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DETERMINATION OF THERMAL NEUTRON MACROSCOPIC CROSS SECTION FOR TWO POLYTHENE BASED SLABS USING MONTE CARLO N-PARTICLE CODE

Igwesi, D. I[1] and Thomas, O. S[2]

1) Dept. of Physics and Industrial Physics, Nnamdi Azikiwe University, Awka, Nigeria.
2) Dept. of Physics, University of Port Harcourt, Port Harcourt, Nigeria.
Corresponding Author e-mail: igwesidavid@yahoo.com

***Abstract*—Thermal neutron macroscopic cross sections and neutron mean free path for pure polythene and borated polythene at different thicknesses were calculated from the result of Monte Carlo N-Particle (MCNP) simulations. The calculations were performed at neutron source-detector distances of 50 cm, 70 cm, and 90 cm by changing the position of the neutron source towards the centre of the water basin. The calculated values showed that the macroscopic cross section and neutron mean free path for pure polythene and borated polythene depend on the thickness of the shielding materials as well as neutron source-detector distance. The increase in the thickness of both shields increases the thermal macroscopic cross section and decreases the mean free path. Also, the thermal microscopic cross section for those shields increased as the neutron source-detector distance increases. The result obtained showed that pure polythene and borated polythene are good enough to be used in neutron shielding and dosimetry calculations.**

Keywords—Macroscopic Cross Section, Polythene, Borated Polythene, Mean Free Path.

1. INTRODUCTION

Neutrons shielding unlike other forms of radiations introduces some complications because of the wide range of energy that must be considered. The fast neutrons produced at high energy are difficult to capture since the probability of the neutron interacting with atomic nuclei is low, therefore for effective shielding, neutrons of high energy have to be slowed down before being captured. The slowing down is achieved by using a material that doesn't become radioactive and of low atomic number to reduce the neutron's energy [1], [2]. Thereafter, appropriate materials are used to shield the thermalized neutron from radiation personnel. The most important factors in determining the effectiveness of any given neutron shielding are the macroscopic cross sections of the shielding material used [3]. Neutron penetration in shielding is characterised by several parameters such as the effective removal cross-sections and the macroscopic thermal neutron cross section [4]. In this work, thermal neutron macroscopic cross sections were calculated for two polythene based shielding materials (pure polythene and borated polythene) in a point-source, point-detector geometry, using the MCNP (version 4c) code. The macroscopic cross sections were calculated at neutron energy of 0.025 eV and for shield thicknesses of 2 cm to 10 cm for pure polythene and 1 cm to 5 cm for borated polythene. The main emphasizes on this work is the dependency of macroscopic cross section both on the thickness of the shielding materials and the neutron source-detector distance.

2. EXPERIMENTAL PROCEDURE

The geometry of this work comprises of an AmBe neutron point isotropic source placed inside a water basin (which serves as a moderator), a shield medium (polythene and borated polythene of different thicknesses) and a He-3 detector as shown in Fig. 1. MCNP version 4C was used to develop the code for this study, with cross-section data from .60c series of ENDF/B-IV library while the material's composition of the polythene, borated polythene and He-3 detector were taken from the DLC-200/MCNPDATA [5]. This code enabled a detailed three dimensional modelling of the actual source and geometry configuration including the shield materials and the detector (Fig. 1). The elemental composition of the materials used in this MCNP neutron shielding modeling is shown in Table 1.

