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CHAPTER 5

SHIELDING EVALUATION

5.1 Discussion and Results

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

Geometric attenuation, enhanced by air and ground attenuation, provides additional dose reduction for distant locations at the restricted area and site boundaries. Figure 5.1-1 shows the configuration of shielding in the cask. Table 5.1-1 lists the compositions of the shielding materials.

In Chapter 2, Table 2.1-4 and Figure 2.1-4 provide separate qualification criteria for low burnup 7x7 fuel and for all other types of BWR fuel, which are acceptable at higher burnups. These fuel classes are specified in more detail in Section 5.2 and the qualification criteria are developed in Section 5.2.5 and Section 5.2.6.

The shielding evaluation presented in this chapter is performed utilizing as a design basis the source terms from the 8x8 fuel assembly with Zircaloy channel and 63 fuel rods. This fuel assembly is bounding because it contains the greatest mass of fuel among the candidate fuel assemblies in the high burnup category.

All shielding calculations are performed using the computer code MCNP[8] which is a 3-dimensional Monte Carlo method. Normal and off-normal conditions are modeled with the TN-68 intact. Average dose rates on the side, top and bottom of the TN-68 cask are calculated. Maximum dose rates are also calculated above and below the radial neutron shield.

Accident conditions assume radially the optional shield ring, the neutron shield and steel shell are removed and axially the polypropylene disk and protective cover are removed. This evaluation bounds the accident conditions in Section 11.2.

The expected dose rates (for normal, off-normal, and accident conditions) from the TN-68 cask are provided in Table 5.1-2. The dose point locations are illustrated in Figure 5.1-2.

The total (direct + skyshine) dose rates around the ISFSI site is presented in Table 5.1-3.

The dose rate around the cask at accessible locations to personnel are presented in Chapter 10. Occupational dose rates from cask operations are also presented in Chapter 10.

The effect of accidents on dose rates in the vicinity of the casks is discussed in Chapter 11.

5.2 Source Specification

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

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

The first three sources of radiation are evaluated using SAS2H/ORIGEN-S modules of the SCALE[1] code with the 44 group ENDF/B-V library. The capture gamma radiation and sub-critical multiplication are handled as part of the shielding analysis which is performed with MCNP4C.

The fuel assemblies acceptable for storage in the TN-68 are listed in Table 2.1-1. This listing of fuel assemblies was collapsed into seven basic designs provided below. The various fuel assembly designs were separated according to fuel assembly array, the maximum metric tons of uranium, and the number of water rods. These three parameters are the significant contributors to the SAS2/ORIGEN-S model. The largest uranium loading results in the largest source term at the design basis enrichment and burnup.

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

Table 5.2-1 provides additional fuel assembly design characteristics for the seven basic fuel designs.

The 8x8 fuel assembly with 63 fuel rods is the design basis fuel for shielding purposes because it has the highest initial heavy metal loading of the 8x8, 9x9, and 10x10 fuels, and therefore results in the highest radioactive source terms for a given irradiation history. Initial enrichment of 2.6 wt% U235, assembly average burnup of 48 GWd/MTU, and cooling time of 7 years complete the specification of the design basis 8x8 fuel. A conservative three-cycle irradiation at a constant specific power of 6 MW/assy is utilized with a 30 day down time between each cycle to calculate the source terms.

For the purpose of developing the 7x7 fuel qualification table, this fuel is specified at 3.3 wt% U235 assembly average enrichment, 40 GWd/MTU burnup, 10 years cooling, and irradiation at a constant power of 5 MW/assy over three cycles.

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

The masses for the materials in the top end fitting, the plenum, and the bottom fitting regions are multiplied by 0.1, 0.2 and 0.15, respectively[6]. These factors are used to correct for the spatial and spectral changes of the neutron flux outside of the fuel zone. The material compositions of the fuel assembly hardware are included in the SAS2H/ORIGEN-S model on a per assembly basis.

Axial variation in the moderator density along the BWR fuel assembly was considered by including a volume averaged density for the moderator around the fuel pins. The following axial variation of temperatures and moderator densities were used to calculated the volume average moderator density for use in the BWR source term models[3].

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

Gamma and neutron source terms are calculated for the 7x7 low burnup and the 8x8 design basis fuel designs. Table 5.2-4 presents the gamma and neutron source terms which confirm the selection of the 8x8 design basis fuel for the shielding analysis.

5.2.1 Axial Source Distribution

Axial source term peaking factors are determined based on typical axial burnup distributions for BWR asssemblies and based upon typical axial water density distribution that occurs during core operation. Using the base SAS2H/ORIGEN-S input for the 8x8 design basis fuel, neutron and

gamma source terms are generated for axial zones as a function of burnup and moderator density. This estimates both the non-linear behavior of the neutron source with burnup and the core operating moderator density effects on the actinide isotopics (neutron source).

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

This axial distribution is conservative at high burnup because the burnup distribution will flatten out with increased burnup resulting in a reduction in the overall peaking factor.

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

The peaking factors and water densities for the 25 axial locations were collapsed into 12 axial zones and utilized in determining the source terms and axial profiles of the sources for the shielding evaluation. The top and bottom 10% of the assembly was divided into two zones each and the middle 80% divided into 8 equal zones. The peaking factors ranged from 0.2357 and 0.2410 at the bottom and top respectively, to a maximum of 1.20 just below the middle. The water densities ranged from 0.3609 at the top zone to 0.7603 at the bottom. See Table 5.2-5.

The burnup and water density axial distribution data was utilized to prepare a 12 axial zone fuel assembly model. Twelve SAS2H calculations were performed for the design basis fuel with the power and water density being variables for each zone. The specific power input was the product of the nominal specific power, (5 MW) and the peaking factor. The water density was that value calculated for the zone as described above. Therefore, the fuel assembly was divided into 12 zones, with each zone having a unique gamma and neutron source term, specifically calculated for the burnup and water density in that zone. The gamma and neutron axial distributions are listed in Table 5.2-6 and shown in Figure 5.2-2.

5.2.2 Gamma Source

The gamma source terms for the 8x8 design basis fuel assembly are provided in Tables 5.2-7 and 5.2-8.

The gamma source spectra are presented in the 18-group structure consistent with the SCALE 27n-18γ cross section library. The conversion of the source spectra from the default ORIGEN-S energy grouping to the SCALE 27n-18γ energy grouping is performed directly through the ORIGEN-S code. The SAS2H/ORIGEN-S input file is provided in Section 5.5.1.

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

Based on the weight fraction of cobalt in each zone of the fuel assembly model (as adjusted by the appropriate flux ratio), the gamma source term in SCALE Energy Groups 36 and 37 are redistributed accordingly.

5.2.3 Neutron Source

Table 5.2-9 provides the total neutron source spectra for the 8x8 design basis fuel assembly. The SAS2H/ORIGEN-S code provides the neutron spectra in the SCALE 27n-18γ energy groups. Table 5.2-9 provides the total neutron source.

5.2.4 Fission Gas Release and Airborne Radioactive Material Sources

Table 5.2-4 presents the total kg moles of fission gas products in each fuel assembly for the 7x7 and 8x8 design basis fuel categories. The fuel rod pressures in the cladding analyses of Appendices 6A and 6B are derived from the fission gas inventory of the design basis 8x8 fuel applied to all fuel types, with the assumption that 30% of the fission gases are released from the fuel matrix.

Three variations on the irradiation history of the 8x8 fuel are evaluated to confirm that the specification of the design basis fuel for shielding also results in the bounding radiological source terms for confinement. The fuel assembly parameters are as follows:

Fuel Description	Burnup	Enrichment	Cooling Time
Design Basis Fuel **(DBF-68)**	48 GWd/MTU	2.6 wt % U235	7 Years
Medium Burnup Fuel **(MBF-68)**	55 GWd/MTU	2.5 wt % U235	10 Years
High Burnup Fuel **(HBF-68)**	60 GWd/MTU	3.2 wt % U235	12 Years

The results in Table 5.2-11 provide the confirmation sought.

Table 5.2-11 provides the design basis Fission Gas Inventory.

5.2.5 Fuel Qualification for 7x7 Fuel

The fuel qualification for the 7x7 fuel was performed with the SCALE 4.3[10] code package.

An evaluation was performed to determine the fuel assembly parameters of burnup, percent initial enrichment and cool time that would result in dose rates less than the current design basis fuel mentioned above and thus would be acceptable for storage in the TN-68 storage cask. The 1-dimensional SAS2H shielding model was used for this evaluation. A 1-D calculation determined the contact dose rate at the midplane on the side of the cask to be 67.9 mrem/hr (3.4 neutron and 64.5 gamma).

A series of SAS2H runs were performed for the design basis 7x7 fuel assembly with variations in the burnup and percent initial enrichment. Cool times were selected such that the total dose rate calculated was approximately 90% or less of the design basis fuel dose rate given above. Table 2.1-4 was generated showing the minimum cool time necessary for various combinations of burnup and bundle average enrichment based on these SAS2H evaluations. This table is used to evaluate burnups from 15,000 MWD/MTU to 40,000 MWD/MTU and enrichments from 1.0 wt% to 3.7 wt%.

Since the neutron dose is a relatively small fraction of the total dose, coupled with the relatively faster reduction of gamma dose with cool time, the minimum cool time is not required to increase significantly above 10 years even for low enrichment/high burnup fuel assemblies. The actual dose rates calculated in SAS2H ranged from a low of 32.5 mrem/hr for 20 GWD/MTU, 1.2 wt%, 10 years decay to 60.8 mrem/hr for 37 GWD/MTU, 1.8 wt%, 11 year decay. The majority of the calculated cases fell between 55 to 58 mrem/hr total dose rate.

Based on the calculated results (which are presented in bold type), other values may be interpolated/extrapolated based on simple logic, i.e. for the same burnup, an increase in enrichment produces a lower source. Some simple algorithms were used to interpolate/ extrapolate.

The dose rate increase can be predicted as a function of the enrichment by raising the ratio of the enrichments to the power of 1.3 for neutrons and 0.45 for gammas. For burnup, the dose rate increase can be predicted by taking the burnup ratio to the power of 4 for neutrons and 1.3 for gammas.

Using a combination of the 19 specific cases calculated using SAS2H and interpolation/extrapolation, Table 2.1-4 can be generated.

5.2.6 Fuel Qualification for 8x8, 9x9 and 10x10 Fuel

This section provides the basis for qualification of the design basis fuel to be loaded in the TN-68 cask from a shielding perspective. The objective is to demonstrate that the fuel assemblies with the parameters corresponding to design basis 8x8 fuel result in the highest calculated dose rate so that bounding shielding analysis can be performed by utilizing these design basis source terms. In order to determine the bounding spent fuel parameters (design basis fuel assembly), the candidate assembly parameters are ranked by their relative radiation source strengths. A simple 1-D shielding calculation based on the TN-68 cask is performed and the radiation dose rates at the cask side surface and 2m from the surface are determined. The spent fuel parameters that yield the highest total dose rate (gamma + neutron) are considered the design basis for shielding calculations.

The SAS2H module of the SCALE [(1)] computer code system is once again utilized to determine the TN-68 dose rates for fuel qualification purposes. The mathematical function developed to determine the heat generation from the spent fuel as a function of enrichment, burnup and cooling time is described in Appendix 4A. For an assembly thermal power of 441 watts, the

Fuel Qualification Table (FQT) for thermal source term purposes is shown in Table 5.2-12. This table shows the maximum burnup as a function of enrichment and cooling time such that the corresponding assembly thermal power is 441 watts.

In order to make a complete and thorough determination of the design basis spent fuel assembly, dose rates from the entries in Table 5.2-12 have to be determined. However, basic shielding analysis principles can be utilized to reduce the number of SAS2H cask analysis calculations. A total of 12 entries in Table 5.2-12 have been utilized to perform the surface dose rate comparison studies. The selection of these candidate assembly parameters is based on the following principles:

- The parameters that affect the dose rates in the increasing order of importance are burnup, enrichment and cooling time. This means that for a given heat load, the fuel assembly with the lowest enrichment and cooling time is expected to produce the largest dose rates.
- For the 2.4 wt % enrichment case, cooling times at 7, 8 and 9 years are analyzed.
- For the 7 years cooling time, the minimum and maximum burnup cases are analyzed.
- The maximum credited burnup (60 GWd/MTU) at the minimum possible cooling time is analyzed.
- Remaining cases that are analyzed are representative for each cooling time.
- Cases are also analyzed at a low enrichment of 1.0 wt % U235 to determine any other restrictions for loading lower enriched fuel.

The cases that are utilized for comparison are representative of all of the entries in Table 5.2-12 and the selection of the design basis from among these candidates is clear and unambiguous. These cases are shown as shaded entries in Table 5.2-12 and the corresponding case number in the subsequent SAS2H shielding analysis is shown in square brackets. The results of these comparison studies are shown in Table 5.2-13 for the cask surface dose rates and Table 5.2-14 for dose rates at 2m.

The results of the dose rate comparison studies from Table 5.2-13 and Table 5.2-14 indicate that the relative ranking in terms of total dose rate does not change with distance. Case ID 4 resulted in the highest dose rate followed by Case ID 1 and then by Case ID 2. As expected, the dose rate was highest at lower enrichment and cooling times. The burnup requirements to achieve the desired decay heat at lower enrichments can be somewhat unrealistic. For example, for 8 years cooled fuel at 1.0 wt % enrichment, the burnup value of 47.5 GWd/MTU is certainly unrealistic since such a burnup cannot be achieved. Similarly, Cases 4 and 9 are analyzed to determine the lower limit for bundle average enrichment. Therefore, it is necessary to restrict some of the spent fuel parameters so that the design basis fuel parameters for shielding that are chosen are realistic and bounding over all of the available BWR spent fuel assemblies.

The fuel parameters corresponding to Case ID 2 are chosen to be the design basis for shielding purposes. Since the dose rates for this case are lower than that for Case ID 1, the fuel assembly with parameters corresponding to Case ID 1 need to be excluded. The ratio of burnup to enrichment (*be-ratio*) is chosen to be the appropriate filter to exclude low enrichment cases at unreasonably high burnups. The *be-ratio* for Case ID 1 is ~ 19.75 and that for Case ID 2 is ~18.5.

Therefore, one of the restrictions for selection of spent fuel parameters is that the *be-ratio* shall not exceed 18.50 for enrichments below 2.6 wt % and decay times below 8 years. The dose rate results at a cooling time of 8 years (Case ID 4 through 8) indicate that the same *be-ratio* restriction need not be imposed at cooling times above 8 years. The results of Case ID 4 indicate that a practically unrestricting *be-ratio* can be imposed for cooling times greater than 8 years. This *be-ratio* is calculated, from the results of Case ID 5, to be 27.0 for enrichments below 1.8 wt % U235. Overall, these restrictions based on the *be-ratio* ensure that the design basis fuel assembly for shielding purposes is based on Case ID 2.

A conservative and practical *be-ratio* as a function of cooling time for enrichments below 2.6 wt % U235 is shown below

The maximum *be-ratio* = 6 x (INT(CT) – 4)
Where CT = Cooling Time (Years) and
INT(CT) = Cooling Time rounded down to the nearest integer value.

Examination of this function indicates that the *be-ratio* at cooling times below 8 years is 18.0 and the *be-ratio* at cooling times between 8 and 9 years is 24.0. These values are lower than those calculated by analysis (18.5 and 27 respectively) and therefore, the above representation of the *be-ratio* is conservative.

This logic is translated into the fuel qualification flowchart shown in Figure 2.1-4. A Fuel Qualification Table generated using that flow chart is shown in Table 5.2-15. The shaded cells in the table represent restrictions imposed due to either the assembly burnup being greater than 60 GWd/MTU or limits due to high burnup to enrichment ratio (*be-ratio*).

The SAS2H input file listing for this fuel qualification evaluation is provided in Section 5.5.1.

5.3 Model Specification

The neutron and gamma dose rates on the surface of the TN-68 cask (Side, Top and Bottom surfaces) and at 1m and 2m from the side surface of the cask are evaluated with the 3-D Monte Carlo transport code MCNP4C2 [8]. The flux-to-dose conversion factors specified by ANSI/ANS 6.1.1-1977 [5] are used and provided in Table 5.3-1.

5.3.1 Description of the Radial and Axial Shielding Configurations

Figure 5.3-1 is a plot of the TN-68 cask MCNP model and shows the axial cross section of the cask. This shielding configuration is utilized to determine the radial (side surface) and axial (top and bottom surface) dose rates around the TN-68 cask for normal, off-normal and accident conditions. This configuration is also utilized to determine the dose rates at long distances from the cask.

Section 5.3.1.1 describes the 3D MCNP model developed for the TN-68 cask under normal and off-normal conditions. Section 5.3.1.2 describes the MCNP model developed for accident conditions. Section 5.3.1.3 describes the MCNP model developed for the TN-68 at far distances from the cask.

5.3.1.1 Shielding Configuration under Normal and Off-Normal Conditions of Storage

A single shielding configuration is utilized for the TN-68 design for both normal and off-normal conditions of storage. The cask bottom dose rates are determined as a consequence of off-normal conditions since these become important only during loading and transfer. During normal conditions of storage the cask is upright and is firmly seated on the concrete pad. A three-dimensional MCNP model discrete fuel assembly model of the TN 68 cask was developed for this purpose. In the MCNP model, the TN 68 cask axis is modeled along the Z-direction. The X and Y axes in the MCNP model represent the cask in the radial direction. The cask is assumed to sit on a concrete pad (-Z direction) that extends out to 2 meters from the cask surface. A 3-D MCNP model with homogenized basket has been utilized to determine dose rates at long distances for lower burnup fuel and is described below.

The MCNP model for these shielding configurations is based on a discrete basket with the homogenized fuel assemblies (with an active height of 146 inches) positioned within fuel compartments. The MCNP model developed in this calculation is essentially based on the design details from the TN 68 drawings, shown in Chapter 1, except for some conservative modifications. The MCNP model developed for these configurations differ from the drawings only in the representation of the basket and the hold-down ring. The cells 500 through 691 represent the discrete basket and associated modeling that is considered a modeling enhancement as opposed to representing the fuel and basket as a single homogenized region.

Figure 5.3-1 is an X-Z plot of the MCNP model of the TN-68 cask (axial section view). All the major details of the cask model are shown in this figure. The basket extends to about 164″ in the axial direction from the bottom of the bottom fitting (-204.19 cm) to midway through the plenum

region (212.37 cm). The active fuel zone is 146″ long with the center at 0 (± 185.42 cm). A simple analog model was used for calculating the neutron dose. For the primary gamma dose rates, a multiple cell sub-layer model was used. The cask trunnions and the resin cutouts are modeled explicitly.

