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RADIOISOTOPE SHIELDING DESIGN MANUAL

Prepared by
William H. Steigelmann

July 1963

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Division of Isotopes Development
United States Atomic Energy Commission

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July, 1963

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I. INTRODUCTION

One of the most frequently occurring tasks in radiation shielding work is to determine the required thickness of a given material that is necessary to reduce the radiation dose rate at a given location to a specified value, when the radiation source is a radioisotope that emits x- and gamma-radiations. A related problem is to estimate the radiation dose rate at a given location due to the radiations from a radioisotope source. In this case there may or may not be a shield structure interposed between the radiation source and the dose point. The purpose of this "Radioisotope Shielding Design Manual" is to present a relatively simple and straightforward method for performing these tasks when the radiation source is one of the more frequently used radioactive isotopes. Information is presented for water, ordinary concrete, iron, lead, and uranium as the shielding materials, and for the following sources of radiation:

N ¹⁶	Kr ⁸⁸	Sb ¹²⁴	Ba ¹⁴⁰
Na ²⁴	Rb ⁸⁸	I ¹²⁵	La ¹⁴⁰
Cr ⁵¹	Sr ⁹⁰	I ¹³¹	Ce ¹⁴⁴ -Pr ¹⁴⁴
Mn ⁵⁴	Y ⁹⁰	Xe ^{131m}	Pm ¹⁴⁷
Mn ⁵⁶	Zr ⁹⁵	I ¹³³	Tm ¹⁷⁰
Fe ⁵⁹	Nb ^{95m}	Xe ^{133m}	Ir ¹⁹²
Co ⁶⁰	Nb ⁹⁵	Xe ¹³³	Au ¹⁹⁸
Br ⁸⁴	Zr ⁹⁷ -Nb ^{97m}	Cs ¹³⁴	Po ²¹⁰
Kr ⁸⁵	Nb ⁹⁷	I ¹³⁵ -Xe ^{135m}	Pu ²³⁸
Kr ⁸⁷	Ru ¹⁰³ -Rh ^{103m}	Xe ¹³⁵	Am ²⁴¹
		Cs ¹³⁷ -Ba ^{137m}	Cm ²⁴²

Mixed fission products for the conditions of an infinitely long irradiation time and five specific decay times also have been treated:

<u>Designation</u>	<u>Decay Time</u>
MFP-A	1 hour
MFP-B	1 day
MFP-C	10 days
MFP-D	1 month
MFP-E	6 months

Since some of the isotopes listed above are radioactive daughters of other of the isotopes, the daughter-parent combination will occur together in the radiation source. When the half-life of the daughter isotope is short compared to the half-life of the parent, the activity of the daughter will eventually become equal to the activity of the parent, and the daughter is said to be in equilibrium with the parent. In the following cases where this situation may be encountered in radiation shielding work, the shield design information is presented for one curie of the combination of isotopes, as well as for them individually: $\text{Kr}^{88} - \text{Rb}^{88}$, $\text{Sr}^{90} - \text{Y}^{90}$, $\text{Zr}^{95} - \text{Nb}^{95\text{m}}$, $\text{Zr}^{97} - \text{Nb}^{97\text{m}} - \text{Nb}^{97}$, and $\text{Ba}^{140} - \text{La}^{140}$. The radiation dose rate and shield attenuation curves are based upon one curie of the combined activity in each case.

The reason for presenting the information both ways is that the half-life of the daughter is long enough so that equilibrium conditions will not necessarily have been established at the time when the dose rate is to be determined. It should be noted that in the case of the isotope combinations $\text{Zr}^{97} - \text{Nb}^{97\text{m}}$, $\text{Ru}^{103} - \text{Rh}^{103\text{m}}$, $\text{I}^{135} - \text{Xe}^{135\text{m}}$, $\text{Cs}^{137} - \text{Ba}^{137\text{m}}$ and $\text{Ce}^{144} - \text{Pr}^{144}$ that were presented in the first listing above, the half-life of the daughter isotope is so short that equilibrium conditions will be virtually always encountered.

The shielding design information is presented in the form of curves for the case of a point source of radiation. Any other geometrical form for the radiation source can be approximated by an array of point sources. The radiation dose rate resulting from the emission of gamma rays, bremsstrahlung (X-rays), and neutrons is considered.

II. METHOD OF CALCULATION

The calculation of the radiation dose rate from a radioisotope source, and the determination of the required thickness of a given material as radiation shielding to reduce the dose rate to a specified level, is often a complicated procedure because most radioisotopes emit gamma photons with a multiplicity of energies, and with a different number of photons per disintegration for each energy. Since the flux-to-dose rate conversion factor and the attenuating properties of the shielding materials are strongly energy dependent, it is necessary to calculate separately the contribution to the total dose rate of each photon energy level or energy group, and then to sum these dose rate components over all the energies for each isotope. When more than a single radioactive isotope species is present, it is necessary to repeat this process for each isotope, and then to add the contributions from each of the radioisotopes in order to determine the total dose rate. The calculations are also complicated somewhat by the variety of units in which the different quantities are given. The basic procedure that is usually followed when attempting to determine the required shielding necessary to meet some specified dose rate criterion involves assuming a number of material thicknesses, calculating the dose rate for each, plotting the results, and interpolating to find the shield thickness corresponding to the permissible dose rate.

The method described and presented in this report makes use of two types of curves: one which gives the unshielded dose rate per curie of source activity as a function of distance from the source, and the other which presents a dose attenuation factor as a function of shield thickness. The energy dependent factors are therefore essentially "pre-calculated" for the shield designer. The remainder of this section will discuss separately the calculation of the gamma, bremsstrahlung, and neutron components of the dose rate. Appendix A presents a detailed tabulation of the particles and radiations emitted by each of the

isotopes considered. It should be noted that self-absorption or shielding of the radiations by the source material itself has been ignored.

