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HIGH ENERGY PHOTON RESPONSE OF MODERATED NEUTRON DETECTORS*

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In recent years there has been a trend toward high-energy accelerators as sources of X-rays and electrons for radiotherapy. Commercial electron linacs are available up to 35 MeV and betatrons up to 45 MeV. For most elements the photoneutron threshold is in the 6-18 MeV energy range so there is an inevitable production of neutrons. The neutrons are relatively unattenuated by the photon collimation and shielding, so they constitute a component of leakage radiation which contributes to the whole-body dose of the patient as well as to the treatment dose. Concern about the effect of the neutron component has been reflected in present and proposed regulations.[1]

In measuring the neutron dose, the experimenter has the problem of detecting neutrons in the presence of a high-energy photon field. It is the purpose of this paper to study the photon response of one type of neutron detector.

A broad class of fast-neutron detectors consists of a hydrogenous moderator to thermalize the fast neutrons and a thermal neutron detector inside of it. The neutron detector may consist of a BF_3 proportional counter, a TLD containing ^{6}Li or ^{10}B, or an element to be made radioactive by neutron capture such as

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gold or indium. Many designs of moderators have been reported in the literature and have usually been attempts to give the instrument a dose-equivalent response ("Rem-Meter") or to give a constant response per n/cm^2 over as broad an energy range as possible. The particular detector we have studied is a commercial moderator* made of a cylinder of low-density polyethylene 6-1/4 inch diameter × 6-1/6 inches long. It is covered with 0.020 inches cadmium and an outer protective cover of 3/16 inch ABS plastic. This moderator is based on a design by Smith.[2]

The thermal neutron detector is an indium foil 2 inches diameter and weighing 2.7 grams. Neutrons are detected by the $^{115}In\ (n,\gamma)\ ^{116}In$ ($T_{1/2}$ = 54 minutes) reaction. The foils were counted by a shielded pancake GM counter. The foils were counted about 1 hour after exposure and half-life measurements indicated no problems with other activation modes (e.g., (γ, n) reactions on either ^{113}In or ^{115}In).

With the above precautions, the only mode of photon detection is by (γ, n) reactions in the materials of the detector assembly. At higher energies, $(\gamma, 2n)$ and (γ, np) reactions may be significant Information on the abundance of elements used in this detector and their (γ, n) threshold energies are listed in Table I. Boron is listed because it is a possible substitute for the cadmium used as a thermal-neutron shield. In the 1st two columns of Table II we list the amount of the various elements in the detector. The ABS plastic cover has been neglected since it is outside the Cd cover and the exact composition is unknown although mostly carbon and nitrogen.

Also tabulated in the third column of Table II are the specific yields for the elements of interest at 25 MeV, in neutrons per mole for a 100 R exposure.

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*Reactor Experiments Inc.

These numbers are not very accurate. The carbon and indium values are taken from Montalbetti et al.[3] It was necessary to extrapolate to 25 MeV since their highest energy was 24 and 23 MeV respectively. Yield data for Cd was found in Price & Kerst[4] at 22 MeV and was the same as In at 22 MeV in Montalbetti so the same value was used at 25 MeV as for In. The deuterium yield data was also taken from Price & Kerst. Since we are so far above the peak of the (γ, n) cross section, no extrapolation was made and their yield at 22 MeV is used. The last column is simply calculated from the first two columns assuming no attenuation in the sample and taking into account the percent abundance of deuterium. It is intended only to give an idea of the relative importance of the various components. The Cd cover which is the largest source of neutrons can be replaced with boron for a thermal-neutron shield. Again using the data of Price & Kerst, we estimated that the neutron yield of an equivalent boron shield would produce 3.2% of the neutrons from the Cd. An equivalent boron shield was fabricated for this experiment. The boron was contained in a rubber material* and the rubber binder itself contributed some photoneutrons. Based on the chemical composition (approximately 6% H, 20% B, 48% C, 8% N and 18% O), we calculated the total photoneutron yield of the boron and its binder would be 11% as much as the Cd shield.

