Contract No. and Disclaimer:

Packaging Certification Program Methodology for Determining Dose Rates for Small Gram Quantities in Shipping Packagings.

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ABSTRACT

The Small Gram Quantity (SGQ) concept is based on the understanding that small amounts of hazardous materials, in this case radioactive materials (RAM), are significantly less hazardous than large amounts of the same materials. This paper describes a methodology designed to estimate an SGQ for several neutron and gamma emitting isotopes that can be shipped in a package compliant with 10 CFR Part 71 external radiation level limits regulations. These regulations require packaging for the shipment of radioactive materials, under both normal and accident conditions, to perform the essential functions of material containment, subcriticality, and maintain external radiation levels within the specified limits. By placing the contents in a helium leak-tight containment vessel, and limiting the mass to ensure subcriticality, the first two essential functions are readily met. Some isotopes emit sufficiently strong photon radiation that small amounts of material can yield a large dose rate outside the package. Quantifying the dose rate for a proposed content is a challenging issue for the SGQ approach. It is essential to quantify external radiation levels from several common gamma and neutron sources that can be safely placed in a specific packaging, to ensure compliance with federal regulations. The Packaging Certification Program (PCP) Methodology for Determining Dose Rate for Small Gram Quantities in Shipping Packagings provides bounding shielding calculations that define mass limits compliant with 10 CFR 71.47 for a set of proposed SGQ isotopes. The approach is based on energy superposition with dose response calculated for a set of spectral groups for a baseline physical packaging configuration. The methodology includes using the MCNP radiation transport code to evaluate a family of neutron and photon spectral groups using the 9977 shipping package and its associated shielded containers as the base case. This results in a set of multipliers for “dose per particle” for each spectral group. For a given isotope, the source spectrum is folded with the response for each group. The summed contribution from all isotopes determines the total dose from the RAM in the container.

INTRODUCTION

The Packaging Certification Program Methodology for Determining Dose Rate for Small Gram Quantities in Shipping Packagings provides bounding mass limits for a set of proposed SGQ isotopes. Calculations were performed to estimate external radiation levels for the 9977 shipping package, including the effects special shielded containers. The MCNP radiation transport code was used to develop a set of response multipliers (Green’s functions) for “dose per particle” for each neutron and photon spectral group. The source spectrum for each isotope generated using the ORIGEN-S and RASTA computer codes was folded with the response multipliers to generate the dose rate per gram of each isotope in the 9977 shipping package and its associated shielded containers.

The maximum amount of a single isotope that could be shipped within the regulatory limits contained in 10 CFR 71.47 for dose rate at the surface of the package is determined. If a package contains a mixture of isotopes, the acceptability for shipment can be determined by a sum of fractions approach. Furthermore, the results of this analysis can be easily extended to additional radioisotopes by simply evaluating the neutron and/or photon spectra of those isotopes and folding the spectral data with the Green's functions provided.

ANALYTIC METHODOLOGY

The Monte Carlo N–Particle (MCNP) code package [1] was used for three dimensional Monte Carlo transport calculations to determine the radiation absorbed dose rates outside the package under Normal Conditions of Transport (NCT). The neutron transport calculations were performed as a series of calculations starting a unit source in each group using the BUGLE forty seven group structure [2]. Similarly, photon transport calculations were performed as a series of calculations with a unit source in each group using a seventy seven group structure. Radiation source spectra and strengths were characterized using ORIGEN-S [3] and RASTA (Radiation Source Term Analysis) [4].

The 9977 is a single containment drum type package with a bolted flange closure and a 6 inch diameter right circular cylinder Containment Vessel (6CV) enclosed by LAST-A-FOAM® and Fiberfrax® insulation, aluminum load distribution fixtures, and a lid filled with TR-19® Block/Min K® 2000. Major materials of construction include stainless steel, polyurethane, and aluminum.

