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Review Article

Effectiveness of Serpentine Concrete as Shielding Material for Neutron Source Facility Using Monte Carlo Code

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In recent years, much attention has been dedicated to finding techniques to reduce exposure doses. This work examines the effectiveness of using serpentine concrete to shield a neutron source using a ^{241}Am-Be neutron source facility at the National Nuclear Research Institute (NNRI) as a case study. The results obtained for both neutrons and gamma indicate that serpentine concrete provides better shielding as compared to ordinary concrete. At a distance of 100 cm from the Am-Be source, when shielded with serpentine concrete, it was found that personnel will receive an average gamma dose of $4.395.395 \pm 0.122\,\mu$Sv/h while a dose of $10.399 \pm 0.083\,\mu$Sv/h will be received for ordinary concrete shield. The average neutron dose equivalent at 100 cm, for ordinary concrete and serpentine concrete were 32.189 ± 0.277 and 9.276 ± 0.505, respectively. All dose equivalents obtained were also within internationally accepted limits.

1. Introduction

There are a variety of applications employing the use of radiation. In employing most of these applications, there is a direct interaction with humans resulting in various levels of radiation exposure to the human body. One of the current challenges of radiation protection is reducing the level of exposure dose. Shielding, time, and distance are seen as the major ways to implement and ensure exposure dose reduction. However, as technology advances and more beneficial applications are found for radiation, humans are brought into closer contact with the radiation source thereby increasing the risk of exposure. The need for high-performance shielding material to reduce dose exposure cannot be overemphasized. Due to the high penetrating ability of neutron and photon radiations, it is evident that particular attention is paid to sources emitting such particles. In this study, the irradiator used contained a ^{241}Am-Be source, a neutron, and a photon emitter. An Am-Be radioactive isotope is an important neutron source and has been used efficiently for many applications in medicine, industry, and scientific research [1]. Due to the long half-life of ^{241}Am-Be ($t_{1/2} = 432.7$y), coupled with its supply of high flux of neutrons and photons, it is essential to provide an appropriate and effective shield to ensure dose exposure reduction. Apart from the neutron, which is desired for the start of the reactor, the Am-Be source has two other important radiations. The alpha (α) radiation, which has a higher yield of about 85% of the primary reaction and gamma (γ) as a result of the activity of the actinide produced from the primary reaction.

To aid in shielding, the neutron which is born at a very high speed (fast neutrons) has to undergo a series of slowing down (thermalization/moderation) until it is finally absorbed in the shielding material. Neutrons undergo scattering and absorption interactions with the matter. These interactions are the basis for techniques used to shield and measure neutron radiation doses. Neutrons interact primarily with the nucleus of the atom. Consequently, the types of materials favoured for neutron shielding are quite different than the dense, high atomic number absorbers which are most effective in the attenuation of gamma radiation.

Studies have been conducted to investigate the effectiveness of concrete mixtures with materials such as iron, hematite, barites, chalcocite, and boron carbide in concrete mixtures for neutron shielding.

In practice, the most commonly used shielding materials include polyethylene, paraffin, water, and different types of concrete. The facility used for this study has a series of barriers or shields it employs in reducing dose exposure, i.e., water, PVC plastic, and ordinary concrete. The thermalization process is mostly made possible by elastic collisions of a neutron with light nuclei such as hydrogen. The suitability of hydrogen as a moderator makes any material with a high content of hydrogen a good candidate for shielding of a ^{241}Am-Be source. In order to provide suitable as well as adequate shielding for photons, materials are chosen according to their linear attenuation coefficient and mass density [2].

In this study, the shield is designed for both escaping fast neutrons and gamma radiation. Alpha particles are not considered because alpha particles have low penetrating abilities and they are shielded by the inner lining of the Am-Be facility and thus do not pose radiation protection risks. Gamma and neutrons typically have high penetration and high energies, and inadequate shielding has the potential of causing damage to the personnel and the environment. Hence for the purpose of radiation protection, it is required that facilities that have the ability to emit gamma and neutrons are well shielded [3].

