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RADIATION DOSE CALCULATIONS WITH THE MONTE CARLO METHOD

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ABSTRACT

For high penetrating radiations such as neutrons and high-energy gamma rays, that deposit their energies uniformly over a large volume of a target medium, the determination of dose is commonly based on calculating the flux and then applying flux-to-dose conversion factors. For charged particles and low-energy gamma and x rays, their energies are deposited locally, and flux-to-dose conversion factors could be complex functions of medium geometry and source conditions. Using an isotropic beta source, we calculated the doses in water layers (used to simulate tissue), and found that dose values depend on the size and shape of the water "cell". We recommend that a "standard" size and shape of tissue equivalent material be established for beta dose calculations.

I. INTRODUCTION

Monte Carlo transport calculations have, for a long time, been used to determine the radiation dose, at the point of interest, due to given radiation sources. In most cases, detailed transport calculations are carried out to determine the flux at a given point and then convert to dose with the established flux to dose conversion factors. For gamma rays and neutrons there exist several sets of conversion factors, such as NCRP 38[1] and ANSI/ANS 77.[2] In the recent ICRP 51 report,[3] data were given not only for neutral particles like neutrons and photons, but also for charged particles such as protons, electrons, muons and pions.

In many cases, the flux to dose conversion factors were established by calculations based on a parallel beam and a semi-infinite medium. These results can not be applied to point isotropic sources. Several Monte Carlo calculations have been performed, with point isotropic [illegible] and neutrons, to illustrate the usefulness of the Monte Carlo method. The low-energy photon case is not considered in this paper. We will discuss our results and some issues associated with dose calculations.

II. MONTE CARLO CODES

There exist many Monte Carlo transport codes at research institutes, universities, and laboratories over the world. The two codes that are used most frequently at the Los Alamos National Laboratory are MCNP[4] and ITS.[5] They are readily available at our institution and are very user friendly.

ITS is the Integrated TIGER Series of coupled electron / photon Monte Carlo transport codes, which were developed at the Sandia National Laboratory. They are time-integrated and can accommodate multimaterials. TIGER is used for 1-D calculations, CYLTRAN is for 2-D cylindrical-symmetry cases, and ACCEPT is a 3-D code. The ITS code is based on ETRAN,[6] combining the conventional single-scattering photon Monte Carlo approach with a condensed-history electron Monte Carlo technique. The photon transport simulates all energetic physical processes including photoelectric absorption, Compton scattering and pair production. The electron transport includes energy loss straggling, multiple elastic scattering, and the production of knock on electrons, continuous bremsstrahlung, characteristic x rays, and annihilation radiation. In all cases, generation and transport of secondary particles are also included down to a preset energy cutoff; the lowest cutoff is 1.0 keV for both photons and electrons. The photoionization and electron impact ionization as well as relaxation by fluorescence and the Auger process are also considered but only in the case of the K shell of the element with the highest atomic number for a given material.

MCNP is a general purpose, continuous energy, generalized geometry, time dependent coupled neutron/photon/electron Monte Carlo transport code, developed at the Los Alamos National Laboratory. For many years, it addressed neutral particle transport problems only. It could be used in one of three modes: neutron transport only, photon transport only, or [illegible] neutron/photon transport, where the photons are produced by neutron [illegible] the production and transport of electrons are added, and one can use the code for many different modes. The neutron energy regime is from 10^{-11} MeV to 20 MeV, and the photon/electron energy regime is from 1 keV to 100 MeV. MCNP uses continuous-energy nuclear data libraries, and more than 500 neutron interaction tables are available for about 100 different isotopes or elements. The code employs very elaborate variance reduction schemes, including geometry splitting, Russian roulette, weight-, time-, and energy-cutoff, energy- and cell-dependent weight windows, DXTRAN,[4] etc., to improve the statistics of the calculations.

III. BETA AND GAMMA DOSE CALCULATIONS

Both Monte Carlo codes can calculate the energy deposited in a given zone (volume, mass). In principle, one can calculate the absorbed dose in the given zone by the definition:

Absorbed Dose = Energy Deposited / Mass.

