Fabrication and Characterization of Shielding Properties of Heavy Mineral Reinforced Polymer Composite Materials for Radiation Protection

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Abstract—Heavy mineral and unsaturated polyester resin (UPR) based composite blocks were prepared for potential shielding of ionizing radiations. Locally available heavy minerals with Ilmenite, Magnetite, Garnet, Rutile, Zirconium contents were used to fabricate the composite blocks for the gamma photons with energies 0.662 MeV - 1.25 MeV. The shielding capacity was evaluated in terms of Half Value Layer (HVL), Tenth Value Layer (TVL), Sixteenth Value Layer (SVL), Linear attenuation coefficient, Mass attenuation coefficient, and reduced % of radiation intensity. The Ilmenite composite exhibits relatively good attenuation performance in the case of 0.662 MeV photons of Cs-137. On the other hand, Zirconium composite demonstrates relatively good attenuation capacity in the case of 1.25 MeV photons of Co-60 in comparison to the ordinary concrete block. The goal of this work is to explore some novel materials to be effectively used as gamma shielding options in radiation facilities at minimal cost.

Index Terms—Shielding, Composite, Half value layer, Linear attenuation coefficient, Mass attenuation coefficient.

I. INTRODUCTION

Ionizing radiation poses significant hazards to living organisms although it has some good applications in socioeconomic development [1]. In a radiation facility, the occupational workers and the resident member of the public are potentially get exposed intentionally or unintentionally due to the wide application of nuclear science and technology in various fields such as nuclear power plant, nuclear research reactor, medical centers, industries, laboratories and research facilities. [2-4]. To diminish this hazard, radiation shielding can conduct a key role in order to ensure a safe environment for the radiation workers. Therefore, adequate and effective shielding is a prerequisite for the installation of such nuclear facilities to keep the radiation exposure below the dose limits recommended by the International Commission on Radiological Protection (ICRP) [5-7]. In recent years, numerous studies have been carried out on shielding materials, especially by using nanotechnology in cement-based materials. Most of these shields are made from different types of concretes, such as ordinary concretes, heavy concretes (e.g., barite, serpentine, steel, magnetite, ilmenite, etc.) [8-11]. Depending on the energy, type of radiation, availability of the shielding material, considering the state of technology, the economics of reducing exposures relative to the benefits to be achieved and other relevant socioeconomic factors. The heavy concretes proved themselves as the best suitable materials for the attenuation of gamma radiation and neutron particle shielding [12-16]. Although several investigators have worked on the shielding properties of concretes and building materials, however, limited research works focused on the evaluation of the shielding properties polymer based composites materials. Now-a-days, combination of polymer and heavy mineral-based composites has become a versatile solution to reduce exposure for both gamma and neutron radiation [17-19]. The objective of this study is to develop heavy mineral reinforced polymer composite materials for radiation shielding as well as generate a data base of attenuation parameters for these materials to facilitate the shielding design events.

II. MATERIALS AND METHODS

Collimated radiation beam of Cs-137 gamma radiation sources was used to estimate the attenuation properties of different types of composite samples such as Zirconium composite, Ilmenite composite, Rutile composite, Garnet composite, and their mixed composites. In this work six types’ polymer concrete blocks were used to investigate their shielding potential with respect to the evaluation of their radiation attenuation capacity for both relatively low and high energy gamma radiation. A Geiger Müller type radiation measuring detector was used to investigate the radiation attenuation capacity of the aforesaid sample materials. The relative low energetic photons of 0.661 MeV from Cs-137 beam outputs were found to be attenuated significantly with the studied sample materials in comparison to the open beam output. The studied samples attenuated the radiation beam due to absorption of the photon beam in the sample matrix. The attenuation capacity of the same studied materials for relative high energetic photons was verified as well with 1.25 MeV gamma photons from Co-60 beam outputs. The intensity of the radiation beam would be attenuated according to the Beer-Lambert’s law when gamma-ray beam traverses an absorber. In present experiment, the attenuation of the transmitted gamma photon intensity through the absorbing materials is
described by this law as:

\[ I = I_0 e^{-\mu t} \]  

(1)

Where \( I_0 \) and \( I \) are the non-attenuated and attenuated gamma ray beam intensities, \( \mu \) (cm\(^{-1}\)) is the linear attenuation coefficient and \( t \) is the thickness of the material (cm). In this study, \( I_0 \) indicates the initial open beam output. The determined linear attenuation coefficient could be useful to determine the mass attenuation coefficients \( \mu / \rho \) \( (\text{cm}^2/\text{g}) \) by applying the bulk densities of the respective samples as follows:

\[ \mu = \frac{1}{\rho} \ln \left( \frac{I_0}{I} \right) \]  

