A Comparative Study Of Dose Transmission Factor Of Polythene And Borated Polythene For High Neutron Source Shielding

David. I. Igwesi, M. Sc., Otobong. S. Thomas, M. Sc.

ABSTRACT: A comparison of the dose transmission factor for Polythene and Borated Polythene against Am/Be neutron source have been investigated to ensure safe working environment for radiation workers. The Dose transmission factors for the materials were estimated using the Monte-Carlo transport code for incident neutron energies ranging from 0.025eV to 10MeV and the measurements were taken at three different neutron source-detector distances of 50cm, 70cm and 90cm by changing the position of the neutron source while the detector remained at fixed position. This study experimentally demonstrates that, by using suitable shielding material around Am/Be neutron source, the dose transmission factor of borated polythene is small compared to the dose transmission factor of polythene for any given thickness, and that the neutron source-detector distance of 90cm being a position where the Am/Be neutron source is very close to the centre of the water tank was found to be the best position for the neutron source to guarantee the safety of the radiation workers.

Keywords: Dose, Transmission Factor, Polythene, Borated Polythene, Neutron Source

1. INTRODUCTION

Neutrons are applied in many industrial/medical researches and can be produced by many processes which include nuclear reactions induced by alpha particles from naturally occurring alpha emitters, nuclear reaction induced by accelerated beams of light ions, fission and spallation reaction. Neutron sources can be classified as nuclear fission reactors, radioisotopes, and particle accelerators. These processes which can either be by natural or artificial means occurs with a wide range of energies of varying intensities. Neutrons from accelerators or nuclear reactors typically emanate as beams and these are readily characterized in terms of fluence of neutrons per unit area (n/cm^2) or fluence rate or flux (n/cm^2 s) [1]. Most laboratory work use radioisotope neutron sources because of their small size, portability, and do not require a high voltage source. These sources utilize (γ, n) or (α, n) reactions to produce neutron. Radioisotope neutron source could be direct or indirect. Direct radioisotope sources are those that emit neutrons in their natural decay processes while indirect sources are sources that rely on charged particle-emitting radionuclide and a stable target nuclide to produce neutrons through a nuclear reaction. These indirect sources involves the alpha-neutron sources which contain beryllium, boron or ^2H mixed with an alpha emitter that changes according to the radioactivity of the alpha source used. The most commonly used alpha-emitting sources are Americium, Plutonium, Radium, or Polonium together with beryllium are encapsulated to make a neutron source [2].

- *David. I. Igwesi, M. Sc. and Otobong. S. Thomas, M. Sc. Email: igwesidavid@yahoo.com*

Am/Be is one of the most commonly used indirect radioisotope neutron source in laboratory measurements. The alpha particle emitted by Americium-241 with decay energy of approximately 5.7 MeV and half-life of 433 years impinges on the Beryllium-9 target to produce neutrons with wide range of energies. The neutrons emitted are mostly high energy neutrons (fast neutrons) and thus needs to be thermalized by an appropriate material before being captured. Being an uncharged particle, neutron shielding is a bit complicated because of the wide range of energies and mass levels that are to be considered. Also, the gamma radiation produced during moderation of neutrons (thermalization) needs to be considered. The basis of neutron shielding is first reducing its energy through moderation (thermalization) and then placing shielding material with high neutron absorption cross section between the object and the source [3]. The use of efficient material for neutron shielding is an important step towards protecting radiation workers from the harmful effect of neutrons. The fast neutrons produced are more difficult to shield because absorption cross sections are much lower at higher energies, thus fast neutrons must first be thermalized either by elastic or inelastic scattering. In general, an efficient neutron shield is a combination of hydrogenous or low mass number materials to moderate neutrons; high absorption cross section materials to absorb the thermal neutrons and high atomic number materials to absorb the generated gamma rays [3]. Therefore, the effectiveness of any given material in shielding against neutron source depends on the material density, material thickness and the geometry of the neutron source being shielded. In most shielding materials the average distance a neutron travels between scattering collisions decreases rapidly as its energy decreases. The study therefore seeks to compare the dose transmission factor of Polythene and Borated Polythene. In their study to evaluate the neutron shielding effects on various materials, Kang et al [4] showed that the shielding effects depend on the thickness of the shielding materials and on the hydrogen content of the material. Thus, an increase in polythene thickness was found by Allen and Futterer [5] to reduce the dose transmission factor of fast neutron and Ochbelagh et al [6], observed a considerable increase in neutron counts when polythene shield was removed from the landmines. The effectiveness of polymeric materials in the attenuation of fast neutrons in shielding materials was studied by Gujrathi and D'auria [7]. Their result showed that borated polythene is a better material for neutron shielding than polythene. This result was supported by Harrison [8], Karni and Greenspan [9], who found that polythene containing boron composites, showed improved results for neutrons attenuation compared to polythene alone.

