AREVA	Form 3.2-1 Calculation Cover Sheet TIP 3.2 (Revision 2)[1]	Calculation No.:	P04F.M2.02-01
		Revision No.:	0
		Page 1 of 49	

DCR NO (if applicable) : NA	PROJECT NAME: HalfPACT Shielded Container
PROJECT NO: P04F.M2.02	CLIENT: Washington TRU Solutions

CALCULATION TITLE: HalfPACT Shielded Container Nuclide Activity Limits

SUMMARY DESCRIPTION:

1) Calculation Summary: This calculation presents activity limits for radioisotopes that may be shipped in the HalfPACT shipping package.

2) Storage Media Description: One compact disc.

If original issue, is licensing review per TIP 3.5 required?

Yes ☐ No ☒ (explain below) Licensing Review No.: _____

This is not a Part 72 project. Also, PacTec is not responsible for the HalfPACT license.

Software Utilized: MCNP5	Version: 1.40
Calculation is complete: Originator Name and Signature: B Day (Gamma analysis) BRAD DAY	Date: 12/6/07
Originator Name and Signature: RJ Migliore (Neutron analysis, combined results) Rick Migliore	Date: 11/7/07
Calculation has been checked for consistency, completeness and correctness: Checker Name and Signature: M Mason Michael Mason	Date: 11/7/07
Calculation is approved for use: Project Engineer Name and Signature: GL Clark Gary Clark	Date: 12/18/07

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1.0 Introduction

The HalfPACT package is used to transport contact-handled transuranic waste. The license is being modified to allow three shielded containers filled with waste of potentially high activity. The purpose of this calculation is to determine the activity limit for each isotope of interest that results in a dose rate of 1000 mrem/hr at a distance of 1 m from the package surface subsequent to an accident. MCNP v1.40 [4] is used for the neutron analysis, and a simple point-kernel approach is used for the gamma analysis.

2.0 Methodology

The methodology follows the established methodology used to determine the nuclide activity limits in the RH-TRU 72B SAR [2]. A list of common nuclides that may be present in the waste is summarized in Table 1. Each nuclide may be gamma emitting, neutron emitting, or both.

The objective is to compute a gamma activity limit, A_G, and a neutron activity limit, A_N, for each isotope of interest so that the hypothetical accident condition (HAC) dose rate limit of 1000 mrem/hr at a distance of 1-m from the package surface cannot be exceeded. If a nuclide is both a gamma and neutron emitter, a combined activity limit A_{GN} is computed as $1/[(1/A_G)+(1/A_N)]$. All radionuclides that do not have gamma energies and do not undergo spontaneous fission or whose maximum allowable activity is calculated to be greater than 1×10^8 curies are classified as "unlimited."

A HalfPACT package may transport up to three shielded containers (SC). It is conservatively assumed under HAC that the radionuclides contained within each SC reconfigure into a point source. These point sources are then assumed to migrate inside of each SC to the closest possible location to the damaged surface of the HalfPACT package. The maximum dose rate will occur when the distance from the sources to the 1-m location is minimized.

Under HAC, the HalfPACT package may impact either in the radial or axial direction. The direction of impact determines the assumed location of the point sources. Because the shielding characteristics and geometry are different in the radial and axial directions, the computed activity limits will depend on the assumed impact direction. The most restrictive radial or axial activity limit is selected as the final limit for each radionuclide.

Once the most restrictive combined neutron and gamma activities A_{GN} are known, the sum of activity partial fractions for any combination of the radionuclides for each SC must be less than or equal to unity, or:

$$\sum_{i=1}^{n} \frac{a_i}{A_{GN_i}} \leq 1$$

where, for a particular payload mix, a_i is the actual curie content of radionuclide "i" and A_{GNi} is the limiting curie content of radionuclide "i".

2.1 Neutron Methodology

For the neutron emitting isotopes, MCNP5 v1.40 [4] is used to compute the dose rates for each isotope of interest. Explicit radial and axial impact models are developed, and the dose rate is computed at 1-m from the surface of the damaged package. ANSI/ANS-6.1.1-1977 flux-to-dose rate conversion factors are used in the calculations [6]. These factors for neutrons are provided in Table 2.

The neutron source is computed in ED-042 [3] for the isotopes of interest. The total neutron spectrum is provided in Table 3 and Table 5 of ED-042. Each spectrum is entered in MCNP as a histogram distribution. Note that the lower energy bound is conservatively set at 0.1 MeV in MCNP, although the lower energy bound is 0 MeV in ED-042. For convenience, the total source strength is set to 1 n/s in all MCNP models.

Note that data for Am-242m are provided for two metastable states (0.048 and 2.2 MeV) in ED-042, although only the 0.048 MeV source is utilized in the current calculation. The excluded 2.2 MeV Am-242m isotope has a half life of only 14 ms (see Table 1 of ED-042) and therefore will not be present in the waste.

No fissile material is included in the models to minimize self-shielding. Because no fissile material is included, no subcritical neutron multiplication is performed by MCNP. It has been determined in conjunction with the RH-TRU 72B SAR that a conservative factor of 2.7 may be utilized to account for subcritical neutron multiplication (see Section 5.5.2 of [2]). This subcritical multiplication factor was determined by surrounding a point source with a sphere comprised of 325 g Pu-239 and a 30% polyethylene/70% water mixture. The U-238 spectrum, which has the lowest average energy of the isotopes under consideration and hence highest subcritical multiplication factor, was conservatively utilized for the point source in that analysis. Given the similarity between the RH-TRU 72B and HalfPACT package payloads, this subcritical neutron multiplication factor is applicable to the current analysis.

Once the neutron dose rate is known for a 1 n/s source, the source strength may be scaled up to the limit of 1000 mrem/hr at 1-m by utilizing the total neutron source strength (per gram) and the specific activity of the nuclide (Ci/g). The total neutron source strength (n/s/g) is provided in Table 2 and Table 4 of ED-042. The details of this calculation are shown in Section 4.1.

2.2 Gamma Methodology

Because of the large number of gamma emitting isotopes (169), the two impact geometries (i.e., radial and axial), and the thick gamma shielding of the SC and HalfPACT package, explicit computation of the activity limits using MCNP for the gamma emitting isotopes is prohibitively expensive. Rather, a simplified point-kernel approach is utilized to produce results that are conservative in comparison to those obtained from a full Monte Carlo analysis.

The dose rate from an isotropic point-source, as a function of gamma energy, is simply [9]:

$$D(E) = \frac{S(E)B(E,x)C(E)}{4\pi R^2} e^{-x}$$

where D(E) is the dose rate as a function of gamma energy, E (mrem/hr), S(E) is the source strength as a function of gamma energy, E (γ/s; 1 curie = $3.7(10)^{10}$ γ/s), B(E,x) is the buildup

factor as a function of gamma energy, E, and mean free paths, x(E), C(E) is the gamma flux-to-dose rate conversion factor (γ/cm²-s to mrem/hr), and R is the radial distance from the source to the receptor (cm).

Gamma energy and intensity data is provided in Kinsey, et al [8]. All gamma energies less than 0.100 MeV are conservatively rounded to 0.100 MeV for the purpose of linear interpolation in Table 3, Table 4, and Table 5.

The number of mean free paths, x(E), is:

$$x(E) = \sum_{i=1}^{n} \mu_i(E) t_i$$

where $\mu_i(E)$ is the attenuation coefficient (as a function of gamma energy) of shield "i", and t_i is the thickness of shield "i" in centimeters. The number of mean free paths is a non-dimensional number.

The attenuation coefficient for shield "i" is:

$$\mu_i(E) = \left[\frac{\mu_1(E)}{\rho}\right]\rho_1 + \left[\frac{\mu_2(E)}{\rho}\right]\rho_2 + \left[\frac{\mu_3(E)}{\rho}\right]\rho_3 + \cdots + \left[\frac{\mu_n(E)}{\rho}\right]\rho_n$$

where, for a material containing "n" different elements, $\mu_x(E)$ and ρ_x are the mass attenuation coefficient (as a function of gamma energy) and partial density for element "x." The units for the attenuation coefficient are cm^{-1}. The units for mass attenuation coefficient are cm²/g. Table 7 presents a summary of the partial density for each major element of the three gamma attenuation materials (i.e., carbon steel, Type 304 stainless steel, and lead). Table 4 summarizes the mass attenuation coefficients for each element as a function of gamma energy, as taken from ANSI/ANS 6.4.3-1991 [10]. Mass attenuation coefficients used in the subsequent shielding calculations are linearly interpolated from the data in Table 4.

Gamma-ray isotropic point-source buildup factors are determined by conservatively assuming iron as the dominant shielding material. Although the actual buildup factors will lie somewhere between iron (atomic number, Z = 26) and lead (Z = 82), use of iron as the buildup factor will conservatively bound the maximum isotopic quantity (curies) allowed for transport because the buildup factor increases as the atomic number decreases.

Buildup factors are determined using the geometric progression (G-P) function as presented in ANSI/ANS 6.4.3-1991 [10]. The G-P function accurately reproduces buildup factor data for deep penetrations in shields (i.e., >20 mean free paths thick). The buildup factor, as a function of gamma energy, E, and mean free paths, x, using the G-P function is:

$$B(E,x) = 1 + (b-1)\frac{(K^x - 1)}{(K-1)} \quad \text{for } K \neq 1$$

$$B(E,x) = 1 + (b-1)x \quad \text{for } K = 1$$

$$K(x) = cx^{a} + d\left[\frac{\tanh\left(\frac{x}{X_k} - 2\right) - \tanh(-2)}{1 - \tanh(-2)}\right]$$

The coefficients a, b, c, d, and X_K, as a function of gamma energy, are provided in Table 5. These coefficients are linearly interpolated for a given gamma energy.

Gamma flux-to-dose rate conversion factors are determined using the values delineated in ANSI/ANS 6.1.1-1977 [6], and presented in Table 3. The data in Table 3 are linearly interpolated to determine the conversion factor for a given gamma energy.

Examples of the gamma dose rate calculations are presented in Section 5.3 and 5.4 for radial and axial geometry, respectively, for Co-60.

Table 1 – Shielded Container Radionuclide Inventory

^{3}H	^{90m}Zr	^{123m}Te	^{160}Tb	^{223}Fr	^{236}Pu
^{10}Be	^{93}Zr	^{125m}Te	^{166m}Ho	^{223}Ra	^{238}Pu
^{14}C	^{95}Zr	^{127}Te	^{168}Tm	^{224}Ra	^{239}Pu
^{22}Na	^{93m}Nb	^{127m}Te	^{182}Ta	^{225}Ra	^{240}Pu
^{32}P	^{94}Nb	^{129}Te	^{198}Au	^{226}Ra	^{241}Pu
^{33}P	^{95}Nb	^{129m}Te	^{207}Tl	^{228}Ra	^{242}Pu
^{35}S	^{95m}Nb	^{125}I	^{208}Tl	^{225}Ac	^{243}Pu
^{45}Ca	^{99}Tc	^{129}I	^{209}Tl	^{227}Ac	^{244}Pu
^{46}Sc	^{99m}Tc	^{131}I	^{209}Pb	^{228}Ac	^{241}Am
^{49}V	^{103}Ru	^{134}Cs	^{210}Pb	^{227}Th	^{242}Am
^{51}Cr	^{106}Ru	^{135}Cs	^{211}Pb	^{228}Th	^{242m}Am
^{54}Mn	^{103m}Rh	^{137}Cs	^{212}Pb	^{229}Th	^{243}Am
^{55}Fe	^{106}Rh	^{133}Ba	^{214}Pb	^{230}Th	^{245}Am
^{59}Fe	^{107}Pd	^{137}Ba	^{207}Bi	^{231}Th	^{240}Cm
^{57}Co	^{108}Ag	^{137m}Ba	^{210}Bi	^{232}Th	^{242}Cm
^{58}Co	^{108m}Ag	^{141}Ce	^{211}Bi	^{234}Th	^{243}Cm
^{60}Co	^{109m}Ag	^{142}Ce	^{212}Bi	^{231}Pa	^{244}Cm
^{59}Ni	^{110}Ag	^{144}Ce	^{213}Bi	^{233}Pa	^{245}Cm
^{63}Ni	^{110m}Ag	^{143}Pr	^{214}Bi	^{234}Pa	^{246}Cm
^{64}Cu	^{109}Cd	^{144}Pr	^{209}Po	^{234m}Pa	^{247}Cm
^{65}Zn	^{113m}Cd	^{144m}Pr	^{210}Po	^{232}U	^{248}Cm
^{73}As	^{115m}Cd	^{146}Pm	^{211}Po	^{233}U	^{250}Cm
^{79}Se	^{114}In	^{147}Pm	^{212}Po	^{234}U	^{247}Bk
^{85}Kr	^{114m}In	^{148}Pm	^{213}Po	^{235}U	^{249}Bk
^{86}Rb	^{115m}In	^{148m}Pm	^{214}Po	^{236}U	^{250}Bk
^{87}Rb	^{119m}Sn	^{146}Sm	^{215}Po	^{237}U	^{249}Cf
^{89}Sr	^{121m}Sn	^{147}Sm	^{216}Po	^{238}U	^{250}Cf
^{90}Sr	^{123}Sn	^{151}Sm	^{218}Po	^{239}U	^{251}Cf
^{88}Y	^{126}Sn	^{150}Eu	^{211}At	^{240}U	^{252}Cf
^{90}Y	^{124}Sb	^{152}Eu	^{217}At	^{237}Np	^{254}Cf
^{90m}Y	^{125}Sb	^{154}Eu	^{219}Rn	^{238}Np	^{252}Es
^{91}Y	^{126}Sb	^{155}Eu	^{220}Rn	^{239}Np	^{253}Es
^{88}Zr	^{126m}Sb	^{152}Gd	^{222}Rn	^{240}Np	^{254}Es
^{90}Zr	^{123}Te	^{153}Gd	^{221}Fr	^{240m}Np	^{254m}Es

Table 2 – Neutron Flux-to-Dose Rate Conversion Factors [6]

