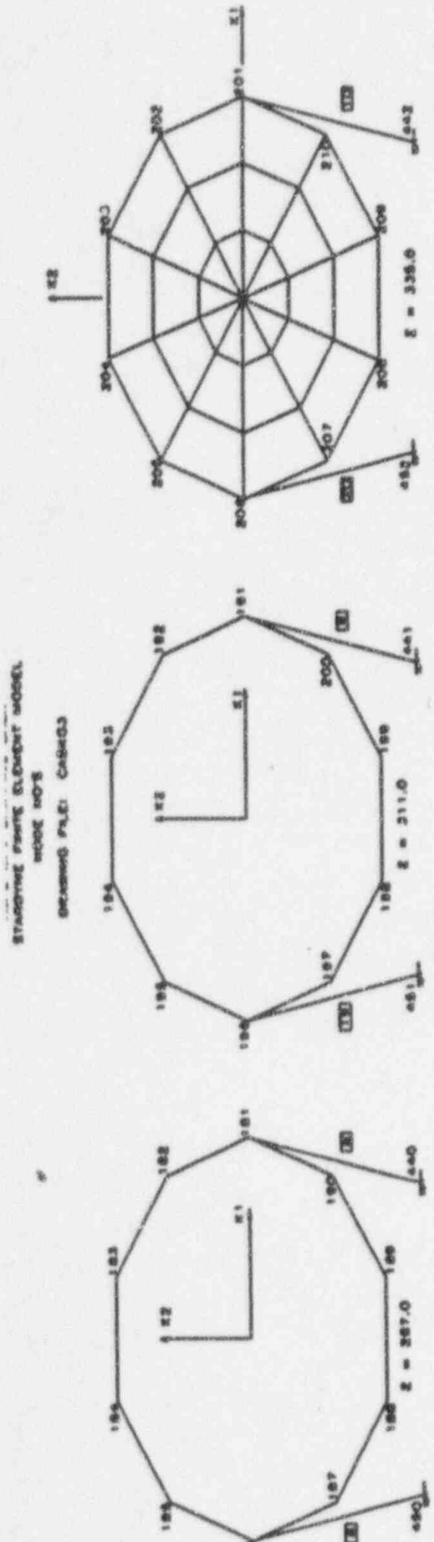
SUBJECT REACTOR PRESSURE VESSEL SHIPPING CASK

PREP BY DRL DATE 3/20/95 REVIEW BY JEP DATE 3/20/95 WORK ORDER NO DD28

STARDYNE FINITE ELEMENT MODEL - NODE NUMBERS



STARDYNE FINITE ELEMENT MODEL - NODE NUMBERS

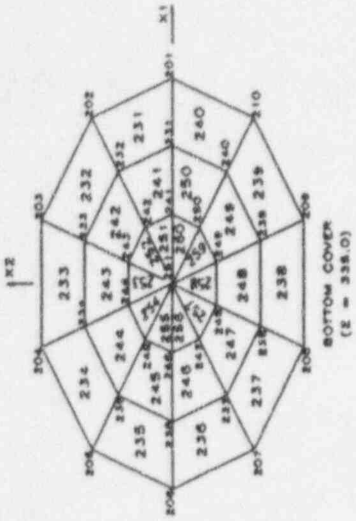


YANKEE ATOMIC ELECTRIC COMPANY CALCULATION NO. YRC-1062 PAGE A7 OF ____

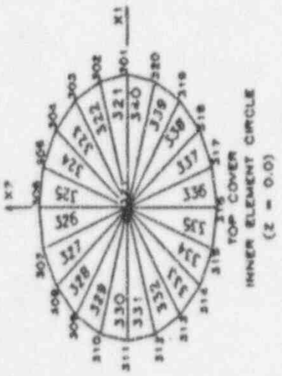
SUBJECT REACTOR PRESSURE VESSEL SHIPPING CASK

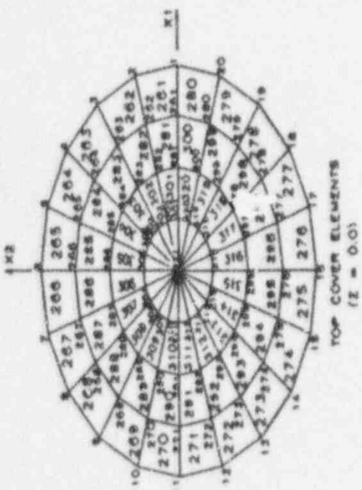
PREP BY DRL DATE 3/20/95 REVIEW BY JEP DATE 3/20/95 WORK ORDER NO DD28

