Chapter 3 Appendix 3

MONTE CARLO NEUTRON AND GAMMA-RAY CALCULATIONS

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Kerma in tissue and the activation produced in sulfur and cobalt due to prompt neutrons from the Hiroshima and Nagasaki bombs were calculated out to 2000 m from the hypocenter in 100 m increments. As neutron sources weapon output spectra calculated by investigators from the Los Alamos National Laboratory (LANL) were used. Other parameters, such as burst height and air and ground densities and compositions, were obtained from recent sources. The Lawrence Livermore National Laboratory (LLNL) Monte Carlo transport code TART was used for these calculations. TART accesses the well-established 1985 ENDL cross-section library, which has built-in reaction cross sections. The zoning for this problem was a full two-dimensional geometry with a ceiling height of 1100 m and a ground thickness of 30 cm. For the Hiroshima calculations (including sulfur activation) an untilted source was used. However, a special sulfur activation problem using a source tilted 15° was run for which the ratios to the untilted case are reported.

The TART code uses a technique for solving the transport equation that is different from that of the Oak Ridge National Laboratory (ORNL) DOT code; it also draws on a specially evaluated cross-section library (ENDL) and uses a larger group structure than DOT. One of the purposes of this work was to instill confidence in the DOT calculations that will be used directly in the dose reassessment of A-bomb survivors. The TART results were compared with values calculated with the DOT code by investigators from ORNL and found to be in good agreement for the most part. However, the sulfur activation comparison is disappointing. Because the sulfur activation is caused by higher energy neutrons (which should have experienced fewer collisions than those causing cobalt activation, for example), better agreement than what is reported here would be expected.

The TART Neutron-Photon Monte Carlo Transport Code

TART is a coupled neutron-photon Monte Carlo transport code available at LLNL.[1] It has been extensively tested against a series of pulsed sphere experiments,[2] where the agreement for materials of interest in this study (i.e., air, nitrogen) is good. (In materials that have very complex resonance structures in their cross sections, such as iron, codes like TART, which work with group-averaged cross sections run into trouble for calculations extending over many mean free paths.) The normal version of TART does not allow for rigorous treatment of thermal neutrons; rather than maintaining a Maxwellian distribution it downscatters neutrons to a designated "floor" energy (0.02 eV) where they remain until captured. A difficulty with all Monte Carlo transport codes is obtaining reasonable statistics after passage through a lot of material. For the case of the neutron kerma at Hiroshima, it was only possible to get good statistics out to 1200 m (though useful data may be derived out to about 1600 m). This limitation is tied to the total amount of computer time that can be invested.

There is a variant of TART called ALICE that alleviates some of the above-mentioned problems. For example, it gets around the "resonance" problem by using a so-called "probability table" method[3] to treat resonances. ALICE also allows proper treatment of a Maxwellian thermal distribution. Comparison runs were made for cobalt activation in Hiroshima; differences between TART and ALICE results were found to be less than 10% out to 1200 m (with the hypocenter showing a difference of 13%). A Maxwellian temperature of 0.025 eV was used in the ALICE calculations. In general, ALICE runs a lot slower than TART and therefore is not routinely used for transport calculations.

Cross Sections

LLNL maintains its own cross-section group that evaluates and maintains cross sections and generates a periodically updated library, called ENDL, which is used for all resident LLNL codes. ENDL has been extensively documented.[4] TART derives its cross-section information from ENDL and forms its own set of group-averaged cross sections (175 groups). Normally the group average is a straight arithmetic average (i.e., not weighted by the fluence); this is what is used.

Problem Geometry

A full 2-D geometry was zoned up for the TART calculations. The code actually works with a full 3-D configuration; however, zonal boundaries were such that there were two-dimensional symmetries that are essential to obtain optimum numbers of Monte Carlo samplings in the tally zones. The air zones consisted of a "stack of pancakes" to take account of changes in air density and zonal weights. Placed under this stack is a "bull's eye" of tally zones that were filled with the material of interest (tissue, sulfur, cobalt). Under this bull's eye there was a 30 cm thick cylinder of soil. The ceiling for this problem was 1095 m. The entire geometry was bounded by a cone of half angle 45° whose apex was located 2500 m above the hypocenter. Cone apex, burst point, and hypocenter were all located on the axis of symmetry for this geometry.

Data Used in These Calculations

The neutron sources used in this study were calculated by workers from LANL.[5] The Nagasaki source was one-dimensional. The Hiroshima source was two-dimensional and was used in the "untilted" position in the calculations. However, a sulfur comparison with a "tilted" position was made that will be discussed later. For kerma in tissue at a point in air and for cobalt activation the sources were split up into a high-energy and a low-energy component. The reason for the split relates to TART not allowing spectral weighting. Such weighting is necessary in Monte Carlo transport problems when there are large differences in amplitude across the energies of a source spectrum. The split takes the place of source weighting and allows the appropriate importance for the high-energy component, which has low amplitude in the neutron spectrum but which produces a significant part of the kerma at long ranges.

Data relating to air and soil density and composition were those provided by ORNL[6] (Tables 1 and 2). The air composition provided by ORNL was converted to atom fractions.

Table 1. Atmospheric Density, Composition, and Zoning (Burst Height: Hiroshima 580 m, Nagasaki 503 m)

Zone Number	Altitude (m)	Density (g/cc)	Composition (Atom Fraction)			
			Hydrogen	Nitrogen	Oxygen	Argon
Hiroshima						
14	0-125	.0011641	.0268	.7352	.2155	.0045
13	125-275	.0011499	.0258	.7544	.2153	.0045
11, 12	275-449	.0011334	.0247	.7557	.2151	.0045
8, 9, 10	449-635	.0011152	.0234	.7572	.2148	.0045
7	635-835	.0010959	.0222	.7587	.2146	.0045
6	835-1095	.0010732	.0207	.7604	.2144	.0045
Nagasaki						
14	0-125	.0011517	.0269	.7531	.2155	.0045
13	125-275	.0011388	.0255	.7547	.2152	.0045
11, 12	275-449	.0011236	.0240	.7565	.2147	.0045
8, 9, 10	449-635	.0011069	.0225	.7583	.2147	.0045
7	635-835	.0010891	.0209	.7602	.2144	.0046
6	835-1095	.0010681	.0191	.7623	.2141	.0046

Private Communications, J. V. Pace, ORNL, dated 29 August 1985 and 2 October 1985

Kerma in Tissue and Reaction Tallies

A major difference between the TART calculations and the ORNL DOT calculations is that "response functions" were not used to convert fluence to other desired quantities. Instead small amounts of the material of interest were put into the tally zones and energy and reactions calculated directly. The materials were placed directly on top of the soil. Both sulfur and cobalt were used alone without any shielding material (like electric insulators or concrete), except for one special calculation in which cobalt was embedded in iron. The composition listed in Table 3 was used for tissue. Reaction cross-section data used by TART for sulfur and cobalt are shown in Tables 4 and 5.

Results and Conclusion

Quantities calculated were neutron and secondary gamma-ray kerma in tissue for Hiroshima and Nagasaki and sulfur and cobalt activation for Hiroshima. Previously TART

Table 2. Summary of Data on Composition of Moist Soil Used in Atmohpheric Radiation Transport Calculations

Element		Atom density (atoms/barn-cm)[a]	Percent of total cross section for thermal neutron-capture gamma rays	Percent by mass of moist soil
Hiroshima				
Hydrogen	H	3.085E−2	66.53	3.04
Carbon	C	7.043E−4	0.02	0.83
Oxygen	O	3.759E−2	0.05	58.74
Sodium	Na	5.451E−4	1.99	1.22
Aluminum	Al	2.061E−3	3.31	5.43
Silicon	Si	9.315E−3	10.36	25.55
Chlorine	Cl	2.541E−6	0.58	0.01
Potassium	K	7.191E−4	10.46	2.75
Calcium	Ca	1.583E−4	0.47	0.62
Titanium	Ti	2.567E−5	1.08	0.12
Manganese	Mn	6.659E−6	0.61	0.04
Iron	Fe	2.482E−4	4.39	1.35
Total			99.85	99.70
Nagasaki				
Hydrogen	H	3.521E−2	66.20	3.47
Carbon	C	1.806E−3	0.04	2.12
Oxygen	O	3.857E−2	0.04	60.27
Sodium	Na	2.556E−4	0.81	0.57
Aluminum	Al	2.816E−3	3.94	7.42
Silicon	Si	6.806E−3	6.60	18.67
Chlorine	Cl	7.512E−6	1.50	0.03
Potassium	K	1.510E−4	1.92	0.58
Calcium	Ca	2.267E−4	0.59	0.89
Titanium	Ti	1.034E−4	3.81	0.48
Manganese	Mn	1.796E−5	1.44	0.10
Iron	Fe	8.247E−4	12.92	4.50
Total			99.81	99.10

[a]Assumes a moist-soil density of 1.7 g/cc (i.e., 1.3 g/cc of dry soil and 0.4 g/cc of water)

This table is contained in a letter from Kerr, ORNL, to R. Christy, CIT, dated 18 April 1985

Table 3. Tissue Composition

Element		Atom Percent
Hydrogen	H	6.30E−1[a]
Oxygen	O	2.40E−1
Carbon (12)	C	1.18E−1
Nitrogen	N	1.08E−2
Calcium	Ca	3.53E−5
Phosphorus	P	2.61E−4
Sulfur (32)	S	3.78E−4
Potassium	K	3.10E−4
Sodium	Na	2.99E−4
Chlorine	Cl	2.31E−4

[a]Read as 6.30×10^{-1} or 0.630, etc.

