DEVELOPMENT AND CHARACTERIZATION OF PROPERTIES OF HEAVY MINERAL REINFORCED POLYMER COMPOSITE MATERIALS FOR RADIATION SHIELDING PROTECTION

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The thesis titled "DEVELOPMENT AND CHARACTERIZATION OF PROPERTIES OF HEAVY MINERAL REINFORCED POLYMER COMPOSITE MATERIALS FOR RADIATION SHIELDING PROTECTION." submitted by Md. Muminul Haque, Student ID-1015280015, Session 2015-2016 has been accepted as satisfactory in partial fulfillment of the requirements for the degree of Master of Engineering in Nuclear Science and Engineering on January 2020.

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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, relaxation length and reduced % of radiation intensity. Comparative assessment between the characterized shielding material and the ordinary shielding material had performed based on the experimental observations. The magnetite 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.

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INTRODUCTION

1.1 General Introduction

Composites are one of the most widely used materials due to their adaptability to different situations and the relative simplicity of combination with other materials to serve specific purposes and desirable properties. Composite materials gain much attention because of their wide applications in many fields such as civil, industrial, military, air and space crafts, automobiles, and packaging applications due to their excellent thermo-mechanical properties [1].

Polymers have replaced many of the conventional metals/materials in various applications. This is possible because of the advantages such as ease of processing, productivity, cost reduction, etc. offered by polymers over conventional materials. Composites are made up of individual materials referred to as constituent materials.There are two main categories of constituent materials: matrix and reinforcement. The matrix material surrounds and supports the reinforcement materials by maintaining their relative positions. The reinforcements impart their special mechanical and physical properties to enhance the matrix properties [2].

These mechanical and physical properties are taken into account to develop a radiation protective composite material. As we know radiation is injurious to human health, despite the fact that they are beneficial in many cases. For such application one need to guarantee appropriate protection against radiation. Radiation protection can be defined as the protection of the people and environment from the harmful effects of the ionizing radiation. The fundamental aim of radiation is to provide a convenient standard of protection without hampering the beneficial uses from radiation.

The hazardous impacts of radiation can be averted by the following ways: [3]

(i) Keeping up a substantial separation from the radiation sources much as possible

(ii) Decreasing the duration time as low as possible near the source, and

(iii) Constructing shields around the nuclear facilities.

The first two ways are not very dependable as the radiation workers need to be really close to the sources, occasionally for even a long time. This protection can be efficiently implemented by providing radiation shielding. As a matter of fact, shielding is the best and most steady way for protection against radiation.

Radiation shielding is the interposing of radiation absorbing materials as a protective barrier between the source of radiation and the corresponding work place and people for the reduction of radiation. One of the most important challenges in shielding engineering is to reduce exposure to radiation to the standard acceptable level in order to protect human beings, equipment, and structures from the harmful effects of radiation.

It is a protective boundary of radiation sources, made of various kinds of materials as per their expense and accessibility in the nation that can diminish the quality or doses of radiation to an acceptable level prescribed by the national and worldwide radiation protection societies. Shielding addresses the issue to contain and limit both the beam and radiation to such an extent that the general population outside the shielding enclosure does not get the doses beyond the acceptable limit. The ICRP has given the following three recommendations for protection against ionizing radiation:

(i) No practice shall be adopted unless its introduction produces a net benefit.

(ii) All exposures shall be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account.

(iii) The equivalent dose to individuals shall not exceed the limit recommended by ICRP for appropriate circumstance.

The type of radiation emitted indicates the shielding material preferred for the facility. Thus, the study on shield should be preceded by analyzing about various radiation types. Nearly all of the radiation fields of intrigue are the blends of various types of radiation. The most significant among them is the neutrons and gamma-ray. [4]

1.2 Radioisotopes

Isotopes that are not stable and emit radiation are called radioisotopes. A radioisotope is an isotope of an element that undergoes spontaneous decay and emits radiation as it decays. During the decay process, it becomes less radioactive over time, eventually becoming stable.

Figure 1.1: Polonium is a rare, extremely radioactive metalloid

This is a photo of a thin film of polonium over a stainless-steel disk, used as an alpha-particle source. [3]

1.3 Classification of Radiation

Radiation is the process in which energy is emitted as particles or waves. There are two types of radiation [3]:

(i) Ionization radiation,

(ii) Non ionization radiation.

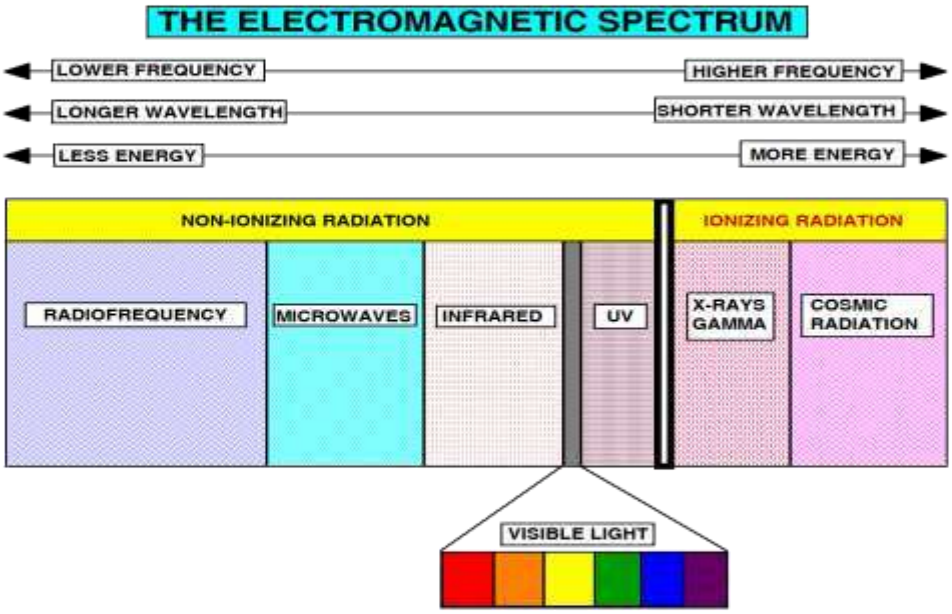

Figure 1.2: Classification of Nuclear Radiations [1]

1.4 Ionizing Radiation

Ionizing radiation is produced by unstable atoms. Unstable atoms differ from stable atoms because they have an excess of energy or mass or both.

Unstable atoms are said to be radioactive. In order to reach stability, these atoms give off, or emit, the excess energy or mass. These emissions are called radiation. The kinds of radiation are electromagnetic (like light) and particulate (i.e., mass given off with the energy of motion). Gamma radiation and X-rays are examples of electromagnetic radiation. Beta and Alpha radiation are examples of particulate radiation. Ionizing radiation can also be produced by devices such as X-ray machines [3].

Radiation	Mass (amu)	Charge	Range in air	Range in tissue
Alpha	4	+2	0.03 m	.0.04 mm
Beta	1/1840	-1	3 m	5 mm
X-ray and Gamma ray	0	0	Very large	Through body
Fast neutron	1	0	Very large	Through body
Thermal neutron	1	0	Very large	0.15 m

Table 1.1: Properties of Ionizing Radiation

1.5 Non-Ionizing Radiation

Non-ionizing radiation has less energy than ionizing radiation; it does not possess enough energy to produce ions. Examples of non-ionizing radiation are visible light, infrared, radio waves, microwaves, and sunlight.

Examples of non-ionizing radiation include:

(i) Microwaves

(ii) Visible light

(iii) Radio waves

(iv) TV waves

(v) Ultraviolet radiation (except for the very shortest wavelengths) [3]

1.6 Radiation Energy

Different radioactive materials produce radiation at different energy levels and at different rates. It is important to understand the terms used to describe the energy and intensity of the radiation. The four terms used most for this purpose are: **energy, activity, intensity and exposure.** And all kinds of ionizing radiation cause the biological effects of human body.

1.6.1 Energy

As mentioned previously, the energy of the radiation is responsible for its ability to penetrate matter. Higher energy radiation can penetrate more and higher density matter than low energy radiation. The energy of ionizing radiation is measured in electron volts (eV). One electron-volt is an extremely small amount of energy so it is common to use kiloelectronvolts (KeV) and mega electron-volt (MeV).

The energy of a radioisotope is a characteristic of the atomic structure of the material. Consider, for example, Iridium-192 Cobalt-60 and Caesium-137 which are three of the more common industrial Gamma ray sources. These isotopes emit radiation in two or three discreet wavelengths. Cobalt-60 will emit 1.33 Mev and 1.17 MeV Gamma rays, and Iridium-192 will emit 0.31 Mev, 0.47 Mev, and 0.60 MeV Gamma rays and Cesium-137 emits 0.617 MeV gamma rays. It can be seen from these values that the energy of radiation coming from Co-60 is about twice the energy of the radiation coming from the Ir-192 and cesium 137. From a radiation safety point of view, this difference in energy is important because the Co-60 has more material penetrating power and, therefore, is more dangerous and requires more shielding [3].

1.6.2 Activity

The strength of a radioactive source is called its activity, which is defined as the rate at which the isotope decays. Specifically, it is the number of atoms that decay and emit radiation in one second. Radioactivity may be thought of as the volume of radiation produced in a given amount of time. It is similar to the current control on a X-ray generator. The International System (SI) unit for activity is the becquerel (Bq), which is that quantity of radioactive material in which one atom transforms per second. The becquerel is a small unit. In practical situations, radioactivity is often quantified in kilobecquerels (kBq) or megabecquerels (MBq). The curie (Ci) is also commonly used as the unit for activity of a particular source material. The curie is a quantity of radioactive material in which 3.7 x 1010 atoms disintegrate per second. This is approximately the amount of radioactivity emitted by one-gram (1g) radium-226. One curie equals approximately 37,037 MBq. New sources of cobalt will have an activity of 20 to over 100 curies, and new sources of iridium will have an activity of similar amounts. Once a radioactive nucleus decays, it is no longer possible for it to emit the same radiation again. Therefore, the activity of radioactive sources decreases with time and the activity of a given amount of radioactive material does not depend upon the mass of material present. Additionally, one-curie sources of Cs-137 might have very different masses depending upon the relative proportion of non-radioactive atoms present in each source. The concentration of radioactivity, or the relationship between the mass of radioactive material and the activity, is called the specific activity. Specific activity is expressed as the number of curies or becquerels per unit mass or volume. The higher the specific activity of a material, the smaller the physical size of the source is likely to be [3].

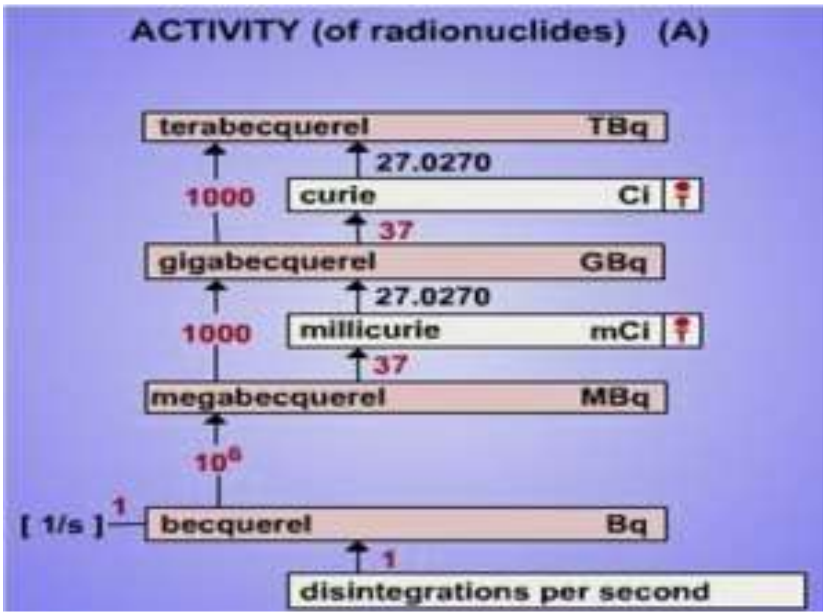

Figure 1.3: Photo of radioactivity nuclides.

1.6.3 Intensity

Radiation intensity is the amount of energy passing through a given area that is perpendicular to the direction of radiation travel in a given unit of time. The intensity of an X-ray or gamma-ray source can easily be measured with the right detector. Since it is difficult to measure the strength of a radioactive source based on its activity, which is the number of atoms that decay and emit radiation in one second, the strength of a source is often referred to in terms of its intensity. Measuring the intensity of a source is sampling the number of photons emitted from the source in some particular time period, which is directly related to the number of disintegrations in the same time period [3].