STARDYNE FINITE ELEMENT MODEL - TOP & BOTTOM COVERS, NODE & ELEM. NO'S

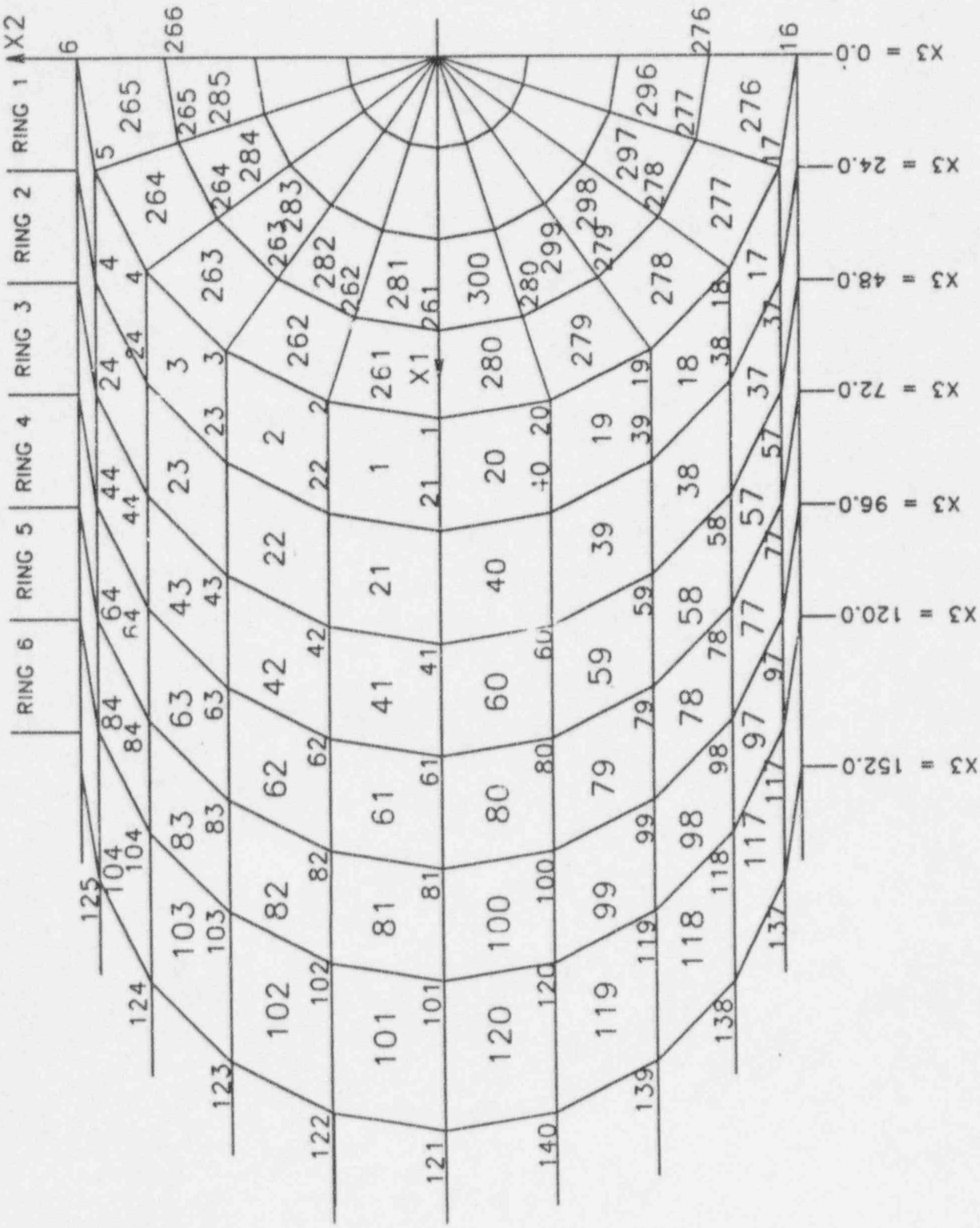


YNPS RPV SHIPPING CASK
STARDYNE FINITE ELEMENT MODEL
COVER ELEMENT AND NODE NO'S
DRAWING FILE: CASK048

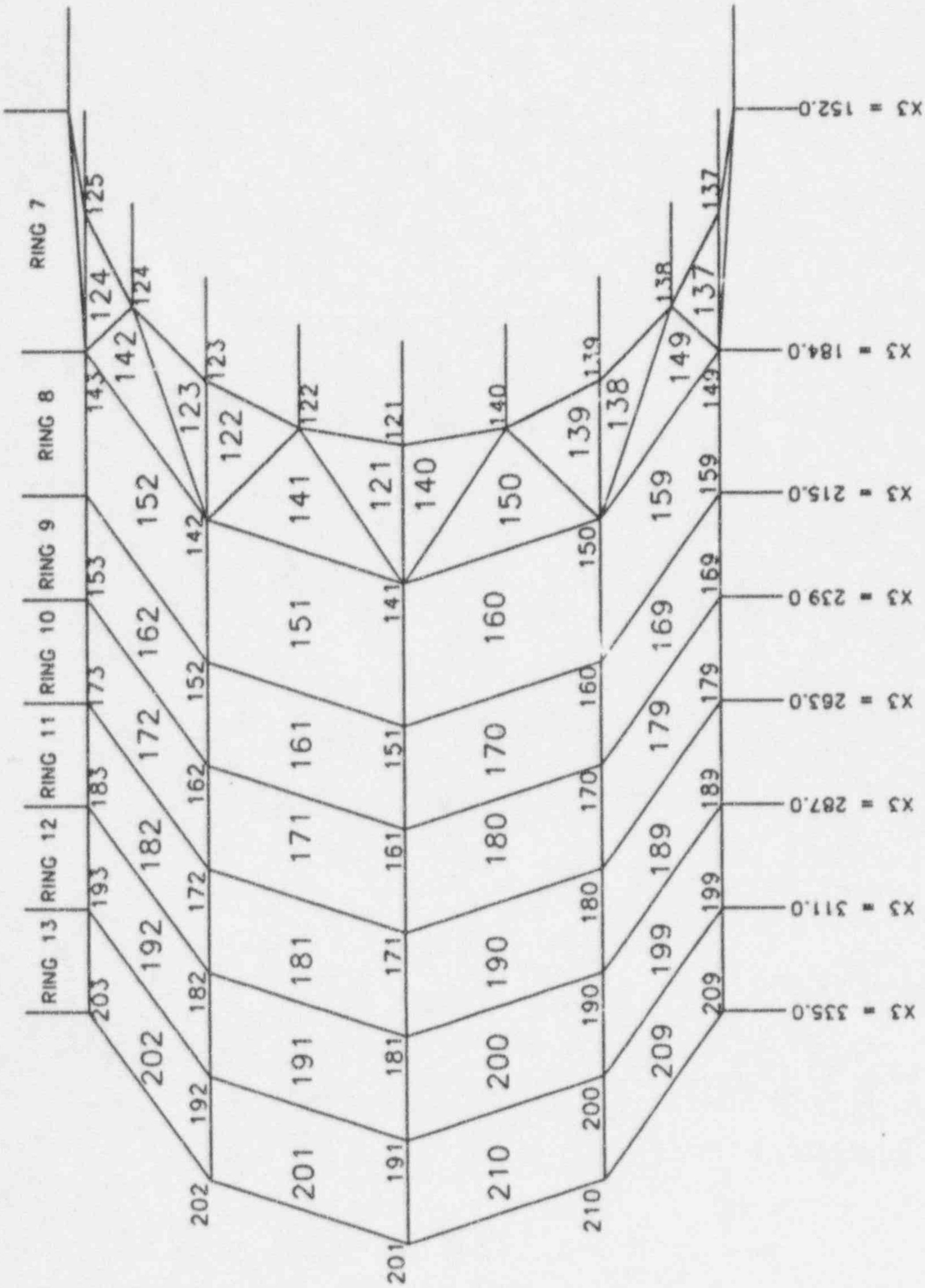




STARDYNE FEM - CASK ELEMENT NO'S, X3 = 0.0 TO 152.0



STARDYNE FEM - CASK ELEMENT NO'S, X3 = 152.0 TO 335.0

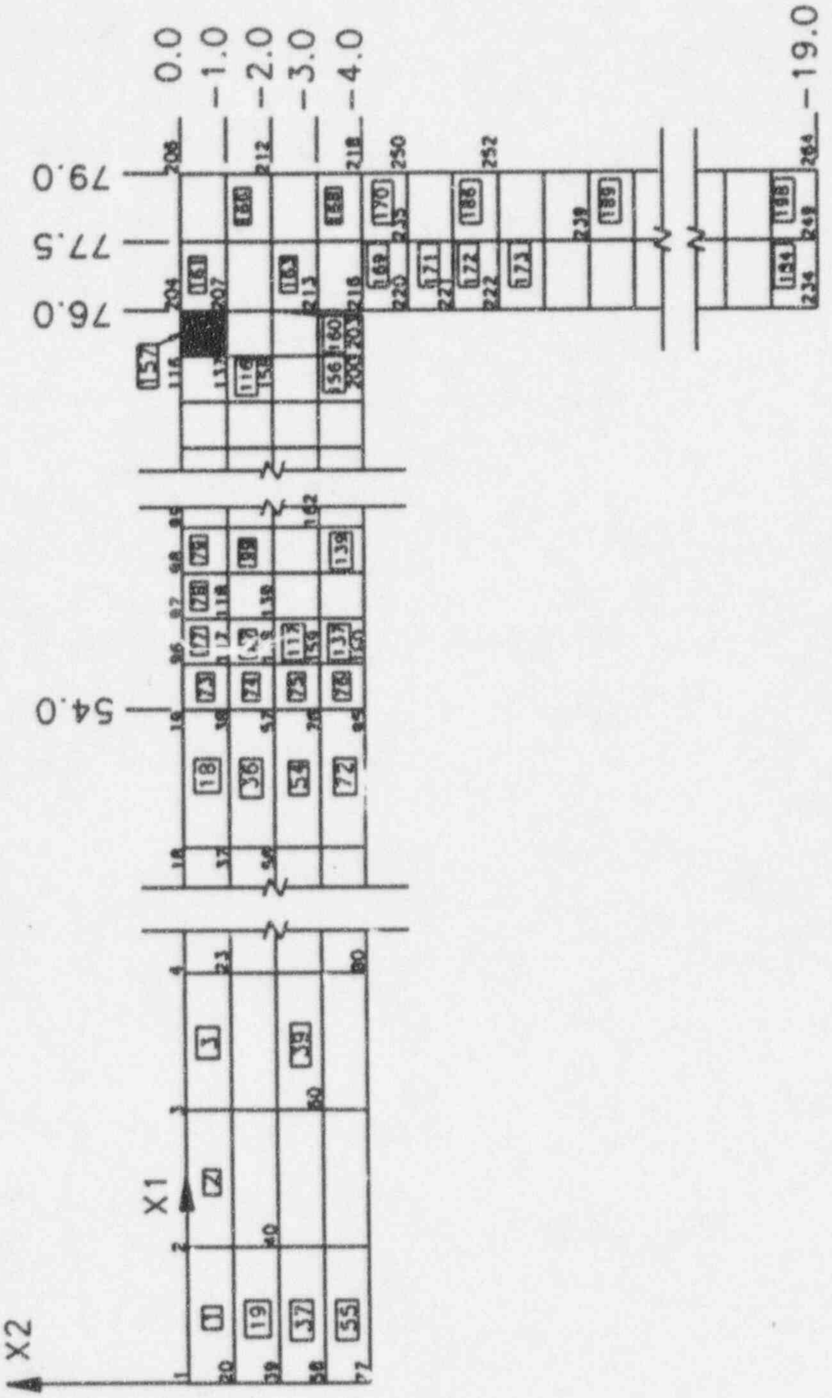


YANKEE ATOMIC ELECTRIC COMPANY CALCULATION NO. YRC-1062 PAGE A10 OF ____

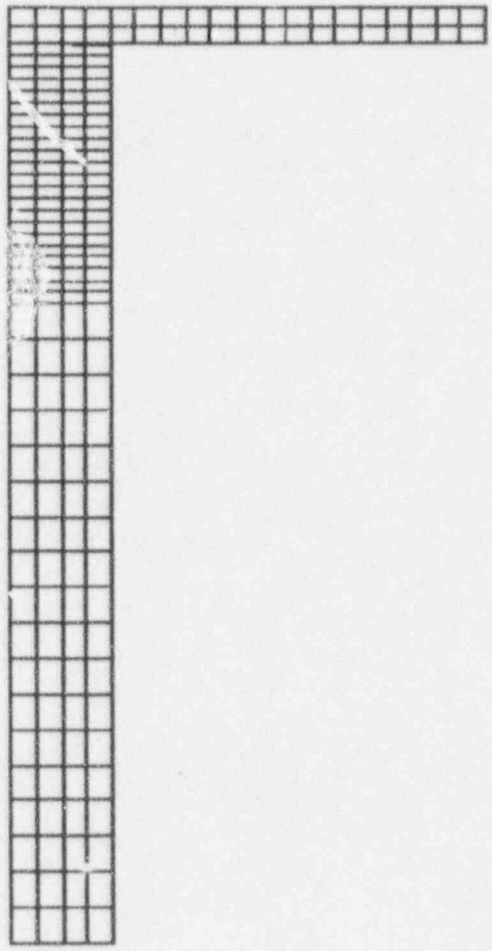
SUBJECT REACTOR PRESSURE VESSEL SHIPPING CASK

P BY DRL DATE 3/20/95 REVIEW BY JEP DATE 3/21/95 WORK ORDER NO DD28

STARDYNE TOP COVER 2-D FEM
Node and Element No's



STARDYNE TOP COVER 2-D FEM



3.0 THERMAL EVALUATION

Chapter 3 provides the results of an evaluation of the limiting temperatures and pressures for the YNPS Reactor Pressure Vessel (RPV) package under the normal conditions of transport specified by 10 CFR 71.71. The results demonstrate that the RPV package complies with the applicable thermal performance requirements of 10 CFR 71.

3.1 Discussion

10 CFR 71 delineates the thermal performance requirements for the RPV package. Per 10 CFR 71.43(f), under the normal conditions of transport defined by 10 CFR 71.71, there must be "no loss or dispersal of radioactive contents, no significant increase in external radiation levels, and no substantial reduction in the effectiveness of the packaging." In addition, 10 CFR 71.43(g) limits the maximum accessible surface temperature of the RPV package to 180°F, since it will be transported as an exclusive use shipment.

This chapter provides the results of the evaluation of the RPV package temperature and pressure for the heat, cold, reduced external pressure, and increased external pressure conditions specified by 10 CFR 71.71(c)(1) through (c)(4). The RPV package temperatures were determined based on steady state conduction of the heat generated within the package to the shipping cask surface, where the heat is transferred to the surrounding air by free convection and radiation. The maximum temperature and pressure evaluations are based on 22 watts of heat from radioactive decay of the activation products in the RPV wall. An additional 18 watts of heat is generated by the radioactive material (or dross) located in the solidified waste package at the core centerline. This is a conservative estimate of the internal heat generation. The minimum temperature and pressure were evaluated without any credit for internal heat generation. Cases with and without solar heating of the package surface were considered.

Only the normal conditions of transport specified in 10 CFR 71.71 were considered in the thermal evaluation. The hypothetical accident conditions specified in 10 CFR 71.73 were not included. Since the Yankee RPV package contains only LSA concentrations of radioactive materials, as indicated in Section 1.2.3, and will be transported as an exclusive use shipment, it is exempt from the requirements of 10 CFR 71.73 (Hypothetical Accident Conditions), as stated in 10 CFR 71.52.

The results of the thermal evaluation are summarized in Table 3.1.1. As shown in Table 3.1.1, the maximum outside surface temperature of the RPV package without solar heating is 102.5°F, well below the 180°F limit. The remaining values in Table 3.1.1 serve as input to the structural evaluation presented in Section 2.6.

3.2 Summary of Thermal Properties of Materials

The RPV package temperatures were determined based on steady state conduction from the interior of the RPV package to the shipping cask surface. The heat generated within the RPV package by radioactive decay was assumed to be conducted radially outward through the concrete inside the solidified waste package located at the core centerline, the solidification liner, the low density concrete between the liner and the inside wall of the RPV, the RPV wall, the RPV insulation, the concrete between the insulation and the shipping cask, the internal shield of the shipping cask, and the shipping cask wall (see Figure 3.4-1). The 1/80" thick, stainless steel insulation wrapper is conservatively replaced by an equivalent thickness of concrete. The heat was then assumed to be transferred from the shipping cask surface to the surrounding air by free convection and radiation. Thermal contact resistance between the different materials in the conduction path was considered.

The material properties needed to solve for the RPV package temperatures under these conditions are the thermal conductivities of the materials in the conduction path and the emissivity of the shipping cask surface. Table 3.2.1 lists the values used in the thermal evaluation for these material properties. Table 3.2.1 also gives the references from which these values were obtained.

3.3 Technical Specifications of Components

The Reactor Pressure Vessel design temperature is 650°F at a pressure of 2500 psig (Reference 3.6.1).

The shipping cask and all associated welds are designed for a temperature range of -40°F to 194°F. The shipping cask is designed for the differential pressures at both hot and cold conditions. Under hot conditions, the cask is designed for a differential pressure of 16.5 psi, with the internal pressure greater than the external pressure. Under normal cold conditions, the cask is designed for a differential pressure of 9.0 psi, with the external pressure greater than the internal pressure.

The evaluations of the RPV package for the limiting temperature and pressure conditions during normal transport are discussed in Section 2.6. As demonstrated therein, the RPV package maintains structural integrity when subject to those limiting conditions.

3.4 Thermal Evaluation for Normal Conditions of Transport

The following sections provide the results of the thermal performance evaluation for the RPV package under the normal conditions of transport. The evaluation included the maximum temperatures (with and without solar heating), the minimum temperature, the maximum internal pressure, the minimum internal pressure, and the maximum differential pressure for the heat, cold, reduced external pressure, and increased external pressure

conditions specified by 10 CFR 71.71(c)(1) through (c)(4). The determination of the heat generated within the RPV package by radioactive decay is also discussed.

3.4.1 Thermal Model

The internal and surface temperatures of the Yankee RPV package were determined assuming one-dimensional conduction from the internal heat sources to the shipping cask surface. The heat was assumed to be transferred to the surrounding air by either free convection (for the case without solar heating), or by a combination of free convection and radiation (for the case including solar heating).

There are two heat sources in the RPV package. The first source of heat is from the decay of the radioactive material (or dross) that is solidified in a liner located at the core centerline. The dross was conservatively treated as a distributed heat source contained in a volume that is half the height and half the diameter of the liner containing the dross. The solidified dross will be uniformly distributed within the concrete contained in the liner and therefore occupies a significantly larger volume within the liner than that assumed. In the analysis, concrete is assumed to occupy the remainder of the liner volume. The second heat source is from radioactive decay of the activation products in the RPV walls. This heat source, which is distributed inside the RPV walls, was assumed to be concentrated at the inside surface of the RPV wall, over the active fuel height (core region), to maximize the RPV temperature.

The RPV package temperatures were determined for the cross section of the RPV package which corresponds to the location of the maximum heat generation from the activation products in the RPV walls (see Figure 3.4-1). This cross section results in the maximum package temperatures. The heat from the dross was assumed to be conducted radially outward through the concrete containing the dross, through the liner, and through the low density concrete between the liner and the inside of the RPV wall. The liner was conservatively neglected, with the thickness of the liner being replaced with concrete. From the inside surface of the RPV wall, the heat from both the dross and the activation products in the RPV walls was assumed to be conducted radially outward through the RPV walls, through the RPV insulation, through the concrete surrounding the outside of the RPV, through the package internal shield, and through the shipping cask wall. The heat was then assumed to be transferred to the surrounding air by either free convection or a combination of free convection and radiation. No credit was taken for axial conduction. Only the surface area over the core region was credited for the transfer of heat out of the package. Thermal contact resistance was conservatively treated as a 1/16 inch air gap between each material in the conduction path.

3.4.2 Maximum Temperature

The maximum temperature at the inside and outside surface of each material in the conduction path was evaluated. The maximum temperatures are based on conservative values

of the decay heat from the activation products in the RPV walls and the solidified dross at the core centerline. The maximum temperatures were determined assuming that the RPV package is surrounded by still air at 100°F, as specified by 10 CFR 71.71(c)(1).

Two cases were evaluated, with and without solar heating of the shipping cask surface. The case without solar heating was used to determine the maximum accessible surface temperature, as specified by 10 CFR 71.43(g). A solar heat input of 400 gcal/cm^2 to the side of the shipping cask, per 10 CFR 71.71(c)(1), also was evaluated. The case with solar heating was used to determine the maximum RPV package temperature and the temperature profile on which the stress calculations in Section 2.6 are based.

3.4.2.1 <u>Internal Heat Generation</u>

The only significant sources of heat in the RPV package are due to radioactive decay of the activation products in the RPV walls and the dross contained in the internal solidified waste package. The heat from each of these two sources was determined assuming that all of the energy from radioactive decay is deposited as heat within the RPV package.

The heat from each source was determined based on the type, the amount, and the energy per decay of the significant radionuclides. The total activity of each radionuclide in both the RPV wall activation products and the dross was taken from the package classifications (Appendices 1.4.2 and 1.4.3). Mn^{54}, Fe^{55}, Co^{60}, and Ni^{63} were the only radionuclides considered in determining the heat. These four nuclides account for more than 99.75 percent of the total activity present for both the activation products and the dross. The heat from other radionuclides is insignificant in comparison.

The heat generated by each of the four nuclides was determined by applying the maximum energy per decay for each nuclide to the nuclide's total activity. The total heat generated is the sum of the heat from each of the four nuclides. For the dross, the heat contributed by each of the four radionuclides was determined by:

$$\textit{Dross Heat}\,(\textit{watts}) = (\textit{Activity})\,(\textit{Decay Energy})$$

where, Activity = Total activity of each radionuclide (decays/sec)
Decay Energy = Energy per decay for each radionuclide (watts)

In determining the heat generated by the activation products, the distribution of the activation products within the RPV wall was included to find the maximum heat generation rate at any location in the wall. For the activation products, the heat contributed by each of the four radionuclides was determined by:

$$\textit{Activation Product Heat}\,(\textit{watts}) = (\textit{Activity})\,(F_Z)\,(\textit{Decay Energy})$$

where, Activity = Average activity of each radionuclide over the core region (decays/sec)

F_Z = Ratio of maximum to average activity over the core region

Decay Energy = Energy per decay for each radionuclide (watts)

The maximum heat generation rate from the activation products in the RPV walls, as of July 1, 1995, is 11 watts. The heat generation rate from the solidified waste package containing the dross, as of July 1, 1995, is 9 watts. To ensure conservatism in determining the maximum RPV package temperatures, the heat generation rates were doubled to 22 watts for the activation products and 18 watts for the waste package.

3.4.2.2 Maximum Package Temperature Without Insolence

The temperature at the inside and outside surface of each material in the conduction path (see Figure 3.4.1) was evaluated assuming steady state conditions. A 22-watt heat source was assumed at the inside wall of the RPV (RPV wall activation products), with an additional 18-watt heat source (dross) evenly distributed within half the height and half the diameter of the liner at the core centerline. The evaluation also assumed a 100°F ambient air temperature, as specified by 10 CFR 71.71(c)(1). This case determined the maximum accessible surface temperature of the RPV package and, per 10 CFR 71.43(g), did not include solar heating of the shipping cask surface.

Heat transfer from the shipping cask surface to the still air surrounding the cask was assumed to occur due only to free convection. Radiation from the shipping cask surface was conservatively neglected. At steady state, crediting only the surface area over the core region, the shipping cask surface temperature is determined by:

$$Q_{DROSS} + Q_{ACTIVATION} = h A (T_{SURFACE} - T_{AIR})$$

where, Q_{DROSS} = Dross heat generation rate (BTU/hr)

$Q_{ACTIVATION}$ = Activation product heat generation rate (BTU/hr)

h = Free convection heat transfer coefficient (BTU/hr-ft^2-°F)

A = Shipping cask surface area over the core region (ft^2)

$T_{SURFACE}$ = Shipping cask surface temperature (°F)

T_{AIR} = Ambient temperature (°F)

The free convection heat transfer coefficient was determined assuming laminar flow conditions to maximize the surface temperature. From Reference 3.6.5, the free convection heat transfer coefficient for laminar flow over a horizontal cylindrical surface is (in BTU/hr-ft^2-°F):

$$h = 0.27 \left(\frac{T_{SURFACE} - T_{AIR}}{Diameter} \right)^{1/4}$$

Thermal contact resistance between the different materials in the conduction path was conservatively treated as a 1/16 inch air gap between each of the materials. The temperature at the inside and outside surface of each material in the conduction path was determined from the shipping cask surface temperature and the temperature difference across each material. Except for the concrete containing the dross, the temperature difference across each material (including the assumed air gaps) was determined using the cylindrical steady state conduction equation:

$$Q = \frac{2 \pi k L (T_{INSIDE} - T_{OUTSIDE})}{\ln\left(\frac{R_{OUTSIDE}}{R_{INSIDE}}\right)}$$

where, Q = Dross and/or activation product heat (BTU/hr)
k = Thermal conductivity (BTU/hr-ft-°F)
L = Core region height (ft)
T_{INSIDE} = Temperature at the inside surface of the material (°F)
$T_{OUTSIDE}$ = Temperature at the outside surface of the material (°F)
R_{INSIDE} = Inside radius (ft)
$R_{OUTSIDE}$ = Outside radius (ft)

Only the heat from the dross was included for conduction from the RPV package centerline to the inside wall of the RPV. The total heat from both the dross and the activation products in the RPV walls was included for conduction from the inside of the RPV wall to the surface of the shipping cask.

The temperature difference across the concrete inside the solidified waste package which contains the dross was determined using the equation shown below for steady state conduction across a cylinder containing an evenly distributed heat source. The concrete containing the dross was conservatively assumed to be half the height and half the diameter of the solidification liner containing the dross-concrete mixture.

$$T_{CENTER} - T_{OUTSIDE} = \frac{Q R^2}{4 k V}$$

where, T_{CENTER} = Temperature at the centerline of the cylinder (°F)
$T_{OUTSIDE}$ = Temperature at the outside surface of the cylinder (°F)
Q = Dross heat (BTU/hr)
R = Radius of the concrete containing the dross (ft)
k = Thermal conductivity (BTU/hr-ft-°F)
V = Volume of the concrete containing the dross (ft^3)

The temperature profile through the side of the RPV package is shown in Table 3.4.1. The maximum surface temperature of the RPV package, 102.5°F, is well below the 180°F limit specified in 10 CFR 71.43(g) for exclusive use shipments.

