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Investigation Of The Effects Of Variation Of Neutron Source-Detector Distance On The Emitted Neutron Dose Equivalent

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***Abstract*—Monte Carlo radiation transport simulations were performed to investigate the effect of variation on the neutron source-detector distances on emitted neutron dose equivalents using pure polythene and borated Polythene as shielding materials. The measurements were taken at three different neutron source-detector distances of 100 cm, 80 cm and 60 cm by varying the position of the neutron source inside the water basin. The experimental set up was modelled using MCNP4 code with a He 3 proportionate counter serving as the detector and different thicknesses of pure polythene and borated polythene as shields. The results from the MCNP simulations were used to calculate the neutron dose equivalents. The calculated results showed that the neutron dose equivalent decreases exponentially as the thickness of both shielding materials increases. The variations observed on the neutron dose equivalents for the three neutron source-detector distances indicated that decreasing the neutron source-detector distance increases the neutron dose equivalent which by implication established that the neutron source-detector distance of 100 cm was obtained as the optimum shielding configuration for any occupational workers among the three distances considered. Furthermore, 6.24 cm of pure polythene and 5.73 cm of 5% borated polythene were obtained as the thickness of the shielding materials that can attenuate 50% of the neutron dose equivalent. Therefore, the results obtained showed that the effectiveness of any shielding material for high neutron source, such as ^{241}Am/Be depends on the neutron source-detector distance and the density of the shielding materials.**

Keywords*—^{241}Am/Be Neutron Source, Polythene, Borated Polythene, Occupational Workers, MCNP4 Code, Half Valued Layer.

1. INTRODUCTION

Nuclear science and technology finds applications in many fields such as scientific research, agriculture, industry and medicine and it offers many advantages but not without some difficulties. Since the radiations involved are ionizing that have damaging effects on human health and environment, it is important therefore to evaluate the risks involved and possibly quantify the level of exposure to such ionizing radiations by radiation workers and subsequently develop technological configuration that guarantees the safety of radiation workers.

Radiation shielding involves placing a shielding material(s) between the sources of ionizing radiations (such as ^{241}Am/Be) and the worker or environment. These ionizing radiations which include alpha particles, beta particles, gamma rays, X-rays, neutrons etc, interact differently with shielding materials. Thus, the efficiency and effectiveness of the shielding varies with the types and energy of the radiation to be shielded as well as the shielding materials.

The best materials for protection against radiation produced by neutron source are mixture of hydrogenous materials (polythene, water and many plastics), heavy elements, and neutron absorbing elements (such as boron, chromium), because they reduce both the intensity of gamma rays and neutrons. Indeed, hydrogenous materials slow down fast and intermediate neutrons energy via inelastic scattering, and they become thermal neutrons which can easily be absorbed by neutron absorbing materials that have a very high neutron absorption cross-section. One material useful for high energy neutron is polythene. Polythene is a good neutron shielding material [1], [2], [3] and more effective when the thickness increases [4]. However, the shielding effect of polythene can be improved by the addition of boron [5], [6], [7], [8], [9], which is a good neutron absorber. Thus, borated polythene materials shield against neutron source better than the pure polythene materials

Unlike other forms of radiations, neutron's shielding introduces some complication because of the wide range of energies needed to be considered. Therefore, fast neutrons are first moderated before being captured. The moderation is achieved by the use of a non-radioactive material of low atomic number, and subsequently, appropriate materials are used to shield the thermalized neutron.

Neutrons are associated with significant health issues as they are highly penetrating and can induce secondary deep body ionizing radiation doses. Large amount of neutron exposure biologically affect cells.

The affected cells can mutate, and in some cases result in cancer. The purpose of this work therefore is to measure the neutron dose emitted by ^{241}Am/Be neutron source positioned at three different points within a water bath. The work specifically involves the simulation of different configurations of the source, shielding slabs and detector using MCNP4 by increasing the thicknesses of polythene and borated polythene slabs. The effective dose equivalent calculated from the MCNP results will be used to estimate the half value layer (HVL) for each of the shielding materials at the three source-detector distance of 100 cm, 80 cm and 60 cm.