1.6.4 Exposure

One way to measure the intensity of neutron or gamma rays is to measure the amount of ionization they cause in air. The amount of ionization in air produced by the radiation is called the exposure. Exposure is expressed in terms of a scientific unit called a roentgen (R or r). The unit roentgen is equal to the amount of radiation that produces in one cubic centimeter of dry air at 0°C and standard atmospheric pressure ionization of either sign equal to one electrostatic unit of charge. Most portable radiation detection safety devices used by a radiographer measure exposure and present the reading in terms of roentgens or roentgens/hour, which is known as the dose rate [3].

1.7 Radiation Detectors

Instruments used for radiation measurement fall into two broad categories:

- rate measuring instruments and
- personal dose measuring instruments.

Figure 1.4: Photo of radiation detector meter.

Rate measuring instruments measure the rate at which exposure is received (more commonly called the radiation intensity). Survey meters, audible alarms and area monitors fall into this category. These instruments present a radiation intensity reading relative to time, such as R/hr or mR/hr. An analogy can be made between these instruments and the speedometer of a car because both are measuring units relative to time. Dose measuring instruments are those that measure the total amount of exposure received during a measuring period. The dose measuring instruments, or dosimeters, that are commonly used in industrial radiography are small devices which are designed to be worn by an individual to measure the exposure received by the individual. An analogy can be made between these instruments and the dosimeter of a car because both are measuring accumulated units. The radiation measuring instruments are commonly used in industrial radiography and medical purposes [3].

1.7.1 Survey Meters

The survey meter is the most important resource a radiographer has to determine the presence and intensity of radiation. A review of incident and over exposure reports indicate that a majority of these type of events occurred when a technician did not have or did not use a survey meter. There are many different models of survey meters available to measure radiation in the field. They all basically consist of a detector and readout display. Analogue and digital displays are available. Figure no. 1.5 showing a standard survey meter which is known as Geiger-Muller Detector. Geiger-Muller Detectors (GM) tubes typically identify gamma and beta radiation. Some GM tubes also detect alpha radiation sources. The GM tube is filled with a low-pressure inert gas (He or Ar). It has two electrodes, the walls of the tube form the cathode, and the anode is a wire in the center of the tube. The voltage across the two electrodes creates a strong electric field. When radiation strikes the tube, some of the gas molecules are ionized. The positive ions are accelerated towards the cathode and the electrons towards the anode. Close to the anode in the "avalanche region" the electrons gain sufficient energy to ionize additional gas molecules and create a large number of electron avalanches which spread along the anode and effectively multiply the effect. The GM tube counts the number of gamma rays or beta particles entering the detector per second. They do not provide identification information unless coupled with alternate detectors.

Figure 1.5: Photo of radiation survey meter (Geiger-Muller Detector)

1.7.2 Radiation dosimeter

A radiation dosimeter is a device used in Medical Health Physics that measures exposure to ionizing radiation. It has three main uses:

1) Human radiation index device.

2) Measurement of dose in both medical and industrial process.

3) Assessment of the ionizing radiation dose absorbed by the human body [3].

Figure 1.6: Type of detector or instrument used

1.8 Biological Effect of Ionizing Radiation

Radiation affects different people in several ways. The harmful effects of this ionizing radiation cannot be completely avoided. The biological effect of radiation is an outcome of the transfer of energy by ionization and the excitation to the cells of body. Whenever ionizing radiation comes to an interaction with human body it creates ionization and excitations in the tissues and weakens the normal functions of the cells. The severity of this radiation damage depends on several factors, namely energy and nature of the radiation, the body part that exposed to radiation, dose rate and total dose, age of the person who exposed to radiation, organ's sensitivity to radiation which is exposed to radiation. The occurrence of particular health effects from exposure to ionizing radiation is a complicated function of numerous factors including:

a) Type of radiation involved. All kinds of ionizing radiation can produce health effects. The main difference in the ability of Neutron and Gamma radiation to cause health effects is the amount of energy they have. Their energy determines how far they can penetrate into tissue and how much energy they are able to transmit directly or indirectly to tissues.

b) Size of dose received. The higher the dose of radiation received, the higher the likelihood of health effects.

c) Rate the dose is received. Tissue can receive larger dosages over a period of time. If the dosage occurs over a number of days or weeks, the results are often not as serious if a similar dose was received in a matter of minutes.

d) Part of the body exposed. Extremities such as the hands or feet are able to receive a greater amount of radiation with less resulting damage than blood forming organs housed in the torso.

e) The age of the individual. As a person ages, cell division slows and the body is less sensitive to the effects of ionizing radiation. Once cell division has slowed, the effects of radiation are somewhat less damaging than when cells were rapidly dividing.

f) Biological differences. Some individuals are more sensitive to the effects of radiation than others. Studies have not been able to conclusively determine the differences.[5]

1.9 Cell Radio-Sensitivity

Radio sensitivity is the relative susceptibility of cells, tissues, organs, organisms, or other substances to the injurious action of radiation. In general, it has been found that cell radio sensitivity is directly proportional to the rate of cell division and inversely proportional to the degree of cell differentiation. In short, this means that actively dividing cells or those not fully mature are most at risk from radiation. The most radio-sensitive cells are those which:

- have a high division rate
- have a high metabolic rate
- are of a non-specialized type
- are well nourished

Examples of various tissues and their relative radio sensitivities are listed below.[3]

High Radio-sensitivity
Lymphoid organs, bone marrow, blood, testes, ovaries, intestines
Fairly High Radio-sensitivity
Skin and other organs with epithelial cell lining (cornea, oral cavity, esophagus, rectum, bladder, vagina, uterine cervix, ureters)
Moderate Radio-sensitivity
Optic lens, stomach, growing cartilage, fine vasculature, growing bone
Fairly Low Radio-sensitivity
Mature cartilage or bones, salivary glands, respiratory organs, kidneys, liver, pancreas, thyroid, adrenal and pituitary glands
Low Radio-sensitivity
Muscle, brain, spinal cord

Table 1.2: Level of radio-sensitivity and their effect on biological tissue

1.10 Mechanisms of Radiation Damage

Radiation damage begins from the cellular level. Radiation that absorbed by a cell has the capability to impact a disparity of critical targets in cell, among them the DNA is the most important one. From several evidences it is indicated that DNA damage is the cause to cell death, carcinogenesis and mutation. The mechanism of this damage can occur via one or two ways [5]. They are

a) Direct action, and

b) Indirect action.

1.10.1 Direct Action:

Radiation may directly impact the DNA by creating ionization in the atoms of DNA molecules. This may be visualized as direct hit' on the DNA by radiation, and thus is an unusual incidence as the size of the target is small. The DNA helix is about 2 nm in diameter. For the occurrence of the action, ionization by the radiation must be produced within that few nanometers of the molecule of DNA. [3,5]

1.10.2 Indirect Action:

In this scenario the interaction of radiation occurs with the target atoms or molecules which are non-critical, usually water. This occurrence produces free radicals which are molecules or atoms having an unpaired electron that makes them highly reactive. Then the critical targets like DNA can be attacked by these free radicals. As they can diffuse in cell from some distance, so the occurrence of initial ionization need not to be close to the DNA to cause any damage. And so, damage from the indirect action is more common than the direct action, specifically for the radiation having low specific ionization.[3]

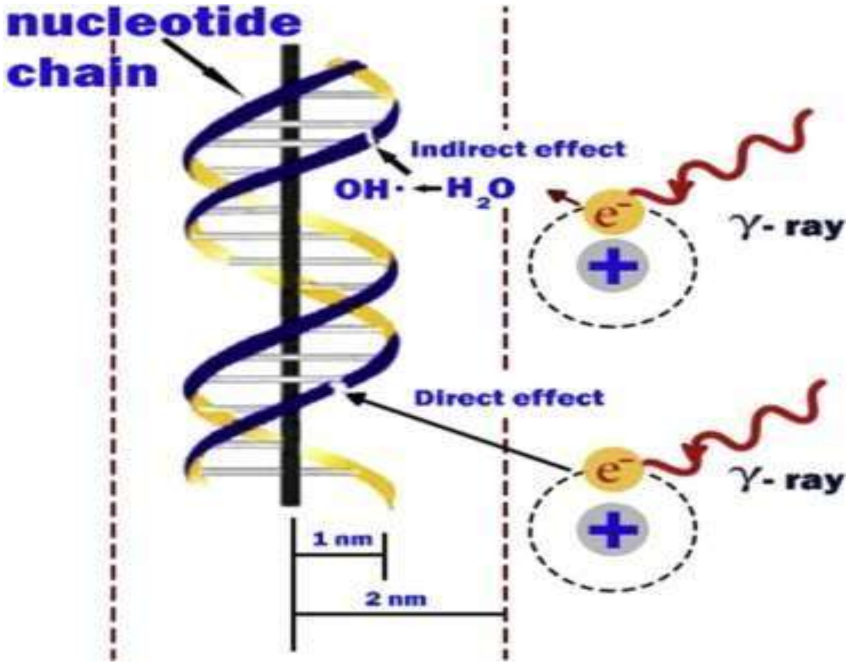

Figure 1.7: Mechanism of ionizing radiation

1.11 Effects of Ionizing Radiation

The biological effects of ionizing radiation can be categorized in accordance with the times of occurrence, characteristics of effects and the object which shows the effects According to the occurring time, the biological effects of ionizing radiation can be classified into two categories; [3]

1.11.1 Short Term Effects:

Radiation that delivered to the body within a short period of time is the acute dose. These biological effects can occur in between several hours or within several days if the amount of absorbed radiation dose is high enough. The total effect may differ from low and transient sickness to death. High doses may kill large number of cells that damages tissues as well as organs. For a group of people who exposed to a whole-body penetrating dose of radiation then the LD50/60 is the lethal dose with minimal care at which 50% of those who are exposed to the dose of 320-360 Rad will die in between 60 days. And the threshold values are about 150 Rad at which the effect will be first observed in the most sensitive individuals who exposed to radiation. The initial symptoms are nausea, fatigue, vomiting, and loss of appetite. Other effects are hair loss, skin burns, Cataracts (clouding of lens of the eye) etc. [3-6]

1.11.2 Long Term Effects

Long term effects are those which may manifest years after the exposure. There is no unique disease linked to the long-term effects of ionizing radiation. It may cause somatic damage that may result in embryological defects, cataracts, increased incident of cancer and lifespan shortening. [3-6]

According to the characteristics of effects, the biological effects of ionizing radiation can be classified into two categories.

1.11.3 Stochastic Effects

These effects are probabilistic in nature and there is not any threshold dose. That is there is no certain exposure condition for the occurrence of this effect. The probability of having this effect is proportional to the dose absorbed but the severity of this effect does not depend on the absorbed dose: This effect can occur due to receive the small exposure over a long period that may cause cancer (e.g. Lymphoma, Leukemia etc.) and also genetic effect by creating change in the coded genetic information that causes various deformation (e.g. death of offspring, mental retardation etc.). [3-6]

1.11.4 Deterministic Effects:

These effects have a particular threshold dose below which these effects do not occur and usually deterministic in nature, Severity of these effects is proportional to the dose received. These effects may be called 'Threshold Effect". Skin erythema, necrosis, vomiting, desquamation, hemorrhage and sometimes even death can occur because of this effect. [3]

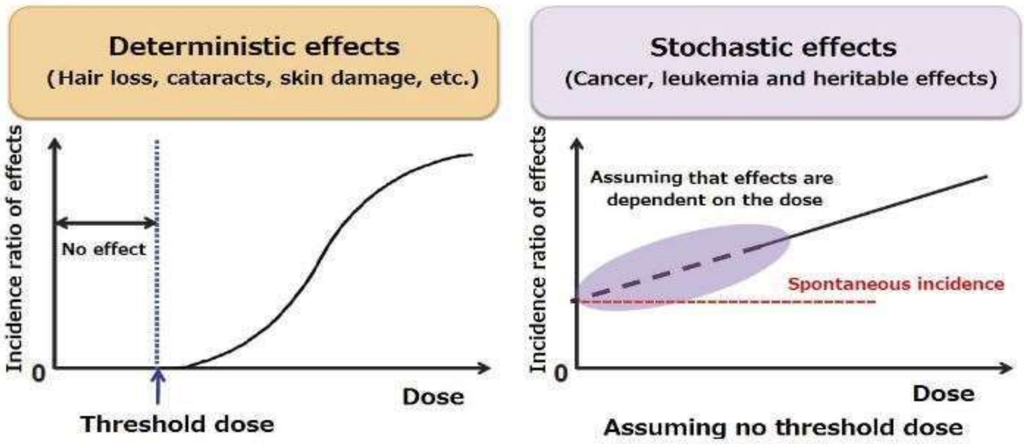

Figure 1.8: Deterministic and Stochastic effect

According to the object that shows the effect, the biological effects of ionizing radiation can be classified into two categories:

1.11.4.1 Somatic Effects:

These effects occur on the person exposed. This may appear immediately right after the exposure. The early stage effects are due to the acute exposure. Persons recovering from early effect still may develop some other types of late effects.