From M.S. Singh, Kerma Factors for Neutrons and Photons in the Energies Below 20 MeV, UCRL-52850, LLNL (1979)

Table 4. Reaction Cross Sections for ^{32}S for Neutrons Above 0.883 MeV

reaction cross sections by energy

react		1	9	9	9	2	2	2	2	
q =		0.	-2.239e+00	-1.504e+01	-8.865e+00	-9.200e-01	-1.269e+01	1.525e+00	8.642e+00	
	mev	elastic	n,n'g	n,2ng	n,n'pg	n,pg	n,tg	n,ag	n,g	total
126	8.834e-01	2.370e+00	0.	0.	0.	0.	0.	4.445e-03	1.214e-04	2.383e+00
127	1.025e+00	2.284e+00	0.	0.	0.	0.	0.	4.667e-03	9.952e-05	2.289e+00
128	1.176e+00	2.478e+00	0.	0.	0.	0.	0.	4.979e-03	9.878e-05	2.483e+00
129	1.338e+00	2.535e+00	0.	0.	0.	0.	0.	9.452e-03	9.799e-05	2.545e+00
130	1.511e+00	2.445e+00	0.	0.	0.	2.394e-04	0.	1.083e-02	9.715e-05	2.456e+00
131	1.694e+00	2.709e+00	0.	0.	0.	5.867e-03	0.	2.798e-02	9.626e-05	2.742e+00
132	1.887e+00	2.594e+00	0.	0.	0.	1.763e-02	0.	7.169e-02	9.532e-05	2.603e+00
133	2.091e+00	2.429e+00	0.	0.	0.	3.708e-02	0.	7.893e-02	9.433e-05	2.545e+00
134	2.305e+00	2.360e+00	4.557e-02	0.	0.	5.975e-02	0.	8.138e-02	9.329e-05	2.547e+00
135	2.530e+00	2.678e+00	1.961e-01	0.	0.	8.155e-02	0.	8.133e-02	9.225e-05	3.037e+00
136	2.741e+00	2.623e+00	2.463e-01	0.	0.	1.056e-01	0.	1.409e-01	9.111e-05	3.124e+00
137	3.011e+00	2.550e+00	3.033e-01	0.	0.	1.419e-01	0.	1.517e-01	8.907e-05	3.147e+00
138	3.267e+00	2.088e+00	3.581e-01	0.	0.	1.898e-01	0.	1.577e-01	8.863e-05	2.794e+00
139	3.533e+00	1.891e+00	4.898e-01	0.	0.	2.398e-01	0.	1.939e-01	8.734e-05	2.734e+00
140	3.811e+00	2.259e+00	4.499e-01	0.	0.	2.867e-01	0.	1.978e-01	8.600e-05	3.193e+00
141	4.069e+00	2.261e+00	4.925e-01	0.	0.	2.858e-01	0.	1.243e-01	8.469e-05	3.164e+00
142	4.396e+00	1.923e+00	5.378e-01	0.	0.	2.664e-01	0.	1.100e-01	8.318e-05	2.837e+00
143	4.784e+00	1.862e+00	5.787e-01	0.	0.	2.491e-01	0.	1.100e-01	8.177e-05	2.800e+00
144	4.991e+00	1.614e+00	5.971e-01	0.	0.	2.460e-01	0.	1.100e-01	8.024e-05	2.567e+00
145	5.353e+00	1.354e+00	5.916e-01	0.	0.	2.692e-01	0.	1.100e-01	7.866e-05	2.325e+00
146	5.658e+00	1.574e+00	5.858e-01	0.	0.	3.044e-01	0.	1.100e-01	7.703e-05	2.575e+00
147	6.042e+00	1.776e+00	5.799e-01	0.	0.	3.241e-01	0.	1.100e-01	7.535e-05	2.798e+00
148	6.367e+00	1.640e+00	5.741e-01	0.	0.	3.310e-01	0.	1.100e-01	7.370e-05	2.663e+00
149	6.737e+00	1.453e+00	5.676e-01	0.	0.	3.389e-01	0.	1.100e-01	7.183e-05	2.469e+00
150	7.156e+00	1.375e+00	5.608e-01	0.	0.	3.478e-01	0.	1.100e-01	6.991e-05	2.393e+00
151	7.548e+00	1.302e+00	5.545e-01	0.	0.	3.546e-01	0.	1.100e-01	6.813e-05	2.321e+00
152	7.918e+00	1.244e+00	5.491e-01	0.	0.	3.600e-01	0.	1.100e-01	6.629e-05	2.264e+00
153	8.322e+00	1.213e+00	5.454e-01	0.	0.	3.646e-01	0.	1.100e-01	6.422e-05	2.233e+00
154	8.787e+00	1.185e+00	5.410e-01	0.	1.896e-05	3.682e-01	0.	1.100e-01	6.219e-05	2.205e+00
155	9.177e+00	1.155e+00	5.382e-01	0.	3.980e-03	3.718e-01	0.	1.100e-01	6.011e-05	2.179e+00
156	9.665e+00	1.124e+00	5.342e-01	0.	1.078e-02	3.758e-01	0.	1.100e-01	5.788e-05	2.155e+00
157	1.012e+01	1.093e+00	5.304e-01	0.	1.741e-02	3.796e-01	0.	1.100e-01	5.570e-05	2.131e+00
158	1.058e+01	1.063e+00	5.267e-01	0.	2.384e-02	3.833e-01	0.	1.100e-01	5.359e-05	2.107e+00
159	1.101e+01	1.047e+00	5.227e-01	0.	3.078e-02	3.726e-01	0.	1.100e-01	5.131e-05	2.083e+00
160	1.155e+01	1.042e+00	5.186e-01	0.	3.785e-02	3.508e-01	0.	1.100e-01	4.898e-05	2.060e+00
161	1.199e+01	1.038e+00	5.146e-01	0.	4.471e-02	3.296e-01	0.	1.100e-01	4.673e-05	2.037e+00
162	1.250e+01	1.032e+00	5.101e-01	0.	5.246e-02	3.058e-01	0.	1.100e-01	4.418e-05	2.011e+00
163	1.307e+01	1.027e+00	5.058e-01	0.	5.998e-02	2.826e-01	2.169e-06	1.100e-01	4.171e-05	1.985e+00
164	1.354e+01	1.023e+00	5.025e-01	0.	6.571e-02	2.649e-01	4.615e-06	1.100e-01	3.983e-05	1.966e+00
165	1.386e+01	1.020e+00	5.000e-01	0.	7.049e-02	2.617e-01	6.391e-06	1.100e-01	3.843e-05	1.953e+00
166	1.413e+01	1.019e+00	4.977e-01	0.	7.811e-02	2.397e-01	8.023e-06	1.100e-01	3.714e-05	1.945e+00
167	1.441e+01	1.018e+00	4.955e-01	0.	8.635e-02	2.274e-01	1.304e-05	1.100e-01	3.584e-05	1.937e+00
168	1.468e+01	1.017e+00	4.922e-01	0.	9.004e-02	2.083e-01	1.023e-04	1.000e-01	3.399e-05	1.924e+00
169	1.519e+01	1.015e+00	4.877e-01	1.380e-05	1.141e-01	1.815e-01	2.581e-04	1.068e-01	3.146e-05	1.906e+00
170	1.575e+01	1.013e+00	4.830e-01	1.916e-04	1.313e-01	1.520e-01	4.004e-04	1.047e-01	2.874e-05	1.006e+00
171	1.633e+01	1.011e+00	4.781e-01	4.492e-04	1.489e-01	1.292e-01	5.697e-04	1.025e-01	2.597e-05	1.871e+00
172	1.692e+01	1.009e+00	4.731e-01	7.505e-04	1.667e-01	1.207e-01	7.330e-04	1.003e-01	2.315e-05	1.872e+00
173	1.752e+01	1.007e+00	4.681e-01	1.054e-03	1.849e-01	1.129e-01	9.088e-04	9.804e-02	2.029e-05	1.873e+00
174	1.813e+01	1.005e+00	4.630e-01	1.324e-03	2.033e-01	1.050e-01	1.071e-03	9.576e-02	1.737e-05	1.875e+00
175	1.875e+01	1.002e+00	4.552e-01	1.729e-03	2.313e-01	9.300e-02	1.320e-03	9.231e-02	1.295e-05	1.877e+00