3.4.2.3 Maximum Package Temperature Including Insolence

The temperatures at the inside and outside surfaces of each material in the conduction path (see Figure 3.4.1) were evaluated assuming steady state conditions with two internal heat sources, similar to Section 3.4.2.2, except for the inclusion of solar heating at the shipping cask surface. A solar heat input of 400 gcal/cm^2 over a 12 hour period, as specified by 10 CFR 71.71(c)(1), was assumed at the curved surface of the shipping cask.

Heat transfer from the shipping cask surface to the still air surrounding the cask was assumed to occur via both radiation and free convection. The temperature of the shipping cask surface was determined from an energy balance, i.e., the heat added at the surface due to internal heat generation and solar heating is equal to the heat removed from the surface by free convection and radiation:

$$Q_{SOLAR} + Q_{DROSS} + Q_{ACTIVATION} = Q_{CONVECTION} + Q_{RADIATION}$$

where, Q_{SOLAR} = Solar heat input (BTU/hr)
Q_{DROSS} = Dross heat generation rate (BTU/hr)
$Q_{ACTIVATION}$ = Activation product heat generation rate (BTU/hr)
$Q_{CONVECTION}$ = $h\ A\ (T_{SURFACE} - T_{AIR})$, as discussed in Section 3.4.2.2

and the radiation term in the heat balance is given by:

$$Q_{RADIATION} = \sigma\, \epsilon\, A \left(T^4_{SURFACE} - T^4_{AIR} \right)$$

where, σ = Stefan-Boltzmann Constant (0.1714×10^{-8} BTU/hr-ft^2-°R^4)
ε = Emissivity of the shipping cask surface
A = Shipping cask surface area over the core region height (ft^2)
$T_{SURFACE}$ = Shipping cask surface temperature (°F)
T_{AIR} = Ambient temperature (°F)

The temperature differences across each material in the conduction path were determined in exactly the same manner, using the same assumptions, as described in Section 3.4.2.2. The resulting temperature profile through the side of the RPV package is shown in Table 3.4.2.

3.4.3 Minimum Temperature

The minimum package temperature was assumed to be -40°F. This corresponds to the minimum ambient temperature specified by 10 CFR 71.71(c)(2). No credit was taken for the heat generated by radioactive decay of the activation products in the RPV walls or the dross contained in the solidified waste package.

3.4.4 Maximum Internal Pressure

The maximum internal pressure was determined assuming that the air trapped in the RPV package is heated from the minimum temperature at the time the shipping cask is sealed to the maximum package temperature. The trapped air was treated as an ideal gas with a constant volume:

$$P_{FINAL} = P_{INITIAL} \left(\frac{T_{FINAL}}{T_{INITIAL}} \right)$$

The initial conditions at the time the shipping cask is sealed were assumed to be normal atmospheric pressure and an ambient temperature of 32°F. The final temperature is the maximum RPV package temperature (193.6°F). The resulting maximum internal package pressure is 19.53 psia. Based on a minimum atmospheric pressure of 3.5 psia, as specified by 10 CFR 71.71(c)(3), the maximum differential pressure across the shipping cask is 16.03 psi.

3.4.5 Minimum Internal Pressure

The minimum internal pressure was determined in the same manner as the maximum internal pressure discussed in Section 3.4.4, assuming that the air trapped in the RPV package is cooled from the maximum temperature at the time the shipping cask is sealed to the minimum package temperature.

The initial conditions at the time the shipping cask is sealed were assumed to be normal atmospheric pressure and an ambient temperature of 100°F. The final temperature is the minimum RPV package temperature (-40°F). The resulting minimum internal package pressure is 11.0 psia. Based on a normal atmospheric pressure of 14.7 psia, the differential pressure across the shipping cask in the maximum cold condition is 3.7 psi. The internal package pressure in the normal cold condition (-20°F), calculated as above, is 11.55 psia. Based on a maximum atmospheric pressure of 20.0 psia as specified by 10 CFR 71.71(c)(4), the pressure differential across the cask in the normal cold condition is 8.45 psi.

3.4.6 Maximum Thermal Stresses

The evaluation of thermal stress is discussed in Sections 2.6.1 and 2.6.2.

3.4.7 Evaluation of Package Performance for Normal Conditions of Transport

The Yankee RPV package meets the thermal performance requirements of 10 CFR 71. As indicated in Table 3.1.1, the maximum accessible surface temperature of the package is well below the limit for exclusive use shipments. In addition, the ranges of temperatures and pressures that the RPV package may experience under the normal conditions of transport are all within the design conditions of the package components. Also, Section 2.6 demonstrates that the maximum thermal stresses are within the allowable limits for the RPV package. Therefore, there will be no loss or dispersal of the radioactive contents, no significant increase in external radiation levels, and no substantial reduction in the effectiveness of the Yankee RPV package under the normal conditions of transport.

3.5 Thermal Evaluation for Hypothetical Accident Conditions

As stated in 10 CFR 71.52, Type B packages which contain only LSA material and which are transported as exclusive use shipments are exempt from the requirements of 10 CFR 71.73, "Hypothetical Accident Conditions." The Yankee RPV package contains only LSA concentrations of radioactive materials, as indicated in Section 1.2.3, and will be transported as an exclusive use shipment. Therefore, a thermal evaluation for hypothetical accident conditions per 10 CFR 71.73 is not required for the RPV package.

3.6 References

3.6.1 Westinghouse Specification for YNPS Reactor Vessel, E-567135-C, "Reactor Vessel", March 7, 1958.

3.6.2 "ASME Boiler and Pressure Vessel Code, Section VIII - Division 2, Pressure Vessels - Alternative Rules", Summer 1985 Addenda.

3.6.3 "Mechanical Engineer's Handbook", Sixth Edition, Edited by T. Baumeister, Published by McGraw Hill Book Co., Inc.

3.6.4 "Specification for Hot Service Thermal Insulation for Yankee Atomic Electric Plant", Stone & Webster, J.O. No. 9699, YS-2304, October 26, 1959.

3.6.5 "Heat Transfer", J. P. Holman, Third Edition, Published by McGraw-Hill Book Company, 1972.

Table 3.1.1

Summary of Thermal Evaluation Results

Parameter[1]	Result
Maximum Cask Surface Temperature	102.5°F[2]
Maximum Cask Temperature	171.6°F
Maximum RPV Temperature	179.1°F
Maximum Package Temperature[4]	193.6°F
Maximum Internal Pressure	19.53 psia
Maximum ΔP Across Cask at Maximum Package Temperature	16.03 psi
Minimum Package Temperature[3]	-40.0°F
Minimum Internal Pressure[3]	11.0 psia
Maximum ΔP Across Cask at Normal Cold Conditions[3]	8.45 psi

NOTES:

(1) The maximum cask surface temperature is determined without solar heating, as specified in 10 CFR 71.43(g). All other parameters are determined with the solar heating specified in 10 CFR 71.71(c)(1), except as noted.

(2) Limit for exclusive use shipments per 10 CFR 71.43(g) is 180°F.

(3) Value conservatively determined without credit for solar heating or heat generation within the package.

(4) Maximum package temperature occurs at the package centerline.

Table 3.2.1

Material Properties Used in Thermal Evaluation

Component	Material	Thermal Conductivity (BTU/hr-ft-°F)	Emissivity
Shipping Cask	A-516, Grade 70 Steel (with exterior paint)	23.6[1]	0.9[2]
Shipping Cask Internal Shield	A-516, Grade 70 Steel	23.6[1]	NA
RPV	SA-302, Grade B Steel	23.3[1]	NA
RPV Insulation	Calcium Silicate with Asbestos Fibers	0.0274[3]	NA
Concrete	100 lb/ft^3 Concrete	0.30[4]	NA
Concrete	75 lb/ft^3 Concrete	0.1962[5]	NA
Low Density Concrete	25 lb/ft^3 Concrete	0.0754[5]	NA
Air	NA	0.016[6]	NA

NOTES:

(1) From Table 1 of Reference 3.6.2.

(2) Low end of the range for various paints given in Table 8-2 of Reference 3.6.5.

(3) From Reference 3.6.4.

(4) From Reference 3.6.3.

(5) Determined based on the thermal conductivity of 80 lb/ft^3 concrete (from Reference 3.6.3), assuming that the reduced density is due to the introduction of air.

(6) From Table A-5 of Reference 3.6.5.

Table 3.4.1

RPV Package Temperatures Without Insolence

Component	Thickness (ft)	Outside Radius (ft)	Inside Radius (ft)	ΔT (°F)	Outside Temp (°F)	Inside Temp (°F)
Shipping Cask	0.2500	6.5833	6.3333	0.005	102.454	102.459
Air Gap	0.0052	6.3333	6.3281	0.147	102.459	102.606
Internal Shield	0.0833	6.3281	6.2448	0.002	102.606	102.608
Air Gap	0.0052	6.2448	6.2396	0.149	102.608	102.757
Concrete (75 lb/ft^3)	0.7709	6.2396	5.4687	1.926	102.757	104.683
Air Gap	0.0052	5.4687	5.4635	0.170	104.683	104.853
RPV Insulation	0.2500	5.4635	5.2135	4.898	104.853	109.751
Air Gap	0.0052	5.2135	5.2083	0.179	109.751	109.930
RPV Wall	0.6667	5.2083	4.5416	0.017	109.930	109.946
Air Gap	0.0052	4.5416	4.5364	0.092	109.946	110.03[illegible]
Concrete (25 lb/ft^3)	1.3594	4.5364	3.177	6.091	110.039	116.130
Air Gap	0.0104	3.177	3.1666	0.264	116.130	116.394
Concrete (100 lb/ft^3, No Dross)*	1.5834	3.1666	1.5832	2.979	116.394	119.373
Concrete (100 lb/ft^3, With Dross)*	1.5833	1.5833	0	5.078	119.373	124.451

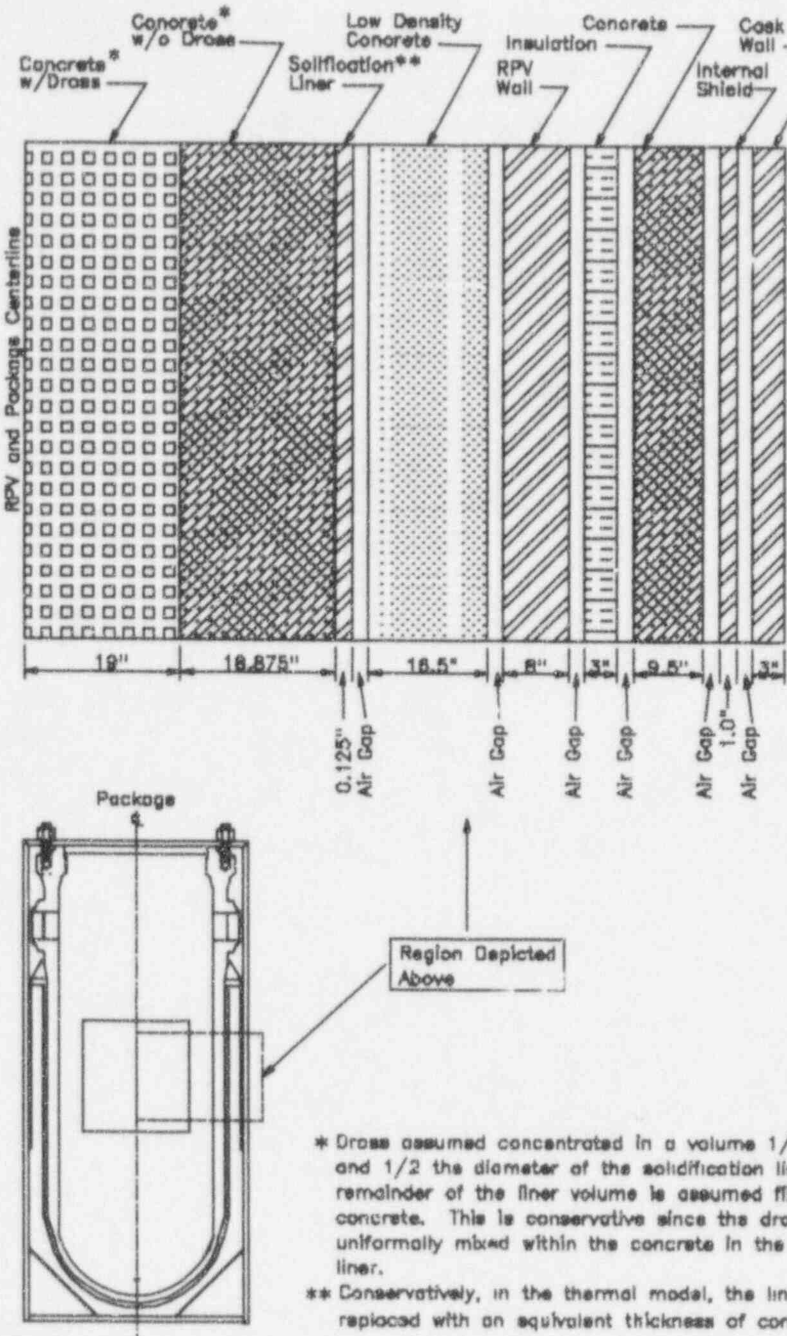
* Dross assumed concentrated in a volume one-half the height and one-half the diameter of the solidification liner. Remainder of liner volume assumed filled with normal concrete.

<u>Table 3.4.2</u>

<u>RPV Package Temperatures Including Insolence</u>

Component	Thickness (ft)	Outside Radius (ft)	Inside Radius (ft)	ΔT (°F)	Outside Temp (°F)	Inside Temp (°F)
Shipping Cask	0.2500	6.5833	6.3333	0.005	171.610	171.615
Air Gap	0.0052	6.3333	6.3281	0.147	171.615	171.762
Internal Shield	0.0833	6.3281	6 2448	0.002	171.762	171.764
Air Gap	0.0052	6.2448	6.2396	0.149	171.764	171.913
Concrete (75 lb/ft^3)	0.7709	6.2396	5.4687	1.926	171.913	173.838
Air Gap	0.0052	5.4687	5.4635	0.170	173.838	174.009
RPV Insulation	0.2500	5.4635	5.2135	4.898	174.009	178.907
Air Gap	0.0052	5.2135	5.2083	0.179	178.907	179.085
RPV Wall	0.6667	5.2083	4.5416	0.017	179.085	179.102
Air Gap	0.0052	4.5416	4.5364	0.092	179.102	179.195
Concrete (25 lb/ft^3)	1.3594	4.5364	3.177	6.091	179.195	185.286
Air Gap	0.0104	3.177	3.1666	0.264	185.286	185.550
Concrete (100 lb/ft^3, No Dross)*	1.5834	3.1666	1.5832	2.979	185.550	188.529
Concrete (100 lb/ft^3, With Dross)*	1.5833	1.5833	0	5.078	188.529	193.607

* Dross assumed concentrated in a volume one-half the height and one-half the diameter of the solidification liner. Remainder of liner volume assumed filled with normal concrete.



* Dross assumed concentrated in a volume 1/2 the height and 1/2 the diameter of the solidification liner. The remainder of the liner volume is assumed filled with concrete. This is conservative since the dross will be uniformally mixed within the concrete in the solidification liner.