1.11.4.2 Genetic Effects:

Genetic effects occur due to the irradiation of reproductive cells. When these cells irradiate, the DNA or the chromosomes of these cells may be affected that can lead to the mutation of the genetic cells, changes in the number of chromosomes or chromosome aberrations. This may result detrimental effects to the descendants of the person exposed. These effects do not appear until the subsequent generations are born. [3-6]

1.12 Radiation Measurements

The size or weight of a container or shipment does not indicate how much radioactivity is in it. The amount of radioactivity in a quantity of material can be determined by noting how many curies of the material are present.

More curies = a greater amount of radioactivity

A large amount of material can have a very small amount of radioactivity; a very small amount of material can have a lot of radioactivity.

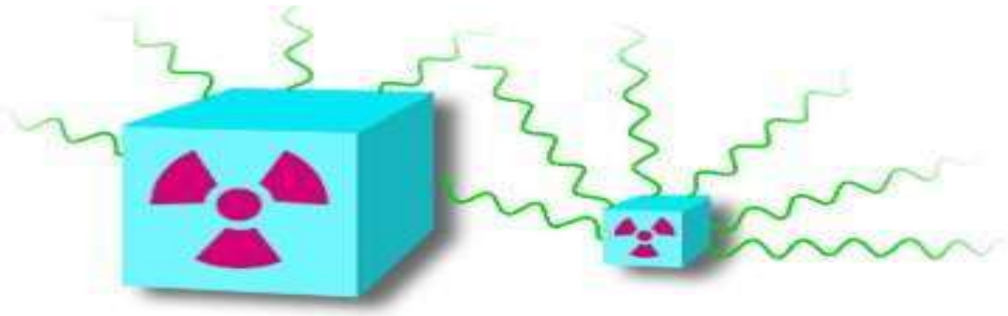

Figure 1.9: Photo of radioactivity material

For example, uranium-238 has 0.00015 curie of radioactivity per pound (0.15 millicurie), while cobalt-60 has nearly 518,000 curie per pound.

In the International System of units (SI), the becquerel (Bq) is the unit of radioactivity. One Bq is 1 disintegration per second (dps). One curie is 37 billion Bq. Since the Bq represents such a small amount, you are likely to see a prefix used with Bq, as shown below [3]:

(i) 1 MBq (27 micro-curie)

(ii) 1 GBq (27 milli-curie)

(iii) 37 GBq (1 curie)

(iv) 1 TBq (27 curies)

1.13 Basic Principles for Radiation Activities

Basic principles applied when performing radiation activities are the following;

(i) Justifiability of Application: Each radiation activity should be planned and implemented in such manner that the use of ionizing radiation sources provides more benefit than the overall damage.

(ii) Optimization of Ionizing Radiation Protection: Each radiation activity must be performed in such manner that the exposure to ionizing radiation shall be as low as objectively possible, considering economic and social factors;

(iii) Limitation of individual exposure: Radiation activity must be planned in such manner that the individual exposures shall always be under prescribed limits. [4]

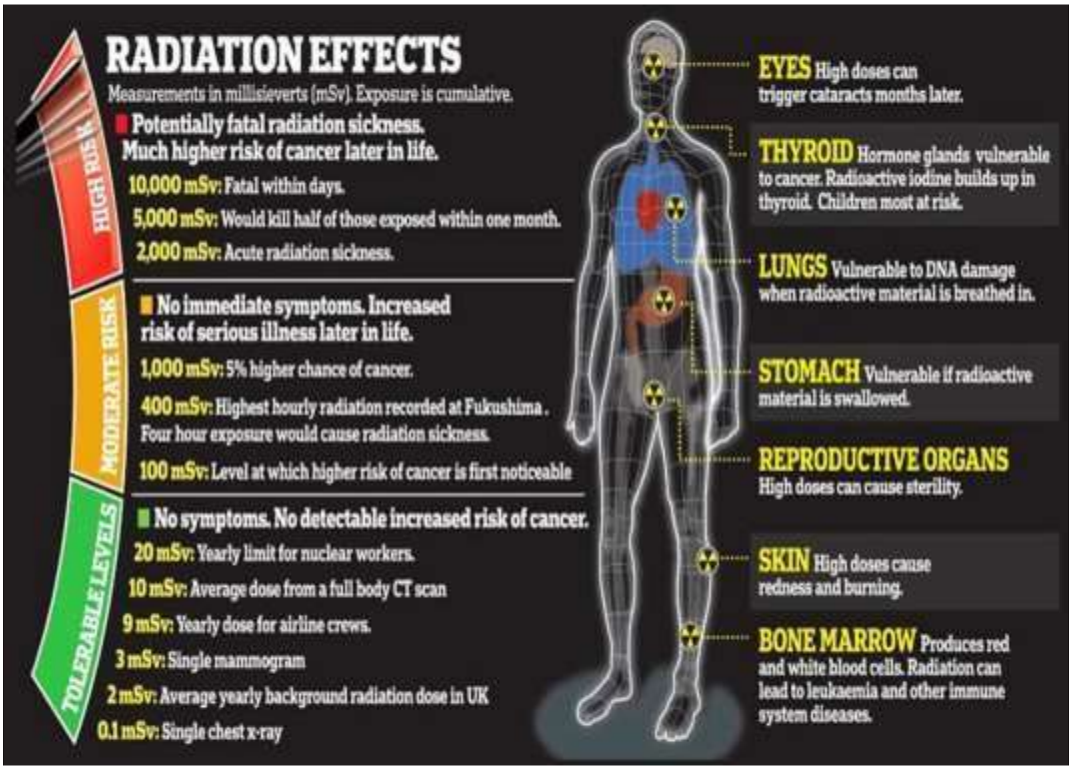

Figure 1.10: Effect of radiation on human body tissues

1.14 Shielding

Shielding reduces the intensity of radiation depending on the thickness of the radiation absorbing material. This is an exponential relationship with gradually diminishing effect as equal slices of shielding material are added. A quantity known as the halving thicknesses is used to calculate this. For example, a practical shield in a fallout shelter with ten halving thicknesses of packed dirt, which is roughly 115 cm (3 ft 9 in) reduces gamma rays to 1/1024 of their original intensity (i.e. $1/2^{10}$).

The effectiveness of a shielding material in general increases with its atomic number, called Z, except for neutron shielding which is more readily shielded by the likes of neutron absorbers and moderators such as compounds of boron e.g. boric acid, cadmium, carbon and hydrogen respectively.

Depending on their different characteristics, different types of ionizing radiation along with their shielding are briefly described below.

1.14.1 Alpha particle:

Alpha particles are also called alpha radiation. It consists of two protons and two neutrons and is identical to helium nucleus. When an atom undergoes an alpha decay, the mass number of that atom decreases by four. Alpha particles have low penetrating power. They can be stopped by the skin or by the few centimeters of air. Only the heavy particles undergo alpha decay. [3,5]

1.14.2 Beta particle:

Beta radiation is the radiation which occurs due to the beta particles that are the electrons (positrons sometimes). The penetrating power of beta radiation is in between the alpha and gamma-ray ray. It can penetrate few meters in the air This decay involves the weak interactions only. [3,5]

1.14.3 Gamma-ray:

Gamma rays consist of high-energy waves that can travel great distances at the speed of light and generally have a great ability to penetrate other materials. For that reason, gamma rays (such as from cobalt-60) are often used in medical applications to treat cancer and sterilize medical instruments. Similarly, x-rays are typically used to provide static images of body parts (such as teeth and bones), and are also used in industry to find defects in welds. Despite their ability to penetrate other materials, in general, neither gamma rays nor x-rays have the ability to make anything radioactive. Several feet of concrete or a few inches of dense material (such as lead) are able to block these types of radiation [3-5].

1.14.3.1 Characteristics of Gamma Radiation

Some characteristics of gamma radiation are given below [3-5]:

i) Gamma radiation and X-rays are electromagnetic radiation like visible light, radio waves, and ultraviolet light. These electromagnetic radiations differ only in the amount of energy they have. Gamma rays and X-rays are the most energetic of these.

ii) Gamma radiation is able to travel many meters in air and many centimeters in human tissue. It readily penetrates most materials and is sometimes called " penetrating radiation."

iii) X-rays are like gamma rays. They, too, are penetrating radiation.

iv) Radioactive materials that emit gamma radiation and X-rays constitute both an external and internal hazard to humans.

v) Dense materials are needed for shielding from gamma radiation. Clothing and turnout gear provide little shielding from penetrating radiation but will prevent contamination of the skin by radioactive materials.

vi) Gamma radiation is detected with survey instruments, including civil defense instruments. Low levels can be measured with a standard Geiger counter, such as the CD V-700. High levels can be measured with an ionization chamber, such as a CD V-715.

vii) Gamma radiation or X-rays frequently accompany the emission of alpha and beta radiation.

viii)Instruments designed solely for alpha detection (such as an alpha scintillation counter) will not detect gamma radiation.

ix) Pocket chamber (pencil) dosimeters, film badges, thermoluminescent, and other types of dosimeters can be used to measure accumulated exposure to gamma radiation.

1.14.3.2 Interactions of Gamma with Matter

There are five major types of interactions causing attenuation of a photon beam by matter: Compton effect, photoelectric effect, pair production, coherent scattering, and photo disintegration. The first three are the most important, as they result in the transfer of energy to electrons. The electrons then transfer this energy to matter in many small Coulomb-force interactions. Coherent scattering is elastic and photo disintegrations are only significant for photon energies above a few MeV, where they may create radiation protection problems with the production of neutrons and consequent radioactivity. The relative importance of Compton effect, photoelectric effect, and pair production depend on both the photon quantum energy and the atomic number of the absorbing medium [3,5]

1.14.4 Neutron:

Neutron is the subatomic particle having the mass number which is slightly higher than that of a proton. It has no electric charge. It is essential in the generation of nuclear power. The interactions of neutrons are highly ionizing. Neutrons can pass through most of the materials. Due to the high kinetic energy, neutron radiations are one of the most dangerous one. Neutrons are uncharged particles and thus have a wide range of energy and mass level that must be blocked. Neutrons being uncharged can easily penetrate dense material; hence lead is inefficient for neutron shielding. Materials of low atomic number components are ideal for ceasing this sort of radiation since they have a higher likelihood of forming cross-section for neutron interaction. Hydrogen as well as hydrogen-based materials is appropriate for this job. Fast neutrons are very effectively slowed down to the thermal neutrons by collision with hydrogen atoms, which may subsequently be eliminated by materials having high thermal neutron absorption cross-section. Compounds having high content of hydrogen is effective neutron barrier, furthermore being! comparatively cheap. The low-density element diffuses the neutron by elastic scattering, whereas heavy materials slow the neutrons through inelastic scattering. [3,5]

1.14.5 X-rays:

X-rays are another form of electromagnetic radiation. They have enough energy that can ionize atoms and break the molecular bonds. X-rays with high energies can travel relatively dense object. They have the energy length much shorter than the visible light. The ionizing capability of x-ray used in medical sectors. From the above discussion it has been concluded that among all types of radiation the X-ray, gamma-ray and neutrons are the most hazardous as they have the higher penetrating power and cause damage to the human body. In the current study the gamma-ray and neutron attenuation properties of different locally developed concrete samples have been investigated. The penetrations of different ionizing radiations are given below. [3,5]

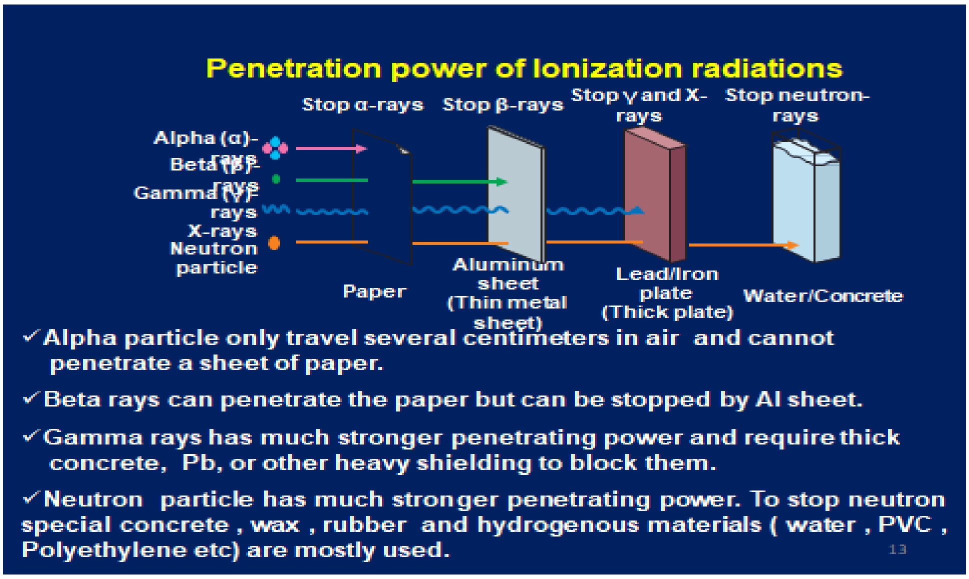

Figure 1.11: Penetration power of Ionization Radiation

1.15 Shielding Parameters and Theoretical formula:

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:

$$I=I_o e^{-\mu t} \quad \text{or} \quad D_u = D_t e^{-\mu t} \tag{1}$$

Where D_u and D_t are the non-attenuated and attenuated gamma ray beam intensities,

μ_l (cm^{-1}) is the linear attenuation coefficient and **t** is the thickness of the material (cm).