Table 5. Reaction Cross Sections for ^{59}Co

react		9	9	2	2	2	2	2	
q =		-2.399e-00	-1.046e+01	-7.030e-01	-5.145e+00	-8.930e+00	3.201e-01	7.490e+00	
	mev	n,n'g	n,2ng	n,pg	n,dg	n,tg	n,ag	n,g	total
1	1.307e-09	0.	0.	0.	0.	0.	0.	1.109e+02	1.176e+02
2	5.227e-09	0.	0.	0.	0.	0.	0.	5.654e+01	6.319e+01
3	2.091e-08	0.	0.	0.	0.	0.	0.	3.670e+01	4.344e+01
4	3.267e-08	0.	0.	0.	0.	0.	0.	3.027e+01	3.693e+01
5	4.704e-08	0.	0.	0.	0.	0.	0.	2.371e+01	3.037e+01
6	8.322e-08	0.	0.	0.	0.	0.	0.	1.852e+01	2.510e+01
7	1.307e-07	0.	0.	0.	0.	0.	0.	1.535e+01	2.201e+01
8	1.882e-07	0.	0.	0.	0.	0.	0.	1.276e+01	1.942e+01
9	2.561e-07	0.	0.	0.	0.	0.	0.	1.106e+01	1.772e+01
10	3.345e-07	0.	0.	0.	0.	0.	0.	9.874e+00	1.654e+01
11	4.234e-07	0.	0.	0.	0.	0.	0.	8.761e+00	1.543e+01
12	5.123e-07	0.	0.	0.	0.	0.	0.	7.593e+00	1.426e+01
13	7.527e-07	0.	0.	0.	0.	0.	0.	6.214e+00	1.289e+01
14	1.176e-06	0.	0.	0.	0.	0.	0.	5.376e+00	1.206e+01
15	1.511e-06	0.	0.	0.	0.	0.	0.	4.594e+00	1.129e+01
16	2.091e-06	0.	0.	0.	0.	0.	0.	3.965e+00	1.060e+01
17	2.741e-06	0.	0.	0.	0.	0.	0.	3.532e+00	1.026e+01
18	3.533e-06	0.	0.	0.	0.	0.	0.	3.102e+00	9.934e+00
19	4.704e-06	0.	0.	0.	0.	0.	0.	2.886e+00	9.503e+00
20	5.658e-06	0.	0.	0.	0.	0.	0.	2.600e+00	9.480e+00
21	6.737e-06	0.	0.	0.	0.	0.	0.	2.430e+00	9.260e+00
22	8.322e-06	0.	0.	0.	0.	0.	0.	2.245e+00	9.180e+00
23	9.620e-06	0.	0.	0.	0.	0.	0.	2.101e+00	8.994e+00
24	1.101e-05	0.	0.	0.	0.	0.	0.	2.026e+00	8.960e+00
25	1.307e-05	0.	0.	0.	0.	0.	0.	1.954e+00	8.929e+00
26	1.460e-05	0.	0.	0.	0.	0.	0.	1.900e+00	8.907e+00
27	1.581e-05	0.	0.	0.	0.	0.	0.	1.843e+00	8.883e+00
28	1.758e-05	0.	0.	0.	0.	0.	0.	1.783e+00	8.858e+00
29	1.882e-05	0.	0.	0.	0.	0.	0.	1.724e+00	8.839e+00
30	2.075e-05	0.	0.	0.	0.	0.	0.	1.695e+00	8.925e+00

Table 5. Continued

react		9	9	2	2	2	2	2	
q =		-2.399e-00	-1.016e+01	-7.030e-01	-5.145e+00	-8.930e+00	3.201e-01	7.498e+00	
	mev	n,n'g	n,2ng	n,pg	n,dg	n,tg	n,ag	n,g	total
31	2.277e-05	0.	0.	0.	0.	0.	0.	1.602e+00	9.029e+00
32	2.417e-05	0.	0.	0.	0.	0.	0.	1.671e+00	9.116e+00
33	2.561e-05	0.	0.	0.	0.	0.	0.	1.660e+00	9.205e+00
34	2.718e-05	0.	0.	0.	0.	0.	0.	1.649e+00	9.297e+00
35	2.862e-05	0.	0.	0.	0.	0.	0.	1.640e+00	9.367e+00
36	2.940e-05	0.	0.	0.	0.	0.	0.	1.635e+00	9.443e+00
37	3.099e-05	0.	0.	0.	0.	0.	0.	1.655e+00	9.602e+00
38	3.345e-05	0.	0.	0.	0.	0.	0.	1.604e+00	9.803e+00
39	3.601e-05	0.	0.	0.	0.	0.	0.	1.714e+00	1.001e+01
40	3.865e-05	0.	0.	0.	0.	0.	0.	1.740e+00	1.019e+01
41	4.040e-05	0.	0.	0.	0.	0.	0.	1.761e+00	1.046e+01
42	4.234e-05	0.	0.	0.	0.	0.	0.	1.777e+00	1.073e+01
43	4.329e-05	0.	0.	0.	0.	0.	0.	1.799e+00	1.109e+01
44	4.619e-05	0.	0.	0.	0.	0.	0.	1.822e+00	1.145e+01
45	4.717e-05	0.	0.	0.	0.	0.	0.	1.839e+00	1.174e+01
46	4.918e-05	0.	0.	0.	0.	0.	0.	1.866e+00	1.212e+01
47	5.123e-05	0.	0.	0.	0.	0.	0.	1.922e+00	1.254e+01
48	5.332e-05	0.	0.	0.	0.	0.	0.	2.002e+00	1.309e+01
49	5.654e-05	0.	0.	0.	0.	0.	0.	2.067e+00	1.354e+01
50	5.763e-05	0.	0.	0.	0.	0.	0.	2.135e+00	1.401e+01
51	6.097e-05	0.	0.	0.	0.	0.	0.	2.252e+00	1.493e+01
52	6.325e-05	0.	0.	0.	0.	0.	0.	2.366e+00	1.580e+01
53	6.557e-05	0.	0.	0.	0.	0.	0.	2.453e+00	1.661e+01
54	6.674e-05	0.	0.	0.	0.	0.	0.	2.572e+00	1.759e+01
55	7.034e-05	0.	0.	0.	0.	0.	0.	2.700e+00	1.874e+01
56	7.155e-05	0.	0.	0.	0.	0.	0.	2.915e+00	2.053e+01
57	7.527e-05	0.	0.	0.	0.	0.	0.	3.177e+00	2.280e+01
58	7.780e-05	0.	0.	0.	0.	0.	0.	3.337e+00	2.418e+01
59	7.900e-05	0.	0.	0.	0.	0.	0.	3.517e+00	2.575e+01
60	8.167e-05	0.	0.	0.	0.	0.	0.	3.886e+00	2.911e+01
61	8.431e-05	0.	0.	0.	0.	0.	0.	4.390e+00	3.375e+01
62	8.834e-05	0.	0.	0.	0.	0.	0.	4.922e+00	3.852e+01
63	9.100e-05	0.	0.	0.	0.	0.	0.	5.648e+00	4.543e+01
64	9.306e-05	0.	0.	0.	0.	0.	0.	6.543e+00	5.403e+01
65	9.660e-05	0.	0.	0.	0.	0.	0.	7.222e+00	6.055e+01
66	9.811e-05	0.	0.	0.	0.	0.	0.	8.330e+00	7.145e+01
67	1.025e-04	0.	0.	0.	0.	0.	0.	1.297e+01	1.105e+02
68	1.099e-04	0.	0.	0.	0.	0.	0.	2.393e+01	2.322e+02
69	1.176e-04	0.	0.	0.	0.	0.	0.	7.350e+01	7.770e+02
70	1.256e-04	0.	0.	0.	0.	0.	0.	5.344e+02	6.156e+03
71	1.338e-04	0.	0.	0.	0.	0.	0.	2.511e+02	3.054e+03
72	1.406e-04	0.	0.	0.	0.	0.	0.	3.937e+01	5.254e+02
73	1.511e-04	0.	0.	0.	0.	0.	0.	1.225e+01	1.021e+02
74	1.601e-04	0.	0.	0.	0.	0.	0.	6.293e+00	1.023e+02
75	1.694e-04	0.	0.	0.	0.	0.	0.	4.309e+00	7.602e+01
76	1.709e-04	0.	0.	0.	0.	0.	0.	3.025e+00	5.618e+01
77	1.807e-04	0.	0.	0.	0.	0.	0.	1.745e+00	3.733e+01
78	1.900e-04	0.	0.	0.	0.	0.	0.	1.340e+00	3.089e+01
79	2.031e-04	0.	0.	0.	0.	0.	0.	6.342e-01	1.048e+01
80	2.741e-04	0.	0.	0.	0.	0.	0.	2.167e-01	1.004e+01
81	3.267e-04	0.	0.	0.	0.	0.	0.	1.222e-01	7.469e+00
82	3.811e-04	0.	0.	0.	0.	0.	0.	7.245e-02	5.838e+00
83	4.704e-04	0.	0.	0.	0.	0.	0.	5.304e-02	5.021e+00
84	4.991e-04	0.	0.	0.	0.	0.	0.	4.262e-02	4.395e+00
85	5.650e-04	0.	0.	0.	0.	0.	0.	3.501e-02	3.914e+00
86	6.042e-04	0.	0.	0.	0.	0.	0.	3.170e-02	3.725e+00
87	6.367e-04	0.	0.	0.	0.	0.	0.	2.652e-02	3.475e+00
88	7.156e-04	0.	0.	0.	0.	0.	0.	2.065e-02	3.203e+00
89	8.322e-04	0.	0.	0.	0.	0.	0.	1.721e-02	3.033e+00
90	9.177e-04	0.	0.	0.	0.	0.	0.	1.455e-02	2.975e+00
91	1.050e-03	0.	0.	0.	0.	0.	0.	1.207e-02	2.610e+00
92	1.307e-03	0.	0.	0.	0.	0.	0.	4.944e-02	2.245e+00
93	1.501e-03	0.	0.	0.	0.	0.	0.	9.822e-03	1.944e+00
94	1.882e-03	0.	0.	0.	0.	0.	0.	1.029e-02	1.773e+00
95	2.200e-03	0.	0.	0.	0.	0.	0.	1.130e-02	1.724e+00
96	2.561e-03	0.	0.	0.	0.	0.	0.	2.734e-01	2.042e+00
97	2.940e-03	0.	0.	0.	0.	0.	0.	1.803e-02	2.239e+00
98	3.345e-03	0.	0.	0.	0.	0.	0.	3.203e-02	4.609e+00
99	3.777e-03	0.	0.	0.	0.	0.	0.	2.232e-01	2.461e+01
100	4.234e-03	0.	0.	0.	0.	0.	0.	3.211e-01	1.535e+02
101	5.763e-03	0.	0.	0.	0.	0.	0.	6.361e-02	2.258e+01
102	7.527e-03	0.	0.	0.	0.	0.	0.	8.671e-02	1.109e+01
103	1.025e-02	0.	0.	0.	0.	0.	0.	5.972e-02	7.797e+00
104	1.511e-02	0.	0.	0.	0.	0.	0.	3.230e-02	7.393e+00
105	2.091e-02	0.	0.	0.	0.	0.	0.	3.781e-02	2.403e+01
106	2.646e-02	0.	0.	0.	0.	0.	0.	2.902e-02	1.301e+01
107	3.267e-02	0.	0.	0.	0.	0.	0.	1.902e-02	1.115e+01
108	3.953e-02	0.	0.	0.	0.	0.	0.	2.147e-02	6.794e+00
109	4.704e-02	0.	0.	0.	0.	0.	0.	9.543e-03	8.393e+00
110	5.761e-02	0.	0.	0.	0.	0.	0.	1.029e-02	6.120e+00
111	7.082e-02	0.	0.	0.	0.	0.	0.	1.277e-02	6.301e+00
112	8.322e-02	0.	0.	0.	0.	0.	0.	1.675e-02	8.370e+00
113	9.891e-02	0.	0.	0.	0.	0.	0.	2.096e-02	7.465e+00
114	1.307e-01	0.	0.	0.	0.	0.	0.	1.820e-02	4.383e+00
115	1.820e-01	0.	0.	0.	0.	0.	0.	1.545e-02	5.082e+00
116	2.075e-01	0.	0.	0.	0.	0.	0.	1.439e-02	4.081e+00
117	2.417e-01	0.	0.	0.	0.	0.	0.	1.359e-02	4.151e+00
118	2.710e-01	0.	0.	0.	0.	0.	0.	1.294e-02	4.376e+00
119	2.940e-01	0.	0.	0.	0.	0.	0.	1.225e-02	4.344e+00
120	3.345e-01	0.	0.	0.	0.	0.	0.	1.152e-02	4.384e+00
121	3.777e-01	0.	0.	0.	0.	0.	0.	1.074e-02	3.599e+00
122	4.234e-01	0.	0.	0.	0.	0.	0.	9.569e-03	4.329e+00
123	5.123e-01	0.	0.	0.	0.	0.	0.	8.320e-03	3.722e+00
124	6.325e-01	0.	0.	0.	0.	0.	0.	8.050e-03	3.782e+00
125	7.527e-01	0.	0.	3.047e-05	0.	0.	0.	8.431e-03	3.376e+00
126	8.034e-01	0.	0.	2.094e-04	0.	0.	0.	8.275e-03	3.383e+00
127	1.025e+00	0.	0.	4.040e-04	0.	0.	0.	7.942e-03	3.231e+00

