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Title: Cf-252 Characterization Documents

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Intended for: DOE Technical Documentation Supporting Cf-252 Characterization for WIPP
Environmental Programs

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Abstract for Cf-252 Technical Documents Authored by Vance and Associates

Six documents were written by Vance and Associates under contract to the Off-Site Source Recovery Project of Los Alamos National Laboratory. These Six documents provided the basis for characterization of Californium-252 sealed sources and for the packaging and manifesting of this material for disposal at the Waste Isolation Pilot Project. The Six documents are:

1. VA-OSR-10, Development of radionuclide distributions for Cf-252 sealed sources.
2. VA-OSR-11, Uncertainty analysis for Cf-252 sealed sources.
3. VA-OSR-12, To determine the radionuclides in the waste drums containing Cf-252 sealed source waste that are required to be reported under the requirements of the WIPP WAC and the TRAMPAC.
4. VA-OSR-13, Development of the spreadsheet for the radiological calculations for the characterization of Cf-252 sources.
5. VA-OSR-14, Relative importance of neutron-induced fission in Cf-252 sources.
6. VA-OSR-15, Determine upper bound of decay product inventories from a drum of Cf-252 sources.

These six documents provide the technical basis for the characterization of Cf-252 sources and will be part of the AK documentation required for submittal to the Central Characterization Project (CCP) of WIPP.

Calculation No. VA-OSR-10
Sheet No. 1 of 11

Vance & Associates
Calculation Cover Sheet

PROJECT: **Sealed Source Radiological Characterization**

SUBJECT:

Development of radionuclide distributions for Cf-252 sealed sources.

RECORD OF ISSUES

Rev. No.	Description	Total No. of Sheets	Last Sheet No.	Originator	Checker	Approved	Date
0	Original Issue	11	11	J. Vance	James H. Holleman	J. Vance	1/17/12

Purpose: To develop radionuclide distributions for Cf-252 sealed sources.

Approach: Californium-252 sources contain a mixture of californium isotopes and impurities of other radionuclides, principally Cs-137 and Sr-90. All of the transcurium materials were produced in the Radiochemical Engineering Development Center (REDC) at ORNL. Processing records that had been retrieved for the purposes of characterizing the REDC RH TRU waste (Reference 1) were used to derive the radionuclide distribution in the Cf-252 source material. The Cf-252 materials were routinely analyzed over the years as the materials were produced and ultimately manufactured into sealed sources for use by the end-users. The data represent the Cf-252 product composition after the final separation of impurities and shortly before the material was manufactured into sealed sources and shipped to the end-users. Twenty two of these batch analyses are extracted from Reference 1 in order to develop the radionuclide distribution in Cf-252 product. In addition one composition provided in an email from Mark Shilton at QSA-Global (Reference 1), a sealed source supplier, was also included. The data on the composition of the Cf-252 is in the form of mass percent of the californium isotopes. Fission products and the uranium, plutonium and curium impurities were typically below detection limits due to the multiple separations that take place in the production of the Cf-252 source materials.

Fission product impurities will exist due to spontaneous fission of Cf-249, Cf-250, Cf-252, Cm-245, Cm-246, and Cm-248. These are evaluated analytically as a function of time to determine the significance of their contribution to the impurity levels in the final Cf-252 product.

Computer Software: Microsoft Office 2010 Excel spreadsheet "Cf-252 Isotope Distribution."

Assumptions:

1. It is assumed that the sealed sources are manufactured reasonably close to the source material production dates.
2. This calculation addresses Cs-137, however, it is assumed that the findings apply to Sr-90 since the yields of the two radionuclides are similar.
3. Impurity levels are not reported for the uranium, plutonium and curium isotopes. Any impurities from these elements would have been removed many times in the processing of the californium product material. Any remaining impurities are negligible and are not considered further.
4. The thermal neutron induced fission yields for Cf-249 and Cm-245 were assumed for the spontaneous fission yields.

Calculations: Reference 2 provides the data from analyses of 22 batches of Cf-252 product shipped from ORNL. The analyses show:

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The following summarizes these data.

	A	B	C	D	E	F
11			Weight %			
12	Data Source	Date	Cf-249	Cf-250	Cf-251	Cf-252
13	Camp 37	1/10/1972	0.21%	11.43%	3.72%	84.64%
14	Camp 35 Rework	5/1/1972	7.72%	10.88%	3.31%	78.09%
15	Camp 37 Rework	11/11/1972	12.26%	12.28%	3.92%	71.55%
16	Camp 38	11/28/1972	6.01%	12.14%	3.45%	78.40%
17	Camp 39 Targets + Rework	2/6/1973	1.81%	8.55%	2.41%	86.62%
18	Camp 51 Rework	9/2/1976	11.92%	9.75%	2.99%	75.35%
19	Camp 51 Targets + Rework	9/19/1976	0.65%	7.93%	2.13%	88.79%
20	Camp 51 + Turf Rework	1/5/1977	7.00%	11.28%	3.63%	78.06%
21	Camp 52 Targets	1/10/1977	0.03%	7.85%	2.09%	88.62%
22	Camp 53 Rework	8/14/1977	11.17%	8.99%	2.65%	77.18%
23	Camp 53	8/31/1977	0.34%	7.70%	2.15%	88.86%
24	Camp 53	9/21/1977	0.79%	7.67%	2.10%	88.94%
25	Camp 54	1/10/1978	1.46%	8.25%	2.31%	87.86%
26	Camp 56 and 57 Rework	1/14/1980	10.97%	8.41%	2.55%	78.07%
27	Camp 63 Targets and Rework	10/8/1982	1.65%	7.92%	2.19%	87.33%
28	Camp 65 Rework	10/15/1985	2.92%	7.38%	2.14%	87.41%
29	Camp 67 Targets	2/2/1987	1.93%	7.21%	2.09%	88.40%
30	Camp 69 Targets	12/31/1994	4.83%	9.67%	2.81%	82.68%
31	Camp 70	4/22/1996	1.77%	8.71%	2.46%	87.05%
32	Camp 71	10/12/1998	0.32%	10.18%	2.84%	86.67%
33	Camp 72	9/11/2000	0.75%	8.68%	2.44%	87.59%
34	Camp 73	2/6/2003	0.79%	8.21%	2.40%	88.61%
35	Shilton email	6/9/2003	3.37%	8.64%	2.59%	85.39%

The isotopic distribution of the Cf-252 material on a g/g of Cf-252 is calculated as follows;

	A	B	G	H	I	J
11			g/g Cf-252			
12	Data Source	Date	Cf-249	Cf-250	Cf-251	Cf-252
13	Camp 37	1/10/1972	2.48E-03	1.35E-01	4.39E-02	1.00E+00
14	Camp 35 Rework	5/1/1972	9.88E-02	1.39E-01	4.24E-02	1.00E+00
15	Camp 37 Rework	11/11/1972	1.71E-01	1.72E-01	5.48E-02	1.00E+00
16	Camp 38	11/28/1972	7.66E-02	1.55E-01	4.40E-02	1.00E+00
17	Camp 39 Targets + Rework	2/6/1973	2.09E-02	9.87E-02	2.79E-02	1.00E+00
18	Camp 51 Rework	9/2/1976	1.58E-01	1.29E-01	3.96E-02	1.00E+00

19	Camp 51 Targets + Rework	9/19/1976	7.30E-03	8.93E-02	2.40E-02	1.00E+00
20	Camp 51 + Turf Rework	1/5/1977	8.97E-02	1.45E-01	4.65E-02	1.00E+00
21	Camp 52 Targets	1/10/1977	3.84E-04	8.86E-02	2.36E-02	1.00E+00
22	Camp 53 Rework	8/14/1977	1.45E-01	1.16E-01	3.43E-02	1.00E+00
23	Camp 53	8/31/1977	3.83E-03	8.67E-02	2.42E-02	1.00E+00
24	Camp 53	9/21/1977	8.89E-03	8.62E-02	2.36E-02	1.00E+00
25	Camp 54	1/10/1978	1.66E-02	9.39E-02	2.63E-02	1.00E+00
26	Camp 56 and 57 Rework	1/14/1980	1.41E-01	1.08E-01	3.27E-02	1.00E+00
27	Camp 63 Targets and Rework	10/8/1982	1.89E-02	9.07E-02	2.51E-02	1.00E+00
28	Camp 65 Rework	10/15/1985	3.34E-02	8.44E-02	2.45E-02	1.00E+00
29	Camp 67 Targets	2/2/1987	2.19E-02	8.16E-02	2.36E-02	1.00E+00
30	Camp 69 Targets	12/31/1994	5.85E-02	1.17E-01	3.40E-02	1.00E+00
31	Camp 70	4/22/1996	2.04E-02	1.00E-01	2.83E-02	1.00E+00
32	Camp 71	10/12/1998	3.68E-03	1.17E-01	3.27E-02	1.00E+00
33	Camp 72	9/11/2000	8.53E-03	9.90E-02	2.79E-02	1.00E+00
34	Camp 73	2/6/2003	8.87E-03	9.26E-02	2.70E-02	1.00E+00
35	Shilton email	6/9/2003	3.95E-02	1.01E-01	3.04E-02	1.00E+00
			5.02E-02	1.10E-01	3.22E-02	1.00E+00

The cell display feature of Excel is shown below for the above calculations:

	A	B	G	H	I	J
11			g/g Cf-252			
12	Data Source	Date	Cf-249	Cf-250	Cf-251	Cf-252
13	Camp 37	1/10/1972	=C13/$F13	=D13/$F13	=E13/$F13	=F13/$F13
14	Camp 35 Rework	5/1/1972	=C14/$F14	=D14/$F14	=E14/$F14	=F14/$F14
15	Camp 37 Rework	11/11/1972	=C15/$F15	=D15/$F15	=E15/$F15	=F15/$F15
16	Camp 38	11/28/1972	=C16/$F16	=D16/$F16	=E16/$F16	=F16/$F16
17	Camp 39 Targets + Rework	2/6/1973	=C17/$F17	=D17/$F17	=E17/$F17	=F17/$F17
18	Camp 51 Rework	9/2/1976	=C18/$F18	=D18/$F18	=E18/$F18	=F18/$F18
19	Camp 51 Targets + Rework	9/19/1976	=C19/$F19	=D19/$F19	=E19/$F19	=F19/$F19
20	Camp 51 + Turf Rework	1/5/1977	=C20/$F20	=D20/$F20	=E20/$F20	=F20/$F20
21	Camp 52 Targets	1/10/1977	=C21/$F21	=D21/$F21	=E21/$F21	=F21/$F21
22	Camp 53 Rework	8/14/1977	=C22/$F22	=D22/$F22	=E22/$F22	=F22/$F22
23	Camp 53	8/31/1977	=C23/$F23	=D23/$F23	=E23/$F23	=F23/$F23
24	Camp 53	9/21/1977	=C24/$F24	=D24/$F24	=E24/$F24	=F24/$F24
25	Camp 54	1/10/1978	=C25/$F25	=D25/$F25	=E25/$F25	=F25/$F25
26	Camp 56 and 57 Rework	1/14/1980	=C26/$F26	=D26/$F26	=E26/$F26	=F26/$F26

27	Camp 63 Targets and Rework	10/8/1982	=C27/$F27	=D27/$F27	=E27/$F27	=F27/$F27
28	Camp 65 Rework	10/15/1985	=C28/$F28	=D28/$F28	=E28/$F28	=F28/$F28
29	Camp 67 Targets	2/2/1987	=C29/$F29	=D29/$F29	=E29/$F29	=F29/$F29
30	Camp 69 Targets	12/31/1994	=C30/$F30	=D30/$F30	=E30/$F30	=F30/$F30
31	Camp 70	4/22/1996	=C31/$F31	=D31/$F31	=E31/$F31	=F31/$F31
32	Camp 71	10/12/1998	=C32/$F32	=D32/$F32	=E32/$F32	=F32/$F32
33	Camp 72	9/11/2000	=C33/$F33	=D33/$F33	=E33/$F33	=F33/$F33
34	Camp 73	2/6/2003	=C34/$F34	=D34/$F34	=E34/$F34	=F34/$F34
35	Shilton email	6/9/2003	=C35/$F35	=D35/$F35	=E35/$F35	=F35/$F35
			=AVERAGE(G13:G35)	=AVERAGE(H13:H35)	=AVERAGE(I13:I35)	=AVERAGE(J13:J35)

The above average isotopic distribution is converted to Ci/Ci of Cf-252 by multiplying the g/g values by their respective specific activities, as follows:

	F	G	H	I	J
38		Ci/Ci Cf-252			
39		Cf-249	Cf-250	Cf-251	Cf-252
40	Average	3.82E-04	2.24E-02	9.48E-05	1.00E+00

	F	G	H	I	J
38		Ci/Ci Cf-252			
39		Cf-249	Cf-250	Cf-251	Cf-252
40	Average	=G36*D44/D47	=H36*D45/D47	=I36*D46/D47	1

The specific activity constants and other constants used in the following spontaneous fission calculations are given as follows:

	A	B	C	D	E	F	G	H
43		half-life, yrs	lambda, yr	SpA	Daughter	half-life, yrs	lambda, yr	SpA
44	Cf-249	350.6	1.98E-03	4.14	Cm-245	8500	8.15E-05	0.174
45	Cf-250	13.08	5.30E-02	111	Cm-246	4750	1.46E-04	0.311
46	Cf-251	900	7.70E-04	1.6	Cm-247	1.56E+07	4.44E-08	9.38E-05
47	Cf-252	2.639	2.63E-01	544	Cm-248	3.39E+05	2.04E-06	0.0043

Spontaneous Fission

Cesium-137 (and Sr-90) will be produced from spontaneous fission of Cf-249, Cf-250, Cf-252, Cm-245, Cm-246, and Cm-248. The following shows the fraction of decays that are spontaneous fissions and the cumulative Cs-137 fission yield for spontaneous fission from the LBL database (Reference 4).

	A	B	C	D	E	F	G	H
52		Cf-249	Cf-250	Cf-252	Cm-245	Cm-246	Cm-248	Cs-137
53	SF fraction of decays	4.40E-09	7.70E-04	3.09E-02	6.10E-09	2.63E-04	8.39E-02	
54	Cumulative Cs-137 yield	0.0562	0.0545	0.0502	0.0622	0.0676	0.0608	
55	lambda (yr-1)	1.98E-03	5.30E-02	2.63E-01	8.15E-05	1.46E-04	2.04E-06	0.023

The amount of Cs-137 produced as a function of time from spontaneous fission of radionuclide *i* is:

$$\frac{dN_{Cs-137}}{dt} = -\lambda_{Cs-137}N_{Cs-137} + f_i y_i \lambda_i N_i$$

where N is in atoms or atom density and the subscript i refers to the spontaneously fissioning californium radionuclides. The californium isotopes' atom densities are given by:

$$N_i(t) = N_i(t=0)e^{-\lambda_i t}$$

For the spontaneous fission of californium isotopes, the solution to these equations is:

$A_{Cs-137}(t) = \frac{\lambda_{Cs-137} f_i y_i A_i(t=0)}{\lambda_{Cs-137}-\lambda_i}\left[e^{-\lambda_i t} - e^{-\lambda_{Cs-137} t}\right]$ where A_j is the activity of the radionuclide.

$$A_i = N_i\lambda_i$$

For the spontaneous fission of the curium isotopes, the solution is a bit more complicated, since they are produced over time through the decay of the californium isotopes. The governing equations are:

$$\frac{dN_{Cs-137}}{dt} = -\lambda_{Cs\text{-}137}N_{Cs\text{-}137} + f_j y_j \lambda_j N_j$$

Where the subscript j refers to the spontaneously fissioning decay daughters.

$$dN_j/dt = -\lambda_j N_j + (1 - f_i)\lambda_i N_i$$

$$N_i(t) = N_i(t=0)\, e^{-\lambda_i t}$$

The analytical solution is:

$$N_j(t) = k_1 e^{-\lambda_j t} + k_2 e^{-\lambda_i t}$$

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$$N_{Cs\text{-}137}(t) = c_1 e^{-\lambda_{Cs\text{-}137} t} + c_2 e^{-\lambda_j t} + c_3 e^{-\lambda_i t}$$

.

The coefficients are:

$$k_2 = [(1-f_i)\lambda_i N_i\,(t=0)] \div (\lambda_j - \lambda_i) = [(1-f_i)A_i\,(t=0)] \div (\lambda_j - \lambda_i)$$

$$k_1 = N_j(t=0) - k_2 = [A_j\,(t=0)/\,\lambda_j] - k_2 = -k_2$$

$$c_2 = [f_j y_j \lambda_j k_1] \div (\lambda_{Cs\text{-}137} - \lambda_j)$$

$$c_3 = [f_j y_j \lambda_j k_2] \div (\lambda_{Cs\text{-}137} - \lambda_i)$$

and

$$c_1 = -(c_2 + c_3).$$

The Cs-137 activity from the spontaneous fission of the decay daughters is:

$$A_{Cs\text{-}137}(t) = \lambda_{Cs\text{-}137}\,[c_1 e^{-\lambda_{Cs\text{-}137} t} + c_2 e^{-\lambda_j t} + c_3 e^{-\lambda_i t}]$$

.

The following shows the amount of Cs-137 from spontaneous fission and from the initial impurity level as a function of time.

