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Determination of linear attenuation coefficient of aggregate serpentine concrete exposed to gamma and neutron radioactive sources

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Abstract

This research was designed to perform linear attenuation calculations for ordinary concrete both through simulation and experiment. The linear attenuation coefficient is an essential parameter for radiation shielding design for both source transport casks and storage bunkers. The shielding properties of concrete designed with Serpentine aggregates of different granule sizes were investigated. The simulation was performed using Radpro computer Software while the experiment involved sourcing for local Serpentine rock, crushing into four different granule sizes of 5 mm, 10 mm, 15 mm and 20 mm respectively and casting the concrete samples, exposure of the samples to gamma and neutron radioactive sources and monitoring with a radiation survey meter, sample weight measurements and concrete sample crushing using a load testing machine for determination of the concrete compressive strength. The results of the linear attenuation calculations showed that there was high consistency in the values obtained by simulation with those obtained via experiment. Very high linear attenuation property was observed when the serpentine concrete samples were exposed to a neutron source, which corroborates the fact that Serpentine rock is a Boron-rich mineral and Boron is known to have high neutron absorption property. The experimentally determined linear attenuation coefficients showed that the values at 15 mm aggregate sizes were higher than those at 5 mm, 10 mm and 15 mm sizes respectively, which demonstrates that better shielding optimization, will be obtained when the concrete cask is fabricated with 15 mm aggregate size.

Keywords: Attenuation; Serpentine; Concrete; Radioactive

1 Introduction

Radiation dose that exceeds the maximum permitted level is extremely harmful to humans. Nuclear particles or electromagnetic waves that are produced as a direct or indirect result of radioactive decay and fission are collectively referred to as radiation [1]. According to the atomic mass of the substance, gamma rays are attenuated. Making a gamma-ray shield out of the densest material that is economically achievable is useful [2]. Due to its low weight per unit, great durability, and effective heat insulation, lightweight concrete, which has been utilized since the early Roman era, has gained popularity. Due to their pozzolanic activities, which react with free and reactive chemicals released during the hydration of the cement, lightweight aggregates, particularly those of volcanic origin, boost the strength, durability, and heat insulation qualities. Depending on the production process and the materials used, lightweight concrete exhibits a wide range of characteristics. Due to radiation emissions from numerous tools and materials once utilized for the benefit of humans, technology has advanced and is now widely used, endangering human health [3].

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Also, it has been reported that concrete is a good substitute material for radiation shielding due to its versatile application in shielding designs as it can easily be cast into various shapes and sizes with different types of mixture constituents depending on the type of radiation to be shielded [4]. The excessive use of radioactive sources cannot be overemphasized, but this usage can cause harm to man and the environment if not properly controlled [5]. It has been established that radiation exposure from poorly shielded radioactive sources can cause several radiation injuries of different magnitudes in humans [4]. Since radioactive sources are extensively used in Nigeria for different purposes such as medical, petroleum industry, agriculture, food preservation, animal husbandry, water resources management, pest control, industry, materials analysis, and mineral exploration as well as in science and research, the country have recorded a stock pile of so many legacy radioactive sources in which some are currently maintained by the Ajaokuta Steel Company Limited. Consequently, there is burden of transporting and storing those legacy radioactive sources in a safe and secured environment to avoid losing to non-state actors. The growing nature of terrorism in Nigeria and the risk that terrorists may acquire, traffic in or use radioactive materials or sources in radiological dispersion devices, have drawn the attention of stakeholders to secure the numerous radioactive sources in use, in temporary storage as well as the legacy sources under the watch of Nigeria Nuclear Regulatory Authority (NNRA). In a bid to join the determination of the international community in securing radioactive materials, Nigeria has signed an agreement of cooperation with the United States Department of Energy (US-DOE) Office of Radiological Security (ORS). The objectives of this corporation are to reduce and protect vulnerable radioactive material located at civilian sites; remove and dispose of excess radiological materials; and protect nuclear and other radiological materials from theft or sabotage [6]. Consequently, Nigeria has since begun processes and preparation for relocation of over 200 legacy sources currently at Ajaokuta Steel Company to a safer and secured storage facility. These processes involve the design and fabrication of a portable, resilient and cost-effective transportation cask that will be used to convey the sources to the storage facility. Hence, in an effort to contribute to the country's determination to relocate the legacy sources, designing a concrete cask that could be used to safely and securely transport the radioactive sources to the temporary radioactive waste management and storage facility at the Centre for Energy Research and Training (CERT) Zaria is necessary. This research determined the linear attenuation coefficient of aggregate serpentine concrete exposed to gamma and neutron radioactive sources using experiment and simulation approach.