Energy (MeV)	C(E) (n/cm²-s to mrem/hr)	Energy (MeV)	C(E) (n/cm²-s to mrem/hr)
2.50E-08	3.67E-03	5.00E-01	9.26E-02
1.00E-07	3.67E-03	1.0	1.32E-01
1.00E-06	4.46E-03	2.5	1.25E-01
1.00E-05	4.54E-03	5.0	1.56E-01
1.00E-04	4.18E-03	7.0	1.47E-01
1.00E-03	3.76E-03	10.0	1.47E-01
1.00E-02	3.56E-03	14.0	2.08E-01
1.00E-01	2.17E-02	20.0	2.27E-01

Table 3 – Gamma Flux-to-Dose Rate Conversion Factors [6]

Energy (MeV)	C(E) (γ/cm²-s to mrem/hr)	Energy (MeV)	C(E) (γ/cm²-s to mrem/hr)
0.100	2.83E-04	0.800	1.68E-03
0.150	3.79E-04	1.000	1.98E-03
0.200	5.01E-04	1.400	2.51E-03
0.250	6.31E-04	1.800	2.99E-03
0.300	7.59E-04	2.200	3.42E-03
0.350	8.78E-04	2.600	3.82E-03
0.400	9.85E-04	2.800	4.01E-03
0.450	1.08E-03	3.250	4.41E-03
0.500	1.17E-03	3.750	4.83E-03
0.550	1.27E-03	4.250	5.23E-03
0.600	1.36E-03	4.750	5.60E-03
0.650	1.44E-03	5.000	5.80E-03
0.700	1.52E-03		

Table 4 – Mass Attenuation Coefficients [10]

γ-Energy (MeV)	Silicon	Chromium	Manganese	Iron	Nickel	Lead
0.100	1.73E-01	2.92E-01	3.10E-01	3.43E-01	4.10E-01	5.36E+00
0.150	1.40E-01	1.67E-01	1.72E-01	1.83E-01	2.05E-01	1.92E+00
0.200	1.25E-01	1.31E-01	1.32E-01	1.38E-01	1.49E-01	9.43E-01
0.300	1.07E-01	1.04E-01	1.03E-01	1.06E-01	1.11E-01	3.77E-01
0.400	9.54E-02	9.04E-02	8.95E-02	9.20E-02	9.53E-02	2.17E-01
0.500	8.70E-02	8.17E-02	8.07E-02	8.28E-02	8.55E-02	1.51E-01
0.600	8.04E-02	7.52E-02	7.42E-02	7.61E-02	7.84E-02	1.18E-01
0.800	7.06E-02	6.57E-02	6.49E-02	6.64E-02	6.83E-02	8.47E-02
1.000	6.34E-02	5.90E-02	5.82E-02	5.96E-02	6.12E-02	6.84E-02
1.500	5.17E-02	4.81E-02	4.75E-02	4.86E-02	4.99E-02	5.10E-02
2.000	4.47E-02	4.20E-02	4.15E-02	4.25E-02	4.37E-02	4.54E-02
3.000	3.67E-02	3.55E-02	3.51E-02	3.61E-02	3.73E-02	4.20E-02
4.000	3.23E-02	3.23E-02	3.20E-02	3.30E-02	3.44E-02	4.18E-02
5.000	2.96E-02	3.05E-02	3.04E-02	3.14E-02	3.28E-02	4.26E-02

Table 5 – Iron Exposure Buildup Factor Coefficients [10]

γ-Energy (MeV)	b	c	a	X_K	d
0.100	1.389	0.557	0.144	14.11	-0.0791
0.150	1.660	0.743	0.079	14.12	-0.0476
0.200	1.839	0.911	0.034	13.23	-0.0334
0.300	1.973	1.095	-0.009	11.86	-0.0183
0.400	1.992	1.187	-0.027	10.72	-0.0140
0.500	1.967	1.240	-0.039	8.34	-0.0074
0.600	1.947	1.247	-0.040	8.20	-0.0096
0.800	1.906	1.233	-0.038	7.93	-0.0110
1.000	1.841	1.250	-0.048	19.49	0.0140
1.500	1.750	1.197	-0.040	15.90	0.0110
2.000	1.712	1.123	-0.021	7.97	-0.0057
3.000	1.627	1.059	-0.005	11.99	-0.0132
4.000	1.553	1.026	0.005	12.93	-0.0191
5.000	1.483	1.009	0.012	13.12	-0.0258

3.0 Shielding Model

3.1 Configuration of Source and Shielding

The calculation is concerned only with dose rates under hypothetical accident conditions (HAC). Under normal conditions of transport (NCT), the dose rates will be determined by measurement.

It is assumed that the waste remains within the shielded container in which it is loaded. However, the waste in each shielded container is conservatively assumed to configure into a point source. Treating the source as point sources conservatively neglects self shielding effects.

The HalfPACT package geometric model is based on several conservative assumptions. The assumptions used to developed the damaged geometry model are summarized below.

1. The inner containment vessel (ICV), honeycomb spacer, outer containment vessel (OCV), and outer containment assembly (OCA) dimensions are established from the HalfPACT package SAR drawing nominal dimensions [1], although these dimensions are modified to neglect any diametrical and axial clearances between the ICV and OCV. Torispherical heads are treated as flat ends.
2. The shielded container, axial dunnage, radial dunnage, and pallet dimensions are established from the Shielded Container SAR drawing nominal dimensions.
3. The HalfPACT package OCA crush damage is established from certification unit testing of the HalfPACT package (and TRUPACT-II package for axial drops). The bounding side crush is provided in Table 2.10.3-3 under test #2 as a 37-in wide by 3-3/4-in deep flat [1]. The bounding top crush is provided in Table 2.10.3-1 under test #4 as a ∅53 in. by 3-7/8-in deep flat [11]. The TRUPACT-II package is considered equivalent to the HalfPACT package in the top-down axial drop orientation.
4. The shielded container lid and bottom is modeled as 2.50-in thick and the lead annulus is modeled as 0.85-in thick to account for HAC drop test damage, embedded material defects and tolerance stack-up conditions in the primary shield materials.
5. Pallet damage is assumed to be negligible such that top-down axial drop is limiting over bottom-down axial drop for distance attenuation. Honeycomb spacer damage is assumed as 70% crush to lock-up of the honeycomb materials. Axial and radial dunnage damage is assumed as 90% crush to lock-up of the foam materials.
6. OCA external shells, OCA foam, honeycomb spacers, radial spacers, and payload pallet are neglected from a material attenuation perspective and credited for distance attenuation only.

A sketch of the undamaged configuration is provided in Figure 1. Under HAC, it is assumed the limiting damage occurs when the package is dropped either radially or axially. If the drop is radial, it is assumed that the three shielded containers will shift to one side, as shown in Figure 2. The dose rate location is 1 m from the crushed outer surface of the package. The minimum distance between the source and the dose rate location is 48.5-in. The sources in the other two shielded containers are a minimum of 62.4375-in from the dose rate location. If the drop is axial, the three shielded containers shift to the top of the package, as shown in Figure 3. The source in each shielded container is assumed to cluster at the centerline of the package to maximize the

maximize the dose rate 1 m from the top of the package. In this case, the distance from each source to the dose rate location is 54.9375-in.

Based upon these considerations, both radial and axial MCNP neutron models are developed. The key dimensions utilized in these models are summarized in Table 6. The "index dimensions" noted in this table are the base dimensions from which the region thicknesses are built around. The MCNP model geometry of the shielded container is shown in Figure 4. The MCNP model geometry of the radial and axial configurations is shown in Figure 5 and Figure 6, respectively. Sample radial and axial MCNP input files are provided in Section 5.2.

The radial and axial gamma models are highly simplified and utilize only the thicknesses of the non-void regions listed in Table 6 and the total source to detector distances noted above.

Table 6 – Key MCNP Model Dimensions (inches)

Shielded Container [7]	
Outer Diameter	23.0
Overall Height (35.75 on [7])	35.25[1]
Inner Shell Thickness (carbon steel)	0.179
Lead Thickness (bounds value from [7])	0.85
Outer Shell Thickness (carbon steel)	0.120
Lead Penetration into Lid (approximate)	0.625
Lid Thickness (carbon steel) (bounds value from [7])	2.5
HalfPACT Package (radial model)	
Compressed Aluminum and Foam Thickness (void)	1.0625
ICV Inner Diameter (index dimension)	72.625
ICV Shell Thickness (stainless steel)	0.25
OCV Shell Thickness (stainless steel)	0.1875
OCA Annulus (void)	6.25
OCA Shell (void)	0.25
HalfPACT Package (axial model)	
Compressed Aluminum and Foam Thickness (void)	0.5
ICV Height (index dimension)	44.75
Honeycomb Spacer (void)	3.75
ICV Upper Head (stainless steel)	0.25
OCV Upper Head (stainless steel)	0.25
OCA Annulus (void)	7.875
OCA Shell (void)	0.25

[1] The overall height of the shielded container is 35.75-in. If the overall height is reduced by 1-in to account for the assumed reduction in shield thickness at the ends, the overall height reduces to 34.75-in. A height of 35.25-in is used in the MCNP models, although this discrepancy has no impact on the calculation.

Figure Withheld Under 10 CFR 2.390

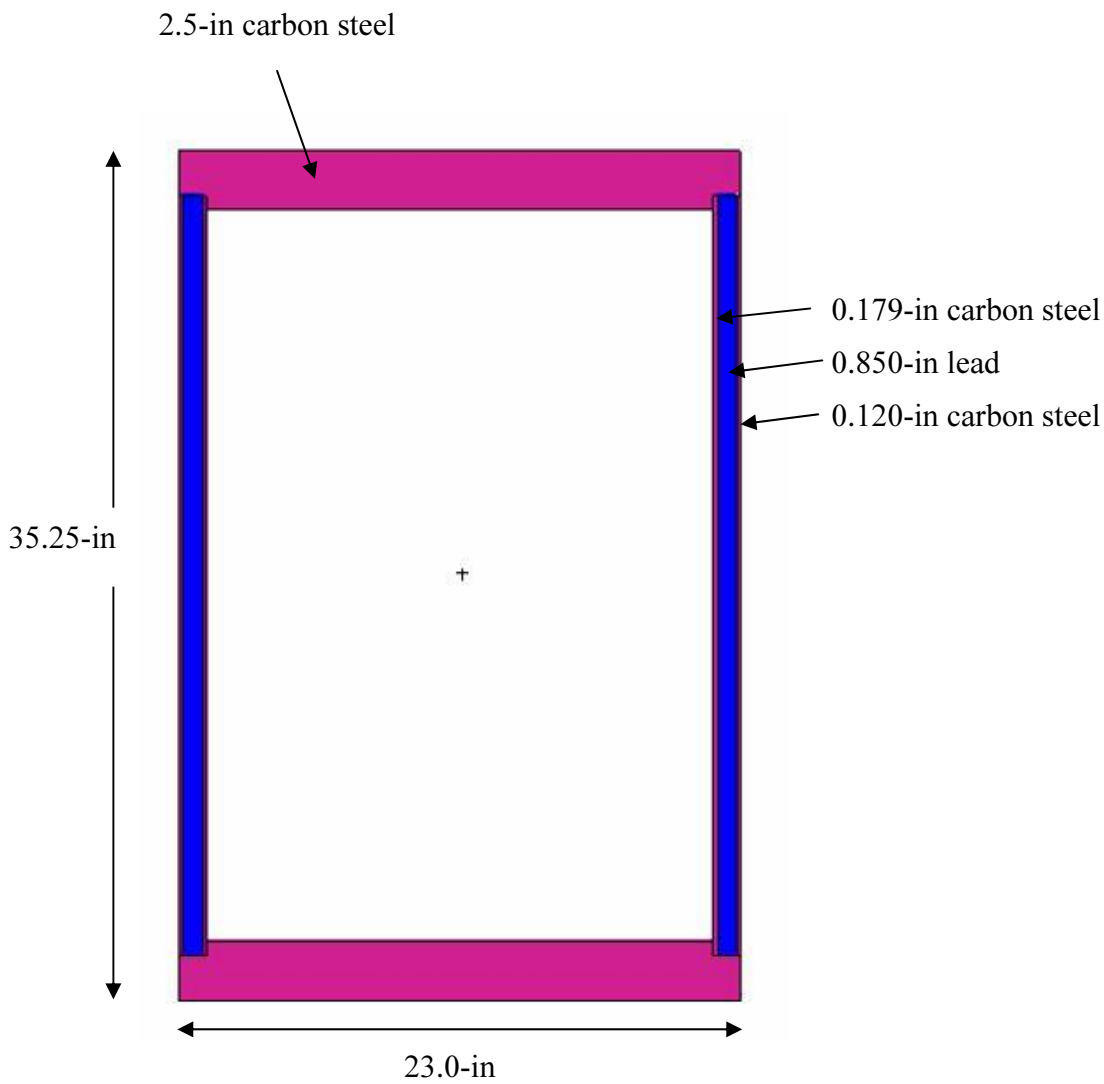
Figure 1 – Preshipment Geometry

Figure Withheld Under 10 CFR 2.390

Figure 2 – HAC Radial Geometry

Figure Withheld Under 10 CFR 2.390

Figure 3 – HAC Axial Geometry



Bottom modeled symmetrical to the top for convenience.