In this study, D_u indicates the initial open beam output. From this equation (1) we can calculate the linear attenuation coefficient.

The determined linear attenuation coefficient could be useful to determine the mass attenuation coefficients by using following formulae

$$\mu_l = \frac{1}{\rho t} ln\left(\frac{I_0}{I}\right) \tag{2}$$

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:

$$\mu_m = \left(\frac{\mu}{p}\right)p \tag{3}$$

Where $\frac{\mu}{\rho}$ is the mass attenuation coefficient (cm^2/g) and ρ 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.

The half value layers (HVL) of any 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:

$$\mathrm{HVL} = \frac{0.693}{\mu} \tag{4}$$

The tenth value layers (TVL) of any 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:

$$\mathrm{TVL} = \frac{2.303}{\mu} \tag{5}$$

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

$$\mathrm{SVL} = \frac{2.773}{\mu} \tag{6}$$

The linear 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:

$$\lambda = \frac{1}{\mu} \tag{7}$$

This is the average distance a single particle travels through a given attenuating medium before interacting. It is also the depth to which a fraction 1/e (~37%) of a large homogeneous population of particles in a beam can penetrate. For example, a distance of three free mean paths, $3/\mu$, reduces the primary beam intensity to 5% . [3,5]

1.16 Benefits of Composite Material as Radiation Shield

An ideal shielding material should have the following requirements;

(i) Easy to Handle and fabricate;

(ii) Be inexpensive;

(iii) Stable in chemical composition and physical dimension;

(iv) Physically strong and durable under operational condition:

(v) Non corrosive and odorless;

(vi) should not change its properties with use; and

(vii)High thermal conductivity.

In recent years, numerous studies have been carried out on shielding materials, especially by using nanotechnology in cement-based materials [7]. So, it is chosen heavy minerals like Magnetite, Ilmenite, Rutile, Garnet and Zirconium for gamma- radiation shielding and polymer is chosen for neutron shielding [7-10]. Combination of polymer and heavy mineral-based composites will be a probable solution to reduce radiation exposure. Heavy mineral reinforced composite shield is by a wide margin the most broadly utilized material, which is additionally viewed as a brilliant and flexible shielding material. It can be extensively used as a radiation shield in various nuclear facilities, power plants and power and research reactors. It also has a widespread use in different medical radio-therapeutic facilities, particle accelerators, laboratory hot cell and may more. Lead has long been considered as a highly effective shielding material for gamma-ray radiation attenuation due to its high density [17]. Then again, in light of the fact that lead is costly and toxic, a few specialists have been examining alternative materials for radiation attenuation. One such material explored by the scientists is heavy mineral reinforced composite concrete, Because of its great mechanical properties, high sturdiness, it tends to be effectively arranged in various pieces and can be effortlessly framed and utilized in construction work. Composite concrete has undeniable superiority over different items in this field since it gives outstanding shielding characteristics in sensible thickness at a particularly minimal cost for each pound. Composite concrete is comparatively affordable and have palatable mechanical properties. It tends to be effortlessly taken care of and cast into complicated structures, it contains a blend of different light and substantial components and an attenuating ability for both photons and neutrons. In the meantime, it meets all typical structure necessities. As the efficiency of any shielding material relies upon its mass, high-density material is being prominently used in shielding industries. The composite concrete aggregates play a crucial role in modifying the physical and mechanical properties of the concrete, which typically affect its shielding properties. The composite concrete aggregates containing a combination of heavy material play a significant role. The utilization of high-density concrete helps to reduce the thickness of the shield in half, i.e. it results in thinner shield wall. As the density of the shielding material is raised, the radiation shielding capacity is developed. High density composite concrete has so far shown some outstanding economic in some large-scale installation [18-21]. High density concrete is extensively used in shielding because of its ready availability, affordability, shielding efficiency and recognized fabrication properties. In adequate density it demonstrates both neutron and gamma-ray attenuation properties efficiently.

1.17 Objectives of the Present Study

(i) To fabricate and develop samples of high-density polymer-based concrete by using different locally available materials,

(ii) To study on gamma-ray attenuation properties of locally developed high density concretes.

(iii) To measure and analyze the suitability of shieling materials in medical treatment facilities.

1.18 Significance of the Study

This research determines linear attenuation coefficients and attenuation factors, among analyzed materials heavy mineral like Magnetite, Ilmenite, Rutile, Garnet and Zirconium based polymer composite concrete for gamma-radiation shielding. These types of materials are available, which will be used in gamma radiation Protection. In this study, it will be helped to determine that which shielding materials will be better to protect gamma radiation. These values are used to determine the shape and thickness of the alloy needed, as well as the radiation dose to use along with the alloy tissue compensator. The availability of appropriate and varying materials for radiation shielding increases the efficiency of using different radiation for various purposes

CHAPTER 2

LITERATURE REVIEW

2.1 Review of The Past Works

Within a few years of emergence of radiation shielding as a subject, huge amount of experimental data has been accumulated on shield test facilities ranging from measurements with simple geometries to detailed test on full scale mockups. On the other hand, theoretical investigations progressed in a relatively haphazard manner. Among all the theoretical studies, the phenomenological approach received the maximum attention. A review of some selected representative previous works is made below.

A. S. Mollah *et al.* [7] investigated the neutron transport and shielding properties of heavy concretes that made from locally available magnetite and ilmenite sand. They used Cf-252 source and as a long counter they used BF3 detector for the experiment. They investigated the thickness dependent removal cross section at the thickness of 5 cm and 100 cm and found the result varies for magnetite and ilmenite concretes.

F. U. Ahmed *et al.* [8] measured the shielding properties of gamma-ray of ilmenite-magnetite concrete polyboron. They used NaI (TI) detector for this measurement. They used NaI (TT) detector for this measurement. They generated the inverted detector response matrix. They used the inverted response matrix to convert the pulse height spectra of gamma rays that transmitted through the shields into photon spectra and then they calculated the respective dose rates. They reported the buildup factors and instantaneous relaxation length for gamma-rays from the source Cf-252 that penetrate I-M concrete and polyboron slabs. They also reported the related coefficients.

S. Ahmed, Ouda [9] presented that high-performance heavy density concrete is developed using different aggregates that are useful for gamma-ray radiation shielding. The concrete mixes with the aggregates mainly used is Magnetite, Barite, Goethite, Supernite in both form (Coarse and Fine). The aggregates are replaced with sand to a fixed amount. Main measuring parameters are workability of fresh concrete, compressive strength, half value layer and tenth value layer, the linear Attenuation coefficient. Among them linear attenuation coefficient is most important in case of gamma-ray shielding properties. From the paper it is observed that magnetite has high density with proper compressive strength and good output of HVL and TVL. Mainly two types of photon energy source are used in case of measuring HVL and TVL. The results of HVL and TVL are inversely proportional with density. It has also noticed that magnetite has higher physico mechanical properties than other concrete. High performance concrete incorporated with magnetite enhances the gamma-ray shielding efficiency.

S. I. Bhuiyan *et al.* [10] measured the gamma ray shielding properties of ilmenite-magnetite(I-M) concrete experimentally. They reported the thickness dependent removal cross section and also the instantaneous relaxation length of neutrons from a CF-252 source that penetrates I-M concrete without and with cadmium sheet. For the calculation of thickness dependence of removal cross section, they developed an empirical formula. They also reported some other related coefficients.

A. M. Madbouly *et al*. [11] illustrated that different types of materials used as radiation shielding result in different outcome. They examined mainly five types of materials Barite (BaSO4), Boron carbide (B_4C), Ferroboron (FeB), Ilmenite (FeTiO3), Galena (PbS) – as shielding for neutron and gamma-ray. Five separate mixtures were made, each comprising of only one of the materials where the material at hand/total ingredient ratio is 0.20. After that, the radiation shielding parameters measured. Mass attenuation coefficient of gamma-rays measured with XCOM program. Mainly the absorption macroscopic cross section of the neutrons, relaxation length (λ) and macroscopic effective removal cross section for fast neutrons measured theoretically. Different types of materials used for the shielding were selected from several studies, which studied thermal and mechanical properties. From the result it is understood that the concrete prepared with galena is preferable as fast neutrons and gamma ray shielding. Concrete with B4C is the most suitable for neutron absorption and barite for scattering of neutrons. Concrete made with galena has high density than other studied material.

Stankovic *et al.* [12] researched to provide radiation protection with different types of materials that are used in making of concrete. Two types were mainly used (ordinary and barite concrete). Results showed that concrete made with barite were more effective than ordinary concrete in case of radiation shielding. The goal of this paper was to show the transmission of gamma-rays through concrete using Monte Carlo calculations. Two different energies were used to calculate gamma-ray radiation absorption. By comparing the mass attenuation coefficient, the values were obtained. The result shows that the barite concrete is more efficient in case of radiation shielding as it has good mass attenuation coefficient.

Najam laith Ahmed *et al.* [13] used granite for gamma-ray shielding in this study. The author practically and theoretically presented the linear attenuation coefficient and mass attenuation coefficient of gamma-rays using granite. The author compared these attenuation properties of granite with lead. Here the author also showed the relationship between attenuation coefficient and energy as well as attenuation coefficient and density.

Eduardo Gallego *et al.* [14] tested high density concrete for neutron shielding material. Here as a neutron shielding material the authors used high density magnetite concrete. In this study the authors characterized the behavior of the material against neutrons, the authors also tested different mixing with boron compounds to improve the efficiency of neutron shielding. They used Am-Be-241 as neutron source. They also performed Monte Carlo calculations. They determined tenth value layer for different mixing.

Y. Elmahroug *et al*. [15] calculated fast neutron removal cross section for several shielding materials which is one of the most important parameters of neutron penetration. Here the authors theoretically calculated the macroscopic effective removal cross section for pure polyethylene, Borated Polyethylene (1%, 5%, 5.45%, 8.97%, 30%), 7.5% Lithium Polyethylene, Bismuth-loaded Polyethylene (78.5%, 90%), Flexi Boron shielding, Borated Silicon, Borated Hydrogen mix, Borated Hydrogen-loaded castable dry mix, Borated-lead Polyethylene, resin 250WD, SUS304, K-resin, Krafton HB and Premadex. The authors also computed the elemental composition, its weight fraction, the partial density and the mass removal cross section for these shielding materials.

Sahadath *et al.* [16] calculated one of the main neutrons shielding properties developing ilmenite magnetite concrete. They mainly calculated the removal cross section of ilmenite magnetite concrete. The neutron emitted mostly fast neutron, which are very difficult to shield because of low absorption cross section at higher energies. For the results effective removal cross section, variation of hydrogen content, relaxation length is calculated analytically and then compared with ordinary concrete. As I-M concrete has heavy aggregates, this is more able to tame the radiation of inelastic gamma-rays. Ordinary concrete is slightly better than I-M concrete for its higher percentage of hydrogen. As I-M concrete has better removal cross shielding results, it is good for using as shielding in nuclear reactor and other nuclear facilities.

R. S. Rajavikraman [17] investigated the result for using nano concrete composites. The concrete is made of galena (Pbs) and nano silica along with carbon nano-tubes. Two types of ratios were used to make the concrete. The result showed that the Nano concrete has higher shielding parameters than other shielding materials. The concrete has high compressive strength and high density. This new type of concrete is useful for preventing radiation in radioactive zone.

S. M. J. Mortazavi *et al*. [18] computed the radiation shielding properties made with Galena and then compared with other concretes. Two types of concrete mixes were produced and then density and compressive strength were measured. In case of gamma-ray radiation, half value layer thickness of galena concrete is less than ordinary result. All the values of the galena concrete measured with ordinary concrete and barite concrete and galena concrete provide better physical properties. For these reason, high density galena concrete is suitable for radiation attenuation.

J. Kazjonovs *et al.* [19] designed high density concrete. In this study they used iron dross (iron oxides mainly) and steel punching as aggregates. They designed two different mixes where the sizes of aggregates are less than 11.2 mm. they replaced the traditional aggregates of concrete by 50% and 100% steel treatment waste. Here they determined the mechanical and physical properties like density, tensile strength and compressive strength, freeze-thaw resistance etc.

