JOINT INSTITUTE FOR NUCLEAR RESEARCH

Dzhelepov Laboratory of Nuclear Problems

FINAL REPORT ON THE INTEREST PROGRAMME

Radiation Protection and the Safety of Radiation Sources

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Abstract

This project delves into the fundamental aspects of radiation protection and radiation sources, exploring various types, units, and quantification methods. It encompasses radiation protection principles, scintillation detectors, peak integration, energy calculation, source identification, attenuation coefficient determination, and alpha particle range in air. Utilizing software like ROOT, Origin Analysis, Excel, and SRIM simulation, the study compares scintillation detectors, emphasizing the superior resolution of NaI over BGO. The project also addresses energy determination for unknown sources, attenuation coefficients for aluminum and copper, and alpha particle range determination using diverse detectors. Overall, the project successfully achieves its objectives, advancing understanding in radiation science.

Key words

Radiation protection, scintillation detectors, NaI detector, BGO detector, energy calibration, , attenuation coefficient, alpha particle range, radioactive materials, safety culture.

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1 Introduction

Radioactivity is a natural phenomenon in the universe, characterized by the spontaneous release of energy or particles when unstable atoms transition to a stable state. This release of energy or particles is known as radiation. Radiation can be classified into two types: ionizing and non-ionizing, based on its ability to penetrate matter. Ionizing radiation has the capability to detach electrons from atoms or molecules, leading to atomic-level changes when it interacts with matter, including living organisms. Alpha particles, beta particles, neutrons, gamma rays, and x-rays are examples of ionizing radiation, which can potentially damage cells, organs, and even cause death at high doses. On the other hand, non-ionizing radiation has lower energy and primarily induces molecular vibration and the production of heat. Microwave radiation, UV light, radio waves, and infrared light are examples of non-ionizing radiation.

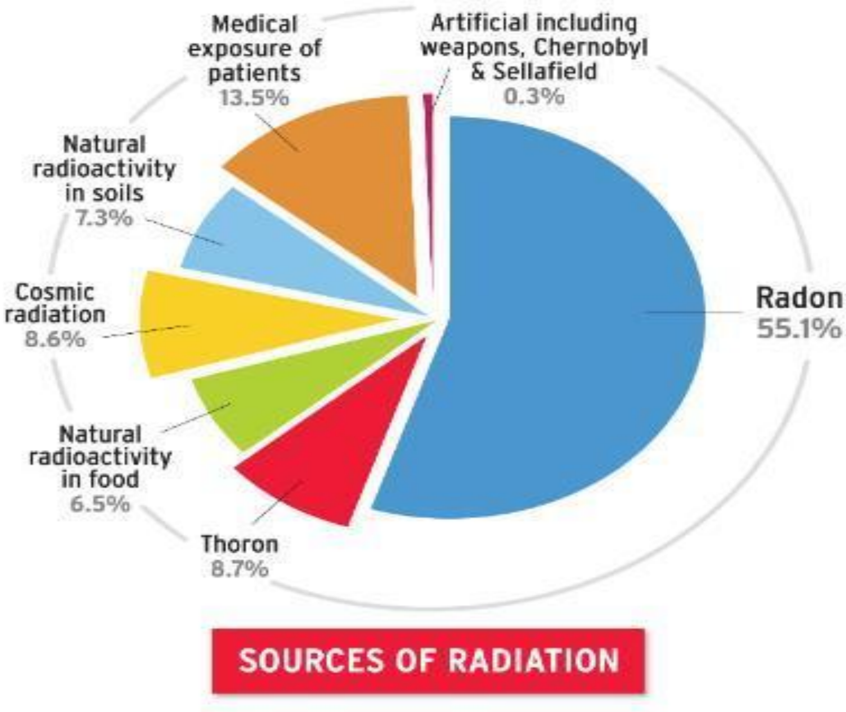
Despite its potential dangers, radiation also plays a crucial role in various fields and can save lives. Radioactive sources are widely used in medicine, energy production, industry, agriculture, space exploration, law enforcement, and more. When used correctly, with appropriate doses and necessary safety measures, ionizing radiation can be harnessed for countless beneficial applications. However, it is essential to assess and control radiation risks to protect workers, the public, and the environment. Safety standards govern activities involving ionizing radiation, such as medical procedures, nuclear facility operations, production and transportation of radioactive materials, and management of radioactive waste.

In response to these concerns, radiation protection practices have been developed to reduce unnecessary radiation exposure and minimize the harmful effects of ionizing radiation. The primary objective of this project is to establish a strong foundation in radiation protection and radiation sources. Additionally, it aims to provide practical skills and basic tools necessary for individuals interested in working in the field of radiation protection and the safe utilization of radiation sources through a series of laboratory works.

2 Background

2.1 Sources of Radiation Exposure

Humans are constantly surrounded by radiation, which can be broadly categorized into two types: natural background radiation and artificial radiation. Natural background radiation originates from three sources: cosmic radiation in space, terrestrial radiation from the Earth's natural environment, and internal radiation within the human body. On the other hand, artificial radiation is generated as a result of human activities, including nuclear weapon testing, medical applications of radiation, consumer products, and more. Artificial radiation exposure can be experienced by two distinct groups: the general public and occupationally exposed individuals. Regulatory measures for radiation protection are tailored to address the specific needs of these groups.



Radiation exposure can also be characterized based on the pathway of irradiation. External irradiation occurs when radioactive substances and environmental radiation sources irradiate the human body from the outside. Alternatively, internal irradiation can occur when individuals inhale radioactive substances in the air, ingest them through food and water, or absorb them through the skin, resulting in radiation within the body. Globally, doses from external and internal exposure are similar.

2.2 Radiation Units

When ionizing radiation interacts with the human body or objects, it transfers energy, leading to the concept of "dose." Dose quantities are used to measure radiation exposure and can vary depending on factors such as the nature and strength of the radiation source, the biological sensitivity of the exposed area, and parameters like time, distance, and shielding. The three commonly used dose measurements are absorbed dose, equivalent dose, and effective dose.

1. Absorbed dose refers to the amount of energy deposited by radiation in a given mass of material, such as tissue or organs. It is expressed in grays (Gy).
2. Equivalent dose is calculated by multiplying the absorbed dose by a radiation factor (WR), which accounts for the varying biological effects of different types of radiation. An equivalent dose is measured in sieverts (Sv).
3. Effective dose takes into consideration both the equivalent dose and organ factors (WT), which reflect the susceptibility and sensitivity of different organs to radiation damage. Effective dose is also expressed in sieverts (Sv).

These dose measurements play a crucial role in assessing and quantifying the potential risks associated with radiation exposure. By understanding the dose received from different radiation sources, scientists and medical professionals can implement appropriate safety measures and ensure the protection of individuals.

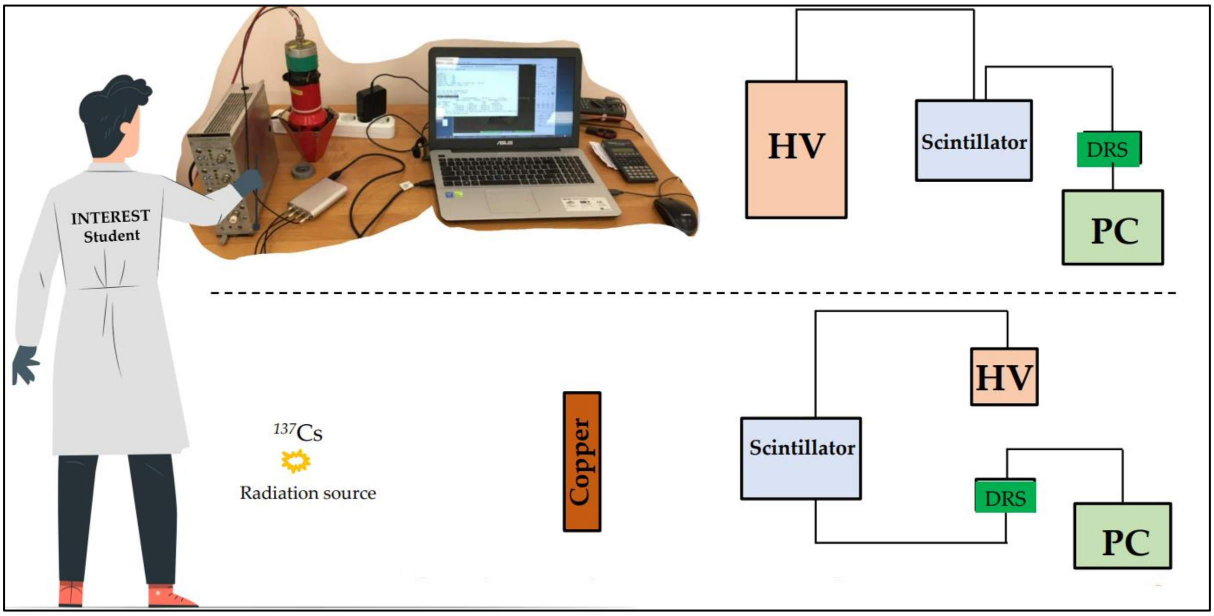
2.3 Principles of Radiation Protection

The principles of radiation protection have been established by the International Commission on Radiological Protection (ICRP) as a framework to safeguard human health and the environment. These principles include "justification," "optimization," and the "application of dose limits."