	A	B	C	D	E	F	G	H	I
56		Cs-137 from SF of Cf-249	Cs-137 from SF of Cf-250	Cs-137 from SF of Cf-252	Cs-137 from SF of Cm-245	Cs-137 from SF of Cm-246	Cs-137 from SF of Cm-248	Total Cs-137	
57	Years	Ci/Ci Cf-252	Ci/Ci Cf-252	Ci/Ci Cf-252	Ci/Ci Cf-252	Ci/Ci Cf-252	Ci/Ci Cf-252	Ci/Ci Cf-252	% from Cf-252
58	1	2.14E-15	2.08E-08	3.10E-05	1.35E-19	7.23E-13	1.03E-10	3.10E-05	99.9%
59	2	4.24E-15	4.01E-08	5.42E-05	5.34E-19	2.68E-12	3.81E-10	5.42E-05	99.9%
60	3	6.28E-15	5.79E-08	7.13E-05	1.19E-18	5.79E-12	7.89E-10	7.13E-05	99.9%
61	95	5.27E-14	2.05E-07	6.22E-05	1.45E-16	4.02E-10	2.08E-08	6.24E-05	99.6%
62	96	5.35E-14	2.03E-07	6.07E-05	1.52E-16	4.15E-10	2.12E-08	6.10E-05	99.6%
63	97	5.43E-14	2.01E-07	5.94E-05	1.59E-16	4.27E-10	2.16E-08	5.96E-05	99.6%

The calculations for the above Cs-137 impurity contribution by spontaneous fission are shown as follows:

	A	B	C	D
56		Cs-137 from SF of Cf-249	Cs-137 from SF of Cf-250	Cs-137 from SF of Cf-252
57	Years	Ci/Ci Cf-252	Ci/Ci Cf-252	Ci/Ci Cf-252
58	1	=($H$55*B$53*B$54*$G$40/($H$55-B$55))*(EXP(-B$55*$A58)-EXP(-$H$55*$A58))	=($H$55*C$53*C$54*$H$40/($H$55-C$55))*(EXP(-C$55*$A58)-EXP(-$H$55*$A58))	=($H$55*D$53*D$54*$J$40/($H$55-D$55))*(EXP(-D$55*$A58)-EXP(-$H$55*$A58))
59	2	=($H$55*B$53*B$54*$G$40/($H$55-B$55))*(EXP(-B$55*$A59)-EXP(-$H$55*$A59))	=($H$55*C$53*C$54*$H$40/($H$55-C$55))*(EXP(-C$55*$A59)-EXP(-$H$55*$A59))	=($H$55*D$53*D$54*$J$40/($H$55-D$55))*(EXP(-D$55*$A59)-EXP(-$H$55*$A59))
60	3	=($H$55*B$53*B$54*$G$40/($H$55-B$55))*(EXP(-B$55*$A60)-EXP(-$H$55*$A60))	=($H$55*C$53*C$54*$H$40/($H$55-C$55))*(EXP(-C$55*$A60)-EXP(-$H$55*$A60))	=($H$55*D$53*D$54*$J$40/($H$55-D$55))*(EXP(-D$55*$A60)-EXP(-$H$55*$A60))
95	38	=($H$55*B$53*B$54*$G$40/($H$55-B$55))*(EXP(-B$55*$A95)-EXP(-$H$55*$A95))	=($H$55*C$53*C$54*$H$40/($H$55-C$55))*(EXP(-C$55*$A95)-EXP(-$H$55*$A95))	=($H$55*D$53*D$54*$J$40/($H$55-D$55))*(EXP(-D$55*$A95)-EXP(-$H$55*$A95))
96	39	=($H$55*B$53*B$54*$G$40/($H$55-B$55))*(EXP(-B$55*$A96)-EXP(-$H$55*$A96))	=($H$55*C$53*C$54*$H$40/($H$55-C$55))*(EXP(-C$55*$A96)-EXP(-$H$55*$A96))	=($H$55*D$53*D$54*$J$40/($H$55-D$55))*(EXP(-D$55*$A96)-EXP(-$H$55*$A96))
97	40	=($H$55*B$53*B$54*$G$40/($H$55-B$55))*(EXP(-B$55*$A97)-EXP(-$H$55*$A97))	=($H$55*C$53*C$54*$H$40/($H$55-C$55))*(EXP(-C$55*$A97)-EXP(-$H$55*$A97))	=($H$55*D$53*D$54*$J$40/($H$55-D$55))*(EXP(-D$55*$A97)-EXP(-$H$55*$A97))

	A	E	F	G	H	I
56		Cs-137 from SF of Cm-245	Cs-137 from SF of Cm-246	Cs-137 from SF of Cm-248	Total Cs-137	
57	Years	Ci/Ci Cf-252	Ci/Ci Cf-252	Ci/Ci Cf-252	Ci/Ci Cf-252	% from Cf-252
58	1	=$H$55*(K$52*EXP(-$H$55*$A58)+K$53*EXP(-$E$55*$A58)+K$54*EXP(-B$55*$A58))	=$H$55*(L$52*EXP(-$H$55*$A58)+L$53*EXP(-$E$55*$A58)+L$54*EXP(-C$55*$A58))	=$H$55*(M$52*EXP(-$H$55*$A58)+M$53*EXP(-$E$55*$A58)+M$54*EXP(-D$55*$A58))	=SUM(B58:G58)	=D58/H58
59	2	=$H$55*($K$52*EXP(-$H$55*A59)+$K	=$H$55*(L$52*EXP(-$H$55*$A59)+L	=$H$55*(M$52*EXP(-$H$55*$A59)+M	=SUM(B59:G59)	=D59/H59

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		$53*EXP(-$E$55*A59)+$K$54*EXP(-$B$55*A59))	$53*EXP(-$E$55*$A59)+L$54*EXP(-C$55*$A59))	$53*EXP(-$E$55*$A59)+M$54*EXP(-D$55*$A59))		
60	3	=$H$55*($K$52*EXP(-$H$55*A60)+$K$53*EXP(-$E$55*A60)+$K$54*EXP(-$B$55*A60))	=$H$55*(L$52*EXP(-$H$55*$A60)+L$53*EXP(-$E$55*$A60)+L$54*EXP(-C$55*$A60))	=$H$55*(M$52*EXP(-$H$55*$A60)+M$53*EXP(-$E$55*$A60)+M$54*EXP(-D$55*$A60))	=SUM(B60:G60)	=D60/H60
61	95	=$H$55*($K$52*EXP(-$H$55*A95)+$K$53*EXP(-$E$55*A95)+$K$54*EXP(-$B$55*A95))	=$H$55*(L$52*EXP(-$H$55*$A95)+L$53*EXP(-$E$55*$A95)+L$54*EXP(-C$55*$A95))	=$H$55*(M$52*EXP(-$H$55*$A95)+M$53*EXP(-$E$55*$A95)+M$54*EXP(-D$55*$A95))	=SUM(B95:G95)	=D95/H95
62	96	=$H$55*($K$52*EXP(-$H$55*A96)+$K$53*EXP(-$E$55*A96)+$K$54*EXP(-$B$55*A96))	=$H$55*(L$52*EXP(-$H$55*$A96)+L$53*EXP(-$E$55*$A96)+L$54*EXP(-C$55*$A96))	=$H$55*(M$52*EXP(-$H$55*$A96)+M$53*EXP(-$E$55*$A96)+M$54*EXP(-D$55*$A96))	=SUM(B96:G96)	=D96/H96
63	97	=$H$55*($K$52*EXP(-$H$55*A97)+$K$53*EXP(-$E$55*A97)+$K$54*EXP(-$B$55*A97))	=$H$55*(L$52*EXP(-$H$55*$A97)+L$53*EXP(-$E$55*$A97)+L$54*EXP(-C$55*$A97))	=$H$55*(M$52*EXP(-$H$55*$A97)+M$53*EXP(-$E$55*$A97)+M$54*EXP(-D$55*$A97))	=SUM(B97:G97)	=D97/H97

The constants k_1, k_2, c_1, c_2 and c_3 are calculated as:

	J	K	L	M
49		For Cm-245	For Cm-246	For Cm-248
50	k1	2.01E-01	4.24E-01	3.69E+00
51	k2	-2.01E-01	-4.24E-01	-3.69E+00
52	c1	2.45E-14	-8.48E-08	-1.83E-06
53	c2	2.72E-13	4.81E-08	1.67E-06
54	c3	-2.96E-13	3.67E-08	1.61E-07

	J	K	L	M
49		For Cm-245	For Cm-246	For Cm-248
50	k1	=-K51	=-L51	=-M51
51	k2	=(1-B53)*G40/(E55-B55)	=(1-C53)*H40/(F55-C55)	=(1-D53)*J40/(G55-D55)
52	c1	=-(K53+K54)	=-(L53+L54)	=-(M53+M54)
53	c2	=E53*E54*E55*K50/(H55-E55)	=F53*F54*F55*L50/($H55-F55)	=G53*G54*G55*M50/($H55-G55)
54	c3	=E53*E54*E55*K51/(H55-B55)	=F53*F54*F55*L51/($H55-C55)	=G53*G54*G55*M51/($H55-D55)

Rev.	Originator	Date	Checker	Date
0	[illegible]	1/17/12	[illegible]	1/17/12

The above results indicate that greater than 99% of the Cs-137 (and Sr-90) impurity in the source material is generated from the spontaneous fissioning of Cf-252. This contribution is accounted for in the characterization spreadsheet using the time period from the date of manufacture to the date of shipment.

Neutron-Induced Fission

Another source of fission products is from fast neutron fission induced by neutrons emitted during spontaneous fission. Reference 3 assesses these and concludes that fission products from neutron-induced fission are negligible compared to the fission products from spontaneous fission.

Summary

The following summarizes the radionuclide distribution of Cf-252 source material on the date the source is manufactured.

Radionuclide	Concentration	Units
U-233	<LLD	Ci/Ci of Cm-244
U-234	<LLD	Ci/Ci of Cm-244
U-235	<LLD	Ci/Ci of Cm-244
U-238	<LLD	Ci/Ci of Cm-244
Pu-238	<LLD	Ci/Ci of Cm-244
Pu-239	<LLD	Ci/Ci of Cm-244
Pu-240	<LLD	Ci/Ci of Cm-244
Pu-242	<LLD	Ci/Ci of Cm-244
Am-241	<LLD	Ci/Ci of Cm-244
Am-243	<LLD	Ci/Ci of Cm-244
Cm-244	<LLD	Ci/Ci of Cm-244
Cm-245	*	Ci/Ci of Cm-244
Cm-246	*	Ci/Ci of Cm-244
Cm-247	*	Ci/Ci of Cm-244
Cm-248	*	Ci/Ci of Cm-244
Cs-137	*	Ci/Ci of Cm-244
Sr-90	*	Ci/Ci of Cm-244

* To be determined by decay from the date of manufacture to the shipment date..

The relative californium isotope activities on the date of manufacture are:

Ci/Ci Cf-252

	Cf-249	Cf-250	Cf-251	Cf-252
Average	3.82E-04	2.24E-02	9.48E-05	1.00E+00

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0	J. N. Vance	1/17/12	James H Holderness	1/17/12

References:

1. email dated 2/12/2008 from Mark Shilton to Mike Pearson at LANL
2. *Central Characterization Project, Remote-Handled Transuranic Radiological Characterization Technical Report For Remote-Handled Transuranic Waste from Oak Ridge National Laboratory, Radiochemical Engineering Development Center*, CCP-AK-ORNL-501.
3. *Neutron-Induced Fission in Cf-252 Sources*, Calculation No. VA-OSR-14.
4. Lawrence Berkeley National Laboratory Isotopes Project Home Page, www.lbl.gov.

Attachments None

Rev.	Originator	Date	Checker	Date
0	J. N. Vance	1/17/12	James H Holdeman	1/17/12

Calculation No. VA-OSR-11
Sheet No. 1 of 8

Vance & Associates
Calculation Cover Sheet

PROJECT: **Sealed Source Radiological Characterization**

SUBJECT:

Uncertainty analysis for Cf-252 sealed sources.

RECORD OF ISSUES

Rev. No.	Description	Total No. of Sheets	Last Sheet No.	Originator	Checker	Approved	Date
0	Original Issue	8	8	James H Holloway	J. Vance	J. Vance	12/21/11

Purpose: To determine the uncertainty in the radionuclide inventory for waste drums containing Cf-252 sealed sources.

Approach: The radionuclide inventory of each sealed source is calculated based on the manufacturer's reported Cf-252 activity and the radionuclide distributions derived from REDC campaign data as described in Reference 1. Sources of uncertainty are summarized as follows.

Source of Uncertainty	Treatment
Manufacturer's uncertainty in mass or activity of source	± 5% per source, with 50 sources per drum
Uncertainty in californium isotopics	Based on variability among REDC campaigns
Uncertainty in other radionuclide impurities	Derived from californium decay
Instrument uncertainty	Included as one source of variability in the REDC data.
Uncertainty in decay time from REDC campaign measurement to source calibration date	± 6 months

Computer Software: Microsoft Office 2010 Excel spreadsheet "Cf-252 Uncertainty Analysis."

Assumptions:
1. It is assumed that manufacturing tolerances for all source manufacturers are similar.
2. It is assumed that each drum will contain 50, or more, sources of approximately equal size.
3. It is assumed that the sources are calibrated within six months of the REDC campaign measurement, and that the sources are, on average, at least ten years old at the time of shipping.

Calculations: Manufacturer's Source Material Uncertainty

The Cf-252 sources were either manufactured by ORNL directly or by other manufacturers that re-encapsulated ORNL-supplied source material. Reference 3 states that the ORNL-supplied material is calibrated to within 3%. Reference 2 is a product catalog for one manufacturer. It specifies that, for various sources (including Cf-252), the source material will be determined to within 2-5%. A value of 5% is assumed for each source. Each drum is assumed to contain at least 50 sources of approximately equal size. Therefore, the drum uncertainty due to the manufacturer's uncertainty is approximately $5/\sqrt{50}$ % = 0.71%.

It is assumed that sources from all manufacturers have a similar level of uncertainty.

Uncertainty in the Californium Isotopics

The isotopic composition of californium produced at ORNL REDC was measured for each production campaign. Reference 1 provides a summary of these for over 20 campaigns. The standard deviation among the campaigns is calculated based on the spreadsheet from Reference 1.

A	B	C	D	E	F	
Data Source	Date	Cf-249	Cf-250	Cf-251	Cf-252	12
Camp 37	1/10/1972	0.21%	11.43%	3.72%	84.64%	13
Camp 35 Rework	5/1/1972	7.72%	10.88%	3.31%	78.09%	14
Camp 73	2/6/2003	0.79%	8.21%	2.40%	88.61%	34
Shilton email	6/9/2003	3.37%	8.64%	2.59%	85.39%	35
Average		3.94%	9.12%	2.67%	84.01%	36
Standard Deviation		4.18%	1.54%	0.57%	5.33%	37
Relative Standard Deviation		106.2%	16.9%	21.4%	6.3%	38

A	B	C	D	E	F	
Data Source	Date	Cf-249	Cf-250	Cf-251	Cf-252	12
Camp 37	1/10/1972	0.21%	11.43%	3.72%	84.64%	13
Camp 35 Rework	5/1/1972	7.72%	10.88%	3.31%	78.09%	14
Camp 73	2/6/2003	0.79%	8.21%	2.40%	88.61%	34
Shilton email	6/9/2003	3.37%	8.64%	2.59%	85.39%	35
Average		=AVERAGE (C13:C35)	=AVERAGE (D13:D35)	=AVERAGE (E13:E35)	=AVERAGE (F13:F35)	36
Standard Deviation		=STDEV (C13:C35)	=STDEV (D13:D35)	=STDEV (E13:E35)	=STDEV (F13:F35)	37
Relative Standard Deviation		=C37/C36	=D37/D36	=E37/E36	=F37/F36	38

Note that for Cf-252, the campaign-to-campaign variability in the Cf-252 fraction is accounted for when the source is calibrated. Therefore, only the manufacturer's uncertainty applies to Cf-252.

It is assumed that 10 sources in each drum are from the same batch of californium and that each drum contains at least 50 sources of approximately equal size. As a result each drum has material from at least 5 batches of californium. The uncertainty is therefore estimated to be the above relative standard deviation values divided by the square root of five.

Uncertainty in Other Radionuclides

The impurity levels for other radionuclides are derived from decay of the californium isotopes. The following shows each radionuclide and its parent. The uncertainty in the radionuclide impurity level is the same as the relative uncertainty of the parent. Note that the fission products Cs-137 and Sr-90 are formed from the spontaneous fission of several californium and curium isotopes. As shown in Reference 1, however, the overwhelmingly dominant contributor is from the spontaneous fission of Cf-252.

Radionuclide	Parent	Uncertainty in Impurity Level
Cm-245	Cf-249	Same as Cf-249
Cm-246	Cf-250	Same as Cf-250
Cm-247	Cf-251	Same as Cf-251
Cm-248	Cf-252	Only manufacturer's uncertainty applies
Cs-137	Cf-252 via spontaneous fission	Only manufacturer's uncertainty applies
Sr-90	Cf-252 via spontaneous fission	Only manufacturer's uncertainty applies

<u>Uncertainty in the Decay Time from REDC Campaign Measurement to the Shipping Date</u>

The californium isotopics are taken from REDC campaign measurements at the time the californium targets are processed. It is assumed that the sources are fabricated and calibrated within a relatively short time after processing, since the principal source material (Cf-252) has a relatively short half-life. This decay is accounted for when calibrating the Cf-252 source and therefore this element of uncertainty does not apply to Cf-252 or its decay products. However, the levels of the other californium isotopes will be affected by decay and the uncertainty in the decay time. The uncertainty in the decay time is assumed to be six months. The uncertainty in the decaying radionuclide is:

$$\frac{\Delta A}{A} \cong \lambda \Delta t .$$

The uncertainty in the decay product is:

$$\frac{\Delta A}{A} \cong \frac{\lambda \Delta t e^{-\lambda t}}{(1-e^{-\lambda t})} .$$

The following shows the uncertainty in each radionuclide due to uncertainty in the decay time as a function of decay time.