2. METHODOLOGY

2.1MCNP Simulations

The MCNP set up (Fig. 1) involves an ^{241}Am/Be neutron source in a water bath of dimension 100 cm by 60 cm, an absorbers (polythene and borated polythene slabs), and a SP9 He-3 proportional counter of dimensions 3.8 cm x 20.8 cm x 25.5 cm (Fig. 1). The atomic percent composition of the water which acts as a moderator is 67% by hydrogen and 33% by oxygen.

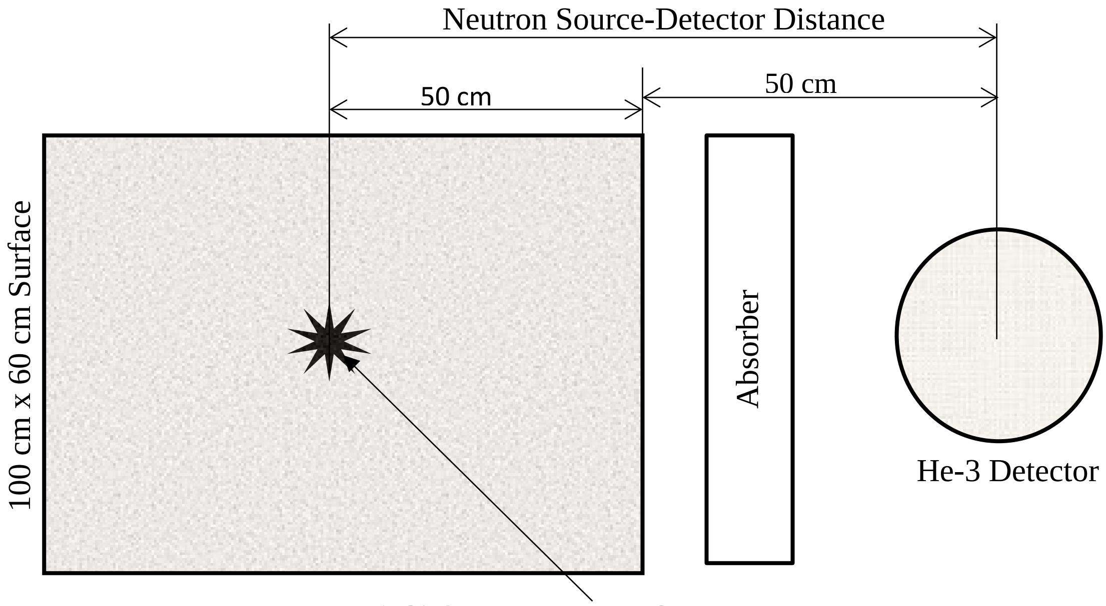

Fig. 1 Geometry set up for the Am-Be neutron source shielding.

MCNP version 4 was used with cross sectional data obtained from .60c series of ENDF/B-IV library while the materials' composition of polythene, borated polythene and He-3 detector were taken from the DLC-200/MCNPDATA [10]. Table 1 shows the elemental composition of the materials used in the work. The ^{241}Am/Be neutron source emit 2.2 x 10^6 neutron per cm^2 per second. The neutron source, absorbers and detector set up was simulated at three different neutron source-detector distances (100 cm, 80 cm, and 60 cm) to ascertain which source-detector distance gives optimum shielding results. In each simulation, neutron histories of 10 x 10^9 were considered and the number of neutron flux incident on the detector were calculated for each of the set up. Conversions from neutron fluxes to doses in rem hr^{-1}/n cm^{-2} s^{-1} were performed using equation 1.

$$H = h\Phi . 1$$

where H = dose equivalent, Φ = the emitted neutron flux, and h = the flux-to-dose-equivalent conversion which varies with neutron energy and was obtained from ANSI/ANS-6.1.1-1977 [11]. The equivalent dose values in rem hr^{-1}/n cm^{-2} s^{-1} were converted to millisievert (mSv) (tables 2, 3 and 4).