Figure 4 – MCNP Shielded Container Model

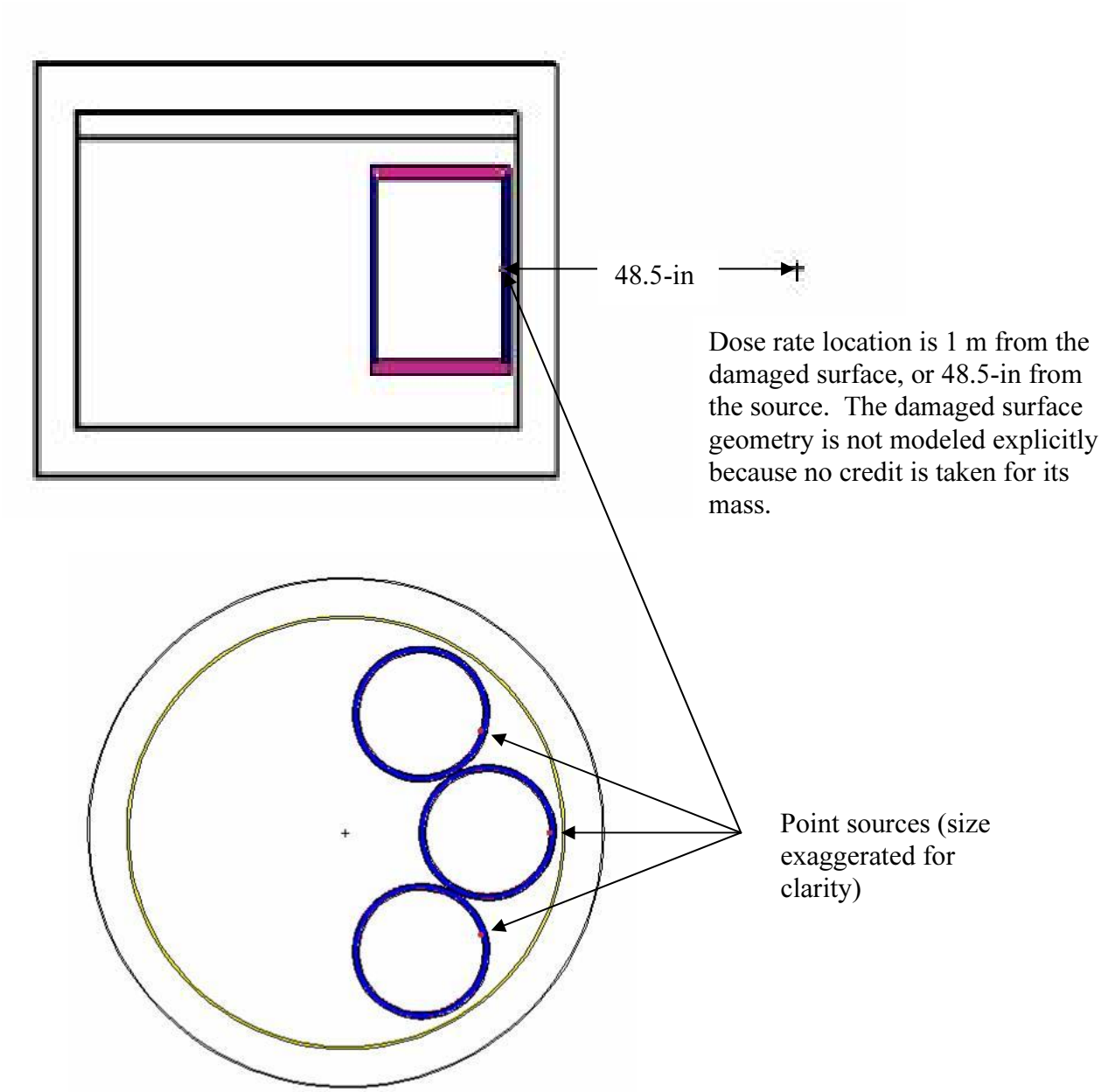


Figure 5 – MCNP Radial Model

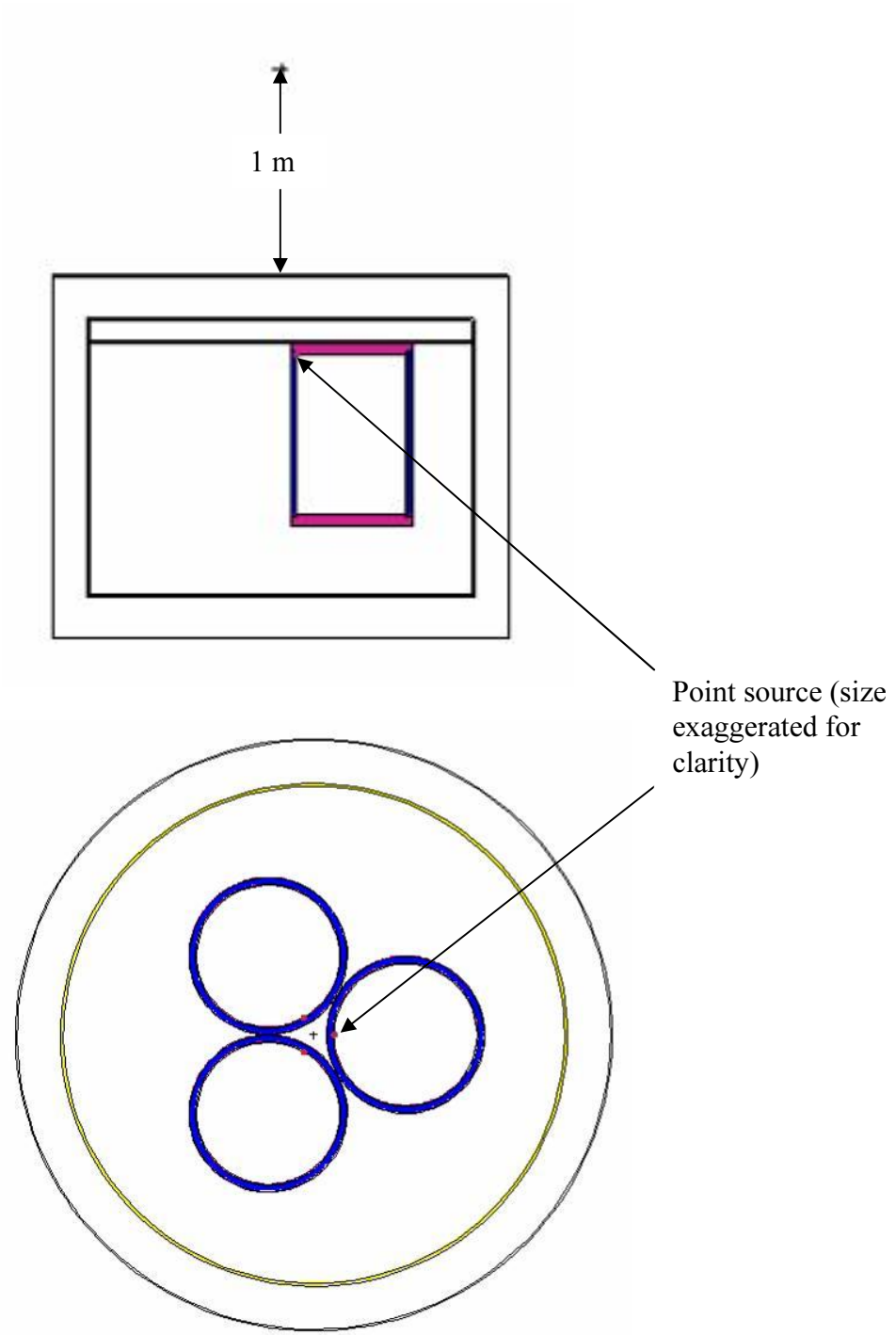


Figure 6 – MCNP Axial Model

3.2 Material Properties

Three materials are used in the analysis: lead, carbon steel, and stainless steel. The composition and densities are selected to be consistent with the RH-TRU 72B SAR [2]. Lead is modeled with a density of 11.35 g/cm^3. Carbon steel is modeled as pure iron with a density of 7.8526 g/cm^3. Stainless steel is modeled with a density of 8.0128 g/cm^3. The shield regional densities for all shield materials are summarized in Table 7. These densities are used in both the point-kernel and MCNP analyses.

Table 7 – Summary of Shield Regional Densities

	Carbon Steel		Type 304 Stainless Steel		Lead	
Element	**Percent**	**Partial Density (g/cc)**	**Percent**	**Partial Density (g/cc)**	**Percent**	**Partial Density (g/cc)**
Silicon	—	—	1%	0.0801	—	—
Chromium	—	—	19%	1.5224	—	—
Manganese	—	—	2%	0.1603	—	—
Iron	100%	7.8526	68%	5.4487	—	—
Nickel	—	—	10%	0.8013	—	—
Lead	—	—	—	—	100%	11.3500
Total	*100%*	*7.8526*	*100%*	*8.0128*	*100%*	*11.3500*

4.0 Results

4.1 Neutron Results

The neutron activity limits for the radial and axial models are provided in Table 8 and Table 9, respectively. The MCNP statistical uncertainty is ≤ 0.15% in all cases, indicating a well-converged solution. The three point sources in the model have a total combined source strength of 1 n/s. Therefore, the dose rate reported in Column B of Table 8 and Table 9 is very small.

To determine the radioisotope mass to achieve a 1 m dose rate of 1000 mrem/hr, the limiting dose rate is divided by the dose rate for a 1 n/s source (including the subcritical neutron multiplication factor of 2.7), and by the total neutron source strength for 1 g of material. This mass (which is a total mass for all 3 sources) is provided in Column C of Table 8 and Table 9. To convert to an activity per shielded container, this total mass is multiplied by the specific activity and divided by 3. The final result is provided in Column E of Table 8 and Table 9.

Comparison of the radial and axial neutron results indicates that the radial activity limits bound the axial activity limits for all radioisotopes.

4.2 Gamma Results

The gamma activity limits are determined by application of the point-kernel model as implemented in MathCAD Version 13.1. Activity limits are determined both for radial and axial models. The MathCAD application considers all three point sources, although the final results are reported per source, not per package. The results are reported in Table 10. For convenience, this table includes all isotopes, including non-gamma emitters, which are assigned an allowable activity of 1.000E99 Ci. Note that the limiting activity may be from either the radial or axial model, as the nuclides have a widely varying energy spectrum.

Example radial and axial calculations are provided in Section 5.3 and 5.4, respectively, for Co-60.

4.3 Combined Results

For isotopes that are both gamma and neutron emitters, the total activity limit is determined by combining the gamma and neutron results using the following equation: $1/[(1/A_G)+(1/A_N)]$, where the combined limit is determined for both the radial and axial configurations. The radial and axial limits are then compared to determine the limiting activity for each radionuclide and reported in Table 11.

The sum of dose rate partial fractions for any combination of the radionuclides must be less than or equal to unity, or:

$$\sum_{i=1}^{n} \frac{a_i}{A_{GN_i}} \leq 1$$

where, for a particular payload mix, a_i is the actual curie content of radionuclide "i" and A_{GNi} is the limiting curie content of radionuclide "i" given in Table 11.

For example, 1.609 curies of ^{60}Co, 8.821 curies of ^{95}Zr, and 446.4 curies of ^{242}Pu each result in a limiting HAC dose rate of 1000 mrem/hr at a distance of 1-m from the package surface. The sum of the partial fractions for a payload containing 1.0 curies of ^{60}Co, 1.0 curies of ^{95}Zr, and 50 curies of ^{242}Pu is:

$$\frac{a_{60_{Co}}}{A_{GN_{60_{Co}}}} + \frac{a_{95_{Zr}}}{A_{GN_{95_{Zr}}}} + \frac{a_{242_{Pu}}}{A_{GN_{242_{Pu}}}} = \frac{1}{1.609} + \frac{1}{8.821} + \frac{50}{446.4} = 0.62 + 0.11 + 0.11 = 0.85 \leq 1.00$$

Thus, the combination of isotopes for the above example will not exceed the HAC limiting dose rate.

Table 8 – Neutron Activity Calculations, Radial Model

	A	B		C = 1000/(2.7*A*B)	D	E = C*D/3
Radio-nuclide	**Source Strength (n/s/g) [3]**	**Dose Rate from 1 n/s source (mrem/hr)**	**σ**	**Total Mass for 1000 mrem/hr at 1 Meter (g)**	**Specific Activity (Ci/g) [2]**	**Activity per Shielded Container for 1000 mrem/hr at 1 Meter, (Ci), A_N**
^{230}Th	9.13E+00	6.67E-07	0.12%	6.08E+07	2.10E-02	4.26E+05
^{232}Th	2.28E-05	6.72E-07	0.12%	2.42E+13	1.10E-07	8.87E+05
^{231}Pa	2.64E+01	6.66E-07	0.12%	2.11E+07	4.70E-02	3.30E+05
^{232}U	1.50E+04	6.68E-07	0.12%	3.70E+04	2.20E+01	2.71E+05
^{233}U	4.83E+00	6.66E-07	0.12%	1.15E+08	9.70E-03	3.73E+05
^{234}U	3.03E+00	6.66E-07	0.12%	1.84E+08	6.20E-03	3.79E+05
^{235}U	1.02E-03	6.59E-07	0.12%	5.51E+11	2.20E-06	4.04E+05
^{236}U	2.94E-02	6.62E-07	0.12%	1.90E+10	6.50E-05	4.12E+05
^{238}U	1.37E-02	6.14E-07	0.12%	4.40E+10	3.40E-07	4.99E+03
^{237}Np	3.43E-01	6.66E-07	0.12%	1.62E+09	7.10E-04	3.84E+05
^{236}Pu	5.33E+05	6.69E-07	0.13%	1.04E+03	5.30E+02	1.84E+05
^{238}Pu	1.62E+04	6.63E-07	0.13%	3.45E+04	1.70E+01	1.95E+05
^{239}Pu	3.86E+01	6.66E-07	0.12%	1.44E+07	6.20E-02	2.98E+05
^{240}Pu	1.17E+03	6.32E-07	0.13%	5.01E+05	2.30E-01	3.84E+04
^{241}Pu	1.35E+00	6.65E-07	0.12%	4.13E+08	1.00E+02	1.38E+10
^{242}Pu	1.72E+03	6.27E-07	0.13%	3.43E+05	3.90E-03	4.46E+02
^{244}Pu	1.90E+03	6.19E-07	0.12%	3.15E+05	1.80E-05	1.89E+00
^{241}Am	2.71E+03	6.70E-07	0.13%	2.04E+05	3.40E+00	2.31E+05
^{242m}Am	1.65E+02	6.38E-07	0.13%	3.52E+06	1.00E+01	1.17E+07
^{243}Am	1.40E+02	6.66E-07	0.12%	3.97E+06	2.00E-01	2.65E+05
^{240}Cm	9.48E+07	6.52E-07	0.13%	5.99E+00	2.00E+04	4.00E+04
^{242}Cm	2.48E+07	6.40E-07	0.13%	2.33E+01	3.30E+03	2.57E+04
^{243}Cm	4.10E+05	6.41E-07	0.13%	1.41E+03	5.20E+01	2.44E+04
^{244}Cm	1.09E+07	6.34E-07	0.13%	5.36E+01	8.10E+01	1.45E+03
^{245}Cm	3.82E+03	6.34E-07	0.13%	1.53E+05	1.72E-01	8.77E+03
^{246}Cm	9.68E+06	6.31E-07	0.13%	6.06E+01	3.10E-01	6.27E+00
^{248}Cm	4.03E+07	6.28E-07	0.13%	1.46E+01	4.20E-03	2.05E-02
^{250}Cm	1.25E+10	6.13E-07	0.13%	4.83E-02	1.50E-01	2.42E-03
^{249}Bk	1.65E+05	6.18E-07	0.13%	3.63E+03	1.60E+03	1.94E+06
^{249}Cf	6.26E+03	6.62E-07	0.13%	8.93E+04	4.10E+00	1.22E+05
^{250}Cf	1.08E+10	6.48E-07	0.13%	5.29E-02	1.10E+02	1.94E+00
^{251}Cf	1.52E+03	6.71E-07	0.13%	3.63E+05	1.60E+00	1.94E+05
^{252}Cf	2.05E+12	6.49E-07	0.13%	2.78E-04	5.40E+02	5.01E-02
^{254}Cf	1.05E+15	6.51E-07	0.13%	5.42E-07	8.50E+03	1.54E-03
^{253}Es	3.13E+08	6.53E-07	0.13%	1.81E+00	2.50E+04	1.51E+04
^{254}Es	9.43E+06	6.57E-07	0.13%	5.98E+01	1.90E+03	3.79E+04
^{254m}Es	1.94E+13	6.50E-07	0.13%	2.94E-05	3.10E+05	3.04E+00