Radionuclide	10-Year Decay	15-Year Decay	20-Year Decay
Cf-249	0.1%	0.1%	0.1%
Cf-250	2.6%	2.6%	2.6%
Cf-251	0.0%	0.0%	0.0%
Cf-252			
Cm-245	5.0%	3.3%	2.5%
Cm-246	3.8%	2.2%	1.4%
Cm-247	5.0%	3.3%	2.5%
Cm-248			
Cs-137			

Sr-90			

As an example, the calculation for the 10-year decay is shown below.

A	B	C		C
Radionuclide	lambda (yr-1)	Uncertainty Due to Decay Time	19	Uncertainty Due to Decay Time
Cf-249	1.98E-03	0.1%	20	=B20*$B$34
Cf-250	5.30E-02	2.6%	21	=B21*$B$34
Cf-251	7.70E-04	0.0%	22	=B22*$B$34
Cf-252	2.63E-01		23	
Cm-245		5.0%	24	=B20*$B$34*EXP(-B20*$B$33)/(1-EXP(-B20*$B$33))
Cm-246		3.8%	25	=B21*$B$34*EXP(-B21*$B$33)/(1-EXP(-B21*$B$33))
Cm-247		5.0%	26	=B22*$B$34*EXP(-B22*$B$33)/(1-EXP(-B22*$B$33))
Cm-248			27	
Cs-137			28	
Sr-90			29	

A	B	C	
Decay Time	10	year	33
Uncertainty in Decay Time	0.5	year	34

It is assumed that the sources are, on average, more than ten years old. Therefore, as an upper bound, the uncertainty values for a 10-year decay period are applied.

Total Uncertainty

The total uncertainty is calculated by combining the above in quadrature. As will be discussed in the next section, the manufacturer's uncertainty is common to all of the radionuclides and is treated separately. Therefore, the combination in quadrature is performed in two steps. First, all uncertainties other than the manufacturer's uncertainty are combined. Then the manufacturer's uncertainty is included to determine the total uncertainty.

A	B	C	D	E	F	
Radionuclide	Uncertainty in Manufacturer's Stated Source Size	Uncertainty in Californium Isotopics	Uncertainty in Decay Time	Total Uncertainty Excluding Mfg. Uncertainty	Total Uncertainty	3
Cm-245	0.71%	47.5%	5.0%	47.8%	47.8%	4
Cm-246	0.71%	7.6%	3.8%	8.5%	8.5%	5
Cm-247	0.71%	9.6%	5.0%	10.8%	10.8%	6
Cm-248	0.71%			0.0%	0.7%	7
Cf-249	0.71%	47.5%	0.1%	47.5%	47.5%	8
Cf-250	0.71%	7.6%	2.6%	8.0%	8.0%	9
Cf-251	0.71%	9.6%	0.0%	9.6%	9.6%	10
Cf-252	0.71%			0.0%	0.7%	11
Sr-90	0.71%			0.0%	0.7%	12
Cs-137	0.71%			0.0%	0.7%	13

A	B	C	
Radionuclide	Uncertainty in Manufacturer's Stated Source Size	Uncertainty in Californium Isotopics	3
Cm-245	=0.05/SQRT(50)	=C8	4
Cm-246	=B4	=C9	5
Cm-247	=B5	=C10	6
Cm-248	=B6		7
Cf-249	=B7	=1.061/SQRT(5)	8
Cf-250	=B8	=0.169/SQRT(5)	9
Cf-251	=B9	=0.214/SQRT(5)	10
Cf-252	=B10		11
Sr-90	=B11		12
Cs-137	=B12		13

A	E	F	
Radionuclide	Total Uncertainty Excluding Mfg. Uncertainty	Total Uncertainty	3
Cm-245	=SQRT(SUMSQ(C4:D4))	=SQRT(SUMSQ(B4,E4))	4
Cm-246	=SQRT(SUMSQ(C5:D5))	=SQRT(SUMSQ(B5,E5))	5
Cm-247	=SQRT(SUMSQ(C6:D6))	=SQRT(SUMSQ(B6,E6))	6
Cm-248	=SQRT(SUMSQ(C7:D7))	=SQRT(SUMSQ(B7,E7))	7
Cf-249	=SQRT(SUMSQ(C8:D8))	=SQRT(SUMSQ(B8,E8))	8
Cf-250	=SQRT(SUMSQ(C9:D9))	=SQRT(SUMSQ(B9,E9))	9
Cf-251	=SQRT(SUMSQ(C10:D10))	=SQRT(SUMSQ(B10,E10))	10
Cf-252	=SQRT(SUMSQ(C11:D11))	=SQRT(SUMSQ(B11,E11))	11
Sr-90	=SQRT(SUMSQ(C12:D12))	=SQRT(SUMSQ(B12,E12))	12
Cs-137	=SQRT(SUMSQ(C13:D13))	=SQRT(SUMSQ(B13,E13))	13

Uncertainty in Derived Parameters

The derived parameters TRU, FGE, PE-Ci, decay heat and total activity are summations over the individual radionuclides multiplied by appropriate weighting factors. The manufacturer's uncertainty applies to all radionuclides and therefore introduces a statistical dependency when determining the uncertainty in the derived parameters. Other parameters also introduce potential statistical dependences. The decay time introduces a dependency among the decaying radionuclides (either too much or too little decay) and the opposite dependency among the decay products (i.e., too little or too much production). These tend to be offsetting. The uncertainty in the californium isotopics affects both the californium isotope and its curium daughter. However, the activities of the curium isotopes are very small so that this dependency is negligible.

Therefore, the only dependency which will explicitly be considered is the manufacturer's uncertainty. The derived parameter is defined as:

$$Derived\ Parameter = \sum_i w_i A_i$$

where A_i is the activity of radionucide i.

The uncertainty in the derived parameter is therefore approximated as:

$$\sigma^2_{derived-parameter} \cong \sigma^2_{derived-parameter,mfg} + \sum_i w_i^2\, \sigma_i^2.$$

where σ_i includes all sources of uncertainty except for the manufacturer's uncertainty [see Column E above], and $\sigma_{derive\,d_{parameter},mfg}$ is the manufacturer's uncertainty (0.71%) multiplied by the value of the derived parameter. Note that the σ parameters in the above are uncertainty values in units of grams, curies, etc. and are not relative percent uncertainties.

References:

1. *Radionuclide Distribution Development: Cf-252 Sources*, Calculation No. VA-OSR-10.

2. *Monsanto Nuclear Products and Services*, Monsanto Research Corporation, 1974.

3. Cf-252 Neutron Source Information Sheet supplied by ORNL to source manufacturer (QSA Global).

Attachments: None.

Calculation No. VA-OSR-12
Sheet No. 1 of 7

Vance & Associates
Calculation Cover Sheet

PROJECT: **Sealed Source Radiological Characterization**

SUBJECT:

To determine the radionuclides in the waste drums containing Cf-252 sealed source waste that are required to be reported under the requirements of the WIPP WAC and the TRAMPAC.

RECORD OF ISSUES

Rev. No.	Description	Total No. of Sheets	Last Sheet No.	Originator	Checker	Approved	Date
0	Original Issue	7	7	G Vance	James H Holderness	J Vance	1/17/12

Purpose: To determine the radionuclides in the waste drums containing Cf-252 sealed source waste that are required to be reported under the requirements of the WIPP WAC and the TRAMPAC.

Approach: The WAC requires that, at a minimum, the ten WIPP-tracked radionuclides (U-233, U-234, U-238, Pu-238, Pu-239, Pu-240, Pu-242, Am-241, Sr-90 and Cs-137) be reported, if present in the waste. The Cf-252 source material contains no detectable uranium, plutonium or americium impurities. However, The source material does contain TRU radionuclides, namely; Cm-245, Cm-246, Cm-247, Cm-248, Cf-249 and Cf-251, are TRU radionuclides and accordingly they will be reported for the TRU determination of the waste drum.

The WAC also requires that, in addition to the ten identified, all other radionuclides that contribute to 95% of the radioactive hazard must be included. The TRAMPAC further requires that any radionuclide that contributes 1% to the decay heat, or that when aggregated contributes to 95% of the total decay heat, be reported.

The TRAMPAC also requires that the FGE be tracked for each waste package. It does not set a lower limit on what is reportable. It simply requires that the total FGE plus two sigma uncertainty bounds be less than the limiting value. Given that the uncertainty is typically on the order of 50-100%, it is unreasonable to report minute quantities of FGE. For this analysis, all radionuclides that, when aggregated, contribute to 95% of the FGE will be reported. This approach is consistent with the radiological hazard reporting requirements.

Radiological Hazard Determination

The A_2 values, where provided, were extracted from 49 CFR 173.435 for the radionuclides and divided into the scaling factor for each radionuclide to get the A_2 fractions. The A_2 fractions were summed for the required ten radionuclides and for the total of all radionuclides. The percentage of the total associated with the required ten radionuclides was calculated to determine if additional radionuclides needed to be reported.

Decay Heat Determination

The total decay heat is also calculated from the scaling factors in the same manner as the radiological hazard. The scaling factor for each radionuclide (Ci) is divided by the specific activity (Ci/g) and multiplied by the specific decay heat value (w/g). The specific activities and specific decay heat values are taken from the TRAMPAC. The percentage contribution of each of the radionuclides and the aggregate total decay heat is then calculated for each of the radionuclides to determine if additional radionuclides need to be reported.

Rev.	Originator	Date	Checker	Date
0	J. N. Vance	1/17/12	James H Holderness	1/17/12

FGE Determination

The total FGE is calculated from the relative radionuclide distribution. The activity quantities for each radionuclide (Ci) is divided by the specific activity (Ci/g) and multiplied by the FGE value (g/g). The specific activities and specific FGE values are taken from the TRAMPAC. The percentage contribution of each of the radionuclides and the aggregate total FGE is then calculated for each of the radionuclides to determine if additional radionuclides need to be reported.

The relative radionuclide distribution was used to evaluate the relative contributions of the individual radionuclides to the radiological hazard, the decay heat and the FGE. The relative distribution was developed by entering an arbitrary quantity of Cf-252 into the characterization spreadsheet (Reference 1) which accounts for decay of source material up to the date of shipment.

Computer Software: Microsoft Office 2010 Excel spreadsheet "Cf-252 Reporting Requirements."

Assumptions:

Calculations:

Evaluation of Radiological Hazard

A	B	C	D	E	F	
Radionuclide	Activity	A2 Value	Hazard	% Contribution to Hazard	Cumulative % Contribution to Hazard	8
Sr-90	5.34E-05	8.1	6.60E-06	0.0%	0.00%	9
Y-90	5.34E-05			0.0%	0.00%	10
Cs-137	4.98E-04	16	3.11E-05	0.0%	0.00%	11
Ba-137m	4.71E-04			0.0%	0.00%	12
Cm-245	3.46E-06	0.024	1.44E-04	0.0%	0.02%	13
Cm-246	2.25E-04	0.024	9.36E-03	0.8%	0.82%	14
Cm-247	4.74E-10	0.027	1.76E-08	0.0%	0.82%	15
Cm-248	4.22E-05	0.008	5.27E-03	0.5%	1.28%	16
Cf-249	1.99E-03	0.022	9.06E-02	7.8%	9.10%	17
Cf-251	5.08E-04	0.019	2.67E-02	2.3%	11.40%	18
Cf-250	4.03E-02	0.054	7.47E-01	64.5%	75.88%	19
Cf-252	2.26E-02	0.081	2.80E-01	24.1%	100.00%	20
		Total	1.16E+00			21

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0		1/17/12		1/17/12

A	B	C	D	E	F	
Radionuclide	Activity	A2 Value	Hazard	% Contribution to Hazard	Cumulative % Contribution to Hazard	8
Sr-90	5.34E-05	8.1	=B9/C9	=D9/$D$21	=E9	9
Y-90	5.34E-05			=D10/$D$21	=E10+F9	10
Cs-137	4.98E-04	16	=B11/C11	=D11/$D$21	=E11+F10	11
Ba-137m	4.71E-04			=D12/$D$21	=E12+F11	12
Cm-245	3.46E-06	0.024	=B13/C13	=D13/$D$21	=E13+F12	13
Cm-246	2.25E-04	0.024	=B14/C14	=D14/$D$21	=E14+F13	14
Cm-247	4.74E-10	0.027	=B15/C15	=D15/$D$21	=E15+F14	15
Cm-248	4.22E-05	0.008	=B16/C16	=D16/$D$21	=E16+F15	16
Cf-249	1.99E-03	0.022	=B17/C17	=D17/$D$21	=E17+F16	17
Cf-251	5.08E-04	0.019	=B18/C18	=D18/$D$21	=E18+F17	18
Cf-250	4.03E-02	0.054	=B19/C19	=D19/$D$21	=E19+F18	19
Cf-252	2.26E-02	0.081	=B20/C20	=D20/$D$21	=E20+F19	20
		Total	=SUM(D9:D20)			21

Based on the above results, beyond the WIPP-tracked fission products of Cs-137 and Sr-90 and the TRU radionuclides, Cf-250 and Cf-252 are required to be reported.

Evaluation of Decay Heat

All of the relative contributions of each radionuclide were divided by the specific activities and multiplied by the specific decay heat values to yield a decay heat fraction as shown.

G	H	I	J	K	L	M	
Radionuclide	Scaling Factor	Specific Decay Heat (w/g)	Specific Activity (Ci/g)	Decay Heat	% Contribution to Decay Heat	Cumulative % Decay Heat	8
Sr-90	5.34E-05	0.16	1.38E+02	6.20E-08	0.0%	0.0%	9
Y-90	5.34E-05	3.01E+03	5.44E+05	2.96E-07	0.0%	0.0%	10
Cs-137	4.98E-04	0.0974	8.80E+01	5.51E-07	0.0%	0.0%	11
Ba-137m	4.71E-04	2.12E+06	5.38E+08	1.86E-06	0.1%	0.1%	12
Cm-245	3.46E-06	5.77E-03	1.74E-01	1.15E-07	0.0%	0.1%	13
Cm-246	2.25E-04	1.02E-02	3.11E-01	7.37E-06	0.2%	0.3%	14
Cm-247	4.74E-10	2.98E-06	9.38E-05	1.51E-11	0.0%	0.3%	15
Cm-248	4.22E-05	5.53E-04	4.30E-03	5.42E-06	0.2%	0.5%	16
Cf-249	1.99E-03	1.54E-01	4.14E+00	7.41E-05	2.2%	2.7%	17
Cf-251	5.08E-04	5.89E-02	1.60E+00	1.87E-05	0.6%	3.3%	18

Cf-250	4.03E-02	4.12E+00	1.11E+02	1.50E-03	45.4%	48.7%	19
Cf-252	2.26E-02	4.06E+01	5.44E+02	1.69E-03	51.3%	100.0%	20
			Total	3.30E-03		100.0%	21

G	H	I	J	K	L	M	
Radionuclide	Scaling Factor	Specific Decay Heat (w/g)	Specific Activity (Ci/g)	Decay Heat	% Contribution to Decay Heat	Cumulative % Decay Heat	8
Sr-90	5.34E-05	0.16	1.38E+02	=H9*I9/J9	=K9/$K$21	=L9	9
Y-90	5.34E-05	3.01E+03	5.44E+05	=H10*I10/ J10	=K10/$K$21	=M9+L10	10
Cs-137	4.98E-04	0.0974	8.80E+01	=H11*I11/ J11	=K11/$K$21	=M10+L11	11
Ba-137m	4.71E-04	2.12E+06	5.38E+08	=H12*I12/ J12	=K12/$K$21	=M11+L12	12
Cm-245	3.46E-06	5.77E-03	1.74E-01	=H13*I13/ J13	=K13/$K$21	=M12+L13	13
Cm-246	2.25E-04	1.02E-02	3.11E-01	=H14*I14/ J14	=K14/$K$21	=M13+L14	14
Cm-247	4.74E-10	2.98E-06	9.38E-05	=H15*I15/ J15	=K15/$K$21	=M14+L15	15
Cm-248	4.22E-05	5.53E-04	4.30E-03	=H16*I16/ J16	=K16/$K$21	=M15+L16	16
Cf-249	1.99E-03	1.54E-01	4.14E+00	=H17*I17/ J17	=K17/$K$21	=M16+L17	17
Cf-251	5.08E-04	5.89E-02	1.60E+00	=H18*I18/ J18	=K18/$K$21	=M17+L18	18
Cf-250	4.03E-02	4.12E+00	1.11E+02	=H19*I19/ J19	=K19/$K$21	=M18+L19	19
Cf-252	2.26E-02	4.06E+01	5.44E+02	=H20*I20/ J20	=K20/$K$21	=M19+L20	20
			Total	=SUM(K9: K20)		=M20+L21	21

Based on the above results, beyond the fission products of Cs-137 and Sr-90 and the TRU radionuclides, Cf-250 and Cf-252 are required to be reported.

Evaluation of FGE

The relative contributions of each of the radionuclides were divided by the specific activities and multiplied by the FGE values to yield a FGE fraction as shown below.