The basket is modeled discretely using the advanced geometry features of MCNP. The fuel is modeled as a cuboid based on a 5.518″ square that includes the Zircaloy channel. The fuel compartment inside dimension is 6.00″ and is modeled with stainless steel with a thickness of 0.187″ surrounded by a 0.15″-thick basket aluminum plates. The borated aluminum (or any other poison material) plates were modeled as pure aluminum and the stainless steel strips (tie plates) were not modeled. The stainless steel and aluminum peripheral rails were modeled explicitly with some conservative simplifications. These are listed in the assumptions section of this report (Section 5.4.7). A small air gap of 0.138″ (0.35 cm) was assumed between the basket and the cask shell. Figure 5.3-2 and Figure 5.3-3 show a radial cross section plot (X-Y plot) of the MCNP model at Z=0 (center of the active fuel zone).

Above the basket, the stainless steel fuel compartments and aluminum plates surrounding the fuel assembly are replaced by air. Due to modeling constraints necessitated by the usage of advanced MCNP geometry to model the basket, the hold down ring is represented in the model in a simplistic fashion. The actual hold down ring is a 69″ diameter, 1.0″ thick cylindrical ring with 6 “basket-like” horizontal and vertical stainless steel plates (0.375″ thick) forming the basket-like (gridded) internal cross section. The hold down ring has a cut-out thickness of 0.44″ at four locations of the ring. The design drawings in Chapter 1 provide a better understanding of the hold down ring. A simplified, yet conservative version of the hold down ring is modeled as a stainless steel cylinder with a maximum thickness of 1.15″ with four cut-out regions with a thickness of about 0.58″. The shielding effect of the basket-like (0.375″ thick) horizontal and vertical plates is included by increasing the thickness of the hold-down ring by 0.15″. Note that the increase in thickness of 0.15″ is 40% of the thickness of the plates of the hold-down ring. Figure 5.3-4 and Figure 5.3-5 show the radial cross section of the MCNP model at Z=220 cm. The simplified hold-down ring and the top trunnions are shown in these figures.

Two MCNP models are developed for determining the normal and off-normal dose rates. The gamma model containing a detailed segmentation of the thicker cask steel body is utilized to calculate the primary gamma dose rates. The neutron model is utilized to calculate the neutron and secondary gamma dose rates. These dose rates are calculated without the presence of the optional 1″ auxiliary shield ring.

The tallies are based on F2 detectors which provide the average flux across a surface. Surfaces 62, 63 and 64 are the radial tally surfaces that represent the cask outer surface, 1 m from cask outer surface and 2m from cask outer surface. Similarly, the cask top and bottom doses are measured on surfaces 81, 82 and 83 that are segmented surfaces 25 cm, 50 cm and 75 cm from the cask center.

5.3.1.2 Shielding Configuration under Accident Conditions of Storage

The MCNP design basis model for accident conditions is almost identical to that of the normal (and off-normal) conditions except that all neutron shielding and the stainless steel skin shell materials are replaced with void. The accident configuration is also modeled without the auxiliary shield ring. This scenario is based on a fire accident that results in the complete removal of the neutron shielding materials including the polypropylene disk at the top. This is implemented in MCNP by replacing the materials in cells 86, 87, 83, 22, 211, 20, 21, 403, 404, 405, 406, 411, 412, 423, 424, 425, 426, 431 and 432 with void.

The tally surfaces and detectors for accident dose rates are identical to those for normal and off-normal dose rates.

5.3.1.3 Shielding Configuration at Long Distances from the Cask

The near field dose rate calculational MCNP model described in Section 5.3.1.1 is modified to determine the long distance dose rates. The near-field MCNP model is extended further to include additional volumetric detectors beyond the immediate vicinity of the cask (about 1000 m from the cask surface) to obtain dose rates at long distances. The MCNP cell # 900-903, 1002-1010, 1101-1110 are the additional cells that describe the geometry and materials beyond the immediate vicinity of the cask. The 1″ auxiliary shield ring is included in the MCNP model. The MCNP calculated dose rates at far distances consist of contributions from direct, air scatter (skyshine) and limited ground scatter (only in the immediate vicinity of the cask). Concrete is modeled as the ground surface around a 7 foot by 7 foot area (about 1 m from the cask surface). Beyond that concrete pad, the soil is assumed to be air. It is expected that the maximum contribution to the ground scatter component occurs at the immediate vicinity of the cask. The dose is calculated as F4 tallies (that calculate the volumetric dose rates) in an annular cylindrical detector, 30 cm thick and 30 cm high. The dose rates are calculated at distances ranging from 10m to 600m from the edge of the concrete pad.

5.3.2 Shield Regional Densities

Table 5.3-2 shows the fuel assembly material composition for the four fuel assembly regions. Based on these material compositions, material densities and number densities for the fuel assembly regions are determined and provided in Table 5.3-3 for unchanneled fuel and in Table 5.3-4 for channeled fuel. The mass of materials in each fuel assembly region is homogenized over the volume of the region (area = 30.45 in^2, based on a 5.518″ x 5.518″ cross-section). Table 5.3-5 provides the shield regional densities for the TN-68 Cask. The actual fuel layout in the TN-68 Cask is an array of fuel assemblies inside stainless steel compartments surrounded by sheets of aluminum material.

The radial resin and aluminum boxes are homogenized into a single composition based on the mass of each component. Dose measurements from similar designs in use have not shown dose streaming effects due to the aluminum boxes.

5.4 Shielding Evaluation

5.4.1 Computer Programs

MCNP[8] is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces. Point-wise (continuous energy) cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation are accounted for in the cross section set. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. Important standard features that make MCNP very versatile and easy to use include a powerful general source; an extensive collection of cross-section data; and an extensive collection of variance reduction techniques that can be employed to track particles through very complex deep penetration problems.

5.4.2 Spatial Source Distribution

The source components are:

- A neutron source due to the active fuel regions of the 68 fuel assemblies,
- A gamma source due to the active fuel regions of the 68 fuel assemblies,
- A gamma source due to the plenum regions of the 68 fuel assemblies,
- A gamma source due to the top nozzle regions of the 68 fuel assemblies, and
- A gamma source due to the bottom nozzle region of the 68 fuel assemblies.

Axial burnup peaking factors for the BWR fuel utilized in this analysis is shown in Table 5.2-5. The gamma and neutron peaking factors and axial power distribution (source distribution) factors shown in Table 5.2-5 are obtained at lower burnups and are conservatively utilized for high burnup fuel. These factors are directly applied to each axial interval in the fuel region in the MCNP model. The average values of the axial peaking distributions are also provided in Table 5.2-5. For the gamma distribution, the average value is around 1.0. However, for the neutron distribution, the average value of the distribution is greater than 1.0. The average value of the axial neutron distribution may be interpreted as the ratio of the true total neutron source in an assembly to the neutron source calculated by SAS2H/ORIGEN-S for an average assembly burnup. Therefore, to properly correct the magnitude of the neutron source, the neutron source per assembly as reported in Table 5.2-10 is multiplied by the average value of the neutron source distribution as reported in Table 5.2-5.

5.4.3 Cross-Section Data

The cross-section data used is the continuous energy ENDF/B provided with the MCNP code. The cross-section data allows coupled neutron/gamma-ray dose rate evaluation to be made to account for secondary gamma radiation (n,γ), if desired. All of the near field dose rate

calculations account for the dose rate due to secondary gamma radiation. For the far field (long distance) dose rate calculation, the dose rate contribution from the secondary gamma radiation is ignored because it is insignificant.

5.4.4 Flux-to-Dose-Rate Conversion

The flux distribution calculated by the MCNP code is converted to dose rates using flux-to-dose rate conversion factors from ANSI/ANS-6.1.1-1977 [5] given in Table 5.3-1.

5.4.5 Model Geometry

Figures 5.3-1 through 5.3-5 are the MCNP models for the TN-68 cask. The model geometry is discussed in Section 5.3.1.

5.4.6 Methodology

The methodology used in the shielding analysis of the TN-68 cask is the 3-D MCNP code. MCNP allows for explicit 3-D modeling of any shielding configuration and reduces the number of approximations needed. The methodology used herein is summarized below.

1. Sources are developed for all fuel regions using the source term data from Section 5.2. Source regions include the active fuel region, bottom end fitting (including all materials below the active fuel region), plenum, and top end fitting (including all materials above the active fuel region).
2. Suitable shielding material densities are calculated for all regions modeled.
3. The 3-D Monte Carlo transport code MCNP is used to calculate dose rates on and around the TN-68 cask. The MCNP4C2 code is selected because of its ability to handle thick, multi-layered shields as in the TN-68 cask body using 3-D geometry. MCNP4C2 results are used to calculate offsite exposures (Chapter 10).
4. For the cask, importance biasing is utilized for variance reduction for both the near and far field dose rate calculation. F2 tallies are utilized for determining near field dose rates while F4 tallies are utilized for determining the far field dose rates.
5. MCNP models are also generated to determine the effects of accident scenarios, such as loss of cask neutron shield.

5.4.7 Assumptions

The following general assumptions are used in the analyses.

5.4.7.1 Source Term and Material Composition Assumptions

- The BWR heavy metal weight is assumed to be 0.188 MTU per assembly to bound the candidate BWR fuel designs.
- The Zircaloy material is conservatively modeled as zirconium.
- The main constituents of the fuel are uranium dioxide, zirconium, stainless and carbon

steel. Inconel is modeled as stainless steel.

5.4.7.2 MCNP Cask Dose Rate Analysis Assumptions (Near Field Dose Rates)

- The fuel assembly is homogenized assuming no blankets, burnable absorbers or fission product poisons. The fuel assembly is homogenized without the presence of the channel and related hardware. Such a homogenization is conservative since the source terms were determined with a 0.100″ Zircaloy channel and about 0.7 kg of stainless steel hardware.
- The fixed poison plates and the stainless steel tie plates in the basket are modeled as aluminum.
- The 0.19″ peripheral aluminum plates are modeled as 0.15″ aluminum plates.
- The attachment access holes and the various weld studs on the aluminum rails are not modeled.
- The hold down ring is modeled as a segmented cylinder with a thickness of 1.15″ (instead of 1.00″) in the thicker portion and a thickness of 0.59″ (instead of 0.44″ + thickness of box for damaged fuel) in the cut-out portion. This representation is conservative because the increase in the ring thickness in the model tends to offset only 40% of the plate thickness in the hold down ring that was not modeled.
- All the accident conditions assume that the auxiliary shield ring, when present, is detached from the cask body.
- The design basis fire accident results in the complete removal of all neutron shielding materials present in the cask and also the neutron shield skin and the top cover. This is a conservative assumption.

5.4.7.3 MCNP Cask Dose Rate Analysis Assumptions (Far Field Dose Rates)

- The 1″ thick auxiliary shield ring is included in the MCNP model.
- The far field computational models assume that the fuel assembly homogenization includes a 0.065″ Zircaloy channel. A channel height of 167.2″ is utilized in the homogenization instead of 167″. However, a channel thickness of 0.065″ is utilized in the homogenization instead of an average value of 0.100″. Due to the nature of the calculation and the uncertainty in the results over large distances, it is expected that there would be no significant difference between the channeled and un-channeled assembly results and that these results could be applied to all BWR fuel assemblies.
- The doses due to capture gamma sources are not calculated since they are insignificant at large distances in comparison to primary gamma and neutron sources.
- The “ground shine” contribution is not calculated since soil is not modeled explicitly beyond the concrete pad of the ISFSI. The ground shine component is that portion of the dose rate that is due to reflection/scatter from the ground. The ground shine is significant only at the immediate vicinity of the cask (source) and is relatively insignificant at far distances.
- The location of the F5 detectors for the ISFSI array MCNP model inherently assume that there is no effect due to the orientation of the casks on the dose rates especially at far distances.

5.4.8 Near Field Dose Rate Calculations

MCNP models are developed to determine the near field dose rates for normal and off-normal conditions as discussed in Section 5.3.1.1 and for accident conditions as discussed in Section 5.3.1.2. The results of the near field dose rate calculations for normal and off-normal conditions are shown in Table 5.4-1 (without the auxiliary shield ring) and Table 5.4-2 (with the 1″ auxiliary shield ring. These results are also shown in Figure 5.4-1 through Figure 5.4-6. The MCNP input file listing for neutron and gamma are shown in Section 5.5.1. Additional cask surface dose rate results above and below the neutron shield for normal and off-normal conditions are shown in Table 5.4-3.

The results of the near field dose rate calculations for accident conditions are shown in Table 5.4-4. These results are also shown in Figure 5.4-7 through Figure 5.4-10.

5.4.9 Far Field Dose Rate Calculations

MCNP models (with the 1″ auxiliary shield ring) are developed to determine the far field dose rates as discussed in Section 5.3.1.3. The results of the far field dose rate calculations are shown in Table 5.4-5. These results are also shown in Figure 5.4-11. The MCNP input file listing for neutron is shown in Section 5.5.1.

5.5 Supplemental Data

5.5.1 Sample Input Files

This Section provides representative input files for the computer codes, SAS2H and MCNP, utilized to calculate the source terms and dose rates for the TN-68 Cask. These input files are provided as part of a separate proprietary compact disc. A listing of the contents of the compact disc and a brief description of the input files utilized as part of the shielding analysis are given below.

File Name	**Description**
d26b48c07.inp	SAS2H Input File for Design Basis Fuel Utilized in FQT Calculation
design_basis.inp	SAS2H Input File for Design Basis Fuel (DBF-68 for Source Term)
n0b0g	MCNP Input for Primary Gamma Dose (Near Field, No Auxiliary Shield Ring)
n0b0ng	MCNP Input for Neutron and Secondary Gamma Dose (Near Field, Without the Auxiliary Shield Ring)
tdn4ff	MCNP Input for Neutron Dose (Far Field, With the 1″ Auxiliary Shield Ring)

5.6 References

1. SCALE 4.4, “Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers,” CCC-545, ORNL

2. GE Proprietary Fuel Assembly Hardware Data.

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

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

5. ANSI/ANS 6.1-1-1977, "Neutron and Gamma Ray Flux-to-Dose Rate Factors"

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

7. “Radiation Shielding,” J. Kenneth Shultis and Richard E. Faw, Pretence Hall, 1996

8. MCNP4C2, “Monte Carlo N-Particle Transport Code System,” Los Alamos National Laboratory, CCC-701, RSIC, June 2001

9. Ludwig, S.B., and J.P. Renier, “Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code,” ORNL/TM-11018, Oak Ridge National Laboratory, December 1989

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

Table 5.1-1
TN-68 Cask Shield Materials

Component	Material	Density (g/cm^3)	Thickness (inches)
Cask Body Wall	Carbon Steel	7.82	7.50
Lid	Carbon Steel	7.82	9.50
Bottom	Carbon Steel	7.82	9.75
Resin [a]	Polyester Resin Styrene Aluminum Hydrate Zinc Borate	1.58	6.00
Aluminum Box	Aluminum	2.7	0.12
Outer Shell	Carbon Steel	7.82	0.75
Basket [b]	Stainless Steel (Inserts) Stainless Steel (Strips) Aluminum Neutron Poison Material [c]	7.92 7.92 2.7	0.19 0.31 0.31
Protective Cover	Carbon Steel	7.82	0.25
Polypropylene Disc	Polypropylene	0.90	4.00
Basket Hold Down Ring	Carbon Steel	7.82	1.0
Shield Ring (Optional)	Carbon Steel	7.82	1.0

Notes:

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

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

[c] See Chapter 9 for description of neutron poison material. This is modeled as aluminum for shielding purposes.

Table 5.1-2
Summary of Average Dose Rates

	Average Dose Rates (mrem/hour) for Normal and Off-Normal Conditions								
	No Shield Ring	With Shield Ring							
	Side Above Shield[1]	Side Above Shield[2]	Side Along Shield	Side Below Shield[3]	Top Surface	Side Surface	Bottom Surface	1m from Side	2m from Side
Gamma	1460	500	98.0	360	155	130	236	56.8	34.4
Neutron	175	104	20.6	275	5.44	31.4	519	11.7	6.70
Total	1630	604	119	635	161	162	755	68.5	41.1
	Average Dose Rates (mrem/hour) for Accident Conditions[4]								
Gamma	-	-	774	-	302	724	238	355	222
Neutron	-	-	2030	-	59.2	1560	527	730	430
Total	-	-	2810	-	362	2280	765	1090	652

1) Maximum surface dose rates above the neutron shield are 1855 mrem/hour (gamma) and 288 mrem/hour (neutron).
2) Maximum surface dose rates above the neutron shield are 620 mrem/hour (gamma) and 140 mrem/hour (neutron).
3) Maximum surface dose rates below the neutron shield are 866 mrem/hour (gamma) and 445 mrem/hour (neutron).
4) The 1″ optional shield ring, neutron shield and skin shell are removed from the sides. The polypropylene disk and protective cover are removed from the top.

Table 5.1-3
Dose Rates at Postulated Site Boundary from One Cask

	Farfield Dose Rates[1] (mrem/hour)		
Distance (meters)	**Gamma Dose Rate**	**Neutron Dose Rate**	**Total Dose Rate**
10	3.67E+00	7.14E-01	4.38E+00
20	1.21E+00	2.23E-01	1.43E+00
40	3.07E-01	5.27E-02	3.60E-01
60	1.34E-01	2.36E-02	1.57E-01
80	7.12E-02	1.16E-02	8.28E-02
100	4.06E-02	6.74E-03	4.73E-02
200	5.30E-03	8.88E-04	6.19E-03
400	4.42E-04	5.62E-05	4.98E-04
600	6.50E-05	6.79E-06	7.18E-05

1 Doses are calculated with the optional shield ring in place.

Table 5.2-1
BWR Fuel Assembly Design Characteristics

Transnuclear, ID	7 x 7 - 49/0	8 x 8 - 63/1	8 x 8 - 62/2	8 x 8 - 60/4	8 x 8 - 60/1	9 x 9 - 74/2	10x10- 92/2
GE Designations	GE2 GE3	GE4	GE-5 GE-Pres GE-Barrier GE8 Type I	GE8 Type II	GE9 GE10	GE11 GE13	GE12
Max Length (in)[a]	176.2	176.2	176.2	176.2	176.2	176.2	176.2
Max Width (in)[a]	5.44	5.44	5.44	5.44	5.44	5.44	5.44
Rod Pitch (in)	0.738	0.640	0.640	0.640	0.640	0.566	0.510
No of Fueled Rods	49	63	62	60	60	66 full 8 partial	78 full 14 partial
Maximum Active Fuel Length (in)	144	146	150	150	150	146" full 90" partial	150" full 93" partial
Fuel Rod OD (in)	0.563	0.493	0.483	0.483	0.483	0.440	0.404
Clad Thickness (in)	0.032	0.034	0.032	0.032	0.032	0.028	0.026
Fuel Pellet OD (in)	0.487	0.416	0.410	0.410	0.411	0.376	0.345
No of Water Rods	0	1	2	4	1	2	2
Water Rod OD (in)	---	0.493	0.591	2 @ 0.591 2 @ 0.483	1.340	0.980	0.980
Water Rod ID (in)	---	0.425	0.531	2 @ 0.531 2 @ 0.419	1.260	0.920	0.920
Maximum MTU/assembly[b]	0.1977	0.1880	0.1886	0.1825	0.1834	0.1766	0.1867
Minimum Plenum Volume (in^3)	2.066	1.595	1.273	1.273	1.291	1.184	0.995
Fill Gas	He	He	He	He	He	He	He
Maximum Initial Rod Pressurization (psia)	24.7	24.7	80	80	80	155	155

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

Table 5.2-1a
BWR Fuel Assembly Design Characteristics
(continued)

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

Table 5.2-2
BWR Fuel Assembly Hardware Characteristics [a]

Item	Material	Average Mass (kg/assembly)
Fuel Zone		
Cladding	Zircaloy	49.2
Spacers	Zircaloy	1.95
Spacer Springs	Inconel	0.36
Fuel-Gas Plenum Zone		
Cladding	Zircaloy	4.89
Springs	Stainless Steel	1.05
Top End Fitting Zone		
Upper Tie Plate	Stainless Steel	2.08
Lock Tab Washers & Nuts	Stainless Steel	0.05
Expansion Springs	Inconel	0.43
End Plugs	Zircaloy	1.26
Bottom End Fitting Zone		
Finger Springs	Inconel	0.05
End Plugs	Zircaloy	1.26
Lower Tie Plate	Stainless Steel	4.70
Channel		
Channel Sleeve	Zircaloy	37.1
Channel Spacer & Rivet[b]	Stainless Steel	0.13
Channel Fastener[b]		
Guard	Stainless Steel	0.46
Spring & Bolt	Inconel	0.13
Total		105

[a] Data from Reference 2.
[b] The channel spacer, rivet and fastener are located at top end fitting zone.