The smaller the mass, the more accurate the absorbed dose value. However, absorbed dose, hereafter referred to as dose, is a macrodosimetry concept; it is not suitable for small objects like DNA or a tissue cell. There exists no standard size for beta dose calculations. We choose a size of 5.0×10^{-4} cm^3, which consists of about 10^7 cells, and is small enough to show the non uniformity of the dose distribution, but large enough to be a macro object.

To illustrate the dose calculation, we use the example by Chabot et al,[7] where a ^{60}Co point isotropic source resides on the surface of protective clothing (0.025 g/cm^2), and an air gap of 1 mm exists. The energy depositions were calculated for a water layer of 0.007 g/cm^2, which simulates skin, and several successive 0.005 g/cm^2 water layers beneath the skin. In each layer we tally the energy depositions in zones of cylinders of different diameters.

^{60}Co emits beta particles with a maximum energy of 318 keV. The spectrum is shown in Fig. 1. ^{60}Co decays to the excited states of ^{60}Ni, followed by two gamma rays with energies of 1.332 and [illegible] MeV.

[illegible] summarize the results, Beta and gamma doses are presented at different layers of water for the different [illegible] every layer of water both beta [illegible] take different amounts of mass we have different dose values. For example, consider the first .005 water layer. The beta dose for a mass in the area of .1cm^2 is 3.410 rad/hr per μCi of ^{60}Co. When we double the area (double the mass), the dose value decreases to 1.852. Note also that for a given area the gamma doses are relatively uniform through different layers, but the beta doses decrease rapidly as the thickness increases.

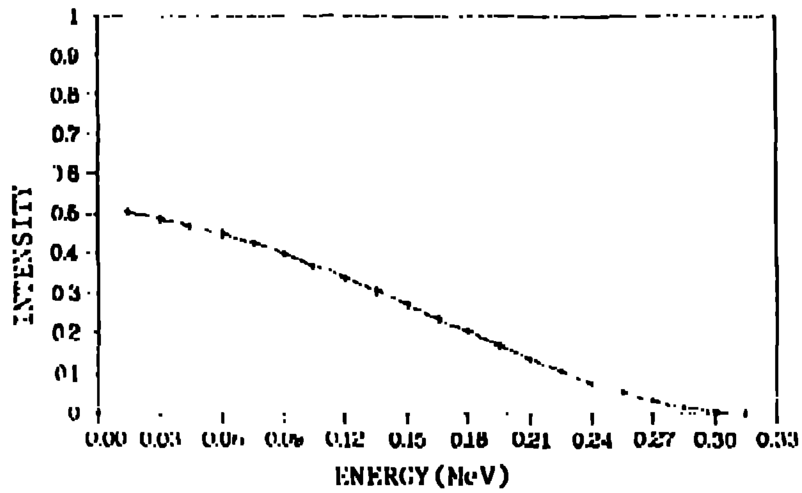


Fig. 1, ^{60}Co Beta Spectrum.

Table 1. Beta and Gamma Doses in Water-layer

Water Layer Thickness (cm)		Dose (rad/hr) per μCi of ^{60}Co — Area .1cm^2	.2cm^2	.5cm^2	1cm^2	2cm^2
.007	β	6.103	3.468	1.471	0.749	0.379
	γ	0.208	0.164	0.122	0.088	0.060
.005	β	3.410	1.852	0.765	0.385	0.200
	γ	0.187	0.156	0.122	0.090	0.062
.005	β	2.106	1.118	0.458	0.230	0.115
	γ	0.188	0.159	0.132	0.100	0.066
.005	β	0.980	0.515	0.207	0.104	0.052
	γ	0.202	0.165	0.128	0.097	0.066
.005	β	0.461	0.249	0.096	0.048	0.024
	γ	0.220	0.193	0.150	0.107	0.070
.005	β	0.208	0.107	0.043	0.022	0.011
	γ	0.193	0.186	0.142	0.104	0.069
[illegible]	β	[illegible]	0.038	0.015	0.0076	0.0039
	γ	[illegible]	0.186	0.132	0.104	[illegible]
[illegible]	β	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
	γ	[illegible]	[illegible]	0.131	[illegible]	[illegible]

