# A COMPARATIVE CALCULATION OF DOSE RATE USING RAD PRO CALCULATOR FOR SHIELDING REQUIREMENTS OF COBALT-60 AND CESIUM-137 SOURCES AT CERT.

By


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## Declaration

I declare that the work in this Project Report entitled COMPARATIVE CALCULATION OF DOSE RATE USING RAD PRO CALCULATOR FOR SHIELDING REQUIREMENTS OF COBALT-60 AND CESIUM-137 SOURCES AT CENTER FOR ENERGY RESEARCH AND TRAINING has been carried out by me in the Department of Physics. The information derived from the literature has been duly acknowledged in the text and a list of references provided. No part of this project report was previously presented for another degree or diploma at this or any other Institution.

Ibrahim Muhammad

## Certification

This project report entitled A COMPARATIVE CALCULATION OF DOSE RATE USING RAD PRO CALCULATOR FOR SHIELDING REQUIREMENTS OF COBALT-60 AND CESIUM-137 SOURCES AT CERT.; By Ibrahim MUHAMMAD meets the regulations governing the award of the degree of Masters of Science (Radiation Biophysics) of the Ahmadu Bello University, and is approved for its contribution to knowledge and literary presentation

Professor S.A.Jonah __________________ _____________

Chairman, Supervisory Committee Signature Date

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## Dedication

To Muhammad Tukur Naiya and Usman Salihu Aliyu.

# Abstract

A Comparative Calculation of Dose Rate Using Rad Pro Calculator for Shielding Requirements of Cobalt-60 and Cesium-137 Sources at CERT. was carried out using seven shielding materials: Lead, Concrete, Iron, Uranium, Lead glass, Aluminum and Tungsten by the Rad Pro Calculator. The study was to understand the attenuation strength of these materials in relation to one another; and the behavior of the photon dose rate in relation to the thickness of the shield. These materials and of varying thicknesses were placed in between the radiation source and the detector and the dose rate were calculated .It has been observed that each of the results has a similar behavior, in that the photon dose rate is inversely proportional to the absorber thickness in an almost exponential form. As the absorber's thickness increases, the photon dose rate decreases; conversely. From a given distance, the result of these seven absorbing materials (Lead, Concrete, Iron, Uranium, Lead glass, Aluminum and Tungsten) look very similar, but on a closer look, one will identify a striking and important differences amongst these seven materials especially tungsten. While only 6cm thickness of tungsten is required to reduce the photon dose rate of 5mCi of $^{60}$Co at 1metre distance from the source to detector to 0.08µSv/h, 6cm of uranium thickness reduces it to 0.11µSv/h , 10cm of Lead glass thickness is required to reduce it to 0.08µSv/h,12cm of lead reduce to 0.07µSv/h,22cm of Iron shielding reduce the dose rate to 0.07µSv/h,42cm of Aluminum shielding reduce the dose rate to 0.11µSv/h, 70cm of concrete shielding require to reduce the dose rate to 0.06µSv/h.This then goes to show that uranium is a far better absorber of $^{60}$Co gamma photon when compared to tungsten, lead glass, lead, iron, aluminum and concrete can be used to shield against gamma ray. Conversely, 3cm of uranium shielding is require to reduce the photon dose rate of 22mCi of Cs-137 at 1m from the source to detector to 0.05µSv/h, 4cm of tungsten to 0.07µSv/h, lead glass to 0.05µSv/h, lead to 6cm to 0.07µSv/h, 16cm of iron shielding to 0.07µSv/h, 32cm of aluminum shielding to 0.09µSv/h, 55cmof concrete shielding to 0.08µSv/h. In conclusion, Rad Pro Calculator simulation is a veritable tool for modeling certain real life situations, and very useful in shielding calculation. The result have also show that out of the seven materials, tungsten and uranium is a better shield because it requires just a few thicknesses of it to cut down the photon dose rate to acceptable limit, then followed by lead glass, lead, iron, aluminum and concrete being the least.

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# CHAPTER ONE

## 1.0 INTRODUCTION

### 1.1 General Background

The term shielding implies the deliberate introduction of material between the radiation and an object to reduce radiation intensity and damage to the object. This is preferred method of controlling radiation because it results in intrinsic safe working conditions (Martin and Harbison, 1980). The type of shielding used and the thickness require depend on the type of radiation, its energy, the flux and the dimension of the source (Esien-umo, 2007).

#### 1.1.1 Shielding of x-ray generators:

The hazards of x rays were recognized within months of Roentgen's 1895 discovery, but dose limitation by time, distance, and shielding was at the discretion of the individual practitioner until about 1913. Only then were there organized professional efforts to establish guides for radiation protection, and not until about 1925 were there instruments available to quantify radiation exposure. In his monumental survey of organization for radiation protection, Taylor (1979) begins with British and German efforts at establishing guidance for x-ray shielding. In 1913, the German Radiological Society on X-Ray Protection Measures issued recommendations that 2 mm of lead shielding was needed, regardless of generator voltage, workload, or filtration. In Britain, the Roentgen Society addressed radiation protection, stressing operator protection, the need for beam collimation, and the importance of scattered X rays. No explicit recommendations on shielding requirements were issued.

### 1.1.2 Shielding of Radium sources:

In 1927, the International Committee on X-Ray and Radium Protection issued the following recommendations for storage of radium sources. Tubes and applicators should have at least 5 cm of lead shielding per 100 mg of radium. Radium solutions required lead shielding ranging from 15 cm for a 0.5 g source to 30 cm for a 2g source. NCRP Report 2 (1934) specified a 3-m protective zone around stored Ra sources, and recommended exhaust fans or hoods for removal of radon and decay products escaping from unsealed sources. Shielding of stored sources was revised to 4 cm of lead for 100 mg to 6 cm for 300 mg. NCRP Report 4 (1938) addressed dosimetry for gamma rays emitted from radium sources. It was not until 1941 that there was established a tolerance dose for radium, expressed in terms of a maximum permissible body burden of 0.1 Ci. This was done largely in consideration of the experiences of early "radium-dial" painters and the need for standards on safe handling of radioactive luminous compounds (NCRP, 1941).

### 1.1.3 Advances in Monte Carlo computational methods:

The Monte Carlo method of simulating radiation transport computationally has its roots in the work of John von Neumann and Stanislaw Ulam at Los Alamos in the 1940's. Neutron transport calculations were performed in 1948 using the ENIAC digital computer, which had commenced operations in 1945. Major advances were made in the 1950's by Kahn (1950), Goertzel and Kalos (1958), and Cashwell and Everett (1959). Berger (1955, 1956) began a Monte Carlo effort at the National Bureau of Standards that thrived for several decades

His decade of the 1960's the 1960's saw the technology of nuclear reactor shielding consolidated in several important publications. Blizard and Abbott (1962) edited and released a revision of a portion of the 1955 Reactor Handbook as a separate volume on radiation shielding, recognizing that reactor shielding had emerged from nuclear reactor physics into a discipline of its own. In a similar vein, the first volume of the Engineering Compendium on Radiation Shielding (Jaeger, 1968) was published. These two volumes brought together contributions from scores of authors and had a great influence on both practice and education in the field

1.1.4 Structure shielding:

Structure shielding against nuclear-weapon fallout required careful examination of the atmospheric transport of gamma rays of a wide range of energies and expression of angular distributions and related data in a manner easily adapted to analysis of structures. There was a need to assess, at points within a structure, the ratio of interior dose rates to that outside the building, called reduction factors. Calculations were completed at the National Bureau of Standards by Lewis Spencer (1962) using the moments method of transport calculations which had been used so successfully in calculation of buildup factors. The engineering methodology was developed by Eisenhauer (1964) and rapidly deployed by the Office of Civil Defense for shelter analysis. The body of theoretical, analytical, and experimental support was documented by Spencer et al., (1980)

1.1.5 Computer applications:

The 1980's and 1990's were decades of revolution for the computational aspects of radiation shield design and analysis. The advent of inexpensive personal computers with rapidly increasing speeds and memory freed the shielding analyst from dependence on a few supercomputers at national laboratories. Many shielding codes that could previously run only on large mainframe computers were reworked to run on small personal computers thereby allowing any shielding analysts to perform detailed calculations that only a privileged few were able to do previously. At the same time many improvements were made to the transport codes and their algorithms. MCNP has gone through a series of improvements adding new capabilities and improvements, such as new variance reduction methods, tallies, and physics models. It was translated to the FORTRAN-77 standard in 1983 and to the FORTRAN-90 standard in 2003. New releases appear every few years, and the version as of this writing is MCNP-5. It has also spun off a second version MCNPX with a capability of treating 34 types of particles with energies up to 150 MeV. General purpose discrete-ordinates codes were extensively improved with many novel acceleration schemes introduced to improve their speeds. An excellent review of many such improvements is given by Adams and Larsen

1.1.6 Radiation dose limit:

For occupational exposure of workers over the age of 18 years, the dose limits are: (a) An effective dose of 20 mSv per year averaged over five consecutive years (100 mSv in 5 years), and of 50 mSv in any single year; (b) An equivalent dose to the lens of the eye of 20 mSv per year averaged over 5 consecutive years (100 mSv in 5 years) and of 50 mSv in any single year; (c) An equivalent dose to the extremities (hands and feet) or the skin of 500 mSv in a year. Additional restrictions apply to occupational exposure for a female worker who has notified pregnancy or is breast-feeding  For occupational exposure of apprentices of 16 to 18 years of age who are being trained for employment involving radiation and for exposure of students of age 16 to 18 who use sources in the course of their studies, the dose limits are: (a) An effective dose of 6 mSv in a year; (b) An equivalent dose to the lens of the eye of 20 mSv in a year; (c) An equivalent dose to the extremities (hands and feet) or the skin65 of 150 mSv in a year. For public exposure, the dose limits are: (a) An effective dose of 1 mSv in a year; (b) In special circumstances, a higher value of effective dose in a single year could apply, provided that the average effective dose over five consecutive years does not exceed 1 mSv per year; (c) An equivalent dose to the lens of the eye of 15 mSv in a year; (d) An equivalent dose to the skin of 50 mSv in a year.

### 1.2 Statement of Research Problems

The provision of adequate shielding barrier is one method of controlling exposure to gamma rays. To ensure that adequate protection is afforded both workers and member of the public, adequate shielding barrier is essential to attenuate the intensity of gamma rays to acceptable limits. The standard or general concept of provision of radiation shielding barrier for radiation installation begins with the designing of radiation shielding structures by qualified expert ( medical or health physicist) to ensure that the required degree of protection is achieved (NCRP, 2005). This is followed by the correct designed radiation shielding structures.(Esien-umo, 2007)

Based on this realization, this study will try to determine the radiation dose rate through a given shield material and Thickness using radcalculator. For the C0-60 and Cs-137 sources at the CERT for deploying them in radiation dosimetry applications.

## 1.3 Aim and Objectives

The aim of the research is to design radiation shielding for (5mCi and 22mCi) 60Co and Cs-137 gamma ray sources at Centre for Energy Research and Training (CERT) ABU-Zaria, using calculated dose rate as function of shielding thickness.

The specific objectives are;

i. To make contribution to the radiation protection and safety of radiation workers and member of the public at Centre for Energy Research and Training, ABU-Zaria by investing different radiation shield barrier using calculated dose rate with Radprocalculator.

ii. Shield design is to find the thickness of a material or a combination of materials that would attenuate the intensity of the radiation to the level that would not adversely affect individuals in the vicinity of the gamma ray radiation field.

**iii.** Proposing shielding design for the sources based on calculation data

## 1.4 Justification of Research Study

The provision of enough shielding barrier in any radiation source installation is very vital as this is the preferred method of control of external radiation (martin and harbison, 1986). The effectiveness of the shielding barrier gives the radiation worker a sense of security when radiation intensity is reduced to the shielding design dose limit and members of the public not exposed above the recommended shielding design dose (Muhammad, 2004).

Two gamma ray sources cobat-60 and cesium-137 were acquired by CERT in 2000 to perform a variety of research projects. The sources will therefore be put to research usage.

For dosimetry applications by investigating different shielding materials for the design of barriers

## CHAPTER TWO

### 2.0 LITREATURE REVIEW

### 2.1Absorbed Dose

Usually the interaction of radiation with matter involves a transfer of energy from the radiation to the matter. Ultimately, the energy transferred either to tissue or to a radiation shield is dissipated as heat. The radiation dose depends on the intensity and energy of the radiation, the exposure time, the area exposed and the depth of energy deposition. Various quantities such as the absorbed dose, the effective dose and the equivalent dose have been introduced to specify the dose received and the biological effectiveness of that dose. One of the earliest established phenomena regarding radiation is its ability to ionize a gas. On this basis, the unit called the roentgen (R) was introduced. The roentgen is defined as the amount of exposure that will create $2.58 \times 10^{-4}$ C of singly charged ions in 1 kg of air at STP. Since about 34 eV of energy is needed to produce one ion pair in air, 1R corresponds to energy absorption per unit mass of 0.0088 J $kg^{-1}$. Today it is more usual to use the quantity called the absorbed dose ($D$) which specifies the amount of radiation absorbed per unit mass of material. The original unit for this quantity, the rad (radiation absorbed dose), is equal to 100ergs per gram (1 erg = $10^{-7}$ J). The modern SI unit is the gray (Gy) where 1 Gy = 1 J $kg^{-1}$ = 100 rad. In dosimetry, it is useful to define an average dose for a tissue or organ

$D_T$. The absorbed dose to the mass $\delta m_T$, is defined as the imparted energy $\delta E_T$ per unit mass of the tissue or organ, i.e. (Magill, 2013)

$$D_T = \delta E_T/\delta m_T \qquad \text{2.0}$$

## 2.2 Dose Rate

The absorbed dose rate is the rate at which an absorbed dose is received. The units are Gy/s, mGy/hr, etc. Biological effects depend not only on the total dose to the tissue but also on the rate at which this dose was received. In organisms, mechanisms exist which enable molecules such as deoxyribonucleic acid (DNA) to recover if they have not been too badly damaged. Hence it is possible for organs to recover from a potentially lethal dose provided that the dose was supplied at a sufficiently slow rate. (Magill, 2013)

## 2.3 Quality or Weighting Factor

The biological effect of radiation is not directly proportional to the energy deposited by radiation in an organism. It depends, in addition, on the way in which the energy is deposited along the path of the radiation, and this in turn depends on the type of radiation and its energy. Thus the Biological effect of the radiation increases with the linear energy transfer (LET) defined as the mean energy deposited per unit path length in the absorbing material (units keV $\mu m^{-1}$). Thus for the same absorbed dose, the biological effect from high LET radiation such as $\alpha$ particles or neutrons is much greater than that from low LET radiation such as $\beta$ or $\gamma$ rays. The quality or weighting factor, $w_R$, is introduced to account for this difference in the biological effects of different types of radiation.(Magill, 2013)

The weighting factors for the various types of radiation and energies are given in Table 2.0

Table 2.0 Quality or weighting factors for different types of radiation

| **Type and energy range** | **Radiation weighting factor, WR** |
|---|---|
| **Gamma rays and x rays** | **1** |
| **Beta particles** | **1** |
| **Neutrons, energy** | |
| **<10 keV** | **5** |
| **> 10 keV to 100 keV** | **10** |
| **> 100 keV to 2 MeV** | **20** |
| **> 2 MeV to 20 MeV** | **10** |
| **> 20 MeV** | **5** |
| **Alpha particles** | **20** |

*Source: ICRP Publication 74. Annals of the ICRP* ***26*** *(3/4), 1996*

$$w_R = \begin{cases} 2.5 + 18.2e^{-[\ln(E_n)]^2/6}, & E_n < 1 \text{ MeV} \\ 5.0 + 17.0e^{-[\ln(2E_n)]^2/6}, & 1 \text{ MeV} \leqslant E_n \leqslant 50 \text{ MeV} \\ 2.5 + 3.25e^{-[\ln(0.04E_n)]^2/6}, & E_n > 50 \text{ MeV} \end{cases}$$

Continuous function in neutron energy, En (MeV),

For the calculation of radiation weighting factors for neutrons

**2.4 Equivalent Dose**

Based on the weighting factors of different types of radiation, the equivalent Dose $H$ is introduced to indicate the biological effects of radiation Exposure. The equivalent dose $H_T$ in tissue or organ is defined as the product of the absorbed dose and the quality or weighting factor, i.e.

$H_T = D_{T,R} \cdot W_R$……………………………………………………………………..2.1

Where $D_{T,R}$ is the averaged absorbed dose in tissue T from a given type of Radiation R. It a point in the body should produce approximately the same biological effect. However, a given should be clear that equal equivalent doses from different Sources of radiation delivered to equivalent dose will in general produce different effects in different parts of the body. A dose to the hand may be considerably less serious than the same dose to blood forming organs. (Magill, 2013)

If there are several types of radiation present, then the equivalent dose is the weighted sum over all contributions, i.e.

$$H_T = \sum_R (D_{T,R} \cdot w_{R,}) \quad \ldots 2.2$$

The SI unit of dose is the Sievert, Sv (1 Sv = 1 J $kg^{-1}$, the old unit is the rem, 1 Sv = 100 rem). This is the equivalent dose arising from an absorbed dose of 1 Gy. Hence for γ rays, where $w_R = 1$, an absorbed dose of 1 Gy gives an equivalent dose of 1 Sv. The same absorbed dose

For α particles, where $w_R = 20$, gives an equivalent dose of 20 Sv. The equivalent dose rate is the rate at which an equivalent dose is received, i.e.

$$dH_T/dt = dD_{T,R}/dt \cdot w_R. \quad \ldots 2.3$$

The equivalent dose rate is expressed in Sv/s or mSv/hr.

**2.5 Effective Dose**

The biological effects to a person exposed to radiation depend on the Nature of the radiation and the parts of the body irradiated. Certain organs of the body are more sensitive to radiation than others. For this reason, tissue weighting factors $W_T$ are introduced to account for the damage Sensitivities to different tissues. On this basis, the effective dose ($E$) is defined as the sum of the equivalent doses to different tissues, i.e. (Magill, 2013)

$E= \sum_{T}(H\mathrm{T} \cdot W\mathrm{T},)$ ……………………………………………………………..2.4

Table 2.1: The values for tissue weighting factors are given in Table below:

| Organ | Tissue weighting factor $W_T$ |
|---|---|
| Gonads | 0.20 |
| Colon | 0.12 |
| Bone marrow (red) | 0.12 |
| Lung | 0.12 |
| Stomach | 0.12 |
| Bladder | 0.05 |
| Chest | 0.05 |
| Liver | 0.05 |
| Thyroid gland | 0.05 |
| Esophagus | 0.05 |
| Skin | 0.01 |
| Bone surface | 0.01 |
| Adrenals, brain, small intestine, kidney, muscle, pancreas, spleen, thymus, uterus (the weighting factor 0.05 is applied to the average dose of these organs) | 0.05 |

∗ ICRP Publication 74, Annals of the ICRP **26** (3/4) (1996)

**2.6** Absorption of Gamma Radiation

The attenuation coefficient is a measure of how photons are removed from the beam under conditions of good geometry. Attenuation is a result of three basic processes: the photoelectric effect (pe), Compton scattering (cs), and pair production (pp) and the total attenuation coefficient is a sum of the attenuation coefficients for these processes, i.e.

$\mu = \mu pe + \mu cs + \mu pp$……………………………………………………………….2.5

Photoelectric absorption results when a photon interacts with a bound electron. If the energy of the photon is greater than or equal to the binding energy of the electron, the electron is released with kinetic energy equal to any excess energy of the photon over the binding energy. This photoelectron then dissipates its energy to the medium mainly by excitation and ionization. Compton scattering results from elastic scattering of the photon with weakly bound or “free” electrons. In this process, the scattered photon has less energy than the incident photon. Since the collision is elastic, the Electron gains this loss in photon energy. Pair production results when the energy of the photon exceeds 1.02MeV. In the neighborhood of a heavy nucleus, such a photon can spontaneously disappear and results in the formation of an electron-positron pair. Photon energy in excess of that needed to form the pair appears as kinetic energy of the particles. The positron and electron are projected in the forward direction (relative to that of the initial photon) and lose their kinetic energy by excitation, ionization, Bremsstrahlung etc. Finally, when the positron has lost its kinetic energy, it will combine with an electron to produce annihilation radiation consisting of two 0.51 MeV photons. The photons may then be lost from the medium or may undergo Compton scattering or photoelectric absorption. The total attenuation coefficient $\mu$ given above is the fraction of the energy of the beam that is *removed* per unit distance in the medium. The energy absorbed in the medium is determined by the energy absorption coefficient $\mu$en. The difference between $\mu$and $\mu$en results from the fact that energy may be lost from the medium through Compton scattering and by annihilation radiation. For dose calculations in tissue for example, the energy absorption coefficient $\mu$en must be used. For shielding calculations (Magill, 2013).