The ^{241}Am-Be neutron source irradiator facility at the National Nuclear Research Institute (NNRI) provides an alternate source of neutrons for irradiation purposes aside from the one produced in the research reactor (GHARR-1). The low neutron flux levels of this facility allow for good sensitivity when used for neutron activation analysis (NAA). Having a stable neutron flux eliminates the need for standard material in the measurement of induced activity in a sample, thus having an advantage over nuclear reactors [4].

Recent works had established that the degree of attenuation of biologically harmful forms of radiation, such as gamma-rays and X-rays, is proportional to the atomic mass (Z) of the shielding material. The high penetration power enables the radiation to penetrate the living bodies thereby causing harmful ionization of the biological cells. Therefore, the concrete material capable of effective attenuation of these rays plays a critical role in ensuring the safety and health of personnel in nuclear facilities [5, 6]. In living tissues, neutrons are biologically damaging, and they are about ten times more carcinogenic than photons or beta rays with the same level of radiation [3]. Hence effective shielding is required to protect the personnel working at the ^{241}Am-Be neutron source of the NNRI. The ^{241}Am-Be facility is currently shielded using ordinary concrete blocks. Aside from ordinary concrete, other concrete types can also be used for shielding.

Li et al., [7] modelled and calculated the neutron and gamma dose rate distributions around the liquid-fueled thorium molten salt reactor (TMSR-LF) after applying a 60 cm serpentine concrete layer as a radiation shield. The analysis of the results showed that the maximum thermal neutron flux density outside the reactor biological shield was $1.89\times 10^3\,\mathrm{cm}^{-2}\,\mathrm{s}^{-1}$, which was below the $1\times 10^5\,\mathrm{cm}^{-2}\,\mathrm{s}^{-1}$ limit. Also, previous works by Kansouh, 2012 on collimated beams from ET-RR-1 reactor showed that serpentine concrete is a better neutron shield than ordinary concretes.

There is an abundance of rock called serpentinite that can be used for concrete intended for shielding radiation sources in Ghana. This work explores the performance of serpentine concrete as an alternative to ordinary concrete for shielding the ^{241}Am-Be neutron source. Serpentinite consists of a hydrous magnesium silicate with water for hydration ($3MgO.\ 2SiO_2.\ 2H_2O$). The rock also contains calcium and aluminium oxide [5]. Having a higher percentage of water content than ordinary concrete makes serpentine suitable for shielding neutrons [6]. Also, serpentine loses its water of crystallization when heated to a very high temperature (above 480°C) and as such serpentine shields could be subjected to high temperatures without a need for complex and expensive cooling systems for radiation shielding. So far, three deposits have been found in Ghana, at Anum, Peki-Dzake and Golokwati [7, 8]. They have been investigated and found to be suitable for photon attenuation [9]. This irradiator is housed in a laboratory with work surfaces and an office a few meters away from the concrete shield, hence the need to ensure that dose to researchers is reduced to the barest minimum. An Am-Be neutron source emits neutrons by the alpha decay of americium-241 (^{241}Am) with a decay energy of 5.638 MeV and a half-life of 432.2 years [10]. These alpha particles, emitted in Am-241 decay, then bombard the target nucleus of the Beryllium-9 (^{9}Be) target, producing neutrons over a broad range of energies with an average energy of around 4.2 MeV and a maximum of around 10 MeV. The neutron fluxes investigated are thermal, epithermal, and fast. The thermal neutron flux energy ranges from 0 to 6.25×10^{-7} MeV, the epithermal neutron flux energy ranges from 6.25×10^{-7} to 8.21×10^{-1} MeV, and that of the fast neutron flux ranges from 8.21×10^{-1} to 20.00 MeV, and these were the energy ranges chosen for this work. The ability of the serpentine concrete to shield high energetic neutrons and the resulting gamma energy produced from the ^{241}Am-Be neutron source has been studied in this work. Calculations of the gamma doses equivalent were computed for the various neutron energy ranges considering that the Am-Be source can emit neutrons up to about 10 MeV. This is of much concern in the area of radiation protection because of the severity of damage it can cause if the personnel, the public, and the environment are exposed.