Table 5.2-3
Material Compositions for Fuel Assembly Hardware Materials

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

[a] Material compositions are taken from the SCALE Standard Composition Library, however, cobalt impurities are taken from Reference 3.

Table 5.2-4
BWR Fuel Assembly Source (with Channels)

Total Gamma Source (γ/sec/assembly)

TN Design ID	Total
7x7-49-0	1.39E15
8x8-63-1	3.60E15

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

TN Design ID	Total
7x7-49-0	8.98E07
8x8-63-1	3.17E08

Total Moles of Fission Gas Products (kg mole/assembly)[a]

7x7-49-0	1.084E-02
8x8-63-1	1.567E-02

[a] Contributions primarily from the isotopes of I, Kr and Xe, including stable isotopes

7x-749-0: 40 GWd/MTU, 3.3 % enrichment, 10 year cooled

8x8-63-1: 48 GWd/MTU, 2.6% enrichment, 7 year cooled

Table 5.2-5
Peaking Factor and Water Density Input
for Determination of Axial Source Distribution

Zone	Fraction of Core Height	Peaking Factor	Water Density (g/cm^3)
12	0.95-1.0	0.2410	0.3609
11	0.90-0.95	0.6330	0.3631
10	0.8-0.9	0.8973	0.3701
9	0.7-0.8	1.0766	0.3861
8	0.6-0.7	1.1515	0.4118
7	0.5-0.6	1.1912	0.4375
6	0.4-0.5	1.2000	0.4708
5	0.3-0.4	1.2000	0.5251
4	0.2-0.3	1.1836	0.5945
3	0.1-0.2	1.0750	0.7008
2	0.05-0.1	0.7746	0.7541
1	0.0-0.05	0.2357	0.7603

Table 5.2-6
Gamma and Neutron Axial Distributions

Axial Height (cm)	Location (% of Core Height)	Design Basis Axial Distribution	
		Gamma Distribution	Neutron Distribution
-185.42	Bottom	0.0000	0.0000
-176.15	0.00 to 0.05	0.2256	0.0018
-157.61	0.05 to 0.10	0.7674	0.1683
-129.79	0.10 to 0.20	1.0854	0.8447
-92.71	0.20 to 0.30	1.2027	1.3859
-55.63	0.30 to 0.40	1.2223	1.5288
-18.54	0.40 to 0.50	1.2244	1.5775
18.54	0.50 to 0.60	1.2164	1.5624
55.63	0.60 to 0.70	1.1227	1.3842
92.71	0.70 to 0.80	1.0964	1.0707
129.79	0.80 to 0.90	0.9053	0.5047
157.61	0.90 to 0.95	0.6255	0.1093
176.15	0.95 to 1.00	0.2303	0.0028
185.42	Top	0.0000	0.0000
Axial Peaking Factor[1]		**0.9890**	**1.3260**

1) The axial peaking factor for gamma is 0.989 and the factor utilized in the shielding analysis is 1.00. The axial peaking factor for neutron is 1.326 and the factor utilized in the shielding analysis is 1.330.

Table 5.2-**7**
Gamma Source Terms for 8x8 Design Basis Fuel
(γ/s/assembly)

Energy (MeV)		**Energy (MeV)**	**Fission Products**	**Actinides**	**Light Elements**	**Total**
1.00E-02	to	5.00E-02	4.41E+14	7.66E+12	3.00E+12	4.52E+14
5.00E-02	to	1.00E-01	1.18E+14	3.37E+12	9.81E+10	1.21E+14
1.00E-01	to	2.00E-01	9.24E+13	3.17E+11	2.87E+11	9.30E+13
2.00E-01	to	3.00E-01	2.65E+13	2.28E+11	1.69E+10	2.67E+13
3.00E-01	to	4.00E-01	1.75E+13	1.86E+10	6.45E+10	1.76E+13
4.00E-01	to	6.00E-01	1.90E+14	3.71E+08	1.41E+12	1.92E+14
6.00E-01	to	8.00E-01	9.31E+14	7.15E+08	7.28E+11	9.32E+14
8.00E-01	to	1.00E+00	8.96E+13	1.06E+08	1.37E+10	8.96E+13
1.00E+00	to	1.33E+00	2.74E+13	3.06E+08	2.37E+13	5.11E+13
1.33E+00	to	1.66E+00	6.55E+12	8.58E+06	6.69E+12	1.32E+13
1.66E+00	to	2.00E+00	1.57E+11	1.05E+08	2.94E+03	1.57E+11
2.00E+00	to	2.50E+00	1.94E+11	6.21E+07	1.59E+08	1.94E+11
2.50E+00	to	3.00E+00	1.05E+10	1.04E+08	2.46E+05	1.06E+10
3.00E+00	to	4.00E+00	1.33E+09	3.23E+07	1.77E-10	1.36E+09
4.00E+00	to	5.00E+00	1.38E-05	1.09E+07	0.00E+00	1.09E+07
5.00E+00	to	6.50E+00	3.98E-06	4.38E+06	0.00E+00	4.38E+06
6.50E+00	to	8.00E+00	5.06E-07	8.60E+05	0.00E+00	8.60E+05
8.00E+00	to	1.00E+01	6.76E-08	1.83E+05	0.00E+00	1.83E+05
		Total	**1.94E+15**	**1.16E+13**	**3.60E+13**	**1.99E+15**

Table 5.2-8
Source Term for Design Basis Fuel Assembly

HARDWARE

	Top Fitting	Plenum	Active Fuel	Bottom Fitting	Total
Total Cobalt (kg)	0.000580	0.000182	0.00318	0.000615	0.00456
% of Total Cobalt	12.7	4.0	69.8	13.5	100

Energy (MeV)		**Energy (MeV)**	**Total Source (γ/sec/assy)**	**Light Elements (γ/sec/assy)**	**Energy Fraction**
				Top Fitting	
				0.127	
1.00E+00	to	1.33E+00	2.37E+13	3.01E+12	0.780
1.33E+00	to	1.66E+00	6.69E+12	8.49E+11	0.220
			Total	3.86E+12	
				Plenum	
				0.04	
1.00E+00	to	1.33E+00	2.37E+13	9.47E+11	0.780
1.33E+00	to	1.66E+00	6.67E+12	2.67E+11	0.220
			Total	1.21E+12	
				Active Fuel	
				0.698	
1.00E+00	to	1.33E+00	2.37E+13	1.65E+13	0.780
1.33E+00	to	1.66E+00	6.69E+12	4.67E+12	0.220
			Total	2.12E+13	
				Bottom Fitting	
				0.135	
1.00E+00	to	1.33E+00	2.37E+13	3.20E+12	0.780
1.33E+00	to	1.66E+00	6.69E+12	9.03E+11	0.220
			Total	4.10E+12	

Table 5.2-8
Gamma Source Terms for 8x8 Design Basis Fuel Assembly

(Continued)
ACTIVE FUEL REGION

Energy (MeV)		**Energy (MeV)**		**Fission Products**	**Actinides**	**Light Elements**	**Total**
				(γ/s/assembly)			
1.00E-02	to	5.00E-02		4.41E+14	7.66E+12	3.00E+12	4.52E+14
5.00E-02	to	1.00E-01		1.18E+14	3.37E+12	9.81E+10	1.21E+14
1.00E-01	to	2.00E-01		9.24E+13	3.17E+11	2.87E+11	9.30E+13
2.00E-01	to	3.00E-01		2.65E+13	2.28E+11	1.69E+10	2.67E+13
3.00E-01	to	4.00E-01		1.75E+13	1.86E+10	6.45E+10	1.76E+13
4.00E-01	to	6.00E-01		1.90E+14	3.71E+08	1.41E+12	1.92E+14
6.00E-01	to	8.00E-01		9.31E+14	7.15E+08	7.28E+11	9.32E+14
8.00E-01	to	1.00E+00		8.96E+13	1.06E+08	1.37E+10	8.96E+13
1.00E+00	to	1.33E+00		2.74E+13	3.06E+08	**1.65E+13**	4.39E+13
1.33E+00	to	1.66E+00		6.55E+12	8.58E+06	**4.67E+12**	1.12E+13
1.66E+00	to	2.00E+00		1.57E+11	1.05E+08	2.94E+03	1.57E+11
2.00E+00	to	2.50E+00		1.94E+11	6.21E+07	1.59E+08	1.94E+11
2.50E+00	to	3.00E+00		1.05E+10	1.04E+08	2.46E+05	1.06E+10
3.00E+00	to	4.00E+00		1.33E+09	3.23E+07	1.77E-10	1.36E+09
4.00E+00	to	5.00E+00		1.38E-05	1.09E+07	0.00E+00	1.09E+07
5.00E+00	to	6.50E+00		3.98E-06	4.38E+06	0.00E+00	4.38E+06
6.50E+00	to	8.00E+00		5.06E-07	8.60E+05	0.00E+00	8.60E+05
8.00E+00	to	1.00E+01		6.76E-08	1.83E+05	0.00E+00	1.83E+05
		Total		1.94E+15	1.16E+13	2.6805E+13	1.98E+15

Table 5.2-9
Neutron Source Terms for the 8x8 Design Basis Fuel

No.	Energy	-	Energy	7 yrs
	MeV		MeV	neutrons / s / FA
1	6.43E+00	-	2.00E+01	5.896E+06
2	3.00E+00	-	6.43E+00	6.650E+07
3	1.85E+00	-	3.00E+00	7.287E+07
4	1.40E+00	-	1.85E+00	4.146E+07
5	9.00E-01	-	1.40E+00	5.639E+07
6	4.00E-01	-	9.00E-01	6.166E+07
7	1.00E-01	-	4.00E-01	1.207E+07
			Total	3.168E+08

Table 5.2-10
Radioactive Inventory for Verification of Bounding
Irradiation History

Classification	Nuclide	DBF-68		MBF-68		HBF-68
		(Ci/assembly)		(Ci/assembly)		(Ci/assembly)
gas	H3[1]	8.89E+01		8.54E+01		8.29E+01
fine	Pu238	9.72E+02		1.13E+03		1.33E+03
fine	Pu239	5.15E+01		5.04E+01		5.30E+01
fine	Pu240	1.24E+02		1.32E+02		1.33E+02
fine	Pu241	1.90E+04		1.67E+04		1.60E+04
fine	Am241	2.78E+02		3.66E+02		4.39E+02
fine	Am243	2.78E+02		3.66E+02		1.86E+01
fine	Cm243	1.35E+01		1.86E+01		8.16E+00
fine	Cm244	2.22E+03		3.33E+03		3.10E+03
gas	Kr 85	1.28E+03		1.12E+03		1.14E+03
volatile	Sr 90	1.46E+04		1.44E+04		1.60E+04
fine	Y 90	1.46E+04		1.44E+04		1.60E+04
volatile	Ru106	1.29E+03		1.87E+02		5.17E+01
fine	Rh106	1.29E+03		1.87E+02		5.17E+01
fine	Sb125	3.68E+02		1.93E+02		1.25E+02
fine	Te125m	8.98E+01		4.71E+01		3.04E+01
gas	I129	8.94E-03		1.03E-02		1.09E-02
volatile	Cs134	4.92E+03		2.17E+03		1.26E+03
volatile	Cs137	2.47E+04		2.62E+04		2.73E+04
fine	Ba137m	2.33E+04		2.47E+04		2.58E+04
fine	Ce144	3.91E+02		2.66E+01		5.35E+00
fine	Pr144	3.91E+02		2.66E+01		5.35E+00
fine	Pm147	4.52E+03		1.98E+03		1.27E+03
fine	Eu154	1.04E+03		9.33E+02		8.91E+02
fine	Eu155	3.15E+02		2.35E+02		1.87E+02
	Total	**1.16E+05**		**1.09E+05**		**1.11E+05**
		SAS2H Results For All Isotopes[2]				
	Light Elements	7.08E+02		5.05E+02		3.78E+02
	Actinides	2.26E+04		2.17E+04		2.11E+04
	Fission Products	9.32E+04		8.70E+04		9.04E+04
	Total	**1.16E+05**		**1.09E+05**		**1.12E+05**

1) The total of H3 from Light Elements and Fission Products.
2) A comparison of the total radioactive inventory indicates that the activity of the selected isotopes represents greater than 99.4% of the total.

Table 5.2-11
Fission Gas Inventory for 8x8 Design Basis Fuel

Nuclide	**SAS2H (g/assembly)**	**Molar Mass (kg/kg-mole)**	**Fission Gas Content (kg-mole /assembly)**
Kr82	3.54E-01	82	4.32E-06
Kr83	8.69E+00	83	1.05E-04
Kr84	3.85E+01	84	4.58E-04
Kr85	2.90E+00	85	3.41E-05
Kr86	5.42E+01	86	6.30E-04
I127	1.56E+01	127	1.23E-04
I129	6.19E+01	129	4.80E-04
Xe128	1.91E+00	128	1.49E-05
Xe130	5.95E+00	130	4.58E-05
Xe131	9.94E+01	131	7.59E-04
Xe132	4.27E+02	132	3.23E-03
Xe134	5.09E+02	134	3.80E-03
Xe136	8.13E+02	136	5.98E-03
SAS2H Total			**1.57E-02**

Table 5.2-12
Fuel Qualification Table for Uniform Heat Load
(441 Watts per Assembly)

Maximum Assembly Burnup in GWd/MTU is shown as a function of Enrichment and Cooling time

Enrichment (wt % U235)	**7 Years**	**8 Years**	**9 Years**	**10 Years**	**12 Years**
1.00		*47.5 [4]*	*50.2 [9]*		
1.80		*49.0 [5]*			
2.40	*47.5 [1]*	*50.2 [6]*	*52.7 [10]*	54.8	58.5
2.50				*55.0 [12]*	
2.60	*47.9 [2]*	50.6	53.1	55.2	58.9
2.80	48.3	*51.0 [7]*	53.5	55.6	59.2
3.00	48.7	51.4	*53.9 [11]*	56.0	59.6
3.20	49.1	51.8	54.2	56.4	*60.0 [14]*
3.40	49.5	52.2	54.6	56.8	
3.60	49.9	52.6	55.0	57.1	
3.80	50.2	*53.0 [8]*	55.4	57.5	
4.00	50.6	53.3	55.7	*57.9 [13]*	
4.20	50.9	53.7	56.1	58.2	
4.40	51.2	54.0	56.4	58.6	
4.60	*51.6 [3]*	54.3	56.8	58.9	

The cases analyzed in the SAS2H dose comparison studies are italicized and the corresponding case numbers are shown in square brackets [ ].