The drum consists of a SS outer shell with a SS liner, aluminum load distribution fixtures and miscellaneous other hardware, as shown in Figure 1. The drum is modeled as a right circular cylinder, simplifying some of the components. The simplifications are conservative since they place the source material closer to the surface of the drum being analyzed.

The 6CV is modeled as a cylindrical main portion with a conical transition at the top, a short upper cylinder, a Cone-Seal Plug, and the Cone-Seal Nut. Some of the components of the 6CV that are simple to model (e.g., the conical transition at the top) are included exactly. Other more complex components are modeled as simpler shapes.

Three shielded containers were modeled:

- The SGQ-SC1 Lead Shielded Container, a right circular cylinder 9.354" tall and 5.625" diameter with a 5.5" tall by 1.625" diameter cavity.
- The SGQ-SC2 Polyethylene Shielded, Container a right circular cylinder 12.25" tall and 5.875" in diameter with a 8.00" tall by 1.625" diameter cavity.
- The SGQ-SC3 Tungsten Shielded Container, a right circular cylinder 7.480" tall and 5.480" diameter with a 3.330" tall by 1.330" diameter cavity.

Representing the shielded containers as right circular cylinders simplifies the geometric modeling without impacting the calculated dose rates since the detectors are located on the source axis or mid-plane.

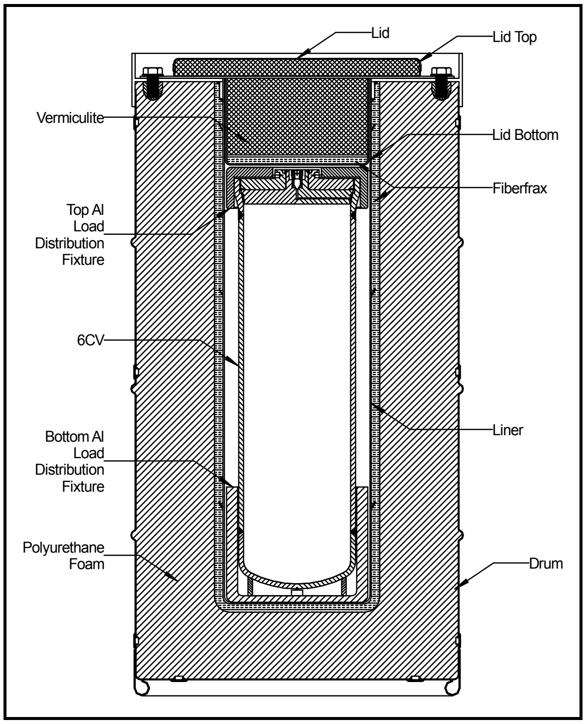

Figure 1. Model of 9977

Ten material compositions, other than the source region, are used to model the 9977 (Table 1).

Table 1. Material Compositions

	Nitronic® 60	SS 304L	6061 Aluminum	Foam	Vermiculite	Min-K® 2000	Fiberfrax®	Lead	Tungsten	HDPE
density (g/cc)	**7.584**	**7.9**	**2.7**	**0.256**	**0.3684**	**0.3204**	**0.2563**	**11.34**	**17**	**0.95**
Element	**Weight Percent**									
H				4.1		0.84				14.37
C		0.03		54.4		1.28				85.63
O				29.4	45.53	48.01	50.03			
N	0.14			12.1						
Si	4	1	0.6		21.22	37.71	23.14			
Ni	8.5	10							6	
Cr	17	19	0.2							
Mn	8	2								
Fe	62.36	67.895			5.4					
P		0.045								
S					0.18					
Ca					12.17					
Al			97.9		5.47	6.39	26.46			
Cu			0.28						4	
Mg			1		8.9					
Ti					0.82	5.53				
Na					1.09		0.37			
K					1.22					
Ga						0.15				
Sb						0.09				
W									90	
Pb								100		

A set of thirty six source isotopes was proposed by personnel at Lawrence Livermore National Laboratory [5]. Thirty five of these isotopes (with the exception being the very short lived Pd-103), listed in Table 2, were analyzed to determine the neutron and photon source strengths. The 47 group structure used to calculate neutron spectra is shown in Table 3. The 77 group structure used to calculate photon spectra is shown in Table 4. The sources were computed as a function of decay time using ORIGEN-S and without decay using RASTA. Actinides were analyzed with varying amounts of beryllium to determine the effects of (α-n) interactions on the neutron source. The added beryllium is modeled as a percent of the actinide mass (e.g., ten percent added beryllium in Pu-238 is modeled as 1 gram of Pu-238 and 0.1 grams of beryllium).