- Justification: The use of radiation or any activity involving radiation exposure should only be permitted if the benefits outweigh the potential risks. This principle applies to three different scenarios: emergency exposure, existing exposure, and planned exposure.
- Optimization: When the benefits of using radiation outweigh the associated risks, efforts should be made to keep all doses as low as reasonably achievable (ALARA) while considering economic and societal factors. Dose constraints and reference levels are employed to promote the optimization principle.

Application of dose limits: In planned exposure situations, the total dose received by an individual from regulated sources should not exceed the appropriate dose limits. These limits are typically specified in two ways: occupational exposure (50 mSv per year and 100 mSv per five years) and public exposure (1 mSv per year). It's important to note that dose limits do not apply to medical exposure, as they could hinder patients from receiving necessary treatment.

3 Experimental Set-up and Methods

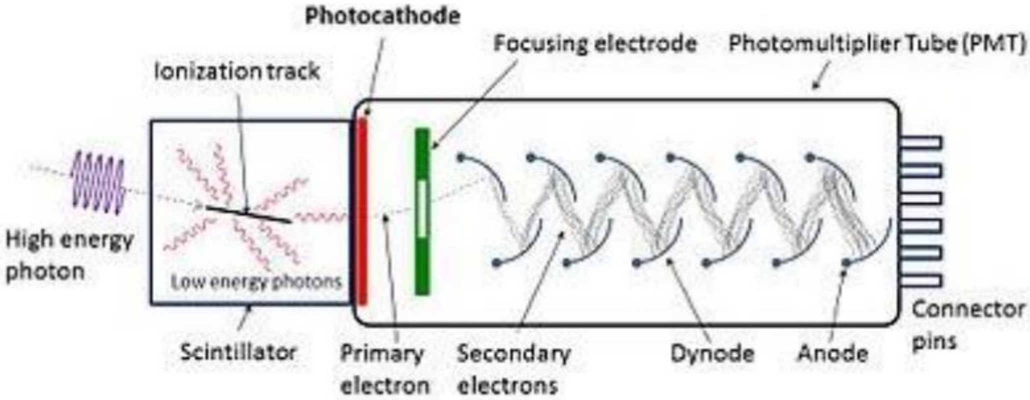


3.1 Scintillator Detectors and Photomultiplier Tube

3.1.1 Scintillator Detectors

Scintillator detectors are devices that produce flashes of light, known as scintillation, when they are excited by ionizing radiation. They have a long history and were initially used with photographic films for radiation measurements. In modern designs, scintillation detectors are often coupled with amplifying devices such as photomultiplier tubes (PMTs), which convert the scintillations into electrical pulses. These pulses are then electronically analyzed and counted to gather information about the radioactive source.

There are two common types of scintillators: inorganic crystals and organic scintillators. In inorganic crystal scintillators, the scintillation mechanism relies on the crystal lattice structure. On the other hand, organic scintillators utilize fluorescence resulting from transitions in the energy levels of a single molecule. The fluorescence can be detected independently of the physical state of the molecule. Various materials are used as scintillators, including inorganic crystals, organic crystals, organic liquids, plastic scintillators, noble gases, and scintillating glass.



The experimental results analyzed in this project primarily utilized two types of inorganic scintillator detectors: BGO and NaI (Tl). Additionally, data were collected from a plastic detector and a pixel detector

Scintillator properties of crystals

Scintillator	Light output	Decay (ns)	Wavelength (nm) max	Density (g/cm2)	Hygroscopic
Na(Tl)	100	250	415	3.67	yes
CsI	5	16	315	4.51	slightly
BGO	20	300	480	7.13	no
BaF2(f/s)	3/16	0.7/630	220/310	4.88	slightly
CaF2	50	940	435	3.18	no
CdWO4	40	14000	475	7.9	no
LaBr3(Ce)	165	16	380	5.29	yes
LYSO	75	41	420	7.1	no
YAG(Ce)	15	70	550	4.57	no

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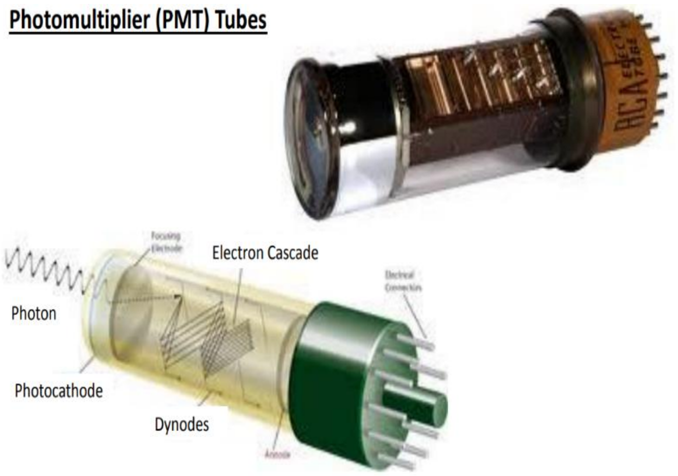
1. BGO detectors, crafted from Bismuth

BGO detectors, crafted from Bismuth stand out as highly effective absorbers of gamma rays, owing to their impressive peak-to-total ratio derived from elevated atomic number, density, and Z values. These crystals exhibit durability, toughness, and resistance to moisture, maintaining their structural integrity without cleaving or significant self-absorption of scintillation light. BGO finds varied applications, including Positron Emission Tomography (PET), Compton suppression spectrometers, geological logging, and specialized functions in space and medical physics.

2. NaI (Tl) detectors

   NaI (Tl) detectors, incorporating Sodium Iodide doped with Thallium, feature exceptional scintillation efficiency and are available in single crystal and polycrystalline forms of various sizes and geometries. Widely employed, Sodium Iodide crystals demonstrate high optical output, minimal absorption of scintillation light, and proficiency in detecting gamma rays of low and intermediate energies. Thallium-doped Sodium Iodide detectors produce substantial signals in a Photomultiplier Tube (PMT) per unit of absorbed radiation. Their applications span Time-of-Flight (ToF) measurements, positron lifetime studies, PET, and specialized roles in nuclear and high-energy physics.

3.1.2 Photomultiplier Tube (PMT)



Photomultiplier tubes are commonly used as photodetectors in scintillator detectors. They consist of a photocathode and a series of dynodes. When the scintillation light strikes the photocathode, photoelectrons are emitted and focused onto the first dynode. This process triggers a cascade of electron multiplication, as the electrons generated at each dynode strike the next dynode, dislodging even more electrons. The amplified signal is collected at the anode of the photomultiplier tube. The magnitude of the resulting electrical signal is proportional to the number of photoelectrons generated.

3.2 Tasks

3.2.1 **Task 1**: Relation between Resolution and Applied Voltage for BGO

BGO is an inorganic scintillator (crystal) that converts incident radiation into light within the visible range. It's coupled with a photon multiplier tube (PMT) that converts the light output of a BGO crystal into voltage pulses. Voltage pulses are processed such that they indicate radiation counts.

Resolution of BGO detectors is its ability to accurately determine the energy of the incoming radiation and separate it between adjacent energy peaks.

It is calculated from the peak at full width half maximum (FWHM) divided by the location of the peak centroid.

Where,

$$Resolution = {}^{\mathrm{FWHM}}/_{\mathrm{Mean}}$$

And,

$$FWHM = \sigma \times 2.35$$

Therefore,

$$Resolution = {}^{\sigma}/_{\mathrm{Mean}} * 2.35$$

The energy resolution provides a measure of the detector's ability to differentiate closely spaced energy peaks. A lower energy resolution signifies a higher capability to distinguish neighboring energy peaks, enabling the identification of different decay processes or radionuclides in the radiation spectrum.

Calculations

- ROOT is used to apply gaussian fitting to given data at different applied voltages.
- Resolution is calculated using equation 1.
- Resolution is plotted against voltage using Origin.

