



A Comparative Calculation of Dose Rate Using Rad Pro Calculator for Shielding Requirements of Cobalt-60 and Cesium-137 Sources.

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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 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 1 meter distance from the source to detector to $0.08\mu\text{Sv/h}$, 6cm of uranium thickness reduces it to $0.11\mu\text{Sv/h}$, 10cm of Lead glass thickness is required to reduce it to $0.08\mu\text{Sv/h}$, 12cm of lead reduce to $0.07\mu\text{Sv/h}$, 22 cm of Iron shielding reduce the dose rate to $0.07\mu\text{Sv/h}$, 42cm of Aluminum shielding reduce the dose rate to $0.11\mu\text{Sv/h}$, 70cm of concrete shielding require to reduce the dose rate to $0.06\mu\text{Sv/h}$. 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 $0.05\mu\text{Sv/h}$, 4 cm of tungsten to $0.07\mu\text{Sv/h}$, lead glass to $0.05\mu\text{Sv/h}$, lead to 6 cm to $0.07\mu\text{Sv/h}$, 16cm of iron shielding to $0.07\mu\text{Sv/h}$, 32cm of aluminum shielding to $0.09\mu\text{Sv/h}$, 55cm of concrete shielding to $0.08\mu\text{Sv/h}$. 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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INTRODUCTION

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

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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). 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 2mm 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.

International Committee on X-Ray and Radium Protection issued the following recommendations for storage of radium sources. Tubes and applicators should have at least 5cm of lead shielding per 100mg of radium. Radium solutions required lead shielding ranging from 15cm for a 0.5g source to 30cm 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 100mg to 6cm 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 0.1Ci. 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).

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.

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).

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

The aim of this paper 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.

THEORY OF METHODS

Absorbed Dose

In dosimetry, it is useful to define an average dose for a tissue or organ DT . The absorbed dose to the mass δmT , is defined as the imparted energy δET per unit mass of the tissue or organ, i.e. (Magill, 2013)
 $DT = \delta ET / \delta mT \dots \dots \dots 2.0$

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)

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. (Magill, 2013)

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 HT in tissue or organ is defined as the product of the absorbed dose and the quality or weighting factor, i.e.

$HT = DT, R \cdot wR \dots 2.1$

Where DT, R is the averaged absorbed dose in tissue T from a given type of Radiation R. If there are several types of radiation present, then the equivalent dose is the weighted sum over all contributions, i.e.

$HT = \sum_R (DT, R \cdot wR) \dots 2.2$

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

$dHT/dt = dDT, R/dt \cdot wR \dots 2.3$

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 wT are introduced to account

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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)

$$E = \sum_T (HT \cdot WT) \dots 2.4$$

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.

$$\mu = \mu_{pe} + \mu_{cs} + \mu_{pp} \dots 2.5$$

Attenuation of Gamma Radiation

Linear Attenuation Shielding Formula:

$$I_B = I_A \cdot e^{-\mu x} \dots 2.6$$

Where: I_B = the shielded dose rate, I_A = the initial dose rate, μ = the linear attenuation coefficient in cm^{-1} , 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.

MATERIALS AND METHODS

Calculation of dose rate involves the use of laboratory equipment and software; these

include Rados dose-rate meter, and Cobalt-60, Cesium 137 and Rad Pro Calculator which were used in this research. All materials were obtained at Centre for Energy Research and Training, A.B.U-Zaria (CERT/ABU).

Materials

Rad pro calculator

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.

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

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.

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 α - 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

Method

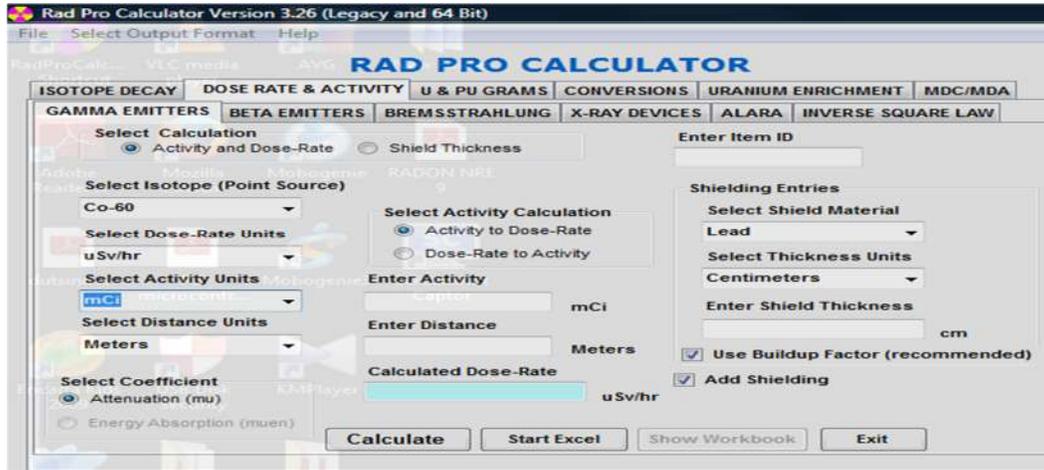


Figure 1: Rad pro calculator (Rad pro calculator software)

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.