The dose value is not only a function of mass, it also varies for different size of cylinders. Table 2 shows that for a given mass, but with different cylinders, the dose values are different. For example, for 5 mg of water, we consider three different cylindrical areas and thichness: .2cm^2 x .25mm (0.766), .5cm^2 x .1mm (0.612), and 1cm^2 x .05mm (0.382). The numbers in the parentheses are the corresponding dose values, the differences can be as much as a factor of two.

Table 2. Shape-Dependence of Dose

Mass of Water (gm)	Geometry Area x Thickness	Dose (rad/hr-μCi) Beta	Gamma
5.0e-4	.1cm^2 x .05mm	3.410	0.187
1.0e-3	.1cm^2 x .10mm	2.756	0.214
	.2cm^2 x .05mm	1.852	0.156
2.0e-3	.1cm^2 x .20mm	1.740	0.199
	.2cm^2 x .10mm	1.485	0.158
2.5e-3	.1cm^2 x .25mm	1.433	0.198
	.5cm^2 x .05mm	0.765	0.122
3.0e-3	.1cm^2 x .30mm	1.207	0.198
	.2cm^2 x .15mm	1.162	0.161
5.0e-3	.2cm^2 x .25mm	0.766	0.172
	.5cm^2 x .10mm	0.612	0.127
	1cm^2 x .05mm	0.385	0.092
1.0e-2	.5cm^2 x .20mm	0.382	0.133
	1cm^2 x .10mm	0.107	0.096

For these calculations, we used one layer of clothing. The gamma doses are smaller than the corresponding beta doses in shallow water layers, and then become larger for deeper cases If we use two layers of clothing, gamma dose is more important, even at shallow sites, since most of beta particles are stopped in the protective clothing. Table 3 shows the results.

Table 3. Doses with Different Clothing

Water Layer Thickness (cm)	One-Layer-Clothing Dose(rad/hr-μCi) Beta	One-Layer-Clothing Gamma	Two-Layer-Clothing Beta	Two-Layer-Clothing Gamma
.007	6.103	0.208	0.292	0.175
.005	3.410	0.187	0.096	0.181
.005	2.106	0.188	0.047	0.201
.005	0.980	0.202	0.019	0.221
.005	0.461	0.220	0.013	0.239

In this calculation, we used a point source. The effect due to self absorption was not included. For real sources of finite size, beta and gamma spectra changes due to self absorption, production of Compton electrons, internal conversion electrons from gamma rays should be considered. We have another article in preparation to address these effects for microsize ^{60}Co sources.