Table 5.2-13
Verification of Fuel Qualification Table
Surface Dose Rate Comparison

Case ID	Description	Surface Dose Rate (mrem/hour)		
		Neutron	Gamma	Total
1	Enrichment 2.40 wt % U235 Burnup 47.5 GWd/MTU Cooling Time 7 Years	10.5	104.1	114.6
2	Enrichment 2.60 wt % U235 Burnup 48.0 GWd/MTU Cooling Time 7 Years	9.9	103.6	113.5
3	Enrichment 4.60 wt % U235 Burnup 52.0 GWd/MTU Cooling Time 7 Years	5.5	96.6	102.1
4	Enrichment 1.00 wt % U235 Burnup 47.5 GWd/MTU Cooling Time 8 Years	20.0	100.2	120.2
5	Enrichment 1.80 wt % U235 Burnup 49.0 GWd/MTU Cooling Time 8 Years	15.0	94.9	109.9
6	Enrichment 2.40 wt % U235 Burnup 50.2 GWd/MTU Cooling Time 8 Years	12.3	92.1	104.4
7	Enrichment 2.80 wt % U235 Burnup 51.0 GWd/MTU Cooling Time 8 Years	10.9	90.7	101.6
8	Enrichment 3.80 wt % U235 Burnup 53.0 GWd/MTU Cooling Time 8 Years	8.0	87.3	95.3
9	Enrichment 1.00 wt % U235 Burnup 50.2 GWd/MTU Cooling Time 9 Years	22.4	90.9	113.3
10	Enrichment 2.40 wt % U235 Burnup 52.7 GWd/MTU Cooling Time 9 Years	13.9	83.1	97.0
11	Enrichment 3.00 wt % U235 Burnup 54.0 GWd/MTU Cooling Time 9 Years	11.7	81.3	93.0
12	Enrichment 2.50 wt % U235 Burnup 55.0 GWd/MTU Cooling Time 10 Years	14.9	75.5	90.4
13	Enrichment 4.00 wt % U235 Burnup 58.0 GWd/MTU Cooling Time 10 Years	9.7	72.6	82.3
14	Enrichment 3.20 wt % U235 Burnup 60.0 GWd/MTU Cooling Time 12 Years	14.0	64.1	78.1

Table 5.2-14
Verification of Fuel Qualification Table
2 M Dose Rate Comparison

Case ID	Description	Dose Rate at 2 m (mrem/hour)		
		Neutron	Gamma	Total
1	Enrichment 2.40 wt % U235 Burnup 47.5 GWd/MTU Cooling Time 7 Years	2.3	29.4	31.7
2	Enrichment 2.60 wt % U235 Burnup 48.0 GWd/MTU Cooling Time 7 Years	2.2	29.2	31.4
3	Enrichment 4.60 wt % U235 Burnup 52.0 GWd/MTU Cooling Time 7 Years	1.2	27.3	28.5
4	Enrichment 1.00 wt % U235 Burnup 47.5 GWd/MTU Cooling Time 8 Years	4.4	28.1	32.5
5	Enrichment 1.80 wt % U235 Burnup 49.0 GWd/MTU Cooling Time 8 Years	3.3	26.7	30.0
6	Enrichment 2.40 wt % U235 Burnup 50.2 GWd/MTU Cooling Time 8 Years	2.7	25.9	28.6
7	Enrichment 2.80 wt % U235 Burnup 51.0 GWd/MTU Cooling Time 8 Years	2.4	25.5	27.9
8	Enrichment 3.80 wt % U235 Burnup 53.0 GWd/MTU Cooling Time 8 Years	1.8	24.6	26.4
9	Enrichment 1.00 wt % U235 Burnup 50.2 GWd/MTU Cooling Time 9 Years	5.0	25.4	30.4
10	Enrichment 2.40 wt % U235 Burnup 52.7 GWd/MTU Cooling Time 9 Years	3.1	23.3	26.4
11	Enrichment 3.00 wt % U235 Burnup 54.0 GWd/MTU Cooling Time 9 Years	2.6	22.8	25.4
12	Enrichment 2.50 wt % U235 Burnup 55.0 GWd/MTU Cooling Time 10 Years	3.3	21.1	24.4
13	Enrichment 4.00 wt % U235 Burnup 58.0 GWd/MTU Cooling Time 10 Years	2.2	20.3	22.5
14	Enrichment 3.20 wt % U235 Burnup 60.0 GWd/MTU Cooling Time 12 Years	3.1	17.9	21.0

Table 5.2-15
Fuel Qualification Table for TN-68 Cask
(Modified to Take Shielding Qualification into Account)

Maximum Assembly Burnup in GWd/MTU is shown as a function of Enrichment and Cooling time

Enrichment (wt % U235)	**7 Years**	**8 Years**	**9 Years**	**10 Years**	**12 Years**
2.40	*47.5*	50.2	52.7	54.8	58.5
2.60	47.9	50.6	53.1	55.2	58.9
2.80	48.3	51.0	53.5	55.6	59.2
3.00	48.7	51.4	53.9	56.0	59.6
3.20	49.1	51.8	54.2	56.4	60.0
3.40	49.5	52.2	54.6	56.8	*60.3*
3.60	49.9	52.6	55.0	57.1	*60.7*
3.80	50.2	53.0	55.4	57.5	*61.0*
4.00	50.6	53.3	55.7	57.9	*61.4*
4.20	50.9	53.7	56.1	58.2	*61.8*
4.40	51.2	54.0	56.4	58.6	*62.1*
4.60	51.6	54.3	56.8	58.9	*62.5*

The shaded and italicized cells represent restrictions imposed due to either the assembly burnup being greater than 60GWd/MTU or a greater than acceptable burnup / enrichment ratio (*be-ratio*)

Table 5.3-1
ANSI Standard-6.1.1-1977 Flux-to-Dose Factors

Photon energy (MeV)	Response Function (rem/hr)/(γ/cm^2-s)	Neutron energy (MeV)	Response Function (rem/hr)/(n/cm^2-s)
0.01	3.96E-06	2.5E-08	3.67E-06
0.03	5.82E-07	1.0E -07	3.67E-06
0.05	2.90E-07	1.0E-06	4.46E-06
0.07	2.58E-07	1.0E-05	4.54E-06
0.10	2.83E-07	1.0E-04	4.18E-06
0.15	3.79E-07	1.0E-03	3.76E-06
0.20	5.01E-07	1.0E-02	3.56E-06
0.25	6.31E-07	1.0E-01	2.17E-05
0.30	7.59E-07	5.0E-01	9.26E-05
0.35	8.78E-07	1.0	1.32E-04
0.40	9.85E-07	2.5	1.25E-04
0.45	1.08E-06	5.0	1.56E-04
0.50	1.17E-06	7.0	1.47E-04
0.55	1.27E-06	10.0	1.47E-04
0.60	1.36E-06	14.0	2.08E-04
0.65	1.44E-06	20.0	2.27E-04
0.70	1.52E-06		
0.80	1.68E-06		
1.0	1.98E-06		
1.4	2.51E-06		
1.8	2.99E-06		
2.2	3.42E-06		
2.6	3.82E-06		
2.8	4.01E-06		
3.25	4.41E-06		
3.75	4.83E-06		
4.25	5.23E-06		
4.75	5.60E-06		
5.0	5.80E-06		
5.25	6.01E-06		
5.75	6.37E-06		
6.25	6.74E-06		
6.75	7.11E-06		
7.5	7.66E-06		
9.0	8.77E-06		
11.0	1.03E-05		
13.0	1.18E-05		
15.0	1.33E-05		

Table 5.3-2
Material Compositions for the BWR Fuel Assembly

Fuel Zone	Height (cm)	UO_2 (kg)	Zircaloy (kg)	Inconel (kg)	SS304 (kg)	Volume (cm^3)
Bottom	18.77	0	1.60 (1.255)	0.05	4.7	3.687E+03
Active	370.84	213.2	72.21 (51.15)	0	0	7.285E+03
Plenum	36.76	0	6.98 (4.89)	0	1.05	7.221E+03
Top	21.18	0	1.88 (1.255)	0.556 (0.430)	2.715 (2.130)	4.161E+03

Note: The BWR fuel assembly is modeled as a homogenized cuboid with a cross section area of 196.446 cm^2 (14.016 cm by 14.016 cm) or about 5.518″ by 5.518″.

The values shown in brackets are for an un-channeled fuel assembly

Table 5.3-3
Atomic Fractions for the Homogenized Design Basis 8x8 Fuel Without Channel

Nuclide ID	Element	Number Density (Atoms/barn-cm)	Atomic Fraction
Active Fuel Region, Density = 3.629 g/cm^3			
8016	8016	8016	8016
92235	92235	92235	92235
92238	92238	92238	92238
40000	40000	40000	40000
Bottom Nozzle Region, Density = 1.615 g/cm^3			
40000	40000	40000	40000
24000	24000	24000	24000
25055	25055	25055	25055
26000	26000	26000	26000
28000	28000	28000	28000
Plenum Region, Density = 0.823 g/cm^3			
40000	40000	40000	40000
24000	24000	24000	24000
25055	25055	25055	25055
26000	26000	26000	26000
28000	28000	28000	28000
Top Nozzle Region, Density = 0.917 g/cm^3			
40000	40000	40000	40000
24000	24000	24000	24000
25055	25055	25055	25055
26000	26000	26000	26000
28000	28000	28000	28000

Table 5.3-4
Atomic Fractions for the Homogenized Design Basis 8x8 Fuel
with a 0.065-Inch Channel

Nuclide ID	Element	Number Density (Atoms/barn-cm)	Atomic Fraction
Active Fuel Region, Density = 3.918 g/cm^3			
8016	Oxygen	1.30561E-02	0.49972
92235	Uranium	1.71844E-04	0.00658
92238	Uranium	6.35622E-03	0.24328
40000	Zirconium	6.54254E-03	0.25042
Bottom Nozzle Region, Density = 1.723 g/cm^3			
40000	Zirconium	2.87066E-03	0.17150
24000	Chromium	2.83405E-03	0.16931
25055	Manganese	2.82344E-04	0.01687
26000	Iron	9.49593E-03	0.56731
28000	Nickel	1.25546E-03	0.07500
Plenum Region, Density = 1.112 g/cm^3			
40000	Zirconium	6.37972E-03	0.80305
24000	Chromium	3.19748E-04	0.04025
25055	Manganese	3.18551E-05	0.00401
26000	Iron	1.07137E-03	0.13486
28000	Nickel	1.41646E-04	0.01783
Top Nozzle Region, Density = 1.238 g/cm^3			
40000	Zirconium	2.98634E-03	0.26080
24000	Chromium	1.72979E-03	0.15106
25055	Manganese	1.72331E-04	0.01505
26000	Iron	5.79591E-03	0.50616
28000	Nickel	7.66280E-04	0.06692

Table 5.3-5
Material Properties for the TN-68 Cask MCNP Models

Material Description	Density (g/cm^3)	Element and Fraction (w_i)[1]		
Air	0.0012	^{14}N -0.75519	^{16}O -0.23179 Ar -0.01288	C -0.00014
Concrete	2.320	^{1}H -0.0056 Mg -0.0024 S -0.0012	^{16}O -0.4983 ^{27}Al -0.0456 K -0.0192 Fe -0.0122	^{23}Na -0.0171 Si -0.3158 Ca -0.0826
Soil	1.625	^{1}H -0.021 Fe -0.011 Si -0.271	^{12}C -0.016 Ca -0.041 ^{16}O -0.577	K -0.013 ^{27}Al -0.050
Active Fuel (no Channel)	3.629	^{16}O 0.53906	^{235}U 0.00710 Zr 0.19141	^{238}U 0.26244
Top Nozzle (no Channel)	0.917	Zr 0.23117 Fe 0.52645	Cr 0.15712 Ni 0.06960	^{55}Mn 0.01565
Plenum Zone (no Channel)	0.823	Zr 0.74067 Fe 0.17757	Cr 0.05300 Ni 0.02348	^{55}Mn 0.00528
Bottom Fitting (no Channel)	1.615	Zr 0.14060 Fe 0.58847	Cr 0.17563 Ni 0.07780	^{55}Mn 0.01750
Active Fuel (with Channel)	3.918	^{16}O 0.49972	^{235}U 0.00658 Zr 0.25042	^{238}U 0.24328
Top Nozzle (with Channel)	1.238	Zr 0.26080 Fe 0.50616	Cr 0.15106 Ni 0.06692	^{55}Mn 0.01505
Plenum Zone (with Channel)	1.112	Zr 0.80305 Fe 0.13486	Cr 0.04025 Ni 0.01783	^{55}Mn 0.00401
Bottom Fitting (with Channel)	1.723	Zr 0.17150 Fe 0.56731	Cr 0.16931 Ni 0.07500	^{55}Mn 0.01687
SS304 Rails	7.920	Fe 0.68826	^{55}Mn 0.02013 Ni 0.08952	Cr 0.20209
Cask Body Carbon Steel	7.8212	Fe 0.95510	C 0.04490	
Aluminum Rails	2.702	^{27}Al 1.0000		
Polypropylene Disk	0.900	^{12}C 0.33480	^{1}H 0.66520	
Resin Aluminum Mix Radial Neutron Shield	1.687	^{27}Al 0.10331 ^{1}H 0.42207	^{12}C 0.24658 ^{10}B 0.00164	^{16}O 0.21985 ^{11}B 0.00655

1) The material composition is shown as atomic fractions (if positive) or weight fraction (if negative), for each component

Table 5.4-1
TN-68 Near Field Dose Rates – Normal and Off-Normal Conditions
(Without the Auxiliary Shield Ring)

Axial Interval	Cask Surface (mrem/hour)			1 M from Cask Surface (mrem/hour)			2 M from Cask Surface (mrem/hour)		
(cm)	gamma	neutron	total	gamma	neutron	total	gamma	neutron	total
-228.96									
-204.19	199	139	338	60.0	22.6	82.6	32.0	9.39	41.4
-195.3	302	129	431	56.8	22.1	78.9	29.0	9.56	38.6
-164.59	81.3	35.7	117	48.0	19.4	67.4	28.6	9.23	37.9
-146.3	74.2	34.2	108	44.2	16.3	60.5	30.9	8.76	39.6
-109.73	103	33.8	137	46.4	14.7	61.1	30.4	8.16	38.6
-73.15	110	21.4	131	51.4	12.3	63.7	31.9	7.68	39.5
-36.53	119	22.4	141	54.6	11.1	65.7	34.0	7.40	41.4
0	123	24.2	147	58.0	11.0	69.0	36.6	6.43	43.0
36.53	121	22.6	143	55.3	9.76	65.1	37.2	6.42	43.6
73.15	113	18.5	131	53.7	8.64	62.3	38.7	6.14	44.9
109.73	102	15.2	117	54.1	7.54	61.6	42.0	5.90	47.9
146.3	83.1	8.88	92.0	57.8	7.37	65.2	51.4	5.63	57.0
164.59	62.1	5.28	67.4	73.0	8.19	81.1	60.4	5.61	66.0
185.42	51.6	4.09	55.7	99.7	9.03	109	71.9	5.81	77.8
200	76.5	6.67	83.2	145	10.4	156	80.7	5.97	86.7
222.18	454	43.7	498	183	11.3	195	87.5	6.29	93.8
243.36	1010	76.3	1080	202	12.6	215	87.8	6.26	94.0
263.81	831	56.8	887	202	12.5	215	92.9	6.02	98.9
282.22	249	33.6	283	175	11.8	187	86.0	5.79	91.8
316.2	94.9	16.7	112	121	10.6	132	73.6	5.71	79.3
Average	**189**	**32.3**	**221**	**83.3**	**11.9**	**95.2**	**49.7**	**6.85**	**56.6**

Radius	Top Surface			Bottom Surface		
(cm)	gamma	neutron	total	gamma	neutron	total
25.0	360	1.48	361	603	1240	1850
50.0	319	2.33	321	558	1080	1640
75.0	232	2.23	234	452	817	1270
101	128	5.91	133	136	438	574
124	56.0	8.21	64.2	32.1	137	169
Average	**156**	**5.36**	**161**	**233**	**520**	**753**

Table 5.4-2
TN-68 Near Field Dose Rates – Normal and Off-Normal Conditions
(With 1″ Auxiliary Shield Ring)

Axial Interval	Cask Surface (mrem/hour)			1 M from Cask Surface (mrem/hour)			2 M from Cask Surface (mrem/hour)		
(cm)	gamma	neutron	total	gamma	neutron	total	gamma	neutron	total
-228.96									
-204.19	196	140	336	59.8	22.8	82.6	30.9	9.46	40.4
-195.3	301	129	430	56.4	22.4	78.7	28.3	9.51	37.8
-164.59	79.4	35.6	115	47.3	19.5	66.8	28.6	9.19	37.8
-146.3	74.3	34.5	109	43.7	16.2	59.8	30.1	8.88	39.0
-109.73	102	33.7	136	46.4	14.8	61.2	30.1	8.19	38.2
-73.15	111	21.7	133	51.0	12.2	63.3	31.0	7.76	38.8
-36.53	118	22.7	141	54.3	11.1	65.4	32.8	7.37	40.2
0	121	24.6	146	56.7	11.0	67.7	34.4	6.49	40.9
36.53	119	22.5	141	54.9	9.88	64.8	34.5	6.39	40.9
73.15	112	18.5	130	51.6	8.56	60.2	34.5	6.06	40.6
109.73	102	14.9	117	50.0	7.62	57.6	33.6	5.71	39.4
146.3	83.2	8.86	92.1	47.3	7.17	54.5	35.3	5.48	40.8
164.59	63.2	5.18	68.4	48.5	7.86	56.4	35.9	5.39	41.3
185.42	48.8	4.37	53.1	56.9	8.59	65.5	39.7	5.58	45.3
200	71.4	6.50	77.9	73.4	9.74	83.2	41.6	5.74	47.4
222.18	204	40.4	244	80.9	10.4	91.3	41.7	5.83	47.5
243.36	404	67.3	472	87.2	11.1	98.4	42.7	5.61	48.3
263.81	327	48.1	375	86.4	11.4	97.8	41.3	5.42	46.7
282.22	130	30.0	160	74.7	10.4	85.1	39.1	5.45	44.6
316.2	72.7	16.2	88.9	51.4	9.74	61.1	32.7	5.19	37.9
Average	**130**	**31.4**	**162**	**56.8**	**11.7**	**68.5**	**34.4**	**6.70**	**41.1**

Radius	Top Surface			Bottom Surface		
(cm)	gamma	neutron	total	gamma	neutron	total
25.0	350	1.66	352	603	1230	1830
50.0	318	2.38	320	573	1080	1650
75.0	228	2.45	230	465	816	1280
101	131	6.08	137	132	438	569
124	54.2	8.13	62.3	34.1	137	171
Average	**155**	**5.44**	**161**	**236**	**519**	**755**

Table 5.4-3
TN-68 Near Field Dose Rates – Normal Conditions
(Dose Rates Above and Below the Neutron Shield)

Axial Interval	**Above Neutron Shield No Shield Ring (mrem/hour)**			**Above Neutron Shield 1-inch Shield Ring (mrem/hour)**		
(cm)	gamma	neutron	total	gamma	neutron	total
200						
222.18	1330	288	1620	451[1]	140[1]	591
243.36	1860	150	2010	620	111	731
263.81	1190	78.9	1260	428	57.8	485
Average	**1460**	**175**	**1630**	**500**	**104**	**604**

Axial Interval	**Below Neutron Shield (mrem/hour)**		
(cm)	gamma	neutron	total
-228.96			
-204.19	179	214	392
-195.3	866	445	1310
Average	**360**	**275**	**635**

1) The MCNP gamma and neutron dose rates between surfaces 72 and 8 (200 cm and 222.18 cm) are doubled to obtain the dose rates shown since the auxiliary shield ring only extends until 211.1 cm.