**2.**7Attenuation of Gamma Radiation

**2.7.1** Linear Attenuation Shielding Formula:

$$I_B = I_A * e^{-\mu x} \text{……………………………………………………………………….} 2.6$$

Where:

$I_B$ = the shielded dose rate

$I_A$= the initial dose rate

µ= the linear attenuation coefficient in –cm

x= the shielding thickness in cm

The linear attenuation coefficient can be considered as the fraction of photons that interact with the shielding medium per centimeter of shielding. This coefficient assumes that all photons that Interact are removed and ignores Compton scatter and pair production photons (underestimates the shielded dose rate and the shielding required). It is also known as narrow beam conditions because the source and detector are assumed to be collimated and the measurement made at a short distance. No photons are scattered. The only way to make this happen is to Side and back shield the source and the detector (collimate). This only applies at close distances though. Further away, the Air scatters the photons in real life. This is idealistic and without the collimation or at a longer distance, the dose-rate is underestimated.

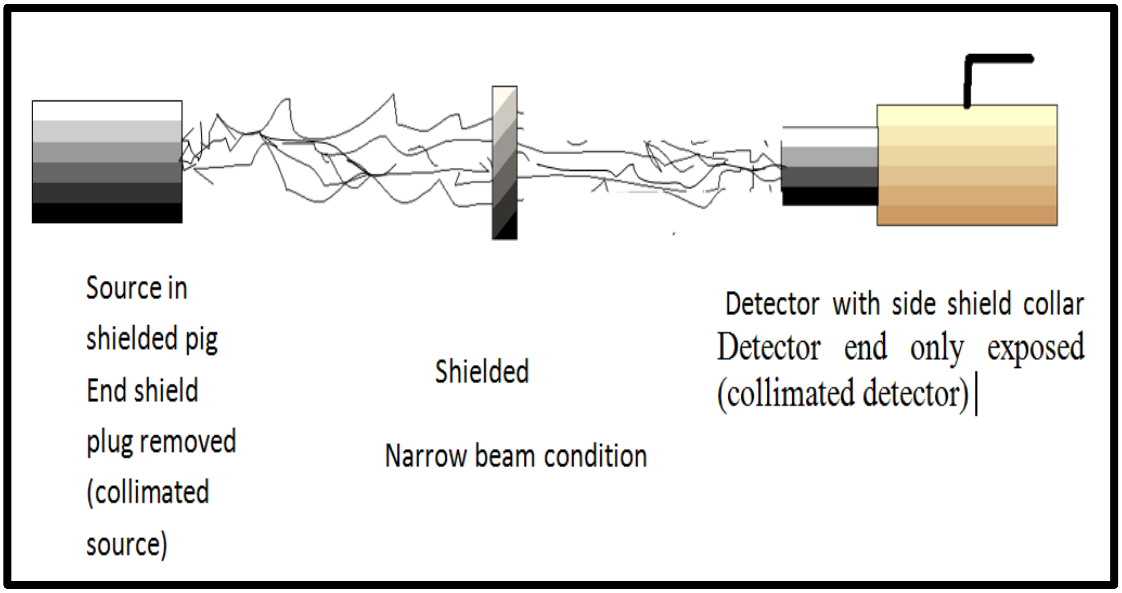



Figure 2.1: Linear Attenuation Shielding (McGinnis, R. MIT Rad Pro Calculator Software Development, Shielding Equations and Buildup Factors Explained. Retrieved from www.radprocalculator.com/Files/ShieldingandBuildup.pdf on November, 2014.)

**2.7.2** Linear Energy Absorption Shielding Formula:

$$I_B = I_A * e^{-\mu en x} \qquad \text{..............} 2.7$$

Where:

$I_B$ = the shielded dose rate

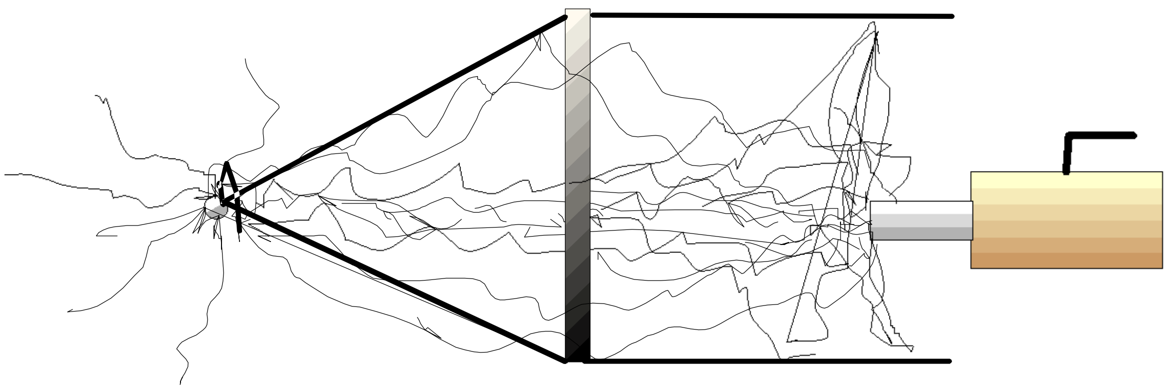
$I_A$ = the initial dose rate

μne= linear energy absorption attenuation coefficient

x = the shielding thickness in cm

The linear energy absorption attenuation coefficient can be considered the fraction of energy removed from the photons by the shielding medium per centimeter of shielding or the fraction of energy absorbed. This coefficient takes into account Compton scatter and pair production photon but it assumes that all scatter photons reach the detector (overestimates the shielded dose rate and the shielding required). It is also known as broad beam conditions because the source and detectors assumed to be uncollimated.

Scattered electromagnetic energy included (But all scattered photons are assumed to reach the detector). This is unrealistic and overestimates the shielded dose-rate.



Unshielded source    Shield    Unshielded detector
Broad Beam Conditions

Figure 2.2: Linear Energy absorption shielding

**2.7.3** Linear Attenuation Shielding Formula with Buildup**:**

$$I_B = I_A * b * e^{-\mu x} \quad \text{……} 2.8$$

Where:

$I_B$ = the shielded dose rate

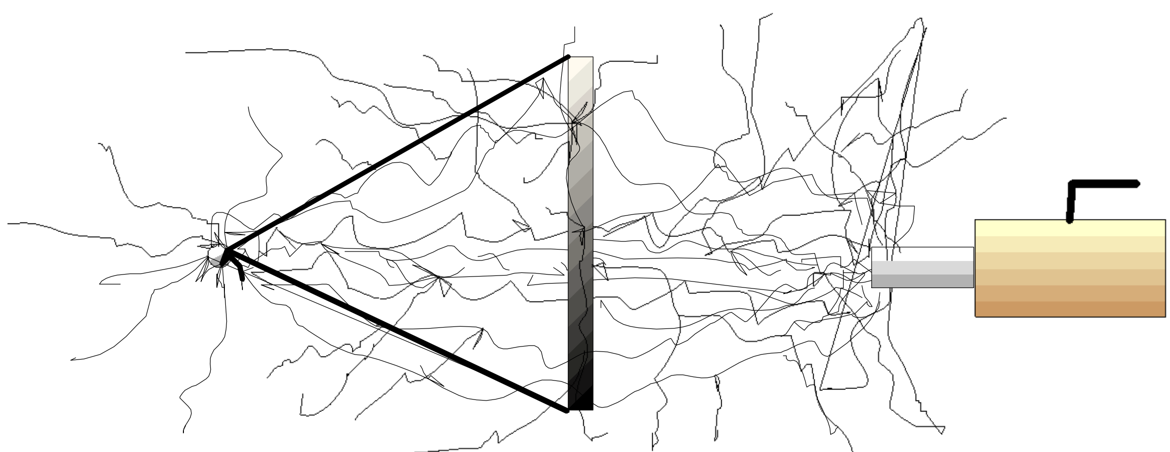
$I_A$ = the initial dose rate

b = the buildup factor for one energy at the shield thickness x

μ= the linear attenuation coefficient in – cm

x = the shield thickness in cm.

This formula attempt to estimate the correct number of scattered photons that reach the detector(close estimates)by using a correction factor to add in the Compton scatter and pair production photons that are ignored by the linear attenuation coefficient formula.



Unshielded source Shield unshielded detector

Figure2.3: Linear Shielding with Buildup Factor

Using Buildup (closer to real-life) since using the attenuation coefficient method under estimate the dose rate on the other side of the shield and using the energy absorption coefficient method over estimates it. A method of getting closer to the real world dose rate was needed. A buildup factor is a correction factor to multiply the number obtained from using the attenuation coefficient by that hopefully gives us the correct result.

## 2.8 Radiation Shielding Requirement

In areas where people are likely to encounter ionizing radiation, it is often necessary to provide shielding to reduce exposure to gamma radiation. Common forms of shielding include rigid materials with limited portability, such as high density concrete, lead bricks, steel plates and cooling pools filled with water. The gamma attenuation of these materials has been widely studied, and the attenuation data is available in resources such as the Nation Institute of Standards and Technology (NIST) XCOM database1.

(McAlister, D.R. (2013), Gamma ray Attenuation Properties of Common Shielding Materials. Retrieved from www.eichrom.com/PDF/gamma-ray-attenuation-white-paper-by-d.m.-rev-4.pdfon January, 2015.)

In addition to these classic, well characterized shielding materials, composite materials are becoming increasingly available from shielding manufacturers. These composite materials range from simple advances, such as lead wool blankets with protective plastic covers, to more advanced materials, such as custom molded components constructed from high density metals dispersed in organic polymers. When evaluating the merits of these composite materials relative to the more classic forms of shielding, it is important to understand the basic principles that lead to gamma ray attenuation. Put simply, shielding, or the attenuation of gamma radiation, occurs through the interaction of the gamma radiation with matter. The degree to which gamma radiation is attenuated is dependent upon the energy of the incident gamma radiation, the atomic number and density of the elements in the shielding material, and the thickness of the shielding. Composite Materials may offer additional benefits in chemical resistance, physical durability, and portability. However, the composite material will not exceed the gamma attenuation characteristics of an equal mass of the components used in its construction. This principle is often referred to as “mass in the path”. For example, a square foot of solid lead sheet will have essentially the same attenuation as a square foot lead wool blanket; as long as the two shields Contain the same mass of lead. However, the lead wool blanket will be physically thicker than the solid lead sheet.

### 2.9 Penetration properties of Radiation

Radiations from radioactive materials can be dangerous and pose health hazards. By knowing the ability of the different types of radiation to penetrate matter allows us to gain an understanding on how best to protect ourselves.

Alpha particles can be absorbed by a thin sheet of paper or by a few centimeters of air. As alpha particles travel through air they collide with nitrogen and oxygen molecules. With each collision they lose some of their energy in ionizing the air molecule until eventually they give up all of their energy and are absorbed. In a sheet of paper the molecules are much close together so the penetration of alpha particles is much less than in air.

The plates below show the penetrating properties of radiations.

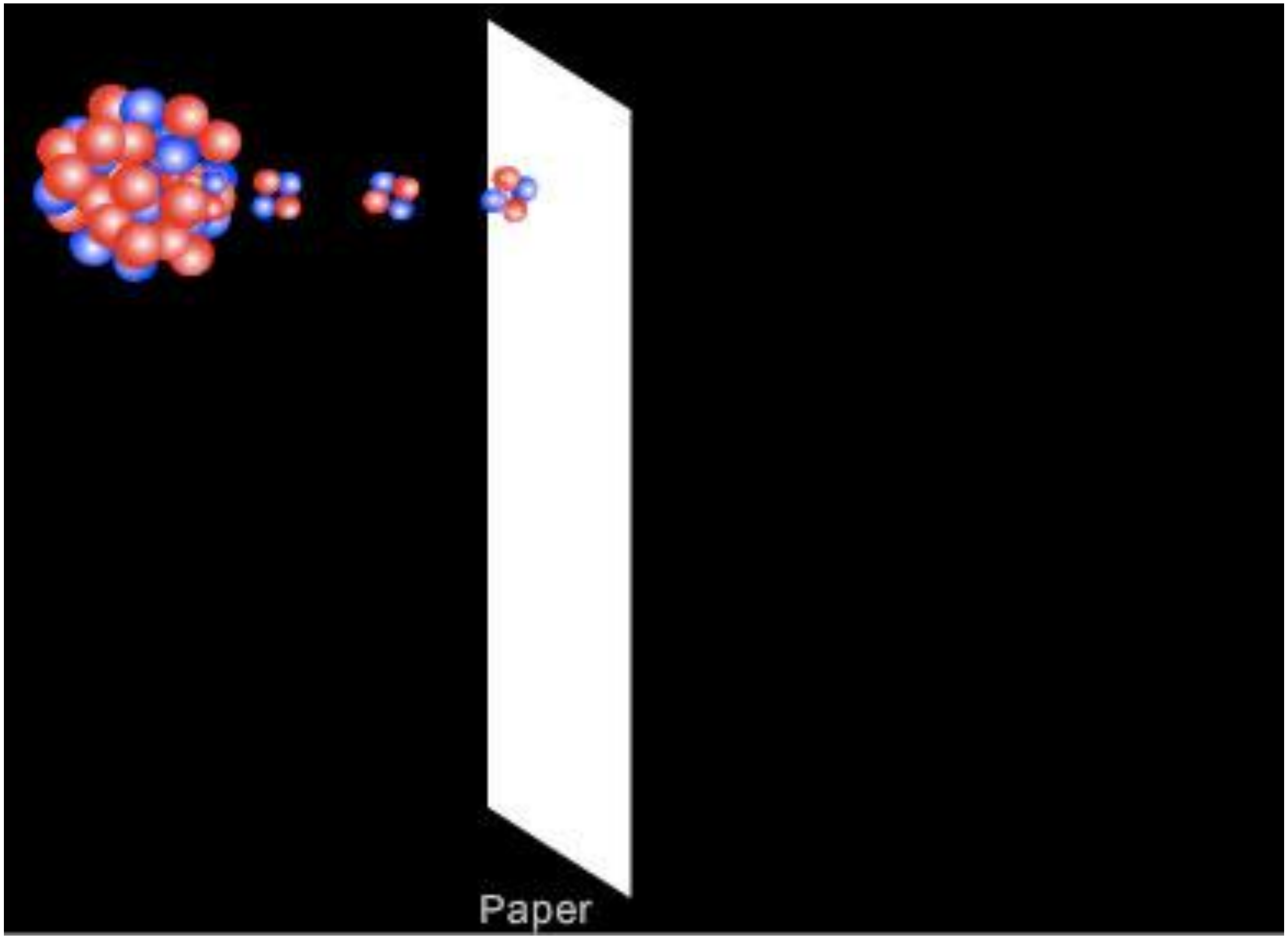



**Plate2.0:** Penetration of Alpha Particles (Penetrating Properties of Radiation pass my exams retrieved from http://www.passmyexams.co.uk/GCSE/physics/penetrating-properties-of-radiation.html on May, 2014)

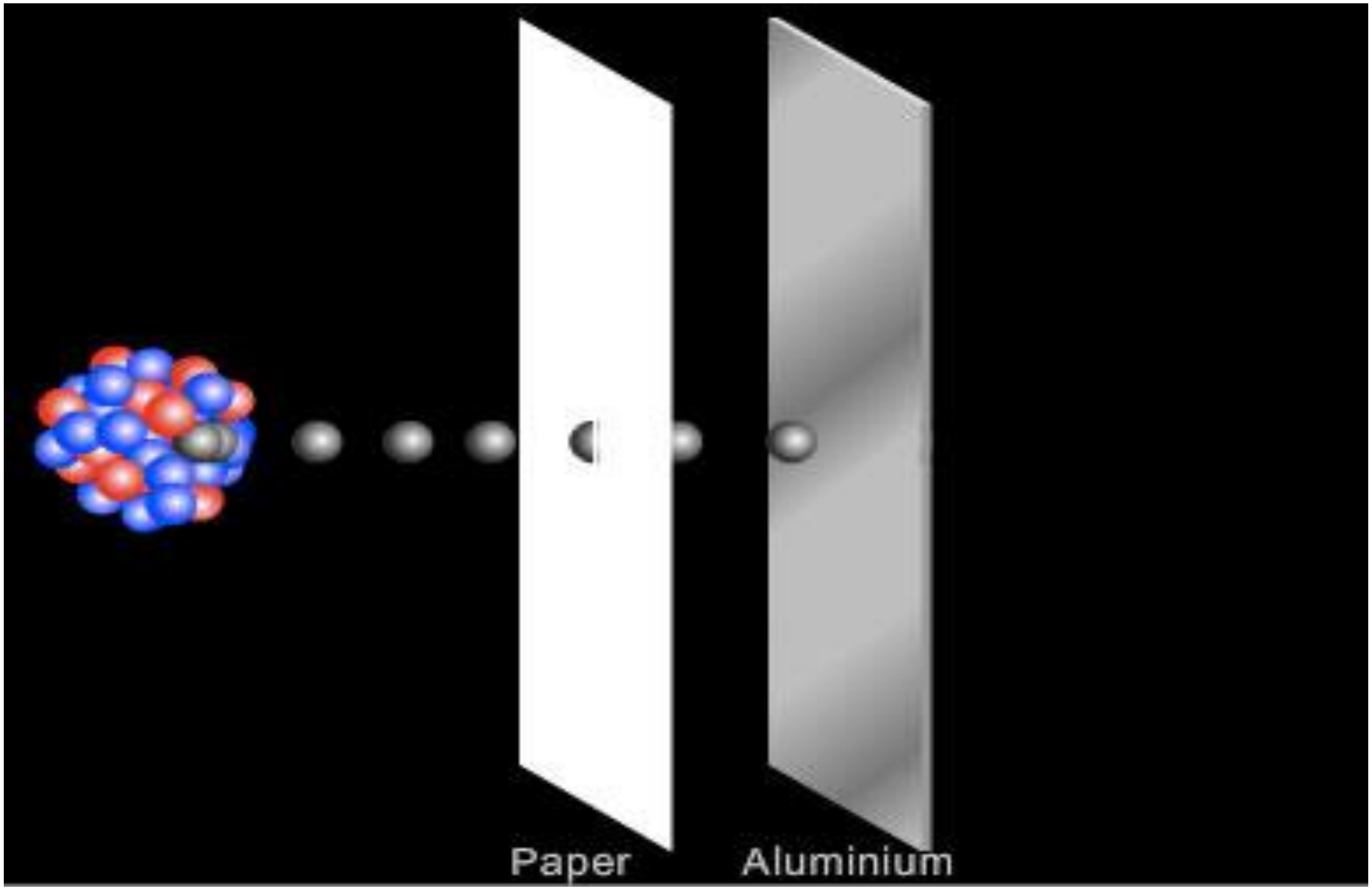



**plate2.1:** Penetration of Beta Particles (Penetrating Properties of Radiation pass my exams retrieved from http://www.passmyexams.co.uk/GCSE/physics/penetrating-properties-of-radiation.html on May, 2014)

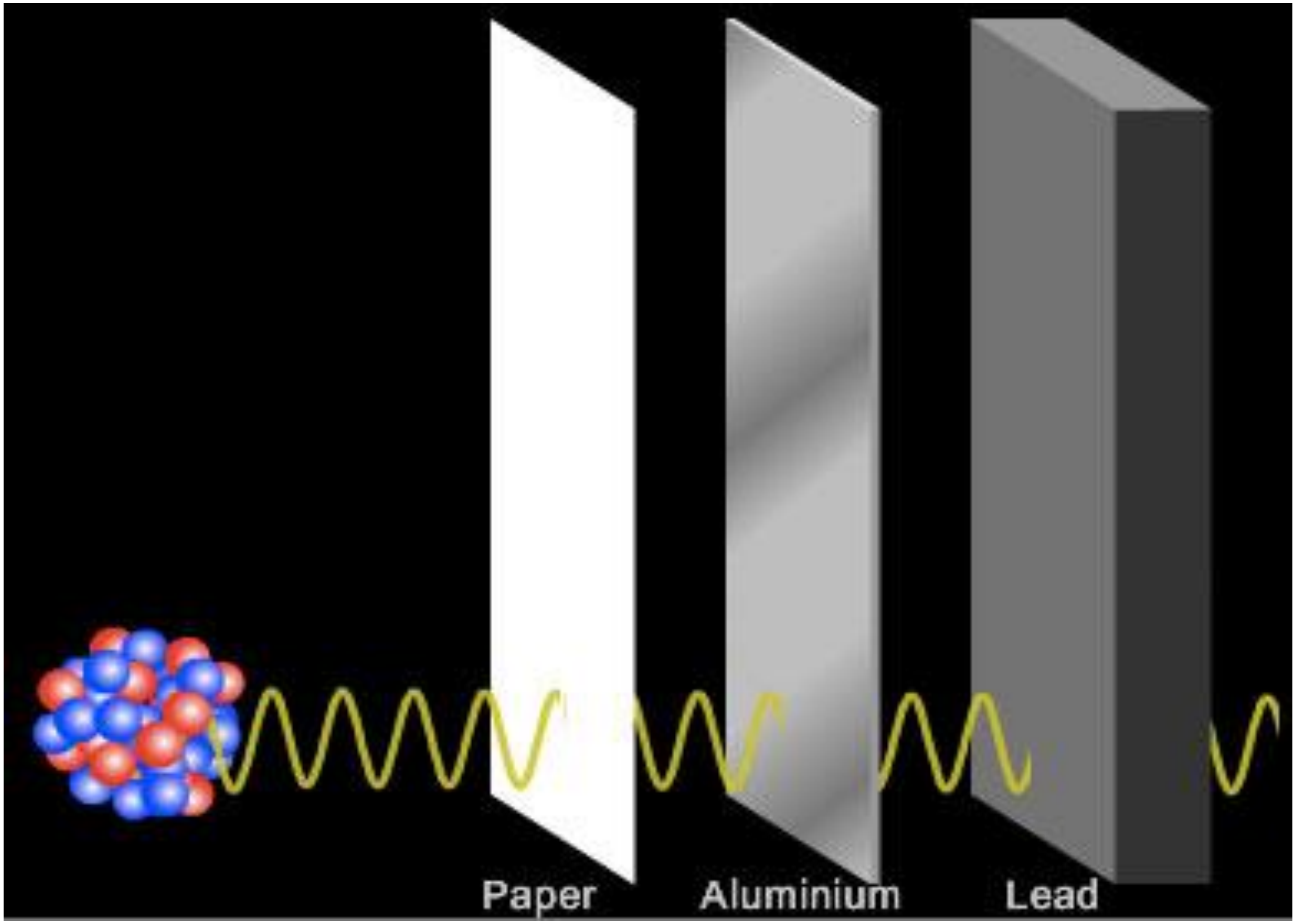



**plate2.2:** Penetration of Gamma rays (Penetrating Properties of Radiation pass my exams retrieved from http://www.passmyexams.co.uk/GCSE/physics/penetrating-properties-of-radiation.html on May, 2014)

Beta particles travel faster than alpha particles and carry less charge (one electron compared to the 2 protons of an alpha particle) and so interact less readily with the atoms and molecules of the material through which they pass. Beta particles can be stopped by a few millimeters of aluminum

Gamma rays are the most penetrating of the radiations. Gamma rays are highly energetic waves and are poor at ionizing other atoms or molecules. It cannot be said that a particular thickness of a material can absorb all gamma radiation. Many centimeters of lead or many meters of concrete are required to absorb high levels of gamma rays.