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Table 5. Continued

	react	9	9	2	2	2	2	2	
	q =	-2.399e+00	-1.016e+01	-7.030e-01	-5.145e+00	-8.930e+00	3.201e-01	7.490e+00	
	mev	n,n'g	n,2ng	n,pg	n,dg	n,tg	n,ag	n,g	total
128	1.176e+00	0.	0.	6.124e-04	0.	0.	0.	7.504e-03	3.323e+00
129	1.338e+00	0.	0.	8.340e-04	0.	0.	0.	6.624e-03	3.154e+00
130	1.511e+00	0.	0.	1.071e-03	0.	0.	0.	5.620e-03	3.342e+00
131	1.694e+00	0.	0.	1.321e-03	0.	0.	0.	4.691e-03	3.255e+00
132	1.887e+00	0.	0.	1.581e-03	0.	0.	0.	4.072e-03	3.257e+00
133	2.091e+00	0.	0.	1.818e-03	0.	0.	0.	3.723e-03	3.292e+00
134	2.305e+00	1.432e-03	0.	2.059e-03	0.	0.	0.	3.417e-03	3.344e+00
135	2.530e+00	1.177e-02	0.	2.299e-03	0.	0.	0.	3.220e-03	3.421e+00
136	2.741e+00	2.797e-02	0.	2.564e-03	0.	0.	0.	3.099e-03	3.363e+00
137	3.011e+00	8.969e-02	0.	3.019e-03	0.	0.	0.	2.959e-03	3.440e+00
138	3.267e+00	2.042e-01	0.	3.620e-03	0.	0.	0.	2.820e-03	3.532e+00
139	3.533e+00	3.246e-01	0.	4.246e-03	0.	0.	0.	2.675e-03	3.508e+00
140	3.811e+00	4.409e-01	0.	4.879e-03	0.	0.	0.	2.535e-03	3.562e+00
141	4.059e+00	5.707e-01	0.	5.976e-03	0.	0.	0.	2.442e-03	3.603e+00
142	4.396e+00	7.110e-01	0.	7.311e-03	0.	0.	0.	2.362e-03	3.650e+00
143	4.704e+00	8.301e-01	0.	8.560e-03	0.	0.	5.757e-06	2.288e-03	3.701e+00
144	4.991e+00	9.534e-01	0.	1.037e-02	6.286e-06	0.	1.774e-04	2.207e-03	3.727e+00
145	5.353e+00	1.050e+00	0.	1.264e-02	8.562e-05	0.	4.740e-04	2.124e-03	3.753e+00
146	5.650e+00	1.141e+00	0.	1.500e-02	1.942e-04	0.	8.917e-04	2.038e-03	3.753e+00
147	6.042e+00	1.194e+00	0.	1.886e-02	3.057e-04	0.	1.509e-03	1.985e-03	3.730e+00
148	6.367e+00	1.219e+00	0.	2.372e-02	4.149e-04	0.	2.346e-03	1.959e-03	3.693e+00
149	6.737e+00	1.246e+00	0.	2.945e-02	5.391e-04	0.	3.536e-03	1.929e-03	3.645e+00
150	7.156e+00	1.260e+00	0.	3.739e-02	6.667e-04	0.	5.084e-03	1.899e-03	3.577e+00
151	7.540e+00	1.286e+00	0.	4.530e-02	7.853e-04	0.	6.605e-03	1.870e-03	3.520e+00
152	7.910e+00	1.291e+00	0.	5.318e-02	9.870e-04	0.	8.300e-03	1.841e-03	3.477e+00
153	8.322e+00	1.264e+00	0.	6.153e-02	1.045e-03	0.	1.034e-02	1.808e-03	3.407e+00
154	8.787e+00	1.235e+00	0.	6.933e-02	1.188e-03	2.710e-06	1.217e-02	1.776e-03	3.344e+00
155	9.177e+00	1.196e+00	0.	7.463e-02	1.318e-03	8.108e-05	1.382e-02	1.743e-03	3.304e+00
156	9.665e+00	1.154e+00	0.	7.969e-02	1.463e-03	1.943e-04	1.554e-02	1.708e-03	3.244e+00
157	1.012e+01	1.125e+00	0.	8.206e-02	1.535e-03	3.047e-04	1.722e-02	1.682e-03	3.174e+00
158	1.058e+01	1.103e+00	9.036e-03	8.340e-02	1.580e-03	4.117e-04	1.886e-02	1.660e-03	3.095e+00
159	1.101e+01	1.013e+00	9.267e-02	8.456e-02	1.620e-03	5.271e-04	2.062e-02	1.636e-03	3.024e+00
160	1.155e+01	8.836e-01	2.282e-01	8.554e-02	1.677e-03	6.448e-04	2.278e-02	1.611e-03	2.960e+00
161	1.199e+01	7.665e-01	3.398e-01	8.563e-02	1.725e-03	7.246e-04	2.516e-02	1.588e-03	2.897e+00
162	1.250e+01	6.419e-01	4.565e-01	8.482e-02	1.778e-03	7.783e-04	2.715e-02	1.561e-03	2.831e+00
163	1.307e+01	5.453e-01	5.407e-01	8.350e-02	1.830e-03	8.305e-04	2.844e-02	1.535e-03	2.768e+00
164	1.354e+01	4.905e-01	6.043e-01	8.239e-02	1.870e-03	8.782e-04	2.900e-02	1.515e-03	2.724e+00
165	1.386e+01	4.579e-01	6.382e-01	8.141e-02	1.900e-03	8.998e-04	2.987e-02	1.502e-03	2.692e+00
166	1.413e+01	4.390e-01	6.589e-01	7.988e-02	1.927e-03	9.270e-04	2.894e-02	1.500e-03	2.662e+00
167	1.441e+01	4.217e-01	6.781e-01	7.823e-02	1.954e-03	9.545e-04	2.870e-02	1.500e-03	2.631e+00
168	1.468e+01	3.903e-01	7.044e-01	7.582e-02	1.990e-03	9.900e-04	2.792e-02	1.500e-03	2.594e+00
169	1.519e+01	3.794e-01	7.208e-01	7.174e-02	2.000e-03	1.000e-03	2.633e-02	1.500e-03	2.554e+00
170	1.575e+01	3.636e-01	7.500e-01	6.727e-02	2.000e-03	1.000e-03	2.444e-02	1.500e-03	2.513e+00
171	1.633e+01	3.514e-01	7.689e-01	6.326e-02	2.000e-03	1.000e-03	2.234e-02	1.500e-03	2.476e+00
172	1.692e+01	3.420e-01	7.844e-01	5.924e-02	2.000e-03	1.000e-03	1.989e-02	1.500e-03	2.446e+00
173	1.752e+01	3.367e-01	7.964e-01	5.517e-02	2.000e-03	1.000e-03	1.755e-02	1.500e-03	2.423e+00
174	1.813e+01	3.350e-01	8.044e-01	5.133e-02	2.000e-03	1.000e-03	1.539e-02	1.500e-03	2.399e+00
175	1.875e+01	3.350e-01	8.057e-01	4.573e-02	2.000e-03	1.000e-03	1.266e-02	1.500e-03	2.360e+00