Table 5.4-4
TN-68 Near Field Dose Rates – Accident Conditions

Axial Interval	Cask Surface (mrem/hour)			1 M from Cask Surface (mrem/hour)			2 M from Cask Surface (mrem/hour)		
(cm)	gamma	neutron	total	gamma	neutron	total	gamma	neutron	total
-228.96									
-204.19	234	309	544	191	497	689	143	351	494
-195.3	477	460	936	220	546	766	157	380	537
-164.59	564	719	1280	275	618	892	181	411	592
-146.3	621	1170	1790	333	727	1060	207	449	656
-109.73	831	1870	2700	386	864	1250	231	493	724
-73.15	946	2660	3600	438	1030	1470	262	547	810
-36.53	999	3120	4120	471	1140	1620	280	580	860
0	1000	3290	4300	479	1200	1680	287	598	885
36.53	982	3200	4180	475	1170	1650	285	587	872
73.15	937	2800	3740	450	1060	1510	279	555	834
109.73	885	2170	3060	420	903	1320	263	503	766
146.3	736	1390	2130	372	718	1090	238	439	677
164.59	540	842	1380	338	579	916	222	384	606
185.42	452	549	1000	316	486	802	219	347	567
200	528	360	889	319	411	730	197	318	516
222.18	624	224	848	309	342	651	188	284	472
243.36	1020	138	1150	289	271	559	171	250	421
263.81	834	88.4	922	257	219	475	159	219	378
282.22	255	58.5	314	211	176	387	139	196	335
316.2	118	44.0	162	142	137	279	114	163	278
Average	**724**	**1560**	**2280**	**355**	**730**	**1090**	**222**	**430**	**652**

Radius	Top Surface		
(cm)	gamma	neutron	total
25.0	665	101	766
50.0	598	91.5	689
75.0	435	74.9	510
101	255	52.5	307
124	121	39.8	161
Average	**302**	**59.2**	**362**

Bottom Surface		
gamma	neutron	total
592	1250	1840
571	1070	1640
474	816	1290
136	439	576
31.9	161	193
238	**527**	**765**

Table 5.4-5
TN-68 - Single Cask Farfield Dose Rates

Distance (meters)	Gamma Dose (mrem/hour)	Error	Neutron Dose (mrem/hour)	Error
Total Dose Rate Results				
10	3.67E+00	0.0199	7.14E-01	0.0198
20	1.21E+00	0.0272	2.23E-01	0.0240
40	3.07E-01	0.0330	5.27E-02	0.0279
60	1.34E-01	0.0365	2.36E-02	0.0307
80	7.12E-02	0.0398	1.16E-02	0.0302
100	4.06E-02	0.0444	6.74E-03	0.0323
200	5.30E-03	0.0513	8.88E-04	0.0362
400	4.42E-04	0.0547	5.62E-05	0.0313
600	6.50E-05	0.0752	6.79E-06	0.0460

Figure 5.1-1
Cask Shielding Configuration

Figure Withheld Under 10 CFR 2.390

FIGURE 5.1-1
CASK SHIELDING
CONFIGURATION

Figure 5.1-2
TN-68 Normal Conditions Dose Point Locations

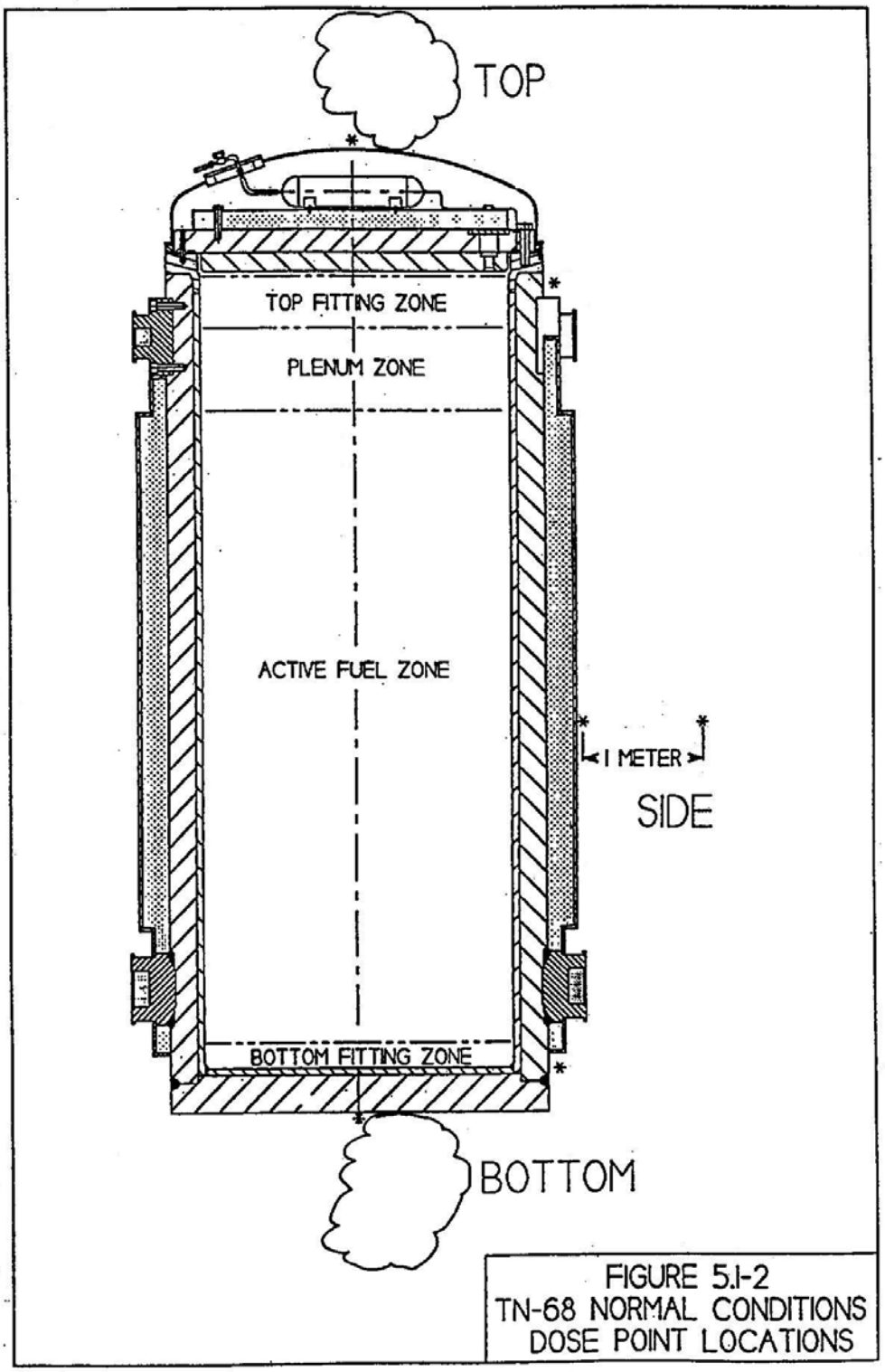
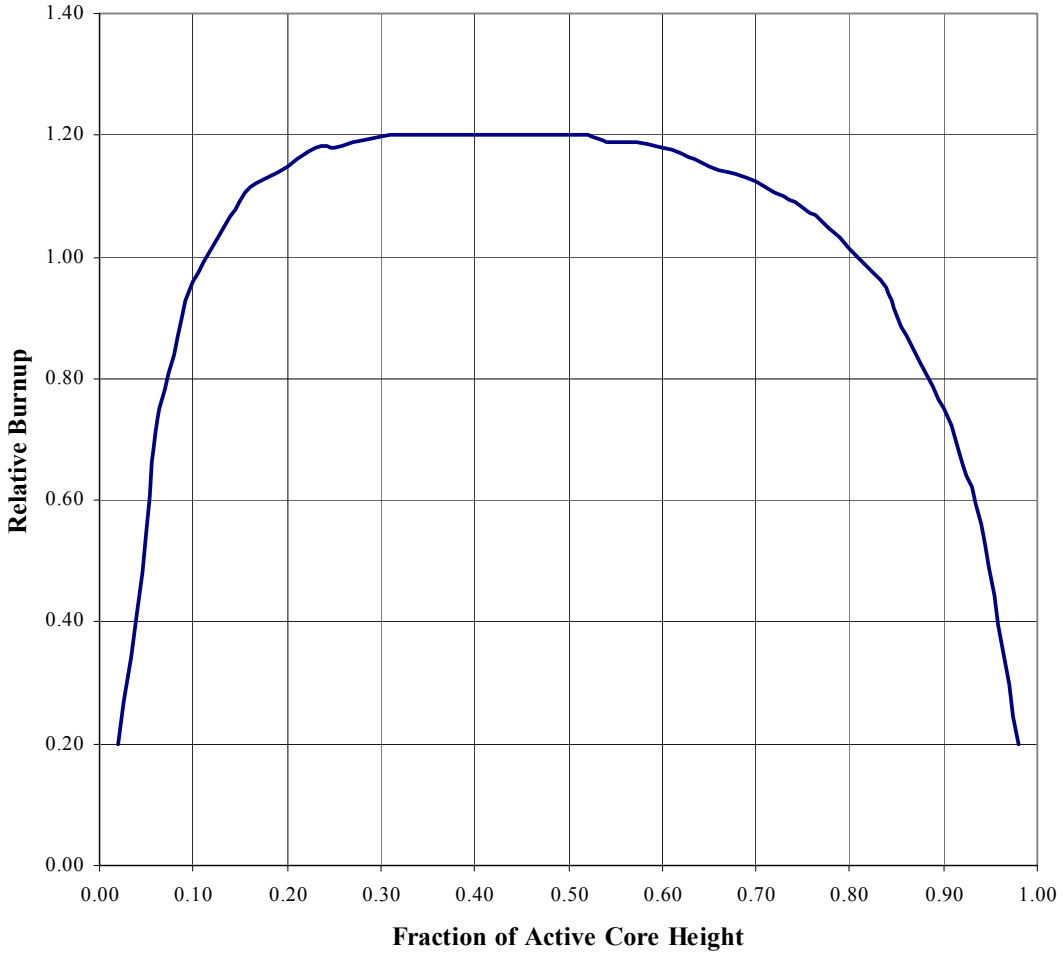