### 2. 10 External Radiation Protection

The three basic methods used to reduce the external radiation hazard are time, distance, and shielding. Good radiation protection practices require optimization of these fundamental techniques; Time, Distance and Shielding. Retrieved from

http://webfiles.ehs.ufl.edu/rssc_stdy_chp_3.pdf May, 2015

#### 2.10.1 Time

The amount of radiation an individual accumulates will depend on how long the individual stays in the radiation field, because:

Dose (μSv) = Dose Rate (μSv/hr). Time (hr)

Therefore, to limit a person's dose, one can restrict the time spent in the area. How long a person can stay in an area without exceeding a prescribed limit is called the "stay time" and is calculated from the simple relationship:

$$\text{Stay Time} = \frac{\text{Limit} \quad (\mu\text{Sv})}{\text{DoseRate} \quad (\mu\text{Sv/hr})} \qquad \text{2.9}$$

### 2.10.2 Distance

The amount of radiation an individual receives will also depend on how close the person is to the source.

1. The Inverse Square Law - Point sources of x- and gamma radiation follow the inverse square law, which states that the intensity of the radiation (*I*) decreases in proportion to the inverse of the distance from the source (*d*) squared:

   $I \propto \frac{1}{d^2}$……………………………………………………………………….2.10

This can be rewritten:

$I \propto k\frac{1}{d^2}$Where *K* is a constant of unknown value

So, for intensity $I_1$at distance$d_1$, and another intensity $I_2$at distance $d_2$:

$I_1 = k\frac{1}{{d_1}^2};\ I_2 = k\frac{1}{{d_2}^2}$

Now solve for the relationship by eliminating *K*:

$$\frac{I_1}{I_2} = \frac{K/{d_1}^2}{K/{d_2}^2}$$

$$\frac{I_1}{I_2} = \frac{{d_2}^2}{{d_1}^2}$$

OR

$I_{1{d_1}^2} = I_2{d_2}^2$ …………………………………………………………………….2.11

Therefore, by knowing the intensity at one distance, one can find the intensity at any other distance.

### 2.10.2.1 Gamma Constants

Gamma radiation levels (μSv/hr) for one curie of many radionuclide's at a distance of one meter have been measured. These gamma constants can be used to determine the expected exposure rate at a given distance (using the inverse square law) for a known quantity of a radionuclide, or the activity of a radionuclide from a measured exposure rate. To determine the gamma radiation level in μSv/hr at one meter per curie.

### 2.10.3 Gamma Exposure Rate Formula

The exposure rate from a gamma point source can be approximated from the following expression:

μSv/hr = $\frac{6CEf}{d^2}$

Where: *C* is the activity of the gamma emitter, in mill curies

*E* is the gamma ray energy in MeV

*f*is the fraction of disintegrations yielding the gamma of energy E

*d*is the distance from the source in feet

If more than one gamma ray is emitted by the radionuclide of interest, then the contribution from each one must be calculated separately and summed. This expression is accurate to about 20% for gamma emitters with energies ranging from 0.07 MeV to 4 MeV.

### 2.10.4 Shielding

When reducing the time or increasing the distance may not be possible, one can choose shielding material to reduce the external radiation hazard. The proper material to use depends on the type of radiation and its energy.

#### 2.10.4.1 Alpha and Beta Radiation

Alpha particles are easily shielded. A thin piece of paper or several cm of air is usually sufficient to stop them. Thus, alpha particles present no external radiation hazard. Beta particles are more penetrating than alpha particles. Beta shields are usually made of aluminum, brass, plastic, or other materials of low atomic number to reduce the production of bremsstrahlung radiation.

#### 2.10.4.2 X and Gamma Radiation

Monoenergetic x- or gamma rays collimated into a narrow beam are attenuated exponentially through a shield according to the following equation:

$$I = I_0 \, e^{-\mu x} \qquad \text{2.13}$$

Where I is the intensity outside of a shield of thickness x

Io is the unshielded intensity

μ is the linear attenuation coefficient of the shielding material

*x* is the thickness of shielding material

The linear attenuation coefficient is the sum of the probabilities of interaction per unit path length by each of the three scattering and absorption processes - photoelectric effect, Compton Effect and pair production: Note that μ has dimensions of inverse length (1/cm). The reciprocal of μ is defined as the mean free path, which is the average distance the photon travels in an absorber before an interaction takes place.

Because linear attenuation coefficients are proportional to the absorber density, which usually does not have a unique value but depends somewhat on the physical state of the material, it is customary to use the mass attenuation coefficient, which removes density dependence:

Mass attenuation coefficient $\mu_m = \frac{\mu}{\rho}$

Where ρ = density (g/cm3)

For a given photon energy, $\mu_m$ does not change with the physical state of a given absorber. For example, it is the same for water whether present in liquid or vapor form. If the absorber thickness is in cm, then $\mu_m$ will have units of $\frac{cm^{-1}}{g/_{cm^3}} = \frac{cm^2}{g}$

Using the mass attenuation coefficient instead of the linear attenuation coefficient, the attenuation equation can be rewritten:

$I = I_0\, e^{-\mu px}$ ……………………………………………………………………………2.14

Lead is a common shielding material for x-rays and gamma radiation because it has a high density, is inexpensive, and is relatively easy to work with. When working with a radionuclide that emits multiple types of radiation such as beta particles and gamma radiation, it is sometimes necessary to shield with several materials. The less penetrating beta radiation can first be shielded with a layer of plastic or Plexiglas, thereby slowing or stopping the beta particles while reducing the production of bremsstrahlung. The more penetrable gamma radiation would require an additional layer of shielding. Types of shielding and amount of shielding vary depending on photon energy. A good rule of thumb: shield the less penetrable radiation type first then proceed to shield the more penetrable type. This usually decreases both scattering and the total amount of shielding material required.

#### 2.10.4.3 Half Value Layer

The half value layer (HVL) is the thickness of a shielding material required to reduce the intensity of radiation at a point to one half of its original intensity. It can be calculated by setting $I = {}^{1}/_{2}\ I_0$ and solving the attenuation equation for *x*:

$$x_{1/2} = \frac{0.693}{\mu} = \text{HVL} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots 2.15$$

When the HVL is known rather than μ, the total attenuation from *n* half value layers can be calculated by using the following equation:

$$I = \frac{I_0}{2} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots 2.16$$

## 2.11 Review of previous work

The two standard data on shielding calculation was published or has been prepared by the Standard Committee of the Atomic Energy Society of Japan. One provides dose conversion coefficients from fluence rate to effective dose rate of photons and neutrons for use in radiation shielding calculations. The other provides gamma-ray buildup factors for use in gamma-ray shielding calculations.

A number of experimental and theoretical works have been performed on radiation shielding, which has large different application areas with different materials, for concretes (Akkurt et al., 2006, 2012), for some aqueous solutions (Singh et al., 2001), for semiconductors and superconductors (Çevik et al., 2006; Baltaş et al., 2005), for alloys (Han et al., 2009) for steels (Akkurt et al., 2011).

Experimental study is a vital work for this kind of purposes as every calculation should be confirmed with the experimental results. Thus in the present work, the g-ray attenuation coefficients of metal matrix composites for different gamma energies (range from 662 to 1332 keV) by using different point radioactive sources ($^{137}$Cs and $^{60}$Co) has been investigated. The measured results have been compared with the calculation.

The first experimental field work of structural shielding was performed in the mid-1950s. McDonald (1956) reported on studies performed in England in which a gamma source was placed at numerous points over an area around and on top of a structure. Similar experiments were performed for brick houses with a Co-60 source by Stewart (1955). Results from this work provided generic shielding data for a variety of building-types. Soon after the publication of McDonald's and Stewart's work, the shielding data was applied to an early consequence study by Jones (1958)

Auxier's experimental design resembled the previous experiments performed in England (McDonald, 1956). Both Cobalt-60 and Cesium-137 sources were used to simulate fallout and Victoreen pocket-type ion chambers were used to record exposure measurements.

To simulate a distributed sources surrounding each of the buildings, four hundred (400) Co-60 and twenty (20) Cs-137 sources were scattering around and on top of each of the structures, as well as two sources located inside to simulate internal deposition. Exposure measurements were taken at a height of three feet above the floor at various locations within each structure. Measurements were then compared to the dose-rate distributions above an extended plane source measured in the "phantom" house considered to be the reference standard for the non-attenuated exposure measurements.

Protection factors were based on the method used by Strickler (1960) by calculating the ratio between the experimental measurement and the hypothetical infinite-plane dose rate (estimated to be 500 (mR/hr) for Co-60).

In 1962, calculations and Tabletop laboratory experiments with monoenergetic gamma-ray sources (Co-60 and Cs-137) were performed by Spencer (1962).

Schoke and Rexford (1963) conducted experiments to verify theoretical calculations of wall thickness effects on the shielding characteristic of a full-scale concrete blockhouse in a uniformly contaminated field. Results were compared to the predictive method developed by Spencer (1962). To simulate a continuous distribution of fallout, investigators divided the area around the test structure into an array of squares and placed two isotropic-point sources (Co-60 and Cs-137) at the center of each. Instead of having sources at each of the points simultaneously, a single source was moved over the successive centers until the total area around the structure was "covered". Due to the symmetry of the experimental structure, only one-eighth of the surrounding fallout field required simulation. (Schmoke, 1963)

The methodology used by Eisenhauer (1964) is based on the primary calculations presented by Spencer (1962). Eisenhauer agrees with the methods presented by Spencer in calculating the relative amount of protection from fallout radiation associated with a wide variety of shielding situations. Comparisons of numerical results with experimental information show fairly good agreement for simple structures surrounded by an infinite plane source with less complex structures. (Eisenhauer, 1964)

## CHAPTER THREE

## 3.0 Materials and Method

### 3.1 Introduction

Calculation of dose rate involves the use of laboratory equipment and software, these include Rados dose-rate meter, and Cobalt-60, Cesium 137 and RadProCalculator which were used in this research. All materials were obtained at Centre for Energy Research and Training, A.B.U-Zaria (CERT/ABU)

### 3.2 Materials:

#### 3.2.1 Radprocalculator

Rad Pro Calculator performs many nuclear calculations that are useful to the health physicist, radiological researcher, radio chemist, radiation safety officer, health physics technician (HP) and other professionals in radiation physics and radiological engineering. It calculates, among other things, radioactivity unit's conversions (SI and US customary) and gamma emitter dose rate and activity.

#### 3.2.2 Cesium -137 Radiation Source

Cs -137 Radiation source activity used in CERT/ABU. It's a sealed source with the following specification: source model: Cs -137, Initial Activity: 30mCi, Date: 6/11/2000 to Present Activity: 22mCi, Date: 5/11/2014, source no: 9716GM, container no: A3451

### 3.2.3 Cobalt- 60 Radiation Source

Co-60 Radiation source activity used in CERT/ABU. It's a sealed source with the following specification: source model: Co-60, Initial Activity: 30mCi, Date: 6/11/2000 to Present Activity: 5mCi, Date: 5/11/2014, source No: L495, Container No: a3449.

### 3.2.4 Rados Dose Rate Meter

This is a hand held portable dose –rate meter with high precision. It has a digital readout and contains probes to isolate and detect highly ionizing radiation like $\alpha$ - and β – particles. Models: RADOS-120, serial number: 20563054 measurements: programmed version: V 1.3 X; make; Rados Technology OY Box 506 FIN-20101 Turku Finland. Calibration accuracy: less than 0.05 at temperature ±20

## 3.3 Method

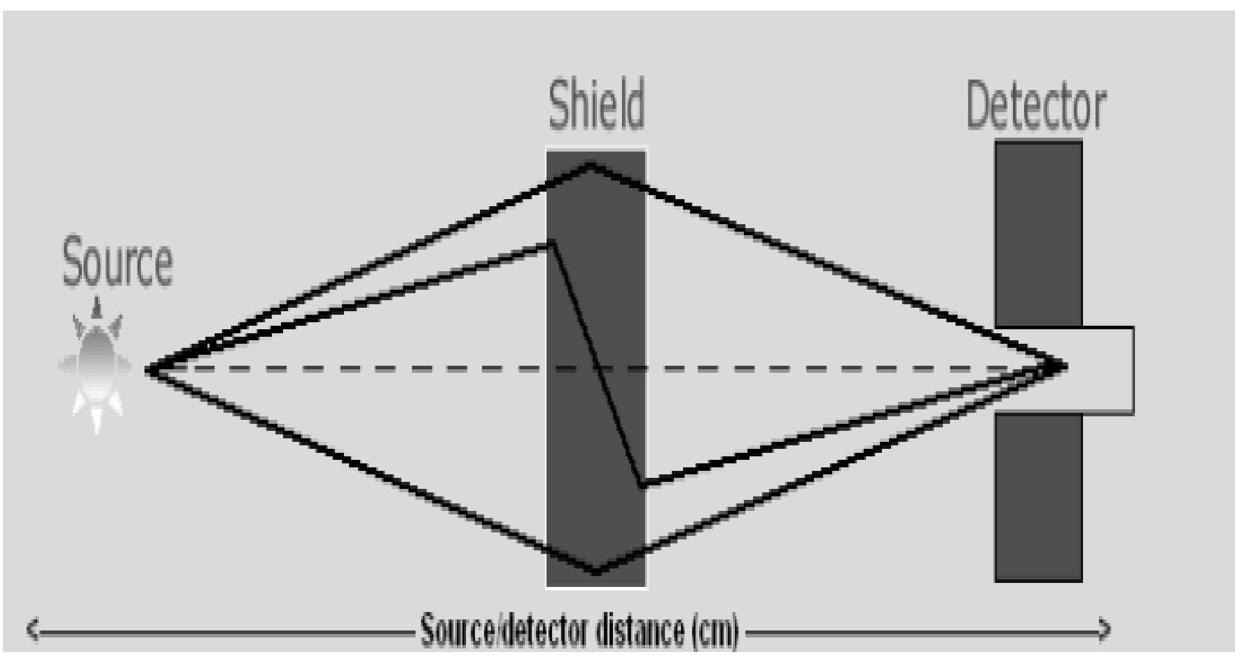



Figure 3.1: Gamma Radiation Attuenation under the condition of broad beam geometry showing the effect of photons scattered into the detector (Majill, 2013)

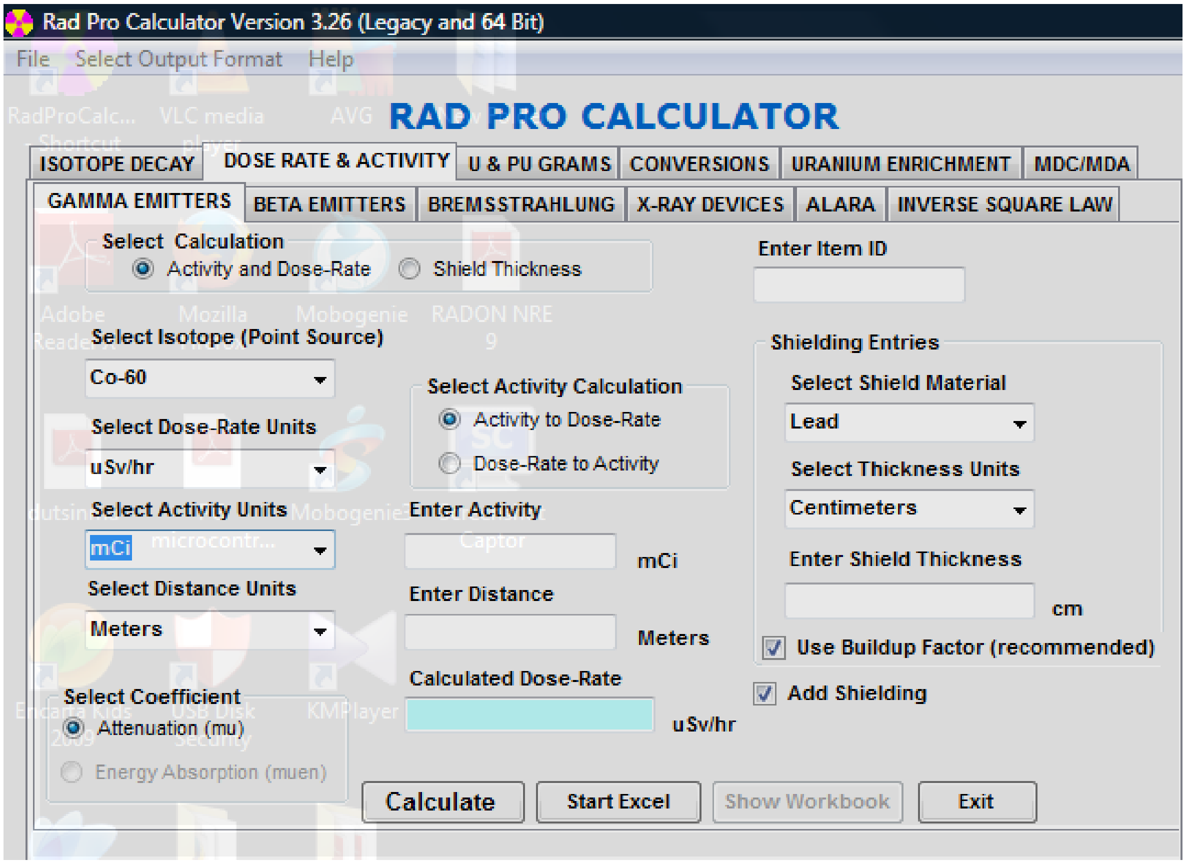



Figure 3.2: Radprocalculator (Rad pro calculator software)

A comparative calculation of dose rate for shielding requirements of cobalt-60 and cesium-137 sources at CERT were made using rad pro calculator. Upon opening Rad Pro Calculator, it is in the calculate dose-rate mode. Select your activity, dose-rate and distance from source to detector units. Select an isotope. Enter your source activity strength value and the measurement distance value from the source to detector. Select your shielding material, thickness units and enter shield thickness value. Press the Calculate for your dose-rate. By default, the answers are outputted in full decimal place numbers until the number is either less than 0.001 or greater than 9999. At that point, it switches to scientific notation with full decimal places (x.xxxxxE+yyy). Calculator has the Excel output feature; you will see a Start Excel button. When this is pressed, an Excel object is created in memory. It's not visible at this time but is there awaiting your input. You may make the Excel spreadsheet visible at any time by pressing the Show Workbook button. You may still input more data after showing it by minimizing the Excel application.

# CHAPTER FOUR

## 4.0 RESULTS AND DISSCUSSIONS

### 4.1RESULTS of Co-60

Calculated dose rate of 5mCi of 60Co at 1 meter distance from the source to detector is tabulated below. Table 4.1 shows the results of lead shielding, Table 4.2 shows the results of concrete shielding, Table 4.3 shows the results of iron shielding,Table4.4 shows the results of uranium shielding, Table 4.5 shows the results of lead glass, Table 4.6 shows the results of aluminum, Table 4.7 shows the results of tungsten.