Table 1 Elemental Composition of the Shielding Materials used

Material	Density (g/cm^3)	Constituents	MCNP ID	Atomic fraction
Polythene	0.92	H-1 C-nat	1001.60C 6000.60C	0.667954 0.332046
Borated Polythene	1.04	H-1 C-nat B-10	1001.60C 6000.60C 5010.60C	0.625741 0.320296 0.053963

2.2Theoretical Background

The dose equivalent of neutrons, H_x, traversing through a thickness, x, of absorber is proportional to the intensity of the neutron source and the thermal neutron total cross section, Σ_t, of the absorbing material [12]: Thus,

$$-\frac{dH}{dx} = \Sigma_t H \quad .(2)$$

If the equation is integrated, the result shows that the intensity of the uncollided neutrons decreases exponentially with the thickness of the absorber as shown in equation 3.

$$H(x) = H_o e^{-\Sigma_t x} \quad .(3)$$

where H_o is the initial neutron dose equivalent and H(x) refers to those neutrons dose equivalent that penetrate a distance x in an absorber without a collision; thus, the attenuation factor, $e^{-\Sigma_t x}$ represents the probability that a given neutron travels a distance x without an interaction. Therefore, 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 [13].

The half value layer (HVL) which is the thickness of the shielding material required to reduce the dose equivalent to a half of its initial value was estimated. From equation 3, the HVL is given as

$$HVL = \frac{0.693}{\Sigma_t} \quad . (4)$$

where Σ_t is the thermal microscopic cross section of the absorbers. The half value layer was calculated for each neutron source-detector distance. In this work, it was considered that 5 % borated polythene and pure polythene have thermal microscopic cross section of 0.121 cm^{-1} [14] and 0.111 cm^{-1} [12] respectively.

3. RESULTS AND DISCUSSIONS

3.1Neutron Dose Equivalent

The results of the dose equivalents presented in tables 2, 3, and 4 show that the neutron dose equivalents for the three neutron source-detector distances decreases as the thickness of the absorber increases [2], [4].

From the results calculated, the dose equivalent values for pure polythene for a given thickness of absorber show remarkable high values than the dose equivalent values for borated polythene. For instance, for a neutron source – detector distance of 100 cm, a neutron dose equivalent of 17.40 mSv/yr was obtained for a pure polythene of thickness 2 cm while for the same thickness of borated polythene, a neutron dose equivalent of 17.22 mSv/yr was obtained. This results agree favourably with Karmi and Greenspan [5]. However, the difference between them are not very wide, supporting the view of Coeck, et al [15] who held that radiation reduction power of pure polythene and borated polythene is not very pronounce when dealing with small thickness but becomes very noticeable as the thickness of the absorbers increase. The dose equivalents for the three neutron source-detector distances were observed to vary significantly. The least values of dose equivalent were obtained at the neutron source-detector distance of 100 cm which indicate that

Table 2 Neutron Dose equivalents for both shields when Am-Be Source and Detector are 100 cm apart

Polythene			Borated Polythene		
Thickness (cm)	Neutron Dose (mSv/h)	Neutron Dose (mSv/yr)	Thickness (cm)	Neutron Dose (mSv/h)	Neutron Dose (mSv/yr)
0	2.346×10^{-3}	20.55	0	2.346×10^{-3}	20.55
2	1.986×10^{-3}	17.40	1	2.154×10^{-3}	18.87
4	1.542×10^{-3}	13.51	2	1.966×10^{-3}	17.22
6	1.230×10^{-3}	10.77	3	1.747×10^{-3}	15.30
8	9.506×10^{-4}	8.33	4	1.525×10^{-3}	13.36
10	7.305×10^{-4}	6.40	5	1.367×10^{-3}	11.98

Table 3 Neutron Dose equivalents for both shields when Am-Be Source and Detector are 80 cm apart

Polythene			Borated Polythene		
Thickness (cm)	Neutron Dose (mSv/h)	Neutron Dose (mSv/yr)	Thickness (cm)	Neutron Dose (mSv/h)	Neutron Dose (mSv/yr)
0	5.583×10^{-3}	48.91	0	5.583×10^{-3}	48.91
2	4.766×10^{-3}	41.75	1	4.095×10^{-3}	45.61
4	3.695×10^{-3}	32.36	2	3.740×10^{-3}	41.47
6	2.933×10^{-3}	25.70	3	3.349×10^{-3}	37.13
8	2.336×10^{-3}	20.46	4	2.902×10^{-3}	32.17
10	1.813×10^{-3}	15.88	5	2.587×10^{-3}	28.68