2. MATERIALS AND METHODS

The MCNP modelling was intended to represent the actual experimental setup as closely as possible. 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 [10]. This code enabled a detailed three dimensional modelling of the actual source and geometry configuration including the shield materials and the detector. 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 modelling

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

The moderator was modelled as a rectangular block of dimension 100 cm by 60 cm, filled with water of mass composition 67% hydrogen to 33% oxygen. Similarly, the Am/Be neutron source was modelled as an isotropic point source placed inside the moderator. Also, the detector was modelled as a helium gas filled spherical material, with diameter 20.8 cm. The neutron source and the detector were place 90 cm apart inside a rectangular world as shown in Figure 1. Furthermore, the polythene shield was modelled as a slab of dimensions 100 cm x 100 cm x 2 cm as shown in Figure 1. The percentage atomic fraction of carbon and hydrogen recorded in Table 1 above were used to develop the MCNP input code.

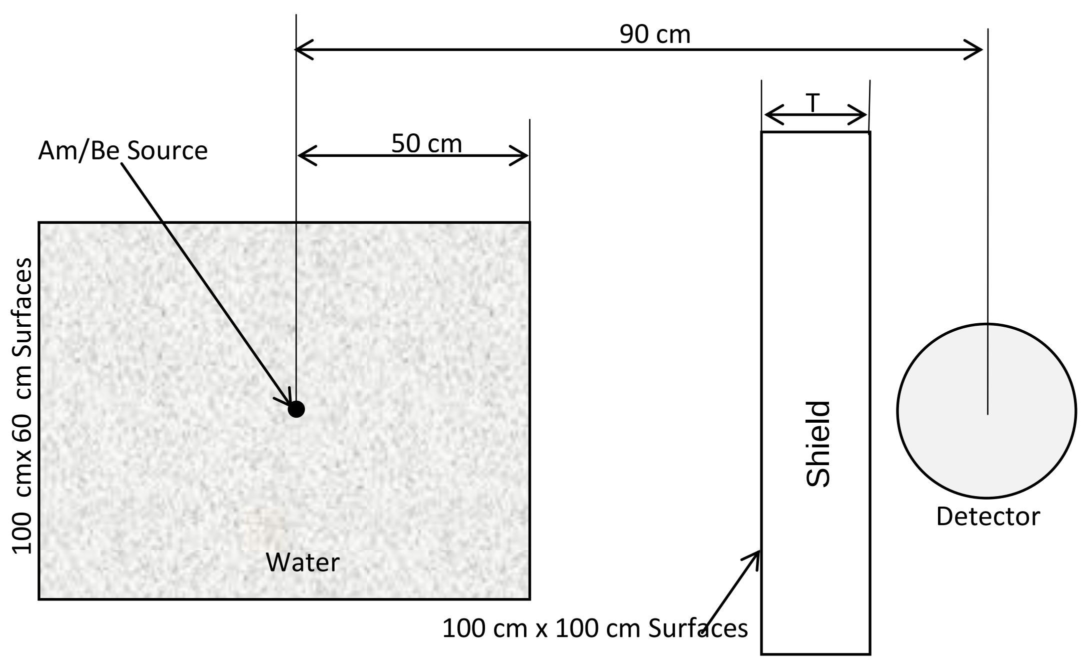

Fig. 3.1 Experimental Set-up for Am/Be neutron source shielding

3. MCNP SIMULATIONS

The neutron source detector model was simulated at three different neutron source-detector distances (50 cm, 70 cm and 90 cm) by moving the Am/Be source closer to the centre of the moderator. In each simulation, neutron histories of 10000000 were considered and the numbers of neutron flux incident (F14) were calculated for each of the model. Conversion from neutron fluxes to doses in rem hr^{-1}/n cm^{-2} s^{-1} was done using the most widely operational dose equivalent quantity for neutrons [11]:

$$H_c = h\Phi_m \qquad (1)$$

Where H_c is the dose equivalent, Φ_m is the emitted neutron fluence obtained from the simulations and h is the fluence-to-dose-equivalent conversion which varies with neutron energy and taken from ANSI/ANS-6.1.1-1977. The resulting dose values in rem hr^{-1}/n cm^{-2} s^{-1} were converted to Sievert by multiplying the result by 2.2 x 10^6 n/s and 100 (being rem to Sievert conversion) as well as the square of the source-detector distance. The neutron doses in Sievert were finally converted to microsievert by multiplying by 1000000. The relative errors of the calculations were kept below 10%. A polythene slab of thickness 2 cm was thereafter incorporated in the model when the neutron source and detector are 50 cm apart and the model was simulated maintaining the same number of particles histories. The thickness of the polythene was increased to 4 cm, 6 cm, 8 cm, and 10 cm and the resulting models were simulated in each case. The tallies for the neutron dose incident on the detector were recorded. The polythene slab was replaced with a slab of borated polythene of thickness 2 cm and the model was simulated for source-detector distance of 50 cm for 10000000 particles histories. The thickness of the borated polythene was increased to 4, 6, 8, and 10 cm and the resulting models were simulated in each case maintaining the same number of particle histories. The whole processes were repeated for both polythene and borated polythene slabs by changing the kind of material sample and thickness for a neutron source-detector distance to 70 cm and 90 cm by moving the source towards the centre of the moderator. In each case, the model was simulated and the tallies for neutron doses were recorded.