(2)

The linear attenuation coefficient reflects the removal of photons from a radiation beam by interaction with electrons of the sample material. The higher the electron density, the more interaction of gamma photons with the sample material occurs. These interactions can cause the absorption of the photons (i.e., removal from the beam) or scattering (i.e., change of direction with reduction in energy). Therefore, it seems appropriate to scale the linear attenuation coefficient with the sample density. The linear attenuation coefficient can also be rewritten as:

\[ \mu_m = \left( \frac{\mu}{\rho} \right) \rho \]  

(3)

Where \( \frac{\mu}{\rho} \) is the mass attenuation coefficient (cm\(^2\)/g) and \( \rho \) is the density (g/cm\(^3\)). The mass attenuation coefficient is approximately constant for different materials in a specified energy range, and therefore the linear attenuation coefficient is strongly determined on density. The linear attenuation coefficient is also strongly energy dependent. In general, lower energetic gamma photons have a higher interaction probability, and hence cause relatively high attenuation. In this study, photon beam transmission was considered in a moderate energy range of 0.662 – 1.25 MeV to verify the shielding applicability of the studied samples in the same energy range.

The half value layers (HVL) of the studied sample materials were estimated based on the experimental observations. The HVL reduces the radiation level by a factor of 2 that is to half the initial level, which is mathematically defined as:

\[ \text{HVL} = \frac{0.693}{\mu} \]  

(4)

The tenth value layers (TVL) of the studied sample materials were estimated as well based on the experimental observations. A shield that would attenuate a radiation beam to 10% of its radiation level is called TVL, which is mathematically presented as:

\[ \text{TVL} = \frac{2.303}{\mu} \]  

(5)

In a similar technique, the sixteenth value layers (SVL) of the studied samples were estimated by using the following equation.

\[ \text{SVL} = \frac{2.773}{\mu} \]  

(6)

The mass attenuation coefficient of a particular shielding material would be useful to determine the relaxation length for a certain photon beam based on the following expression:

\[ \lambda = \frac{1}{\mu} \]  

(7)

III. EXPERIMENTAL DETAILS

3.1 SAMPLE PREPARATION

The studied samples were prepared with the composition of various locally available cost effective materials. Naturally occurring heavy minerals and dense gravel materials are preferred to fabricate the shielding blocks in order to check its attenuation property of gamma radiation. In this perspective, various heavy mineral sand, such as, Magnetite, Ilmenite, Rutile, Garnet and Zirconium was used to make the shielding blocks. The unsaturated polyester resin (UPR) was used as the binder of the heavy mineral sand and gravels.

In the first step of sample preparation, the mixed aggregates were placed in a concrete mixer container. The aggregates of heavy mineral, ordinary sand and stone chunks were mixed with a ratio of 75:75:300, respectively. After preparing the mixed aggregate, the UPR binder liquid was added uniformly by concrete mixer machine. For each sample Methyl ethyl ketone peroxide (MEKP) was used as a cross linker to settle down the UPR and mixed aggregates, except ordinary concrete sample. The MEKP and UPR were used with a ratio of 2%-98% by their weight.

The polymer-based effective radiation attenuating composite shielding samples were fabricated with heavy mineral sand in combination of UPR by concrete mix design process. The flow diagram of sample preparation is presented in Figure 1. The ratio for the concrete mix design was 18:15:15:52 in accordance to the civil engineering standard for UPR, heavy mineral, ordinary sand and stone chunks respectively. Subsequently, the mixed aggregates were poured into a molding box, and stirred for their close-fitting and settling. After that the molding box was stored in a cool place to settle down the concrete for twenty-four hour. For the convenience of experiments, the dimension of the molding box was relatively small \((5"\times5"\times1.5")\) to prepare handy samples.

In this study, six types of concrete block were fabricated as shown in Figure 2. The fabricated samples were used in a series of experiments with Cs-137 and Co-60 radiation sources at the secondary standard dosimeter laboratory.
3.2. EXPERIMENTAL PROCESS

In the present experiments the collimated beams from two radiation sources of Cesium-137 and Cobalt-60 gamma radiation sources were used to investigate the radiation attenuation capacity of the prepared samples. The open radiation beam data was recorded without placing any sample on the collimator. Then, the attenuated radiation beam data was recorded for all the prepared samples. Fabricated composite blocks were placed in front of the collimator of the gamma ray emitting calibrator for experimental evaluation of its attenuation capacity, as shown in Figure 3. The diversity in radiation attenuation performance of the composite block was justified as per the variation in material compositions of respective sample. The breakthrough result was further analyzed numerically to determine the shielding performance parameters to evaluate the shielding capacity of the studied samples. The shielding capacity were evaluated in terms of HVL, TVL, SVL, linear attenuation coefficient, mass attenuation coefficient, and reduced % of radiation intensity. In addition, a comparative assessment was performed between the characterized composite shielding materials and the conventional shielding material based on the experimental result.