For NCT, the package is considered to be intact and dose rates are calculated on contact at the side, bottom, and top of the package.

The neutron source was modeled as a cylinder 2.5 cm in diameter by 4 cm tall. The source cylinders were run initially as void and then as reduced density plutonium oxide without shielded containers to determine the effects of self-shielding. The gamma source was modeled as a cylinder 1 cm in diameter by 2 cm tall. The source cylinders were run initially as void and then as reduced density (1 g/cc) iron[1] without shielded containers to determine the effects of self-shielding.

Table 2. Typical Sealed Source Isotopes

Nuclide	Half Life (days)	E-max (MeV)		Nuclide	Half Life (days)	E-max (MeV)	
		Gamma	Alpha			Gamma	Alpha
Ac-227	7.95E+03	0.242	4.953	Pm-147	9.58E+02	0.121	0
Am-241	1.58E+05	0.060	5.49	Po-210	1.38E+02	-	5.304
Am-241/Be	1.58E+05	0.060	5.49	Pu-238	3.20E+04	0.014	5.5
Am-243	2.69E+06	0.075	5.28	Pu-238/Be	3.20E+04	0.014	5.5
Am-243/Be	2.69E+06	0.075	5.28	Pu-239	8.80E+06	0.014	5.16
Cd-109	4.63E+02	0.088	0	Pu-239/Be	8.80E+06	0.014	5.16
Cf-252	9.66E+02	0.160	6.12	Pu-240	2.40E+06	0.014	5.168
Cm-244	6.61E+03	0.153	5.81	Pu-240/Be	2.40E+06	0.014	5.168
Cm-244/Be	6.61E+03	0.153	5.81	Pu-241	5.26E+03	0.060*	4.896
Cm-248	1.27E+08	-	5.078	Pu-241/Be	5.26E+03	0.060*	4.896
Co-60	1.93E+03	1.333	0	Pu-242	1.37E+08	0.014	4.9
Cs-137	1.10E+04	0.662*	0	Pu-242/Be	1.37E+08	0.014	4.9
Eu-152	4.93E+03	1.410	0	Ra-226	5.84E+05	0.186	4.78
Fe-59	4.45E+01	1.292	0	Ru-106	3.73E+02	0.622*	0
Gd-153	2.42E+02	0.103	0	Sc-46	8.38E+01	1.120	0
Hf-181	4.24E+01	0.482	0	Se-75	1.20E+02	0.401	0
Ho-166m	4.38E+05	0.951	0	Sm-145	3.40E+02	0.061	0
Ir-192	7.38E+01	0.612	0	Sr-90	1.05E+04	0.018*	0
Mn-54	3.12E+02	0.835	0	Tm-170	1.29E+02	0.084	0
Np-237	7.82E+08	0.415*	4.79	Yb-169	3.20E+01	0.198	0
Np-237/Be	7.82E+08	0.415*	4.79	Zn-65	2.44E+02	1.116	0
Pb-210	8.15E+03	0.047	0	Zr-95	6.40E+01	0.757	0

* Radiation from daughter isotopes

The source cylinders are placed at the bottom of the CV or at the base of the shielded containers with the cylinder centerline on the package centerline. The source region is modeled as a void for the shielded containers.

[1] Iron was chosen for the self-shielding material since it should be representative of several of the isotopes being analyzed.

Separate MCNP cases were run with a unit source in each of the neutron groups (Table 3) and each of the photon groups (Table 4).