**Table4.1** Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the source to detector using lead shielding at various thicknesses.

| Isotope | Activity (mCi) | Measurement Distance (m) | Shield Coefficient(/cm) | Shield Material | Shield Thickness (cm) | Calculated Dose-Rate (μSv/hr) | $Log_{10}$ Dose-Rate |
|---|---|---|---|---|---|---|---|
| Co-60 | 5 | 1 | 0.25 | Lead | 1 | 38.93 | 1.59 |
| Co-60 | 5 | 1 | 0.25 | Lead | 2 | 23.40 | 1.36 |
| Co-60 | 5 | 1 | 0.25 | Lead | 3 | 12.89 | 1.11 |
| Co-60 | 5 | 1 | 0.25 | Lead | 4 | 7.39 | 0.87 |
| Co-60 | 5 | 1 | 0.25 | Lead | 5 | 4.09 | 0.61 |
| Co-60 | 5 | 1 | 0.25 | Lead | 6 | 2.35 | 0.37 |
| Co-60 | 5 | 1 | 0.25 | Lead | 7 | 1.34 | 0.13 |
| Co-60 | 5 | 1 | 0.25 | Lead | 8 | 0.76 | -0.11 |
| Co-60 | 5 | 1 | 0.25 | Lead | 9 | 0.42 | -0.37 |
| Co-60 | 5 | 1 | 0.25 | Lead | 10 | 0.23 | -0.63 |
| Co-60 | 5 | 1 | 0.25 | Lead | 11 | 0.13 | -0.88 |
| Co-60 | 5 | 1 | 0.25 | Lead | 12 | 0.06 | -1.16 |

**Table4.2** Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using concrete shielding at various thicknesses

| Isotope | Measurement Activity (mCi) | Distance (m) | Shield Coefficient(/cm) | Shield Material | Shield Thickness (cm) | Calculated Dose-Rate (μSv/hr) | $Log_{10}$ Dose-Rate |
|---|---|---|---|---|---|---|---|
| Co-60 | 5 | 1 | 0.05 | Concrete | 5 | 47.58 | 1.67 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 10 | 36.75 | 1.56 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 15 | 21.37 | 1.32 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 20 | 12.87 | 1.11 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 25 | 8.18 | 0.91 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 30 | 4.94 | 0.69 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 35 | 2.96 | 0.471 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 40 | 2.17 | 0.33 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 45 | 1.10 | 0.04 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 50 | 0.77 | -0.11 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 55 | 0.43 | -0.36 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 60 | 0.21 | -0.67 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 65 | 0.13 | -0.85 |
| Co-60 | 5 | 1 | 0.05 | Concrete | 70 | 0.06 | -1.21 |

**Table4.3** Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using iron shielding at various thicknesses

| Isotope | Measurement Activity (mCi) | Distance (m) | Shield Coefficient(/cm) | Shield Material | Shield Thickness (cm) | Calculated Dose-Rate (μSv/hr) | $Log_{10}$ Dose-Rate |
|---|---|---|---|---|---|---|---|
| Co-60 | 5 | 1 | 0.13 | Iron | 2 | 38.65 | 1.58 |
| Co-60 | 5 | 1 | 0.13 | Iron | 4 | 23.69 | 1.37 |
| Co-60 | 5 | 1 | 0.13 | Iron | 6 | 14.05 | 1.14 |
| Co-60 | 5 | 1 | 0.13 | Iron | 8 | 7.39 | 0.86 |
| Co-60 | 5 | 1 | 0.13 | Iron | 10 | 4.56 | 0.65 |
| Co-60 | 5 | 1 | 0.13 | Iron | 12 | 2.21 | 0.34 |
| Co-60 | 5 | 1 | 0.13 | Iron | 14 | 1.25 | 0.09 |
| Co-60 | 5 | 1 | 0.13 | Iron | 16 | 0.69 | -0.15 |
| Co-60 | 5 | 1 | 0.13 | Iron | 18 | 0.31 | -0.50 |
| Co-60 | 5 | 1 | 0.13 | Iron | 20 | 0.18 | -0.73 |
| Co-60 | 5 | 1 | 0.13 | Iron | 22 | 0.06 | -1.16 |

**Table4.4** Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using Uranium shielding at various thicknesses

| | Measurement | | | | Shield | Calculated | |
|---|---|---|---|---|---|---|---|
| Isotope | Activity (mCi) | Distance (m) | Shield Coefficient(/cm) | Shield Material | Thickness (cm) | Dose-Rate (µSv/hr) | $Log_{10}$ Dose-Rate |
| Co-60 | 5 | 1 | 0.47 | Uranium | 1 | 23.87 | 1.37 |
| Co-60 | 5 | 1 | 0.47 | Uranium | 2 | 8.24 | 0.91 |
| Co-60 | 5 | 1 | 0.47 | Uranium | 3 | 2.80 | 0.44 |
| Co-60 | 5 | 1 | 0.47 | Uranium | 4 | 0.94 | -0.02 |
| Co-60 | 5 | 1 | 0.47 | Uranium | 5 | 0.31 | -0.49 |
| Co-60 | 5 | 1 | 0.47 | Uranium | 6 | 0.10 | -0.96 |

**Table4.5** Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using Lead glass shielding at various thicknesses

| Isotope | Measurement Activity (mCi) | Distance (m) | Shield Coefficient | Shield Material | Shield Thickness (cm) | Calculated Dose-Rate (μSv/hr) | Log $_{10}$ Dose-Rate |
|---|---|---|---|---|---|---|---|
| Co-60 | 5 | 1 | 0.28 | Lead Glass | 1 | 29.38 | 1.46 |
| Co-60 | 5 | 1 | 0.28 | Lead Glass | 2 | 15.23 | 1.18 |
| Co-60 | 5 | 1 | 0.28 | Lead Glass | 3 | 7.90 | 0.89 |
| Co-60 | 5 | 1 | 0.28 | Lead Glass | 4 | 4.10 | 0.61 |
| Co-60 | 5 | 1 | 0.28 | Lead Glass | 5 | 2.13 | 0.32 |
| Co-60 | 5 | 1 | 0.28 | Lead Glass | 6 | 1.11 | 0.04 |
| Co-60 | 5 | 1 | 0.28 | Lead Glass | 7 | 0.57 | -0.23 |
| Co-60 | 5 | 1 | 0.28 | Lead Glass | 8 | 0.30 | -0.52 |
| Co-60 | 5 | 1 | 0.28 | Lead Glass | 9 | 0.15 | -0.80 |
| Co-60 | 5 | 1 | 0.28 | Lead Glass | 10 | 0.08 | -1.08 |

**Table4.6** Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using Aluminum shielding at various thicknesses

| | Measurement | | | | Shield | Calculated | |
|---|---|---|---|---|---|---|---|
| | Activity | Distance | Shield | Shield | Thickness | Dose-Rate | $Log_{10}$ |
| Isotope | (mCi) | (m) | Coefficient | Material | (cm) | (μSv/hr) | Dose-Rate |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 2 | 42.14 | 1.6247 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 4 | 31.31 | 1.49 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 6 | 23.26 | 1.36 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 8 | 17.28 | 1.23 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 10 | 12.84 | 1.10 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 12 | 9.54 | 0.97 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 14 | 7.09 | 0.85 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 16 | 5.27 | 0.72 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 18 | 3.92 | 0.59 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 20 | 2.91 | 0.46 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 22 | 2.16 | 0.33 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 24 | 1.61 | 0.20 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 26 | 1.19 | 0.07 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 28 | 0.89 | -0.04 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 30 | 0.66 | -0.17 |

| | | Measurement | | | Shield | Calculated | |
|---|---|---|---|---|---|---|---|
| | Activity | Distance | Shield | Shield | Thickness | Dose-Rate | $Log_{10}$ |
| Isotope | (mCi) | (m) | Coefficient | Material | (cm) | (μSv/hr) | Dose-Rate |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 32 | 0.49 | -0.30 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 34 | 0.36 | -0.43 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 36 | 0.27 | -0.56 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 38 | 0.20 | -0.69 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 40 | 0.15 | -0.82 |
| Co-60 | 5 | 1 | 0.06 | Aluminum | 42 | 0.11 | -0.94 |

**Table4.7**: Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using Tungsten shielding at various thicknesses

| Isotope | Measurement<br>Activity (mCi) | Distance (m) | Shield Coefficient(/cm) | Shield Material | Shield Thickness (cm) | Calculated Dose-Rate (μSv/hr) | $Log_{10}$ Dose-Rate |
|---|---|---|---|---|---|---|---|
| Co-60 | 5 | 1 | 0.46 | Tungsten | 1 | 19.22 | 1.28 |
| Co-60 | 5 | 1 | 0.46 | Tungsten | 2 | 6.52 | 0.81 |
| Co-60 | 5 | 1 | 0.46 | Tungsten | 3 | 2.22 | 0.34 |
| Co-60 | 5 | 1 | 0.46 | Tungsten | 4 | 0.75 | -0.12 |
| Co-60 | 5 | 1 | 0.46 | Tungsten | 5 | 0.25 | -0.58 |
| Co-60 | 5 | 1 | 0.46 | Tungsten | 6 | 0.08 | -1.05 |

**4.2 RESULTS of Cs-137** Calculated dose rate of 22mCi of Cs-137 at 1 meter distance from the source to detector is tabulated as follows Table4.8 shows the results of lead shielding, Table4.9 shows the results concrete shielding, Table4.10 shows the results of iron shielding, Table4.11 shows the results of uranium shielding, Table4.12 shows the results of lead glass shielding, Table4.13 shows the results of aluminum shielding, Table4.14 shows the results of tungsten shielding.

**Table4.8** Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using lead shielding at various thicknesses

| | Measurement | | | | Shield | Calculated | |
|---|---|---|---|---|---|---|---|
| | Activity | Distance | Shield | Shield | Thickness | Dose-Rate | Log $_{10}$ |
| **Isotope** | **(mCi)** | **(m)** | **Coefficient** | **Material** | **(cm)** | **(μSv/hr)** | **Dose-Rate** |
| Cs-137 | 22 | 1 | 0.50 | Lead | 1 | 23.34 | 1.36 |
| Cs-137 | 22 | 1 | 0.50 | Lead | 2 | 7.66 | 0.88 |
| Cs-137 | 22 | 1 | 0.50 | Lead | 3 | 2.47 | 0.39 |
| Cs-137 | 22 | 1 | 0.50 | Lead | 4 | 0.73 | -0.13 |
| Cs-137 | 22 | 1 | 0.50 | Lead | 5 | 0.22 | -0.64 |
| Cs-137 | 22 | 1 | 0.50 | Lead | 6 | 0.07 | -1.14 |

**Table4.9** Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using concrete shielding at various thicknesses

| | Measurement | | | | Shield | Calculated | |
|---|---|---|---|---|---|---|---|
| | **Activity** | **Distance** | **Shield** | **Shield** | **Thickness** | **Dose-Rate** | **Log $_{10}$** |
| **Isotope** | **(mCi)** | **(m)** | **Coefficient** | **Material** | **(cm)** | **(μSv/hr)** | **Dose-Rate** |
| Cs-137 | 22 | 1 | 0.06 | Concrete | 5 | 40.40 | 1.60 |
| Cs-137 | 22 | 1 | 0.06 | Concrete | 10 | 24.03 | 1.38 |
| Cs-137 | 22 | 1 | 0.06 | Concrete | 15 | 15.35 | 1.18 |
| Cs-137 | 22 | 1 | 0.06 | Concrete | 20 | 8.90 | 0.94 |
| Cs-137 | 22 | 1 | 0.06 | Concrete | 25 | 4.85 | 0.68 |
| Cs-137 | 22 | 1 | 0.06 | Concrete | 30 | 2.53 | 0.40 |
| Cs-137 | 22 | 1 | 0.06 | Concrete | 35 | 1.28 | 0.10 |
| Cs-137 | 22 | 1 | 0.06 | Concrete | 40 | 0.63 | -0.19 |
| Cs-137 | 22 | 1 | 0.06 | Concrete | 45 | 0.34 | -0.46 |
| Cs-137 | 22 | 1 | 0.06 | Concrete | 50 | 0.12 | -0.91 |
| Cs-137 | 22 | 1 | 0.06 | Concrete | 55 | 0.08 | -1.04 |

**Table4.10** Dose rate Calculation of 22mCi ofCs-137 at 1 meter distance from the detector using iron shielding at various thicknesses

| | Measurement | | | | Shield | Calculated | |
|---|---|---|---|---|---|---|---|
| | Activity | Distance | Shield | Shield | Thickness | Dose-Rate | Log $_{10}$ |
| **Isotope** | **(mCi)** | **(m)** | **Coefficient** | **Material** | **(cm)** | **(μSv/hr)** | **Dose-Rate** |
| Cs-137 | 22 | 1 | 0.19 | Iron | 1 | 55.05 | 1.74 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 2 | 43.71 | 1.64 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 3 | 23.90 | 1.37 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 4 | 20.37 | 1.30 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 5 | 11.09 | 1.04 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 6 | 8.75 | 0.94 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 7 | 6.58 | 0.81 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 8 | 3.56 | 0.55 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 9 | 2.58 | 0.41 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 10 | 1.39 | 0.14 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 11 | 0.98 | -0.01 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 12 | 0.68 | -0.16 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 13 | 0.36 | -0.43 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 14 | 0.26 | -0.57 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 15 | 0.14 | -0.83 |
| Cs-137 | 22 | 1 | 0.19 | Iron | 16 | 0.07 | -1.10 |

**Table4.11** Dose rate Calculation of 22mCi ofCs-137 at 1 meter distance from the detector using tungsten shielding at various thicknesses

| | Measurement | | | | Shield | Calculated | |
|---|---|---|---|---|---|---|---|
| | Activity | Distance | Shield | Shield | Thickness | Dose-Rate | $Log_{10}$ |
| Isotope | (mCi) | (m) | Coefficient | Material | (cm) | $Log_{10}$ | Dose-Rate |
| Cs-137 | 22 | 1 | 0.74 | Tungsten | 1.0 | 13.36 | 1.12 |
| Cs-137 | 22 | 1 | 0.74 | Tungsten | 1.5 | 6.11 | 0.78 |
| Cs-137 | 22 | 1 | 0.74 | Tungsten | 2.0 | 2.66 | 0.42 |
| Cs-137 | 22 | 1 | 0.74 | Tungsten | 2.5 | 1.12 | 0.05 |
| Cs-137 | 22 | 1 | 0.74 | Tungsten | 3.0 | 0.47 | -0.32 |
| Cs-137 | 22 | 1 | 0.74 | Tungsten | 3.5 | 0.19 | -0.71 |
| Cs-137 | 22 | 1 | 0.74 | Tungsten | 4.0 | 0.07 | -1.10 |

**Table4.12** Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using uranium shielding at various thicknesses

| | Measurement | | | | Shield | Calculated | |
|---|---|---|---|---|---|---|---|
| | Activity | Distance | Shield | Shield | Thickness | Dose-Rate | Log $_{10}$ |
| Isotope | mCi | Meters | Coefficient | Material | cm | μSv/hr | Dose-Rate |
| Cs-137 | 22 | 1 | 1.03 | Uranium | 0.5 | 21.75 | 1.33 |
| Cs-137 | 22 | 1 | 1.03 | Uranium | 1.0 | 6.89 | 0.83 |
| Cs-137 | 22 | 1 | 1.03 | Uranium | 1.5 | 2.14 | 0.33 |
| Cs-137 | 22 | 1 | 1.03 | Uranium | 2.0 | 0.61 | -0.20 |
| Cs-137 | 22 | 1 | 1.03 | Uranium | 2.5 | 0.18 | -0.72 |
| Cs-137 | 22 | 1 | 1.03 | Uranium | 3.0 | 0.05 | -1.24 |

**Table4.13** Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using lead glass shielding at various thicknesses

| | Measurement | | | | Shield | Calculated | |
|---|---|---|---|---|---|---|---|
| | Activity | Distance | Shield | Shield | Thickness | Dose-Rate | $Log_{10}$ |
| **Isotope** | **(mCi)** | **(m)** | **Coefficient** | **Material** | **(cm)** | **(µSv/hr)** | **Dose-Rate** |
| Cs-137 | 22 | 1 | 0.51 | Lead Glass | 1 | 19.01 | 1.27 |
| Cs-137 | 22 | 1 | 0.51 | Lead Glass | 2 | 5.88 | 0.77 |
| Cs-137 | 22 | 1 | 0.51 | Lead Glass | 3 | 1.82 | 0.26 |
| Cs-137 | 22 | 1 | 0.51 | Lead Glass | 4 | 0.56 | -0.24 |
| Cs-137 | 22 | 1 | 0.51 | Lead Glass | 5 | 0.17 | -0.75 |
| Cs-137 | 22 | 1 | 0.51 | Lead Glass | 6 | 0.05 | -1.26 |

**Table4.14** Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using aluminum shielding at various thicknesses

| | Measurement | | | | Shield | Calculated | |
|---|---|---|---|---|---|---|---|
| | Activity | Distance | Shield | Shield | Thickness | Dose-Rate | Log $_{10}$ |
| **Isotope** | **(mCi)** | **(m)** | **Coefficient** | **Material** | **(cm)** | **(μSv/hr)** | **Dose-Rate** |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 2 | 40.95 | 1.61 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 4 | 27.30 | 1.43 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 6 | 18.20 | 1.26 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 8 | 12.14 | 1.08 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 10 | 8.09 | 0.90 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 12 | 5.39 | 0.73 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 14 | 3.59 | 0.55 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 16 | 2.39 | 0.38 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 18 | 1.59 | 0.20 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 20 | 1.06 | 0.02 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 22 | 0.71 | -0.14 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 24 | 0.47 | -0.32 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 26 | 0.31 | -0.50 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 28 | 0.21 | -0.67 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 30 | 0.14 | -0.85 |
| Cs-137 | 22 | 1 | 0.08 | Aluminum | 32 | 0.09 | -1.02 |

Table 4.15 Comparison of linear attenuation coefficient ($cm^{-1}$) of Rad pro calculator and theoretical (XCOM data base) of Cs-137 using different shielding materials.

| | **Linear Attenuation coefficient μ ($cm^{-1}$)** | | |
|---|---|---|---|
| **Shield Material** | **Rad pro** | **XCOM** | **%deviation** |
| LEAD | 0.5 | 0.49 | 2% |
| URANIUM | 1.03 | 0.88 | 17% |
| TUNGSTEN | 0.74 | 0.79 | 6% |
| ALUMINIUM | 0.08 | 0.079 | 1% |
| CONCRETE | 0.055 | 0.054 | 1% |
| IRON | 0.2 | 0.21 | 4% |

### 4.3 Co-60 GRAPHS

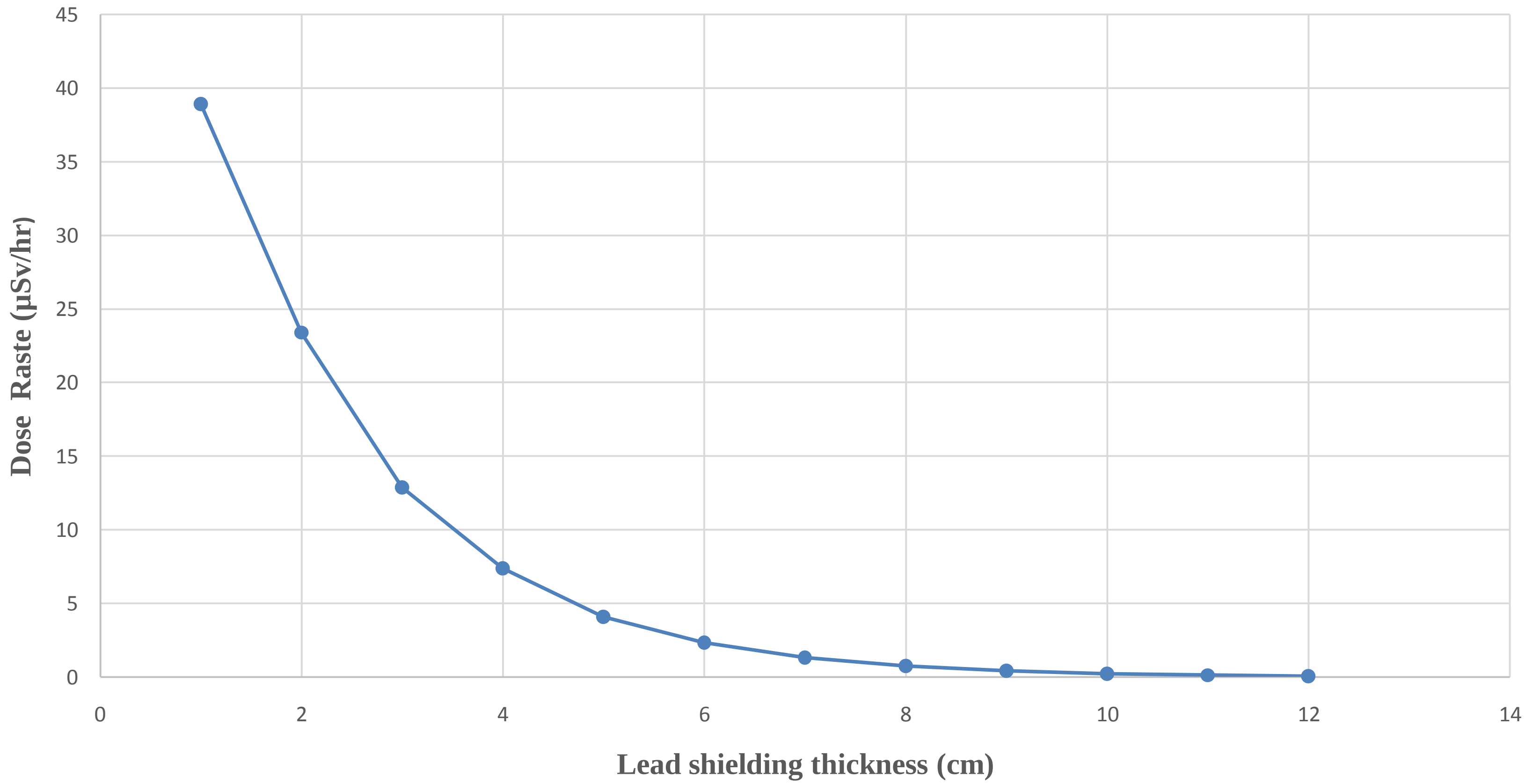



Figure 4.1 Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using lead shielding at various thicknesses