A. Gamma Radiation

The dose rate, D_i , at a distance R feet from a point source containing A_i curies of radioisotope i , with t inches of material m as shielding, is

$$D_i = \frac{3.7 \times 10^{10} A_i}{4\pi [30.48 R]^2} \sum_j F_\gamma(E_j) N_i(E_j) G_m(E_j, b_m(E_j)) \text{ mrem/hr} \quad (1)$$

where

$$G_m(E_j, b_m(E_j)) = B_m(E_j, b_m(E_j)) e^{-b_m(E_j)} \quad (2)$$

$$b_m(E_j) = 2.54 \mu_m(E_j) t \quad (3)$$

and where

3.7×10^{10} is a conversion constant; disintegrations/second/curie

30.48 is a conversion constant; centimeters/foot

E_j is the j^{th} energy level at which gamma photons are emitted from isotope i

$N_i(E_j)$ is the number of gamma photons emitted with energy E_j per disintegration from isotope i

$F_\gamma(E_j)$ is a conversion factor from gamma flux to dose rate and is a function of gamma-ray energy; (millirem/hour)/(gamma rays with energy E_j /square centimeter-second)

$G_m(E_j, b_m(E_j))$ is the dose attenuation factor for the gamma photons with original energy E_j

$b_m(E_j)$ is the thickness of the shield expressed in terms of attenuation lengths

2.54 is a conversion constant; centimeters/inch

$\mu_m(E_j)$ is the total linear attenuation coefficient for material m and is a function of gamma-ray energy; cm^{-1}

$B_m(E_j, b_m(E_j))$ is the dose buildup factor that expresses the ratio of the total dose rate (resulting from both the original, uncollided photons with energy E_j and the photons with lower energies that arise from scattering collisions between the original photons and the electrons in the shield material) to the dose rate resulting from the uncollided photons only

$e^{-b_m(E_j)}$ is the shielding reduction factor for the original gamma photons with energy E_j

The functions F_γ , μ , and B are presented in Appendix B.

Equation (1) can be re-written in the following equivalent form

$$D_i = A_i \hat{D}_i \alpha_i \quad \text{mrem/hr} \quad (4)$$

where

$$\hat{D}_i = \frac{3.17 \times 10^6}{R^2} \sum_j F_\gamma(E_j) N_i(E_j) \quad \text{mrem/hr-curie} \quad (5)$$

$$\alpha_i = \frac{\sum_j F_\gamma(E_j) N_i(E_j) G_m(E_j, b_m(E_j))}{\sum_j F_\gamma(E_j) N_i(E_j)} \quad (6)$$

The function $\hat{D}_i$ is therefore the unshielded dose rate per curie of radioisotope i, as a function of distance R; and α_i is the shield attenuation factor. These two functions are presented in the form of "shielding

design curves" in the following section of this report. The function $\hat{D}_1$ is shown for each isotope as a function of R, while α_1 is shown as a function of thickness t for each of the shielding materials considered.

The α_1 curves presented here can also be used in the case where a shield material has a different density from that used in these calculations (as listed in Appendix B) by reading the curves at thickness $t = [\rho(\text{actual})/\rho(\text{App. B})] t(\text{actual})$. The attenuation resulting from the inherent air shielding was not considered in any of the cases treated, since this attenuation is in most cases negligible for distances less than several hundred feet. It should be noted that it is not always permissible to extrapolate the α_1 curves because of the frequent occurrence of high-energy photons that cause the α_1 curves to have a decreasing slope.

For the mixed-fission products, the gamma-ray emission is expressed in terms of $N_1^*(E_j)$, photons with energy E_j per second per watt of operating power for an essentially infinite operating time in a nuclear fission reactor. In this case, Equations (1), (4), and (5) become, respectively

$$D_1 = \frac{P}{4\pi[30.48R]^2} \sum_j F_\gamma(E_j) N_1^*(E_j) G_m(E_j, b_m(E_j)) \text{ mrem/hr} \quad (7)$$

$$D_1 = P \hat{D}_1^* \alpha_1 \quad (8)$$

$$\hat{D}_1^* = \frac{8.56 \times 10^{-5}}{R^2} \sum_j F_\gamma(E_j) N_1^*(E_j) \text{ mrem/hr-watt} \quad (9)$$

where P is the reactor operating power, in watts. Equation (6) now uses N_1^* rather than N_1 .

B. Bremsstrahlung

Bremsstrahlung is the so-called "braking radiation" that is emitted within a material when it is traversed by a charged particle. Since the range of the charged particles is very short, the bremsstrahlung, which are electromagnetic radiations just as are gamma rays and X-rays, are produced within the source itself. The calculation of the total energy intensity emitted during the deceleration of an electron (beta particle) is a very complex procedure. Evans* presents the following equation as a useful approximation.

$$I = kZE_0^2 \quad (10)$$

where I is the bremsstrahlung intensity, E_0 is the energy of the electron, Z is the atomic number of the stopping material, and k is a constant. Numerical values for the constant k range from 4×10^{-4} to 1.1×10^{-3} , with 7×10^{-4} considered to be the "best" value,* when I and E_0 are expressed in Mev.

The energy spectrum of the bremsstrahlung energy intensity can be approximated as a linearly decreasing function of energy, with the maximum energy equal to E_0 . The number of the photons with energy E_0 is, however, zero. Therefore, the energy spectrum and number spectrum are

$$I(E) dE = 2kZ(E_0 - E) dE \quad (11)$$

$$N(E) dE = 2kZ \left(\frac{E_0}{E} - 1 \right) dE \quad (12)$$

where E is the photon energy.

The problem is complicated by the fact that the beta rays (particles) that are emitted during the radioactive decay of the isotopes

* R. D. Evans, "The Atomic Nucleus", McGraw-Hill Book Co., Inc., New York, 1955.

are not mono-energetic, but are emitted with a spectral distribution that ranges from zero energy up to some energy $E_{\max}$, with the number of beta particles being zero at $E_{\max}$.

The procedure that was followed in the calculation of the bremsstrahlung dose rate was to make first an approximate calculation for each isotope listed in Appendix A. For those isotopes for which the bremsstrahlung dose rate showed the possibility of being significant when compared with the gamma dose rate, the beta spectrum was determined by the method given by Marshall.* Each of the beta spectra were then divided into eight equivalent energy groups, and the gamma spectrum was computed by Equation (12) for each of these energy groups. (The same eight energies were used for both the gamma rays and the beta rays.) Finally, the bremsstrahlung dose rate and shield attenuation curves were determined through the use of Equations (1) through (6), but with $N_i(E_j)$ and E_j being the values computed for the bremsstrahlung.

A value of 26 was used for Z in Equation (12). This value corresponds to the effective Z for strontium titanate, and is also the value for steel. For other values of Z , the $\hat{D}_i$ values can be adjusted by the ratio $Z/26$. It was found that the only isotopes for which the bremsstrahlung dose rate exceeded five percent of the gamma-ray dose rate were Kr^{85} , Kr^{87} , Rb^{88} , Sr^{90} , Y^{90} , Ce^{144} - Pr^{144} , Pm^{147} , and Tm^{170} ; thus these are the only bremsstrahlung dose rates shown on the curves in Section III. The bremsstrahlung photon emission data for these isotopes are presented in Table I. The two highest energy levels have been combined into a single group.