Figure 5.2-1

Axial Burnup Profile for Design Basis Fuel



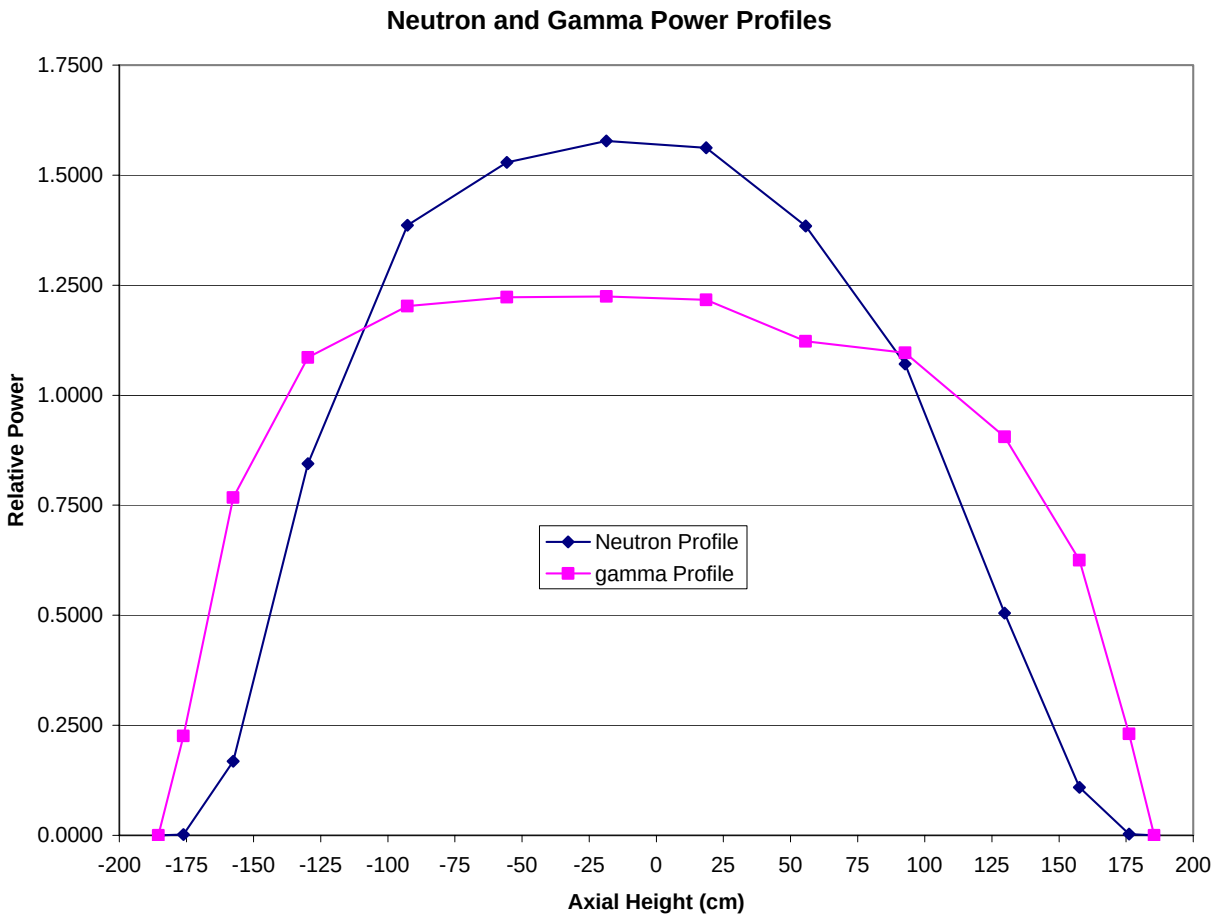


Figure 5.2-2
Axial Neutron and Gamma Profiles

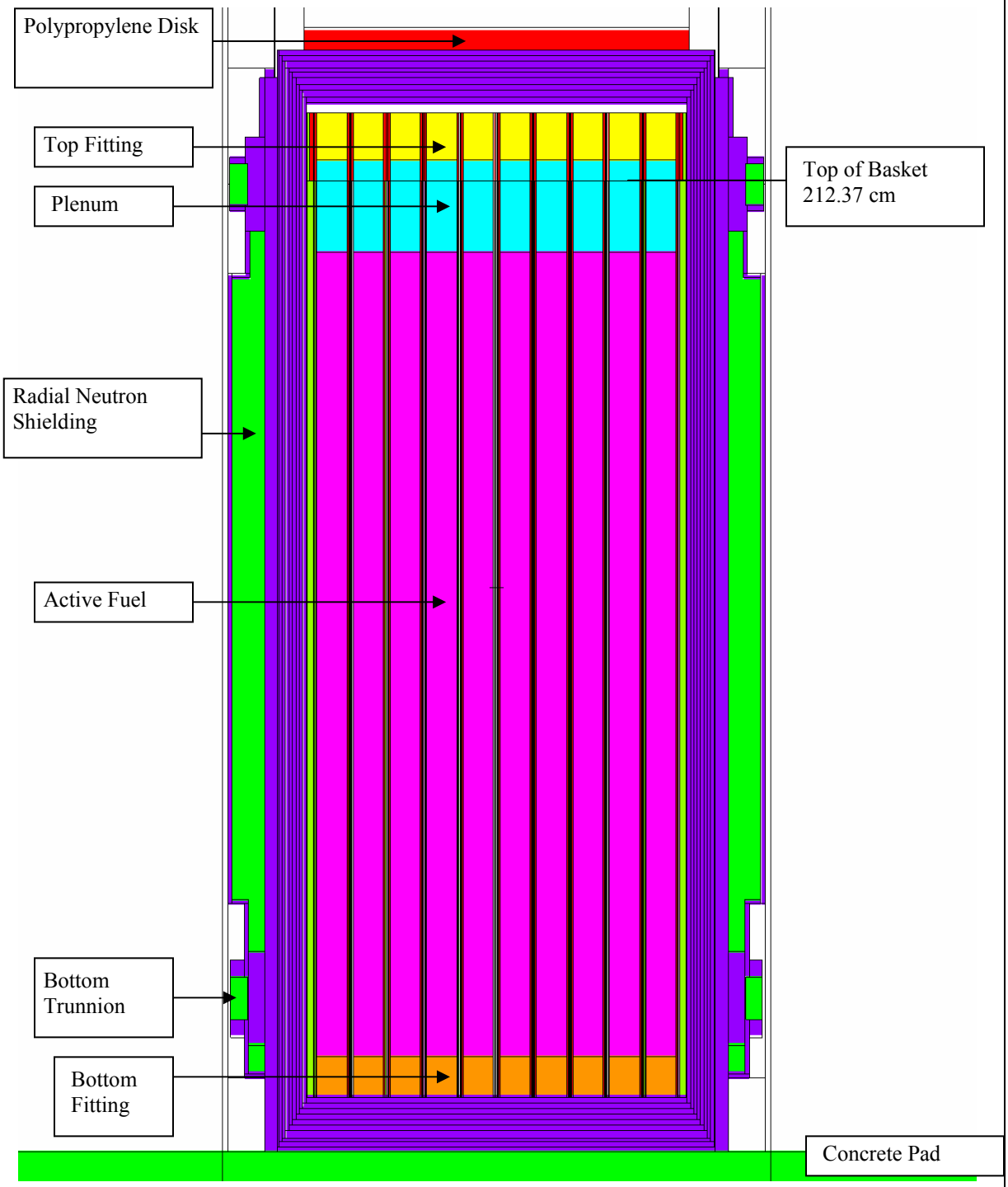


Figure 5.3-1
TN-68 MCNP Gamma Model

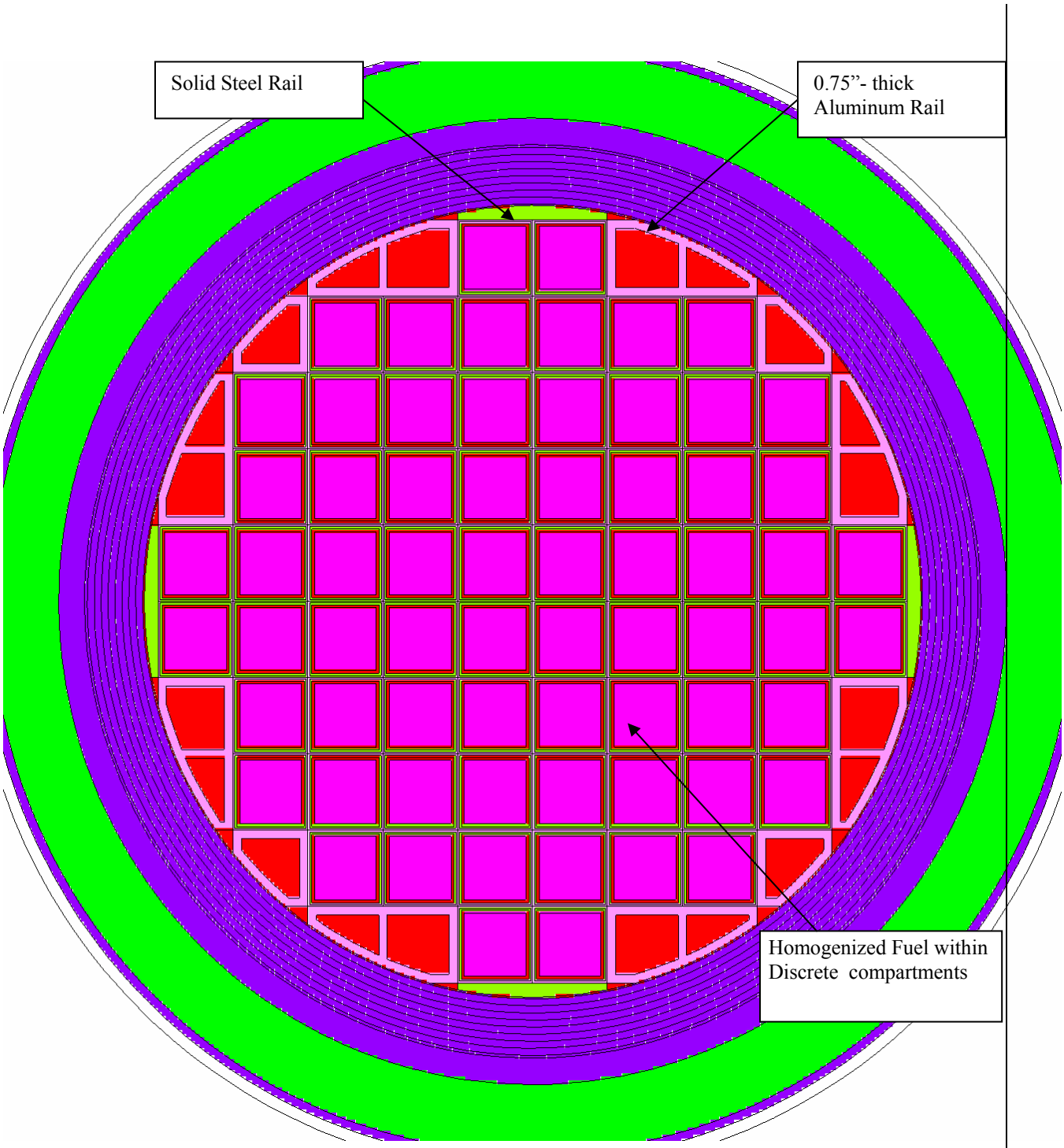


Figure 5.3-2
TN-68 Discrete Basket MCNP Model – XY Plot at Z=0 cm

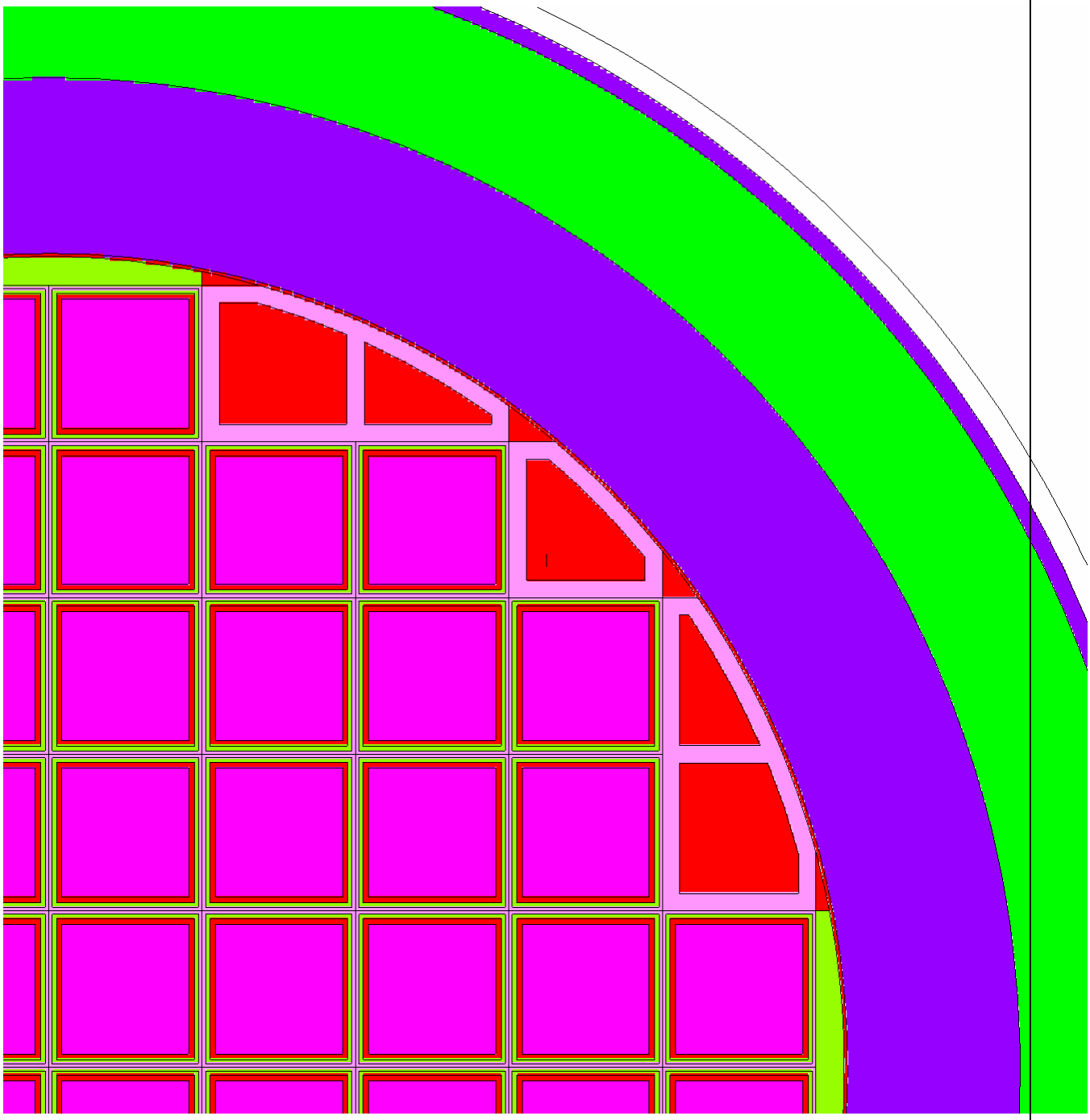


Figure 5.3-3
TN-68 MCNP Quarter Model – XY Plot At Z=0 cm

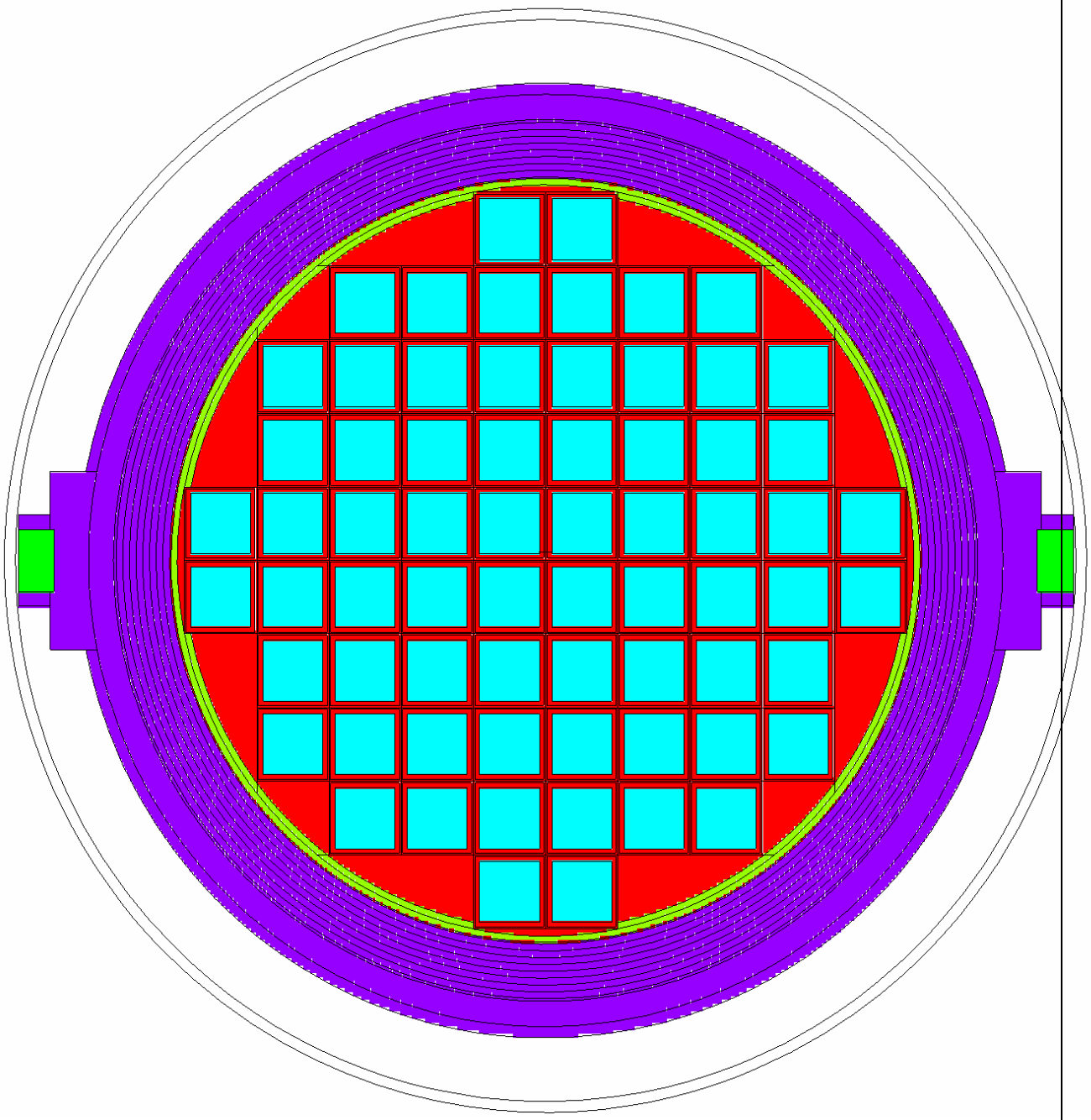


Figure 5.3-4
TN-68 Discrete Basket MCNP Model – XY Plot At Z=220 cm

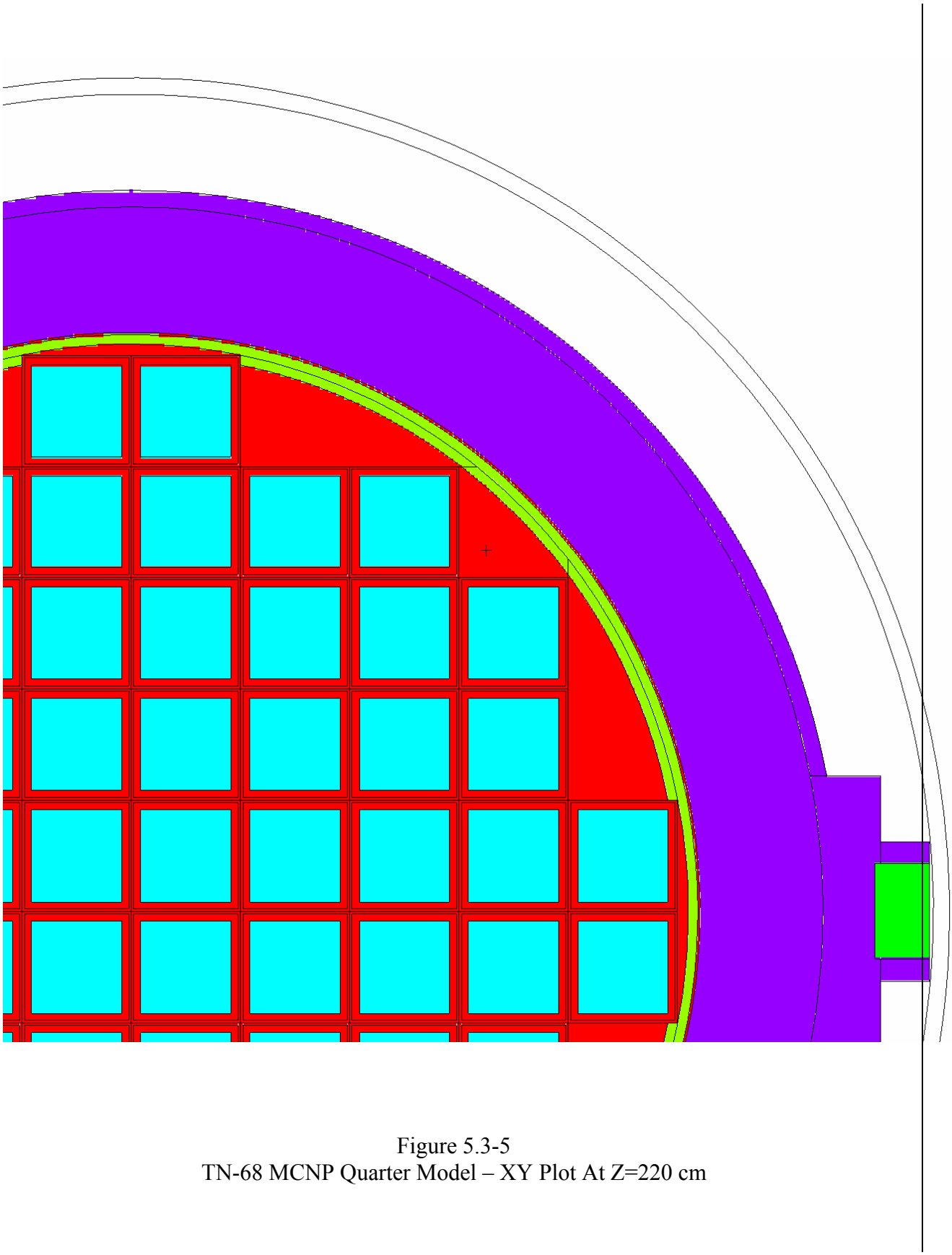


Figure 5.3-5
TN-68 MCNP Quarter Model – XY Plot At Z=220 cm

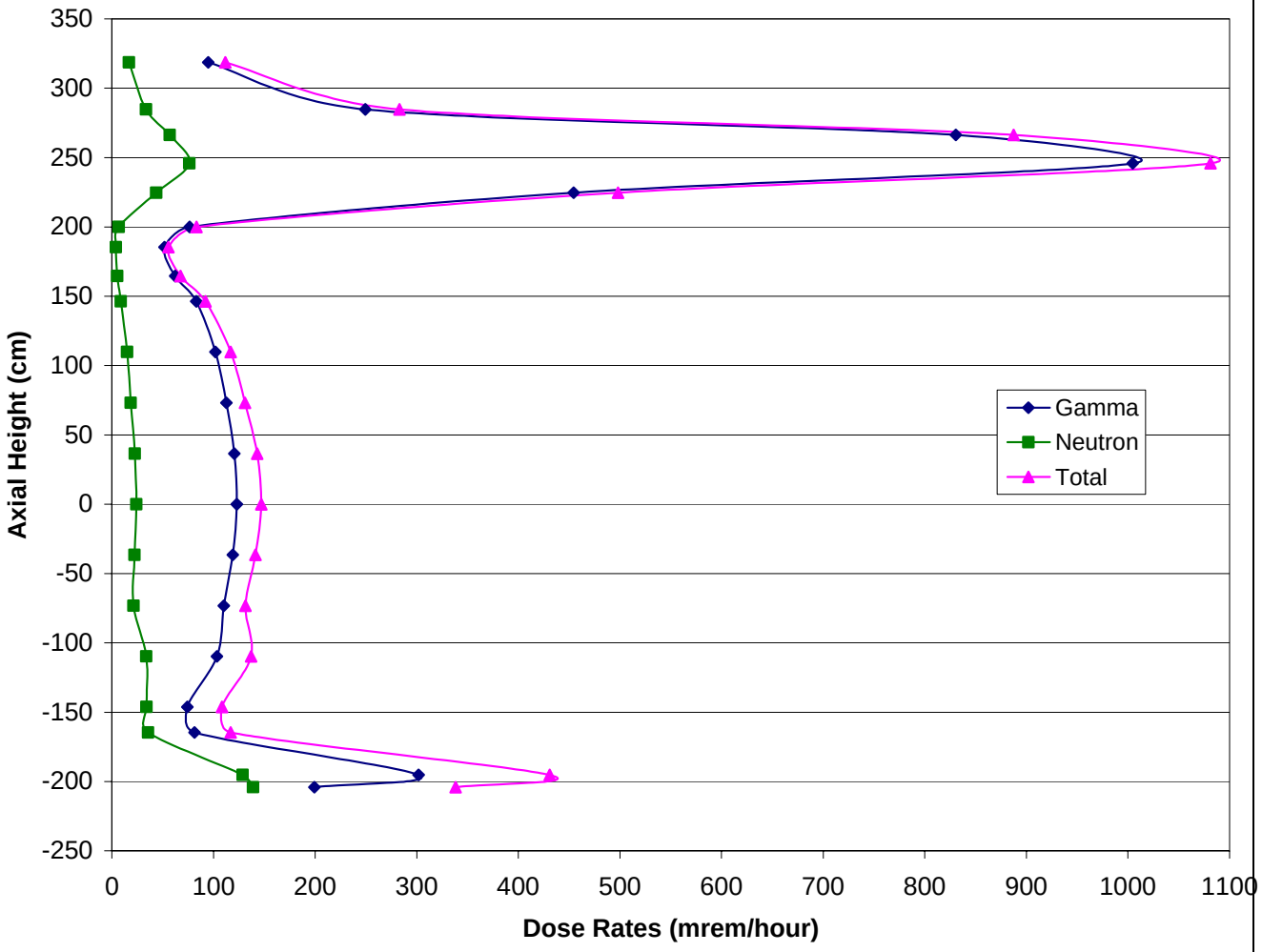


Figure 5.4-1
TN-68 Cask Side Surface Dose Rates
Normal and Off-Normal Conditions
(Without Auxiliary Shield)

