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RELATIVE ATTENUATION CHARACTERISTICS OF SOME
SHIELDING MATERIALS FOR PuB NEUTRONS

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RELATIVE ATTENUATION CHARACTERISTICS OF SOME
SHIELDING MATERIALS FOR PuB NEUTRONS

P. S. Bringham and J. B. McCaslin

Lawrence Berkeley Laboratory
University of California
Berkeley, California 94720

In the course of determining materials for shielding against 2.2 MeV neutrons from the d-d reactions of neutral beam injectors, the following items must be taken into consideration:

1. Attenuation length of neutrons in the shielding material
2. Flammability
3. Cost
4. Structural properties.

This paper deals briefly with relative attenuation measurements of up to 12 inches of shielding material.

Paraffin, although it would be the material of first choice because of its excellent shielding properties, is not considered here because in large quantities it constitutes a substantial fire hazard.

The following shielding materials were tested in the array shown in Figure 1:

1. Polyethylene (0.937 g/cc)
2. Water (1.00 g/cc)
3. Spodumene-gypsum (1.8 g/cc)
4. Gypsum, wet and dry (1.63 and 1.35 g/cc)
5. Concrete (2.31 g/cc).

Table 1 lists the constituents of the spodumene-gypsum, and gypsum shields. These materials were wrapped to prevent loss of water.

DETECTORS

A moderated BF_3 detector was used to measure the neutron flux density, and a polyethylene-lined proton recoil gas proportional counter was used to measure the energy flux density. From these two detectors the average neutron energy can be inferred. Reference 1 gives a detailed description of these two detectors. A Victoreen Model 440 ion chamber was used to measure the gamma component. All three detectors were alternately placed in a standard fixed position for the measurements, and the shielding material to be tested was positioned between the source and the detectors.

RESULTS

Neutron flux density through the shielding materials is shown in Figure 2, and the corresponding average neutron energy is shown in Figure 3. Flux to dose-equivalent conversion factors are those taken from ICRP 21²⁾ and have been fitted to an analytical expression of the form

$$g(E) = k(E)^{-x}$$

as suggested by Rindi³⁾ (see Appendix I). From this, a normalized plot of the neutron dose-equivalent rate was produced and is shown in Figure 4.

To convert these relative values to absolute values, multiply the ordinate by 381.

A PuB source of 7.0×10^6 n/s was used, the average energy of which is about 2.4 MeV. For comparative purposes the results of calculations by Allen and Futterer⁴⁾ which appear in Figure 5 and 6 show much larger dose attenuation factors for 2 MeV neutrons. Presumably, because of the high-energy tail from PuB neutron sources which extends to several MeV, our results are more comparable to the calculations for 5 MeV neutrons which are shown in Figure 6.

Figure 7 shows the results of the gamma dose rate measurements, and Figure 8 shows neutron to gamma dose-equivalent ratios.

TABLE I

SPODUMENE-GYPSUM SHIELD

- 30% Spodumene by weight
- 40% Gypsum by weight
- 30% Water by weight

GYPSUM SHIELD

- 70% Gypsum by weight
- 30% Water by weight

CONCRETE SHIELD

- 16.7% Portland Cement by volume
- 33.3% Sand by volume
- 50% 1/2 to 3/4-inch Rock by volume
- Water to produce a slump of 4 to 7 inches

Spodumene-Gypsum and "wet" gypsum shields were allowed to cure approximately 48 hours before wrapping to prevent water loss. The "dry" gypsum shield was oven baked to remove the water in liquid form leaving only chemically combined water.

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(Academic Press, New York and London, 1973).
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Publication 21 (Pergamon Press, Oxford, 1973).
- 3) Rindi, A., "An Analytical Expression for the Neutron Flux-to-Dose
Conversion Factors," Health Physics 27 (1974).
- 4) Allen, F. J. and Futterer, A. T., "Neutron Transmission Data,"
Nucleonics 21, 8, 120 (1963); or, see National Council on
Radiation Protection and Measurements, Report No. 38 (1971).