Given Data

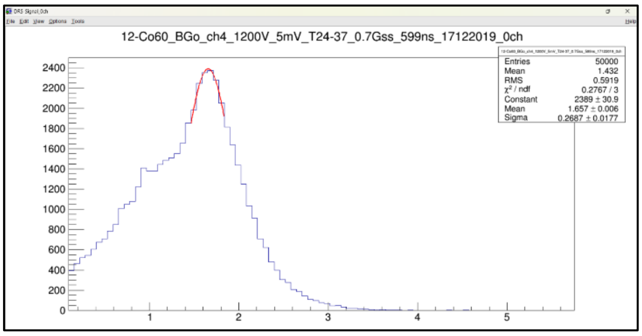


Figure 1 1200V Applied

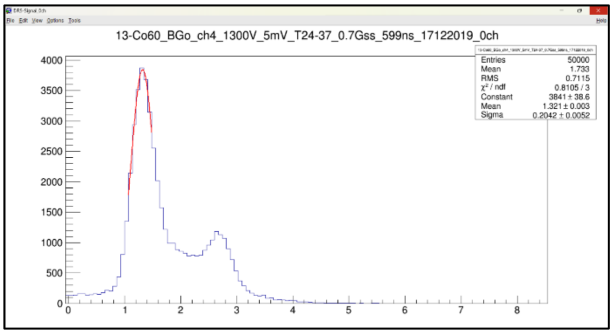


Figure 2 1300V Applied

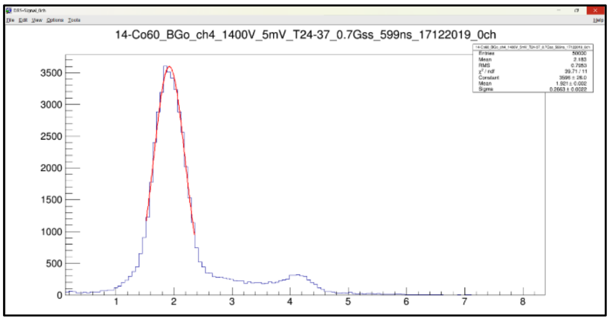


Figure 3 1400V Applied

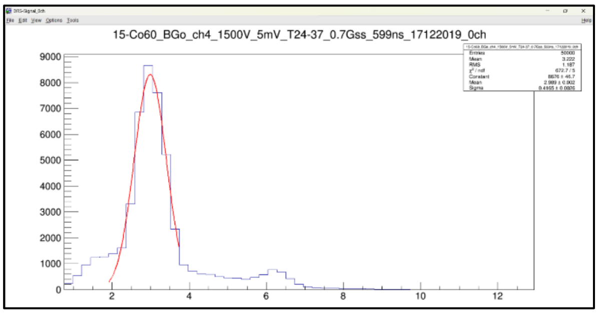


Figure 4 1500V Applied

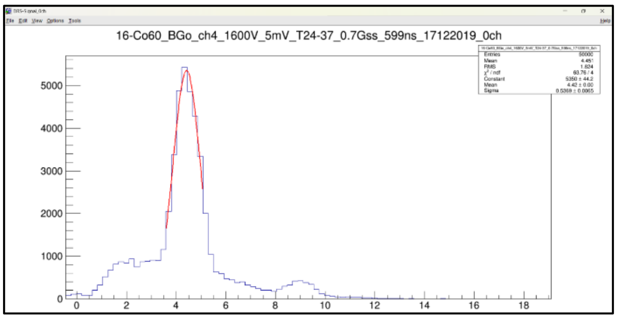


Figure 5 1600V Applied.

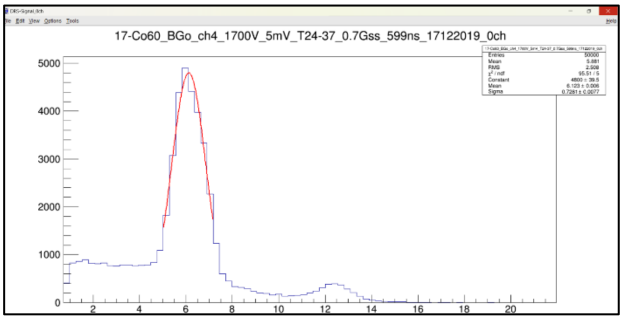


Figure 6 1700V Applied.

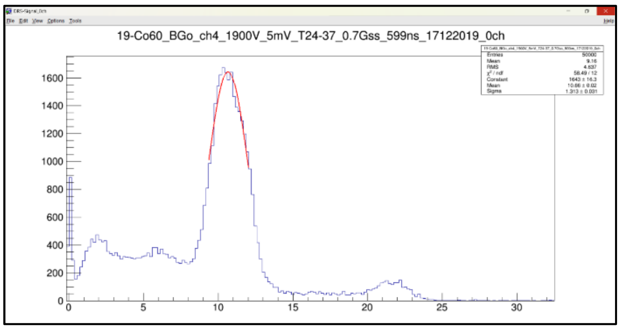


Figure 7 1900V Applied.

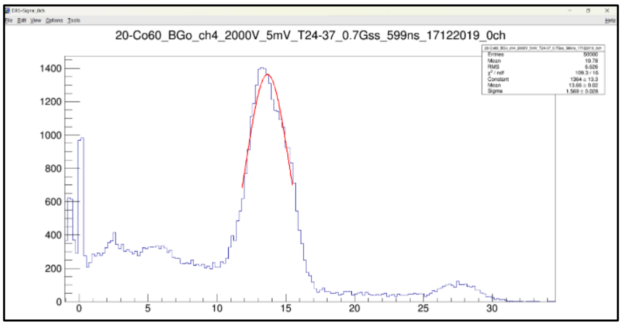


Figure 8 2000V Applied

Results

Applied voltage	σ	Mean	Resolution
1200	0.2687	1.657	0.381077
1300	0.2042	1.321	0.363263
1400	0.2663	1.921	0.325770
1500	0.4165	2.989	0.327459
1600	0.5369	4.420	0.285456
1700	0.7281	6.123	0.279444
1900	1.3131	10.66	0.289451
2000	1.5691	13.66	0.269923

Table 1: Resolution (%) of BGO detector corresponding to the applied voltage.

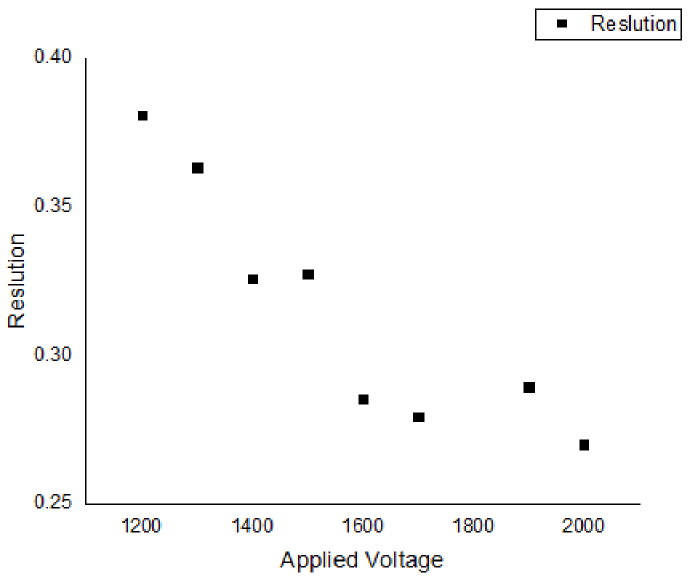


Figure 9, The relation between resolution and applied voltage for BGO detector. Resolution improves with increasing voltage.

3.2.2 **Task 2**: Energy Calibration of BGO Detectors

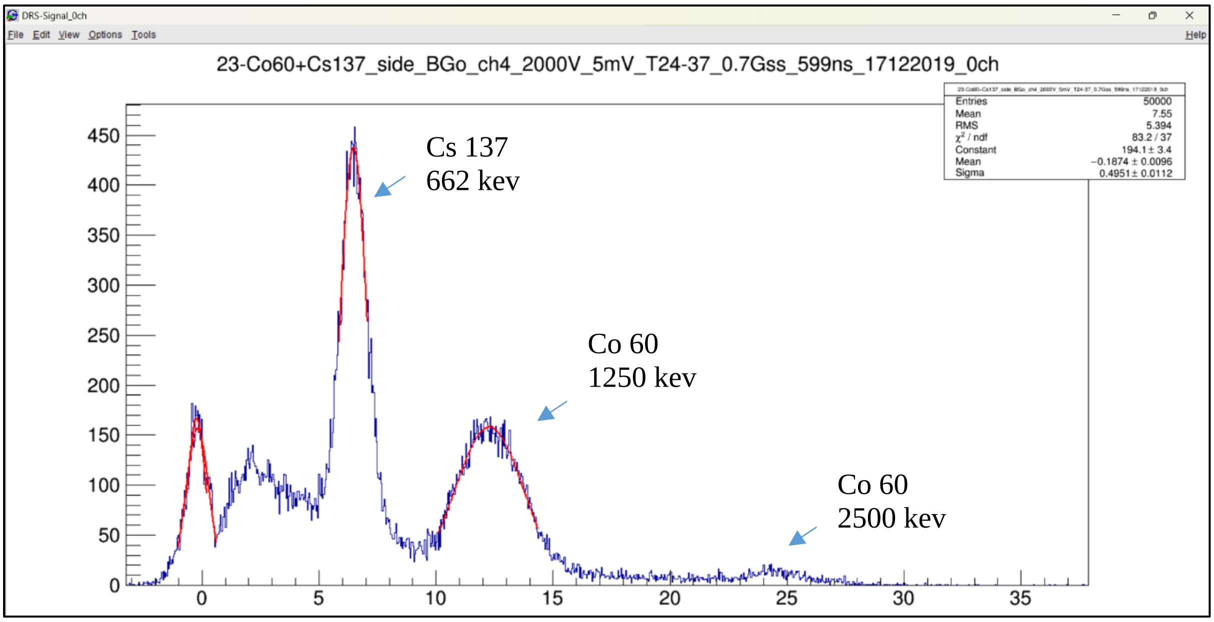


Figure 10 The energy spectrum of Cs-137 and Co-60 from BGO detector measurements at 2000V.