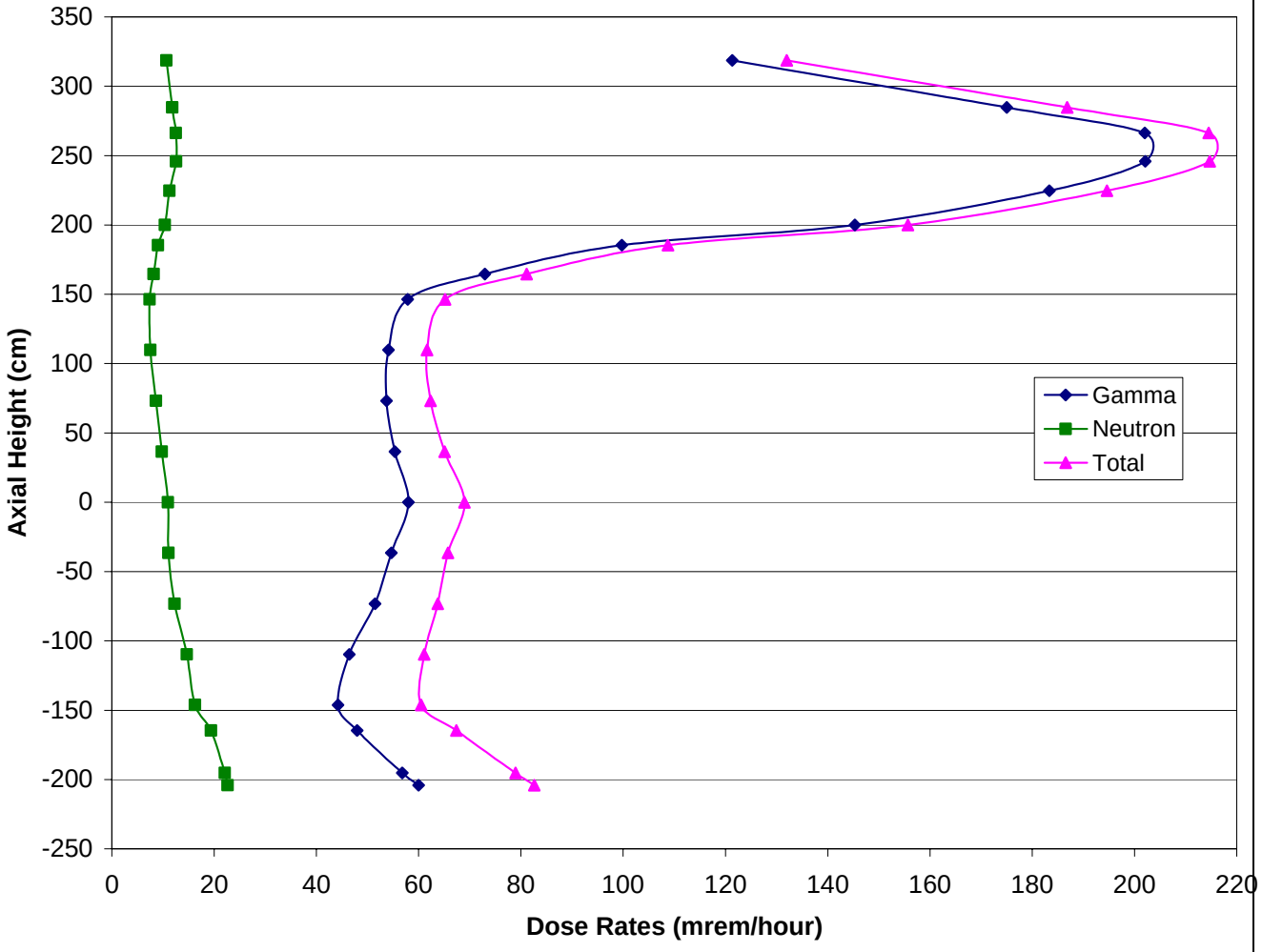


Figure 5.4-2
TN-68 Cask Dose Rates 1 m from Side Surface
Normal and Off-Normal Conditions
(Without Auxiliary Shield Ring)

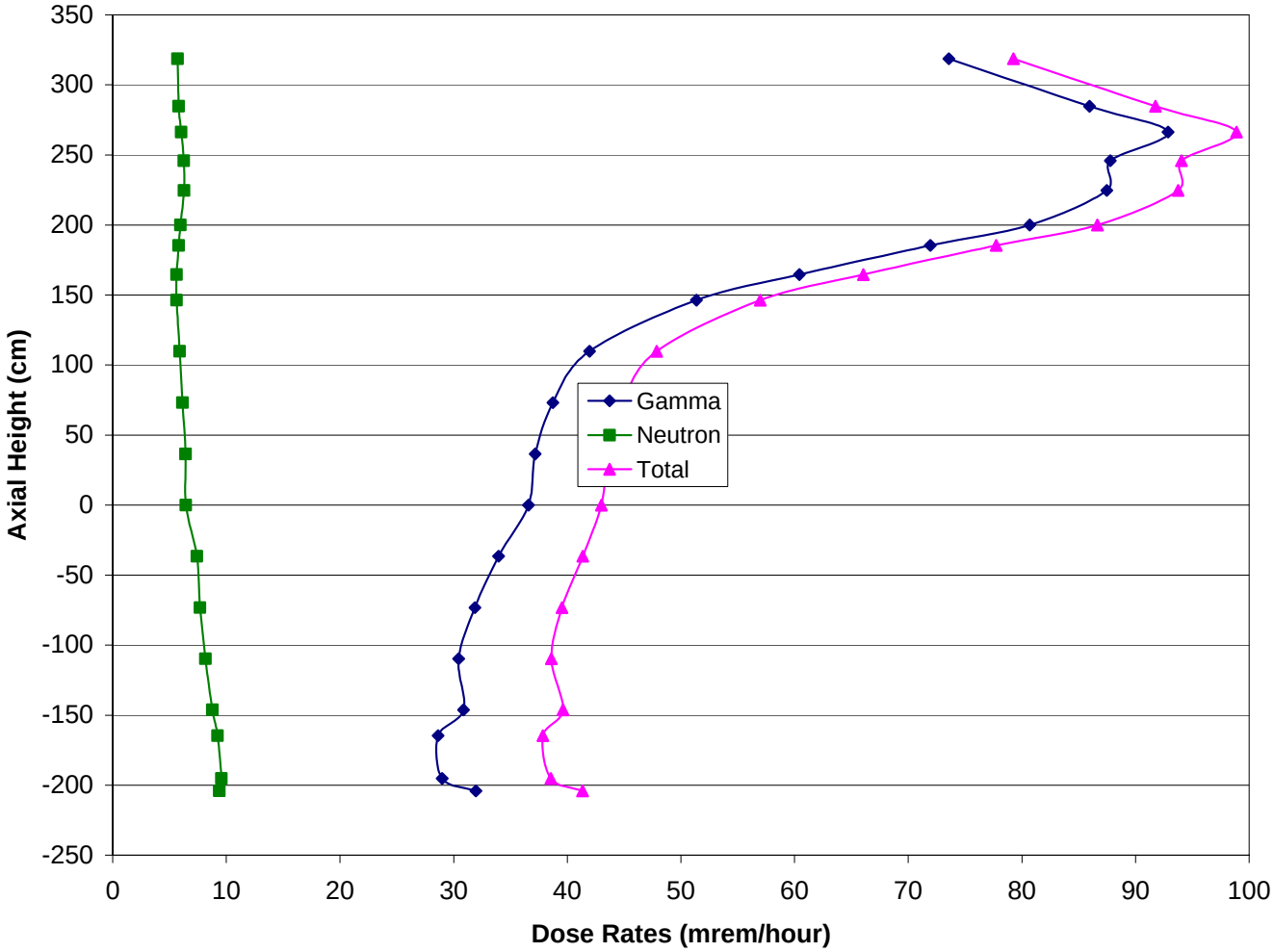


Figure 5.4-3
TN-68 Cask Dose Rates 2 m from Side Surface
Normal and Off-Normal Conditions
(Without Auxiliary Shield Ring)

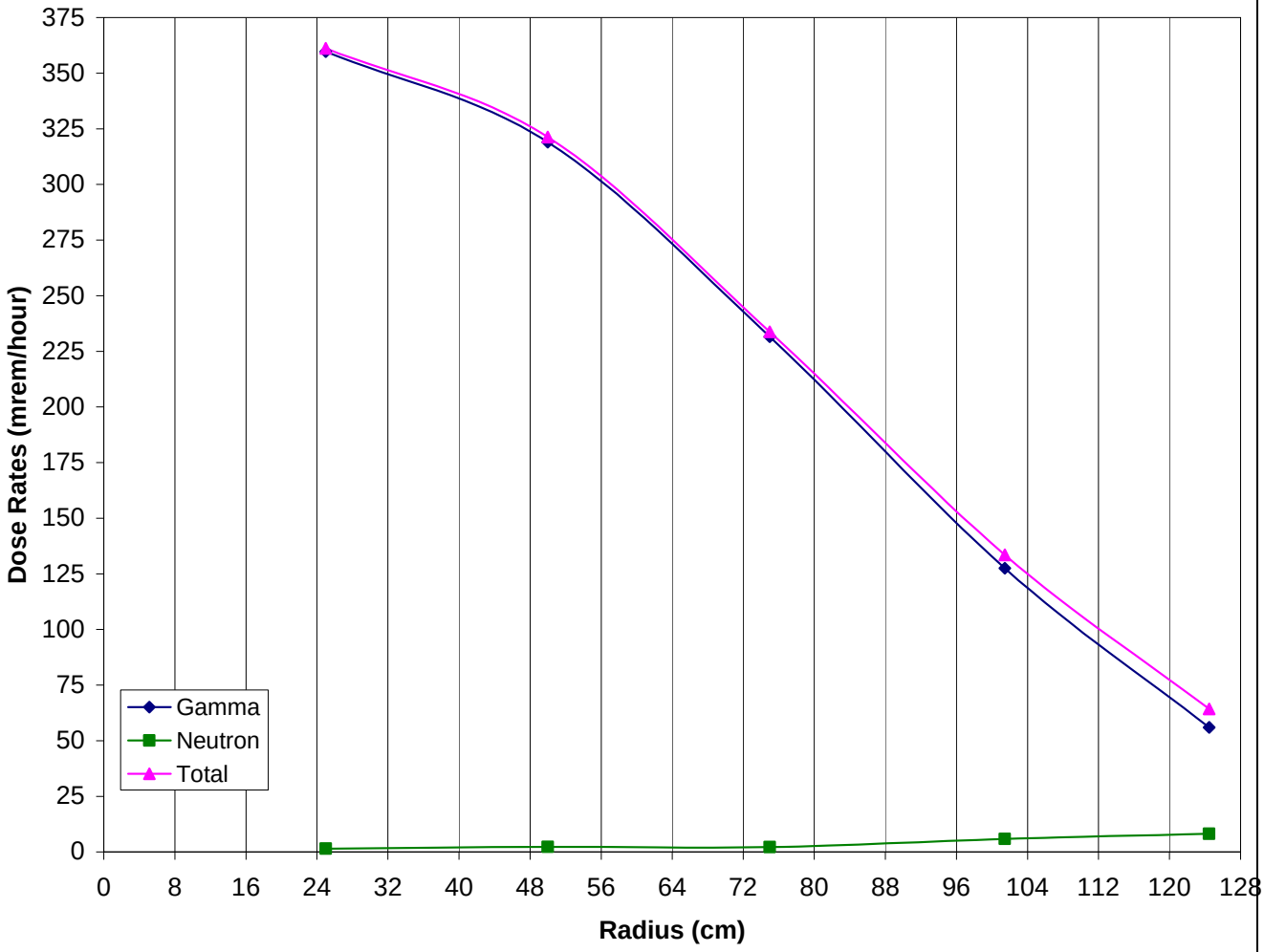


Figure 5.4-4
TN-68 Cask Top Surface Dose Rates
Normal and Off-Normal Conditions

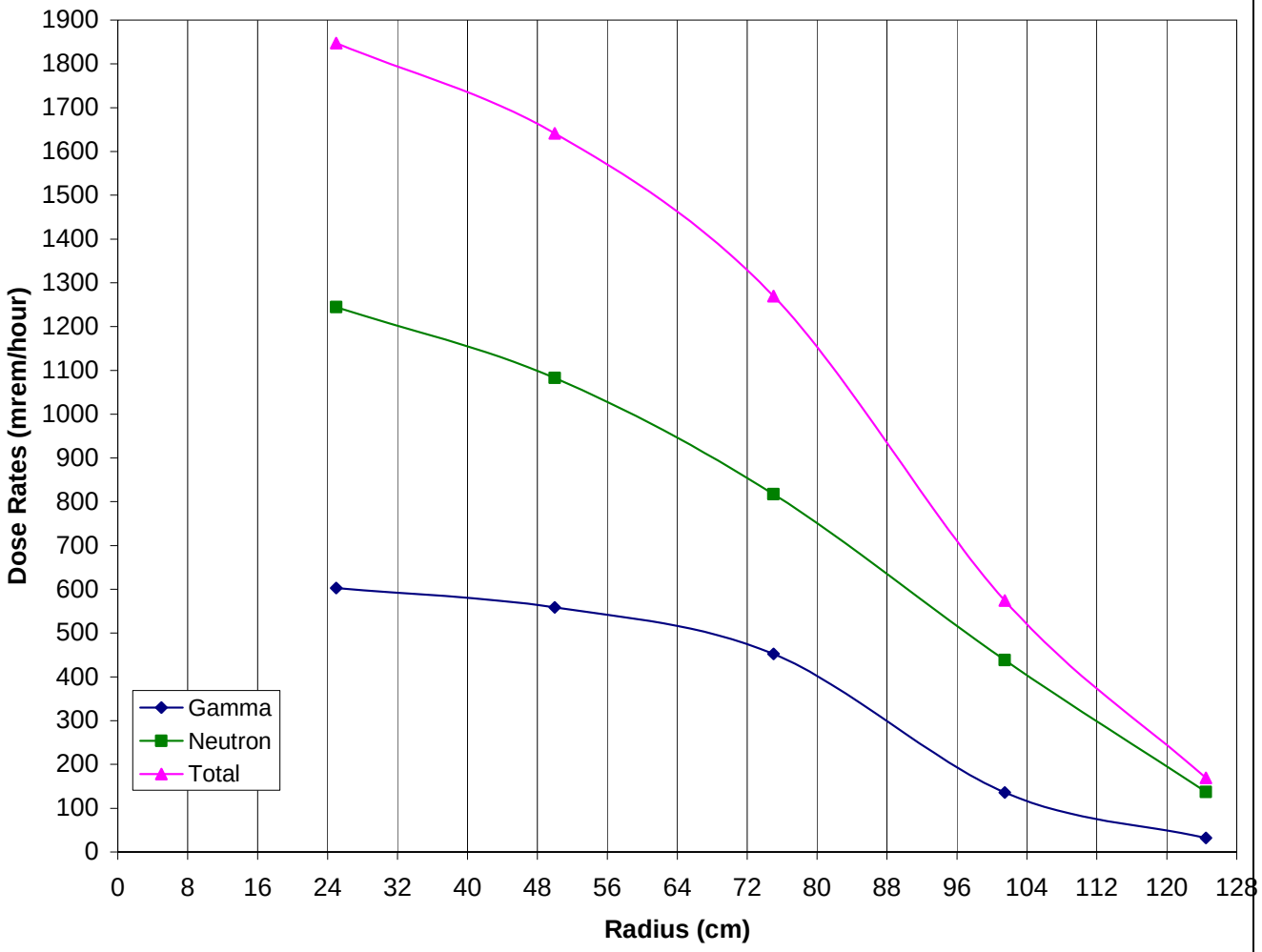


Figure 5.4-5
TN-68 Cask Bottom Surface Dose Rates
Normal and Off-Normal Conditions

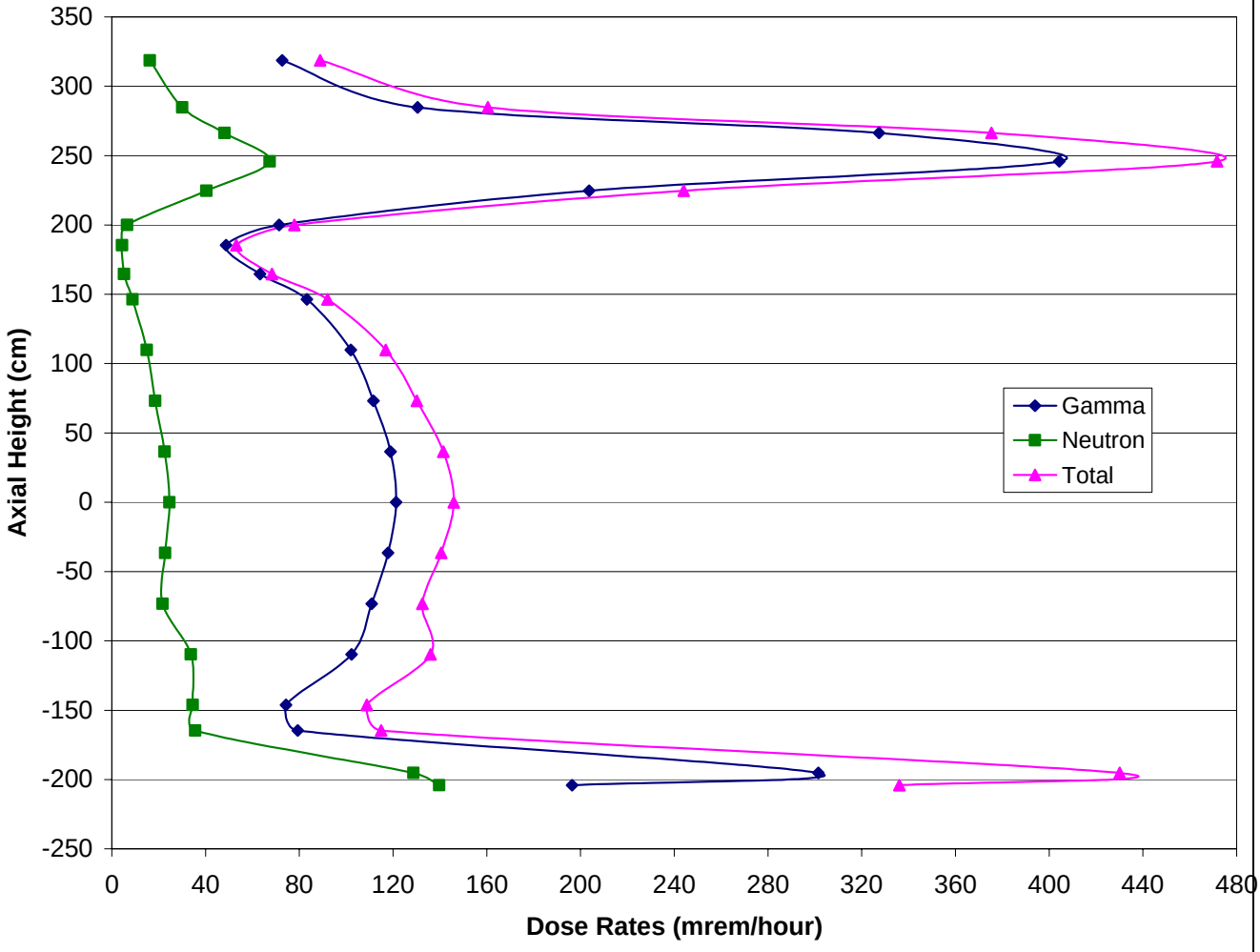


Figure 5.4-6
TN-68 Cask Side Surface Dose Rates
Normal and Off-Normal Conditions
(With 1″ Auxiliary Shield Ring)