Table 3. Neutron Spectra 47 Group Structure

Group	E-upper (MeV)	E-lower (MeV)	Group	E-upper (MeV)	E-lower (MeV)
47	1.00E-07	1.00E-11	23	4.98E-01	3.69E-01
46	4.14E-07	1.00E-07	22	6.08E-01	4.98E-01
45	8.76E-07	4.14E-07	21	7.43E-01	6.08E-01
44	1.86E-06	8.76E-07	20	8.21E-01	7.43E-01
43	5.04E-06	1.86E-06	19	1.00E+00	8.21E-01
42	1.07E-05	5.04E-06	18	1.35E+00	1.00E+00
41	3.73E-05	1.07E-05	17	1.65E+00	1.35E+00
40	1.01E-04	3.73E-05	16	1.92E+00	1.65E+00
39	2.14E-04	1.01E-04	15	2.23E+00	1.92E+00
38	4.54E-04	2.14E-04	14	2.35E+00	2.23E+00
37	1.58E-03	4.54E-04	13	2.37E+00	2.35E+00
36	3.35E-03	1.58E-03	12	2.47E+00	2.37E+00
35	7.10E-03	3.35E-03	11	2.73E+00	2.47E+00
34	1.50E-02	7.10E-03	10	3.01E+00	2.73E+00
33	2.19E-02	1.50E-02	9	3.68E+00	3.01E+00
32	2.42E-02	2.19E-02	8	4.97E+00	3.68E+00
31	2.61E-02	2.42E-02	7	6.07E+00	4.97E+00
30	3.18E-02	2.61E-02	6	7.41E+00	6.07E+00
29	4.09E-02	3.18E-02	5	8.61E+00	7.41E+00
28	5.74E-02	4.09E-02	4	1.00E+01	8.61E+00
27	1.11E-01	5.74E-02	3	1.22E+01	1.00E+01
26	1.83E-01	1.11E-01	2	1.42E+01	1.22E+01
25	2.97E-01	1.83E-01	1	1.73E+01	1.42E+01
24	3.69E-01	2.97E-01			

Note: Group structure is for RASTA. ORIGEN-S group structure differs for groups 1, 27, and 28. ORIGEN-ARP Group 1 Upper Bound is 1.96E+01, Group 27 Lower Bound/ Group 28 Upper Bound is 6.74 E-02.

Neutron and photon flux-to-dose-rate conversion factors were obtained from the 1977 American Nuclear Society Standard [6]. The 1977 values were used rather than those from the 1991 standard because the neutron dose conversion factors more closely reflect those provided in federal regulations, and the photon dose conversion factors more closely correspond to the response measured by instrumentation.

Dose points were modeled as both segmented surface tallies and point detectors located:

- At the bottom surface on the centerline of the package
- At the top surface on the centerline of the package
- At the side surface of the package axially centered on the source

Table 4. Photon Spectra 77 Group Structure

Group	E-upper (MeV)	E-lower (MeV)	Group	E-upper (MeV)	E-lower (MeV)	Group	E-upper (MeV)	E-lower (MeV)
77	0.020	0.010	51	0.280	0.270	25	0.900	0.875
76	0.030	0.020	50	0.290	0.280	24	0.925	0.900
75	0.040	0.030	49	0.300	0.290	23	0.950	0.925
74	0.050	0.040	48	0.325	0.300	22	0.975	0.950
73	0.060	0.050	47	0.350	0.325	21	1.000	0.975
72	0.070	0.060	46	0.375	0.350	20	1.250	1.000
71	0.080	0.070	45	0.400	0.375	19	1.500	1.250
70	0.090	0.080	44	0.425	0.400	18	1.750	1.500
69	0.100	0.090	43	0.450	0.425	17	2.000	1.750
68	0.110	0.100	42	0.475	0.450	16	2.250	2.000
67	0.120	0.110	41	0.500	0.475	15	2.500	2.250
66	0.130	0.120	40	0.525	0.500	14	2.750	2.500
65	0.140	0.130	39	0.550	0.525	13	3.000	2.750
64	0.150	0.140	38	0.575	0.550	12	3.500	3.000
63	0.160	0.150	37	0.600	0.575	11	4.000	3.500
62	0.170	0.160	36	0.625	0.600	10	4.500	4.000
61	0.180	0.170	35	0.650	0.625	9	5.000	4.500
60	0.190	0.180	34	0.675	0.650	8	5.500	5.000
59	0.200	0.190	33	0.700	0.675	7	6.000	5.500
58	0.210	0.200	32	0.725	0.700	6	6.500	6.000
57	0.220	0.210	31	0.750	0.725	5	7.000	6.500
56	0.230	0.220	30	0.775	0.750	4	7.500	7.000
55	0.240	0.230	29	0.800	0.775	3	8.000	7.500
54	0.250	0.240	28	0.825	0.800	2	9.000	8.000
53	0.260	0.250	27	0.850	0.825	1	10.000	9.000
52	0.270	0.260	26	0.875	0.850			