cpu 0 io 0 sys 0

showed good agreement when compared with pulsed sphere experiments with air and nitrogen (at least out to seven mean free paths). The results compare well with those derived from DOT S_n calculations performed at ORNL by J. Pace. However, the DOT code appears to have difficulty calculating values at and near the hypocenter; an observation previously made and verified by ORNL through comparison with ANISN calculations. TART, being a Monte Carlo code, has difficulty calculating values with good statistics at long distances. By and large, however, considering the differences in calculational method, energy group structures, and evaluated cross-section data bases, the agreement between TART and DOT is gratifying and reassuring. The calculational results are contained in Tables 6 to 12 and Figures 1 to 5. A comparison of the ratio of TART and DOT results at various ranges is contained in Figures 6 and 7. TART and ALICE calculations were also compared for bare cobalt activation and their ratio is plotted in Figure 8; the differences are small.

A brief comparison of the new data (particularly for Hiroshima) with certain previously published modern calculational results[7] is in order. These previous results were obtained with a different neutron source (1-D rather than 2-D for the case of Hiroshima) calculated by Preeg of LANL in 1976.[8] For neutron kerma, the new results for Hiroshima are lower by a factor of between 0.6 and 0.7 for the range of 1000 to 1500 m when compared with the previous values. The same comparison yields reasonable agreement for secondary gamma

Table 6. Calculated Neutron Kerma in Tissue (rad per indicated unit) Using the 2-D Hiroshima Source Spectrum.

Tally zone	Outside radius (m)	Tissue mass (g)	Tissue kerma (rad) per mole of neutrons			Tissue kerma per unit yield (rad kt^{-1})
			E<.823 MeV	E>.823 MeV	Total	
15	100	31.4	410.	723.	1134.	201.
16	200	94.2	337.	625.	962.	170.
17	300	157	236.	480.	716.	127.
18	400	220	130.	333.	463.	82.0
19	500	283	63.6	208.	272.	48.1
20	600	346	29.2	122.	151.	26.7
21	700	408	12.7	69.0	81.7	14.5
22	800	471	5.15	37.4	42.6	7.54
23	900	534	2.17	20.0	22.2	3.93
24	1000	597	(.795)	10.4	11.2	1.98
25	1100	660	(.297)	5.30	5.60	.991
26	1200	723	(.147)	2.78	2.93	.519
27	1300	785	h	1.53		
28	1400	848	h	.861		
29	1500	911	h	.406		
30	1600	974	h	.222		
31	1700	1037	h	(.117)		
32	1800	1100	no data	(.0741)		
33	1900	1162	no data	(.0605)		
34	2000	1225	no data	h		

h = high statistics; data not trustworthy

no data = no recorded particle histories

Values in parentheses: Standard Deviation 10-20%; caution advised

Normalization. For units per mole of neutrons; neutrons below .823 MeV: factor used = 5.59 X 10 exp.23 neutrons above .823 MeV: factor used = 4.28 X 10 exp.22. For units per kt (of total yield): factor = .177 moles per kt

Tissue Tally Zones. Tissue Tally Zones (15 through 34) consist of thin cylindrical annuli having the listed outside radius. The inside radius of these zones is the outside radius of the previous zone, except for Zone 1 (which is a thin cylinder of radius 100 m). The zones contain the indicated weight of tissue whose composition is listed on Table 3.

Table 7. Calculated Kerma in Tissue from Secondary Gamma Rays (rad per indicated unit) Using the 2-D Hiroshima Source Spectrum.

Tally zone	Outside radius (m)	Tissue mass (g)	Tissue kerma (rad) per mole of neutrons			Tissue kerma per unit yield (rad kt^{-1})
			E<.823 MeV	E>.823 MeV	Total	
15	100	31.4	1551.	252.	1803.	319.
16	200	94.2	1342.	222.	1564.	277.
17	300	157	1027.	180.	1201.	213.
18	400	220	728.	128.	856.	152.
19	500	283	471.	87.3	558.	98.8
20	600	346	311.	56.3	367.	65.0
21	700	408	193.	34.9	228.	40.4
22	800	471	128.	21.5	150.	26.6
23	900	534	82.9	13.3	96.2	17.0

→

Table 7. Continued

Tally zone	Outside radius (m)	Tissue mass (g)	Tissue kerma (rad) per mole of neutrons E<.823 MeV	E>.823 MeV	Total	Tissue kerma per unit yield (rad kt^{-1})
24	1000	597	55.4	7.98	63.4	11.2
25	1100	660	37.3	4.97	42.3	7.48
26	1200	723	26.4	3.21	29.6	5.24
27	1300	785	17.9	2.00	19.9	3.52
28	1400	848	12.9	1.35	14.3	2.53
29	1500	911	8.30	.911	9.21	1.63
30	1600	974	6.14	.588	6.73	1.19
31	1700	1037	4.50	.402	4.90	.867
32	1800	1100	2.90	.265	3.17	.561
33	1900	1162	1.81	(.233)	2.04	.361
34	2000	1225	1.37	.141	1.51	.267

Values in parentheses: Standard Deviation 10-20%; caution advised
For normalization and tissue tally zones see footnote Table 6

Table 8. Calculated (n,p) Reactions in ^{32}S Using the 2-D Hiroshima Source Spectrum, "Untilted" Weapon

Tally zone	Outside radius (m)	Sulfur mass (g)	(n,p) Reactions per source neutron ($\times 10^{14}$)	Standard deviation (%)	(n,p) Reactions per unit yield (mg kt^{-1})
15	100	31.4	2.68	.79	6469
16	200	94.2	7.35	.53	5915
17	300	157	10.2	.44	4902
18	400	220	11.1	.44	3827
19	500	283	10.1	.40	2705
20	600	346	8.25	.47	1813
21	700	408	6.27	.53	1172
22	800	471	4.55	.71	733
23	900	534	3.10	.84	441
24	1000	597	2.06	1.2	261
25	1100	660	1.34	1.3	154
26	1200	723	.878	1.6	92.0
27	1300	785	.552	2.2	53.3
28	1400	848	.336	2.9	30.0
29	1500	911	.207	3.4	17.3
30	1600	974	.137	4.2	10.7
31	1700	1037	.0801	5.2	5.86
32	1800	1100	.0510	6.6	3.52
33	1900	1162	.0258	9.3	1.69
34	2000	1225	.0191	10.4	1.19

Normalization. Only source neutrons above .823 MeV were used to run this problem. For normalization we used .0126 moles of neutrons above .823 MeV per kiloton (kt) of yield for the Hiroshima Little Boy weapon.

Sulfur Tally Zones. Sulfur Tally Zones (15 through 34) consist of thin cylindrical annuli having the listed outside radius. The inside radius of these zones is the outside radius of the previous zone, except for Zone 1 (which is a thin cylinder of radius 100 m).