Calibration results in a relationship between the number of a given channel (mean) and its corresponding energy. Therefore, to obtain the relation, a source of known energy peaks (in this case, Cobalt and Cesium) will be used.

- The first peak on the spectrum is noise that originated due to the resolution of the detector, so it’s not a peak of energy from either element.
- Cesium-137 has one energy peak, which is the first from the left. While Cobalt-60 has two peaks.

The channel number (mean) is obtained by making a Gaussian fit using the ROOT software. From the plot of energy vs. channel number (mean), a calibration curve is made, and the equation of the line is generated.

Isotope	Mean	Energy(keV)
Cs-137	6.467	662
Co-60	12.274	1250
	24.467	2500

Table 2 Mean and energy of Cs-137 and Co-60 peaks from a BGO detector.

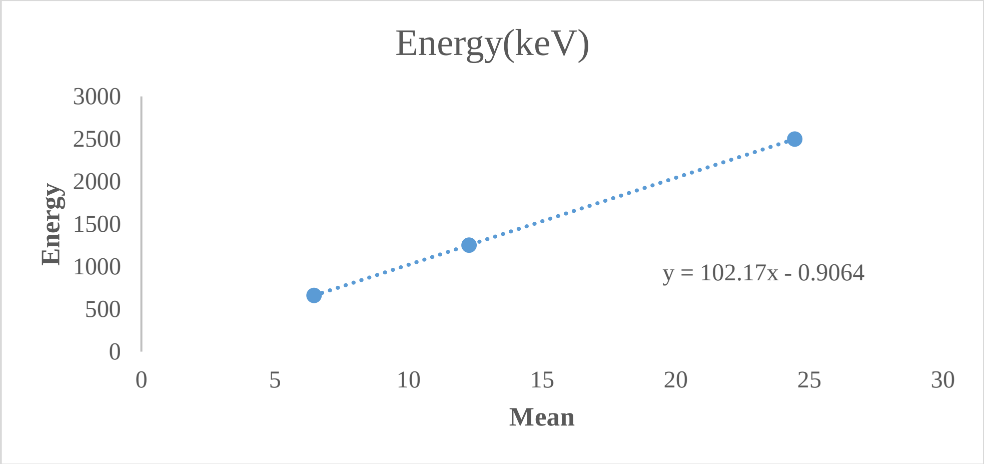


Figure 11, Energy calibration function for Cs-137 and Co-60 spectrum from BGO detector measurements.

The equation of the energy calibration line for BGO detector is:

$y = 102.17x - 0.9064$

Where:

x = channel number (mean),

y = energy of the peaks (in keV)

3.2.3 **Task 3**: Relation of Resolution Against Applied Voltage for NaI Detector

The Sodium Iodide detector functions as a scintillation detector, converting radiation into photons. It consists of a photomultiplier and an electronic circuit, similar to the previously mentioned BGO detector.

In terms of resolution, the NaI detector outperforms the BGO detector by easily distinguishing between peaks. However, both detectors have a maximum achievable resolution of 23%. Consequently, when analyzing Cobalt-60 peaks, the overlapping of its two peaks cannot be avoided.

Resolution of NaI detectors is its ability to accurately determine the energy of the incoming radiation and separate it between adjacent energy peaks.

It is calculated from the peak at full width half maximum (FWHM) divided by the location of the peak centroid.

Where,

$$Resolution = {}^{\mathrm{FWHM}}/_{\mathrm{Mean}}$$

And,

$$FWHM = \sigma \times 2.35$$

Therefore,

$$Resolution = {}^{\sigma}/_{\mathrm{Mean}} * 2.35$$

The energy resolution provides a measure of the detector's ability to differentiate closely spaced energy peaks. A lower energy resolution signifies a higher capability to distinguish neighboring energy peaks, enabling the identification of different decay processes or radionuclides in the radiation spectrum.

Calculations

- ROOT is used to apply gaussian fitting to given data at different applied voltages.
- Resolution is calculated using equation 1.
- Resolution is plotted against voltage using Origin.

Given Data

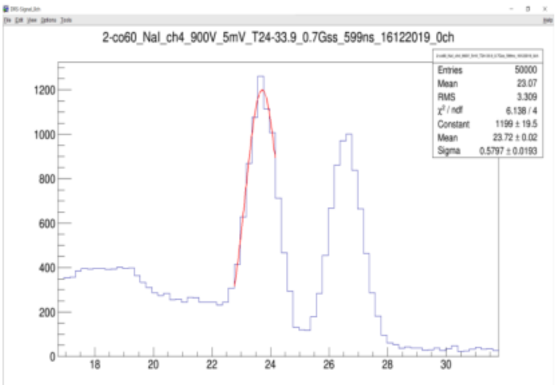


Figure 12 900V Applied

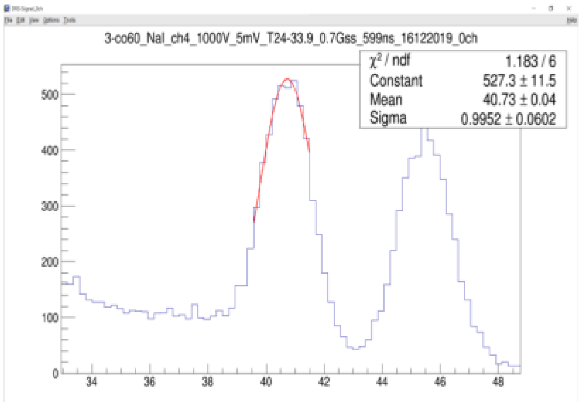


Figure 13 1000V Applied.

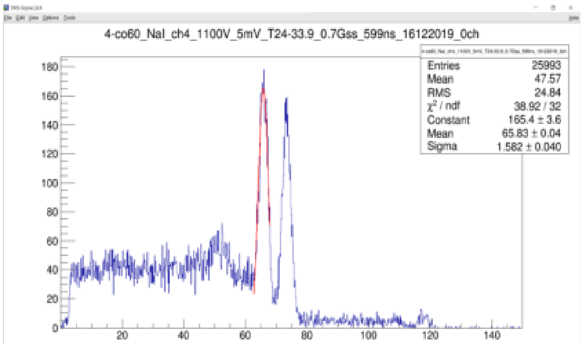


Figure 14 1100 V Applied

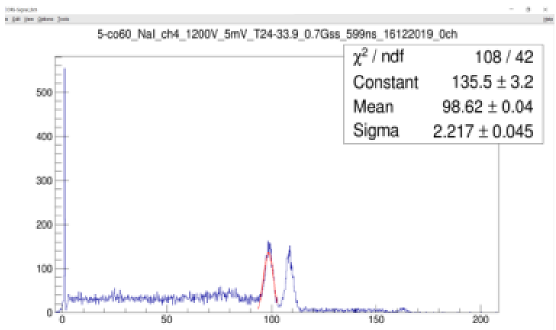


Figure 15 1200V Applied.

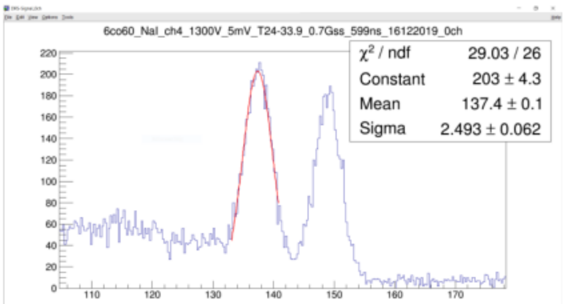


Figure 16 1300V Applied.

Results

Applied voltage	**σ**	Mean	Resolution (%)
900	0.5797	23.37	5.7432
1000	0.9952	40.73	5.7420
1100	1.5620	65.83	5.6474
1200	2.2170	98.62	5.2829
1300	2.4930	137.4	4.2639

Table 3: Resolution (%) of NaI detector corresponding to the applied voltage.

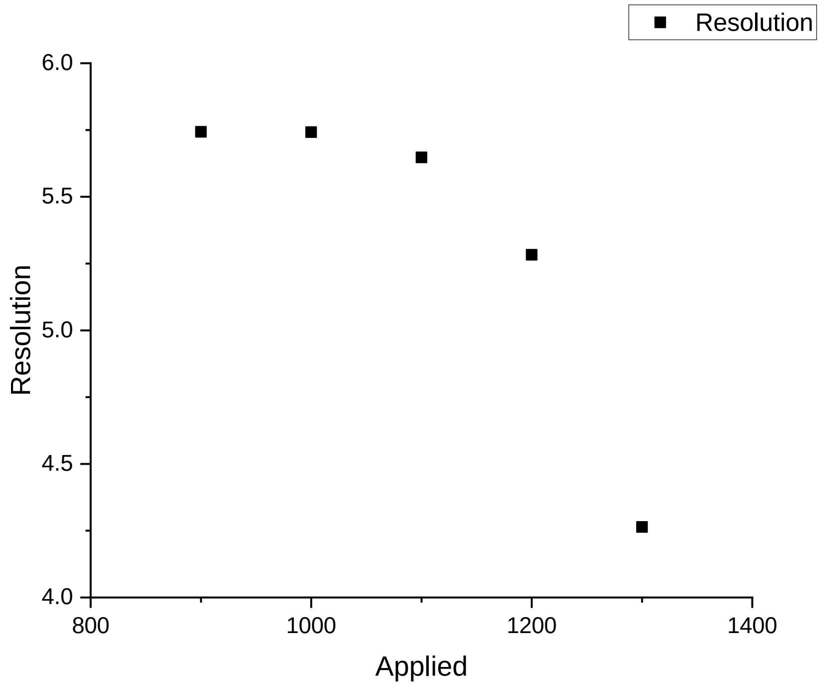


Figure 17, The relation between resolution and applied voltage for NAI detector. Resolution improves with increasing voltage.