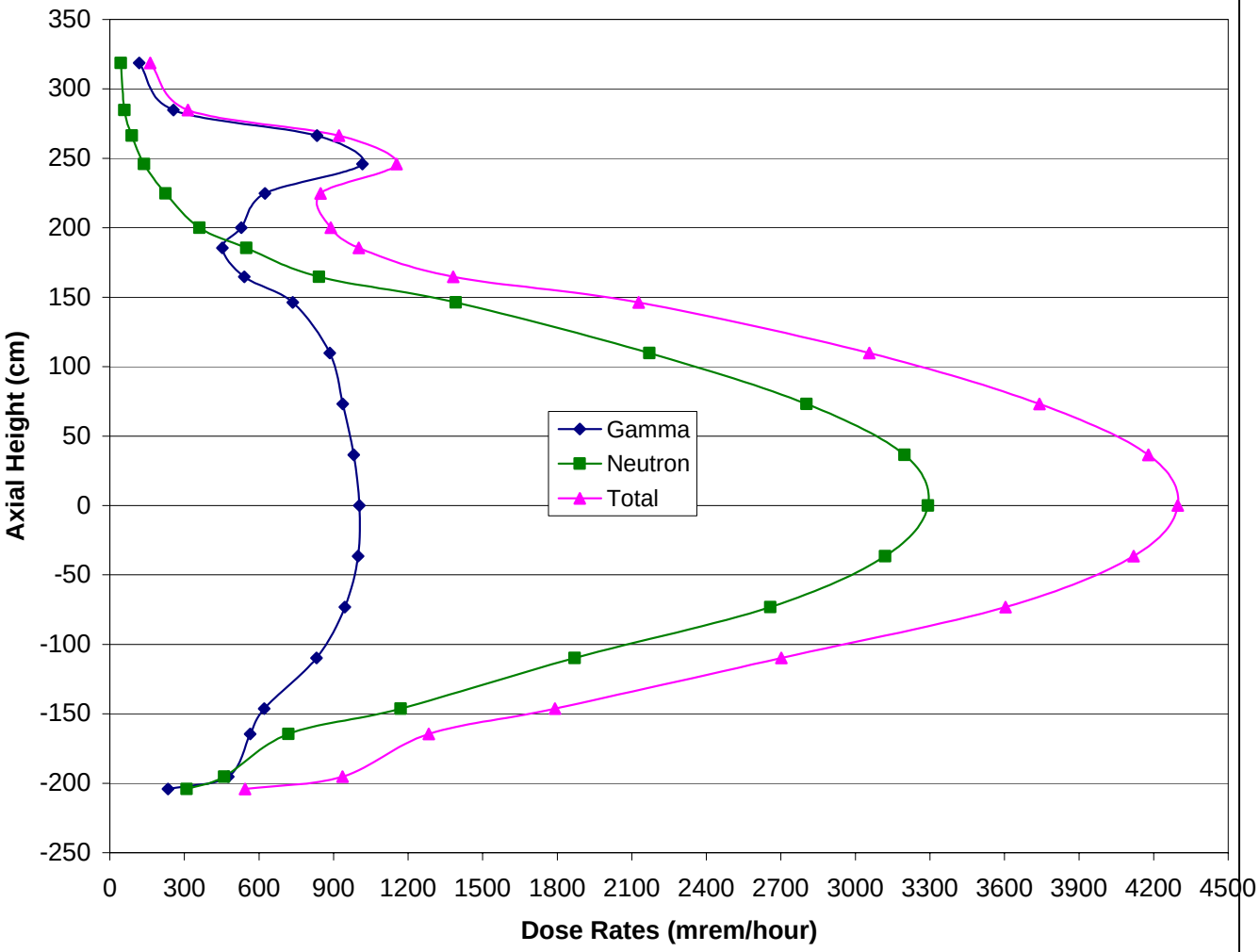


Figure 5.4-7
TN-68 Cask Side Surface Dose Rates – Accident Conditions

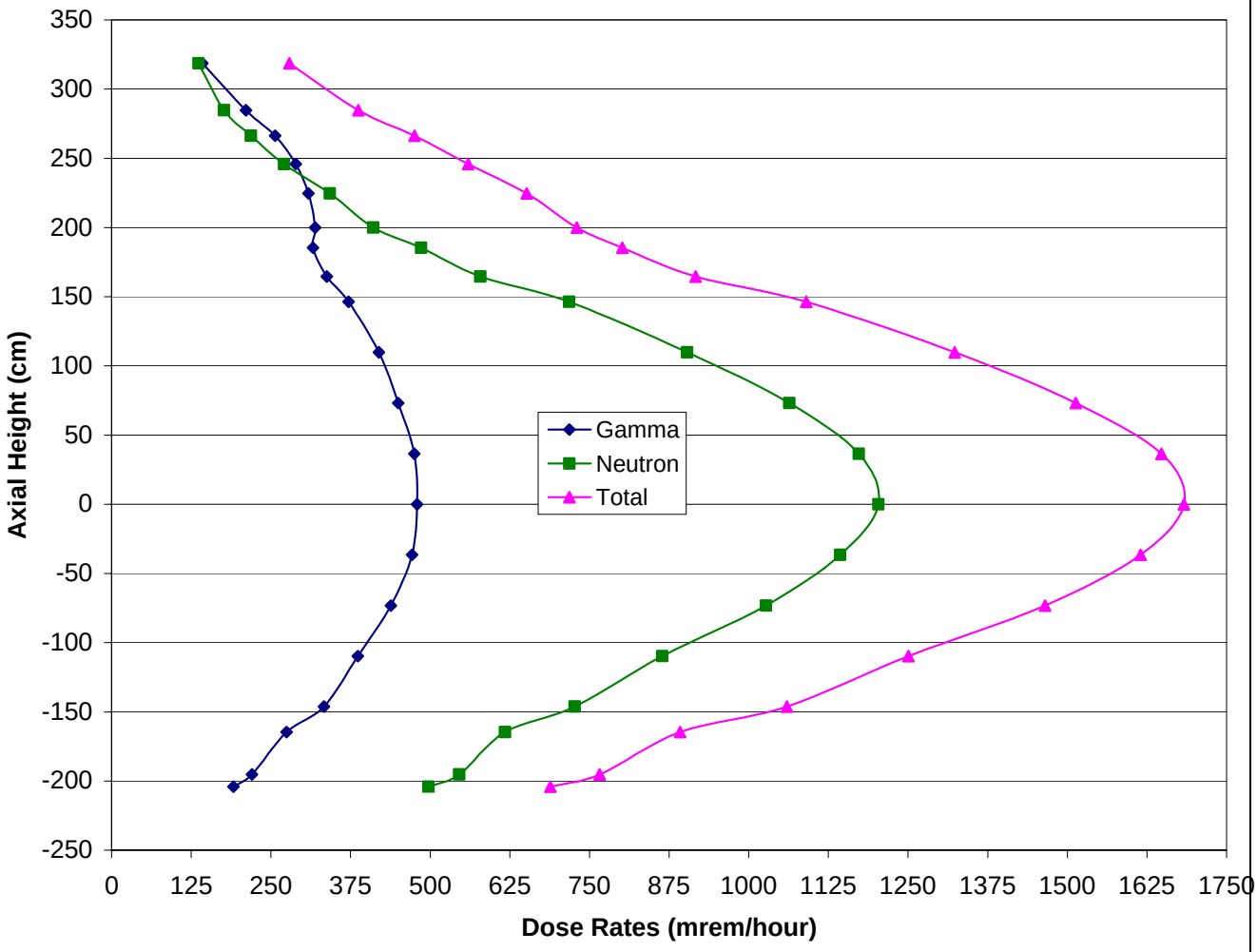


Figure 5.4-8
TN-68 Cask Dose Rates 1 m from Side Surface – Accident Conditions

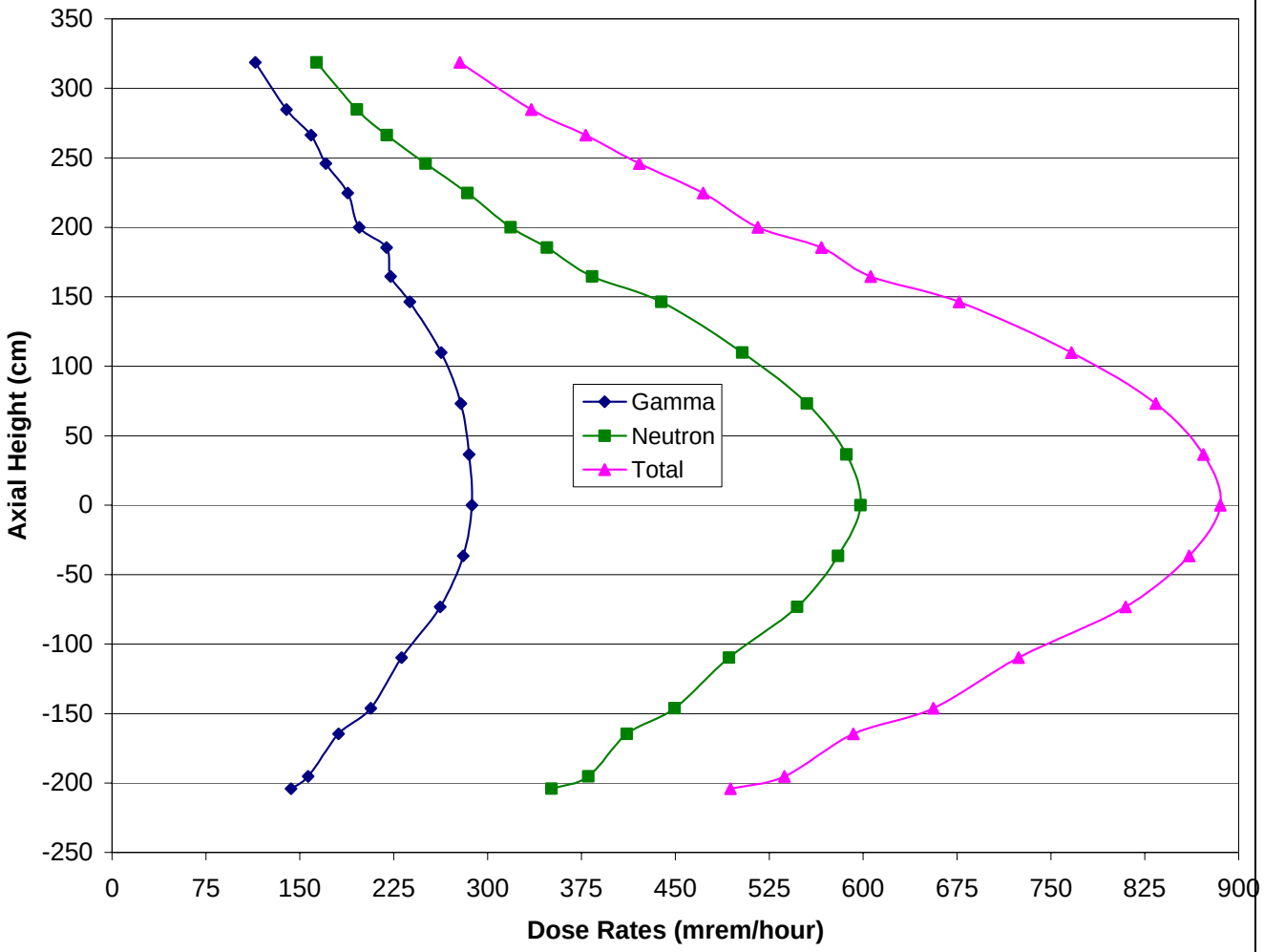


Figure 5.4-9
TN-68 Cask Dose Rates 2 m from Side Surface – Accident Conditions

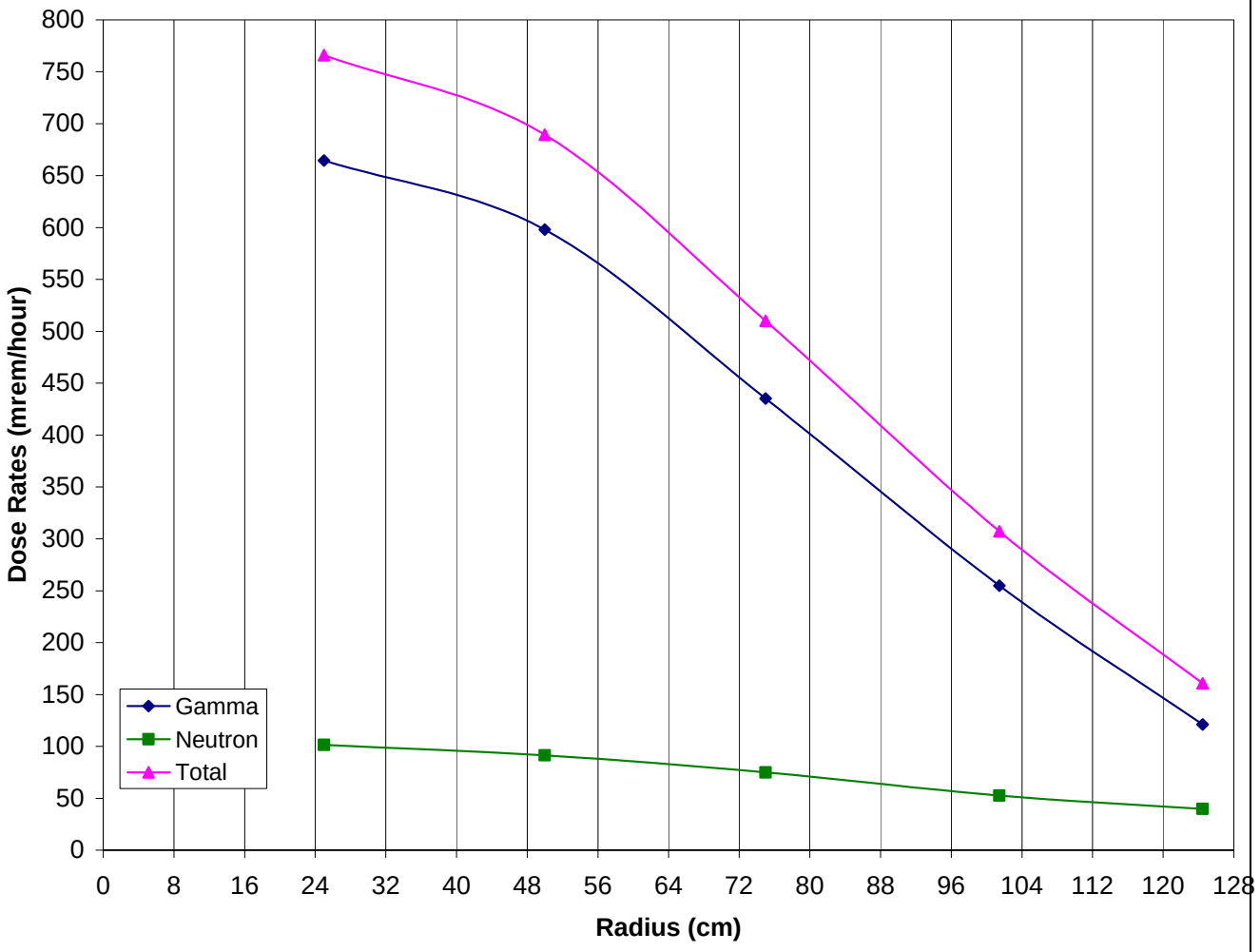


Figure 5.4-10
TN-68 Cask Top Surface Dose Rates – Accident Conditions

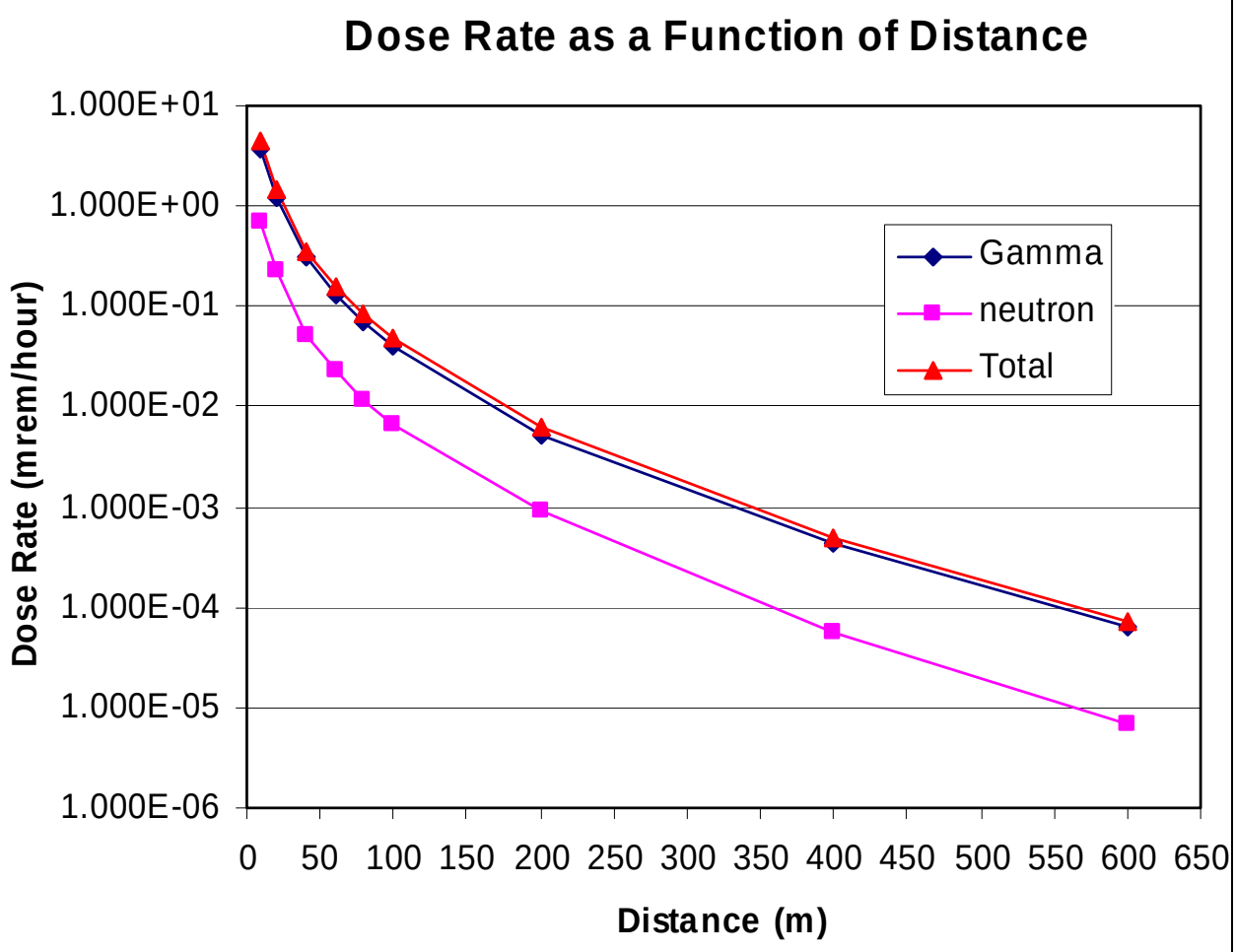


Figure 5.4-11
TN-68 Cask – Dose Rate as a Function Of Distance