Study on the shielding materials for low-energy gamma sources

A H Aminordin Sabri¹,a), M Z Abdul Aziz², S F Olukotun³, F Tabbakh⁴, S M Tajudin¹

¹Faculty of Health Sciences, Universiti Sultan Zainal Abidin (UniSZA), Terengganu, Malaysia.
²Oncological and Radiological Science Cluster, Advance Medical and Dental Institute, Universiti Sains Malaysia, Penang, Malaysia.
³Department of Physics and Engineering Physics, Obafemi Awolowo University, Ile-Ife, Nigeria.
⁴Nuclear Science and Technology Research Institute, Tehran, Iran.

a)adilahanim@unisza.edu.my

Abstract. Photon dosimetry is indispensable in designing an irradiation facilities shielding. Monte Carlo simulation was utilized to investigate the efficiency of relatively new developed clay and Gadolinum (Gd)-doped polymer as a radiation shielding material for low energy gamma sources (Am-241 and Co-57). The calculated linear attenuation coefficient (µ) of Am-241 and Co-57 for clay is higher within 6.6 % and 0.9 % compared to ordinary concrete, respectively. The µ value for Gd-doped polymer is higher by a factor of 9 and 5 compared to clay for Am-241 and Co-57, respectively. A thickness of 2 cm and 5 cm from both clay and concrete were adequate to attenuate almost 90 % incident photons from Am-241 and Co-57, respectively. The same thickness of 2 cm by Gd-doped polymer could attenuate almost 95 % of Co-57 photons. 3 cm thickness of clay and concrete could shield the gamma source dose rate of Am-241 (1 MBq) down to 0.05 µSv/hr, while almost 10 cm needed for Co-57 source. Gd-doped polymer with thickness of 2 cm could shield almost 94 % the dose rate from Co-57 source. For higher energy gamma sources, clay and ordinary concrete need to be doped with a higher Z element to ensure safety of the radiation.

1. Introduction
Photon dosimetry is indispensable in designing an irradiation facilities shielding. For example, lead (Pb-82) material and ordinary concrete in the order of millimeter (mm) and centimeter (cm) thickness, respectively had been widely used to attenuate the photons, either used as a shielding material for the wall thickness of an irradiation facility room [1][2], such as our radiotherapy room (150 keV) or to shield the emission of photons from a gamma source in a source pig.

The development and properties of non-lead and non-concrete material as a radiation-shielding material have been studied previously by many researchers [3-10]. Lead material already known for its toxicity, expense and limitation in abundance [11-13]. For example, Olukotun et al. proposed the use of
clay and kaolin as an effective shielding material for photons [9], Tabbakh studied carbohydrate based compounds as new shielding materials [8] for gamma radiation shielding, while Akkurt and Canakci doped the clay with boron as a shielding material for higher-energy gamma sources [14]. Previous studies conducted by Kaura et al. and Agar et al. were proposed the use of metallic alloy as a potential gamma-shielding material by doping with a high atomic number (Z) element [15][16].

The primary concern in many cases of photon shielding studied was for the transmitted photons, both scattered and unscattered photons that penetrate the shielding materials. It was depends on the point at which the quantity of interest will be calculated or measured from the attenuating material. As example, Sayyed et al. had studied the attenuation properties for selected germanate glasses based shielding compound through theoretical calculation [17]. Another studies of radiation attenuation had been performed by theoretical calculation for various concretes [18] and by experimental work for glass based shielding compound [19][20]. In addition to attenuation coefficient (µ) and its related parameters, it is necessary to evaluate intensity transmitted or reflected due to the shielding material, particularly for a newly developed shielding material and photon energy (keV). As we need to ensure the radiation dose rate is as low as possible as per background radiation, such evaluation is important when designing a radiation shielding facility or for source storage. Thus, we conduct the study of our relatively new developed clay [10] and Gadolinium (Gd)-doped polymer [28] as well as ordinary concrete [27] as a radiation shielding material for low energy gamma sources (Am-241 and Co-57).

In this study, the transmitted photons and its ambient dose equivalent (Sv/photon) of the clay and Gadolinium (Gd)-doped polymer materials were evaluated by Monte Carlo simulation (EGS5 code [21][22]). These compound as a shielding material will be optimized by calculating for photon (gamma) energy below 150 keV as based on our irradiation facility to have an Am-241 gamma source at UniSZA, which has a maximum setting of 150 keV. In the first case, photon dosimetry using clay and Gadolinium (Gd)-doped polymer materials were calculated for gamma ray of low-energy source such as Am-241 (16.1 keV, 26.3 keV, and 59.5 keV photons). In the latter case, we include the calculation for gamma ray of Co-57 (14.4 keV, 122 keV, and 136 keV photons).