* J. H. Marshall, "How to Figure Shapes of Beta Ray Spectra, Nucleonics", 13:8, p. 34, August 1955.

TABLE I: Bremsstrahlung Emission Data

<u>Radioisotope</u>	<u>Energy, E, Mev</u>	<u>Emission Per Disintegration*</u>	
		<u>$N_{\beta}(E)$</u>	<u>$N_x(E)$</u>
Kr ⁸⁵	0.610	0.062	1.3×10^{-5}
	0.480	0.103	4.2×10^{-5}
	0.390	0.132	1.3×10^{-4}
	0.305	0.138	3.1×10^{-4}
	0.220	0.162	7.3×10^{-4}
	0.130	0.198	1.8×10^{-3}
	0.043	0.205	7.9×10^{-3}
Kr ⁸⁷	3.30	0.049	4.7×10^{-5}
	2.60	0.073	1.7×10^{-4}
	2.15	0.125	5.4×10^{-4}
	1.66	0.149	1.5×10^{-3}
	1.20	0.168	3.6×10^{-3}
	0.71	0.222	9.3×10^{-3}
	0.24	0.214	4.0×10^{-2}
Rb ⁸⁸	4.64	0.080	1.2×10^{-4}
	3.64	0.126	4.0×10^{-4}
	2.98	0.161	8.3×10^{-4}
	2.32	0.188	2.8×10^{-3}
	1.66	0.192	6.5×10^{-3}
	1.00	0.165	1.6×10^{-2}
	0.33	0.088	6.7×10^{-2}
Sr ⁹⁰	0.480	0.054	9.0×10^{-6}
	0.375	0.099	3.0×10^{-5}
	0.305	0.137	9.3×10^{-5}
	0.240	0.138	2.3×10^{-4}
	0.170	0.162	5.5×10^{-4}
	0.102	0.196	1.4×10^{-3}
	0.034	0.214	6.0×10^{-3}

TABLE I (Continued)

Radioisotope	Energy, E, Mev	Emission Per Disintegration*	
		$N_{\beta}(E)$	$N_x(E)$
Y^{90}	2.00	0.077	5.2×10^{-5}
	1.55	0.130	1.8×10^{-4}
	1.27	0.157	5.1×10^{-4}
	0.98	0.162	1.3×10^{-3}
	0.70	0.167	2.9×10^{-3}
	0.42	0.174	7.1×10^{-3}
	0.14	0.133	2.9×10^{-2}
$Ce^{144} - Pr^{144}$	2.75	0.079	6.3×10^{-5}
	2.16	0.142	2.2×10^{-4}
	1.80	0.158	8.2×10^{-4}
	1.38	0.163	1.5×10^{-3}
	0.98	0.165	3.9×10^{-3}
	0.59	0.166	9.4×10^{-3}
	0.20	1.127	4.3×10^{-2}
Pm^{147}	0.21	0.045	3.1×10^{-6}
	0.16	0.108	1.2×10^{-5}
	0.13	0.142	3.9×10^{-5}
	0.102	0.106	9.8×10^{-5}
	0.073	0.148	2.3×10^{-4}
	0.044	0.208	5.8×10^{-4}
	0.015	0.243	2.5×10^{-3}
Tm^{170}	0.84	0.038	8.2×10^{-6}
	0.66	0.089	3.9×10^{-5}
	0.54	0.132	1.3×10^{-4}
	0.42	0.149	3.6×10^{-4}
	0.30	0.169	8.7×10^{-3}
	0.18	0.198	2.3×10^{-3}
	0.060	0.225	1.0×10^{-2}

* N_{β} = Number of beta particles in each energy group
 N_x = Number of bremsstrahlung photons in each group.

C. Neutrons

The only isotopes for which neutron emission is treated here are Pu²³⁸ and Cm²⁴², both of which undergo spontaneous fission. It must be noted, however, that these isotopes and others such as Po²¹⁰ and Am²⁴¹ also emit alpha particles and these can cause (α,n) reactions with other materials that may be present with the isotope. The (α,n) reaction may result in a much greater neutron flux than that from spontaneous fission. For these reasons, and because of the scarcity of data concerning the neutron emission data for spontaneous fission, the neutron dose rate and shielding calculations have been treated in an approximate manner. Because only spontaneous fission is considered here, the user of these curves should exercise caution to see whether (α,n) reactions need to be considered also.

Corresponding to Equation (5), the unshielded neutron dose rate per unit activity, $\hat{D}_i$ was calculated from the equation:

$$\hat{D}_i = \frac{3.17 \times 10^6}{R^2} \bar{F}_n v_i \text{ mrem/hr-curie} \quad (13)$$

Where $\bar{F}_n$ is an average neutron flux: dose rate conversion factor and v_i is the average number of neutrons released per disintegration. Data on v_i are presented in Appendix A. Referring to Figure B-1 in Appendix B, it can be seen the value of $\bar{F}_n$ is approximately constant for neutron energies greater than about one Mev. Since most of the neutrons released during spontaneous fission probably have energies in excess of one Mev, a value of 0.13 (mrem/hr)/(n/cm²-sec) will be used for $\bar{F}_n$.

With regard to the dose attenuation curves for the two neutron sources, approximate shielding information will be presented, as discussed in Glasstone and Sesonske* and Blizard.⁺ In all cases, the shielding

*S.Glasstone and A. Sesonske, Nuclear Reactor Engineering, D. Van Nostrand Co., Inc., New York, 1963

⁺E.P. Blizard (Ed.), Reactor Handbook, 2nd Ed; Volume III, Part B, Shielding, Interscience Publishers, New York, 1962.

curves will be based on data for the fission neutron spectrum of U-235. For water, the expression for the point kernel given in Eg. 10.32 of Glasstone and Sesonske will be used as the basis for the α_1 curves. For concrete, a relaxation length of 12 cm (as reported in both references) will be used. For steel and lead, relaxation lengths of 16 cm and 24 cm, respectively, will be assumed (from the Blizard reference). For uranium, it will be assumed that the attenuation is negligible, since (a) the relaxation length is probably large, and (b) additional neutrons will be produced within the shield as a result of the neutron-induced fast-fission reaction.