2 Material and methods

The materials used during this research in order to achieve the stated objectives are divided into two parts, which including both experimental and simulation.

2.1 Materials for Simulation

The data for parameters (comprising the radionuclide type, initial activity of the radionuclide, reference date of the initial activity as well as the final activity and the expiration date) were obtained for the two hundred and twenty-seven legacy sources which working live expired since 1993, Microsoft soft Excel software (2016 version) and Rad Pro calculator software (version 3.26) were used for statistical and theoretical analysis.

2.2 Rad Pro Calculator

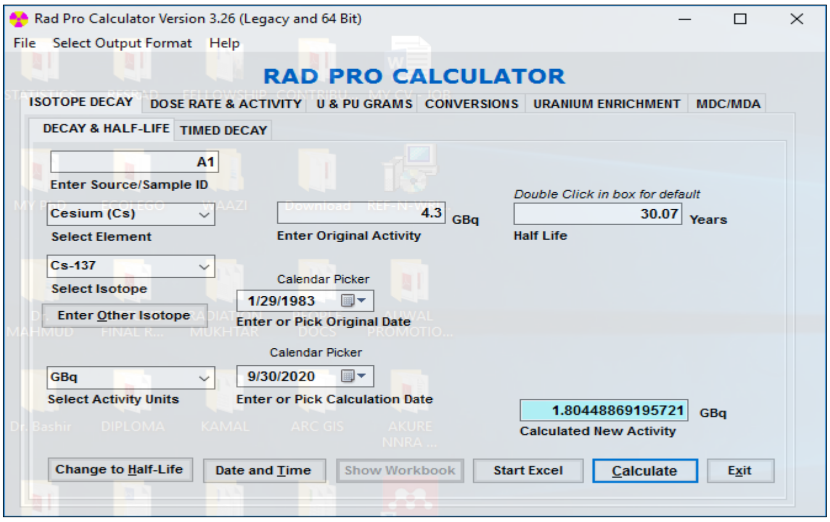

Figure 1 Rad Pro Activity Calculation Interface

The present activity of each source was calculated by putting the original activity, the reference date and the calculation date (which was 30th September, 2020) under the decay and half-life section of the isotope decay on the Rad pro calculator. Figure 1, illustrates the calculation for the group A1 sources combined.

The total present activity of the sources under the same group was obtained by multiplying the present activity of one source with the total number of sources in the group [7]. To obtain the total present activity of all the Cs-137 sources at Ajaokuta steel company, the activities of the source groups A1 - A29 were aggregated. The same was done for the C0-60 sources B1-B5 and for Pu-239 sources.

2.3 Dose rate in air at 100cm from the source

The dose rate in air at 100cm from each aggregated source activity was obtained from Rad Pro Calculator Software by selecting the required input parameters such as: the radioactive isotope, the dose rate unit, distance, the item ID as well as the combined activity on Rad pro calculator input deck. Figure 2, illustrate the calculation for the aggregated Cs-137 sources activity.