** Conservatively, in the thermal model, the liner wall is replaced with an equivalent thickness of concrete

Figure 3.4-1

Conduction Path and Dimensions

4.0 CONTAINMENT

Containment for the YNPS RPV package is provided by the cylindrical shipping cask. The cask cylinder is fabricated from 3-inch thick, A516, Grade 70 steel. The top and bottom covers of the cask are 4-inch thick, A516, Grade 70 steel. The bottom cover will be welded to the cylinder during shop fabrication. The top cover will be welded to the cask at YNPS, after the RPV is placed inside. All penetrations listed in Table 4.1.1 will be capped or plugged, and seal welded. The bolts used for attaching trunnions to the cask will not penetrate the cask wall. The bolts will be removed and threaded plugs installed and seal welded.

Performance requirements for the package containment are provided by 10 CFR 71.43(f). The requirements are that "a package must be designed, constructed, and prepared for shipment so that under the test specified in §71.71 (Normal Conditions of Transport) there would be no loss or dispersal of radioactive contents, no significant increase in external radiation, and no substantial reduction in the effectiveness of the packaging." As shown in Section 1.2.3, the RPV package contains only LSA radioactive materials, and is therefore exempted from the requirements of 10 CFR 71.51, including the hypothetical accident conditions specified in 10 CFR 71.73 as the package will be transported as exclusive use.

Chapter 2 discusses the evaluations performed to ensure that the shipping cask meets the above requirements. As demonstrated in Chapter 2, the cask will maintain its integrity when subject to the normal conditions of transport and thereby meets the performance requirements of 10 CFR 71.43(f).

4.1 Containment Boundary

4.1.1 Containment Vessel

The design, shop fabrication and welding, and seal welding of the shipping cask will be performed in accordance with the ASME Boiler & Pressure Vessel (B&PV) Code, Section VIII, Division 2 (Reference 2.10.1). The weld, welding process, and non-destructive examination (NDE) of the top cover will be qualified using the ASME code as a guide. See Figures 1.2-2 through 1.2-4 for required NDE. The use of ASME Section VIII is consistent with the fabrication and welding recommendations of NUREG/CR-3019 and NUREG/CR-3854 (References 4.5.1 and 4.5.2).

Stresses in the cask when subject to the applicable normal conditions of transport per 10 CFR 71.71 are less than the ASME Section VIII allowables. The ASME Section VIII allowable stresses are consistent with the allowable stresses in Regulatory Guide 7.6 (Reference 2.10.2).

Several mechanisms exist to prevent the dispersal of radioactive contaminants. The primary mechanism is the containment provided by the shipping cask. Secondary mechanisms are:

(a) The majority of radioactive material is bound in the neutron activated RPV walls (approximately 68.7 percent) and the internal solidified waste package (approximately 31.1 percent). This leaves only about 0.2 percent of the radioactive material as surface contamination.

(b) The concrete placed in the RPV and in the RPV/cask annulus will fix any loose surface contamination in place.

4.1.2 Containment Penetrations

The shipping cask is penetrated by fill and vent ports in the cylinder and top cover, and holes in the top cover for installation of 22 RPV closure head studs. The fill and vent ports are used to place the concrete in the annulus between the RPV and the shipping cask and the low density concrete into the RPV. The penetrations are shown on Figures 1.2-2 and 1.2-4, and listed in Table 4.1.1. All penetrations will be plugged or capped, and seal welded. The bolts used for attaching trunnions to the cask will not penetrate the cask wall. The bolts will be removed and threaded plugs installed and seal welded. All seal welding will be performed in accordance with the requirements of the ASME B&PV Code, Section VIII.

4.1.3 Seals and Welds

There are no gasketed closures in the YNPS RPV package shipping cask. All openings are capped or plugged and seal welded closed.

Welds affecting the containment are shown in Figures 1.2-2 through 1.2-4. As stated above, all shop and field seal welding and weld inspections will be performed in accordance with the requirements of the ASME B&PV Code, Section VIII. Welding and inspection of the top cover to cylinder weld will be performed using the ASME code as a guideline. Inspection of the cover to cylinder weld will include volumetric and surface examinations.

4.1.4 Closure

There are no bolted closures which are part of the containment. The reactor head studs are used to connect the RPV to the top cover for lifting of the RPV only. Once the RPV is placed into the shipping cask, the cover will be welded in place as shown in Figure 1.2-4. Caps, also shown in Figure 1.2-4, will be placed over the studs and welded in place.

4.2 Requirements for Normal Conditions of Transport

The YNPS RPV package must meet the containment requirements of 10 CFR 71.43(f) when subject to the initial and test conditions for normal transport specified in 10 CFR 71.71. The requirements of 10 CFR 71.43(f) for a cask subject to the conditions of normal transport are that "there must be no loss or dispersal of radioactive contents, no significant increase in external radiation, and no substantial reduction in the effectiveness of the packaging."

The shipping cask has been evaluated for each applicable 10 CFR 71.71 specified test condition at the appropriate initial condition. The results of the evaluations are discussed in Chapter 2, Section 2.6 and Chapter 3. Section 2.6 shows that under all the applicable initial and test conditions for normal transport specified in 10 CFR 71.71(c), the shipping cask remains intact, and its shielding effectiveness is maintained. Furthermore, the placement of concrete in the package ensures the stability of the radioactive contents. Therefore, there is no increase in external radiation or reduction in the effectiveness of the packaging.

Sections 4.2.1 through 4.2.3 below demonstrate that the RPV package will prevent loss or dispersal of radioactive contamination during normal conditions of transport.

4.2.1 Containment of Radioactive Material

The single use shipping cask is the primary mechanism for the containment of radioactive material. All penetrations, listed in Table 4.1.1, are plugged or capped and then seal welded. The top cover will be welded in place to the cask cylinder. The cask penetrations are shown on Figures 1.2-1 and 1.2-4. The cask, plugs, caps, and all welds are designed for the applicable 10 CFR 71.71 normal conditions of transport. The results of the cask evaluations for these conditions show margins to conservative allowable stresses providing assurance that the cask will retain its integrity and prevent dispersal of radioactive material.

As discussed in Section 4.1.1, two secondary mechanisms also exist to contain radioactive material. These mechanisms are the integration of the majority of activity within the RPV walls and the internal solidified waste package, and the fixing in place of contamination by the concrete fill. Based on this discussion, the RPV package will provide for containment of radioactive materials when subject to normal conditions of transport.

4.2.2 Pressurization of Containment Vessel

The only vapors or gases which could form inside the shipping cask would be the result of air trapped inside prior to sealing the cask, or water vapor present in the concrete fill. Pressurization of the cask could only occur with a change in temperature and/or atmospheric pressure. The cask has been evaluated for the worst combination of temperature and pressure as specified in 10 CFR 71.71. As discussed in Section 3.4.4, this results in the cask being subject to a maximum internal pressure of 16.03 psi. This pressure produces small

stresses in the cask and therefore has no effect on the effectiveness of the shipping cask as containment.

There are no other conditions which could subject the shipping cask to an increased external or internal pressure.

4.2.3 Containment Criterion

Containment is provided by the shipping cask, its welds, and any plugs, caps, and seal welds. All welding and NDE of the welding will be performed in accordance with ASME B&PV Code, Section VIII. This provides assurance that there are no leak paths for the escape of radioactive materials. Dispersal of radioactive materials are further prevented by the concrete fill and sealing of the RPV. The integrity of the shipping cask is demonstrated by the results of the evaluations performed. Therefore, the shipping cask meets the containment requirements of 10 CFR 71.43(f).

4.3 Containment Requirements for Hypothetical Accident Conditions

As shown in Section 1.2.3, the YNPS RPV package contains only LSA radioactive materials, and is therefore exempted from the requirements of 10 CFR 71.51, including the hypothetical accident conditions specified in 10 CFR 71.73 since the package will be transported as exclusive use.

4.4 Special Requirements

The YNPS RPV package is not subject to the special requirements for plutonium shipments specified in 10 CFR 71.63. This section is therefore not applicable.

4.5 References

4.5.1 NUREG/CR-3019, "Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials".

4.5.2 NUREG/CR-3854, "Fabrication Criteria for Shipping Containers".

Table 4.1.1

Shipping Cask Penetrations

Penetration No.	Description	Location	
		Height[(1)]	θ[(2)]
C1	Concrete Injection Port	9'-0"	0°
C2	Concrete Injection Port	9'-0"	90°
C3	Concrete Injection Port	9'-0"	180°
C4	Concrete Injection Port	9'-0"	270°
C5	Concrete Injection Port	20'-0"	0°
C6	Concrete Injection Port	20'-0"	90°
C7	Concrete Injection Port	20'-0"	180°
C8	Concrete Injection Port	20'-0"	270°
T1-T4	Concrete Fill/Vent Port, Top Cover, RPV/Cask Annulus	6'-0"[(3)]	0°,90°, 180°,270°
T5, T6	Concrete Fill Port, Top Cover, RPV Interior	4'-0"[(3)]	90°,270°
T7	Vent Port, Top Cover, RPV Interior	0'-0"[(3)]	0°
T8 - T29	Top Cover, Holes for Reactor Head Studs	5'-3¾"[(3)]	See Figure 1.2-4

NOTES:

(1) Height is measured from the bottom of the cask. See Figure 1.2-2.

(2) θ is measured from the called North axis. See Figure 1.2-4.

(3) Measurement is radius from the center of the cask cover.

5.0 SHIELDING EVALUATION

10 CFR 71.47 provides the external radiation standards for packages for the transport of radioactive materials. For packages, such as the YNPS RPV package, which are transported as exclusive use by rail or highway, radiation levels external to the package are limited to:

(a) 200 millirem/hour on the accessible external surface of the package (10 CFR 71.47(a)).

(b) 200 millirem/hour at any point on the outer surface of the transport vehicle. In the case of an open vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the package, and on the lower external surface of the vehicle (10 CFR 71.47(b)).

(c) 10 millirem/hour at any point two meters from, in the case of an open vehicle, the vertical planes projected from the outer edges of the conveyance (10 CFR 71.47(c)).

The calculated dose rates at 0 and 2 meters from the surface of the RPV package are given in Table 5.1.1. These dose rates are well below the 10 CFR 71.47 limits. Therefore, the RPV package design, as shown in Figure 1.2-1, provides adequate shielding for shipment of the RPV package. Prior to transport of the package, actual dose rates will be measured as required by 10 CFR 71.87(j) to verify compliance with 10 CFR 71.47. The capability of the package to retain shielding integrity under normal conditions of transport is demonstrated in Chapters 2 and 3.

5.1 Shielding Design Features

The RPV will be placed in a 3-inch thick, carbon steel, cylindrical shipping cask. An additional 1-inch thick carbon steel cylinder is located inside the cask opposite the RPV core region. The cask is 28'-3" long with an outside diameter of 13'-2". The top and bottom ends of the cask are 4-inch thick carbon steel. Low density concrete will be used to fill the interior of the RPV. A 75 pcf concrete will be placed in the annulus between the RPV and the cask. A cross-section of the RPV package is shown in Figure 1.2-1.

5.1.1 Shield Description

5.1.1.1 Radial Shielding

Radial shielding is provided by: (a) the low density concrete placed inside the RPV, (b) the 8-inch thick carbon steel wall of the RPV, (c) 3-inch thick calcium silicate block insulation attached to the outside wall of the RPV, (d) a 75 pcf concrete placed in the annulus between the RPV and shipping cask, (e) the 1-inch thick cylindrical internal shield opposite

the core region, and (e) the 3-inch thick cylindrical wall of the shipping cask. The RPV package is shown on Figure 1.2-1.

The radial thicknesses of the concrete in the RPV/cask annulus are:

(a) 10.5 inches from the bottom head to the 1-inch thick internal shield,
(b) 9.5 inches in the core region at the internal shield, and
(c) a minimum of 1.75 inches at the RPV support lugs and main coolant nozzles.

5.1.1.2 Main Coolant Nozzle Shielding

There are 8 main coolant nozzles in the RPV. Each nozzle will be fitted with an 8-inch thick, carbon steel shield plug. This is equivalent to the RPV wall thickness around the core region. The nozzle plugs are shown in Figure 1.2-5.

5.1.1.3 Top Shielding

The shipping cask top cover provides the shielding at the top of the package. The top cover is 4 inches thick. Details of the top cover are shown in Figure 1.2-4.

5.1.1.4 Bottom Shielding

Shielding at the bottom of the RPV package is provided by the 4-inch thick carbon steel RPV bottom hemispherical head, 3-inch thick calcium silicate block insulation attached to the bottom head, 6-inch (minimum) thickness of 75 pcf concrete, ½-inch thick carbon steel cone plates, and the 4-inch thick carbon steel bottom of the shipping cask. Details of the bottom shielding are shown in Figure 1.2-1.

5.1.2 Shield Effectiveness

The QAD-UE computer code (Reference 5.6.1) was used to determine dose rates at 0 and 2 meters from the surface of the RPV package. Dose rate calculation points are shown on Figure 5.1-1. The internal solidified waste package (discussed below) and the RPV are modeled independently. Dose rates from the two evaluations are combined to determine the total dose rates for the package. The calculated dose rates on 1/1/97 at 0 and 2 meters from the surface of the RPV package are shown in Table 5.1.1. These dose rates are significantly lower than the 10 CFR 71.47 limits. Therefore, the package design as shown in Figure 1.2-1 provides adequate shielding for transport of the package.

5.2 Source Specification

There are two principal sources of radioactivity present in the RPV package: the activated RPV components, and the internal solidified waste package containing the dross

material. The total number of curies of each isotope in the RPV and in the dross are given in Table 1.2.2. The two sources are discussed below.

5.2.1 Reactor Pressure Vessel (RPV)

For the shielding analysis performed, beltline average source data with a simplified axial distribution was assumed. Expected variation in the azimuthal direction has not been explicitly accounted for, nor has the detailed axial distribution calculated from Section 1.2.3 been assumed. Comparisons of measured dose rates presented in Section 1.2.3 confirm the use of the average source versus peak location sources due to conservative assumptions made in both the activation analysis and the shielding analysis performed.

The radioactive material present due to neutron activation was analyzed using the DORT and ORIGEN computer codes. This analysis is discussed in Section 1.2.3. Three activated components of the RPV are considered: the 8-inch thick carbon steel RPV wall, the 0.109-inch thick stainless steel internal clad , and the 1/80-inch thick stainless steel insulation wrapper. The neutron flux at the outer diameter of the RPV during irradiation was significantly lower than the flux at the inner diameter, due to attenuation and backscatter of the neutrons. This was taken into account in the analysis by dividing the RPV wall into 20 regions. This 20 region modelling is consistent with the vessel mesh spacing in the activation calculation. The average neutron flux in each region was used to calculate the radial activity profile in the RPV wall region. Activities for the reactor vessel insulation wrapper, a 1/80th of an inch thick stainless steel covering on the vessel insulation, were determined from the specific activities of the vessel inner clad. The ratio of the thermal flux at the wrapper location to the thermal flux at the vessel inner clad was determined from the DORT radial analysis and was conservatively adjusted to bound the wrapper activity. This process was performed because the wrapper was not explicitly represented in the neutron transport calculations. The vessel inner clad is stainless steel like the wrapper and more representative of the specific activity of the wrapper than the vessel which is carbon steel. The activity of the clad is higher than the outside of the vessel because: (1) stainless steel contains a higher cobalt impurity than carbon steel (1414 ppm vs 120 ppm) and (2) the wrapper activities determined by ratio were increased by 50 percent to account for uncertainties in the ratioing. Fourteen radionuclides were calculated in these components, although only two, Co^{60} and Mn^{54}, are of any consequence in the shielding evaluation. These two nuclides account for more than 99 percent of the gamma activity. The calculated activity distribution in the RPV is given in Table 1.2.2. The source terms used in the shielding calculation in μCi/g and in MeV/cm^3-sec around the active core region are shown in Table 5.2.1. The source terms listed in Table 5.2.1 are the estimated gamma source terms for the activated RPV components on January 1, 1994. The shipment date of the RPV package will be after 1/1/97. Thus, the dose rates calculated with these source terms were decay-corrected by a decay factor of 0.674 to obtain the the dose rates on 1/1/97.