The appropriate expressions for α_1 are therefore

$$\text{For water:} \quad \alpha_1 = 0.892 e^{-0.328t} + 0.108 e^{-0.231t} \quad (14a)$$

$$\text{For concrete:} \quad \alpha_1 = e^{-0.212t} \quad (14b)$$

$$\text{For steel:} \quad \alpha_1 = e^{-0.159t} \quad (14c)$$

$$\text{For lead:} \quad \alpha_1 = e^{-0.106t} \quad (14d)$$

$$\text{For uranium:} \quad \alpha_1 = 1 \quad (14e)$$

where t is expressed in inches.

III. DOSE RATE AND SHIELDING CURVES

The radiation dose rate curves and shield attenuation curves are presented in sixteen sets of figures, each set presenting information on one group of radioisotopes. In each set, the first figure, A, shows the shielded dose rate, $\hat{D}_1$ or $\hat{D}_1^*$, as a function of distance from a point source of the radioisotope. The dose rate information is presented separately for gamma rays, bremsstrahlung X-rays, and neutrons. (Where only gamma radiation is considered, the curves are not designated with regard to the type of radiation.) In all cases, it is assumed that the charged particle radiations (α -rays and β -rays) are completely stopped within the source and its container or cladding (the dose rate from these radiations is not considered in this report). The remaining figures in each set (B,C,D,E, and F) present the shield attenuation factors, α_1 , for the cases where water, ordinary concrete, steel, lead, and uranium are the shield materials. The radioisotopes are grouped, and the groups are presented in order of increasing mass number. Some deviations from the mass number sequence has been necessary, however, in the interest of minimizing the confusion as to which curve corresponds to which isotope. Table II presents an index to the figures and shows where the information on the various isotopes may be found.

In using these curves, it should be remembered that the unshielded dose rate does not include the self-shielding contributed by the source itself (except as regards the charged particles, as discussed above). In some instances, as shown in the shield attenuation factor curves, a very small thickness of material can make a great difference in the dose rate, and therefore the source self-shielding can at times be quite significant.

It may be noted that some of the α_1 curves initially rise to values greater than unity instead of monotonically decreasing. This behavior occurs with low-energy photons in light-weight shielding

materials (such as water) and is the result of the buildup and accumulation of the very low energy photons which are biologically very harmful. (The F_{γ} curve in Figure B-1 increases with decreasing energy for photon energies less than about 60 Kev.) Another reason for the initial rise is that the curves are based upon infinite medium buildup factors. Since these very low energy photons are readily absorbed by a heavier material (such as the container for the water) and since a finite shield will have a smaller buildup factor than an infinite shield because the backscatter will be less, the initial rise in the α_i curves should not be taken at face value. Also, the shape of the curve in the "peaked" region is not well defined. The curves are conservative in that they predict dose rates values that are greater than those expected to occur in an actual situation.

TABLE II: Index to the Shielding Design Curves

RADIOISOTOPE	FIGURE NO.					
	D_1 or D_1^*	α_1 for water	α_1 for concrete	α_1 for steel	α_1 for lead	α_1 for uranium
N ¹⁶	1A	1B	1C	1D	1E	1F
Na ²⁴	2A	2B	2C	2D	2E	2F
Cr ⁵¹	2A	2B	2C	2D	2E	2F
Mn ⁵⁴	1A	1B	1C	1D	1E	1F
Mn ⁵⁶	2A	2B	2C	2D	2E	2F
Fe ⁵⁹	1A	1B	1C	1D	1E	1F
Co ⁶⁰	2A	2B	2C	2D	2E	2F
Br ⁸⁴	4A	4B	4C	4D	4E	4F
Kr ⁸⁵	3A	3B	3C	3D	3E	3F
Kr ⁸⁷	3A	3B	3C	3D	3E	3F
Kr ⁸⁸	4A	4B	4C	4D	4E	4F
Rb ⁸⁸	4A	4B	4C	4D	4E	4F
Kr ⁸⁸ - Br ⁸⁸	4A	4B	4C	4D	4E	4F
Sr ⁹⁰	5A	5B	5C	5D	5E	5F
Y ⁹⁰	5A	5B	5C	5D	5E	5F
Sr ⁹⁰ - Y ⁹⁰	5A	5B	5C	5D	5E	5F
Zr ⁹⁵	6A	6B	6C	6D	6E	6F
Nb ^{95m}	6A	6B	6C	6D	6E	6F
Zr ⁹⁵ - Nb ^{95m}	6A	6B	6C	6D	6E	6F
Nb ⁹⁵	7A	7B	7C	7D	7E	7F
Zr ⁹⁷ - Nb ^{97m}	8A	8B	8C	8D	8E	8F
Nb ⁹⁷	8A	8B	8C	8D	8E	8F
Zr ⁹⁷ - Nb ^{97m} - Nb ⁹⁷	8A	8B	8C	8D	8E	8F

TABLE II: Index to the Shielding Design Curves (Continued)

RADIOISOTOPE	FIGURE NO.					
	D_1 or D_1^*	α_1 for water	α_1 for concrete	α_1 for steel	α_1 for lead	α_1 for uranium
Ru ¹⁰³ - Rh ^{103m}	7A	7B	7C	7D	7E	7F
Sb ¹²⁴	9A	9B	9C	9D	9E	9F
I ¹²⁵	9A	9B	9C	9D	9E	9F
I ¹³¹	9A	9B	9C	9D	9E	9F
Xe ^{131m}	9A	9B	9C	9D	9E	9F
I ¹³³	10A	10B	10C	10D	10E	10F
Xe ^{133m}	10A	10B	10C	10D	10E	10F
Xe ¹³³	10A	10B	10C	10D	10E	10F
Cs ¹³⁴	10A	10B	10C	10D	10E	10F
I ¹³⁵ -Xe ^{135m}	11A	11B	11C	11D	11E	11F
Xe ¹³⁵	11A	11B	11C	11D	11E	11F
Cs ¹³⁷ - Ba ^{137m}	11A	11B	11C	11D	11E	11F
Ba ¹⁴⁰	12A	12B	12C	12D	12E	12F
La ¹⁴⁰	12A	12B	12C	12D	12E	12F
Ba ¹⁴⁰ - La ¹⁴⁰	12A	12B	12C	12D	12E	12F
Ce ¹⁴⁴ - Pr ¹⁴⁴	14A	14B	14C	14D	14E	14F
Pm ¹⁴⁷	13A	13B	13C	13D	13E	13F
Tm ¹⁷⁰	3A	3B	3C	3D	3E	3F
Ir ¹⁹²	15A	15B	15C	15D	15E	15F
Au ¹⁹⁸	14A	14B	14C	14D	14E	14F
Po ²¹⁰	13A	13B	13C	13D	13E	13F
Pu ²³⁸	14A	14B	14C	14D	14E	14F
Am ²⁴¹	15A	15B	15C	15D	15E	15F
Cm ²⁴²	15A	15B	15C	15D	15E	15F
MFP-A	16A	16B	16C	16D	16E	16F
MFP-B	16A	16B	16C	16D	16E	16F
MFP-C	16A	16B	16C	16D	16E	16F
MFP-D	16A	16B	16C	16D	16E	16F
MFP-E	16A	16B	16C	16D	16E	16F

