Title
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RELATIVE ATTENUATION CHARACTERISTICS OF SOME SHIELDING MATERIALS FOR PuB NEUTRONS

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For Reference
Not to be taken from this room
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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 21 and have been fitted to an analytical expression of the form

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

as suggested by Rindi$^3$ (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 \(7.0 \times 10^6\) n/s was used, the average energy of which is about 2.4 MeV. For comparative purposes the results of calculations by Allen and Futterer\(^4\) 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

<table>
<thead>
<tr>
<th>SPODUMENE-GYPSUM SHIELD</th>
</tr>
</thead>
<tbody>
<tr>
<td>30% Spodumene by weight</td>
</tr>
<tr>
<td>40% Gypsum by weight</td>
</tr>
<tr>
<td>30% Water by weight</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>GYPSUM SHIELD</th>
</tr>
</thead>
<tbody>
<tr>
<td>70% Gypsum by weight</td>
</tr>
<tr>
<td>30% Water by weight</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>CONCRETE SHIELD</th>
</tr>
</thead>
<tbody>
<tr>
<td>16.7% Portland Cement by volume</td>
</tr>
<tr>
<td>33.3% Sand by volume</td>
</tr>
<tr>
<td>50% 1/2 to 3/4-inch Rock by volume</td>
</tr>
<tr>
<td>Water to produce a slump of 4 to 7 inches</td>
</tr>
</tbody>
</table>

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.
REFERENCES


Work performed under the auspices of the U. S. Energy Research and Development Administration.
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 inside the body is often convenient to introduce the conversion factor

\[ H = \int E \phi(E)/g(E) dE \]  

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,

\[ g(E) = k(E)^{-a} \]  

where \( g(E) \) is flux density to dose rate conversion factor expressed in \( \text{cm}^{-4} \text{sec}^{-1} \text{mrem}^{-1} \text{hr}^{-1} \)

\( E \) is 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 \(-4.2\%\), 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.

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LETTERS TO THE EDITOR

References


Concrete blocks 7\%x7\%x7\% B

Detector

Shield samples

Floor slab 5\" thick

Drain hole 7\" 10\"

View A-A

Source

Source-shield array

Fig. 1

XBL774-708
Fig. 2
Fig. 3
Fig. 5

1 in of Polyethylene = 1.165 in of Water
= 1.85 in of Concrete
= 2.28 in of Nevada Test Site Soil (100% Saturated)
= 3.2 in of Nevada Test Site Soil (Area 7)

0°, Concrete or NTS
45°, Concrete or NTS
70°, Concrete or NTS
0°, Polyethylene or Water
45°, Polyethylene or Water
70°, Polyethylene or Water

Relative dose

Thicknness of polyethylene (in)
1 in of Polyethylene = 1.21 of Water
= 1.80 in of Concrete
= 2.38 in of Nevada Test Site Soil (100% Saturated)
= 3.25 in of Nevada Test Site Soil (Area 7)

Relative dose

Thickness of polyethylene (in)

0°, Concrete or NTS
45°, Concrete or NTS
70°, Concrete or NTS
0°, Polyethylene or Water
45°, Polyethylene or Water
70°, Polyethylene or Water

Fig. 6
Source-detector distance = 225"

Dose-rate mrem/hr (γ)

Shield thickness (inches)

- Gypsum (wet)
- Concrete
- Water
- Polyethylene
- Gypsum–spodumene

Fig. 7
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