3.2.4 **Task 4**: Energy Calibration for NaI Detector

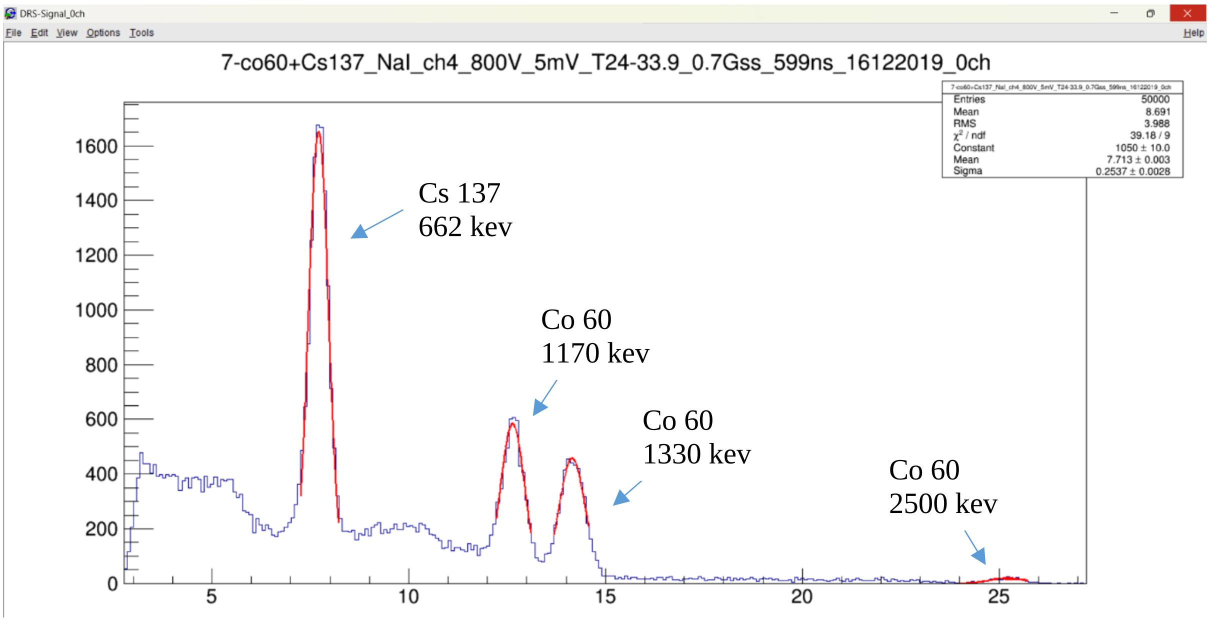


Figure 18, The energy spectrum of Cs-137 and Co-60 from NaI detector measurements at 800V.

Calibration results in a relationship between the number of a given channel (mean) and its corresponding energy. Therefore, to obtain the relation, a source of known energy peaks (in this case, Cobalt and Cesium) will be used.

- The first peak on the spectrum is noise that originated due to the resolution of the detector, so it's not a peak of energy from either element.
- Cesium-137 has one energy peak, which is the first from the left. While Cobalt-60 has three peaks.

The channel number (mean) is obtained by making a Gaussian fit using the ROOT software. From the plot of energy vs. channel number (mean), a calibration curve is made, and the equation of the line is generated.

Isotope	Mean	Energy(keV)
Cs-137	7.4676	662
Co-60	12.274	1170
	14.691	1330
	24.467	2500

Table 4 Mean and energy of Cs-137 and Co-60 peaks from a NaI detector.

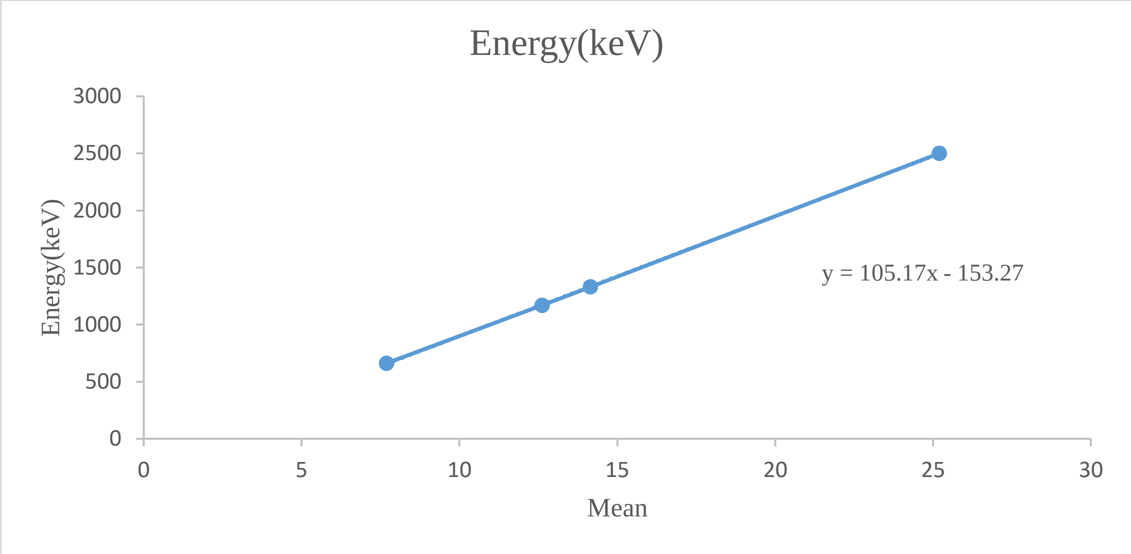


Figure 19, Energy calibration function for Cs-137 and Co-60 spectrum from NaI detector measurements.

The equation of the energy calibration line for NaI detector is:

$y = 105.17x - 153.27$

Where:

x = channel number (mean),

y = energy of the peaks (in keV)

3.2.5 **Task 5**: Identification of Unknown Source by NaI Detector

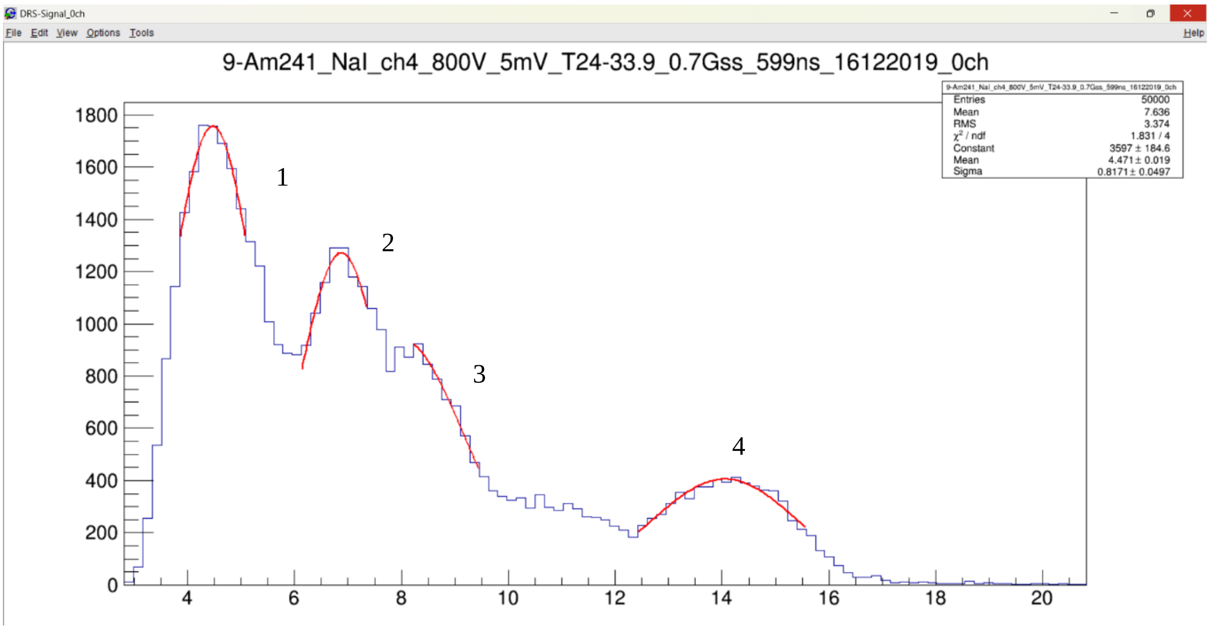


Figure 19, The energy spectrum Unknown source NaI detector measurements at 800V

Calibration results in a relationship between the number of a given channel (mean) and its corresponding energy. Therefore, to obtain the relation, a source of known energy peaks. After obtaining the calibration equation for an NaI detector, it's now possible to identify nuclei based on their characteristic energies.