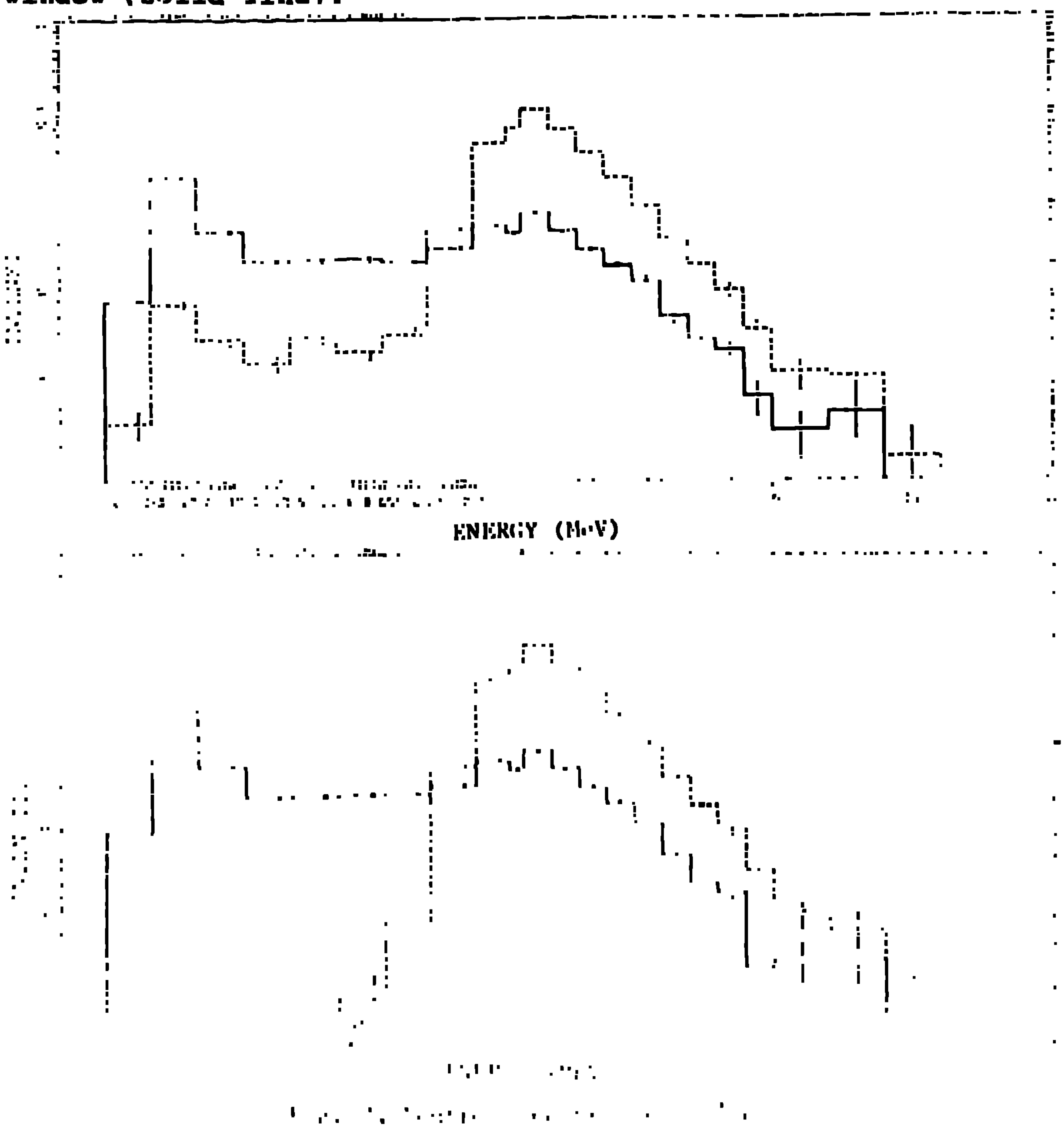
Our results differ from those of Chabot et al. For one layer of clothing and a 1 mm air gap their point-to-point dose was 7.98 rad/hr per 1 μCi of ^{60}Co. Their dose averaged over 1 cm^2 at a tissue (water) depth of 7 mg/cm^2 was 0.272 rad/hr per 1 μCi of ^{60}Co. We understand that Chabot et al. calculated the beta dose with semi-empirical equations based on flux-to-dose conversion method. Such an approach depends on the calculation of the conversion factors. One must consider parallel beam vs point isotropic-source incidence, one dimensional medium vs three dimensional geometry, the mass and shape of the "cell" concerned, and the depth profiles, etc. For example, if these semi empirical equations were established through calculations with a parallel beam, they may not be applied to this point isotropic source case.

IV. NEUTRON DOSE CALCULATION

If we disregard how the flux to dose conversion factors were established, the neutron dose calculation is straight forward. We used the point detector tally in MCNP to calculate the flux at the point of interest, and then [illegible] (that is the dose equivalent for a human body at [illegible]). However, [illegible] are produced by [illegible] therefore a separate gamma dose tally should be made.