Sulfur Isotope. Sulfur listed in this table refers to sulfur (Isotope 32)

Table 9. Calculated (n,γ) Reactions in ^{59}Co Using the 2-D Hiroshima Source Spectrum

Tally zone	Outside radius (m)	Cobalt mass (g)	(n,g) Reactions in cobalt per source neutron		(n,g) Reactions per mg of cobalt	
			E<.823 MeV ($\times 10^{10}$)	E>.823 MeV ($\times 10^{10}$)	Total per mole	Total per kt
15	100	31.4	.768	2.21	1.67e+06	2.96e+05
16	200	94.2	1.84	5.81	1.36e+06	2.41e+05
17	300	157	2.09	7.59	9.51e+05	1.68e+05
18	400	220	1.65	7.38	5.63e+05	9.97e+04
19	500	283	1.03	6.02	2.94e+05	5.20e+04
20	600	346	.577	4.29	1.46e+05	2.58e+04
21	700	408	.268	2.90	6.71e+04	1.19e+04
22	800	471	.128	1.78	3.14e+04	5560.
23	900	534	.0532	1.06	1.41e+04	2500.
24	1000	597	.0190	.606	6120.	1080.
25	1100	660	(.00809)	.341	2900.	513.
26	1200	723	(.00234)	.175	1220.	216.
27	1300	785	h	.0945		
28	1400	848	h	.0531		
29	1500	911	h	.0251		
30	1600	974	h	(.0163)		
31	1700	1037	h	(.00675)		
32	1800	1100	no data	(.00598)		
33	1900	1162	no data	h		
34	2000	1225	no data	h		

h = high statistics; data not trustworthy
no data = no recorded particle histories
Values in parenthesis: Standard Deviation 10-20%; caution advised

Normalization. For units per mole of neutrons; neutrons below .823 MeV: factor used = 5.59 X10 exp.23 neutrons above .823 MeV: factor used = 4.28 X10 exp.22. For units per kt (of total yield): factor = .177 moles per kt

Tissue Tally Zones. Tissue Tally Zones (15 through 34) are thin cylindrical annuli with indicated outside radius. The inside radius of each zone is the outside radius of the previous zone, except for Zone 15 (which is a thin cylinder of radius 100 m). These zones contain the indicated mass of cobalt.

Table 10. Calculated Kerma in Tissue from Neutrons (rad per indicated unit), Nagasaki

Tally zone	Outside radius (m)	Tissue mass (g)	Tissue kerma (rad) per mole of neutrons			Tissue kerma per unit yield (rad kt^{-1})
			E<1.24 keV	E>1.24 keV	Total	
15	100	31.4	8.05	216.	224.	61.2
16	200	94.2	5.54	186.	192.	52.5
17	300	157	2.38	137.	139.	38.0
18	400	220	.719	91.8	92.5	25.3
19	500	283	.164	56.6	56.8	15.5
20	600	346	(.0252)	33.1	33.1	9.05
21	700	408	h	18.6	18.6	5.09
22	800	471	h	10.2	10.2	2.79
23	900	534	h	5.67	5.67	1.55
24	1000	597	no data	3.15	3.15	.861

→

Table 10. Continued

Tally zone	Outside radius (m)	Tissue mass (g)	Tissue kerma (rad) per mole of neutrons E<1.24 keV	E>1.24 keV	Total	Tissue kerma per unit yield (rad kt^{-1})
25	1100	660		1.68	1.68	.459
26	1200	723		.867	.867	.237
27	1300	785		.499	.499	.136
28	1400	848		.302	.302	.0826
29	1500	911		.150	.150	.0410
30	1600	974		.0823	.0823	.0225
31	1700	1037		(.0482)	(.0482)	(.0132)
32	1800	1100		(.0287)	(.0287)	(.00785)
33	1900	1162		(.0172)	(.0172)	(.00472)
34	2000	1225	no data	h	h	h

h = high statistics; data not trustworthy
no data = no recorded particle histories
Values in parenthesis: Standard Deviation 10-20%; caution advised
For normalization and tissue tally zones see footnote Table 6

Table 11. Calculated Kerma in Tissue from Secondary Gamma Rays (rad per indicated unit), Nagasaki

Tally zone	Outside radius (m)	Tissue mass (g)	Tissue kerma (rad) per mole of neutrons E<1.24 keV	E>1.24 keV	Total	Tissue kerma per unit yield (rad kt^{-1})
15	100	31.4	1261.	52.2	1313.	359.
16	200	94.2	1048.	45.8	1094.	299.
17	300	157	833.	34.5	868.	237.
18	400	220	599.	24.1	623.	170.
19	500	283	423.	15.8	439.	120.
20	600	346	284.	9.72	294.	80.4
21	700	408	192.	5.94	198.	54.1
22	800	471	128.	3.69	132.	36.1
23	900	534	87.8	2.21	90.0	24.6
24	1000	597	58.4	1.42	59.8	16.3
25	1100	660	40.2	.884	41.1	11.2
26	1200	723	26.5	.551	27.1	7.41
27	1300	785	18.8	.339	19.1	5.22
28	1400	848	12.7	.216	12.9	3.53
29	1500	911	8.76	.134	8.89	2.43
30	1600	974	6.00	.105	6.11	1.67
31	1700	1037	4.34	.0626	4.40	1.20
32	1800	1100	3.34	(.0492)	3.39	.927
33	1900	1162	2.15	(.0296)	2.18	.596
34	2000	1225	1.67	(.0222)	1.69	.462

Values in parentheses: Standard Deviation 10-20%; caution advised
For normalization and tissue tally zones see footnote Table 6

Table 12. Calculated Kerma in Tissue from Source Gamma Rays (rad per indicated unit), Nagasaki

Tally zone	Outside radius (m)	Tissue mass (g)	Tissue kerma (rad) per mole of gamma rays			Tissue kerma per unit yield (rad kt^{-1})
			E<5 MeV	E>5 MeV	Total	
15	100	31.4	2513.	38.6	2552.	158.
16	200	94.2	2170.	34.9	2205.	137.
17	300	157	1663.	28.1	1691.	105.
18	400	220	1152.	21.2	1173.	72.7
19	500	283	756.	15.5	772.	47.8
20	600	346	473.	10.9	484.	30.0
21	700	408	293.	7.54	301.	18.6
22	800	471	177.	5.16	182.	11.3
23	900	534	108.	3.55	112.	6.92
24	1000	597	64.5	2.49	67.0	4.15
25	1100	660	40.0	1.73	41.7	2.58
26	1200	723	24.0	1.19	25.2	1.56
27	1300	785	14.8	.846	15.7	.970
28	1400	848	8.91	.594	9.50	.589
29	1500	911	5.72	.426	6.15	.381
30	1600	974	3.42	.297	3.72	.231
31	1700	1037	2.31	.205	2.52	.156
32	1800	1100	1.40	.140	1.54	.0954
33	1900	1162	.826	.101	.927	.0575
34	2000	1225	.564	.071	.635	.0394

Normalization. For units per mole of gamma rays. Source gamma rays below 5 MeV: factor used = 6.004 X10 exp.23, source gamma rays above 5 MeV: factor used = 1.625 X10 exp.21. For units per kt (of total yield): factor = .0620 moles per kt

For tissue tally zones see footnote Table 6

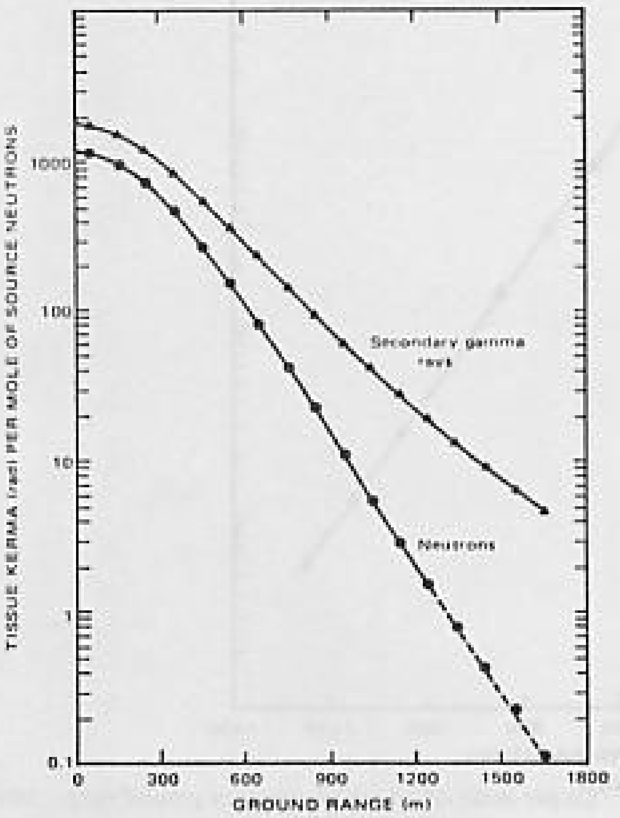

Figure 1. Kerma in tissue due to prompt source neutrons vs ground range, Hiroshima