FIGURE 1A

FIGURE 1B

FIGURE 1C

FIGURE 1 D

FIGURE 1 E

FIGURE 1F

FIGURE 2A

FIGURE 2 B

FIGURE 2 C

FIGURE 2D

FIGURE 2 E

FIGURE 2F

FIGURE 3A

FIGURE 3 B

FIGURE 3C

FIGURE 3D

FIGURE 3 E

FIGURE 3F

FIGURE 4A

FIGURE 4 B

FIGURE 4C

FIGURE 4D

FIGURE 4 E

FIGURE 4F

FIGURE 5 A

FIGURE 5 B

FIGURE 5C

FIGURE 5 D

FIGURE 5E

FIGURE 5 F

FIGURE 6A

FIGURE 6B

FIGURE 6C

FIGURE 6D

FIGURE 6 E

FIGURE 6F

FIGURE 7A

FIGURE 7B

FIGURE 7C

FIGURE 7D

FIGURE 7E

FIGURE 7F

FIGURE 8A

FIGURE 8B

FIGURE 8C

FIGURE 8 D

FIGURE 8E

FIGURE 8F

FIGURE 9A

FIGURE 9B

FIGURE 9 C

FIGURE 9 D

FIGURE 9E

FIGURE 9F

FIGURE 10A

FIGURE 10 B

FIGURE 10 C

FIGURE 10 D

FIGURE 10E

FIGURE 10F

FIGURE 11A

FIGURE 11 B

FIGURE 11 C

FIGURE 11 D

FIGURE 11 E

FIGURE 11F

FIGURE 12A

FIGURE 12 B

FIGURE 12 C

FIGURE 12 D

FIGURE 12 E

FIGURE 12F

FIGURE 13A

FIGURE 13 B

FIGURE 13 C

FIGURE 13 D

FIGURE 13 E

FIGURE 13F

FIGURE 14A

FIGURE 14 B

FIGURE 14 C

FIGURE 14 D

FIGURE 14 E

FIGURE 14F

FIGURE 15A

FIGURE 15 B

FIGURE 15 C

FIGURE 15 D

FIGURE 15 E

FIGURE 15F

FIGURE 16A

FIGURE 16 B

FIGURE 16 C

FIGURE 16 D

FIGURE 16 E

FIGURE 16F

IV. USE OF THE CURVES

This section will present some examples of the use of the $\hat{D}_i$ and α_i curves that appeared in the preceding section of this report.

Example 1:

What is the dose rate at six feet from a 500 curie Co^{60} source shielded by nine inches of steel?

Step 1: From Figure 2A, $\hat{D}_i = 400 \text{ mrem/hr-C}$ at $R = 6 \text{ ft.}$

Step 2: The unshielded dose rate that results from a 500 curie Co^{60} source equals $(400)(500) = 2 \times 10^5 \text{ mrem/hr} = 200 \text{ rem/hr.}$

Step 3: From Figure 2D, $\alpha_i = 9 \times 10^{-4}$ for $t = 9 \text{ in.}$

Step 4: Actual dose rate: $D_i = \hat{D}_i \alpha_i = (2 \times 10^5)(9 \times 10^{-4}) = 180 \text{ mrem/hr.}$

Example 2:

How much lead shielding is required around a small vial containing 250 millicuries of I^{131} in order not to exceed a dose rate of 0.75 mrem/hr at a distance of one yard?

Step 1: From Figure 9A, $\hat{D}_i = 300 \text{ mrem/hr-C}$ at $R = 3 \text{ ft.}$

Step 2: The unshielded dose rate from a 250 mc I^{131} source = $(300)(.25) = 75 \text{ mrem/hr.}$

Step 3: The required dose attenuation factor is $\alpha_i = \frac{\text{Allowed Dose Rate}}{\text{Unshielded Dose Rate}} = 0.75/75 = 10^{-2}.$

Step 4: From Figure 9E, $t = 1-1/4 \text{ inches}$ for $\alpha_i = 10^{-2}.$

Example 3:

What is the unshielded dose rate at six inches from a point source containing a mixture 2000 curies of Sr^{90} and 600 curies of Y^{90} ? Assume that the beta particles are stopped in a material with $Z = 35$.

Step 1: From Figure 5A, $\hat{D}_i = 24 \text{ mrem/hr-C}$ for Sr^{90} and $\hat{D}_i = 260 \text{ mrem/hr-C}$ for Y^{90} , at $R = 0.5 \text{ ft}$.

Step 2: Correction for $Z = 35$:

$$\hat{D}_i(\text{corrected}) = 24 \left(\frac{35}{26} \right) = 32.3 \text{ mrem/hr-C for } \text{Sr}^{90}$$

$$\hat{D}_i(\text{corrected}) = 260 \left(\frac{35}{26} \right) = 350 \text{ mrem/hr-C for } \text{Y}^{90}.$$

Step 3: Actual dose rate: $D_i = (32.3)(2000) + (350)(600) = 2.7 \times 10^5 \text{ mrem/hr} = 270 \text{ rem/hr}$.

Example 4:

Depleted uranium is used as a shield for transporting spent fuel elements. What shield thickness is required to limit the dose rate at ten feet to 2.5 mr/hr , when the source is a group of four fuel elements that have been discharged from reactor that had been operating at a power level of 60 Mw? The reactor core contains 800 such fuel elements, and one month has elapsed since the fuel elements were removed from the reactor.

Step 1: From Figure 16A for MFP-D, $\hat{D}_i^* = 5.8 \text{ mrem/hr-watt}$ for $R = 10 \text{ ft}$.

Step 2: The operating power level for the group of four fuel elements is $P = \frac{(4)(60)}{(800)} = 0.3 \text{ Mw}$.

Step 3: The unshielded dose rate is $\hat{D}_i^* P = (5.8)(3 \times 10^5) = 1.74 \times 10^6 \text{ mrem/hr}$.