Table 1. Elemental Composition of the Shielding Materials used in this work

Material	Density (g/cm3)	Constituents	MCNP ID	Atomic fraction
		H-1	1001. 60C	0.667954
Polythene	0.92	C-nat	6000. 60C	0.332046
		H-1	1001. 60C	0.625741
Borated Polythene	1.04	C-nat	6000. 60C	0.320296
		B-10	5010. 60C	0.053963

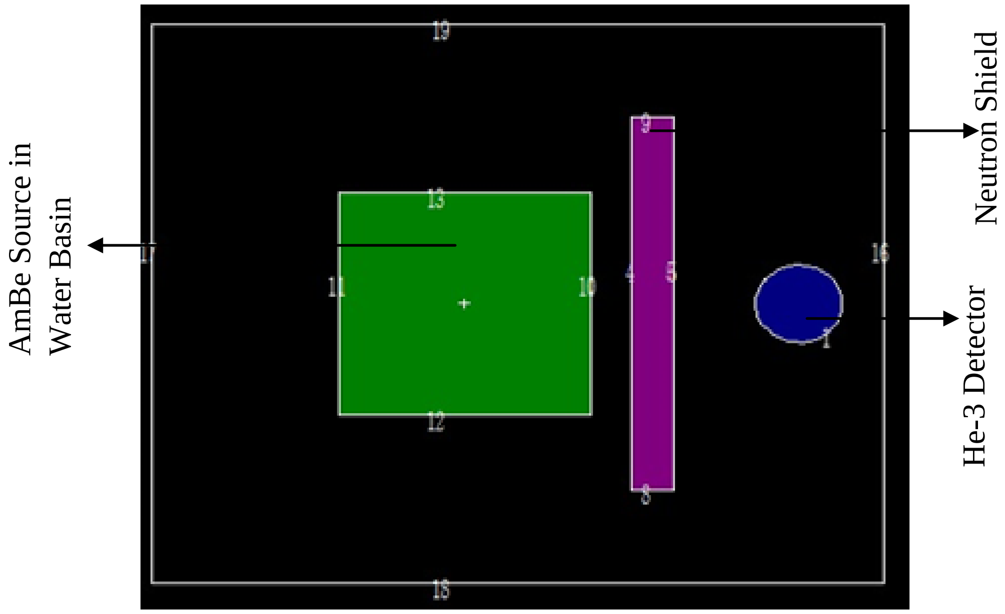

Fig. 1. MCNP output of the experimental set-up

The MCNP code simulates the transport of neutrons produced by the AmBe source through the shielding medium (pure polythene and borated polythene) to the detector. The average dose equivalent per neutron source at the detector point is obtained by converting the neutron fluence from the simulations to dose rate using ANSI/ANS-6.1.1-1977 [6].

The measurement of neutron shielding capabilities lies on its macroscopic cross sections. It is represented by Σ and its unit is cm^{-1}. The macroscopic cross sections is applicable to a thick samples that often contain a mixture of elements. The dose equivalent rate for a beam of neutron

transmitted through a material of thickness, x could be express by the equation:

$$D_x = D_0 e^{-N\sigma_t x} \quad (1)$$

Where D_x is the known values of dose equivalent rates when a material (shield) is placed between source and detector, D_o is dose rate when there is no material between the source and detector, N is the number of nuclei per unit volume and σ_t is the total cross section.

But the number of nuclei per unit volume, N and the total cross section, σ_t are related to the total macroscopic cross section, Σ_t by

$$\Sigma_t = N\sigma_t \quad (2)$$

Thus, the interaction of a neutron with a certain volume of material depend not only on the microscopic cross section of the individual nuclei, rather, it also depends on the number of nuclei within that volume [7]. Σ_t can be regarded as the probability per unit length that a neutron will undergo an interaction as it moves through an absorber and be removed from the beam by either absorption or scattering [8].

And $\Sigma_t = \Sigma_{scatter} + \Sigma_{capture} + \Sigma_{fission} + \ldots\ldots$

We finally have that

$$D_x = D_0 e^{-\Sigma_t x} \quad (3)$$

And solving equation 3, we obtained

$$\Sigma_t = \frac{ln\left(\frac{D_x}{D_0}\right)}{x} \text{ cm}^{-1} \quad (4)$$

The values of Σ_t for each of the two materials for different thicknesses were calculated from the result of MCNP for the neutron source-detector distance of 50 cm. The processes were repeated for neutron source-detector distance of 70 cm and 90 cm and the corresponding values of Σ_t were calculated. From the macroscopic cross section values, the mean free path (which is the mean distance a neutron travels between interactions) is calculated using

$$\lambda = \frac{1}{\Sigma_t} \quad (5)$$

This mean free path could be assumed to be the average thickness of a medium in which an interaction is likely to occur and is similar to mean life of a radioactive atom [9].