A. Delnavaz *et al.* [20] investigated the transmission rate of gamma radiation from heavy concrete which contained iron oxide and barite aggregates. They made cylindrical samples and tested the compressive strength of the samples. For gamma radiation tests the authors used Cs-137 source. In heavy concrete they used 25%, 50%, 75% and 100% of barite and iron oxide. They measured density, attenuation coefficient, half value layer for the concretes. The authors presented the relation between density and attenuation coefficient that is how attenuation coefficient increases or decrease with the increase or decrease of density. They also showed how compressive strength changes with the amount of heavy aggregates added in the concrete.

İ. Akkurta *et al.* [21] investigated half value layer for different concrete. Here the authors used concrete which contained different ratios of limonite, siderite and barite. For the beam of gamma-ray they used Co-60 sources. They also evaluated the total linear attenuation coefficient. The authors exhibited the relationship between the transmission rate and the thickness of concrete.

E. Yilmaz *et al.* [22] studied gamma-ray and neutron shielding of 12 concrete samples. They studied these concretes without and with mineral additives. The authors calculated linear and total mass attenuation coefficients, half value thickness, effective electron densities, effective atomic numbers and atomic cross-sections where the photon energies are of 59.5 and 661 keV. Here they compared the calculated values. For neutron shielding they computed macroscopic removal cross section. They conducted the calculation of attenuation coefficients by the Win X Com and N X com programs.

P. S. Vishwanath *et al*. [23] studied properties of gamma-ray and neutron shielding of some soil samples. Here they used soil samples of five different types, namely clay, loam, clay loam, sandy loam and sandy clay loam. Here the authors studied the effectiveness of gamma-ray shielding of soil by half value layer, mass attenuation coefficient and exposure build-up factors. They also calculated the fast neutron removal cross section by the method of partial density. They used GP factor formula for the calculation of build-up factor.

A. El-Sayed Abdo [24] calculated total mass attenuation coefficients (wp) for the gamma-rays and also the effective removal cross-sections (Cr) for the fast neutrons theoretically. They used four types of concrete which have different density namely

(1) dolomite-sand (p=2.5 g/cc), (2) magnetite-limonite (p=3.6 g/cc) (3) barite-barite (p=3.49 g/cc) and (4) ilmenite-ilmenite (p=3.69 g/cc). They used the XCOM computer program to calculate the total mass attenuation coefficient where the range of energies was from 10 keV to 1 GeV. They used the elemental composition of the concrete mixes to calculate effective removal cross-section. They compared the calculated results with those previously measured.

Sarıyer *et al.* [25] designed different types of special concretes by addition of Shielding materials such as iron and boron carbide aggregates. That boron is good thermal neutron absorber material and that iron is instrumental material in slowing down fast neutrons below 1 MeV are well known. Using FULKA Monte Carlo code, they calculated the shielding thickness of the concrete containing B_4C and Fe - B at different ratios. The concrete made with B_4C had a density of 2.52 g/cc and the density of Fe - B concrete was 7.51 g/cc. The code was used to calculate the minimum shielding thickness and this thickness reduced the dose rate available beyond surface of the shield to less than 0.1uSv/hour for general public area. They argued that high density Fe - B added to concrete was better neutron shielding material than B_4C and other ordinary concrete.

F. Bouzarjomehri *et al.* [26] conducted the evaluation of efficiency of shielding of different concrete made with barite mixtures. Diverse size variations of barite aggregates mixed with five different water/cement ratios were examined. These aggregates were mainly used for measuring the higher densities of the concrete. For the result the specimen was irradiated by Co-60gamma-ray beam. The result showed that among all samples, 350 kg/m3 cement and equal amounts of coarse and fine barite sands had nearly lowest half value layer (HVL), and maximum compression strength, so the specimen was considered as the excellent barite concrete sample and recommended for shielding facilities.

Gencel *et al*. [27] studied the effects of various concentrations of hematite (10-50 volume% at 10% intervals) on neutron and gamma-ray shielding properties concrete. For making concrete a fixed water-to-cement ratio of 0.42 kg/m and 400 kg/m of cement were selected. Compressive strength increased so high for mixing the hematite with plain concrete. Among all ratios adding 50% hematite with the concrete provides the best result for shielding purpose as it has a higher compressive strength. The compressive strength was depended on the increased amount of hematite. Although concretes containing hematite have desired gamma- ray shielding workability, there was no effect of hematite for inclusion in concrete.

2.1 Summary of review

In recent years, numerous studies have been carried out on shielding materials, especially by using nanotechnology in cement-based materials. From reviewing the above literature, we can see that different aggregates like lead, barite, magnetite, ilmenite, hematite, granite, B4C and Fe - B etc. were used in several ratios to make different shield for radiation. These shielding materials are suitable for construction of shielding barrier for different radioactive sources.

So, taking that into account a study has been undertaken for developing high density polymer composite concrete by using polymer, locally available materials and minerals like Magnetite, Ilmenite, Rutile, Garnet and Zirconium for all type of radiation shielding. Combination of polymer and heavy mineral-based composites are good solution to reduce radiation exposure.

CHAPTER 3

METHODOLGY

3.1 Overview

The context of this chapter briefly describes about materials selection, sample preparation and experimental instrumentation that adopted in this study for individual sample used in the current experiments. The preceding part of this chapter describes the calibrator systems of SSDL laboratory that generates gamma radiation. In this study two types of radiation sources and one types of radiation survey meter were used. We also used thermometer, barometer, etc., to estimate the ambiance condition of the experiments. These are described in the subsequent sections:

3.2 Material Selection

One of the basic principles of radiation protection is shielding. Shielding has a demand which is shielding has to be good at reducing the absorbed dose rate to the most possible extent. Radiation shielding consists of many materials. The radiation shield is composed of heavy mineral based polymer concrete, which is made of several types of aggregates Countries differ with regard to available raw materials. The goal, hence, is to make high density concrete using locally available heavy aggregates.

Selection of high-density aggregates is determined by physical properties, availability etc. Usually locally available heavy aggregates are desirable. A number of minerals are used in composition of the heavyweight aggregate – synthetic or natural or artificial.

The most common of all-natural aggregates are hematite, barite, magnetite, limonite, goethite, and ilmenite. In this study we used Unsaturated polymer resign (UPR) as matrix , sand and stone along with magnetite, ilmenite, garnet , rutile and zirconium.

3.2.1 Ilmenite

Ilmenite sand shown in Figure 3.1 is basically a black heavy ore of iron and titanium. It is used in manufacturing Ferro-alloys. Ilmenite is dense mineral; and its density is 4.7 g/cm^3 which are higher than other minerals. For its high density, it makes the composite concrete space effective which is one of the reasons that it used in radiation shielding. The type reactivity of ilmenite sand is inert.

Figure 3.1: Heavy mineral Ilmenite

3.2.2 Magnetite

Magnetite sand shown in Figure 3.2 is the most common minerals in heavy mineral fraction of sand and one of the main iron ores having the chemical formula Fe_30_4. It is usually black or brownish-black. The important property of magnetite is ferromagnetism. Magnetite is dense mineral; density is 5.20 g/cm^3 which are higher than other minerals. For its high density, it makes the composite concrete space effective which is one of the reasons that it used in radiation shielding.

Figure 3.2: Heavy mineral Magnetite

3.2.3 Garnet

Garnet is a common mineral in sand but usually in low quantities. There are some hard to find exceptions – red and very beautiful sand where garnet is indeed the most abundant sand forming mineral. Source rocks of garnet is usually either metamorphic (garnet schist, rarely eclogitic) or igneous (aluminum-rich granite). They form under the same high temperatures and / or pressures that form those types of rocks. Garnet sand is a sub-type of heavy mineral sand. It is a more complex orthosilicate.

Garnet is a moderate type dense mineral; its density varies between $3.5g/cm^3$ to $4.3g/cm^3$. For its high density, it used in radiation shielding.

Figure 3.3: Heavy mineral Garnet

3.2.4 Rutile

Rutile is a titanium oxide mineral with a chemical composition of TiO_2. It is found in igneous, metamorphic and sedimentary rocks throughout the world. Rutile also occurs as needle-shaped crystals in other minerals.

Figure 3.4 : Heavy mineral Rutile

The primary uses of rutile and titanium oxide made from rutile are: manufacturing titanium oxide pigments, manufacturing refractory ceramics, and production of titanium metal. Rutile is dense mineral; and its density is 4.23 g/cm^3 which are higher than other minerals. For its high density, it helpful for radiation shielding.

3.2.5 Zirconium

Zirconium is a very strong, malleable, ductile, lustrous silver-gray metal. Its chemical and physical properties are similar to those of titanium. Zirconium is extremely resistant to heat and corrosion. Zirconium is lighter than steel and its hardness is similar to copper. When it is finely divided, the metal can spontaneously ignite in air, especially at high temperatures. Zirconium powder is black and is regarded as very dangerous fire hazard. Zirconium does not dissolve in acids and alkalis. Zirconium is dense mineral; specific gravity is 6.51 g/cm^3 which are higher than other minerals. For its high density, it makes the concrete space effective which is one of the reasons that it used in radiation shielding.

Figure 3.5: Heavy mineral Zirconium

3.2.6 Unsaturated polymer resin

Unsaturated polyester resin (UPR) as a polymer obtained by the polycondensation reaction between polyacids and polyalcohol. The development of water is the byproduct of this polycondensation process. Specifically, the unsaturated polyester resin, also known by the English acronym UPR, is an easily printable liquid polymer which, once cured (cross-linked with styrene, by the use of particular substances, organic peroxides, named hardeners), keeps the solid shape taken in the mold.

Figure 3.6 : Matrix material Unsaturated polyester resin

The items so realized have exceptional strength and durability characteristics. Unsaturated polyester resins are mostly used in combination with reinforcing materials. The unsaturated polyester resin is used with great success in many industrial sectors, such as in watersports for the creation of windsurfers and pleasure boats. This polymer has been at the center of a real revolution in the boat industry, because it can provide great performances and a very high flexibility of use. The unsaturated polyester resins are also commonly used in the automotive sector (car industry), for their great design versatility, light weight, lower system costs and mechanical strength. This material is used also for buildings, especially in the manufacture of hobs for cookers, tiles for roofs, bathrooms accessories, but also pipes , ducts and tanks.

3.3 Sample Preparation

The primary purpose of this study is to prepare heavy mineral reinforced polymer composite concrete. For this purpose, a number of rectangular samples were made. Some research works have already been conducted for heavy mineral reinforced polymer composite materials for Radiation Protection.

The studied samples were prepared with the composition of various locally available cost-effective materials. Naturally occurring heavy minerals and stones chips 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 stone chips.

In the first step of sample preparation, the mixed aggregates were placed in a concrete mixer tray. 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 3.7. The ratio for the concrete mix design was 18:15:15:52 in accordance to the civil engineering empirical 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”×5”×1.5”) to prepare handy samples.

Figure 3.7: Flow diagram of sample preparation

In this study, six types of concrete block were fabricated. The fabricated samples were used in a series of experiments with Cs-137 and Co-60 radiation sources at the secondary standard dosimeter laboratory (SSDL).

3.4 Description of the Survey Meter, Thermometer and G-10 Gamma Calibrator

3.4.1 Graetz Survey Meter

Graetz Survey Meter is a G-M type Dose rate measuring device with dose indication and alarm function for personal radiation protection when handling ionizing radiation, with RS-232 interface and connection facility for an external probe.

PTB-approved dose rate measuring range: 1.0 µSv/h – 20 mSv/h

Dose rate indication range: 0 nSv/h - 20 mSv/h

Dose indication range: 0 nSv - 10 Sv

Energy range: 40 keV – 1.3 MeV

Dose rate alarm thresholds: 4, free programmable, 1 µSv/h - 20 mSv/h

Dose alarm thresholds: 4, free programmable, 1 µSv - 10 Sv

Temperature range: -30°C up to +60°C

Dimensions / Weight: (152 x 82 x 39) mm approx. 400 g

PTB-Approval No.: 23.51/04.01

German Fire Brigades Approval No.: DL/FW/IdF 080221/1

The photographic view of the **Graetz Survey Meter** used for gamma radiation monitoring (as shown in Figure 3.8).

Figure 3.8: Graetz Survey Meter

3.4.2 Thermometer

A thermometer is a device that measures temperature. Digital precision pocket thermometer (GTH 175/Pt) was used for measuring the temperature of the room. It is handy and very strong precision thermometer with stainless steel-Insertion sensor and PTFE-cable and PTFE-Griff.

Measuring range: -199.9 ^{0}C To +199. 9 ^{0}C

Resolution: 0, 1° C

Accuracy: (at nominal temperature =25 °C) 0,1 %of m.v. ±2 digit (within range of: (-70.0...+199.9°C), (at nom. temperature) probe is calibrated to the device, i.e. the error in the range of 0 to 100°C will be approx. 0,1°C ±1 digit.