Table 9 – Neutron Activity Calculations, Axial Model

	A	B		C = 1000/(2.7*A*B)	D	E = C*D/3
Radio-nuclide	**Source Strength (n/s/g) [3]**	**Dose Rate from 1 n/s source (mrem/hr)**	**σ**	**Total Mass for 1000 mrem/hr at 1 Meter (g)**	**Specific Activity (Ci/g) [2]**	**Activity per Shielded Container for 1000 mrem/hr at 1 Meter, (Ci), A_N**
^{230}Th	9.13E+00	6.41E-07	0.14%	6.32E+07	2.10E-02	4.43E+05
^{232}Th	2.28E-05	6.45E-07	0.13%	2.52E+13	1.10E-07	9.23E+05
^{231}Pa	2.64E+01	6.41E-07	0.14%	2.19E+07	4.70E-02	3.43E+05
^{232}U	1.50E+04	6.40E-07	0.14%	3.86E+04	2.20E+01	2.83E+05
^{233}U	4.83E+00	6.43E-07	0.14%	1.19E+08	9.70E-03	3.86E+05
^{234}U	3.03E+00	6.41E-07	0.14%	1.91E+08	6.20E-03	3.94E+05
^{235}U	1.02E-03	6.33E-07	0.14%	5.74E+11	2.20E-06	4.21E+05
^{236}U	2.94E-02	6.37E-07	0.14%	1.98E+10	6.50E-05	4.28E+05
^{238}U	1.37E-02	5.90E-07	0.14%	4.58E+10	3.40E-07	5.19E+03
^{237}Np	3.43E-01	6.42E-07	0.14%	1.68E+09	7.10E-04	3.98E+05
^{236}Pu	5.33E+05	6.39E-07	0.15%	1.09E+03	5.30E+02	1.92E+05
^{238}Pu	1.62E+04	6.34E-07	0.14%	3.60E+04	1.70E+01	2.04E+05
^{239}Pu	3.86E+01	6.39E-07	0.14%	1.50E+07	6.20E-02	3.10E+05
^{240}Pu	1.17E+03	6.03E-07	0.14%	5.25E+05	2.30E-01	4.03E+04
^{241}Pu	1.35E+00	6.38E-07	0.14%	4.30E+08	1.00E+02	1.43E+10
^{242}Pu	1.72E+03	6.00E-07	0.14%	3.59E+05	3.90E-03	4.67E+02
^{244}Pu	1.90E+03	5.94E-07	0.14%	3.28E+05	1.80E-05	1.97E+00
^{241}Am	2.71E+03	6.42E-07	0.14%	2.13E+05	3.40E+00	2.41E+05
^{242m}Am	1.65E+02	6.12E-07	0.15%	3.67E+06	1.00E+01	1.22E+07
^{243}Am	1.40E+02	6.39E-07	0.14%	4.14E+06	2.00E-01	2.76E+05
^{240}Cm	9.48E+07	6.21E-07	0.15%	6.30E+00	2.00E+04	4.20E+04
^{242}Cm	2.48E+07	6.10E-07	0.15%	2.45E+01	3.30E+03	2.69E+04
^{243}Cm	4.10E+05	6.13E-07	0.15%	1.47E+03	5.20E+01	2.56E+04
^{244}Cm	1.09E+07	6.03E-07	0.15%	5.63E+01	8.10E+01	1.52E+03
^{245}Cm	3.82E+03	6.06E-07	0.15%	1.60E+05	1.72E-01	9.18E+03
^{246}Cm	9.68E+06	6.03E-07	0.14%	6.34E+01	3.10E-01	6.55E+00
^{248}Cm	4.03E+07	6.00E-07	0.14%	1.53E+01	4.20E-03	2.15E-02
^{250}Cm	1.25E+10	5.88E-07	0.14%	5.04E-02	1.50E-01	2.52E-03
^{249}Bk	1.65E+05	5.92E-07	0.14%	3.79E+03	1.60E+03	2.02E+06
^{249}Cf	6.26E+03	6.32E-07	0.15%	9.37E+04	4.10E+00	1.28E+05
^{250}Cf	1.08E+10	6.18E-07	0.15%	5.55E-02	1.10E+02	2.04E+00
^{251}Cf	1.52E+03	6.41E-07	0.15%	3.80E+05	1.60E+00	2.03E+05
^{252}Cf	2.05E+12	6.18E-07	0.15%	2.92E-04	5.40E+02	5.26E-02
^{254}Cf	1.05E+15	6.21E-07	0.15%	5.68E-07	8.50E+03	1.61E-03
^{253}Es	3.13E+08	6.21E-07	0.15%	1.90E+00	2.50E+04	1.59E+04
^{254}Es	9.43E+06	6.25E-07	0.15%	6.28E+01	1.90E+03	3.98E+04
^{254m}Es	1.94E+13	6.18E-07	0.15%	3.09E-05	3.10E+05	3.19E+00

Table 10 – Gamma Activity Calculations

Radio-nuclide	Activity per SC, Radial (Ci), A_G	Activity per SC, Axial (Ci), A_G	Limiting Case
^{3}H	1.000E+99	1.000E+99	NA
^{10}Be	1.000E+99	1.000E+99	NA
^{14}C	1.000E+99	1.000E+99	NA
^{22}Na	3.150E+00	5.829E+00	Radial
^{32}P	1.000E+99	1.000E+99	NA
^{33}P	1.000E+99	1.000E+99	NA
^{35}S	1.000E+99	1.000E+99	NA
^{45}Ca	6.741E+63	6.375E+16	Axial
^{46}Sc	2.254E+00	4.235E+00	Radial
^{49}V	1.000E+99	1.000E+99	NA
^{51}Cr	4.185E+04	1.570E+03	Axial
^{54}Mn	6.337E+00	1.168E+01	Radial
^{55}Fe	1.000E+99	1.000E+99	NA
^{59}Fe	3.496E+00	6.533E+00	Radial
^{57}Co	6.607E+03	1.033E+04	Radial
^{58}Co	6.583E+00	1.205E+01	Radial
^{60}Co	1.609E+00	2.983E+00	Radial
^{59}Ni	1.000E+99	1.000E+99	NA
^{63}Ni	1.000E+99	1.000E+99	NA
^{64}Cu	6.136E+02	1.125E+03	Radial
^{65}Zn	7.504E+00	1.417E+01	Radial
^{73}As	1.939E+57	1.834E+10	Axial
^{79}Se	1.000E+99	1.000E+99	NA
^{85}Kr	7.462E+03	7.665E+03	Radial
^{86}Rb	4.606E+01	8.738E+01	Radial
^{87}Rb	1.000E+99	1.000E+99	NA
^{89}Sr	5.595E+04	1.050E+05	Radial
^{90}Sr	1.000E+99	1.000E+99	NA
^{88}Y	1.484E+00	2.567E+00	Radial
^{90}Y	1.214E+08	1.940E+08	Radial
^{90m}Y	5.419E+01	4.310E+01	Axial
^{91}Y	1.313E+03	2.452E+03	Radial
^{88}Zr	2.412E+02	7.368E+01	Axial
^{90}Zr	1.000E+99	1.000E+99	NA
^{90m}Zr	1.000E+99	1.000E+99	NA
^{93}Zr	3.631E+61	3.434E+14	Axial
^{95}Zr	8.821E+00	1.502E+01	Radial
^{93m}Nb	3.542E+61	3.350E+14	Axial

Radio-nuclide	Activity per SC, Radial (Ci), A_G	Activity per SC, Axial (Ci), A_G	Limiting Case
^{94}Nb	3.752E+00	6.624E+00	Radial
^{95}Nb	7.919E+00	1.387E+01	Radial
^{95m}Nb	1.912E+04	3.159E+03	Axial
^{99}Tc	3.111E+61	2.942E+14	Axial
^{99m}Tc	3.891E+09	4.654E+05	Axial
^{103}Ru	3.557E+01	3.540E+01	Axial
^{106}Ru	1.000E+99	1.000E+99	NA
^{103m}Rh	2.974E+59	2.813E+12	Axial
^{106}Rh	4.601E+01	6.407E+01	Radial
^{107}Pd	1.000E+99	1.000E+99	NA
^{108}Ag	6.444E+02	9.054E+02	Radial
^{108m}Ag	6.082E+00	8.617E+00	Radial
^{109m}Ag	5.466E+57	5.169E+10	Axial
^{110}Ag	2.609E+02	4.018E+02	Radial
^{110m}Ag	1.957E+00	3.450E+00	Radial
^{109}Cd	5.466E+57	5.169E+10	Axial
^{113m}Cd	6.377E+09	1.998E+06	Axial
^{115m}Cd	1.380E+02	2.552E+02	Radial
^{114}In	2.127E+03	3.860E+03	Radial
^{114m}In	2.001E+02	1.197E+02	Axial
^{115m}In	4.429E+03	2.828E+02	Axial
^{119m}Sn	1.255E+57	1.186E+10	Axial
^{121m}Sn	1.093E+58	1.034E+11	Axial
^{123}Sn	6.204E+02	1.176E+03	Radial
^{126}Sn	3.172E+56	3.000E+09	Axial
^{124}Sb	2.638E+00	4.397E+00	Radial
^{125}Sb	3.517E+01	4.052E+01	Radial
^{126}Sb	2.925E+00	4.537E+00	Radial
^{126m}Sb	6.057E+00	8.751E+00	Radial
^{123}Te	1.000E+99	1.000E+99	NA
^{123m}Te	4.403E+12	4.015E+04	Axial
^{125m}Te	2.656E+33	2.211E+10	Axial
^{127}Te	1.319E+04	5.524E+03	Axial
^{127m}Te	8.662E+04	1.308E+05	Radial
^{129}Te	2.580E+02	2.927E+02	Radial
^{129m}Te	2.423E+02	3.942E+02	Radial
^{125}I	3.027E+57	2.863E+10	Axial
^{129}I	2.693E+57	2.547E+10	Axial

Radio-nuclide	Activity per SC, Radial (Ci), A_G	Activity per SC, Axial (Ci), A_G	Limiting Case
^{131}I	1.168E+02	7.331E+01	Axial
^{134}Cs	4.410E+00	7.301E+00	Radial
^{135}Cs	1.000E+99	1.000E+99	NA
^{137}Cs	1.441E+01	2.213E+01	Radial
^{133}Ba	1.075E+03	1.214E+02	Axial
^{137}Ba	1.000E+99	1.000E+99	NA
^{137m}Ba	1.361E+01	2.090E+01	Radial
^{141}Ce	7.037E+24	3.210E+05	Axial
^{142}Ce	1.000E+99	1.000E+99	NA
^{144}Ce	1.460E+34	1.548E+07	Axial
^{143}Pr	7.266E+08	1.236E+09	Radial
^{144}Pr	1.534E+02	2.504E+02	Radial
^{144m}Pr	4.666E+03	7.993E+03	Radial
^{146}Pm	1.325E+01	1.880E+01	Radial
^{147}Pm	2.189E+19	4.167E+11	Axial
^{148}Pm	8.190E+00	1.429E+01	Radial
^{148m}Pm	4.300E+00	6.482E+00	Radial
^{146}Sm	1.000E+99	1.000E+99	NA
^{147}Sm	1.000E+99	1.000E+99	NA
^{151}Sm	6.440E+59	6.091E+12	Axial
^{150}Eu	7.318E+00	9.948E+00	Radial
^{152}Eu	4.506E+00	8.111E+00	Radial
^{154}Eu	3.970E+00	7.256E+00	Radial
^{155}Eu	2.279E+27	2.312E+08	Axial
^{152}Gd	1.000E+99	1.000E+99	NA
^{153}Gd	3.718E+19	3.617E+07	Axial
^{160}Tb	4.519E+00	8.359E+00	Radial
^{166m}Ho	4.816E+00	8.081E+00	Radial
^{168}Tm	6.124E+00	1.047E+01	Radial
^{182}Ta	3.574E+00	6.684E+00	Radial
^{198}Au	1.321E+02	6.193E+01	Axial
^{207}Tl	2.106E+03	3.937E+03	Radial
^{208}Tl	1.340E+00	2.007E+00	Radial
^{209}Tl	2.261E+00	3.812E+00	Radial
^{209}Pb	1.000E+99	1.000E+99	NA
^{210}Pb	4.758E+57	4.500E+10	Axial
^{211}Pb	1.303E+02	2.031E+02	Radial
^{212}Pb	7.309E+04	1.432E+03	Axial
^{214}Pb	3.088E+02	1.599E+02	Axial
^{207}Bi	3.761E+00	6.474E+00	Radial