Step 4: The required dose attenuation factor is

$$\alpha_1 = \frac{2.5}{1.74 \times 10^6} = 1.4 \times 10^{-6}.$$

Step 5: From Figure 16F, the required shield thickness is 5.6 in.
uranium (by extrapolation).

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APPENDIX A

RADIOISOTOPE DATA

The following table presents the half-life and gamma and beta emission data, states the relationships between the various isotopes, indicates the general quality of the data, and cites the reference from which the data were obtained. In addition to the specific references cited, others were used in a general way to confirm the data selected for specific isotopes. It should be noted that for many of the isotopes, the decay scheme and radiation emission are not accurately known, and numerous discrepancies exist between the data from different sources. The numerical values presented herein are believed to be the most nearly correct. In a few cases where a large number of gamma photons are emitted per disintegration, the least significant photons (i.e., low energy or relatively small emission) have been combined with other photons to form a smaller number of energy groups. The neutron emission data are based upon the release of three neutrons per spontaneous fission.

NOMENCLATURE

E	Energy
$E_{\max}$	Maximum or end-point energy
$\bar{E}$	Average energy
$N(E)$	Number of photons or particles emitted with energy E per disintegration
$N^*(E)$	Number of photons or particles emitted with energy E per second per watt of power level (infinite irradiation time).

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RADIOISOTOPE DATA

<u>Radio-isotope</u>	<u>Half-life</u>	<u>Gamma Emission</u>		<u>Beta Emission</u>			<u>Remarks</u>	<u>Quality of Data</u>	<u>Ref.</u>
		<u>E, Mev</u>	<u>N(E)</u>	<u>E_{max}, Mev</u>	<u>Ē, Mev</u>	<u>N(E)</u>			
N ¹⁶	7.35 sec	8.85	.0007	10.4	-	.22	From neutron activation of water	Good	1,2
		7.11	.051	4.39	-	.73			
		6.13	.73	3.32	-	.05			
		2.75	.0095						
		1.90	.0004						
		1.72	.001						
Na ²⁴	14.97 hr	3.85	.0004	4.17	-	3 x 10 ⁻⁵	From neutron activation of sodium	Very Good	3
		2.75	1.0	1.39	.56	1.0			
		1.37	1.0						
Cr ⁵¹	27.8 days	.325	.10	-	-	0	From neutron activation of stainless steel	Very Good	3
		.005	.20						
Mn ⁵⁴	280 days	.840	1.0	-	-	0	From neutron activation of steel	Good	1,4
Mn ⁵⁶	2.58 hr	2.98	.004	2.86	-	.60	From neutron activation of steel	Good	1,4
		2.65	.018	1.05	-	.25			
		2.11	.14	.75	-	.15			
		1.81	.23						
		.84	.99						
Fe ⁵⁹	45.3 days	1.29	.44	1.56	.62	.003	From neutron activation of steel	Very Good	3
		1.10	.56	.462	.148	.53			
		.336	.003	.271	.080	.46			
		.191	.025						
		.145	.008						

<u>Radio-isotope</u>	<u>Half-Life</u>	<u>Gamma Emission</u>		<u>Beta Emission</u>			<u>Remarks</u>	<u>Quality of Data</u>	<u>Ref.</u>
		<u>E, Mev</u>	<u>N(E)</u>	<u>E_{max}, Mev</u>	<u>Ē, Mev</u>	<u>N(E)</u>			
Co ⁶⁰	5.26 yr	2.16	1.2 x 10 ⁻⁵	1.48	-	.0001	From neutron activation	Very Good	3
		1.33	1.0	.314	.093	1.0			
		1.17	1.0						
Br ⁸⁴	32 min.	3.93	.125	4.68	-	.40	Fission product	Fair (Deduced from decay scheme)	4
		3.15	.07	3.56	-	.09			
		2.50	.10	2.53	-	.16			
		2.11	.04	1.72	-	.35			
		1.85	.23						
		1.10	.14						
		.88	.59						
		.60	.145						
Kr ⁸⁵	10.4 yr	.52	.007	.695	.232	.993	Fission product	Good	1,4
				.150	-	.007			
Kr ⁸⁷	78 min	2.57	.14	3.8	-	.70	Fission product	Fair (Deduced from decay scheme)	4
		.85	.16	3.3	-	.05			
		.403	.84	1.27	1.01	.25			
Kr ⁸⁸	2.8 hr	2.40	.35	2.8	-	.20	Fission product (Parent of Rb ⁸⁸)	Fair (Deduced from decay scheme)	1,4
		2.19	.19	.90	-	.12			
		1.55	.14	.52	.331	.68			
		.85	.23						
		.36	.05						
		.191	.35						
		.166	.07						

Radio isotope	Half-life	Gamma Emission		Beta Emission			Remarks	Quality of Data	Ref.
		E, Mev	N(E)	E _{max} , Mev	Ē, Mev	N(E)			
Rb ⁸⁸	17.8 min	4.7	.008	5.3	-	.78	Fission product (Daughter of Kr ⁸⁸)	Fair (Deduced from decay scheme)	4
		3.52	.005	3.6	-	.13			
		3.24	.006	2.5	1.61	.09			
		3.01	.006						
		2.68	.05						
		2.11	.02						
		1.85	.45						
		.91	.30						
Sr ⁹⁰	28 yr	0	0	.544	.194	1.0	Fission product (Parent of Y ⁹⁰)	Very Good	3
Y ⁹⁰	64 hr	0	0	2.25	.93	1.0	Fission product (Daughter of Sr ⁹⁰)	Very Good	3
				1.75	1.75	.0002			
				.48	-	.0002			
Zr ⁹⁵	65 days	.757	.54	.89	-	.03	Fission product (Parent of Nb ^{95m} (3%) and Nb ⁹⁵ (97%))	Good	4
		.724	.43	.396	-	.43			
				.364	.127	.54			
Nb ^{95m}	90 hr	.231	1.0	-	-	0	Fission product (Daughter of Zr ⁹⁵ and parent of Nb ⁹⁵ (100%))	Good	1,4
Nb ⁹⁵	35 days	.768	.99	.93	-	.01	Fission product (Daughter of Zr ⁹⁵ and Nb ^{95m})	Good	1,4
		.204	.01	.158	-	.99			