RESULTS

The neutron and photon sources were calculated in the group structure shown in Tables 3 and 4 using both ORIGEN-S and RASTA. The response from a unit source in each neutron and photon group was calculated using MCNP5 with each unshielded and shielded container configuration. Effects of self-shielding on both neutron and photon response were evaluated by including either plutonium oxide or iron in the source region for the case with no shielded container.

The calculated dose rate form one gram of each isotope considered is given in Table 5 (sources without impurities) and Table 6 (sources with impurities). Tables 7 and 8 present the maximum amount of a single isotope that could be shipped in the Model 9977 Package (or packagings having the same or larger external dimensions as well as similar structural materials) and have the external radiation level within the regulatory dose limits at the surface of the package. The estimates of the mass limits presented would also serve as conservative limits for both the Models 9975 and 9978 packages. If a package contains a mixture of isotopes, the acceptability for shipment can be determined by a sum of fractions approach.

For the cases of actinides mixed with light elements, beryllium is the bounding light element. The added beryllium (10 to 90 percent of the actinide mass) in the cases studied represents between 9 and 47 percent concentration of the total mixture mass. For beryllium concentrations larger than 50 percent, the increase in the neutron source term and dose rate tend to increase at a much lower rate than at concentrations lower than 50%. The intimately mixed actinide-beryllium form used in these models is very conservative and thus the limits presented in this report are practical bounds on the mass that can be safely shipped.

It should be noted that the SGQ masses presented in this report represent limits that would comply with the external radiation limits under 10CFR Part 71. They do not necessarily bound lower limits that may be required to comply with other factors such as heat load of the package.