Display: 3½ digit, approx. 13 mm high

Nominal temperature: +25 °C

Working temperature: -30 ^{0}C to +45 ^{0}C

Storage temperature: -30 to +70 °C

Power supply: 9V battery type IEC 6F22 (included)

Battery service life: approx. 200 operating hours

Low battery warning: “BAT”

Dimension of device: approx. 106 x 67 x 30 mm (H x W x D). impact resistant ABS plastic housing

Weight: approx. 190 g

The photographic view of the **digital precision pocket thermometer** used for temperature measurement (as shown in Figure 3.9).

Figure 3.9: A digital precision pocket thermometer (GTH 175/Pt).

3.4.3 G-10 Gamma Beam Irradiation

The model G10 is a dual source gamma beam irradiation with a 740 GBq (20 Ci) Cs-137 source and a 38 GBq (1 Ci) Co-60 source. The sources are doubly encapsulated, hermetically sealed, special form source. The source capsule is fabricated of stainless steel. Stated source activity is ±20% but typically falls within ±10%. The source is housed in a stainless steel and tungsten rod. Tungsten is above and below the source limits radiation along the axis of the source rod. The source is moved either the shielded position or exposed position via pneumatic air cylinder in less than 3 seconds. Sensors on the source rod indicate source position. The source rod is designed to be fail – safe: it will return to the shielded position if power or air pressure is lost. In the shielded position, the source is shielded on all sides with lead and tungsten.

The shield is a steel encased lead cylinder. Beam centerline can range from 100 cm to 130 cm, with the standard centerline of 110 cm. the beam height is specified during the design phase of each project. Lead surrounds the tube through which the source travels and provides sufficient shielding to limit radiation level to <6 µSv/hr at 1 meter from the surface to shield when the source is in the shielded position. Overall, size is typically 61 cm wide × 61 cm deep × 76 cm tall for frame and 36 cm× 76 cm for shield with a weight of 820 kg. The beam port holds a removable collimator that is designed based on ISO 4037. It has a series of tungsten disks with steps to trap scattered radiation. It provides a circular radiation beam with a 16.5 degree angle.

The photographic view of the Co-60 and Cs-137 source used as a high energy gamma radiation (as shown in Figure 3.10).

Figure 3.10: Co-60 and Cs-137 Gamma radiation source (Calibrator G10).

3.5 Characteristics of Composite Sample

In the present study, six types of potential shielding material were used to analyze the attenuation property. The fabricated samples were used in a series of experiments with ^{137}Cs and ^{60}Co radiation sources at the secondary standard dosimeter laboratory (SSDL). The photographic images are shown in bellow:

Figure 3.11: Photographs of some prepared samples

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 3.12. The diversity in radiation attenuation performance of the composite block was justified as per the variation in material compositions of respective sample. The break through 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.

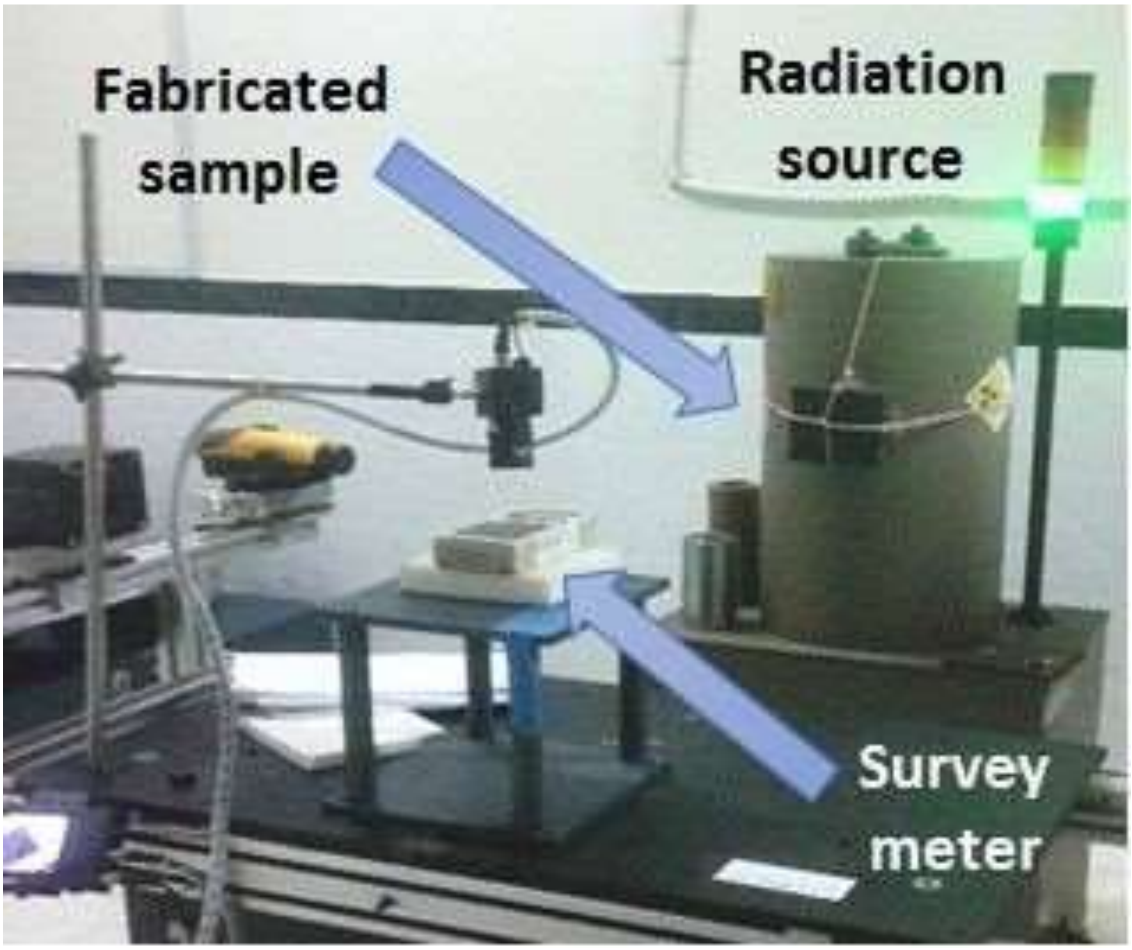

Figure 3.12: Experimental assembly for data collection process

3.6 Photography of the High Energy Gamma Radiation Source with Sample

The representative photographic views of the experimental set-up with some study materials are shown in (Figure 3.13 to 3.18). In the following figures, Magnetite, Ordinary concrete, Rutile , Zirconium, Ilmenite and Garnet, composite and samples are presented for the high energy gamma radiation shielding experiments.

Figure 3.13: Magnetite composite with Co-60 source

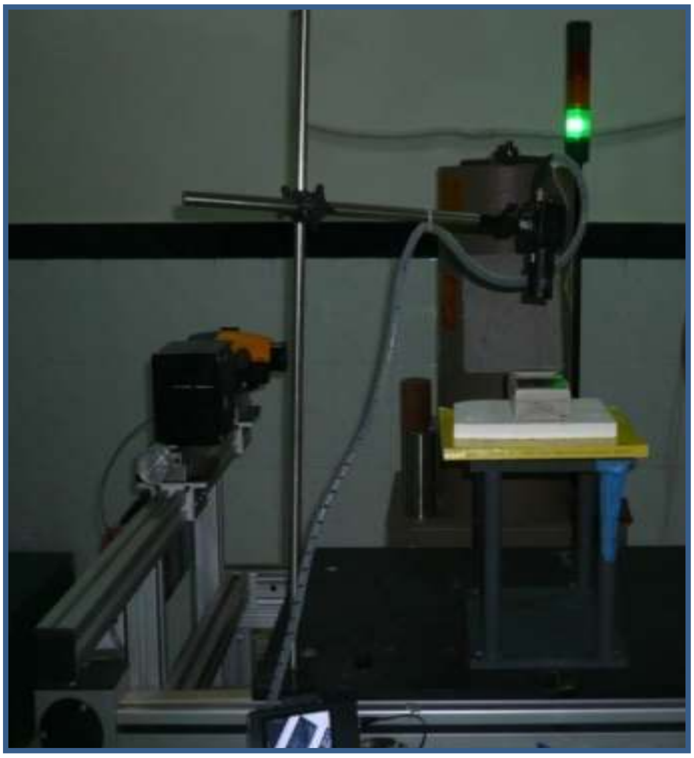

Figure 3.14: Concrete composite with Co-60 source

Figure 3.15: Ilmenite composite with Cs-137 source

Figure 3.16: Zirconium composite with Cs-137source

Figure 3.17: Rutile composite with Co-60 source

Figure 3.18: Garnet composite with Cs-137 source

The photographic view of the control panel system of **Gamma calibrator (G10)** is shown in Figure 3.19

Figure 3.19: The Control panel system of Gamma Calibrator

CHAPTER 4

DISCUSSION ON RESULTS AND RELEVANCE

4.1 Overview

The context of this chapter briefly describes all experimental observations that are presented in the form of numeric data, respective to the individual sample used in the current experiments. A current study on the gamma ray attenuation properties of the six types of locally prepared heavy mineral reinforced polymer composite sample are carried out, which are briefly discussed below.

4.2 Experimental Result for The Gamma Radiation Sources

Table 4.1: Average values of different shielding parameters for **Cs-137** radiation source

Sample name	Thickness **t** (**t** ±0.0114) (cm)	Mass **m** (gm)	Density ***p*** (***p*** ± 0.004) (gm/cm^3)	Initial dose rate D_u (D_u ± 1.58) (mSv/hr)	Dose rate after shield D_t (mSv/hr)	Percent of reduced activity
Magnetite composite	3.81	1890	3.07	57	24.51	58
Ilmenite composite	3.81	1840	2.99	57	26.52	54
Rutile composite	3.81	1752	2.85	57	27.31	52
Garnet composite	3.81	1845	3.01	57	25.37	56
Zirconium composite	3.81	1888	3.07	57	26.01	55
Ordinary concrete	3.81	1415	2.3	57	32.01	43

Table 4.1: (Continue)

Sample name	Linear attenuation co-efficient **μ_l** (cm^{-1})	Mass attenuation co-efficient **μ_m** (cm^2/ g)	Half value layer **HVL** (cm)	Tenth value layer **TVL** (cm)	Sixteenth value layer **SVL** (cm)	Mean free path λ (cm)
Magnetite composite	0.221	0.074	3.13	10.42	12.55	4.53
Ilmenite composite	0.201	0.067	3.49	11.26	13.8	4.96

Table 4.1:(continue)

Sample name	Linear attenuation co-efficient μ_l (cm^{-1})	Mass attenuation co-efficient μ_m (cm^2/g)	Half value layer **HVL** (cm)	Tenth value layer **TVL** (cm)	Sixteenth value layer **SVL** (cm)	Mean free path λ (cm)
Rutile composite	0.193	0.068	3.59	11.93	14.37	5.18
Garnet composite	0.212	0.067	3.27	10.86	13.08	4.72
Zirconium composite	0.206	0.067	3.26	11.18	13.46	4.85
Ordinary concrete	0.151	0.064	4.60	15.25	18.41	43.5

Table 4.2: Average values of different shielding parameters for **Co-60** radiation source

Sample name	Thickness **t** **(t ±0.0114)** (cm)	Mass **m** (gm)	Density ***p*** **(p± 0.004)** (gm/cm^3)	Initial dose rate D_u ($D_u \pm 0.57$) (mSv/hr)	Dose rate after shield D_t (mSv/hr)	Percent of reduced activity
Magnetite composite	3.81	1890	3.07	8.03	4.03	50
Ilmenite composite	3.81	1840	2.99	8.03	4.45	45
Rutile composite	3.81	1752	2.85	8.03	4.62	43
Garnet composite	3.81	1845	3.01	8.03	4.42	45
Zirconium composite	3.81	1888	3.07	8.03	4.01	49
Ordinary concrete	3.81	1415	2.3	8.03	5.35	33

Table 4.2 (continue)

Sample name	Linear attenuation co-efficient μ_l (cm^{-1})	Mass attenuation co-efficient μ_m (cm^2/g)	Half value layer **HVL** (cm)	Tenth value layer **TVL** (cm)	Sixteenth value layer **SVL** (cm)	Mean free path λ (cm)
Magnetite composite	0.181	0.057	3.83	12.72	15.30	5.53
Ilmenite composite	0.161	0.054	4.30	14.30	17.20	6.21
Rutile composite	0.150	0.053	4.62	15.35	18.50	6.67
Garnet composite	0.160	0.053	4.33	14.40	17.30	6.25
Zirconium composite	0.182	0.057	3.80	12.65	15.20	5.49
Ordinary concrete	0.106	0.045	6.54	21.73	26.16	9.43

4.2 Discussion on Experimental Result

The bulk density of the fabricated samples was determined prior to the subsequent experimental evaluations. The density profile is presented in figure 4.1, which indicates a small density variation of different among the heavy mineral composite blocks. The highest density of 3.1 (g/cm^3) 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/cm^3. The observed density profile of the studied samples was found relatively high than the density 2.3 g/cm^3 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.