2. Description of ^{241}Am-Be Source Facility

The ^{241}Am-Be irradiator of the NNRI used in this study is arranged in a cylindrical configuration. The outer containment vessel is made of plastic having a thickness of 0.54 cm, a height of 122.00 cm, and an inner diameter of 97.00 cm, respectively. For neutron moderation, the irradiator uses deionised water. The neutron source for the irradiator is a single 20 Ci ^{241}Am-Be neutron source of $4.4\times 10^7\,\mathrm{n\,s}^{-1}$ source strength and having a quoted emission of 2.2×10^6 n $\mathrm{s}^{-1}\,\mathrm{Ci}^{1}$ positioned centrally and perpendicularly to the vessel base at a height of 47.30 cm from the base of the cylindrical vessel [11]. Figures 1 and 2 show the longitudinal and cross-sectional views of the Am-Be neutron source. The neutron source is embedded in a Lucite and Plexiglas rod, 5.16 cm in diameter which is then encased in an aluminium tube of diameter 6.00 cm. There are two irradiation channels made of aluminium on either side of the neutron source. They are 2.31 cm and 2.72 cm in radius and placed at distances of 13.10 cm and 4.2 cm, respectively, from the source (Figure 1). The irradiator is mainly used for neutron activation analysis (NAA) which aids in the qualitative and quantitative analysis of materials in different research areas such as biological and geological samples [13, 14]. The stability of the irradiator's neutron flux is a major advantage. However, the low magnitude of this flux compared to the one generated in a particle accelerator or nuclear reactor requires that large samples are irradiated for good sensitivity [4, 15].

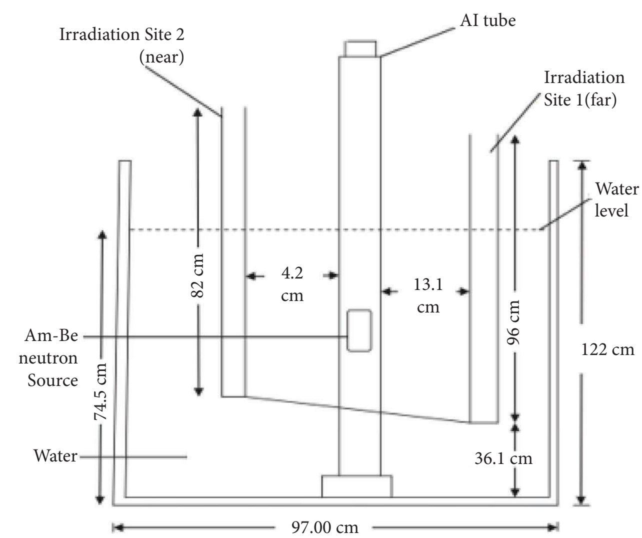

Figure 1: Schematic longitudinal view of ^{241}Am-Be neutron source [12].

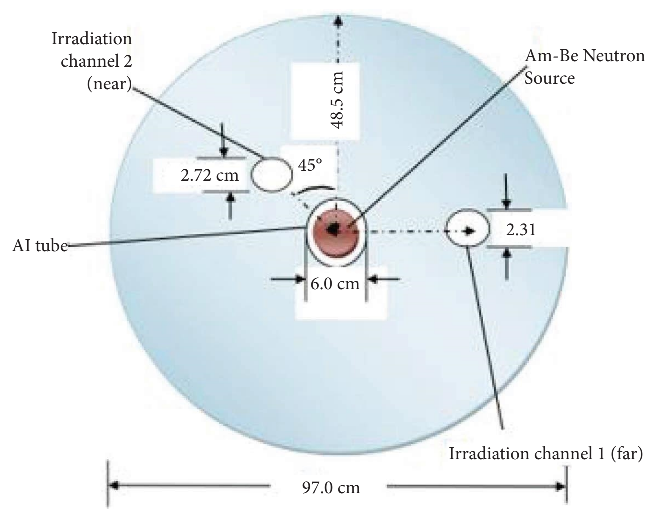

Figure 2: Schematic cross-sectional view of ^{241}Am-Be neutron source [12].