Radio-isotope	Half-life	Gamma Emission		Beta Emission			Remarks	Quality of Data	Ref.
		E, Mev	N(E)	E _{max} , Mev	$\bar{E}$, Mev	N(E)			
Zr ⁹⁷ - Nb ^{97m}	17.0 hr -1.0 min.	2.20 1.62 1.35 .75	.01 .02 .04 .96	1.91 .94 .89	.64 - -	.96 .025 .015	Fission products (Parents of Nb ⁹⁷)	Fair	4,5
Nb ⁹⁷	72 min	1.02 .665	.01 .99	1.27 .92	- -	.99 .01	Fission product (Daughter of Zr ⁹⁷ -Nb ^{97m})	Good	1,4,5
Ru ¹⁰³ - Rh ^{103m}	40 days -57 min	.61 .56 .495 .44 .055 .040	.065 .005 .90 .005 .005 .975	.715 .221 .110	- .08 -	.03 .90 .07	Fission products	Good	4,5
Sb ¹²⁴	60 days	2.09 1.69 1.36 .98 .716 .646 .603	.067 .48 .067 .043 .14 .06 .98	2.39 1.68 1.04 .63 .28	- - - -	.22 .06 .04 .56 .12	Fission product	Fair (Deduced from decay scheme)	1,4,5
I ¹²⁵	60 days	.0355	1.0	-	-	0	Fission product	Very Good	1,4,5
I ¹³¹	8.07 days	.724 .638 .365 .284 .080 .030 .005	.03 .09 .79 .06 .02 .05 .006	.812 .608 .335 .250	.272 .193 .098 .071	.007 .872 .093 .028	Fission product (Parent of Xe ^{131m} (0.7%))	Very Good	3

Radio-isotope	Half-life	Gamma Emission		Beta Emission			Remarks	Quality of Data	Ref.
		E, Mev	N(E)	E _{max.} Mev	E, Mev	N(E)			
Xe ^{131m}	12 days	.164	1.0	-	-	0	Fission product (Daughter of I ¹³¹)	Very Good	3
I ¹³³	21 hr	1.4	.01	1.6	-	.01	Fission product (Parent of Xe ^{133m} (1%) & Xe ¹³³ (99%))	Very Good	5
		.85	.05	1.3	-	.93			
		.53	.99	.4	-	.06			
Xe ^{133m}	2.3 days	.233	1.0	-	-	0	Fission product (Daughter of I ¹³³ and parent of Xe ¹³³ (100%))	Very Good	1,4,5
Xe ¹³³	5.27 days	.081	1.0	.35	.115	1.00	Fission product (Daughter of I ¹³³ and Xe ^{133m})	Good	4,5
Cs ¹³⁴	2.1 yr	1.37	.04	.66	-	.75	Fission product	Fair (Conflict between sources; data deduced from decay scheme)	4,5
		1.10	.08	.09	-	.25			
		.798	1.12						
		.662	.02						
		.606	.97						
		.568	.34						
		.204	.08						

Radio-isotope	Half-life	Gamma Emission		Beta Emission			Remarks	Quality of Data	Ref.
		E, Mev	N(E)	E _{max} , Mev	$\bar{E}$, Mev	N(E)			
I ¹³⁵ -	6.7 hr	1.80	.11	1.4	-	.25	Fission product (Parent of Xe ¹³⁵)	Good	1,4
Xe ^{135m}	-15.3 min	1.72	.19	1.0	-	.40			
		1.46	.12	.5	.308	.35			
		1.28	.34						
		1.14	.37						
		1.04	.09						
		.86	.11						
		.53	.36						
Xe ¹³⁵	9.2 hr	.61	.03	.91	-	.97	Fission product (Daughter of I ¹³⁵ -Xe ^{135m})	Very Good	1,4,5
		.36	.001	.55	.302	.03			
		.25	.97						
Cs ¹³⁷ -	30 yr -	.662	.92	1.17	-	.08	Fission product	Very Good	1,4,5
Ba ^{137m}	2.6 min			.52	.192	.92			
Ba ¹⁴⁰	12.8 days	.54	.25	1.02	-	.60	Fission product (Parent of La ¹⁴⁰)	Fair (Deduced from decay scheme)	1,4,5
		.40	.15	.48	.268	.40			
		.03	.75						
La ¹⁴⁰	40.2 hr	2.6	.05	2.20	-	.08	Fission product (Daughter of Ba ¹⁴⁰)	Fair (Very complex; data deduced from decay scheme)	1,4
		1.62	.88	1.62	-	.14			
		.875	.40	1.36	.50	.30			
		.490	.44	1.15	-	.20			
		.438	.05	.86	-	.12			
		.328	.35	.42	-	.16			

<u>Radio-isotope</u>	<u>Half-life</u>	<u>Gamma Emission</u>		<u>Beta Emission</u>			<u>Remarks</u>	<u>Quality of Data</u>	<u>Ref.</u>
		<u>E, Mev</u>	<u>N(E)</u>	<u>E_{max}, Mev</u>	<u>Ē, Mev</u>	<u>N(E)</u>			
Ce ¹⁴⁴	285	2.19	.01	3.15	-	.95	Fission product	Fair	5
Pr ¹⁴⁴	days-	1.48	.01	2.45	-	.03			
	17.3								
	min	.695	.04	.90	-	.02			
		.134	.17	.309	-	.76			
		.081	.07	.175	-	.24			
		.054	.07						
Pm ¹⁴⁷	2.6	.121	.0001	.223	.074	1.00	Fission product	Good	1,4,5
	yr								
Tm ¹⁷⁰	129	.084	.22	.96	-	.78	From neutron activation	Good	1,4
	days			.88	-	.22			
Ir ¹⁹²	74	.885	.02	.67	-	.48	From neutron activation of iridium	Fair (Deduced from decay scheme)	4,5
	days	.61	.19	.54	-	.41			
		.59	.04	.26	-	.07			
		.468	.54						
		.317	.88						
		.308	.30						
		.295	.29						
		.15	.10						
Au ¹⁹⁸	2.7	1.089	.0016	1.37	.475	.00025	From neutron activation of gold	Very Good	3
	days	.676	.0082	.959	.314	.99			
		.412	.95	.282	.079	.0098			
		.07	.027						
		.009	.013						