For the following example, we considered a 20-gm ^{240}Pu metal ball in a glovebox of 90 cm x 90 cm x 90 cm with five iron walls of 2 mm thickness, and a front wall of 10 cm CH_2. The tallies were made at several points 10 cm from the CH_2 surface but at different heights relative to the center of the ball. Calculations were also carried out without iron walls to compare dose contributions from iron scatterings.

Figure 2 shows the neutron spectra from ^{240}Pu before entering the CH_2 window (dashed line) and after exiting the CH_2 window (solid line).



The lower figure is for the case without iron walls. The dashed line shows the ideal neutron spectrum from Pu metal, and it is seen that not many low-energy neutrons are present. The upper figure is for the case with iron walls. One can now see the presence of low-energy neutrons scattered from the iron walls, but most of them are absorbed in the CH_2 so that the spectrum after CH_2 is almost the same as for the other case

Table 4 summarizes the neutron and gamma dose values at four different points. Point A is at 10 cm from the CH_2 surface and at the same height as the Pu ball. Point B is 5 cm below A. Point C is 10 cm below A. Point D is 20 cm below A. The numbers in parentheses are the associated percent uncertainties. Both gamma and neutron doses with and without iron walls are about the same within uncertainties at all four points. Note that gammray and neutron doses peak at different places. There was about one gamma ray produced for every two neutrons, produced primarily in the Pu metal ball. The gamma rays following the decay of ^{240}Pu and from the fission products are not included in this calculation. A separate calculation should be carried out to accurately evaluate the gamma dose.

Table 4, Neutron and Gamma Doses from ^{240}Pu

Position	Dose (mrem/hr per gm of ^{240}Pu)			
	Neutron		Gamma	
	With Iron	No Iron	With Iron	No Iron
A	7.02 (3)	7.06 (3)	.148 (2)	.146 (3)
B	6.87 (3)	6.91 (3)	.144 (2)	.145 (3)
C	6.22 (3)	6.28 (3)	.186 (3)	.194 (6)
d	4.03 (3)	4.16 (3)	.123 (3)	.119 (3)

V. CONCLUSION

Monte Carlo calculations conducted with [illegible] to [illegible] conversion factors [illegible] a very powerful [illegible] the [illegible] of a point [illegible] [illegible]. The [illegible] penetrating radiation such as neutrons and high-energy gamma rays. For isotropic sources of low-energy gamma rays and electrons, the dose value may depend strongly on the "cell" size and shape. We recommend that a reasonable standard size and shape of a tissue equivalent material be established and internationally adopted for beta dose calculations.

VI. REFERENCES

1. NCRP Scientific Committee 4 on Heavy Particles. H. H. Rossi, Chairman, "Protection Against Neutron Radiation." NCRP-38, National Council on Radiation Protection and Measurements (Jan. 1971)
2. ANS-6.1.1 Working Group, M. E. Battat, Chairman, "American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors." ANSI/ANS-6.1.1-1977 (N666), American Nuclear Society, LaGrange Park, ILL (1977)
3. ICRP Committee 3 Task Group, "Data for Use in Protection Against External Radiation", ICRP-51, International Commission on Radiological Protection, Pergamon Press (March 1987)
4. MCNP-A General Monte Carlo Code for Neutron and Photon Transport, LA-7396-M UC-32, Los Alamos National Laboratory, Los Alamos, NM, 1986
5. Halbleib, J. "Structure and Operation of the ITS Code System." and "Applications of the ITS Codes." Chapters 10 and 11 of "Monte Carlo Transport of Electrons and Photons" ed by Jenkins, T. M., Nelson, W. R. and Rindi, A. Plenum Press, 1988
6. Berger, M. J. and Seltzer, S. M. "The ETRAN System." Chapters 7, 8, and 9 of "Monte Carlo Transport of Electrons and Photons" ed by Jenkins, T. M., Nelson, W. R. and Rindi, A. Plenum Press, 1988
7. Chabot, G. E., Skrable, K. W. and French, C. S., "When Hot Particles Are Not on the Skin." Rad. Prot. Manag. vol 5 pp. 31-42, 1988