Table 4 Neutron Dose equivalents for both shields when Am-Be Source and Detector are 60 cm apart

Polythene			Borated Polythene		
Thickness (cm)	Neutron Dose (mSv/h)	Neutron Dose (mSv/yr)	Thickness (cm)	Neutron Dose (mSv/h)	Neutron Dose (mSv/yr)
0	16.445×10^{-3}	144.06	0	16.445×10^{-3}	144.06
2	14.201×10^{-3}	124.40	1	15.487×10^{-3}	135.67
4	11.371×10^{-3}	99.61	2	14.061×10^{-3}	123.17
6	8.568×10^{-3}	75.06	3	12.668×10^{-3}	110.97
8	6.757×10^{-3}	59.19	4	11.104×10^{-3}	97.27
10	5.218×10^{-3}	45.71	5	9.692×10^{-3}	84.90

From figure 2, it is observed that at neutron source-detector distance of 100 cm, the values of neutron dose equivalents for both pure polythene and borated polythene are less than the International Commission on Radiological Protection (ICRP) recommended dose limit of 20 mS/yr for occupational workers [16], [17]. The results therefore showed that the position of the neutron source inside the water bath has significant effect on the number of emitted neutron dose equivalent and that the farther the neutron source-detector distance, the lesser dose equivalent obtained. This result is very useful when considering an optimum position for neutron source for the maximum protection of occupational workers.

3.2The half value layer (HVL)

The results of the half value layer calculated show that 6.24 cm of pure polythene is required to attenuate 50% of the neutron dose equivalent while 5.73 cm of borated polythene is required for the same percentage. Thus, making borated polythene a better shielding material than pure polythene [5].

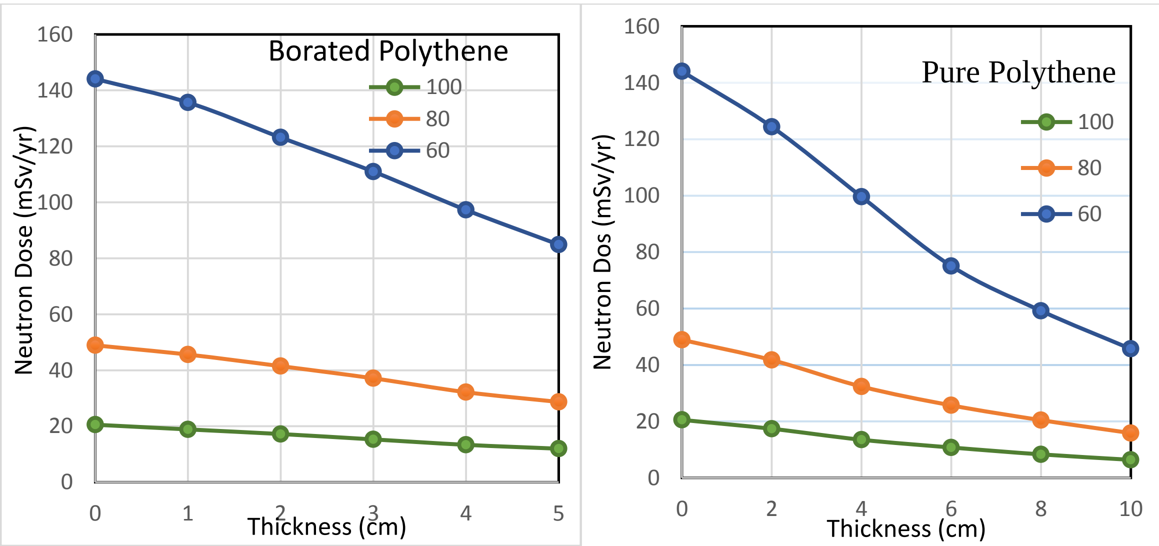

Fig. 2 Variation of Neutron Source-Detector Distances for shielding materials

4. CONCLUSION

It can be concluded that the thermalization and subsequent absorption of neutrons produced by ^{241}Am/Be source depends on the neutron source-detector distance and the density of the shielding materials. The results obtained show that the increase in the thickness of the shielding material decreases the neutron dose equivalent. It was also observed from the results that decreasing the neutron source-detector distance increases the neutron dose equivalent and as such, the neutron source-detector distance of 100 cm produced an optimum shielding configuration for an occupational workers.