RESULTS

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

	Activity	Measurement	Shield	Shield	Shield	Calculated	
Isotope	(mCi)	Distance (m)	Coefficient(/cm)	Material	Thickness (cm)	(μ Sv/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

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	Activity	Measurement	Shield	Shield	Shield Thickness	Calculated Dose-Rate	Log ₁₀ Dose-Rate
Isotope	(mCi)	Distance (m)	Coefficient(/cm)	Material	(cm)	(μ Sv/hr)	
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

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

	Activity	Measurement	Shield	Shield	Shield Thickness	Calculated Dose-Rate	Log ₁₀ Dose-Rate
Isotope	(mCi)	Distance (m)	Coefficient(/cm)	Material	(cm)	(μ Sv/hr)	
Co-60	5	1	0.05	Concrete		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	/.05	Concrete	70	0.06	-1.21

Table 4.3 Dose rate Calculation of 5mCi of ⁶⁰Co at 1 meter distance from the detector using iron shielding at various thicknesses

	Activity	Measurement	Shield	Shield	Shield Thickness	Calculated Dose-Rate	Log ₁₀ Dose-Rate
Isotope	(mCi)	Distance (m)	Coefficient(/cm)	Material	(cm)	(μ Sv/hr)	
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

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Table 4.4 Dose rate Calculation of 5 mCi of ⁶⁰Co at 1 meter distance from the detector using Uranium shielding at various thicknesses

Isotope	Activity (mCi)	Measurement		Shield Material	Shield Thickness (cm)	Calculated Dose-Rate (μSv/hr)	Log ₁₀ Dose-Rate
		Distance (m)	Shield Coefficient(/cm)				
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

Table 4.5 Dose rate Calculation of 5mCi of ⁶⁰Co at 1 meter distance from the detector using Lead glass shielding at various thicknesses

Isotope	Activity (mCi)	Measurement		Shield Material	Shield Thickness (cm)	Calculated Dose-Rate (μSv/hr)	Log ₁₀ Dose-Rate
		Distance (m)	Shield Coefficient				
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

Table 4.6 Dose rate Calculation of 5mCi of ⁶⁰Co at 1 meter distance from the detector using Aluminum shielding at various thicknesses

Isotope	Activity (mCi)	Measurement		Shield Material	Shield Thickness (cm)	Calculated Dose-Rate (μSv/hr)	Log ₁₀ Dose-Rate
		Distance (m)	Shield Coefficient				
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

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Table 4.7: Dose rate Calculation of 5mCi of ⁶⁰Co at 1 meter distance from the detector using Tungsten shielding at various thicknesses

Isotope	Activity (mCi)	Measurement		Shield Material	Shield Thickness (cm)	Calculated Dose-Rate (μSv/hr)	Log ₁₀ Dose-Rate
		Distance (m)	Shield Coefficient(/cm)				
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

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 Table 4.8 shows the results of lead shielding, Table 4.9 shows the results concrete shielding, Table 4.10 shows the results of iron shielding, Table 4.11

shows the results of uranium shielding, Table 4.12 shows the results of lead glass shielding, Table 4.13 shows the results of aluminum shielding, Table 4.14 shows the results of tungsten shielding.

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

Isotope	Activity (mCi)	Measurement		Shield Material	Shield Thickness (cm)	Calculated Dose-Rate (μSv/hr)	Log ₁₀ Dose-Rate
		Distance (m)	Shield Coefficient				
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

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

Isotope	Activity (mCi)	Measurement		Shield Material	Shield Thickness (cm)	Calculated Dose-Rate (μSv/hr)	Log ₁₀ Dose-Rate
		Distance (m)	Shield Coefficient				
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

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Table 4.10 Dose rate Calculation of 22mCi ofCs-137 at 1 meter distance from the detector using iron shielding at various thicknesses

Isotope	Activity (mCi)	Measurement		Shield Material	Shield Thickness (cm)	Calculated Dose-Rate (µSv/hr)	Log ₁₀ Dose-Rate
		Distance (m)	Shield Coefficient				
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

Table 4.11 Dose rate Calculation of 22mCi ofCs-137 at 1 meter distance from the detector using tungsten shielding at various thicknesses

Isotope	Activity (mCi)	Measurement		Shield Material	Shield Thickness (cm)	Calculated Dose-Rate (µSv/hr)	Log ₁₀ Dose-Rate
		Distance (m)	Shield Coefficient				
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