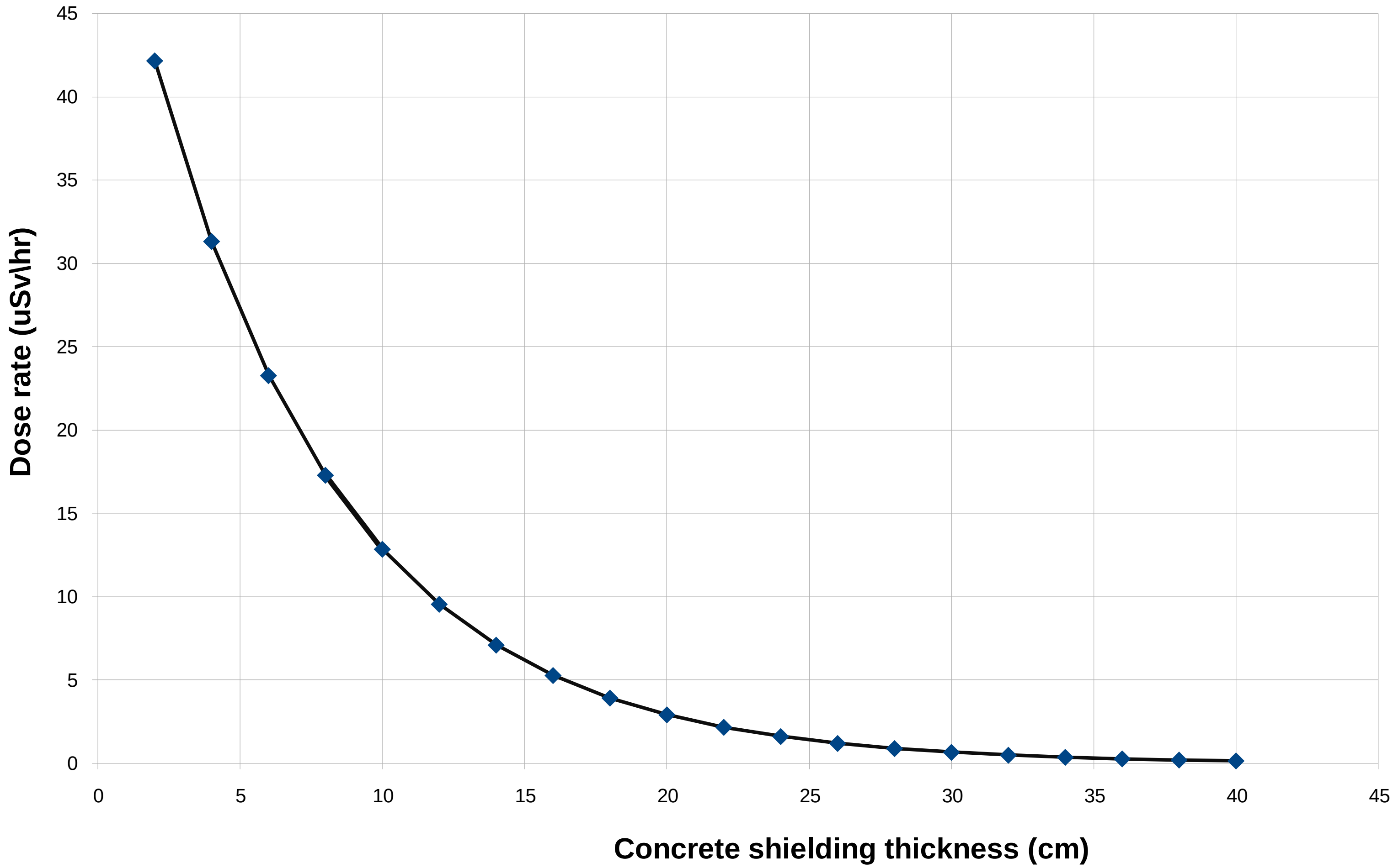



Figure 4.2 Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using concrete shielding at various thicknesses

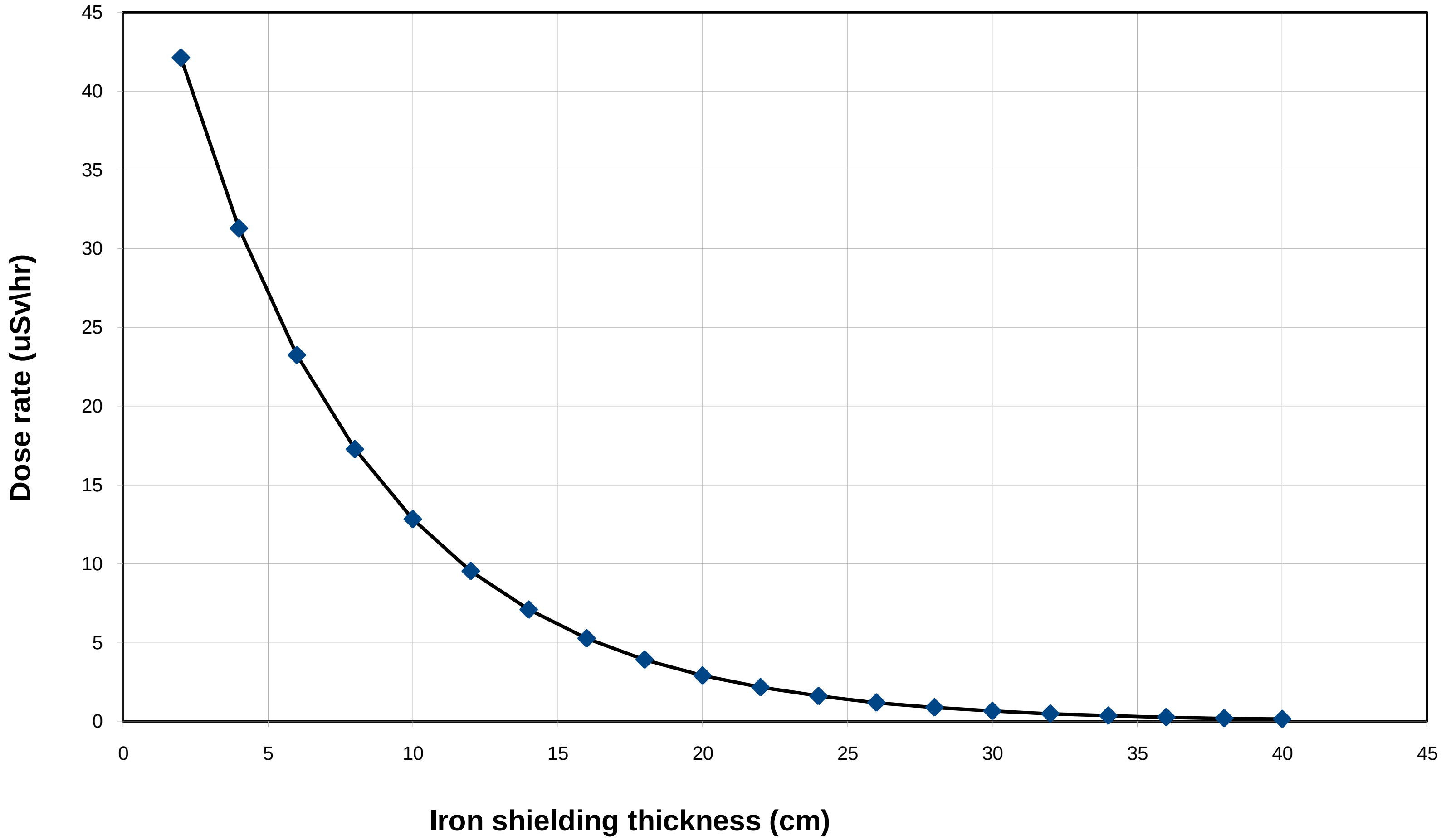



Figure4.3 Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using iron shielding at various thicknesses

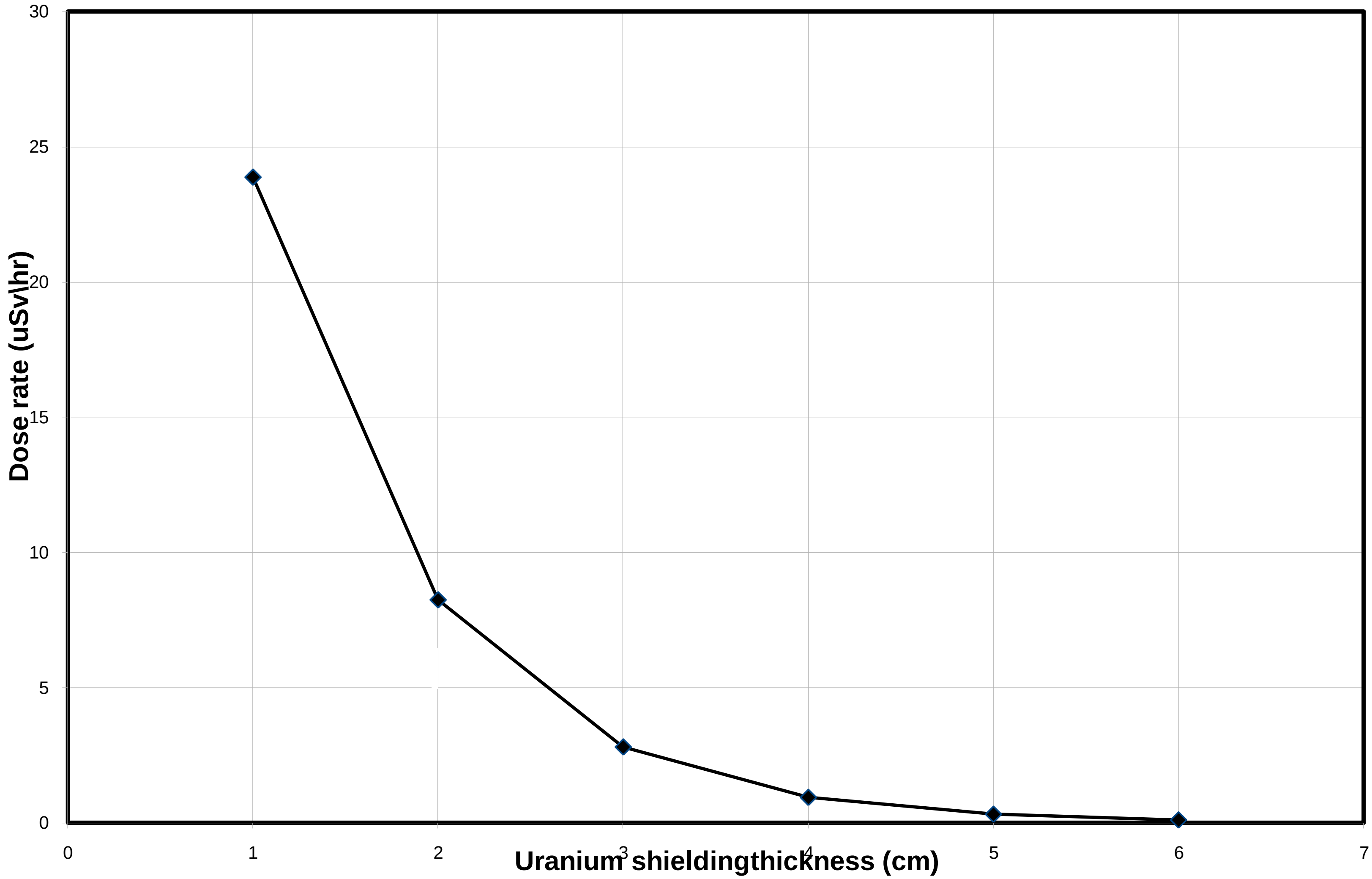



Figure4.4 Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using uranium shielding at various thicknesses.

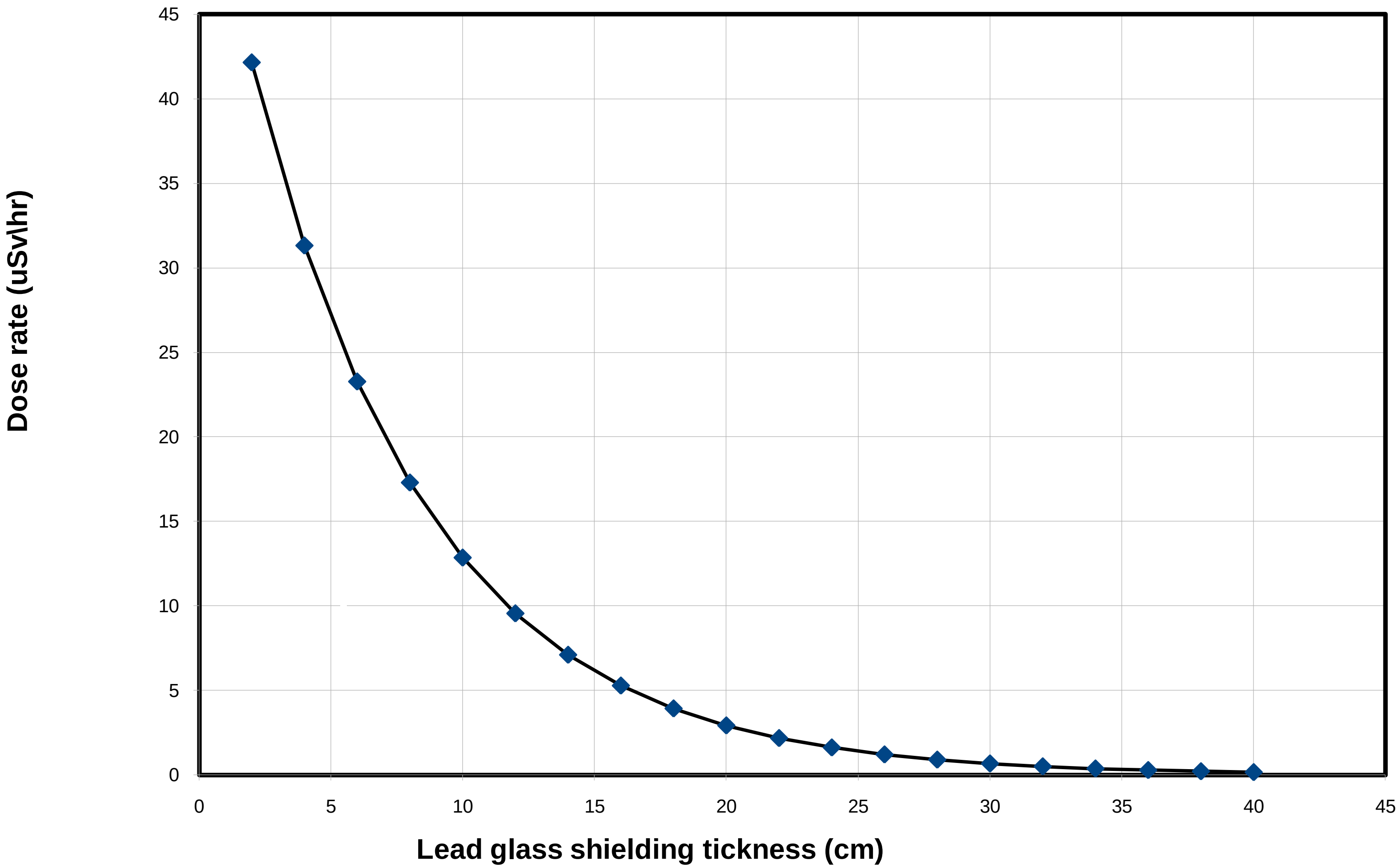



Figure4.5 Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using lead glass shielding at various thicknesses.

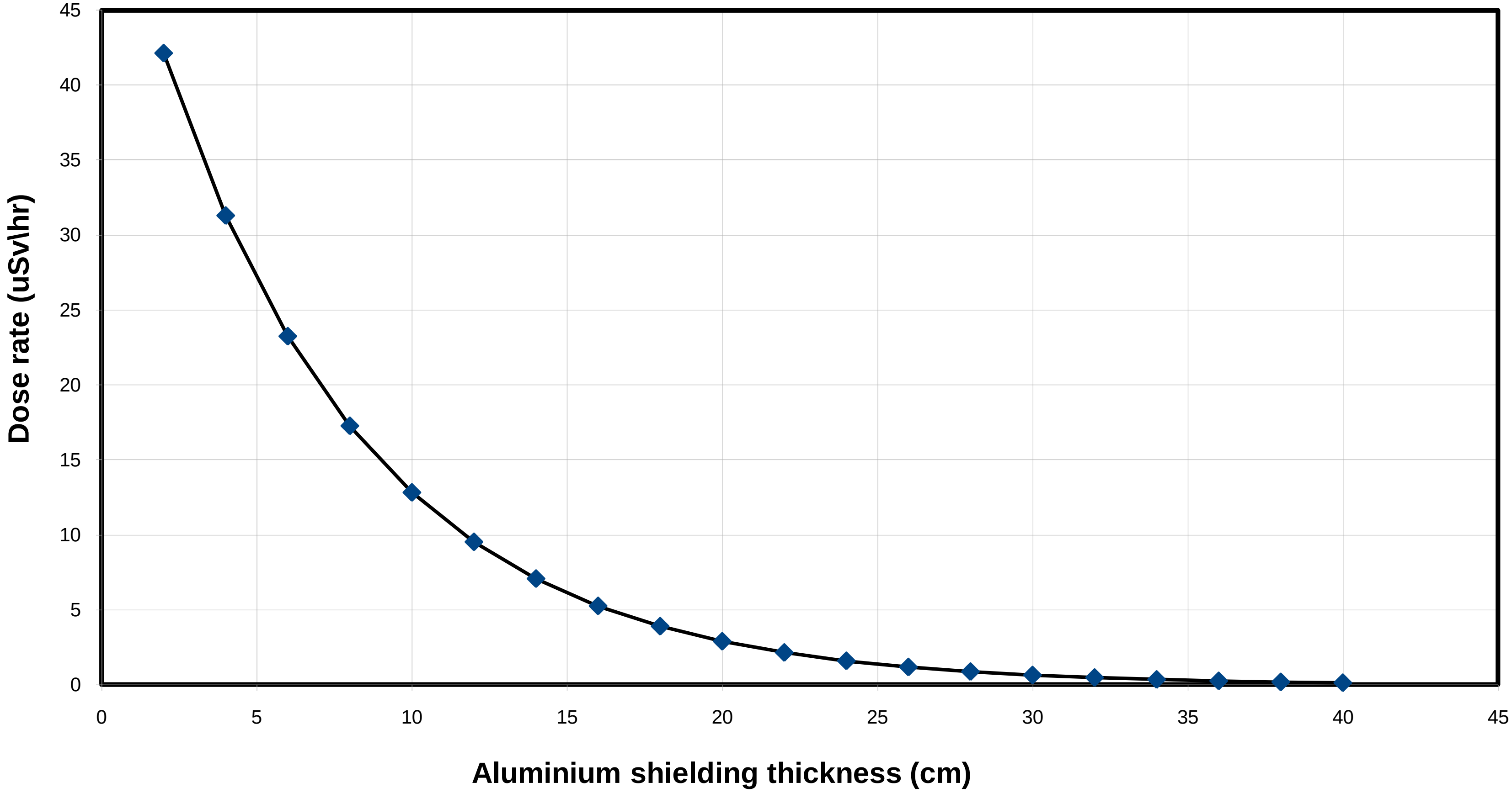



Figure4.6 Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using Aluminum shielding at various thicknesses.