The channel number (mean) is obtained by making a Gaussian fit using the ROOT software. From the plot of energy vs. channel number (mean), a calibration curve is made, and the equation of the line is generated.

Peak	Mean	Energy(keV)	Unknown source
1	4.551	325	Sn-125
2	6.876	668	Cs-137 / Ba-137
3	8.168	706.8	Tc-125
4	13.995	1358	Mg-28

Table 5, Energy, Mean, and Unknown source of the unknown sources identified using the Nuclide Datasheet.

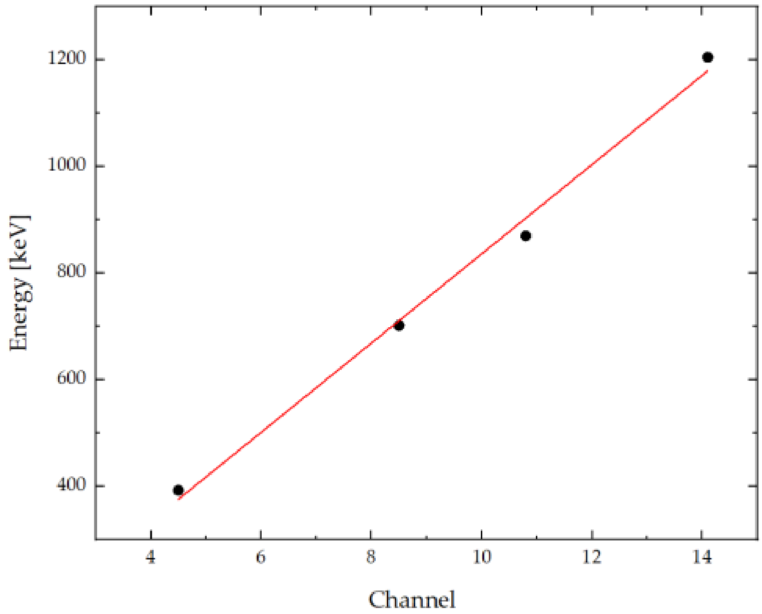


Figure 20, Energy calibration Unknown source NaI detector measurements

The energy of each peak was determined using the linear energy calibration curve.

The equation of the energy calibration line for NaI detector is:

$y = (0.0098\,x) + 1.3316$

Where:

x = channel number (mean),

y = energy of the peaks (in keV)

3.2.6 **Task 6**: Range of Alpha Particles in Air

Alpha particles are positively charged particles consisting of two protons and two neutrons, essentially the same as a helium-4 nucleus. Due to their relatively large mass and double positive charge, alpha particles are characterized by low penetration ability but high ionization potential. This makes them effective at interacting with matter over short distances.

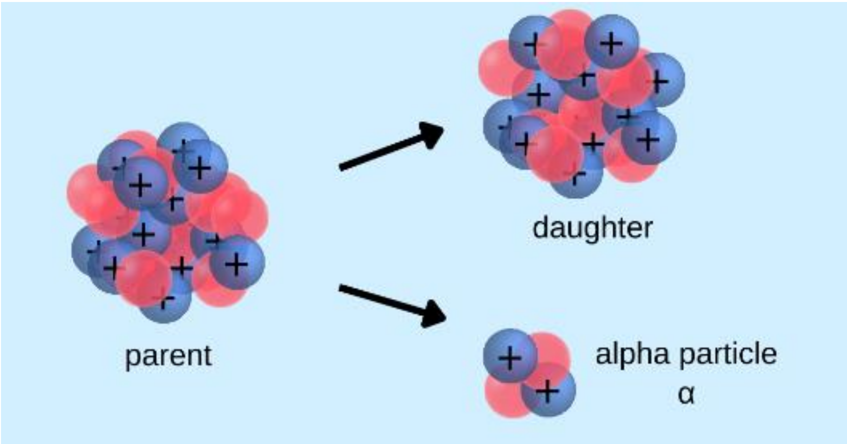


Figure 21, Alpha Particles

In various materials, the range of alpha particles can vary. However, in air, the range is relatively limited due to interactions with air molecules.

The range of alpha particles in air depends on their initial energy, and here are approximate values:

- 3 MeV alpha particle:

Range is about 3 to 4 centimeters in air.

- 5 MeV alpha particle:

Range is about 4 to 6 centimeters in air.

- 7 MeV alpha particle

Range is about 6 to 7 centimeters in air.

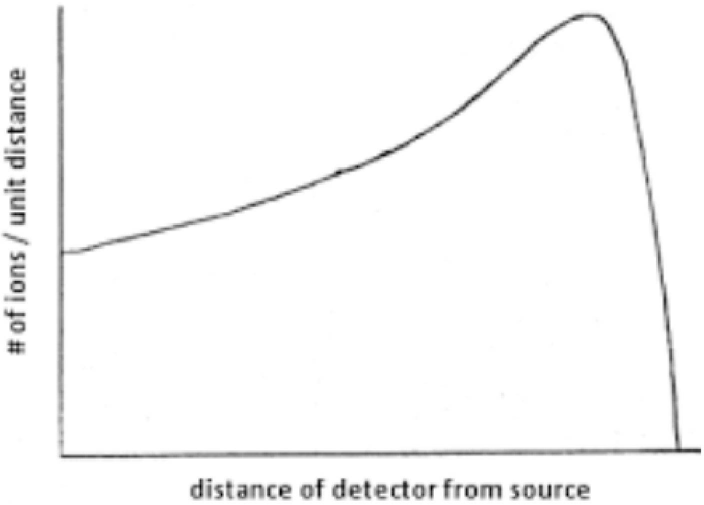


Figure 22, Alpha Particles Range

The interaction mechanisms include ionization and excitation as alpha particles collide with air molecules. This leads to energy loss and a gradual decrease in the ability of alpha particles to penetrate materials. It's crucial to recognize that alpha particles are typically stopped by a few centimeters of air, making them less of an external radiation hazard in open air.

In this experiment, plastic detector is used instead of a BGO detector. This is because BGO detector has a thin aluminum foil layer and shielding can occur, leading to energy loss and inaccurate measurements.

Experimental Equipment:

- Radioactive Source: Pu-239
- Energy of He: 5.5 MeV
- Detector: Plastic Detector
- Voltage: 2000V

Given Data

Distance	Counts
0	440
0.5	390
1	360
1.5	340
2	320
2.5	300
3	280
3.5	260
3.8	260
4	260

Table 6, Alpha radiation counts observed by plastic.

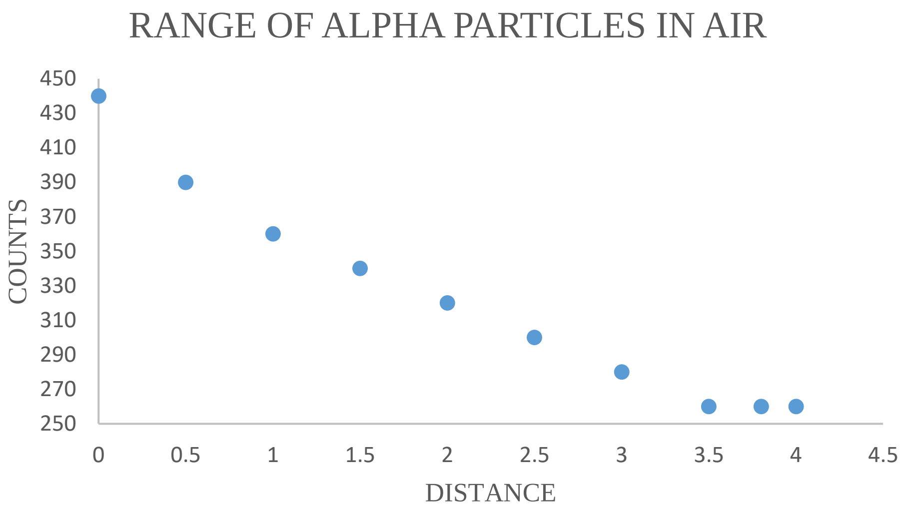
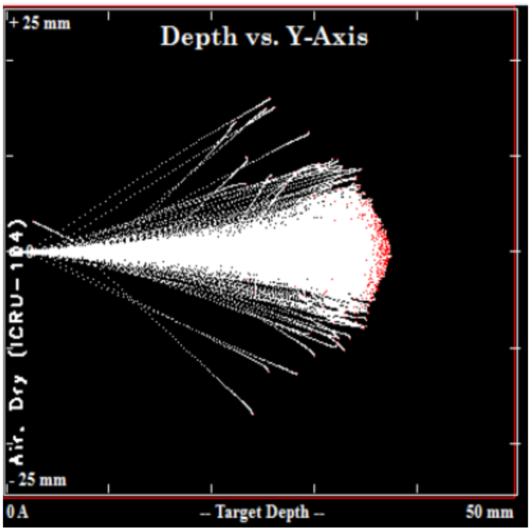


Figure23, Alpha radiation counts observed by a plastic detector.