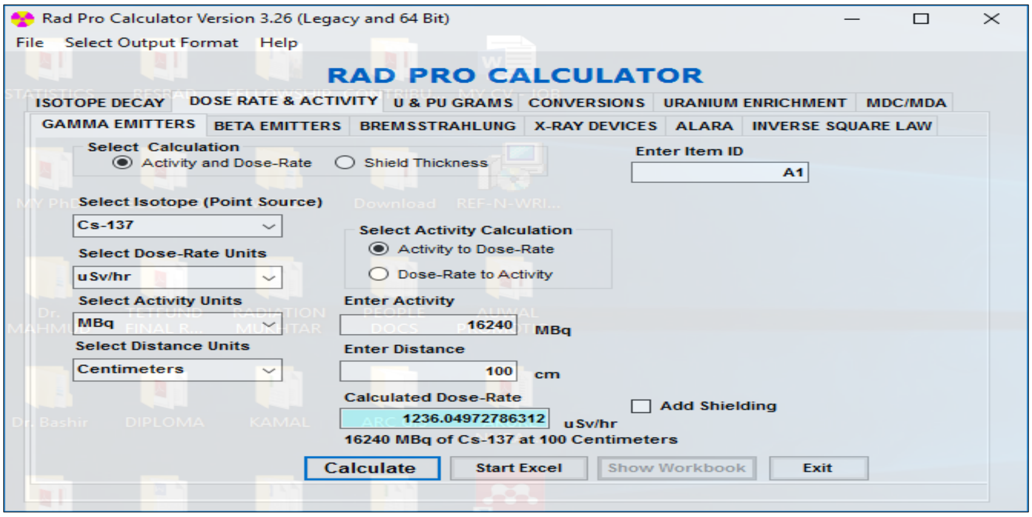

Figure 2 Rad Pro Dose Rate Calculation Input Interface

2.4 Determination of attenuation coefficients

The attenuation coefficients were calculated for Cs-137, Co-60 and Pu-239 sources. The equation used was adopted from [8-10]

$$D = D_O e^{-\mu x} \qquad ....(1)$$

Where, D is the dose rate (µSv/hr) at 100cm when a concrete shielding material of thickness (x) was used to shield the radioactive source, D_O is the dose rate at 100cm from the combined activity of each radioactive source when there is no concrete shield. For Cs-137 and Co-60, the thicknesses used in estimating the attenuation coefficients were 10, 20, 30, 40, 50, 60 and 70cm respectively. While for Pu-239 source the thicknesses used were 0.1, 0.2, 0.3, 0.4, 0.5 and 0.6cm respectively. These arbitrary thicknesses were so chosen based on the aggregated activity of the radioactive sources. The dose rates were computed using the Rad Pro Calculator Software and the obtained results were plotted using Microsoft Excel Version 2016. In order to obtain the attenuation coefficients, Equation 1, was linearized into equation 2.

$$In\left(\frac{D}{D_O}\right) = \mu x \qquad ....(2)$$

2.5 Materials Used for Experiment

2.5.1 Fabrication of concrete blocks with different aggregate sizes

An experimental determination of linear attenuation coefficients of a Serpentine concrete was carried out to benchmark with the simulated data. To achieve this, samples of Serpentinide rock was obtained from Jibiya Local Government of

Katsina State, Nigeria. The same was transported to a quarry site at Zaria, Kaduna State for grinding into fine aggregates of sizes 5 mm, 10 mm, 15 mm and 20 mm respectively which were accurately obtained with the aid of sieves. 200 kg of sharp river sand was collected at river Kubanni in Zaria and a sieve was also used to obtained sand of sizes ranging between 4-5 mm. 100L of treated water was collected at Zaria Water Treatment Plant located at Congo-Zaria.

With the aid of a hand trowel and shovel, a concrete mixture was produced using the standard mixes ratio of 1:2:4 for cement-sand-aggregate and 2:5 for cement-water mixture. Using 20 constructed 15 cm by 25 cm metallic molds, concrete mixed with each aggregate size was loaded into 5 molds of different depths to produce concrete blocks of different thicknesses (9cm, 8cm, 7cm, 6cm and 5cm respectively). Vibrating table was used to ensure uniform distribution of the aggregates in the concrete matrix. Hence a set of 4 concrete blocks (5 in each set with thicknesses numbered 1-5) were fabricated and labelled A, B, C, and D Figure 3.