3. Methodology

The purpose of the present study as stated earlier was to investigate the use of serpentine concrete as an alternative to ordinary concrete in shielding the Am-Be neutron source facility at NNRI by comparing their shielding abilities. The Monte Carlo N-Particle code version 5 (X-5 Monte Carlo Team, 2003) was used to simulate the ^{241}Am-Be neutron source facility having both ordinary concrete and serpentine concrete as a shield (Figures 3 and 4).

For the study, two MCNP models of the Am-Be facility were set up; one with ordinary concrete and another with serpentine concrete as shields. Table 1 shows the percentage composition for ordinary concrete and serpentine concrete. Simulations were run for 215,000,000 source particle histories (500,000 source particles and 430 KCODE cycles). The code was run on a personal computer with the following specifications: Intel(R) Core (TM) i5-4200U CPU 2.3 GHz processor, 8.00 GB of RAM, and Windows 8.1 (64-bit). In order to determine the possible neutron and gamma doses received by the staff working with the neutron source in the laboratory, specific points were selected from where measurements were taken. At each of these points, a cylinder (dose cell) of 165 cm height (average height of Ghanaian), and with a radius of 10 cm was modelled and filled with water.

MCNP treats every particle using its path as defined by the specified geometry. The probability of a particle travelling a distance, ℓ, without collision is shown by the following equation [16]:

$$P(\ell) = \operatorname{Exp}\left(-\sum_t \ell\right). \tag{1}$$

The probability of a particle undergoing a collision between ℓ and $\ell + d\ell$ is given by the following equation:

$$P(\ell)d\ell = \operatorname{Exp}\left(\sum_t \ell\right) * \sum_t d\ell. \tag{2}$$

Integrating equation (2) to determine the length of the path and equating to a random number, ξ, gives the equation

$$\xi = \int_0^1 \operatorname{Exp}\left(-\sum_t s\right) * \left(\sum_t ds\right), \tag{3}$$

and simplifying the equation further gives

$$\xi = 1 - \operatorname{Exp}\left(\sum_t \ell\right), \tag{4}$$

then, we solve ℓ by using the following equation:

$$\ell = -\frac{1}{\sum t} \ln (1 - \xi), \tag{5}$$

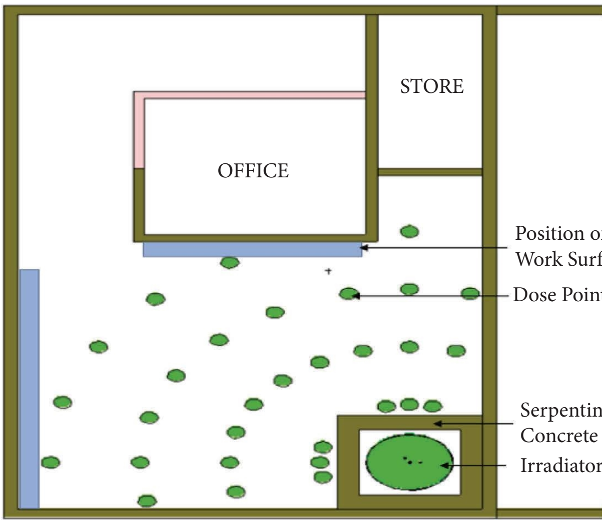

Figure 3: MCNP5 model of ^{241}Am-Be facility with serpentine concrete as shield.