The active core region is 91 inches (231.1 cm) long. The core region starts at 119.3 inches (303.0 cm) and ends at 210.3 inches (534.2 cm) above the bottom head. The core

center line is 164.8 inches (418.6 cm) above the bottom head. The relative axial source in the active core region was assumed to be the core average values as provided in Table 5.2.1. At 74.8 inches (190 cm) above and at 59.1 inches (150 cm) below the core region, the source was assumed to be zero due to negligible relative axial thermal flux. Between the above two regions, the source terms were conservatively assumed to be one tenth of those in the active core region. Figure 5.2-1, provides a comparison of the actual relative axial thermal flux profile with the values used in the shielding evaluation.

5.2.2 Dross

The activated dross deposited at the bottom of the RPV during the cutting of core internals will be uniformly mixed with concrete in a solidification liner creating an internal solidified waste package. The solidified waste package will be located in the core region of the RPV (Figure 1.2-1). The density of the concrete in the waste package will be 100 pcf (1.6 g/cm^3) (minimum). The dimensions of the dross-concrete matrix are assumed to be the same as the internal dimensions of the solidification liner, i.e., a 38 inch radius and a 77 inch height.

The solidified waste package is LSA material (see discussion in Section 1.2.3). The Co^{60} nuclide accounts for more than 99 percent of the gamma activity in the dross (see Table 1.2.2). The dross will be uniformly mixed with concrete in the liner, therefore, the source will be equally distributed through the dross-concrete volume. For conservatism, this dross-concrete matrix is assumed to contain 0.3 mCi/g of Co^{60} (the LSA limit for Co^{60}), although the actual curie content is much less. Each disintegration of Co^{60} emits two gammas with an average energy of 1.25 MeV. Thus, the source term for the dross is 4.44E+07 MeV/cm^3-sec.

5.3 Model Specification

The radial and the top axial shielding configurations were modeled separately. The bottom axial shielding configuration was not modeled. The dose points above the top of the package are more limiting than the corresponding dose points below the bottom of the package (see discussion in Section 5.4 below).

5.3.1 Description of Shielding Configuration

5.3.1.1 Radial Shielding Model for Dose Rate at Package Side

The cylindrical radial shield model for calculation of dose rates at the side of the package is shown in Figure 5.3-1. The model consists of eight concentric cylindrical regions, A through H.

- Region A: 3-inch thick carbon steel shipping cask wall with a 1-inch thick carbon steel internal shield around the core region.

- Region B: annulus between the RPV and the cask which is uniformly filled with 75 pcf concrete.

- Region C: 1/80-inch thick stainless insulation wrapper.

- Region D: 3-inch thick block insulation which is modeled as air.

- Region E: 8-inch thick carbon steel RPV wall.

- Region F: 0.109-inch thick stainless steel internal clad.

- Region G: 25 pcf concrete fill within RPV.

- Region H: Solidification liner containing activated dross uniformly mixed in 100 pcf concrete.

Dimensions of the regions are shown in Figure 5.3-1. The dose points are located at the core center line elevation, at 0 and 2 meters from the surface of the shipping cask wall (Figure 5.1-1).

5.3.1.2 Top Axial Shielding Model for Dose Rate Above Package Top

The cylindrical top axial shielding cylindrical model for calculation of dose rates above the top of the RPV package is shown in Figure 5.3-2. The model is similar to the radial shielding model with the side wall of the cask (Region A) and annulus concrete (Region B) removed and the 4-inch thick carbon steel cask cover added. The dose points are located on the package centerline at 0 and 2 meters from the surface of the cover (see Figure 5.1-1).

5.3.2 Source and Shield Regional Densities

The source and shielding materials are identified in Figures 5.3-1 and 5.3-2. The densities of the materials are given in Table 5.3.1.

5.4 Shielding Evaluation

The dose rates at 0 and 2 meters from the surface of the RPV package were calculated by using the sources and the models described in Sections 5.2 and 5.3. The computer code QAD-UE (Reference 5.6.1) was used to perform the calculation. The iron dose buildup factor was used in the calculation. The dose rate contribution from each source was calculated individually. The dose rate at each dose point was obtained by summing the contributions

from all sources. At the side of the RPV package, the dose rate contribution from the activated dross is negligible. On the surface of the shipping cask cover, dose rate contributions from the RPV and the dross sources are 2.0 and 2.6 mrem/hr, respectively. At 2 meters from the surface of the shipping cask cover, dose rate contributions from the RPV and the dross sources are 3.3 and 1.2 mrem/hr, respectively. The calculated surface dose rate contribution from the RPV at the outside surface of the shipping cask cover is less than the dose rate at 2 meters from the cover. This is a result of the conservative assumption regarding the axial distribution of the Co-60 activation product in the RPV as shown in Figure 5.2-1. As can be seen from Figure 5.3-2, there is more attenuation through the concrete and the steel cover for radiation reaching the surface dose point than there is for radiation reaching the 2 meter receptor location. This attenuation is sufficient to overcome the effects of buildup and distance reduction, resulting in a lower calculated dose rate at the surface than 2 meters. If a more realistic axial source distribution had been chosen, the activity above the core region would be much less, and the calculated dose rate contribution from the RPV at 2 meters would have been less than the surface dose rate. Table 5.1.1 summarizes the calculated dose rates. These dose rates are well below the 10 CFR 71.47 limits.

The dose points below the bottom of the package are shielded by the 4-inch thick carbon steel RPV bottom head and 6 inches (minimum) of 75 pcf concrete, in addition to the 4-inch thick carbon steel cask cover and the low density concrete in the vessel (see Figure 1.2-1). The bottom head is more than 60 inches below the core region and is not significantly activated. This can be seen in Figure 5.2-1, a plot of the relative axial thermal flux profile. In comparison, the dose points above the top of the cask are only shielded by the 4-inch thick carbon steel cask cover and the low density concrete in the RPV. The half-value layer of iron for Co^{60} is about one inch. The half-value layer of concrete for Co^{60} is about 2.5 inches. Therefore, the dose rates at locations below the cask bottom are at least ten times smaller than the dose rates at locations above the top of the cask.

Each Main Coolant nozzle is fitted with an 8" carbon steel plug (see Figures 1.2-1 and 1.2-5). This is equivalent to the 8" vessel wall around the core region. Since the nozzle elevation is more than 3.5 feet above the top of the core region, the source strength near the nozzle is at least 10 times less than the source strength of the RPV wall around the core region. This can be seen in Figure 5.2-1. Thus, at the same distance from the side surface of the package, the dose rate at the nozzle elevation will be less than that at the core center line elevation.

5.5 Analysis Verification

5.5.1 Dose Rate Verification

The dose rate in water inside the RPV at 6 inches from the stainless steel clad and the dose rate outside the RPV in contact with the 1/80" stainless steel insulation wrapper were calculated for comparison with actual dose rate measurements. The geometric model for

these two dose rate calculations is similar to Figure 5.3-1 with the cask removed and the low density concrete in the RPV replaced by water. A comparison of the calculated and the measured dose rates is shown in Table 5.5.1. The calculated dose rates are higher than the measured dose rates. Thus, the source term and the model used in the verification analysis produce conservative results relative to the measured values.

5.5 QAD-UE

QAD-UE is a point-kernel code for calculating fast-neutron and gamma-ray penetration through various shield configurations defined by combinatorial geometry specifications. This version of QAD incorporates a revised gamma-ray point kernel source volume integration option to reduce user setup time and provide increased computational efficiency for large volume source problems. It contains all the options and features of QAD-CG.

QAD-UE uses a point kernel ray-tracing technique for gamma-ray calculations and either a modified Albert-Welton kernel or kernels obtained from the moments method solution of the Boltzmann equation for neutron penetration calculation. The new technique performs a coupled ray angular integration at the detector position as opposed to an integration over the source volume.

5.6 References

5.6.1 ORNL RSIC Code Package CCC-448, "QAD-UE, A Revised Numerical Integration Option for Gamma-Ray Volume Source Problems in the QAD-CG Point Kernel Shielding Code System", N. L. DeGangi, 1/1994.

Table 5.1.1

Summary of Calculated Dose Rates on 1/1/97

	Surface Dose Rate[1] (mrem/hr)	2 meter Dose Rate[2] (mrem/hr)
Side of RPV Package[3]	16.4[3]	6.6[4]
Top of RPV Package[3][4]	4.6[5]	4.5[5]

NOTES:

(1) Maximum allowable dose rate is 200 mrem/hr (10 CFR 71.47).

(2) Maximum allowable dose rate is 10 mrem/hr (10 CFR 71.47).

(3) See Figure 5.1-1 for locations of dose points.

(4) The dose rates for the corresponding dose points at the bottom of the package are at least 10 times smaller. See discussion in Section 5.4.

(5) See discussion in Section 5.4 regarding relative values of surface and 2-meter dose rates.

Table 5.2.1

Source Terms at Active Core Region (1/1/94)

SourceNumber	Source Term in μCi/g		Source Term in MeV/cm^3-sec	
	Co-60	Mn-54	1.25 MeV gamma	0.835 MeV gamma
1	508.000	2.970	3.69E+08	7.21E+05
2	30.664	4.020	2.23E+07	9.76E+05
3	18.223	3.710	1.32E+07	9.01E+05
4	15.310	3.270	1.11E+07	7.94E+05
5	13.730	2.930	9.98E+06	7.11E+05
6	12.510	2.670	9.10E+06	6.48E+05
7	11.500	2.460	8.36E+06	5.97E+05
8	10.610	2.270	7.71E+06	5.51E+05
9	9.818	2.100	7.14E+06	5.10E+05
10	9.093	1.940	6.61E+06	4.71E+05
11	8.425	1.800	6.13E+06	4.37E+05
12	7.801	1.670	5.67E+06	4.05E+05
13	7.217	1.540	5.25E+06	3.74E+05
14	6.665	1.420	4.85E+06	3.45E+05
15	6.142	1.310	4.47E+06	3.18E+05
16	5.645	1.200	4.10E+06	2.91E+05
17	5.171	1.100	3.76E+06	2.67E+05
18	4.719	1.010	3.43E+06	2.45E+05
19	4.294	0.917	3.12E+06	2.23E+05
20	3.896	0.832	2.83E+06	2.02E+05
21	3.554	0.759	2.58E+06	1.84E+05
22	22.400	0.080	1.63E+07	1.94E+04

NOTES:

(a) Source #1 is the 0.109" SS clad of the inner surface of the RPV.

(b) Source #22 is the 1/80" SS wrapper of the block insulation on the RPV.

(c) The 8" RPV wall is evenly divided into 20 sources, #2 to #21.

Table 5.3.1

Densities of Source and Shielding Materials

Material	Density (g/cm^3)
Air	0.001293
Carbon Steel (Fe)	7.86
Stainless Steel (Fe)	7.86
25 pcf concrete	0.40
100 pcf concrete	1.60
75 pcf concrete	1.20

Table 5.5.1

Comparison of Measured and Calculated Dose Rates

	Measured Dose Rate (R/hr)	Calculated Dose Rate (R/hr)	Ratio C/M(4)
6" from the RPV SS Internal Clad (in water)	60(1)	87(2)	1.450
Contact with 1/80" SS Insulation Wrapper	8(3)	12.5(2)	1.563

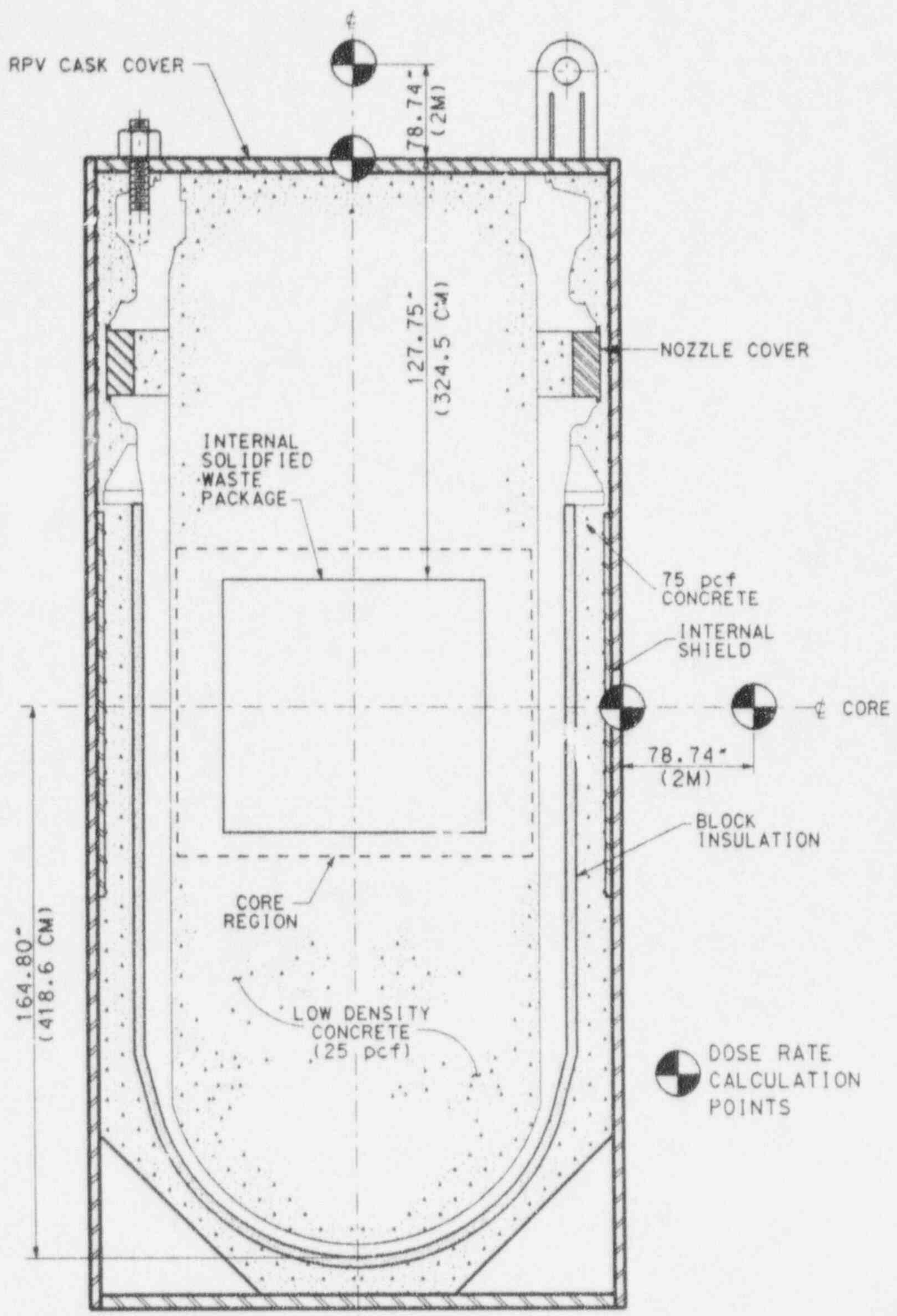
NOTES:

(1) Measured internal beltline average dose rate on 7/20/94.