After determining the densities of the aforesaid composite blocks, their linear attenuation coefficient $\mu_l$, mass-attenuation coefficient $\mu_m$, HVL, TVL, SVL, and reduced percentage of beam transmission was determined, and presented in Table-1 and Table 2 for Cs-137 and Co-60 source respectively.

The variation of the experimental $\mu_l$ and $\mu_m$ were mainly due to their dependence on the respective sample density. The comparative evaluations of radiation attenuation capacity in terms of $\mu_l$ between the fabricated shielding material and the conventional shielding material for gamma photons of Cs-137 and Co-60 are shown in Figure 4. A similar comparison of $\mu_m$ is presented in Figure 5. The estimated experimental values of $\mu_l$ and $\mu_m$ were cross checked by numerical software RADPro, and reasonably good agreement was found.

IV. RESULTS AND DISCUSSION

The bulk density of the fabricated samples was determined prior to the subsequent experimental evaluations. The density profile is presented in Figure 4, which indicates a small density variation of different among the heavy mineral composite blocks. The highest density of 3.1 (g/cm3) was attained in the case of Garnet composite block. Whereas, the density profile was found same in the cases of 50% magnetite containing sand-based, and Zirconium composite blocks. As different heavy mineral based composite blocks exhibited different unit density, hence the fabricated polymer composite blocks was found within the density range of 2.85 to 3.1 g/cm3. The observed density profile of the studied samples was found relatively high than the density 2.3 g/cm3 of the ordinary concrete block. Thus, this figure evidently indicates that the studied samples of the heavy mineral composites had relatively high density in comparison to the ordinary concrete block.
The comparative evaluations of radiation attenuation capacity in terms of μ₁ between the fabricated shielding material and the conventional shielding material for gamma photons of Cs-137 and Co-60 are shown in Figure 5. A similar comparison of μ₁ was presented in Figure 6. The estimated experimental values of μ₁ and μₘ₁ were cross checked by numerical software RADPro, and reasonably good agreement was found. In the current experiments, detector was placed relatively at distant position (i.e., 1 meter) from the sample so as to ensure the detection of the attenuated secondary beam that escaped after the interaction with the sample. In the experiments with Cs-137 and Co-60 gamma radiation sources, the μ₁ and μₘ₁ of all the studied heavy mineral composite blocks are found significantly larger than that of the ordinary concrete block, as shown in Figure 5 and Figure 6 respectively. The highest values of μ₁ and μₘ₁ were found to be 0.221 cm⁻¹ and 0.074 cm²/gm for the Ilmenite composite block with Cs-137 source. For other samples, μ₁ and μₘ₁ were found closely similar, where ordinary concrete block showed very poor results. It is obvious that the interaction at lower energy photon is much dominant than the higher energy, and hence the attenuation coefficient was relatively decreased with the increase in photon energy from 0.662 MeV (Cs-137) to 1.25 MeV (Co-60). The μ₁ and μₘ₁ for heavy mineral composite blocks were found relatively larger than the ordinary concrete blocks as shown in Figures 5 and Figures 6 respectively. The evaluated μ₁ and μₘ₁ were found closely similar for every composite block with both radiation emitting source, and they were around 0.20 cm⁻¹ and 0.06 cm²/g for Cs-137 source, whereas for Co-60 source, μ₁ and μₘ₁ were around 0.16cm⁻¹ and 0.05 cm²/g. On the other hand, the ordinary concrete block showed very poor radiation attenuation behavior in terms of lower values of μ₁ and μₘ₁. The present study illustrate that the shielding effectiveness of any shielding material depends on its density, the types of chemical composition and the