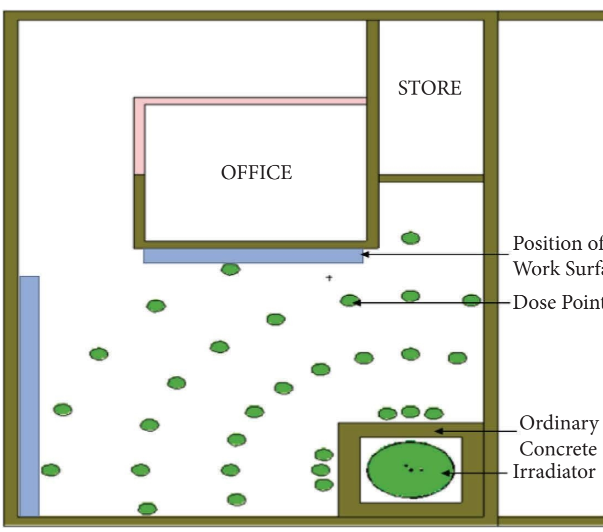

Figure 4: MCNP5 model of ^{241}Am-Be facility with ordinary concrete as shield.

Table 1: Percentage composition of ordinary concrete and serpentine.

Raw material	Concrete type	
	Ordinary concrete	Serpentine
Portland cement	11.82	15.94
Sand	26.71	27.35
Gravel	54.96	—
Water	6.51	4.53
Serpentine	—	48.33
Density (gcm^{-3})	2.30	2.60

which shows an even distribution of the random numbers ξ, hence the distance covered by a particle to collision follows the equation:

$$\ell = -\frac{1}{\sum t}\ln(\xi). \qquad (6)$$

Basic dose distribution data are usually measured in water phantom which closely approximates the radiation absorption and scattering properties of muscle and other soft tissue [17], hence the use of water to fill the dose cells. These cylinders termed dose cells were arranged in concentric circles around the neutron source at intervals of 1, 2, 3, and 4 meters from the centre of the neutron source (Figures 3 and 4).

In the simulations, the dose cells were used to calculate both the neutron flux and the absorbed dose rate as well as the gamma flux and the absorbed dose rate. The neutron and gamma fluxes were estimated using the tally F4 (cm^2) with MODE N and P, respectively, which calculates the average flux over a cell (particles cm^2). The absorbed neutron and gamma dose rates were obtained through the tally F6 (MeV g^{-1}) with MODE N and P, which considers the energy deposition average over the cell. This tally gives the energy deposited per unit mass of the material. The absorbed dose rate is calculated from the tallies obtained. This was then multiplied with the radiation weighting factors (W_R) for neutron and gamma to obtain the respective equivalent doses.

4. Results and Discussion

The graphs (Figures 5 and 6) show similar trends for the attenuation of both neutrons and gamma-rays for both ordinary concrete and serpentine concrete.

In the case of neutrons (Figure 5), it was observed that, as expected, the dose equivalent decreased as the distance from the source increased. At a distance of 100 cm from the centre of the ^{241}Am-Be source, the average neutron dose equivalents were found to be 32.189 μSv/h and 9.276 μSv/h for ordinary concrete and serpentine concrete, respectively. This means that the observed neutron dose equivalent of serpentine concrete was only 28.83% of that observed for ordinary concrete (Table 2). Similarly, at distances of 200 cm, 300 cm, and 400 cm from the source, the observed neutron dose equivalent of serpentine concrete was found to be 32.91%, 38.65%, and 42.50%, respectively, of that observed for ordinary concrete.

As described for the case of neutrons (Figure 5), similar observations were made for the gamma dose equivalents which decreased with increasing distance from the source (Figure 6). From Table 3, it can be seen that at distances of 100 cm, 200 cm, 300 cm, and 400 cm from the source, the observed gamma dose equivalents of serpentine concrete were found to be 42.31%, 36.71%, 39.77%, and 42.34%, respectively, of that observed for ordinary concrete.

Figures 7 and 8 present a comparative analysis of the results obtained with existing results on ordinary concrete and serpentine concrete shielding properties for neutrons and gamma doses [18]. The results compared well with the existing results in the literature and hence verify the methodology employed in this work.