(2) Calculated dose rate is based on 1/1/94 source given in Table 5.2.1.

(3) Maximum dose rate on 1/6/95 based on extrapolation of measured values to the peak location.

(4) Ratio of the calculated value to the measured value.



TITLE: DOSE RATE CALCULATION POINTS

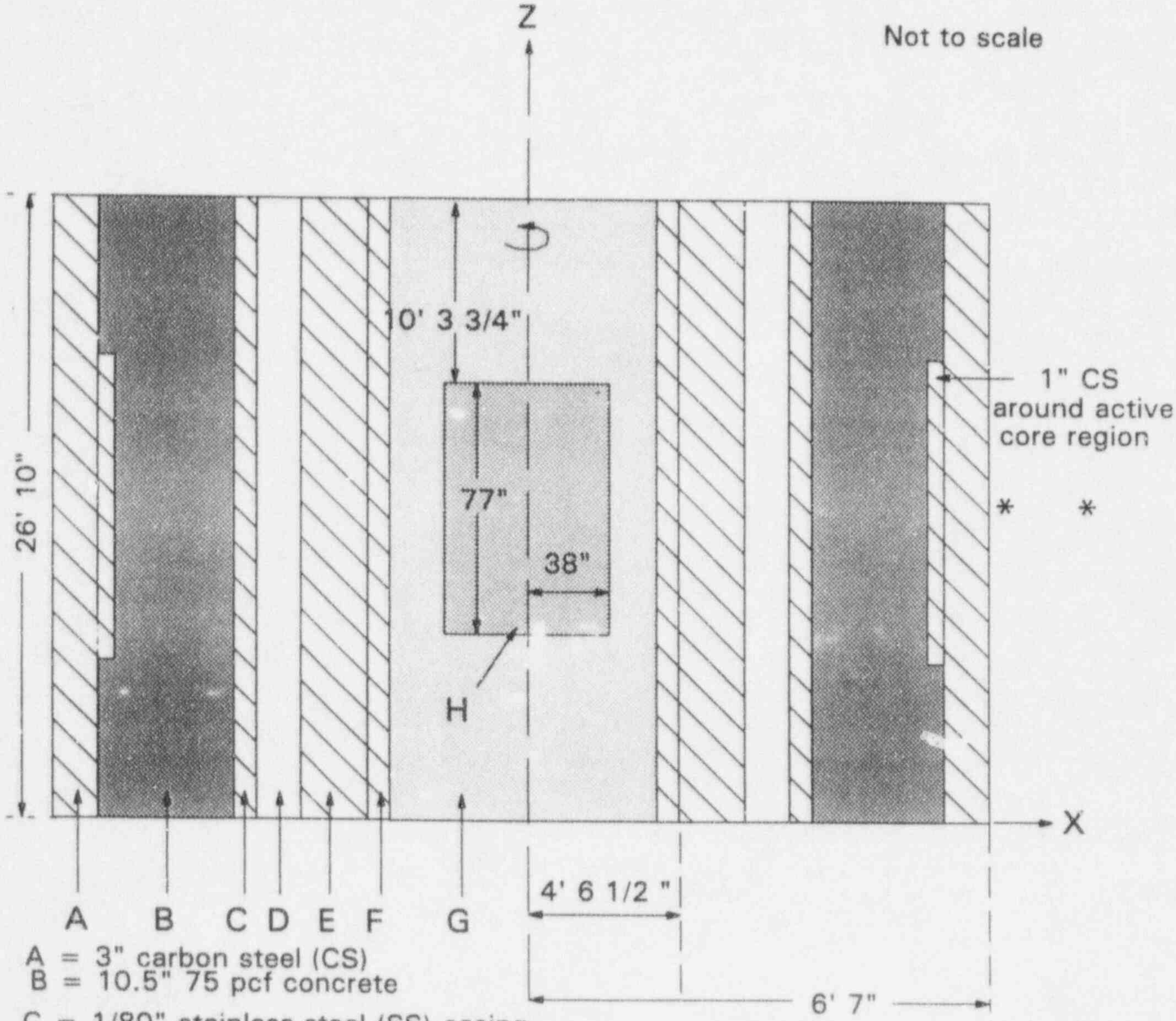
SCALE: NONE

FIGURE 5.1-1

E:/DGN/YRDECOM/F10511.DGN

Figure 5.3-1

Radial Shielding Model

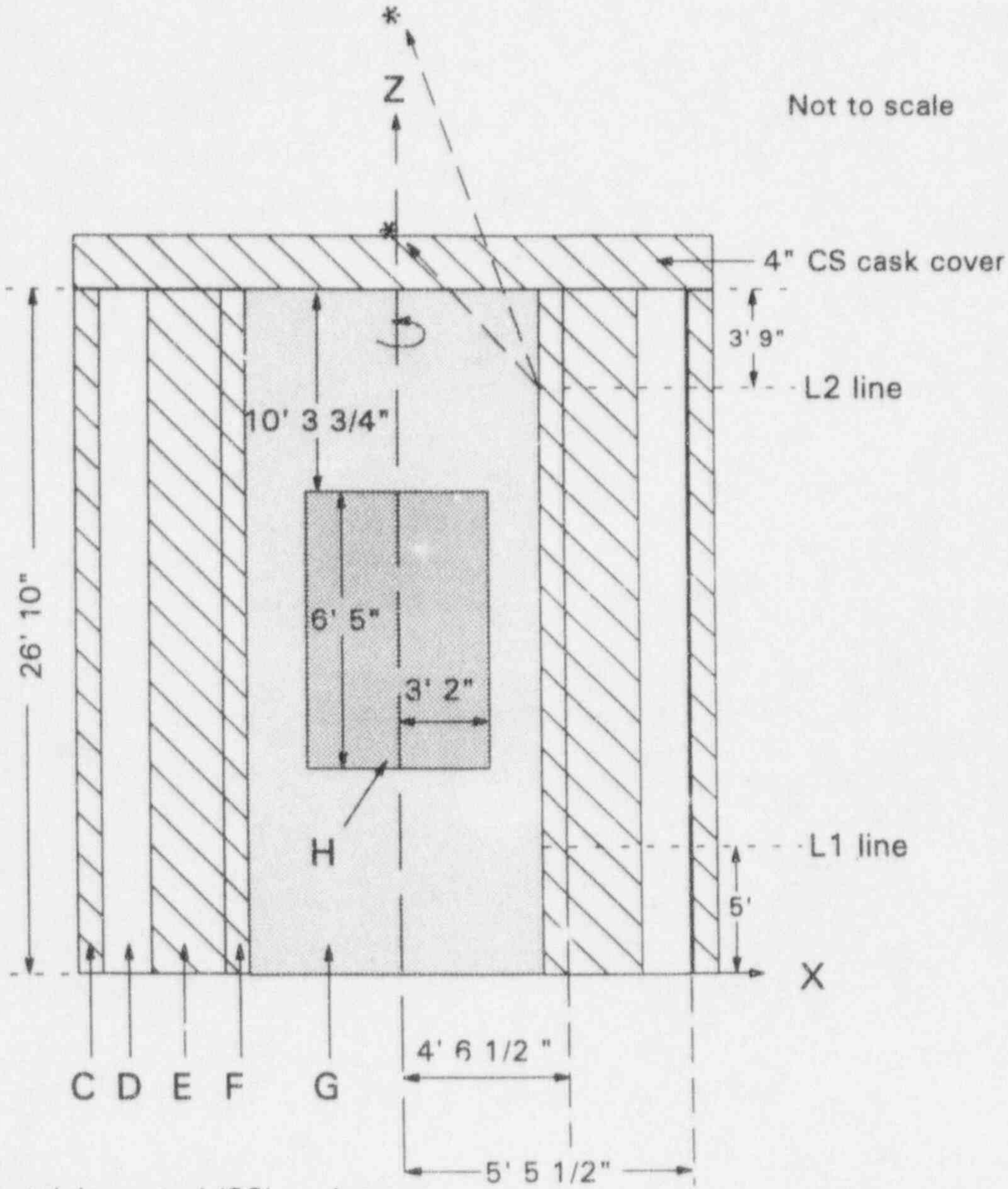


A = 3" carbon steel (CS)
B = 10.5" 75 pcf concrete
C = 1/80" stainless steel (SS) casing
D = 3" block insulation
(modeled as air)
E = 8" CS RPV wall
F = 0.109" SS clad
G = 25 pcf concrete in RPV cavity
H = Assumed liner contains 0.3 mCi/g of Co-60
uniformly mixed in 100 pcf concrete

* Dose points on surface
and at 2 meter distances

Figure 5.3-2

Top Axial Shielding Model



C = 1/80" stainless steel (SS) casing
D = 3" block insulation
(modeled as air)
E = 8" CS RPV wall
F = 0.109" SS clad
G = 25 pcf concrete in RPV cavity
H = Assumed liner contains 0.3 mCi/g of Co-60
uniformly mixed in 100 pcf concrete

* Dose points on surface and at 2 meter distances

Note: Activation above L2 line and below L1 line is negligible.

6.0 CRITICALITY EVALUATION

Only a very small concentration of fissile radionuclides is present in the RPV package. These radionuclides were created from trace concentrations of parent nuclides in the RPV base metal and in the dross. There are less than 15 grams of fissile radionuclides in the RPV package. Therefore, the RPV package contents meet the requirements for exemption from classification as fissile material as delineated in 10 CFR 71.53. A criticality evaluation is therefore not performed.

7.0 OPERATING PROCEDURES

As presented in Regulatory Guide 7.9, this chapter is intended to describe the operating procedures to be used in the preparation for and performance of the process of loading and unloading the package. In the case of the YNPS RPV package, the package will be loaded, but is single use and will not be unloaded. Therefore, this chapter will be used to describe the procedures for loading the package and the methods to be used in transporting the package off-site.

7.1 Procedures for Loading the Package

7.1.1 Removal Activities Prior to Lifting the RPV

Two major operations are required prior to lifting the RPV and placing it into the shipping cask. The first operation will be the removal of the upper neutron shield tank (NST). Removal of the upper NST is required to gain access to the main coolant piping for cutting and removal, and to allow clearance to lift the RPV. The upper NST is comprised of horizontal and vertical plates which will be cut using remote methods wherever possible. See Figure 7.1-1 for an elevation view of the RPV, NST, and MC piping.

The second major operation will be the cutting of the main coolant (MC) piping. The eight (8) main coolant pipes will be cut after removal of the upper NST. The RPV is totally supported by twenty-eight (28) lugs which rest on a concrete corbel. The MC piping therefore does not provide support to the RPV, and may be cut without providing temporary supports. The MC piping will be cut at the RPV nozzles and also at approximately three (3) feet away from the RPV, near the Biological Shield wall. The sections of MC piping will be removed and the pipe and RPV nozzle openings will be plugged. The RPV nozzle plugs are shown in Figure 1.2-5.

7.1.2 Dross Solidification

The 1.0 ft^3 of contaminated residue ("dross") present on the bottom of the RPV will be collected into a liner. The dross will be solidified into a homogeneous monolith within the liner. The solidified dross will be encapsulated within the RPV, resulting in a single solidified LSA waste package. The classification of the solidified waste package and RPV are discussed in Section 1.2.3. The process of solidification is discussed below.

The solidification process will be performed in accordance with CNSI's Process Control Program (PCP) (Reference 7.6.1). The PCP has been used to solidify radioactive waste at many sites, and is approved by the State of South Carolina Department of Health and Environmental Control (DHEC). A site specific procedure will be developed to control the

dross solidification. All procedures used for solidification will be reviewed and approved by the YNPS Plant Operations Review Committee prior to use. Project specific equipment will be designed and fabricated to make use of existing components for equipment installation. All project specific equipment will be tested prior to use. All personnel directly involved in the solidification will be qualified by CNSI's qualification and training program.

The site solidification process will start with the addition of chemicals to the water in the RPV. The chemicals will improve water clarity for visual surveillance, insure that the dross is settled to the bottom of the RPV, and also improve the pumpability of the dross. The next step will be to collect any large debris (tie wraps, rags, etc.) from the bottom of the RPV which could hinder action of the dross collection pumps. Any debris collected will be disposed of in accordance with YNPS procedures. The mixing platform with pumps will be installed on the RPV thermal shield support lugs next and the water in the RPV will be lowered to just below the platform. Water removed from the RPV during solidification will be processed using the normal plant liquid radwaste processing system. The water will not be returned to the RPV or placed in the solidification liner. After the water level is lowered, the solidification liner, fill head and associated hardware will be installed on the liner platform. The water level will continue to be lowered until approximately 120 ft^3 of water containing the 1.0 ft^3 of dross remains. The conceptual solidification hardware is shown in Figure 7.1-2.

The remaining water and dross will then be pumped into the solidification liner. The bottom of the RPV will be visually inspected using a camera either mounted on the mixing platform or operated from above the RPV. If material is observed on the RPV bottom, the RPV will be rinsed using demineralized water, and the water/dross mixture will be pumped into the liner. Again utilizing the camera, a final visual inspection will be performed verifying that the dross has been removed.

The volume of water left in the RPV to be pumped into the liner is calculated based on a total solidified volume of 175 ft^3 with a target density of 100 lb/ft^3 and a water/cement ratio of about 1.0. If water in excess of this amount is pumped into the liner, the water in the liner will be decanted, and the excess removed. Any water removed from the liner will be processed by the plant liquid radwaste system. The target volumes and density of the solidified waste are based on the classification discussed in Section 1.2.3. A solidified waste form with a target volume of 175 ft^3 and a density of 100 pcf will have isotopic concentrations at 61 percent of the 10 CFR 71 LSA limit.

The water/cement volumes may be adjusted based on a sampling of the liner contents performed prior to solidification. Sample results may require the addition of chemicals (such as lime to adjust pH) and other proprietary CNSI materials to assure a properly solidified material. The water and cement volumes will be adjusted as required after the addition of chemicals to achieve the target volume and density. A sample solidification will be performed in accordance with the CNSI PCP. The sampling and sample solidification are a normal part of any

solidification process and provide assurance that the full contents of the liner will be properly solidified.

After sampling is completed, the volume of water in the liner will be verified as correct or adjusted as required, and the cement and any additional solidification chemicals will be introduced. The addition of the cement, and solidification chemicals will be performed at a prescribed rate. Mixing will commence and continue at a predetermined rate for a planned time. The liner contents will then be allowed to cure and properly solidify prior to lifting the RPV with the solidified liner inside. Proper solidification will be verified.

This solidification process is unique only in its location. The process is routinely performed at many sites which process radioactive waste materials. The mixing equipment (liner, mixing blades, motor, etc.) has been used for other solidification programs. The solidification process described above will result in an inherently stable final product meeting the requirements of 10 CFR 71 for LSA. The dross solidification process described above provides the safest and most reliable solution for disposal of the dross while practicing ALARA and minimizing personnel exposure.

7.1.3 RPV Preparation Prior to Lifting

Prior to lifting the RPV, the shipping cask cover, with lifting clevis installed, will be bolted to the RPV using 22 of the existing 5¼-inch diameter reactor head studs. The top cover, lifting clevis, and reactor head studs have been evaluated for the loads imposed during lifting the RPV, and meet the requirements of the AISC Design Specification (Reference 2.10.11) and ASME Section VIII for the studs. See Figure 1.2-4 for details of the top cover and bolting.