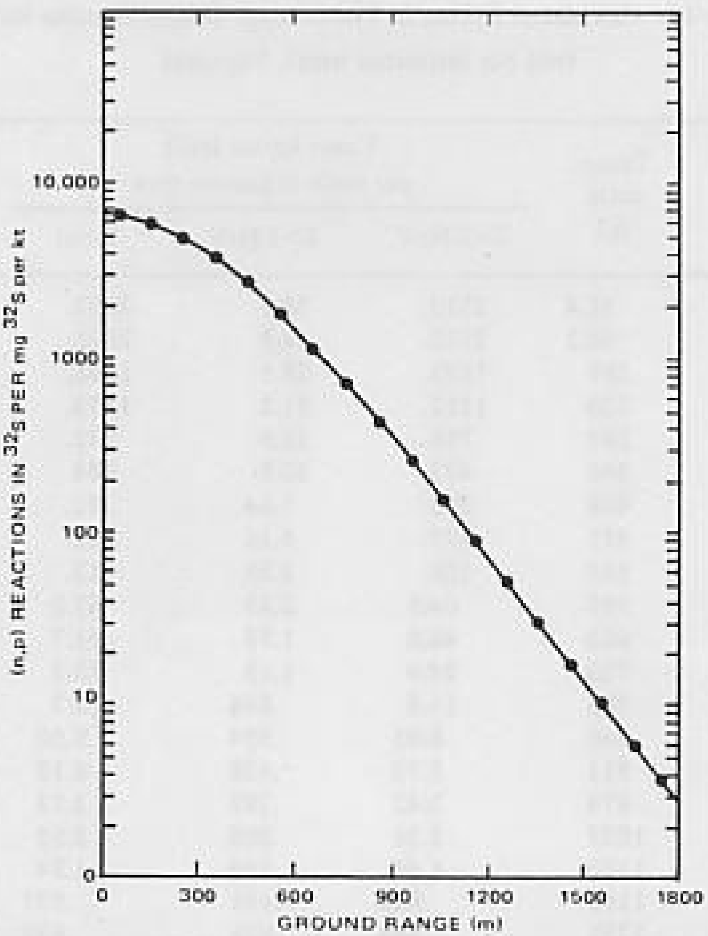

Figure 2. Calculated (n,p) reactions in ^{32}S per mg of ^{32}S per kt vs ground range, Hiroshima

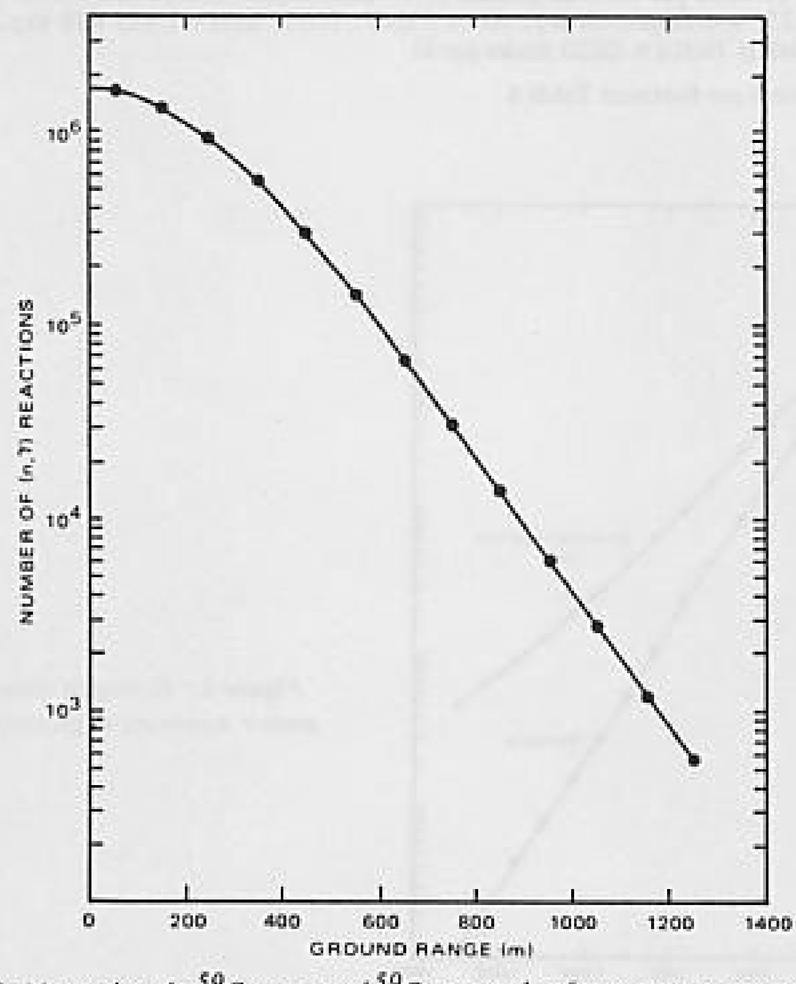

Figure 3. Calculated (n,γ) reactions in ^{59}Co per μg of ^{59}Co per mole of source neutrons vs ground range, Hiroshima

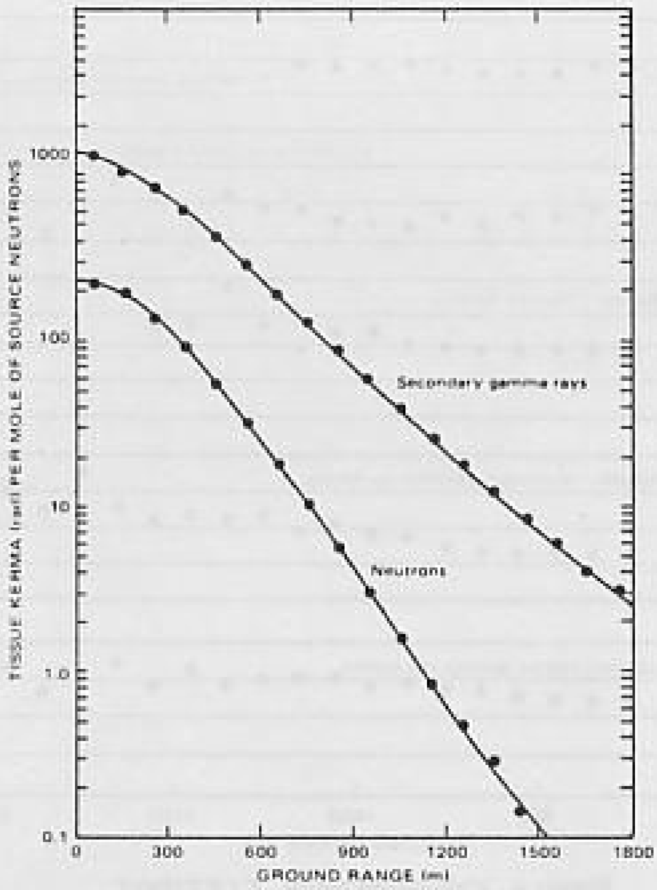

Figure 4. Kerma in tissue due to prompt source neutrons vs ground range, Nagasaki