### Table I: AVERAGE VALUES OF DIFFERENT SHIELDING PARAMETERS FOR Cs-137 RADIATION SOURCE

| Sample name               | Thickness (cm) | Mass (g) | Density (g/cm³) | Initial Radiation Intensity I₀ | Final Radiation Intensity I | Reduced Radiation Intensity (%) | M₁ (cm⁻¹) | Mₘ₁ (cm²/g) | HVL (cm) | TVL (cm) | SVL (cm) |
|---------------------------|----------------|----------|-----------------|-------------------------------|-----------------------------|--------------------------------|------------|-------------|---------|---------|---------|
| Magnetite composite       | 3.81           | 1890     | 3.07            | 57                            | 26.5                        | 53.5                           | 0.201      | 0.067       | 3.49    | 11.26   | 12.55   |
| Ilmenite composite        | 3.81           | 1840     | 2.99            | 57                            | 24.5                        | 57.5                           | 0.221      | 0.074       | 3.13    | 10.42   | 13.8    |
| Rutile composite          | 3.81           | 1752     | 2.85            | 57                            | 27.3                        | 52.1                           | 0.193      | 0.068       | 3.59    | 11.93   | 14.37   |
| Garnet composite          | 3.81           | 1845     | 3.01            | 57                            | 25.37                       | 55.5                           | 0.212      | 0.067       | 3.27    | 10.86   | 13.08   |
| Zirconium composite       | 3.81           | 1888     | 3.07            | 57                            | 26                          | 54.9                           | 0.206      | 0.067       | 3.26    | 11.18   | 13.46   |
| Ordinary concrete         | 3.81           | 1415     | 2.3             | 57                            | 46.43                       | 18.01                          | 0.052      | 0.023       | 13.33   | 44.28   | 53.32   |

### Table II: AVERAGE VALUES DIFFERENT SHIELDING PARAMETERS FOR Co-60 RADIATION SOURCE

| Sample name               | Thickness (cm) | Mass (g) | Density (g/cm³) | Initial Radiation Intensity I₀ | Final Radiation Intensity I | Reduced Radiation Intensity (%) | M₁ (cm⁻¹) | Mₘ₁ (cm²/g) | HVL (cm) | TVL (cm) | SVL (cm) |
|---------------------------|----------------|----------|-----------------|-------------------------------|-----------------------------|--------------------------------|------------|-------------|---------|---------|---------|
| Magnetite composite       | 3.81           | 1890     | 3.07            | 8.03                          | 4.03                        | 49.8                           | 0.181      | 0.057       | 3.83    | 12.72   | 15.30   |
| Ilmenite composite        | 3.81           | 1840     | 2.99            | 8.03                          | 4.45                        | 44.6                           | 0.161      | 0.054       | 4.30    | 14.30   | 17.20   |
| Rutile composite          | 3.81           | 1752     | 2.85            | 8.03                          | 4.62                        | 42.5                           | 0.150      | 0.053       | 4.62    | 15.35   | 18.50   |
| Garnet composite          | 3.81           | 1845     | 3.01            | 8.03                          | 4.42                        | 45.0                           | 0.160      | 0.053       | 4.33    | 14.40   | 17.30   |
| Zirconium composite       | 3.81           | 1888     | 3.07            | 8.03                          | 4.01                        | 48.9                           | 0.182      | 0.057       | 3.80    | 12.65   | 15.20   |
| Ordinary concrete         | 3.81           | 1415     | 2.3             | 8.03                          | 6.75                        | 16.00                          | 0.046      | 0.020       | 15.10   | 50.06   | 60.27   |

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concentration of the elements that it contains.

To design and select an appropriate shielding material, all the nuclear parameters associated with it needs to be studied thoroughly.

![Graph](image)

Fig. 5. Linear attenuation coefficients of various composite samples for Cs-137 and Co-60 source.

![Table](image)

Fig. 6. Comparative view of mass attenuation coefficients of various composite samples for Cs-137 and Co-60 source.

Figure 7 demonstrates the competence of our prepared shielding blocks. It shows the significant differences in necessity of shielding thickness. It was observed much better result for all type of heavy mineral containing sand-based composite and the HVL and TVL was lies between 3.13 to 4 cm and 11 cm to 15 cm respectively for both Cs-137 and Co-60 source. Whereas the ordinary concretes had the HVL and TVL values of 13 cm to and 60 cm respectively for both radiations emitting source. This graph shows effectiveness of heavy mineral concrete with respect to ordinary concrete.

![Graph](image)

Fig. 7. Comparative evaluations of experimental HVL, TVL and SVL for the fabricated samples.

V. CONCLUSION

The polymer-based effective radiation attenuating composite shielding samples were fabricated in combination of heavy mineral sand and UPR. These heavy mineral reinforced polymer composite materials exhibited relatively good radiation shielding capacity in comparison to the ordinary concrete block. Fabricated composite materials would be a good option for cost effective and less space consuming shielding material against gamma radiation. The shielding competence of Ilimenite and Zirconium composites was found relatively good for Cs-137 and Co-60 radiation source respectively. Thus, they could be used as a biological shield for the medical, industrial, radiation laboratories as well as commercial gamma ray irradiation facilities. The technical database of this study would be useful to explore further novel materials for gamma shielding options in radiation facilities at minimal cost.

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