7.1.4 Shipping Cask Preparation

The shipping cask will be moved into position directly beneath the equipment hatch during preparations for lifting the RPV. The cask will be maneuvered beneath the hatch in a horizontal position and upended to a vertical position. The upending operation is necessary since there is not adequate clearance beneath containment to transport the cask while in a vertical position. Similar upending operations were required during plant construction to bring the steam generators, pressurizer, and RPV into containment.

7.1.5 Lifting the RPV

The YNPS polar crane will be used to lift the RPV. The crane will lift the RPV straight up until the bottom of the RPV is above the charging floor. Once at this elevation, the RPV will be moved in a straight line directly to a position over the containment equipment hatch. The RPV will then be lowered into the shipping cask. The load path is shown in Figure 7.1-3.

7.1.6 Preparing the Package for Transport

Once the RPV is secured in the cask, the polar crane hooks will be removed. Next, the concretes will be placed into the annulus between the RPV and cask and into the RPV interior, and allowed to cure. The annulus concrete serves to stabilize the RPV in the cask and avoid any "hard spots" by eliminating contact between the RPV and cask. The fill/vent ports in the cover will not be sealed until the concretes have cured, allowing excess moisture from the concretes to escape from the package. Threaded pipe plugs will not be used for any seals.

The cask will be downended onto a transporter or steel bed after the annulus concrete has cured and the cask cover is welded in place. The exterior of the package will be surveyed for both contamination and dose rate once it is in a horizontal position. The exterior of the package will not be extensively exposed to radioactive contamination, and it is expected that there will be minimal, if any, surface contamination. The exterior survey will insure that the package meets the contamination requirements of 10 CFR 71.87(2) for an exclusive use package.

7.1.7 ALARA Considerations During Package Loading

All activities associated with removal of the RPV and placing it into the shipping cask will be planned and engineered to keep radiation exposure ALARA. All work will be performed in accordance with reviewed and approved procedures and will be performed under the direction of experienced Radiation Protection personnel.

The following specific measures will be implemented to help keep exposure ALARA.

(1) A temporary RPV cover/radiation shield will be installed on the RPV flange after RPV closure head removal. The temporary cover will incorporate shielded access ports for performance of internal RPV work. The cover will be used as a personnel shield until the shipping cask cover is installed on the RPV.

(2) Cutting and removal operations will be performed using remote and/or automated methods whenever possible. As required, areas will be tented to reduce airborne contamination.

(3) No work will be performed on the RPV during the lift. A minimum of personnel will be allowed inside containment during the actual lift and traverse of the RPV. Monitoring of the lift will be performed remotely to the extent possible.

(4) Exhaust from the package vent ports will be routed through HEPA filters and then to the plant primary vent stack. This allows control and monitoring of any airborne contamination from the package during the concreting operations.

7.2 Procedures for Unloading Package

The YNPS RPV package is a single use package. The package will be welded closed prior to shipment, and will not be reopened. Therefore, the requirements of 10 CFR 71.89 are not applicable.

7.3 Preparation of Empty Package for Transport

As previously discussed, the YNPS RPV package is single use. Therefore, this section is not applicable.

7.4 Transport - YNPS to Rail Site

The RPV package will be transported from the plant to the railway using a heavy haul transporter, specially selected for this application. The route will traverse a private dam and bridge, and public roads. This section will discuss the route and the process used to insure that the route is adequate for transport.

7.4.1 Pre-Transport Preparation and Checks

Prior to leaving the site, the package will be prepared for transport. This includes removing clevises and trunnions, loading the package onto the heavy hauler, installing the tie-down system, pre-transport checks, and acceptance tests. Acceptance tests, discussed in Chapter 8, are performed to insure that the package meets the requirements of 10 CFR 71.87.

The following pre-transport checks will be performed. The transporter will be checked and tested to insure that load leveling systems and safety components are functioning. The tie-downs will be inspected to insure proper installation. The package will be marked in accordance with 49 CFR 172.

7.4.2 Roadway Transport Route

The roadway transport route begins at the YNPS site boundary in the town of Rowe and ends approximately six miles away at the railway transfer site. The package will be transported across the Sherman Dam, across the Sherman Dam spillway bridge, and then onto Readsboro Road, a public road in the town of Monroe. The package will proceed south along Readsboro

Road into the town of Florida. In Florida, the road name changes to River Road. The package will continue along River Road to the railway transfer site.

The transport route is located entirely within the Commonwealth of Massachusetts. The route involves two towns: Monroe, located in Franklin County, and Florida, located in Berkshire County.

The following evaluations of the route will be performed prior to transport of the package:

(1) Slope stability along the public roadway, the Sherman Dam crest roadway, and at the Sherman Dam spillway bridge will be evaluated.

(2) The existing Sherman Dam spillway bridge will be evaluated, and if necessary, replaced prior to transport of the RPV package. Evaluation of the existing bridge, and if required, design of the replacement bridge will be performed in accordance with AASHTO (Reference 2.10.12).

(3) Culverts on public roads will be evaluated to AASHTO standards for the package loads.

7.4.3 <u>Heavy Haul Transporter</u>

The package will be transported using a dedicated heavy haul transporter. The hauler is comprised of a prime mover and a hydraulically [illegible]eled trailer. The trailer will have a capacity exceeding the combined package and tiedown system weight. The prime mover will consist of one or more units to provide adequate pulling power.

7.5 <u>Rail Transport</u>

The RPV package will be transported from the railway transfer site to a licensed low-level radioactive waste disposal facility using a dedicated train. The rail car will have a rated capacity greater than the combined package and tie down system weights. The package will meet railway requirements for size and center of gravity location. The tie-down system was designed using accelerations 50 percent higher than those required by AAR (Reference 2.10.5). A comparison of ANSI, AAR, and design accelerations is provided in Table 2.6.4.

7.5.1 Transfer to Rail Car

A suitable surface will be prepared at the railway transfer site for moving the package from the heavy hauler to the rail car. The hauler will be parked on the surface parallel to the tracks. Cribbing will be placed adjacent to the trailer, and steel beams will then be placed between the package and the hauler. The steel beams will rest on the hauler, railcar, and cribbing. The saddles and wire ropes tieing the package to the hauler will be disconnected from the hauler. The package will then be transferred to the railcar using the steel beams. The saddles will remain attached to the package during the transfer. The steel beams will be removed and the package tied to the rail car. At no time will the package be suspended or lifted by an overhead lifting device.

7.5.2 Rail Transport Route

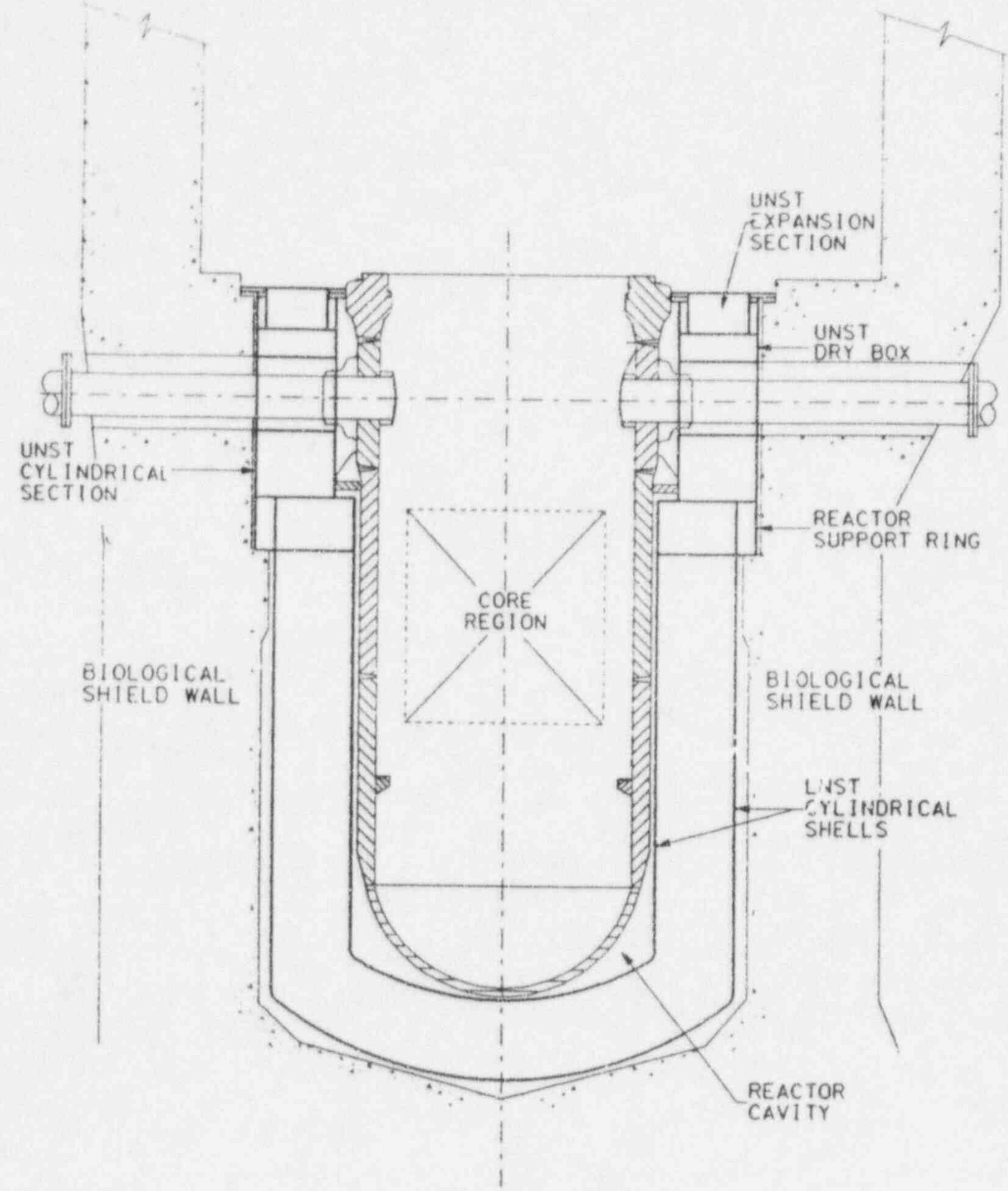
The RPV package will be transported to the waste disposal facility along a pre-defined route. The route may require the services of several rail companies. Transitions from one rail line to another will be planned prior to shipment. The transitions will be coordinated to insure proper turnover and to minimize the transition time.

7.5.3 Route Evaluation

Prior to rail shipment, each rail company involved in the package transport will evaluate the portion of the route within their jurisdiction. The route will be evaluated to insure adequate capacity of the railway and adequate clearances. The RPV package tie-down system will be inspected by rail car inspectors who are AAR rules qualified.

7.6 References

7.6.1 Chem-Nuclear Systems, Inc., Document No. SD-OP-003, "Process Control Program for Solidification of Stable Waste Forms," Revision 22.