Experimental. It is very difficult to distinguish between the photon and neutron response, since there is no way to produce high-energy photons without at the same time producing neutrons. Initially measurements were made on a medical electron accelerator at 25 MeV. Two approaches were tried and the results compared.

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*Flex/Shield, Reactor Experiments Inc.

In the first approach a comparison between the expected neutron fluence and the measured apparent-neutron fluence was made. The apparent-neutron fluence was measured by exposing an In foil in a moderator one meter from the photon target. We measured an apparent-neutron fluence of 4.85×10^5 n/cm^2 per photon rad at the isocenter. Photon dose is measured at the maximum of the depth-dose curve in water.

The total neutron production of the accelerator was determined by measurement of gold-foil activation using the method of Patterson & Wallace.[5] This gave us a source strength of 6.2×10^{10} n/rad. It was found that there were two sources of neutrons about 75 cm apart. The room was quite small and scattering was so severe that it was impossible to tell just how intense each source was. Reasonable estimates could only tell us that the apparent neutron fluence was of the right order of magnitude.

The second approach used was to try to separately measure the different components of the photon response. Identical exposures were made of In foils in the standard moderator and in a moderator with the Cd shield replaced by a boron shield. The boron-shield measurement gave a response about 14% smaller. The difference between these two results corresponds to an apparent neutron fluence of 5.98×10^4 n/cm^2 per photon rad. Correcting for the photoneutron production of the boron shield increases this to 6.72×10^4 n/cm^2 per photon rad due to the cadmium. A moderator without any shield (either Cd or B) gave meaningless numbers that reflected only the presence of a high intensity fluence of thermal neutrons inside the concrete room.

It is also possible to compare the results of the Cd contribution with Table II. The solid angle of the moderator as seen by the Cd is about $0.44 \times 4\pi$.* The

*The authors wish to thank Prof. Robin Gardner for making the angle calculations for us.

cross-sectional area of the moderator is 198 cm^2 from the ends and 244 cm^2 from the sides. The efficiency for counting neutrons is the same from either direction, however. If we use the average of the two areas then the total neutron production in the Cd is $\frac{6.72 \times 10^4}{.44} \times 221\ cm^2 = 3.4 \times 10^7$ n/rad. We must correct for nonuniform distribution of photons due to the varying distance of parts of the moderator from the target and to attenuation of the high-energy photons (> 6.42 MeV) in the moderator. We convert absorbed dose to exposure using 0.9 rads/R, correct to 100 R, and end up with 5.1×10^9 n/100 R, about 34% more than predicted.

The neutrons produced in the polyethylene moderators were measured by the resulting ^{11}C activation. A bare moderator was exposed in the same way as before. The resulting ^{11}C was measured in two ways; first with a NaI (Tl) scintillation counter and a correction made for the distributed source[6]; second, exposing a plastic scintillator of similar size in an identical manner and determining the activity in it by mounting it on a photomultiplier and measuring integral counting rate. This absolute measurement was then compared with the moderator activity and corrections applied for slightly differing chemical composition and physical size. The results were as follows:

NaI (Tl) method	9.5×10^6 n/rad
Plastic scintillator method	1.11×10^7 n/rad
Average =	1.03×10^7 n/rad

in quite satisfactory agreement. This number can be compared directly to the yield in the last column in Table II if we correct for the nonuniformity of the exposure due to inverse square and attenuation of the high energy (> 18.72 MeV) photons, correct to 100 R, and convert our rad dose to R assuming 0.9 rad/R. The average intensity of the high-energy photons is 92% of what it is at one meter.

After these corrections we have 1.24×10^9 n/100 R. The neutron yield from carbon comes from the reactions on both ^{12}C and ^{13}C, and we have measured only the $^{12}C\ (\gamma, n)\ ^{11}C$. Montalbetti et al. measured the yield of this reaction separately and obtained 7.6×10^6 n/100 R per mole. That portion of the total yield due to this reaction then is 1.43×10^9 n/100 R or 15% higher than our measured value.