Figure 3 Fabricated Serpentine Concrete Samples

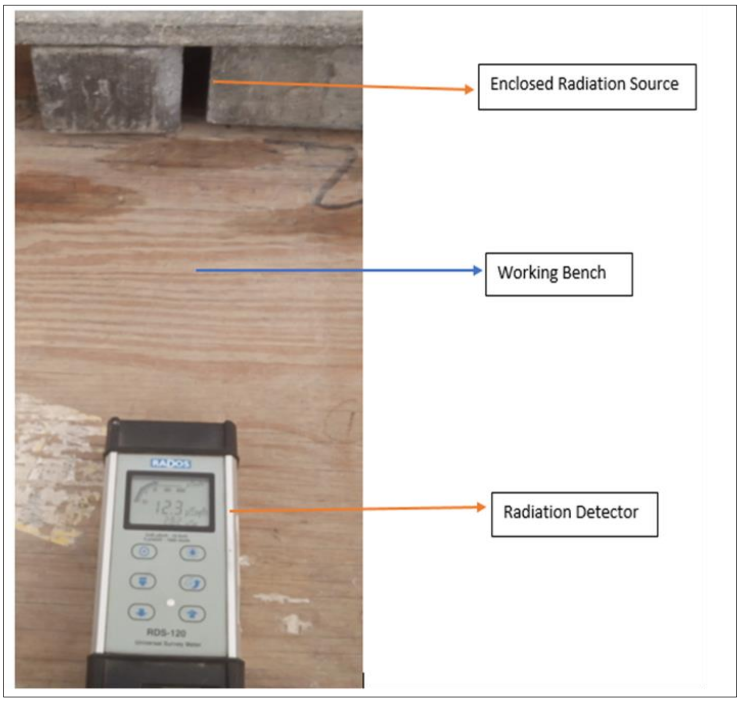

Figure 4 Source-Detector Configuration

The fabricated concrete blocks were transported to Centre for Energy Research and Training for linear attenuation coefficient experiment. To achieve this, the source-detector distance was chosen and kept constant at 30cm (Figure 4) while source-sample and sample-detector distance was also kept constant at about 15cm (Figure 5). The background dose rate of the laboratory was recorded as B$\dot{D}$ (the same was subtracted from every dose rate measurement taken when the source was opened and when the concrete samples were inserted). The source enclosure was then removed and the dose rate was measured and recorded as $\dot{D}o$. A concrete block labelled A_1, was inserted between the detector and a collimated beam of gamma radiation from cobalt-60 source Figure 3 and the value of dose rate $\dot{D}_1$ was taken from the detector display five times and an average was computed and recorded. The same measurements were obtained for all other concrete samples and the results were tabulated. These same measurements were also carried out using cesium-137 gamma source and Am-Be neutron source (Figure 4). The linear attenuation coefficients were obtained graphically using Equation 1. A graph of $\ln(\frac{\dot{D}}{D_o})$ versus 'x' is linear with a slope equals to 'μ'. All the terms retained their meaning as defined in this work.

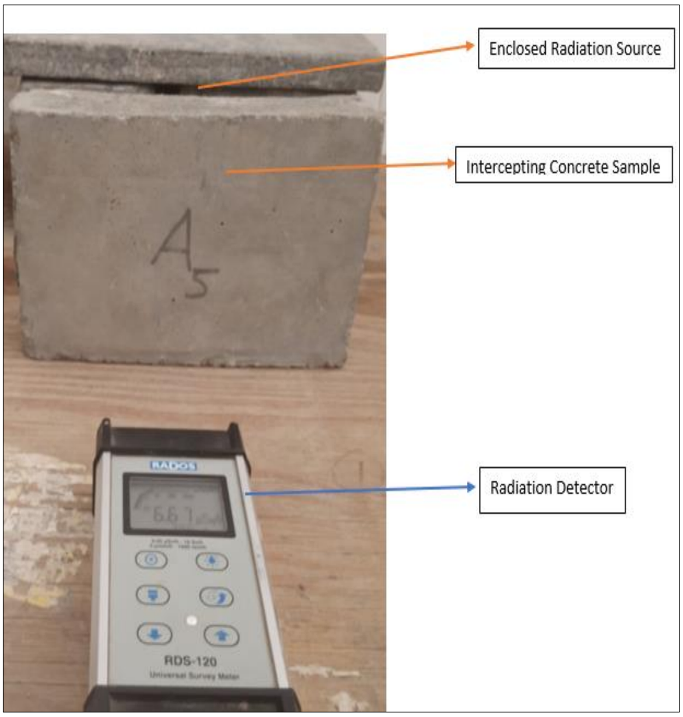

Figure 5 Source-Concrete Sample-Detector Configuration

3 Results and discussion

Linear attenuation coefficient μ (cm^{-1}) which is the fractional decrease in the photon intensity per unit path length in the absorber depends strongly on the physical density of the absorber. This parameter is useful in radiation dosimetry and radiation protection in order to have a precise knowledge of gamma ray photon attenuation and consequent absorption [11]. The graph of $ln\ (\frac{D}{D_o})$ against the.thickness (cm) is a linear graph whose slope represents the attenuation coefficients (Simon et al., 2019). The simulated results is presented in Figure 6 to 8 presents the graphs of $ln\ (\frac{D}{D_o})$ against the thickness, x, (cm) for Cs-137, Co-60 and Pu-239. The slope of the graphs (the linear attenuation coefficients μ) were obtained from the graphs as $0.142 \pm 0.005cm^{-1}$, $0.104 \pm 0.003cm^{-1}$ and $0.471 \pm 0.002cm^{-1}$ for Cs-137, Co-60 and Pu-239 respectively.