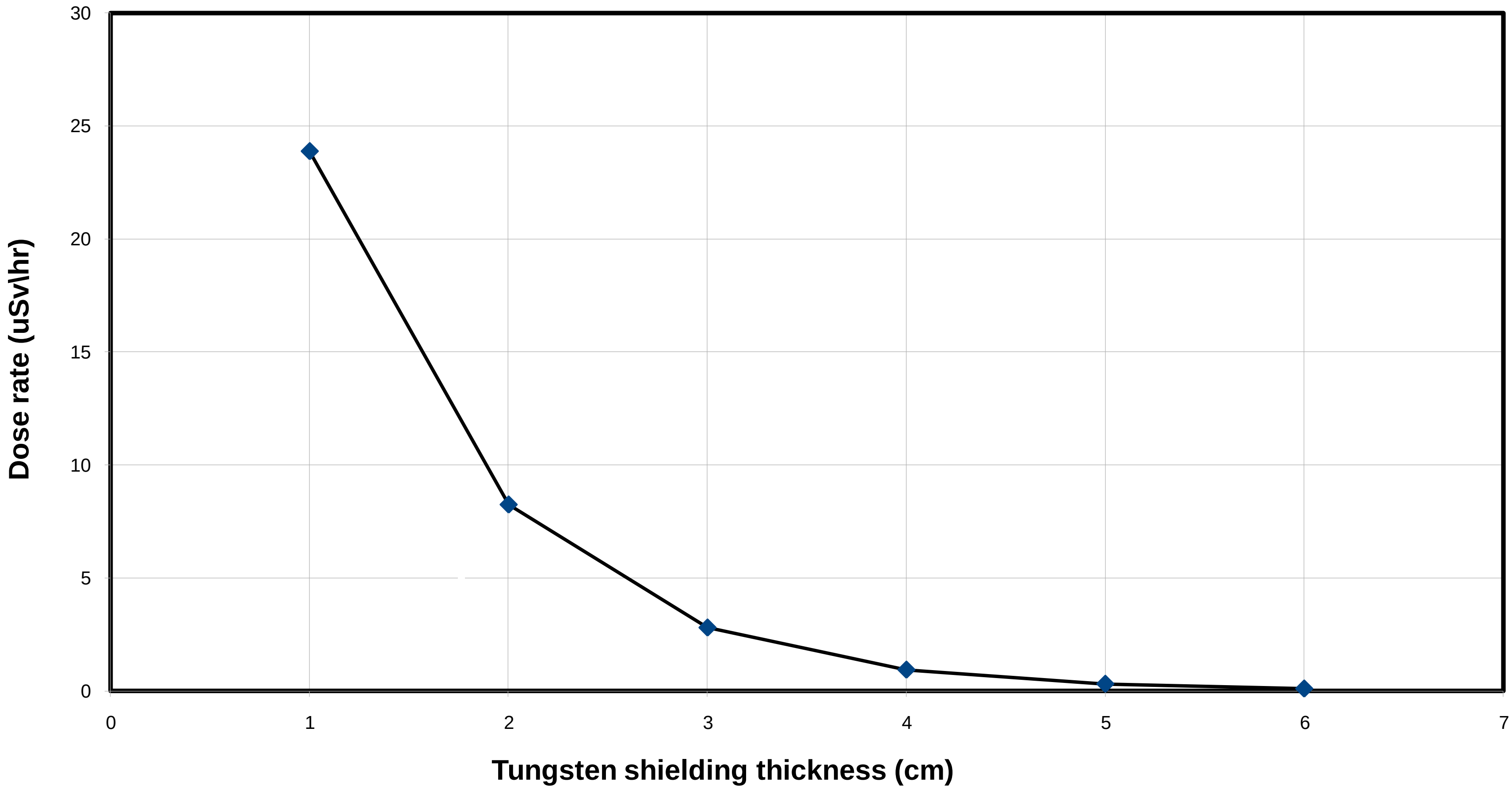



Figure4.7 Dose rate Calculation of 5mCi of 60Co at 1 meter distance from the detector using Tungsten shielding at various thicknesses.

**Cs-137 GRAPH**

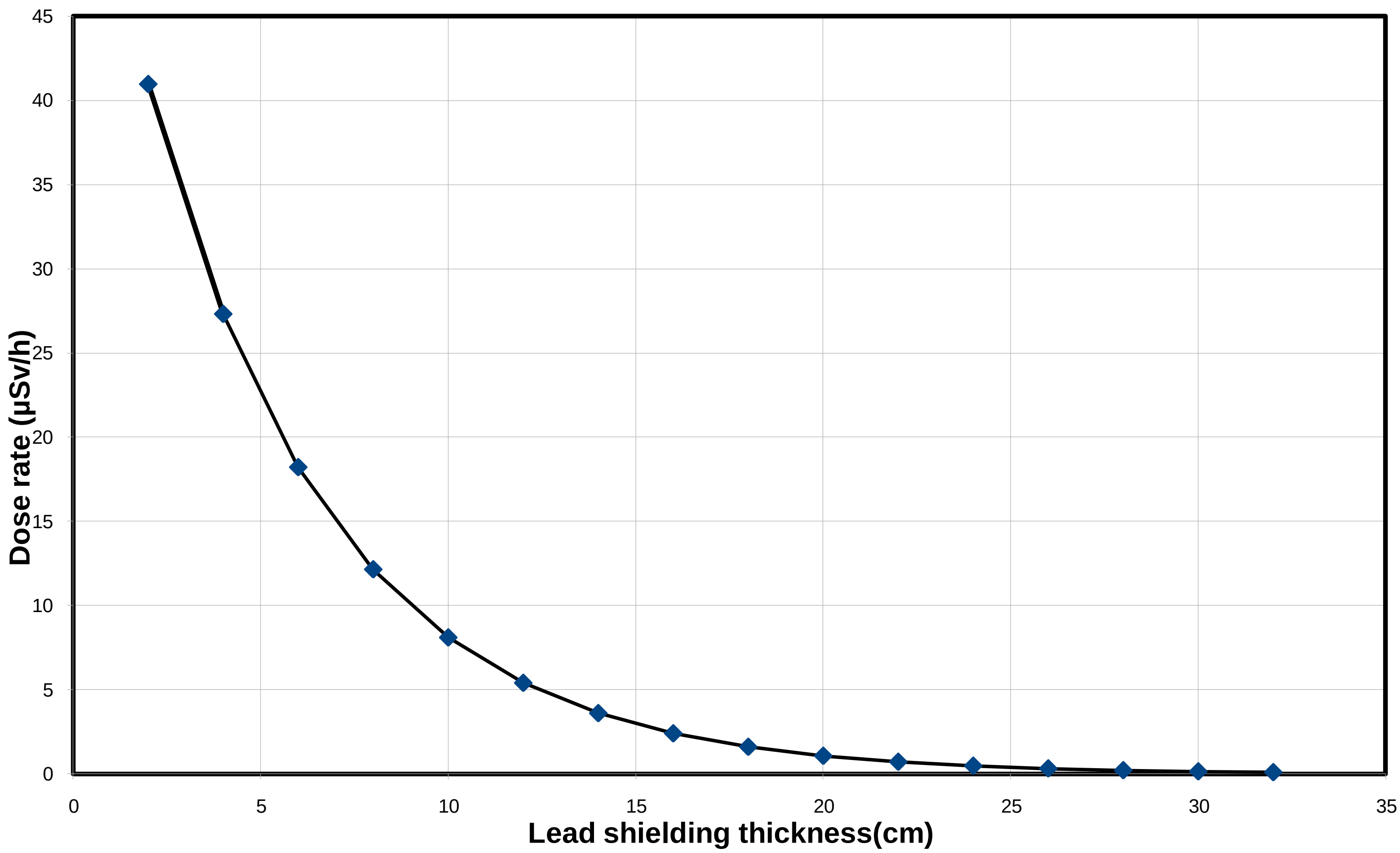



Figure 4.8 Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using lead shielding at various thicknesses.

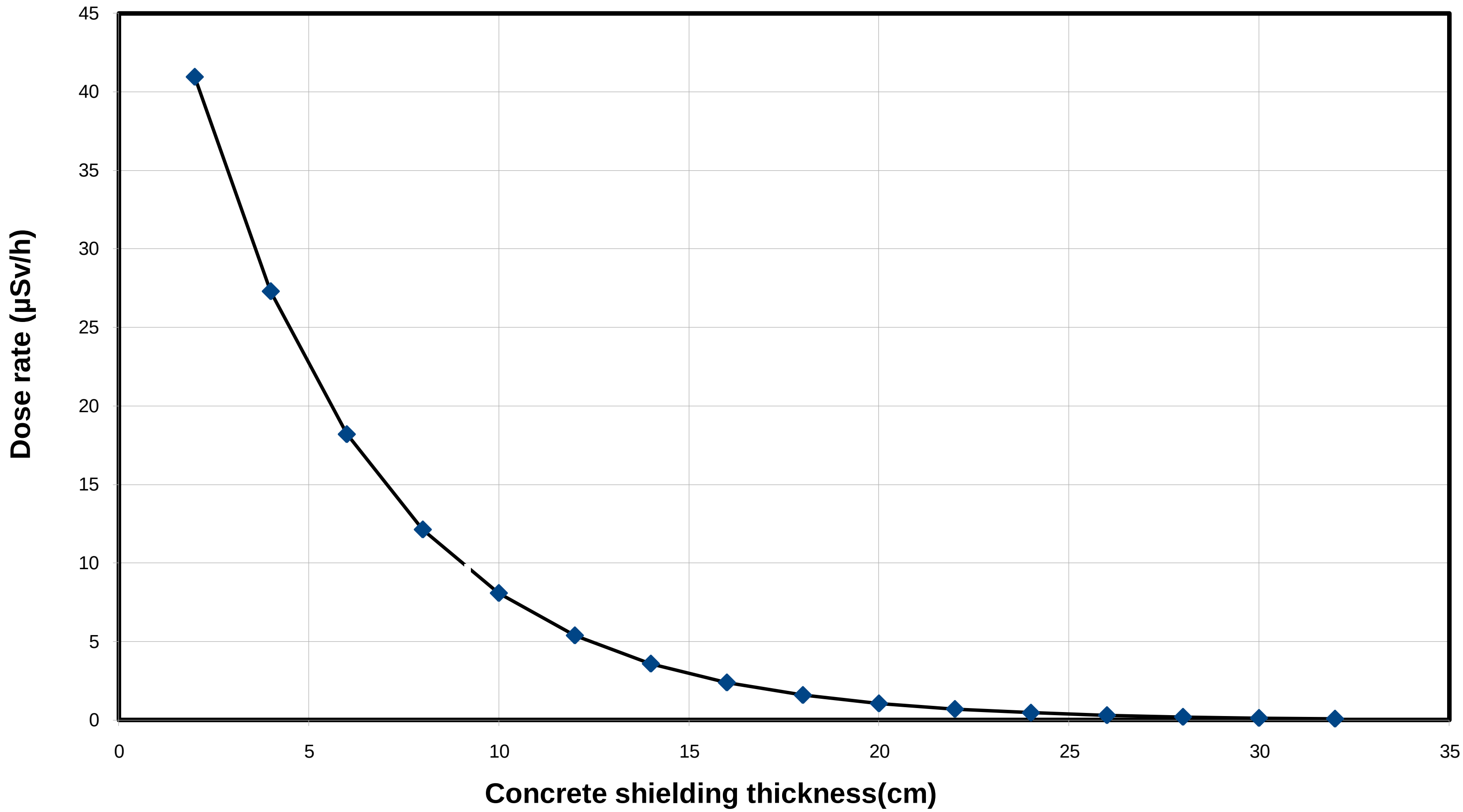



Figure 4.9 Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using concrete shielding at various thicknesses.

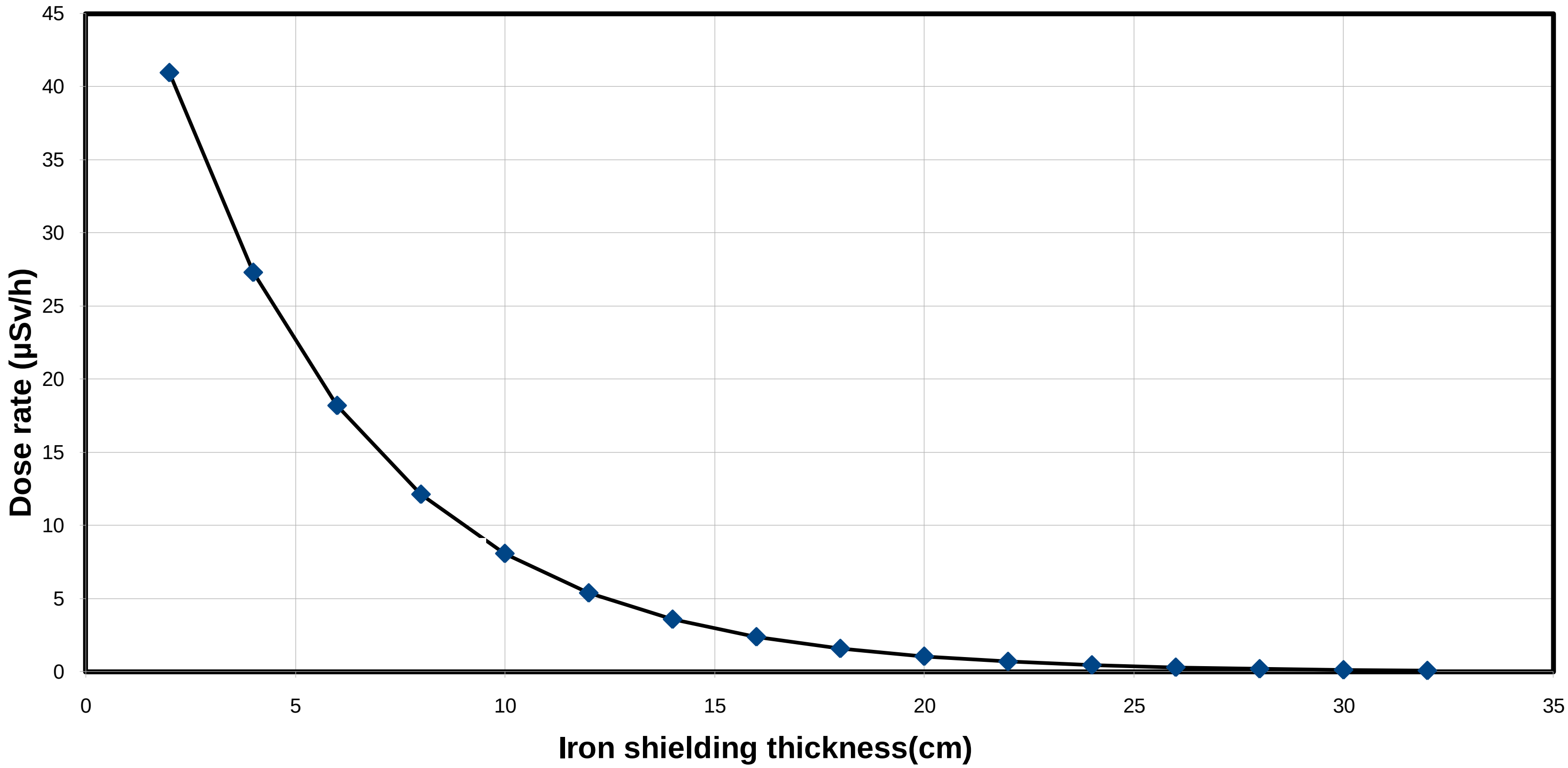



Figure 4.10 Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using Iron shielding at various thicknesses.

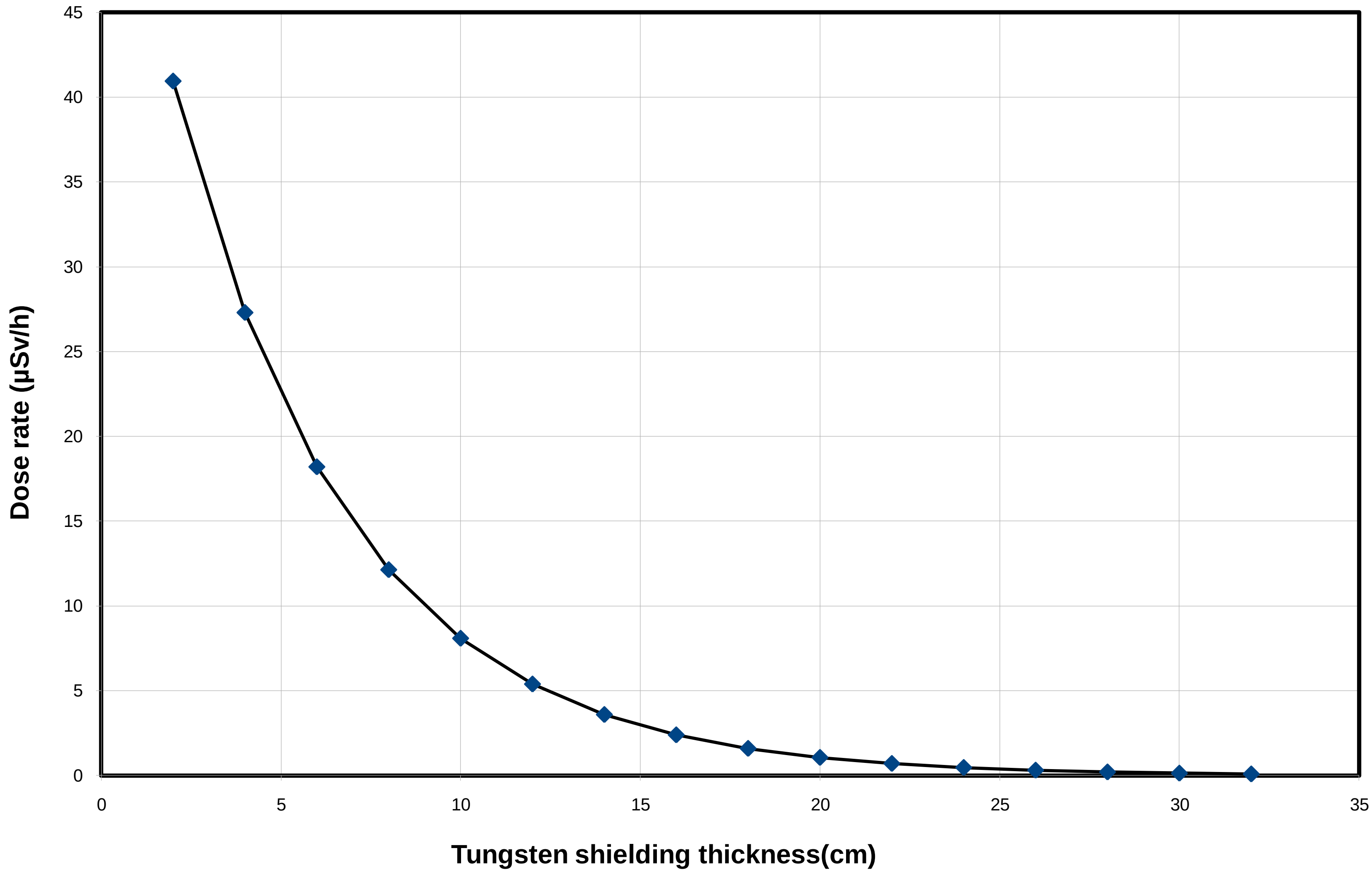



Figure 4.11 Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using Tungsten shielding at various thickness

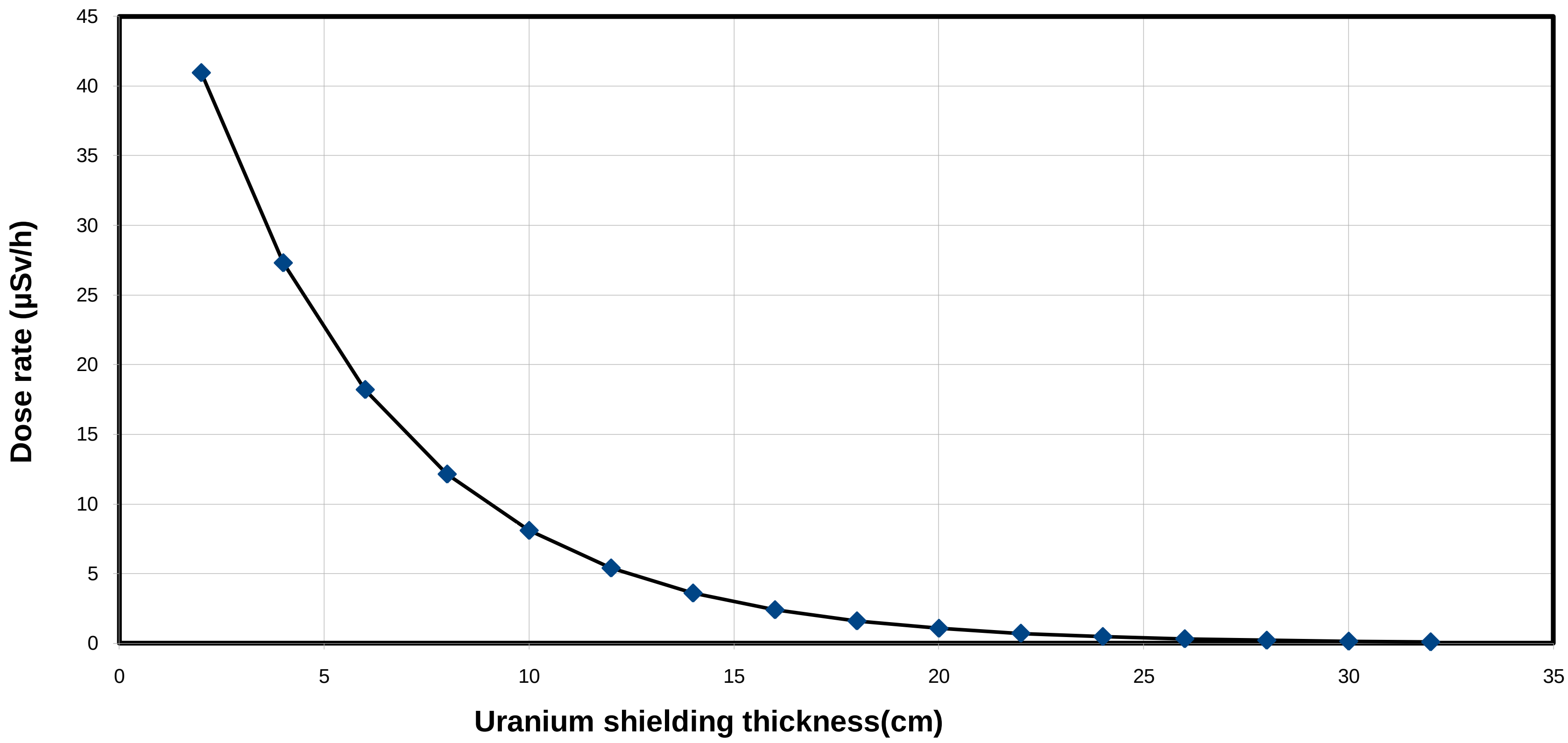



Figure 4.12 Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using Uranium shielding at various thicknesses