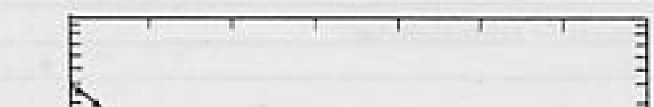

Figure 5. Kerma in tissue due to prompt gamma rays vs ground range, Nagasaki

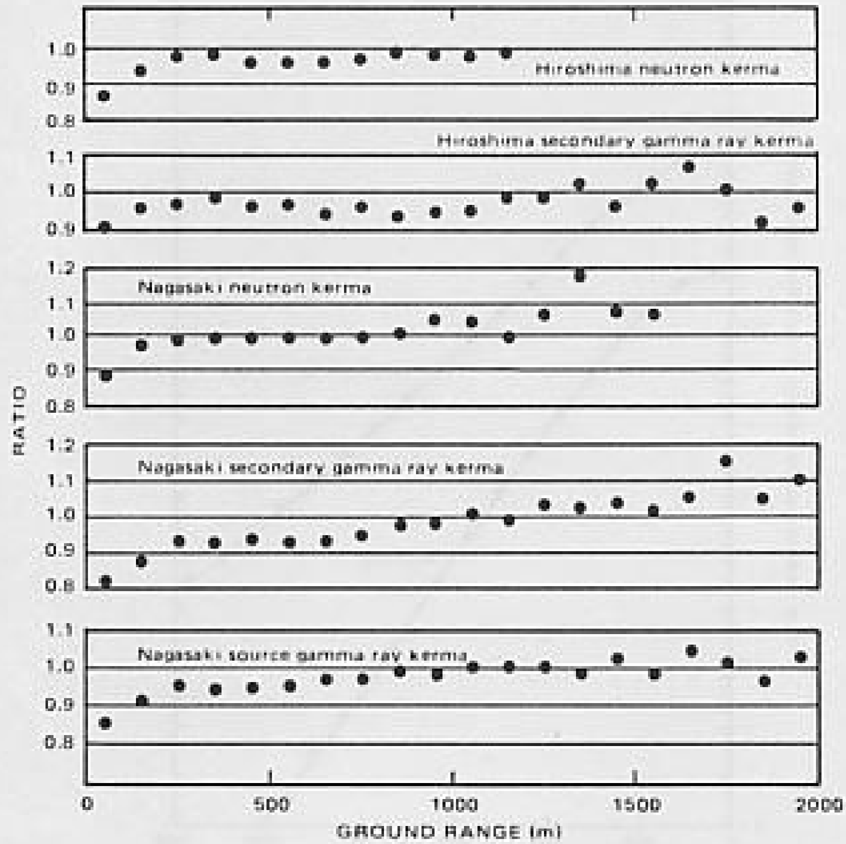

Figure 6. Calculated ratios, TART/DOT

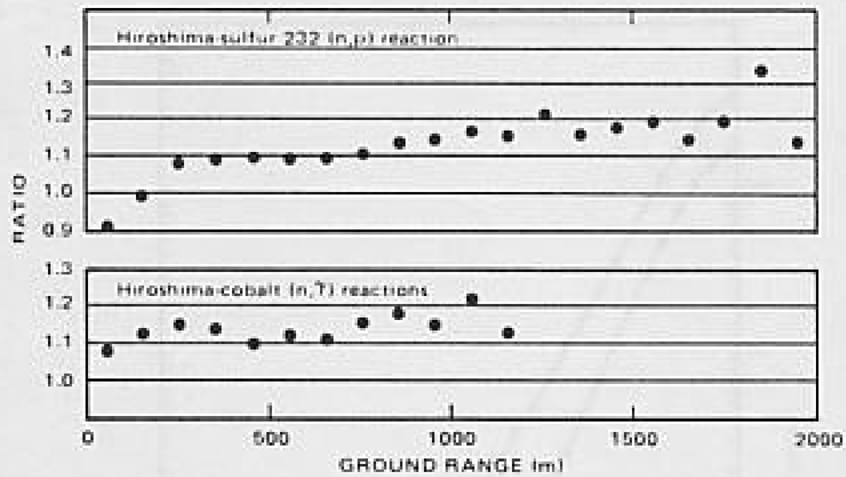

Figure 7. Calculated ratios, TART/DOT

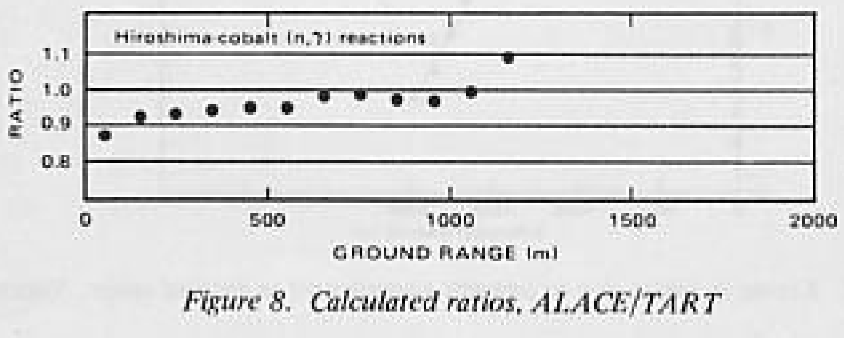

Figure 8. Calculated ratios, ALACE/TART

rays. The reasons for these differences in neutron kermas have not been investigated in detail. A major cause could be the smaller number of high energy neutrons in the new source when compared with the old one (0.0043 mol/kt above 1.33 MeV compared to 0.0073 mol/kt).

A special calculation was done for sulfur activation using the LANL two-dimensional Hiroshima neutron source but tilting the source 15°, which corresponds to the inclination of the bomb at the time of burst. The results, presented as ratios of tilted to untilted values, can be seen in Figure 9. To obtain reasonable Monte Carlo statistics, the angular subdivision of the angular tally cylinders on the ground was limited to four quadrants. Because two of the quadrants occupy the same relative position, the actual angular definition is limited to three values. At any rate, the maximum difference between tilted and untilted activation is about 25%.

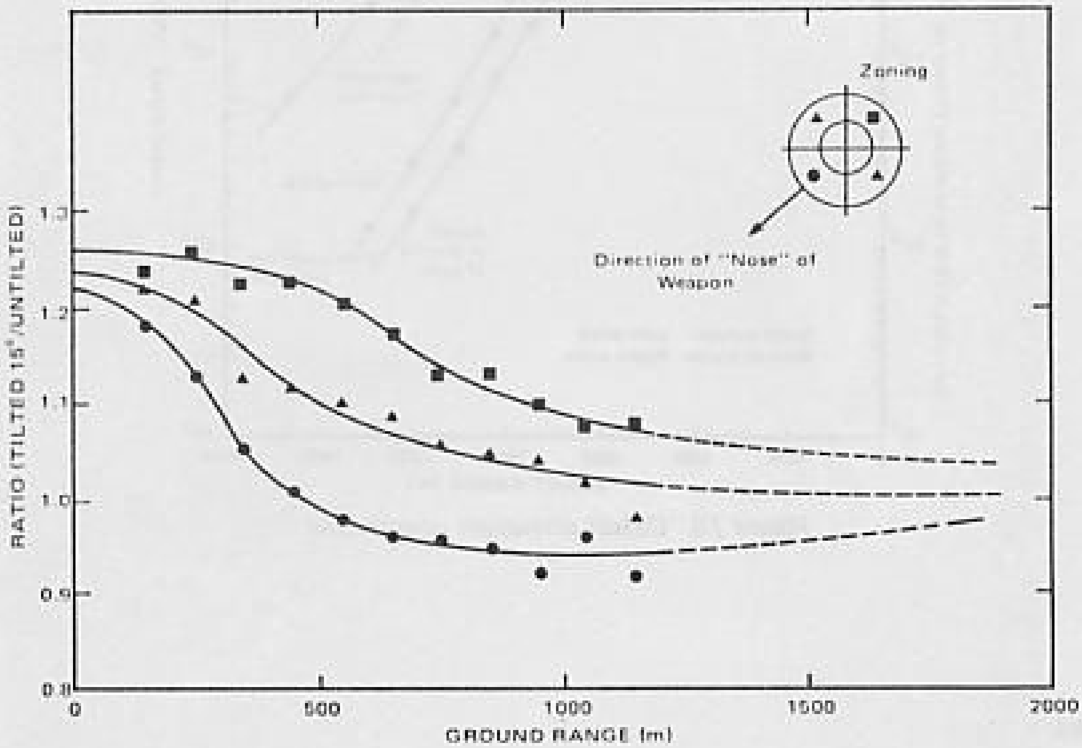

Figure 9. The effect of weapon tilt on sulfur activation in Hiroshima

A few special cobalt calculations were made and the results are shown in Figure 10. Note that the abscissa is slant range, not ground range as in the other figures. There is a comparison of bare cobalt on ground and cobalt embedded in a 2 cm thick sheet of iron placed on top of the ground (cobalt concentration: 1 ppm); the indicated difference is about a factor of two, which is caused by the iron absorbing low energy neutrons. A curve that shows calculated bare cobalt activation due to a fission source spectrum was also plotted. Shown on the same graph are the "iron ring" cobalt activation data of Hashizume et al.[9] Note that the shape of the iron ring curve is quite similar to the fission spectrum curve; indeed, making allowance for cobalt embedded in iron, the two curves might come close together. Thus one could infer the possibility of a fission like spectrum with total source strength of approximately 1 mole of neutrons giving rise to the iron ring data; a conjecture unsupported by other data.

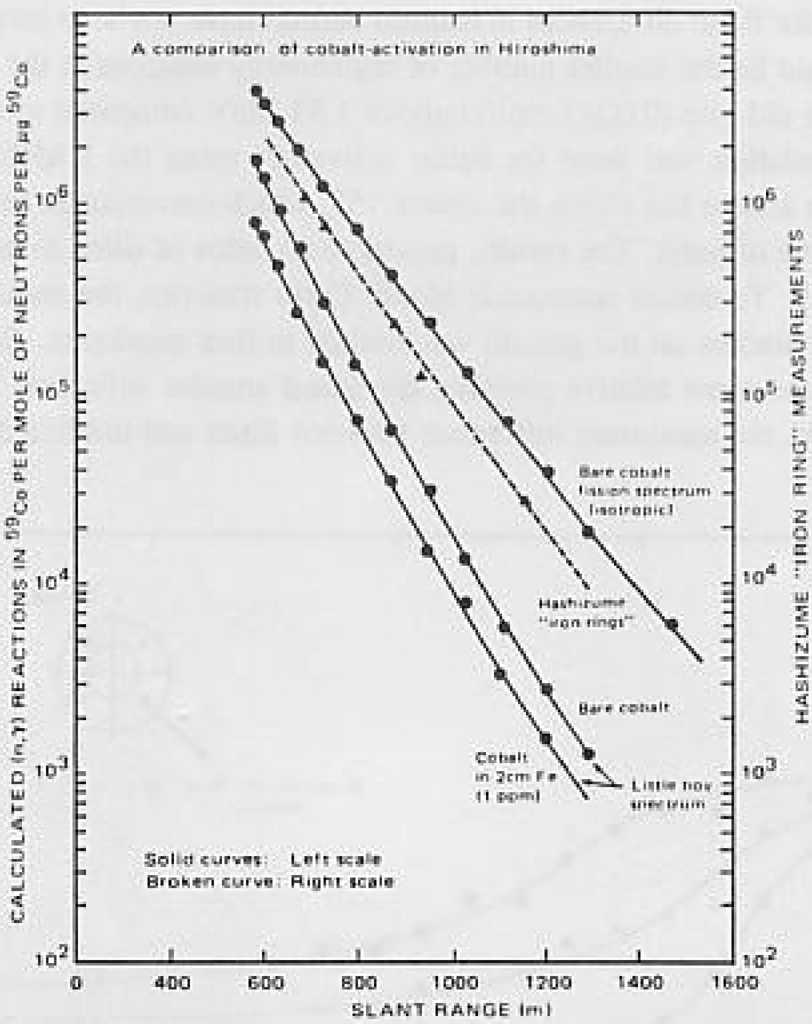

Figure 10. Cobalt activation comparison

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