Table 5. Maximum Dose Rate without Impurities

Isotope	Dose Rate from 1 gram of Isotope (rem/hr)			
	No Shielded Container	SC1 Lead Shielded Container	SC2 Polyethylene Shielded Container	SC3 Tungsten Shielded Container
Ac-227	4.84E+02	3.80E-01	1.41E+02	7.48E-02
Am-241	9.11E-03		1.73E-03	
Am-243	6.31E-01		1.65E-01	
Cd-109	3.52E+01	0.00E+00	7.18E+00	0.00E+00
Cf-252	1.05E+05		2.32E+04	
Cm-244	5.08E-01		1.09E-01	
Cm-248	1.83E+00		3.78E-01	
Co-60	6.10E+04	1.64E+03	2.09E+04	4.43E+02
Cs-137	1.24E+03	1.86E+00	4.01E+02	3.03E-01
Eu-152	4.52E+03	7.54E+01	1.49E+03	1.97E+01
Fe-59	1.28E+06	3.32E+04	4.37E+05	8.90E+03
Gd-153	1.44E+03	3.34E-33	3.14E+02	5.80E-38
Hf-181	2.21E+05	1.33E+01	6.69E+04	5.45E+00
Ho-166m	7.56E+01	2.78E-01	2.41E+01	5.47E-02
Ir-192	1.98E+05	2.92E+01	5.93E+04	4.83E+00
Mn-54	1.55E+05	9.84E+02	5.15E+04	1.88E+02
Np-237	6.27E-03		1.01E-03	
Pb-210	7.12E+00	6.67E-04	1.92E+00	1.16E-04
Pm-147	7.50E-01	9.72E-22	1.66E-01	3.21E-22
Po-210	9.25E-01	5.02E-03	3.06E-01	9.51E-04
Pu-238	1.80E-03		4.24E-04	
Pu-239	5.73E-05		1.61E-05	
Pu-240	6.74E-05		1.44E-05	
Pu-241	8.54E-03		1.66E-03	
Pu-242	7.94E-05		1.66E-05	
Ra-226	3.79E+01	8.61E-01	1.25E+01	2.73E-01
Ru-106	2.15E+04	8.59E+01	6.79E+03	2.35E+01
Sc-46	1.56E+06	2.31E+04	5.24E+05	5.43E+03
Se-75	1.30E+05	1.25E-01	3.56E+04	8.07E-03
Sm-145	1.39E+01	8.54E-05	2.63E+00	3.61E-05
Sr-90	6.92E+01	1.42E-01	2.00E+01	3.43E-02
Tm-170	7.43E+02	2.73E-02	1.97E+02	4.52E-03
Yb-169	9.43E+04	2.05E-02	2.46E+04	3.15E-03
Zn-65	1.06E+05	2.10E+03	3.60E+04	5.19E+02
Zr-95	4.33E+05	1.49E+03	1.42E+05	2.59E+02

Table 6. Maximum Dose Rate with Impurities

Isotope	Dose Rate from 1 gram of Isotope (rem/hr)								
	10% Be	20% Be	30% Be	40% Be	50% Be	60% Be	70% Be	80% Be	90% Be
No Shielded Container									
Am-241	1.37E-01	2.13E-01	2.64E-01	2.99E-01	3.26E-01	3.47E-01	3.64E-01	3.77E-01	3.89E-01
Am-243	7.59E-01	8.35E-01	8.86E-01	9.21E-01	9.48E-01	9.69E-01	9.86E-01	9.99E-01	1.01E+00
Cm-244	4.23E+00	6.43E+00	7.90E+00	8.94E+00	9.71E+00	1.03E+01	1.08E+01	1.12E+01	1.15E+01
Np-237	3.58E-03	3.59E-03	3.60E-03	3.60E-03	3.61E-03	3.61E-03	3.61E-03	3.61E-03	3.61E-03
Pu-238	6.42E-01	1.02E+00	1.28E+00	1.46E+00	1.59E+00	1.70E+00	1.78E+00	1.85E+00	1.90E+00
Pu-239	1.91E-03	3.00E-03	3.73E-03	4.24E-03	4.63E-03	4.93E-03	5.16E-03	5.36E-03	5.52E-03
Pu-240	6.90E-03	1.09E-02	1.36E-02	1.55E-02	1.69E-02	1.80E-02	1.89E-02	1.96E-02	2.02E-02
Pu-241	1.23E-01	1.90E-01	2.35E-01	2.67E-01	2.91E-01	3.09E-01	3.24E-01	3.36E-01	3.46E-01
Pu-242	1.78E-04	2.36E-04	2.74E-04	3.01E-04	3.21E-04	3.37E-04	3.49E-04	3.59E-04	3.68E-04
SC2 Polyethylene Shielded Container									
Am-241	4.13E-02	6.48E-02	8.04E-02	9.15E-02	9.98E-02	1.06E-01	1.11E-01	1.16E-01	1.19E-01
Am-243	1.67E-01	1.68E-01	1.69E-01	1.70E-01	1.70E-01	1.70E-01	1.70E-01	1.71E-01	1.71E-01
Cm-244	1.28E+00	1.97E+00	2.43E+00	2.76E+00	3.00E+00	3.19E+00	3.34E+00	3.47E+00	3.57E+00
Np-237	1.02E-03	1.02E-03	1.02E-03	1.03E-03	1.03E-03	1.03E-03	1.03E-03	1.03E-03	1.03E-03
Pu-238	1.98E-01	3.16E-01	3.95E-01	4.50E-01	4.92E-01	5.24E-01	5.50E-01	5.71E-01	5.89E-01
Pu-239	5.80E-04	9.14E-04	1.13E-03	1.29E-03	1.41E-03	1.50E-03	1.57E-03	1.63E-03	1.68E-03
Pu-240	2.10E-03	3.33E-03	4.14E-03	4.72E-03	5.15E-03	5.49E-03	5.75E-03	5.97E-03	6.15E-03
Pu-241	3.70E-02	5.79E-02	7.17E-02	8.16E-02	8.89E-02	9.46E-02	9.92E-02	1.03E-01	1.06E-01
Pu-242	4.64E-05	6.40E-05	7.55E-05	8.37E-05	8.98E-05	9.45E-05	9.82E-05	1.01E-04	1.04E-04