<u>Radio-isotope</u>	<u>Half-life</u>	<u>Gamma Emission</u>		<u>Beta Emission</u>			<u>Remarks</u>	<u>Quality of Data</u>	<u>Ref.</u>
		<u>E, Mev</u>	<u>N(E)</u>	<u>E_{max}, Mev</u>	<u>Ē, Mev</u>	<u>N(E)</u>			
Po ²¹⁰	138.4 days	.803	1.2x10 ⁻⁵	-	-	0	Also emits 5.3 Mev α	Good	1,2,5
Am ²⁴¹	458 yr.	.128	5x10 ⁻⁵	-	-	0	Also emits 5.48 Mev (85%), 5.44 Mev (13%), and 5.3 Mev (2%) alphas.	Good	4,5
		.103	1.9x10 ⁻⁴						
		.099	2.3x10 ⁻⁴						
		.0596	.359						
		.0264	.025						
Pu ²³⁸	86 yr.	.800	10 ⁻⁸	-	-	0	Also emits 5.49 Mev (72%) and 5.45 Mev (28%) alphas and ~ 10 ⁻⁹ n/dis by spontaneous fission.	Good	4,5
		.203	7x10 ⁻⁸						
		.153	5x10 ⁻⁵						
		.0996	.00135						
		.0435	.28						
Cm ²⁴²	163 days	1.00	4x10 ⁻⁸	-	-	0	Also emits 6.11 Mev (74%) and 6.07 Mev (26%) alphas and ~ 10 ⁻⁶ n/dis by spontaneous fission.	Good	4,5
		.890	3x10 ⁻⁷						
		.605	1.1x10 ⁻⁶						
		.211	3x10 ⁻⁷						
		.158	6.1x10 ⁻⁵						
		.102	3.5x10 ⁻⁴						
		.044	.27						

<u>Radio- Isotope</u>	<u>Half- life</u>	<u>Gamma Emission</u>		<u>Beta Emission</u>			<u>Remarks</u>	<u>Quality of Data</u>	<u>Ref.</u>
		<u>E.Mev</u>	<u>N*(E)</u>	<u>E_{max.} Mev</u>	<u>E.Mev</u>	<u>N*(E)</u>			
MFP-A	-	2.8	1.21×10^9				Mixed Fission Products at t = 1 hour	Good	6
		2.5	1.56×10^8						
		2.2	8.64×10^8						
		1.7	3.82×10^9						
		1.3	1.31×10^9						
		.8	1.50×10^{10}						
		.4	4.0×10^9						
MFP-B	-	2.8	2.14×10^6				Mixed Fission Products at t = 1 day	Good	6
		2.5	1.07×10^8						
		2.2	3.57×10^7						
		1.7	1.75×10^9						
		1.3	9.0×10^7						
		.8	9.38×10^9						
		.4	3.22×10^9						
MFP-C	-	2.8	~ 0				Mixed Fission Products at t = 10 days	Good	6
		2.5	6.8×10^7						
		2.2	1.05×10^7						
		1.7	1.3×10^9						
		1.3	~ 0						
		.8	5.19×10^9						
		.4	1.2×10^9						

<u>Radio-isotope</u>	<u>Half-life</u>	<u>Gamma Emission</u>		<u>Beta Emission</u>		<u>Remarks</u>	<u>Quality of Data</u>	<u>Ref.</u>
		<u>E.Mev</u>	<u>N*(E)</u>	<u>E_{max.} Mev</u>	<u>Ē.Mev</u> <u>N*(E)</u>			
MFP-D	-	2.8	~ 0			Mixed fission products at t = 1 month	Good	6
		2.5	2.52x10 ⁷					
		2.2	7.36x10 ⁶					
		1.7	4.03x10 ⁸					
		1.3	~ 0					
		.8	3.38x10 ⁹					
		.4	4.85x10 ⁸					
MFP-E	-	2.8	~ 0			Mixed fission products at t = 6 months	Good	6
		2.5	5.6x10 ⁵					
		2.2	4.91x10 ⁶					
		1.7	2.0x10 ⁶					
		1.3	~ 0					
		.8	1.28x10 ⁹					
		.4	1.12x10 ⁷					

APPENDIX B

FLUX: DOSE-RATE CONVERSION FACTOR AND SHIELDING PROPERTIES OF MATERIALS

The number-flux: dose-rate conversion factor for gamma rays and for neutrons were calculated from the data in Tables 2-1 and 2-6 of Goldstein¹, and are presented here in Figure B-1.

The densities of the various shielding materials and the concrete composition used in the calculations for the curves in this manual are tabulated in Table B-1. The gamma-ray linear attenuation coefficients for the five materials appear as a function of energy in Figure B-2. These data were mainly also taken from Goldstein, but Walker and Grotenhuis² was used to supply the data for the concrete. The data shown here for steel are those for iron in Ref. 1; there being no difference between the two insofar as gamma radiation shielding is concerned.

Gamma-ray dose buildup factors for the shielding materials considered in this manual appear in Figures B-3 through B-7. These five sets of curves are all for the case of a "point source" of radiation in an infinite shielding medium. The symbol b represents the shielding thickness expressed in attenuation lengths, $b = \mu t$, where μ is the linear attenuation coefficient (from Fig. B-2) and t is the shield thickness. Data for concrete were again obtained from Ref. 2, while for the other materials Ref. 1 was again the source.

REFERENCES

1. H. Goldstein, Fundamental Aspects of Reactor Shielding, Addison-Wesley Publishing Co., Inc., Reading, Mass., 1959.
2. R. Walker and M. Grotenhuis, A Summary of Shielding Constants for Concrete, USAEC Report ANL-6443, Argonne National Laboratory, November 1961.

TABLE B-1: Shielding Materials Data

Densities:

<u>Material</u>	<u>Density</u>	
	<u>lb/ft³</u>	<u>g/cm³</u>
Water	62.4	1.0
Ordinary Concrete	144	2.3
Steel	490	7.85
Lead	708	11.35
Uranium	1170	18.75

Composition of Ordinary Concrete²:

<u>Element</u>	<u>Weight Fraction</u>
H	0.010
C	0.0010
O	0.530
Na	0.0160
Mg	0.0022
Al	0.0339
Si	0.337
K	0.0130
Ca	0.0435
Fe	0.0139

FIG B-1 GAMMA AND NEUTRON RADIATION FLUX DOSE RATE CONVERSION FACTORS VERSUS ENERGY

FIG. B-2 LINEAR ATTENUATION COEFFICIENT VERSUS ENERGY

FIG. B-3. DOSE BUILDUP FACTOR FOR WATER

FIG. B-4. DOSE BUILDUP FACTOR FOR ORDINARY CONCRETE

FIG B-5. DOSE BUILDUP FACTOR FOR IRON

FIG. B-6a. DOSE BUILDUP FACTOR FOR LEAD

FIG B-6b. DOSE BUILDUP FACTOR FOR LEAD

FIG. B-7a. DOSE BUILDUP FACTOR FOR URANIUM

FIG. B-7d. DOSE BUILDUP FACTOR FOR URANIUM