From the table and the plot, it can be observed that the counts per second decrease as the distance increases, until reaching a point where the number of counts is constant. It means that there is no more signal detected. Therefore, the range of alpha particles in air is about **3.5 cm.**

3.2.6.1 Range of Alpha Particles in Air by SRIM Simulation (Monte Carlo)

Using the SRIM software, it is possible to observe the simulation of the total path length traveled by alpha particles in air. Two plots are obtained: the depth vs. y-axis and the ionization (Bragg peak/curve) of the alpha particles. The Bragg curve represents the energy loss rate as a function of the distance through a stopping medium. The Bragg peak is the maximum, and beyond that, the energy deposition drops sharply.



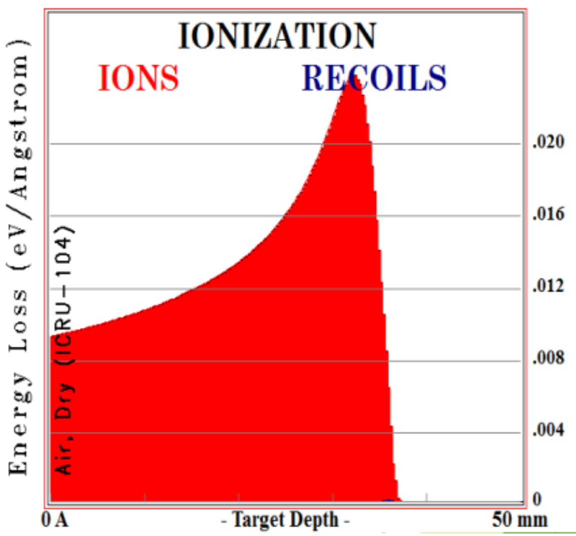


Figure24, Depth and Bragg curve of alpha particles in 5 cm air.

3.2.7 **Task 7**: Attenuation of γ radiation coefficient

The attenuation coefficient of gamma radiation represents the rate at which gamma rays lose their intensity as they pass through a material. It is a crucial factor in understanding how materials interact with and absorb gamma radiation. The attenuation process is influenced by factors such as the material's density and atomic composition.

Mathematically, the attenuation of gamma radiation (I) can be described using the exponential attenuation law:

$$I = I_0 e^{-\mu x}$$

where:

- I is the intensity of gamma radiation after passing through a material.
- I_0 is the initial intensity of the radiation.
- μ is the linear attenuation coefficient of the material.
- x is the thickness of the material.

The linear attenuation coefficient μ is a measure of how effectively a material attenuates gamma radiation per unit thickness. It depends on factors such as the atomic number of the material, the energy of the gamma rays, and the density of the material.

The linear attenuation coefficient μ can be calculated using the following formula:

$$\mu=\ln(I_0/I)$$

where:

- I is the intensity of gamma radiation after passing through a material.
- I_0 is the initial intensity of the radiation.
- x is the thickness of the material.

In practical terms, materials with higher atomic numbers and densities tend to have higher attenuation coefficients for gamma radiation. Common materials used for shielding against gamma radiation include lead and concrete, as they effectively reduce the intensity of gamma rays due to their dense and high atomic number nature.

Experimental Equipment:

- Detector: BGO detector
- Voltage: 2000V
- Radioactive source: Cs-137, E_{Cs}=662 keV
- Attenuation material: Aluminum and Copper

Procedure to calculate it:

1. Measure Initial Intensity I_0

Use a gamma radiation source with a known initial intensity I_0.

Measure the intensity of the gamma radiation emitted by the source without any intervening material.

2. Measure Final Intensity I

- Introduce the material through which gamma radiation will pass.
- Measure the intensity I of the gamma radiation after it has passed through the material.

3. Determine Material Thickness x

- Measure the thickness x of the material that the gamma radiation traverses.

4. Apply the Attenuation Coefficient Formula: $\mu=\ln(I_0/I)$ to calculate the linear attenuation coefficient.

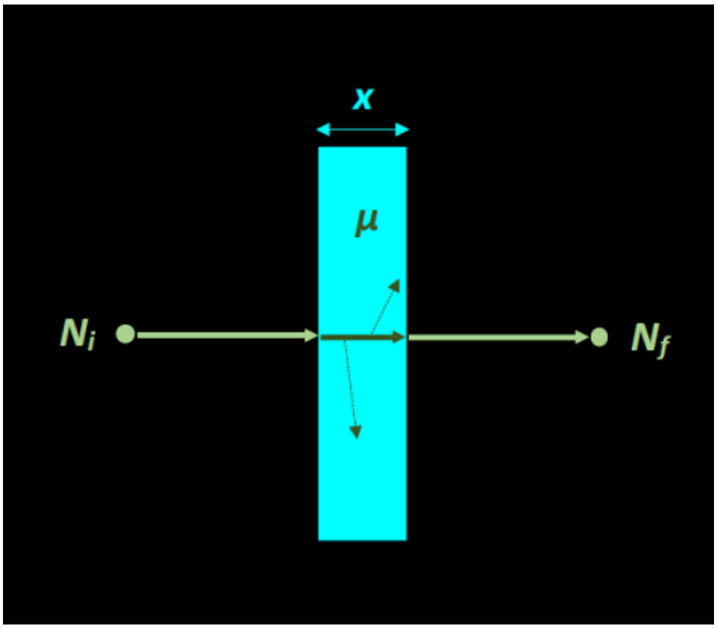


Figure 25, Energy Attenuation of γ radiation coefficient

Results

Higher linear attenuation coefficient indicates greater attenuation, meaning the material is more effective at reducing gamma radiation intensity.

Attenuation coefficient FOR

A) Aluminum

Thickness (cm)	I/I_0
0	1
0.15	0.75573
0.3	0.71623
0.45	0.70569
0.75	0.68596
0.9	0.67155
1.08	0.66103
1.26	0.63939

Table 7

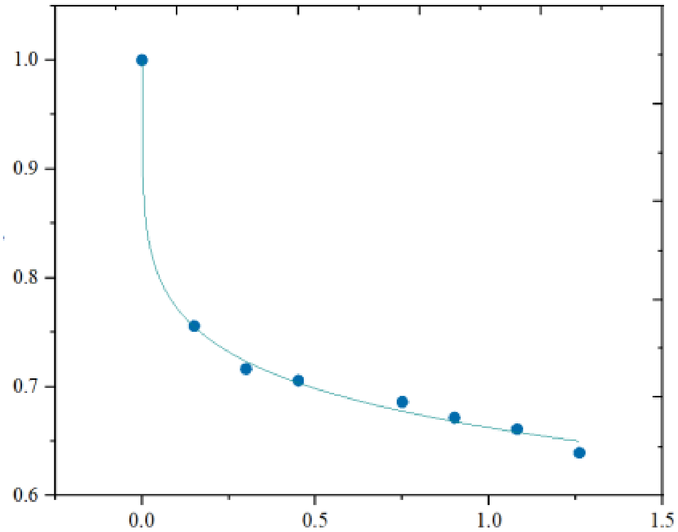


Figure26

Determination of attenuation coefficient for Al using BGO scintillation detector and the radiation source 137Cs (661 keV)

Therefore, μ for copper = 0.238 cm^2 /gm

B) Copper

Thickness (cm)	I/I_0
0	1
0.1	0.73931
0.25	0.7357
0.4	0.68065
0.8	0.58611
1	0.53827
1.2	0.48042

Table 8

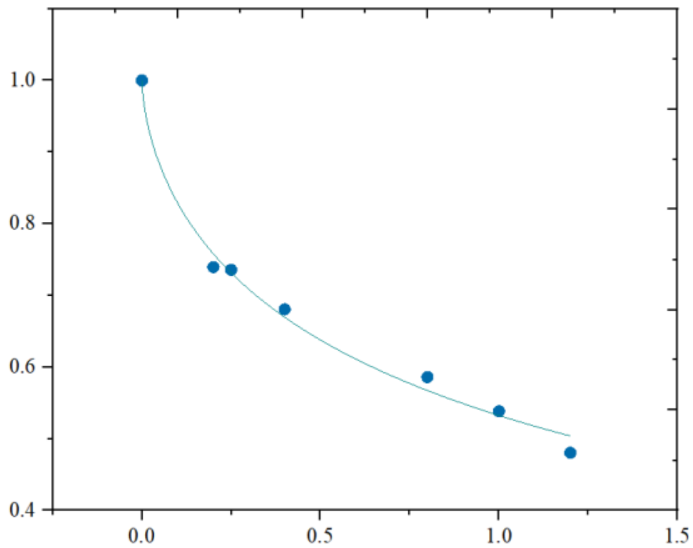


Figure 27

Determination of attenuation coefficient for Cu using BGO scintillation detector and the radiation source 137Cs (661 keV)

Therefore, μ for copper = 0.628 cm^2 /gm

3.2.8 **Task 8**: Pixel detectors

A pixel detector is a type of radiation detector that is designed with a pixelated structure, where the sensing element is divided into an array of small individual pixels. Each pixel in the detector can independently register the presence and intensity of ionizing radiation, providing detailed spatial information about the radiation field.