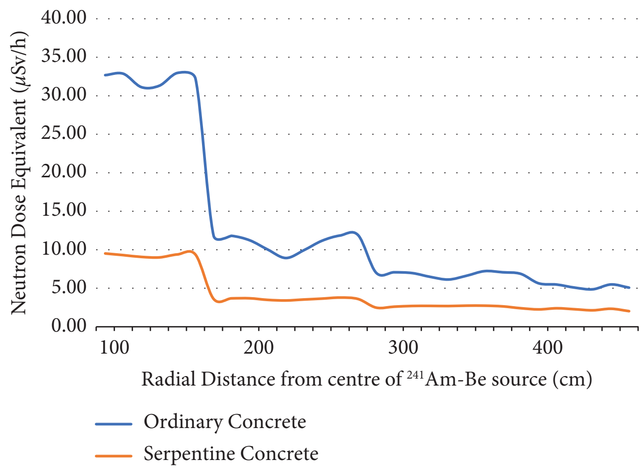

Figure 5: Graph of neutron dose equivalent versus radial distance from the centre of ^{241}Am-Be source.

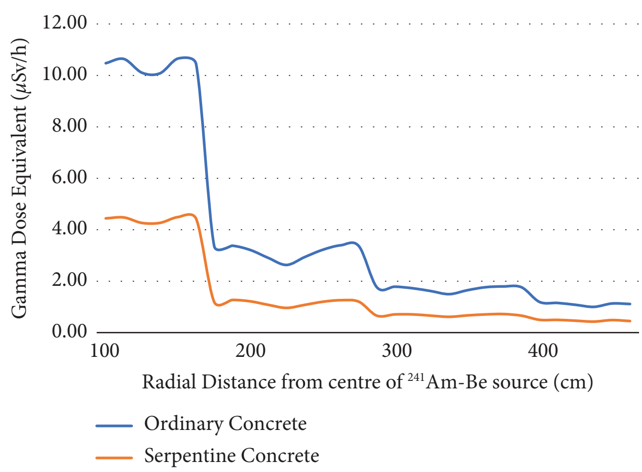

Figure 6: Graph of gamma dose equivalent versus radial distance from the centre of ^{241}Am-Be source.

Table 2: Ratio of average neutron dose equivalent of serpentine concrete to ordinary concrete.

Distance from source (cm)	Average neutron dose equivalent (μSv/h)		(NDESC/NDEOC) × 100
	Ordinary concrete	Serpentine concrete	
100	32.189 ± 0.277	9.276 ± 0.505	28.82
200	10.965 ± 0.462	3.610 ± 0.776	32.92
300	6.829 ± 0.579	2.636 ± 0.918	38.60
400	5.273 ± 0.653	2.239 ± 0.973	42.46

NDESC, neutron dose equivalent of serpentine concrete; NDEOC, neutron dose equivalent of ordinary concrete.

Table 3: Ratio of average gamma dose equivalent of serpentine concrete to ordinary concrete.

Distance from source (cm)	Average gamma dose equivalent (μSv/h)		(GDESC/GDEOC) × 100
	Ordinary concrete	Serpentine concrete	
100	10.399 ± 0.083	4.395 ± 0.122	42.26
200	3.156 ± 0.134	1.161 ± 0.224	36.78
300	1.705 ± 0.185	0.678 ± 0.291	39.77
400	1.111 ± 0.228	0.470 ± 0.621	42.30

GDESC, gamma dose equivalent of serpentine concrete; GDEOC, gamma dose equivalent of ordinary concrete.