Note: 90% Beryllium is modeled as 1 gram of actinide and 0.9 grams of beryllium.

CONCLUSIONS

Bounding shielding calculations for a set of proposed SGQ isotopes were performed using the MCNP transport code to develop a set of response multipliers for "dose per particle" for each neutron and photon spectral group. The source spectrum for each isotope generated using the ORIGEN-S and RASTA computer codes was folded with the response multipliers to generate the dose rate per gram of each isotope in the 9977 shipping package and its associated shielded containers.

Table 7. Allowed Mass for Shipment – NCT without Impurities

Isotope	Allowed Mass of Isotope (g)			
	No Shielded Container	SC1 Lead Shielded Container	SC2 Polyethylene Shielded Container	SC3 Tungsten Shielded Container
Ac-227	4.1E-04	5.3E-01	1.4E-03	2.7E+00
Am-241	2.2E+01		1.2E+02	
Am-243	3.2E-01		1.2E+00	
Cd-109	5.7E-03	Unlimited*	2.8E-02	Unlimited*
Cf-252	1.9E-06		8.6E-06	
Cm-244	3.9E-01		1.8E+00	
Cm-248	1.1E-01		5.3E-01	
Co-60	3.3E-06	1.2E-04	9.6E-06	4.5E-04
Cs-137	1.6E-04	1.1E-01	5.0E-04	6.6E-01
Eu-152	4.4E-05	2.7E-03	1.3E-04	1.0E-02
Fe-59	1.6E-07	6.0E-06	4.6E-07	2.2E-05
Gd-153	1.4E-04	Unlimited*	6.4E-04	Unlimited*
Hf-181	9.1E-07	1.5E-02	3.0E-06	3.7E-02
Ho-166m	2.6E-03	7.2E-01	8.3E-03	3.7E+00
Ir-192	1.0E-06	6.8E-03	3.4E-06	4.1E-02
Mn-54	1.3E-06	2.0E-04	3.9E-06	1.1E-03
Np-237	3.2E+01		2.0E+02	
Pb-210	2.8E-02	3.0E+02	1.0E-01	1.7E+03
Pm-147	2.7E-01	Unlimited*	1.2E+00	Unlimited*
Po-210	2.2E-01	4.0E+01	6.5E-01	2.1E+02
Pu-238	1.1E+02		4.7E+02	
Pu-239	3.5E+03		1.2E+04	
Pu-240	3.0E+03		1.4E+04	
Pu-241	2.3E+01		1.2E+02	
Pu-242	2.5E+03		1.2E+04	
Ra-226	5.3E-03	2.3E-01	1.6E-02	7.3E-01
Ru-106	9.3E-06	2.3E-03	2.9E-05	8.5E-03
Sc-46	1.3E-07	8.7E-06	3.8E-07	3.7E-05
Se-75	1.5E-06	1.6E+00	5.6E-06	2.5E+01
Sm-145	1.4E-02	2.3E+03	7.6E-02	5.5E+03
Sr-90	2.9E-03	1.4E+00	1.0E-02	5.8E+00
Tm-170	2.7E-04	7.3E+00	1.0E-03	4.4E+01
Yb-169	2.1E-06	9.8E+00	8.1E-06	6.3E+01
Zn-65	1.9E-06	9.5E-05	5.5E-06	3.9E-04
Zr-95	4.6E-07	1.3E-04	1.4E-06	7.7E-04

* Unlimited denotes that the allowed mass of the isotope is restricted only by the capacity of the container.