UNST - UPPER NEUTRON SHIELD TANK
LNST - LOWER NEUTRON SHIELD TANK

TITLE: IN SITU RPV	
SCALE: NONE	FIGURE 7.1-1

e:/dgn/yrdecom/sar711.dgn

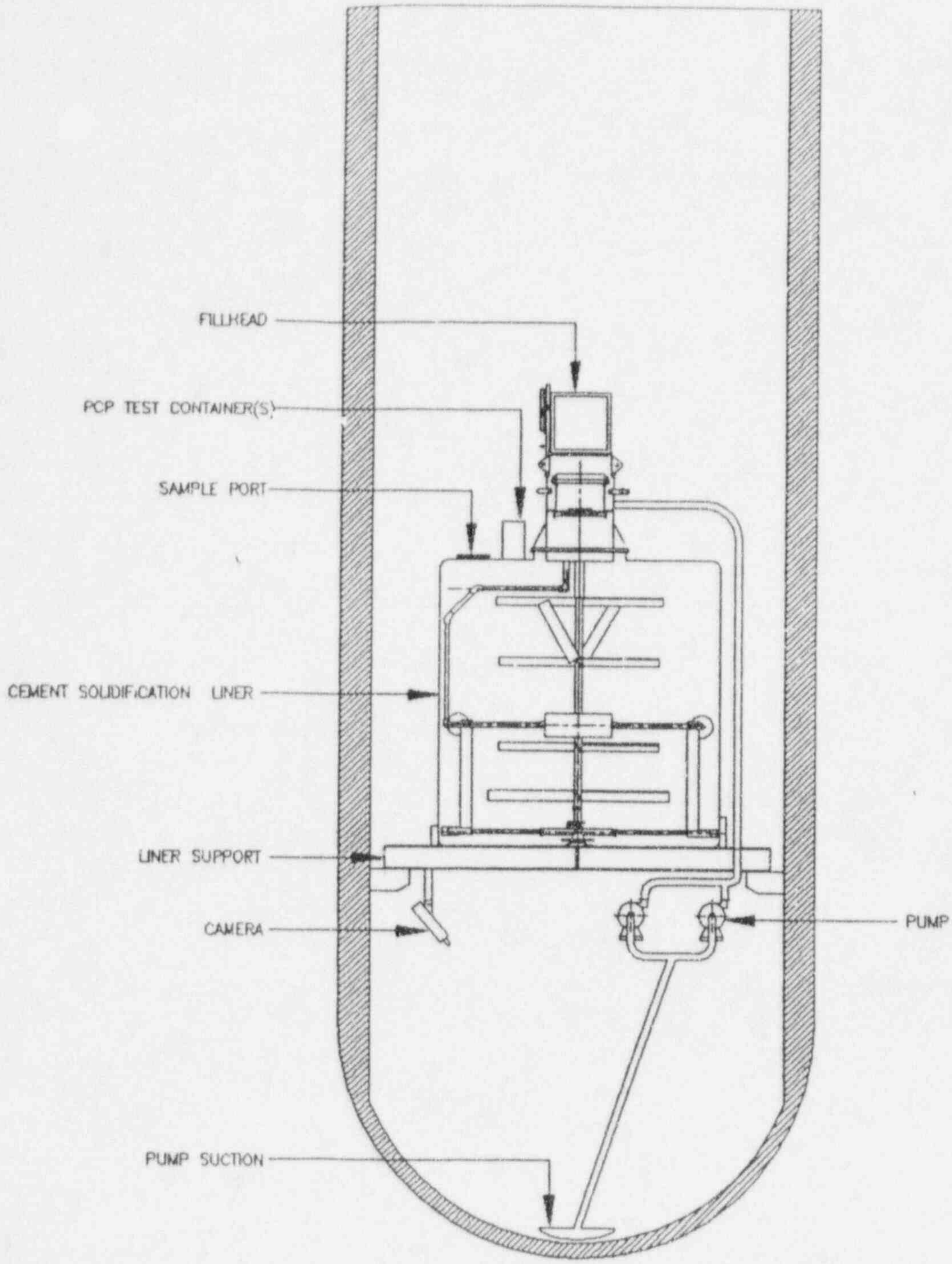
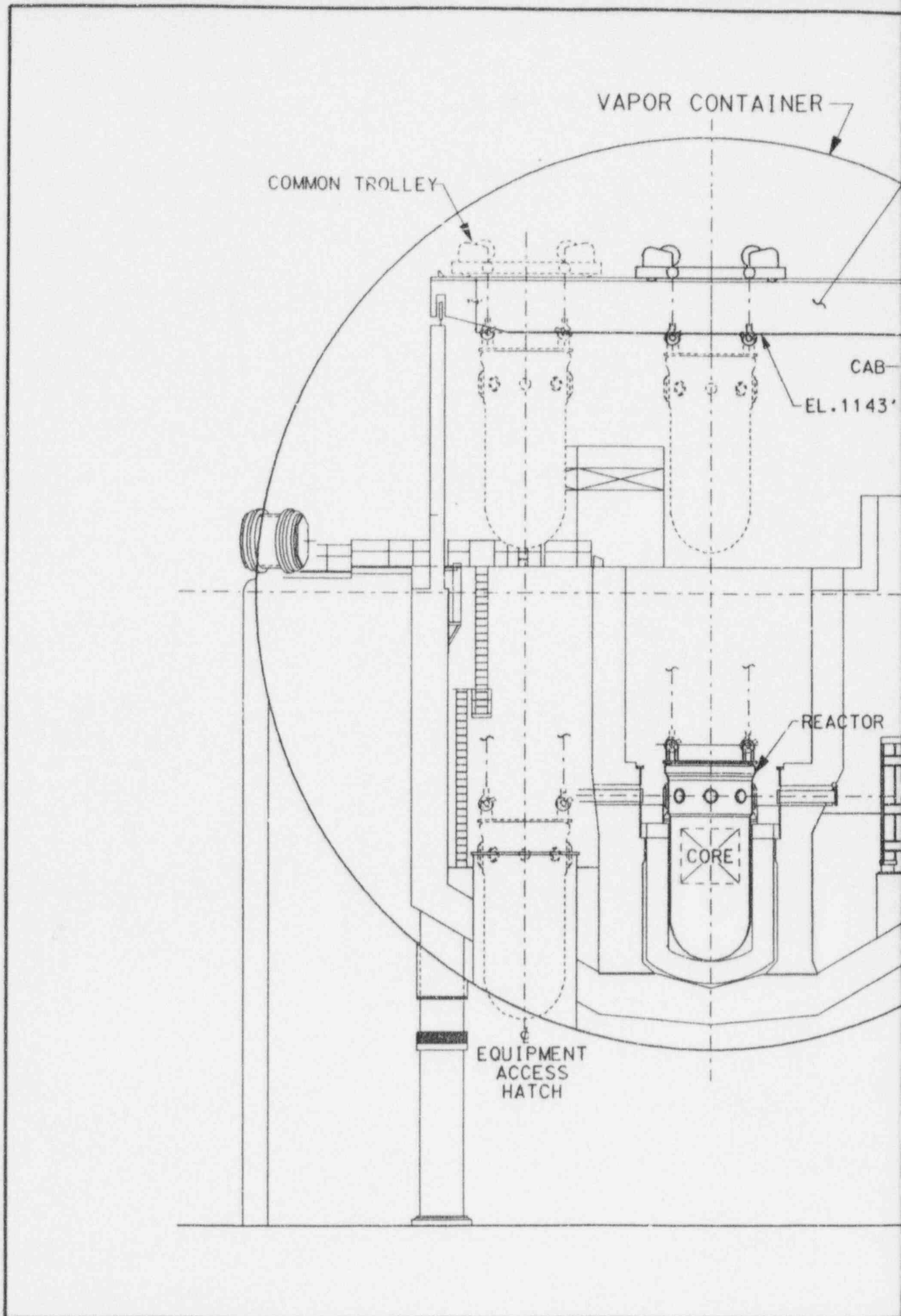
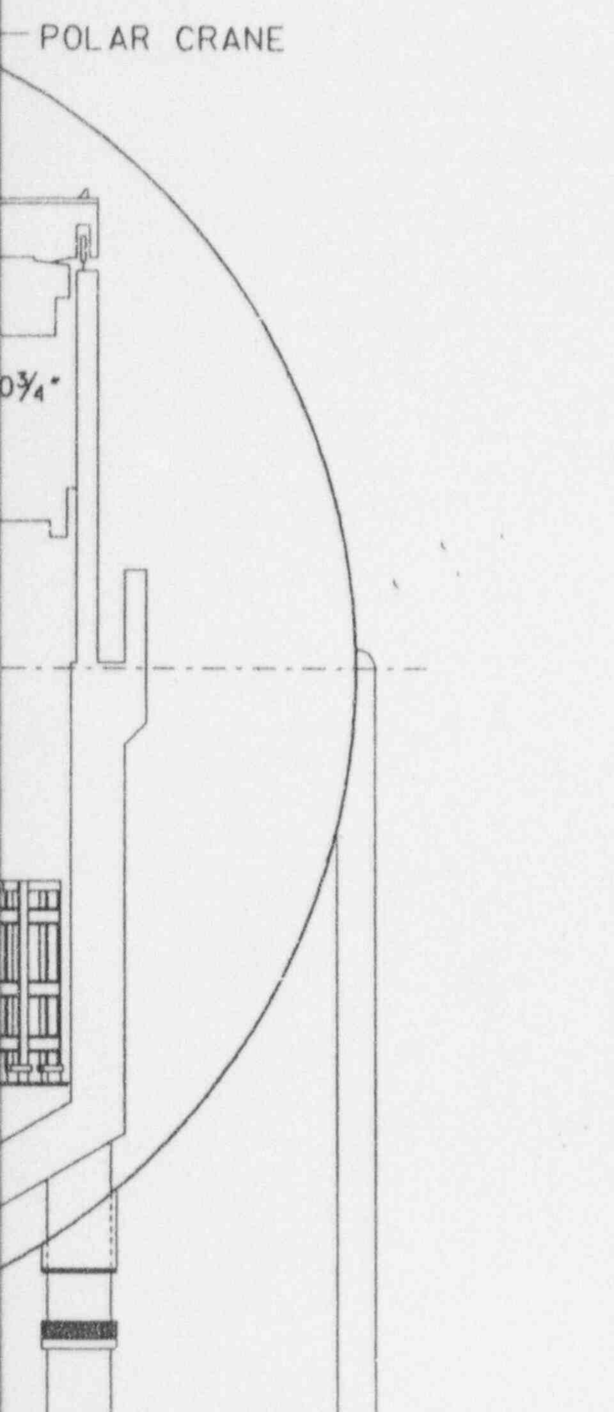


Figure 7.1-2

Conceptual Solidification Hardware


VAPOR CONTAINER
COMMON TROLLEY
CAB
EL.1143'
REACTOR
CORE
EQUIPMENT
ACCESS
HATCH



9611190240-09

TITLE:
RPV LIFT LOAD PATH

SCALE:
NONE

FIGURE 7.1-3

e:\dan\vrdecom\sar713.dan

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

This chapter describes the acceptance tests which will be performed prior to transport to insure that the YNPS RPV package meets the requirements of 10 CFR 71.85, "Preliminary Determinations" and 10 CFR 71.87, "Routine Determinations." The RPV package will not require a maintenance program since it is a single use package. If a waste disposal facility is not available at the time of package loading, a surveillance program described in this chapter will be implemented while the package is in storage. The surveillance program will assure that the effectiveness of the package is preserved during the storage period.

8.1 Acceptance Tests for Preliminary Determination

The acceptance tests described below provide assurance that the requirements of 10 CFR 71.85 are met.

8.1.1 Visual Inspection

10 CFR 71.85(a) requires that prior to the first use of the RPV package "the licensee shall ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects which could significantly reduce the effectiveness of the packaging." The RPV shipping cask will be shop fabricated and inspected in accordance with Section VIII, Division 2 of the ASME Boiler and Pressure Vessel Code (Reference 2.10.1). The acceptance criteria therein, including required non-destructive examinations (NDE), provide assurance that there will be no defects which could significantly reduce the effectiveness of the cask.

Additionally, the RPV package will be visually inspected prior to shipment, after all preparations have been completed. The visual inspection will be performed using written procedures. The results of the inspection will be appropriately documented. Any repairs resulting from the inspection will be performed in accordance with ASME Section VIII.

Seal welds will be inspected as required by ASME Section VIII. This inspection will insure that there are no cracks, pinholes, uncontrolled voids, or other defects which could significantly reduce the effectiveness of the packaging in these welds. The results of the inspection will be appropriately documented. Any repairs resulting from the inspection will be performed in accordance with ASME Section VIII.

8.1.2 Structural and Pressure Tests

NDE will performed on all shop and seal welds as required by Section VIII, Division 2 of the ASME Boiler and Pressure Vessel Code. Required NDE for these welds is shown on Figures 1.2-2 through 1.2-4. Acceptance criteria will be as required by ASME Section VIII,

Division 2. NDE of the top cover weld will be performed using ultrasonic and magnetic particle testing. The examinations and acceptance criteria for the top cover weld will be developed using ASME Section VIII as a guide.

Pressure testing is required by 10 CFR 71.85(b) if the maximum normal operating pressure will exceed 34.3 kilopascal (5 psi) gauge. The maximum normal operating pressure is defined as the difference between the maximum internal pressure during hot conditions (19.53 psia per Table 3.1.1) and normal atmospheric pressure (14.7 psia). This produces a maximum normal operating pressure of 4.83 psig. Therefore, pressure testing is not required.

8.1.3 Leak Tests

Leak testing requirements are stated in 10 CFR 71.51. As discussed in Section 1.2.3, the RPV package contains LSA material and will be transported as exclusive use. The package is exempted from the additional requirements of 10 CFR 71.51 by 10CFR52. Therefore, leak testing is not required.

8.1.4 Component Tests

There are no active components included as part of the RPV package. Therefore, component testing is not required.

8.1.5 Package Marking

10 CFR 71.85(c) requires that the package be conspicuously and durably marked with its model number, gross weight, and the package identification number assigned by the NRC. 10 CFR 71.85(c) also requires that, prior to applying the markings, the licensee shall determine that the package has been fabricated in accordance with the design approved by the NRC.

The RPV package will be marked as required by 10 CFR 71.85(c). The marking will be verified and documented prior to transport. The marking will include the following: Model No: YNPS RPV PACKAGE; Gross Weight: 328 Tons; NRC Identification No.____.

The fabrication and preparation of the RPV package will be performed in accordance with the YAEC 10 CFR 71, Subpart H Quality Assurance program. The fabrication and preparation of the package will be documented, and will be verified to insure that the requirements of the package Certificate of Compliance issued by the NRC are met. The requirements of 10 CFR 71.85(c) will be met.

8.2 Acceptance Tests for Routine Determination

Prior to shipment, the requirements for routine determination as given by 10 CFR 71.87 must be met. The tests that will be performed to meet those requirements are described below.

8.2.1 Proper Packaging

10 CFR 71.87(a) requires that the licensee determine that the package is proper for its contents. The RPV package is conservatively designed based on sampling and surveys of the RPV, and calculations. The design is discussed in Chapters 1 through 5. The package design is in accordance with the requirements of 10 CFR 71. Package fabrication and preparation will be performed under the YAEC 10 CFR 71, Subpart H, Quality Assurance Program. The package is therefore proper for its contents.

8.2.2 Package Physical Condition

10 CFR 71.87(b) requires that the licensee determine that the package is in unimpaired physical condition, except for superficial defects such as marks or dents. As discussed in Sections 8.1.1 and 8.1.2 above, the package will be subject to visual inspections and NDE. This will insure that the package is in unimpaired physical condition. Therefore, the package will meet the requirements of 10 CFR 71.87(b).

8.2.3 Closure Devices

10 CFR 71.87(c) requires that the licensee determine that each package closure device is properly installed and secured, and free from defects. The RPV package has no bolted or gasketed closures. The package top cover and vent and port covers will be inspected as described in Sections 8.1.1 and 8.1.2 above. These inspections will insure that package closures are properly installed and secured, and free from defects. Therefore, the package will meet the requirements of 10 CFR 71.87(c).

8.2.4 Liquid Containment

10 CFR 71.87(d) requires that the licensee determine that the package is adequately sealed and has adequate space or other specified provisions for expansion of contained liquid. The only liquid which would be present in the RPV package is water introduced during concrete placement. Free water will be minimized by venting the package during concreting operations and while the concrete cures, and ensuring that the concrete does not contain excess water. The concrete will be able to absorb the expansion of the minimal water which may exist in the package after sealing. The NDE and visual inspections will assure that the package and seal welds will adequately contain any free water. The package therefore will be adequately

sealed and have adequate space for expansion of contained liquid (water). Therefore, the package will meet the requirements of 10 CFR 71.87(d).

8.2.5 Pressure Relief Devices

10 CFR 71.87(e) requires that pressure relief devices are operable and set in accordance with written procedures. The RPV package has no pressure relief devices. Therefore, the requirements of 10 CFR 71.87(e) are not applicable.

8.2.6 Loading and Closure

10 CFR 71.87(f) requires that the package be loaded and closed in accordance with written procedures. The package will be loaded and closed as described in Sections 7.1 and 8.1. All operations associated with loading and closure will be performed as required by written procedures. Therefore, the package will meet the requirements of 10 CFR 71.87(f).

8.2.7 Fissile Material

10 CFR 71.87(g) requires that, if required for fissile material, any moderator or neutron absorber is present and in proper condition. There is no moderator or neutron absorber required for the RPV package. Therefore, the requirements of 10 CFR 71.87(g) are not applicable.

8.2.8 Lifting and Tie Devices

10 CFR 71.87(h) requires that any structural part of the package which could be used to lift or tie down the package during transport be rendered inoperable for that purpose unless it satisfies the design requirements of 10 CFR 71.45. This requirement is applicable to two parts of the package: the clevises welded to the top cover, and the trunnions attached to the shipping cask cylinder. These parts will be removed prior to shipment of the package. Therefore, the package will meet the requirements of 10 CFR 71.87(h).

8.2.9 External Removable Surface Contamination

10 CFR 71.87(i)(2) provides the acceptable levels of external removable surface contamination for exclusive use packages shipped by highway or rail. This requirement is therefore applicable to the RPV package. The shipping cask exterior will not be subject to a highly contaminated environment. Also, the shipping cask will be painted on all exterior surfaces to facilitate decontamination should it be necessary. Prior to shipment, surface contamination will be measured in accordance with 10 CFR 71.87(i)(1) as a minimum to insure that contamination requirements are met. The above actions will insure that the RPV package will meet the requirements of 10 CFR 71.87(i)(2).

8.2.10 External Radiation Levels

10 CFR 71.87(j) requires that the licensee determine that external radiation levels around the package and around the vehicle, if applicable, will not exceed the levels specified in 10 CFR 71.47 at any time during transport. Surveys of the package will be performed prior to shipment to ensure that radiation levels do not exceed the 10 CFR 71.47 levels. If the levels exceed those calculated in Chapter 5, the package will be reviewed to determine if action is required. Therefore, the package will meet the requirements of 10 CFR 71.87(j).

8.2.11 Accessible Surface Temperatures

10 CFR 71.87(k) requires that the licensee determine that accessible package surface temperatures will not exceed the limits specified in 10 CFR 71.43(g) at any time during transport. As discussed in Chapter 3, calculated accessible package surface temperatures do not exceed the limits specified in 10 CFR 71.43(g) when the package is subject to the conditions specified. Therefore, the package will meet the requirements of 10 CFR 71.87(k).

8.3 Maintenance (Surveillance) Program

The RPV package will be a single use package. As such, a maintenance program is not necessary to insure package performance for repeated uses.

The potential exists that the RPV package may be stored for a period of time prior to shipment. To insure the continued performance of the package, a surveillance program consisting of periodic visual inspections, contamination measurements, and radiation surveys will be implemented during the storage period. The inspections, measurements, and surveys will be performed using written procedures. Any repairs required as a result of the surveillances will be performed in accordance with ASME Section VIII. These actions insure that the effectiveness of the package is maintained during storage.

Additionally, the package will be tested as described in Sections 8.1.1 and 8.2 after removal from storage, and prior to shipment. The package will therefore, after removal from storage and preparation for transport, satisfy the requirements of the package Certificate of Compliance issued by the NRC.