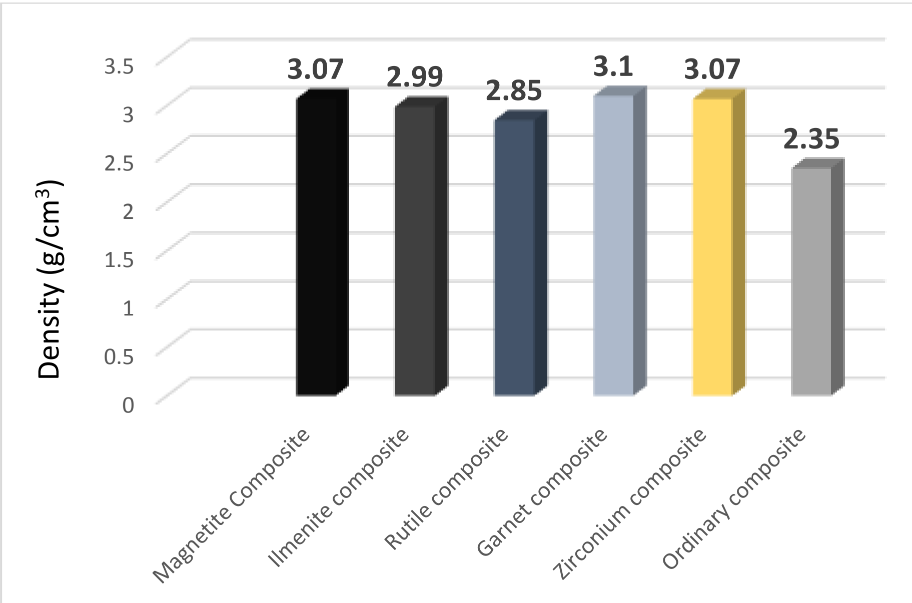

Figure 4.1: Density profile of the fabricated samples

After determining the densities of the aforesaid composite blocks, their Linear attenuation coefficient μ_l, Mass-attenuation coefficient μ_m, Half value layer (HVL) , Tenth value layer (TVL), Sixteenth value (SVL), Relaxation length λ and reduced percentage of beam transmission was determined, and presented in Table 4.1 and Table 4.2 for ^{137}Cs and ^{60}Cosource respectively.

The variation of the experimental μ_l and μ_m were mainly due to their dependence on the respective sample density. The comparative evaluations of radiation attenuation capacity in terms of μ_l between the fabricated shielding material and the conventional shielding material for gamma photons of Cs-137 and Co-60 are shown Figure 4.2. A similar comparison of μ_m is presented in figure 4.3. The estimated experimental values of μ_l and μ_m 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., 1meter) 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 μ_l and μ_m of all the studied heavy mineral composite blocks are found significantly larger than that of the ordinary concrete block, as shown in figure 4.2 and figure 4.3 respectively. The highest values of μ_l and μ_m were found to be 0.227cm^{-1} and 0.0182 cm^2/gm for the Zirconium composite block with Cs-137 source. For other samples, μ_l and μ_m 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 μ_l and μ_m for heavy mineral composite blocks were found relatively larger than the ordinary concrete blocks as shown in figure 4.2 and figure 4.3 respectively. The evaluated μ_l and μ_m

were found closely similar for every composite block with both radiation emitting source, and they were around 0.20 cm^{-1} and 0.06 cm^2 /g for Cs-137 source, whereas for Co-60 source, μ_l and μ_m were around 0.16cm^{-1} and 0.05 cm^2/g. On the other hand, the ordinary concrete block showed very poor radiation attenuation behavior in terms of lower values of μ_l and μ_m. The present study illustrate that the shielding effectiveness of any shielding material depends on its density, the types of chemical composition and the 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.

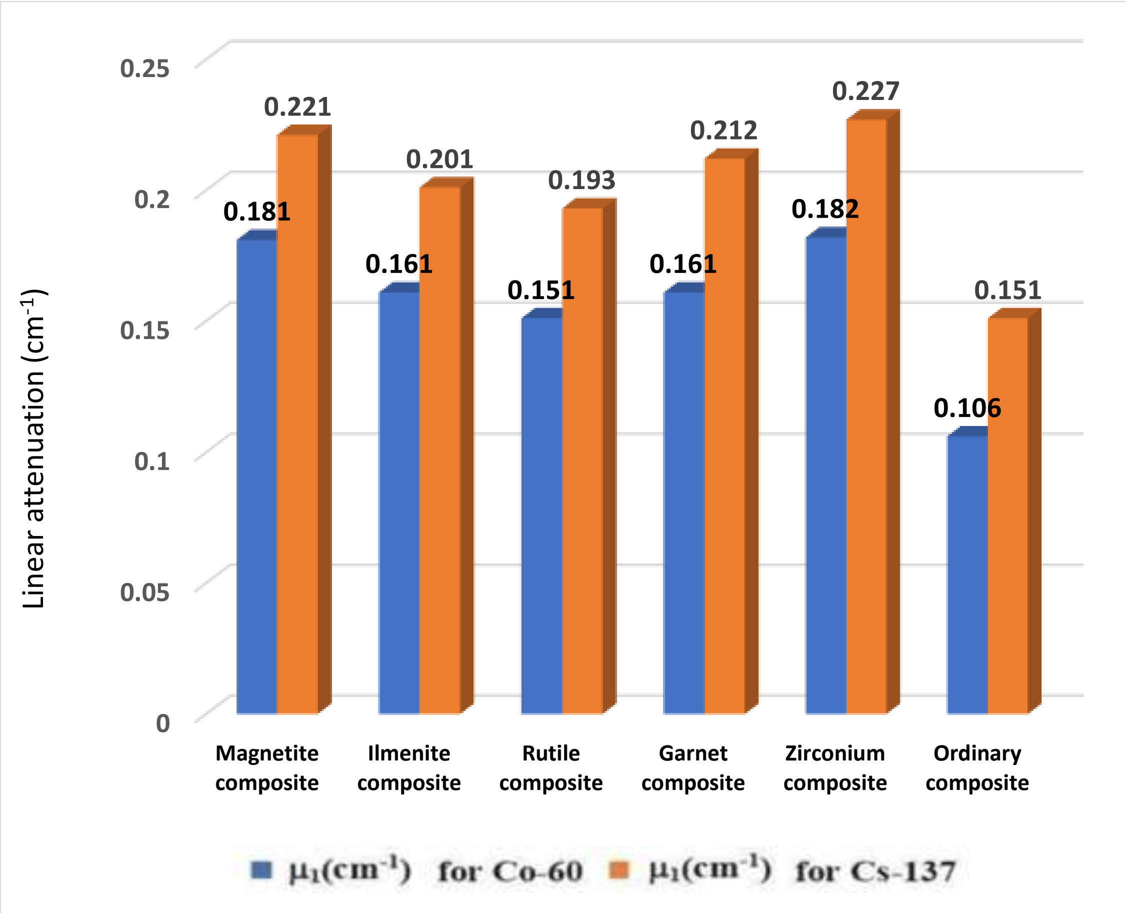

Figure 4.2: Linear attenuation coefficients of various composite samples for Cs- 137 and Co-60 source

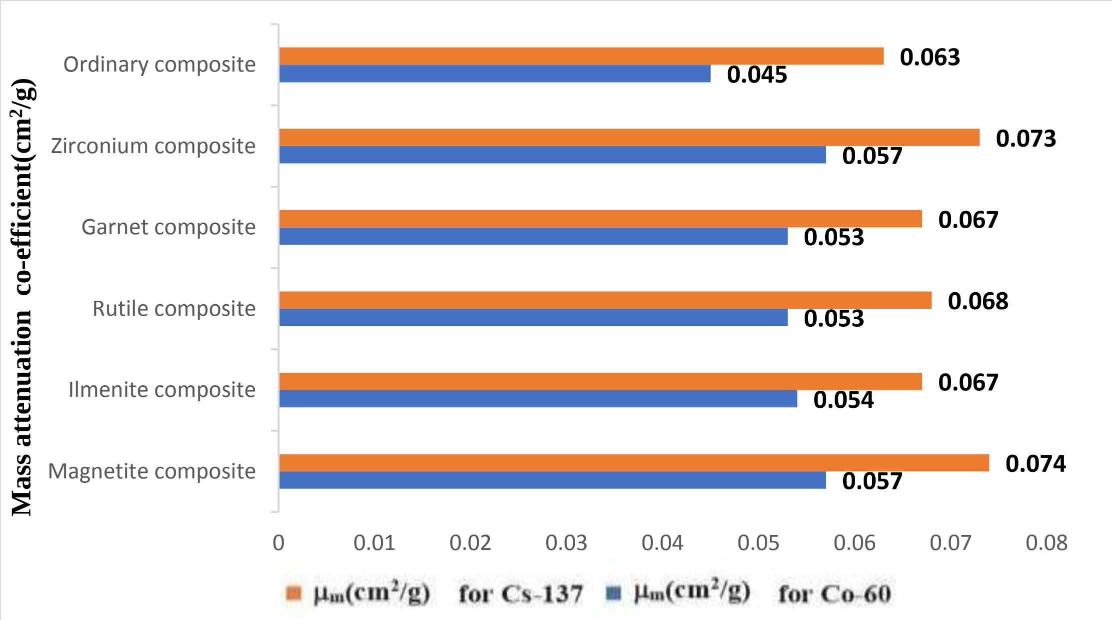

Figure 4.3: Comparative view of mass attenuation coefficients of various composite samples for Cs-137 and Co-60 source.

Figure 4.4 and figure 4.5 demonstrates the competence of our prepared shielding blocks against Cs-137 source and Co-60 respectively and Figure 4.6 demonstrates the comparison between our prepared shielding blocks against low and high energy gamma radiation. 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 5 cm to and 27 cm respectively for both radiations emitting source. figure 4.7 represents the relaxation length of different sample against low and high energy radiation sources. It was observed much better result for all type of heavy mineral containing sand-based composite and the mean free path lies between 4.53 cm to 6.67 cm for low and high energy radiation sources. Whereas the ordinary composite showed very poor results. This figure shows effectiveness of heavy mineral composite with respect to ordinary composite.

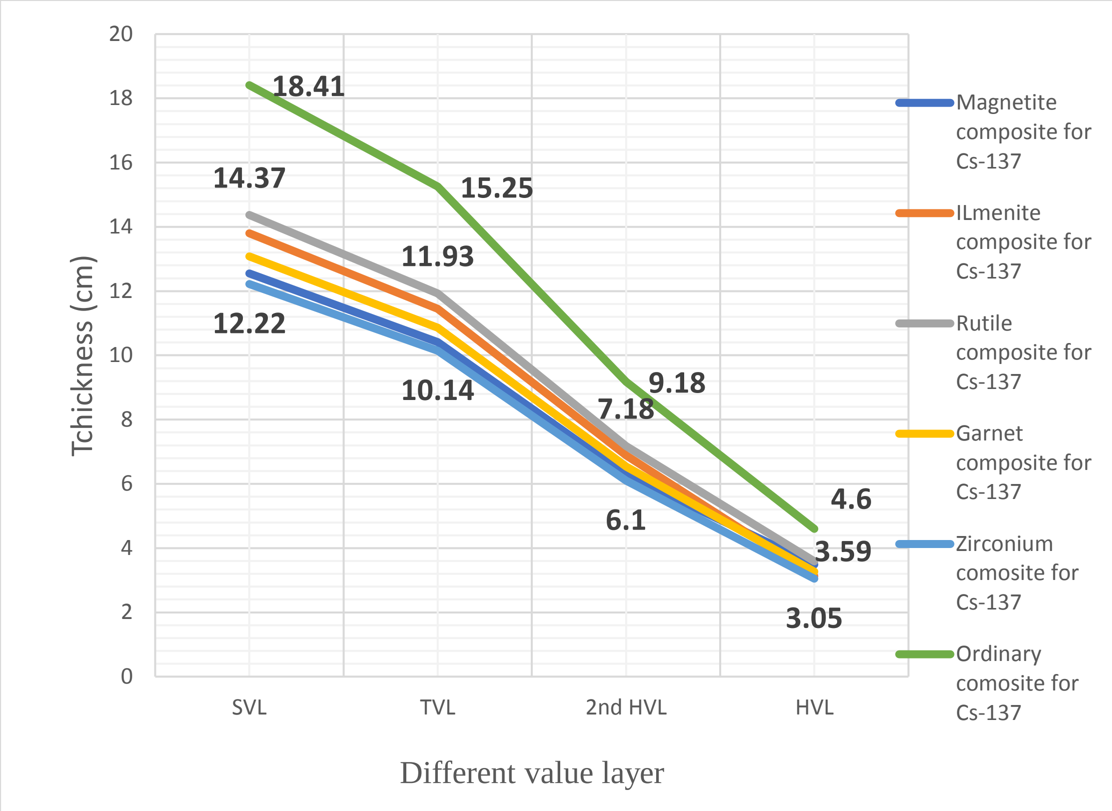

Figure 4.4: Comparative evaluations of experimental HVL, TVL and SVL for the fabricated samples for Cs-137 source.