A shielding thickness of 6.24 cm of pure polythene and 5.73 cm of 5% borated polythene were obtained to be capable of attenuating 50% of the neutron dose equivalent.

REFERENCE

[1].D. R. Ochbelagh, H. M. Hakimabadb, and I. Najafabadi, The investigation of Am–Be neutron

source shield effect used on landmine detection. Nuclear Instruments and Methods in Physics Research A 2007; 577: 756–761.

[2].H. Y. Kang, C. J. Park, K. S. Seo, and J. S. Yoon, Evaluation of Neutron Shielding Effects on Various Materials by Using a Cf-252 Source. Journal of the Korean Physical Society, 2008; Vol. 52, No. 6, pp. 1744-1747.

[3].A. Donzella, G. Bonomi, E. Giroletti, and A. Zenoni, Biological shielding assessment and dose rate calculation for a neutron inspection portal. Radiation Physics and Chemistry, 2012; 81, 414-420.

[4].F. J. Allen, and A. T. Futterer, Neutron Transmission Data. Nucleonics, 1963; 21(8), 120.

[5].Y. Karni, E, and Greenspan, Medium density hydrogenous materials for shielding against nuclear radiation. Proceedings, 8th International Conference on Radiation Shielding, April 24-28, 1994, Arlington, Texas USA, 1994; Vol 1, pp 43-50.

[6].D. A. Torres, R. D. Mosteller, and J. E. Sweezy, Comparison of MCNP5 and Experimental Results on Neutron Shielding Effects for Materials. Submitted to 2004 Annual Meeting of the American Nuclear Society, June 13-17, 2004. Pittsburgh, PA.

[7].R. C. Singleterry, and T. A. Sheila, Materials for Low-Energy Neutron Radiation Shielding. NASA Langley Technical Report Server, 2000.

[8].S. C. Gujrathi, and J. M. D'auria. The Attenuation of Fast Neutrons in Shielding Materials. Nuclear Instruments and Methods, 1972; 100, 445-452.

[9].C. E. Harrison, High Density Polythene/Boron containing Composites for Radiation Shielding Applications. An unpublished Ph.D Thesis, University of Kentucky, 2008.

[10].J. F. Briesmeiter, (Ed). MCNP- A general Monte Carlo N-Particle transport code, Version 4C. LA-13709-M, Los Alamos National Laboratory, 2000.

[11].ANS-6.1.1 Working Group, M. E. Battat (Chairman), "American National Standard Neutron and Gamma-Ray Flux-to-Dose rate Factors," ANSI/ANS-6.1.1-1977 (N666), American Nuclear Society, Lagrange Park, Illinois (1977).

[12].J. E. Martin, Physics for Radiation Protection. WILEY-VCH Verlag GmbH & Co. KGaA. Weinheim, 2006.

[13].J. R. Lamarsh, and A. J. Baratta, Introduction to Nuclear Engineering. Prentice-Hall, Inc. Upper Saddle River, New Jersey 07458, 2001.

[14].M. P. C. De Medeiros, J. J. S. Estrada, W. F. R. Junior, R. G. Gomes, R. F. G. dos Santos, C. F. E. Alves, and A. X. da Silva, Shielding of a Room for Installation of a calibration services for Neutron Monitors. A presentation at the 2013 International Nuclear Atlantic Conference, Brazil, 24 – 29, 2013.

[15].M. Coeck, F. Vermeersch, and F. Vanhavere, Neutron shielding evaluation for a small fuel transport case. Nuclear Instruments and Methods in Physics Research A, 2002; 476, 273–276.

[16].A. D. Wrixon. New ICRP Recommendation. Journal of Radiological Protection, 2008; 28: 161 – 168.

[17].ICRP, 1991, Recommendations of the International Commission on Radiological Protection. ICRP Publication 60. Oxford: Pergamon Press, 1990.