2. Calculated transmitted photons for multienergy gamma sources

Previously in our study, clay had been successfully demonstrated to decrease transmitted photon compared to ordinary concrete for low-incident photon energy because of dominant photoelectric absorption [9]. A pencil beam of photons was incident to the center of cylinder clay. The photon’s interaction, such as pair productions, Compton scattering (including Rayleigh scattering) and photoelectric absorption were considered in our calculation. However, Compton scattered photons that penetrate the clay-shielding material were ignored for comparison with the attenuation coefficients generated by XCOM [23]. Figure 1 shows the calculation of transmitted photons as a function of ball clay thickness for a pencil beam of 150 keV incident photon. The full square points were fitted with an exponential function to obtain linear attenuation coefficient (µ) values of 1.3892 cm⁻¹. The calculated µ value has agreed with the theory, within 1.06% to justify the subsequent calculations. The error of 0.23% shown in the figure for the calculated µ value comes from exponential fitting error.
For the latter point, $\mu$ value was calculated for multi-energies gamma source of Am-241. The branching ratio of the gamma source was sampled according to JRIA data book [24]. Table 1 shows the gamma energies and its corresponding branching ratio adopted in the calculation.

From Table 2, the calculated linear attenuation coefficient ($\mu$) of Am-241 and Co-57 for clay is higher within 6.6 % and 0.9 % compared to ordinary concrete, respectively. The $\mu$ value for Gd-doped polymer is higher by a factor of 9 and 5 compared to clay for Am-241 and Co-57, respectively. A thickness of 2 cm clay is adequate to attenuate up to 90% of the photons from Am-241 source. The calculated $\mu$ value is $0.7527 \text{ cm}^{-1}$, which is higher, within 5.47%, than the theoretical values of 59.5 keV (0.7126 cm$^{-1}$). The amount of difference was as expected, as the Am-241 source has lower gamma energies of 16.1 and 26.3 keV photons, which are not considered in the theoretical value.

Meanwhile, a thickness of 2.0 cm Gadolinum (Gd)-doped polymer is adequate to attenuate up to 95% of the photons from Co-57 source. The calculated $\mu$ value is $1.5601 \text{ cm}^{-1}$, which is higher, within 13.67%, than the theoretical values for 122 keV. A thickness of 6 cm clay is adequate to attenuate up to 90% of the photons from Co-57 source. The calculated $\mu$ value is $0.3304 \text{ cm}^{-1}$, has a good agreement, within 1.59% with the theory.
4

Table 1. The list of gamma energies and its branching ratio [24].

| Source | Energy (MeV) | Branching ratio (%) |
|--------|--------------|---------------------|
|        | 0.0263       | 2.4                 |
| Am-241 | 0.0595       | 35.9                |
|        | 0.0161       | 37.9                |
|        | 0.0144       | 9.2                 |
| Co-57  | 0.122        | 85.6                |
|        | 0.136        | 10.7                |

Table 2. The calculated linear attenuation coefficient (µ) cm\(^{-1}\) for gamma sources.

| Sources | Calculated Linear Attenuation Coefficient (µ) cm\(^{-1}\) |
|---------|----------------------------------------------------------|
|         | Ordinary concrete [27] Clay [10] Gd-Doped Polymer [28] |
| Am-241  | 0.7043 0.7527 6.9471                                     |
| Co-57   | 0.3274 0.3304 1.5601                                     |

3. Calculated ambient dose equivalent for multi-energy gamma source

Calculation of ambient photon dose (Sv/photon) becomes important for the public and radiation workers, as the radiation must be estimated for the radiation safety area. It is necessary to ensure that the dose rate from the gamma source attenuated down to background radiation level for radiation safety, particularly if the storage location of the gamma source is easily assessable. As an example, the measured background dose rate at our X-ray laboratory is 0.055 ± 0.008 mSv/hr. Ambient dose equivalent is one of the dose unit as an operational quantity by ICRU [25] body for radiation protection. For example, an environmental dosimeter or ionization chamber survey meter was calibrated to measure ambient dose in the unit of Sievert (Sv) to read the photon dose equivalent at 1 cm depth (H*(10)). In such a case, the calculated and measured values of photon dose rate are comparable in absolute values [26].