N	O	P	Q	R	S	T	
Radionuclide	Scaling Factor	Specific Activity (Ci/g)	FGE (g/g)	FGE	% Contribution to FGE	Cumulative % FGE	8
Sr-90	5.34E-05	1.38E+02	0	0.00E+00	0.0%	0.0%	9

Y-90	5.34E-05	5.44E+05	0	0.00E+00	0.0%	0.0%	10
Cs-137	4.98E-04	8.80E+01	0	0.00E+00	0.0%	0.0%	11
Ba-137m	4.71E-04	5.38E+08	0	0.00E+00	0.0%	0.0%	12
Cm-245	3.46E-06	1.74E-01	15	2.98E-04	0.6%	0.6%	13
Cm-246	2.25E-04	3.11E-01	0	0.00E+00	0.0%	0.6%	14
Cm-247	4.74E-10	9.38E-05	0.5	2.53E-06	0.0%	0.6%	15
Cm-248	4.22E-05	4.30E-03	0	0.00E+00	0.0%	0.6%	16
Cf-249	1.99E-03	4.14E+00	45	2.17E-02	42.9%	43.5%	17
Cf-251	5.08E-04	1.60E+00	90	2.85E-02	56.5%	100.0%	18
Cf-250	4.03E-02	1.11E+02	0	0.00E+00	0.0%	100.0%	19
Cf-252	2.26E-02	5.44E+02	0	0.00E+00	0.0%	100.0%	20
				5.05E-02			21

N	O	P	Q	R	S	T	
Radionuclide	Scaling Factor	Specific Activity (Ci/g)	FGE (g/g)	FGE	% Contribution to FGE	Cumulative % FGE	8
Sr-90	5.34E-05	1.38E+02	0	=O9*Q9/P9	=R9/$R$21	=S9	9
Y-90	5.34E-05	5.44E+05	0	=O10*Q10/P10	=R10/$R$21	=T9+S10	10
Cs-137	4.98E-04	8.80E+01	0	=O11*Q11/P11	=R11/$R$21	=T10+S11	11
Ba-137m	4.71E-04	5.38E+08	0	=O12*Q12/P12	=R12/$R$21	=T11+S12	12
Cm-245	3.46E-06	1.74E-01	15	=O13*Q13/P13	=R13/$R$21	=T12+S13	13
Cm-246	2.25E-04	3.11E-01	0	=O14*Q14/P14	=R14/$R$21	=T13+S14	14
Cm-247	4.74E-10	9.38E-05	0.5	=O15*Q15/P15	=R15/$R$21	=T14+S15	15
Cm-248	4.22E-05	4.30E-03	0	=O16*Q16/P16	=R16/$R$21	=T15+S16	16
Cf-249	1.99E-03	4.14E+00	45	=O17*Q17/P17	=R17/$R$21	=T16+S17	17
Cf-251	5.08E-04	1.60E+00	90	=O18*Q18/P18	=R18/$R$21	=T17+S18	18
Cf-250	4.03E-02	1.11E+02	0	=O19*Q19/P19	=R19/$R$21	=T18+S19	19
Cf-252	2.26E-02	5.44E+02	0	=O20*Q20/P20	=R20/$R$21	=T19+S20	20
				=SUM(R9:R20)			21

Based on the above results, beyond the fission products of Cs-137 and Sr-90 and the TRU radionuclides, there are no addition radionuclides required to be reported based on FGE.

Conclusion

For the drums of Cf-252 sealed source waste, all of the measured isotopes of californium and their curium decay daughters need to be added to the WIPP-tracked radionuclides of Cs-137 and Sr-90. It is noted that there is no cutoff contribution for reporting the TRU radionuclides. Accordingly, all of the TRU radionuclides will be reported.

References:

1. *Radionuclide Calculation Spreadsheet for Cf-252 Sources*, Calculation No. VA-OSR-13.

Attachments: None.

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Vance & Associates
Calculation Cover Sheet

PROJECT: **Sealed Source Radiological Characterization**

SUBJECT:

Development of the spreadsheet for the radiological calculations for the characterization of Cf-252 sources.

RECORD OF ISSUES

Rev. No.	Description	Total No. of Sheets	Last Sheet No.	Originator	Checker	Approved	Date
0	Original Issue	21	21	J. Vance	James H. Holleman		
1	Added less-than values for source decay products	21	21	J. Vance	James H. Holleman		
2	Re-formatted Summary Report Output	20	20	J. Vance	James H. Holleman		

Purpose: Development of the spreadsheet for the radiological calculations for the characterization of Cf-252 sources.

Approach: The spreadsheet includes four worksheets; (1) an input sheet, (2) a decay calculation sheet, (3) a characterization summary report sheet and (4) a sheet that contains the constants that support the characterization calculations shown on the characterization summary report sheet. The input sheet displays the required information needed for the radiological characterization of a drum of waste. The following are the required inputs to this Input sheet:

- NDA/Radiological Characterization BDR#;
- Run Date for the drum characterization;
- Waste Stream Designation
- Container Number;
- Special Form Capsule Model Number; and
- Special Form Capsule Model Number (if a second SFC is added to the drum).
- Estimated Waste Weight
- The specific information on each individual source loaded into the SFC is:

  Source ID
  Cf-252 Content of Source in curies or grams
  Date of Source Manufacture

The Decay Calculation sheet adds the activities of the impurity radionuclides to each source and performs the decay calculation for each individual source loaded into the drum and finally sums the activities of all of the radionuclides in the drum. The Characterization Report sheet includes the radiological data and a summary of the resulting characterization in a suitable format for printing. The sheet also contains the uncertainty factors and the calculated uncertainty quantities for the radionuclides.

The Constants worksheet contains the conversion constants used in the characterization.

The Cf-252 isotopic distribution is taken from Reference 1. Specific activities taken from the CH-TRAMPAC [Ref. 2] were used to convert between curies and gram masses for the radionuclides. Values for the decay heat and Pu-239 fissile gram equivalent were also taken from the CH-TRAMPAC. The PE-Ci weighting factors were extracted from the CH TRU WAC for WIPP [Ref. 3].

The uncertainties are calculated using the methodology from Reference 4.

Computer Software: Microsoft Office 2010 Excel spreadsheet "Cf-252 Characterization Spreadsheet."

Assumptions:

Calculations: The "Source Input" sheet contains the following input information:

A	B	C	D	
	Container Characterization Report			1
		Cf-252 Sources		2
				3
Data Input Sheet				4
NDA/Radiological Characterization BDR#			LA11-OSR-CH-00X	5
Run Date			6/15/2011	6
Waste Stream Designation			Cf-252 Sealed Sources	7
Container Number			1234	8
Estimate Waste Weight, kg			7.7	9
				10
Source ID	**Manufacture Date**	**Source Contents, Curies**	**Source Contents, Grams**	11
91-191	1/1/1991		0.01	12
91-192	1/1/1991	3.00E-02		13
91-193	1/1/1991	3.00E-02		14
91-194	1/1/1991	3.00E-02		15
91-195	1/1/1991	3.00E-02		16
91-196	1/1/1991	3.00E-02		17
91-197	1/1/1991	3.00E-02		18
91-198	1/1/1991	3.00E-02		19
91-199	1/1/1991	3.00E-02		20
91-200	1/1/1978	3.00E-02		21

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The upper portion of the "Decay Calculations" worksheet contains the radionuclide-specific decay constants, specific activities and the impurity concentrations in Cf-252 source material.

B	C	D	E	F	G	H	I	J	K	L	M
		Cf-249	Cf-250	Cf-251	Cf-252	Cm-245	Cm-246	Cm-247	Cm-248	Cs-137	Sr-90
Decay Constants	yr^{-1}	1.98E-03	5.30E-02	7.70E-04	2.63E-01	8.15E-05	1.46E-04	4.44E-08	2.04E-06	2.30E-02	2.42E-02
Specific Activity	(Ci/g)	4.14	111	1.6	544	0.174	0.311	9.38E-05	0.0043	8.80E+01	1.38E+02
Concentrations from VA-OSR-10	(Ci/Ci of Cm-244)	3.82E-04	2.24E-02	9.48E-05	1.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

The lower portion of the "Decay Calculations" worksheet contains the calculations for the activities of all radionuclides, including impurities, in each source and the decay calculations.

Columns A and B are echoes of the input from the worksheet "Source Inputs" showing the Source ID and the Manufacture Date.

Column C echoes the inputs for each source from the worksheet "Source Inputs" for the activity input or converts the mass input to activity by multiplying the mass times the specific activity for Cf-252.

Columns D through G account for the decay of californium isotopes from the date of source manufacture to the run date.

Columns H through K account for the in-growth of the curium isotope daughters from the decay of the californium isotopes in the time period from the date of manufacture to the run date.

Column N calculates the decay period, in years, for each source by subtracting the "Source Manufacture Date" from the "Run Date" input by the user.

Columns L and M calculate the in-growth of the fission products of Cs-137 and Sr-90 from the spontaneous fissioning of Cf-252.

Row 8 sums the activities of the individual sources to yield the total activities of all of the radionuclides in the drum.

The lower portion of the "Decay Calculations" worksheet is shown in the following:

	A	B	C	D	E	F	G	H	I	J	K	L	M
8	Cf-252 Sources		Sum	2.09E-03	4.23E-02	5.33E-04	2.38E-02	3.63E-06	2.36E-04	4.98E-10	4.43E-05	5.22E-04	5.61E-05
9	Source ID	Manufacture Date	Source Contents, Curies	Cf-249	Cf-250	Cf-251	Cf-252	Cm-245	Cm-246	Cm-247	Cm-248	Cs-137	Sr-90
10	91-191	01/01/91	5.44E+00	1.99E-03	4.03E-02	5.08E-04	2.26E-02	3.46E-06	2.25E-04	4.74E-10	4.22E-05	4.98E-04	5.34E-05
11	91-192	01/01/91	3.00E-02	1.10E-05	2.22E-04	2.80E-06	1.25E-04	1.91E-08	1.24E-06	2.62E-12	2.33E-07	2.74E-06	2.95E-07
12	91-193	01/01/91	3.00E-02	1.10E-05	2.22E-04	2.80E-06	1.25E-04	1.91E-08	1.24E-06	2.62E-12	2.33E-07	2.74E-06	2.95E-07
13	91-194	01/01/91	3.00E-02	1.10E-05	2.22E-04	2.80E-06	1.25E-04	1.91E-08	1.24E-06	2.62E-12	2.33E-07	2.74E-06	2.95E-07
14	91-195	01/01/91	3.00E-02	1.10E-05	2.22E-04	2.80E-06	1.25E-04	1.91E-08	1.24E-06	2.62E-12	2.33E-07	2.74E-06	2.95E-07
15	91-196	01/01/91	3.00E-02	1.10E-05	2.22E-04	2.80E-06	1.25E-04	1.91E-08	1.24E-06	2.62E-12	2.33E-07	2.74E-06	2.95E-07
16	91-197	01/01/91	3.00E-02	1.10E-05	2.22E-04	2.80E-06	1.25E-04	1.91E-08	1.24E-06	2.62E-12	2.33E-07	2.74E-06	2.95E-07
17	91-198	01/01/91	3.00E-02	1.10E-05	2.22E-04	2.80E-06	1.25E-04	1.91E-08	1.24E-06	2.62E-12	2.33E-07	2.74E-06	2.95E-07
18	91-199	01/01/91	3.00E-02	1.10E-05	2.22E-04	2.80E-06	1.25E-04	1.91E-08	1.24E-06	2.62E-12	2.33E-07	2.74E-06	2.95E-07
19	91-200	01/01/91	3.00E-02	1.10E-05	2.22E-04	2.80E-06	1.25E-04	1.91E-08	1.24E-06	2.62E-12	2.33E-07	2.74E-06	2.95E-07

The cell display feature of Excel for the above calculations is given as follows:

	A	B	C	D	E	F	G	H	I	J	K	L	M
8	Cf-252 Sources		Sum	2.09E-03	4.23E-02	5.33E-04	2.38E-02	3.63E-06	2.36E-04	4.98E-10	4.43E-05	5.22E-04	5.61E-05
8	Cf-252 Sources		Sum	=SUM(D10:D107)	=SUM(E10:E107)	=SUM(F10:F107)	=SUM(G10:G107)	=SUM(H10:H107)	=SUM(I10:I107)	=SUM(J10:J107)	=SUM(K10:K107)	=SUM(L10:L107)	=SUM(M10:M107)
9	Source ID	Manufacture Date	Source Contents, Curies	Cf-249	Cf-250	Cf-251	Cf-252	Cm-245	Cm-246	Cm-247	Cm-248	Cs-137	Sr-90
10	='Source Inputs'!A12	='Source Inputs'!B12	='Source Inputs'!C12+544*'Source Inputs'!D12	=$C10*D$6*EXP(-D$4*$N10)	=$C10*E$6*EXP(-E$4*$N10)	=$C10*F$6*EXP(-F$4*$N10)	=$C10*G$6*EXP(-G$4*$N10)	=$C10*D$6*(1-EXP(-D$4*$N10))*H$4/D$4	=$C10*E$6*(1-EXP(-E$4*$N10))*I$4/E$4	=$C10*F$6*(1-EXP(-F$4*$N10))*J$4/F$4	=$C10*G$6*(1-EXP(-G$4*$N10))*K$4/G$4	=L$4*$P$3*$P$2*$C10/(L$4-$G$4)*(EXP(-$G$4*$N10)-EXP(-L$4*$N10))	=M$4*$P$4*$P$2*$C10/(M$4-$G$4)*(EXP(-$G$4*$N10)-EXP(-M$4*$N10))
11	='Source Inputs'!A13	='Source Inputs'!B13	='Source Inputs'!C13+	=$C11*D$6*EXP(-	=$C11*E$6*EXP(-	=$C11*F$6*EXP(-	=$C11*G$6*EXP(-	=$C11*D$6*(1-	=$C11*E$6*(1-	=$C11*F$6*(1-	=$C11*G$6*(1-	=L$4*$P$3*$P$2*	=M$4*$P$4*$P$2*

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			544*'Source Inputs'!D13	D$4*$N11)	E$4*$N11)	F$4*$N11)	G$4*$N11)	EXP(-D$4*$N11))*H$4/D$4	EXP(-E$4*$N11))*I$4/E$4	EXP(-F$4*$N11))*J$4/F$4	EXP(-G$4*$N11))*K$4/G$4	$C11/(L$4-$G$4)*(EXP(-$G$4*$N11)-EXP(-L$4*$N11))	$C11/(M$4-$G$4)*(EXP(-$G$4*$N11)-EXP(-M$4*$N11))
12	='Source Inputs'!A14	='Source Inputs'!B14	='Source Inputs'!C14+544*'Source Inputs'!D14	=$C12*D$6*EXP(-D$4*$N12)	=$C12*E$6*EXP(-E$4*$N12)	=$C12*F$6*EXP(-F$4*$N12)	=$C12*G$6*EXP(-G$4*$N12)	=$C12*D$6*(1-EXP(-D$4*$N12))*H$4/D$4	=$C12*E$6*(1-EXP(-E$4*$N12))*I$4/E$4	=$C12*F$6*(1-EXP(-F$4*$N12))*J$4/F$4	=$C12*G$6*(1-EXP(-G$4*$N12))*K$4/G$4	=L$4*$P$3*$P$2*$C12/(L$4-$G$4)*(EXP(-$G$4*$N12)-EXP(-L$4*$N12))	=M$4*$P$4*$P$2*$C12/(M$4-$G$4)*(EXP(-$G$4*$N12)-EXP(-M$4*$N12))
13	='Source Inputs'!A15	='Source Inputs'!B15	='Source Inputs'!C15+544*'Source Inputs'!D15	=$C13*D$6*EXP(-D$4*$N13)	=$C13*E$6*EXP(-E$4*$N13)	=$C13*F$6*EXP(-F$4*$N13)	=$C13*G$6*EXP(-G$4*$N13)	=$C13*D$6*(1-EXP(-D$4*$N13))*H$4/D$4	=$C13*E$6*(1-EXP(-E$4*$N13))*I$4/E$4	=$C13*F$6*(1-EXP(-F$4*$N13))*J$4/F$4	=$C13*G$6*(1-EXP(-G$4*$N13))*K$4/G$4	=L$4*$P$3*$P$2*$C13/(L$4-$G$4)*(EXP(-$G$4*$N13)-EXP(-L$4*$N13))	=M$4*$P$4*$P$2*$C13/(M$4-$G$4)*(EXP(-$G$4*$N13)-EXP(-M$4*$N13))
14	='Source Inputs'!A16	='Source Inputs'!B16	='Source Inputs'!C16+544*'Source Inputs'!D16	=$C14*D$6*EXP(-D$4*$N14)	=$C14*E$6*EXP(-E$4*$N14)	=$C14*F$6*EXP(-F$4*$N14)	=$C14*G$6*EXP(-G$4*$N14)	=$C14*D$6*(1-EXP(-D$4*$N14))*H$4/D$4	=$C14*E$6*(1-EXP(-E$4*$N14))*I$4/E$4	=$C14*F$6*(1-EXP(-F$4*$N14))*J$4/F$4	=$C14*G$6*(1-EXP(-G$4*$N14))*K$4/G$4	=L$4*$P$3*$P$2*$C14/(L$4-$G$4)*(EXP(-$G$4*$N14)-EXP(-L$4*$N14))	=M$4*$P$4*$P$2*$C14/(M$4-$G$4)*(EXP(-$G$4*$N14)-EXP(-M$4*$N14))
15	='Source Inputs'!A17	='Source Inputs'!B17	='Source Inputs'!C17+544*'Source Inputs'!D17	=$C15*D$6*EXP(-D$4*$N15)	=$C15*E$6*EXP(-E$4*$N15)	=$C15*F$6*EXP(-F$4*$N15)	=$C15*G$6*EXP(-G$4*$N15)	=$C15*D$6*(1-EXP(-D$4*$N15))*H$4/D	=$C15*E$6*(1-EXP(-E$4*$N15))*I$4/E	=$C15*F$6*(1-EXP(-F$4*$N15))*J$4/F	=$C15*G$6*(1-EXP(-G$4*$N15))*K$4/	=L$4*$P$3*$P$2*$C15/(L$4-$G$4)*(E	=M$4*$P$4*$P$2*$C15/(M$4-$G$4)*(E