Radio-nuclide	Activity per SC, Radial (Ci), A_G	Activity per SC, Axial (Ci), A_G	Limiting Case
^{210}Bi	1.000E+99	1.000E+99	NA
^{211}Bi	9.166E+03	8.561E+02	Axial
^{212}Bi	5.100E+01	8.828E+01	Radial
^{213}Bi	2.289E+02	1.679E+02	Axial
^{214}Bi	3.027E+00	5.167E+00	Radial
^{209}Po	1.170E+03	2.105E+03	Radial
^{210}Po	5.639E+05	1.031E+06	Radial
^{211}Po	7.756E+02	1.321E+03	Radial
^{212}Po	1.000E+99	1.000E+99	NA
^{213}Po	1.561E+05	2.779E+05	Radial
^{214}Po	1.000E+99	1.000E+99	NA
^{215}Po	2.378E+05	1.295E+05	Axial
^{216}Po	3.576E+05	6.540E+05	Radial
^{218}Po	1.000E+99	1.000E+99	NA
^{211}At	4.143E+03	6.574E+03	Radial
^{217}At	1.414E+05	1.614E+05	Radial
^{219}Rn	2.424E+03	7.733E+02	Axial
^{220}Rn	2.106E+04	2.477E+04	Radial
^{222}Rn	4.408E+04	4.459E+04	Radial
^{221}Fr	9.242E+04	8.873E+03	Axial
^{223}Fr	7.879E+02	1.223E+03	Radial
^{223}Ra	3.207E+03	7.708E+02	Axial
^{224}Ra	2.233E+05	1.825E+04	Axial
^{225}Ra	6.741E+56	6.375E+09	Axial
^{226}Ra	2.851E+06	1.466E+05	Axial
^{228}Ra	1.057E+58	9.992E+10	Axial
^{225}Ac	1.534E+04	1.217E+04	Axial
^{227}Ac	2.172E+12	4.286E+07	Axial
^{228}Ac	6.161E+00	1.104E+01	Radial
^{227}Th	1.733E+04	9.810E+02	Axial
^{228}Th	2.887E+07	5.473E+05	Axial
^{229}Th	1.279E+08	2.323E+04	Axial
^{230}Th	4.094E+08	4.468E+06	Axial
^{231}Th	1.549E+08	1.271E+06	Axial
^{232}Th	3.537E+31	1.834E+09	Axial
^{234}Th	1.717E+50	1.292E+10	Axial
^{231}Pa	3.253E+04	1.906E+03	Axial
^{233}Pa	3.190E+03	2.815E+02	Axial
^{234}Pa	4.122E+00	7.325E+00	Radial
^{234m}Pa	3.020E+02	5.552E+02	Radial

Radio-nuclide	Activity per SC, Radial (Ci), A_G	Activity per SC, Axial (Ci), A_G	Limiting Case
^{232}U	4.438E+07	3.391E+06	Axial
^{233}U	2.112E+07	7.512E+05	Axial
^{234}U	6.256E+07	6.311E+07	Radial
^{235}U	4.965E+05	6.641E+03	Axial
^{236}U	2.779E+51	5.262E+11	Axial
^{237}U	1.267E+05	4.116E+03	Axial
^{238}U	1.600E+51	7.855E+11	Axial
^{239}U	6.490E+02	1.130E+03	Radial
^{240}U	6.528E+07	5.128E+05	Axial
^{237}Np	4.991E+09	2.611E+05	Axial
^{238}Np	6.943E+00	1.323E+01	Radial
^{239}Np	5.472E+04	1.172E+03	Axial
^{240}Np	6.340E+00	1.093E+01	Radial
^{240m}Np	2.360E+01	3.662E+01	Radial
^{236}Pu	3.550E+06	4.751E+06	Radial
^{238}Pu	2.110E+07	3.705E+07	Radial
^{239}Pu	3.328E+06	1.208E+06	Axial
^{240}Pu	7.113E+07	1.059E+08	Radial
^{241}Pu	1.399E+26	4.098E+10	Axial
^{242}Pu	3.692E+24	7.582E+09	Axial
^{243}Pu	5.422E+04	1.141E+04	Axial
^{244}Pu	1.000E+99	1.000E+99	NA
^{241}Am	1.449E+06	1.800E+06	Radial
^{242}Am	1.465E+60	1.386E+13	Axial
^{242m}Am	7.279E+21	8.796E+07	Axial
^{243}Am	8.400E+05	1.267E+06	Radial
^{245}Am	7.377E+06	8.619E+03	Axial
^{240}Cm	1.000E+99	1.000E+99	NA
^{242}Cm	4.972E+06	7.123E+06	Radial
^{243}Cm	1.031E+06	1.745E+03	Axial
^{244}Cm	5.031E+06	7.823E+06	Radial
^{245}Cm	1.353E+15	1.160E+05	Axial
^{246}Cm	1.000E+99	1.000E+99	NA
^{247}Cm	2.393E+02	8.881E+01	Axial
^{248}Cm	1.000E+99	1.000E+99	NA
^{250}Cm	1.000E+99	1.000E+99	NA
^{247}Bk	4.065E+06	1.485E+03	Axial
^{249}Bk	1.683E+10	7.350E+08	Axial
^{250}Bk	4.933E+00	9.414E+00	Radial
^{249}Cf	4.114E+02	1.005E+02	Axial

Radio-nuclide	Activity per SC, Radial (Ci), A_G	Activity per SC, Axial (Ci), A_G	Limiting Case
^{250}Cf	1.446E+60	1.367E+13	Axial
^{251}Cf	3.155E+06	6.648E+03	Axial
^{252}Cf	2.940E+25	9.084E+09	Axial
^{254}Cf	1.000E+99	1.000E+99	NA
^{252}Es	3.251E+01	5.826E+01	Radial
^{253}Es	3.178E+05	1.091E+05	Axial
^{254}Es	4.207E+05	4.230E+04	Axial
^{254m}Es	1.757E+01	2.725E+01	Radial

Table 11 – Summary of HAC Activity Limits

Radio-nuclide	Gamma Emitter	Neutron Emitter	SC HAC @ 1-m Maximum Allowable Activity (Ci), A_{GN}	Limiting Case
^{3}H	·	·	unlimited	NA
^{10}Be	·	·	unlimited	NA
^{14}C	·	·	unlimited	NA
^{22}Na	×	·	3.150E+00	Radial
^{32}P	·	·	unlimited	NA
^{33}P	·	·	unlimited	NA
^{35}S	·	·	unlimited	NA
^{45}Ca	×	·	unlimited	NA
^{46}Sc	×	·	2.254E+00	Radial
^{49}V	·	·	unlimited	NA
^{51}Cr	×	·	1.570E+03	Axial
^{54}Mn	×	·	6.337E+00	Radial
^{55}Fe	·	·	unlimited	NA
^{59}Fe	×	·	3.496E+00	Radial
^{57}Co	×	·	6.607E+03	Radial
^{58}Co	×	·	6.583E+00	Radial
^{60}Co	×	·	1.609E+00	Radial
^{59}Ni	·	·	unlimited	NA
^{63}Ni	·	·	unlimited	NA
^{64}Cu	×	·	6.136E+02	Radial
^{65}Zn	×	·	7.504E+00	Radial
^{73}As	×	·	unlimited	NA
^{79}Se	·	·	unlimited	NA
^{85}Kr	×	·	7.462E+03	Radial
^{86}Rb	×	·	4.606E+01	Radial
^{87}Rb	·	·	unlimited	NA
^{89}Sr	×	·	5.595E+04	Radial
^{90}Sr	·	·	unlimited	NA
^{88}Y	×	·	1.484E+00	Radial
^{90}Y	×	·	unlimited	NA
^{90m}Y	×	·	4.310E+01	Axial
^{91}Y	×	·	1.313E+03	Radial
^{88}Zr	×	·	7.368E+01	Axial
^{90}Zr	·	·	unlimited	NA
^{90m}Zr	·	·	unlimited	NA
^{93}Zr	×	·	unlimited	NA
^{95}Zr	×	·	8.821E+00	Radial

Radio-nuclide	Gamma Emitter	Neutron Emitter	SC HAC @ 1-m Maximum Allowable Activity (Ci), A_{GN}	Limiting Case
^{93m}Nb	×	·	unlimited	NA
^{94}Nb	×	·	3.752E+00	Radial
^{95}Nb	×	·	7.919E+00	Radial
^{95m}Nb	×	·	3.159E+03	Axial
^{99}Tc	×	·	unlimited	NA
^{99m}Tc	×	·	4.654E+05	Axial
^{103}Ru	×	·	3.540E+01	Axial
^{106}Ru	·	·	unlimited	NA
^{103m}Rh	×	·	unlimited	NA
^{106}Rh	×	·	4.601E+01	Radial
^{107}Pd	·	·	unlimited	NA
^{108}Ag	×	·	6.444E+02	Radial
^{108m}Ag	×	·	6.082E+00	Radial
^{109m}Ag	×	·	unlimited	NA
^{110}Ag	×	·	2.609E+02	Radial
^{110m}Ag	×	·	1.957E+00	Radial
^{109}Cd	×	·	unlimited	NA
^{113m}Cd	×	·	1.998E+06	Axial
^{115m}Cd	×	·	1.380E+02	Radial
^{114}In	×	·	2.127E+03	Radial
^{114m}In	×	·	1.197E+02	Axial
^{115m}In	×	·	2.828E+02	Axial
^{119m}Sn	×	·	unlimited	NA
^{121m}Sn	×	·	unlimited	NA
^{123}Sn	×	·	6.204E+02	Radial
^{126}Sn	×	·	unlimited	NA
^{124}Sb	×	·	2.638E+00	Radial
^{125}Sb	×	·	3.517E+01	Radial
^{126}Sb	×	·	2.925E+00	Radial
^{126m}Sb	×	·	6.057E+00	Radial
^{123}Te	·	·	unlimited	NA
^{123m}Te	×	·	4.015E+04	Axial
^{125m}Te	×	·	unlimited	NA
^{127}Te	×	·	5.524E+03	Axial
^{127m}Te	×	·	8.662E+04	Radial
^{129}Te	×	·	2.580E+02	Radial
^{129m}Te	×	·	2.423E+02	Radial

Radio-nuclide	Gamma Emitter	Neutron Emitter	SC HAC @ 1-m Maximum Allowable Activity (Ci), A_{GN}	Limiting Case
^{125}I	×	·	unlimited	NA
^{129}I	×	·	unlimited	NA
^{131}I	×	·	7.331E+01	Axial
^{134}Cs	×	·	4.410E+00	Radial
^{135}Cs	·	·	unlimited	NA
^{137}Cs	×	·	1.441E+01	Radial
^{133}Ba	×	·	1.214E+02	Axial
^{137}Ba	·	·	unlimited	NA
^{137m}Ba	×	·	1.361E+01	Radial
^{141}Ce	×	·	3.210E+05	Axial
^{142}Ce	·	·	unlimited	NA
^{144}Ce	×	·	1.548E+07	Axial
^{143}Pr	×	·	unlimited	NA
^{144}Pr	×	·	1.534E+02	Radial
^{144m}Pr	×	·	4.666E+03	Radial
^{146}Pm	×	·	1.325E+01	Radial
^{147}Pm	×	·	unlimited	NA
^{148}Pm	×	·	8.190E+00	Radial
^{148m}Pm	×	·	4.300E+00	Radial
^{146}Sm	·	·	unlimited	NA
^{147}Sm	·	·	unlimited	NA
^{151}Sm	×	·	unlimited	NA
^{150}Eu	×	·	7.318E+00	Radial
^{152}Eu	×	·	4.506E+00	Radial
^{154}Eu	×	·	3.970E+00	Radial
^{155}Eu	×	·	unlimited	NA
^{152}Gd	·	·	unlimited	NA
^{153}Gd	×	·	3.617E+07	Axial
^{160}Tb	×	·	4.519E+00	Radial
^{166m}Ho	×	·	4.816E+00	Radial
^{168}Tm	×	·	6.124E+00	Radial
^{182}Ta	×	·	3.574E+00	Radial
^{198}Au	×	·	6.193E+01	Axial
^{207}Tl	×	·	2.106E+03	Radial
^{208}Tl	×	·	1.340E+00	Radial
^{209}Tl	×	·	2.261E+00	Radial
^{209}Pb	·	·	unlimited	NA
^{210}Pb	×	·	unlimited	NA
^{211}Pb	×	·	1.303E+02	Radial

Radio-nuclide	Gamma Emitter	Neutron Emitter	SC HAC @ 1-m Maximum Allowable Activity (Ci), A_{GN}	Limiting Case
^{212}Pb	×	·	1.432E+03	Axial
^{214}Pb	×	·	1.599E+02	Axial
^{207}Bi	×	·	3.761E+00	Radial
^{210}Bi	·	·	unlimited	NA
^{211}Bi	×	·	8.561E+02	Axial
^{212}Bi	×	·	5.100E+01	Radial
^{213}Bi	×	·	1.679E+02	Axial
^{214}Bi	×	·	3.027E+00	Radial
^{209}Po	×	·	1.170E+03	Radial
^{210}Po	×	·	5.639E+05	Radial
^{211}Po	×	·	7.756E+02	Radial
^{212}Po	·	·	unlimited	NA
^{213}Po	×	·	1.561E+05	Radial
^{214}Po	·	·	unlimited	NA
^{215}Po	×	·	1.295E+05	Axial
^{216}Po	×	·	3.576E+05	Radial
^{218}Po	·	·	unlimited	NA
^{211}At	×	·	4.143E+03	Radial
^{217}At	×	·	1.414E+05	Radial
^{219}Rn	×	·	7.733E+02	Axial
^{220}Rn	×	·	2.106E+04	Radial
^{222}Rn	×	·	4.408E+04	Radial
^{221}Fr	×	·	8.873E+03	Axial
^{223}Fr	×	·	7.879E+02	Radial
^{223}Ra	×	·	7.708E+02	Axial
^{224}Ra	×	·	1.825E+04	Axial
^{225}Ra	×	·	unlimited	NA
^{226}Ra	×	·	1.466E+05	Axial
^{228}Ra	×	·	unlimited	NA
^{225}Ac	×	·	1.217E+04	Axial
^{227}Ac	×	·	4.286E+07	Axial
^{228}Ac	×	·	6.161E+00	Radial
^{227}Th	×	·	9.810E+02	Axial
^{228}Th	×	·	5.473E+05	Axial
^{229}Th	×	·	2.323E+04	Axial
^{230}Th	×	×	4.028E+05	Axial
^{231}Th	×	·	1.271E+06	Axial
^{232}Th	×	×	8.866E+05	Radial
^{234}Th	×	·	unlimited	NA