Table 3. The calculated and theoretical dose rates for 1MBq gamma sources.

| Source | 1 MBq Dose Rate (µSv/hr) at 20 cm | % of difference |
|--------|-----------------------------------|-----------------|
|        | Calculated | Theoretical |               |
| Am-241 | 0.3796     | 0.3775      | 0.54           |
| Co-57  | 0.5552     | 0.5650      | 1.72           |

Surface crossing was used to calculate the ambient dose on the surface of 2 cm x 2 cm scoring region at a distance of 20 cm from the gamma source in air. In this part of the calculation, the solid angle of the source photons was 4π that was isotropically incident to the clay. To obtain ambient dose equivalent, air absorbed dose (Gy) was calculated as a first step by using kerma approximation, which calculates collision kerma by multiplying mass energy absorption coefficient to the energy fluence. The following conversions were used to convert the unit of MeV g\(^{-1}\) to Gy;
1 MeV = 1.602 x 10^{-13} J
1 MeV/g = 1.602 x 10^{-13} (J/MeV) x 1000(g/kg) = 1.602 x 10^{-10} Gy

A function was adopted in the calculation to evaluate the ratio of ambient dose equivalent (Sv) to air absorbed dose. Then the ambient dose equivalent was calculated by multiplying the ratio of ambient dose to air absorbed dose. As a first step, the calculated ambient dose rate without the clay was compared to the theory 1 cm depth dose rate at a distance of 20 cm for Am-241. For the theoretical dose rate, the dose rate conversion factor ($\mu$Sv.m$^{-2}$.MBq$^{-1}$.hr$^{-1}$) for above 10 keV was obtained from radioisotope data book [24]. The constant value for Am-241 at 1-meter distance is 0.0151 $\mu$Sv.m$^{-2}$.MBq$^{-1}$.hr$^{-1}$. Table 3 shows the result of calculated dose rate compared to the theoretical dose rate ($\mu$Sv/hr) at a distance of 20 cm.

Figure 2 (a) and (b) shows the calculated dose rate at a distance of 20 cm for 1 MBq of Am-241 and Co-57 sources as a function of clay, ordinary concrete and Gadolinium (Gd)-doped polymer thickness. From both graphs, a thickness of 2 cm and 5 cm from both clay and concrete were adequate to attenuate almost 90% incident photons from Am-241 and Co-57, respectively. The same thickness of 2 cm by Gd-doped polymer could attenuate almost 95% of Co-57 photons. 3 cm thickness of clay and concrete could shield the gamma source dose rate of Am-241 (1 MBq) down to 0.05 $\mu$Sv/hr, while almost 10 cm needed for Co-57 source. Gd-doped polymer with thickness of 2 cm could shield almost 94% the dose rate from Co-57 source, which could be considered a safe value as per our background radiation value.

![Figure 2](image_url)

**Figure 2.** The calculated dose rate ($\mu$Sv/hr) at a distance of 20 cm for 1 MBq of (a) Am-241 and, (b) Co-57 sources. The open square points are the ordinary concrete, the green circle is the clay material, the blue square is the Gadolinium (Gd)-doped polymer, while the cross point at 0 cm (without clay) is the theoretical dose rate.

4. Conclusion
An approach of calculations for photon energy spectra and dose rate either it is transmitted or reflected one must be considered, particularly in exploring the new element or compound materials when designing a radiation shielding. For clay and ordinary concrete, a thickness of 2 and 6 cm is adequate to attenuate up to 90% of the photons from Am-241 and Co-57 sources, respectively. The clay $\mu$ is higher within 6.6% and 0.9% compared to ordinary concrete for Am-241 and Co-57 sources, respectively. While the $\mu$ values for the proposed Gd (8%)-doped polymer is higher by a factor of 9 and 5 for the same sources, respectively. A thickness of 3 cm clay and concrete were adequate to shield dose rate of 1 MBq Am-241 down to 0.05 $\mu$Sv/hr at 20 cm distance, while almost 10 cm needed for Co-57 source.
Consequently, for higher energy gamma sources such as brachytherapy room, the developed clay and ordinary concrete need to be doped with a higher Z element to ensure radiation safety for adjacent area.

5. References
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