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								$4	$4	$4	G$4	XP(-$G$4*$N15)-EXP(-L$4*$N15))	XP(-$G$4*$N15)-EXP(-M$4*$N15))
16	='Source Inputs'!A18	='Source Inputs'!B18	='Source Inputs'!C18+544*'Source Inputs'!D18	=$C16*D$6*EXP(-D$4*$N16)	=$C16*E$6*EXP(-E$4*$N16)	=$C16*F$6*EXP(-F$4*$N16)	=$C16*G$6*EXP(-G$4*$N16)	=$C16*D$6*(1-EXP(-D$4*$N16))*H$4/D$4	=$C16*E$6*(1-EXP(-E$4*$N16))*I$4/E$4	=$C16*F$6*(1-EXP(-F$4*$N16))*J$4/F$4	=$C16*G$6*(1-EXP(-G$4*$N16))*K$4/G$4	=L$4*$P$3*$P$2*$C16/(L$4-$G$4)*(EXP(-$G$4*$N16)-EXP(-L$4*$N16))	=M$4*$P$4*$P$2*$C16/(M$4-$G$4)*(EXP(-$G$4*$N16)-EXP(-M$4*$N16))
17	='Source Inputs'!A19	='Source Inputs'!B19	='Source Inputs'!C19+544*'Source Inputs'!D19	=$C17*D$6*EXP(-D$4*$N17)	=$C17*E$6*EXP(-E$4*$N17)	=$C17*F$6*EXP(-F$4*$N17)	=$C17*G$6*EXP(-G$4*$N17)	=$C17*D$6*(1-EXP(-D$4*$N17))*H$4/D$4	=$C17*E$6*(1-EXP(-E$4*$N17))*I$4/E$4	=$C17*F$6*(1-EXP(-F$4*$N17))*J$4/F$4	=$C17*G$6*(1-EXP(-G$4*$N17))*K$4/G$4	=L$4*$P$3*$P$2*$C17/(L$4-$G$4)*(EXP(-$G$4*$N17)-EXP(-L$4*$N17))	=M$4*$P$4*$P$2*$C17/(M$4-$G$4)*(EXP(-$G$4*$N17)-EXP(-M$4*$N17))
18	='Source Inputs'!A20	='Source Inputs'!B20	='Source Inputs'!C20+544*'Source Inputs'!D20	=$C18*D$6*EXP(-D$4*$N18)	=$C18*E$6*EXP(-E$4*$N18)	=$C18*F$6*EXP(-F$4*$N18)	=$C18*G$6*EXP(-G$4*$N18)	=$C18*D$6*(1-EXP(-D$4*$N18))*H$4/D$4	=$C18*E$6*(1-EXP(-E$4*$N18))*I$4/E$4	=$C18*F$6*(1-EXP(-F$4*$N18))*J$4/F$4	=$C18*G$6*(1-EXP(-G$4*$N18))*K$4/G$4	=L$4*$P$3*$P$2*$C18/(L$4-$G$4)*(EXP(-$G$4*$N18)-EXP(-L$4*$N18))	=M$4*$P$4*$P$2*$C18/(M$4-$G$4)*(EXP(-$G$4*$N18)-EXP(-M$4*$N18))
19	='Source Inputs'!A21	='Source Inputs'!B21	='Source Inputs'!C21+544*'Source Inputs'!D21	=$C19*D$6*EXP(-D$4*$N19)	=$C19*E$6*EXP(-E$4*$N19)	=$C19*F$6*EXP(-F$4*$N19)	=$C19*G$6*EXP(-G$4*$N19)	=$C19*D$6*(1-EXP(-D$4*$N19))*H$4/D$4	=$C19*E$6*(1-EXP(-E$4*$N19))*I$4/E$4	=$C19*F$6*(1-EXP(-F$4*$N19))*J$4/F$4	=$C19*G$6*(1-EXP(-G$4*$N19))*K$4/G$4	=L$4*$P$3*$P$2*$C19/(L$4-$G$4)*(EXP(-$G$4*$N19)-	=M$4*$P$4*$P$2*$C19/(M$4-$G$4)*(EXP(-$G$4*$N19)-

					EXP(-L$4*$N19))	EXP(-M$4*$N19))

The worksheet "Characterization Report" contains a summary of the radiological characterization for the drum. The upper portion of the worksheet is shown as follows:

A	B	C D	E	F G	
		Container Characterization Report			5
		CF-252 Sources		**Version 1.0**	6
					7
					8
NDA/Radiological Characterization BDR#		LA11-OSR-CH-00X			9
Run Date		11/15/2011			10
Waste Stream Designation		CF-252 Sealed Sources			11
Container Number		1234			12
Net Waste Weight		7.7	kg		13

Cells D9, D10, D11, D12 and D13 are echoes of the information entered by the user in the "Source Input" worksheet.

The Excel cell display feature of the above cells is shown below:

A	B	C D	E	F G	
		Container Characterization Report			5
		CF-252 Sources		**Version 1.0**	6
					7
					8
NDA/Radiological Characterization BDR#		='Source Inputs'!D5			9
Run Date		='Source Inputs'!D6			10
Waste Stream Designation		='Source Inputs'!D7			11
Container Number		='Source Inputs'!D8			12
Net Waste Weight		='Source Inputs'!D9	kg		13

The lower portion of the worksheet "Characterization Report" presents the summary of the radiological data for the drum of sealed sources.

Column C gives the total activity of each radionuclide in the drum from the summation calculations in Row 8 of the worksheet "Decay Calculations".

The remaining columns are calculated by multiplying the activity (or mass) by the appropriate constants from the worksheet "Constants". The bottom portion of the worksheet calculates the radiological parameters for the drum.

A	B	C	D	E	F	G	H	I	
Nuclide	% Type A Limit	Activity (Ci)	Grams	FGE	PECi	Watts	Uncertainty in Curies	Uncertainty in Grams	15
Cm-245	1.51E-06	3.63E-06	2.09E-05	3.13E-04	3.87E-06	1.21E-07	1.74E-06	9.98E-06	16
Cm-246	9.82E-05	2.36E-04	7.58E-04	0.00E+00	2.51E-04	7.73E-06	2.00E-05	6.43E-05	17
Cm-247	6.15E-10	4.98E-10	5.31E-06	2.65E-06	4.79E-10	1.58E-11	5.38E-11	5.74E-07	18
Cm-248	8.20E-03	4.43E-05	1.03E-02	0.00E+00	1.64E-04	5.69E-06	3.13E-07	7.28E-05	19
Cf-249	2.58E-03	2.09E-03	5.05E-04	2.27E-02	2.25E-03	7.78E-05	9.94E-04	2.40E-04	20
Cf-250	7.84E-03	4.23E-02	3.81E-04	0.00E+00	1.83E-02	1.57E-03	3.40E-03	3.07E-05	21
Cf-251	2.80E-04	5.33E-04	3.33E-04	3.00E-02	5.85E-04	1.96E-05	5.11E-05	3.20E-05	22
Cf-252	1.70E+00	2.38E-02	4.37E-05	0.00E+00	6.09E-03	1.77E-03	1.68E-04	3.09E-07	23
Sr-90	6.93E-04	5.61E-05	4.07E-07	0.00E+00	0.00E+00	6.50E-08	3.97E-07	2.87E-09	24
Cs-137	9.67E-04	5.22E-04	5.94E-06	0.00E+00	0.00E+00	5.78E-07	3.69E-06	4.20E-08	25
U-233	0.00E+00	<5.97E-17	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	26
U-234	0.00E+00	<4.36E-21	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	27
U-238	0.00E+00	<1.12E-16	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	28
Pu-238	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	29
Pu-239	0.00E+00	<2.34E-15	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	30
Pu-240	0.00E+00	<6.56E-14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	31
Pu-242	0.00E+00	<3.68E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	32
Am-241	0.00E+00	<2.87E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	33
Totals	1.72	6.96E-02	1.23E-02	5.30E-02	2.76E-02	3.46E-03			34
									35

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2	[signature]	7/31/12	James H Hollen[signature]	7/31/12

	Value	(one Sigma)	Units	36
TRU Alpha Activity Concentration	3.78E+02	4.16E+01	nCi/g	37
TRU Alpha Activity	2.91E-03	3.20E-04	Ci	38
Total Pu-239 Equiv Activity	2.76E-02	2.11E-04	Ci	39
Total Pu-239 Fissile Gram Equiv	5.30E-02	1.62E-02	G	40
Total Decay Heat	3.46E-03	2.46E-05	W	41
Volume Activity	6.13E-04	4.38E-06	Ci/L	42

The Excel cell display feature for the above calculations is shown as follows:

A	B	C	D	E	F	G	H	I	
Nuclide	**% Type A Limit**	**Activity (Ci)**	**Grams**	**FGE**	**PECi**	**Watts**	**Uncertainty in Curies**	**Uncertainty in Grams**	**15**
Cm-245	=C16/Constants!G8*100	='Decay Calculations'!H8	=C16/Constants!B8	=D16*Constants!D8	=C16/Constants!E8	=D16*Constants!F8	=Constants!$B46*C16	=Constants!$B46*D16	16
Cm-246	=C17/Constants!G9*100	='Decay Calculations'!I8	=C17/Constants!B9	=D17*Constants!D9	=C17/Constants!E9	=D17*Constants!F9	=Constants!$B47*C17	=Constants!$B47*D17	**17**
Cm-247	=C18/Constants!G10*100	='Decay Calculations'!J8	=C18/Constants!B10	=D18*Constants!D10	=C18/Constants!E10	=D18*Constants!F10	=Constants!$B48*C18	=Constants!$B48*D18	18
Cm-248	=C19/Constants!G11*100	='Decay Calculations'!K8	=C19/Constants!B11	=D19*Constants!D11	=C19/Constants!E11	=D19*Constants!F11	=Constants!$B49*C19	=Constants!$B49*D19	19
Cf-249	=C20/Constants!G12*100	='Decay Calculations'!D8	=C20/Constants!B12	=D20*Constants!D12	=C20/Constants!E12	=D20*Constants!F12	=Constants!$B50*C20	=Constants!$B50*D20	20
Cf-250	=C21/Constants!G13*100	='Decay Calculations'!E8	=C21/Constants!B13	=D21*Constants!D13	=C21/Constants!E13	=D21*Constants!F13	=Constants!$B51*C21	=Constants!$B51*D21	21
Cf-251	=C22/Constants!G14*100	='Decay Calculations'!F8	=C22/Constants!B14	=D22*Constants!D14	=C22/Constants!E14	=D22*Constants!F14	=Constants!$B52*C22	=Constants!$B52*D22	22
Cf-252	=C23/Constants!G15*100	='Decay Calculations'!G8	=C23/Constants!B15	=D23*Constants!D15	=C23/Constants!E15	=D23*Constants!F15	=Constants!$B53*C23	=Constants!$B53*D23	23
Sr-90	=C24/Constants!G16*100	='Decay Calculations'!M8	=C24/Constants!B16	=D24*Constants!D16	0	=D24*Constants!F16	=Constants!$B54*C24	=Constants!$B54*D24	24

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2	J. N. Vance	7/31/12	[illegible]	7/31/12

Cs-137	=C25/Constants!G17*100	='Decay Calculations'!L8	=C25/Constants!B17	=D25*Constants!D17	0	=D25*Constants!F17	=Constants!$B55*C25	=Constants!$B55*D25	25
U-233	0	<5.97E-17	0	0	0	0	0	0	26
U-234	0	<4.36E-21	0	0	0	0	0	0	27
U-238	0	<1.12E-16	0	0	0	0	0	0	28
Pu-238	0	0	0	0	0	0	0	0	29
Pu-239	0	<2.34E-15	0	0	0	0	0	0	30
Pu-240	0	<6.56E-14	0	0	0	0	0	0	31
Pu-242	0	<3.68E-08	0	0	0	0	0	0	32
Am-241	0	<2.87E-07	0	0	0	0	0	0	33
Totals	=SUM(B16:B33)	=SUM(C16:C33)	=SUM(D16:D33)	=SUM(E16:E33)	=SUM(F16:F33)	=SUM(G16:G33)			34
									35

	Value	(one Sigma)	Units	
				36
TRU Alpha Activity Concentration	=(G38*1000000000)/(D13*1000)	=G37*Constants!B56	nCi/g	37
TRU Alpha Activity	=C16+C17+C18+C19+C20+C22	=G38*Constants!B56	Ci	38
Total Pu-239 Equiv Activity	=F34	=G39*Constants!B58	Ci	39
Total Pu-239 Fissile Gram Equiv	=E34	=G40*Constants!B57	g	40
Total Decay Heat	=G34	=G41*Constants!B59	W	41
Volume Activity	=C34/(Constants!B25/1000)	=G42*Constants!B60	Ci/L	42

The worksheet "Constants" contains the constants used in the radiological calculations.
The upper portion of the worksheets contains the radiological conversion constants.

A	B	C	D	E	F	G	
Radionuclide or Derived Parameter	SpA (CI/g)	TRU/Ci	FGE/g of radionuclide	PE-Ci/Ci	W/g of radionuclide	Type A Limit	7
Cm-245	1.74E-01	1.00E+00	1.50E+01	0.94	5.77E-03	2.40E+02	8

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2	J. N. Vance	7/31/12	James H Holleman	7/31/12

Cm-246	3.11E-01	1.00E+00	0.00E+00	0.94	1.02E-02	2.40E+02	9
Cm-247	9.38E-05	1.00E+00	5.00E-01	1.04	2.98E-06	8.10E+01	10
Cm-248	4.30E-03	1.00E+00	0.00E+00	0.27	5.53E-04	5.40E-01	11
Cf-249	4.14E+00	1.00E+00	4.50E+01	0.93	1.54E-01	8.10E+01	12
Cf-250	1.11E+02	0.00E+00	0.00E+00	2.32	4.12E+00	5.40E+02	13
Cf-251	1.60E+00	1.00E+00	9.00E+01	0.91	5.89E-02	1.90E+02	14
Cf-252	5.44E+02	0.00E+00	0.00E+00	3.90	4.06E+01	1.40E+00	15
Sr-90	1.38E+02	0.00E+00	0.00E+00	0.00	1.60E-01	8.10E+00	16
Cs-137	8.80E+01	0.00E+00	0.00E+00	0.00	9.74E-02	5.40E+01	17

The lower portion includes the uncertainties derived in Reference 4 as follows:

A	B	
Radionuclide or Derived Parameter	Source Total Uncertainty	45
Cm-245	47.8%	46
Cm-246	8.5%	47
Cm-247	10.8%	48
Cm-248	0.7%	49
Cf-249	47.5%	50
Cf-250	8.0%	51
Cf-251	9.6%	52
Cf-252	0.7%	53
Sr-90	0.7%	54
Cs-137	0.7%	55
TRU	11.0%	56
FGE	30.5%	57
PE-Ci	0.8%	58
Decay Heat	0.7%	59
Total Activity	0.7%	60

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The uncertainties in the derived parameters of TRU, FGE, PE-Ci, Decay Heat and Total Activity are calculated below, based on the description in Reference 4.