Radio-nuclide	Gamma Emitter	Neutron Emitter	SC HAC @ 1-m Maximum Allowable Activity (Ci), A_{GN}	Limiting Case
^{231}Pa	×	×	1.896E+03	Axial
^{233}Pa	×	·	2.815E+02	Axial
^{234}Pa	×	·	4.122E+00	Radial
^{234m}Pa	×	·	3.020E+02	Radial
^{232}U	×	×	2.611E+05	Axial
^{233}U	×	×	2.549E+05	Axial
^{234}U	×	×	3.770E+05	Radial
^{235}U	×	×	6.537E+03	Axial
^{236}U	×	×	4.121E+05	Radial
^{237}U	×	·	4.116E+03	Axial
^{238}U	×	×	4.987E+03	Radial
^{239}U	×	·	6.490E+02	Radial
^{240}U	×	·	5.128E+05	Axial
^{237}Np	×	×	1.577E+05	Axial
^{238}Np	×	·	6.943E+00	Radial
^{239}Np	×	·	1.172E+03	Axial
^{240}Np	×	·	6.340E+00	Radial
^{240m}Np	×	·	2.360E+01	Radial
^{236}Pu	×	×	1.746E+05	Radial
^{238}Pu	×	×	1.937E+05	Radial
^{239}Pu	×	×	2.468E+05	Axial
^{240}Pu	×	×	3.836E+04	Radial
^{241}Pu	×	×	unlimited	NA
^{242}Pu	×	×	4.464E+02	Radial
^{243}Pu	×	·	1.141E+04	Axial
^{244}Pu	·	×	1.891E+00	Radial
^{241}Am	×	×	1.995E+05	Radial
^{242}Am	×	·	unlimited	NA
^{242m}Am	×	×	1.074E+07	Axial
^{243}Am	×	×	2.014E+05	Radial
^{245}Am	×	·	8.619E+03	Axial
^{240}Cm	·	×	3.996E+04	Radial
^{242}Cm	×	×	2.555E+04	Radial
^{243}Cm	×	×	1.634E+03	Axial
^{244}Cm	×	×	1.446E+03	Radial
^{245}Cm	×	×	8.507E+03	Axial
^{246}Cm	·	×	6.267E+00	Radial
^{247}Cm	×	·	8.881E+01	Axial
^{248}Cm	·	×	2.050E-02	Radial

Radio-nuclide	Gamma Emitter	Neutron Emitter	SC HAC @ 1-m Maximum Allowable Activity (Ci), A_{GN}	Limiting Case
^{250}Cm	·	×	2.416E-03	Radial
^{247}Bk	×	·	1.485E+03	Axial
^{249}Bk	×	×	1.937E+06	Radial
^{250}Bk	×	·	4.933E+00	Radial
^{249}Cf	×	×	1.004E+02	Axial
^{250}Cf	×	×	1.941E+00	Radial
^{251}Cf	×	×	6.437E+03	Axial
^{252}Cf	×	×	5.008E-02	Radial
^{254}Cf	·	×	1.536E-03	Radial
^{252}Es	×	·	3.251E+01	Radial
^{253}Es	×	×	1.386E+04	Axial
^{254}Es	×	×	2.050E+04	Axial
^{254m}Es	×	×	2.589E+00	Radial

5.0 Appendices

5.1 References

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5.2 Sample MCNP Input Files

Sample radial MCNP input file (R01TH230)

```
HalfPACT Shielded Canister
10     0         13 -20 -30 #20 #21 #22 imp:n=1 $ main inner cavity
11     0         20 -21 -30             imp:n=1 $ axial dunnage as void
12     0         21 -22 -30             imp:n=1 $ honeycomb as void
13     3 -8.0128 (22:30:-13) 12 -23 -31 imp:n=1 $ SS
14     0         (23:31:-12) 11 -24 -32 imp:n=1 $ foam as void
15     0         (24:32:-11) 10 -25 -33 imp:n=1 $ SS as void
c
20     0         40 -41 -42   fill=1(1) imp:n=1 $ canister 1
21     0         40 -41 -43   fill=1(2) imp:n=1 $ canister 1
22     0         40 -41 -44   fill=1(3) imp:n=1 $ canister 1
c
50     0          (-10:25:33) -999      imp:n=1
999    0         999                    imp:n=0
```

```
c
c      Universe 1: Shielded canister
c
100    0         102 -103 -110                   u=1 imp:n=1 $ inner cavity
101    1 -7.8526 101 -104 110 -111               u=1 imp:n=1 $ CS inner
102    2 -11.35  101 -104 111 -112               u=1 imp:n=1 $ lead
103    1 -7.8526 101 -104 112 -113               u=1 imp:n=1 $ CS outer
104    1 -7.8526 (103 -104 -110):(104 -105 -113) u=1 imp:n=1 $ lid
105    1 -7.8526 (101 -102 -110):(100 -101 -113) u=1 imp:n=1 $ bottom
106    0         -100:105:113                    u=1 imp:n=1 $ outside

10     pz -88.265
11     pz -87.63
12     pz -67.6275
13     pz -66.3575
c
20     pz 55.5625 $ max container may shift to
21     pz 56.8325
22     pz 66.3575
23     pz 67.6275
24     pz 87.63
25     pz 88.265
c
30     cz 92.2338
31     cz 93.345
32     cz 109.22
33     cz 109.855
c
40  4  pz -44.7676 $ filling cell canister 1
41  4  pz  44.7676
42  1  cz -29.2101
43  2  cz -29.2101
44  3  cz -29.2101
c
100    pz -44.7675
101    pz -40.005
102    pz -38.4175
103    pz  38.4175
104    pz  40.005
105    pz  44.7675 $ top of canister
c
110    cz 26.2915
111    cz 26.7462
112    cz 28.9052
113    cz 29.21
c
999    so 500

mode n
sdef    pos=d1 erg=d7
si1     L 86.616 0 0  57.3 43.863 0 57.3 -43.863 0
sp1     1 1 1
sc7   TH230
si7   0.1 0.5     1.0     2.0     3.0     4.0     6.0     8.0
          10.0  15.0
sp7    0.00E+00  3.59E-01  3.27E-01  2.65E+00  4.84E+00
       9.24E-01  3.63E-02  6.16E-06  7.58E-07  9.36E-08
c
tr1     60.325         0 0  $ shift canister 1 fill
tr2     32.0372  51.1148 0  $ shift canister 2 fill
```

```
tr3     32.0372 -51.1148 0  $ shift canister 3 fill
tr4          0         0 0  $ axial shift filling surfaces
c
c       ANSI/ANS-6.1.1-1977 Neutron Flux to Dose Factors (mrem/hr)
de0     2.5e-08    1.0e-07    1.0e-06    1.0e-05    1.0e-04
        1.0e-03    1.0e-02    1.0e-01    5.0e-01    1.0
        2.5        5.0        7.0        10.0       14.0
        20.0
df0     3.67e-03   3.67e-03   4.46e-03   4.54e-03   4.18e-03
        3.76e-03   3.56e-03   2.17e-02   9.26e-02   1.32e-01
        1.25e-01   1.56e-01   1.47e-01   1.47e-01   2.08e-01
        2.27e-01
c
c     Tally
c
f5:n    209.806 0 0 0
prdmp   j j 1 2
ctme    1
m1      26000  1  $ carbon steel
m2      82000  1  $ lead
m3      14000 -1  $ stainless steel
        24000 -19
        25055 -2
        26000 -68
        28000 -10
```

Sample axial MCNP input file (A05U233)

```
HalfPACT Shielded Canister
10     0         13 -20 -30 #20 #21 #22 imp:n=1 $ main inner cavity
11     0         20 -21 -30             imp:n=1 $ axial dunnage as void
12     0         21 -22 -30             imp:n=1 $ honeycomb as void
13     3 -8.0128 (22:30:-13) 12 -23 -31 imp:n=1 $ SS
14     0         (23:31:-12) 11 -24 -32 imp:n=1 $ foam as void
15     0         (24:32:-11) 10 -25 -33 imp:n=1 $ SS as void
c
20     0         40 -41 -42   fill=1(1) imp:n=1 $ canister 1
21     0         40 -41 -43   fill=1(2) imp:n=1 $ canister 1
22     0         40 -41 -44   fill=1(3) imp:n=1 $ canister 1
c
50     0          (-10:25:33) -999      imp:n=1
999    0         999                    imp:n=0
c
c      Universe 1: Shielded canister
c
100    0         102 -103 -110                   u=1 imp:n=1 $ inner cavity
101    1 -7.8526 101 -104 110 -111               u=1 imp:n=1 $ CS inner
102    2 -11.35  101 -104 111 -112               u=1 imp:n=1 $ lead
103    1 -7.8526 101 -104 112 -113               u=1 imp:n=1 $ CS outer
104    1 -7.8526 (103 -104 -110):(104 -105 -113) u=1 imp:n=1 $ lid
105    1 -7.8526 (101 -102 -110):(100 -101 -113) u=1 imp:n=1 $ bottom
106    0         -100:105:113                    u=1 imp:n=1 $ outside

10     pz -88.265
11     pz -87.63
12     pz -67.6275
13     pz -66.3575
c
20     pz 55.5625 $ max container may shift to
```

```
21     pz 56.8325
22     pz 66.3575
23     pz 67.6275
24     pz 87.63
25     pz 88.265
c
30     cz 92.2338
31     cz 93.345
32     cz 109.22
33     cz 109.855
c
40  4  pz -44.7676 $ filling cell canister 1
41  4  pz  44.7676
42  1  cz -29.2101
43  2  cz -29.2101
44  3  cz -29.2101
c
100    pz -44.7675
101    pz -40.005
102    pz -38.4175
103    pz  38.4175
104    pz  40.005
105    pz  44.7675 $ top of canister
c
110    cz 26.2915
111    cz 26.7462
112    cz 28.9052
113    cz 29.21
c
999    so 500

mode n
sdef    pos=d1 erg=d7 $ rad=0.1
si1     L 7.4385 0 49.2113  -3.7193 6.4419 49.2113 -3.7193 -6.4419 49.2113
sp1     1 1 1
sc7   U233
si7   0.1 0.5     1.0     2.0     3.0     4.0     6.0     8.0
          10.0  15.0
sp7   0.00E+00  2.12E-01  1.75E-01  1.31E+00  2.52E+00
      5.90E-01  2.15E-02  1.52E-05  3.03E-06  6.64E-07
c
tr1     33.729       0 10.7948  $ shift canister 1 fill
tr2    -16.865  29.211 10.7948  $ shift canister 2 fill
tr3    -16.865 -29.211 10.7948  $ shift canister 3 fill
tr4          0       0 10.7948  $ axial shift filling surfaces
c
c       ANSI/ANS-6.1.1-1977 Neutron Flux to Dose Factors (mrem/hr)
de0     2.5e-08    1.0e-07    1.0e-06    1.0e-05    1.0e-04
        1.0e-03    1.0e-02    1.0e-01    5.0e-01    1.0
        2.5        5.0        7.0        10.0       14.0
        20.0
df0     3.67e-03   3.67e-03   4.46e-03   4.54e-03   4.18e-03
        3.76e-03   3.56e-03   2.17e-02   9.26e-02   1.32e-01
        1.25e-01   1.56e-01   1.47e-01   1.47e-01   2.08e-01
        2.27e-01
c
c     Tally
c
f5:n    0 0 188.265 0
prdmp   j j 1 2
```

```
ctme    1
m1      26000  1  $ carbon steel
m2      82000  1  $ lead
m3      14000 -1  $ stainless steel
        24000 -19
        25055 -2
        26000 -68
        28000 -10
```

5.3 Example Radial Gamma Calculation (Co-60)

HAC at 1m Point Source Gamma Shielding Analysis for the Shielded Container

Unit Definition:

Units not standard to MathCAD and additional constants are defined as follows:

$\gamma := Bq \cdot s$ $\quad \nu := \gamma$ $\quad Ci := 3.7 \cdot 10^{10} \cdot Bq$ $\quad eV := 1.60 \cdot 10^{-19} \cdot J$

$KeV := 1 \cdot 10^{3} \cdot eV$ $\quad MeV := 1 \cdot 10^{6} \cdot eV$ $\quad mrem := \frac{hr \cdot \gamma}{cm^{2} \cdot s}$

Mass Attentuation Coefficients from ANSI/ANS 6.4.3-1991:

Tabular mass attenuation coefficients as a function of gamma energy for 1) silicon, 2) chromium, 3) manganese, 4) iron, 5) nickel, and 6) lead are imported and assigned to the following variables:

$Ti := C:\backslash shieldcalc\backslash mass_attenuation_coefficients.xls$ $\quad i := 0 .. rows(Ti) - 1$

$Ei_i := Ti_{i,0} \cdot MeV$ $\quad \mu 1_{\rho_i} := Ti_{i,1} \cdot \frac{cm^2}{g}$ $\quad \mu 2_{\rho_i} := Ti_{i,2} \cdot \frac{cm^2}{g}$ $\quad \mu 3_{\rho_i} := Ti_{i,3} \cdot \frac{cm^2}{g}$

$\mu 4_{\rho_i} := Ti_{i,4} \cdot \frac{cm^2}{g}$ $\quad \mu 5_{\rho_i} := Ti_{i,5} \cdot \frac{cm^2}{g}$ $\quad \mu 6_{\rho_i} := Ti_{i,6} \cdot \frac{cm^2}{g}$

Iron Exposure Build Up Factor Coefficients from ANSI/ANS 6.4.3-1991:

Tabular iron exposure build up factor coefficients as function of gamma energy are imported and assigned to the following variables:

$Tj := C:\backslash shieldcalc\backslash iron_buildup_coefficients.xls$ $\quad j := 0 .. rows(Tj) - 1$

$Ej_j := Tj_{j,0} \cdot MeV$ $\quad a_j := Tj_{j,1}$ $\quad b_j := Tj_{j,2}$ $\quad c_j := Tj_{j,3}$ $\quad d_j := Tj_{j,4}$ $\quad x_{k_j} := Tj_{j,5}$

Flux to Dose Rate Conversion Factors from ANSI/ANS 6.1.1-1977:

Tabular flux to dose rate conversion factors as a function of gamma energy are imported and assigned to the following variables:

$Tk := C:\backslash shieldcalc\backslash flux_doserate_factors.xls$ $\quad k := 0 .. rows(Tk) - 1$

$Ek_k := Tk_{k,0} \cdot MeV$ $\quad Kk_k := Tk_{k,1}$

Shield Composition Definition:

The following table defines the thickness, density, and percent composition of the shielding materials. The percent composition is defined by 1) silicon, 2) chromium, 3) manganese, 4) iron, 5) nickel, and 6) lead.