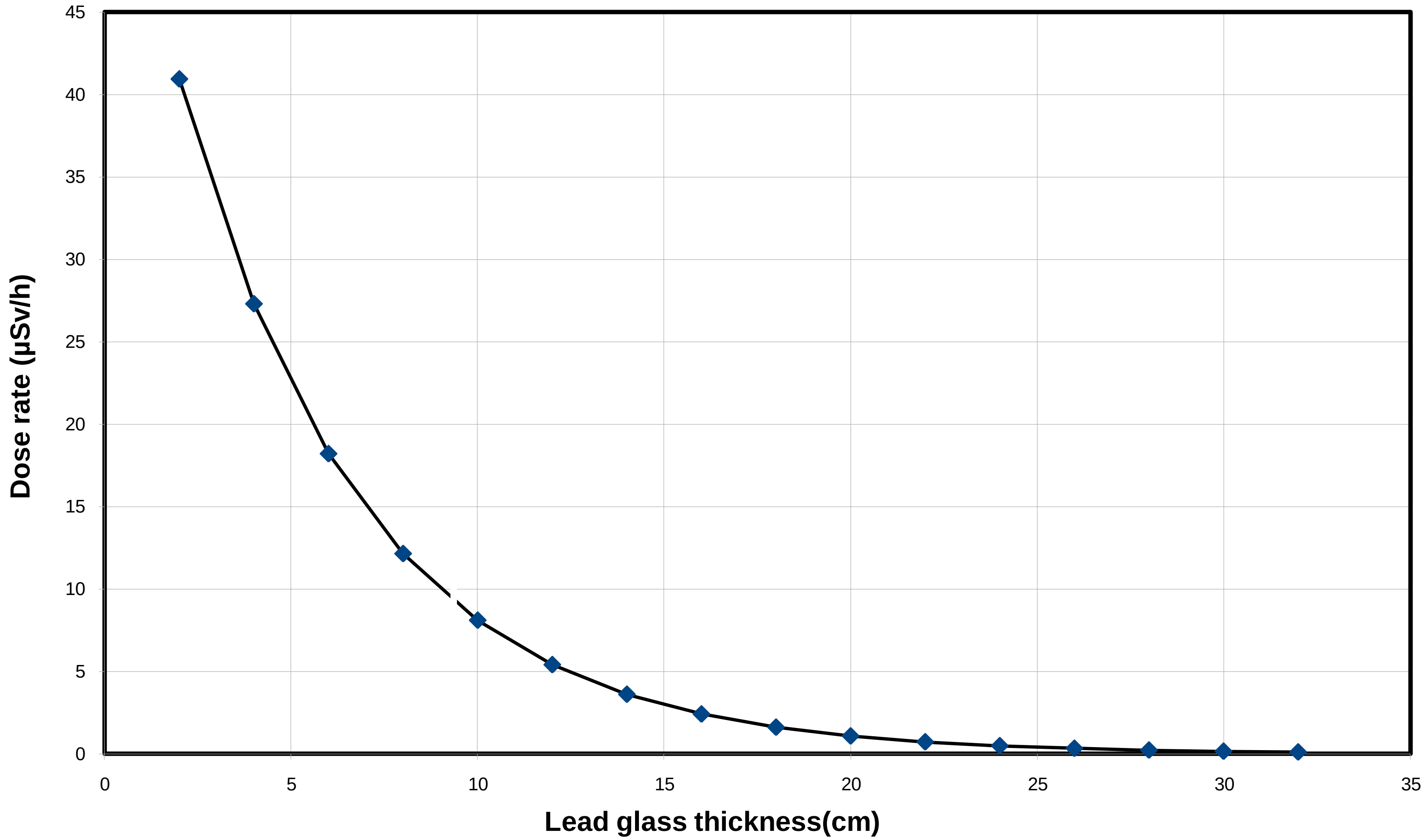



Figure 4.13 Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using Lead glass shielding at various thicknesses

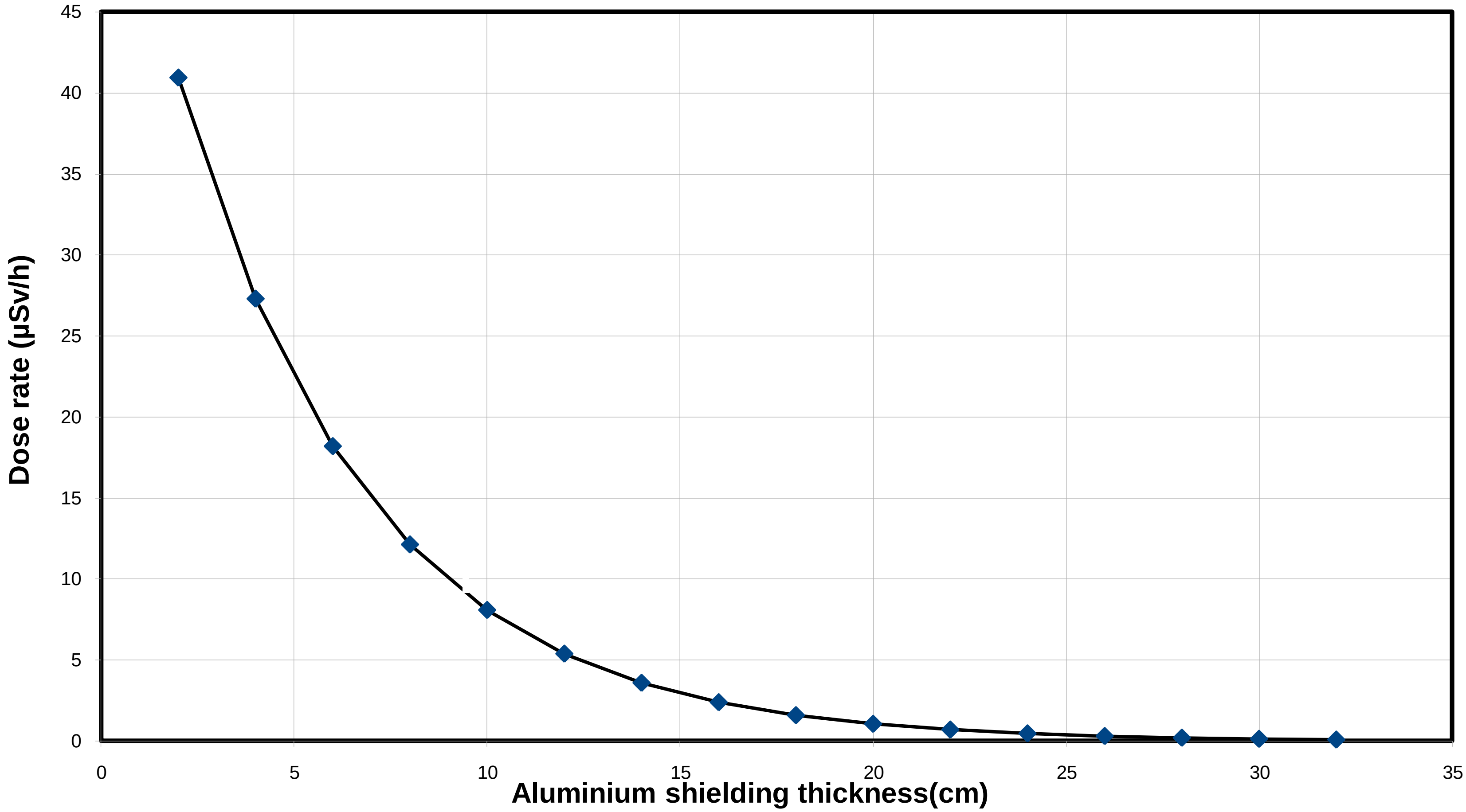



Figure 4.14 Dose rate Calculation of 22mCi of Cs-137 at 1 meter distance from the detector using Aluminum shielding at various thicknesses

PRESENTATION OF COMPARISON GRAPHS

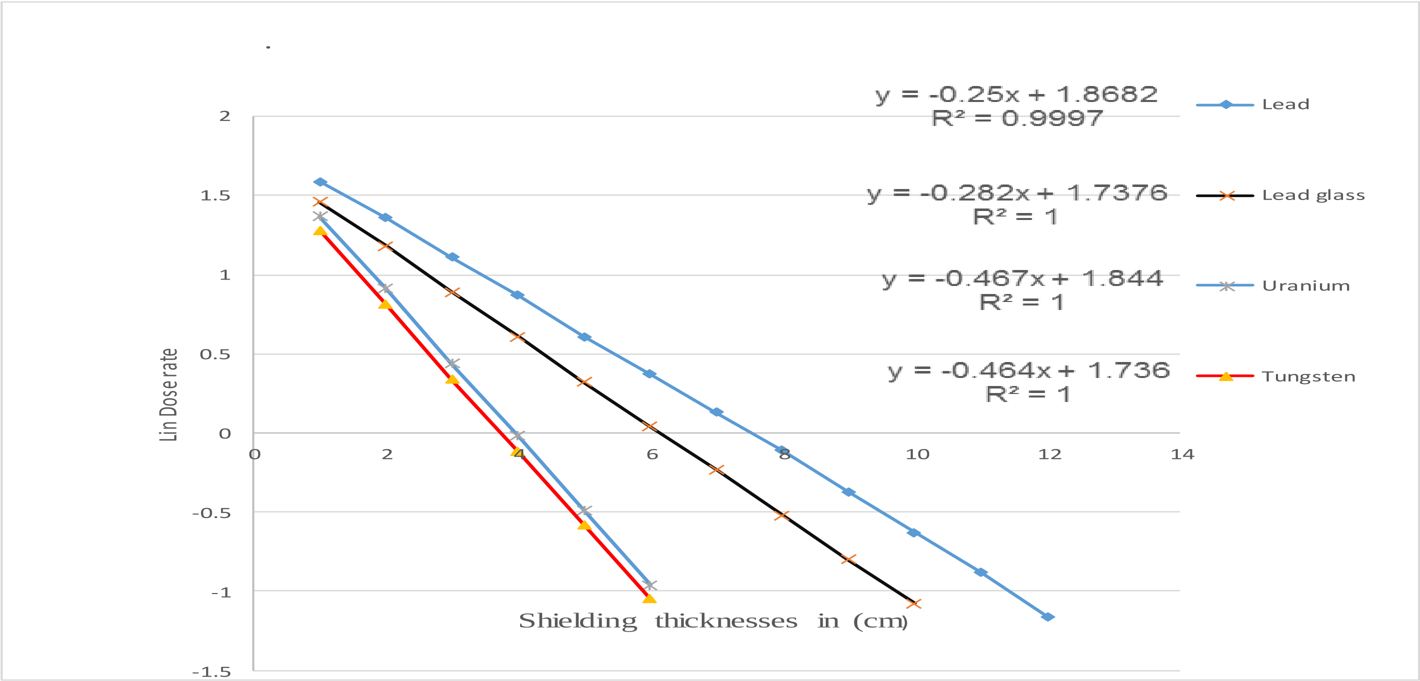



Fig 4.15 Comparisons of attenuation coefficients of 5mCi of Co-60 at 1 meter distance from the source to detector using lead, lead glass, uranium and tungsten at various thicknesses

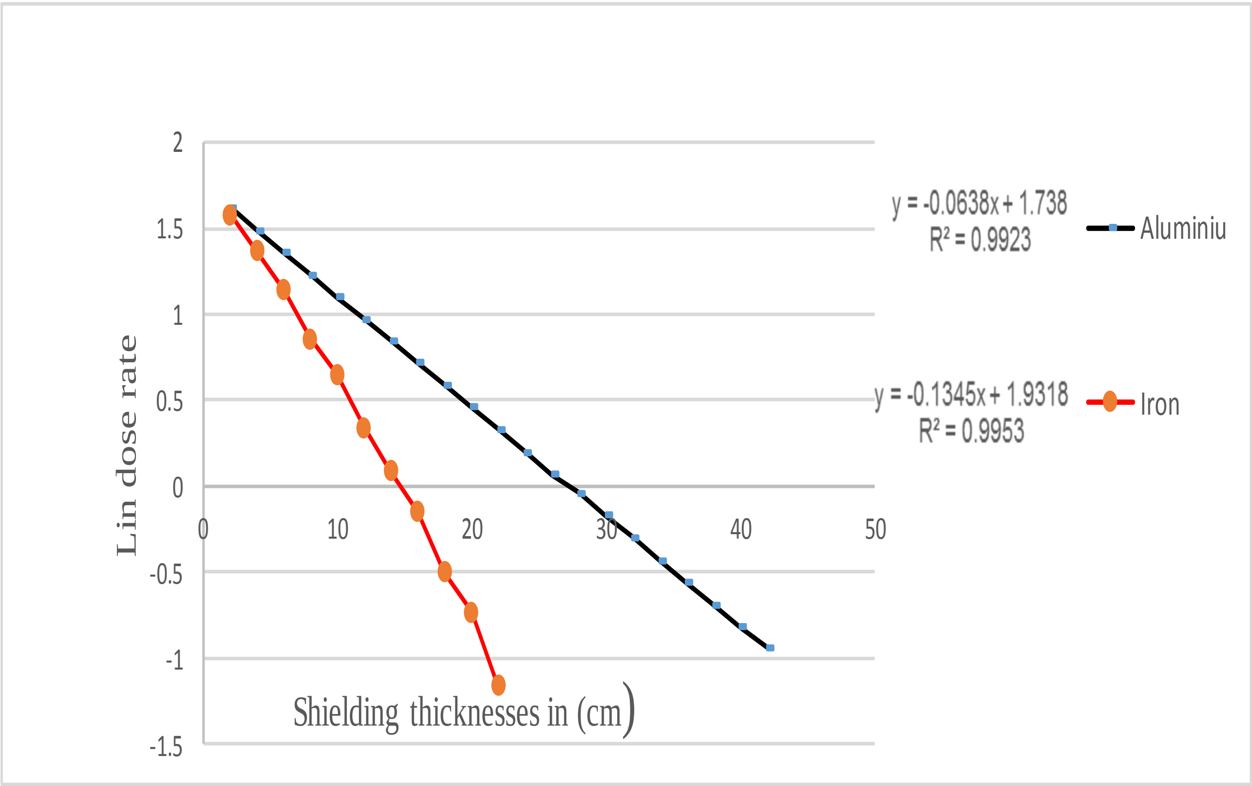



Fig 4.16 Comparisons of attenuation coefficients of 5mCi of Co-60 at 1 meter distance from the source to detector using Aluminum, Iron at various thicknesses.

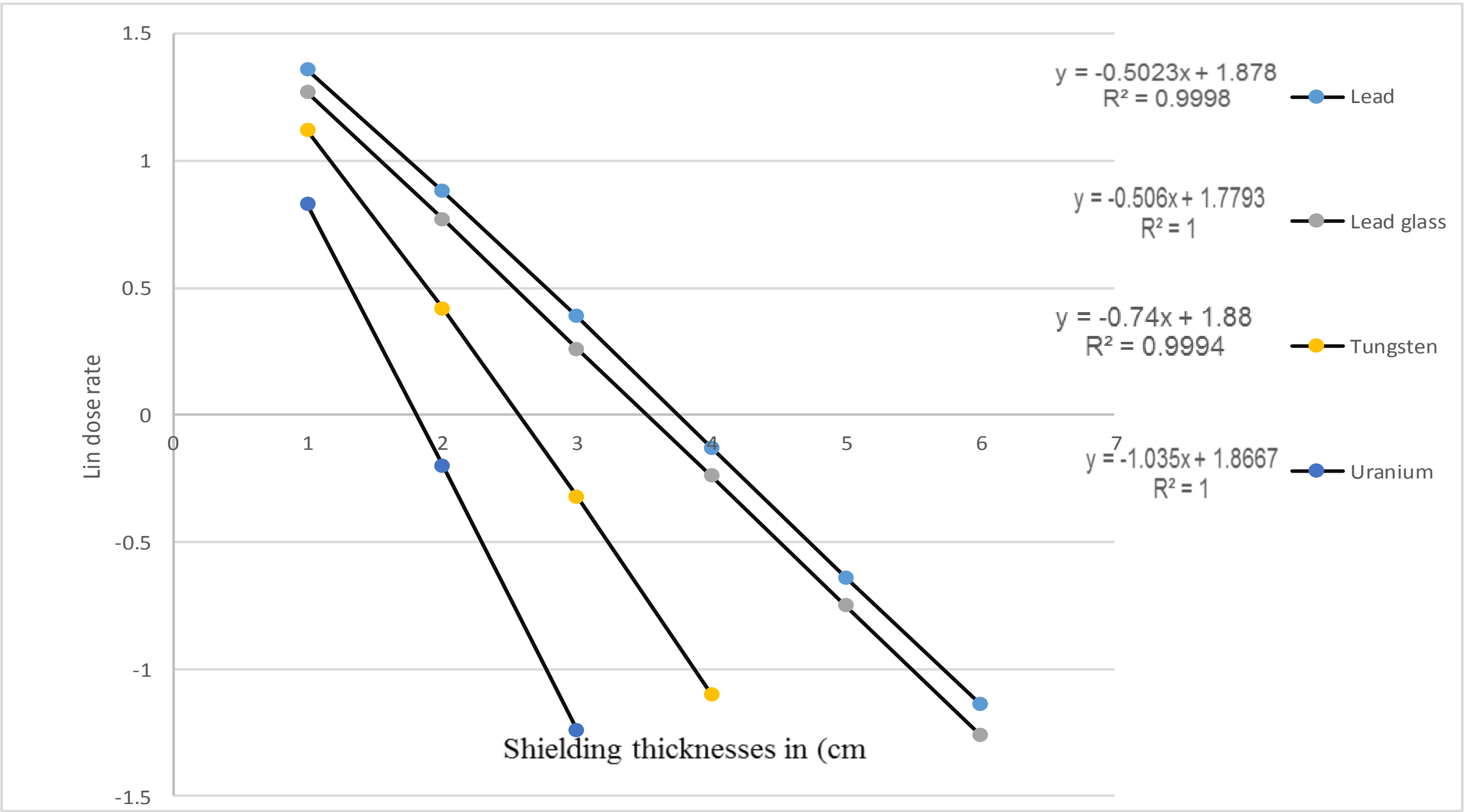



Fig 4.17Comparisons of attenuation coefficients of 22mCi of Cs-137 at 1 meter distance from the source to detector using lead, lead glass, uranium and tungsten at various thicknesses.