Resolution: Pixel detectors offer high spatial resolution, allowing precise localization of radiation interactions. This is particularly important in applications where detailed imaging or tracking of radiation events is crucial.

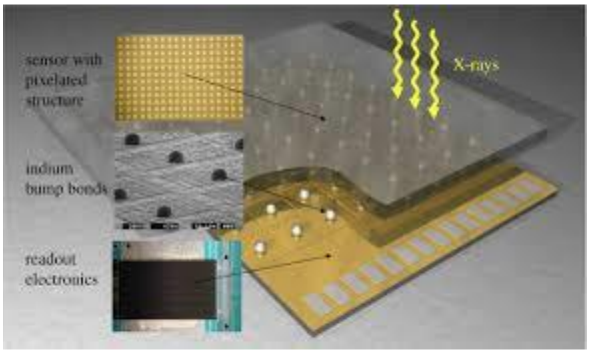


Figure 28 Pixel detectors

Types of Pixel Detectors:

- Silicon Pixel Detectors: Commonly used in many applications, these detectors use silicon as the sensing material. When ionizing radiation interacts with the silicon, it produces charge carriers that can be detected and analyzed.
- CMOS (Complementary Metal-Oxide-Semiconductor) Pixel Detectors: CMOS technology allows for the integration of pixel detectors with readout electronics on the same chip, providing compact and efficient systems.

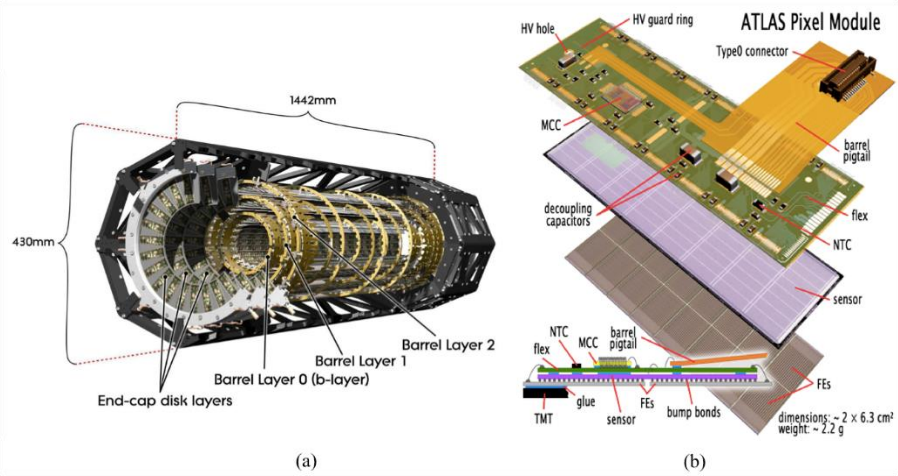


Figure 29 Pixel detectors

Characteristics of pixel detectors:

- Advanced detectors, like a digital camera.
- They consist out of 3 parts:
  1) Sensor (Si)
  2) Electronic chip
  3) USB
- The size of the sensor is 1.5x1.5cm and it has 256 x 256 pixels.
- The pixel size is 55µm x 55µm.
- These detectors have high resolution and are used for registration of different types of radiation.

Determination of the range of Alpha particles with (241Am) energy about 4 MeV in air using pixel detector.

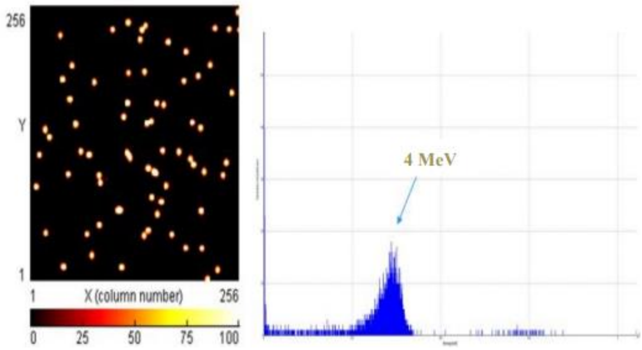


Figure 30, Absorption of αparticle energy at 0 cm in the

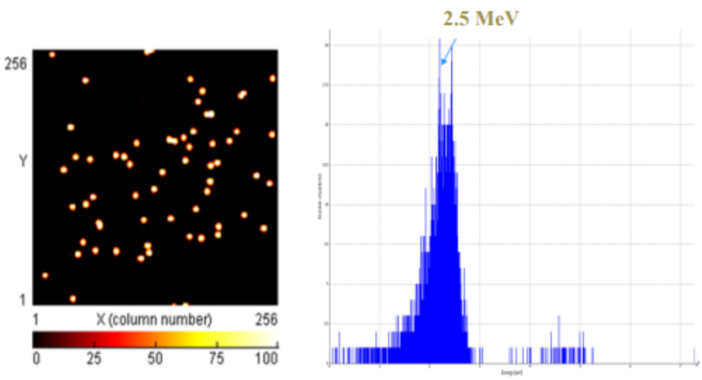


figure 31, Absorption of αparticle energy at 1 cm in the air

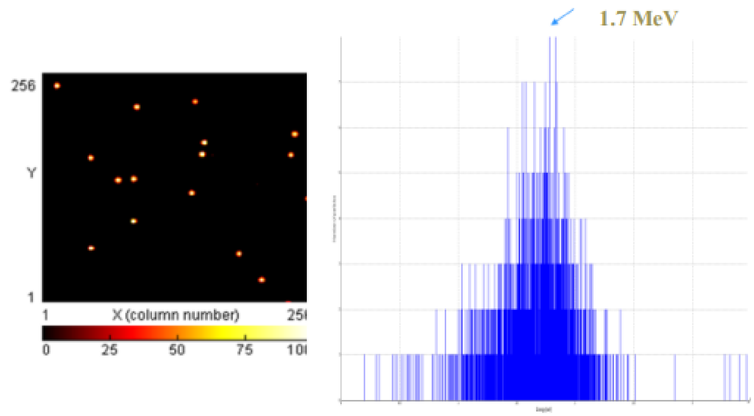


Figure 32, Absorption of αparticle energy at 2 cm in the air

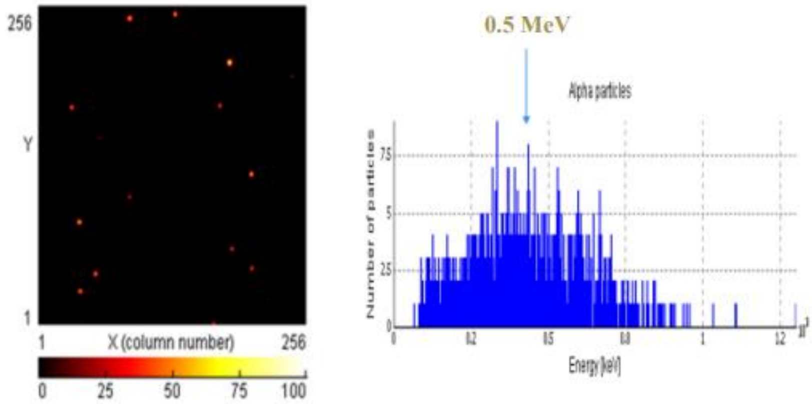


figure 33, Absorption of αparticle energy at 2.5 cm in the air

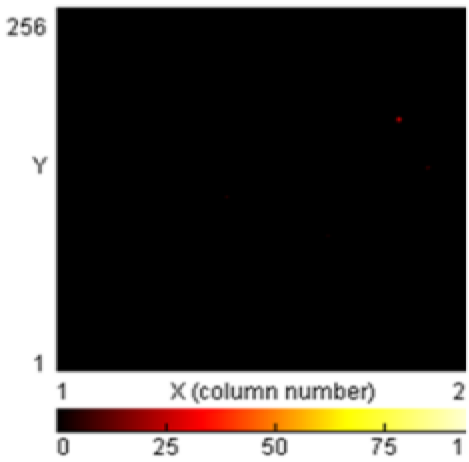


Figure34, No α-particles are detected which means maximum of α-particle range is 3 cm in this case.

Result

At a 3 cm distance from the source, there are no alpha particles detected. Therefore, the maximum range of alpha particles in air as measured by a pixel detector is about 3 cm.

4 Conclusions

In conclusion, this project significantly contributes to our understanding of radiation protection and safety. Through theoretical exploration and hands-on experiments, various skills and experiences were acquired, ranging from detecting radiation sources to assessing attenuation coefficients. The comparative analysis of BGO and NaI detectors reveals nuanced characteristics, emphasizing the importance of proper resolution. The project successfully identifies unknown sources, calculates resolutions, determines alpha particle ranges, and assesses attenuation coefficients. This comprehensive exploration underscores the critical role of promoting a culture of safety and responsibility in activities involving radioactive materials, showcasing the broader implications of the study beyond the realm of nuclear research.

5 References

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