$m := 0 .. 4$

Tm :=

	Thickness	Density	%Silicon	%Chromium	%Manganese	%Iron	%Nickel	%Lead
	0	1	2	3	4	5	6	7
0	0.1790	7.8526	0.0000	0.0000	0.0000	100.0000	0.0000	0.0000
1	0.8500	11.3500	0.0000	0.0000	0.0000	0.0000	0.0000	100.0000
2	0.1200	7.8526	0.0000	0.0000	0.0000	100.0000	0.0000	0.0000
3	0.2500	8.0128	1.0000	19.0000	2.0000	68.0000	10.0000	0.0000
4	0.1875	8.0128	1.0000	19.0000	2.0000	68.0000	10.0000	0.0000
5	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
6								
7								

$t_m := Tm_{m,0} \cdot in$ $\quad p1_m := Tm_{m,2} \cdot \%$ $\quad p3_m := Tm_{m,4} \cdot \%$ $\quad p5_m := Tm_{m,6} \cdot \%$

$\rho_m := Tm_{m,1} \cdot \frac{g}{cm^3}$ $\quad p2_m := Tm_{m,3} \cdot \%$ $\quad p4_m := Tm_{m,5} \cdot \%$ $\quad p6_m := Tm_{m,7} \cdot \%$

Detector Distance and Dose Limit Definition:

The dose limit is defined as $D := 1000 \cdot mrem \cdot hr^{-1}$ when measured at a distance of 1 meter from the surface of the package. The following table defines the horizontal and vertical distance from each point source to the detector and calculates the resulting radial distance.

$nn := 2$ $\quad n := 0 .. nn$

Tn :=

	Horizontal	Vertical
	0	1
0	48.5000	0.0000
1	62.4375	0.0000
2	62.4375	0.0000
3	0.0000	0.0000
4	0.0000	0.0000
5	0.0000	0.0000
6	0.0000	0.0000
7	0.0000	0.0000
8	0.0000	0.0000
9	0.0000	0.0000
10	0.0000	0.0000
11	0.0000	0.0000
12	0.0000	0.0000
13	0.0000	0.0000

$R_n := \sqrt{(Tn_{n,0})^2 + (Tn_{n,1})^2} \cdot in$

Radionuclide Gamma Energy and Intensity:

For the specified radionuclide, the gamma energies and gamma intensities are imported and assigned to the following variables:

$Tl := C:\backslash shieldcalc\backslash radionuclide_data.xls$ $\quad$ $index := in0$

$$ll := \left| \begin{array}{l} i \leftarrow 0 \\ \text{while } Tl_{i,0} \neq index \\ \quad i \leftarrow i + 1 \\ ll \leftarrow Tl_{i,1} - 1 \end{array} \right. \qquad l := 0 .. ll$$

$$E := \left| \begin{array}{l} i \leftarrow 0 \\ \text{while } Tl_{i,0} \neq index \\ \quad i \leftarrow i + 1 \\ ll \leftarrow Tl_{i,1} - 1 \\ \text{for } j \in 0 .. ll \\ \quad E_j \leftarrow \left| \begin{array}{ll} 100 & \text{if } Tl_{i+j,2} < 100 \\ Tl_{i+j,2} & \text{otherwise} \end{array} \right. \\ E \cdot KeV \end{array} \right. \qquad I := \left| \begin{array}{l} i \leftarrow 0 \\ \text{while } Tl_{i,0} \neq index \\ \quad i \leftarrow i + 1 \\ ll \leftarrow Tl_{i,1} - 1 \\ \text{for } j \in 0 .. ll \\ \quad I_j \leftarrow Tl_{i+j,3} \\ I \cdot \% \end{array} \right.$$

Free Mean Path Calculations:

The following equations utilize mass attenuation coefficients as a function of gamma energy and the shield composition definition to calculate the number of mean free paths for each of the gamma energies and shield materials:

$$\mu_{m,l} := \left(\text{linterp}\left(Ei, \mu 1_\rho, E_l\right) \cdot \rho_m \cdot p1_m\right) + \left(\text{linterp}\left(Ei, \mu 2_\rho, E_l\right) \cdot \rho_m \cdot p2_m\right) + \left(\text{linterp}\left(Ei, \mu 3_\rho, E_l\right) \cdot \rho_m \cdot p3_m\right) \ldots$$
$$+ \left(\text{linterp}\left(Ei, \mu 4_\rho, E_l\right) \cdot \rho_m \cdot p4_m\right) + \left(\text{linterp}\left(Ei, \mu 5_\rho, E_l\right) \cdot \rho_m \cdot p5_m\right) + \left(\text{linterp}\left(Ei, \mu 6_\rho, E_l\right) \cdot \rho_m \cdot p6_m\right)$$

$$x_{m,l} := \mu_{m,l} \cdot t_m$$

$$x_{total_l} := \sum x^{\langle l \rangle}$$

Build Up Factor Calculations:

The following G-P geometric progression equations (ANSI/ANS 6.4.3-1991) is utilized along with iron buildup factor coefficients and the number of mean free paths to calculate a buildup factor for each of the gamma energies:

$$Kx_l := \mathrm{linterp}(Ej, c, E_l)\cdot\left(x_{total_l}\right)^{\mathrm{linterp}(Ej, a, E_l)} + \mathrm{linterp}(Ej, d, E_l)\cdot\frac{\tanh\left[\left(\frac{x_{total_l}}{\mathrm{linterp}(Ej, x_k, E_l)}\right) - 2\right] - \tanh(-2)}{1 - \tanh(-2)}$$

$$B_l := \begin{cases} 1 + \left(\mathrm{linterp}(Ej, b, E_l) - 1\right)\cdot\left[\frac{\left(Kx_l\right)^{x_{total_l}} - 1}{Kx_l - 1}\right] & \text{if } Kx_l \neq 1 \\ 1 + \left(\mathrm{linterp}(Ej, b, E_l) - 1\right)\cdot x_{total_l} & \text{if } Kx_l = 1 \end{cases}$$

Maximum Activity Calculations:

The following equations calculate the total gamma dose resulting from 1 Ci of the specified radionuclide, then ratio the total dose to calculate the activity required to meet the limiting dose rate. The limiting activity is then utilized to calculate the individual gamma dose contributions from each point source. The gamma dose calculations utilize Rockwell's methodology.

$$K_l := \mathrm{linterp}(Ek, Kk, E_l)$$

$$D_{ratiocomp_{l,n}} := \left[\frac{Ci\cdot I_l\cdot B_l\cdot K_l}{4\cdot\pi\cdot\left(R_n\right)^2}\right]\cdot e^{-x_{total_l}}$$

$$D_{ratiototal_n} := \sum D_{ratiocomp}\langle n\rangle$$

$$D_{ratiolimit} := \sum D_{ratiototal}$$

$$A_{limit} := \begin{cases} Ci\cdot\frac{D}{D_{ratiolimit}} & \text{if } D_{ratiolimit} \neq 0 \\ Ci\cdot 10^{99} & \text{otherwise} \end{cases}$$

$$D_{limitcomp_{l,n}} := \left[\frac{A_{limit}\cdot I_l\cdot B_l\cdot K_l}{4\cdot\pi\cdot\left(R_n\right)^2}\right]\cdot e^{-x_{total_l}}$$

$$D_{limittotal_n} := \sum D_{limitcomp}\langle n\rangle$$

$$D_{limit} := \sum D_{limittotal}$$

Summary of Results:

The gamma energies and intensities for radionuclide $\text{index} = 16$ are as follows:

$$E^T = \left(346.9300 \quad 826.2800 \quad 1.1732 \times 10^3 \quad 1.3325 \times 10^3 \quad 2.1588 \times 10^3 \quad 2.5050 \times 10^3\right) \text{KeV}$$

$$I^T = \left(7.6 \times 10^{-3} \quad 7.6 \times 10^{-3} \quad 99.974 \quad 99.986 \quad 1.11 \times 10^{-3} \quad 2 \times 10^{-6}\right) \%$$

The attenuation coefficients and the number of mean free paths for each of the gamma energies and shield materials are

$$\mu^T = \begin{pmatrix} 0.781 & 3.427 & 0.781 & 0.797 & 0.797 \\ 0.514 & 0.937 & 0.514 & 0.525 & 0.525 \\ 0.438 & 0.708 & 0.438 & 0.447 & 0.447 \\ 0.411 & 0.645 & 0.411 & 0.419 & 0.419 \\ 0.326 & 0.509 & 0.326 & 0.333 & 0.333 \\ 0.308 & 0.496 & 0.308 & 0.315 & 0.315 \end{pmatrix} \frac{1}{\text{cm}} \text{ and}$$

$$x^T = \begin{pmatrix} 0.355 & 7.398 & 0.238 & 0.506 & 0.38 \\ 0.234 & 2.023 & 0.157 & 0.334 & 0.25 \\ 0.199 & 1.528 & 0.134 & 0.284 & 0.213 \\ 0.187 & 1.393 & 0.125 & 0.266 & 0.2 \\ 0.148 & 1.099 & 0.099 & 0.211 & 0.158 \\ 0.14 & 1.07 & 0.094 & 0.2 & 0.15 \end{pmatrix},$$

such that the total mean free path for each gamma energy is

$$x_{total}^T = (8.877 \quad 2.998 \quad 2.358 \quad 2.17 \quad 1.716 \quad 1.654) \ .$$

The buildup factors and flux-to-dose rate conversion factors for each of the gamma energies are

$$B^T = (13.775 \quad 4.212 \quad 3.154 \quad 2.869 \quad 2.242 \quad 2.137) \text{ and}$$

$$K^T = \left(8.707 \times 10^{-4} \quad 1.719 \times 10^{-3} \quad 2.21 \times 10^{-3} \quad 2.421 \times 10^{-3} \quad 3.376 \times 10^{-3} \quad 3.725 \times 10^{-3}\right) \ .$$

The distances from each of the point-sources to the detector location are

$$R^T = (48.50 \quad 62.44 \quad 62.44) \text{ in},$$

such that the individual gamma dose contributions from each gamma energy and point sources are

$$D_{limitcomp}^T = \begin{pmatrix} 3.972 \times 10^{-5} & 8.575 \times 10^{-3} & 205.696 & 247.443 & 4.714 \times 10^{-3} & 9.503 \times 10^{-6} \\ 2.397 \times 10^{-5} & 5.174 \times 10^{-3} & 124.113 & 149.303 & 2.844 \times 10^{-3} & 5.734 \times 10^{-6} \\ 2.397 \times 10^{-5} & 5.174 \times 10^{-3} & 124.113 & 149.303 & 2.844 \times 10^{-3} & 5.734 \times 10^{-6} \end{pmatrix} \frac{\text{mrem}}{\text{hr}}.$$

Therefore, the total limiting dose of $D_{limit} = 1000 \frac{\text{mrem}}{\text{hr}}$ is obtained with an individual container point source activity of $A_{limit} = 1.609 \text{ Ci}$ for each of the $nn + 1 = 3$ sources.

5.4 Example Axial Gamma Calculation (Co-60)

HAC at 1m Point Source Gamma Shielding Analysis for the Shielded Container

Unit Definition:

Units not standard to MathCAD and additional constants are defined as follows:

$\gamma := Bq \cdot s$ $\quad \nu := \gamma$ $\quad Ci := 3.7 \cdot 10^{10} \cdot Bq$ $\quad eV := 1.60 \cdot 10^{-19} \cdot J$

$KeV := 1 \cdot 10^{3} \cdot eV$ $\quad MeV := 1 \cdot 10^{6} \cdot eV$ $\quad mrem := \frac{hr \cdot \gamma}{cm^{2} \cdot s}$

Mass Attentuation Coefficients from ANSI/ANS 6.4.3-1991:

Tabular mass attenuation coefficients as a function of gamma energy for 1) silicon, 2) chromium, 3) manganese, 4) iron, 5) nickel, and 6) lead are imported and assigned to the following variables:

$Ti := C:\backslash shieldcalc\backslash mass_attenuation_coefficients.xls$ $\quad i := 0 .. rows(Ti) - 1$

$Ei_i := Ti_{i,0} \cdot MeV$ $\quad \mu 1_{\rho_i} := Ti_{i,1} \cdot \frac{cm^2}{g}$ $\quad \mu 2_{\rho_i} := Ti_{i,2} \cdot \frac{cm^2}{g}$ $\quad \mu 3_{\rho_i} := Ti_{i,3} \cdot \frac{cm^2}{g}$

$\mu 4_{\rho_i} := Ti_{i,4} \cdot \frac{cm^2}{g}$ $\quad \mu 5_{\rho_i} := Ti_{i,5} \cdot \frac{cm^2}{g}$ $\quad \mu 6_{\rho_i} := Ti_{i,6} \cdot \frac{cm^2}{g}$

Iron Exposure Build Up Factor Coefficients from ANSI/ANS 6.4.3-1991:

Tabular iron exposure build up factor coefficients as function of gamma energy are imported and assigned to the following variables:

$Tj := C:\backslash shieldcalc\backslash iron_buildup_coefficients.xls$ $\quad j := 0 .. rows(Tj) - 1$

$Ej_j := Tj_{j,0} \cdot MeV$ $\quad a_j := Tj_{j,1}$ $\quad b_j := Tj_{j,2}$ $\quad c_j := Tj_{j,3}$ $\quad d_j := Tj_{j,4}$ $\quad x_{k_j} := Tj_{j,5}$

Flux to Dose Rate Conversion Factors from ANSI/ANS 6.1.1-1977:

Tabular flux to dose rate conversion factors as a function of gamma energy are imported and assigned to the following variables:

$Tk := C:\backslash shieldcalc\backslash flux_doserate_factors.xls$ $\quad k := 0 .. rows(Tk) - 1$

$Ek_k := Tk_{k,0} \cdot MeV$ $\quad Kk_k := Tk_{k,1}$

Shield Composition Definition:

The following table defines the thickness, density, and percent composition of the shielding materials. The percent composition is defined by 1) silicon, 2) chromium, 3) manganese, 4) iron, 5) nickel, and 6) lead.