The neutrons produced in the polyethylene will not be detected as efficiently as an equal number of neutrons striking the moderator since some of them are produced in the outer part and are initially directed outwards. We estimate the average efficiency to be approximately 80% based on geometry and hydrogen cross section. With this correction and again using the average cross-sectional area of the moderator, the apparent neutron yield from this component would be 1.03×10^7 n/rad $\times$ .8 $\times \frac{1}{221\ \text{cm}^2} = 3.73 \times 10^4$ n/cm^2 per photon rad. This number must be increased to allow for the contribution of the $^{13}C\ (\gamma, n)\ ^{12}C$ reaction which we did not measure. Using data of Montalbetti et al., this increased our value to 4.14×10^4 n/cm^2 per photon rad.

The total apparent neutron yield from the polyethylene and the cadmium would be 1.09×10^5 n/cm^2 per photon rad or 22.5% of our observed response. There is not much other data with which to compare our results. One method would be to measure inside and just outside the beam on the assumption the photon shielding would have little or no effect on the neutron fluence (it would have an effect on the dose since the neutron energy would be decreased). Unfortunately the diffuse nature of the neutron source in this accelerator produces a nonuniform neutron field even over small distances. Lawrence[7] obtained such data measuring outside the field in both directions along the axis of rotation. The ratio of the apparent neutron fluence in the beam to outside the beam was 1.11

in one direction and 1.48 in the other. Our number of 1.21 lies in between these values.

It is possible to calculate the energy dependence of the photon sensitivity. For the energy dependence of the carbon yield we will use Montalbetti et al. No energy dependent yield data for cadmium is known except Price & Kerst measured at two energies. We will use the Montalbetti results for silver (Ag) which fits the Price & Kerst data reasonably. Both energy responses will be normalized to our measured photon response at 25 MeV. The results are shown in Fig. 1.

The data from Montalbetti et al. was taken with an unflattened beam from a betatron. Our 25 MeV measurements were made with a beam flattened by an iron flattening filter. The relatively good agreement with yields predicted by the Montalbetti et al. data is due to the following.

1. At 25 MeV the photon spectrum is relatively unchanged in penetrating an iron filter.
2. The two significant neutron reactions (Cd and C) peak at much different energies and it happened that our errors in these yields partially compensated each other.

We would not expect this situation at other energies. The energy dependence shown in Fig. 1 is what we would expect for unflattened, relatively thin target photon spectra. For comparison, we measured at three energies on a linear accelerator with a tungsten flattener and got the points shown on Fig. 1. We would expect values between the points and the curves for an accelerator with a low or medium atomic number flattener. Calculations based on Monte Carlo generated photon spectra confirm these assumptions but accurate calculations cannot be made without cross section data for cadmium.

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TABLE I

Isotope	Percent Abundance	(γ, n) Threshold (MeV)
^{113}In	4.28	9.34
^{115}In	95.72	9.03
^{106}Cd	1.215	12.19
^{108}Cd	0.875	10.42
^{110}Cd	12.39	9.84
^{111}Cd	12.75	6.97
^{112}Cd	24.07	9.29
^{113}Cd	12.26	6.42
^{114}Cd	28.86	9.05
^{116}Cd	7.58	8.64
^{12}C	98.893	18.72
^{13}C	1.107	4.95
^{10}B	19.61	8.44
^{11}B	80.39	11.45
^{2}H	0.01492	2.23

TABLE II

Element	Weight (grams)	Specific Yield $\left(\frac{\text{neutrons}}{\text{mole-100R}}\right)$	Total Yield $\left(\frac{\text{neutrons}}{\text{100R}}\right)$
C	2292	8.5×10^{6}	1.6×10^{9}
In	2.7	8×10^{8}	1.9×10^{7}
Cd	540	8×10^{8}	3.8×10^{9}
H	383	5×10^{7}	7.1×10^{5}
B*	143	9×10^{4}	1.2×10^{8}

*Note: Possible Boron shield to replace the Cd.