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APPENDIX I

Health Physics Pergamon Press 1974. Vol. 27 (September), pp. 322-323. Printed in Northern Ireland

An Analytical Expression for the Neutron Flux-to-Dose Conversion Factors

(Received 22 February 1974; accepted 4 April 1974)

Dear Sir:

THE ICRP (1969) states that "... an alternative approach to the estimation of the dose equivalent by the use of quality factors and an assessment of absorbed dose is to convert the particle fluence incident upon the body directly by the use of conversion factors. . ." The conversion factors for neutrons recommended by ICRP (1973), which lead to the highest value of the dose equivalent, are given in the form of a table or a curve. The estimation of the dose equivalent rates ($\dot{H}$) in mixed neutron fields is often performed through the computation of the following integral, which expresses the sum of the maxima of the dose equivalent inside the body at each energy and then is an upper limit for the maximum dose equivalent inside the body:

$$\dot{H} = \int_{E_1}^{E_2} \phi(E)/g(E)dE \quad (1)$$

where $\phi(E)$ is the flux density and $g(E)$ is the conversion factor at the energy E .

It is often convenient to introduce the $g(E)$ into the computer program in an analytical form rather than as a data matrix. THOMAS (1967) proposed an analytical expression for $g(E)$, based on the values suggested by ICRP (1962).

The ICRP has recently recommended slightly different values of $g(E)$ (ICRP, 1973).

The following updated analytical expression is proposed to fit the present ICRP recommendations.

$$g(E) = k(E)^{-x} \quad (2)$$

where $g(E)$ = flux density to dose rate conversion factor expressed in $n \text{ cm}^{-2} \text{ sec}^{-1}/\text{mrem hr}^{-1}$

E = neutron energy in MeV

k and x are parameters whose values change over different energy ranges and are given in Table 1.

In Table 2 we compare the values of $g(E)$ as tabulated by ICRP (1973) and calculated with the above formula. Deviations of $g(E)$ calculated using equation 2 from the ICRP values are also shown. The errors are within the range +8.3% and -4.2%.

Table 1.

Energy range (MeV)	k	x
$2.5 \times 10^{-8} \leq E \leq 10^{-6}$	117.6	0.0453
$10^{-6} < E \leq 10^{-2}$	316	-0.0262
$10^{-2} < E \leq 1$	8.5	0.75
$1 < E \leq 10^2$	8	0.0774
$10^2 < E \leq 5 \times 10^3$	22.7	0.29
$5 \times 10^3 < E$	42	0.425

Table 2.

Neutron energy (MeV)	$g(E)$ ICRP values	$g(E)$ Calculated	Error
2.5×10^{-8}	260	260	0
1×10^{-7}	240	244	+1.7%
1×10^{-6}	220	220	0
1×10^{-5}	230	233	+1.6%
1×10^{-4}	240	248	+3.4%
1×10^{-3}	270	263	-2.3%
1×10^{-2}	280	280	0
1×10^{-1}	48	47.8	-0.4%
5×10^{-1}	14	14.3	+2.1%
1	8.5	8.5	0
2	7	7.58	+8.3%
5	6.8	7.05	+3.8%
10	6.8	6.69	-1.5%
20	6.5	6.34	-2.4%
50	6.1	5.9	-3.1%
1×10^2	5.6	5.6	0
2×10^2	5.1	4.88	-4.2%
5×10^2	3.6	3.74	+3.9%
1×10^3	2.2	2.23	+1.3%
1×10^3	1.6	1.66	+3.8%
3×10^3	1.4	1.4	0

and are rather randomly spread along the energy interval.

J. McCASLIN (personal communication) has calculated the dose equivalent from a neutron spectrum of the $1/E$ type covering 10 energy decades; he shows that the dose equivalent calculated using the above formula is only 2% higher than that obtained by using ICRP tables.

ALESSANDRO RINDI

Health Physics Department
Lawrence Berkeley Laboratory
Berkeley, California 94720

LETTERS TO THE EDITOR

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Fig. 1

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Fig. 2

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Fig. 3

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Fig. 4

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Fig. 5

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Fig. 6

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Fig. 7

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Fig. 8

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