	A	B	C	D	E	I	J
82		Activity	Total Uncertainty Excluding Mfg. Uncertainty	SpA (CI/g)	TRU	TRU Activity	TRU Uncertainty
83		Ci		(Ci/g)	(Ci/Ci)	(Ci/Ci Cf-252)	(Ci/Ci Cf-252)
84	Cm-245	1.51E-06	47.8%	1.74E-01	1.00E+00	1.51E-06	7.23E-07
85	Cm-246	9.82E-05	8.5%	3.11E-01	1.00E+00	9.82E-05	8.31E-06
86	Cm-247	6.15E-10	10.8%	9.38E-05	1.00E+00	6.15E-10	6.63E-11
87	Cm-248	8.20E-03	0.0%	4.30E-03	1.00E+00	8.20E-03	0.00E+00
88	Cf-249	2.58E-03	47.5%	4.14E+00	1.00E+00	2.58E-03	1.23E-03
89	Cf-250	7.84E-03	8.0%	1.11E+02	0.00E+00	0.00E+00	0.00E+00
90	Cf-251	2.80E-04	9.6%	1.60E+00	1.00E+00	2.80E-04	2.68E-05
91	Cf-252	1.70E+00	0.0%	5.44E+02	0.00E+00	0.00E+00	0.00E+00
92	Sr-90	6.93E-04	0.0%	1.38E+02	0.00E+00	0.00E+00	0.00E+00
93	Cs-137	9.67E-04	0.0%	8.80E+01	0.00E+00	0.00E+00	0.00E+00
94					Totals	1.12E-02	1.23E-03
95							10.99%
96					Mfg. Uncertainty		0.71%
97					Total Uncertainty		11.02%

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	A	B	C	D	E	I	J
82		Activity	Total Uncertainty Excluding Mfg. Uncertainty	SpA (Cl/g)	TRU	TRU Activity	TRU Uncertainty
83		Ci		(Ci/g)	(Ci/Ci)	(Ci/Ci Cf-252)	(Ci/Ci Cf-252)
84	Cm-245	1.51E-06	47.8%	1.74E-01	1.00E+00	=E84*B84	=I84*C84
85	Cm-246	9.82E-05	8.5%	3.11E-01	1.00E+00	=E85*B85	=I85*C85
86	Cm-247	6.15E-10	10.8%	9.38E-05	1.00E+00	=E86*B86	=I86*C86
87	Cm-248	8.20E-03	0.0%	4.30E-03	1.00E+00	=E87*B87	=I87*C87
88	Cf-249	2.58E-03	47.5%	4.14E+00	1.00E+00	=E88*B88	=I88*C88
89	Cf-250	7.84E-03	8.0%	1.11E+02	0.00E+00	=E89*B89	=I89*C89
90	Cf-251	2.80E-04	9.6%	1.60E+00	1.00E+00	=E90*B90	=I90*C90
91	Cf-252	1.70E+00	0.0%	5.44E+02	0.00E+00	=E91*B91	=I91*C91
92	Sr-90	6.93E-04	0.0%	1.38E+02	0.00E+00	=E92*B92	=I92*C92
93	Cs-137	9.67E-04	0.0%	8.80E+01	0.00E+00	=E93*B93	=I93*C93
94					Totals	=SUM(I84:I93)	=SQRT(SUMSQ(J84:J93))
95							=J94/I94
96					Mfg. Uncertainty		=0.0071
97					Total Uncertainty		=SQRT(SUMSQ(J95:J96))

	A	B	C	D	F	K	L
82		Activity	Total Uncertainty Excluding Mfg. Uncertainty	SpA (Cl/g)	FGE	FGE	FGE Uncertainty
83		Ci		(Ci/g)	(FGE/g)	(FGE/Ci Cf-252)	(FGE/Ci Cf-252)

84	Cm-245	1.51E-06	47.8%	1.74E-01	1.50E+01	1.31E-04	6.23E-05
85	Cm-246	9.82E-05	8.5%	3.11E-01	0.00E+00	0.00E+00	0.00E+00
86	Cm-247	6.15E-10	10.8%	9.38E-05	5.00E-01	3.28E-06	3.54E-07
87	Cm-248	8.20E-03	0.0%	4.30E-03	0.00E+00	0.00E+00	0.00E+00
88	Cf-249	2.58E-03	47.5%	4.14E+00	4.50E+01	2.81E-02	1.33E-02
89	Cf-250	7.84E-03	8.0%	1.11E+02	0.00E+00	0.00E+00	0.00E+00
90	Cf-251	2.80E-04	9.6%	1.60E+00	9.00E+01	1.58E-02	1.51E-03
91	Cf-252	1.70E+00	0.0%	5.44E+02	0.00E+00	0.00E+00	0.00E+00
92	Sr-90	6.93E-04	0.0%	1.38E+02	0.00E+00	0.00E+00	0.00E+00
93	Cs-137	9.67E-04	0.0%	8.80E+01	0.00E+00	0.00E+00	0.00E+00
94					Totals	4.40E-02	1.34E-02
95							30.51%
96					Mfg. Uncertainty		0.71%
97					Total Uncertainty		30.52%

	A	B	C	D	F	K	L
82		Activity	Total Uncertainty Excluding Mfg. Uncertainty	SpA (CI/g)	FGE	FGE	FGE Uncertainty
83		Ci		(Ci/g)	(FGE/g)	(FGE/Ci Cf-252)	(FGE/Ci Cf-252)
84	Cm-245	1.51E-06	47.8%	1.74E-01	1.50E+01	=B84*F84/D84	=K84*C84
85	Cm-246	9.82E-05	8.5%	3.11E-01	0.00E+00	=B85*F85/D85	=K85*C85
86	Cm-247	6.15E-10	10.8%	9.38E-05	5.00E-01	=B86*F86/D86	=K86*C86
87	Cm-248	8.20E-03	0.0%	4.30E-03	0.00E+00	=B87*F87/D87	=K87*C87
88	Cf-249	2.58E-03	47.5%	4.14E+00	4.50E+01	=B88*F88/D88	=K88*C88
89	Cf-250	7.84E-03	8.0%	1.11E+02	0.00E+00	=B89*F89/D89	=K89*C89
90	Cf-251	2.80E-04	9.6%	1.60E+00	9.00E+01	=B90*F90/D90	=K90*C90

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2	J. N. Vance	7/31/12	James H Hollenass	7/31/12

91	Cf-252	1.70E+00	0.0%	5.44E+02	0.00E+00	=B91*F91/D91	=K91*C91
92	Sr-90	6.93E-04	0.0%	1.38E+02	0.00E+00	=B92*F92/D92	=K92*C92
93	Cs-137	9.67E-04	0.0%	8.80E+01	0.00E+00	=B93*F93/D93	=K93*C93
94					Totals	=SUM(K84:K93)	=SQRT(SUMSQ(L84:L93))
95							=L94/K94
96					Mfg. Uncertainty		=0.0071
97					Total Uncertainty		=SQRT(SUMSQ(L95:L96))

	A	B	C	D	G	M	N
82		Activity	Total Uncertainty Excluding Mfg. Uncertainty	SpA (CI/g)	PE-Ci	PE-Ci	PE-Ci Uncertainty
83		Ci		(Ci/g)	(Ci/Ci)	(Ci/Ci Cf-252)	(Ci/Ci Cf-252)
84	Cm-245	1.51E-06	47.8%	1.74E-01	0.94	1.61E-06	7.69E-07
85	Cm-246	9.82E-05	8.5%	3.11E-01	0.94	1.05E-04	8.84E-06
86	Cm-247	6.15E-10	10.8%	9.38E-05	1.04	5.91E-10	6.38E-11
87	Cm-248	8.20E-03	0.0%	4.30E-03	0.27	3.04E-02	0.00E+00
88	Cf-249	2.58E-03	47.5%	4.14E+00	0.93	2.78E-03	1.32E-03
89	Cf-250	7.84E-03	8.0%	1.11E+02	2.32	3.38E-03	2.71E-04
90	Cf-251	2.80E-04	9.6%	1.60E+00	0.91	3.08E-04	2.95E-05
91	Cf-252	1.70E+00	0.0%	5.44E+02	3.90	4.35E-01	0.00E+00
92	Sr-90	6.93E-04	0.0%	1.38E+02	0.00		0.00E+00
93	Cs-137	9.67E-04	0.0%	8.80E+01	0.00		0.00E+00
94					Totals	4.72E-01	1.35E-03
95							0.29%
96					Mfg. Uncertainty		0.71%
97					Total Uncertainty		0.77%

Rev.	Originator	Date	Checker	Date
2	[signature]	7/31/12	James H Hollenes	7/31/12

	A	B	C	D	G	M	N
82		Activity	Total Uncertainty Excluding Mfg. Uncertainty	SpA (CI/g)	PE-Ci	PE-Ci	PE-Ci Uncertainty
83		Ci		(Ci/g)	(Ci/Ci)	(Ci/Ci Cf-252)	(Ci/Ci Cf-252)
84	Cm-245	1.51E-06	47.8%	1.74E-01	0.94	=B84/G84	=M84*C84
85	Cm-246	9.82E-05	8.5%	3.11E-01	0.94	=B85/G85	=M85*C85
86	Cm-247	6.15E-10	10.8%	9.38E-05	1.04	=B86/G86	=M86*C86
87	Cm-248	8.20E-03	0.0%	4.30E-03	0.27	=B87/G87	=M87*C87
88	Cf-249	2.58E-03	47.5%	4.14E+00	0.93	=B88/G88	=M88*C88
89	Cf-250	7.84E-03	8.0%	1.11E+02	2.32	=B89/G89	=M89*C89
90	Cf-251	2.80E-04	9.6%	1.60E+00	0.91	=B90/G90	=M90*C90
91	Cf-252	1.70E+00	0.0%	5.44E+02	3.90	=B91/G91	=M91*C91
92	Sr-90	6.93E-04	0.0%	1.38E+02	0.00		=M92*C92
93	Cs-137	9.67E-04	0.0%	8.80E+01	0.00		=M93*C93
94					Totals	=SUM(M84:M93)	=SQRT(SUMSQ(N84:N93))
95							=N94/M94
96					Mfg. Uncertainty		=0.0071
97					Total Uncertainty		=SQRT(SUMSQ(N95:N96))

	A	B	C	D	H	O	P
82		Activity	Total Uncertainty Excluding Mfg. Uncertainty	SpA (CI/g)	Decay Heat	Decay Heat	Decay Heat Uncertainty
83		Ci		(Ci/g)	(W/g)	(W/Ci Cf-252)	(W/Ci Cf-252)
84	Cm-245	1.51E-06	47.8%	1.74E-01	5.77E-03	5.02E-08	2.40E-08
85	Cm-246	9.82E-05	8.5%	3.11E-01	1.02E-02	3.22E-06	2.72E-07

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2	J. R. Vance	7/31/12	James H Hollemans	7/31/12

86	Cm-247	6.15E-10	10.8%	9.38E-05	2.98E-06	1.95E-11	2.11E-12
87	Cm-248	8.20E-03	0.0%	4.30E-03	5.53E-04	1.05E-03	0.00E+00
88	Cf-249	2.58E-03	47.5%	4.14E+00	1.54E-01	9.61E-05	4.56E-05
89	Cf-250	7.84E-03	8.0%	1.11E+02	4.12E+00	2.91E-04	2.33E-05
90	Cf-251	2.80E-04	9.6%	1.60E+00	5.89E-02	1.03E-05	9.88E-07
91	Cf-252	1.70E+00	0.0%	5.44E+02	4.06E+01	1.27E-01	0.00E+00
92	Sr-90	6.93E-04	0.0%	1.38E+02	1.60E-01	8.03E-07	0.00E+00
93	Cs-137	9.67E-04	0.0%	8.80E+01	9.74E-02	1.07E-06	0.00E+00
94					Totals	1.28E-01	5.12E-05
95							0.04%
96					Mfg. Uncertainty		0.71%
97					Total Uncertainty		0.71%

	A	B	C	D	H	O	P
82		Activity	Total Uncertainty Excluding Mfg. Uncertainty	SpA (CI/g)	Decay Heat	Decay Heat	Decay Heat Uncertainty
83		Ci		(Ci/g)	(W/g)	(W/Ci Cf-252)	(W/Ci Cf-252)
84	Cm-245	1.51E-06	47.8%	1.74E-01	5.77E-03	=B84*H84/D84	=O84*C84
85	Cm-246	9.82E-05	8.5%	3.11E-01	1.02E-02	=B85*H85/D85	=O85*C85
86	Cm-247	6.15E-10	10.8%	9.38E-05	2.98E-06	=B86*H86/D86	=O86*C86
87	Cm-248	8.20E-03	0.0%	4.30E-03	5.53E-04	=B87*H87/D87	=O87*C87
88	Cf-249	2.58E-03	47.5%	4.14E+00	1.54E-01	=B88*H88/D88	=O88*C88
89	Cf-250	7.84E-03	8.0%	1.11E+02	4.12E+00	=B89*H89/D89	=O89*C89
90	Cf-251	2.80E-04	9.6%	1.60E+00	5.89E-02	=B90*H90/D90	=O90*C90
91	Cf-252	1.70E+00	0.0%	5.44E+02	4.06E+01	=B91*H91/D91	=O91*C91
92	Sr-90	6.93E-04	0.0%	1.38E+02	1.60E-01	=B92*H92/D92	=O92*C92

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93	Cs-137	9.67E-04	0.0%	8.80E+01	9.74E-02	=B93*H93/D93	=O93*C93
94					Totals	=SUM(O84:O93)	=SQRT(SUMSQ(P84:P93))
95							=P94/O94
96					Mfg. Uncertainty		=0.0071
97					Total Uncertainty		=SQRT(SUMSQ(P95:P96))

	A	B	C	D		Q	R
82		Activity	Total Uncertainty Excluding Mfg. Uncertainty	SpA (Cl/g)		Total Activity	Total Activity Uncertainty
83		Ci		(Ci/g)		(Ci/Ci Cf-252)	(Ci/Ci Cf-252)
84	Cm-245	1.51E-06	47.8%	1.74E-01		1.51E-06	7.23E-07
85	Cm-246	9.82E-05	8.5%	3.11E-01		9.82E-05	8.31E-06
86	Cm-247	6.15E-10	10.8%	9.38E-05		6.15E-10	6.63E-11
87	Cm-248	8.20E-03	0.0%	4.30E-03		8.20E-03	0.00E+00
88	Cf-249	2.58E-03	47.5%	4.14E+00		2.58E-03	1.23E-03
89	Cf-250	7.84E-03	8.0%	1.11E+02		7.84E-03	6.28E-04
90	Cf-251	2.80E-04	9.6%	1.60E+00		2.80E-04	2.68E-05
91	Cf-252	1.70E+00	0.0%	5.44E+02		1.70E+00	0.00E+00
92	Sr-90	6.93E-04	0.0%	1.38E+02		6.93E-04	0.00E+00
93	Cs-137	9.67E-04	0.0%	8.80E+01		9.67E-04	0.00E+00
94					Totals	1.72E+00	1.38E-03
95							0.08%
96					Mfg. Uncertainty		0.71%
97					Total Uncertainty		0.71%

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2		7/31/12		7/31/12

	A	B	C	D		Q	R
82		Activity	Total Uncertainty Excluding Mfg. Uncertainty	SpA (CI/g)		Total Activity	Total Activity Uncertainty
83		Ci		(Ci/g)		(Ci/Ci Cf-252)	(Ci/Ci Cf-252)
84	Cm-245	1.51E-06	47.8%	1.74E-01		=B84	=Q84*C84
85	Cm-246	9.82E-05	8.5%	3.11E-01		=B85	=Q85*C85
86	Cm-247	6.15E-10	10.8%	9.38E-05		=B86	=Q86*C86
87	Cm-248	8.20E-03	0.0%	4.30E-03		=B87	=Q87*C87
88	Cf-249	2.58E-03	47.5%	4.14E+00		=B88	=Q88*C88
89	Cf-250	7.84E-03	8.0%	1.11E+02		=B89	=Q89*C89
90	Cf-251	2.80E-04	9.6%	1.60E+00		=B90	=Q90*C90
91	Cf-252	1.70E+00	0.0%	5.44E+02		=B91	=Q91*C91
92	Sr-90	6.93E-04	0.0%	1.38E+02		=B92	=Q92*C92
93	Cs-137	9.67E-04	0.0%	8.80E+01		=B93	=Q93*C93
94					Totals	=SUM(Q84:Q93)	=SQRT(SUMSQ(R84:R93))
95							=R94/Q94
96					Mfg. Uncertainty		=0.0071
97					Total Uncertainty		=SQRT(SUMSQ(R95:R96))

References:

1. *Radionuclide Distribution Development for Cf-252 Sources*, Calculation No. VA-OSR-10.
2. *Contact Handled Transuranic Waste Authorized Methods for Payload Control (CH TRAMPAC)*, Revision 3, February 2009.
3. *Transuranic Waste Acceptance Criteria for the Waste Isolation Pilot Plant*, Revision 7.0, December 2010.
4. *Uncertainty Analysis for Cf-252 Sealed Sources*, Calculation No. VA-OSR-11.

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2	J. N. Vance	7/31/12	James H Hollemans	7/31/12

Calculation No. VA-OSR-14
Sheet No. 1 of 6

Vance & Associates
Calculation Cover Sheet

PROJECT: **Sealed Source Radiological Characterization**

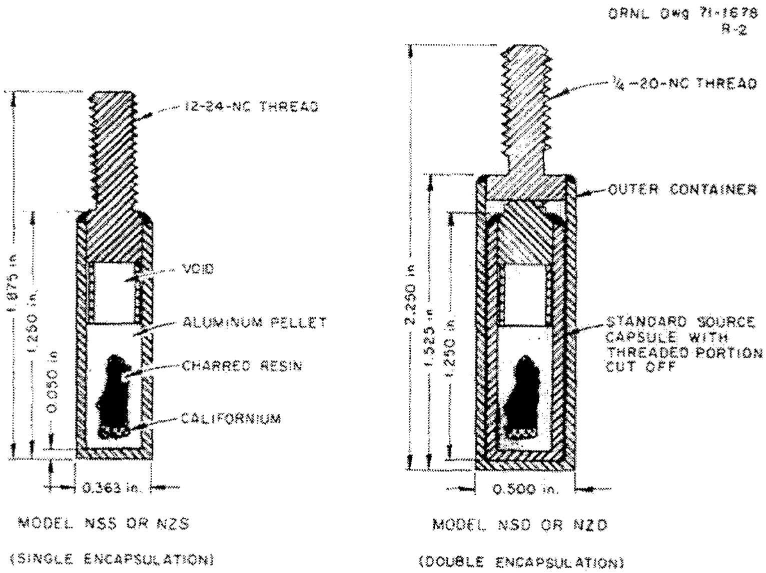
SUBJECT:

Relative importance of neutron-induced fission in Cf-252 sources.