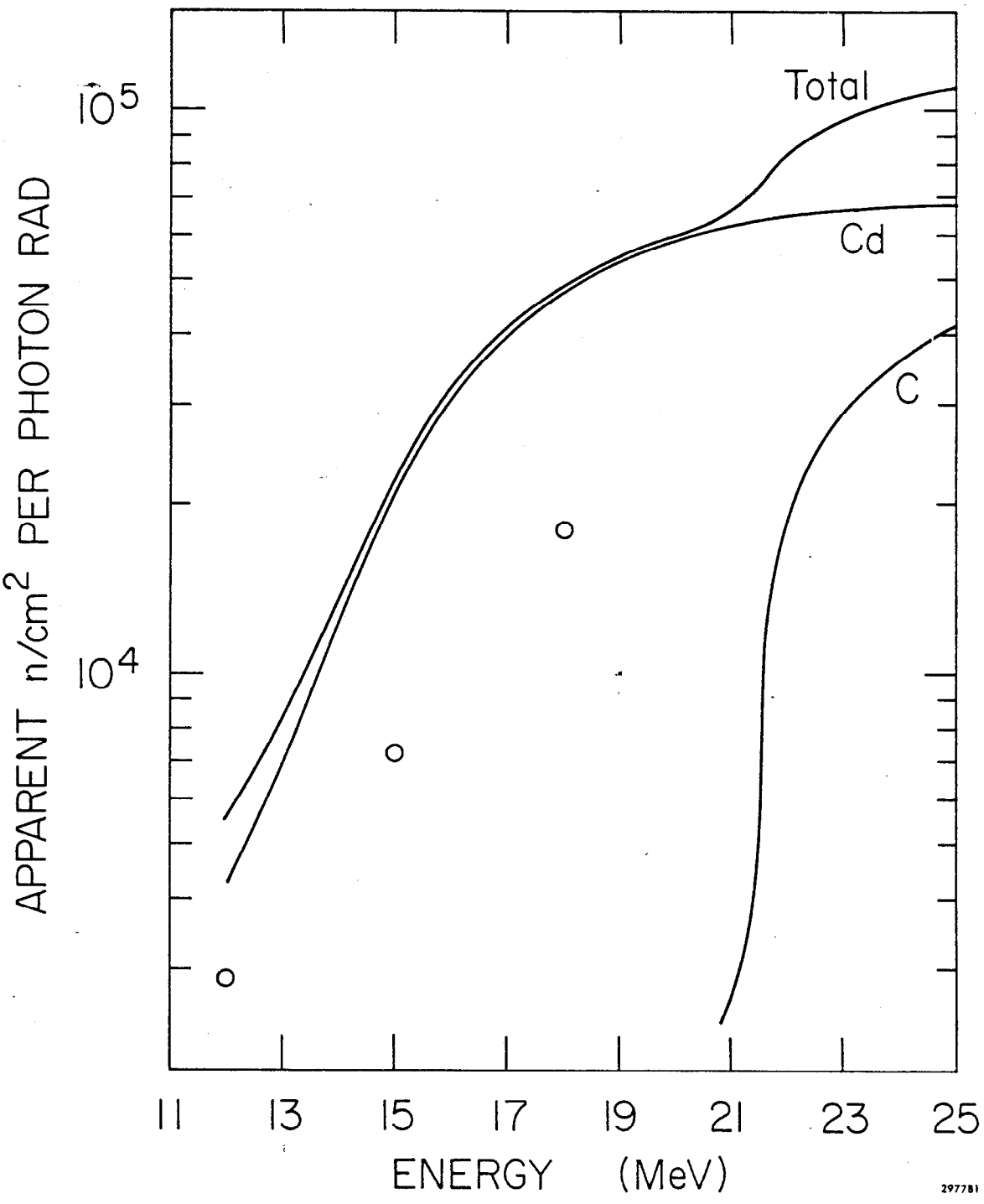
APPARENT n/cm² PER PHOTON RAD
10⁵
10⁴
Total
Cd
C
11
13
15
17
19
21
23
25
ENERGY (MeV)
297781

Fig. 1