Table 8. Allowed Actinide Mass for Shipment – NCT with Impurities

Isotope	Allowed Mass of Isotope (g)								
	10% Be	20% Be	30% Be	40% Be	50% Be	60% Be	70% Be	80% Be	90% Be
	No Shielded Container								
Am-241	1.5E+00	9.4E-01	7.6E-01	6.7E-01	6.1E-01	5.8E-01	5.5E-01	5.3E-01	5.1E-01
Am-243	2.6E-01	2.4E-01	2.3E-01	2.2E-01	2.1E-01	2.1E-01	2.0E-01	2.0E-01	2.0E-01
Cm-244	4.7E-02	3.1E-02	2.5E-02	2.2E-02	2.1E-02	1.9E-02	1.9E-02	1.8E-02	1.7E-02
Np-237	5.6E+01	5.6E+01	5.6E+01	5.6E+01	5.5E+01	5.5E+01	5.5E+01	5.5E+01	5.5E+01
Pu-238	3.1E-01	2.0E-01	1.6E-01	1.4E-01	1.3E-01	1.2E-01	1.1E-01	1.1E-01	1.0E-01
Pu-239	1.0E+02	6.7E+01	5.4E+01	4.7E+01	4.3E+01	4.1E+01	3.9E+01	3.7E+01	3.6E+01
Pu-240	2.9E+01	1.8E+01	1.5E+01	1.3E+01	1.2E+01	1.1E+01	1.1E+01	1.0E+01	9.9E+00
Pu-241	1.6E+00	1.1E+00	8.5E-01	7.5E-01	6.9E-01	6.5E-01	6.2E-01	6.0E-01	5.8E-01
Pu-242	1.1E+03	8.5E+02	7.3E+02	6.6E+02	6.2E+02	5.9E+02	5.7E+02	5.6E+02	5.4E+02
	SC2 Polyethylene Shielded Container								
Am-241	4.8E+00	3.1E+00	2.5E+00	2.2E+00	2.0E+00	1.9E+00	1.8E+00	1.7E+00	1.7E+00
Am-243	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00	1.2E+00
Cm-244	1.6E-01	1.0E-01	8.2E-02	7.3E-02	6.7E-02	6.3E-02	6.0E-02	5.8E-02	5.6E-02
Np-237	2.0E+02	2.0E+02	2.0E+02	2.0E+02	1.9E+02	1.9E+02	1.9E+02	1.9E+02	1.9E+02
Pu-238	1.0E+00	6.3E-01	5.1E-01	4.4E-01	4.1E-01	3.8E-01	3.6E-01	3.5E-01	3.4E-01
Pu-239	3.4E+02	2.2E+02	1.8E+02	1.5E+02	1.4E+02	1.3E+02	1.3E+02	1.2E+02	1.2E+02
Pu-240	9.5E+01	6.0E+01	4.8E+01	4.2E+01	3.9E+01	3.6E+01	3.5E+01	3.3E+01	3.2E+01
Pu-241	5.4E+00	3.5E+00	2.8E+00	2.5E+00	2.2E+00	2.1E+00	2.0E+00	1.9E+00	1.9E+00
Pu-242	4.3E+03	3.1E+03	2.6E+03	2.4E+03	2.2E+03	2.1E+03	2.0E+03	2.0E+03	1.9E+03

Note: 90% Beryllium is modeled as 1 gram of actinide and 0.9 grams of beryllium.

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