$m := 0..2$

Tm :=

	Thickness	Density	%Silicon	%Chromium	%Manganese	%Iron	%Nickel	%Lead
	0	1	2	3	4	5	6	7
0	2.5000	7.8526	0.0000	0.0000	0.0000	100.0000	0.0000	0.0000
1	0.2500	8.0128	1.0000	19.0000	2.0000	68.0000	10.0000	0.0000
2	0.2500	8.0128	1.0000	19.0000	2.0000	68.0000	10.0000	0.0000
3	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
4	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
5	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000	0.0000
6								
7								

$t_m := Tm_{m,0} \cdot in$ $\quad p1_m := Tm_{m,2} \cdot \%$ $\quad p3_m := Tm_{m,4} \cdot \%$ $\quad p5_m := Tm_{m,6} \cdot \%$

$\rho_m := Tm_{m,1} \cdot \frac{g}{cm^3}$ $\quad p2_m := Tm_{m,3} \cdot \%$ $\quad p4_m := Tm_{m,5} \cdot \%$ $\quad p6_m := Tm_{m,7} \cdot \%$

Detector Distance and Dose Limit Definition:

The dose limit is defined as $D := 1000 \cdot mrem \cdot hr^{-1}$ when measured at a distance of 1 meter from the surface of the package. The following table defines the horizontal and vertical distance from each point source to the detector and calculates the resulting radial distance.

$nn := 2$ $\quad n := 0..nn$

Tn :=

	Horizontal	Vertical
	0	1
0	54.9375	0.0000
1	54.9375	0.0000
2	54.9375	0.0000
3	0.0000	0.0000
4	0.0000	0.0000
5	0.0000	0.0000
6	0.0000	0.0000
7	0.0000	0.0000
8	0.0000	0.0000
9	0.0000	0.0000
10	0.0000	0.0000
11	0.0000	0.0000
12	0.0000	0.0000
13	0.0000	0.0000

$R_n := \sqrt{\left(Tn_{n,0}\right)^2 + \left(Tn_{n,1}\right)^2} \cdot in$

Radionuclide Gamma Energy and Intensity:

For the specified radionuclide, the gamma energies and gamma intensities are imported and assigned to the following variables:

$Tl := C:\backslash shieldcalc\backslash radionuclide_data.xls$ $\quad$ $index := in0$

$$ll := \left| \begin{array}{l} i \leftarrow 0 \\ \text{while } Tl_{i,0} \neq index \\ \quad i \leftarrow i + 1 \\ ll \leftarrow Tl_{i,1} - 1 \end{array} \right. \qquad l := 0 .. ll$$

$$E := \left| \begin{array}{l} i \leftarrow 0 \\ \text{while } Tl_{i,0} \neq index \\ \quad i \leftarrow i + 1 \\ ll \leftarrow Tl_{i,1} - 1 \\ \text{for } j \in 0 .. ll \\ \quad E_j \leftarrow \left| \begin{array}{ll} 100 & \text{if } Tl_{i+j,2} < 100 \\ Tl_{i+j,2} & \text{otherwise} \end{array} \right. \\ E \cdot KeV \end{array} \right. \qquad I := \left| \begin{array}{l} i \leftarrow 0 \\ \text{while } Tl_{i,0} \neq index \\ \quad i \leftarrow i + 1 \\ ll \leftarrow Tl_{i,1} - 1 \\ \text{for } j \in 0 .. ll \\ \quad I_j \leftarrow Tl_{i+j,3} \\ I \cdot \% \end{array} \right.$$

Free Mean Path Calculations:

The following equations utilize mass attenuation coefficients as a function of gamma energy and the shield composition definition to calculate the number of mean free paths for each of the gamma energies and shield materials:

$$\mu_{m,l} := \left(\text{linterp}\left(Ei, \mu 1_\rho, E_l\right) \cdot \rho_m \cdot p1_m\right) + \left(\text{linterp}\left(Ei, \mu 2_\rho, E_l\right) \cdot \rho_m \cdot p2_m\right) + \left(\text{linterp}\left(Ei, \mu 3_\rho, E_l\right) \cdot \rho_m \cdot p3_m\right) \ldots$$
$$+ \left(\text{linterp}\left(Ei, \mu 4_\rho, E_l\right) \cdot \rho_m \cdot p4_m\right) + \left(\text{linterp}\left(Ei, \mu 5_\rho, E_l\right) \cdot \rho_m \cdot p5_m\right) + \left(\text{linterp}\left(Ei, \mu 6_\rho, E_l\right) \cdot \rho_m \cdot p6_m\right)$$

$$x_{m,l} := \mu_{m,l} \cdot t_m$$

$$x_{total_l} := \sum x^{\langle l \rangle}$$

Build Up Factor Calculations:

The following G-P geometric progression equations (ANSI/ANS 6.4.3-1991) is utilized along with iron buildup factor coefficients and the number of mean free paths to calculate a buildup factor for each of the gamma energies:

$$Kx_l := \mathrm{linterp}(Ej, c, E_l)\cdot\left(x_{total_l}\right)^{\mathrm{linterp}(Ej, a, E_l)} + \mathrm{linterp}(Ej, d, E_l)\cdot\frac{\tanh\left[\left(\frac{x_{total_l}}{\mathrm{linterp}(Ej, x_k, E_l)}\right) - 2\right] - \tanh(-2)}{1 - \tanh(-2)}$$

$$B_l := \begin{cases} 1 + \left(\mathrm{linterp}(Ej, b, E_l) - 1\right)\cdot\left[\frac{\left(Kx_l\right)^{x_{total_l}} - 1}{Kx_l - 1}\right] & \text{if } Kx_l \neq 1 \\ 1 + \left(\mathrm{linterp}(Ej, b, E_l) - 1\right)\cdot x_{total_l} & \text{if } Kx_l = 1 \end{cases}$$

Maximum Activity Calculations:

The following equations calculate the total gamma dose resulting from 1 Ci of the specified radionuclide, then ratio the total dose to calculate the activity required to meet the limiting dose rate. The limiting activity is then utilized to calculate the individual gamma dose contributions from each point source. The gamma dose calculations utilize Rockwell's methodology.

$$K_l := \mathrm{linterp}(Ek, Kk, E_l)$$

$$D_{ratiocomp_{l,n}} := \left[\frac{Ci\cdot I_l\cdot B_l\cdot K_l}{4\cdot\pi\cdot\left(R_n\right)^2}\right]\cdot e^{-x_{total_l}}$$

$$D_{ratiototal_n} := \sum D_{ratiocomp}\langle n\rangle$$

$$D_{ratiolimit} := \sum D_{ratiototal}$$

$$A_{limit} := \begin{cases} Ci\cdot\frac{D}{D_{ratiolimit}} & \text{if } D_{ratiolimit} \neq 0 \\ Ci\cdot 10^{99} & \text{otherwise} \end{cases}$$

$$D_{limitcomp_{l,n}} := \left[\frac{A_{limit}\cdot I_l\cdot B_l\cdot K_l}{4\cdot\pi\cdot\left(R_n\right)^2}\right]\cdot e^{-x_{total_l}}$$

$$D_{limittotal_n} := \sum D_{limitcomp}\langle n\rangle$$

$$D_{limit} := \sum D_{limittotal}$$

Summary of Results:

The gamma energies and intensities for radionuclide $\text{index} = 16$ are as follows:

$$E^T = \left(346.9300 \quad 826.2800 \quad 1.1732 \times 10^3 \quad 1.3325 \times 10^3 \quad 2.1588 \times 10^3 \quad 2.5050 \times 10^3\right) \text{KeV}$$

$$I^T = \left(7.6 \times 10^{-3} \quad 7.6 \times 10^{-3} \quad 99.974 \quad 99.986 \quad 1.11 \times 10^{-3} \quad 2 \times 10^{-6}\right) \%$$

The attenuation coefficients and the number of mean free paths for each of the gamma energies and shield materials are

$$\mu^T = \begin{pmatrix} 0.781 & 0.797 & 0.797 \\ 0.514 & 0.525 & 0.525 \\ 0.438 & 0.447 & 0.447 \\ 0.411 & 0.419 & 0.419 \\ 0.326 & 0.333 & 0.333 \\ 0.308 & 0.315 & 0.315 \end{pmatrix} \frac{1}{\text{cm}} \text{ and}$$

$$x^T = \begin{pmatrix} 4.958 & 0.506 & 0.506 \\ 3.266 & 0.334 & 0.334 \\ 2.782 & 0.284 & 0.284 \\ 2.607 & 0.266 & 0.266 \\ 2.069 & 0.211 & 0.211 \\ 1.958 & 0.2 & 0.2 \end{pmatrix},$$

such that the total mean free path for each gamma energy is

$$x_{total}^T = (5.97 \quad 3.934 \quad 3.35 \quad 3.14 \quad 2.491 \quad 2.358) .$$

The buildup factors and flux-to-dose rate conversion factors for each of the gamma energies are

$$B^T = (8.578 \quad 5.513 \quad 4.282 \quad 3.887 \quad 2.864 \quad 2.662) \text{ and}$$

$$K^T = \left(8.707 \times 10^{-4} \quad 1.719 \times 10^{-3} \quad 2.21 \times 10^{-3} \quad 2.421 \times 10^{-3} \quad 3.376 \times 10^{-3} \quad 3.725 \times 10^{-3}\right) .$$

The distances from each of the point-sources to the detector location are

$$R^T = (54.94 \quad 54.94 \quad 54.94) \text{ in},$$

such that the individual gamma dose contributions from each gamma energy and point sources are

$$D_{limitcomp}^T = \begin{pmatrix} 6.538 \times 10^{-4} & 6.359 \times 10^{-3} & 149.641 & 183.681 & 4.009 \times 10^{-3} & 8.466 \times 10^{-6} \\ 6.538 \times 10^{-4} & 6.359 \times 10^{-3} & 149.641 & 183.681 & 4.009 \times 10^{-3} & 8.466 \times 10^{-6} \\ 6.538 \times 10^{-4} & 6.359 \times 10^{-3} & 149.641 & 183.681 & 4.009 \times 10^{-3} & 8.466 \times 10^{-6} \end{pmatrix} \frac{\text{mrem}}{\text{hr}}.$$

Therefore, the total limiting dose of $D_{limit} = 1000 \frac{\text{mrem}}{\text{hr}}$ is obtained with an individual container point source activity of $A_{limit} = 2.983 \text{ Ci}$ for each of the $nn + 1 = 3$ sources.

5.5 Computer Run Record

	COMPUTER RUN RECORD
Computer Run Number	See Input File Names
Analysis Software	MCNP5 v1.40
Hardware Description	Windows XP/Xeon Processor QA verification for MCNP5 v1.40 has been performed for this hardware platform [5]. Models were executed on computer ADANC491TC.
Disk Storage Description	Compact Disc

```
MCNP5 Input/Output Files
09/28/2007  11:03 AM              3,601 A01Th230.i
09/28/2007  11:04 AM              3,606 A02Th232.i
09/28/2007  11:04 AM              3,611 A03Pa231.i
09/28/2007  11:05 AM              3,596 A04U232.i
09/28/2007  11:05 AM              3,600 A05U233.i
09/28/2007  11:05 AM              3,600 A06U234.i
09/28/2007  11:06 AM              3,600 A07U235.i
09/28/2007  11:06 AM              3,600 A08U236.i
09/28/2007  11:06 AM              3,600 A09U238.i
09/28/2007  11:07 AM              3,601 A10Np237.i
09/28/2007  11:07 AM              3,601 A11Pu236.i
09/28/2007  11:07 AM              3,601 A12Pu238.i
09/28/2007  11:08 AM              3,601 A13Pu239.i
09/28/2007  11:08 AM              3,601 A14Pu240.i
09/28/2007  11:08 AM              3,601 A15Pu241.i
09/28/2007  11:09 AM              3,601 A16Pu242.i
09/28/2007  11:09 AM              3,601 A17Pu244.i
09/28/2007  11:09 AM              3,601 A18Am241.i
09/28/2007  11:10 AM              3,609 A19Am242m.i
09/28/2007  11:10 AM              3,601 A20Am243.i
09/28/2007  11:11 AM              3,601 A21Cm240.i
09/28/2007  11:11 AM              3,601 A22Cm242.i
09/28/2007  11:11 AM              3,601 A23Cm243.i
09/28/2007  11:12 AM              3,601 A24Cm244.i
09/28/2007  11:12 AM              3,594 A25Cm245.i
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09/28/2007  11:14 AM              3,602 A30Cf249.i
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09/28/2007  11:15 AM              3,602 A33Cf252.i
09/28/2007  11:15 AM              3,602 A34Cf254.i
09/28/2007  11:16 AM              3,602 A35Es253.i
09/28/2007  11:16 AM              3,602 A36Es254.i
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09/28/2007  09:02 AM             3,550 R09U238.i
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09/28/2007  09:46 AM            30,278 R30Cf249.o
09/28/2007  09:47 AM            30,252 R31Cf250.o
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09/28/2007  09:52 AM            30,278 R36Es254.o
09/28/2007  09:53 AM            30,252 R37Es254m.o
```

	COMPUTER RUN RECORD
Analysis Software	EXCEL and MathCADv13.1
Disk Storage Description	Compact Disc

The following EXCEL spreadsheets are utilized in the analysis. SC_SHIELDCALC_HAC_13.1.XLS is embedded with a MathCADv13.1 application that computes the gamma activities. The spreadsheet is constructed to automatically loop through each of the gamma emitting isotopes. This spreadsheet also summarizes the results of the neutron activity calculations and combines the results. The remaining EXCEL spreadsheets are used as input to the gamma point-kernel analysis as implemented in MathCAD.

```
sc_shieldcalc_hac_13.1.xls
flux_doserate_factors.xls
iron_buildup_coefficients.xls
mass_attenuation_coefficients.xls
radionuclide_data.xls
```