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

Isotope	Activity (mCi)	Measurement		Shield Material	Shield Thickness (cm)	Calculated Dose-Rate (µSv/hr)	Log ₁₀ Dose-Rate
		Distance (m)	Shield Coefficient				
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

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

Isotope	Activity (mCi)	Measurement		Shield Material	Shield Thickness (cm)	Calculated Dose-Rate ($\mu\text{Sv/hr}$)	Log ₁₀ Dose-Rate
		Distance (m)	Shield Coefficient				
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

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

Isotope	Activity (mCi)	Measurement		Shield Material	Shield Thickness (cm)	Calculated Dose-Rate ($\mu\text{Sv/hr}$)	Log ₁₀ Dose-Rate
		Distance (m)	Shield Coefficient				
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.

Shield Material	Linear Attenuation coefficient μ (cm^{-1})		%deviation
	Rad pro	XCOM	
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%



DISCUSSION

Table 4.1 shows the results of calculated dose rate of 5mCi of ^{60}Co 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. The result shows that 12cm of lead shielding can reduce the dose rate of 5mCi of ^{60}Co at 1m distance from the source to detector to $0.06\mu\text{Sv/h}$ which is within the allowable limit.

Table 4.2 shows the results of calculated dose rate of 5mCi of ^{60}Co 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. The result show that 70cm of concrete shielding can reduce the dose rate of 5mCi of ^{60}Co at 1m distance from the source to detector to $0.01\mu\text{Sv}$ however, low density materials can compensate for the disparity with increased thickness which is as significant as density in shielding application.

Table 4.3 shows the result of calculated dose rate of 5mCi of ^{60}Co 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 that 22cm of iron shielding can reduce the photon dose rate of 5mCi of ^{60}Co at 1m distance from the source to detector to $0.06\mu\text{Sv/h}$.

Table 4.4 shows the results of calculated dose rate of 5mCi of ^{60}Co 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. The result shows that 6cm of uranium shielding can reduce the dose rate of 5mCi of ^{60}Co at 1m distance from the source to detector to $0.11\mu\text{Sv/h}$.

Table 4.5 shows the results of calculated dose rate of 5mCi of ^{60}Co 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. The result shows that 10cm of lead glass shielding can reduce the dose rate of 5mCi of ^{60}Co at 1m distance from the source to detector to $0.08\mu\text{Sv/h}$.

Table 4.6 shows 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 is 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. The result shows that 42cm of aluminum shielding can reduce the dose rate of 5mCi of ^{60}Co at 1m distance from the source to detector to $0.11\mu\text{Sv/h}$.

Table 4.7 shows the results of calculated dose rate of 5mCi of ^{60}Co at 1m distance from the source to detector using tungsten shielding at various thickness. The simulation results obtained with tungsten showing decreasing photon dose rate with increasing tungsten shielding thickness. The result shows that 6cm of tungsten shielding can reduce the dose rate of 5mCi of ^{60}Co at 1 m distance from the source to detector to $0.08\mu\text{Sv/h}$.

Table 4.8 shows the results of calculated dose rate of 22mCi 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. The result shows that 6cm of lead shielding can reduce the dose rate of 22mCi of Cs-137 at 1m distance from the source to detector to $0.07\mu\text{Sv/h}$. 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 22mCi 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. The result shows 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 $0.08\mu\text{Sv/h}$. using the same shielding material as the case of ^{60}Co but due to the energies difference ^{60}Co 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. The result shows that 16cm of iron shielding can reduce the dose rate of 22mCi of Cs-137 at 1m distance from the source to detector to $0.07\mu\text{Sv/h}$. unlike that of ^{60}Co 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. The result shows that 4 cm of iron shielding can reduce the dose rate of 22mCi of Cs-137 at 1m distance from the source to detector to $0.07\mu\text{Sv/h}$.

Table 4.12 showed the results of calculated dose rate of 22mCi 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. The result shows that 3 cm of uranium shielding can reduce the dose rate of 22mCi of Cs-137 at 1 m distance from the source to detector to $0.05\mu\text{Sv/h}$.

Table 4.13 showed the results of calculated dose rate of 22mCi 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. The result shows that 6cm of lead glass shielding can reduce the dose rate of 22mCi of Cs-137 at 1m distance from the source to detector to 0.05 μ Sv/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. The result shows that 6cm of aluminum shielding can reduce the dose rate of 22mCi of Cs-137 at 1m distance from the source to detector to 0.09 μ Sv/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).

CONCLUSION

In conclusion, Rad Pro Calculator simulation is a veritable tool for modeling certain real life situations, and very useful in shielding calculation. The above results shows 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.

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