RECORD OF ISSUES

Rev. No.	Description	Total No. of Sheets	Last Sheet No.	Originator	Checker	Approved	Date
0	Original Issue	6	6	James H Hollmann	J. Vance	J. Vance	12/20/4

Purpose:	Cesium-137 and Sr-90 are produced as fission products from both spontaneous fission of Cf-252 and from neutron-induced fission. The purpose of this calculation is to determine the relative contributions from each of these sources.
Approach:	An MCNP5 model of a Cf-252 source is developed. The source particles are from spontaneous fission of Cf-252. For this calculation 10^7 neutrons are randomly generated. This represents approximately 2.68×10^6 spontaneous fissions (with an average of 3.725 neutrons emitted per spontaneous fission – Ref. 1). MCNP5 tracks the randomly produced neutron-induced reactions, including the number of neutron-induced fissions. This number is compared to the 2.68×10^6 spontaneous fissions to determine the relative number of fissions of each type. These are multiplied by the Cs-137 and Sr-90 fission yields for spontaneous fission of Cf-252 and from neutron-induced fission in order to determine the relative yields from each type of fission.
Computer Software:	MCNP Version 5.
Assumptions:	1. The source is assumed to be a 5.0 Ci Cf-252 source. This is a relatively large source which tends to maximize the potential for neutron-induced fission.
Calculations:	Typical Cf-252 source designs are shown in the figure below (Ref. 5). The source may have single or double encapsulation and the capsule may consist of stainless steel or Zircaloy-2. The stainless steel double encapsulation source design is the most common. The californium may be converted to oxysulfate microspheres and pressed into a pellet with aluminum powder (Ref. 6). This is the design shown in the figure below.



An MCNP5 model is developed for the stainless steel double encapsulation source design. The model includes five volumes:

- the californium source, which is considered to be a small cylinder of material near the bottom of the pellet;
- the aluminum powder pellet;
- the void space within the capsule;
- the stainless steel capsules (modeled as a single capsule with double the wall thickness); and
- an external volume.

```
c
c      ************************************************************
c                        Cell Cards
c      ************************************************************
c
1    1   -1.50   3  -4  -11                 imp:n=1  $ Source volume
2    2   -1.50   2  -5  -12  (-3:4:11)      imp:n=1  $ Aluminum pellet
3    0           5  -6  -12                 imp:n=1  $ Internal void
4    3   -7.86   1  -7  -13  (-2:6:12)      imp:n=1  $ Capsule
5    0          -1:7:13                     imp:n=0  $ External volume
```

The 5.0 Ci source contains (5.0/544) = 9.19 mg of Cf-252. The weight percent of a Cf-252 source is approximately 84.0% (Reference 2), so the total mass of californium is 10.9 mg. The californium is assumed to have the same density as the pressed aluminum powder, which has an assumed density of 1.5 g/cm^3. This gives a source volume of 7.29x10^{-3} cm^3. The source volume is assumed to be 0.2" (0.508 cm) in diameter. The thickness of the source volume is calculated to be 3.56x10^{-2} cm. Other capsule dimensions are from Reference 1 or are scaled from the above figure.

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0	James H Hollerman	12/20/11	J. G. Vance	12/20/11

```
c       ***********************************************************
c                   Surface Cards
c       ***********************************************************
c
1   pz    0.000           $ Bottom of capsule
2   pz    0.254           $ Bottom of pellet
3   pz    0.508           $ Bottom of source
4   pz    0.5436          $ Top of source
5   pz    1.842           $ Top of pellet
6   pz    2.540           $ Top of void space
7   pz    3.874           $ Top of capsule
11  cz    0.254           $ Radius of source
12  cz    0.381           $ Pellet radius
13  cz    0.635           $ Capsule radius
```

The source is assumed to be uniformly distributed throughout Cell 1. MCNP5 particles are generated within the source volume at random radial positions varying from 0 to the source radius and at random heights varying from the 0 to the thickness of Cell 1. The reference point for the generated particle locations is the bottom centerline position of Cell 1.

The following shows the neutron source model. The neutron energy is randomly selected from the Watt fission spectrum for the spontaneous fission of Cf-2 (Ref. 3).

```
c
c       *********************************************************
c                               Source
c       *********************************************************
c
sdef    erg=d8  pos=frad=d7  rad=d1 par=1
        ext=frad=d4 axs=0 0 1
c
si1  S 2
sp1     1.000
si2     0  0.254
sp2     -21  1
ds4  S 5
si5     0  0.0356
ds7  L    0.0    0.0    0.508        $ Bottom position of source
sp8 -3  1.025 2.926                  $ Watt fission spectrum for Cf-252
```

Other MCNP5 inputs are:

```
c       **********************************************************
c                            Problem Type
c       **********************************************************
c
mode n
c
c
c       *********************************************************
c                             Material
c       *********************************************************
c
c  Source
m1         98249.60c  0.0400 98250.60c 0.0921 98251.60c 0.0268
           98252.60c  0.8412
c  Aluminum
m2         13027.60c 1
c  Steel
```

```
m3        26056.60c 1c
c       ********************************************************
c                            Cutoff Cards
c       ********************************************************
c
ctme 60
nps  1e7
c
```

These inputs define the problem type as being a neutron transport problem, the material composition for the source, pellet and capsule, and the problem cut-off (either 60 minutes of computer run time, or 10^7 source particles).

Reference 2 provides the average isotopic distribution for californium sources.

	w/o (Ref. 2)	at. Wt.	Atom Density	a/o
Cf-249	3.94%	249	1.58E-04	4.00%
Cf-250	9.12%	250	3.65E-04	9.21%
Cf-251	2.67%	251	1.06E-04	2.68%
Cf-252	84.00%	252	3.33E-03	84.12%
Total			3.96E-03	

Results

The following shows the results of the MCNP5 calculation. Spontaneous fission accounts for over 99.6% of the Cs-137 and Sr-90 production. As a result Cs-137 and Sr-90 production via neutron-induced fission is negligible and can be ignored.

	MCNP5 Results: Number of Fissions	Cs-137 Integral Fission Yield	Cs-137 Production (atoms)	Relative Contribution to Cs-137 Production
Spontaneous Fission	2.68E+06	0.0502 (Ref. 4)	1.35E+05	99.82%
Neutron-Induced Fission	3756	0.0637*	2.39E+02	0.18%
Total			1.35E+05	

	MCNP5 Results: Number of Fissions	Sr-90 Integral Fission Yield (Ref. 4)	Sr-90 Production (atoms)	Relative Contribution to Sr-90 Production
Spontaneous Fission	2.68E+06	0.00522 (Ref. 4)	1.40E+04	99.61%

Neutron-Induced Fission	3756	0.0147*	55.2E+01	0.39%
Total			1.41E+04	

* Cumulative yields for neutron-induced fission of Cf-252 from Ref. 1 fast reactor library FFTFC.

References:

1. *ORIGEN2.2 – Isotope Generation and Depletion Code*, CCC-371, ORIGEN2.2, Radiation Safety Information Computational Center, Oak Ridge National Laboratory.
2. *Radionuclide Distribution Development: Cf-252 Sources*, Calculation No. VA-OSR-10.
3. *MCNP – A General Monte Carlo N-Particle Transport Code – Version 5*, LA-UR-03-1987, Los Alamos National Laboratory.
4. Lawrence Berkeley National Laboratory Isotopes Project Home Page, www.lbl.gov.
5. L. J. King, et al., *Transuranium Processing Plant Semiannual Report of Production, Status and Plans for Period Ending June 30, 1973*, ORNL-4921.
6. R. C. Martin, et al., *Production, Distribution and Applications of Calcifornium-252 Neutron Sources*, 4th Topical Meeting on Industrial Radiation and Radioisotope Measurement Applications, Raleigh, NC, October 3-7, 1999.

Attachments: 1. MCNP5 Output File

Attachment 1
MCNP5 Output File

```
         Thread Name & Version = MCNP5_RSICC, 1.30
                                              _
         ._ _    _  ._    ._    |_
         | | |  (_  | |   |_)    _)
                          |
 +---------------------------------------------------------------------+
 |   This program was prepared by the Regents of the University of    |
 |California at Los Alamos National Laboratory (the University) under |
 |  contract number W-7405-ENG-36 with the U.S. Department of Energy  |
 |(DoE).  The University has certain rights in the program pursuant to|
 |  the contract and the program should not be copied or distributed  |
 | outside your organization.  All rights in the program are reserved |
 |by the DoE and the University.  Neither the U.S. Government nor the |
 | University makes any warranty, express or implied, or assumes any  |
 |     liability or responsibility for the use of this software.      |
 +---------------------------------------------------------------------+

1mcnp     version 5     ld=06212004                    12/20/11 13:15:29
 *************************************************************************                   probid =  12/20/11 13:15:29
 i=cf.txt o=cf.o

         5.0 Ci Cf-252 Source
         c
         c     ************************************************************
         c                      Cell Cards
         c     ************************************************************
         c
         1    1   -1.50   3  -4  -11                 imp:n=1  $ Source volume
         2    2   -1.50   2  -5  -12  (-3:4:11)      imp:n=1  $ Aluminum pellet
         3    0           5  -6  -12                 imp:n=1  $ Internal void
         4    3   -7.86   1  -7  -13  (-2:6:12)      imp:n=1  $ Capsule
         5    0          -1:7:13                     imp:n=0  $ External volume

         c      ***********************************************************
         c                  Surface Cards
         c      ***********************************************************
         c
         1   pz    0.000           $ Bottom of capsule
         2   pz    0.254           $ Bottom of pellet
         3   pz    0.508           $ Bottom of source
         4   pz    0.5436          $ Top of source
         5   pz    1.842           $ Top of pellet
         6   pz    2.540           $ Top of void space
```

```
7   pz    3.874           $ Top of capsule
11  cz    0.254           $ Radius of source
12  cz    0.381           $ Pellet radius
13  cz    0.635           $ Capsule radius

c      *********************************************************
c      *********************************************************
c                          Problem Type
c      *********************************************************
c
mode n
c
c      ********************************************************
c                             Source
c      ********************************************************
c
sdef   erg=d8  pos=frad=d7  rad=d1 par=1
       ext=frad=d4 axs=0 0 1
c
si1  S 2
sp1    1.000
si2    0  0.254
sp2    -21  1
ds4  S 5
si5    0  0.0356
ds7  L   0.0   0.0   0.508        $ Bottom position of source
sp8 -3  1.025 2.926               $ Watt fission spectrum for Cf-252
c
c      ********************************************************
c                           Material
c      ********************************************************
c
c  Source
m1        98249.60c  0.0400 98250.60c 0.0921 98251.60c 0.0268
          98252.60c  0.8412
c  Aluminum
m2        13027.60c 1
c  Steel
m3        26056.60c 1
c
c      ********************************************************
c                         Cutoff Cards
c      ********************************************************
c
ctme 60
nps  1e7
c
```

```
 warning. there are no tallies in this problem.

 warning.   1 materials had unnormalized fractions. print table 40.
1cells                                                                                                    print table 60

                      atom        gram                                            neutron
       cell  mat    density     density      volume       mass      pieces importance

    1     1    1  3.58811E-03 1.50000E+00 7.21551E-03 1.08233E-02      1  1.0000E+00
    2     2    2  3.34787E-02 1.50000E+00 7.16971E-01 1.07546E+00      1  1.0000E+00
    3     3    0  0.00000E+00 0.00000E+00 3.18314E-01 0.00000E+00      1  1.0000E+00
    4     4    3  8.46220E-02 7.86000E+00 3.86496E+00 3.03786E+01      1  1.0000E+00
    5     5    0  0.00000E+00 0.00000E+00 0.00000E+00 0.00000E+00      0  0.0000E+00

 total                                    4.90746E+00 3.14649E+01

    minimum source weight = 1.0000E+00    maximum source weight = 1.0000E+00

 ***************************************************
 * Random Number Generator  =                    1 *
 * Random Number Seed       =       19073486328125 *
 * Random Number Multiplier =       19073486328125 *
 * Random Number Adder      =                    0 *
 * Random Number Bits Used  =                   48 *
 * Random Number Stride     =               152917 *
 ***************************************************


        2 warning messages so far.
1cross-section tables                                                                                     print table 100

     table    length

                          tables from file endf60

  13027.60c   24448  13-al-27 from endf/b-vi                                               mat1325      11/25/93
  26056.60c   92647  endf/b-vi.1 fe56a                                                     mat2631      11/25/93
  98249.60c   32411  98-cf-249 from endf-vi                          prompt nu             mat9852      11/28/93

 warning. nubar of  98250.60c may be either prompt or total.
  98250.60c   26807  98-cf-250 from patched endf-vi                                        mat9855      11/28/93
  98251.60c   25861  98-cf-251 from patched endf-vi                  prompt nu             mat9858      11/28/93
  98252.60c   30004  98-cf-252 from patched endf-vi                  prompt nu             mat9861      11/28/93

  total      232178

 ***********************************************************************************************************
 dump no.    1 on file runtpo       nps =           0     coll =              0     ctm =        0.00   nrn =                 0
```

```
         3 warning messages so far.
1problem summary

      run terminated when    10000000  particle histories were done.
+                                                                                                12/20/11 13:17:06
      5.0 Ci Cf-252 Source                                                             probid =  12/20/11 13:15:29
0
 neutron creation    tracks      weight        energy            neutron loss        tracks      weight        energy
                                 (per source particle)                                           (per source particle)

 source            10000000    1.0000E+00    2.3057E+00          escape            10013876    1.0011E+00    2.2532E+00
                                                                 energy cutoff            0    0.            0.
                                                                 time cutoff              0    0.            0.
 weight window            0    0.            0.                  weight window            0    0.            0.
 cell importance          0    0.            0.                  cell importance          0    0.            0.
 weight cutoff            0    4.0765E-08    2.9749E-09          weight cutoff            2    1.7783E-08    1.2979E-09
 e or t importance        0    0.            0.                  e or t importance        0    0.            0.
 dxtran                   0    0.            0.                  dxtran                   0    0.            0.
 forced collisions        0    0.            0.                  forced collisions        0    0.            0.
 exp. transform           0    0.            0.                  exp. transform           0    0.            0.
 upscattering             0    0.            0.                  downscattering           0    0.            5.4121E-02
 photonuclear             0    0.            0.                  capture                  0    2.5787E-04    1.2962E-03
 (n,xn)                 173    1.6719E-05    1.3463E-05          loss to (n,xn)          86    8.3097E-06    9.2545E-05
 prompt fission       17547    1.7483E-03    4.0264E-03          loss to fission       3756    3.7410E-04    9.6438E-04
 delayed fission          0    0.            0.
     total         10017720    1.0018E+00    2.3097E+00              total         10017720    1.0018E+00    2.3097E+00

   number of neutrons banked                     13878        average time of (shakes)              cutoffs
   neutron tracks per source particle     1.0018E+00          escape            5.3720E-02          tco   1.0000E+33
   neutron collisions per source particle 1.4050E-01          capture           3.2956E-02          eco   0.0000E+00
   total neutron collisions                    1404977          capture or escape 5.3715E-02          wc1  -5.0000E-01
   net multiplication              1.0014E+00 0.0000          any termination   5.3695E-02          wc2  -2.5000E-01

 computer time so far in this run     1.58 minutes            maximum number ever in bank         7
 computer time in mcrun               1.56 minutes            bank overflows to backup file       0
 source particles per minute            6.4188E+06
 random numbers generated               132311529             most random numbers used was        156 in history     4399955

 range of sampled source weights = 1.0000E+00 to 1.0000E+00
1neutron  activity in each cell                                                                         print table 126

                   tracks     population   collisions   collisions     number        flux       average      average
        cell      entering                               * weight     weighted     weighted   track weight   track mfp
                                                      (per history)    energy       energy     (relative)      (cm)

     1     1      10069949     10013834        14995    1.4995E-03   1.5765E+00   2.2963E+00   9.9997E-01   3.6607E+01
     2     2      10245065     10013845       406423    4.0637E-02   1.5481E+00   2.2676E+00   9.9986E-01   1.0586E+01
     3     3        208624       202403            0    0.0000E+00   1.5049E+00   2.2216E+00   9.9959E-01   0.0000E+00
```

```
    4     4    10184924     10013878       983559    9.8326E-02   1.5160E+00   2.2299E+00   9.9971E-01   5.4788E+00

    total      30708562     30243960      1404977    1.4046E-01

************************************************************************************************************************
dump no.    2 on file runtpo       nps =     10000000     coll =        1404977     ctm =        1.56   nrn =         132311529

        3 warning messages so far.

run terminated when    10000000  particle histories were done.

computer time =    1.58 minutes

mcnp     version 5     06212004                     12/20/11 13:17:06                     probid =  12/20/11 13:15:29
```

Vance & Associates
Calculation Cover Sheet

PROJECT: **Sealed Source Radiological Characterization**

SUBJECT:

Determine upper bound of decay product inventories from a drum of Cf-252 sources.

RECORD OF ISSUES

Rev. No.	Description	Total No. of Sheets	Last Sheet No.	Originator	Checker	Approved	Date
1	Calculate upper bound activities for WIPP-reportable actinides	4	4	James H Hollaman	J. Vance	J. Vance	3/4/12

Purpose: To determine the upper bound inventories of the WIPP-reportable actinides as decay products of Cf-252 sources.

Approach: The Cf-252 sources initially contain Cf-249, Cf-250, Cf-251 and Cf-252 (Reference 1). These undergo alpha decay to Cm-245, Cm-246, Cm-247 and Cm-248, respectively. Further decays along the decay chain may produce very small quantities of the WIPP-reportable actinides U-233, U-234, U-238, Pu-238, Pu-239, Pu-240, Pu-242 and Am-241. In order to determine the upper bound drum inventories of these radionuclides, the 50-year decay of a 1-curie Cf-252 source is calculated using the RadDecay code. As noted below, a limiting drum is assumed to contain 10 curies of Cf-252 sources, so the RadDecay results are multiplied by a factor of 10.