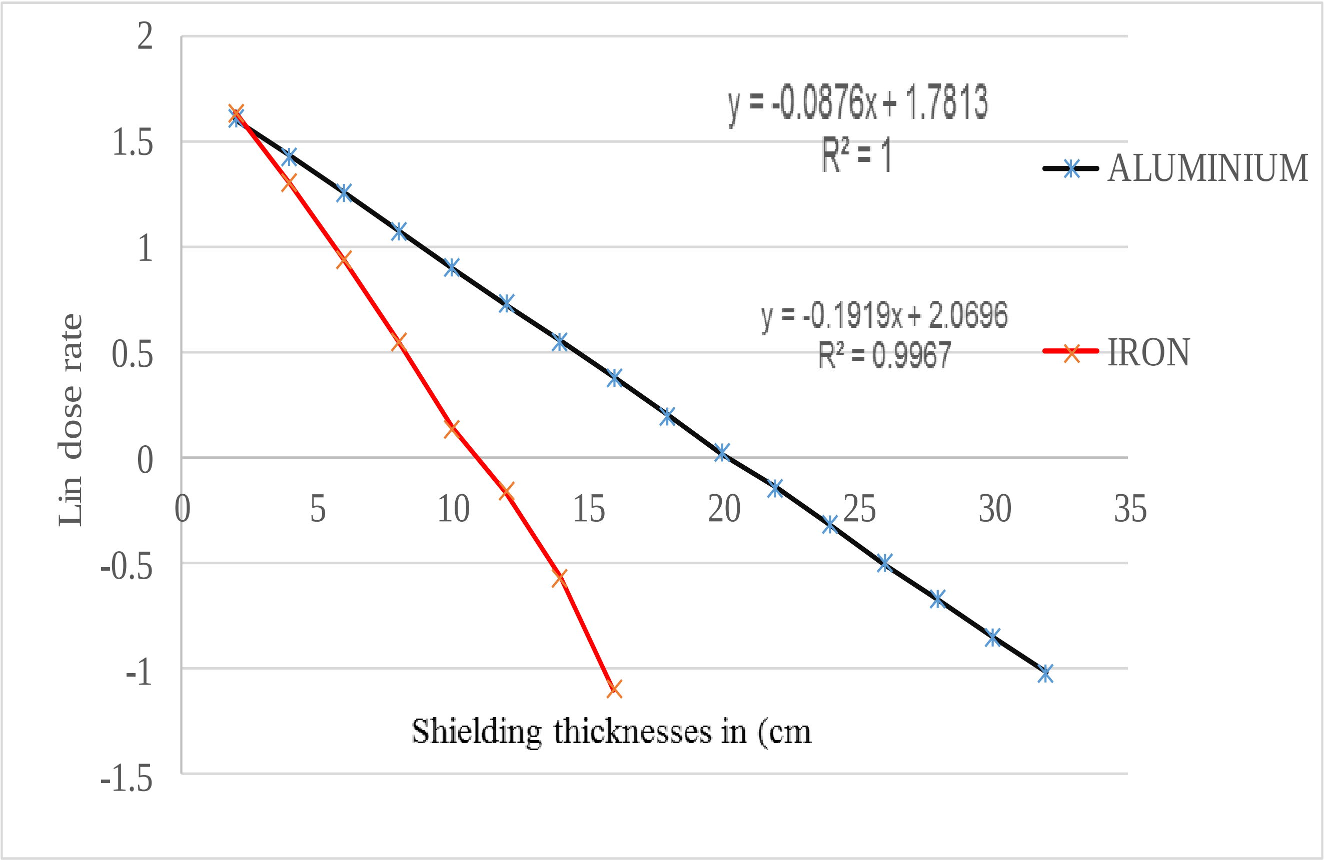



Fig 4.18 Comparisons of attenuation coefficients of 22mCs-137 at 1 meter distance from the source to detector using Aluminum, Iron at various thicknesses

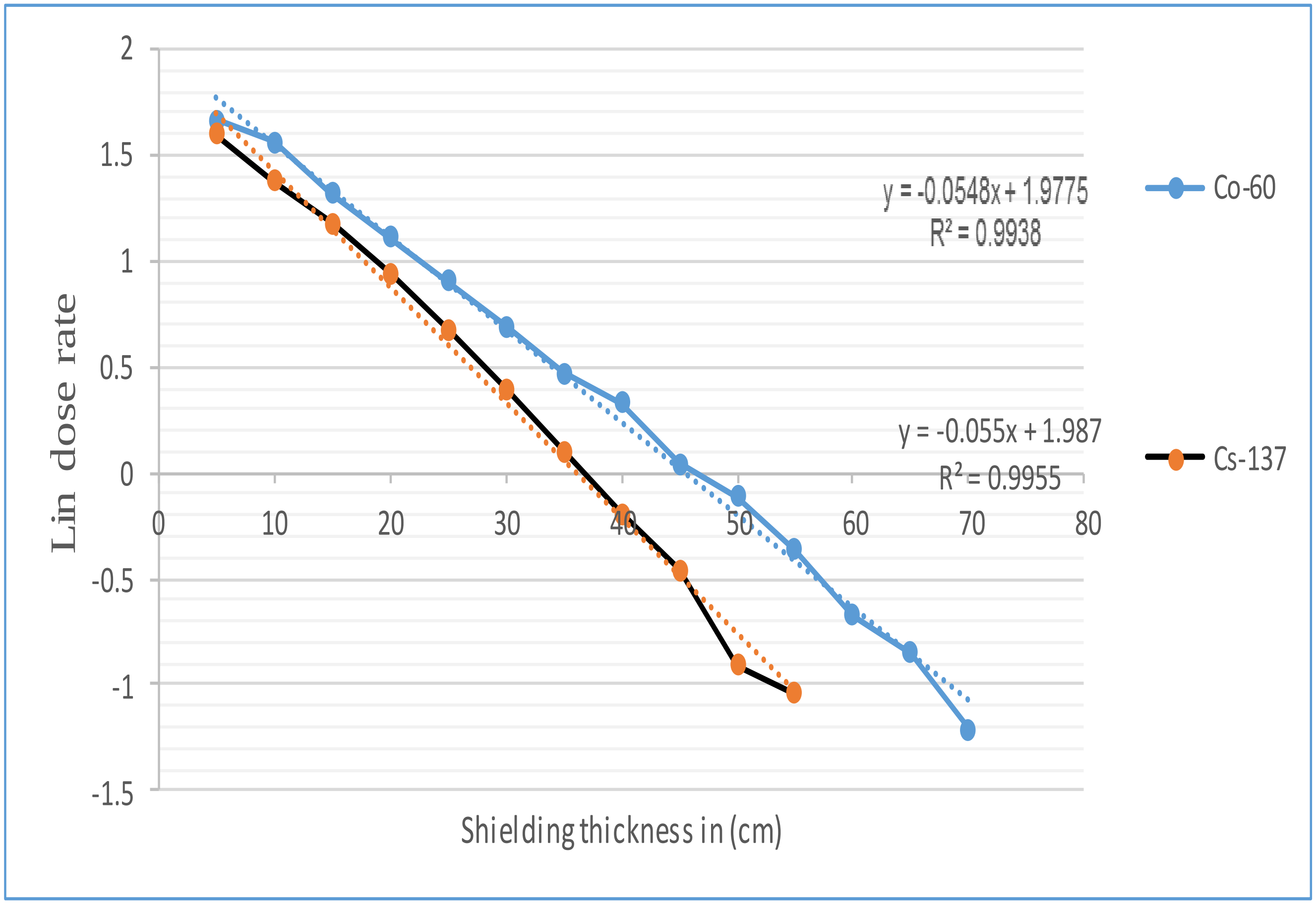



Fig 4.19 Comparisons of attenuation coefficients of Co-60 and 22mCs-137 at 1 meter distance from the source to detector using concrete shielding at various thicknesses.

## 4.5 DISCUSSION

Table4.1 showed the results of calculated dose rate of 5mCi of 60Co at 1m distance from the source to detector using lead shielding at various thicknesses. For decades, lead has been considered the gold standard in radiation shielding, it's cheap, easy to processes, and provide effective shielding. Materials for shielding gamma rays are typically measured by the thickness required to reduce the intensity of the gamma rays (Kaplan, 1989) when we make shielding calculations, we are more concerned among other things about the thickness of the material used in the shielding, the minimum thickness that could give us maximum shielding from the emitting source.

The simulation results obtained with lead showing decreasing photon dose rate with increasing lead thickness has been show on Table4.1 and Figure4.1

The result show that12cm of lead shielding can reduce the dose rate of 5mCi of 60Co at 1m distance from the source to detector to 0.06 μSv/h which is within the allowable limit

Table 4.2 show the results of calculated dose rate of 5mCi of 60Co at 1m distance from the source to detector using concrete shielding at various thicknesses. Most designers and builders today are familiar with the advantages of using very high density concrete for radiation shielding not so well known is excellent economy which can results from the use of normal site-cast concretes with locally available aggregates when space and other factor do not absolutely demand that the desired protection be achieved within minimum dimensional limits. The effectiveness of any biological shielding material is related only to its mass, and concrete has an obvious advantage in this highly specialized field of construction because of its exceptional low cost per pound.

The simulation results obtained with concrete showing decreasing photon dose rate with increasing concrete shielding thickness has been show on Table4.2 and Figure 4.2

The result show that70cm of concrete shielding can reduce the dose rate of 5mCi of 60Co at 1m distance from the source to detector to 0.01uSv however, low density materials can compensate for the disparity with increased thickness which is as significant as density in shielding application.

Table 4.3 show the result of calculated dose rate of 5mCi of 60Co at 1m distance from the source to detector using iron shielding at various thickness. Iron are common gamma ray shield materials and is often used in the situation where the size and configuration of the lead shield alone would make its construction too expensive in such situation the iron will be compromised.

The simulation result show that22cm of iron shielding can reduce the photon dose rate of 5mCi of 60Co at 1m distance from the source to detector to 0.06µSv/h.

Table 4.4 show the results of calculated dose rate of 5mCi of 60Co at 1m distance from the source to detector using uranium shielding at various thickness. The uranium has been in casks as shielding because its high density provides the needed gamma attenuation for the lowest-weight and smallest casks. Studies have assessed the use of uranium for shielding in both spent fuel and high level waste casks uranium fabrication and production costs showed that uranium was expensive than other common shielding materials. Therefore, the primary application for uranium shielding is for transportation casks, where the most stringent total-package size and weight limit exist and where high-cost, uranium shielding can be justified.

Also, there is an added benefit to the nuclear community if this use as shielding consumes large quantity of source.

The simulation results obtained with uranium showing decreasing photon dose rate with increasing uranium shielding thickness has been show on Table4.4 and Figure 4.4

The result show that 6cm of uranium shielding can reduce the dose rate of 5mCi of 60Co at 1m distance from the source to detector to 0.11µSv/h.

Table 4.5 show the results of calculated dose rate of 5mCi of 60Co at 1m distance from the source to detector using lead glass shielding at various thickness. Radiation shielding lead glass and radiation protector glass can be used in any facility which requires protection from x-ray radiation. Lead glass can be used for x-ray observation equipment, electron beam/ plasma generators and x-ray TV detectors, lead glass protects doctors and staff from x-ray irradiation with no glass discoloration or deterioration in viewing quality, lead glass protect people from airport luggage inspection equipment in airports, lead glass can be used for observation windows at radioactive storage stations, nuclear fuel developments and reprocessing plants and for application near nuclear reactors.

The simulation results obtained with lead glass showing decreasing photon dose rate with increasing uranium shielding thickness has been show on Table4.5 and Figure 4.5

The result shows that10cm of lead glass shielding can reduce the dose rate of 5mCi of 60Co at 1m distance from the source to detector to 0.08µSv/h.

Table 4.6 show the results of calculated dose rate of 5mCi of $^{60}$Co at 1m distance from the source to detector using aluminum shielding at various thickness. Aluminum are common gamma ray shielding material. And is often used because of its low cost but not much good observer but it can be compensate by maximum thickness to achieve proffer shield.

The simulation results obtained with lead glass showing decreasing photon dose rate with increasing aluminum shielding thickness has been show on Table4.6 and Figure 4.6

The result shows that 42cm of aluminum shielding can reduce the dose rate of 5mCi of 60Co at 1m distance from the source to detector to 0.11µSv/h.

Table 4.7 show the results of calculated dose rate of 5mCi of 60Co at 1m distance from the source to detector using tungsten shielding at various thickness. With rapid development of medical sciences, more and more of tungsten shielding radiation shielding are used in our lives. It's essential that people paid attention to radiation and even more important for the institutions to protect public from radiation exposure and to make sure to protect every radiation sources of x-ray radiation s. in order to protect patients, doctors, nurses and other people who may be exposed to radiation, sources of radiation must be safely separated and shielded. It's crucial that holding and delivery instruments for radioactive materials would keep the radiation levels low enough, not to create harmful effects of ionizing radiation such as breast cancer, skin cancer, etc. lead and steel are the traditional protection materials, but tungsten shielding is without a doubt the best solution excellent radiation- absorption and radiation shielding, twice the density of lead and good physical resistance are main reasons to use tungsten radiation shielding. The density of a material is related to its radiation protection ability. Higher density means better stopping and absorption radiation ray. Due to a higher density than most other materials, tungsten radiation shielding has a much higher ability of absorbing and stopping almost ray than others such as traditional radiation shielding. Tungsten radiation shielding greater attenuation of gamma radiations means that less is required for equal shielding. Alternatively equal amounts of tungsten shielding provide diminished exposure risks than equivalent lead. Tungsten is the right material for radiation protection, as its combination of radiographic density ( more than 60% dense lead), good corrosion resistance, high radiation absorption (superior to lead and steel), simplified life cycle and high strength. Tungsten radiation can provide the same degree of protection  as lead while significantly reducing the overall volume and thickness of shields and containers beside compare with or uranium, tungsten radiation shielding is more friendly for the environment both lead and uranium for its no any toxic.

The simulation results obtained with tungsten showing decreasing photon dose rate with increasing tungsten shielding thickness has been show on Table 4.7 and Figure 4.7

The result show that 6 cm of tungsten shielding can reduce the dose rate of 5 mCi of 60Co at 1 m distance from the source to detector to 0.08 µSv/h.

Table 4.8 showed the results of calculated dose rate of 22 mCi of Cs-137 at 1m distance from the source to detector using lead shielding at various thicknesses.

The simulation results obtained with lead showing decreasing photon dose rate with increasing lead thickness has been show on Table 4.8 and Figure 4.8

The result show that6cm of lead shielding can reduce the dose rate of 22 mCi of Cs-137 at 1m distance from the source to detector to 0.07 µSv/h. compare with Table 4.1 the difference because of high energy of cobalt-60 compared.

Table 4.9 showed the results of calculated dose rate of 22 mCi of Cs-137 at 1m distance from the source to detector using concrete shielding at various thicknesses.

The simulation results obtained with concrete showing decreasing photon dose rate with increasing concrete thickness has been show on Table 4.9 and Figure 4.9

The result show that 55 cm of concrete shielding can reduce the dose rate of 22mCi of Cs-137 at 1m distance from the source to detector to 0.08 µSv/h. using the same shielding material as the case of C0-60 but due to the energies difference Co-60 required more thickly than Cs-137.

Table 4.10 showed the results of calculated dose rate of 22mCi of Cs-137 at 1m distance from the source to detector using iron shielding at various thicknesses.

The simulation results obtained with concrete showing decreasing photon dose rate with increasing iron thickness has been show on Table 4.10 and Figure 4.10

The result show that16cm of iron shielding can reduce the dose rate of 22mCi of Cs-137 at 1m distance from the source to detector to 0.07 µSv/h. unlike that of 60Co which has higher energy.

Table 4.11 showed the results of calculated dose rate of 22mCi of Cs-137 at 1m distance from the source to detector using tungsten shielding at various thicknesses.

The simulation results obtained with tungsten showing decreasing photon dose rate with increasing tungsten thickness has been show on Table 4.11 and Figure 4.11

The result shows that 4 cm of iron shielding can reduce the dose rate of 22 mCi of Cs-137 at 1m distance from the source to detector to 0.07 µSv/h.

Table 4.12 showed the results of calculated dose rate of 22 mCi of Cs-137 at 1 m distance from the source to detector using uranium shielding at various thicknesses.

The simulation results obtained with uranium showing decreasing photon dose rate with increasing uranium thickness has been show on Table 4.12 and Figure 4.12

The result shows that 3 cm of uranium shielding can reduce the dose rate of 22 mCi of Cs-137 at 1 m distance from the source to detector to 0.05µSv/h.

Table 4.13 showed the results of calculated dose rate of 22 mCi of Cs-137 at 1 m distance from the source to detector using lead glass shielding at various thicknesses.

The simulation results obtained with lead glass showing decreasing photon dose rate with increasing lead glass thickness has been show on Table 4.13 and Figure 4.13

The result shows that 6cm of lead glass shielding can reduce the dose rate of 22 mCi of Cs-137 at 1m distance from the source to detector to 0.05 µSv/h.

Table 4.14 showed the results of calculated dose rate of 22mCi of Cs-137 at 1m distance from the source to detector using aluminum shielding at various thicknesses.

The simulation results obtained aluminum with showing decreasing photon dose rate with increasing aluminum thickness has been show on Table 4.14 and Figure 4.14

The result shows that 6cm of aluminum shielding can reduce the dose rate of 22 mCi of Cs-137 at 1m distance from the source to detector to 0.09 µSv/h.

Table 4.15 shows the comparison of attenuation coefficient of Rad pro calculator with the tabulations based upon the results of photon cross section database (XCOM) is done by calculating the Percentage deviation. Linear attenuation coefficient Results of the Rad Pro Calculator have been compared with tabulations based upon the results of photon cross section database (XCOM). The calculated Rad Pro Calculator attenuation coefficients were found to be very close to the (XCOM) values.

The linear attenuation coefficient is obtained by multiplying the mass attenuation coefficient of the material by its density. The Rad pro calculator values of attenuation coefficient and that of (XCOM) values shows excellent accuracy. The (XCOM) values with Rad pro calculator confirm the considerations of the contribution of various processes such as photoelectric effect, the Compton scattering and the pair production. The data is useful in radiation dosimetry and other fields. To decide the radiation to be delivered without any harm to normal cells it is necessary to have a precise knowledge of gamma ray photon attenuation and consequent absorption (Pawar,P.P and Mahajan,C.S, 2013).

Fig 4.15Comparisons of attenuation coefficients of 5mCi of Co-60 at 1 meter distance from the source to detector using lead, lead glass, uranium and tungsten at various thicknesses as 0.25/cm, 0.28/cm, 0.47/cm, 0.46/cm.

Fig 4.16Comparisons of attenuation coefficients of 5mCi of Co-60 at 1 meter distance from the source to detector using Aluminum, Iron at various thicknesses as 0.06/cm, 0.13/cm.

Fig 4.17Comparisons of attenuation coefficients of 22mCi of Cs-137 at 1 meter distance from the source to detector using lead, lead glass, uranium and tungsten at various thicknesses as 0.50/cm, 0.51/cm, 0.74/cm, 1.03/cm

Fig 4.18 Comparisons of attenuation coefficients of 22mCs-137 at 1 meter distance from the source to detector using Aluminum, Iron at various thicknesses as 0.09/cm, 0.19/cm

Fig 4.19 Comparisons of attenuation coefficients of Co-60 and 22mCs-137 at 1 meter distance from the source to detector using concrete shielding at various thicknesses as 0.05/cm, 0.04 /cm.

## CHAPTER FIVE

### 5.0 CONCLUSION

The graphs have a similar behavior, in that the photon dose rate is inversely proportional to the absorber thickness in an almost exponential form. As the absorber's thickness increases, the photon dose rate decreases; conversely .This is reasonably true, since the absorber is to shields, absorbs or reduces the number of photons dose rate going into the detector, then the more you increase its thickness the more it does this work and the lesser the photons dose rate that get to the detector and then the lower the photons dose rate .The exponential relationship for photon absorption suggests that, theoretically, complete absorption photon radiation never really occur, (Nwosu, O. B 2015). From a distance, the graphs of these seven absorbing materials (Lead, Concrete, Iron, Uranium, Lead glass, Aluminum and Tungsten) look very similar, but on a closer look, one will identify a striking and important differences amongst these seven materials especially tungsten. While only 6 cm thickness of tungsten is required to reduce the photon dose rate of 5 mCi of 60Co at 1 meter distance from the source to detector to 0.088 µSv/h, about 6 cm of uranium thickness to 0.10 µSv/h , 10 cm of Lead glass thickness is required to reduce to 0.08 µSv/h, 12 cm of lead reduce to 0.06 µSv/h, 22 cm of Iron shielding reduce the dose rate to 0.06 µSv/h, 42 cm of Aluminum shielding reduce the dose rate to 0.11 µSv/h, 70 cm of concrete shielding require to reduce the dose rate to 0.06 µSv/h. This then goes to show that tungsten is a far better absorber of 60C0 gamma photon when compared to uranium, lead glass, lead, iron, aluminum and concrete can be used to shield against gamma ray. and for the case of cesium 3 cm of uranium shielding is require to reduce the photon dose rate of 22m mCi of Cs-137 at 1 m from the source to detector to 0.05 µSv/h, 4 cm of tungsten to 0.07 µSv/h, lead glass to 0.05 µSv/h, lead to 6 cm to 0.07 µSv/h, 16 cm of iron shielding to 0.07 µSv/h, 32 cm of aluminum shielding to 0.091 µSv/h, 55 cm of concrete shielding to 0.08 µSv/h. In conclusion, Rad Pro Calculator simulation is a veritable tool for modeling certain real life situations, and very useful in shielding calculation. The above graphs have also show that out of the seven materials, uranium and tungsten is a better shield because it requires just a few thicknesses of it to cut down the photon dose rate to acceptable limit, then followed by lead glass, lead, iron, aluminum and concrete being the least. Materials for shielding gamma rays are typically measured by the thickness required to reduce the intensity of the gamma rays(Kaplan,1989) when we make shielding calculations, we are more among other things about the thickness of the material used in the shielding, the minimum thickness that could give us maximum shielding from the emitting source. Uranium and tungsten shielding because of their high density provides the needed gamma attenuation for the lowest-weight and smallest casks. Casks production and fabrication costs showed that Uranium and tungsten was more expensive than other common shielding materials such as lead glass, lead, iron, aluminum and concrete. Therefore, the primary application for uranium metal shielding is for transportation casks, where the most stringent total-package size and weight limits exist and where high-cost, shielding can be justified. Also, there is an added benefit to the nuclear community if this use as shielding consumes large quantities of sources.

In the situation where by construction of shield with high density material will be expensive, however low density materials will compromise, low density materials can compensate with increased thickness. This is as significant as density in shielding application.

## 5.1 RECOMMENDATIONS

1.0 Materials for shielding gamma rays are typically measured by the thickness required to reduce the intensity of the gamma rays( Kaplan,1989) when we make shielding calculations, we are more among other things about the thickness of the material used in the shielding, the minimum thickness that could give us maximum shielding from the emitting source. Uranium and tungsten shielding because of their high density provides the needed gamma attenuation for the lowest-weight and smallest casks. Casks production and fabrication costs showed that Uranium and tungsten was more expensive than other common shielding materials such as lead glass, lead, iron, aluminum and concrete. Therefore, the primary application for uranium metal shielding is for transportation casks, where the most stringent total-package size and weight limits exist and where high-cost, shielding can be justified. Also, there is an added benefit to the nuclear community if this use as shielding consumes large quantities of sources. Therefore I recommended using uranium and tungsten shielding materials.

2.0 It’s also recommended that a more comprehensive code like the MCNP can be deployed for future work.

3.0 For further research there is need to carry out research on measurement since this one is purely computational.

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