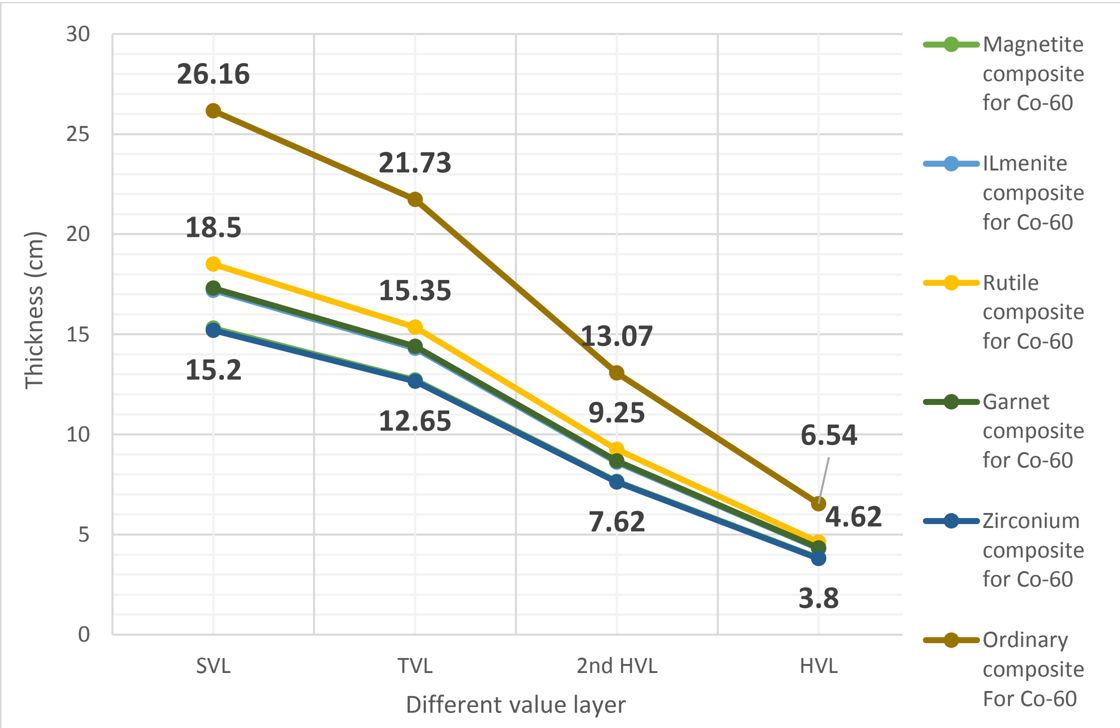

Figure 4.5: Comparative evaluations of experimental HVL, TVL and SVL for the fabricated samples for Co-60 source.

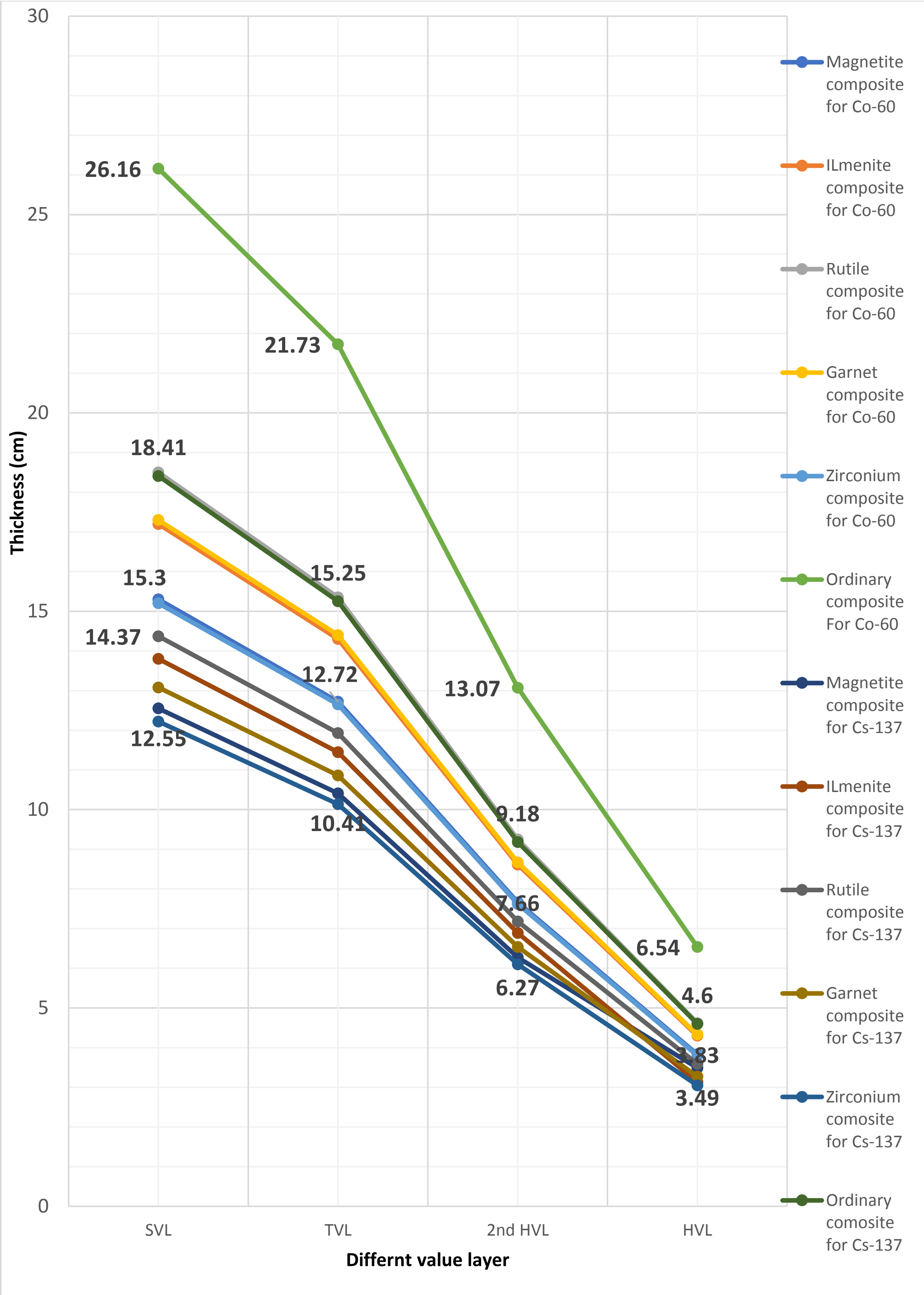

Figure 4.6: Comparison between experimental HVL, TVL and SVL for the fabricated samples for Co-60 and Cs-137 radiation source.

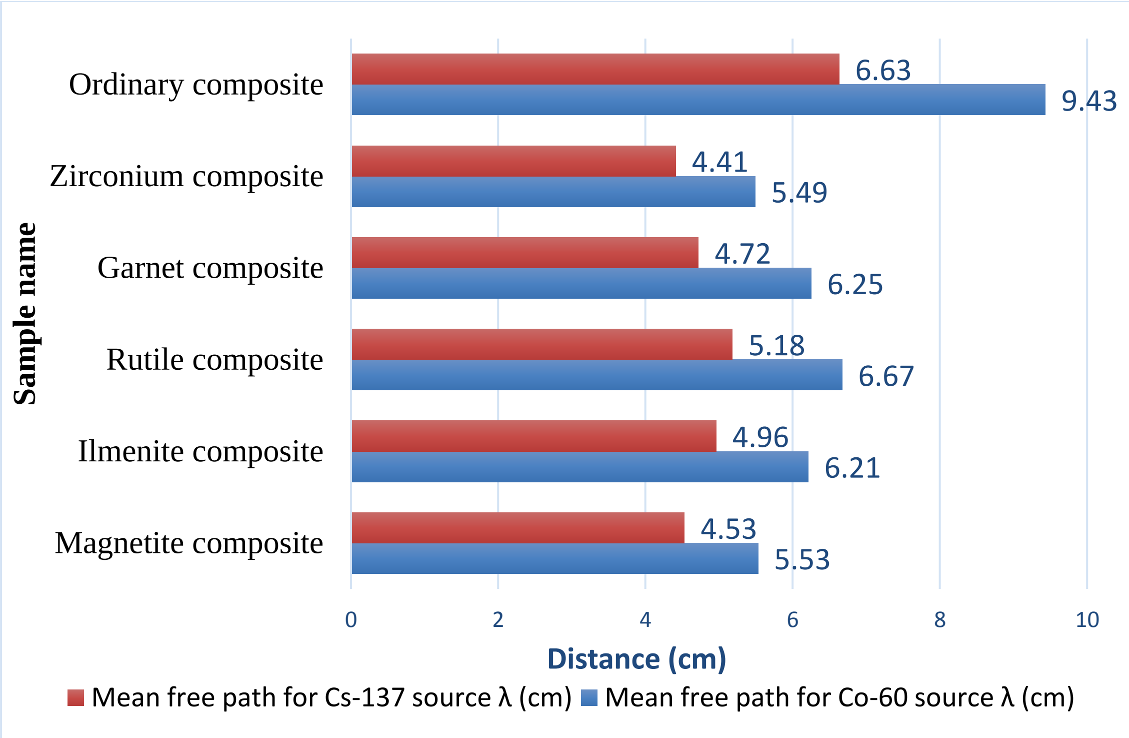

Figure 4.7: Mean free path of different sample against high and low energy radiation source.

CHAPTER 5

CONCLUSION AND RECOMMENDATIONS FOR FUTURE WORK

Radiation detection and protection is very important in nuclear facilities. The objective of this work to Fabrication and Characterization of Shielding Properties of Heavy Mineral Reinforced Polymer Composite Materials which might be useful for radiation protection of gamma radiation. G-10 gamma emitting calibrator contains the radioactive sources Cs-137 (0.651 MeV) and Co-60 (1.25 MeV). The high and low energy gamma radiation shielding property of the current study materials were investigated using Co-60 source with activity (1 Ci) and Cs-137 source with activity (20 Ci) respectively. The Greatz Survey meter is a G-M type survey is used for measuring gamma dose rate without and with the studied shielding material.

In this practical experiment 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. Investigation of attenuation coefficient and attenuation factor has been determined with several potential shielding materials. Comparative assessment between the characterized shielding material and the ordinary shielding material had performed based on the experimental observations. The Magnetite composite exhibits relatively good attenuation performance in the case of 0.661 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. 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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Appendix

Appendix 1: Average thickness for different composite samples

Sample name	Average Thickness		SD
Magnetite composite	3.81	±	0.0114
Ilmenite composite			
Rutile composite			
Garnet composite			
Zirconium composite			
Ordinary concrete			

Appendix 2: Average Density for different composite samples

Sample name	Average Density		SD
Magnetite composite	3.07	±	0.004
Ilmenite composite	2.99		
Rutile composite	2.85		
Garnet composite	3.01		
Zirconium composite	3.07		
Ordinary concrete	2.3		

Appendix 3: Average Initial activity for different composite samples of Cs-137 source

Sample name	Initial Dose rate		SD
Magnetite composite	57	±	1.58
Ilmenite composite			
Rutile composite			
Garnet composite			
Zirconium composite			
Ordinary concrete			

Appendix 4: Average Initial activity for different composite samples of Co-60 source

Sample name	Initial Dose rate		SD
Magnetite composite	8.03	±	0.57
Ilmenite composite			
Rutile composite			
Garnet composite			
Zirconium composite			
Ordinary concrete			

List of Publications

Conference papers

1. **“Fabrication and Characterization of Heavy Mineral Reinforced Polymer- based Concrete Composite for Gamma Radiation Shielding.**”

Md. Muminul Haque*, Abu Z. M. Salahuddin, Ruhul A. Khan.

4th International Conference on Structure, Processing and Properties of Materials 2018.Organized by Department of Materials and Metallurgical Engineering (MME). Bangladesh University of Engineering and Technology (BUET), Dhaka- 1000.

2. **“Construction and Characterization of Heavy Mineral Armored Polymer- based Concrete Composite for Gamma Radiation Protection.”**

Md. Muminul Haque*, Md. Borhan Uddin , Abu Z. M. Salahuddin, Ruhul A. Khan.

International Conference on Physics -2018. Organized by Bangladesh Physical Society. Department of Physics, University of Dhaka, Dhaka-1000.

Published papers

1. “Fabrication and Characterization of Shielding Properties of Heavy Mineral Reinforced Polymer Composite Materials for Radiation Protection”

M. M. Haque, M. B. Uddin, A. Z. M. Salahuddin, Ruhul A. Khan.

European Journal of Engineering Research and Science .Vol. 4, No. 3, March 2019