Computer Software: RadDecay Version 4.01, Grove Software, Inc..

Assumptions:

1. The total Cf-252 activity of the sources within a single drum is assumed to be less than 10 Ci. This is the approximate total of the fifty largest Cf-252 sources in the LANL OSRP database. It is therefore extremely unlikely that this amount of Cf-252 would be placed in a single drum and this value therefore represents an upper bound of a single drum's Cf-252 inventory.
2. The sources are assumed to decay over a 50-year period. This is longer than the age of the oldest source and will tend to maximize the potential ingrowth of the reportable radionuclides.

Calculations: From Reference 1, the initial composition of a 1-curie Cf-252 source is as follows:

Radionuclide	Curies
Cf-249	3.82×10^{-4}
Cf-250	2.24×10^{-2}
Cf-251	9.48×10^{-5}
Cf-252	1.00

Each of these is decayed for 50 years, using the RadDecay code. Attachment 1 contains the results of each decay calculation.

Summary of Results

The following table summarizes results of the decay calculations. It shows the reportable actinide decay product activities, along with the curium decay product activities and the remaining source activities.

Radionuclide	Cf-249	Cf-250	Cf-251	Cf-252	Total (Ci)
Initial Activity (Ci)	3.82E-04	2.24E-02	9.48E-05	1.00E+00	
Decay Products: (Ci)					
U-233	5.97E-18				5.97E-18
U-234		4.36E-22			4.36E-22
U-238		1.12E-17			1.12E-17
Pu-238					0.00E+00
Pu-239			2.34E-16		2.34E-16
Pu-240				6.56E-15	6.56E-15
Pu-242		3.68E-09			3.68E-09
Am-241	2.87E-08				2.87E-08
Cm-245	1.48E-06				1.48E-06
Cm-246		5.70E-05			5.70E-05
Cm-247			2.07E-10		2.07E-10
Cm-248				7.54E-06	7.54E-06
Remaining Source Activity (Ci)					
Cf-249	3.46E-04				3.46E-04
Cf-250		1.58E-03			1.58E-03
Cf-251			9.12E-05		9.12E-05
Cf-252				1.98E-06	1.98E-06

The above are based on decay of a 1-curie Cf-252 source. Since the upper bound drum inventory is assumed to be 10 curies, the above values are multiplied by a factor of 10. The resulting upper-bound drum inventories for the WIPP-reportable radionuclides are shown below. The below "less than" values will be reported for each drum containing Cf-252 sources. Note that Pu-238 is not a decay product of any of the Californium isotopes and its activity will therefore be reported as zero.

Radionuclide	Total (Ci)
U-233	< 5.97E-17
U-234	< 4.36E-21
U-238	< 1.12E-16
Pu-238	0.00E+00
Pu-239	< 2.34E-15
Pu-240	< 6.56E-14
Pu-242	< 3.68E-08

Am-241	< 2.87E-07

References: 1. *Radionuclide Distribution Development: Cf-252 Sources*, Calculation No. VA-OSR-10.

Attachments: 1. RadDecay Case Outputs

Rev.	Originator	Date	Checker	Date
1	James H Hollemans	3/4/12	J. H. Vance	3/4/12

Attachment 1
RadDecay Case Outputs

<u>Cf-249 Decay</u>

RadDecay 4.01

File Tools Options Window Help

New Open Save Print Nuclide Info Periodic Table

Nuclide Info: Cf-249 - Library: GROVE (Kocher)

Nuclide Decay Info: Cf-249 - Library: GROVE (Kocher)

Activity (Ci)

3.8200E-04

Decay

5.0000E+01

Years

Table Summary Chain Graph

	Nuclide	Half-Life	Branching	Fraction	Activity (Ci)
1	Cf-249	3.5060E+02 Ye	Parent	9.0588E-01	3.4605E-04
2	Cm-245	8.5000E+03 Ye	1.0000E+00	3.8743E-03	1.4800E-06
3	Pu-241	1.4400E+01 Ye	1.0000E+00	2.4372E-03	9.3100E-07
4	U-237	6.7500E+00 Da	2.4500E-05	5.9666E-08	2.2792E-11
5	Np-237	2.1400E+06 Ye	1.0000E+00	3.7951E-13	1.4497E-16
6	Pa-233	2.7000E+01 Da	1.0000E+00	3.7745E-13	1.4419E-16
7	U-233	1.5920E+05 Ye	1.0000E+00	2.2623E-17	8.6418E-21
8	Th-229	7.3400E+03 Ye	1.0000E+00	2.2766E-20	8.6967E-24
9	Ra-225	1.4800E+01 Da	1.0000E+00	2.2641E-20	8.6488E-24
10	Ac-225	1.0000E+01 Da	1.0000E+00	2.2557E-20	8.6169E-24
11	Fr-221	4.8000E+00 Mir	1.0000E+00	2.2558E-20	8.6173E-24
12	At-217	3.2300E-02 Sec	1.0000E+00	2.2558E-20	8.6172E-24
13	Bi-213	4.5650E+01 Mir	1.0000E+00	2.2558E-20	8.6171E-24
14	Tl-209	2.2000E+00 Mir	2.1600E-02	4.8726E-22	1.8613E-25
15	Pb-209	3.2530E+00 Ho	1.0000E+00	4.8721E-22	1.8611E-25
16	Po-213	4.2000E-06 Sec	9.7840E-01	2.2071E-20	8.4310E-24
17	Pb-209	3.2530E+00 Ho	1.0000E+00	2.2070E-20	8.4306E-24
18	Am-241	4.3220E+02 Ye	9.9998E-01	7.5129E-05	2.8699E-08
19	Np-237	2.1400E+06 Ye	1.0000E+00	3.3762E-10	1.2897E-13
20	Pa-233	2.7000E+01 Da	1.0000E+00	3.3504E-10	1.2798E-13
21	U-233	1.5920E+05 Ye	1.0000E+00	1.5614E-14	5.9646E-18
22	Th-229	7.3400E+03 Ye	1.0000E+00	1.2903E-17	4.9291E-21
23	Ra-225	1.4800E+01 Da	1.0000E+00	1.2798E-17	4.8889E-21
24	Ac-225	1.0000E+01 Da	1.0000E+00	1.2722E-17	4.8598E-21
25	Fr-221	4.8000E+00 Mir	1.0000E+00	1.2731E-17	4.8634E-21
26	At-217	3.2300E-02 Sec	1.0000E+00	1.2731E-17	4.8634E-21
27	Bi-213	4.5650E+01 Mir	1.0000E+00	1.2712E-17	4.8561E-21
28	Tl-209	2.2000E+00 Mir	2.1600E-02	2.7417E-19	1.0473E-22
29	Pb-209	3.2530E+00 Ho	1.0000E+00	2.7355E-19	1.0450E-22
30	Po-213	4.2000E-06 Sec	9.7840E-01	1.2438E-17	4.7512E-21
31	Pb-209	3.2530E+00 Ho	1.0000E+00	1.2456E-17	4.7583E-21

Cf-250 Decay

RadDecay 4.01

File Tools Options Window Help

New Open Save Print Nuclide Info Periodic Table

Nuclide Info: Cf-250 - Library: GROVE (Kocher)

Nuclide Decay Info: Cf-250 - Library: GROVE (Kocher)

Activity (Ci)

2.2400E-02

Decay

5.0000E+01

Years

Table | Summary | Chain | Graph

	Nuclide	Half-Life	Branching	Fraction	Activity (Ci)
1	Cf-250	1.3080E+01 Ye	Parent	7.0676E-02	1.5831E-03
2	Cm-246	4.7500E+03 Ye	9.9923E-01	2.5441E-03	5.6988E-05
3	Pu-242	3.7580E+05 Ye	9.9974E-01	1.6424E-07	3.6789E-09
4	U-238	4.4680E+09 Ye	1.0000E+00	5.0070E-16	1.1216E-17
5	Th-234	2.4100E+01 Da	1.0000E+00	4.9828E-16	1.1161E-17
6	Pa-234m	1.1700E+00 Mir	1.0000E+00	4.9828E-16	1.1161E-17
7	Pa-234	6.7000E+00 Ho	1.6000E-03	7.9720E-19	1.7857E-20
8	U-234	2.4450E+05 Ye	1.0000E+00	3.1169E-23	6.9819E-25
9	Th-230	7.7000E+04 Ye	1.0000E+00	2.5029E-27	5.6064E-29
10	Ra-226	1.6000E+03 Ye	1.0000E+00	1.8314E-27	4.1022E-29
11	Rn-222	3.8235E+00 Da	1.0000E+00	1.4181E-27	3.1765E-29
12	Po-218	3.0500E+00 Mir	1.0000E+00	3.9860E-27	8.9287E-29
13	Pb-214	2.6800E+01 Mir	9.9980E-01	1.7604E-27	3.9434E-29
14	Bi-214	1.9900E+01 Mir	1.0000E+00	2.5450E-27	5.7007E-29
15	Po-214	1.6370E-04 Sec	9.9979E-01	2.4749E-27	5.5438E-29
16	Pb-210	2.2260E+01 Ye	1.0000E+00	0.0000E+00	0.0000E+00
17	Bi-210	5.0130E+00 Da	1.0000E+00	0.0000E+00	0.0000E+00
18	Po-210	1.3838E+02 Da	1.0000E+00	0.0000E+00	0.0000E+00
19	U-234	2.4450E+05 Ye	9.9840E-01	1.9452E-20	4.3571E-22
20	Th-230	7.7000E+04 Ye	1.0000E+00	3.0802E-24	6.8996E-26
21	Ra-226	1.6000E+03 Ye	1.0000E+00	1.5346E-24	3.4376E-26
22	Rn-222	3.8235E+00 Da	1.0000E+00	1.8001E-24	4.0322E-26
23	Po-218	3.0500E+00 Mir	1.0000E+00	2.0142E-24	4.5119E-26
24	Pb-214	2.6800E+01 Mir	9.9980E-01	1.9702E-24	4.4132E-26
25	Bi-214	1.9900E+01 Mir	1.0000E+00	1.1151E-24	2.4979E-26
26	Po-214	1.6370E-04 Sec	9.9979E-01	1.0715E-24	2.4002E-26
27	Pb-210	2.2260E+01 Ye	1.0000E+00	3.5902E-25	8.0420E-27
28	Bi-210	5.0130E+00 Da	1.0000E+00	0.0000E+00	0.0000E+00
29	Po-210	1.3838E+02 Da	1.0000E+00	1.1051E-24	2.4755E-26

Cf-251 Decay

RadDecay 4.01

File Tools Options Window Help

New Open Save Print Nuclide Info Periodic Table

Nuclide Info: Cf-251 - Library: GROVE (Kocher)

Nuclide Decay Info: Cf-251 - Library: GROVE (Kocher)

Activity (Ci)

9.4800E-05

Decay

5.0000E+01

Years

Table | Summary | Chain | Graph

	Nuclide	Half-Life	Branching	Fraction	Activity (Ci)
1	Cf-251	9.0000E+02 Ye	Parent	9.6222E-01	9.1219E-05
2	Cm-247	1.5600E+07 Ye	1.0000E+00	2.1794E-06	2.0661E-10
3	Pu-243	4.9560E+00 Ho	1.0000E+00	2.1794E-06	2.0660E-10
4	Am-243	7.3800E+03 Ye	1.0000E+00	5.1419E-09	4.8746E-13
5	Np-239	2.3550E+00 Da	1.0000E+00	5.1400E-09	4.8728E-13
6	Pu-239	2.4131E+04 Ye	1.0000E+00	2.4682E-12	2.3399E-16
7	U-235	7.0380E+08 Ye	1.0000E+00	3.0456E-20	2.8872E-24
8	Th-231	2.5520E+01 Ho	1.0000E+00	3.0265E-20	2.8691E-24
9	Pa-231	3.2760E+04 Ye	1.0000E+00	7.7816E-23	7.3770E-27
10	Ac-227	2.1773E+01 Ye	1.0000E+00	0.0000E+00	0.0000E+00
11	Fr-223	2.1800E+01 Mir	1.3800E-02	0.0000E+00	0.0000E+00
12	Ra-223	1.1434E+01 Da	9.9994E-01	0.0000E+00	0.0000E+00
13	Rn-219	3.9600E+00 Se	1.0000E+00	0.0000E+00	0.0000E+00
14	Po-215	1.7780E-03 Sec	1.0000E+00	0.0000E+00	0.0000E+00
15	Pb-211	3.6100E+01 Mir	1.0000E+00	0.0000E+00	0.0000E+00
16	Bi-211	2.1300E+00 Mir	1.0000E+00	0.0000E+00	0.0000E+00
17	Po-211	5.1600E-01 Sec	2.7300E-03	0.0000E+00	0.0000E+00
18	Tl-207	4.7700E+00 Mir	9.9727E-01	0.0000E+00	0.0000E+00
19	Th-227	1.8718E+01 Da	9.8620E-01	0.0000E+00	0.0000E+00
20	Ra-223	1.1434E+01 Da	1.0000E+00	0.0000E+00	0.0000E+00
21	Rn-219	3.9600E+00 Se	1.0000E+00	0.0000E+00	0.0000E+00
22	Po-215	1.7780E-03 Sec	1.0000E+00	0.0000E+00	0.0000E+00
23	Pb-211	3.6100E+01 Mir	1.0000E+00	0.0000E+00	0.0000E+00
24	Bi-211	2.1300E+00 Mir	1.0000E+00	0.0000E+00	0.0000E+00
25	Po-211	5.1600E-01 Sec	2.7300E-03	0.0000E+00	0.0000E+00
26	Tl-207	4.7700E+00 Mir	9.9727E-01	2.0126E-22	1.9079E-26

Cf-252 Decay

RadDecay 4.01

File Tools Options Window Help

New Open Save Print Nuclide Info Periodic Table

Nuclide Info: Cf-252 - Library: GROVE (Kocher)

Nuclide Decay Info: Cf-252 - Library: GROVE (Kocher)

Activity (Ci)

1.0000E+00

Decay

5.0000E+01

Years

Table | Summary | Chain | Graph

	Nuclide	Half-Life	Branching	Fraction	Activity (Ci)
1	Cf-252	2.6390E+00 Yea	Parent	1.9793E-06	1.9793E-06
2	Cm-248	3.3900E+05 Yea	9.6908E-01	7.5432E-06	7.5432E-06
3	Pu-244	8.2600E+07 Yea	9.1740E-01	2.6826E-12	2.6826E-12
4	U-240	1.4100E+01 Hou	9.9875E-01	2.6791E-12	2.6791E-12
5	Np-240m	7.4000E+00 Mir	1.0000E+00	2.6791E-12	2.6791E-12
6	Np-240	6.5000E+01 Mir	1.1000E-03	2.9470E-15	2.9470E-15
7	Pu-240	6.5690E+03 Yea	1.0000E+00	7.2187E-18	7.2187E-18
8	U-236	2.3415E+07 Yea	1.0000E+00	3.3178E-24	3.3178E-24
9	Th-232	1.4050E+10 Yea	1.0000E+00	3.4286E-29	3.4286E-29
10	Ra-228	5.7500E+00 Yea	1.0000E+00	3.4286E-29	3.4286E-29
11	Ac-228	6.1300E+00 Hou	1.0000E+00	1.0286E-28	1.0286E-28
12	Th-228	1.9132E+00 Yea	1.0000E+00	6.8573E-29	6.8573E-29
13	Ra-224	3.6200E+00 Day	1.0000E+00	3.4286E-29	3.4286E-29
14	Rn-220	5.5610E+01 Sec	1.0000E+00	3.4286E-29	3.4286E-29
15	Po-216	1.4600E-01 Sec	1.0000E+00	0.0000E+00	0.0000E+00
16	Pb-212	1.0643E+01 Hou	1.0000E+00	3.4286E-29	3.4286E-29
17	Bi-212	6.0550E+01 Mir	1.0000E+00	3.4286E-29	3.4286E-29
18	Tl-208	3.0530E+00 Mir	3.5930E-01	1.2319E-29	1.2319E-29
19	Po-212	2.9800E-07 Sec	6.4070E-01	2.1967E-29	2.1967E-29
20	Pu-240	6.5690E+03 Yea	9.9890E-01	6.5553E-15	6.5553E-15
21	U-236	2.3415E+07 Yea	1.0000E+00	3.0129E-21	3.0129E-21
22	Th-232	1.4050E+10 Yea	1.0000E+00	0.0000E+00	0.0000E+00
23	Ra-228	5.7500E+00 Yea	1.0000E+00	0.0000E+00	0.0000E+00
24	Ac-228	6.1300E+00 Hou	1.0000E+00	0.0000E+00	0.0000E+00
25	Th-228	1.9132E+00 Yea	1.0000E+00	0.0000E+00	0.0000E+00
26	Ra-224	3.6200E+00 Day	1.0000E+00	0.0000E+00	0.0000E+00
27	Rn-220	5.5610E+01 Sec	1.0000E+00	0.0000E+00	0.0000E+00
28	Po-216	1.4600E-01 Sec	1.0000E+00	0.0000E+00	0.0000E+00
29	Pb-212	1.0643E+01 Hou	1.0000E+00	0.0000E+00	0.0000E+00
30	Bi-212	6.0550E+01 Mir	1.0000E+00	0.0000E+00	0.0000E+00
31	Tl-208	3.0530E+00 Mir	3.5930E-01	0.0000E+00	0.0000E+00
32	Po-212	2.9800E-07 Sec	6.4070E-01	0.0000E+00	0.0000E+00