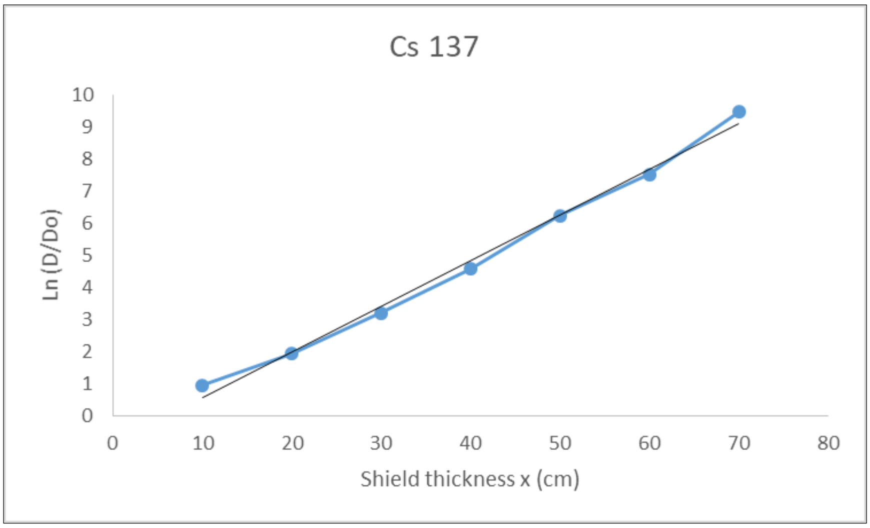

Figure 6 Graph of $\boldsymbol{ln}\ (\frac{D}{D_o})$ against the thickness (cm) for Cs-137

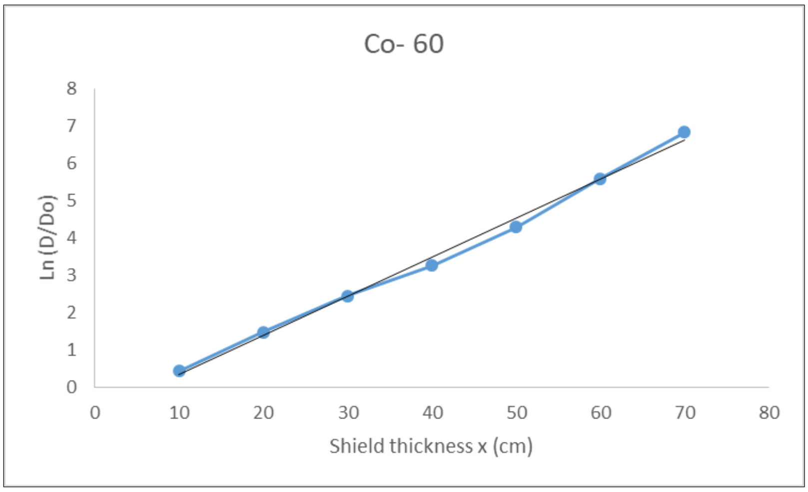

Figure 7 Graph of $\boldsymbol{ln}\ (\frac{D}{D_o})$ Against the Thickness (cm) for Co-60