3. RESULT AND DISCUSSION

The macroscopic cross sections (Σ_t) and mean free paths calculated from the simulated MCNP results are presented in tables 2 to 4 for pure polythene and borated polythene slabs. For each shield, the macroscopic cross section and the mean free path have been calculated for the neutron source-detector distances of 50 cm, 70 cm and 90 cm. The calculated values indicated that the macroscopic cross section of the shielding materials slightly depends on the thickness and the distance of separation of the neutron source and the detector as shown in Tables 2, 3, and 4. The increase in shield thickness increases the values of Σ_t, which means that the probability of interaction increases as the shielding materials became thicker. Similarly, an increase in the shield thickness decreases the mean free path length of the neutron. It was also observed that as the neutron source-detector distance increases, there is a slight increase on the macroscopic cross section for both shields as shown in Table 2, thus decreasing the neutron mean free path. Therefore, as the neutron source is moved further away from the detector towards the centre of the water basin, the neutron mean free path decreases and the probability of interaction increases [9], [10], [11]. The calculated values of macroscopic cross section for pure polythene are higher than that of the borated polythene for any given thickness of the shield materials. This could be understood because borated polythene slab has neutron absorption material (boron) incorporated in it.

Table 2: Macroscopic cross section for shield materials at Source-Detector distance of 50 cm

Polythene Shield			Borated Polythene Shield		
Thickness (cm)	Σ_t (cm^{-1})	Mean Free Path (cm)	Thickness (cm)	Σ_t (cm^{-1})	Mean Free Path (cm)
2	0.07335	13.6	1	0.06003	16.7
4	0.09225	10.8	2	0.07832	12.8
6	0.10567	9.5	3	0.08699	11.5
8	0.10618	9.4	4	0.09819	10.2
10	0.11079	9.0	5	0.10574	9.5

Table 3: Macroscopic cross section for shield materials at Source-Detector distance of 70 cm

Polythene Shield			Borated Polythene Shield		
Thickness (cm)	Σ_t (cm^{-1})	Mean Free Path (cm)	Thickness (cm)	Σ_t (cm^{-1})	Mean Free Path (cm)
2	0.07914	12.6	1	0.07429	13.5
4	0.10322	9.7	2	0.08250	12.1
6	0.10726	9.3	3	0.09180	10.9
8	0.10892	9.2	4	0.10471	9.6
10	0.11248	8.9	5	0.10676	9.4

Table 4: Macroscopic cross section for shield materials at Source-Detector distance of 90 cm

Polythene Shield			Borated Polythene Shield		
Thickness (cm)	Σ_t (cm^{-1})	Mean Free Path (cm)	Thickness (cm)	Σ_t (cm^{-1})	Mean Free Path (cm)
2	0.08316	12.0	1	0.08520	11.7
4	0.10482	9.5	2	0.08844	11.3
6	0.10763	9.3	3	0.09832	10.2
8	0.11291	8.9	4	0.10768	9.3
10	0.11666	8.6	5	0.10997	9.1

4. CONCLUSION

Pure Polythene and borated polythene are good neutron shielding materials. This work estimated the thermal neutron macroscopic cross sections and mean free path for these shielding materials at different thicknesses from the result of the MCNP simulations. The investigation was performed at neutron source-detector distances of 50 cm, 70 cm, and 90 cm. From the experimental results, neutron macroscopic cross section and mean free path were found to depend on the thickness of the shielding materials and the neutron source-detector distance.

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