4. DOSE TRANSMISSION FACTOR

The rate at which each of these materials reduces neutron doses can be determined by the absorption cross section which is related to the transmission factor of the materials. Thus, according to Sorenson and Phelps [12], dose transmission factor is considered to be the ratio of the shielded neutron dose to the unshielded neutron dose:

$$Dose Transmission\ Factor = \frac{Dose\ with\ shield}{Dose\ without\ shield}$$

It represents the fraction of neutron dose transmitted by polythene or borated polythene. This shows that the transmission factor of small dose is better off than the transmission factor of large dose values. The dose transmission factor at zero thickness (no shield) of the shielding was calculated and afterwards for polythene and borated polythene at different thickness when the neutron source-detector distances are 50, 70 and 90 cm.

5. RESULTS AND DISCUSSIONS

Table 4.7 Calculated dose transmission factor for a source-detector distances

Borated Polythene				Polythene			
Thickness (cm)	Dose transmission factor			Thickness (cm)	Dose transmission factor		
	50 cm	70 cm	90 cm		50 cm	70 cm	90 cm
2	0.855	0.848	0.838	2	0.864	0.853	0.847
4	0.675	0.658	0.650	4	0.691	0.662	0.658
6	0.504	0.497	0.492	6	0.521	0.516	0.509
8	0.333	0.328	0.322	8	0.411	0.408	0.405
10	0.157	0.151	0.149	10	0.317	0.314	0.311

The dose transmission factors of the two shielding materials investigated in this study are compared in table 4.7 below. It is observed from the table that an increase in the thickness of both borated polythene and pure polythene reduces the dose transmission factor [5]. The difference in dose rate for a given thickness of both borated polythene and pure polythene is not very pronounced because according Coeck et al [13], the reduction power of the two shields is not very much when dealing with small thickness but very noticeable as the thickness of the materials increase. However, the result shows that the dose transmission factor for borated polythene is small compared to the dose transmission factor for polythene for a given thickness, and this shows that borated polythene is a better material for shielding of neutron source than pure polythene, supporting the work of Gujrathi and D'auria [6], and Singleterry and Sheila, [14]. Generally, it is observed that the transmission factor depends also on the neutron source-detector distance. As the neutron source-detector distance increases, the dose transmission factor decreases, making the source-detector distance of 50 cm to be of high dose transmission factor compared to source-detector distance of 70 cm and 90 cm.

6. CONCLUSION

It is observed that the transmission factor depends also on the neutron source-detector distance. As the neutron source-detector distance increases, the dose transmission factor decreases, making the source-detector distance of 50 cm to be of high dose transmission factor compared to source-detector distance of 70 cm and 90 cm Generally, it was observed from the results that polythene and borated polythene provide significant reductions in the neutron doses to radiation personnel working near the Am/Be neutron source and this suggest that the optimisation of the shielding system is necessary and very relevant for radiological protection.

REFERENCES

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

[2]. J. K. Shultis, and R. E. Faw. "Fundamentals of Nuclear Science and Engineering", Second Edition. CRC Press, 2002.

[3]. P. Rinard, "Neutron Interactions with Matter. In Passive Nondestructive Assay of Nuclear Materials", (ed. D. Reilly, N. Ensslin, and H. Smith Jr), Nuclear Regulatory Commission, NUREG/CR-5550. Pp. 357-377, 1991.

[4]. 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, Vol. 52, No. 6, pp. 1744-1747, 2008.

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

[6]. D. R. Ochbelagh, H. M. Hakimabadb, and R.I. Najafabadi. "The investigation of Am–Be neutron source shield effect used on landmine detection". Nuclear Instruments and Methods in Physics Research A 577, 756–761, 2007.

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

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

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

[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]. Zheng, Y., Newhauser, W., Fontenot, J., Koch, N. and Mohan, R. (2007). Monte Carlo simulations of stray neutron radiation exposures in proton therapy, Journal of Nuclear Materials 361 289-297.

[12]. Sorenson, J. A. and Phelps, M. E. (1987). Physics in Nuclear Medicine, W. B. Saunders Company, Second Edition.

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

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

ABOUT THE AUTHORS:

David. I. Igwesi is a Lecturer at the Department of Physics and Industrial Physics, Nnamdi Azikiwe University, Awka. He holds a Master of Science (M. Sc.) in Nuclear Science and Technology from the University of Liverpool, UK. His research interest are in the areas of Radiation Shielding, Nuclear Materials and Radiation Detectors Efficiency. He is a member of Nigerian Institute of Physics.

Otobong. S. Thomas is a Lecturer at the Department of Physics, University of Port Harcourt. He holds a Master of Science (M. Sc.) in Nuclear Science and Technology from the University of Liverpool, UK. His research interest are in the areas of Radiation Shielding, Nuclear Materials and Radiation Detectors Efficiency. He is a member of Nigerian Institute of Physics.