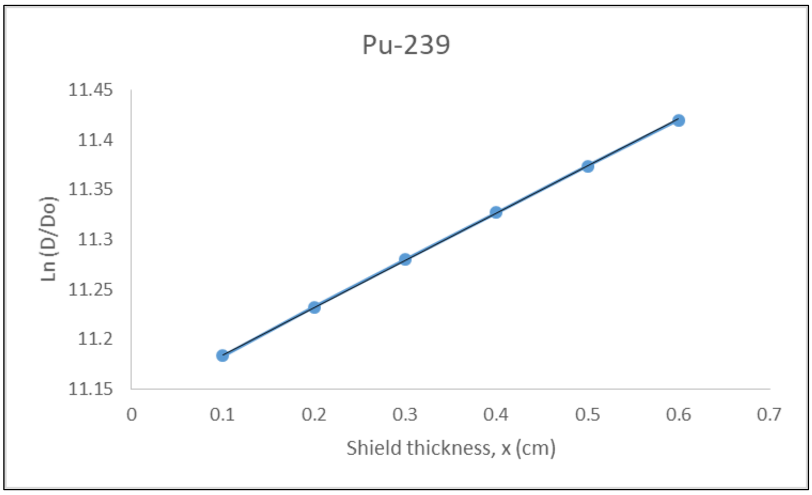

Figure 8 Graph of $\boldsymbol{ln}\ (\frac{D}{D_o})$ against the thickness (cm) for Pu-239

The linear attenuation coefficients of the fabricated concretes were determined from Figure 9 to 11 from the plots of measured dose rates versus thicknesses of the concretes. The obtained linear attenuation coefficient of the serpentine concrete with aggregate sizes 5 mm, 10 mm, 15 mm, and 20 mm respectively against Cs-137 gamma source, Co-60 gamma source and Am-Be neutron sources are presented in Table 1.

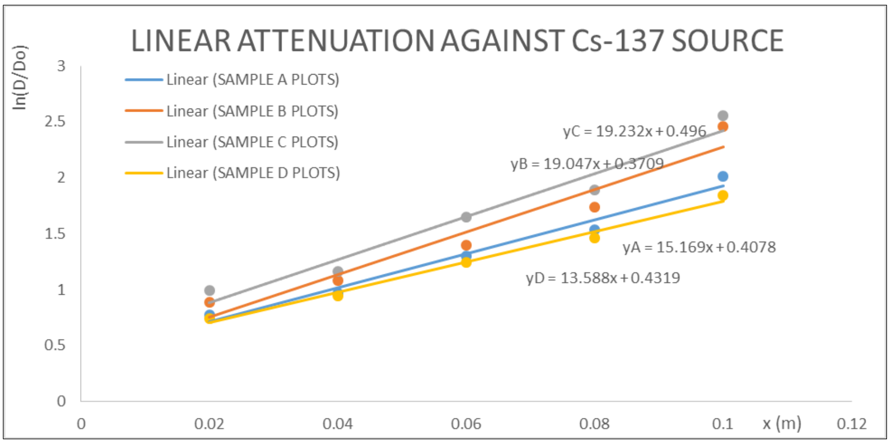

Figure 9 Determination of attention coefficient μ(m^{-1}) of Serpentine Concrete Samples for Cs-137 Gamma Source

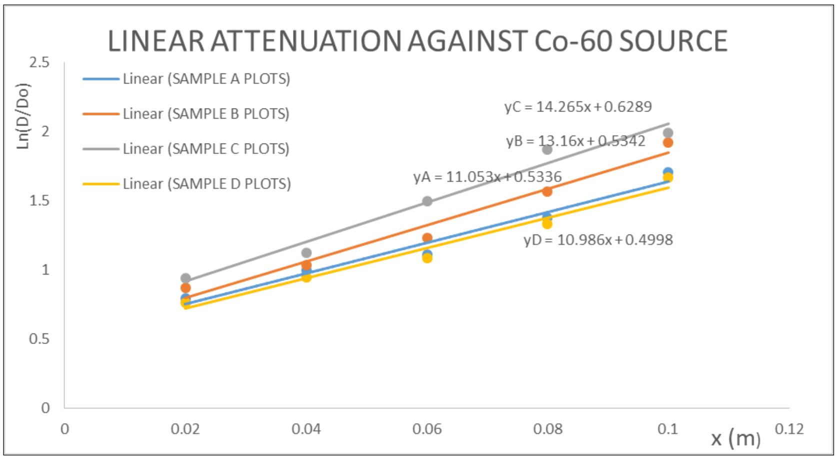

Figure 10 Determination of attention coefficient μ (m^{-1}) of Serpentine Concrete Samples for Co-60 Gamma Source