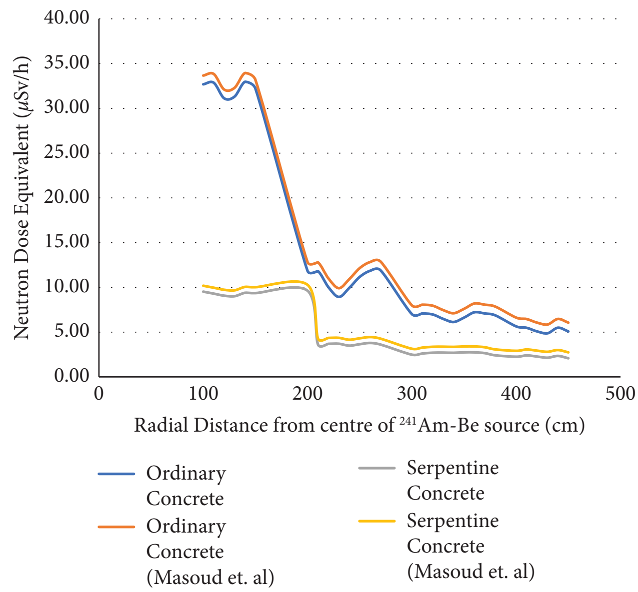

Figure 7: Comparison of results of neutron dose equivalent versus radial distance from the centre of ^{241}Am-Be source with literature.

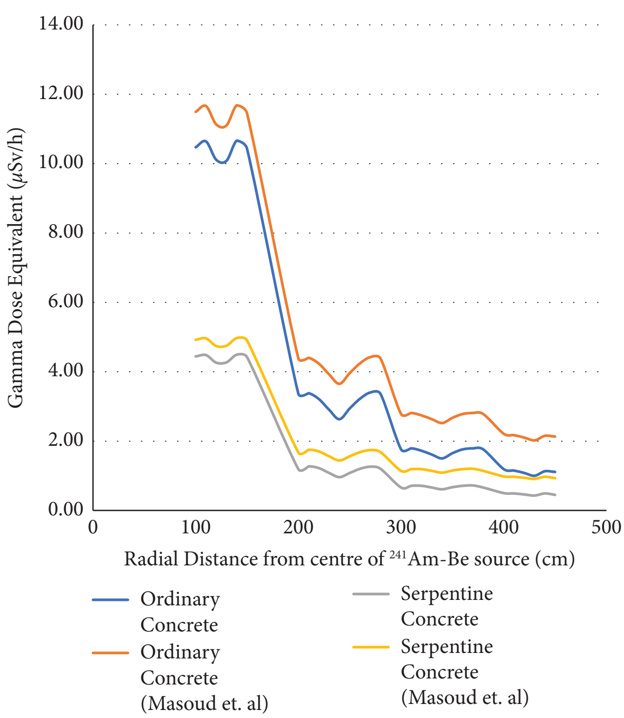

Figure 8: Comparison of results of gamma dose equivalent versus radial distance from the centre of ^{241}Am-Be source with literature.

5. Conclusion

The doses recorded were all found to be within acceptable limits set by both International Commission on Radiological Protection (ICRP) and Nuclear Regulatory Authority (NRA), Ghana.

The results obtained for both neutrons and gamma indicate that serpentine concrete provides better shielding as compared to ordinary concrete. At 100 cm from the Am-Be source, when shielded with serpentine concrete, it was found that the personnel receives an average gamma dose of $4.395.395 \pm 0.122\,\mu$Sv/h while a dose of $10.399 \pm 0.083\,\mu$Sv/h was received for ordinary concrete shield. The average neutron dose equivalent at 100 cm, for ordinary concrete and serpentine concrete were 32.189 ± 0.277 and 9.276 ± 0.505, respectively. On an average, the serpentine concrete reduces the neutron dose recorded for the ordinary concrete by a factor of approximately 3. In the case of gamma, the serpentine concrete reduced the dose by a factor of approximately 2. To further ensure the safety of the personnel, where available, serpentine concrete can be used as a preferred alternative over ordinary concrete to provide more effective shielding for neutron and gamma facilities. Newer facilities should consider the use of serpentine concrete as a preferred alternative to ordinary concrete for enhanced protection of the personnel, the public, and the environment.

Data Availability

The data used to support the findings of the study can be obtained through this publication in print or online.

Conflicts of Interest

The authors declare that they have no conflicts of interest.

Acknowledgments

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