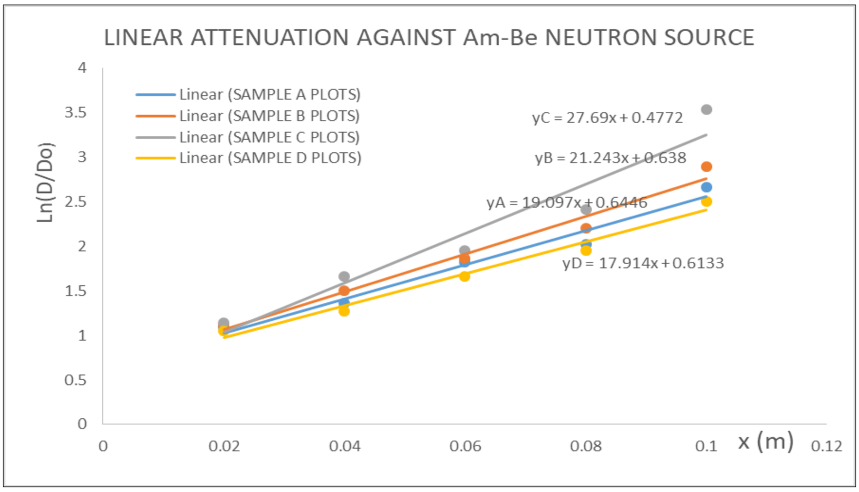

Figure 11 Determination of attention coefficient μ(m^{-1}) of Serpentine Concrete Samples for Am-Be Neutron Source

Table 1 Attenuation Coefficients of Fabricated Concrete Samples

Concrete Sample ID	Sample Aggregate Sizes (mm)	Radiation Attenuation	Cs-137 source	Co-60 source	Am-Be Source
A	5.0	$\bar{\mu}$ (cm^{-1})	0.15 ± 0.03	0.11 ± 0.02	0.19 ± 0.03
B	10.0	$\bar{\mu}$ (cm^{-1})	0.19 ± 0.03	0.13 ± 0.02	0.21 ± 0.03
C	15.0	$\bar{\mu}$ (cm^{-1})	0.19 ± 0.03	0.14 ± 0.02	0.28 ± 0.03
D	20.0	$\bar{\mu}$ (cm^{-1})	0.14 ± 0.03	0.11 ± 0.02	0.18 ± 0.03
SIMULATION	-	$\bar{\mu}$ (cm^{-1})	0.142 ± 0.03	0.104 ± 0.03	0.471 ± 0.03

From the values of linear attenuation coefficients presented in Table 1, an increase in shielding performance of the concrete was observed with increase in aggregate size. But this trend dropped significantly past 15 mm size of aggregate. This observation was similar to that reported by [12] using granite aggregates. This shows that an optimized shielding will be obtained when the transportation cask is fabricated with 10 mm or 15 mm Serpentinide aggregate sizes. It was also observed (Table 1), that very high linear attenuation property was observed when the serpentine concrete samples were exposed to a neutron source. This buttress the fact that Serpentinide rock is a Boron rich mineral and Boron is known to have high neutron absorption property. Hence, Serpentine concrete is a good shielding material for design of neutron source transportation cask. A comparison of simulation and experimental determination of linear attenuation coefficients is presented in Table 1, and the bias between the simulation and experimental data was found to be extremely small as the data demonstrated a high correlation.

4 Conclusion

The linear attenuation coefficients of Serpentine concrete were determined by simulation using Radpro computer software and experimentally using a radiation detector for benchmarking. The experimental determination of the linear attenuation coefficients was carried out at four different granule sizes of the Serpentine aggregate used in the fabrication of the concrete samples. The calculations of the linear attenuation coefficients are necessary for the design of a concrete cask for the transportation of Gamma and Neutron radioactive legacy sources at Ajeokuta steel company limited. The linear attenuation coefficients obtained by simulation were found to have high precision with the experimentally determined values. Very high linear attenuation property was observed when the serpentine concrete samples were exposed to a neutron source, which buttresses the fact that Serpentine rock is a Boron-rich mineral and Boron is known to have high neutron absorption property. The experimentally determined linear attenuation coefficients showed that the values at 15 mm aggregate sizes were higher than those at 5 mm, 10 mm and 15 mm sizes respectively, which demonstrates that better shielding optimization will be obtained when the concrete cask is fabricated with 15 mm aggregate size.

Compliance with ethical standards

Acknowledgments

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